Individual Plant Examination

of the

Perry Nuclear Power Plant

July 1992

Cleveland Electric Illuminating Company

Abstract

A Level 2 probabilistic risk assessment of the Perry Nuclear Power Plant has been performed in response to the NRC Generic Letter No. 88-20 dated November 23, 1988 requesting an Individual Plant Examination for Severe Accident Vulnerabilities - 10CFR50.54(*f*).

The primary results are a delineation of the likely frequency of core damage as the result of internal initiating events including internal flooding and the expected frequency and magnitude of fission product release as the result of the containment response following core damage. In addition the specific contribution of decay heat removal failure to core damage has been assessed.

The total core damage frequency is approximately 1.3 X 10^{-5} per year. anticipated transients without scram and loss of offsite power are the major contributors to core damage. Containment failure as the result of steam over pressure is the dominant contributor to offsite releases.

Acknowledgments

The Perry Individual Plant Examination was performed by a team of personnel from Cleveland Electric Illuminating Company, Halliburton NUS Environmental Corporation, Gilbert Commonwealth, and Gabor Kenton and Associates. The project has benefited from advice and review from RAPA in the case of the Level 1 portion of the study and ERIN in the case of the Level 2.

The provision of information and performance of review by the operational and engineering staff at Perry was a key factor in ensuring the accuracy of the final plant model.

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TABLE OF ACRONYMS

A	Large LOCA
AC	Alternating Current
ADS	Automatic Depressurization System
APET	Accident Progression Event Tree
ARI	Alternate Rod Insertion or Alarm Response Instructions
ASEP	Accident Sequence Evaluation Program
ATWS	Anticipated Transient Without Scram
BOP	Balance of Plant
BWR	Boiling Water Reactor
CCCW	Control Complex Chilled Water System
CCI	Core Concrete Interaction
CD	Core Damage
CDF	Core Damage Frequency
CET	Containment Event Tree
CM	Mechanical Failure of Control Rods (Basic Event)
CRD	Control Rod Drive
CS	Containment Spray
CST	Condensate Storage Tank
CTS	Condensate Transfer System
Cv	Core Vulnerable
DAM	Disallowed Maintenance
DC	Direct Current
DET	Decomposition Event Tree
D/G	Diesel Generator
ECC	Emergency Closed Cooling System
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure

TABLE OF ACRONYMS

EPG Emergency Procedure Guideline

EPRI Electric Power Research Institute

ERIN ERIN Engineering and Research, Inc.

ESF Engineered Safety Feature

ESW Emergency Service Water System

EVNTRE Event Progression Analysis Program

FMEA Failure Modes and Effects Analysis

FPCC Fuel Pool Cooling and Clean-up System

FW Feedwater

G/C Gilbert/Commonwealth, Inc.

HEP Human Error Probability

HI Human Interaction

HPCS High Pressure Core Spray

HRA Human Reliability Analysis

HVAC Heating Ventilation and Air Conditioning

IA Instrument Air

INPO Institute of Nuclear Power Operations

IOI Integrated Operating Instruction

IORV Inadvertent Open Relief Valve

IREP Interim Reliability Evaluation Program

LER Licensee Event Report

LOCA Loss of Coolant Accident

LOOP Loss of Offsite Power

LPCI Low Pressure Coolant Injection

LPCS Low Pressure Core Spray

MAAP Modular Accident Analysis Program

MCC Motor Control Center

TABLE OF ACRONYMS

MFP Motor Feed Pump

MOV Motor Operated Valve

MSIV Main Steam Isolation Valve

NCC Nuclear Closed Cooling System

NPRDS Nuclear Plant Reliability Data System

NPSH Net Positive Suction Head

NRC Nuclear Regulatory Commission

NSSS Nuclear Steam Supply System

NSSSS Nuclear Steam Supply Shutoff System

NUCAP NUS Containment Accident Processes Workstation

NUPRA NUS Probabilistic Risk Assessment Workstation

NUS Haliburton NUS Corporation

P&ID Piping and Instrumentation Drawing

PCS Power Conversion System

PDS Plant Damage State

PEI Plant Emergency Instructions

PRA Probabilistic Risk Assessment

RAPA Reliability and Performance Associates

RCIC Reactor Core Isolation Cooling

RFBP Reactor Feed Booster Pump

RHR Residual Heat Removal

RPS Reactor Protection System

RPT Recirculation Pump Trip

RPV Reactor Pressure Vessel

RSSMAP Reactor Safety Study Methodology Application Program

S1 Intermediate LOCA

S2 Small LOCA

TABLE OF ACRONYMS

SA	Service Air
SBO	Station Blackout
SIA	Safety Related Instrument Air
SLC	Standby Liquid Control System
SOI	System Operating Instruction
SPC	Suppression Pool Cooling
SPMU	Suppression Pool Make-Up
SRV	Safety Relief Valve
SW	Service Water
T1	Loss of Offsite Power Transient
T2	Transient with loss of PCS
T3A	Transient with PCS initially available
T3B	Transient caused loss of Feedwater
T3C	Transient caused by IORV
TAF	Top of Active Fuel
TCW	Transient caused by loss of CCCW
TIA	Transient caused by loss of Instrument Air
TSW	Transient caused by loss of Service Water
USAR	Updated Final Safety Analysis Report

1.0 EXECUTIVE SUMMARY

This document presents the results of the Level 2 Probabilistic Risk Assessment (PRA) performed in response to the NRC Generic Letter (NRC, 1988) which requested each licensee to perform a plant examination of each of their nuclear power plants to search for vulnerabilities to severe accidents and propose cost effective safety improvements that reduce or eliminate the important vulnerabilities. The content and format of this report is in accordance with the NRC submittal guidance document (NRC, 1989) and provides sufficient information to show how the results were obtained and how they can be reproduced.

This summary includes a discussion on the background and objectives (1.1), the plant familiarization activities (1.2), the overall methodology (1.3), and a summary of the major findings of the study (1.4).

1.1 BACKGROUND AND OBJECTIVES

The Commission issued a policy statement in 1985 to the effect that, based on available information, existing plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on generic rulemaking or other regulatory requirements for these plants. However it was decided that a systematic evaluation of each plant would be beneficial in that it would provide information on any plant-specific vulnerabilities to accidents and, through their resolution, enhance safety. In addition to identifying each vulnerability it was decided that unlike many earlier PRAs, the IPE should be performed by a team in which utility personnel played a major part as recommended in the generic letter. Thus the IPE would ensure that utility personnel also achieved the following:

1. The development of an appreciation of severe accident behavior.

2. An understanding of the most likely severe accident sequences that could occur at the plant.

3. The gaining of a more quantitative understanding of the overall probabilities of core damage and fission product release.

4. If necessary, the reduction of the overall frequency of core damage and fission product release by modification, where appropriate, of hardware or procedures that would likely prevent or mitigate severe accidents.

It was considered that these objectives, and the additional CEI objective of having a living PRA, would best be achieved by performing a PRA. The advantage of having a living PRA is that it could be used in the future for some or all of the following uses:

1. To compare the costs and benefits of various proposed equipment modifications.

2. To establish what, if any, improvements should be made to procedures.

3. To identify which accident sequences should be used for operator retraining.

4. To determine the variations in risk for a range of maintenance options for a given plant state.

5. To evaluate proposed changes to Technical Specifications or limiting conditions of operation.

It should be emphasized that the involvement of CEI personnel in all aspects of the analysis and the development of the plant models and appropriate support documentation has ensured that the PRA represents the risk posed by the Perry Plant as accurately as possible, and therefore enables the above objectives to be met.

1.2 PLANT FAMILIARIZATION

The project team consisted of five CEI engineers (3 full time and 2 part time) and a number of engineers from Halliburton NUS Environmental Corporation, Gilbert Commonwealth and Associates and Gabor Kenton and Associates. One of the CEI personnel had over seven years experience in the operations department at Perry, where he served as Operations Engineering Human Factors Unit Lead Engineer, and played a major part in developing the emergency operating procedures, as a member of the BWR Owners Group Emergency Operating Procedures Committee. A second member of the group was also SRO certified. The other CEI engineers had considerable plant experience as members of either the Independent Safety Engineering Group or Mechanical Design Section on site at the Perry Plant.

As the study was performed on site at Perry, visits were made to the unit whenever plant-specific information concerning the layout of components, maintenance practices, or specific operational information was required. In addition, there was ready access to the latest drawings, operational and training experience, and the appropriate design, licensing and maintenance engineers. All work products were reviewed by CEI personnel directly involved in plant operation and design as well as by independent review teams.

1.3 OVERALL METHODOLOGY

The methodology used to perform the IPE for Perry is the performance of a Level 2 PRA. The specific approach used is very similar to that used in the recent Sandia PRA of Grand Gulf for the NRC (NUREG/CR-4550, Drouin, 1989). Fundamentally this is based on the use of event trees to develop the sequence of events following a plant transient or loss of coolant accident and fault trees to model the potential system failures at each phase of the sequence of events. Each individual core damage sequence following a transient or loss of coolant accident is quantified by linking together the fault trees for the potential system and support system failures. The results of the quantification define the combination of component or other failures that will lead to inadequate cooling of the core and failures of the containment systems designed to prevent or mitigate the release of fission products. The fission product source term resulting from core damage is determined by extending the analysis of events through the phases of core damage, vessel failure, containment failure, and, ultimately, release of fission products. In order that this process may be represented in a logical manner, an Accident Progression Event Tree [similar to that used in the Grand Gulf study reported in NUREG/CR-4551, (Brown, 1990)] is developed to represent the range of possible events associated with containment loading mechanisms such as steam generation, hydrogen generation and combustion and the subsequent variations in pressure and temperature within containment. In order to investigate these events the MAAP code was used to model core meltdown, vessel failure, hydrogen generation and burning, and the corresponding variations in containment pressure. A containment capacity analysis was performed to determine the various failure modes and ultimate capacity. The comparison of the pressure transients and their frequency with the results of the containment ultimate strength analysis enables the frequency of containment failure in the different modes to be determined.

Finally the source terms for the various containment failure scenarios identified above were estimated based on the work done by Sandia National Laboratories for the evaluation of severe risks (Brown, 1990) and the use of new MAAP analyses for dominant sequences in each release category.

The methodology used to perform the analysis is described in more detail in Chapter 2. As the PRA aimed at identifying specific strengths and weaknesses of the design and operation at Perry, particular emphasis was placed on ensuring that the information used to perform the study (P&ID, operating procedures, maintenance experiences) represented the current condition of the plant (January 1990).

Particular attention was paid to the relationship between system failures following a plant trip and the actions that the operators would take in accordance with the Plant Emergency Instructions (PEI). In this way the PEIs were integrated into the system and sequence analysis so that not only individual operator actions could be assessed but also combinations of operator action required to prevent development or continuation of a sequence of failures that would lead to core damage. The evaluation was based on a careful review of the PEIs related to each scenario. Training and operational staff were interviewed to enhance the qualitative understanding of the way in which the operators use the procedures in the course of a sequence, and to assist in quantifying the failure probabilities of key actions. Measurements were made of the time taken to perform a number of actions outside the control room in order to ensure that proper credit was given for the operators performance following various system failures.

Plant-specific data were collected and analyzed for system maintenance outages and initiating event frequencies.

One very important feature of the methodology is that the entire Level 1 analysis including the internal flooding was performed on the NUPRA workstation (Fulford, 1989). The Level 2 analysis is captured on the NUCAP+ (Fulford and Sherry, 1991) and the NUREG-1150 EVNTRE software, which enabled the project team to develop a completely integrated model from initiating event to source term release, for use on a personal computer. This is essential for the establishment of a living PRA, i.e., one which can be easily maintained and modified as changes are made in design and operation of the plant. The workstation model is fully supported by a comprehensive set of analysis files (engineering calculations), which detail the assumptions and information sources used at each stage of model development.

A formal Project Plan was developed for the project to ensure the appropriate level of review and documentation. The work products were reviewed at each stage, both initially by project team members and externally by engineering and operations personnel at Perry. In addition an independent review was performed to ensure consistency within the overall methodology. All comments received have been addressed and retained within the appropriate analysis files.

An independent verification program as defined by our quality assurance program and engineering criterion for technical practices will be implemented to allow utilization of the PRA as an approved methodology for review and analysis by the Perry staff.

1.4 SUMMARY OF MAJOR FINDINGS

1.4.1 Summary of Core Damage Frequency for Internal Events

In order to differentiate between core damage and fission product release the various accident sequences were developed for two sets of bounding conditions. The first set of conditional event trees were developed to determine the frequency of accident sequences in which an initiating event and subsequent system failures would lead to core damage. Core damage is defined as failure to maintain the water level in the vessel above the Minimum Zero Injection Water Level or, in the case of ATWS, failure to maintain the maximum cladding temperatures below 2,200°F, with no possibility of recovery of injection in the short term. This set of event trees were defined as the core damage event trees.

A second set of plant damage state event trees were developed for the plant damage states, and in this case consideration was given to recovery of in-vessel cooling to prevent vessel failure, and to operator actions that affect the accident progression, containment loading, and fission product behavior. This second set of event trees gives the frequency of the contribution to the various fission product releases or source terms. In the case of the loss of offsite power and station blackout events these were developed to accurately model the various stages at which offsite power recovery would enable recovery of late injection either before vessel or containment failure.

The point estimate frequency of core damage from internal events excluding flooding is approximately 1.2×10^{-5} per reactor year. This was composed of 21 core damage frequencies with an annual frequency of greater than 10^{-7} and which contributed 86% of the overall core damage frequency, and an additional 89 sequences with a point estimate frequency of greater than 10^{-10}/yr which contributed the remaining 14%. The accident grouping by initiating event is shown in Table 1-1. The quantification is described in detail in section 3.4.1.1.

The cumulative distribution function for the core damage frequency is shown in Figure 1-1. The significant parameters of the uncertainty analysis frequency distribution function are as follows.

Mean	1.4×10^{-5}
Standard Deviation	3.9×10^{-5}
95th Percentile	2.5×10^{-5}
Median	1.1×10^{-5}
5th Percentile	6.2×10^{-6}

The dominant accident initiating event type is anticipated transient without scram (ATWS) at 40.7 percent. Transients contribute 25.0 percent, station blackout 19.3 percent, and loss of offsite power 12.4 percent. As a complete class, LOCAs contribute 2.6 percent. The individual sequences contributing to each initiator are shown in Table 1-2 and those contributing to 95 percent of the core damage frequency in Table 1-3.

An event importance analysis was performed on the overall core damage model. In this analysis the relative importance of each basic event was calculated with respect to three different measures. The three measures are Fussell-Vesely, risk reduction, and risk achievement.

The dominant basic events ranked in order by Fussell-Vesely and risk reduction measures are shown in Table 1-4. The Fussell-Vesely importance is a measure of the contribution of the given component to the overall core damage frequency by comparing the sum of all cutsets in which that basic event occurs with the sum of all cutsets. The risk reduction measure shows the ratio of the original core damage frequency to the reduced core damage frequency if the component was perfect or its failure probability is zero.

It should noted that the ranking of events by the Fussell-Vesely measure and the risk reduction measure are identical so the highest ranked items for these two measure are discussed in the following paragraphs.

The most important basic event for risk reduction is the mechanical failure of the control rods (CM) preventing insertion into the core given a signal to shut down the reactor. This is the single event following any transient initiator which will lead to the ATWS scenarios which in turn contribute 40.7 percent to the core damage frequency.

The second important basic event for risk reduction is the loss of offsite power (T1). This initiator leads to station blackout sequences (19.3 percent) and loss of offsite power sequences (12.4 percent). Core damage sequences following the loss of offsite power initiator thus contribute 31.7 percent to the core damage frequency. The Fussell-Vesely importance is 0.32 with a risk reduction of 1.47.

Failure of the operator to maintain availability of the power conversion system (PCS) (NSHICPEC5-2-L1T3) for an ATWS resulting from a transient with PCS initially available or loss of feedwater transient, is the third most important basic event. It has been assessed that the feedwater runback leading to loss of RPV level will result in closure of the MSIV in all cases. This basic event was set to 1.0 and occurs in the dominant sequences of the

The Fussell-Vesely importance is 0.27 with a risk reduction of 1.38. The initiating event transient with PCS available (T3A) is the fourth most important basic event. This event contributes to the dominant ATWS sequences and has a Fussell-Vesely importance of 0.25 with a risk reduction of 1.34.

Failure of the operator to re-open the motor feed pump control valves or manually depressurize the RPV (FWHICPEL-2-FDW-V) during an ATWS event following a transient with a loss of PCS is the fifth most important basic event with a Fussell-Vesely importance of 0.25 with a risk reduction of 1.33.

The initiating event transient with PCS not available (T2) is the sixth most important basic event. This event contributes to the dominant ATWS sequences and has a Fussell-Vesely importance of 0.23 with a risk reduction of 1.30.

The seventh important basic event is the failure of the operator to inhibit ADS (ADHICPC5-1-ADS-O) for ATWS scenarios where the feedwater system has failed. The Fussell-Vesely importance of this basic event is 0.22 with a risk reduction of 1.28. The failure to inhibit ADS in other sequences is modeled by different basic events as they have different values. The sensitivity of the results to this event is discussed in section 3.4.1.6.

The failure of the containment anchorage (CV05) is the eighth most important basic event. This is included in any sequence in which RPV injection is successful but long-term containment heat removal fails. The Fussell-Vesely importance is 0.15 with a risk reduction of 1.17.

Non-recovery of offsite power in 3 hours (R15) leads to sequences in LOOP and station blackout contributing 12.8 percent to the core damage frequency. The Fussell-Vesely importance is 0.13 with a risk reduction of 1.15.

The failure of the 4,160 VAC bus EH12 can fail all equipment which requires division 2 power. The Fussell-Vesely importance is 0.12 with a risk reduction of 1.13. Failure of this bus will prevent the opening of the valve in the vent path inside containment and therefore result directly in failure of the ability to vent containment. Failure of the division 1 power to the outboard valve can be overcome by manual opening of the valve, thus failure of division 1 power has a much lower importance value.

Of the next six basic events in importance, four are failures of either the Division 1 and 2 diesel generators to supply power or the non-recovery of offsite power. These basic events contribute to both LOOP and station blackout sequences.

Two of the next eight basic events are failures to provide alternate injection to the RPV via the fire protection system. The fire protection system may be used when other alternate injection systems are unavailable due to non-recovery of offsite power.

Similar information was generated for risk achievement worth. Risk achievement worth is derived by calculating the core damage frequency with a given event failure probability set equal to 1.0. This is equivalent to determining the core damage frequency if the component is failed at the time of the initiating event. The dominant basic events ranked in order by risk achievement worth are shown in Table 1-5.

The dominant basic event as measured by risk achievement is the mechanical failure of the control rods to insert into the core (CM). In this case, the probability that the control rods fail leading to an ATWS following all initiating events is 1.0 and shutdown is only achieved by use of the standby liquid control system.

The next basic event is the common cause failure of the ECCS pump room coolers (EPFACCEPRCS). These coolers provide heat removal from the pump rooms. The common cause failure was assessed to be low enough such that setting it to 1.0 would significantly increase its contribution to core damage frequency.

The third basic event in importance as ranked by risk achievement is the failure of the Division 2 4,160 VAC bus, EH12, (DGBALC1R22S0006). Failure of this bus would cause a loss of all equipment powered from Division 2 including low pressure coolant injection, heat removal equipment, and containment venting equipment.

Common cause failure of the batteries (DCBTCC) is the fourth most important basic event. As DC power is required for all systems and the batteries are required for any station blackout event, the common cause failure of the batteries is relatively important.

The next three basic events are common cause failures of diesel building ventilation fans, dampers and louvers (DBMFCC, DBMDCC, and DBLVCC). These components support the operation of the diesel generators and are required following a loss of offsite power.

The initiating event for a Large LOCA (A) is the eighth basic event in risk achievement importance.

Of the next ten ranked basic events, six are common cause basic events. Common cause is very high in risk achievement worth because the redundant components are failed at the same time. Setting the common cause basic events to 1.0 therefore significantly increases the core damage frequency.

1.4.1.1 Vulnerability Screening

A concise definition of vulnerability has not been given in either the documentation associated with the performance and reporting of the IPE or at a number of gatherings held to discuss the performance of the IPE. In the response to questions in Appendix C to the Submittal Guidance Document (NRC, 1989), mention is made of examining sequences that are above the screening criteria in order to determine if a weakness exists. Thus the word weakness replaces the word vulnerability but, neither is defined in numerical or comparative terms. In another response it is suggested that a vulnerability is an outlier. The NUMARC Severe Accident Issues Closure Guidelines (NUMARC, 1992) proposes a set of guidelines based on a combination of the core damage frequency for a group of sequences and the individual contribution from a sequence group. If the contribution from a given initiator or system failure is greater than 50 percent to the total core damage frequency it is interpreted as a significant vulnerability, if it contributes 20-50% it is interpreted as a potential vulnerability to be investigated. Similarly, contributions from sequence groups between a core damage frequency of 10^{-5} to 10^{-4} should be reviewed to determine if there is an effective plant procedure or hardware change which would reduce the frequency of the sequences.

The functional accident sequence groups, the definition of each group and the frequency of sequences in each group are shown in Table 1-6. It can be seen from this table that there are no significant vulnerabilities as defined in the previous section as all the accident sequence groups have a frequency below 10^{-5} and no group contributes more than 50% to the overall core damage frequency. However there are two groups of accident sequences that contribute between 20 and 50 percent. Group 4 which is made up of accident sequences involving anticipated transients without scram leading to core damage, containment failure and subsequent loss of inventory, and Group 2 which is made up of accident sequences involving loss of containment heat removal leading to containment failure and subsequent failure of coolant inventory make-up.

Group 4

The contribution to core damage from sequences in this group comes primarily from ATWS sequences in which the motor feed pump has failed to inject water and ADS has not been inhibited resulting in rapid depressurization of the RPV and injection of low pressure ECCS. This leads to a series of reactivity oscillations resulting in generation of large quantities of steam and ultimately containment failure and core damage. In these sequences, the potential vulnerability is the failure to inhibit ADS. However, the sensitivity analysis that uses the plant operating data for cycles 2 and 3 will reduce the frequency of the initiators which contribute to this group and consequently directly reduces the contribution to core damage frequency of these sequences from 34% to approximately 15%, which is no longer a potential vulnerability in terms of the NUMARC criteria.

Group 2

The contribution to core damage from sequences in this group comes from a failure of containment heat removal leading to containment failure and subsequent loss of injection. One of the reasons that this is a significant contributor is that the containment design at Perry is such that approximately 15% of containment failures lead to injection failure. One potential modification considered, a passive vent, if fitted would reduce the containment failure frequency, with the core damage frequency also being reduced. This would also have an impact on source term magnitude and is further discussed in the containment evaluation in section 1.4.3.2.

1.4.2 SUMMARY OF CORE DAMAGE FREQUENCY FROM INTERNAL FLOODING

The contribution to core damage from floods is 1.5×10^{-6} per reactor year which is approximately 12 percent of the overall core damage frequency. The contribution from each flood area is shown in Table 1-1 and Figure 1-4.

The most significant contribution is from floods in Zone 13 in the control complex at elevation 576 ft. The total contribution from floods in this area is 8.8×10^{-7} per year or 7 percent of the total core damage frequency which is approximately the same as the contribution from loss of instrument air.

In the event of a flood in this zone, the majority of the damage is done to the support systems (control complex chilled water, emergency closed cooling pumps, and instrument and service air).

The second highest contributor is flooding in Zone 17 in the control complex at the 599' level with a frequency of 3.2×10^{-7} per year (2.4%). Floods in this area will flow through doors and down to Zone 13, which then has the same impact on the support equipment as a flood in Zone 13. The results of the flooding analysis are discussed in section 3.3.7.

1.4.2.1 Flooding Core Damage Vulnerabilities

There are no vulnerabilities associated with internal flooding following the NUMARC definition. The total contribution to core damage frequency is 12 percent from all floods, and flooding in the most significant area, Zone 13, only contributes 7 percent to the overall core damage frequency. This sequence is based on conservative assumptions assuming the availability of injection systems and therefore the current procedures for maintaining vessel injection will ensure that optimum use will be made of the systems, and the potential core damage frequency minimized.

1.4.3 SUMMARY OF CONTAINMENT EVALUATION

The containment evaluation was conducted in two phases. In the first phase, the strength of the containment was determined and in the second, the accident progression and fission product release following core damage were evaluated.

The results of the containment analysis performed by the Perry plant architect that designed the containment indicate that the median pressure capacity is 64.3 psig compared with the design pressure of 15 psig. The potential failure modes under severe accident conditions will vary according to the accident progression, and may also affect the accident progression. The two failure modes of concern are penetration failures resulting in steam release from the shield building to the auxiliary building through the penetration seals, and anchorage failure which would result in loss of the suppression pool as the water is expelled to the shield building and adjacent buildings. The latter event would also result in drywell bypass. Additionally, anchorage rupture can result in a direct radiological release to the environment. Penetration failures would commence with a small leakage area and increase with containment pressure. Concrete and steel anchorage failure would result in gross failure of the containment vessel from the mat foundation.

The level one analysis identified a number of sequences in which gradual steam overpressurization would occur as the result of failure of containment heat removal, but successful injection of water to the vessel. In this case it is necessary to identify what proportion of the containment failure will lead to failure of the injection systems and therefore core damage in a failed containment. This analysis resulted in the identification of a function designated Cv in the level one analysis. The exact value of this function is dependent upon the injection system operating at the time of containment failure as well as the containment failure mode. The total contribution to core damage resulting from internal event sequences in which a gradual steam overpressurization results in containment failure is 2.6 x 10^{-6}/per year or 22 per cent of the core damage frequency. These sequences are included in Group 2 in Table 1-6. In the importance analysis in Table 3.4.1-7 it can be seen that the highest ranked function is CV05 (eighth) and that functions CV01 and CV03 are ranked 31st and 38th respectively. As discussed in section 1.4.1.1, a means of mitigating excessive containment pressure buildup such as the fitting of a passive vent, could reduce the frequency of containment failure and the frequency of failure of the injection system, and the frequency of core damage within a failed containment.

The level two analysis plant damage state grouping also shows that Critical (non-shutdown) ATWS results in a failed containment before core damage with a frequency of 5.6 x 10^{-7} which represents 4.4% of the total core damage frequency.

For those sequences where the containment is intact at the time of core damage, the progression of the accident was evaluated by use of the Modular Accident Analysis Program (MAAP) and the construction of an Accident Progression Event Tree in order to account for the uncertainties surrounding events during the course of the accident. Particular attention was paid to the generation of hydrogen and the impact of hydrogen burns, the potential for pedestal overpressurization, and pedestal core concrete interactions.

The plant damage state profile from the level 1/2 interface is: 0% containment bypass; 77% containment intact at core damage (9% - Station Blackout, and 68% - transients and other event types with AC power available), 23% - containment failed at core damage (4.4% - critical (not shutdown) ATWS sequences, 4.3% - Loss Of Offsite Power and Station Blackout, and 14% - other events).

The Accident Progression Event Tree summary results for containment performance are: no containment failure - represents a 39% conditional probability given core damage and a frequency of 5.0 x 10^{-6}, containment venting - represents a 29% conditional probability and a frequency of 3.7 x 10^{-6}, and containment structural failure - represents a 32% conditional probability and a frequency of 4.0 x 10^{-6}. The conditional probability estimates of the detailed containment failure modes evaluation are: 50.8% - in-vessel cooling and no RPV failure; and the balance of the sequences with RPV failure (49.2%) is composed of: 12.4% - no containment failure, 10% - venting with a damaged core; 7.4% - late containment failure; 3.4% - early containment failure with no pool bypass; and 16.1% containment failure with pool bypass. These results are shown in Table 1-7.

As in the case of the other BWR/6 plants with Mark III containments, the hydrogen ignitors are AC powered. Consequently Station Blackout sequences result in loss of the ignitors. However, as the total contribution to core damage from Station Blackout sequences is only 9% and the contribution to containment structure failure as a result of hydrogen burns is less than 5 percent, the addition of a backup power supply would not significantly impact the source term release given the other contributions to containment failure at the present time.

1.4.4 COMPARISON OF RESULTS WITH OTHER STUDIES

1.4.4.1 Core Damage Frequency

The major purpose of this study was to ensure that the PRA model developed by the project team was understood by the CEI staff at Perry and represented the as-built as-operated condition at Perry as far as possible. Guidance for performing the IPE indicated that heavy reliance could be placed on the results of previous studies for similar plants, in this case the Grand Gulf study performed as part of the NRC program in 1987 and reported in NUREG/CR-4550 and NUREG/CR-4551. However it was decided that if the completed PRA was to be used as a living PRA the success criteria and plant models would have to be developed for the as-built condition of the Perry plant and incorporate the latest understanding, as presented by the BWR Owners Group, of the required response by the operations to ATWS events, and the thermal hydraulic performance of the core when the water level is below the top of active fuel.

In addition, a plant-specific human reliability analysis was performed so that the impact of the current emergency operating procedures on the operator's response to the initiating events would be correctly incorporated in the event and fault trees.

It can be seen in Table 1-8 that as the result of incorporating the plant-specific insights into the models and developing a completely new set of event and fault trees, the core damage frequency from internal events at Perry is approximately 1.2 X 10^{-5} compared with 4.0 X 10^{-6} at Grand Gulf (NUREG/CR-4550). The design features are compared in Table 1-9. The results from the analysis of BWR/6s in Taiwan (Kuosheng) and Spain (Cofrentes) are also included in Table 1-8 for comparison purposes. It was not possible to compare the results with the IPEs for Grand Gulf, Clinton or River Bend as the IPE submissions were not available at the time of production of this report.

The only significant differences between Perry and NUREG/CR-4550 (Drouin, 1989) are in the results for anticipated transient without scram, transients without the power conversion system available and loss of instrument air. Some of the reasons for these differences are discussed in the following paragraphs.

It can be seen from the table that there is a wide variation in the assessment of core damage frequency following an ATWS in the four studies, ranging from less than 10^{-7} to 2.6 x 10^{-5}, a factor of over 250. The specific reasons for the differences between the NUREG/CR-4550 core damage frequency and the Perry frequency are the result of different success criteria being used in the two studies. In the Perry study the latest information from the BWR/6 Mark III Issues Committee of the BWR Owners Group that the High Pressure Core Spray can not be used to maintain vessel level has been incorporated. In the NUREG/CR-4550 study it was assumed that if HPCS started, successful injection would be achieved. When it is assumed that HPCS can not be used for high pressure injection the feedwater system becomes important. At Perry runback occurs following ATWS requiring the operator to take control to restore feedwater injection and maintain level. The net effect is to place dependence on the operator to achieve the required plant status. The quantification of the operator actions based on a detailed evaluation of the PEI combined with the system failures resulted in the ATWS core damage frequency of 4.74×10^{-6}.

The only other two initiators for which there is a significant difference between the Perry and NUREG/CR-4550 studies are the Transient with loss of PCS (1.7×10^{-6} at Perry compared with 1.3×10^{-8} in NUREG/CR-4550), and loss of Instrument Air (1.0×10^{-6} compared with less than 10^{-7}). In the case of loss of PCS the failure of recovery of PCS in the NUREG/CR-4550 study is two orders of magnitude lower than at Perry. This is not in line with the generally applied value which has been used in the Perry study. Similarly in the loss of instrument air it is considered that non-conservative recovery actions have been used for the NUREG/CR-4550 study.

1.4.4.2 Containment Evaluation

The Perry IPE level 2 containment analysis uses the same Event Progression Analysis (EVNTRE) code applied in the Grand Gulf NUREG/CR-4551 study (Brown, 1990) to transpose the severe accident analysis phenomenological framework of the template Mark III reference study and to model the many dependencies associated with containment loading mechanisms such as steam generation, hydrogen generation and combustion, and the subsequent variations in pressure and temperature within the containment. The Perry IPE Accident Progression Event Tree consists of 68 questions which address the four general time frames of accident progression: initial, early, intermediate and late. The more extensive NUREG/CR-4551 Grand Gulf evaluation consisted of 125 questions.

To evaluate the impact of interruption of power to the hydrogen ignitors, the Perry APET addresses the hydrogen combustion phenomena in a similar manner to the NUREG/CR-4551 Grand Gulf APET. The Perry APET models the recovery of offsite power before RPV failure as well as after RPV failure, and possible variations in hydrogen concentration at the time of power restoration are evaluated including the possibility of detonable concentrations.

The Perry IPE use of the same event progression analysis code as the NUREG/CR-4551 template study enables a good transfer of phenomenological modeling as well as of the quantification bases. The Perry APET routinely references the NUREG/CR-4551 template study and transfers many of the event models, such as the bounding model of alpha mode steam explosions. However, the primary reference for modeling phenomenological parameters is the EPRI maintained MAAP code 3.0B.

One parameter was found to be important when performing the sensitivity analysis of the Accident Progression Event Tree. That was large in-vessel steam explosions leading to bottom head failure. These phenomena are not modeled in the MAAP code nor addressed in the EPRI Recommended Sensitivity Analyses discussed above in section 7.2. The probabilities applied in the NUREG/CR-4551 template plant study appear to over estimate the risk and are reduced by a factor of 10 in the base case. However, if the NUREG/CR-4551 values for large in-vessel steam explosion failure are used, the base case all sequence estimate of 51% for in-vessel cooling and no RPV failure is reduced to 21%, a 58% reduction.

2.0×10^{-6} to 4.2×10^{-7}.

The contribution from individual maintenance activities is not large and contribution to planned maintenance for the HPCS and RCIC has already been reduced. The maintenance activities for the HPCS and RCIC were revised as a result of the schedule optimization prior to the third operating cycle. There is a more extensive Systematic Maintenance Optimization (SMO) in progress which will result in the optimization of maintenance for all ECCS systems. Currently preventive maintenance of the HPCS system is only performed during an outage. Thus, this component of the maintenance outage is now zero.

It is considered, in view of the low frequency of the contribution to core damage of the failure of the decay heat function, that no immediate action is required to modify the plant design and that this analysis satisfies the requirements for resolution of unresolved safety issue A-45. The ongoing systematic maintenance optimization program will ensure that the reliability of the ECCS system will be maintained at least above that assumed in the performance of the IPE.

1.4.6 POTENTIAL IMPROVEMENTS

The A-45 evaluation discussed in the previous section and the vulnerability analysis described in section 3.4.2 shows that there are no vulnerabilities associated with core damage. However two items are discussed which would lead to a reduction in the core damage frequency. The improvements and impact on core damage frequency are summarized in Table 1-10.

Similarly the containment analysis discussed in section 4 identified the contributors to containment failure and a number of changes which would reduce the likelihood of containment failures, given core damage. The impact on containment failure frequency is shown in Table 1-7.

The potential plant improvements considered are discussed in the following sections.

1.4.6.1 Potential Plant Improvements to Prevent Core Damage

Passive Containment Vent Path

One of the sensitivity analyses performed was on the impact of containment failure to loss of RPV injection and subsequent core damage. The addition of means of mitigating slow excessive containment pressure buildup, such as a passive containment vent path that does not depend on AC power, would reduce the core damage frequency for the non-ATWS sequences in which loss of decay heat removal leads to containment failure and results in an overall reduction of approximately 19% from approximately 1.3×10^{-5} to 1.1×10^{-5} (includes flooding events).

Automatic ADS Inhibit for ATWS

One of the contributors to core damage frequency for ATWS is manually inhibiting ADS. By installing an automatic inhibit of ADS, those ATWS sequences in which manual inhibit fails would drop out. The overall core damage frequency is reduced by 21 percent from approximately 1.3×10^{-5} to 1×10^{-5}. As the sequences resulting from this failure result in an uncontrolled flow to the vessel from the low pressure injection systems with subsequent core damage and containment failure, the installation of the auto inhibit would reduce the frequency of this set of sequences.

Maintenance

Performing maintenance on systems and components is important from two aspects. First, maintenance is needed to provide assurance that the systems can perform their function when called upon. Second, because systems are out of service while maintenance is being performed, maintenance can also have a detrimental impact on core damage frequency. Current practice is such that HPCS preventive maintenance is now only scheduled during plant outage. A Systematic Maintenance Optimization (SMO) for all ECCS systems is now in progress, and it is expected that this will result in changes in planned maintenance activities which will reduce the core damage frequency.

1.4.6.2 Design Consideration for Reduction in Containment Failure

The containment bypass as the result of an unisolated breach of the primary system outside the containment was determined to be less than 10^{-8} as reported in the Grand Gulf study and therefore does not require any action at Perry.

Supplement No. 3 of Generic Letter 88-20 identified that Mark III containment owners were expected to evaluate the vulnerability to interruption of power to the hydrogen ignitors. The modification of the electrical supply to the hydrogen ignitors to ensure availability during station blackout would remove the possibility of high containment loads from hydrogen deflagrations and detonations. The overall impact of this change on the base case assessment is very minor: 1) the containment failure with early/late pool bypass frequency changes from 2.04×10^{-6} to 2.00×10^{-6} (-2% change), 2) the containment structural failure frequency changes from 4.03×10^{-6} to 3.76×10^{-6} (-6.7% change). Thus, a hardware upgrade to provide uninterrupted electrical supply to the hydrogen ignitors is not warranted.

In the base case analysis the frequency of RPV Failure and Early Containment Failure with Pool Bypass is 2.0×10^{-6} or 16 per cent of the core damage frequency. In section 1.4.6.1 two modifications were identified which would lower the core damage frequency. The implementation of the two modifications would also reduce the frequency of core damage with early containment failure. The implementation of a passive vent would reduce the contribution to RPV failure and early containment failure with pool bypass from 2.0×10^{-6} to 4.5×10^{-7} which is a 78% reduction. The inclusion of an automatic ADS inhibit in the event of an ATWS would reduce the RPV failure and early containment failure with pool bypass from 2.0×10^{-6} to 1.8×10^{-6}, an approximately 15% reduction. The combined effect of performing both modifications would be to reduce the contribution to RPV failure and early containment failure with pool bypass from 2.0×10^{-6} to 1.6×10^{-7} a 92% reduction.

It should also be noted that if frequencies of scrams at Perry based on operating cycles 2 and 3 is used, the contribution to RPV failure and early

The results of the Perry IPE containment performance analysis cannot be directly compared to the NUREG/CR-4551 Grand Gulf study, due to differences in containment failure modes, plant damage state group frequencies, and phenomenological modeling assumptions. The NUREG/CR-4551 study was dominated by Station Blackout (97% plant damage state frequency) and determined that hydrogen combustion was the dominate cause of containment failure. The Perry IPE is dominated by shutdown ATWS sequences (ATWS with successful SLC) and other transients with Station Blackout accounting for 9% of the plant damage state frequency. The modeling for debris cooled in-vessel is similar, with the one variance noted above regarding the estimated value for large in-vessel steam explosions. When the Perry APET event for large in-vessel steam explosion is set to the NUREG/CR-4551 value, the estimated results for No RPV Failure compare closely, with 21% for the Perry APET and 18% for the Grand Gulf APET. The Perry APET estimates for the conditional probability of Containment Performance are shown below with the NUREG/CR-4551 Grand Gulf APET estimates:

	Perry	NUREG/CR-4551 Grand Gulf
No RPV Failure	51%	18%
RPV Failure and No Containment Failure	12%	5%
RPV Failure and Venting	10%	4%
RPV Failure and Late Containment Failure	7%	28%
RPV Failure and Early Containment Failure With No Pool Bypass	3%	22%
RPV Failure and Early Containment Failure With Pool Bypass	16%	21%

1.4.5 DECAY HEAT REMOVAL EVALUATION - ISSUE A-45

The objectives of Task Action Plan A-45 are to evaluate the safety adequacy of Decay Heat Removal (DHR) Systems in existing light water reactor nuclear power plants and to assess the value and impact (benefit/cost) of alternative measures for improving the overall reliability of the DHR function if required.

Some potential accidents which could result in core melt were excluded from the A-45 studies performed by Sandia National Laboratories (S.W. Hatch, 1987). Since the purpose of the program is to study the adequacy of shutdown decay heat removal systems, large LOCAs, reactor vessel ruptures, interfacing system LOCAs and anticipated transients without scram (ATWS) are excluded. The study focused on events occurring from power or in hot standby.

The delineation of the accident sequences, system analysis, core damage and sequence quantification are fully described in sections 3.1 through 3.4.1. The concern in issue A-45 is to identify the specific vulnerabilities associated with sequences identified as potentially leading to core damage if all injection to the vessel and decay heat removal are lost.

The breakdown of functional contribution to core damage is shown in Table 1-6. The three groups which result in failure of decay heat removal as the result of failure of injection contribute 22 percent to core damage, with individual contributions as follows:

Group	Description	Percentage
1B	Loss of offsite power and make-up	13
1E	Loss of coolant inventory make-up at low pressure	8
1A	Loss of high pressure make-up and failure to depressurize	<1

The contribution to core damage frequencies as the result of loss of decay heat removal from containment leading to containment failure and consequential loss of injection (class 2) is 22 percent of the overall core damage frequency. Thus 44 percent of the contribution to core damage is the direct result of failure of decay heat removal either from the vessel or containment. Although the overall frequency is not high (5.1×10^{-6}) it is of interest to identify any potential improvements which, if they could be addressed cost effectively, would reduce the contribution of decay heat removal failures to the overall core damage frequency.

The importance analysis discussed in section 1.4.1 and listed in Table 1-4 identified the individual components whose improvement would make the greatest contribution to the reduction in core damage frequency. The highest ranked component in terms of decay heat removal is the event representing failure of injection as the result of containment failure (CV05). This indicates that the mode of containment failure following overpressurization is more significant than any individual containment heat removal system failure. Therefore an alternative means to prevent containment failure will have the largest impact in reducing the core damage frequency.

The contributions from the remaining individual components is small. Therefore the improvement of an individual component will not make a significant difference to the core damage frequency.

1.4.5.1 Summary of A-45 Evaluation

The core damage frequency resulting from failure of decay heat removal systems is 5.1×10^{-6} per year representing 44 percent of the contribution to core damage from internal events. This frequency is below 10^{-5} and therefore not a significant vulnerability in frequency terms. However a review of the contributions to this failure identified one event which contributed more than any other to the core damage frequency and for which a possible modification could be considered. The mode of containment failure results in a significant probability of injection failure following containment failure and subsequent core melt in a failed containment, with consequent increase in offsite release. The addition of means of mitigating slow excessive containment pressure buildup, such as a passive containment vent path, would reduce the frequency of core damage and result in a reduction in offsite releases. The overall core damage frequency, excluding flooding, would be reduced from approximately 1.2×10^{-5} to 9.0×10^{-6} and the frequency of core damage sequences with containment failed at core damage is reduced from containment failure with pool bypass is reduced from 2.0×10^{-6} to 1.0×10^{-6} and with the fitting of the passive vent to 2.0×10^{-7}.

There results are summarized in Tables 1-7 and 1-10.

1.4.7 CONCLUSIONS

The performance of the level 2 PRA in response to the NRC's request in Generic Letter 88-20 for an Individual Plant Examination of the Perry plant has resulted in the CEI gaining a number of insights into the contribution to risk at the plant. The outcome was the performance of the following improvements during the course of the study.

Loss of Offsite Power Instructions

- Retention of RCIC isolation bypass for high steam tunnel temperature
- Enhanced process for crosstieing Unit 1 and 2 batteries
- Enhanced process for offsite power recovery to HPCS and alternate injection system bus bars.

Flooding Instructions

- Enhanced alarm response instructions to flooding scenarios.

In addition the following improvements are expected to be made in the near future.

- Implementation of automatic RPV depressurization for non-ATWS events (requires industry Emergency Procedures Group and UNSRC reviews and approvals prior to design and implementation)
- "Fast Firewater" tie between Fire Protection and HPCS
- Permanent Division 3 to Division 2 "quick" connect
- Reduction of Out of Service Time for certain critical components (already achieved for HPCS and RCIC)

The result of the improvements is an overall core damage frequency for the internal and internal flooding events of 1.3×10^{-5} per year and an RPV failure with early containment failure with pool bypass frequency of 2.0×10^{-6} per year. In the event automatic RPV depressurization is not implemented, core dmage frequency will not increase as plant specific operating data when incorporated will more than compensate. Beyond the base case, a number of additional enhancements discussed in the previous section have been identified and are being evaluated further. However, careful analysis is required before any further improvements beyond those identified above are made.

When performing a major analysis of this type, it is necessary to fix the date for collection of design and operational information, in this case January 1, 1990. At this time Perry had only completed one full cycle of operation and therefore little or no operational data was available. Since that time, it has been possible to include data for maintenance outages on a small number of components. It is clear from the experience in operating cycles 2 and 3 that the initiating event frequencies are significantly lower than the generic values used in the NRC Grand Gulf study and therefore in the

Perry study. Therefore, before any further decisions are made concerning plant improvements the first step will be to update the living PRA to include this data, and any design changes made since the freeze date of January 1, 1990. In the case of the latter a brief review of the work performed during the first two cycles indicate that there have been no major design changes to the ECCS.

The enhancements discussed earlier improve the decay heat removal capabilities following an initiating event. It is considered that the current core damage frequency, as the result of decay heat removal failures, is within the current guidelines and therefore the results of this study represent satisfactory resolution of Unresolved Safety Issue A-45 for the Perry Nuclear Power Plant.

There were no specific vulnerabilities identified with regard to containment performance in the Perry IPE. The Perry backend containment analysis indicates that the Perry containment response to severe core damage accidents is generally similar to that for other BWR/6 Mark III plants [e.g., NUREG-1150 Grand Gulf (NRC, 1989a)].

The containment performance summary results for the Perry Mark III containment were: no containment failure - represents a 39% conditional probability given core damage and a frequency of 5.0×10^{-6}, containment venting - represents a 29% conditional probability and a frequency of 3.7×10^{-6}, and containment structural failure - represents a 32% conditional probability and a frequency of 4.0×10^{-6}. The conditional probability of RPV failure and early containment failure with pool bypass given core damage was estimated to be 0.16 in the Perry IPE (compared to 0.21 for Grand Gulf in NUREG-1150). The differences in these results between the Perry IPE and NUREG-1150 mainly result from significant differences in the type of sequence contributing to core damage, from different containment failure modes, and from the phenomenological assumption made in the Perry IPE regarding the probability of steam explosions failing the lower RPV head.

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Table 1-1

Summary of Core Damage Frequency by Initiating Event & Flood Zone

	Core Damage Freq	Percent of CDF	
Loss of Offsite Power			
T1	1.80×10^{-7}	1.5	(Loss of Offsite Power)
R	7.19×10^{-7}	6.2	(Loss of Offsite Power and No Offsite Power Recovery at 3 hrs)
U	4.16×10^{-7}	3.6	(Loss of Offsite Power w/no HPCS or RCIC
T1P1	7.62×10^{-8}	0.7	(Loss of Offsite Power and 1 SORV)
T1P1U	1.53×10^{-8}	0.1	(Loss of Offsite Power and 1 SORV w/no HPCS or RCIC)
T1P2	3.89×10^{-8}	0.3	(Loss of Offsite Power and 2 SORVs)
Total	1.44×10^{-6}	12.4	
Station Blackout			
B	2.11×10^{-6}	18.1	(Station blackout)
BP1	8.36×10^{-8}	0.7	(Station blackout 1 SORV)
BP2	5.91×10^{-8}	0.5	(Station blackout and 2 SORVs)
Total	2.25×10^{-6}	19.3	
Transients			
T3A	$< 10^{-8}$	0.0	(Transient w/ PCS)
T3AP1	$< 10^{-8}$	0.0	
T3AP2	$< 10^{-8}$	0.1	
T3B	$< 10^{-8}$	0.0	(Loss of feedwater)
T3C	1.38×10^{-7}	1.2	(Inadvertent open SRV)
T2	1.64×10^{-6}	14.1	(Transient w/o PCS)
T2P1	2.47×10^{-8}	0.2	
T2P2	$< 10^{-8}$	0.0	
TIA	1.01×10^{-6}	8.7	(Loss of instrument air)
TIAP1	$< 10^{-8}$	0.1	
TIAP2	$< 10^{-8}$	0.0	
TSW	6.68×10^{-8}	0.6	(Loss of service water)
TSWP1	$< 10^{-8}$	0.0	
TSWP2	$< 10^{-8}$	0.0	
Total	2.90×10^{-6}	25.0	

Table 1-1 continued

	Core Damage Freq	Percent of CDF	
LOCAs			
A	2.11×10^{-7}	1.8	(Large LOCA)
S1	6.18×10^{-8}	0.5	(Intermediate LOCA)
S2	3.34×10^{-8}	0.3	(Small LOCA)
Total	3.06×10^{-7}	2.6	
ATWS			
T1-C	3.61×10^{-8}	0.3	
T3A-C	$< 10^{-8}$	0.1	
T3B-C	5.42×10^{-7}	4.6	
[illegible]	9.38×10^{-8}	0.8	
[illegible]	4.02×10^{-6}	34.5	
TIAC	4.33×10^{-8}	0.4	
Total	4.74×10^{-6}	40.7	

Total Core Damage Frequency (internal initiators) 1.17×10^{-5} (88%)

Flooding		
Zone 13	8.84×10^{-7}	57
Zone 17	3.22×10^{-7}	21
TPC	8.70×10^{-8}	6
Zone 1	1.9[illegible] $\times 10^{-7}$	12
Zone 1A	$< 10^{-8}$	< 1
Zone 8	2.80×10^{-8}	2
Zone 16	1.10×10^{-8}	1

Total Core Damage Frequency (flooding) 1.54×10^{-6} (12%)

Total Core Damage Frequency (internal initiators & flooding) 1.32×10^{-5}

Table 1-2

<u>Sequence Core Damage Frequencies Grouped by Initiator</u>

T2-C	Sum = 4.02E-006	34.5%	
T2-CS30	2.27E-006	19.5%	T2-C-U3-X'
T2-CS20	6.25E-007	5.4%	T2-C-Lc-C1
T2-CS28	3.12E-007	2.7%	T2-C-U3-X
T2-CS11	2.90E-007	2.5%	T2-C-C1
T2-CS12	2.37E-007	2.0%	T2-C-X'
T2-CS06	1.58E-007	1.4%	T2-C-V
T2-CS05	1.25E-007	1.1%	T2-C-V'
B	Sum = 2.11E-006	18.1%	
BS24	7.71E-007	6.6%	B-U1-Va-R
BS34	5.26E-007	4.5%	B-U1-U2-R-Va1
BS17	3.36E-007	2.9%	B-U1-R
BS07	1.60E-007	1.4%	B-R-Y-Cv
BS12	1.04E-007	0.9%	B-HI-R
BS22	5.96E-008	0.5%	B-U1-Va-V
BS35	5.15E-008	0.4%	B-U1-U2-R-X
BS33	5.11E-008	0.4%	B-U1-U2-R-Y-Cv
BS29	3.37E-008	0.3%	B-U1-U2-Va1
BS30	1.81E-008	0.2%	B-U1-U2-X
T2	Sum = 1.64E-006	14.1%	
T2S04	1.62E-006	13.9%	T2-W-Y-Cv
T2S09	1.90E-008	0.2%	T2-U3-U2-W-Y-Cv
T2S18	9.56E-009	0.1%	T2-U3-U2-U1-V-Va
TIA	Sum = 1.01E-006	8.7%	
TIAS14	7.53E-007	6.5%	TIA-U2-U1-V-Va
TIAS05	2.57E-007	2.2%	TIA-U2-W-Y-Cv
R	Sum = 7.19E-007	6.2%	
RS20	6.04E-007	5.2%	R-Ws-V-Va
RS19	8.73E-008	0.7%	R-Ws-V-Cv
RS10	2.75E-008	0.2%	R-Ws-W-Y-Cv
T3B-C	Sum = 5.42E-007	4.6%	
T3B-CS29	2.76E-007	2.4%	T3B-C-U3-X'
T3B-CS19	7.60E-008	0.7%	T3B-C-Q-Lc-C1
T3B-CS09	5.32E-008	0.5%	T3B-C-Q-X
T3B-CS27	3.80E-008	0.3%	T3B-C-U3-X
T3B-CS10	3.45E-008	0.3%	T3B-C-Q-C1
T3B-CS11	2.88E-008	0.2%	T3B-C-Q-X'
T3B-CS08	1.78E-008	0.2%	T3B-C-Q-V
T3B-CS07	1.52E-008	0.1%	T3B-C-Q-V'
U	Sum = 4.14E-007	3.6%	
US29	3.34E-007	2.9%	U-R1-V-Va
US12	5.99E-008	0.5%	U-V-Va
US28	1.63E-008	0.1%	U-R1-V-W-Y-Cv

Table 1-2 continued

A Sum =	2.11E-007	1.8%	
AS09	2.10E-007	1.8%	A-U1-V
T1 Sum =	1.80E-007	1.5%	
T1S08	1.47E-007	1.3%	T1-R2-W-Y-Cv
T1S04	1.94E-008	0.2%	T1-W-Y-Cv
T1S33	1.33E-008	0.1%	T1-U1-Ws-V-Va
T3C Sum =	1.38E-007	1.2%	
T3CS04	1.38E-007	1.2%	T3C-W-Y-Cv
T3C-C Sum =	9.38E-008	0.8%	
T3C-CS27	5.49E-008	0.5%	T3C-C-U3-X'
T3C-CS17	1.40E-008	0.1%	T3C-C-Lc-C1
T3C-CS07	9.81E-009	0.1%	T3C-C-X
BP1 Sum =	8.36E-008	0.7%	
BP1S27	4.85E-008	0.4%	BP1-U1-U2
BP1S17	1.69E-008	0.1%	BP1-U1-V
BP1S26	1.13E-008	0.1%	BP1-U1-Va-R
T1P1 Sum =	7.62E-008	0.7%	
T1P1S31	7.24E-008	0.6%	T1P1-U1-Ws-V
TSW Sum =	6.68E-008	0.6%	
TSWS10	5.49E-008	0.5%	TSW-U2-U1-V
TSWS14	1.00E-008	0.1%	TSW-C
S1 Sum =	6.18E-008	0.5%	
S1S13	5.85E-008	0.5%	S1-U1-V-Va
BP2 Sum =	5.91E-008	0.5%	
BP2S13	5.91E-008	0.5%	BP2-U1
TIAC Sum =	4.33E-008	0.4%	
TIACS09	3.31E-008	0.3%	TIAC-X'
T1P2 Sum =	3.89E-008	0.3%	
T1P2S11	2.39E-008	0.2%	T1P2-U1-V
T1P2S04	1.49E-008	0.1%	T1P2-W-Y-Cv
T1-C Sum =	3.61E-008	0.3%	
T1-CS09	2.19E-008	0.2%	T1-C-X'
T1-CS08	8.44E-009	0.1%	T1-C-C1
S2 Sum =	3.24E-008	0.3%	
S2S20	3.00E-008	0.3%	S2-C
T2P1 Sum =	2.47E-008	0.2%	
T2P1S04	2.46E-008	0.2%	T2P1-W-Y-Cv
T1P1U Sum =	1.53E-008	0.1%	
T1P1US22	1.37E-008	0.1%	T1P1U-R1-V

Table 1 3

Sequence within the Upper 95% of Total Core Damage

T2-CS30	2.27E-006	19.5%	T2-C-U3-X'
T2S04	1.62E-006	13.9%	T2-W-Y-Cv
BS24	7.71E-007	6.6%	B-U1-Va-R
TIAS14	7.53E-007	6.5%	TIA-U2-U1-V-Va
T2-CS20	6.25E-007	5.4%	T3-C-Lc-C1
RS20	6.04E-007	5.2%	R-Ws-V-Va
BS34	5.26E-007	4.5%	B-U1-U2-R-Va1
BS17	3.36E-007	2.9%	B-U1-R
US29	3.34E-007	2.9%	U-R1-V-Va
T2-CS28	3.12E-007	2.7%	T2-C-U3-X
T2-CS11	2.90E-007	2.5%	T2-C-C1
T3B-CS29	2.76E-007	2.4%	T3B-C-U3-X'
TIAS05	2.57E-007	2.2%	TIA-U2-W-Y-Cv
T2-CS12	2.37E-007	2.0%	T2-C-X'
AS09	2.10E-007	1.8%	A-U1-V
BS07	1.60E-007	1.4%	B-R-Y-Cv
T2-CS06	1.58E-007	1.4%	T2-C-V
T1S08	1.47E-007	1.3%	T1-R2-W-Y-Cv
T3CS04	1.38E-007	1.2%	T3C-W-Y-Cv
T2-CS05	1.25E-007	1.1%	T2-C-V'
BS12	1.04E-007	0.9%	B-HI-R
RS19	8.73E-008	0.7%	R-Ws-V-Cv
T3B-CS19	7.60E-008	0.7%	T3B-C-Q-Lc-C1
T1P1S31	7.24E-008	0.6%	T1P1-U1-Ws-V
US12	5.99E-008	0.5%	U-V-Va
BS22	5.96E-008	0.5%	B-U1-Va-V
BP2S13	5.91E-008	0.5%	BP2-U1
S1S13	5.85E-008	0.5%	S1-U1-V-Va
TSWS10	5.49E-008	0.5%	TSW-U2-U1-V
T3C-CS27	5.49E-008	0.5%	T3C-C-U3-X'
T3B-CS09	5.32E-008	0.5%	T3B-C-Q-X
BS35	5.15E-008	0.4%	B-U1-U2-R-X
BS33	5.11E-008	0.4%	B-U1-U2-R-Y-Cv
BP1S27	4.85E-008	0.4%	BP1-U1-U2
T3B-CS27	3.80E-008	0.3%	T3B-C-U3-X
T3B-CS10	3.45E-008	0.3%	T3B-C-Q-C1
BS29	3.37E-008	0.3%	B-U1-U2-Va1
TIACS09	3.31E-008	0.3%	TIAC-X'
S2S20	3.00E-008	0.3%	S2-C
T3B-CS11	2.88E-008	0.2%	T3B-C-Q-X'
RS10	2.75E-008	0.2%	R-Ws-W-Y-Cv

Table 1-4

Importance Ranking of Basic Events (Fussell-Vesely)

Rank	Event Name	Point Est	F-V Imp	Risk Ach	Risk Red
1	CM	1.000E-005	4.110E-001	41100.41	1.698
2	T1	6.090E-002	3.202E-001	5.94	1.471
3	NSHICPEC5-2-L1T3	1.000E+000	2.735E-001	1.00	1.377
4	T3A	4.510E+000	2.530E-001	0.80	1.339
5	FWHICPEL-2-FDW-V	5.003E-003	2.472E-001	50.17	1.328
6	T2	1.620E+000	2.314E-001	0.91	1.301
7	ADHICPC5-1-ADS-O	7.200E+000	2.165E-001	0.81	1.276
8	CV05	1.400E-001	1.468E-001	1.90	1.172
9	R15	8.230E-002	1.279E-001	2.43	1.147
10	DGBALC1R22S0006	9.600E-005	1.171E-001	1220.42	1.133
11	DHDGFS1E22S0001	3.000E-002	1.164E-001	4.76	1.132
12	U207B	8.752E-001	1.028E-001	1.01	1.115
13	FPOFFSITEPUMPER	6.000E-001	1.003E-001	1.07	1.112
14	DGDGFS1R43S0001B	3.000E-002	9.750E-002	4.15	1.108
15	T1A	9.200E-002	9.134E-002	1.90	1.101
16	DGDGFS1R43S0001A	3.000E-002	8.504E-002	3.75	1.093
17	R36	5.290E-001	8.427E-002	1.08	1.092
18	FPDPFR0P54C0001	1.745E-001	7.890E-002	1.37	1.086
19	LCLCUMA	1.930E-002	7.724E-002	4.92	1.084
20	U202B	7.823E-001	6.915E-002	1.02	1.074
21	LCLCUMRHRALPRC	9.220E-003	6.623E-002	8.12	1.071
22	FWHICPSN27-4:1IA	1.196E-001	6.408E-002	1.47	1.068
23	SPHICPPS4:5SPCU	1.000E+000	6.344E-002	1.00	1.068
24	FWHICPEC5-3:2	1.000E-002	6.151E-002	7.09	1.066
25	SLHICPEQ-6-SLCX	1.000E+000	6.128E-002	1.00	1.065
26	MESC133	7.760E-003	5.075E-002	7.49	1.053
27	T3B	7.600E-001	4.707E-002	1.01	1.049
28	MESA133	7.760E-003	4.603E-002	6.89	1.048
29	MESB133	7.760E-003	4.306E-002	6.61	1.046
30	R39	1.230E-002	4.258E-002	4.42	1.044
31	CV01	4.300E-001	4.132E-002	1.05	1.043
32	FPHICPPS4:2RCIC1	2.998E-001	3.813E-002	1.09	1.040
33	DGDGFR1R43S0001B	7.864E-003	3.627E-002	5.58	1.038
34	ESMPCC	3.417E-004	3.505E-002	103.55	1.036
35	CVHICPEPC-FPCC	1.000E-001	3.482E-002	1.31	1.036
36	CVHICPEPC-COM	1.001E-003	3.482E-002	35.76	1.036
37	FPHICPPS4:2RCIC4	1.000E-001	3.424E-002	1.31	1.035
38	CV03	4.000E-002	3.337E-002	1.80	1.035
39	ECECUMA	1.980E-002	3.322E-002	2.64	1.034
40	LCLCUMLPCIBLPCIC	1.770E-002	3.199E-002	2.78	1.033
41	ESESUMA	1.890E-002	3.131E-002	2.63	1.032
42	VA04B	7.327E-001	3.056E-002	1.01	1.032
43	ECECUMB	1.980E-002	3.023E-002	2.50	1.031
44	ESESUMB	1.890E-002	2.929E-002	2.52	1.030
45	FPHICPPS4:2RCIC2	1.000E-001	2.927E-002	1.26	1.030
46	DGDGUM1R43S0001B	3.080E-002	2.900E-002	1.91	1.030
47	CVHICPEPC-RHR	1.000E-002	2.879E-002	3.85	1.030
48	RCHICPS51-LDTRIP	4.999E-002	2.859E-002	1.54	1.029

Table 1-5

Importance Ranking of Basic Events (Risk Achievement Worth)

Rank	Event Name	Point Est	F-V Imp	Risk Ach	Risk Red
1	CM	1.000E-005	4.110E-001	41100.41	[illegible]
2	EPFACCEPRCS	3.750E-006	6.664E-003	1778.20	[illegible]
3	DGBALC1R22S0006	9.600E-005	1.171E-001	1220.42	[illegible]
4	DCBTCC	1.370E-005	5.433E-003	397.55	1.005
5	DBMFCC	2.930E-005	8.087E-003	277.00	1.008
6	DBNDCC	2.930E-005	8.087E-003	277.00	1.008
7	DBLVCC	2.930E-005	8.087E-003	277.00	1.008
8	A	1.000E-004	1.814E-002	182.34	1.018
9	ESMPCC	3.417E-004	3.505E-002	103.55	1.036
10	ESMVCC	9.250E-005	7.284E-003	79.73	1.007
11	ADSRCCADS	8.000E-006	5.934E-004	75.18	1.001
12	ECMPCC	1.190E-005	7.606E-004	64.92	1.001
13	DGBALC1R22S0007	9.600E-005	5.953E-003	63.01	1.006
14	FWHICPEL-2-FDW-V	5.003E-003	2.472E-001	50.17	1.328
15	ECMVCC	9.250E-005	4.238E-003	46.81	1.004
16	CVHICPEPC-COM	1.001E-003	3.482E-002	35.76	1.036
17	DGDGCC	3.675E-004	8.743E-003	24.78	1.009
18	S1	3.000E-004	5.280E-003	18.60	1.005
19	DCBTLC1R42S0002	1.367E-003	1.071E-002	8.83	1.011
20	CVMVNO1G41F0140	2.930E-003	2.212E-002	8.53	1.023
21	SCMVNC1E12F0048A	2.930E-003	2.210E-002	8.52	1.023
22	LCLCUMRHRALPRC	9.220E-003	6.623E-002	8.12	1.071
23	MESC133	7.760E-003	5.075E-002	7.49	1.053
24	SLHICPEQ-6-SLC1	1.251E-003	7.903E-003	7.31	1.008
25	HIHICPEC5-5-CRIT	2.001E-003	1.259E-002	7.28	1.013
26	SLEVCC	2.930E-004	1.826E-003	7.23	1.002
27	SLMPCC	2.930E-004	1.826E-003	7.23	1.002
28	ADHICPC5-1-ADS-A	3.793E-003	2.346E-002	7.16	1.024
29	HIHICPEC5-3:2-F	1.001E-003	6.127E-003	7.12	1.006
30	FWHICPEC5-3:2	1.000E-002	6.151E-002	7.09	1.066
31	SLMVCC1G33	9.250E-005	5.577E-004	7.03	1.001
32	SLMVCC	9.250E-005	5.577E-004	7.03	1.001
33	SLCVNO1C41F0007	1.000E-004	6.028E-004	7.03	1.001
34	SLCVNO1C41F0006	1.000E-004	6.028E-004	7.03	1.001
35	SLXVPL1C41F0036	4.499E-005	2.659E-004	6.91	1.000
36	MESA133	7.760E-003	4.603E-002	6.89	1.048
37	TSW	1.000E-003	5.731E-003	6.73	1.006
38	LCMPCC	2.930E-004	1.677E-003	6.72	1.002
39	P2	1.600E-003	9.113E-003	6.69	1.009
40	MESB133	7.760E-003	4.386E-002	6.61	1.046
41	ECMPFS1P42C0001A	2.930E-003	1.464E-002	5.98	1.015
42	HPMVNO1E22F0015	2.930E-003	1.454E-002	5.95	1.015
43	HPMVNO1E22F0004	2.930E-003	1.453E-002	5.95	1.015
44	HPMPFS1E22C0001	2.930E-003	1.453E-002	5.95	1.015
45	HPMVNO1E22F0012	2.930E-003	1.453E-002	5.95	1.015
46	T1	6.090E-002	3.202E-001	5.94	1.471
47	LCLCUMLPCIALPCS	5.290E-003	2.622E-002	5.93	1.027
48	ECHXPL1P42B0001A	2.049E-003	9.627E-003	5.69	1.010

Table 1-6

Accident Sequence Grouping Criteria and Results

Functional Accident Sequence	Definition	CDF	Percent CDF
1A	Accident Sequences Involving Loss of Coolant Inventory Makeup in Which Reactor Pressure Remains High.	7.0E-8	< 1
1B	Accident Sequence Involving a Loss of AC Power and Loss of Coolant Inventory Makeup.	1.5E-6	13
1C	Accident Sequence Involving a Loss of All AC Power and No Recovery of AC Power.	1.2E-6	10
1D	Accident Sequences Involving a Loss of Coolant Inventory Makeup and ATWS.	4.4E-7	4
1E	Accident Sequence Involving a Loss of Coolant Inventory Makeup in which Reactor Pressure has been successfully reduced.	9.4E-7	8
2	Accident Sequences Involving Loss of Containment Heat Removal Leading to Containment Failures and Subsequent Loss of Coolant Inventory Makeup.	2.5E-6	22
3A	Vessel Rupture Leading beyond makeup capability.	< 1.0E-7	< 1
3B	Accident Sequence Initiated or resulting in a small or medium LOCA for which reactor cannot be depressurized and inventory makeup is inadequate.	< 1.0E-7	< 1
3C	Accident sequences initiated or resulting in medium or large LOCA for which the reactor is at low pressure and inadequate coolant inventory makeup is available.	3.9E-7	3
3D	Accident sequences which are initiated by a LOCA or failure for which vapor suppression is inadequate.	< 1.E-7	< 1
4	Accident sequences involving an ATWS leading to containment failure due to high pressure and subsequent loss of inventory.	4.0E-6	34
5	Unisolated LOCA outside containment leading to loss of effective coolant inventory makeup.	< 1E-7	< 1

TABLE 1-7 CONTAINMENT PERFORMANCE COMPARISON OF DESIGN CHANGE CONSIDERATIONS

	GENERIC INITIATING EVENT FREQUENCY			UPDATED INITIATING EVENT FREQUENCY		
Bypass	Core Damage	Containment Structural Failure	RPV Failure And Early Containment Failure With Pool Bypass	Core Damage	Containment Structural Failure	RPV Failure And Early Containment Failure With Pool
Base Case	1.3E-5	4.0E-6	2.0E-6	8.1E-6	2.6E-6	1.0E-6
Passive Vent	1.0E-5 (-18% CHG)	9.9E-7 (-75% CHG)	4.5E-7 (-78% CHG)	6.7E-6 (-17% CHG)	5.3E-7 (-80% CHG)	2.0E-7 (-81% CHG)
ATWS Mods: Alt Shutdown & ADS Inhibit	1.0E-5 (-19% CHG)	3.5E-6 (-13% CHG)	1.8E-6 (-14% CHG)	7.3E-6 (- 9% CHG)	2.5E-6 (- 5% CHG)	9.5E-7 (- 7% CHG)
Passive Vent & ATWS Mods	8.0E-6 (-37% CHG)	4.6E-7 (-89% CHG)	1.6E-7 (-92% CHG)	5.9E-6 (-26% CHG)	4.0E-7 (-85% CHG)	1.2E-7 (-89% CH
Passive Vent, ATWS Mods & Ignitor Power	8.0E-6 (-37% CHG)	1.8E-7 (-96% CHG)	1.1E-7 (-94% CHG)	5.9E-6 (-26% CHG)	1.2E-7 (-96% CHG)	7.2E-8 (-93% CHG)

NOTE: These results are based on an analysis of the core damage sequences included in the plant damage state trees. Thus there are small differences in the impact of changes reported in this table compared with those reported for internal event core damage sequences in section 3.4. These differences do not change the overall conclusions.

Table 1-8

Comparison of Perry IPE with Kuosheng, Cofrentes, & NUREG/CR-4550 Results

Initiating Event	Perry	Kuosheng	Cofrentes	NUREG/CR-4550
Loss of Offsite Power	1.4×10^{-6}	1.0×10^{-6}	1.7×10^{-7}	3.9×10^{-6}
Station Blackout	2.2×10^{-6}	3.3×10^{-6}		
Transient w/ PCS (T3A)	$< 10^{-8}$	1.1×10^{-6}		$< 10^{-7}$
Loss of feedwater (T3B)	$< 10^{-8}$	3.0 X 10-7		$< 10^{-7}$
Inadvertent open SRV (T3C)	1.4×10^{-7}	3.0×10^{-8}		$< 10^{-7}$
Transient w/o PCS (T2)	1.7×10^{-6}	1.3×10^{-5}	$< 10^{-7}$	1.3×10^{-8}
Loss of instrument air (TIA)	1.0×10^{-6}	-NA-		$< 10^{-7}$
Loss of service water (TSW)	6.7×10^{-8}	-NA-		-NA-
Large LOCA (A)	2.1×10^{-7}	$< 10^{-7}$		$< 10^{-7}$
Intermediate LOCA (S1)	6.2×10^{-8}	4.1×10^{-7}	$< 10^{-7}$	$< 10^{-7}$
Small LOCA (S2)	3.3×10^{-8}	$< 10^{-7}$		$< 10^{-7}$
ATWS	4.7×10^{-6}	2.5×10^{-5}	$< 10^{-7}$	1.1×10^{-7}
Vessel Rupture	$< 10^{-7}$	2.7×10^{-7}	$< 10^{-7}$	$< 10^{-7}$
Total Core Damage Freq*	1.2×10^{-5}	3.4×10^{-5}	2.6×10^{-7}	4.0×10^{-6}
Internal Flooding CDF	1.5×10^{-6}	5.7×10^{-7}	-NA-	-NA-

* Total CDF does not include flooding.

Table 1-9

Comparison of Plant Design Features

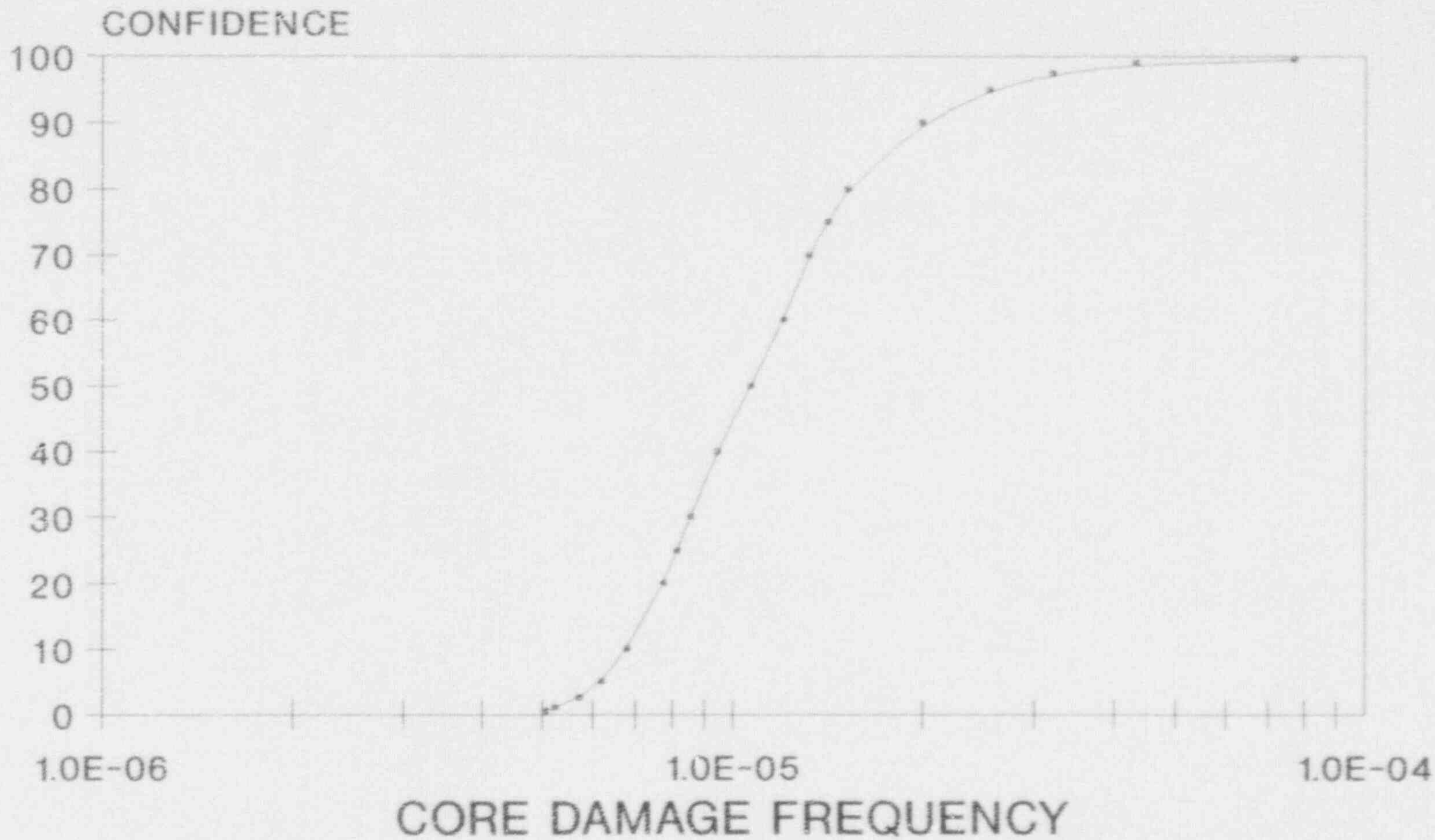
Plant Feature/Parameter	GGNS	Clinton	Perry	River Bend
Rated Power, Mwt	3833	2894	3579	2894
AE	Bechtel	S&L	GILBERT/ COMMONWEALTH	SWEC
ECCS Systems (number x gpm)				
LPCS	1 X 7115	1 X 5010	1 X 6000	1 X 5010
LPCI	3 X 7450	3 X 5050	3 X 6500	3 X 5050
HPCS	1 X 1650	1 X 1400	1 X 1550	1 X 1400
RCIC	1 X 800	1 X 600	1 X 700	1 X 600
ADS Valves/Non-ADS	8 / 12	7 / 9	8 / 11	7 / 9
CONDENSATE/FEEDWATER				
Turbine-driven Pumps	2	2	2	0
Motor-driven Pumps	0	1	1	3
Feed Booster	0	0	4	0
Condensate Pumps	3	4	3	3
Cond Booster Pumps	3	4	3	0
Turbine Bypass Capacity (%)	35	35	35	1[illegible]
Service Water				
Safety Related	3 independent	3 independent	3 independent	4 pumps/2 loops
Non-safety Related	1 loop/8 pumps	1 loop	1 loop/4 pumps	1 loop
AC Power				
Offsite Power Circuits	3	4	4	6
ESF Buses	3	3	3	3
Standby Diesels	3	3	3	3
DC Power				
Safety Related Batteries	3	4	3+3(Unit 1&2)	3
Non-Safety Related Batteries	4	2		3
Design Battery Life (hours)	4(Div 1&2) 2(Div 3)	4	7 22(x-tie & shedding)	4(Div 1&2) 2(Div 3)

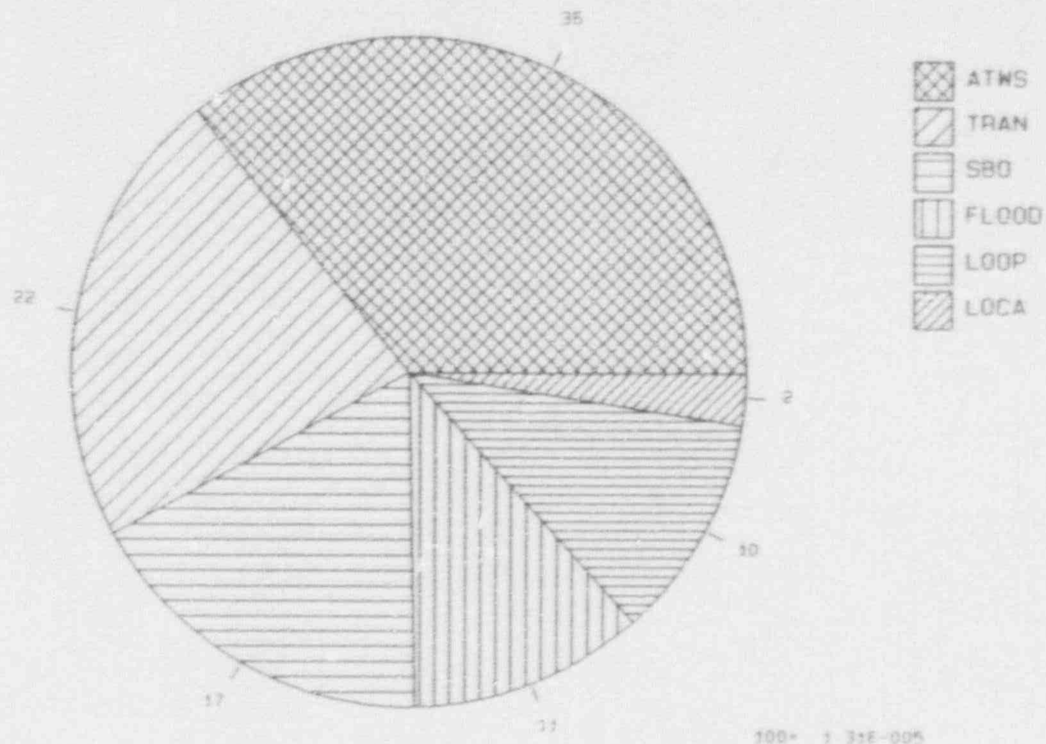
TABLE 1-10 SUMMARY OF CONTAINMENT FAILURE FREQUENCY

		GENERIC INITIATING EVENT FREQUENCY BASE CASE	UPDATED INITIATING EVENT FREQUENCY BASE CASE	UPDATED INITIATING EVENT FREQUENCY PASSIVE VENT
No RPV Failure:	No Containment Failure	3.39E-6 (26.7%)	1.22E-6 (15.2%)	1.22E-6 (18.3%)
	Vent	2.45E-6 (19.3%)	2.36E-6 (29.4%)	2.82E-6 (42.3%)
	Containment Failure	6.18E-7 (4.9%)	4.70E-7 (5.8%)	6.74E-8 (1.0%)
Subtotal No RPV Failure Core Damage Frequency:		6.46E-6 (50.8%)	4.06E-6 (50.3%)	4.11E-6 (61.6%)
RPV Failure:	No Containment Failure	1.58E-6 (12.4%)	6.22E-7 (7.7%)	6.22E-7 (9.3%)
	Vent	1.27E-6 (10.0%)	1.23E-6 (15.3%)	1.47E-6 (22.1%)
	Late Containment Failure	9.38E-7 (7.4%)	9.25E-7 (11.5%)	2.45E-7 (3.7%)
	Early CF: No Pool Bypass	4.30E-7 (3.4%)	1.88E-7 (2.3%)	2.00E-8 (0.3%)
	Late Pool Bypass	1.54E-6 (12.1%)	7.51E-7 (9.3%)	8.40E-8 (1.3%)
	Early PB, Spray	6.12E-8 (0.5%)	2.82E-8 (0.3%)	2.78E-8 (0.4%)
	Early PB, No Spray	4.45E-7 (3.5%)	2.49E-7 (3.1%)	8.76E-8 (1.3%)
Subtotal RPV Failure Core Damage Frequency:		6.27E-6 (49.2%)	4.00E-6 (49.7%)	2.56E-6 (38.4%)
TOTAL CORE DAMAGE FREQUENCY:		1.27E-5 (100%)	8.05E-6 (100%)	6.67E-6 (100%)
Subtotal Containment Venting Frequency:		3.72E-6 (29.2%)	3.59E-6 (44.7%)	4.29E-6 (64.4%)
Subtotal Cntmt Structural Failure Frequency:		4.03E-6 (31.7%)	2.61E-6 (32.5%)	5.34E-7 (8.0%)
TOTAL CONTAINMENT FAILURE & VENTING FREQUENCY:		7.76E-6 (60.9%)	6.21E-6 (77.1%)	4.83E-6 (72.4%)
RPV FAILURE & EARLY CONTAINMENT FAILURE WITH POOL BYPASS FREQUENCY:		2.04E-6 (16.1%)	1.03E-6 (12.8%)	1.99E-7 (3.0%)

CORE DAMAGE FREQ DISTRIB

FIGURE 1-1





CONTRIBUTION OF ACCID. INITIATORS TO CDF

Figure 1-2

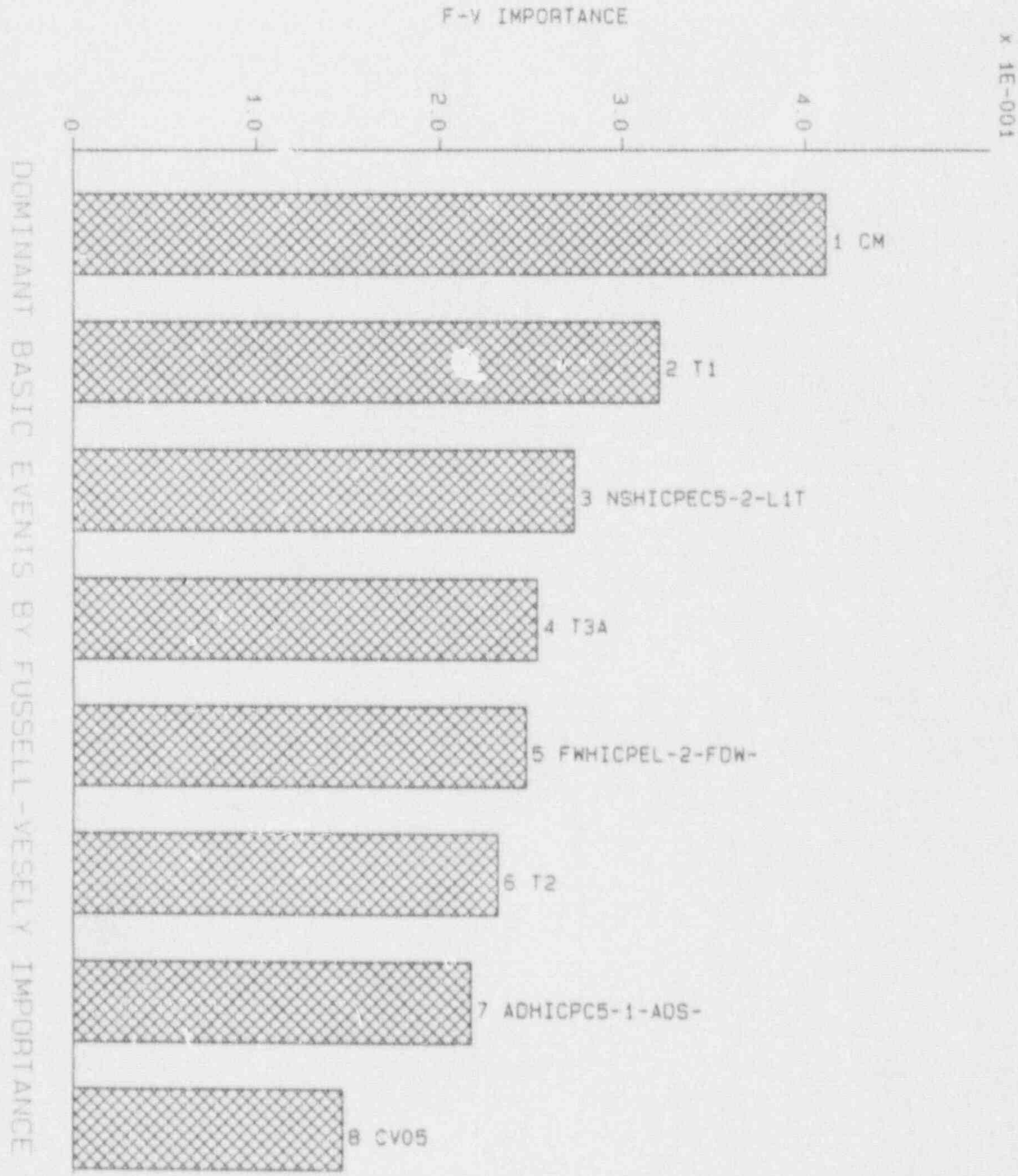
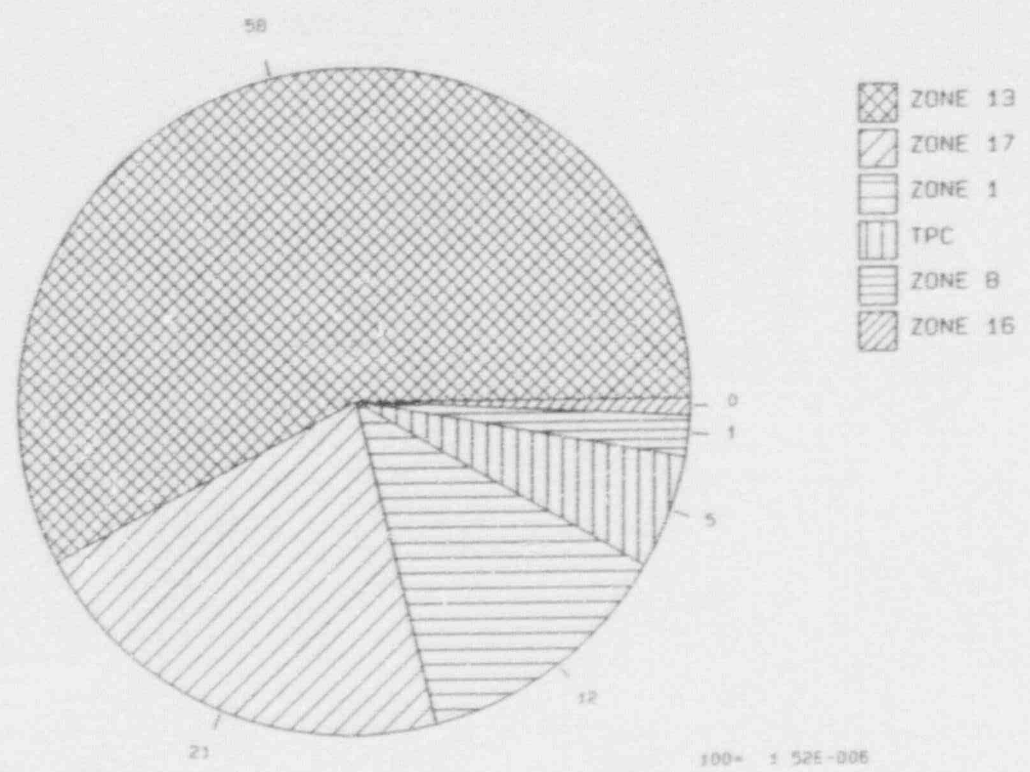

DOMINANT BASIC EVENTS BY FUSSELL-VESELY IMPORTANCE
F-V IMPORTANCE
x 1E-001
0
1.0
2.0
3.0
4.0
1 CM
2 T1
3 NSHICPEC5-2-L1T
4 T3A
5 FWHICPEL-2-FDW-
6 T2
7 ADHICPC5-1-ADS-
8 CV05

Figure I-3



CONTRIBUTION OF EACH FLOOD ZONE TO CORE DAMAGE

Figure 1-4

2.0 EXAMINATION DESCRIPTION

2.1 INTRODUCTION

The Perry IPE project is aimed at meeting two objectives, satisfying the NRC requirements for an individual plant examination and providing plant models for use in assisting in maintaining the safe operation of the Perry Nuclear Power Plant. In this section it is demonstrated that the project conforms with the NRC requirements. The general methodology and the information used in the course of the study are also described.

2.2 CONFORMANCE WITH GENERIC LETTER AND SUPPORTING MATERIAL

Generic Letter 88-20 and its supplements identify a number of requirements in the areas of the examination process, methodology, treatment of unresolved safety issues and reporting. This IPE conforms with the requirements laid out in the generic letter and uses the current supporting material appropriate to the analysis of a BWR/6 Mark III. The software used in the Level 1 and Level 2 sections of the analysis has been developed in accordance with NRC or EPRI QA requirements and is in general use in the industry. Conformance within each of the specific areas is discussed in the following sections.

2.2.1 Examination Process

It is the belief of the NRC as stated in the generic letter that the quality and comprehensiveness of the results derived from the IPE depend on the vigor with which the utility applies the method of examination and on their commitment to the intent of the IPE. It can be seen from the project organization shown in Figure 2-1 that the CEI personnel were involved in the performance of all tasks. In fact the majority of the Level 1 and 2 analyses was performed by Perry personnel with assistance from a number of consultants in key areas such as performance of the containment ultimate strength analysis and the core melt phenomenology.

In house review of systems analysis was performed by Perry staff conversant with system design, maintenance and operation. In addition to the in house review further review of the examination process and the results were performed by independent consultants.

The combination of experienced consultants, Perry personnel and independent review in the performance of the analysis has ensured that the quality and comprehensiveness of the Perry IPE results meet the highest standards and will be usable in the future as a living PRA and for the development of accident management strategies.

2.2.2 External Events

The current study includes only core damage and fission product release assessment following an internal initiating event or internal flood. A number of plant walkdowns have been conducted for the flooding analysis and for the investigation of common cause failures. The information gained from these walkdowns has been included in the appropriate analysis files and will be available when performing external event analyses.

2.2.3 Methods of Examination

The approach used to satisfy the IPE was to perform a Level II PRA based on the procedures in NUREG/CR-2300 (Hickman 1983) and information provided in NUREG/CR-4550 (Drouin, 1989) for the Grand Gulf analysis. In order to ensure that methods used for the human reliability and common cause analysis represent the current state of the art in these areas the latest information from either EPRI or the NRC was used to perform the analysis. As the PRA was performed at the Perry site the latest design information (January 1990) was used in the performance of the study. As the quality assurance for the development of the plant model has required full recording of all documentation used, it will be a straightforward process to update the model as design change packages are complete. The methodology is fully described in Section 2.3.

Details of the approach to the Containment Structural Analysis are given in Section 4 of this report.

2.2.4 Resolution of Unresolved Safety Issue A-45

The PRA specifically addresses the performance of the shutdown decay heat removal systems following transients from high power or loss of coolant accidents. The overall contribution to core damage is assessed and any individual system vulnerabilities identified.

2.2.5 Severe Accident Sequence Selection

The results of the analysis which are reported in Sections 3.4 and 4.7 have been based on the screening analysis defined in Appendix 2 of the generic letter and amplified in the guidance document (NRC, 1989). Any identified vulnerabilities, unique safety features, and potential improvements are discussed in Section 6.

2.2.6 Documentation of Examination Results

As it is intended that the IPE shall be widely used within CEI in the future, considerable care has been taken in developing the documentation for the study. The documentation of the completed study has three components. The first is this report which summarizes the results and findings of the PRA given the current plant status. The second is the Perry plant model which exists in the PC software NUPRA/NUCAP+/EVNTRE. This model can be updated as and when changes in design or procedures are made. The third is the set of analysis files which document all the assumptions, boundary conditions and documents used in developing the plant model on the PC. It is these documents which form the basis for the living PRA in that they provide the basis of comparison of new designs, data, operational procedures, training, testing, etc with that used for the base scope study, and thus enable the changes that need to be made to the model to be identified.

One of the key items in this third tier of documentation is the listing of all the documents (such as the Updated Safety Analysis Report (USAR), P&IDs, etc.) used and reports produced in the study.

2.3 GENERAL METHODOLOGY

The methodology used to perform the analysis for the Individual Plant Examination is designed to answer the following questions.

1. What events at the plant will result in a reactor trip or the necessity for the reactor to trip?

2. What requirements for reactivity control, pressure control or level control will arise?

3. How can decay heat be removed following reactor trip?

4. How often (with what frequency) will core damage occur as the result of failure to remove decay heat?

5. How likely is containment failure to occur following core damage?

6. What fission product release is likely to occur and how often?

7. What are the significant contributors to the frequency of core damage, containment failure, and fission product release?

The answers to the above questions are determined by carrying out the following basic steps shown in the flow diagrams in Figures 2-2 and 2-3.

1. Identifying the causes and frequency of plant trips (Initiating Events).

2. Developing a model which represents the many different ways in which decay heat removal can fail following the various initiating events (event trees and fault trees) taking into account the reactivity, pressure and level control requirements to achieve decay heat removal.

3. Assembling data for the component failures, maintenance unavailabilities, operator actions, etc. included in the event and fault tree models. (Data and common cause failures)

4. Quantifying the core damage frequency and performing importance analysis to identify the dominant contributors to core damage (quantification, importance analysis).

5. Performing a containment building strength analysis to determine the likelihood of failure over a range of pressures and temperatures.

6. Determine the range of pressures and temperatures in the containment building for a range of possible core damage scenarios. Each scenario is identified by a specific set of conditions at the time of core damage known as the plant damage state. (containment analysis)

7. Combine 5 and 6 to determine the likelihood of containment building failure and the magnitude of the fission product release.

The way in which each of these steps is performed is discussed in the following paragraphs. In order to ensure that the analysis was as complete as possible and that all the interfaces between the various sub-tasks of the PRA were clearly recognized, the first step in performing the analysis was to develop a set of comprehensive procedures to identify the way in which the analysis in each sub-task was to be performed. The procedures used are listed in Table 2-1.

2.3.1 Initiating Events and Success Criteria

Initiating event information was derived from the plant operating experience for the past two years, from past studies of BWRs and the NRC study of Grand Gulf. Particular attention was paid to the Failure Mode and Effects analysis of support systems which would result in possible initiating events associated with the loss of switchgear room cooling, loss of service water, loss of alternating current (AC) and direct current (DC) bus bars and loss of instrument air.

The safety functions required in response to each of the initiating events were defined and the systems which could perform these functions were identified. The success criteria were determined by reviewing earlier analyses, the USAR, General Electric analyses and by performing thermal hydraulic analyses using the MAAP code to confirm the requirements and to obtain estimates of timing. These are discussed in section 3.1.1.

Following the identification of the success criteria for each initiating event, it was possible to group together events which had similar system success criteria in order to reduce the number of event trees to be developed.

2.3.2 Accident Sequence Development

An event tree was developed for each initiating event or event group using the information developed for the system success criteria and including the operator actions identified in the various emergency procedures (section 3.1.2). An individual path through each event tree (an accident sequence) then specifies the combination of system failures and successes and operator actions which will lead either to successful core cooling or to core damage through lack of cooling. As the Perry IPE is effectively a Level 2 PRA, this analysis also considered the impact that the containment building systems could have on the potential for core damage. In order to clearly define the boundary conditions in terms of timing for each sequence, it was necessary to perform a number of MAAP analyses. The results of these analyses enabled the timing of actuation signals and the time available for operators to perform back up or recovery actions to be identified. The development of the plant damage states is described in Section 2.3.8.

2.3.3 Data Base

Plant-specific data has been used to determine the maintenance unavailability for all front line and important support systems, and for the frequency of transients initiating events for comparison with the generic transient initiating event frequencies. Generic data has been used for all other items.

2.3.4 Dependent Failures and Interaction

In addition to the failures directly associated with the components in the system performing the safety function, it is necessary to model the dependent failures. Direct functional dependencies (i.e. electric power, pump bearing cooling, initiating signals, etc.) are modeled by developing a dependency matrix for each system and including these dependencies in the system fault tree. The system performance is then evaluated by linking the fault trees for the supporting systems into the appropriate points of the system fault tree. These dependencies are summarized in section 3.2 and shown in detail in the fault trees in Appendix A.

Multiple failures of some components may occur as the result of influences which it is neither possible to identify nor explicitly model in the system fault tree of the individual equipment failures. In addition, there may also be interactions between systems as the result of location or the fluid being pumped by the system. An attempt has been made to model such failures by including common cause events in the fault trees for components which are assessed to be subject to such common failures, for example diesel generators, motor-driven pumps, motor-operated valves, and check valves. The quantification of these events has been based on a review of both plant design and industry data. The evaluation and quantification of these dependencies is described in section 3.3.4 and Appendix C.

The dependency of systems on environmental conditions has been included in the analysis. For each system the running time and room temperature have been evaluated, over the range of accident scenarios, to determine if room cooling is required. The impact of suppression pool water temperature on the systems which take suction from the suppression pool has been taken into account. This information is included in the system analysis supporting documentation and summarized in section 3.2.

The dependencies between initiating events and systems required to prevent core damage or the release of fission products are modeled in the event and fault trees. Similarly, the dependency of a function in an event tree on the success or failure of a previous function is the basis for the construction of the event trees. The dependency between function and initiating events is represented in the system fault trees by the use of house events. These events are then set to one or zero depending on the system being modeled and the initiating event. The setting of these events for each of the sequences are given in Appendix E.

2.3.5 System Modeling

As insufficient data exist for determining the probability of failure of each system directly, it is necessary to model each system in a logical way, breaking it down to individual components for which failure data are available or can be estimated. The basic model used in this study, as is in other PRAs, is the fault tree.

The systems for which fault tree models are required are defined in the development of the success criteria and event trees. In addition to the front line systems, the list includes the support systems such as electric power, cooling water, control air, and instrumentation and control. For each system a fault tree has been developed either to the component level or, if it is not important relative to core damage frequency, to a lesser level of detail, but always including the dependency on supporting systems. (Section 3.2 and Appendix A)

2.3.6 Human Interaction Analysis

System and safety function failures may result from the operators' failure to perform certain actions or the performance of incorrect actions. Therefore operator actions are included in the event and fault trees. Two types of actions are identified.

1. Errors occurring prior to the plant trip which result in equipment (e.g. valves) left in the wrong position at the time of the demand on the system or result in miscalibration of instruments such that a system is not initiated.

2. Post trip errors which result in the operator failing to take the correct action in response to the plant state either to start a system or failure to perform a backup action if auto initiation should fail.

The second group can in fact be broken down into the following three subgroups: failing to take the correct action in accordance with procedure, taking the incorrect action, and failing to take appropriate recovery actions following system failures.

A systematic approach based on the work performed at the Electric Power Research Institute (EPRI) was used to perform the analysis. There are three essential steps in this method. The first is identification by the event and fault tree analyst of the human interactions that can occur which will impact the operation of the system either before or after the plant trip. This is achieved by a detailed review of test, maintenance, and operating procedures.

The second step is to evaluate each action in terms of how the action would be performed for various combinations of prior system failures, how long the operator has to perform the action, what cues he has to indicate the necessities to perform the action, and how it relates to the procedures. At this stage it may be necessary to modify the event or fault tree in order to more accurately model the human interaction. It is also necessary to examine the combinations of human actions that occur in a given sequence of events in order to determine the relationship between success and failure of one action to the success and/or failure of the second and/or subsequent actions in the combination.

The third and final stage is the quantification of each of the actions ultimately modeled in the event and fault trees. The quantification of each of the important actions was based either on THERP or methods based on those developed as a result of the Operator Reliability Experiments (ORE) program performed by EPRI. Quantification of the less important actions was based on estimates used in NUREG/CR-4550 (Drouin, 1989) for the Grand Gulf PRA for similar actions. (Section 3.3.3)

2.3.7 Accident Sequence Quantification

In this step the information gathered in the preceding steps is used to quantify the individual core damage sequences. In the NUPRA code the first step is to develop linked and quantified fault trees for all the functions in the event tree. Care is taken to ensure that for each function the system fault tree used reflects the boundary conditions in terms of initiating event and actuation conditions appropriate to the sequence in which the function appears. Each sequence is then quantified by combining the results from the quantification of the linked fault trees taking care to ensure that both success and failure paths are accounted for in each core damage sequence. The final result is entered into a data base which stores the frequency of each sequence, the various functional failures and successes, and a list of the combinations of basic event failures contributing to the sequence frequency. (Section 3.3.5 and 3.3.6)

2.3.8 Plant Damage State Analysis

The interface between Level 1 and Level 2 is the set of plant damage states which reflect the operability of plant systems and containment building systems, as well as the Reactor Pressure Vessel state at the onset of core damage. In order to group the individual core damage sequences into the various plant damage states, two requirements have to be met. The first is to define the grouping criteria. The criteria have been developed based on those aspects of plant and containment building systems or reactor state at the onset of core damage which have a significant impact on the accident progression and ultimate fission product release. (Section 3.1.4)

The second is to develop a set of event trees which include all the information necessary to apply the grouping logic developed above. This requires extending the core damage event trees to include not only functions which can prevent core damage, but also functions which will prevent containment building failure or significantly impact fission product release.

2.3.9 Containment Evaluation

The Perry IPE containment evaluation is performed using an Accident Progression Event Tree (APET) similar to that used in the Grand Gulf NUREG/CR-4551 analysis (Brown, 1990). The initial accident progression analysis begins by grouping the functional characteristics determined in the plant damage state analysis into plant damage state groups. Each Plant Damage State contains Front-End sequences with sufficient similarity of the system functional characteristics that the containment accident progression for all sequences in the group can be considered to lead to the same thermal hydraulic transient containment response. The APET is then extended into early, intermediate and late time periods to determine: the coolability of the core debris in-vessel, the mode of containment failure and the mode of suppression pool bypass. Containment performance and source term category groups are determined using a binner function with a specified grouping logic. Source term category grouping logic selects those sequence characteristics important to source term assessment. Release category source terms are estimated based on the results of Modular Accident Analysis Program (MAAP) computer runs.

2.3.10 Risk Contribution Evaluation

The results of both the core damage quantification and the source term releases have been analyzed to identify the dominant contributors to each in terms of initiating events, system and component failures, operator actions, containment capacity, and phenomenological sensitivity. This has been achieved by performing three sets of analyses: sensitivity, importance, and uncertainty. The use of importance analysis leads to the identification of the dominant contributors to the frequency of core damage and the frequency of a given source term. The identification of vulnerabilities in this way (dominant or significant contributors that are initiating events, component failures, operator actions, maintenance items, or containment building failure modes) enables the areas in which improvements would have the most impact to be identified. The important contributions to decay heat removal failure and the evaluation of these vulnerabilities is addressed in the risk contribution evaluation task.

2.4 INFORMATION ASSEMBLY

The first step in the performance of the IPE was the assembly and review of the documentation necessary to perform this study. This consisted of the USAR, the PRA performed for Grand Gulf (Drouin, 1989), the piping and instrumentation diagrams, normal and emergency operating procedures, control room logs, maintenance records, and selected thermal hydraulic analyses performed by General Electric and the architect/engineer Gilbert/Commonwealth. In addition, the PRA studies performed for the BWR/6s at Kuosheng in Taiwan (ROC, 1985) and Cofrentes in Spain, (Hydroelectrica, 1991) were made available to the project team.

The members of the project team from CEI were all very familiar with the Perry plant. The members of the project team from Gilbert/Commonwealth were also very familiar with the Perry plant, having been involved in the construction and startup of the plant. All the Halliburton NUS personnel had previously performed a PRA for the BWR/6 at Kuosheng and were therefore familiar with the plant layout and performance. Visits were made by personnel during the course of the study to acquire specific information relating to the flooding analysis and to system performance.

One of the major concerns when developing a living IPE/PRA model is that it represents the as-built as operated plant. In order to ensure that the Perry model described in this report meets this requirement, the following specific actions were taken:

1. The study was performed by the Independent Safety Engineering Section at the Perry site so that design documentation was directly available.

2. Analysis files were set up for each phase of the model development to ensure that the documents used and decisions made on the basis of information in a given document were recorded. This ensures that comparison between the model and subsequent design change packages can be made in a controlled manner.

3. The design engineers reviewed all the system models for correctness of assumptions concerning design, alignment and operation.

4. Operations reviewed all the event trees.

5. The current set of operating procedures were used in performing the human reliability analysis and many of the actions were discussed with training and operations personnel.

6. Maintenance data was acquired directly from plant operating experience.

7. A significant number of visits were made to the plant to walkdown systems which would lead to flooding and trace potential flood propagation pathways.

8. The Containment Building Strength Evaluation and portions of the internal flooding analysis were performed by Gilbert/Commonwealth, the architect/engineer (A/E) for the Perry Nuclear Power Plant.

9. In addition to reviews of each of the system analyses by the appropriate design engineers, intermediate reviews of the work products and the draft report were performed by key personnel from the operations, training, and engineering departments.

2.4.1 Plant Layout and Containment Building Information

The Perry Nuclear Power Plant is a two unit site on which only Unit 1 has been completed and is operating. Unit 2 was approximately 40% complete when all work was stopped. The site is therefore a single unit site as far as interactions are concerned. The unit is a General Electric BWR/6 with a Mark III containment. The balance of plant systems were engineered by Gilbert/Commonwealth. The unit started commercial operation in 1987. Some important design features of the Perry plant are described in Table 2-2.

The applicability of the unique features associated with Mark III containment design to the progression of an accident are discussed in Section 4.1 of the report.

2.4.2 Review of PRA Studies of Plants Similar to Perry

Three BWR/6 Mark III plants have been the subject of previously published PRAs. These are the Kuosheng Nuclear Power Plant (ROCAEC, 1985), Grand Gulf (Druin, 1989) and Cofrentes (Hydroelectrica, 1991). The NRC has also published a document summarizing generic risk insights for General Electric Boiling Water Reactors (Travis, 1991). Other studies reviewed for information pertaining to either internal events, internal flooding or containment evaluation analysis were GESSAR II (General Electric, 1982), the EPRI document recommending sensitivity analysis to be performed when using the MAAP code (Kenton, 1991) and the evaluation of severe accident risk performed for Grand Gulf reported in NUREG/CR-4551 (Brown, 1990).

For the Level 1 analysis, the above studies were reviewed to identify the dominant contributions to core damage and any plant-specific vulnerabilities associated with equipment or systems similar to equipment or systems at Perry. In particular, careful attention was paid to the Grand Gulf study to determine what assumptions had been made with regard to room cooling requirements, the availability and modeling of support systems and the evaluation of the emergency operating procedures and recovery actions. The results of this study are compared with the results of the NUREG/CR-4550 and other BWR/6 PRA results in sections 1.4.4 and 3.4. Important insights from some of these studies are listed in Table 2-3 and the core damage frequencies for a number of plants in Table 2-4.

For the Level 2 analysis the above studies were reviewed to identify 1) Plant systems and components important to accident progression and mitigation, 2) Phenomenological events and processes which are important to drywell and containment failure and source term definition, and potential containment loading and failure mechanisms and modes. These studies were also reviewed to determine the criteria and parameters used in the definition of plant damage states and the physical processes, system operation, operator actions, etc., which were considered of sufficient importance to containment accident progression to be included as a containment event tree heading in the analysis. The results from the containment analysis are compared with the results from the analysis performed for Grand Gulf in sections 1.4.4 and 4.7.4.

2.4.3 Walkthrough Activities

In order to ensure that all members of the project team were familiar with the Perry Plant, a plant familiarization task was included in the work scope. This consisted of reviewing the system descriptions and making a plant visit at the beginning of this study so that analysts could familiarize themselves with the general information and layout of the plant.

As the study was conducted at the Perry site visits were made to the unit as and when the need arose. The areas of analysis which resulted in such visits are discussed in the following sections.

2.4.3.1 Human Reliability Analysis

Throughout the human interaction modeling task, communication was maintained with the Operations Procedure Unit personnel in order to thoroughly perform the plant logic model construction. Particularly, close communication was maintained with the Plant Emergency Instruction Coordinator to clearly understand the Plant Emergency Instruction Flow Charts. Several Plant Emergency Instructions relating to alternate injection and containment venting were walked down in the plant. The assessment of human interaction timing was determined from interviews with the Operator Training Unit, as well as plant operator walkdown.

2.4.3.2 Data Analysis

Because of the short operating history of the plant generic failure data was used for component failure rates. For unavailability due to maintenance for the HPCS and RCIC systems, Perry-specific data from the third operating cycle was used as the core damage frequency was most sensitive to the maintenance unavailability of these systems. Earlier operating cycle data was used for other systems. The maintenance data was taken from the Unit Log, Active LCO Tracking sheets and the Fire Protection Log for the periods of time when the unit was at power. Maintenance data for the Motor Feed Pump was taken from Work Orders. These documents were obtained from Document Control at the site.

2.4.3.3 System Modeling

As part of the system modeling task plant walkdowns were performed. These walkdowns are documented in the analysis files of the system modeling task. Walkdowns were done on an area by area basis without trying to trace a system from beginning to end. General location of components, external crud buildup, rust, leaks, and general cleanliness were all examined as part of the walkdown process. A team of two individuals performed all system modeling walkdowns.

2.4.3.4 Dependencies and Common Cause

Included in the walkdowns performed for the system modeling task were examinations of potential dependencies such as the proximity of systems and components to high energy pipes and sources of water damage, sources of heat or enclosed areas subject to heat buildup. Multiple systems located in a given vicinity were noted. This is documented in the system modeling analysis files.

2.4.3.5 Internal Flooding

Several walkdowns were performed by a team of two persons covering the general plant areas with emphasis on the critical identified zones. Direction of door opening, penetrations, and equipment heights were investigated. In addition numerous one man walkdowns took place. These typically involved checking the gaps under doors, measuring critical heights on skid and panel mounted equipment, and various other concerns such as counting pipe welds where this factor was needed for input to the flood initiation frequency.

2.4.3.6 Containment Evaluation

The containment evaluation walkdown focused on modeling the modes of containment failure impact on RPV injection and on suppression pool loss. Containment walkdowns were made to Unit 1 to provide general familiarization of detail. When the containment anchorage failure mode was identified, the walkdowns were extended to the Unit 2 containment (now under construction) to better examine the containment construction within the open shield building annulus.

References

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Table 2-1

List of Procedures

Description	Number
IPE Project Plan	-NA-
IPE Technical Assignment Plan	80793
Initiating Event/Accident Sequence Task Plan	80744/80745
System Modeling Task Plan	80746
Dependent Failure Task Plan	80747
Data Base Task Plan	80748
Human Interaction Task Plan	80749
Risk Contribution Evaluation Task Plan	80750
Internal Flooding Task Plan	80751
Containment Evaluation Task Plan	80752
Final Report Task Plan	80755

Table 2-2

Summary of Design Features - Perry

1. High Pressure Injection Systems

   a. High pressure core spray; diesel backed motor driven pump.

   b. Reactor core isolation cooling; turbine driven pump.

   c. Condensate/Feedwater system; turbine and motor driven pumps.

2. Low Pressure Injection Systems

   a. Low pressure coolant injection system; three independent trains.

   b. Low pressure core spray system; one train.

   c. Condensate transfer alternate injection system.

   d. Fire Protection system; diesel driven pump.

3. Depressurization System

   a. Automatic depressurization system - 8 relief valves

   b. Manual depressurization using ADS or non-ADS valves - 19 relief valves.

4. Decay Heat Removal

   a. Power conversion system.

   b. Two trains of RHR with one heat exchanger in each train which can be used in the following modes:

      Suppression pool cooling mode;
      Containment spray mode.

5. Emergency Service Water

   Two trains of emergency service water with one pump in each train which provides cooling to the RHR heat exchangers. One train provides an alternative injection path to the RHR train B.

6. Electrical Design

   a. Number of offsite circuits = 4

   b. Number of auxiliary power circuits = 4
      1 unit auxiliary transformer;
      1 startup transformer (per unit).

c. Number of preferred power circuits of ESF buses = 2.

d. Number of ESF buses per unit = 3.

e. Number of standby AC power sources per unit = 3 (1/ESF bus).

f. Number of 125VDC systems = 6 (2/ESF bus).

g. Sharing of standby power supplies and interconnection between safety buses - Division 3 D/G (HPCS) can be connected to ESF Bus 2.

7. Containment Structure

a. Type: Mark III, steel containment with pressure suppression, enclosed by reinforced concrete shield building. Containment enclosed drywell and suppression pool.

b. Construction: Steel shell enclosed by reinforced concrete cylindrical structure with hemispherical head. The internal design temperature is 185°F, the design pressure 15 psig and the free volume 1.165 X 10^6 ft^3.

c. Drywell: Reinforced concrete; basically cylindrical with a flat concrete roof with a steel refueling head. The internal design temperature is 330°F and design pressure is 30 psi. The external design pressure is 21 psi. The free volume is 277,700 ft^3.

d. Suppression Pool: Reinforced concrete, steel lined; basically annualr. The minimum water volume is 115,612 ft^3.

Table 2-3

Significant Insights From Other Studies

NUREG/CR-4550 Grand Gulf IPE (1989)

1. Station Blackout Sequences are the dominant contributor to core damage (97%).

   Significant contributions are:
   - Operator failure to recover AC power
   - RCIC turbine driven pump failure
   - Failure to recover diesel generators
   - Diesel generator hardware failures
   - Battery common cause failure

2. ATWS sequences contribute 3%, all other sequences are at or below a core damage frequency of 1.0E-8/yr.

Kuosheng Study (AEC, 1985)

1. ATWS sequences are the dominant contributor to core damage (76%).

   Significant contributions are:
   - Failure of the standby liquid control systems (SLC)
   - Failure of the operator to initiate SLC
   - Failure of RCIC pumps

2. Loss of offsite power and station blackout contributes 13% and all other transients 8%. Loss of coolant accidents contribute 3%.

Cofrentes Study (Hydroelectrica, 1991)

1. Loss of offsite power and station blackout accidents are the dominant contributors to core melt (66.5%).

   Significant contributions:
   - Failure of the RCIC turbine driven pump
   - Common cause failure to operate the diesel generators A and B
   - Operator failure to depressurize following a station blackout
   - Common cause failure of Div I and II essential service water pump discharge valve to open

2. Loss of coolant accidents contribute 15.5% and transients resulting in isolation of the reactor, loss of feedwater, loss of instrument air and loss of service water contribute 16%. ATWS sequences only contribute 2%.

Table 2-3 continued

Generic Risk Insights for BWRs (NUREG/CR-5692, 1991)

This study summarizes the results from seven studies.

1. For 3 plants loss of offsite power and station blackout are the dominant sequences.

2. For 3 plants transients or small LOCA with failure of high pressure injection and a failure to depressurize are dominant.

3. For one plant transients or LOCA followed by loss of containment heat removal is dominant.

4. For one plant the maximum contribution from failure to scram is 43%. For all the other plants it is 12% or less.

Table 2-4

Comparison of Core Damage Frequency From Various PRAs

Plant PRA	Mean Core Damage Frequency for Internal Events
Cofrentes	2.6 E -7
Grand Gulf	4.0 E -6
Perry	1.2 E -5
Kuosheng	3.4 E -5
Surrey	7.4 E -5
Oconee	1.4 E -4
Seabrook	1.7 E -4
Three Mile Island	4.4 E -4

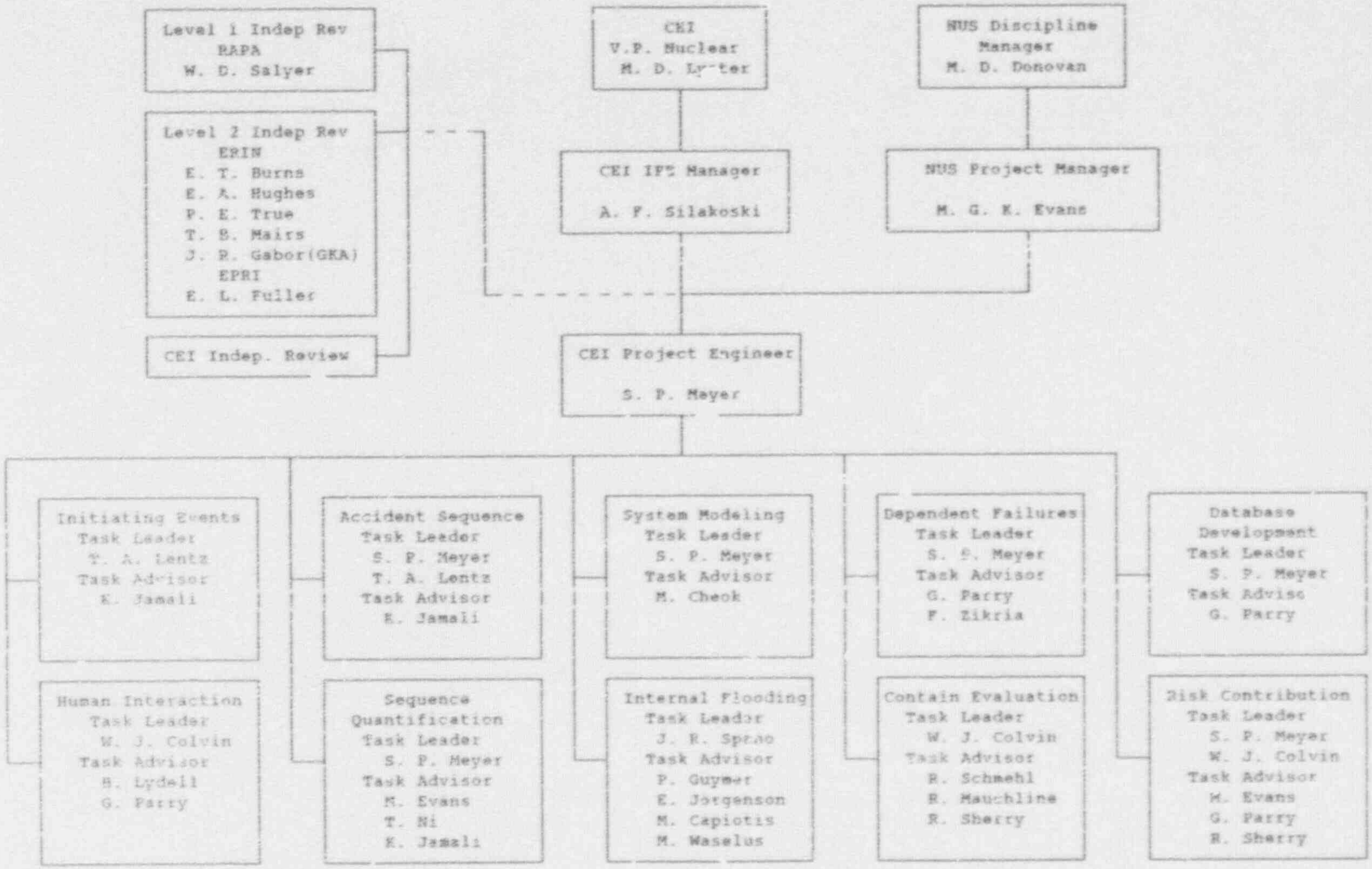


Figure 2-1 Project Organization

FIGURE 2-2 OVERVIEW OF LEVEL 1 PRA METHODOLOGY

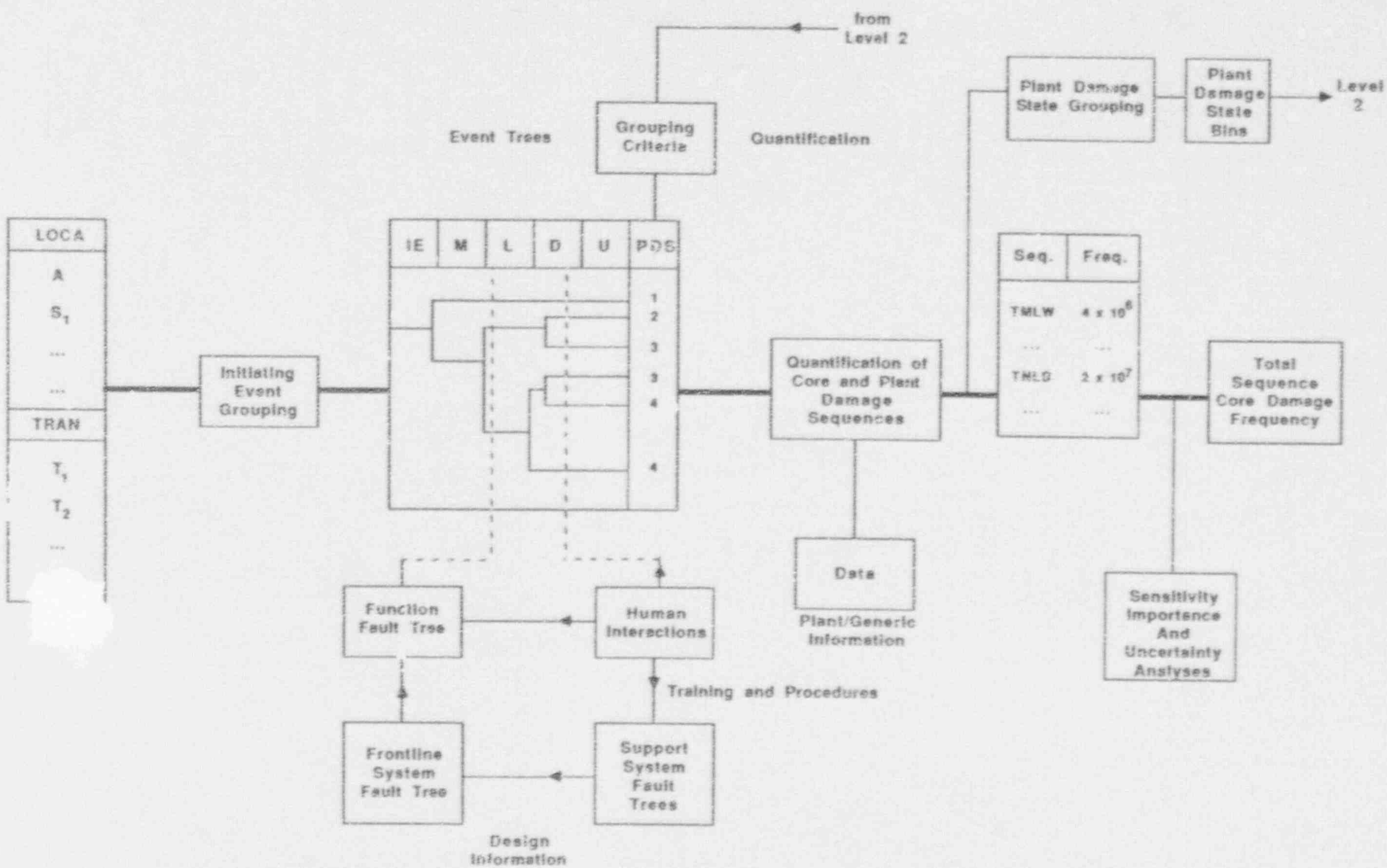
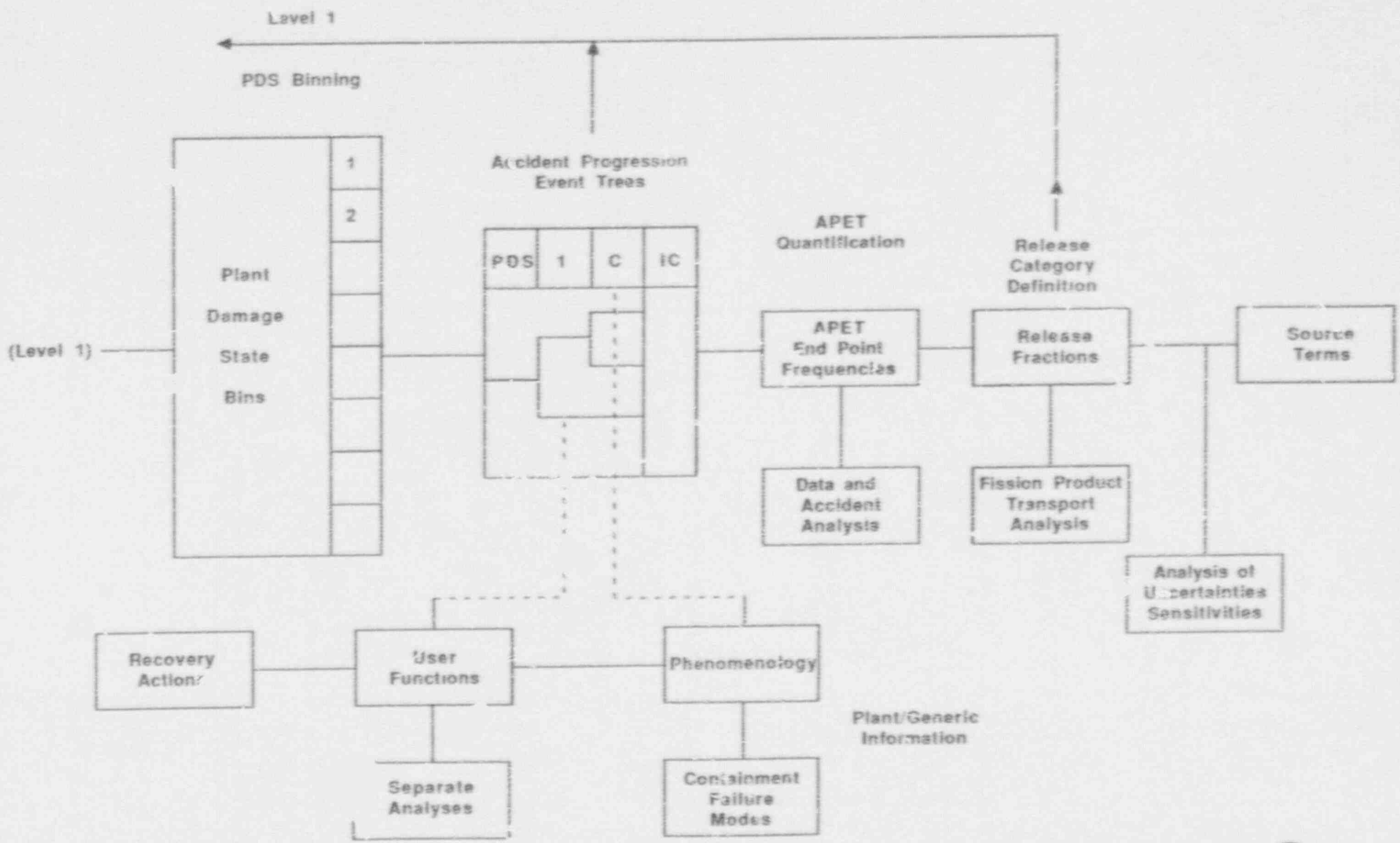


FIGURE 2-3 OVERVIEW OF LEVEL 2 PRA METHODOLOGY



3.0 FRONT END ANALYSIS

3.1 ACCIDENT DELINEATION

The reporting of the accident delineation process is divided into four sections. In the first section the selection and grouping of the initiating events is described. The event trees associated with each initiating event group and its success criteria are developed in sections 3.1.2 and 3.1.3 for the Front Line and Special event trees respectively. Finally the sequence grouping into the various plant damage states in order to achieve the Level 1/Level 2 interface is described in section 3.1.4.

3.1.1 Initiating Events

For the purpose of this analysis, initiating events are defined as those plant occurrences, during normal plant operation, which cause a rapid shutdown of the plant (automatic scram), or an immediate need to manually trip the plant (manual scram), so as to challenge the capability of the plant systems to bring the reactor to a safe shutdown condition. Manual orderly shutdowns of the plant for outages or administrative reasons (i.e. Technical Specifications) are not considered. Events during plant startup, shutdown or below 12% power are additionally excluded from the scope of this study. The decision to define events at power as those above 12% is that this is the point at which the mode switch is placed to RUN.

This section discusses the identification of the initiating events which are analyzed in the Perry IPE, and the grouping of these events into classes. The list of Perry IPE initiating events is identified in Table 3.1.1-1. This list provides input for the event tree task. Event trees, which define the plant response to the given initiator, are developed for each initiator.

The scope of this analysis includes internally initiated events only. That is, those abnormal occurrences within the plant systems which have the potential to cause a plant challenge. This includes internal flooding, which is discussed in the report of the flooding analysis in Section 3.3.7.

3.1.1.1 Initiating Events - Identification

The significant issue in the development of the list of initiating events is completeness. To minimize the possibility of omitting a significant initiator, the task of identifying initiating events is subdivided into three categories.

A. Transients
B. LOCAs
C. Special Initiators

Each task is defined and discussed in the following subsections.

3.1.1.1.1 Transients

For this analysis a transient is defined as any event which requires or results in a reactor scram. The first step involved with identifying a list of PNPP specific transient events is to prepare a list of all transients which have occurred in other BWRs. NUREG/CR-3862 (NRC, 1985a) provides a summary and categorization of all industry transient events and is used as a basis for this analysis. NUREG/CR-3862 updates and validates the transient categorizations identified in EPRI NP-2230 (McClymont, 1982). Thirty-seven BWR Transient Category Definitions were defined in the EPRI report. The same categorization was used in NUREG/CR-3862.

All PNPP post scram restart reports from plant startup through scram 1-90-01 were reviewed to determine the basic cause of scram, plant response following scram and to identify any unique features of the PNPP design. These events were then classified using the EPRI BWR Transient Category Definitions to verify the completeness and applicability of the transient definitions to Perry.

The transient categorization used in NUREG/CR-4550 (Drouin, 1989) for the Grand Gulf analysis was also reviewed to ensure that no unique transient categories were identified for a plant of similar design configuration.

The EPRI BWR Transient Category Definitions as described by NUREG/CR-3862 are listed in Table 3.1.1-2. The review of all Perry scram reports verified that all the transient events which have occurred at Perry were represented by the industry operating experience described by the EPRI BWR Transient Categories. Also, the review determined that the Grand Gulf Initiating Events Identification as described by NUREG/CR-4550 referenced the EPRI BWR Transient Categories as a summary of transient initiating events, and did not identify any additional transient categories.

During the review of the Perry scrams, it was noted that the thirty-seven EPRI BWR Transient Categories provide only three associated with startup and shutdown operation: Low Feedwater Flow During Startup, High Feedwater Flow During Startup or Shutdown, and High Flux Due to Rod Withdrawal at Startup. The Perry IPE analysis considers that a more refined transient categorization would include the classification of reactor power level for each applicable category. The transfer from startup or shutdown operation to power operation was selected to be at 12% reactor power. 12% reactor power corresponds to the limiting reactor power level when the reactor mode switch is placed in "run" during startup operation. As the IPE specifically excludes consideration of initiating events other than those which occur at power, no further investigation was made of events below 12% power.

3.1.1.1.2 LOCAs

Loss of Coolant Accidents (LOCAs) comprise one major class of initiating events. LOCAs are examined because they would cause a plant trip and could require emergency core cooling systems to operate.

A review was made of Probabilistic Risk Assessments (PRAs) performed for the Kuosheng Nuclear Power Station (AEC, 1985), Grand Gulf Nuclear Station (Drouin, 1989), and the General Electric 238 Nuclear Island (GESSAR II, GE, 1988). Kuosheng is a BWR/6-218 and Grand Gulf is a BWR/6-251.

Each of the studies examined at least three different break size ranges defined as Large LOCA, Intermediate LOCA, and Small LOCA. Vessel rupture was also included in each of the studies. LOCAs bypassing the containment were addressed by the General Electric and Kuosheng PRAs. System interfacing LOCAs were addressed by the Grand Gulf PRA (NUREG/CR-4550) and Kuosheng PRA. Grand Gulf included one LOCA not addressed in the other studies reviewed. A leak from the Recirculation pump seals was termed a Small-Small LOCA.

LOCA break sizes are determined by success criteria for the LOCA definitions provided below. The break sizes used by Kuosheng were determined through the use of General Electric (GE) NEDO-24708A (GE, 1980) based on BWR/6-218 plants and calculations made by the computer code MARCH. The Grand Gulf break sizes were scaled up based on GE NEDO-24708A. GESSAR II used the GE licensing codes incorporating best estimate decay heat and modeling of core heat up to account for the steam cooling effects following core uncovery.

Large LOCA

A Large LOCA (Type A) is defined as a break large enough such that the reactor vessel will depressurize and allow the low pressure injection systems to inject into the reactor vessel.

The Kuosheng study defined this as a break of greater than or equal to 0.3 sqft for both liquid and steam breaks as determined from GE NEDO-24708A and MARCH code calculations.

The Grand Gulf study took the data from the same GE report as Kuosheng but scaled the break size up based on a reactor vessel diameter of 251 inches. A break size of greater than or equal to 0.4 sqft was used by Grand Gulf for both liquid and steam breaks.

In the GESSAR II PRA, GE calculated the break sizes to be greater than or equal to 0.5 sqft for a liquid break and greater than or equal to 0.3 sqft for a steam break. This study was based on BWR/6-238 plants.

Since Perry is a BWR/6-238 model plant, break sizes of greater than or equal to 0.5 sqft for a liquid break and greater than or equal to 0.3 sqft for a steam break taken from the GESSAR II PRA will be used for a large LOCA.

Intermediate LOCA

An Intermediate LOCA (Type S1) is defined as a break large enough such that the Reactor Core Isolation Cooling system alone is not sufficient to mitigate the accident and not large enough to depressurize the reactor vessel to where the low pressure ECCS would inject into the reactor vessel.

The Kuosheng PRA used break sizes of between 0.005 and 0.3 sqft for liquid breaks and 0.1 to 0.3 sqft for steam breaks with an assumed Reactor Core Isolation Cooling (RCIC) system flow rate of 600 gpm.

The Grand Gulf study used data from the NEDO-24708A and scaled the break size up based on a reactor vessel diameter of 251 inches. Break sizes of between 0.007 to 0.4 sqft for liquid breaks and 0.13 to 0.4 sqft for steam breaks were used by Grand Gulf with an assumed RCIC flow rate of 825 gpm.

In the GESSAR II PRA, GE calculated the break sizes to between 0.01 to 0.5 sqft for a liquid break and 0.1 to 0.3 sqft for a steam break with an assumed

RCIC flow rate of 700 gpm.

Since Perry is a BWR/6-238 model plant and has the same RCIC flow rate as that used in the GESSAR II PRA, break sizes of 0.01 to 0.5 sqft for a liquid break and 0.1 to 0.3 sqft for a steam break will be used for an intermediate LOCA.

Small LOCA

A Small LOCA (Type S2) is defined as a break small enough such that the Reactor Core Isolation Cooling system alone can maintain the core covered.

The Kuosheng PRA used break sizes of less than or equal to 0.005 sqft for liquid breaks and less than or equal to 0.1 sqft for steam breaks.

The Grand Gulf study used data from the NEDO-24708A and scaled the break size up based on a rector vessel diameter of 251 inches. Break sizes of less than or equal to 0.007 sqft for liquid breaks and less than or equal to 0.13 sqft for steam breaks were used by Grand Gulf.

In the GESSAR II PRA, GE calculated the break sizes to be less than or equal to 0.01 sqft for a liquid break and less than or equal to 0.1 sqft for a steam break.

Since Perry is a BWR/6-238 model plant and has the same RCIC flow rate as that used in the GESSAR II PRA, break sizes of less than or equal to 0.01 sqft for a liquid break and less than or equal to 0.1 sqft for a steam break will be used for a small LOCA. The lower bound for a small LOCA will be taken to be the Technical Specification leakage from the reactor coolant system.

Small - Small LOCA

The Grand Gulf study considered a Small-Small LOCA (Type S3) defined as a recirculation pump seal leak. Although this type of LOCA is easily detected and isolated, because of its higher frequency, it was addressed. Leaks on the order of 50 to 100 gpm could occur on a per pump basis.

At Perry the recirculation pump seals are well instrumented and a seal leak would easily be detected. Applicable PNPP operating instructions list the recirculation pump seals as possible leakage sources. Operating instructions direct operators to trip and isolate the recirculation pump given a seal leak.

It is not expected that the Reactor Protection System would be activated due to the capability of the feedwater system to make up the leakage and the length of time it would take for the Drywell to pressurize sufficiently to reach the reactor scram setpoint.

From NUREG/CR-4550, Table 4.9-26, the mean probability of a Small-Small LOCA occurring is 3.0E-2. Table 4.9-25 provides a probability of failing to detect and isolate the leakage as 1.0E-2. Therefore, the probability that a Small-Small LOCA would exist and not be detected and isolated is 3.0E-4. The probability of a Small LOCA is given by Table 4.9-26 as 3.0E-3. Since the

Small-Small LOCA event is an order of magnitude less than the Small LOCA with the same outcome, event type S3 as defined by NUREG/CR-4550 will not be considered further.

Interfacing System LOCA

An Interfacing System LOCA (Type V) is defined as a break of a high pressure to low pressure interface with the primary system.

The Kuosheng PRA identified the following lines as susceptible to an interfacing LOCA: Low Pressure Core Spray injection line, Shutdown Cooling discharge line (A and B), Shutdown Cooling suction line, Low Pressure Coolant Injection lines (A, B, and C), and the Residual Heat Removal Head Spray line.

The Grand Gulf study identified the Emergency Core Cooling Systems' (ECCS) injection lines and the Reactor Water Cleanup system.

The GESSAR II PRA did not explicitly address the interfacing system LOCA event.

Perry will use the exclusion criteria described in NUREG/CR-5124 (Chu, 1989) as the basis for determining the lines susceptible to interfacing LOCAs. A review of the Perry design was performed with the following lines identified as potentially susceptible to interfacing LOCAs: MSIV Leakage Control lines (A, B, C, and D), RHR Head Spray line, Reactor Core Isolation Cooling suction line, RHR Steam Condensing suction lines, RHR Shutdown Cooling suction line, RHR Shutdown Cooling return lines (A and B), RHR Low Pressure Coolant Injection lines (A, B, and C), Low Pressure Core Spray injection line, Reactor Water Cleanup suction line, Standby Liquid control suction lines, and Feedwater pump suction lines.

Table 3.1.1-3 provides a list of valves which must fail. Each of these valves either receives a closure signal on line break, is normally closed or is a check valve. the closure of any one of the valves in each pathway will prevent high pressure fluid from challenging low pressure piping, thus preventing an interfacing LOCA. The frequency of these failures is evaluated in the data section.

Containment Bypass LOCAs

Containment Bypass LOCAs (Type U) are all LOCAs occurring in high pressure portions of systems that result in the discharge of reactor coolant outside of the containment.

The Kuosheng PRA identified the following lines as potentially susceptible to initiating a LOCA that bypasses the Containment: Main Steam lines (A, B, C, and D), Feedwater lines (A and B), High Pressure Core Spray injection line, Main Steam to RCIC and RHR, and the Reactor Water Cleanup pump suction.

The Grand Gulf study did not explicitly model LOCAs bypassing the Containment as a separate group from Interfacing LOCAs. Several of the ECCS injection lines fall into the category defined by LOCA Bypassing Containment.
A review of the Perry design was made with the following lines identified as being potentially susceptible to a LOCA that bypasses the containment: Main

Steam lines (A, B, C, and D), Main Steam Drain lines (A, b, C, and D), MSIV Leakage Control lines (A, B, C, and D), RCIC injection line, Main Steam to RCIC and RHR, Reactor Water Cleanup suction lines (A and B), Reactor Water Cleanup return lines (A and B), High Pressure Core Spray injection line, Standby Liquid Control injection lines (A and B), and Feedwater lines (A and B).

Table 3.1.1-4 identifies the valves which must fail. Each of these valves either receives a closure signal on line break, is normally closed or is a check valve. The closure of any one of the valves in each line up will isolate a LOCA that bypasses containment.

Reactor Vessel Rupture

LOCAs from Reactor Vessel ruptures (Type R) can be divided into two categories. In the first category, the Reactor Vessel ruptures are of a size and in a location such that they are essentially equivalent to a large, intermediate, or small pipe break. This type is covered in the large, intermediate, and small LOCA initiators. In the second category the Reactor Vessel ruptures are greater than the size of a double-ended recirculation line break or are located in the region below the core shroud.

The above definition comes from the Grand Gulf study The Kuosheng and GESSAR II PRAs simply postulate a random Reactor Vessel rupture that exceeds the capability of the ECCS systems.

Perry will use the Grand Gulf definition with the location being top of the jet pump instead of below the core shroud. This provides a reflood of 2/3 core height capability.

3.1.1.1.3 Special Initiators

In addition to the evaluation of transient and LOCA initiator categories, a failure modes and effects analysis (FMEA), of support systems was conducted. The purpose of the FMEA was to identify special initiators unique to the PNPP design. A special initiator is defined as any internal initiating event which will:

- Cause a reactor scram.
- Is additionally capable of degrading an accident mitigation system required in response to the initiator.

The FMEA was divided into five categories addressing the loss of both safety and non-safety support systems. The scope of each category is defined below:

- Electrical Systems: (Safety and Non-safety)
  - 13.8 KV Buses
  - 4.16 KV Buses
  - 480 VAC Buses
  - 120 VAC Buses
  - 125 VDC Buses
- Air Systems: (Safety and Non-safety)

- Instrument Air System (P52)
- Service Air System (P51)
- Safety Related Instrument Air System (P57)

- Water Systems: (Safety and Non-safety)

  - Service Water System (P41)
  - Emergency Service Water System (P45)
  - Nuclear Closed Cooling System (P43)
  - Emergency Closed Cooling System (P42)
  - Turbine Closed Cooling system (P44)
  - Turbine Building Chilled Water System (P46)
  - Control Complex Chilled Water System (P47)
  - Containment Vessel Chilled Water (P50)

- HVAC Systems: (Safety and Non-safety)

  - Containment Vessel Cooling System (M11)
  - Drywell Cooling System (M13)
  - Controlled Access and Misc. Equip. Area HVAC (M21)
  - MCC, Switchgear and Misc. Elec. Areas HVAC (M23)
  - Battery Room Exhaust (M24)
  - Control Room HVAC and Recirculation System (M25/M26)
  - Computer Room HVAC (M27)
  - Emergency Closed Cooling Pump Area Cooling (M28)
  - ECCS Pump Room Cooling system (M39)
  - Steam Tunnel Cooling System (M47)
  - Turbine Power Complex Ventilation System (M42)

- Control Cabinets

Plant design information from the PNPP Off-normal Instructions (ONIs), System Operating Instructions (SOIs), Alarm Response Instructions (ARIs), P&IDs, Electrical One-line Diagrams, post scram reports, Licensee Event Reports (LERs) design calculations, System Description Manuals (SDMs), walkdowns and discussion with operations support personnel provided the basis for the evaluation. The evaluation process assumed the complete loss of each of subject support system. Plant response was characterized from the references described above.

Special Initiators Identified:

From this review the following special initiators were retained for further analysis:

- Loss of Instrument Air
- Loss of Service Water

Loss of Instrument Air

Loss of Instrument Air would cause a plant scram through the dependency of multiple plant systems upon instrument air. A reactor scram would be expected due to the MSIV closure caused by loss of air, or on high water level in the scram discharge volume (SDV) caused by drifting of the scram valves and the closure of the vent and drain valves.

The MSIVs will close on decreasing air pressure causing a loss of the Power Conversion System (PCS). Loss of the Steam Jet Air Ejectors will also lead to a loss of the PCS. Due to various system dependencies the loss of instrument air will also lead to the loss of make-up air to the non-ADS SRVs, probable loss of feedwater flow, loss of the RHR steam condensing mode and loss of CRD as an alternate flow injection source.

For the non-ADS SRVs immediate operability will not be affected due to the accumulators which provide a source of backup air. The MSIVs are also backed up by accumulators. Feedwater flow is expected to be lost due to the lockup of the Hot Surge Tank level control valves, the failing open of the feedwater and feedwater booster pump recirculation valves and the lockup of the Motor Feedpump flow control valve. The CRD flow control valve will fail closed on loss of air causing a decrease in the drive water flow to the CRDs.

Based on the dependencies described above, the loss of instrument air meets the criteria to be considered as a special initiator.

Loss of Service Water

The loss of Service Water event will lead to the loss of Steam Tunnel Cooling event since Service Water provides the ultimate cooling source for the Steam Tunnel Cooling System. This results in a reactor scram on high steam tunnel temperature and MSIV closure. In addition RCIC isolation would occur after a delay of 30 minutes. A reactor scram would also be expected due to the loss of stator cooling water (turbine trip) or on a high drywell temperature due to loss of Drywell Cooling. A loss of feedwater and CRD flow would also be expected as both systems would be required to be manually shutdown due to loss of component motor bearing and/or lube oil cooling. The Service Air/Instrument Air compressors would also be expected to trip on high temperature. This event is treated as a special initiator.

Special Initiators Eliminated by Screening:

Through the process of the failure modes and effects (FMEA) analysis, several initiators which may have been considered for analysis in other PRAs or which may have initially appeared significant were eliminated. Such considerations are addressed below.

The review of plant electrical systems did not identify any bus loss which would meet the criteria established in this study for a special initiator. Electrical loads are distinctly separated between safety and non-safety loads. No single loss of a safety bus was identified which would cause a reactor scram. Additionally, loss of a safety bus would not cause more than the loss of one division of plant safety systems. Several non-safety buses were identified that would cause a reactor scram due to variety of plant responses. In fact some are documented by actual plant scrams. However, none were identified which would cause a loss of an accident mitigation system other than feedwater or the PCS. Each of these events are already identified as transient initiating events (T2 and T3B).

The loss of Safety Related Instrument Air is not a special initiator. The only loads provided by this system are the ADS accumulators and one non-ADS SRV. Loss would not result in a reactor scram or the immediate loss of any accident mitigation system.

The loss of the various plant water systems identified only the loss of Service Water (P41) as a special initiator. Loss of the safety related service water systems, including Emergency Service Water (P45) and Emergency Closed Cooling (P42), were not included as special initiators as neither would cause a reactor scram upon system loss.

The HVAC systems were reviewed for special initiators. None were retained. The loss of Control Room HVAC was reviewed, however it met neither of the two criteria for special initiators. A review of engineering calculations indicated that under realistic analysis criteria (i.e., credit for heat sinks, realistic heat loads) that the control room heatup would remain below the control room qualification temperature of 120 degrees Fahrenheit, thus not challenging the capability of control room equipment. Additionally, the remote shutdown room, cooled by a separate HVAC system would be available to provide control capability to achieve safe shutdown.

The loss of the MCC, Switchgear and Miscellaneous Electrical Equipment Area HVAC was reviewed. Two redundant, divisionally separate loops, each with redundant fans, are available to individually provide 100% room cooling. Assuming the loss of all cooling, no interaction was identified which would cause a reactor scram or requirement for immediate shutdown. Additionally, engineering calculations indicated a very long time lag (>24 hrs)prior to any exceedance of equipment temperature qualification limits. It is assumed that corrective action could be taken prior to any potentially significant equipment loss.

The loss of Control Complex Chilled Water (CCCW) does not meet the specific criteria of a special initiator since no immediate automatic or manual scram is expected or required. However, due to a Technical Specification requirement for plant shutdown within six hours upon the loss of Control Room HVAC (which is a load supplied by CCCW) this event will lead to reactor shutdown. However recent analysis has confirmed that the Switchgear room heatup will not lead to loss of functioning equipment, therefore all ECCS will be available and this event does not need to be treated as a special initiator.

The failure modes and effects analysis (FMEA), included a limited look at control cabinets to identify any potential special initiators. The review scope and criteria included the following:

-Only Control Room cabinets were reviewed, excluding the "horseshoe control panels.

-Each cabinet was reviewed to ensure that each cabinet was either divisionally separated or that it controlled only one plant "function". Examples of plant functions include, RPV level control, RPV Pressure control, HVAC, etc.

- The analysis consisted of a Control Room walkdown and review of panel drawings.

The results of this analysis indicated that all cabinets were either divisionally or functionally separated. No special initiators were identified.

3.1.1.2 Initiating Events - Grouping

The process for transient identification was described in section 3.1.1.1 and the results are provided in Table 3.1.1-5. Each of these transients were reviewed and grouped according to plant response. This grouping reduces the total number of events which must be individually analyzed, while retaining the detail of the individual events. Table 3.1.1-5 identifies the criteria and rationale for grouping each of the transient events identified in Table 3.1.1-2 into the final initiating event group.

By comparison to NUREG/CR-4550, several of the transients which for Grand Gulf were categorized as transients with loss of PCS (T2), were recategorized as loss of feedwater transients (T3B) for Perry. This is due to the Perry specific design which causes a trip of the RFP turbines on a high RPV water level (L8).

By their definition LOCA, special initiators and internal flooding each has a unique plant response. No grouping of these initiators is required or possible.

3.1.2 FRONTLINE EVENT TREES

This section discusses the development of event trees for each initiating event as identified in section 3.1.1 of this report (see Table 3.1.1-1) and the acceptance criteria used to define success or failure in a given function to prevent core damage following an initiating event and the front line and support systems which will perform those functions.

3.1.2.1 Acceptance Criteria

Fuel Boundaries

In the IPE we are concerned with two levels of modeling, the identification of core damage, and the risk posed by the release of fission products following core degradation and melting. For the core damage analysis the conditions defined are very similar to that defined in the Grand Gulf study (NUREG/CR-4550). For Perry, core damage is defined as exceedance of any of the fuel boundary criteria in the USAR or an occurrence of significant core uncovery such that the water level falls below -112 inches and there is no expectation of imminent reflooding of the core. As the bottom of the active fuel is at -150 inches this is slightly higher that that defined in the Grand Gulf study (24 inches above its bottom of the active fuel). However the timing difference between reaching the level of -112 compared with -126 is so small that this will make negligible difference to the assessed core damage frequency. The result in both cases is prolonged uncovery of the core which leads to damaged fuel and an expected release of fission products.

Reactor Coolant Boundary

The acceptance criteria for maintaining integrity of the reactor coolant boundary is based on not exceeding the corresponding American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Level C service limit stress criterion during any transient. The number of safety relief valves required to open for a given event is based on this criterion.

Containment Building Integrity

Analysis performed by Gilbert Commonwealth (G/C, 1992) showed that the Containment Capacity Threshold Limit (i.e. a 1 percent probability of containment failue) for the Perry containment is 50 psig. This has been taken as the basis for the timing of the requirement of long term containment heat removal.

System Acceptance Criteria

Besides the above overall acceptance criteria, there are also specific acceptance criteria for components and systems. Two of them which have an influence on the construction of the event and fault trees and the development of the success criteria are:

1. Pumped fluid temperatures for pumps connected to the suppression pool.

2. Maximum ambient temperature for all safety equipment.

The first of these is addressed in developing the various event trees for initiating events which are going to result in injection systems taking suction from the suppression pool and the second in evaluating the performance of each system in response to the various initiating events.

3.1.2.2 Front Line and Support Systems

The basic requirements for prevention of core damage and prevention or mitigation of the release of fission products can be divided into the following four functions.

1. Reactivity Control
2. Reactor Coolant System Overpressure Protection
3. Emergency Core Cooling
4. Containment Overpressure Protection and Fission Product Control.

For a given accident initiating event the systems that directly perform one or more of these functions are defined as front line systems. Support systems are those that affect the course of the sequence only by means of their effect on the operation of a front line system. The list of front line and support systems is given in Table 3.1.2-1 and the dependency between them in Table 3.1.2-2. The dependency of some support systems on other support systems is shown in Table 3.1.2-3. The level of support varies from dependency to dependency. For example, cooling dependency is continuous. But power may only be required for a very short period of time when system initiation is required. Room cooling may only be required when a system is operating and the system may be able to operate for an extended period of time without room cooling. These requirements are identified in the development of the event and fault trees.

3.1.2.3 Event Tree Development

The event tree model is the central analytical tool used in the determination of the frequency of core damage, and the various ways in which it can occur. As the principles of its development are well documented in the PRA Procedures Guide (NRC, 1983), and the Interim Reliability Evaluation Program Procedures Guide (NRC, 1982a), they are not described in detail in this report. Some discussion is, however, provided on those aspects of the development of the event trees that are specific to the present study.

The initiating event task defined the LOCA and transient initiating events, grouped according to the major functions required to prevent core damage.

The first task in the development of the event tree is to define the frontline system requirements to meet the acceptance criteria for each of the functions defined in section 3.1.2.2. The second task is to identify the impact of the initiating event on the frontline or support systems required to perform each of the functions.

The most important aspect of developing the event trees from the above information is to reflect the inherent functional and physical dependencies between each phase of the sequence, and at the same time, the interaction between operators and systems as the sequence unfolds. Thus the event tree is developed by first considering those functions (reactivity control, overpressure protection) that are required early, and then those which are required in the long term. In this way, it is relatively straight forward to model dependencies between functions. For example, in the case of a large LOCA, injection is required immediately. If injection fails, it will not be necessary to consider long term containment heat removal in the context of core damage, although it will be considered later in terms of the impact of its loss on long term containment integrity. This is discussed in section 3.1.4.

Operator interactions, such as cooling down and depressurizing the RPV when called for in the emergency procedures, by use of the ADS either in the automatic or manual mode, are specifically identified, so that the relationship between the success or failure of the operators interaction can be clearly identified.

Finally, it will be noted that for initiating events that are not related to loss of a support system, the event tree addresses only systems which perform the functions identified in Section 3.1.2.2. The dependency of each of these systems on the various support systems is developed in the analysis of individual systems. In the case of loss of a support system, such as loss of offsite power, recovery of the support system is included in the event trees. The results of this phase of the study are the identification of the individual core damage sequences, and the detailed analysis requirements for determining the timing and progression of each accident sequence. The timing is required in order to evaluate the impact of the operator actions, and the time of occurrence of the automatic systems initiation signals.

The front line event trees described in this section are for the following initiating events.

Transients with loss of the Power Conversion System	T2
Transients with Power Conversion System initially available	T3A
Transients involving loss of feedwater, but with the Power Conversion System initially available	T3B
Large Loss of Coolant Accident	A
Intermediate Loss of Coolant Accident	S1
Small Loss of Coolant Accident	S2

3.1.2.4 TRANSIENT WITH A LOSS OF PCS EVENT TREE

3.1.2.4.1 General Description

Transients with a loss of the Power Conversion System (PCS) are defined as class T2 transients, consistent with the nomenclature utilized in NUREG/CR-4550.

The T2 transient is characterized by the occurrence of an event or action which causes a loss of the PCS. A plant scram due to a MSIV isolation or a loss of condenser vacuum are examples of such a transient.

Table 3.1.1-5 identifies the EPRI classified transients which comprise this transient category.

For the purpose of this analysis, the PCS is defined as the main steam system, the turbine bypass system, the main condenser and its primary supporting systems, (i.e., Circulating Water System, the Steam Jet Air Ejectors, etc.).

3.1.2.4.2 Success Criteria

Success criteria for the loss of PCS transient is listed in Table 3.1.2-4. The incorporation of these success criteria into the event tree is discussed in the following section.

3.1.2.4.3 Event Tree

The event tree for this transient is provided on Figure 3.1.2-1 sheets 1-3. Heading definitions are provided below in the approximate chronological order that would be expected following a transient with a loss of PCS.

T2 A loss of PCS transient occurs which disrupts the normal operation of the plant and requires mitigation.

C Success or failure of RPV reactivity control with RPS automatic or manual scram or ARI. Success implies that a sufficient number of control rods have inserted into the core to provide a shutdown rod pattern. Thus, a successful reactor scram has occurred. Failure implies the rods have not inserted to a subcritical rod configuration. Failure sequences transfer to the ATWS event tree.

P1 Success or failure of RPV pressure control with the SRVs opening and reclosing. Success implies that the required SRVs opened and reclosed. Failure implies that of the SRVs which successfully opened, one failed to reclose. Failure sequences are developed in a transfer event tree. The sizing of the SRVs is such that one open SRV results in leakage and depressurization of the RPV at the same rate as in the small LOCA. The success criteria is the same as the small lOCA and the event tree (Sheet 2) is developed accordingly.

P2 Success or failure of RPV pressure control with the SRVs opening and reclosing. Success implies that the required SRVs opened and reclosed. Failure implies that of the SRVs which successfully opened, two failed to reclose. Failure sequences are developed in a transfer event tree. The sizing of the SRVs is such that two open SRVs results in leakage and depressurization of the RPV at the same rate as the intermediate LOCA. The success criteria is the same as for an intermediate LOCA and the event tree (Sheet 3) is developed accordingly.

U3 Success or failure of RPV level control with the Motor Feedpump. Success implies that the Motor Feedpump (MFP) is manually initiated by the plant operators and adequate RPV level control is maintained. Failure implies that the Motor Feedpump could not be, or was not started such that RPV level control cannot be maintained with the MFP.

U2 Success or failure of RPV level control with the RCIC system. Success implies that either RCIC automatically actuated at RPV level 2 or that it was manually actuated and is providing adequate RPV level control. It further implies that long-term containment heat removal with RHR in Suppression Pool Cooling mode successfully maintains containment conditions for long-term operability of RCIC. Failure implies that RCIC is not maintaining RPV level control or that long-term containment heat removal is not maintained, resulting in the loss of RCIC.

U1 Success or failure of RPV level control with the HPCS system. Success implies that either HPCS automatically actuated at RPV level 2 or that it was manually actuated and is providing adequate RPV level control. Failure implies that RPV level control cannot be maintained with HPCS.

X Success or failure of emergency RPV depressurization. Success implies that the ADS was automatically or manually initiated or that the operator manually depressurized the vessel with at least 4 SRVs. Failure implies that ADS failed and that the operator failed to manually emergency depressurize with the required number of SRVs causing the reactor vessel to remain at a higher pressure than that required by the PNPP Plant Emergency Instructions (PEIs) for successful RPV level control with low pressure injection.

V Success of failure of RPV level control with the ECCS low pressure make-up systems. Success implies that, at a minimum, LPCS or one loop of LPCI either automatically initiated or was manually placed in service and is providing adequate RPV level control. Failure implies that RPV level control cannot be maintained by an ECCS low pressure injection system.

Va Success or failure of RPV level control with the Condensate Transfer Alternate Injection alignment. Success implies, that the CTS alternate injection alignment was manually started by the plant operators and that the alignment successfully provided RPV level control. Failure implies that CTS alternate injection was not aligned or that it was unable to maintain RPV level control.

W Success or failure of long-term containment heat removal with RHR to maintain containment pressure below the Containment Capacity Threshold Limit of 50 psig. Success implies that long-term heat removal can be achieved by at least one of the following modes of RHR. Failure implies that no long-term containment heat removal mode of RHR was aligned or that pressure cannot be maintained below the Containment Capacity Threshold Limit.

- o One RHR train in Suppression Pooling Cooling mode.
- o One RHR train in the Containment Spray mode.

Y Success or failure of long-term containment heat removal by venting of the containment. Success implies that one of the below listed venting paths were aligned, per the PEIs, prior to containment pressure exceeding the Containment Capacity Threshold Limit of 50 psig and that the aligned path maintains pressure below that limit. Failure implies that no path was successfully aligned or that the Containment Capacity Threshold Limit was exceeded.

- o Fuel Pool Cooling and Cleanup System. Manual alignment of vent path through the fuel pool skimmers into the Fuel Handling Building.
- o RHR Containment Spray Loop A or B. Manual alignment of vent path through the containment spray header into the Fuel Handling Building.

Cv Defines the susceptibility of the core to damage, dependent upon containment conditions. Success implies that given containment failure, the core is still maintained in a safe configuration. Failure implies that the containment failure has led to a degradation of plant systems such that the core is susceptible to damage. The evaluation of this function is discussed in Section 4.4.

The development of the functional fault tree for each of these functions with the appropriate system boundary conditions defined by the initiating event is described in Appendix E.

3.1.2.5 TRANSIENT WITH PCS AVAILABLE EVENT TREE

3.1.2.5.1 General Description

Transients with the Power Conversion system (PCS) available are defined as class T3A transients, consistent with the nomenclature utilized in NUREG/CR-4550.

For the purpose of this analysis, the PCS is defined as the main steam system, the turbine bypass system, the main condenser and its primary supporting systems. (i.e., Circulating Water System, the Steam Jet Air Ejectors, etc.)

The T3A transient is characterized as a "typical" plant scram (i.e., all systems and functions are potentially available to respond to the transient, including the feedwater system and the PCS).

Table 3.1.1-5 identifies the EPRI classified transients which comprise this transient class. Plant scrams due to a turbine trip or manual scram are examples of such a transient.

3.1.2.5.2 Success Criteria

Success criteria for the PCS available transient (T3A) is listed in Table 3.1.2-5. The incorporation of these success criteria into the event tree is discussed in the following section.

3.1.2.5.3 Event Tree

The event tree for this transient is provided on Figure 3.1.2-2 Sheets 1-3. Heading definitions are provided below.

T3A A transient with PCS available occurs which disrupts the normal operation of the plant and requires mitigation.

Q success or failure of the PCS to remain available. Success implies that the PCS is available to provide RPV pressure control and to remove decay heat from the RPV for 24 hours following the initiation of the transient. Failure implies that it is not available. This transfers to the transient with loss of PCS event tree.

U3 Success or failure of RPV level control with the Feedwater system. Success implies that one of two Reactor Feedpumps (RFPs) or one Motor Feedpump (MFP) started or continued to run and provided successful RPV level control. Failure implies that the feedwater system could not maintain RPV level control.

U1 Success or failure of RPV level control with the HPCS system. Success implies that either HPCS automatically actuated at RPV level 2 or that it was manually actuated and is providing adequate RPV level control. Failure implies that RPV level control cannot be maintained with HPCS. As PCS is available, no containment heat removal is required following successful HPCS operation.

The remaining functions C, P1, P2, U2, X, V, Va, W, Y, and Cv are as described above in the previous section for the T2 transient. The development of the functional fault trees for each of the functions with the appropriate boundary conditions is described in Appendix E.

3.1.2.6 LOSS OF FEEDWATER TRANSIENT EVENT TREE

3.1.2.6.1 General Description

Transients with a loss of feedwater are defined as class T3B transients, consistent with the nomenclature of NUREG/CR-4550.

The T3B transient is characterized as a transient with the Power Conversion System (PCS) available, further characterized by the loss of the feedwater system.

For the purpose of this analysis, the loss of feedwater is defined as loss of the two normally operating Reactor Feedpumps (RFPs). The Motor Feedpump (MFP) is assumed to be able to be recovered by operator action. The Reactor Feedwater Booster Pumps and the condensate system are assumed available to support MFP operation for this transient.

The complete loss of feedwater (including the loss of the MFP) is modeled in the T3B event tree path in which recovery with the MFP has failed. (Failure of function U3, discussed below).

The PCS is defined as being available for this transient. The PCS is defined as the main steam system, the turbine bypass system, the main condenser and its primary supporting systems, (i.e., Circulating Water System, the Steam Jet Air Ejectors, etc.)

Table 3.1.1-5 identifies the EPRI classified transients which comprise this transient class.

3.1.2.6.2 Success Criteria

Success criteria for the loss of feedwater transient (T3B) is listed in Table 3.1.2-6.

3.1.2.6.3 Event Tree

The event tree for this transient is provided in Figure 3.1.2-3. Heading definitions are provided below. As by definition the PCS remains available the SRVs are not challenged, therefore events P1, P2, are excluded from the tree.

T3B A loss of feedwater transient occurs which disrupts the normal operation of the plant and requires mitigation.

Q Success or failure of the PCS to remain available. Success implies that the PCS is available to provide RPV pressure control and to remove decay heat from the RPV for 24 hours following the initiation of the transient. Failure implies that it is not available.

U3 Success or failure of RPV level control with the Motor Feedpump (MFP). Success implies that the MFP is manually initiated by the plant operators and adequate RPV level control is maintained. Failure implies that the Motor Feedpump could not be, or was not started such that RPV level control cannot be maintained with the MFP. As PCS is available, no containment heat removal is required following successful MFP recovery.

U1 Success or failure of RPV level control with the HPCS system. Success implies that either HPCS automatically actuated at RPV level 2 or that it was manually actuated and is providing adequate RPV level control. Failure implies that RPV level control cannot be maintained with HPCS. As PCS is available, no containment heat removal is required following successful HPCS operation.

The remaining functions C, U2, X, V, Va, W, Y and Cv are the same as those described in the T2 event tree in section 3.1.2.4. The development of the functional fault tree for each of these functions with the appropriate system boundary conditions defined by the initiating event is described in Appendix E.

3.1.2.7 LARGE LOCA EVENT TREE

3.1.2.7.1 General Description

The Large LOCA event is defined as a class A event, consistent with the nomenclature utilized in NUREG/CR-4550.

A large LOCA is defined as a break large enough such that the reactor pressure vessel (RPV) will depressurize without the assistance of Automatic Depressurization System (ADS) or the Safety Relief Valves (SRVs) and permit the low pressure injection systems to inject into the reactor vessel.

This event is expected to result in a rapid loss of RPV water inventory, resulting in a trip of the reactor and the main turbine generator (RPV level 3) and the loss of the power conversion system (PCS) due to closure of the Main Steam Isolation Valves (MSIVs), (RPV level 1).

Only Emergency Core Cooling Systems (ECCS) are immediately available for injection of water into the RPV. The Reactor Core Isolation Cooling System (RCIC) is not available due to the absence of steam pressure to drive the RCIC turbine.

3.1.2.7.2 Success Criteria

The success criteria for a large lOCA is provided in Table 3.1.2-7.

Depressurization of the reactor vessel and closure of the MSIVs eliminates use of the turbine driven feedwater pumps as a source of water to the reactor vessel.

3.1.2.7.3 Event Tree

The event tree for this event is provided on figure 3.1.2-4. Heading definitions are provided below.

A A large LOCA occurs which disrupts the normal operation of the plant and requires mitigation.

C Success or failure of RPV reactivity control with RPS automatic or manual scram or ARI. Success implies that a sufficient number of control rods have inserted into the core to provide shutdown rod pattern. Thus, a successful reactor scram has occurred. Failure implies the rods have not inserted to a subcritical rod configuration.

Failure sequences transfer to the ATWS event tree.

U1 Success or failure of RPV level control with the HPCS system. Success implies that either HPCS automatically actuated at RPV level 2 or that it was manually actuated and is providing adequate RPV level control. Failure implies that RPV level control cannot be maintained with HPCS.

V Success or failure of RPV level control with the ECCS low pressure make-up systems. Success implies that, at a minimum, LPCS or one loop of LPCI either automatically initiated or was manually placed in service and is providing adequate RPV level control. Failure implies that RPV level control cannot be maintained by an ECCS low pressure injection system.

W Success or failure of long-term containment heat removal with RHR to maintain containment pressure below the Containment Capacity Threshold Limit of 50 psig. Success implies that long-term heat removal can be achieved by at least one of the following modes of RHR. Failure implies that no long-term containment heat removal mode of RHR was aligned or that pressure cannot be maintained below the Containment Capacity Threshold Limit.

- o One RHR train in Suppression Pool Cooling mode.
- o One RHR train in the Containment Spray mode.

Y Success or failure of long-term containment heat removal by venting of the containment. Success implies that one of the below listed venting paths were aligned, per the PEIs, prior to containment pressure exceeding the Containment Capacity Threshold Limit of 50 psig and that the aligned path maintains pressure below that limit. Failure implies that no path was successfully aligned or that the containment limit was exceeded.

- o Fuel Pool Cooling and Cleanup System. Manual alignment of vent path through the fuel pool skimmers into the Fuel Handling Building.
- o RHR Containment Spray Loop A or B. Manual alignment of vent path through the containment spray header into the Fuel Handling Building.

Cv Defines the susceptibility of the core to damage, dependent upon containment conditions. Success implies that given containment failure, the core is still maintained in a safe configuration. Failure implies that the containment failure has led to a degradation of plant systems such that the core is susceptible to damage.

The vacuum breakers are normally closed motor operated valves so as in the case of the Grand Gulf study the potential for suppression pool bypass was omitted from the event.

Calculations were performed to determine if suppression pool make up was required to provide vapor suppression, i.e. to maintain the containment pressure below the containment failure pressure. It was established that suppression pool make-up is not required either to maintain pump NPSH or prevent vortex limits to be exceeded. It was also shown that containment design pressure would not be exceeded. These analyses are reported in the analysis files developed for this task.

The development of the functional fault trees for each of the functions with the approximate system boundary conditions defined by the initiating event and subsequent system successes or failures is described in Appendix E.

3.1.2.8 INTERMEDIATE LOCA EVENT TREE

3.1.2.8.1 General Description

The intermediate LOCA event is defined as a class S1 event, consistent with the nomenclature utilized in NUREG/CR-4550.

An intermediate LOCA is defined as a break large enough such that the Reactor Core Isolation Cooling (RCIC) alone is not sufficient enough to mitigate the accident and not large enough to depressurize the reactor vessel to where the low pressure ECCS would inject into the RPV.

This event is expected to result in a rapid loss of RPV water inventory, resulting in a trip of the reactor and the main turbine generator (RPV level 3) and the loss of the PCS due to closure of the Main Steam Isolation Valves (MSIVs), (RPV level 1).

Only Emergency Core Cooling Systems (ECCS) are immediately available for injection of water into the RPV. RCIC capacity is defined as insufficient for the event.

3.1.2.8.2 System Success Criteria

System Success Criteria for an Intermediate LOCA are listed in Table 3.1.2.8.

During an Intermediate LOCA the condensate/feedwater systems would attempt to maintain water level in the reactor vessel but would be unable to do so prior to a reactor scram. Eventually a trip of the reactor and the main turbine generator would be received and the loss of PCS due to a closure of the Main Steam Isolation Valves would be experienced. Depressurization of the reactor vessel by means of the Main Steam Safety Relief Valves is necessary for the operation of the low pressure systems to mitigate the accident. Alternate injection from Condensate Transfer Alternate Injection alignment will provide adequate low pressure injection.

3.1.2.8.3 Event Tree Heading Definition

The event tree for this event is provided on Figure 3.1.2-5. Heading definitions are provided below.

S1 An intermediate LOCA occurs which disrupts the normal operation of the plant and requires mitigation.

X Success or failure of emergency RPV depressurization. Success implies that the ADS was automatically or manually initiated or that the operator manually depressurized the vessel with at least 2 SRVs. Failure implies that ADS failed and that the operator failed to manually emergency depressurize with the required number of SRV causing the reactor vessel to remain at a higher pressure than that required by the PNPP Plant Emergency Instructions (PEIs) for successful RPV level control with low pressure injection.

Va Success or failure of RPV level control with the Condensate Transfer Alternate Injection alignment. Success implies, that the CTS alternate injection alignment was manually started by the plant operators and that the alignment successfully provided RPV level control. Failure implies that CTS alternate injection was not aligned or that it was unable to maintain RPV level control or that the ESW cross tie was not used.

The remaining functions C, U1, V, W, I, and Cv are the same as those described in the previous section for the type A event tree.

The vapor suppression function is not modeled as calculations performed to determine if it was necessary showed that suppression pool make-up would not be required. This is summarized in the analysis files for this task.

The development of the functional fault trees for each of the event tree functions with the appropriate boundary conditions defined by the initiating event and preceding function success or failures is described in Appendix E.

3.1.2.9 SMALL LOCA EVENT TREE

3.1.2.9.1 General Description

The small LOCA event is defined as a class S2 event, consistent with the nomenclature utilized in NUREG/CR-4550.

The small LOCA event is defined as a break small enough such that the RCIC system alone can maintain the core covered. However, the break is defined as larger in size than that which would be handled in an orderly plant shutdown. A small LOCA will not depressurize the reactor vessel fast enough, without the assistance of the SRVs, to prevent uncovery of the core before the low pressure systems inject into the vessel.

The event is expected to cause only minor perturbation to the RPV water level. The capacity of feedwater system is more than sufficient to make-up the [illegible]ory [illegible] due to the break. However, automatic reactor scram is ex[illegible] [illegible] occur due to a high drywell pressure or due to low RPV water level [illegible] 1 3), if feedwater fails to provide level control.

3.1.2.[illegible] System Success Criteria

System success criteria for a Small LOCA are listed in Table 3.1.2.9.

During a Small LOCA the condensate/feedwater systems can maintain water level in the reactor vessel prior to a reactor scram. Eventually a trip of the reactor and [illegible] main turbine generator would be received and the loss of PCS due to a [illegible] of the Main Steam Isolation valves would be experienced.

RCIC and HPCS inject into the vessel to recover reactor vessel level. Depressurization of the reactor vessel by means of the Main Steam Safety Relief Valves is necessary for operation of the low pressure systems to mitigate the accident. An alternate source of water is the Condensate Transfer Alternate Injection.

3.1.2.9.3 Event tree

The event tree for this event is provided on Figure 3.1.2-6. Heading definitions are provided below.

S2 A small LOCA occurs which disrupts the normal operation of the plant and requires mitigation.

U3 Success or failure of RPV level control with the feedwater system. Success implies that one of two Reactor Feedpumps (RFPs) or one Motor Feedpump (MFP) starts or continues to run and provide successful RPV level control. Failure implies that the feedwater system cannot maintain RPV level control.

U2 Success or failure of RPV level control with the RCIC system. Success implies that either RCIC automatically actuates at RPV level 2 or that it is manually actuated and is providing adequate RPV level control. It further implies that long-term containment heat removal with RHR in Suppression Pool Cooling mode successfully maintains containment conditions for long-term operability of RCIC. Failure implies that RCIC is not maintaining RPV level control or that long-term containment heat removal is not maintained, resulting in the loss of RCIC.

X Success or failure of emergency RPV depressurization. Success implies that the ADS was automatically or manually initiated or that the operator manually depressurized the vessel with at least 3 SRVs. Failure implies that ADS failed and that the operator failed to manually emergency depressurize with the required number of SRV causing the reactor vessel to remain at a higher pressure than that required by the PNPP Plant Emergency Instructions (PEIs) for successful RPV level control with low pressure injection.

Va Success or failure of RPV level control with the Condensate Transfer Alternate Injection alignment. Success implies, that the CTS alternate injection alignment was manually started by the plant operators and that the alignment successfully provided RPV level control. Failure implies that CTS alternate injection was not aligned or that it was unable to maintain RPV level control or the ESW cross tie.

The remaining functions C, U1, V, W, Y and Cv are the same as those described in the type A event tree in section 3.1.1.7.

For the small LOCA, failure of the Vapor Suppression System was considered to be a very low frequency event and therefore not included in the event tree. With a small LOCA, the RPV will not immediately depressurize and only a high pressure system would be operating. Flow would be limited by the break size. Drawdown of the suppression pool would not occur quickly and even if all high pressure systems fail multiple low pressure systems are available to provide water from external sources.

The development of the functional fault trees for each of the event tree functions with the appropriate boundary conditions defined by the initiating event and preceding function success or failures is described in Appendix E.

3.1.3 SPECIAL EVENT TREES

The definition of the Acceptance Criteria and methodology for developing the event trees is described in section 3.1.2.3. The same criteria and methodology is used to develop the event trees for the following special initiating events.

Loss of Offsite Power and Station Blackout	(T1/B)
Transient with Inadvertent Open Relief Valve	(T3C)
Loss of Instrument Air	(TIA)
Loss of Service Air	(TSW)
Anticipated Transient Without Scram	(ATWS)

No event trees have been developed for initiating events

Interfacing System LOCA	(V)
Containment Bypass (High Pressure System) LOCA	(O)
Vessel Rupture	(R)

Each of these events for which a tree has been developed are discussed in the following section.

3.1.3.1 LOSS OF OFFSITE POWER EVENT TREE

3.1.3.1.1 General Description

Loss of offsite power transients are defined as class T1 transients, consistent with the nomenclature utilized in NUREG/CR-4550.

Offsite AC power is normally supplied from the 345 kV grid. The onsite emergency diesel generators are designed to supply AC power to the Engineered-Safety-Features (ESF) systems in the case of failure of all offsite power.

A loss of offsite power causes a trip of the reactor and the main turbine generator and the loss of the PCS due to the closure of the main steam isolation valves. Only Emergency Core Cooling Systems (ECCS) and the Reactor Core Isolation Cooling (RCIC) System are immediately available for injection. Alternate injection can be provided by the diesel driven fire pump prior to restoration of offsite power. Other alternate injection alignments can be provided from additional systems only when the offsite power is restored.

The Perry plant has three electrical divisions. Division 1 provides power for the LPCS system and to Loop A of the RHR system, and is essential for long-term power to the DC power system, which provides control power for RCIC. Division 2 provides power for the remaining two loops of the RHR system, and Division 3 is dedicated to the HPCS system. If the Division 1 and 2 diesel generators fail to start or fail while running before offsite power is restored, station blackout occurs.

3.1.2.1.2 Success Criteria

Success criteria for Loss of Offsite Power is listed in Table 3.1.3-1.

The condensate/feedwater pumps have not been included in the Emergency Core Cooling success criteria upon power recovery, due to water hammer concerns on system restart. For a Hotwell pump start, the discharge valve is required to be manually throttled down. After the initial recovery portion of the event, the operators would be reluctant to start a Reactor Feedwater Booster Pump due to the potential for voiding in the feedwater header.

3.1.2.1.3 Event Tree

The event tree for this transient is provided on Figure 3.1.3-1 Sheets 1-6. Heading definitions are provided below.

T1 A loss of offsite power transient occurs which disrupts the normal operation of the plant and requires mitigation.

C Success or failure of RPV reactivity control with RPS automatic or manual scram or ARI. Success implies that a sufficient number of control rods have inserted into the core to provide shutdown rod pattern. Thus, a successful reactor scram has occurred. Failure implies the rods have not inserted to a subcritical rod configuration. Failure sequences transfer to the ATWS event tree.

P1 Success or failure of RPV pressure control with the SRVs opening and reclosing. Success implies that the required SRVs opened and reclosed. Failure implies that of the SRVs which successfully opened, one failed to reclose. This is equivalent to a small LOCA and is developed on Sheet 4.

P2 Success or failure of RPV pressure control with the SRVs opening and reclosing. Success implies that the required SRVs opened and reclosed. Failure implies that of the SRVs which successfully opened, one failed to reclose. This is equivalent to an intermediate LOCA event and is developed on Sheet 6.

B1 Success or failure of onsite emergency power to either the division 1 or division 2 buses. Success implies that one or both buses are available and supplying power to their respective loads. Failure implies that both buses fail and that power is not being supplied to division 1 and division 2 loads. This transfers to the station blackout event tree.

U1 Success or failure of RPV level control with the HPCS system. Success implies that either HPCS automatically actuated at RPV level 2 or that it was manually actuated and is providing adequate RPV level control. Failure implies that RPV level control cannot be maintained with HPCS.

U2 Success or failure of RPV level control with the RCIC system. Success implies that either RCIC automatically actuated at RPV level 2 or that it was manually actuated and is providing adequate RPV level control. It further implies that long-term containment heat removal with RHR in Suppression Pool Cooling mode successfully maintains containment conditions for long-term operability of RCIC. Failure implies that RCIC is not maintaining RPV level control or that long-term containment heat removal is not maintained, resulting in the loss of RCIC.

R1 Recovery or nonrecovery of offsite power. Recovery of offsite power implies that offsite power was recovered and all safety related and non-safety related buses have power. Nonrecovery implies that offsite power was not recovered.

Upon failure of HPCS and success of RCIC, R1 is based on recovery in 3.3 hours. Upon failure of both HPCS and RCIC, R1 is based on recovery in 0.4 hours. For failure of both HPCS and RCIC and one stuck open relief valve, R1 is based on recovery in 0.26 hours.

Ws Success or failure of maintaining the suppression pool less than 185°F. Success implies that at least one train of RHR is in operation in the suppression pool cooling mode. Failure implies that the suppression pool could not be maintained below 185°F.

X Success or failure of emergency RPV depressurization. Success implies that the ADS was automatically or manually initiated or that the operator manually depressurized the vessel with at least 4 SRVs. Failure implies that ADS failed and that the operator failed to manually emergency depressurize with the required number of SRV causing the reactor vessel to remain at a higher pressure than that required by the PNPP Plant Emergency Instructions (PEIs) for successful RPV level control with low pressure injection.

V Success or failure of RPV level control with the ECCS low pressure make-up systems. Success implies that, at a minimum, LPCS or one loop of LPCI either automatically initiated or was manually placed in service and is providing adequate RPV level control. Failure implies that RPV level control cannot be maintained by an ECCS low pressure injection system.

Hv Initial analysis indicated that switchgear room cooling was required to prevent breaker failure. Recent analysis showed that this was not the case. The function is still in the event tree but the probability of failure is set to zero and success to one.

R2 Success or failure of the ability to recover offsite power. Success implies offsite power is recovered. Failure implies that offsite power was not recovered within the following time frames.

With success of HPCS, R2 is based on failure to recover in 12 hours. Following failure of HPCS, R2 is based on failure to recover in 4 hours.

X2 Success or failure of emergency RPV depressurization. Success implies that the ADS was automatically or manually initiated or that the operator manually depressurized the vessel with at least 4 SRVs.

Failure implies that ADS failed and that the operator failed to manually emergency depressurize with the required number of SRV causing the reactor vessel to remain at a higher pressure than that required by the PNPP Plant Emergency Instructions (PEIs) for successful RPV level control with low pressure injection. This event occurs much later than X above as it follows failure of high pressure injection after failure of the room cooling (Hv).

Va Success or failure of RPV level control with the alternate low pressure injection systems. Success implies that either the alternate injection mode of the condensate transfer system or the fire protection system crosstie to ESW B and RHR B was manually placed in service and is maintaining RPV level control. If offsite power is not available the only alternate injection system modeled is the fire protection system. Failure implies that RPV level control cannot be maintained by either the condensate transfer system or the fire protection system.

W Success or failure of long-term containment heat removal with RHR to maintain containment pressure below the Containment Capacity Threshold Limit of 50 psig. Success implies that long-term heat removal can be achieved by at least one of the following modes of RHR. Failure implies that no long-term containment heat removal mode of RHR was aligned or that pressure cannot be maintained below the Containment limit.

o One RHR train in Suppression Pool cooling mode.

o One RHR in the Containment Spray mode.

Y Success or failure of long-term containment heat removal by venting of the containment. Success implies that one of the below listed venting paths were aligned, per the PEIs, prior to containment pressure exceeding the Containment Capacity Threshold Limit of 50 psig and that the aligned path maintains pressure below that limit. Failure implies that no path was successfully aligned or that the containment limit was exceeded.

o Fuel Pool Cooling and Cleanup System. Manual alignment of vent path through the fuel pool skimmers into the Fuel Handling Building.

o RHR Containment Spray Loop A or B. Manual alignment of vent path through the containment spray header into the Fuel Handling Building.

Cv Defines the susceptibility of the core to damage, dependent upon containment conditions. Success implies that given containment failure, the core is still maintained in a safe configuration. Failure implies that the containment failure has led to a degradation of plant systems such that the core is susceptible to damage.

The development of the functional fault trees for each of the event tree functions with the appropriate boundary conditions defined by the initiating event and preceding success or failures is described in Appendix E.

3.1.3.2 STATION BLACKOUT EVENT TREE

3.1.3.2.1 General Description

The loss of offsite power transients which are further compounded by the failure of both emergency AC Divisions 1 and 2 which supply electric power to the safe shutdown RHR systems are defined as station blackout (SBO) sequences.

Station Blackout sequences are defined as "B" sequences, consistent with the nomenclature utilized in NUREG/CR-4550.

Multiple plant instructions address the station blackout event at Perry. In addition to the BWR Owners Group EPG Revision 4 based Perry Plant Emergency Instructions (PEIs), off-normal instruction, ONI-R10, Loss of AC Power provides scenario specific plant instructions.

A station blackout event is initiated by the occurrence of a loss of offsite power and the subsequent loss of both Division 1 and Division 2 diesel generators. The Power Conversion System (PCS), is lost due to the closure of the Main Steam Isolation Valves (MSIVs). All ECCS systems are immediately lost, with the exception of the Division 3 diesel generator backed High Pressure Core Spray (HPCS) System and the AC independent Reactor Core Isolation Cooling (RCIC) System. Alternate injection with the diesel driven Fire Water pump is also available.

Upon the loss of motive power caused by a Station Blackout, the containment AC motor-operated isolation valves will remain in the normal plant lineup. In most cases this is insignificant as the open penetrations lead to closed systems outside containment. The upper containment fuel pool return line penetration is not automatically isolated under SBO conditions and thus provides a containment vent path, however the operators are procedurally directed to close this penetration.

The containment air-operated isolation valves generally reposition in the closed position following the loss of power; however, air-operated inboard MSIV before seat normal drain valve, 1B13-F033, does not close and provides a pathway from the reactor vessel to the low pressure condenser through a one inch line. This line is isolated by the operators during a station blackout by closing 1B21-F019. The front end analysis will assume that this manual isolation is always completed.

3.1.3.2.2 Success Criteria

The success criteria for station blackout is listed in Table 3.1.3-2.

3.1.3.2.3 Event Tree

The event tree for the station blackout event is provided on Figure 3.1.3-2 Sheets 1-3. Heading definitions are provided below.

B A loss of offsite power transient occurs with a subsequent failure of onsite Division 1 and 2 power (Station Blackout) such that no AC power is available. Initial mitigation is only possible with HPCS and RCIC.

BP1 A loss of offsite power transient occurs with a subsequent failure of onsite Division 1 and 2 power (Station Blackout) such that no AC power is available. In addition, one SRV is stuck open. Initial mitigation is only possible with HPCS and RCIC. Failure of HPCS and RCIC is assumed to lead to core damage.

BP2 A loss of offsite power transient occurs with a subsequent failure of onsite Division 1 and 2 power (Station Blackout) such that no AC power is available. In addition, two SRVs are stuck open. Initial mitigation is only possible with HPCS. Failure of HPCS is assumed to lead to core damage.

U1 Success or failure of the HPCS system. Success implies that either HPCS automatically actuated at level 2 or that it was manually actuated and is maintaining coolant make-up into the reactor vessel. Failure implies that HPCS is not maintaining coolant make-up.

HI Success or failure of operator to take action to extend HPCS operation. Success implies that the operators crosstie the Unit 1 and Unit 2 [illegible] batteries enabling HPCS to provide injection for the complete mission time. Failure implies that this was not successfully completed and that HPCS injection fails.

U2 Success or failure of the RCIC system. Success implies that either RCIC automatically actuated at Level 2 or that it was manually actuated and is maintaining coolant make-up into the reactor vessel until its limiting system parameter is reached. Failure implies that RCIC is not maintaining coolant make-up for its defined mission time.

Va Success or failure of the diesel-driven fire water pump. With success of RCIC upon the suppression pool temperature reaching 185°F the operators are instructed to depressurize the RPV. Given a failure to depressurize, RCIC will continue to operate. Therefore, success of the diesel driven fire pump implies that the RPV was depressurized causing a loss of RCIC and that the fire pump is injecting to the RPV and will continue to run until limiting system parameters are reached. Failure implies that the system could not maintain injection for its defined mission time.

R Success or failure of power restoration either from the offsite grid or an onsite Division 1 or 2 diesel generator to the associated safety-related 4.16 bus(es) at the limiting critical time for each accident sequence. Success implies the restoration of power does not occur in time to prevent core damage.

With HPCS success and successful operator actions to extend HPCS operation, R is based on recovery in 13 hours. If the operator action to extend HPCS operation fails, R is based on recovery in 11 hours. With failure of HPCS and success of RCIC and alternate injection with the fire pump, R is based on recovery in 13 hours. If the fire pump fails, only 3 hours is available for recovery. If both HPCS and RCIC fail, R is based on recovery in 0.4 hours.

With one stuck open relief valve the above recovery times are slightly more restrictive. With HPCS success and successful operator action, R is based on recovery in 11 hours. If the operator action fails, R remains based on recovery in 11 hours. With failure of HPCS and success of RCIC and alternate injection with the fire pump, R is based on 11 hours. If the fire pump fails, only 2 hours is available for recovery.

The times with two stuck open relief valves are the same as with one stuck open relief valve.

X Success or failure of emergency depressurization (automatic or manual) following the recovery of divisional power supplies. Success implies that the RPV is depressurized using the appropriate number of SRVs for the sequence (4, 3 or 2) . Failure implies that the reactor has not been depressurized and no ECCS system will be capable to provide injection into the vessel.

V Success or failure of low pressure make-up. Success implies that a low pressure ECCS system is providing adequate make-up into the RPV. Failure implies no low pressure ECCS is capable of providing RPV coolant make-up.

Val Success or failure of condensate transfer alternate injection or fire protection alternate injection crosstied to the HPCS injection line. Success implies that the transfer systems were aligned to the RPV and are providing injection. Failure implies that the systems could not provide coolant makeup.

W Success or failure of long-term containment heat removal by RHR suppression pool cooling or containment spray, to maintain containment pressure below the Containment Capacity Threshold Limit of 50 psig. Success implies that heat removal can be demonstrated after power restoration by operation of at least one heat exchanger. Failure implies that heat removal by RHR is not maintaining containment pressure below the limit.

Y Success or failure of containment heat removal by venting to maintain containment pressure below the Containment Capacity Threshold Limit of 50 psig. Success implies that steady-state heat removal can be demonstrated after the power restoration time by a venting operation performed either manually or with motor-operated valves. Failure implies that heat removal by venting is not maintaining containment pressure below the limit.

- Venting with the Fuel Pool Cooling and Cleanup system requires the manual alignment of vent path through the fuel pool skimmers into the Fuel Handling Building.

- Venting with the RHR Containment Spray Loop A or B requires the manual alignment of vent path through the applicable containment spray header into the Fuel Handling Building. Operation of the containment motor operated valves requires AC power.

Following failure of offsite AC power restoration success of venting implies that the division 3 bus has been crosstied to the division 2 bus and the inboard FPCC isolation vavle was successfully opened and that the operators manually opened the Fuel Pool Cooling and Cleanup (FPCC) outboard isolation valve. Failure implies that the FPCC line is isolated.

Cv Success or failure of operating injection systems following loss of containment integrity. Success implies that adequate core cooling can be maintained by the operating injection system. Failure implies that the reactor core was vulnerable to damage as a result of the failure of an operating injection system coincident with loss of containment integrity.

The development of the functional fault trees for each of the event tree functions with the appropriate boundary conditions defined by the initiating event and preceding success or failures is described in Appendix E.

3.1.3.3 TRANSIENT WITH INADVERTENT OPEN RELIEF VALVE

3.1.3.3.1 General Description

The inadvertent open relief valve (IORV), event is defined as a class T3C transient, consistent with the nomenclature utilized in NUREG/CR-4550.

The IORV transient is defined as the spurious or inadvertent opening, during normal plant operation, of an SRV that then sticks in the full open position. With an inadvertent open relief valve, reactor steam is discharged directly into the suppression pool. Alarms, temperature monitors and plant response will readily indicate the existence of an IORV. The PNPP procedure for an IORV calls for attempts by the operators to close the valve. PNPP Technical Specifications call for an immediate plant shutdown if the valve cannot be closed prior to the suppression pool exceeding 110°F.

A MAAP analysis indicates that it will take approximately 8 minutes for the pool temperature to increase from 90 to 110°F.

For an IORV transient, it is assumed that the Power Conversion System (PCS) is initially available to eliminate any challenge to the SRVs for RPV pressure control.

RCIC is capable of providing injection for the IORV event. Additionally, an IORV is assumed to not depressurize the reactor vessel fast enough, without the assistance of additional open SRVs or the ADS system, to prevent uncovery of the core before the low pressure systems inject into the vessel.

The event is expected to cause only minor perturbation to the RPV water level. The capacity of the feedwater system is sufficient to make-up the inventory lost due to the IORV.

No automatic scram is assumed to occur until the suppression pool heatup causes an increase in the drywell pressure and a resultant automatic reactor scram.

This event is treated separately from the transient induced stuck open relief valve (SORV), since for the IORV, plant shutdown will not occur until suppression pool temperature approaches 110°F. For the SORV, a plant scram will have occurred prior to the open SRV. Thus, there is more critical demand upon the containment heat removal systems for the IORV than for the SORV.

3.1.3.3.2 Success Criteria

The success criteria for a transient with an inadvertent open relief valve is listed in Table 3.1.3-3.

3.1.3.3.3 Event Tree

The event tree for this event is provided on Figure 3.1.3-3. Heading definitions are provided below.

T3C An IORV transient occurs which disrupts the normal operation of the plant and requires mitigation.

C Success or failure of RPV reactivity control with RPS automatic or manual scram or ARI. Success implies that a sufficient number of control rods have inserted into the core to provide a shutdown rod pattern. Thus, a successful reactor scram has occurred. Failure implies the rods have not inserted to a subcritical rod configuration. Failure sequences transfer to the ATWS event tree.

U3 Success or failure of RPV level control with the feedwater system. Success implies that one of two Reactor Feedpumps (RFPs) or one Motor Feedpump (MFP) starts or continues to run and provides successful RPV level control. Failure implies that the feedwater system cannot maintain RPV level control.

U2 Success or failure of RPV level control with the RCIC system. Success implies that either RCIC automatically actuates at RPV level 2 or that it is manually actuated and is providing adequate RPV level control. It further implies that long-term containment heat removal with RHR in Suppression Pool Cooling mode successfully maintains containment conditions for long-term operability of RCIC. Failure implies that RCIC is not maintaining RPV level control or that long-term containment heat removal is not maintained, resulting in the loss of RCIC.

U1 Success or failure of RPV level control with the HPCS system. Success implies that HPCS automatically actuated at RPV level 2 or that it was manually actuated and is providing adequate RPV level control. Failure implies that RPV level control cannot be maintained with HPCS.

X Success or failure of emergency RPV depressurization. Success implies that the ADS was automatically or manually initiated or that the operator manually depressurized the vessel with at least 4 SRVs. Failure implies that ADS failed and that the operator failed to manually emergency depressurize with the required number of SRV causing the reactor vessel to remain at a higher pressure than that required by the PNPP Plant Emergency Instructions (PEIs) for successful RPV level control with low pressure injection.

V Success or failure of RPV level control with the ECCS low pressure make-up systems. Success implies that, at a minimum, LPCS or one loop of LPCI either automatically initiated or was manually placed in service and is providing adequate RPV level control. Failure implies that RPV level control cannot be maintained by an ECCS low pressure injection system.

Va Success or failure of RPV level control with the Condensate Transfer Alternate Injection alignment. Success implies, that the CTS alternate injection alignment was manually started by the plant operators and that the alignment successfully provided RPV level control. Failure implies that CTS alternate injection was not aligned or that it was unable to maintain RPV level control.

W Success or failure of long-term containment heat removal with RHR to maintain containment pressure below the Containment Capacity Threshold Limit of 50 psig. Success implies that long-term heat removal can be achieved by at least one of the following modes of RHR. Failure implies that no long-term containment heat removal mode of RHR was aligned or that pressure cannot be maintained below the limit.

- o One RHR train in Suppression Pool Cooling mode.
- o One RHR in the Containment Spray mode.

Y Success or failure of long-term containment heat removal by venting of the containment. Success implies that one of the below listed venting paths were aligned, per the PEIs, prior to containment pressure exceeding the Containment Capacity Threshold Limit and that the aligned path maintains pressure below that limit. Failure implies that no path was successfully aligned or that the limit was exceeded.

- o Fuel Pool Cooling and Cleanup system. Manual alignment of vent path through the fuel pool skimmers into the Fuel Handling Building.
- o RHR Containment Spray Loop A or B. Manual alignment of vent path through the containment spray header into the Fuel Handling Building.

Cv Defines the susceptibility of the core to damage, dependent upon containment conditions. Success implies that given containment failure, the core is still maintained in a safe configuration. Failure implies that the containment failure has led to a degradation of plant systems such that the core is susceptible to damage.

The development of the functional fault trees for each of the event tree functions with the appropriate boundary conditions defined by the initiating event and preceding success or failures is described in Appendix E.

3.1.3.4 LOSS OF INSTRUMENT AIR EVENT TREE

3.1.3.4.1 General Description

The loss of instrument air transient is defined as the TIA transient. It is a special initiator.

The loss of instrument air event is defined as the complete unrecoverable loss of instrument air during normal plant operation. The loss of instrument air event is similar to a loss of PCS transient (T2). It is developed as a special initiator as it will lead to the potential loss of some additional accident mitigating systems.

The loss of instrument air event is considered a subset of the loss of Service Water event which is expected to fail both Turbine Building Closed Cooling and Nuclear Closed Cooling (and thus instrument air).

Upon the complete unrecoverable loss of instrument air, plant operators are directed to perform a fast reactor shutdown. However, if no immediate operator action is taken, it is expected that a plant scram will occur due to a variety of plant conditions. The worst case initiator is assumed to be a MSIV isolation caused by either the loss of instrument air to the MSIV accumulators (and the subsequent drifting closed of the MSIVs) or due to a low condenser vacuum isolation signal caused by the loss of the Steam Jet Air Ejectors.

A plant scram could also occur due to turbine trip (caused by loss of air to the main turbine front standard), or loss of condenser vacuum (caused by the loss of the Steam Jet Air Ejectors) or due to a high scram discharge volume (caused by drifting of the scram valves and closure of the SDV vent and drain valves).

Feedwater is lost for this transient due to the loss of steam to the Reactor Feedpumps (RFPs), resulting from the MSIV isolation caused by a loss of condenser vacuum. The Motor Feedpump (MFP), is unable to be started due to the lockup of the MFP Flow Control Valve (1N27-F0010) in its normally closed position.

Additionally, the Hot Surge Tank level control valves will fail as is and the feedwater and feedwater booster pump recirculation valves will fail open. these failures are assumed to inhibit recovery of the feedwater system.

CRD pumps will also be unavailable for the transient as the loss of instrument air will cause the closure of the flow control valve, resulting in a decreased drive water and cooling water flow to the CRDs.

Long-term air supply to the non-ADS SRVs (excepting F0051D), is lost with the loss of instrument air. Accumulators provide an immediate air supply for short term SRV operability.

3.1.3.4.2 Success Criteria

The success criteria for loss of instrument air are the same as for the loss of PCS transient listed in Table 3.1.2-4.

3.1.3.4.3 Event Tree

The event tree for this transient is provided on Figure 3.1.3-4 sheets 1-3. Heading definitions are provided below.

TIA A loss of instrument air transient occurs which disrupts the normal operation of the plant and requires mitigation.

C Success or failure of RPV reactivity control with RPS automatic or manual scram or ARI. Success implies that a sufficient number of control rods have inserted into the core to provide a shutdown rod pattern. Thus, a successful reactor scram has occurred. Failure implies the rods have not inserted to a subcritical rod configuration. Failure sequences transfer to the ATWS event tree.

P1 Success or failure of RPV pressure control with the SRVs opening and reclosing. Success implies that the required SRVs opened and reclosed. Failure implies that of the SRVs which successfully opened, one failed to reclose. The event tree for the equivalent small LOCA is developed on Sheet 2.

P2 Success or failure of RPV pressure control with the SRVs opening and reclosing. Success implies that the required SRVs opened and reclosed. Failure implies that of the SRVs which successfully opened, two failed to reclose. The event tree for the equivalent intermediate LOCA event is developed on Sheet 3.

U2 Success or failure of RPV level control with the RCIC system. Success implies that either RCIC automatically actuated at RPV level 2 or that it was manually actuated and is providing adequate RPV level control. It further implies that long-term containment heat removal with RHR in Suppression Pool Cooling mode successfully maintains containment conditions for long-term operability of RCIC. Failure implies that RCIC is not maintaining RPV level control or that long-term containment heat removal is not maintained, resulting in the loss of RCIC.

U1 Success or failure of RPV level control with the HPCS system. Success implies that either HPCS automatically actuated at RPV level 2 or that it was manually actuated and is providing adequate RPV level control. Failure implies that RPV level control cannot be maintained with HPCS.

X Success or failure of emergency RPV depressurization. Success implies that the ADS was automatically or manually initiated or that the operator manually depressurized the vessel with at least 4 SRVs. Failure implies that ADS failed and that the operator failed to manually emergency depressurize with the required number of SRV causing the reactor vessel to remain at a higher pressure than that required by the PNPP Plant Emergency Instructions (PEIs) for successful RPV level control with low pressure injection.

V Success or failure of RPV level control with the ECCS low pressure make-up systems. Success implies that, at a minimum, LPCS or one loop of LPCI either automatically initiated or was manually placed in service and is providing adequate RPV level control. Failure implies that RPV level control cannot be maintained by an ECCS low pressure injection system.

Va Success or failure of reactor feed booster pump alternate injection. Success implies that at least one reactor feed booster pumps is manually aligned and provides injection to the RPV. Failure implies that the reactor feed booster pumps were not aligned or that the pumps are unable to maintain RPV level control.

W Success or failure of long-term containment heat removal with RHR to maintain containment pressure below the Containment Capacity Threshold Limit of 50 psig. Success implies that long-term heat removal can be achieved by at least one of the following modes of RHR. Failure implies that no long-term containment heat removal mode of RHR was aligned or that pressure cannot be maintained below the limit.

o One RHR train in Suppression Pool Cooling mode.

o One RHR train in the Containment Spray mode.

Y Success or failure of long-term containment heat removal by venting of the containment. Success implies that one of the below listed venting paths were aligned, per the PEIs, prior to containment pressure exceeding the Containment Capacity Threshold Limit and that the aligned path maintains pressure below that limit. Failure implies that no path was successfully aligned or that the limit was exceeded.

o Fuel Pool Cooling and Cleanup System. Manual alignment of vent path through the fuel pool skimmers into the Fuel Handling Building.

o RHR Containment Spray Loop A or B. Manual alignment of vent path through the containment spray header into the Fuel Handling Building.

Cv Defines the susceptibility of the core to damage, dependent upon containment conditions. Success implies that given containment failure, the core is still maintained in a safe configuration. Failure implies that the containment failure has led to a degradation of plant systems such that the core is susceptible to damage.

The development of the functional fault trees for each of the event tree functions with the appropriate boundary conditions defined by the initiating event and preceding success or failures is described in Appendix E.

3.1.3.5 LOSS OF SERVICE WATER EVENT TREE

3.1.3.5.1 General Description

The loss of service water transient is defined as the TSW transient. It is a special initiator.

The loss of Service Water event is defined as the complete loss of Service Water during normal plant operation. It is assumed that the loss of Service

Water will lead to the additional loss of the Nuclear Closed Cooling System, the Turbine Closed Cooling System and the systems which are supported by these systems (including Instrument Air).

The loss of Service Water event is similar to a loss of PCS transient (T2). It is developed as a special initiator as a loss of Service Water will lead to the potential loss of additional accident mitigating systems. The loss of Service Water event is expected to bound other similar initiators, such as loss of Turbine Building Closed Cooling and Nuclear Closed Cooling. Failure of these systems is conservatively included in the loss of Service Water initiator frequency.

Upon loss of Service Water, plant operators are directed to perform a fast reactor shutdown in anticipation of the loss of cooling to the various plant loads. If no immediate operator action is taken, it is expected that a plant scram will occur due to a variety of plant conditions.

The worst case initiator for the ISW transient is assumed to be a MSIV isolation. The isolation can be initiated form various mechanisms, including an isolation signal on high steam tunnel temperature (loss of steam tunnel cooling) or on low condenser vacuum (loss of Steam Jet Air Ejectors) or by a loss of Instrument Air to the MSIVs and their subsequent isolation.

A plant scram is also postulated due to high drywell pressure (loss of drywell cooling), a turbine trip (due to loss of Stator Water Cooling), high scram discharge volume (due to drifting of the scram valves and closure of the SDV vent and drain valves).

Feedwater is assumed lost for this transient due to a loss of component motor bearing or lube oil cooling. Compounding the loss of feedwater would be the loss of Instrument Air which would cause a lockup of the Hot Surge Tank level control valves, the failing open of the feedwater and feedwater booster pump recirculation valves and the lockup of the Motor Feedpump flow control valve. Loss of the MSIVs would also isolate the steam supply to the Reactor Feedpumps (RFPs). The Motor Feedpump (MFP) is assumed to be unable to be started due to the lockup of the MFP flow control valve.

The Power Conversion System (PCS) is assumed lost for this transient. The Steam Jet Air Ejectors (SJAE) will be lost upon loss of Instrument Air or due to tripping of the Off-Gas compressors, which would require SJAE shutdown. While ONI-P41 directs the operators to start the mechanical vacuum pumps, these will also be lost due to loss of an ultimate heat sink for the TBCC system which cools the vacuum pumps.

The Service Air and Instrument Air compressors will trip on high temperature resulting from the loss of Nuclear Closed Cooling. Actual loss of instrument air is not expected to occur until the system is bled down by leakage or component usage.

CRD pumps are assumed to be lost due to loss of component cooling. Loss of Instrument Air will lead also to the closure of the CRD flow control valve.

RCIC isolation may also occur if the RCIC isolation signal on high steam tunnel temperature is not overridden by the plant operators prior to the thirty minute timer timing out.

Long-term air supply to the non-ADS SRVs (excepting F0051D), is lost with the loss of Instrument Air. Accumulators provide an immediate air supply for short term SRV operability.

3.1.3.5.2 Success Criteria

The success criteria for loss of service water are the same as for the loss of PCS transient with the addition identified above and are in Table 3.1.2-4.

3.1.3.5.3 Event Tree

The event tree for this transient is provided in Figure 3.1.3-5 sheets 1-3.

TSW A loss of service water transient occurs which disrupts the normal operation of the plant and requires mitigation.

C Success or failure of RPV reactivity control with RPS automatic or manual scram or ARI. Success implies that a sufficient number of control rods have inserted into the core to provide a shutdown rod pattern. Thus, a successful reactor scram has occurred. Failure implies the rods have not inserted to a subcritical rod configuration. Failure sequences transfer to the ATWS event tree.

P1 Success or failure of RPV pressure control with the SRVs opening and reclosing. Success implies that the required SRVs opened and reclosed. Failure implies that of the SRVs which successfully opened, one failed to close. The event tree for the equivalent small LOCA is developed on Sheet 2.

P2 Success or failure of RPV pressure control with the SRVs opening and reclosing. Success implies that the required SRVs opened and reclosed. Failure implies that of the SRVs which successfully opened, two failed to reclose. The event tree for the equivalent intermediate LOCA is developed on sheet 3.

U2 Success or failure of RPV level control with the RCIC system. Success implies that either RCIC automatically actuated at RPV level 2 or that it was manually actuated and is providing adequate RPV level control. It further implies that long-term containment heat removal with RHR in Suppression Pool Cooling mode successfully maintains containment conditions for long-term operability of RCIC. Failure implies that RCIC is not maintaining RPV level control or that long-term containment heat removal is not maintained, resulting in the loss of RCIC.

U1 Success or failure of RPV level control with the HPCS system. Success implies that neither HPCS automatically actuated at RPV level 2 or that it was manually actuated and is providing adequate RPV level control. Failure implies that RPV level control cannot be maintained with HPCS.

X Success or failure of emergency RPV depressurization. Success implies that the ADS was automatically or manually initiated or that the operator manually depressurized the vessel with at least 4 SRVs.

Failure implies that ADS failed and that the operator failed to manually emergency depressurize with the required number of SRV causing the reactor vessel to remain at a higher pressure than that required by the PNPP Plant Emergency Instructions (PEIs) for successful RPV level control with low pressure injection.

V Success or failure of RPV level control with the ECCS low pressure make-up systems. Success implies that, at a minimum, LPCS or one loop of LPCI either automatically initiated or was manually placed in service and is providing adequate RPV level control. Failure implies that RPV level control cannot be maintained by an ECCS low pressure injection system.

W Success or failure of long-term containment heat removal with RHR to maintain containment pressure below the Containment Capacity Threshold Limit of 50 psig. Success implies that long-term heat removal can be achieved by at least one of the following modes of RHR. Failure implies that no long-term containment heat removal mode of RHR was aligned or that pressure cannot be maintained below the limit.

o One RHR train in Suppression Pool Cooling mode.

o One RHR train in the Containment Spray mode.

Y Success or failure of long-term containment heat removal by venting the containment. Success implies that one of the below listed venting paths were aligned, per the PEIs, prior to containment pressure exceeding the Containment Capacity Threshold Limit and that the aligned path maintains pressure below that limit. Failure implies that no path was successfully aligned or that the limit was exceeded.

o Fuel Pool Cooling and Cleanup System. Manual alignment of vent path through the fuel pool skimmers into the Fuel Handling Building.

o RHR Containment Spray Loop A or B. Manual alignment of vent path through the containment spray header into the Fuel Handling Building.

Cv Defines the susceptibility of the core to damage, dependent upon containment conditions. Success implies that given containment failure, the core is still maintained in a safe configuration. Failure implies that the containment failure has led to a degradation of plant systems such that the core is susceptible to damage.

The development of the functional fault trees for each of the event tree functions with the appropriate boundary conditions defined by the initiating event and preceding success or failures is described in Appendix E.

3.1.3.6 INTERFACE LOCA EVENT TREE

3.1.3.6.1 General Description

No event tree was developed for the interface LOCA event. An evaluation was performed on the potential paths of an interface LOCA. The paths noted in section 4.4.15 of NUREG/CR-4550 were identified as the dominant paths contributing to interface LOCA for the Perry plant. The Grand Gulf and Perry designs are identical for these paths. Using the data in NUREG/CR-4550 the core damage frequency for an interfacing LOCA event is less than 10^{-8}/yr.

3.1.3.7 CONTAINMENT BYPASS LOCA EVENT TREE

3.1.3.7.1 General Description

No event tree was developed for the containment bypass LOCA event. An evaluation was performed on the potential paths for containment bypass LOCAs. The potential for an unisolable break in these lines was much less than for an interface LOCA event. Therefore, the core damage frequency for a containment bypass LOCA event is also less than 10^{-8}/yr.

3.1.3.8 VESSEL RUPTURE

3.1.3.8.1 General Description

No event tree will be developed for this event. All occurrences of this event are assumed to result in core damage. Using the data in NUREG/CR-4550, the frequency of a vessel rupture is less than 10¯8/yr.

3.1.3.9 INTERNAL FLOODING

3.1.3.9.1 General Description

This event is being treated as a separate task in the IPE and is fully described in section 3.3.7.

3.1.3.10 ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) EVENT TREE

3.1.3.10.1 General Description

Transients with a failure of automatic scram are defined as ATWS events and are further defined as type "C" events, consistent with the nomenclature utilized in NUREG/CR-4550.

When a transient initiating event is followed by failure of reactivity control to provide rod insertion by automatic/manual Reactor Protection System scram and by automatic/manual Redundant Reactivity Control System Alternate Rod Insertion, the operators will attempt shutdown of the reactor by manually inserting rods and/or initiating the two Standby Liquid Control System (SLC) pumps. The progression of an anticipated transient without automatic scram (ATWS) depends on the initiating events and the subsequent success in the operation of plant systems and in the success of human interactions.

ATWS events can result during the development of all initiators analyzed. Six ATWS event trees (Loss of Offsite Power, Transient Without PCS, Transient With PCS Available, Transient With Loss of Feedwater, Inadvertent Stuck Open Relief Valve, and Loss of Instrument Air) transfer from the initiating trees and bound all ATWS events. These ATWS event trees detail the accident progression for ATWS events due to mechanical failure of control rods to insert. Electrical failure of the reactivity control systems without mechanical failure is considered to be negligible and is not developed.

3.1.3.10.2 Success Criteria

The success criteria are given in Table 3.1.3-4.

Additional key assumptions upon which this analysis is based include the following:

1. For the bounding ATWS condition of all control rods out, the reactor power was modeled with an adjusted Chexal-Layman Correlation (to account for the allowable 120% rod line). At the rated RPV pressure of 1,039 psia, the adjusted Chexal-Layman Correlation calculates 21% rated reactor power with the downcomer water level at the TAF when the decay heat power is 2.5%.

2. HPCS is not utilized as a high pressure injection source in the ATWS event trees, since the BWR Owners Group EPG Rev 4 does not direct the operation of inside the shroud injection systems until all other injection sources have been exhausted.

3. Exceeding the suppression pool heat capacity design temperature limit does not lead immediately to a loss of containment integrity, and will not lead to core damage provided containment heat removal is initiated following shutdown by injection of borated water (SLC).

4. Reactor power remains constant until hot shutdown is achieved which is approximately 44 minutes for 1 pump SLC injection.

5. Containment integrity is not threatened until the Containment Capacity Threshold Limit (approximately 50 psig) is reached.

6. No credit is taken for manually inserting individual rods.

3.1.3.10.3 Event Tree

Reactivity control is provided by multiple, redundant design features and systems. Following the occurrence of a transient which disrupts normal plant operation the front line reactivity control systems are challenged to insert all the control rods into the core to at least position 02 or to a shutdown pattern. The frontline reactivity control capabilities addressed in the transient event trees include Reactor Protection System (RPS), Redundant Reactivity Control System (Alternate Rod Insertion - ARI) and manual scram. Success of any of these control functions implies that the reactor has been successfully shutdown and continues the accident sequence progression through the transient tree. Failure of all these functions implies that the reactivity control electrical portion is not capable of shutting down the reactor. With recent design improvements this frequency is projected to be negligible.

A fraction of reactivity control failures can be attributed to mechanical failures associated with the reactivity control systems. Mechanical failure would eliminate the capability to insert the control rods by any alternate means. The mechanical failure of control rod insertion systems is explicitly addressed in the ATWS event trees. Failure of this event assumes that <u>all</u> of the rods remain in the withdrawn position.

For ATWS scenarios due to mechanical failure, tripping of the recirculation pumps is assumed to have occurred. The trip will occur automatically due to Recirculation Pump Trip (RPT), manual operator action, mechanical overspeed or due to cavitation/flow induced failures when RPV level falls towards the Top of Active Fuel (TAF). For some transients (i.e., loss of offsite power) plant conditions will cause the pump trip. Success implies that the RPT has functioned resulting in a natural circulation reactor power of approximately 50% at the bounding 120% rod line. ATWS scenarios with a failure of the RPT are not developed further in this analysis.

Six events are analyzed in this ATWS analysis. As noted above, the transients are characterized by the occurrence of the transient, the mechanical failure of reactivity control such that no control rods are inserted into the core and the occurrence of the Recirculation Pump Trip (RPT). The six events analyzed include Loss of Offsite Power, Transient without PCS, Transient with PCS Available, Transient with Loss of Feedwater, Inadvertent Stuck Open Relief Valve and Loss of Instrument Air. The associated ATWS event trees are provided on Figures 3.1.3-7 to 3.1.3-12. The structure of each event tree is similar. Heading definitions are provided below:

Txxx-C A transient initiating event has occurred which disrupted normal operation. A mechanical failure of the front line reactivity control systems (Reactor Protection System and Redundant Reactivity Control) has occurred such that a failure to insert all control rods into the core to at least position 02 or to a shutdown pattern has occurred. Recirculation Pump Trip (RPT) has occurred resulting in a natural circulation reactor power of approximately 50% at the bounding 120% rod line.

M Success or failure of the safety relief valves (SRVs) to relieve RPV pressure. Success implies that all SRVs have opened against the spring preventing a rupture of the RPV due to overpressurization. Failure implies that at least one SRV failed to open.

RPT Success or failure of the recirculation pump trip. Success implies that the recirculation pumps have received a trip signal and tripped. Failure implies that the recirculation pumps have not tripped and are still operating.

Q Success or failure of the Power Conversion System (PCS) to remain available. Success implies that the Main Steam Isolation Valves have remained open. Success is contingent on the operators defeating the MSIV Low RPV Level Isolation by repositioning four back panel switches before the rector water level decreases to Level 1. Failure implies the MSIVs did not remain open which results in a loss of PCS and steam to the Turbine Driven Feed Pumps (TDFPs).

U3 Success or failure of RPV level control with at least one pump in the feedwater system. Success implies that either the Motor Feed Pump

(MFP) or one of the two Turbine Driven Feed Pumps operate until hot shutdown and the RPV water level is maintained at or neat the Top of Active Fuel (TAF) per the PNPP Plant Emergency Instructions (PEIs). Failure implies all feedwater pumps failed to function.

Lc Success or failure of RPV level control. Success implies that the operators control the RPV water level at TAF to control RPV power. Failure implies that that level is not deliberately lowered.

X' ADS successfully or not successfully inhibited. Success implies that the operators have bypassed the automatic depressurization system to prevent rapid RPV overfill with cold water from the low pressure injection systems. Failure implies that the ADS is not inhibited and the RPV will depressurize on the automatic initiation of the ADS system allowing a cold water reactivity transient with the potential for core damage and increased containment threat from high reactor power.

C1 Success or failure of Standby Liquid Control (SLC) system injection with one of two pumps. Success implies that the operators have initiated the SLC system such that hot shutdown is achieved with 1 SLC pump injecting 43 gpm of sodium pentaborate into the reactor before containment integrity is threatened. Failure implies the operators did not initiate SLC soon enough or that both SLC pumps failed to operate such that SLC could not inject to avoid threatening containment integrity.

X Success or failure of emergency RPV depressurization. Success implies that the operator manually depressurized the vessel with at least 4 SRVs. Failure implies that the operator failed to manually emergency depressurize with the required number of SRVs causing the reactor vessel to remain at a higher pressure than that required for successful core recovery with low pressure injection.

V Success or failure of adequate core reflood with a low pressure system. Success implies that at least one of the following ECCS systems was successfully aligned and slowly throttled to avoid a large power excursion which might result in core damage, and yet initiated in time to recover the core before the fuel clad temperature becomes excessive. Failure implies that these ECCS pumps were not properly aligned or controlled per the the PEIs.

o Low Pressure Coolant Injection Train A

o Low Pressure Coolant Injection Train B

o Low Pressure Core Spray

V' Success or failure of not overfilling the vessel after shutdown. Success implies the operator slowly increases vessel level to the normal water level control band. Failure implies the operator overfills the vessel and flushes out the boron and returns the vessel to a critical conditions.

W Success or failure of long-term containment heat removal with one loop of RHR to maintain containment pressure below the Containment Capacity Threshold Limit. Success implies that long-term heat removal can be achieved during 24 hours by at least one of the following modes. Failure implies that long-term heat removal by RHR is not maintaining containment pressure below the limit.

o One RHR train in Suppression Pool Cooling.

o One RHR train in Containment Spray.

Y Success or failure of long-term containment heat removal by venting to maintain containment pressure below the Containment Capacity Threshold Limit. Success implies that long-term heat removal can be achieved during 24 hours by at least one of the following modes. Failure implies that long-term heat removal by venting is not maintaining containment pressure below the limit.

o Containment Venting by Fuel Pool Cooling and Cleanup

o Containment Venting with RHR Containment Spray Headers

Cv Defines the susceptibility of the core to damage, dependent upon containment conditions. Success implies that given containment failure, the core is still maintained in a safe configuration. Failure implies that the containment failure has led to a degradation of plant systems such that the core is susceptible to damage.

The development of the functional fault trees for each of the event tree functions with the appropriate boundary conditions defined by the initiating events and preceding success or failures is described in Appendix E.

3.1.4 SEQUENCE GROUPING AND BACK END INTERFACE

3.1.4.1 Plant Damage State Grouping Criteria

The interface between the Level 1 Systems Analysis and the Level 2 Containment Analysis is defined by the plant damage state at the time of core melt. In order to avoid performing a full analysis of the phenomenology of each core melt sequence to determine the potential release of fission products the individual core damage sequences are assigned to one of the plant damage state (PDS) groups. Each group is defined by a set of functional characteristics for system operation which are important in terms of the accident progression, containment failure and source term definition. Each PDS contains Level 1 sequences with sufficient similarity in system functional characteristics such that the containment accident progression for all sequences in the group can be considered essentially the same.

The important functional characteristics for each PDS are developed in section 4.3. These characteristics are interpreted in terms of success or failure of the various systems used to prevent core damage and mitigate the release of the fission products. Therefore the core damage event trees are extended to include all systems which will impact fission product release. The sequence characteristics which are important to the progression of the accident in the order in which they are asked in the grouping process are shown in Figure 3.1.4-19 and are as follows.

1. Not a containment bypass sequence.

2. Containment status at core damage.

3. Event Type:

   3.1 For Containment Intact At Core Damage
   Station Blackout (SBO)
   LOOP With No HVAC
   Other Types than above

   3.2 For Containment Failed At Core Damage
   Critical (Non-Shutdown) ATWS
   Loss of Offsite Power and Station Blackout
   Others than above

4. Initial Containment Heat Removal With Suppression Pool Cooling (only for LOOP With No HVAC)

5. Containment Vent Isolated at RPV Failure (only for SBO)

6. RPV Injection Failure Time

7. Offsite Power Recovery Time

8. Containment Heat Removal With RHR Spray Loop

9. Containment Heat Removal With Vent

10. Late In-Vessel Injection and Pedestal Cavity Supply

11. RPV depressurized during core damage

Each of the above grouping criteria, the plant damage state characteristics and the binning results are fully described in section 4.3.

An examination of the above criteria shows that the event trees discussed in section 3.1.2 and 3.1.3 which are based on the success criteria for the prevention of the onset of core damage, do not include all the functions necessary to group them into the various plant damage states. For example where core damage is the result of failure of injection containment heat removal is not included. Similarly, following loss of offsite power only recovery to prevent core damage has been considered in the core damage event trees. It is necessary to modify the initial core damage event trees to include the functioning of the containment systems for all sequences. As the majority of the function in the plant damage state trees are those that lead to core damage only those functions which have been added in order to enable the PDS grouping to be performed are described in detail in this section. The additional containment systems added to the earlier event trees to give the new PDS trees are modeled in exactly the same way as the earlier functions. Fault trees are developed for each of the containment functions and all functions are linked to give combinations of failures leading to a given sequence of events. This will ensure that any dependencies between the system preventing core damage and those preventing containment failure will be correctly modeled.

3.1.4.2 Plant Damage State Event Trees

The amended event trees for all initiating events are shown in Figures 3.1.4-1 through 3.1.4-18. There are a number of basic functions which have been included in all the trees and a further group of functions only applicable to the loss of offsite power and station blackout trees. The functions applicable to all trees are associated with decay heat removal, depressurization and late injection and are discussed first, followed by a discussion of events only applicable to loss of offsite power and station blackout scenarios.

Containment Heat Removal (Wc, Ws)

The containment heat removal function has been divided into two in order to differentiate between sequences where containment heat removal is achieved through the containment spray system and those where the containment spray system is not operational but containment heat removal is achieved through suppression pool cooling.

Wc is defined as failure of containment heat removal as the result of failure of the containment spray mode of RHR or supporting systems.

Ws is defined as failure of containment heat removal as the result of failure of the suppression pool cooling mode of RHR or supporting systems following the failure of Wc.

Reactor Pressure Vessel Depressurization (X3)

In a number of sequences it is postulated that core damage will occur at high pressure as the result of failures associated with the containment or as the result of the failure of the operator to depressurize in time for low pressure injection systems to inject and prevent core damage. Late depressurization, after core damage, but before vessel failure will have a significant impact on fission product release so this function has been added to all high pressure core damage sequences.

Late Vessel Injection (Li)

If late depressurization is successful then it is possible to inject using low pressure systems with the potential for achieving in vessel debris cooling. In some cases late injection into the containment is also significant. This function has therefore been added to the event tree for all success branches of the function X3.

It has also been added as a special event in the Large and Intermediate LOCA event trees to reflect the fact that alternative low pressure systems which are not available in the short term to prevent core damage could be aligned later to provide invessel cooling of the debris.

Loss of Offsite Power and Station Blackout

In the event of a loss of offsite power or station blackout actions in addition to those discussed can be taken to mitigate the release of fission products. Probably the most important of these is the recovery of offsite (or onsite) power before vessel or containment failure. Each of the functions is discussed in the following paragraphs.

R3 Recovery of offsite power prior to vessel failure. The time available for the recovery of power is dependent upon the specific sequence. The timing for this function is sequence dependent - See Appendix E.

R4 Recovery of offsite power prior to containment failure. The time available to recover offsite power prior to containment failure is considerably longer than that available prior to vessel failure. Recovery will enable containment systems to be restarted and therefore reduce the frequency of containment failure. The time available for recovery is sequence dependent and discussed in Appendix E.

Y2 Long term containment heat removal with venting. As this has not been asked earlier in some sequences it is necessary to include it in the trees. In order to avoid having the venting function appear twice with the same identifier, the second time it is used it is given the designator Y2. The functional requirements are the same as for Y.

Wt Long term containment heat removal with suppression pool cooling. This is the same function as Ws, but again is required at a different time for certain sequences, thus, as in the case of the function Y a different designator is used.

I Isolation of the containment. If power is lost initiation of the Fuel Pool Closed Cooling can only be achieved by the operator or restoration of electrical power. This function models failure to achieve this following a station blackout.

Val Alternate Low Pressure makeup. This is the same function as Va but as in the case of Y and Ws above requires to be modeled at a different position in the development of the sequences.

3.1.4.3 Plant Damage State Sequences

A review of the plant damage state event tree will reveal that not all the core damage sequences in the core damage event trees have been included in the PDS trees. In order to avoid analysis of sequences which do not contribute significantly to the core damage or offsite release all sequences below 1.0×0^{-8} were excluded from the PDS evaluation. In order to ensure that any sequence omitted did not pose a risk to the containment significantly different from those included the characteristics of all excluded sequences were reviewed and compared with those which were included. The resulting PDS trees contain sequences which contribute 99.9% to the overall core damage frequency.

The plant damage state to which a sequence contributes is shown in the status column on the event tree and the grouping logic used to define the plant damage states is shown in Figure 3.14-19. The characteristics of the individual plant damage states and the deriviation of the grouping logic is discussed fully in section 4.3. The contribution of each sequence to a specific plant damage state is also discussed in this section and in the summay of the quantification is section 3.4.

3.2 SYSTEM ANALYSIS

The Perry IPE was performed using the "small event tree - large fault tree" approach, as described in NUREG/CR-2300 (Hickman, 1983) In using this type of method, the fault tree analysis of the plant systems becomes the major underlying task of the Level 1 analysis.

The analysis of the initiating events and the required plant response, to maintain decay heat removal and containment integrity, result in the identification of the safety functions required to achieve these aims. Each of the safety functions can be achieved by either automatic operation of a system or a combination of system operation and operator actions. By analyzing the system requirements in response to each of the groups of initiating events it is possible to identify the response required from each individual system. To summarize, the accident sequence analysis results in the identification of the functional requirements which in turn are translated into individual system requirements. Each individual system requirement is then the starting point for the fault tree development for that system. In the majority of cases the success criteria do not specifically identify the requirements for the support systems, that is electric power, room cooling, motor cooling, instrument air, etc. The top event for the fault trees for these systems, are determined by the requirements identified in the development of the frontline system fault trees.

The system analysis was conducted in accordance with the task plan which provided specific guidance on the modeling of components and the general assumptions to be made when developing a fault tree. This included specifying component boundaries, level of detail of modeling, guidelines for restoration errors, test and maintenance unavailability, etc. The task procedure was developed to meet all the applicable requirements of NUREG-1335 (NRC, 1989) and NUREG/CR-2300 for fault tree analysis.

Each system is briefly described in this section with details of the success criteria, the functions performed and the dependency matrix. The detailed analysis of each system is recorded in a series of analysis files specifically developed for the project. The fault trees are shown in Appendix A.

The quantification of individual fault trees has not been included in this section. The contribution of individual component and function failures to the overall core damage frequency is discussed in section 3.4.

3.2.1 RPV DEPRESSURIZATION SYSTEM, B21

The Automatic Depressurization System (ADS) is part of the Emergency Core Cooling System (ECCS) and its function is to reduce the pressure in the reactor pressure vessel in the event that the high pressure injection systems fail to perform their function. If the high pressure injection systems cannot maintain adequate core cooling, then ADS reduces the RPV pressure to a point at which the Low Pressure Core Spray (E21) and the Low Pressure Coolant Injection (E12) systems can be used to provide inventory make-up. The non-ADS safety relief valves can also be used to reduce the RPV pressure by opening each of the valves individually. Simplified diagrams for the system are shown in Figures 3.2.1-1 and 3.2.1-2.

3.2.1.1 System Description

Eight of the nineteen installed safety relief valves (SRVs) on the main steam lines are used for the ADS function. There are two ADS valves in each of the main steam lines A through D. These valves are mechanically self actuating under conditions of high reactor steam pressure (to prevent overpressure in the vessel), or are electrically actuated either manually or by relay logic circuits to provide pressure relief (low-low set or relief mode) or depressurization if the low pressure injection systems are required for inventory make-up.

The eleven installed SRVs not used for ADS are also mechanically self actuating under conditions of high reactor steam pressure or may be electrically actuated either manually or by relay logic circuits to provide pressure relief (low-low set or relief mode).

ADS operation is achieved by actuation of one of two solenoid valves which will allow air from the safety related instrument air system (P57) into the pneumatic operating cylinder of the ADS valve, thus opening the ADS valve. There are two accumulators associated with each ADS SRV. These accumulators ensure an adequate air supply in the event of a safety related instrument air system failure. The accumulator capacity is sufficient to provide two valve actuations during accident conditions.

The discharge from each of the ADS and non-ADS valves is piped to the suppression pool, with the discharge line exhaust submerged to ensure steam condensation within the suppression pool whenever a valve operates.

3.2.1.2 System Operation

Automatic initiation of ADS will occur if the following conditions occur: a) low reactor water level, Level 3: b) low reactor water level, Level 1: c) time delay of 105 seconds after low level is reached - this timer can be reset manually and is automatically reset every time level is raised above Level 1: and d) availability of the LPCS or LPCI pumps. Automatic operation can be overridden if required.

ADS logic can be started manually by placing the manual initiation switch collars in the armed position and depressing the switches. This manual initiation signal bypasses both the level requirements for initiation and the 105 second timer. A LPCI/LPCS pump must still be running in order to open the ADS valves. Both the ADS and non-ADS sets of valves can be individually opened by the operators from the control room.

3.2.1.3 System Interface and Dependencies

The system interfaces and dependencies are shown in Table 3.2.1-1.

The ADS depends on the safety related Instrument Air System for opening the valves and on DC buses ED-1-A and ED-1-B for valve operation and control.

The non-ADS SRVs depend on the Instrument Air system for valve operation and on the ED-1-A and ED-1-B buses for valve operation and control.

3.2.1.4 Success Criteria

Success of the ADS implies that at least four of the eight ADS valves open when all the present conditions are satisfied. Valves can be opened either by automatic or manual initiation. Given failure of the above, credit will be taken for the manual opening of at least four of the remaining 11 non-ADS Safety Relief Valves. The fault trees developed for this system are one of the inputs to the failure to the depressurize function in the event trees (X, X2, X3). The exact relationship is shown in the development of the function fault trees in Appendix E.

3.2.2 STANDBY LIQUID CONTROL, C41

The standby liquid control system shuts down the reactor by pumping a neutron absorbing solution (sodium pentaborate) into the reactor pressure vessel in sufficient concentration and quantity to provide the required reactor shutdown margin without control rod movement and to overcome the maximum positive reactivity resulting from cooldown and xenon decay. A simplified diagram for the system is shown in Figure 3.2.2-1.

3.2.2.1 System Description

The standby liquid control system takes suction from a storage tank containing a highly concentrated solution of sodium pentaborate. Two parallel positive displacement pumps rated at 43 gpm are available to inject the solution into the reactor pressure vessel. Each pump suction contains a normally closed motor operated valve. The pump suctions are crosstied downstream of the motor operated valves to ensure suction to both pumps in the event that one of the suction valves fails closed. Two parallel explosive valves are downstream of the pump discharge. The pump discharges are also crosstied upstream of the explosive valves to ensure a flow path to both pumps in the event that one of the explosive valves fails. Downstream of the explosive valves, the A and B trains combine to provide a single injection line to the reactor pressure vessel.

Reactor water cleanup receives an isolation signal upon initiation of the standby liquid control system to prevent depletion of the boron solution in the reactor, following injection.

3.2.2.2 System Operation

There is no automatic actuation of the standby liquid control system. However, the storage tank outlet valves open, the pumps start, and the explosive valves fire by moving the keylock switches for each train to "ON".

3.2.2.3 System Interface and Dependencies

The system interfaces and dependencies are shown in Table 3.2.2-1.

The storage tank outlet valves and the SLC pumps are dependent on the associated diesel backed 480 VAC buses (EF-1-A and EF-1-B). The explosive valves receive 120 VAC power from both divisions 1 and 2. A loss of a single division therefore would not fail either of the explosive valves.

Standby liquid control is dependent upon the isolation of reactor water cleanup. If left operating, the reactor water cleanup system would deplete the boron concentration in the reactor pressure vessel after injection.

In addition to AC power, the standby liquid control system is dependent on the heat tracing and antifreeze protection system to maintain adequate temperature in the SLC pump suction lines and the storage tank. The two-bed demineralized water system provides a keep fill function for the standby liquid control system. The instrument air system provides air for mixing during the addition of chemicals into the storage tank. None of these systems were considered necessary for the injection of sodium pentaborate into the reactor pressure vessel and were not incorporated into the logic model.

3.2.2.4 Success Criteria

A single pump with a suction flow path from the storage tank and a discharge flowpath to the reactor pressure vessel is sufficient to shutdown the reactor. The reactor water cleanup valves must also be closed for success. The fault tree developed for this system is one of the inputs to the failure of the standby liquid control function in the event trees (C1). The exact relationship is shown in the functional fault trees in Appendix E.

3.2.3 RESIDUAL HEAT REMOVAL, E12

The functions of the Residual Heat Removal (RHR) system are to remove heat from the reactor vessel and suppression pool under normal and accident conditions, to automatically restore and maintain the desired water level in the reactor vessel following a loss of coolant accident (LOCA), and to control or reduce pressure in the reactor and the containment following a LOCA. Simplified diagrams of the system are shown in Figures 3.2.3-1 through 3.2.3-7.

3.2.3.1 System Description

RHR is a 3 train system consisting of motor operated valves and motor driven pumps. The 3 pumps are rated at 7,260 gpm at 125 psi. Trains A and B each have 2 heat exchangers in series downstream of the pumps. Cooling water to the heat exchangers is not required for the Low Pressure Coolant Injection (LPCI) mode of operation. Train C has no heat exchangers and is dedicated to the LPCI mode of operation. The RHR pumps take suction from the suppression pool for the low pressure coolant injection, suppression pool cooling and containment spray modes of operation.

Flow through the pump minimum flow lines is limited to 1,650 gpm by valves F0018A/B/C.

3.2.3.2 System Operation

The following modes of operation are considered in preventing core damage following a plant trip from power.

Low Pressure Coolant Injection

LPCI involves restoration of the water level in the reactor vessel to a height sufficient to provide adequate core cooling after a LOCA. LPCI is a low pressure, high flow system which automatically starts giving flow from all 3 pumps to the reactor vessel when the vessel water level decreases to Level 1 or the drywell pressure increases to 1.68 psi. One pump is sufficient to maintain the integrity of the fuel cladding.

During LPCI operation all 3 pumps draw water from the suppression pool. The bypass valves around the heat exchangers open and may not be closed for 10 minutes because heat rejection is not required during the time it takes to flood the reactor. Water flows to the reactor vessel through the LPCI injection valves.

Containment Spray

The containment spray mode is used to remove heat from the containment following a LOCA. This mode uses either train A or B pumps to pump water from the suppression pool through the heat exchangers and out of the nozzles in the ring spray headers in the dome of the containment. The spray condenses any steam that may exist in the containment thus reducing containment pressure. The water returns to the suppression pool through drainage.

The containment spray mode is actuated automatically 10 minutes after LPCI initiation if the drywell pressure is 1.68 psig (setpoint) and the containment pressure is 22.42 psia (setpoint). Operating procedures, however, instruct the operators to bypass automatic actuation and manually initiate containment spray as required. Injection to the reactor vessel is isolated for the train in the containment spray mode of operation.

Suppression Pool Cooling

The suppression pool cooling mode is used to maintain the suppression pool less than 90°F during normal plant operation to ensure adequate condensation of the steam in the event of a LOCA. It may also be used to reduce suppression pool temperature following Reactor Core Isolation Cooling (RCIC) or Main Steam Safety Relief Valve (SRV) operation and following a LOCA.

Either train A or B may be used for this mode. The flow path is from the suppression pool to the pump through the heat exchangers and back to the suppression pool through the test return valve.

Operator action is required to place RHR into suppression pool cooling mode of operation.

RHR Loop B Containment Flooding

This mode of operation may be used to flood the containment with water from the Emergency Service Water (ESW) system. Two manual ESW inter-tie valves are opened with handwheels to admit ESW water to RHR loop B downstream of heat exchanger outlet valve. The water then flows to the reactor vessel through the LPCI injection line and out of the break thereby flooding the containment.

The systems can be manually aligned to the three principle modes as follows:

LPCI - The RHR system may be manually aligned to the LPCI mode of operation by arming and depressing push-button 1E21-S9 for train A or 1E12A-S21 for trains B and C.

Containment Spray - The RHR system may be manually aligned to the containment spray mode of operation by arming and depressing push-button 1E12A-S63A for train A or 1E12A-S63B for train B. Manual actuation of containment spray bypasses the time delay.

Suppression Pool Cooling - The RHR system must be manually aligned to the suppression pool cooling mode of operation by closing valves F004A/B and F0027A/B and opening valve F0024A/B.

3.2.3.3 System Interface and Dependencies

The system interfaces and dependencies are shown in Table 3.2.3-1.

The major system dependencies are DC control power for actuation. AC power for operating the pumps and valves, cooling to the pumps, and pump room cooling. AC and DC power is divisionally separated with train A being dependent on Division 1 and trains B and C being dependent on Division 2.

In addition to the dependencies common to all the modes of operation, the train A LPCI actuation sensors are shared with the Low pressure Core Spray system. Trains B and C share LPCI actuation sensors and instrumentation.

Although the containment spray mode of operation is normally manually initiated, the system is designed to be automatically initiated by high containment and drywell pressures with a 10 minute time delay. At the end of 10 minutes if the high pressures still exist, the LPCI injection valve, F0042A will be closed and the containment spray valves F0028A and F0537A will be opened. The train B valves will operate 35 seconds after the train A valves.

The suppression pooling cooling mode is manually initiated. If either a LPCI injection or a containment spray signal is generated after initiation of the suppression pool cooling mode of operation, the system will automatically realign to the respective mode of operation demanded by the signal.

3.2.3.4 Success Criteria

Low Pressure Coolant Injection

Injection to the reactor pressure vessel from anyone of loops A, B or C with the flow path through the heat exchangers or the heat exchanger bypass valve is considered a success. The fault trees developed for each train form the primary input to function V as shown in Appendix E.

Containment Spray

Flow to the containment spray headers from either loop A or B with the flow path through the heat exchangers is considered a success. The fault trees developed for each train form one of the primary inputs to function W as shown in Appendix E.

Suppression Pool Cooling

Flow to the suppression pool from either loop A or B with the flow path through the heat exchangers is considered a success. The fault trees developed for each train form one of the primary inputs to function W as shown in Appendix E.

3.2.4 LOW PRESSURE CORE SPRAY, E21

The purpose of the Low Pressure Core Spray (LPCS) system is to automatically provide coolant to the reactor pressure vessel during accidents when the pressure is low. The Automatic Depressurization System (ADS) can be used in conjunction with LPCS to attain a low enough system pressure for injection to occur. A simplified system diagram is shown in Figure 3.2.4-1.

3.2.4.1 System Description

The LPCS system consists of a single train with motor-operated and manual valves and a motor driven pump. Suction is taken from the suppression pool through a suction strainer and injection is through the reactor pressure vessel via a sparger which surrounds the core just inside the core shroud.

The LPCS pump is designed to operate with a maximum discharge head of 520 psig. The pump is designed to deliver rated flow of 6,110 gpm at a discharge head of 128 psig within 40 seconds of an initiation (including diesel generator start and pump start time delay).

3.2.4.2 System Operation

The LPCS system is automatically initiated and controlled. Automatic initiation occurs at a low reactor pressure vessel water level of Level 1 or a high drywell pressure of 1.68 psig. This energizes the pump and aligns the valves.

Normally no operator actions are required for system initiation. However, if automatic initiation fails for any reason, the LPCS system can be manually initiated by depressing a manual initiation push button. This will energize the LOCA relays which will automatically cause the above mentioned automatic actions to occur.

3.2.4.3 System Interface and Dependencies

The system interfaces and dependencies are shown in Table 3.2.4.-1.

The LPCS system major dependencies are DC control power for initiating the actuation relay logic and LPCS pump breaker, AC power for operating the LPCS pump and valves, and LPCS pump room cooling.

The DC power is provided by Division 1 125VDC bus ED-1-A. Power for the LPCS pump is provided by Division 1 4,160VAC Bus EH-11, and power for the valves and room cooler is provided by Division 3 480VAC bus EF-1-A.

LPCS room cooling is provided by the ECCS Pump Room Cooling System, M39. Room cooling is not required for the initiation of the system. However, long term LPCS operation is dependent upon the room cooling system to maintain the environmental conditions within the qualified limits.

3.2.4.4 Success Criteria

Success of the LPCS system implies that either LPCS was automatically actuated at Level 1 or that it was manually actuated, and that coolant make-up to the reactor pressure vessel is being carried out at the rated flow. The fault tree developed for this system is one of the inputs to the failure of the low pressure injection system function in the event trees (V). The exact relationship is shown in the functional fault trees in Appendix E.

3.2.5 HIGH PRESSURE CORE SPRAY, E22

The purpose of the High Pressure Core Spray (HPCS) system is to automatically provide coolant to the reactor vessel during accidents when the pressure remains high. Sufficient water inventory is maintained until the reactor is depressurized to a level where the Low Pressure Core Spray and Low Pressure Coolant Injection systems can be placed into operation. The HPCS system is also capable of performing the functions of the Reactor Core Isolation Cooling (RCIC) system in the event of RCIC system failure. A simplified diagram of the system is shown in Figure 3.2.5-1.

3.2.5.1 System Description

The HPCS system consists of a single train with motor-operated valves and a motor driven pump. Suction is taken from two possible water sources. The Condensate Storage Tank (CST) is the preferred source. When a low level in the CST or a high level in the suppression pool is sensed, suction is automatically aligned to the suppression pool. Injection to the reactor vessel is via a sparger which surrounds the core just inside the core shroud.

The HPCS system is designed to pump water into the reactor vessel over a wide range of pressures. When the system is started, initial flow rate is established by primary system pressure. For reactor pressures of 1,177 psid (differential pressure between the reactor vessel and the suction source), the minimum rated flow is 517 gpm. As vessel pressure decreases flow will increase. When vessel pressure reaches 200 psid the system rated core spray flow is 6,110 gpm. A restricting orifice in the pump discharge limits flow to a maximum designed runout flow of 7,800 gpm with reactor depressurized. The HPCS pump and motor are capable of delivering rated flow at full reactor pressure within 27 seconds of an initiation. This includes the time for the Division 3 diesel generator to start and supply power to bus EH-13 if necessary.

3.2.5.2 System Operation

The HPCS system is initiated automatically if a low reactor vessel water level of Level 2 or a high drywell pressure of 1.68 psig is sensed. This energizes the pump, starts the HPCS diesel generator, and aligns the valves.

Since the CST is designed to be the initial suction source for the HPCS system, the CST suction valve, F001, is normally open and F0015, the suppression pool suction valve, is closed. Suction is automatically switched to the suppression pool upon either low CST level or high suppression pool level. The CST suction valve closes when the suppression pool suction valve is fully open.

The HPCS system is automatically isolated when the reactor water level reaches Level 8. At this pint, the HPCS injection valve closes and the minimum flow valve to the suppression pool opens. The HPCS pump continues to run. A Level 2 signal will reopen the valve.

3.2.5.3 System Interface and Dependencies

The system interfaces and dependencies are shown in Table 3.2.5-1.

The HPCS system major dependencies are DC control power for initiating the actuation relay logic and HPCS pump breaker, AC power for operating the HPCS pump and valves, and HPCS pump room cooling.

The DC power is provided by Division 3 125VDC bus ED-1-C. Power for the HPCS pump is provided by Division 3 4,160VAC Bus EH-13, and power for the valves and room cooler is provided by Division 3 480VAC bus EF-1-E. It should be noted that Division 3 electric power (AC and DC) is dedicated to HPCS and its supporting systems.

HPCS room cooling is provided by the ECCS Pump Room Cooling System, M39. The system is not required for the initiation of the system. However, long term operation is dependent upon the room cooling system to maintain the environmental conditions within the qualified limits.

The HPCS and RCIC systems share a common CST suction valve F0518. This is a normally open manual valve. Failure of this valve will fail the CST as a suction source to both HPCS and RCIC.

3.2.5.4 Success Criteria

Success of the HPCS system implies that either HPCS was automatically actuated at Level 2 or that it was manually actuated, and that coolant make-up to the reactor vessel is being carried out at the rated flow. It also implies that switchover from CST suction to suppression pool suction is carried out when CST level is low. The fault tree developed for this system is one of the inputs to the failure of the high pressure core spray function in the event trees (U1). The exact relationship is shown in the functional fault tree in Appendix E.

3.2.6 REACTOR CORE ISOLATION COOLING, E51

The purpose of the Reactor Core Isolation cooling (RCIC) System is to ensure that sufficient reactor water inventory is maintained in the reactor vessel during vessel isolation conditions to permit adequate core cooling to take place.

During accident conditions, the RCIC system is capable of providing coolant injection into the RPV for high pressure accident scenarios. A simplified diagram for the system is shown in Figure 3.2.6-1.

3.2.6.1 System Description

The RCIC system is a single train system which consists of a steam-driven turbine, an associated pump assembly, valve, and instrumentation. Suction is taken from either the condensate storage tank (CST), or the suppression pool. The RCIC pump discharges the water through a spray nozzle inside the reactor pressure vessel (RPV) head.

The RCIC pump is capable of providing a constant 725 gpm flow rate, for RPV pressures ranging between 1,192 psia and 165 psia. MAAP (Modular Accident Analysis Program) assumes RCIC operation down to 20 psig. 25 gpm of the system flow is recirculated to the RCIC pump lube oil cooler, providing 700 gpm injection into the RPV.

3.2.6.2 System Operation

During normal reactor operation, the RCIC system is in standby configuration. The system can be activated and shutdown manually or automatically. Automatic initiation occurs on a low reactor water level (Level 2). Automatic initiation of the RCIC system causes a trip of the main turbine and the Reactor Feed Pump Turbines. In addition, automatic alignment of required RCIC components will occur. Required cooling systems will receive concurrent automatic initiation signals on low reactor water level.

The RCIC system can be manually initiated by plant operators by arming and depressing the RCIC manual initiation switch. Plant and system response will be the same as for automatic initiation.

The RCIC suction supply can be manually transferred between the CST and the suppression pool.

The operators are directed to override the RCIC suction shift from the CST to the suppression pool for a Station Blackout. If the transfer has occurred, they are directed to return the suction configuration to the CST. The above manual actions are noted as they are specifically modeled in the RCIC fault tree.

3.2.6.3 System Interface and Dependencies

The system interfaces and dependencies are shown in Table 3.2.6-1.

The major RCIC dependency is upon DC power. By design, RCI is capable of initiation and operation independent of AC power, plant service air, and any external cooling water systems. All components and control systems necessary for the initiation of the RCIC system are powered from the plant emergency DC electrical system.

Other system components are supplied from emergency and non-emergency backed AC power supplies. None of these components are required for system initiation. The emergency backed components are typically associated with isolation requirements. The non-emergency backed components are not required for any safety or reliability based system function.

Instrument Air is provided only to RCIC steam supply and exhaust drain line valves.

RCIC room cooling is provided by the ECCS Pump Room Cooling System, M39. The system is not required for the initiation of the system.

3.2.6.4 Success Criteria

RCIC is assumed to be a success path for the following initiating events:

1. Small LOCA (S2)
2. Loss of PCS Transient (T2)
3. PCS Available Transient (T3A)
4. Loss of Feedwater Transient (T3B)
5. Loss of Offsite Power (T1)
6. Station Blackout (T1-B)

For each of the initiating events in which RCIC is considered a success path, the RCIC system is required to provide full flow (700 gpm) into the reactor vessel for a twenty-four hour accident duration. The fault tree developed for this system is one of the inputs to the failure of the reactor core isolation cooling function in the event trees (U2). The exact relationship is shown in the functional fault tree in Appendix E.

3.2.7 EMERGENCY CLOSED COOLING, P42

The ECC system provides cooling to selected loads for selected modes of normal reactor operation as well as during a loss of coolant accident (LOCA) or a loss of offsite power (LOOP). Motor operated valves automatically align on a LOCA or LOOP signal to provide cooling to the control complex chillers. These chillers normally receive cooling water from the Nuclear Closed Cooling (NCC) system, P43. A simplified diagram of the system is shown in Figures 3.2.7-1 and 3.2.7-2.

3.2.7.1 System Description

The ECC system is divided into two independent loops each consisting of a pump, heat exchanger, surge tank, valves and interconnecting piping. A chemical addition tank is shared by the two loops. The ECC heat exchangers are cooled by the Emergency Service Water system. The pumps and motor operated valves receive power from diesel backed buses.

The ECC is designed to yield a maximum expected equipment cooling water temperature of 95°F.

ECC is designed such that in the event of a single active or passive failure in the system, cooling water can be supplied to either of the Engineered Safeguards Features divisions.

3.2.7.2 System Operation

ECC is required to supply equipment cooling to the Residual Heat Removal pumps and room coolers and the Reactor Core Isolation Cooling room cooler during reactor isolation in hot standby. The initiation of the system for this mode of operation is a remote-manual function.

ECC is required to supply cooling water to support the Residual Heat Removal system in the normal shutdown mode. The initiation of ECC for this mode is a manual operation and is dependent on the specific RHR system requirements for cooling water.

The ECC system provides cooling to selected loads during a loss of coolant accident (LOCA) or a loss of offsite power (LOOP). Motor operated valves automatically align on a LOCA or LOOP signal to provide cooling to the control complex chillers. These chillers normally receive cooling water from the Nuclear Closed Cooling (NCC) system, P43.

ECC is designed such that in the event of a single active or passive failure in the system, cooling water can be supplied to either of the Engineered Safeguards Features divisions.

3.2.7.3 System Interface and Dependencies

The system interfaces and dependencies are shown in Table 3.2.7-1.

The system is dependent upon the following electrical supplies:

EF-1-A (A valves)	ED-1-A (Pump A controls)
EF-1-B (Pump A)	ED-1-B (Pump B controls)
EF-1-C (B valves)	
EF-1-D (Pump B)	

The ECC major dependencies are DC control power for initiation logic and control logic, AC power for pumps and valves, and Emergency Service Water. Dependencies on Instrument Air, P52, and Two-Bed Demineralized Water, P21, or Emergency Service Water, P45, for make-up water were not needed for the fault tree models developed. The dependency on Emergency Pump Area Cooling for HVAC was not incorporated into the fault tree model as not being needed for the duration of the event.

DC power for Division 1 control logic and initiation logic is supplied by Bus ED-1-A. Bus ED-1-B supplies DC power to the Division 2 control logic and initiation logic. Because train A ECC is initiated during Reactor Core Isolation Cooling, there is an initiation signal from Division 2 to train A ECC. AC power is supplied from divisionally separated AC 480 volt buses.

3.2.7.4 Success Criteria

ECC is required to supply cooling water for 24 hours following a LOCA or transient. The fault tree developed for this system provides inputs to the frontline systems.

3.2.8 EMERGENCY SERVICE WATER, P45

The Emergency Service Water (ESW) system supplies cooling water to equipment (including RHR, diesel generators, and emergency closed cooling (tube side of heat exchanger)) required for normal and emergency shutdown of the reactor. It can also provide water to Fire Protection, Fuel Pool Cooling and Cleanup, Emergency Closed Cooling (shell side of heat exchanger), Residual Heat Removal (for containment flooding), Standby Liquid Control, and the ESW Screen Wash (for deicing). A simplified diagram of the system is shown in Figure 3.2.8-1.

3.2.8.1 System Description

The source of water for the ESW system is Lake Erie. The ESW system is made up of three independent trains. Each train consists of a motor driven pump, motor operated valves, and heat exchangers. Train C is dedicated to the High Pressure Core Spray system.

The Loop A and B pumps are designed for 12,900 gpm and are driven by 800 hp 4,160VAC motors powered from EH-11 and EH-12, respectively. The Loop C pump is designed for 960 gpm and is driven by a 75 hp 480VAC motor powered from EF-1-E.

The system is designed to operate with a maximum water temperature of 85°F.

The ESW pumps are all located in the Emergency Service Water Pumphouse. Because of the relative location of the system components, local access to the ESW system would not be affected by either containment venting or failure. Because of the large size of the pumphouse and the presence of louvers on the walls, ample room ventilation was assumed available and a loss of ESW Pumphouse Ventilation, M32, was not considered to fail the pumps.

3.2.8.2 System Operation

ESW loops A and B automatically start within 18.5 seconds following receipt of any of the following signals: Division 1 or 2 diesel starts, or drywell high pressure, or RPV level 1, or loss of offsite power, or (for loop A only) initiation of RCIC.

ESW loop C automatically starts 28 seconds after receipt of any of the following signals: Division 3 diesel start, or high drywell pressure, or RPV level 2.

ESW loop B can be manually aligned to RHR loop B to provide the capability to flood the containment.

ESW loops A and B can be manually aligned to provide an emergency make-up to the ECC surge tanks for the respective loops.

ESW loop B can be manually aligned to provide an emergency make-up to the SLC surge tank for the respective loops.

3.2.8.3 System Interface and Dependencies

The system interfaces and dependencies are shown in Table 3.2.8-1.

The Emergency Service Water system major dependencies are DC control power to the pumps and AC power for operating the valves and pumps. ESW train A receives DC power form ED-1-A, train B from ED-1-B, and train C from ED-1-C. AC power is supplied from divisionally separated sources.

Dependencies on Instrument Air were not needed for the fault trees and not developed. The dependency on ESW Pumphouse Ventilation system was not incorporated into the ESW fault tree model as not being needed for the duration of the events.

3.2.8.4 Success Criteria

ESW must supply water for 24 hours following a LOCA or transient. The fault trees developed for this support system provide inputs to the front line systems and also one of the inputs to the failure of the low pressure injection (ESW crosstie) function in the event trees (Va). The exact relationship is shown in the functional fault tree in Appendix E.

3.2.9 SERVICE/INSTRUMENT AIR SYSTEMS, P51/P52

The purpose of the Instrument Air system is to supply clean, dry, oil free air to various control and instrumentation functions throughout the plant. For the IPE, the purpose of the Service Air system is to supply air to the Instrument Air system. A simplified diagram of the service and instrument air systems is shown in Figure 3.2.9-1.

3.2.9.1 System Description

The Service Air and Instrument Air system each consist of two separate trains, one for each Unit. The Service Air trains are interconnected to through two motor operated valves (1P51-F0090 and 2P51-F0090). The Service Air trains are also connected to the Instrument Air trains through check valves (1P51-F0532 and 2P52-F0532) and pneumatic valves (1P52-F0050 and 2P52-F0050). The Instrument Air trains are interconnected through two motor operated valves (1P52-F0210 and 2P52-F0210). Each train of the Service Air and Instrument Air systems consist of an air compressor and a receiver tank. Compressed air flows from the receiver tank through the distribution piping to components in the plant.

3.2.9.2 System Operation

Under normal plant operating conditions, the instrument air for both Units will be supplied by one Service or Instrument Air compressor. Preferred system configuration is to have one compressor running with at least one compressor in standby. The preferred alignment is to have a compressor from the opposite unit in standby to ensure an air supply on a loss of either the

Unit 1 or 2 supply bus. Normal system operating pressure range is 120-130 psig.

3.2.9.3 System Interface and Dependencies

The system interfaces and dependencies are shown in Table 3.2.9-1.

The major dependencies of the Service and Instrument Air systems are electrical power and Nuclear Closed Cooling.

Service and Instrument Air systems are non-safety systems and thus are not supplied by diesel backed electrical power. The exception is for the Instrument Air Containmert and Drywell isolation valves which are safety related.

The Unit 1 Service and Instrument Air Compressors are supplied from 4.6KV BUS H-12. The Unit 2 compressors are supplied from Bus H-22. The 480VAC, 120VAC, and 125VDC supplied Service and Instrument Air components are powered from the corresponding Units 4.16KV Bus.

Component cooling is provided by the Nuclear Closed Cooling System. Room cooling for the systems is provided by the Controlled Access and Miscellaneous Equipment Area HVAC System (M21). Cooling under accident conditions is also provided indirectly by the Emergency Closed Cooling Pump Area Cooling System (M28) due to the proximity of the compressors to the Emergency Closed Cooling System. It is assumed for this analysis that neither room cooling system is required for Instrument and Service Air System success for the accident duration.

3.2.9.4 Success Criteria

The success criteria for Service and Instrument Air is not dependent upon plant initiators.

The system is required for long-term successful operation of the non-ADS Safety Relief Valves (SRVs). System success is defined as provided at least one operable compressor and a flow path to the SRV Instrument Air distribution system.

The fault trees developed for the P51/P52 systems are linked to other systemic fault trees as they are support systems.

3.2.10 SAFETY RELATED INSTRUMENT AIR SYSTEM P57

The purpose of the Safety Related Instrument Air (SRIA) System is to supply clean, dry, oil free air to the Automatic Depressurization System (ADS), Safety Relief Valve Accumulators and to the accumulator for one non-ADS SRV (B21-F051D). A simplified diagram of the system is shown in Figure 3.2.10-2.

3.2.10.1 System Description

The Safety Related Instrument Air System consists of an air compressor, air filters, air receiver tanks, two air storage tanks, and two separate air headers for distribution. The "A" header charges the four accumulators associated with the ADS relief valves and the one non-ADS valve. The "B" header charges the four accumulators associated with the other four ADS valves.

The Safety Related Instrument Air System fulfills its normal function without operator action. The ADS Storage Tanks supply make-up air to the ADS accumulators for any leakage from the ADS system. When air pressure decreases to approximately 160 psig, the air compressor will automatically start to recharge the system to approximately 170 psig and will automatically stop.

The ADS Storage Tanks provide a volume of 1,350 cubic feet of compressed air. Extrapolation of an engineering design calculation indicates that up to 336 actuations per train can be achieved. This accounts for system leakage over a seven day period. Pipe fittings are provided for temporary connection of portable air cylinders to each header as a backup source of compressed air. Once connected, these cylinders would be used to restore system pressure and then disconnected.

3.2.10.2 System Operation

In normal operation, the Safety Related Instrument Air System Compressor will be instandby configuration, cycling on and off to maintain system pressure between 160 psig and 170 psig.

In standby configuration, the compressor will auto start when system pressure decreases to approximately 160 psig. The compressor will automatically shutdown when system pressure increases to approximately 170 psig.

3.2.10.3 System Interface and Dependencies

The system interfaces and dependencies are shown in Table 3.2.10-1.

The only major dependency of the Safety Related Instrument Air System is upon the plant electrical system. The containment and drywell isolation valves are supplied by diesel-backed electrical power. 480VAC Bus EF-1-A supplies valves 1P57-F015A and 1P57-F020A. 480VAC Bus EF-1-C supplies valves 1P57-F015B and 1P57-F020B. The compressor is supplied from a non-emergency power supply (Bus F-1-C).

No component cooling system is required for system operation.

Room cooling is provided by the Auxiliary Building HVAC System (M38) and the Intermediate Building Ventilation System (M33). The HVAC systems were not modeled.

3.2.10.4 Success Criteria

System success is defined per individual train of the Safety Related Instrument Air System.

Individual train success requires the operability of a flow path from the air storage tank to the ADS valves for the event duration. However for long term accident scenarios, success criteria will additionally require the operability of the Safety Related Instrument Air Compressor or connection of air cylinders to the air system.

The system is required for successful operation of the ADS valves and the one non-ADS valve.

The fault trees developed for the P57 system are linked to the ADS systemic fault trees as P57 is a support system.

3.2.11 FIRE PROTECTION, P54

For the IPE the purpose of the Fire Protection system is to provide an alternate supply of water to the suppression pool or the reactor pressure vessel in the event that the Emergency Core Cooling systems are unavailable. A simplified diagram for the system is shown in Table 3.2.11-1.

3.2.11.1 System Description

Although the Perry Fire Protection system includes both a motor driven pump and a diesel driven pump, only the diesel driven pump will be included in the model. The Diesel Fire Service Pump with a design capacity of 2,500 gpm at 125 psig is directly coupled to a diesel engine through a right angle coupling with a non-reversing mechanism to prevent backspin of the diesel. The diesel engine is equipped with an electric starter which is supplied by 2 independent battery banks. A battery charger is supplied to automatically maintain the batteries in a fully charged condition. During operation the diesel control and alarm circuitry is powered by a shaft driven alternator.

3.2.11.2 System Operation

For the Perry IPE there are no automatic functions associated with the fire protection system.

The diesel driven fire pump is started and the valves aligned to enable fire water to be pumped into the vessel through the feedwater injection line.

3.2.11.3 System Interface and Dependencies

The system interfaces and dependencies are shown in Table 3.2.11-1.

The portion of the fire protection system modeled for the IPE is not dependent on any of the other support systems within the IPE model. The diesel driven pump has its own battery sets and the shaft driven alternator provides control power. Each of the valves is manipulated by hand.

3.2.11.4 Success Criteria

The diesel driven fire pump starts and injects into the Reactor Pressure Vessel.

The fault trees developed for this system are one of the inputs to the failure of the fire protection function in events trees (Va). The exact relationship is shown in the functional fault trees in Appendix E.

3.2.12 D.C. ELECTRICAL SYSTEM, R42

The purpose of the Class 1E DC power system is to provide continuous 125 VDC power for control and switching of the components in those systems needed for the safe shutdown of the plant.

The emergency DC power system consists of three divisions. These divisions are completely separate and independent with each designed to provide DC power to the respective divisional loads required for safe shutdown of the plant. A simplified diagram for the system is shown in Figure 3.2.12-1, 2, and 3.

3.2.12.1 System Description

Division 1 and 2

Unit 1 Division 1 and 2 buses (ED-1-A and ED-1-B) are each connected to a 60 cell lead acid 1,200 ampere-hour battery. The batteries are sized to supply the required DC loads for a minimum of 2 hours without any cell voltage decreasing to 1.75 Volts/cell or the total battery voltage decreasing to less than 105VDC.

The Class 1E DC buses are also connected to 400 ampere battery chargers and 400 ampere reserve battery chargers. All Class 1E battery chargers are solid state and are backed by the respective division 1 or 2 diesel generators.

Division 3

The Division 3 bus (ED-1-C) is connected to a 60 cell lead acid battery rated at 100 ampere-hours. ED-1-C is also connected to a 50 ampere battery charger and a 50 ampere reserve battery charger. The battery chargers are backed by the Division 3 diesel generator.

The Unit 2 DC buses, ED-2-A and ED-2-B and ED-2-C, may be manually crosstied to the respective Unit 1 DC buses in the event that the Unit 1 batteries and chargers are not available due to either maintenance (equalizing charge on batteries, charger maintenance, etc.) or random failure. The Unit 2 battery chargers are not diesel backed.

3.2.12.2 System Operation

If the battery chargers and reserve battery chargers become inoperable or if AC power is lost, the batteries will automatically pick up the load.

3.2.12.3 System Interface and Dependencies

The system interfaces and dependencies are shown in Table 3.2.12-1.

The Class 1E DC power system is dependent on the availability of the AC power system for long term operation. Following Station Blackout with manual load shedding and crosstieing Unit 1 and 2 batteries, the batteries have sufficient capacity for 22 hours.

The DC power system is also dependent on the MCC, Switchgear and Miscellaneous Electrical Equipment Areas HVAC, M23, to maintain the ambient temperatures within the design limits. Battery Rooms Exhaust System, M24, is also needed to dissipate the hydrogen generated from the operation of the batteries. These systems were not modeled as being necessary for the duration of the event.

3.2.12.4 Success Criteria

The DC power system (batteries, chargers and reserve chargers) must supply at least 105VDC for 2 hours following a LOCA or transient other than a Station Blackout.

The batteries must supply at least 105 VDC for 22 hours following a Station Blackout. After 22 hours AC power must be restored. The Unit 1 and Unit 2, Divisions 1 and 2 must also be crosstied and nonessential loads shed for the batteries to last 22 hours.

The fault trees developed for the D.C. electrical system are linked to other systemic fault trees.

3.2.13 AC POWER, STANDBY DIESEL GENERATOR SYSTEM, R43
HIGH PRESSURE CORE SPRAY DIESEL GENERATOR SYSTEM, E22B

The AC power system provides the source of AC power to those systems needed to safely shutdown the plant. A simplified diagram for the system is shown in Figure 3.2.13-1.

3.2.13.1 System Description

The AC power system consists of three 4,160VAC Class 1E buses, their associated stub buses and switchgear, and the diesel generator systems. The Division 1 and 2 buses (EH-11 and EH-12) provide an independent source of AC power to the majority of the engineered safeguard features (ESF) of Perry. The Division 3 bus (EH-13) is dedicated to the High Pressure Core Spray system and its support systems.

The Unit 1 interbus transformer is considered the preferred source of power to the 4,160VAC Class 1E buses with the Unit 2 interbus transformer being the alternate preferred source and the diesels being the emergency source. The interbus transformers receive power from 13.8KV buses L10 and L20 which receive power from the Unit 1 and 2 startup transformers.

3.2.13.2 System Operation

The Division 1 and 2 diesel generators automatically start on an RHR LOCA signal or an undervoltage signal on the associated bus. Large loads are sequenced at approximately 5 second intervals to ensure that large motors will have attained rated speed and that voltage and frequency will have stabilized before succeeding loads are applied.

The Division 1 and 2 stub buses (XH-11 and XH-12) are stripped from buses EH-11 and EH-12 during an automatic diesel generator start due to an RHR LOCA signal. The stub bus tie breaker may be reclosed after placing the LOCA

Bypass Keylock Switch to BYPASS then taking the control switch for the breaker to CLOSE. During a bus undervoltage event without an RHR LOCA signal, the stub buses are not stripped.

The Division 3 diesel generator starts and loads on a HPCS LOCA signal or an undervoltage signal of Bus EH-13.

3.2.13.3 System Interface and Dependencies

The system interfaces and dependencies are shown in Table 3.2.13-1.

Each of the Division 1 and 2 diesel generators depend on 5 systems for its operation: Standby Diesel Generator Starting Air, R44: Standby Diesel Generator Fuel Oil, R45: Standby Diesel Generator Jacket Water Cooling, R46: Standby Diesel Generator Lube Oil, R47; and Standby Diesel Generator Exhaust/Intake Crankcase, R48. The Division 3 diesel generator has these same systems incorporated into its design. All of these systems are treated as part of the diesel generator.

In addition, each of the diesel generators also depend on several other systems for long term operation. Emergency Service Water, P45, transports a large portion of the heat generated during operation of the diesel generators to Lake Erie. Diesel Generator Building Ventilation, M43, dissipates the balance of the heat generated to the atmosphere. Emergency DC Power, R42, provides control power to the diesel generators. Each of these systems provides support to the diesel generators by separate divisions. There are no dependencies which are interdivisional.

3.2.13.4 Success Criteria

Emergency AC power system supplies AC power for at least 24 hours following a LOCA or transient other than a Station Blackout.

Following a Station Blackout, AC power must be restored within 22 hours or within the time specified in the sequence.

Given a failure of the automatic initiation, the diesel generators must be manually started in the event of loss of offsite power.

The fault trees developed for this system are input as support systems to frontline systems and one of the inputs to the failure of the AC power function in the loss of offsite power event trees (B1). The exact relationship is shown in the functional fault trees in Appendix E.

3.2.14 SUPPRESSION POOL MAKE-UP, G43
DRYWELL VACUUM RELIEF, M16

The Suppression Pool Make-up System (SPMU) provides a rapid means of gravity feeding the suppression pool from the upper containment pool to compensate for any conservable water loss associated with a Loss-of-Coolant-Accident (LOCA). A simplified diagram for the system is shown in Figure 3.2.14-1.

The Drywell Vacuum Relief System provides a means to limit the buildup of negative pressure in the drywell in order to protect the drywell from flooding following a small break in the containment. The system also provides for drywell to containment isolation during LOCA conditions. A simplified diagram for the system is shown in Figure 3.2.14-2.

3.2.14.1 System Description

The Suppression Pool Make-up system is divided into two 100% capacity, independent trains, each consisting of piping and two downstream motor operated isolation valves. Train A is physically separated from Train B. The suppression pool volume, between the normal low water level (LWL) and minimum post-accident pool level, plus the make-up volume from the upper pool is adequate to supply all possible post-accident entrapment volumes for suppression pool water. The long term post-accident containment pressure and suppression pool temperature takes credit for the volume added from the upper containment pool.

The system gravity dump time through one of the two redundant lines is less than or equal to the minimum pump time. Pump time is determined by dividing pumping volume (upper pool make-up volume plus volume in the suppression pool stored between LWL and minimum top vent coverage) by the total maximum runout flow rate from all five ECCS pumps.

The Drywell Vacuum Relief System is divided into two independent relief lines. Each line consists of a 10" vent line, a motor-operated drywell vacuum relief valve, and a swing check valve. If a vacuum is sensed in the drywell, the vacuum relief valves (1M16-F010A,B) open. The differential pressure between the drywell and containment will cause air to be drawn into the vent pipe, through the swing check valves (1M16-F020A,B) and the drywell vacuum relief valves (1M16-F010A,B) and into the drywell.

The Drywell Vacuum Relief System is designed to limit a buildup of negative pressure inside the drywell and to prevent suppression pool water from overflowing the suppression pool weir wall which would result in partial flooding of the drywell following a small pipe break in the containment.

The drywell vacuum relief lines are spaced around the drywell to meet space separation requirements so that direct "shine" or "streaming" of radiation from inside the drywell is minimized.

3.2.14.2 System Operation

The opening of the SPMU system valves is signaled by a series combination of a Lo-Lo suppression pool level and a LOCA signal. The Lo-Lo level is 18 inches below the normal low water level. Since maximum ECCS pump flow lowers the suppression pool at a rate of approximately .88 feet per minute, there is approximately 1.5 minutes between the start of ECCS flow and generation of a dump initiation signal taking into account only for maximum ECCS system of flow from the suppression pool. The actual time between a LOCA and suppression pool low level signal is actually one to two minutes longer than this because vessel inventory mass is added to the suppression pool during blowdown steam condensation. This built-in delay assures that the Drywell pressure transient due to vessel blowdown has ended prior to dumping of the upper pool and corresponding increase of vent submergence.

The SPMU system dump valves can also be signaled to open by a LOCA signal in series with a 30 minute timer where the timer itself is started by the LOCA signal. This path of initiation logic is independent of suppression pool level and is specifically directed towards insuring that the combined upper pool and suppression pool volumes are available as a heat sink for "small" breaks which do not lower the suppression pool to the Lo-Lo level trip, but continue to dump vessel blowdown energy into the suppression pool. The minimum suppression pool volume, however, is adequate to meet all heat sink requirements without an upper pool dump.

Adequate water can be provided to the suppression pool even if one of the SPMU trains fails to initiate. One line can deliver the entire make-up volume in about 8 minutes and 40 seconds. In the remote possibility that both lines fail to initiate, the Control Room operator can manually initiate the system. This is done by first checking that a LOCA permissive signal is present. The Control Room operator then initiates the system by arming and depressing the Train A and Train B Manual Initiation Switches.

If a drywell vacuum condition is sensed, the isolation valve will automatically open until the vacuum signal is no longer present, then the valve will automatically reshut. If a LOCA signal (BOP isolation - RPV water level 2 or High drywell pressure) is received, the vacuum relief isolation valve will close. The valve will remain closed until either a drywell vacuum signal is received or the operator places the control switch to OPEN. It should be noted that the isolation valve will remain in the OPEN position indefinitely. In order to shut the isolation valve, the vacuum signal must be cleared and the operator must place the control switch to the CLOSE position.

The drywell vacuum relief check valves (M16-F020A,B) have no controls or interlocks associated with their operation. They function on a differential pressure across the valves to open and gravity to close.

3.2.14.3 System Interface and Dependencies

The system interfaces and dependencies for G43 are shown in Table 3.2.14-1. The system interfaces and dependencies for the M16 are shown in Table 3.2.14-2.

Electric Buses (required): for Suppression Pool Makeup	EF-1-A (A valves) EF-1-C (B valves) EK-1-A1 (A valve controls) EK-1-B1 (B valve controls)
Electric Buses (required): for Drywell Vacuum Relief	EF-1-B (480 VAC for A isolation valve) EF-1-D (480 VAC for B isolation valve) EK-1-A1 (120 VAC for indication) EK-1-B1 (120 VAC for indication)

The SPMU dependencies are 120VAC control power for initiation logic and control logic, 480VAC power for the main flow path motor operated valve operation, and the water supply coming from inventory in the upper containment pools.

The AC power for the Division 1 control logic and initiation logic is supplied by Bus EK-1-A1. Bus EK-1-B1 supplies AC power to the Division 2 control logic and the initiation logic. Motor operated valve power is supplied from divisionally separated 480 VAC buses.

The Drywell Vacuum Relief System dependencies are 120VAC control power for isolation valve logic, 480VAC power for the motor operated isolation valve operation, and instrument air for testing of the check valves.

Bus EF-1-B supplies Division 1, 480VAC power to the M16-F001A isolation valve for operation and control. Bus EF-1-D supplies Division 2, 480VAC power to the M16-F0010B isolation valve for operation and control. Three 120 VAC sources (buses EK-1-A1 and EK-1-B1 and distribution panel K-1-H) supply power for indicator lights, ERIS, and check valve testing. Instrument Air, P52, is used to open the relief check valves for periodic exercising so that operability is assured.

3.2.14.4 <u>Success Criteria</u>

SPMU is required to provide a rapid means of gravity feeding the suppression pool from the upper containment pool to compensate for any conceivable water loss associated with a LOCA. This ensures that there is an adequate water volume in the suppression pool to keep the suppression pool vents covered for all break sizes.

The dump valves are required to open for a Lo-Lo level in the suppression with a simultaneous LOCA signal. The dump valves are also required to open for a LOCA signal that has been continually present for 30 minutes following its initiation. In this latter case the Lo-Lo level in the suppression is not required. This ensures an adequate long term heat sink is available regardless of the break size.

The fault trees developed for this system are one of the inputs to the failure of the containment venting function (Y) in the ATWS event trees. The exact relationship is shown in the functional fault trees in Appendix E.

The Drywell Vacuum Relief System is required to limit the buildup of negative pressure inside the drywell. This ensures that the suppression pool water will not overflow the suppression pool weir wall which would result in partial flooding of the drywell following a small break in the containment.

Following the blowdown phase of a LOCA, air initially contained in the drywell is full of steam. During this period the ECCS is injecting cooling water from the suppression pool into the reactor pressure vessel. When the reactor pressure vessel is flooded to the level of the break, water begins spilling onto the drywell, condensing the steam and causing rapid depressurization of the drywell, and a possible vacuum condition to be created. The Drywell Vacuum Relief System ensures that the vacuum condition will be mitigated.

The fault trees developed for this system are one of the inputs to the failure of the containment venting function (Y) in the ATWS event trees. The exact relationship is shown in the function fault trees in Appendix E.

3.2.15 DIESEL GENERATOR BUILDING VENTILATION, M43

The Diesel Generator Building Ventilation system functions to provide ventilating air to the diesel generator rooms. This air is provided to dissipate the heat generated by the diesel generators and miscellaneous equipment during operation. A simplified diagram for the M16 system is shown in Figure 3.2.15-1.

3.2.15.1 System Description

The Diesel Generator Building Ventilation system is designed to maintain the Diesel Generator Rooms at 120°F maximum with an outside temperature of 95°F and at 40°F minimum with an outside temperature of -5°F.

The system serves the three diesel generator rooms. Each room is served by two 100% capacity air fans, four 50% capacity motor operated exhaust louvers, outside air and recirculation dampers, and ductwork. One of the four exhaust louvers in the Division 1 and 2 diesel generator rooms is normally open, the other three are normally closed. All of the exhaust louvers in the Division 3 diesel generator room are normally closed. The system for each room is divided into two trains.

Both trains start when the diesel generator in that room starts. The amounts of outside and recirculated return air used for cooling are automatically controlled by outside air and recirculation dampers. Only one ventilation train is normally required to provide adequate room cooling. The other train can be shutdown. Inactive supply fans are automatically isolated by the outside air dampers. For the Division 1 and 2 diesel generator rooms, when neither fan in the rooms operate, the outside air dampers modulate to promote natural recirculation and the auxiliary room cooling fan operates to maintain acceptable room temperatures during standby conditions.

When the carbon dioxide fire suppression system is activated, outside air and exhaust air dampers close, the recirculation air dampers open, and all fans stop.

3.2.15.2 System Operation

Both trains of M43 in a room start when the diesel generator in that room starts. The amounts of outside and recirculated return air used for cooling are automatically controlled by outside air and recirculation dampers.

3.2.15.3 System Interface and Dependencies

The system interfaces and dependencies are shown in Table 3.2.15-1.

The diesel generator building ventilation system is dependent only on the AC power system. There are no room coolers to be supported by a cooling water supply. There is no dependency on DC power.

3.2.15.4 Success Criteria

The diesel generator ventilation system must supply air to the diesel generator rooms within 15 minutes of the start of a diesel generator. Only one train is required to start and run for each division. The fault trees developed for these systems are input as support systems to the diesel generator fault trees.

3.2.16 CONDENSATE TRANSFER SYSTEM

The Plant Emergency Instructions, PEI-B13, provide six alternate methods for the injection of water into the reactor pressure vessel under accident conditions. These systems are utilized after the primary injection systems (i.e., ECCS, Feedwater and RCIC) fail to maintain RPV level above the Top of Active Fuel (TAF). A simplified diagram for the system is shown in Figure 3.2.16-1.

3.2.16.1 System Description

This alignment provides make-up into the RPV using the Condensate Transfer and Storage flush Water Supply to RHR Shutdown cooling to Feedwater Line or LPCI injection line for injection either outside or inside the shroud. This alignment requires one local valve in the AB 620' elevation to be operated. A simplified diagram for the Condensate Transfer Alternate Injection path is shown in Figure 3.2.16-1.

3.2.16.2 System Operation

Suction for this injection path is taken from the Condensate Storage Tank, pumped by Condensate Transfer Pumps, 1P11-C001A and B and through Feedwater Flush Supply Valve 1E12-F300A or B into the RHR System. From there, flow can be diverted inside or outside the RPV shroud by directing flow through the LPCI injection line or the Shutdown Cooling to feedwater line.

3.2.16.3 System Interface and Dependencies

The system interfaces and dependencies are shown in Table 3.2.16-1.

Electrical power is provided by offsite power.

Instrument Air is supplied to valve 1E12-F0300A/B/C. Failure of Instrument Air will cause the loss of this valve function and thus a loss of the injection path.

No component or room cooling is required for system operation.

The water supply is provided by the Condensate Storage Tank.

3.2.16.4 Success Criteria

For the Condensate Transfer Alternate Injection System, one 1,000 gpm pump is assumed to be required for system success.

The fault trees developed for each injection path form the primary input to function Va as shown in Appendix E.

3.2.17 CONTAINMENT VENTING (FPCC)

Plant Emergency Instruction PEI-D23-2 provides for 2 mechanisms for venting the containment in the event that the containment pressure cannot be maintained below the Containment Pressure Limit. One vent path is through the Fuel Pool Cooling and Cleanup system skimmers and piping. A simplified diagram for the system is shown in Figure 3.2.17-1.

3.2.17.1 System Description

The Fuel Pool and Cleanup system normally removes decay heat generated by spent fuel stored in the Fuel Storage pools and maintains the purity, clarity and level of the water in the upper pools. This system may also be used to vent the containment if the containment pressure cannot be maintained below the Containment Pressure Limit provided in PEI-D23-2.

The FPCC system is aligned to vent the containment through the 11 skimmers in the Fuel Transfer and Storage Poo the Reactor Well and the Separator Storage Well, to the FPCC Surge Tank and into the fuel Handling Building atmosphere through the 5 skimmers in the Spent fuel Storage Pool.

3.2.17.2 System Operation

Venting of the containment must be manually aligned.

a. Verify that at least 1 Fuel Handling Building HVAC Exhaust fan (OM40-C0002A, B or C) is in operation.

b. Open Containment isolation motor operated valve 1G41-F0145.

c. Open Containment isolation motor operated valve 1G41-F0140.

3.2.17.3 System Interface and Dependencies

The system interfaces and dependencies are shown in Table 3.2.17-1.

The systems are dependent upon the following electrical supplies:

EF-1-A (1G41-F0145)
EF-1-B (OM40-C0002A)
EF-1-C (1G41-F0140)
EF-1-D (OM40-C0002B)
EF-2-D (OM40-C0002C)

The dependency for this vent path is on electric power only if a containment isolation should occur prior to the loss of electric power. In this case the normally open inboard isolation valve 1G41-F0140 would be closed with no way for an operator to open it. The Fuel Handling Building Ventilation System exhaust fans, although dependent on electric power are not assumed to be needed for containment venting success.

3.2.17.4 Success Criteria

Any one vent path is considered a success. This system is the major input to function Y. Full details of the development of this function fault tree are given in Appendix E.

3.2.18 CONTAINMENT VENTING (RHR)

Plant Emergency Instruction PEI-D23-2 provides for 2 mechanisms for venting the containment in the event that the containment pressure cannot be maintained below the Containment Pressure Limit. One vent path is through the RHR Containment Spray headers and out through the Spent Fuel Storage Pool. Simplified diagrams are shown in figures 3.2.18-1 and 2.

3.2.18.1 System Description

In addition to providing containment heat removal through the containment spray mode of operation, the RHR containment spray headers can be used to vent the containment as described in PEI-D23-2.

The containment is vented by aligning the RHR containment spray headers by means of the RHR piping to the FPCC Supplement Cooling connection and into the Fuel Handling Building atmosphere through the Spent Fuel Storage Pool.

3.2.18.2 System Operation

Venting of the containment must be manually aligned.

a. Open the FPCC to RHR Supply manual valve 1G41-F0559A.

b. Verify that at least 1 Fuel Handling building HVAC Exhaust fan (0M40-C0002A, B or C) is in operation.

c. Verify RHR valve 1E12-F0027A(B) is open.

d. If the containment spray mode has not been initiated or if the RHR valves to the spray headers are closed, stop RHR pump A(B) and arm and depress the Containment Spray Manual Initiation push-button, 1E12A-63A(B).

   If the loop to be placed in the venting mode is in operation, then depress the Containment Spray Loop A(B) Seal-In Reset push-button, 1E12A-S64A(B) to reset the initiation logic. Then stop RHR pump A(B).

e. Close the RHR heat exchanger bypass motor operated valve 1E12-F0048A(B).

f. Close the RHR heat exchanger outlet motor operated valve 1E12-F0003A(B).

g. Open the RHR FPCC Supplement Cooling discharge manual valve 1E12-F0099A(B).

3.2.18.3 System Interface and Dependencies

The systems are dependent upon the following electrical supplies:

EF-1-A (RHR A valves)
EF-1-B (OM40-C002A)
EF-1-D (RHR B valves)
(OM40-C0002B)
EF-2-D (OM40-C0002C)

3.2.18.4 Success Criteria

Any one vent path is considered a success. This system is the major input to function Y. Full details of the development of this function fault tree are given in Appendix E.

3.3 SEQUENCE QUANTIFICATION

3.3.1 LIST OF GENERIC DATA

NUREG/CR-4550 (Drouin, 198 was adopted as the primary source for generic data. A review of the data used in this study was made against other sources of data (NUS BWR Generic (NUS, 1982), NUREG/CR-1363 (Miller, 1982), NUREG/CR-1740 (Trojovsky, 1984), NUREG/CR-3831 (Kahl, 1985), IEEE-500, WASH-1400 (NRC, 1985), GESSAR II (GE, 1982), and Kuosheng PRA, (AEC, 1985)) to establish that NUREG/CR-4550 data are of the same order of magnitude. Since the data contained in NUREG/CR-4550 are a compilation of many sources, a review of the base documents was also made, when possible, to verify the data being used and to assist in establishing component boundaries for use in system modeling. For those components not contained in NUREG/CR-4550, other sources of generic data were used. For a very few components not described elsewhere, assumptions of similarity were made to estimate failure rates.

3.3.2 PLANT SPECIFIC DATA AND ANALYSIS

Because of the short operating history of Perry, insufficient data was available to provide a meaningful statistical sample of failures upon which to build a data base. It was assumed that the greater population base of generic data offered a better estimate of failure data than the Perry data that was available. Perry failure values that were based on zero or one failure from a small population were not used. Large components such as Emergency Core Cooling Pumps have not experienced failures at Perry to date. Generic failure data was used for these components. The diesel generators do have a failure history for some modes of failure. This history was analyzed to estimate a failure rate for the diesels.

Due to the variance of system unavailability from testing and maintenance from plant to plant, Perry Unit 1 documents were used to determine unavailability of the Emergency Core Cooling Systems and the Reactor Core Isolation Cooling system due to testing and maintenance.

A lognormal distribution was assumed for nearly all of the data contained in NUREG/CR-4550 and other documents examined. This distribution was also assumed for the Perry specific unavailabilities and failures used.

Table 3.3.2-1 presents the basic event data used in the Perry IPE. The table is sorted by component.

3.3.3 HUMAN FAILURE DATA (GENERIC AND PERRY)

3.3.3.1 Introduction

The objective of the human reliability analysis (HRA) is to provide estimates of the probabilities of the human interaction basic events included in the Perry Plant IPE logic model. In general, human interactions considered for inclusion in a PRA can be divided into three general classes according to the time phase in which they occur.

Type A HIs arise before an initiating event, when plant personnel can affect availability and safety of the plant by inadvertently leaving equipment disabled following test, maintenance and calibration activities.

Type B HIs are those HIs that result in, or contribute to, initiating events. Examples are: plant trips following mistakes during testing, failures to control feedwater leading to plant trip, etc. Type B events are almost invariably incorporated implicitly in the initiating event frequencies obtained from plant operating experience. No explicit consideration of Type B HIs has been included in this study.

Type C HIs cover a wide range of specific actions following an accident. There are two sub-categories of Type C: (1) operator action performed in response to an Emergency Operating Procedure (EOP), including manual backup on failure of automatic initiation of systems, and (2) recovery actions in response to unavailability of a safety function that failed because of equipment malfunction. Events in the first sub-category (Type CP) can either appear as headings in the event trees or as basic events in system or functional fault trees, while recovery actions (Type CR) are addressed at the accident sequence cutset level.

The major effort in this study was the analysis of the Type CP events, and this is discussed in Section 3.3.3.3.

3.3.3.2 Pre-Initiating Event Human Interactions

Miscalibration of sensors leave an instrument channel such that it cannot actuate a given system as required. Common cause failure of actuation due to miscalibration was also considered. The potential for common cause miscalibration of multiple sensors of a given type or function due to human interaction was characterized as one tenth of the sensor failure probability.

The potential for incorrect system configuration following maintenance or surveillance was reviewed by verifying that the performance test would reveal an incorrect configuration fault. For the Perry IPE, restoration faults were not postulated for those systems or components with readily available control room indication of inoperability and bypassed indication, with daily position verification, or with administrative control for locked valve position verification. With these Type A guidelines, restoration faults were modeled for only the non-safety systems.

3.3.3.3 Post-Initiating-Event Human Interactions

The plant logic model, the event trees and fault trees, were constructed to include human interaction basic events. For inclusion in the event or fault trees, these events are adequately defined in terms of the failure mode they represent e.g., operators fail to depressurize the reactor following a loss of high pressure injection. However, in order to quantify these events, i.e., estimate their probabilities, it is essential to define them in greater detail. For example, it is necessary to understand what cues and procedures the operators use to guide them to perform the required function, what they have to do to successfully accomplish that function, the time available, and other factors that might influence their probability of success or failure. These factors all are scenario specific. The first step in the HRA was, therefore, to define the events as clearly as possible in preparation for the quantification. This was done by studying the scenarios to which the human interaction events contributed, and understand, among other things, the time line of the events.

Another function of this step of the HRA task is an identification of potential dependence between the human interaction events that occur in the model. Functional dependencies of the type "if event A occurs, event B cannot be successful," are handled in the overall structure of the model, i.e., they are hardwired into the event tree structure. What is principally of concern here is the influence of success or failure in a preceding event on the probability of success or failure of another event. There are a variety of reasons why the events may be probabilistically dependent: one important issue is that the cognitive processes needed to recognize the need for multiple actions may have common elements. To assist in the identification of such cognitively correlated HIs, the following groundrules are adopted:

(a) If two HI events are associated with responses to the same plant status (e.g, initiate HPCS pump, initiate RCIC on failure of auto initiation at Level 2), the cognitive part of the failure probabilities are considered to be totally dependent.

(b) As a corollary to this, if, in the chronological development of the scenarios, an HI failure event follow a successful HI, and the procedural instructions for both events are closely related, the cognitive failure probability of the second HI should be very small and can be neglected, since the success in the first event implies a successful recognition of the scenario.

(c) If human interactions are i) separated by a significant time (i.e., time between cues or required responses is long), or ii) separated chronologically in the sequence by a successful action, or iii) responses to different cues in different parts of the EOPs, they may be regarded as being independent.

(d) In addition, the easily memorized responses may be regarded as independent from these actions for which the procedures are expected to be providing the direction.

Other types of dependency, such as the fact that performing one function may take resources away from another is also considered by addressing, in the evaluation of the HEPs, the role of crew personnel, both in performing the actions called for, and in recovering from failure to execute correctly.

3.3.3.4 Quantification Approach

The model of human interactions used for the evaluation of a human error probabilities is the simple one that splits the response into two components, a detection, diagnosis and decision (DDD) phase and an execution phase. This is compatible with the ASEP methodology (Swain, 1987), the more recent EPRI proposed methodology (EPRI, 1991), and the HRA Handbook (Swain, 1983), all of which were used in the quantification. Reference is made to these documents for details.

For the key time-critical human interactions, the time-reliability curve approach of the EPRI methology (EPRI, 1991) was used to estimate the probability of failure in the DDP phase. The alternate approach of the EPRI methodology was used to evaluate the HEPs for those HIs considered dependent on the time critical HI or which are not time critical. A simplified THERP or ASEP approach was used to estimate the HEP for the execution phase. The details can be found in Appendix D. It should be noted that the quantification was performed on a sequence by sequence basis to more completely address the dependency issues.

The HEPs are given in Table 3.3.3-1.

3.3.4 COMMON CAUSE FAILURE DATA

The common cause failure analysis was performed following the general guidelines of NUREG/CR-4780 (Mosleh, 1988). Stages 3 and 4 of the analysis rely on the existence of a data base that provides detailed descriptions of historical events related to both single as well as multiple component failures. However since CCFs are rare events, very limited plant specific experience of these failures is expected. Therefore, in this study, as recommended in NUREG/CR-4780, industry experience, as reported in Nuclear Power Experience (Stoller Corporation) has been used to derive common cause data. Since the information in NPE is obtained primarily from LERs, its value as a complete data base, reporting single as well as multiple failure events was lessened with the introduction of the new LER reporting rules in January 1984. Thus, only the data up to 1984 is used in the quantification analysis.

As part of this analysis, the current Halliburton NUS component failure data base, which is based on the summary of information provided by the NPE event reports was modified to account for the Perry design. This data base contains an assessment of the historical events as they occurred at various plants. Due to the level of reporting, it is not always easy to interpret or assess the data objectively hence in some cases events have been subjectively assessed. The approach taken in NUREG/CR-4780, of recording the interpretation in terms of impact vectors, was followed in this study. The re-interpretation of the generic events with respect to the Perry plant results in a pseudo PNPP specific data base. The data analysis is described in more detail in Appendix C2.

In order to determine which CCF events contribute significantly to the overall core damage frequency, an importance analysis of the core damage cutsets was performed. The measure of significance adopted was the Fussell-Vesely importance measure and a beta factor of 0.1 was used in the initial modeling of common cause failures. Events where importance measures resulted in a contribution to core damage frequencies of greater than 0.1% were included as significant.

On the basis of the above criteria the following CCFs were analyzed in detail:

- ESW motor operated valves (ESMVCC)
- Diesel generators (DGDGCC)

- ESW motor driven pumps (ESMPCC)
- ECC motor driven pumps (ECMPCC)

The CCF probabilities used for each of the components is shown in Table 3.3.4-1. For all the events not analyzed in detail the screening values were retained in the final analysis.

3.3.5 QUANTIFICATION OF UNAVAILABILITIES OF SYSTEMS AND FUNCTIONS

The Perry IPE was performed using a linked fault tree approach as described in NUREG/CR-2300 using the NUPRA PRA workstation. In performing a PRA in this manner, the functions defined by the event tree headings are the main building blocks of the quantification process. The quantified functions can be used to determine the overall contribution of a given function failure, such as decay heat removal, to core damage. Each function is representative of the failure of one or more front line systems and/or human actions. Support systems are linked into the front line systems they support. A given function or combination of systems may be quantified several times to reflect different boundary conditions as the result of different initiating events. The results of the quantification of each function are summarized in Table 3.3.5-1. The quantification process and the results are presented in detail in Appendix E of this report. The individual systems were quantified in the course of development of the fault trees. The results are reported in the individual system analysis files.

3.3.5.1 Summary of function Quantification Process

The accident sequence analysis task identifies safety functions that must be provided in order to prevent core damage following an initiating event. The function is defined by the systems success criteria, the mission time, important operator actions, and any sequence specific equipment unavailability, environmental, or phenomenological conditions that exist following a specific initiating event. The combination of the above used to develop the functional fault tree for a given initiator are first combined and linked with the appropriate support systems. In order to avoid developing a separate fault tree for each condition in which a system is required to respond, a "switch" known as a "house event" is used to switch in or out various sections of the system fault tree for each condition. A house event Basic Event Data (BED) file is constructed for each initiating event. This defines the status (on or off) of each house event for that initiating event. All the house events and combinations of house events are described in Appendix E.

The functional quantification process is very straight forward. The appropriate fault tree(s) for the function of interest are updated against the house event file appropriate for the initiator of interest. This updating will turn on or off gates as required. The fault tree is then solved and quantified using appropriate truncation values. The result is a Boolean equation which is used in the sequence quantification as well as giving a quantified value for that function. Some functions may be single events and would not require this process. A full descriptions of the quantification of all functions is given in Appendix E.

3.3.5.2 Summary of Function Unavailabilities

The functional unavailabilities are shown in Table 3.3.5-1. This table identifies the function and describes the success criteria. Each function is named after the function identifier in the event tree. For example, the function U1 represents failure of High Pressure Core Spray (HPCS). In the event tree it is found that HPCS is required for a number of sets of conditions. Thus the functions are identified as U101, U102, U103, etc. The exact function used in each sequence of each event tree is printed on the event tree and developed in Appendix E.

3.3.6 QUANTIFICATION OF SEQUENCE FREQUENCIES

This section will describe how the quantification of sequence frequencies was accomplished. The methodology is discussed and a brief explanation of the computer code modules used to perform the quantification is given. The full quantification documentation and detailed quantification of an example sequence is given in Appendix F.

Sequence frequency quantification is accomplished by merging the combinations of failures for each function resulting from the quantification of the function unavailabilities (Section 3.3.5) as determined by the event tree structure (Sections 3.1.2 and 3.1.3). The event tree defines core damage sequences, transfer sequences and non-core damage sequences. The NUPRA workstation was used to perform the task of sequence quantification. The non-core damage sequences (Status categorized as OK) are shown on the event trees for convenience, but are not quantified since their results have no application. These non-core damage sequences will not be discussed further.

3.3.6.1 Initial Quantification of Sequence Frequencies

Each core damage sequence is quantified in a number of stages. In the first stage all the combinations of events contributing to each functional failure, and the initiating event, are merged to give a set of combinations of failures (known as minimal cutsets) which will lead to core damage. However, in any given sequence a number of functions may have been successful and, therefore, some of the combinations may not be applicable as the particular piece of equipment may have operated satisfactorily. If, for example, a specific electrical supply appears in both a successful and failed function, it cannot be both failed and successful. In the second stage, the combinations of failures are compared with the successful functions and all the failure combinations which include components which are operational are removed.

In the third stage, the remaining combinations are compared with disallowed combinations of failures. For example, a combination of both Division 1 and Division 2 diesel generator in maintenance is not allowed by the Technical Specifications. These disallowed combinations are removed.

The final list of combinations (cutsets) represent the contributors to core damage, and the sum of the frequencies of all the combinations in a sequence, represents the sequence core damage frequency.

The PRA code NUPRA was used to perform the quantification in the following manner.

The NUPRA Sequence Function Equation Assignment Module was used to automatically generate quantification control (merge) files for each sequence in each page of the event trees based on the event tree construction and function assignment. This file is then used to automatically perform the quantification of all the sequences in the event tree. The cutoff selected for each sequence was a truncation value of 1.0×10^{-10}. The results from system and function quantification used in the quantification of each sequence are in the form of equations containing the minimum combinations of failure cutsets above, typically 1.0×10^{-8}. Then each sequence is quantified automatically by merging the cutsets for each function.

The event tree merge control files perform three processes for each core damage and transfer sequence. The first process is to combine all the function equations for each failed function with the initiating event function equation using Boolean binary conjunction, AND, logic. The second process is to delete the functional successes by comparing the cutset function equation for each function which is a success in the event tree with the combined cutsets for all the failed functions using Boolean binary subtraction of each basic event that is successful. The remaining cutsets then represent the failures which would contribute to the sequence.

The third process is to modify the sequence equation and printout files to take account of Technical Specification limitations, to delete successful function cutsets for the transfer sequences or to correct the value for success branches of event trees when single basic events of the failure branch is greater than 0.1. Each of these actions requires editing of the merge control file for each of the event trees as described in the following paragraphs.

Technical Specification, Limiting Conditions for Operation prohibit certain redundant equipment from being inoperable during power operation. A fault tree, DAM, was created to provide cutsets which represent disallowed maintenance combinations based on Technical Specification. This fault tree was then quantified to give a cut set equation of all the disallowed combinations. The cutset function equation for disallowed maintenance, DAM.EQN, was used to perform Boolean subtraction from every sequence by adding a step in the quantification of each sequence in the event tree merge control files.

Transfer sequences were used as the initiating event frequency for several of the transfer event trees. The sequence equation files were used as input directly to these transfer event trees. For example, the Station Blackout event tree B used Loss of Offsite Power sequence T1S39 as the initiating event frequency. Each of such transfers is shown on the appropriate event tree. All the successful functions from the tree from the sequence is transferred here to be inserted in the merge control file for the transfer time, i.e. all success function sequences T1S319 have to be included for each sequence in event tree B.

When the failure probability of a function is greater than 0.1 then the success branch approximation of 1.0 is no longer valid. NUPRA always uses a success approximation value of 1.0. To compensate for this approximation all success paths for functions which contain a basic event with a failure probability of greater than 0.1 were multiplied by the complement of the function failure rate.

After this final processing, the sequence equation files and the sequence printout files were then ready to be used to determine total core damage frequency or as input for back end analysis.

The event tree merge control files were named using the corresponding event tree name with OCL as the file name extension. For example, for Large LOCA (event tree A) this would be A.OCL. The output from the quantification consists of two files. One output file is the sequence equation file which contains all of the cutsets which are greater than the truncation value. It should be emphasized that this file contains only cutsets and no data or cutset values. A truncation of 1.0 X 10^{-10} was used unless the equation resulted in greater than 5,400 cutsets. Then a slightly higher truncation was used which would result in less than 5,400 cutsets.

Another output file is the sequence printout file which contains the desired number of cutsets (usually 50) ranked in order of magnitude using data in the data base at the time of quantification. Appendix F contains the first page from the printout files for every sequence resulting in a core damage frequency greater than 1.0 X 10^{-7}. The sequence file names are the same as the event tree names in combination with the sequence number. For example, sequence 4 of the Large LOCA event tree would be AS04. The equation files have a file name extension of EQN and the printout files have a file name extension of MGP.

The quantification of an example sequence is fully described in Appendix E.

3.3.6.2 Summary of Quantification

The sequence quantification described in the previous section was performed for three sets of event trees: the first to give the core damage frequency from internal initiating events, the second to give the plant damage state frequencies for use in the quantification of the source term release frequencies and the third to give the conditional core damage frequencies for internal flooding initiated events. The location in this report of information on the results and details of the quantification are as follows:

Internal Events	Section 3.4	Appendix F
Plant Damage States	Section 4.3.4	Appendix H
Internal Flooding	Section 3.3.7	Appendix G

3.3.7 INTERNAL FLOODING

3.3.7.1 Introduction

The objective of the analysis reported here is to estimate the contribution of internal plant flooding to core damage and fission product release category frequencies at PNPP Unit 1.

Internal plant flooding encompasses the effects from the accumulation, spraying or dripping of fluids arising from the rupture, cracking or incorrect operation of components within the plant. In practice major internal floods have occurred in nuclear power plants, for example, from the rupture of pipes, valves and expansion joints as well as from operator errors during plant maintenance activities. All potential internal flood sources and causes are considered in this analysis, including those that result in loss of primary or secondary reactor coolant outside the Containment (i.e., interfacing LOCAs or steam line breaks with failure to isolate).

Internal flooding like other so called "external events" merits consideration as a potentially significant risk contributor because of it's potential for common cause equipment failures and/or human actions which may result in an accident initiating event (e.g., loss of Main Feedwater) and loss of one or more accident mitigating systems. The detailed analysis of such events is very plant specific, since their likelihood of progression and subsequent impact on plant systems is highly dependent on factors such as layout, piping arrangements, drainage as well as the prevailing flood protection features and programs. Furthermore in evaluating the frequency of flood induced accident sequences the probability of coincident random equipment failures and operator errors must also be taken into account.

In theory at least, the risk from all flood sources anywhere in the plant could be assessed in a detailed realistic manner. However this is impractical due to the large number of potential flood sources, and unnecessary since floods in many areas can be shown to be insignificant contributors by simple bounding arguments and analyses. For the sake of efficiency the analysis was performed in two levels of detail. At the conclusion of the screening analysis some plant areas (or particular flood scenarios) are determined not to be significant while others are singled out for further analysis. Plant specific information required to evaluate the impact of flooding was obtained mainly from the Appendix R safe shutdown submittal PNPP USAR, panel and conduit layout, general arrangement and piping drawings and P&IDs. Previous flooding analyses were also reviewed. In addition information obtained during plant walkdowns and in discussions with plant staff proved invaluable.

The flooding analysis is fully described in Appendix G and a summary of the results presented in this section. The screening analysis described in section 3.3.7.2, the detailed analysis in section 3.3.7.3 and the summary and conclusions given in section 3.3.7.4.

3.3.7.2 Screening Analysis

3.3.7.2.1 Designation of Independent Flood Areas

In this sub-task various areas of the plant are designated independent with respect to internal flooding. An area is termed independent if flooding outside the area cannot intrude into the area without the failure of an enclosing flood barrier. Conversely, in the absence of a flood barrier, flooding in an independent area will not be the direct cause of failures outside the area. The concept is useful because it will allow the analyst to define the extent of common-cause failure attributed to a particular flood, and to assign frequencies to the flood.

The physical layout of the plant buildings together with the location and size of potential flood sources is considered in determining the independence of an area. These areas can be easily identified as independent with respect to internal flooding because they are distinct structures with only a few interconnecting pathways. These interconnections are generally in the form of personnel or equipment access ways or, in some case, shared drainage systems. A review of the plant design was performed to identify design features (such as watertight doors, check valves in common drainage lines, and differences in floor elevations between buildings) that tend to inhibit any significant propagation of water from one building to the other. Other factors that may contribute to the independence of an area are physical separation, and equilibrium flood heights compared with the lowest elevation of any opening in a flood barrier.

For plant structures containing safety related equipment the areas of independence were defined as smaller areas within a larger independent area. If possible, these areas were defined such that they contain components pertaining to a particular mitigating system. It should be noted that even non-watertight doors may, under certain circumstances, be regarded as effectively providing independence; for example, a door in a stairwell will funnel virtually all water flowing down the stairwell to a lower elevation. Thus, the elevation to which the door provides access will not be susceptible to significant amounts of water inflow from a higher elevation, even if the stairwell door is not watertight.

In considering the possibility of flood barrier failure, the failure modes will include: operational errors (e.g., watertight doors or hatchways left open) leakage through unsealed doors and door failures due to hydrostatic head. Building gaps (rattle spaces) were also considered.

In general, the collapse of walls or leakage through construction joints were not considered to be important. (Although there have been instances of leakage through wall seams, the leakage rates have been minor and easily accommodated by installed drainage systems.) However portions of the TPC/CDB building will involve collapse of a wallboard stairwell during circ water flooding.

3.3.7.2.2 Evaluation of Bounding Flood Frequencies

For the purpose of screening, flood frequencies are generally determined on the basis of US plant experience from significant incidents reported in Nuclear Power Experience (NPE) (Stoller Power Inc., 1990). However, because of incomplete event descriptions and differences in plant layout and flood sources, this data was applied in a conservative fashion.

Note that this approach is appropriate for screening, since it encompasses all flood sources and no information on flooding rates is required at this stage.

3.3.7.2.3 Identification of Flood Induced Initiating Events and Mitigating System Damage

For each area in which a flood can occur, it is necessary to examine the flood susceptible equipment to determine which of the initiating events defined in the internal events study may occur and which accident mitigating systems may fail. Generally all electrical equipment is assumed to be susceptible to damage except junction boxes which have a double heat shrink wrap covering and cable raceways. In the screening analysis susceptible equipment is assumed to be damaged given that a flood occurs in an area or propagates to it unless damage can be easily ruled out on the basis of an inadequate maximum flood height and minimal possibility of spraying. Flood propagation is assumed to occur if a given pathway exists, unless it requires the failure of a physical barrier whose failure probability is independent of flood height. In this case propagation will be assumed to occur with a probability equal to the probability of the barrier failure.

In certain areas it may be possible for floods to cause more than one type of transient (floods are not usually capable of causing LOCAs). Since the objective of the screening analysis is to be conservative the most severe transient is assumed.

3.3.7.2.4 Summary of Screening Analysis

The results of the screening analysis are summarized in Table 3.3.7-1. This table identifies each of the independent areas, the flood frequency and the estimated core damage frequency. Of the 36 areas six areas are identified as requiring detailed analysis. The estimated core damage frequency from each of the other areas is below 10^{-7} using the conservative assumption that all equipment in the area is damaged immediately on occurrence of the flood. The six areas selected for detailed analysis are:

Zone	Description
1	Turbine Building
13	Lowest Level Control Complex
17	Second Level in Control Complex
8	Auxiliary Building Corridors - lowest level
16	Auxiliary Building Corridors - second level
1A	Steam Tunnel

3.3.7.3 Detailed Flooding Analysis of Designated Areas

3.3.7.3.1 Definition of Flood Damage States

Following a flood incident, damage to some equipment in the local vicinity may occur immediately, due to spraying and dripping. However, for a flood area of reasonable size, much of the equipment will not sustain damage until the flood level rises to a critical level (e.g., for switchgear room MCCs this is generally 6" to 1 ft). Likewise flood propagation to an adjacent area may not occur until the flood level rises above a curb for example, and then additional time will be required before the level reaches a critical height in that area. Therefore, there is usually a basis for defining a set of distinct flood damage states, each corresponding to a progressively increasing severity equipment loss. Each flood damage state is therefore defined in terms of the time at which it occurs after the initial flooding incident together with a set of accident mitigating systems which are damaged. Subsequent analysis then predicts the frequency of each flood damage state and the conditional core damage frequency given it does occur.

The growth (or rate of rise) of flood level is then determined by taking into account the flooding rate, the free cross-sectional area of the flood area, and the capability of the drainage mechanisms (floor drains, and leakage pathways to adjacent areas under doorways). In some cases an equilibrium flood height is established when the outgoing flow through the drains or under doors equals the flood rate. Flood growth may be halted at any time either by automatic or operator action leading to isolation of the flood source, or by exhaustion of the flooding source itself. Factors considered in evaluating the probability of suppression include: the means of detecting occurrence of the flood (alarm, area occupancy, etc.); the means of detecting and isolating the specific source of the flood, the time available for the operator to isolate the flood source before equipment damage occurs and applicable flood procedures and operator training.

3.3.7.3.2 Flood Growth Event Tree

The individual Flood Damage states are determined by developing flood growth event trees for each of the floods in each of the flood areas identified in the screening analysis. In order to more accurately estimate the effects of flooding in an individual zone, the flood rate is divided into three groups, severe, moderate, and small. The basis for this is fully described in Appendix G. The trees then consist of the flood initiating event (and frequency) and the various stages of the containment of the flood. The stopping of the flood at a given stage will lead to a flood damage state in which a given set of equipment is damaged and therefore unable to prevent core damage. The defined flood damage states and equipment affected for the floods in each of the above areas is shown in Table 3.3.7-2. Also included is the conditional core damage frequency, that is the probability of the random failure of the remaining ECCS equipment not already failed by the flood.

3.3.7.3.3 Results of Flooding Analysis

The core damage frequency as the result of flooding is estimated to be 1.5 X 10^{-6}. The contribution from flooding in each of the most significant zones is shown in Table 3.3.7-3 and Figure 3.3.7-1 and discussed in the following paragraphs.

<u>Zone 13 Control Complex Elevation 576'10"</u>

The core damage frequency from floods in this area is 8.8 X 10^{-7} per year which is 57 percent of the total contribution from flooding. This zone consists of a large open area at the lowest level of the control complex. Principle equipment in this area is the control complex chilled water pumps (CCCW), emergency closed cooling pumps (ECC), and the instrument and service air (IA/SA) compressors. Failure of the IA/SA occurs at a flooding level of 1 ft and the ECC and CCCW 22" above the floor defining the two plant damage states.

The severe flood rate was defined at 105,000 gpm based on floods from the service water 30 in and 42 in pipework. The moderate flood was defined as 35,000 gpm and the small flood as 16,000 gpm.

A flood in this area is assumed to lead to loss of instrument air initiating event and eventually failure of the RCIC system, low pressure injection and loss of containment heat removal. The low pressure systems are lost as a result of loss of ECC. As the low pressure systems are not failed immediately, there is time to restore low pressure injection via one of the alternate injection paths. RCIC fails as the result of flow under the door to the Unit 1 auxiliary building and flooding of the RCIC local panel.

The largest contribution comes from the medium flood (29%) followed by a small flood (18%). The frequency of a severe flood is low so the total contribution from this flood is approximately 10%.

Zone 17 Control Complex 599'

The core damage frequency from floods in this area is 3.2×10^{-7} per year which is 21 per cent of the total contribution from flooding. The source of a severe flood in this zone is the service water piping in the Nuclear Closed Cooling (NCC) heat exchangers. A flood in this area would flow through doors and down to zone 13 with the same consequential damage. The doors to the stairwell swing open and would therefore fail before the swing closed double doors to zone 1B. NCC is assumed lost in zone 17 as the result of the loss of SW cooling, NCC equipment and cooling to the instrument air compressor are affected in the course of the flood.

The impact of the flood is the same as that for area 13 however the flood frequencies are lower as the contribution from the moderate, small and severe floods are 11, 6 and 4 percent respectively.

Zone 1 Turbine Building and Turbine Power Complex

The core damage frequency from floods in the turbine building and turbine power complex is 2.8×10^{-7} per year which is 18 percent of the total contribution from flooding. The major sources of water considered in detail in the turbine building are the circulating and service water systems. All other sources were assumed capable of only causing a small flood and therefore the frequency of such floods was based on general industry experience. In order to obtain an understanding of the impact of floods in this area the major flood sources were broken down into the following subgroups: floods from pipework in the turbine building, floods from the expansion joints in the circulating water system and floods from circulating water pipework in the Turbine Power Complex, where it passes through the east and west side rooms.

The three damage states of concern in the turbine building are (1) 3 to 5 ft above the 574'10" floor level which will fail the hotwell and condensate booster pumps (2) the 593'6" level at which the preferred alternate injection pumps (condensate transfer) will fail and (3) the 599' level at which the door to the safety related auxiliary building is located.

Failure at the first and second damage level will only fail equipment in the turbine building and therefore do not make a significant contribution to core damage. For any circulating water flood in the main turbine building the third level cannot be reached with the initial inventory in the system and basin, therefore a severe flood will not reach level 3. The only possible way for this level to be reached is for a relatively moderate or small circ water flood to slowly fill the heater bay and turbine power complex while the basin make-up system adds water to the basin. Floods will only reach the third stage following failure of flood logic and very delayed operator action.

In the turbine power complex the CW return header passes through the east and wests side rooms at the second floor level. An estimated flood rate of 750,000 gpm is postulated if a gross rupture of the pipe occurs. There are no large valves or expansion joints in the area. It has been assessed that a full pipe rupture in the east side room would collapse the wall board stairwell to elevation 568 west room which quickly fills. The double door fails at elevation 568 to the larger west room but the level continues to rise until the level in 593 east rises to the top of the auxiliary building door at elevation 599 at which point the door is assumed to fail. This is estimated to occur 73 sec after the pipe rupture.

For gross rupture in the west side, if the west side elevator door fails inward before 13ft level of water is achieved (top of door to auxiliary building) downward flow of water through the elevator shaft will fill the lower level. Two doors to the turbine building and the elevator door on elevation 577 will open and allow the water into the turbine building where the flood alarm and isolation switches will activate to isolate the flood before the flood causes failure of the door to the auxiliary building at the 599 level.

As the result of the frequencies of the various floods the highest contribution to core damage comes from floods as the result of a small expansion joint failure (11%) followed by floods is either the turbine power complex west or east rooms (3% each). The remaining floods contribute approximately 2% to the overall flooding CDF.

Zone 8 Auxiliary Building Elevation 568'

The core damage frequency from floods in this area is 2.8 X 10^{-8} which is approximately 2 per cent of the total contribution from flooding. Floods in the auxiliary building basement level corridors will not directly affect the ECCS systems as each of the 6 rooms is protected by a watertight door. However, all the local panels are mounted in the corridor at the 568 ft level with the exception of the HPCS panel at the 574 ft level. The major flooding sources within the zone are the emergency service water and condensate transfer. Flooding will lead to the loss of the ADS permissive, which can by bypassed by the operator in the control room, and loss of the RCIC due to multiple trips from flooding of the local panel.

If the flood is not isolated then eventually water will build up and flood under the door gap into the control complex zone 13 with consequential damage to equipment in that area. The time available for flood isolation is such that this is not a significant contributor to the overall core damage frequency from flooding.

Zone 16 Auxiliary Building Elevation 599'

The core damage frequency from floods in this area is 1.1×10^{-8} which is less than 1 per cent of the total contribution from flooding. Floods in this area were found to affect equipment on the floor directly below before a critical height for zone 16 equipment would be reached; however water spray could cause failure of the TBCC chillers in the west zone area. Moreover as the flood rates range from 16,000 gpm (severe) to 3000 gpm (small) the time available to isolate the flood and the relatively minor damage early on ensures that floods in this area are not a significant contribution to core damage.

3.3.7.4 Summary and Conclusions

Sequences leading to core damage initiated by internal floods contribute approximately 12% to the overall core damage frequency from internal events and internal floods. The most significant sequence will arise from floods in Zone 13 in the control complex at elevation 576 ft. The total contribution from small, medium and severe floods in this area is 8.8×10^{-7} on 7 per cent of the total core damage frequency, which is less than the contribution from either ATWS, loss of offsite power or station blackout.

It is concluded that as the highest contribution to flooding in a given area is 8.8×10^{-7}, based on a number of conservative assumptions and the overall contribution from flooding in all areas is 1.5×10^{-6} no specific plant improvements will be considered.

3.4 RESULTS AND SCREENING PROCESS

Core damage for the Perry plant is defined in the USAR as follows:

1. The maximum fuel element cladding shall not exceed 2,200°F.

2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the cladding metal in the fuel region of the core were to react.

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

For the purpose of the Perry IPE non-ATWS sequences, this is translated to define core damage as occurring upon the onset of reactor pressure vessel (RPV) steam cooling, at the Minimum Zero Injection Water Level (MZIWL), given that a high capacity injection system is not immediately available to recover core cooling. For RCIC, injection recovery is assumed to be required by the top of active fuel (TAF) to avoid core damage. The definition of steam cooling is consistent with that defined in the BWR Owners Group Emergency Procedure Guidelines (EPGs). For ATWS sequences, core damage was assumed to occur at fuel clad temperatures of 2,200°F as determined by the MAAP computer code.

Although these criteria are slightly conservative, the analysis shows that the additional time available to recover cooling systems before serious core degradation occurs is small and would not significantly lower the core damage frequency for the dominant sequences. However, in evaluating the frequency of release of fission products, consideration has been given to recovery of systems and re-establishment of decay heat removal capability prior to RPV failure or prior to containment failure. Thus, the slight conservatism in the arbitrary definition of core damage has not been carried over into the frequency of fission product release.

In this section the results and screening process used to identify the core damage frequency are discussed. Results associated with the level 2 analysis, that is contribution of sequences to the containment failure frequency and radionuclide source term category frequencies, are summarized in Sections 4.7 and 7.0.

3.4.1 Application of Generic Letter Screening Data

All of the event trees were constructed with the headings representing functions combining one or more systems and operator actions. The following screening criteria have been used for inclusion of the results in this section of the report. As many of the functions are effectively single systems the criteria for systemic sequences rather than functional sequences have been used.

1. Any sequence that contributes 1×10^{-7} or more per reactor year to core damage has been included in the summary of core damage frequencies by initiating event. (Table 3.4.1-1)

2. Any sequence that contributes 1×10^{-7} or more to the total core damage frequency, grouped by initiator. (Table 3.4.1-2)

3. All sequences that are within the upper 95% of the total core damage frequency. (Table 3.4.1-3)

4. Systemic sequences that contribute to a containment bypass frequency in excess of 1.0×10^{-8} per reactor year. (Table 3.4.1-4)

5. Any systemic sequences that are determined from previous applicable PRAs or by engineering judgment to be important contributors to core damage frequency or poor containment performance which are not already included in Tables 3.4.1-1 through 3.4.1-4.

6. All systemic sequences greater than 1% within the upper 95% of the total containment failure frequency. (Tables 3.4.1-11)

In order to meet the above screening criteria, the core damage frequency calculations were performed using a truncation of 1.0×10^{-10} for the sequence cutsets. Recovery actions were included after the initial quantification by modifying the event trees. No sequences were cutoff after adding the recovery actions. Therefore, information was retained on all sequences that meet the original screening criteria of 1×10^{-7}.

The submittal guidance requires that all sequences that, but for low human error rates in post operator actions, would have been above the applicable core damage frequency screening criteria be identified. By using an initial cut off three orders of magnitude below the basic screening criteria and not eliminating any sequences after applying recovery action, all such sequences are retained and reported in Table 3.4.1-5.

3.4.1.1 Core Damage Frequency

The internal events portion of the IPE identified 21 core damage sequences with an annual frequency of greater than 1×10^{-7} contributing 86.3 percent of the overall core damage frequency (CDF). An additional 89 sequences with a point estimate frequency of greater than 1×10^{-10} per year contributing the remaining 13.7 percent of the overall core damage frequency. Each initiating event's contribution to the overall CDF is shown in Table 3.4.1-1. The CDF results given, except where stated otherwise are point estimates.

The point estimate frequency of core damage is 1.2×10^{-5} per reactor year. The combined frequency of the 89 sequences below the 1×10^{-7} cut off is less than 1.4×10^{-6} per reactor year. An uncertainty analysis was performed to evaluate the uncertainty on core damage frequency resulting from the uncertainties on the parameter values of the core damage model. The cumulative distribution function for the core damage frequency as shown in Figure 3.4.1-1, some significant parameters of the core damage frequency distribution are as follows:

Mean	1.4×10^{-5}
Standard Deviation	3.9×10^{-5}
95^{th} Percentile	2.5×10^{-5}
Median	1.1×10^{-5}
5^{th} Percentile	6.2×10^{-6}

The difference between the mean value, obtained from the uncertainty analysis, and the point estimate results from the correlation of the samples of the basic event probabilities that are based on the same parametric value distribution. This is called state of knowledge correlation (Apostolakis and Kaplan, 1991).

On review of the cutsets it does not appear that the overall characterization of the safety of the plant in terms of the contributions and their relative importance, would be significantly altered by using the uncertainty analysis for the estimation of core damage frequency. Therefore, the point estimate results are used in the remainder of the discussion and interpretation of results. In further support of this approach, it should be noted that the point estimate values for the parameters have been chosen to be either realistic (when sufficient data are available) or conservative.

The summary of sequences grouped by initiator is shown in Table 3.4.1-2 and those contributing 95 percent of the core damage frequency in Table 3.4.1-3. The containment bypass sequences are given in Table 3.4.1-4.

The dominant accident initiating event type is anticipated transient without scram (ATWS) at 40.7 percent. Transients contribute 25.0 percent, station blackout 19.3 percent, and loss of offsite power 12.4 percent. As a complete class, LOCAs contribute 2.6 percent. These results are summarized in Table 3.4.1-6 and Figure 3.4.1-2.

An event importance analysis was performed on the overall core damage model. In this analysis the relative importance of each basic event was calculated with respect to three different measures. The three measures are Fussell-Vesely, risk reduction, and risk achievement.

The dominant basic events ranked in order by Fussell-Vesely and risk reduction measures are shown in Table 3.4.1-7. The Fussell-Vesely importance is a measure of the contribution of the given component to the overall core damage frequency by comparing the sum of all cutsets in which that basic event occurs with the sum of all cutsets. The risk reduction measure shows the ratio of the original core damage frequency to the reduced core damage frequency if the component was perfect or its failure probability is zero.

It should noted that the ranking of events by the Fussell-Vesely measure and the risk reduction measure are identical so the highest ranked items for these two measure are discussed in the following paragraphs.

The most important basic event for risk reduction is the mechanical failure of the control rods (CM) preventing insertion into the core given a signal to shut down the reactor. This is the single event following any transient initiator which will lead to the ATWS scenarios which in turn contribute 40.7 percent to the core damage frequency.

The second important basic event for risk reduction is the loss of offsite power (T1). This initiator leads to station blackout sequences (19.3 percent) and loss of offsite power (12.4 percent). Sequences following the loss of offsite power initiator then contribute 31.7 percent to the core damage frequency. The Fussell-Vesely importance is 0.32 with a risk reduction of 1.47.

Failure of the operator to maintain the power conversion system (PCS) available (NSHICPEC5-2-L1T3) for an ATWS resulting from a transient with PCS initially available or loss of feedwater transient, is the third most important basic event. It has been assessed that the feedwater runback leading to a loss of RPV level will result in closure of the MSIV in all cases. This basic event was set to 1.0 as the operators will not have time to restore feedwater before MSIV closure. It occurs in the dominant sequences of the IPE. This HI effectively becomes the transfer flag for these events to the MSIV closure ATWS tree (T2-C) based on the current feedwater design. The Fussell-Vesely importance is 0.27 with a risk reduction of 1.38.

The initiating event transient with PCS available (T3A) is the fourth most important basic event. This event contributes to the dominant ATWS sequences and has a Fussell-Vesely importance of 0.25 with a risk reduction of 1.34.

Failure of the operator to re-open the motor feed pump control valves or manually depressurize the RPV (FWHICPEL-2-FDW-V) during an ATWS event following a transient with a loss of PCS is the fifth most important basic event with a Fussell-Vesely importance of 0.25 with a risk reduction of 1.33.

The initiating event transient with PCS not available (T2) is the sixth most important basic event. This event contributes to the dominant ATWS sequences and has a Fussell-Vesely importance of 0.23 with a risk reduction of 1.30.

The seventh important basic event is the failure of the operator to inhibit ADS (ADHICPC5-1-ADS-O) for ATWS scenarios where the feedwater system has failed. The Fussell-Vesely importance of this basic event is 0.22 with a risk reduction of 1.28. The failure to inhibit ADS in other sequences is modeled by different basic events as they have different values. The sensitivity of the results to this event is discussed in section 3.4.1.6.

The failure of the containment anchorage (CV05) is the eighth important basic event. This is included in any sequence in which RPV injection is successful but long-term containment heat removal fails. The Fussell-Vesely importance is 0.15 with a risk reduction of 1.17.

Non-recovery of offsite power in 3 hours (R15) is the next basic event. This basic event leads to sequences in LOOP and station blackout contributing 12.8 percent to the core damage frequency. The Fussell-Vesely importance is 0.13 with a risk reduction of 1.15.

The failure of the 4,160 VAC bus EH12 is the next basic event. This basic event can fail all equipment which requires division 2 power. The Fussell-Vesely importance is 0.12 with a risk reduction of 1.13. This failure results in the failure to vent as the inboard FPCC isolation valve is powered from the Division 2 bus. In the event of failure of the Division 1 bus, the outboard FPCC isolation valve can be locally opened by the operator.

Of the next six basic events in importance, four are failures of either the Division 1 and 2 diesel generators to supply power or the non-recovery of offsite power. These basic events contribute to both LOOP and station blackout sequences.

Two of the next eight basic events are failures to provide alternate injection to the RPV via the fire protection system. The fire protection system may be used when other alternate injection systems are unavailable due to non-recovery of offsite power.

Similar information was generated for risk achievement. Risk achievement is derived by calculating the core damage frequency with a given event failure probability set equal to 1.0. This is equivalent to determining the core damage frequency if the component is failed at the time of the initiating event. The dominant basic events ranked in order by risk achievement measures are shown in Table 3.4.1-8.

The dominant basic event as measured by risk achievement is the mechanical failure of the control rods to insert into the core (CM). Assuming that the control rods fail in an ATWS following all initiating events with reduced systems available for RPV injection and the required reactivity.

The next basic event is the common cause failure of the ECCS pump room coolers (EPFACCEPRCS). These coolers provide heat removal from the pump rooms. The common cause failure was assessed to be low enough such that setting it to 1.0 would significantly increase its contribution to core damage frequency.

The third basic event in importance as ranked by Risk Achievement is the failure of the Division 2 4160 VAC bus (DGBALC1R22S0006). Failure of this bus would cause a loss of all equipment powered from Division 2 including low pressure coolant injection, heat removal equipment, and containment venting equipment.

Common cause failure of the batteries (DCBTCC) is the fourth most important basic event. As DC power is required for all systems and the batteries are required for any station blackout event, the common cause failure of the batteries is relatively important.

The next three basic events are common cause failures of diesel building ventilation fans, dampers and louvers (DBMFCC, DBMDCC, and DBLVCC). These components support the operation of the diesel generators and are required following a loss of offsite power.

The initiating event for a Large LOCA (A) is the eighth basic event in risk achievement importance.

Of the next ten basic events ranked, 5 are common cause basic events. Common cause is very high in risk achievement because the redundant components are failed at the same time. Setting the common cause basic events to 1.0 therefore significantly increases the core damage frequency.

3.4.1.2 Functional Failure Leading to Core Damage

In order to evaluate the relative contribution of the failure of the various systems or events, other than initiating events, to the overall core damage frequency it is possible to group the core damage sequences by functional failures. The percentage contribution for the following functional failures are shown in Table 3.4.1-9.

Loss of injection (U1, U2, U3, V, V1)

Failure of decay heat removal (Q, M)

Failure of containment heat removal (W, Y, Ws)

Loss of HVAC (Hv, CC)

Failure to restore offsite AC power (R, R1, R2)

Failure to depressurize (X)

Failure to inhibit ADS (X')

The sum of these events is greater than 100% as a number of the sequences contribute to more than one category of functional failure. For example, some sequences involve both loss of offsite power and failure of injection.

It can be seen that failure of injection, both high pressure and low pressure, (68%) is the dominant contributor with failure to depressurize in the event of failure of the high pressure injection systems a very small contributor (4%). This implies that the reliability of the ADS and procedures associated with depressurization are satisfactory and little reduction in core damage frequency would be achieved by improving them.

As anticipated transients without scram is the dominant contributor to core damage, failure to inhibit ADS is a significant contributor to core damage (25%).

Failure of containment heat removal or venting will lead to the so called W sequences, that is sequences in which injection is successful but the failure to remove decay heat from containment results in containment failure and consequential failure of injection leading to core damage. Contribution from this function is 28%. The contribution from failure to recover offsite power is 22% which reflects the fact that loss of offsite power and station blackout events are also a relatively important contribution to core damage. The failure of containment leading to core damage is also significant at 22%.

3.4.1.3 Dominant Accident Sequences

The top 15 dominant accident sequences contributing 81 percent to the core damage frequency are discussed in detail in this section. A complete list of the sequences which contribute 1 percent or more to the core damage frequency and the dominant cutsets for each sequence is given in Appendix F.

Sequence (T3A + T2)-C-U3-X' (T2-CS30)

Frequency 2.27×10^{-6} Contribution 19.5%

A transient has occurred. The PCS may be lost either due directly to the transient or due to subsequent conditions which may result in MSIV isolation. The control rods fail to insert into the core and the reactor remains at power. The motor feed pump has failed to inject into the RPV to maintain RPV level control. The operators have failed to inhibit ADS resulting in rapid depressurization of RPV and injection of low pressure ECCS resulting in a reactivity excursion leading to core damage.

The initiating event T3A (transient with PCS available) contributes almost three times as much to the failure of this sequence than does T2 (transient without PCS). This is due to the failure of the operators to maintain PCS available for an ATWS scenario. Following the initiating events T3A and T2 given the mechanical failure of the control rods, the dominant contributors to this sequence are failure of the operators to re-open the motor feed pump control valves and manually depressurize the RPV and inhibit ADS.

Sequence T2-~~W~~-Y-Cv (T2S04)

Frequency 1.62×10^{-6} Contribution 13.9%

A loss of PCS transient has occurred with a reactor scram and successful SRV operation to maintain RPV pressure control. The motor feed pump has started and is successfully maintaining high pressure RPV level control. The RHR system and venting have failed to provide long-term containment heat removal. Without containment heat removal the containment ruptures disabling the injection path from the motor feed pump and leading to core damage.

The dominant contributors to the failure of this sequence are failure of the injection path upon failure of the containment and failure of 4,160 VAC Division 2 Bus EH12. The maintenance of RHR train A and the failure of the operators to align a containment vent path also contribute to the frequency of this sequence.

Sequence T1-B1-U1-Va-R (BS24)

Frequency 7.71×10^{-7} Contribution 6.6%

A loss of offsite power has occurred and the Division 1 and 2 diesel generators have failed to provide onsite AC power. HPCS has failed to provide high pressure RPV level control. RCIC has successfully provided high pressure RPV level control for 3 hours at which time the suppression temperature limit of 185°F has been exceeded and RCIC fails. The operators have successfully depressurized the RPV, but the fire protection system has failed to provide adequate low pressure RPV level control and offsite power was not recovered at 3 hours leading to core damage.

The dominant contributors to the failure of this sequence are unavailability of the fire protection system to provide alternate injection due to failure of the offsite pumper, failure of the diesel fire pump to run, and failure of the operators to align the fire protection system for RPV injection after RCIC is lost.

Sequence TIA-U1-U2-V-Va (TJAS14)

Frequency 7.53 X 10^{-7} Contribution 6.5%

A loss of instrument air has occurred with a reactor scram and successful SRV operation to maintain RPV pressure control. RCIC and HPCS have failed to provide adequate RPV level control at high pressure. The RPV has been successfully depressurized. With the RPV depressurized low pressure ECCS make-up and low pressure alternate injection have failed to provide RPV level control leading to core damage.

The dominant contributors to the failure of this sequence are failure of the operators to successfully align the reactor feed booster pumps or suppression pool cleanup for alternate low pressure injection. Common cause failure of Emergency Service Water pumps A and B, and other random failures of Emergency Service Water trains A, B and C also contribute to the core damage frequency for this sequence.

Sequence (T3A + T2)-C-U3-X' (T2-CS20)

Frequency 6.25 X 10^{-7} Contribution 5.4%

A transient has occurred. The PCS may be lost either due directly to the transient or due to subsequent conditions which may result in MSIV isolation. The control rods fail to insert into the core and the reactor remains at power. The motor feed pump has successfully injected into the RPV but the operators have failed to control RPV level. The operators have successfully inhibited ADS but have subsequently failed to initiate standby liquid control leading to core damage.

The initiating event T3A (transient with PCS available) contributes almost three times as much to the failure of this sequence than does T2 (transient without PCS). This is due to the failure of the operators to maintain PCS available for an ATWS scenario.

Sequence T1-U1-R1-Ws-V-Va (RS20)

Frequency 6.04 X 10^{-7} Contribution 5.2%

A loss of offsite power has occurred with a reactor scram and successful SRV operation to maintain RPV pressure control. HPCS has failed, but RCIC has successfully provided high pressure RPV level control. At 3 hours RCIC failed due to failure of the Suppression Pool Cooling mode of RHR and non-recovery of offsite power. The RPV has been successfully depressurized. Low pressure ECCS make-up and alternate low pressure make-up have failed leading to core damage.

The dominant contributors to the failure of this sequence are the failure of the operators to align fire protection for alternate injection after RCIC fails and the failure of the Division 3 diesel generator to start. Failure of the Division 1 and 2 diesel generators to start, the failure of the offsite pumper to provide adequate low pressure alternate injection and the failure of the diesel driven fire pump also contribute to the failure of this sequence.

Sequence T1-B1-U1-U2-R-Va (BS34)

Frequency 5.26 X 10^{-7} Contribution 4.5 %

A loss of offsite power has occurred and the Division 1 and 2 diesel generators have failed to provide onsite AC power. HPCS and RCIC have failed to provide high pressure RPV level control. Offsite power was not recovered at 0.4 hours. The operators successfully depressurized the RPV, but fire protection alternate injection failed to provide adequate RPV level control.

The dominant contributors to the failure of this sequence are the failure of the offsite pumper, failure of the operators to bypass the RCIC isolation on high steam tunnel temperature, running failure of the diesel fire pump, failure of the operators to align fire water in a timely manner, and start failure of the division 3 diesel generator. Start failures of the division 1 and 2 diesel generators also contribute to this sequence.

Sequence T1-B1-U1-R (BS17)

Frequency 3.36 X 10^{-7} Contribution 2.9%

A loss of offsite power has occurred and the Division 1 and 2 diesel generators have failed to provide onsite AC power. HPCS has failed to provide high pressure RPV level control. RCIC has successfully provided high pressure RPV level control for 3 hours at which time the operators have successfully depressurized the RPV and aligned fire water as alternate low pressure injection. The batteries fail at 7 hours and offsite power was not recovered by 13 hours. There is no containment heat removal leading to failure of the containment and subsequent failure of RPV injection leading to core damage.

The dominant contributors to the failure of this sequence are maintenance and starting failures of the division 1, 2, and 3 diesel generators.

Sequence T1-U1-U2-R1-V-Va (US29)

Frequency 3.34 X 10^{-7} Contribution 2.9%

A loss of offsite power has occurred with a reactor scram and successful SRV operation to maintain RPV pressure control. HPCS and RCIC have failed to provide successful RPV level control at high pressure. Offsite power was not recovered by 0.4 hours, but the RPV has been successfully depressurized. Depressurization may be delayed until the MZIWL is reached dependent on the injection system alignment. With the RPV depressurized low pressure ECCS make-up and fire protection alternate injection have failed to provide RPV level control leading to core damage.

The dominant contributors to the failure of this sequence are failure of the division 3 diesel generator to start and maintenance of residual heat removal train A, LPCS, and RCIC. Failure of the offsite pumper and failure of the operators to align fire water in a timely manner, and failure of the RCIC turbine driven pump also contribute to the core damage frequency for this sequence.

Sequence (T3A + T2)-C-U3-X (T2-CS28)

Frequency 3.12×10^{-7} Contribution 2.7%

A transient has occurred The PCS may be lost either due directly to the transient or due to subsequent conditions which may result in MSIV isolation. The control rods fail to insert into the core and the reactor remains at power. The motor feed pump was not successfully placed into operation. ADS inhibit and standby liquid control are successful, but depressurization of the RPV by the operators was unsuccessful resulting in core damage.

The dominant contributor to the failure of this sequence is the failure of the operators to re-open the motor feed pump control valves and depressurize the RPV. For the IPE for a transient with PCS available coupled with an ATWS, it was assumed that the MSIV isolation at RPV level was not bypassed. This is also one of the dominant contributors to this sequence.

Sequence (T3A + T2)-C-C1 (T2-CS11)

Frequency 2.90×10^{-7} Contribution 2.5%

A transient has occurred. The PCS may be lost either due directly to the transient or due to subsequent conditions which may result in MSIV isolation. The control rods fail to insert into the core and the reactor remains at power. The operators successfully controlled RPV level with the motor feed pump. ADS inhibit was successful, but standby liquid control was unsuccessful resulting in core damage.

Given a transient leading to a loss of PCS with an ATWS, the dominant contributor to this sequence is the failure of the operators to initiate standby liquid control.

Sequence T3B-C-U3-X' (T3B-CS29)

Frequency 2.76×10^{-7} Contribution 2.4 %

A loss of feedwater transient has occurred, the control rods have failed to insert into the core and the reactor remains at power. Recovery of the motor feed pump has failed. The operators have also failed to inhibit ADS resulting in rapid depressurization of the RPV and initiation of low pressure ECCS causing a reactivity excursion leading to core damage.

The dominant contributors to the failure of this sequence are the failure of the operators to inhibit ADS, re-open the motor feed pump control valves and manually depressurize the RPV.

Sequence T1A-U2-W-Y-Cv (T1AS05)

Frequency 2.57×10^{-7} Contribution 2.2%

A loss of instrument air has occurred with a reactor scram and successful SRV operation to maintain RPV pressure control. RCIC has failed to adequately control RPV level. HPCS has initiated and is successfully maintaining high pressure RPV level control. The RHR system and venting have failed to provide long-term containment heat removal. Without containment heat removal the containment ruptures disabling the injection path from HPCS and leading to core damage.

The dominant contributors to core damage are the failure of the containment and subsequent release of steam into the HPCS pump room resulting in failure of HPCS and failure of the 4,160 VAC bus EH12. Maintenance of RHR train A and the failure of the containment anchorage also contribute to failure of this sequence.

Sequence (T3A + T2)-C-X' (T2-CS12)

Frequency 2.37×10^{-7} Contribution 2.0%

A transient has occurred. The PCS may be lost either due directly to the transient or due to subsequent conditions which may result in MSIV isolation. The control rods fail to insert into the core and the reactor remains at power. The operators successfully controlled RPV level with the motor feed pump, but fail to inhibit ADS resulting in core damage.

The initiating event T3A (transient with PCS available) contributes almost three times as much to the failure of this sequence than does T2 (transient without PCS). This is due to the failure of the operators to maintain PCS available for an ATWS scenario.

Sequence A-U1-V (AS09)

Frequency 2.10×10^{-7} Contribution 1.8%

A large LOCA has occurred with successful reactor scram. HPCS and low pressure ECCS have failed to injection water into the RPV leading t core damage.

The dominant contributors to this sequence are the failure of trains A and B of Emergency Service Water. The maintenance of trains A and B of Emergency Closed Cooling and trains A and B of Emergency Service Water also contribute to this sequence.

3.4.1.4 Contribution to Containment Failure

The development of the containment response following core damage for the various plant damage states is fully described in Chapter 4. This includes a complete analysis of which sequences and plant damage states contribute to the various containment failure modes, and source terms. The significant sequences most likely to contribute to containment failure are summarized in this section. The frequency of containment structure failure is 7.8×10^{-6}

per year and the frequency of RPV failure and early containment failure with pool bypass is 2.0×10^{-6} per year. As shown in Figure 3.4.1-3 sequences resulting in containment bypass (from Event V) represent less than 1% of the core damage frequency, and those core damage sequences resulting in containment failure represent 23%. The sequences contributing 95% of the total to each of the failures and, where appropriate, the plant damage states in which the sequences occur, are discussed below.

Containment Failed Prior to Core Damage for Internal Initiators

In the case of failure of containment prior to core damage, denoted in the event tree by function Cv, the containment failure can occur at the anchorage, in which case the suppression pool would be lost and the containment bypassed. In section 4.4.4 it is shown that for slow overpressurization sequences the ratio of anchorage failure to other failures is 0.15/0.85. The Level 1 sequences without flooding which contribute to overpressure failure are shown in Table 3.4.1-4. The frequency of the Cv sequences is 2.6×10^{-6}. However the frequency of pool bypass is this frequency multiplied by 0.15. Thus the contribution to the RPV failure and early containment failure with pool bypass frequency is 3.9×10^{-7}, which is approximately 3% of the overall core damage frequency. The dominant contribution to this is the loss of feedwater followed by successful injection but failure of all containment heat removal and venting. The contribution from this sequence is 62%. The next highest sequences associated with loss of instrument air contributes 10%. Event V contributes less than 1%.

Containment Failure

The containment structural failure and venting frequency from all sequences is 7.8×10^{-6} per reactor year. The containment venting frequency is 3.7×10^{-6} and the contanment structural failure frequency is 4.0×10^{-6}. The breakdown of the containment failure modes for all plant damage state sequences is shown in Table 3.4.1-10. The plant damage state failure frequency from sequences in which containment fails prior to core damage is 2.9×10^{-6}, so the total containment structural failure and venting frequency after core melt is approximatley 4.9×10^{-6} per reactor year, which is approximately 38 percent of the core damage frequency. The contributors to this from venting after core damage is 3.7×10^{-6} or 77%. Thus the frequency of a containment structural failure without the frequency of the containment failed at core damage is 1.1×10^{-6} which is approximately 9% of the core damage frequency. The frequency of containment structural failure for all station blackout sequences is 6.0×10^{-7} which represents 4.7 percent of the core damage frequency. The plant damage states which contribute 95% of the containment failure frequency and the sequences which contribute to each of these plant damage states are listed in Table 3.4.1-11. It will be noted that for all plant damage states which contribute 95% of the containment failure frequency there is a 100% probability of containment failure given the conditions identified for that plant damage state. The assessment of containment failure for the 16 plant damage states which contribute 95 percent of containment failure and venting is described in Section 4.5.3.

The plant damage states contributing to containment failure and the dominant sequences (above 1% CDF) in each plant damage state are shown in

Table 3.4.1-11.

3.4.1.5 Comparison of Results With and Without Recovery Actions

The quantification of the core damage sequences was based on inclusion of both the automatic initiation of systems and operator actions specifically called out in the procedures in the fault and event tree models. For example following a loss of feedwater the falling water level in the vessel will result in auto initiation of the HPCS, RCIC, or motor feed pump, and the operator is instructed to confirm that the systems are operating. If the systems have failed, he is instructed to perform a series of operations which will result in depressurization of the vessel and use of the low pressure systems to inject water into the vessel. The operator actions associated with use of these systems have been included in the base case model as have all the other actions in the procedures which he would be using.

When the initial quantification of the core damage sequences were complete each combination of failures was reviewed to determine if it represented a direct path to core damage or if as a result of following procedures, the operator would use additional systems to prevent core damage. This resulted in the identification of the following recovery actions which are now modeled in the event trees.

Use of Alternative Low Pressure Injection (including fire protection system) following failure of the Low Pressure Coolant Injection and Low Pressure Core Spray. This is included in the event tree as function Va.

The operator takes the necessary actions to cross connect dc power from the Unit 2 batteries following loss of offsite power to extend the HPCS operating time by providing indication of water level (HI) or provides power to maintain the SRVs open and allow the fire protection system to be used for injection.

The operator manually opens the outboard fuel pool cooling and cleanup isolation valve when there is no electric power available or motor operator failure in order to vent the containment (Y).

The operator crossties the division 3 and division 2 AC buses in a station blackout to allow opening of the inboard fuel pool cooling and cleanup isolation valve.

In the plant damage state trees additional recovery actions are included.

The operator depressurizes after core damage and before vessel failure following early failure to depressurize.

Operator initiates both standby liquid control pumps at a later stage following an ATWS, failure of level control, and before containment failure.

The applicable operator actions are listed in Table 3.4.1-12, and the impact of adding each group of actions associated with a given function, on the core damage frequency is given in Table 3.4.1-13. The overall impact of removing the recovery actions would be to increase the core damage frequency from

1.2×10^{-5} to 4.2×10^{-5} a factor of approximately 3.5. The most significant contribution is the use of the alternate low pressure injection sources, in the event of failure of the low pressure coolant injection and low pressure core spray, followed by the use of the Unit 1/Unit 2 dc cross-tie.

It should be emphasized that these recovery actions have been included in the model because the operator has specific procedures which direct him to use the identified systems to perform the specific functions modeled.
The two recovery actions which have been included in the plant damage state event trees do not lower the core damage frequency but result in a reduction of the frequency of a number of plant damage states. The late injection will reduce the frequency of plant damage states 63 through 66 (Figure 3.4.1-3) and hence lead to a significant reduction in the frequency of release from these plant damage states. The late depressurization will impact all the plant damage states in which the vessel pressure was high at the time of core damage. The transfer of 90% of these sequences to a plant damage state representing low pressure at vessel failure will again have a significant impact on fission product release. A fuller discussion of the impact on fission product release is given in section 4.

3.4.1.6 Sensitivity Analysis

The dominant contribution to core damage comes from sequences associated with ATWS, loss of offsite power and station blackout. The sequences resulting from each of these initiations were reviewed to establish if there were any areas where the success criteria or other assumptions could be construed as either conservative or non conservative. In addition, the human reliability, common cause failure and maintenance contributions to the core damage frequency were reviewed to determine the overall sensitivity of the results to the modeling and data in these areas. One of the reasons this was done was to ensure that sequences which had a low core damage frequency as a result of operator actions were identified. The results of the sensitivity analysis are summarized in Table 3.4.1-14.

Initiating Event Frequency

The initiating event frequency for the initiation is based on the information given in the Grand Gulf study, as, at the beginning of the Perry IPE study information was only available from the first operating cycle. At the time of completion of the study the second and third operating cycles had been completed providing further information on the number of plant trips. In fact in these two operating cycles there was only one inadvertent scram from power, which was the result of a loss of feedwater. Based on this information the initiating event frequencies for the transients T2 (loss of power conversion), T3A (trip with PCS available) and T3B (loss of feedwater) would be lower than those used in the base case. As no T2 or T3A events occurred a value of 0.5/year is assumed for the sensitivity analysis. In the case of T3B one trip occurred in two years; therefore, a frequency of 0.5/year is used. The use of these frequencies reduces the core damage frequency from 1.2×10^{-5} per year to 7.0×10^{-6} per year, a 40% reduction in the core damage frequency.

It is clear from the experience of the last two operating cycles that the initiating event frequencies at Perry for the above transient categories are significantly lower than the industry values used in the Grand Gulf study. As the Perry IPE is intended to be a living PRA the plant specific information on the initiating event frequencies (and other applicable data such as maintenance unavailabilities) will be included at the first updating of the model.

Another significant impact of this change in the frequency of the initiating events is the change in the relative contributions to the overall core damage frequency. It can be seen in Table 3.1.4-15 that the dominant contribution is now station blackout at 32% followed by loss of offsite power at 20%. The ATWS contribution is now 16.5%, which is less than the combined transient contribution of approximately 23%. The importance of these results when considering plant improvements is discussed in Chapter 6.

Anticipated Transient Without Scram

The critical function of borated water injection, RPV level control and revention of depressurization all require the intervention of the operator. The core damage frequency following an ATWS event is therefore sensitive to the assumptions made in determining the operator error rates in each of these functions. For example, if the operator failure to inhibit ADS is increased to 1.0 the overall core damage frequency goes from 1.2×10^{-5} to 8.6×10^{-5}. However the ATWS sequences frequency goes from 4.7×10^{-6} to 7.9×10^{-5}, a factor of approximately 17. Similar increases occur if the values of the operator error rate for the other functions are increased.

As stated earlier this dependency arises because of the following assumptions and design features at Perry.

1. It is now considered that operation of the HPCS is not an acceptable method of maintaining vessel inventory following failure to insert the control rods.

2. The standby liquid control system has to be manually initiated.

3. The feedwater runback reduces feedwater injection to zero and therefore manual control is required to restore feedwater.

4. It is necessary to inhibit ADS to prevent depressurization and injection of water from the high capacity low pressure systems. This action has to be performed by the operator.

Although the automation of a number of these actions will not make a major reduction in the core damage frequency, it will reduce the dependency on the operator to perform the overt action. For example, the introduction of an automatic ADS inhibit following ATWS would reduce the core damage frequency for 1.2×10^{-5} to 8.8×10^{-6}, approximately 24% but it would also remove any dependency on the operator for the primary response to the particular requirement.

Loss of Offsite Power Sequences

The impact of the assumption that the unit 2 battery can be used to supply dc and that the firewater can be used as an injection source have already been discussed in the previous section (3.4.1.5) on recovery actions. In addition two other assumptions have been made in evaluating the base case core damage frequency.

1. The operator will override the RCIC leak detection trip on loss of offsite power.

2. Room cooling is not required for the switchgear rooms to prevent failure of ac power from the diesels following loss of offsite power.

For the first assumption if the RCIC trip is not overridden, the overall core damage frequency increases from 1.2×10^{-5} to 1.8×10^{-5} and if switchgear room cooling is required, from 1.2×10^{-5} to 1.5×10^{-5}. The current analysis gives a high degree of confidence in the validity of both the above assumptions.

Impact of Containment Failure

The current containment failure analysis indicates that a proportion of containment failures, as the result of loss of containment heat removal following successful reactor vessel injection, will lead to a loss of injection and thus core damage with a failed containment. As indicated in Table 3.1.4-9, such sequences contribute 22% to the core damage frequency. Thus if a passive vent were fitted, it could be shown that containment failure will not lead to loss of injection and the core damage frequency would be reduced by 22% from 1.2×10^{-5} to 9×10^{-6}. This reduction would be even more significant in terms of fission product release as it would eliminate a class of core damage sequences with a failed containment, which result in high fission product release.

Maintenance Contribution to Core Damage

Maintenance of the ECCS and associated support systems results in periods of unavailability of the systems. The data used in the study is based on the plant specific information available at the time of the quantification of the event trees in 1991 and updated for cycle 3 for the HPCS and RCIC systems. A sensitivity study has been performed to determine the contribution of maintenance outages on the core damage frequency.

It can be seen that by putting all maintenance to zero, the core damage frequency is reduced by 45% indicating that this is the maintenance contribution to the core damage frequency. It is significant to note that the contribution from HPCS and RCIC for which the cycle 3 information has been used, is approximately 1%. There is an extensive Systematic Maintenance Optimization (SMO) program at the plant for all ECCS systems. When this is complete the plant specific data can be used to update the values used in the current base case.

Human Reliability Modeling

The inclusion of recovery actions is discussed in section 3.4.1.5 however the sensitivity of the core damage frequency to the other human reliability data is also of interest. This has already been discussed to a limited extent in the early paragraphs on the ATWS sequences.

The current values for all human reliability basic events have been based on a systematic approach to the quantification of the human actions. If it is assumed that there is a bias in this analysis such that all events were assumed to be too low then the core damage frequency would be higher. To assess the overall sensitivity to the human error probabilities the values were increased by a factor of ten if the base value was less than 0.1 and to a value of 1.0 if the base case was between 0.1 and 1.0. The core damage frequency increased from 1.2×10^{-5} to 4.3×10^{-4}, a factor of 28. This indicates that operator actions do play a significant part in determining the core damage frequency and emphasize the importance of the instruction and guidance in the emergency procedures, particularly as they relate to the 41 actions identified in the core damage sequences which contribute to the core damage frequency. Particular emphasis has been placed on the analysis of multiple interdependent actions. This is fully described in Appendix D.

The sequences which but for operator actions would have been above the screening value of 1.0×10^{-7} are shown in Table 3.4.1-5. The sequences fall into three classes, those associated with ATWS events those associated with failure of decay heat removal and those associated with manual depressurization. In all cases those sequences which are increased to the 10^{-3} to 10^{-4} range contain operator actions which are currently in sequences above the cut off and for which dependency analysis has been performed. This is particularly so in the case of the ATWS events and the failure to initiate containment heat removal or venting. It is concluded that there are no significant sequences which would be above 10^{-7} if a more detailed human reliability analysis was performed. Finally if all the operators performed the required actions perfectly the core damage frequency would be reduced to 3.9×10^{-6}.

Common Cause Failures

If the common cause failures were all increased by a factor of 10, the core damage frequency would increase from 1.2×10^{-5} to 2.2×10^{-5} which is less than a factor of 2. As the result of the major contribution to core damage from the ATWS sequences common cause failures of the cooling and support systems such as the emergency service water or diesel generators do not contribute significantly to the overall core damage frequency. However, as described in section 3.3, a detailed analysis has been made of the critical components in these systems.

3.4.2 VULNERABILITY SCREENING

A concise definition of vulnerability is not given in the documentation associated with the performance and reporting of the IPE. In the response to questions in Appendix C to the Submittal Guidance Document (NRC, 1989), mention is made of examining sequences that are above the screening criteria in order to determine if a weakness exists. Thus the word weakness replaces the word vulnerability, neither of which is defined in numerical or comparative terms. In another response it is suggested that a vulnerability is an outlier. The NUMARC Severe Accident Issues Closure Guidelines (NUMARC, 1992) proposes a set of guidelines based on a combination of the core damage frequency for a group of sequences and the individual contribution from a sequence group. If the contribution from a given initiator or system failure is greater than 50 per cent to the total core damage frequency it is interpreted as a significant vulnerability, if it contributes 20-50% it is interpreted as a potential vulnerability to be investigated. Similarly contribution from sequence groups between a core damage frequency of 10^{-5} to 10^{-4} are reviewed to determine if there is an effective plant procedure or hardware change which would reduce the frequency of the sequences.

In this study Importance and Sensitivity measures have been used to determine the most significant contributions to the core damage frequency, containment system performance, and decay heat removal functions as discussed in section 3.4.1.

3.4.2.1 Internal Event Core Damage Vulnerabilities

The breakdown of core damage by initiating event is shown in Table 3.4.1-6 and by sequence with function failure in Table 3.4.1-3. However in order to evaluate the vulnerabilities in terms of the criteria proposed by NUMARC, it is appropriate to regroup the sequences into a series of functional accident groups according to the criteria shown in Table 3.4.2-1.

The list of sequences and frequency of each group is shown in Table 3.4.2-2. It can be seen from this table that there are no significant vulnerabilities as defined in the previous section as all the accident sequence groups have a frequency below 1.0×10^{-5} and no group contributes more than 50% to the overall core damage frequency. However there are two groups of accident sequences that contribute between 20 and 50 per cent. Group 4 which is made up of accident sequences involving an ATWS leading to containment failure due to high pressure and subsequent loss of inventory, and Group 2 which is made up of accident sequences involving loss of containment heat removal leading to containment failure and subsequent failure of coolant inventory make-up.

Group 4

The contribution to core damage from sequences in this group comes primarily from ATWS sequences in which the motor feed pump has failed to inject water and ADS has not been inhibited resulting in rapid depressurization of the RPV and injection of low pressure ECCS. This leads to a series of reactivity oscillations resulting in generation of large quantities of steam and ultimately containment failure and core damage. In these sequences the potential vulnerability is the failure to inhibit ADS. However it should be noted from the sensitivity analysis that the use of the plant operating data for cycle 2 and 3 will reduce the frequency of the initiators which contribute to this group and thus the contribution to core damage frequency of these sequences from 33% to approximately 15% which is no longer a potential vulnerability.

Group 2

The contribution to core damage from sequences in this group comes from failure of containment heat removal leading to containment failure and subsequent loss of injection. One of the reasons that this is a significant contributor is that the containment design is such that in approximately 15% of cases containment failure leads to injection failure. Thus if a passive vent was fitted, the core damage frequency of these sequences would be reduced. This also has an impact on source term magnitude and is further discussed in section 3.4.2.3 under containment vulnerabilities. However it should be emphasized that the overall frequency of this group 2.6×10^{-6} is well below 1.0×10^{-5}.

3.4.2.2 Flooding Core Damage Vulnerabilities

There are no vulnerabilities associated with internal flooding. the total contribution to core damage frequency is 12 per cent from all floods and flooding in the most significant area, Zone 13 only contributes 7 per cent to the overall core damage frequency. The sensitivity analysis shows that it is important to ensure that the supply valves to the Unit 2 heat exchanger, which is in dry lay-up with the bell ends removed, cannot be opened.

3.4.2.3 Containment Vulnerabilities

The Perry containment design pressure is 15 psig. The median containment pressure capacity is estimated to be 64.3 psig. Results from the containment accident progression event tree analysis indicate a containment failure can occur from hydrogen burns and from gradual overpressurization. The systems analysis indicates a relatively high probability of loss of venting capability for LOOP sequences with loss of containment heat removal. The loss of both RHR and venting will eventually result in containment overpressure failure (assuming failure from some other mechanism has not previously occurred).

The Perry Mark III steel containment has failure modes which will have an impact on accident progression. If containment failure should result in penetration failure, then ECCS pump failure may occur due to steam release pathways from the Shield Building to the Auxiliary Building through penetration seals. If containment failure results in anchorage failure, then the ECCS pumps may fail as the result of piping failure and loss of suppression pool. Suppression pool water loss and drywell bypass can occur during anchorage failure when the water inventory is expelled to the Shield Building and adjacent reactor buildings. Additionally, anchorage rupture can result in a direct radiological release to the environment. The containment performance is discussed in more detail in Section 4.

3.4.3 DECAY HEAT REMOVAL EVALUATION - ISSUE A-45

The objectives of Task Action Plan A-45 are to evaluate the safety adequacy of Decay Heat Removal (DHR) Systems in existing light water reactor nuclear power plants and to assess the value and impact (benefit-cost) of alternative measures for improving the overall reliability of the DHR function if required.

Some potential accidents which could result in core melt were excluded from the A-45 studies performed by Sandia National Laboratories (S.W. Hatch, 1987). Since the purpose of the program is to study the adequacy of shutdown decay heat removal systems, large LOCAs, reactor vessel ruptures, interfacing system LOCAs and anticipated transients without scram (ATWS) are excluded. The study focused on events occurring from power or in hot standby.

a failed containment. with consequent increase in offsite release. The addition of an alternate passive vent path will reduce the frequency of core damage and result in a reduction in offsite release. That is the overall core damage frequency would be reduced from 1.2×10^{-5} to 9.0×10^{-6} but the overall frequency of core damage sequence with the suppression pool bypassed is reduced from 2.0×10^{-6} to 4.5×10^{-7} which will have an impact on the source terms.

It is considered that in view of the low frequency of the core damage sequences resulting from failure of decay heat removal (5.1×10^{-6} per year) the performance of these systems is adequate and that this analysis provides the necessary information for adequate resolution of unresolved safety issue A-45 for the Perry Nuclear Power Plant.

McClymount, A.S. and B.W. Pohlman, 1982, ATWS: A Reappraisal: Part 3 Frequency of Anticipated Transients, EPRI-NP-2230, Electric Power Research Institute, Palo Alto, CA.

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WASH-1400 (NUREG-75/014), Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, October 1975.

The current values for all human reliability basic events have been based on a systematic approach to the quantification of the human actions. If it is assumed that there is a bias in this analysis such that all events were assumed to be too low then the core damage frequency would be higher. To assess the overall sensitivity to the human error probabilities the values were increased by a factor of ten if the base value was less than 0.1 and to a value of 1.0 if the base case was between 0.1 and 1.0. The core damage frequency increased from 1.2×10^{-5} to 4.3×10^{-4}, a factor of 28. This indicates that operator actions do play a significant part in determining the core damage frequency and emphasize the importance of the instruction and guidance in the emergency procedures, particularly as they relate to the 41 actions identified in the core damage sequences which contribute to the core damage frequency. Particular emphasis has been placed on the analysis of multiple interdependent actions. This is fully described in Appendix D.

The sequences which but for operator actions would have been above the screening value of 1.0×10^{-7} are shown in Table 3.4.1-5. The sequences fall in three classes, those associated with ATWS events, those associated with failure of decay heat removal and those associated with manual depressurization. In all cases those sequences which are increased to the 10^{-3} to 10^{-4} range contain operator actions which are currently in sequences above the cut off and for which dependency analysis has been performed. This is particularly so in the case of the ATWS events and the failure to initiate containment heat removal or venting. It is concluded that there are no significant sequences which would be above 10^{-7} if a more detailed human reliability analysis was performed. Finally if all the operators performed the required actions perfectly the core damage frequency would be reduced to 3.9×10^{-6}.

Common Cause Failures

If the common cause failures were all increased by a factor of 10, the core damage frequency would increase from 1.2×10^{-5} to 2.2×10^{-5} which is less than a factor of 2. As the result of the major contribution to core damage from the ATWS sequences common cause failures of the cooling and support systems such as the emergency service water or diesel generators do not contribute significantly to the overall core damage frequency. However, as described in section 3.3, a detailed analysis has been made of the critical components in these systems.

3.4.2 VULNERABILITY SCREENING

A concise definition of vulnerability is not given in the documentation associated with the performance and reporting of the IPE. In the response to questions in Appendix C to the Submittal Guidance Document (NRC, 1989), mention is made of examining sequences that are above the screening criteria in order to determine if a weakness exists. Thus the word weakness replaces the word vulnerability, neither of which is defined in numerical or comparative terms. In another response it is suggested that a vulnerability is an outlier. The NUMARC Severe Accident Issues Closure Guidelines (NUMARC, 1992) proposes a set of guidelines based on a combination of the core damage frequency for a group of sequences and the individual contribution from a sequence group. If the contribution from a given initiator or system failure is greater than 50 per cent to the total core damage frequency it is interpreted as a significant vulnerability, if it contributes 20-50% it is interpreted as a potential vulnerability to be investigated. Similarly contribution from sequence groups between a core damage frequency of 10^{-5} to 10^{-4} are reviewed to determine if there is an effective plant procedure or hardware change which would reduce the frequency of the sequences.

In this study Importance and Sensitivity measures have been used to determine the most significant contributions to the core damage frequency, containment system performance, and decay heat removal functions as discussed in section 3.4.1.

3.4.2.1 Internal Event Core Damage Vulnerabilities

The breakdown of core damage by initiating event is shown in Table 3.4.1-6 and by sequence with function failure in Table 3.4.1-3. However in order to evaluate the vulnerabilities in terms of the criteria proposed by NUMARC, it is appropriate to regroup the sequences into a series of functional accident groups according to the criteria shown in Table 3.4.2-1.

The list of sequences and frequency of each group is shown in Table 3.4.2-2. It can be seen from this table that there are no significant vulnerabilities as defined in the previous section as all the accident sequence groups have a frequency below 1.0 x 10^{-5} and no group contributes more than 50% to the overall core damage frequency. However there are two groups of accident sequences that contribute between 20 and 50 per cent. Group 4 which is made up of accident sequences involving an ATWS leading to containment failure due to high pressure and subsequent loss of inventory, and Group 2 which is made up of accident sequences involving loss of containment heat removal leading to containment failure and subsequent failure of coolant inventory make-up.

Group 4

The contribution to core damage from sequences in this group comes primarily from ATWS sequences in which the motor feed pump has failed to inject water and ADS has not been inhibited resulting in rapid depressurization of the RPV and injection of low pressure ECCS. This leads to a series of reactivity oscillations resulting in generation of large quantities of steam and ultimately containment failure and core damage. In these sequences the potential vulnerability is the failure to inhibit ADS. However it should be noted from the sensitivity analysis that the use of the plant operating data for cycle 2 and 3 will reduce the frequency of the initiators which contribute to this group and thus the contribution to core damage frequency of these sequences from 33% to approximately 15% which is no longer a potential vulnerability.

Group 2

The contribution to core damage from sequences in this group comes from failure of containment heat removal leading to containment failure and subsequent loss of injection. One of the reasons that this is a significant contributor is that the containment design is such that in approximately 15% of cases containment failure leads to injection failure. Thus if a passive vent was fitted, the core damage frequency of these sequences would be reduced. This also has an impact on source term magnitude and is further discussed in section 3.4.2.3 under containment vulnerabilities. However it should be emphasized that the overall frequency of this group 2.6×10^{-6} is well below 1.0×10^{-6}.

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The Perry containment design pressure is 15 psig. The median containment pressure capacity is estimated to be 64.3 psig. Results from the containment accident progression event tree analysis indicate a containment failure can occur from hydrogen burns and from gradual overpressurization. The systems analysis indicates a relatively high probability of loss of venting capability for LOOP sequences with loss of containment heat removal. The loss of both RHR and venting will eventually result in containment overpressure failure (assuming failure from some other mechanism has not previously occurred).

The Perry Mark III steel containment has failure modes which will have an impact on accident progression. If containment failure should result in penetration failure, then ECCS pump failure may occur due to steam release pathways from the Shield Building to the Auxiliary Building through penetration seals. If containment failure results in anchorage failure, then the ECCS pumps may fail as the result of piping failure and loss of suppression pool. Suppression pool water loss and drywell bypass can occur during anchorage failure when the water inventory is expelled to the Shield Building and adjacent reactor buildings. Additionally, anchorage rupture can result in a direct radiological release to the environment. The containment performance is discussed in more detail in Section 4.

3.4.3 DECAY HEAT REMOVAL EVALUATION - ISSUE A-45

The objectives of Task Action Plan A-45 are to evaluate the safety adequacy of Decay Heat Removal (DHR) Systems in existing light water reactor nuclear power plants and to assess the value and impact (benefit-cost) of alternative measures for improving the overall reliability of the DHR function if required.

Some potential accidents which could result in core melt were excluded from the A-45 studies performed by Sandia National Laboratories (S.W. Hatch, 1987). Since the purpose of the program is to study the adequacy of shutdown decay heat removal systems, large LOCAs, reactor vessel ruptures, interfacing system LOCAs and anticipated transients without scram (ATWS) are excluded. The study focused on events occurring from power or in hot standby.

The delineation of the accident sequences, system analysis and quantifications are fully described in the preceding sections 3.1 through 3.4.1. The identification and ranking of the plant vulnerabilities are described in section 3.4.2. However the concern in issue A-45 is to identify the specific vulnerabilities associated with sequences identified as potentially leading to core damage if all injection to the vessel and decay heat removal is lost.

The breakdown of functional contribution to core damage is shown in Table 3.4.2-1. From this table it can be seen that failure of decay heat removal as the result of failure of injection contributes 22 percent to core damage with individual function contributions as follows:

Group	Description	Percentage
1B	Loss of offsite power and make-up	13
1E	Loss of coolant inventory make-up at low pressure	8
1A	Loss of high pressure make-up and failure to depressurize	<1

The contribution to core damage frequency as the result of loss of decay heat removal from containment leading to containment failure and consequential loss of injection (class 2) is 22 percent of the overall core damage frequency. Thus 44 per cent of the contribution to core damage is as the direct result of failure of decay heat removal either from the vessel or containment. Although the overall frequency is low (5.1×10^{-6}) it is of interest to identify any potential vulnerabilities which, if they could be addressed, in a cost effective manner would reduce the contribution of decay heat removal failure to the overall core damage frequency.

The importance analysis discussed in section 3.4.1.1. and listed in Table 3.4.1-7 identifies the individual components whose improvement would make the greatest contribution to the reduction in core damage frequency. The highest ranked component in terms of decay heat removal is the event representing failure of injection as the result of containment failure (CV05). This indicates that the mode of containment failure following overpressurization is more significant than any individual containment heat removal system failure. Therefore an alternative means to prevent containment failure will have the biggest impact in reducing the core damage frequency.

The contributions from the remaining individual components is small therefore no improvement of an individual component will have a significant impact on the contribution to decay heat removal failure.

3.4.3.1 Summary and Conclusions

The core damage frequency resulting form failure of decay heat removal systems is 5.1×10^{-6} representing 44 percent of the contribution to core damage form internal events. A review of the contribution to this failure identified one event which contributed twice as much as any others to the core damage frequency for which a possible modification may be considered. The mode of containment failure results in a significant probability of injection failure following containment failures and subsequent core melt in a failed containment, with consequent increase in offsite release. The addition of an alternate passive vent path will reduce the frequency of core damage and result in a reduction in offsite release. That is the overall core damage frequency would be reduced from 1.2×10^{-5} to 9.0×10^{-6} but the overall frequency of core damage sequence with the suppression pool bypassed is reduced from 2.0×10^{-6} to 4.5×10^{-7} which will have an impact on the source terms.

It is considered that in view of the low frequency of the core damage sequences resulting from failure of decay heat removal (5.1×10^{-6} per year) the performance of these systems is adequate and that this analysis provides the necessary information for adequate resolution of unresolved safety issue A-45 for the Perry Nuclear Power Plant.

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Swain, Alan D., 1987, Accident Sequence Evaluation Program Human Reliability Analysis Procedure, NUREG/CR-4772, Sandia National Laboratories, Albuquerque, NM.

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WASH-1400 (NUREG-75/014), Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, October 1975.

Table 3.1.1-1

PNPP IPE Initiating Event List

Designation*	Description	Mean Frequency
T1	Loss of Offsite Power Transient	6.09 E-2
T2	Transients with the loss of the Power Conversion System (PCS)	1.62
T3A	Transients with PCS initially available	4.51
T3B	Transients involving loss of feedwater, but with the PCS initially available	7.60 E-1
T3C	Transients caused by an Inadvertent Open Relief Valve (IORV) on the RPV	1.40 E-1
TIA	Transient caused by a loss of Instrument Air	9.20 E-2
TSW	Transient caused by a loss of Service Water	1.0 E-3
A	Large Loss of Coolant Accident (LOCA)	1.0 E-4
S1	Intermediate LOCA	3.0 E-4
S2	Small LOCA	3.0 E-3
V	Interface Systems LOCA	<1.0 E-8
O	Containment Bypass LOCA	<1.0 E-8
R	Vessel Rupture	1.0 E-7

* The letters used to designate a specific initiating event Group are, where possible, those used in the Grand Gulf Study, and generally in other BWR studies.

Table 3.1.1-2

EPRI BWR Transient Category Definitions

EPRI Category	Title and Definition
1.	Electrical Load Rejection - the electrical load rejection transient occurs when electrical grid disturbances result in significant loss of load on the generator. Also included are intentional generator trips.
2.	Electrical Load Rejection with Turbine Bypass Valve Failure - this transient is identical to No. 1 except that the turbine bypass valves do not open simultaneously with shutdown of the turbine.
3.	Turbine Trip - a turbine trip transient occurs when any one of a number of turbine or nuclear system malfunctions requires the turbine to be shut down. Turbine trips which occur as a by-product of other transients such as loss of condenser vacuum or reactor high level trip are not included. Intentional turbine trips are included.
4.	Turbine Trip with Turbine Bypass Valve Failure - this transient is identical to No. 3 except that the turbine bypass fails to open.
5.	Main Steam Isolation Valve (MSIV) Closure - the MSIV closure transient occurs when any one of various steam line and nuclear system malfunctions require termination of steam flow from the vessel automatically or by operator action.
6.	Inadvertent Closure of One MSIV - only one MSIV closes, the rest remaining open, due to operator or equipment error.
7.	Partial MSIV Closure - partial closure of one or more MSIVs results from a hardware or human error.
8.	Loss of Normal Condenser Vacuum - either a complete loss or decrease in condenser vacuum results from a hardware or human error.
9.	Pressure Regulator Fails Open - either the controlling pressure regulator or backup regulator fails in the open direction. The failure causes a decreasing coolant inventory as the mass flow of water entering the vessel decreases.
10.	Pressure Regulator Fails Closed - either the controlling pressure regulator or backup regulator fails in the closed direction. This failure causes increasing pressure and thus decreasing steam flow from the vessel.
11.	Inadvertent Opening of a Safety/Relief Valve (Stuck) - a safety/relief valve sticks open. Due to an operator error or equipment error a single safety/relief valve can be opened, increasing steam flow from the vessel. If the valve cannot be closed, a scram is initiated. This transient only includes those openings that cannot be subsequently closed before a scram occurs.

Table 3.1.1-2 continued

EPRI BWR Transient Category Definitions

EPRI Category	Title and Definition
12.	Turbine Bypass Fails Open - equipment or operator error results in inadvertent or excessive opening of turbine bypass valves so as to decrease the vessel level.
13.	Turbine Bypass or Control Valves Cause Increased Pressure (closed) - either operator error or equipment failure causes the turbine bypass or control valves to close, resulting in increased system pressure.
14.	Recirculation Control Failure - Increasing Flow - a failure of a flow controller, either in one loop or the master flow controller, causes an increasing flow in the core.
15.	Recirculation Control Failure - Decreasing Flow - flow controller failure failure causes a decreased flow to the core.
16.	Trip of One Recirculation Pump - one recirculation pump trips due to a hardware or human error.
17.	Trip of All Recirculation Pumps - the simultaneous loss of all recirculation pumps occurs.
18.	Abnormal Startup of Idle Recirculation Pump - an idle recirculation pump is started at an improper power and flow condition. The increased flow could cause a flux spike, or, if the loop has been idle so as to allow coolant in the pump loop to cool, core inlet subcooling.
19.	Recirculation Pump Seizure - the failure of a recirculation pump is such that no coastdown occurs, and a sudden flow decrease is experienced.
20.	Feedwater - Increasing Flow at Power - event causes increasing feedwater flow at power. Excluded (see Category 26) are increasing flow events during startup or shutdown, when manual control is being utilized.
21.	Loss of Feedwater Heater - the loss-of-feedwater heating is such that the reactor vessel receives feedwater cool enough to exceed core scram parameters.
22.	Loss of All Feedwater Flow - the simultaneous loss-of-feedwater flow, excluding that due to the loss-of-offsite power (see Category 31), occurs.
23.	Trip of One Feedwater Pump (or Condensate Pump) - the loss of one feedwater or condensate pump is such that a partial loss of feedwater is experienced.

Table 3.1.1-2 continued

EPRI BWR Transient Category Definitions

EPRI Category	Title and Definition
24.	Feedwater - Low Flow - plant occurrence causes decreasing feedwater flow at power. Excluded are events at low power (see Category 25).
25.	Low Feedwater Flow During Startup - event results in low feedwater flow at essentially zero power; this definition includes only startup or shutdown operations.
26.	High Feedwater Flow During Startup or Shutdown - excessive feedwater flow occurs during startup or shutdown. This reactor is essentially at zero power.
27.	Rod Withdrawal at Power - one or more control rods are withdrawn inadvertently in the power range of plant operation.
28.	High Flux Due to Rod Withdrawal at Startup - the inadvertent withdrawal of a control rod causes a local power increase.
29.	Inadvertent Insertion of Rod or Rods - malfunction causes an inadvertent insertion of rod or rods during power operation.
30.	Detected Fault in Reactor Protection System - a scram is initiated due to an indicated fault in the reactor protection system. An example is the indication of a high level in the scram discharge volume.
31.	Loss of Offsite Power - all power to the plant from external sources (the grid of dedicated transmission lines to another plant) is lost. This event requires the plant emergency power sources to be available.
32.	Loss of Auxiliary Power (Loss of Auxiliary Transformer) - the loss of incoming power to a plant results from onsite failures such as the loss of an auxiliary transformer.
33.	Inadvertent Startup of HPCI/HPCS - one of the systems supplying high pressure cold water to the vessel inadvertently starts up. In general, a BWR will have either a high pressure coolant injection (HPCI) system or a high pressure core spray (HPCS) system.
34.	Scram Due to Plant Occurrences - an automatic or manual scram is initiated by an occurrence that does not cause an out-of-tolerance condition in the primary system, but requires shutdown. Examples are turbine vibration, off-gas explosion, fire, excess conductivity of reactor coolant, etc.
35.	Spurious Trip Via Instrumentation, RPS Fault - a scram resulting from hardware or human error in instrumentation or logic circuits occurs.

Table 3.1.1-2 continued

EPRI BWR Transient Category Definitions

EPRI Category	Title and Definition
36.	Manual Scram - No Out-of-Tolerance Condition - a manual initiation of a scram, either purposely or by error, occurs and there are no out-of-tolerance conditions.
37.	Cause Unknown - a scram occurs, but the cause was not determinable.

Table 3.1.1-3

Interfacing LOCA Evaluation

System	Valves	Failure Mode	Valve Type	Initial Position
MSIV Leakage Control	1E32-F001	Fails to Remain Closed	MOV	Closed
(discharges to	1E32-F002	Fails to Remain Closed	MOV	Closed
Annulus)	1E32-F003	Fails to Remain Closed	MOV	Closed
(discharges to Aux	1E32-F006	Fails to Remain Closed	MOV	Closed
Building and Annuls)	1E32-F007	Fails to Remain Closed	MOV	Closed
(discharges to	1E32-F008	Fails to Remain Closed	MOV	Closed
Annulus)	1E32-F009	Fails to Remain Closed	MOV	Closed
RCIC (water side)	1E51-F066	Fails to Prevent Backflow	Check	Closed
(pressurizes RCIC	1E51-F065	Fails to Prevent Backflow	Check	Closed
pump suction)	1E51-F013	Fails to Remain Closed	MOV	Closed
(pressurizes RHR)	1E51-F066	Fails to Prevent Backflow	Check	Closed
	1E51-F065	Fails to Prevent Backflow	Check	Closed
	1E12-F019	Fails to Prevent Backflow	Check	Closed
	1E12-F023	Fails to Remain Closed	MOV	Closed
(discharges to CST)	1E51-F066	Fails to Prevent Backflow	Check	Closed
	1E51-F065	Fails to Prevent Backflow	Check	Closed
	1E51-F013	Fails to Remain Closed	MOV	Closed
	1E51-F022	Fails to Remain Closed	MOV	Closed
	1E51-F059	Fails to Remain Closed	MOV	Closed
RHR-Steam Condensing		Fails to Close	MOV	Open
(pressurizes RHR)	1E51-F064	Fails to Close	MOV	Open
	1E12-F052A,B	Fails to Remain Closed	MOV	Closed
	1E12-F087A,B	Fails to Remain Closed	MOV	Closed
	1E12-F055A,B	Fails to Open	RV	Closed

Table 3.1.1-3 continued

Interfacing LOCA Evaluation

System	Valves	Failure Mode	Valve Type	Initial Position
(pressurizes RHR)	1E51-F063	Fails to Close	MOV	Open
	1E51-F064	Fails to Close	MOV	Open
	1E12-F052A,B	Fails to Remain Closed	MOV	Closed
	1E12-F051A,B	Fails to Remain Closed	PCV	Closed
	1E12-F055A,B	Fails to Open	RV	Closed
RHR-Shutdown Cooling (suction) (pressurizes RHR)	1E12-F009	Fails to Remain Closed	MOV	Closed
	1E12-F008	Fails to Remain Closed	MOV	Closed
RHR-Shutdown Cooling (return) (pressurizes RHR)	1N27-F559A,B	Fails to Prevent Backflow	Check	Closed
	1B21-F032A,B	Fails to Prevent Backflow	Check	Closed
	1E12-F050A,B	Fails to Prevent Backflow	Check	Closed
	1E12-F053A,B	Fails to Remain Closed	MOV	Closed
RHR-LPCI (pressurizes RHR)	1E12-F041A,B,C	Fails to Prevent Backflow	Check	Closed
	1E12-F042A,B,C	Fails to Remain Closed	MOV	Closed
LPCS (pressurizes LPCS)	1E21-F006	Fails to Prevent Backflow	Check	Closed
	1E21-F005	Fails to Remain Closed	MOV	Closed
RWCU (suction) (pressurizes low pressure side of RWCU)	1G33-F001	Fails to Close	MOV	Open
	1G33-F004	Fails to Close	MOV	Open
	1G33-F054	Fails to Close	MOV	Open
	1G33-F053	Fails to Close	MOV	Open
RWCU (return) (pressurizes low pressure side of RWCU)	1N27-F559A,B	Fails to Prevent Backflow	Check	Open
	1B21-F032A,B	Fails to Prevent Backflow	Check	Open
	1G33-F052A,B	Fails to Prevent Backflow	Check	Open
	1G33-F039	Fails to Close	MOV	Open
	1G33-F040	Fails to Close	MOV	Open

Table 3.1.1-3 continued

Interfacing LOCA Evaluation

System	Valves	Failure Mode	Valve Type	Initial Position
HPCS	1E22-F005	Fails to Prevent Backflow	Check	Closed
(pressurizes HPCS	1E22-F004	Fails to Remain Closed	MOV	Closed
pump suction)	1E22-F024	Fails to Prevent Backflow	Check	Closed
(discharges to CST)	1E22-F005	Fails to Prevent Backflow	Check	Closed
	1E22-F004	Fails to Remain Closed	MOV	Closed
	1E22-F010	Fails to Remain Closed	MOV	Closed
	1E22-F011	Fails to Remain Closed	MOV	Closed
Standby Liquid Control	1C41-F007	Fails to Prevent Backflow	Check	Closed
(pressurizes SLC	1C41-F006	Fails to Prevent Backflow	Check	Closed
pump suction)	1C41-F004A,B	Fails to Remain Closed	Squib	Closed
	1C41-F033A,B	Fails to Prevent Backflow	Check	Closed
	1C41-F029A,B	Fails to Remain Closed	RV	Closed
Feedwater	1N27-F559A,B	Fails to Prevent Backflow	Check	Open
(pressurizes FW	1B21-F032A,B	Fails to Prevent Backflow	Check	Open
pump suction)	1N27-F514A,B	Fails to Prevent Backflow	Check	Open

Table 3.1.1-4

LOCA Bypassing Containment Evaluation

System	Valves	Failure Mode	Valve Type	Initial Position
Main Steam	1B21-F022A,B,C,D	Fails to Close	MSIV	Open
	1B21-F028A,B,C,D	Fails to Close	MSIV	Open
MSIV Leakage Control (inboard drains)	1N22-F016	Fails to Close	MOV	Closed
	1N22-F019	Fails to Close	MOV	Closed
(outboard drains)	1N22-F067A,B,C,D	Fails to Remain Closed	MOV	Closed
RCIC (water side)	1E51-F066	Fails to Prevent Backflow	Check	Closed
	1E51-F065	Fails to Prevent Backflow	Check	Closed
RCIC (steam side) & RHR-Steam Condensing	1E51-F063	Fails to Close	MOV	Open
	1G33-F004	Fails to Close	MOV	Open
RWCU (suction)	1G33-F001	Fails to Close	MOV	Open
	1G33-F004	Fails to Close	MOV	Open
RWCU (return)	1N27-F559A,B	Fails to Prevent Backflow	Check	Open
	1B21-F032A,B	Fails to Prevent Backflow	Check	Open
	1G33-F052A,B	Fails to Prevent Backflow	Check	Open
	1G33-F039	Fails to Close	MOV	Open
	1G33-F040	Fails to Close	MOV	Open
HPCS	1E22-F005	Fails to Prevent Backflow	Check	Closed
	1E22-F004	Fails to Remain Closed	MOV	Closed
Standby Liquid Control	1C41-F007	Fails to Prevent Backflow	Check	Closed
	1C41-F006	Fails to Prevent Backflow	Check	Closed
Feedwater	1N27-F559A,B	Fails to Prevent Backflow	Check	Open
	1B21-F032A,B	Fails to Prevent Backflow	Check	Open

Table 3.1.1-5

EPRI Events Included in the Transient Categories

Initiating Event Group	EPRI Transient Category	Rationale For Inclusion
T1 (Transient that causes LOOP)	31	A loss of the offsite grid will result in a reactor scram and loss of normal AC power. The onsite emergency diesel generators are required to start to supply emergency loads. (USAR 15.2.6)
	32	Loss of the auxiliary transformer would result in a reactor scram and a loss of normal AC power. Emergency loads would still be maintained by offsite power, however, it is conservatively assumed that diesel generator start is required to supply emergency loads. (USAR 15.2.6)
T2 (Transient with loss of PCS)	2	Generator load rejection with failure of the turbine bypass valves will result in a fast closure of the turbine control valves which will result in a reactor scram. The failure of the turbine bypass valves to open will result in the loss of the condenser. (USAR 15.2.2)
	4	Turbine trip with failure of the turbine bypass valves. Similar to No. 2 above. Turbine trip causes the reactor scram. A failure of the turbine bypass valves to open will result in the loss of the condenser. (USAR 15.2.3)
	5	An MSIV isolation will result in a reactor scram and loss of the condenser. (USAR 15.2.4)
	6	Inadvertent closure of one MSIV may cause a high steam flow isolation signal resulting in closure of the remaining MSIVs and subsequent loss of the condenser. MSIV Closure will cause a reactor scram. (USAR 15.2.4)
	7	Partial closure of one MSIV. Sequence of events are assumed to be identical to No. 6 above. (USAR 15.2.4)
	8	A loss of normal condenser vacuum will cause both a reactor scram and loss of the condenser. (USAR 15.2.5)

Table 3.1.1-5 continued

EPRI Events Included in the Transient Categories

Initiating Event Group	EPRI Transient Category	Rationale For Inclusion
T2 (Transient with loss of PCS) (Continued)	9	A failure of the pressure regulator in the open position will cause the turbine control valves to go wide open, a reactor scram would occur on high RPV water level and the MSIVs will close on low steamline pressure which will cause isolation of the condenser. (USAR 15.1.3)
	10	A failure of the pressure regulator in the closed position will cause a high neutron flux reactor scram. The turbine control valves will close and the turbine bypass valves will not open causing a loss of the condenser. (USAR 15.2.1)
	12	A failure of the turbine bypass valves in the open position will cause a decrease in the main steamline pressure resulting in a closure of the MSIVs, isolating the condenser and causing a reactor scram. (USAR 15.2.4)
	13	A failure of the turbine bypass or control valves in the closed position will isolate the reactor from the condenser. Turbine stop valve closure will initiate the reactor scram. (USAR 15.2.3)
	37	Transients of unknown cause are assumed to result in the loss of the condenser. (No applicable USAR reference)
T3A (Transient with PCS available)	1	A generator load rejection will cause the fast closure of the turbine control valves which will result in a turbine trip and reactor scram. The condenser will be available. (USAR 15.2.2)
	3	A turbine trip will cause a reactor scram on the fast closure of the turbine control valves. The condenser will be available. (USAR 15.2.3)
	14	A recirculation control failure-increasing flow will result in a high neutron flux reactor scram. The turbine control valves will close on falling turbine pressure. The MSIVs will remain open and the turbine trip will initiate turbine bypass valve operation. (USAR 15.4.5)

Table 3.1.1-5 continued

EPRI Events Included in the Transient Categories

Initiating Event Group	EPRI Transient Category	Rationale For Inclusion
T3A (Transient with PCS available) (Continued)	18	The abnormal startup of an idle recirculation pump is not expected to cause a reactor scram. However, if plant response is exacerbated, the startup could cause a reactor scram on high neutron flux. The MSIVs will remain open and the turbine trip will initiate turbine bypass valve operation. (USAR 15.4.4)
	21	A loss of feedwater heating could result in a high neutron flux reactor scram. The MSIVs will remain open and the turbine trip will initiate turbine bypass valve operation. (USAR 15.1.1)
	27	A rod withdrawal at power is not expected to cause a reactor scram due to the rod withdrawal limiter function (RWL) of the rod control and limiter system (RC&IS). However, if plant response is exacerbated, it is assumed that the event will cause a reactor scram on high neutron flux. The MSIVs will remain open and the turbine bypass valves will be operable. (USAR 15.4.2)
	29	The inadvertent insertion of a control rod or rods with the assumed failure of the RC&IS will cause a turbine trip on low steam flow. The MSIVs will remain open and the turbine bypass valves will be operable. (No applicable USAR reference)
	30	A detected fault in the reactor protection system can result in a reactor scram. The MSIVs will remain open and the turbine bypass valves will remain open. (No applicable USAR reference)
	34	A scram due to a plant occurrence is not assumed to cause an MSIV isolation or a turbine bypass valve failure. (No applicable USAR reference)
	35	A spurious trip via reactor protection system instrumentation is not assumed to cause an MSIV isolation or the failure of the turbine bypass valves. (No applicable USAR reference)
	36	A manual scram with no out of tolerance conditions is assumed to not cause an MSIV isolation or the failure of the turbine bypass valves. (No applicable USAR reference)

Table 3.1.1-5 continued

EPRI Events Included in the Transient Categories

Initiating Event Group	EPRI Transient Category	Rationale For Inclusion
T3B (Transient with a loss of feedwater)	15	A recirculation control failure-decreasing flow will cause a reactor scram on high RPV water level, resulting in a loss of feedwater. The MSIVs will remain open and the turbine trip will initiate turbine bypass valve operation. (USAR 15.3.2)
	16	The trip of one recirculation pump will not cause a reactor scram. However it is conservatively assumed that the plant response will be similar to the loss of two recirculation pumps. See No. 17 below. (USAR 15.3.1)
	17	The trip of both recirculation pumps will result in a reactor scram on high RPV water level, resulting in a loss of feedwater. The MSIVs will remain open and the turbine trip will initiate turbine bypass valve operation. (USAR 15.3.1)
	19	A recirculation pump seizure would cause a reactor scram on high RPV water level, resulting in the loss of feedwater. The MSIVs will remain open and the turbine trip will initiate turbine bypass valve operation. (USAR 15.3.3)
	20	Increasing feedwater flow at power will cause a reactor scram on high RPV water level, resulting in the loss of feedwater. The MSIVs will remain open and the turbine trip will initiate turbine bypass valve operation. (USAR 15.1.2)
	22	A loss of all feedwater flow will cause a reactor scram on low RPV water level. With no feedwater available the HPCS and RCIC systems will initiate to recover level. It is assumed that the plant operators will take the reactor mode switch to shutdown following the scram thus eliminating the possibility of MSIV closure on low steamline pressure. (USAR 15.2.7)
	23	A trip of one feedwater (or condensate) pump may not result in a reactor scram if the feedwater control system can compensate. However, it is assumed that such an event would cause a scram on low RPV water level. The MSIVs will remain open and the turbine bypass valves will remain operable. (No USAR ref.)

Table 3.1.1-5 Last

EPRI Events Included in the Transient Categories

Initiating Event Group	EPRI Transient Category	Rationale For Inclusion
T3B (Transient with loss of feedwater) (Continued)	24	A low flow feedwater transient not compensated by the feedwater control system is expected to cause a reactor scram on low RPV water level. The MSIVs will remain open and the turbine bypass valves will remain operable. (No applicable USAR reference)
	33	The inadvertent startup of HPCS will require the proper compensation by the feedwater control system. Any failure of the control system will potentially result in a reactor scram on either low or high water level. Conservatively, it is assumed to cause a scram on high water level which will result in the loss of feedwater. The MSIVs will remain open and the turbine bypass valves will remain operable. (USAR 15.5.1)
T3C (IORV Transient)	11	An inadvertent opening of a safety/relief valve (stuck) could cause the suppression pool to exceed technical specification limits. A manual scram would be procedurally required. The MSIVs will remain open and the turbine bypass valves will remain operable. (USAR 15.1.4)
N/A	25	A low feedwater flow transient during plant startup or shutdown is not considered within the scope of this study. (See section 3.1.1)
	26	A high feedwater flow transient during plant startup or shutdown is not considered within the scope of this study. (See section 3.1.1)
	28	A high flux due to rod withdrawal at startup is not considered within the scope of this study. (See section 3.1.1)

Table 3.1.2-1

List of Front Line Systems and Support Systems

Frontline Systems

RPV Depressurizaiton

Standby Liquid Control

Residual Heat Removal - Low Pressure Coolant Injection Mode

- Containment Spray Mode

- Suppression Pool Cooling Mode

Low Pressure Core Spray

High Pressure Core Spray

Reactor Core Isolation Cooling

Condensate/Feedwater

Fire Protection Alternate Injection

Condensate Transfer Alternate Injection

ESW B/RHR B Cross-Tie Alternate Injection

Reactor Feed Booster Pump Alternate Injection

Containment Venting by Fuel Pool Cooling and Cleanup

Containment Venting by RHR Containment Spray

Support Systems

Suppression Pool Make-up

Drywell Vacuum Relief

ECCS Pump Room Cooling

Diesel Generator Room Ventilation

Emergency Closed Cooling

Nuclear Closed Cooling

Table 3.1.2-1 continued . . .

List of Front Line Systems and Support Systems

Emergency Service Water

Safety Related Instrument Air

Service/Instrument air

Emergency DC Power

Emergency AC Power

Service Water

Turbine Building Ventilation

Heater Bay Ventilation

Turbine Building Closed Cooling

Table 3.1.2-2

Support System to Frontline System Dependency Matrix

Frontline Systems	Offsite Power	Emergency AC Power 1	Emergency AC Power 2	Emergency AC Power 3	Class 1E DC Power A	Class 1E DC Power B	Class 1E DC Power C	Non-Class 1E DC Power	Emergency Serv Water A	Emergency Serv Water B	Emergency Serv Water C	ECCS Pump Rm Clng E12A	ECCS Pump Rm Clng E12B	ECCS Pump Rm Clng E12C	ECCS Pump Rm Clng E21	ECCS Pump Rm Clng E22	ECCS Pump Rm Clng E51	FHB Vent	Instr Air	Sfty Rel Inst Air	ECC A	ECC B	NCC	CVCW	MCC Batt Vent
ADS, B21	B				C	C														C					
SRV, B21	B				C	C													C						
MSIV, B21	B																		C						
CRDH, C11 A	B	C																C					B		C
B	B		C															C					B		C
RCIS, C11	B							C																	
RRCS, C22 Div 1	B				C																				
Div 2	B					C																			
SLC, C41 A	B	C																							C
B	B		C																						C
RPS, C71																									
LPCI, E12 A	B	C			C				B			C													C
B	B		C			C				B			C												C
C	B		C			C								C											C
LPCS, E21	B	C			C										C										C
HPCS, E22	B			C			C									C									
RCIC, E51	B				C												C								
FPCC, G41 A	B	C							C	C													B		C
B	B			C					C	C													B		C
CVC, M11																								B	C
Cond, N21	B																		B						C
FW, N27	B																		B						C
TBP, N32	B																		B						

Legend: A = Interdependent
B = Complete dependence
C = Partial or delayed dependence

Table 3.1.2-3 Support System to Support System Dependency Matrix

Support Systems	Offsite Power	Emergency AC Power 1	Emergency AC Power 2	Emergency AC Power 3	Class 1E DC Power A	Class 1E DC Power B	Class 1E DC Power C	Non 1E DC Pwr	Emergency Serv Water A	Emergency Serv Water B	Emergency Serv Water C	SW	ESW Pmp Hs Vent	FHB Vent	Instr Air	Sfty Rel Inst Air	ECC A	ECC B	MCC	CVCW	Emerg Pmp Area Clng A	Emerg Pmp Area Clng B	CCCW A	CCCW B	Misc Vent	MCC Batt Vent	DG Vent A	DG Vent B	DG Vent C
Offsite Power																													
Sm AC, R43 Div 1					B				A																	A	A		
Div 2						B				A																A		A	
E22 Div 3							B				A															A			A
1E DC, R42 A	C	C																								A			
B	C		C																							A			
C	C			C																						A			
Non 1E DC, R42	C																									A			
ESW, P45 A	B	A											C													C			
B	B		A										C													C			
C	B			A									C													C			
EPRC, M39 E12A	B	C															B									C			
E12B	B		C															B								C			
E12C	B		C															B								C			
E21	B	C															B									C			
E22	B			C							B															C			
E51	B	C															B									C			
EPHV, M32 A	B	C																								C			
B	B		C																							C			
FHBV, M40	B	C																								C			
IA, P52	B																		B		C	C				C			
SRIA, P57	B																									C			
ECC, P42 A	B	C			C				B												C	C				C			
B	B		C			[illegible]				B											C	C				C			

Legend: A = Interdependent
B = [illegible]ete dependence
C = [illegible]al or delayed dependence

Table 3.1.2-3 Support System to Support [illegible] Dependency Matrix (continued)

Support Systems		Offsite Power	Emergency AC Power 1	2	3	Class 1E DC Power A	B	C	Non 1E DC Pwr	Emergency Serv Water A	B	C	SW	ESW Pmp Hs Vent	FHB Vent	Instr Air	Sfty Inst	Rel Air	ECC A	B	MCC	CVCW	Emerg Pmp Area Clng A	B	CCCW A	B	Misc Vent	MCC Batt Vent	DG Vent A	B	C
MCC, P43		B	C	C		C	C						B			D												C			
CVCW, P50		B							C							B					B							C			
EPAC, M28	A	B	C																						A			C			
	B	B		C																						A		C			
CCCW, P47	A	B	C			C													C		B		A	A				A			
	B	B		C			C													C	B		A	A				A			
	C	B							C										C	C	B		A	A				A			
SW, P41		B		C			C		C																		C	C			
MV, M46		B																										C			
MBV, M23/24		B	C	C																						A		A			
DGV, M43	A	B	A																									C			
	B	B		A																								C			
	C	B			A																							C			

Legend: A = Interdependent
B = Complete dependence
C = Partial or delayed dependence

Table 3.1.2-4

Success Criteria for the Loss of PCS Transient (T2)

Initiator	RPV Reactivity Control	Rx Coolant System Overpressure Protection	Emergency Core Cooling [a]	Containment Overpressure Protection
T2	RPS or ARI	SRVs Open & Close	HPCS or RCIC or MFP or ADS (4 SRVs open automatically or manually) and LPCS or 1 of 3 LPCI or Condensate Transfer Alternate Injection [see Note b]	1 of 2 RHR and HX (in SPC or Spray mode) or Containment Venting

NOTES:
(a) If emergency depressurization is required, then 4 or more open SRVs are necessary for injection.
(b) Condensate Transfer Alternate Injection provides 2,000 gpm @ 75 psig using the Condensate Transfer System (CTS) flush connection to the RHR Shutdown Cooling to FDW line. This alignment requires one local valve to be operated in the Auxiliary Building.

Table 3.1.2-5

Success Criteria for the Transient with PCS Available (T3A)

Initiator	RPV Reactivity Control	Rx Coolant System Overpressure Protection	Emergency Core Cooling [a]	Containment Overpressure Protection
T3A	RPS or ARI	PCS or SRV	HPCS or RCIC or 1 FW or ADS (4 SRV open automatically or manually) and LPCS or 1 of LPCI or Condensate Transfer Alternate Injection [see Note b]	1 of 2 RHR and HX (in SPC or Spray mode) or PCS or Containment Venting

NOTES:

(a) If emergency depressurization is required, then 4 or more open SRVs are necessary for success.

(b) Condensate Transfer Alternate Injection provides 2,000 gpm @ 75 psig using the Condensate Transfer System (CTS) flush connection to the RHR Shutdown cooling to FDW line. This alignment requires one local valve to be operated in the Auxiliary Building.

Table 3.1.2-6

Success Criteria for the Loss of Feedwater Transient (T3B)

Initiator	RPV Reactivity Control	Rx Coolant System Overpressure Protection	Emergency Core Cooling [a]	Containment Overpressure Protection
T3B	RPS or ARI	PCS or SRV	HPCS or RCIC or 1 FW or ADS (4 SRV open automatically or manually) and LPCS or 1 of 3 LPCI or Condensate Transfer Alternate Injection [see Note b]	PCS or 1 of 2 RHR and HX (in SPC or Spray Mode) or Containment Venting

NOTES: (a) Condensate Transfer Alternate Injection provides 2,000 gpm @ 75 psig using the Condensate Transfer System (CTS) flush connection to the RHR Shutdown Cooling to FDW line. This alignment requires one local valve to be operated in the Auxiliary Building.

Table 3.1.2-7

Success Criteria for Large LOCA

Initiator	Reactivity Control	Emergency Core Cooling	Containment Overpressure Protection
A	RPS or ARI	HPCS or LPCS or 1 of 3 LPCI	1 of 2 RHR and Hx (in SPC or Spray mode) or Containment Venting

Table 3.1.2-8

Success Criteria for Intermediate LOCA

Initiator	Reactivity Control	Emergency Core Cooling [a]	Containment Overpressure Protection
S1	RPS or ARI	HPCS or Emergency Depressurization with 2 SRVs and LPCS or 1 of 3 LPCI or Condensate Transfer Alternate Injection [see Note b]	1 of 2 RHR and Hx (in SPC or Spray mode) or Containment Venting

NOTES:

(a) If emergency depressurization is required, then 2 or more SRVs are necessary for success. For the intermediate LOCA, the break size is assumed equivalent to one open SRV.

(b) Condensate Transfer Alternate Injection provides 2,000 gpm @ 75 psig using the Condensate Transfer system (CTS) flush connection to the RHR Shutdown Cooling to FDW line. This alignment requires one local valve to be operated in the Auxiliary Building.

Table 3.1.2-9

Success Criteria for Small LOCA

Initiator	Reactivity Control	Emergency Core Cooling [a]	Containment Overpressure Protection
S2	RPS or ARI	HPCS or RCIC or 1FW or Emergency Depressurization with 3 SRVs and LPCS or 1 of 3 LPCI or Condensate Transfer Alternate Injection [see Note b]	1 of 2 RHR and Hx (in SPC or Spray mode) or Containment Venting

NOTES:
(a) If emergency depressurization is required, then 3 or more SRVs are necessary for success.
(b) Condensate Transfer Alternate Injection provides 2,000 gpm @ 75 psig using the Condensate Transfer system (CTS) flush connection to the RHR Shutdown Cooling to FDW line. This alignment requires one local valve to be operated in the Auxiliary Building.

Table 3.1.3-1

Success Criteria for the Loss of Offsite Power

Initiator	RPV Reactivity Control	Rx Coolant System Overpressure Protection	Emergency Core Cooling [a]	Containment Overpressure Protection
T1	RPS or ARI	SRVs open & close [see Note a]	HPCS or RCIC or Emergency depressurization with 4 or more SRVs and LPCS or 1 of 3 LPCI or Fire Water Cross-tie [see Note b]	1 of 2 RHR and HX (in SPC or Spray Mode) or Containment Venting

NOTES:
(a) If Emergency Depressurization is required then 4 or more open SRVs are necessary for success.
(b) The Fire Water Cross-tie provides 100 gpm @ 120 psig using the ESW cross-tie into the RHR B Shutdown Cooling to Feedwater line. The alignment requires several local valves to be operated.

Table 3.1.3-2

Success Criteria for Station Blackout

Initiator	RPV Reactivity Control	Rx Coolant System Overpressure Protection	Emergency Core Cooling [a]	Containment Overpressure Protection
B	RPS or ARI	SRVs open & close [see Note a]	HPCS or RCIC or Emergency depressurization and LPCS or 1 of 3 LPCI [see Note b] or Fire Water Cross-tie [see Note d]	1 of 2 RHR and HX (in SPC or Spray Mode) or Containment Venting [see note c]

NOTES:

(a) If Emergency Depressurization is required then 4 or more open SRVs are necessary for success.

(b) Available only upon restoration of either offsite or division 1/2 onsite power.

(c) Fuel Pool Cooling & Cleanup Venting may be performed manually; RHR Containment Spray venting is available only upon restoration of either offsite or division 3 to division 2 crosstie.

(d) The Fire Water Cross-tie provides 800 gpm @ 20 psig using the ESW cross-tie into the RHR B Shutdown Cooling to Feedwater line. The alignment requires several local valves to be operated. Fire water can also be crosstied to the HPCS injection line.

Table 3.1.3-3

Success Criteria for IORV Transient (T3C)

Initiator	RPV Reactivity Control	Rx Coolant System Overpressure Protection	Emergency Core Cooling [a]	Containment Overpressure Protection
T3C	RPS or ARI	PCS [see Note b]	HPCS or RCIC or 1FW or Emergency depressurization and LPCS or 1 of 3 LPCI or Condensate Transfer Alternate Injection [see Note c]	1 of 2 RHR and HX (in SPC or Spray Mode) or Containment Venting

NOTES:

(a) If emergency depressurization is required, then 3 or more open SRVs are required for success. Open SRVs, due to the transient are included in the required number of open valves.

(b) for the IORV transient, it is assumed that the PCS is initially available to mitigate the initial pressure transient upon reactor scram. With the PCS available, the SRVs are required to lift to relieve RPV pressure.

(c) Condensate Transfer Alternate Injection provides 2,000 gpm @ 75 psig using the Condensate Transfer System (CTS) flush connection to the RHR Shutdown Cooling to FDW line. This alignment requires one local valve to be operated in the Auxiliary Building.

Table 3.1.3-4

Success Criteria for ATWS

Initiator	RPV Reactivity Control	Rx Coolant System Overpressure Protection	Emergency Core Cooling [a]	Containment Overpressure Protection
T1-C T2-C T3A-C or T3B-C T3C-C T1A-C [see Note a & b]	RPT + 1 of 2 SLC	SRVs open & close [see Note b]	MFP or 1 of 2 TDFPs [see Note d] or 1 of 2 LPCI A/B [see Note f] or LPCS	1 of 2 RHR and HX (in SPC or Spray Mode) PCS [see Note e] or Containment Venting

NOTES:
(a) The ATWS initiators are direct transfers from the Initiating Event Trees.
(b) Transfer from TSW is not shown, since it is not significant. Transfers from A, S1 and S2 (i.e., LOCAs) are not developed, as not significant.
(c) SRVs are required to cycle as necessary in each transient to prevent RPV overpressurization. If Emergency Depressurizaiton is required then conservatively eight open SRVs are necessary for success. The eight SRVs assumed required for depressurization is based on a conservative estimate that at TAF with RPT, power level is approximately 20%. 20% power correlates to the capacity of four open SRVs. An additional four SRVs are assumed required to depressurize. Eight SRVs equals the number of ADS valves.
(d) The Turbine Driven Feed Pumps are potentially available, if offsite AC power and PCS are available and the MSIVs are open. The MFP is available if offsite AC power is available. For this base case ATWS analysis no credit is taken for the Reactor Feedwater Booster Pumps following RPV depressurization.
(e) The main turbine bypass portion of the PCS can accommodate 35% of reactor thermal power.
(f) LPCI flow is directed through the Feedwater Return throttle valve to inject outside the shroud.

Table 3.2.1-1

RPV Depressurization System, B21
Dependency Matrix

		ADS Valves (8 Total)	Non-ADS Valves (11 Total)
Class 1E DC Power	A	*	*
	B	*	*
	C		
Safety Related Inst Air, P57		*	
Inst Air P52			*

* Indicates that the column identifier is dependent on the row identifier

Table 3.2.2-1

Standby Liquid Control System, C41
Dependency Matrix

	Train "A" SLC	Train "B" SLC
Emergency 1	*	
2		*
AC Power 3		

* Indicates that the column identifier is dependent on the row identifier

Table 3.2.3-1

Residual Heat Removal System, E12
Dependency Matrix

	Train A	Train B	Train C
Emergency AC Power 1	*		
Emergency AC Power 2		*	*
Emergency AC Power 3			
Class 1E DC Power A	*		
Class 1E DC Power B		*	*
Class 1E DC Power C			
Emergency Service Water A	*		
Emergency Service Water B		*	*
Emergency Service Water C			
ECCS Pump Room Cooling M39	*	*	*
Emergency Closed Cooling P42 A	*		
Emergency Closed Cooling P42 B		*	*

* Indicates that the column identifier is dependent on the row identifier

Table 3.2.4-1

Low Pressure Core Spray System, E21
Dependency Matrix

	LPCS Injection
Emergency 1 2 AC Power 3	*
Class 1E A B DC Power C	*
Emergency A Closed B Cooling P42	*
ECCS Pump Room Cooling M39	*

* Indicates that the column identifier is dependent on the row identifier

Table 3.2.5-1

High Pressure Core Spray System, E22
Dependency Matrix

	High Pressure Core Spray Operation
Emergency 1 2 AC Power 3	*
Class 1E A B DC Power C	*
Emergency A Service B Water P45 C	*
ECCS Pump Room Cooling M39	*

* Indicates that the column identifier is dependent on the row identifier

Table 3.2.6-1

Reactor Core Isolation Cooling System, E51
Dependency Matrix

		RCIC System Operation
Class 1E	A	*
	B	*
DC Power	C	

* Indicates that the column identifier is dependent on the row identifier

Table 3.2.7-1

Emergency Closed Cooling System, P42
Dependency Matrix

	Unit 1 TRAIN A 1 ECC	Unit 1 TRAIN B 2 ECC
Emergency 1	*	
2		*
Ac Power 3		
Class 1E A	*	
B		*
DC Power C		
Emergency A	*	
Service B		*
P45		

* Indicates that the column identifier is dependent on the row indentifier

Table 3.2.8-1

Emergency Service Water System, P45
Dependency Matrix

No Dependencies Modeled

Table 3.2.9-1

Service/Instrument Air System, P51/52
Dependency Matrix

	P51/52
Offsite Power	*
Nulcear Closed Cooling P43	*

*Indicates that the column identifier is dependent on the row identifier.

Table 3.2.10-1

Safety Related Instrument Air System, P57
Dependency Matrix

	P57
Emergency 1	*
2	*
AC Power 3	
Offsite Power	*

* Indicates that the column identifier is dependent on the row identifier

Table 3.2.11-1

Fire Protection System, P54
Dependency Matrix

	Diesel Driven Fire Pump Injection to RPV
Residual Heat Removal E12 Note: (Various E12 Manual Valves Are Only Required)	*
ESW P45 Note: (Various P45 Manual Valves Are Only Required)	*

* Indicates that the column identifier is dependent on the row identifier.

Table 3.2.12-1

D.C. Electrical System, R42
Dependency Matrix

	Division I 125 VDC	Division II 125 VDC	Division III 125 VDC (HICS)
Emergency 1	*		
2		*	
AC Power 3			*

* Indicates that the column identifier is dependent on the row identifier

Table 3.2.13-1

AC Power
Standby Diesel Generator System, R43
High Pressure Core Spray Diesel Generator System, E22B
Dependency Matrix

		Division 1 Diesel Generator	Division 2 Diesel Generator	Division 3 HPCS Diesel
Class 1E	A	*		
	B		*	
DC Power	C			*
Emergency	A	*		
Service	B		*	
Water P45	C			*

* Indicates that the column identifier is dependent on the row identifier

Table 3.2.14-1

Suppression Pool Make Up System, G43
Dependency Matrix

	Train A	Train B
Emergency 1	*	
2		*
AC Power 3		

* Indicates that the column identifier is dependent on the row identifier

Table 3.2.14-2

Drywell Vacuum Relief System, M16
Dependency Matrix

	Motor Operated Valves	Swing Check Valves
Emergency 1	*	
2	*	
AC Power 3		

* Indicates that the column identifier is dependent on the row identifier

Table 3.2.15-1

Diesel Generator Building Ventilation System, M43
Dependency Matrix

		DGBV Div 1	DGBV Div 2	DGBV Div 3
Emergency	1	*		
	2		*	
AC Power	3			*

* Indicates that the column identifier is dependent on the row identifier

Table 3.2.16-1

Alternate Injection System
Dependency Matrix

	Condensate Transfer Alternate Injection
Offsite Power	*
Emergency 1 2 AC Power 3	* (This is an RHR system * dependency)
Inst Air P52	*

*Indicates that the column identifier is dependent on the row identifier

Table 3.2.17-1

Containment Venting System, D23
Dependency Matrix

		Fuel Pool Cooling and Cleanup Venting	RHR Containment Spray Venting
Emergency	1	*	*
	2	*	*
AC Power	3		

* Indicates that the column identifier is dependent on the row identifier

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
A	1.00E-004	LARGE LOSS OF COOLANT ACCIDENT (LARGE LOCA)
ADADUMA	2.04E-003	DIVISION 1 ADS UNAVAILABLE DUE TO TEST & MAINTENANCE
ADADUMB	2.04E-003	DIVISION 2 ADS UNAVAILABLE DUE TO TEST & MAINTENANCE
ADCLDECHNLALOGIC	1.60E-003	DIVISION 1 ACTUATION LOGIC FAILS
ADCLDECHNLBLOGIC	1.60E-003	DIVISION 2 ACTUATION LOGIC FAILS
ADCVCCADSNC	1.00E-005	1B21-F0039 VALVES COMMON CAUSE FAIL TO CLOSE
ADCVCCNONADSNC	1.00E-005	1B21-F036 VALVES COMMON CAUSE FAIL TO CLOSE
ADCVN01B21F0032A	1.00E-004	1B21-F0032A CHECK VALVE NC - FAILS TO OPEN
ADCVN01B21F0032B	1.00E-004	1B21-F0032B CHECK VALVE NC - FAILS TO OPEN
ADHICPC5-1-ADS-A	3.79E-003	OPERATOR FAILS TO INHIBIT ADS ATWS W/ FEEDWATER
ADHICPC5-1-ADS-I	3.60E+000	OPERATOR FAILS TO INHIBIT ADS ATWS W/O FDW & IORV
ADHICPC5-1-ADS-L	3.60E-002	OPERATOR FAILS TO INHIBIT ADS ATWS W/ LOOP
ADHICPC5-1-ADS-O	7.20E+000	OPERATOR FAILS TO INHIBIT ADS ATWS W/O FEEDWATER
ADHICPEC2-ADS-LR	9.99E-005	FAILS TO RECOVER FROM RPV DEPRESS CD FAILURE
ADHICPEC2-ADS-R	1.00E-001	FAILS TO RECOVER FROM RPV DEPRESS CD FAILURE
ADHICPEC2-ADS-T	1.00E-003	FAILS TO EMERGENCY RPV DEPRESS TRANSIENT
ADHICPEC5-ADS-FL	7.00E-003	FAILS TO EMERGENCY RPV DEPRESS - ATWS W/FDW & LVL CNTRL
ADHICPEC5-ADS-FX	1.40E-002	FAILS TO EMERGENCY RPV DEPRESS - ATWS W/FDW & NO LVL CNT
ADSRCCADS	8.00E-006	ADS VALVES COMMON CAUSE FAILURE
ADSRCCNONADS	1.00E+000	NON-ADS VALVES COMMON CAUSE FAILURE
ADSRN01B21F0041A	1.00E-002	1B21-F0041A SAFETY RELIEF VALVE NC - FAILS TO OPEN
ADSRN01B21F0041B	1.00E-002	1B21-F0041B SAFETY RELIEF VALVE NC - FAILS TO OPEN
ADSRN01B21F0041E	1.00E-002	1B21-F0041E SAFETY RELIEF VALVE NC - FAILS TO OPEN
ADSRN01B21F0041F	1.00E-002	1B21-F0041F SAFETY RELIEF VALVE NC - FAILS TO OPEN
ADSRN01B21F0047D	1.00E-002	1B21-F0047D SAFETY RELIEF VALVE NC - FAILS TO OPEN
ADSRN01B21F0047H	1.00E-002	1B21-F0047H SAFETY RELIEF VALVE NC - FAILS TO OPEN
ADSRN01B21F0051C	1.00E-002	1B21-F0051C SAFETY RELIEF VALVE NC - FAILS TO OPEN
ADSRN01B21F0051G	1.00E-002	1B21-F0051G SAFETY RELIEF VALVE NC - FAILS TO OPEN
ARI	1.00E-002	ALTERNATE ROD INSERTION FAILS
ASABLFOP61B0001A	5.00E-002	0P61-B0001A AUXILIARY BOILER FAILS TO FUNCTION
ASABLFOP61B0001B	5.00E-002	0P61-B0001B AUXILIARY BOILER FAILS TO FUNCTION
C	1.00E-005	REACTOR SCRAM CONTROL RODS IN
C108B	0.00E+000	COMPLEMENT TO C108
CAHICPI008-4:2	1.00E-003	OPERATORS FAIL TO SHIFT CONDENSER AIR REMOVAL TO VP & AB
CAVPFR1N62C0001A	7.20E-004	1N62-C0001A VACUUM PUMP FAILS TO RUN
CAVPFR1N62C0001B	7.20E-004	1N62-C0001B VACUUM PUMP FAILS TO RUN
CAVPFS1N62C0001A	2.93E-003	1N62-C0001A VACUUM PUMP FAILS TO START
CAVPFS1N62C0001B	2.93E-003	1N62-C0001B VACUUM PUMP FAILS TO START
CBMVFC1M17F0015	2.40E-006	1M17-F0015 MOTOR VALVE NO - FAILS CLOSED
CBMVFC1M17F0025	2.40E-006	1M17-F0025 MOTOR VALVE NO - FAILS CLOSED
CBMVFC1M17F0035	2.40E-006	1M17-F0035 MOTOR VALVE NO - FAILS CLOSED
CBMVFC1M17F0045	2.40E-006	1M17-F0045 MOTOR VALVE NO - FAILS CLOSED
CBVBCC	3.44E-004	M17 VACUUM BREAKERS COMMON CAUSE FAILURES
CBVBF01M17F0010	3.44E-003	1M17-F0010 VACUUM BREAKER NC - FAILS TO OPEN
CBVBF01M17F0020	3.44E-003	1M17-F0020 VACUUM BREAKER NC - FAILS TO OPEN
CBVBF01M17F0030	3.44E-003	1M17-F0030 VACUUM BREAKER NC - FAILS TO OPEN
CBVBF01M17F0040	3.44E-003	1M17-F0040 VACUUM BREAKER NC - FAILS TO OPEN
CCCCLFA	0.00E+000	CONTROL COMPLEX CHILLED WATER TRAIN A TRIPS
CCCCUMA	0.00E+000	CCCW TRAIN A UNAVAILABLE DUE TO MAINTENANCE
CCCCUMB	0.00E+000	CCCW TRAIN B UNAVAILABLE DUE TO MAINTENANCE

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
CCCCUMC	0.00E+000	CCCW TRAIN C UNAVAILABLE DUE TO MAINTENANCE
CCCHCC	0.00E+000	CHILLERS COMMON CAUSE FAILURE
CCCHFROP47B0001A	0.00E+000	OP47-B0001A CHILLER FAILS TO RUN
CCCHFROP47B0001B	0.00E+000	OP47-B0001B CHILLER FAILS TO RUN
CCCHFROP47B0001C	0.00E+000	OP47-B0001C CHILLER FAILS TO RUN
CCCHFSOP47B0001A	0.00E+000	OP47-B0001A CHILLER FAILS TO START
CCCHFSOP47B0001B	0.00E+000	OP47-B0001B CHILLER FAILS TO START
CCCHFSOP47B0001C	0.00E+000	OP47-B0001C CHILLER FAILS TO START
CCHICP	0.00E+000	OPERATOR FAILS TO BYPASS LOCA SIGNAL IN 4.5 HOURS
CCHICPSP47-5:4	0.00E+000	OPERATOR FAILS TO REALIGN CCCW LOOP C IN 4.5 HOURS
CCMPCC	0.00E+000	CCCW MOTOR PUMP COMMON CAUSE FAILURES
CCMPFROP47C0001A	3.00E-011	OP47-C0001A MOTOR PUMP FAILS TO RUN
CCMPFROP47C0001B	3.00E-011	OP47-C0001B MOTOR PUMP FAILS TO RUN
CCMPFROP47C0001C	3.00E-011	OP47-C0001C MOTOR PUMP FAILS TO RUN
CCMPFSOP47C0001A	2.93E-003	OP47-C0001A MOTOR PUMP FAILS TO START
CCMPFSOP47C0001B	2.93E-003	OP47-C0001B MOTOR PUMP FAILS TO START
CCMPFSOP47C0001C	2.93E-003	OP47-C0001C MOTOR PUMP FAILS TO START
CCMVNCOP47F0550	2.93E-003	OP47-F0550 MOTOR VALVE FAILS TO CLOSE - NO
CCMVNCOP47F0551	2.93E-003	OP47-F0551 MOTOR VALVE FAILS TO CLOSE - NO
CCTRNARECOVERED4	0.00E+000	CCCW TRAIN A RECOVERED IN 4 HOURS
CDHICPPS2:1-XH11	1.00E-001	OPERATOR FAILS TO BYPASS THE RHR LOCA SIGNAL - XH11
CDHICPPS2:1-XH12	1.00E-001	OPERATOR FAILS TO BYPASS THE RHR LOCA SIGNAL - XH12
CDHICPPS2:1-XH1X	1.00E-003	OPERATOR FAILS TO BYPASS THE RHR LOCA SIGNAL - XH1X
CE	2.00E-005	ELECTRICAL FAILURE ROD INSERTION SIGNAL
CF	1.00E+000	CONTAINMENT FAILS DUE TO EXTERNAL PRESSURE
CICLDEISOLOGIC	1.60E-003	FLOOD PROTECTION ACTUATION LOGIC
CICLLF1N71F0020A	1.25E-004	1N71-F0020A CONTROL LOGIC FAILS TO FUNCTION
CICLLF1N71F0020B	1.25E-004	1N71-F0020B CONTROL LOGIC FAILS TO FUNCTION
CICLLF1N71F0020C	1.25E-004	1N71-F0020C CONTROL LOGIC FAILS TO FUNCTION
CICLLF1N71F0030C	1.25E-004	1N71-F0030C CONTROL LOGIC FAILS TO FUNCTION
CICLLF1N71F0030D	1.25E-004	1N71-F0030D CONTROL LOGIC FAILS TO FUNCTION
CICLLF1N71F0140A	1.25E-004	1N71-F0140A CONTROL LOGIC FAILS TO FUNCTION
CICLLF1N71F0140B	1.25E-004	1N71-F0140B CONTROL LOGIC FAILS TO FUNCTION
CICLLF1N71F0140C	1.25E-004	1N71-F0140C CONTROL LOGIC FAILS TO FUNCTION
CICLLF1N71F0140D	1.25E-004	1N71-F0140D CONTROL LOGIC FAILS TO FUNCTION
CIMVLF1N71F0030A	5.00E-001	1N71-F0030A VALVE FAILS DUE TO PIPE BREAK
CIMVLF1N71F0030B	2.50E-001	1N71-F0030B VALVE FAILS DUE TO PIPE BREAK
CIMVNC1N71F0020A	2.93E-003	1N71-F0020A MOTOR VALVE FAILS TO CLOSE - NO
CIMVNC1N71F0020B	2.93E-003	1N71-F0020B MOTOR VALVE FAILS TO CLOSE - NO
CIMVNC1N71F0020C	2.93E-003	1N71-F0020C MOTOR VALVE FAILS TO CLOSE - NO
CIMVNC1N71F0030C	2.93E-003	1N71-F0030C MOTOR VALVE FAILS TO CLOSE - NO
CIMVNC1N71F0030D	2.93E-003	1N71-F0030D MOTOR VALVE FAILS TO CLOSE - NO
CIMVNC1N71F0140A	2.93E-003	1N71-F0140A MOTOR VALVE FAILS TO CLOSE - NO
CIMVNC1N71F0140B	2.93E-003	1N71-F0140B MOTOR VALVE FAILS TO CLOSE - NO
CIMVNC1N71F0140C	2.93E-003	1N71-F0140C MOTOR VALVE FAILS TO CLOSE - NO
CIMVNC1N71F0140D	2.93E-003	1N71-F0140D MOTOR VALVE FAILS TO CLOSE - NO
CM	1.00E-005	MECHANICAL FAILURE CONTROL RODS
CNAVFO1N21F0220	2.40E-006	1N21-F0220 AIR VALVE NC - FAILS OPEN
CNAVFO1N27F0305	2.40E-006	1N27-F0305 PNEUMATIC VALVE NC - FAILS OPEN
CNAVNC1N21F0230	2.00E-003	1N21-F0230 PNEUMATIC VALVE NO - FAILS TO CLOSE

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
CNMVNO1N27F0200	2.93E-003	1N27-F0200 MOTOR VALVE NC - FAILS TO OPEN
CNXVNO1N27F0508	1.00E-004	1N27-F0508 MANUAL VALVE NC - FAILS TO OPEN
CNXVNO1P81F0553	1.00E-004	1P81-F0553 MANUAL VALVE NC - FAILS TO OPEN
CSCLDEA	1.60E-003	TRN A CNTNMNT SPRAY ACTUATION LOGIC HARDWARE FAILURE
CSCLDEB	1.60E-003	TRN B CNTNMNT SPRAY ACTUATION LOGIC HARDWARE FAILURE
CSHICPET-2:P-1	1.70E-002	OPERATOR FAILS TO INITIATE RHR CONTAINMENT SPRAY
CSHICPET-2:P-1-A	1.70E-002	OPERATOR FAILS TO INITIATE TRN A RHR CONTAINMENT SPRAY
CSHICPET-2:P-1-B	1.70E-002	OPERATOR FAILS TO INITIATE TRN B RHR CONTAINMENT SPRAY
CSMVCC	9.25E-005	CNTNMNT SPRAY VLVS COMMON CAUSE FAILURE
CSMVNC1E12F0042A	2.93E-003	1E12-F0042A MOTOR VALVE NO - FAILS TO CLOSE
CSMVNC1E12F0042B	2.93E-003	1E12-F0042B MOTOR VALVE NO - FAILS TO CLOSE
CSMVNO1E12F0028A	2.93E-003	1E12-F0028A MOTOR VALVE NC - FAILS TO OPEN
CSMVNO1E12F0028B	2.93E-003	1E12-F0028B MOTOR VALVE NC - FAILS TO OPEN
CSMVNO1E12F0537A	2.93E-003	1E12-F0537A MOTOR VALVE NC - FAILS TO OPEN
CSMVNO1E12F0537B	2.93E-003	1E12-F0537B MOTOR VALVE NC - FAILS TO OPEN
CSPXCCCNT	9.59E-006	CONTAINMENT PRESS COMMON CAUSE MISCALIBRATION
CSPXDE1E12N0662A	9.59E-005	1E12-N0662A CNT PRS PRESSURE INSTRUMENT FAILS TO FUNCTION
CSPXDE1E12N0662B	9.59E-005	1E12-N0662B CNT PRS PRESSURE INSTRUMENT FAILS TO FUNCTION
CSPXDE1E12N0662C	9.59E-005	1E12-N0662C CNT PRS PRESSURE INSTRUMENT FAILS TO FUNCTION
CSPXDE1E12N0662D	9.59E-005	1E12-N0662D CNT PRS PRESSURE INSTRUMENT FAILS TO FUNCTION
CSTILF1E12K93A	2.93E-004	1E12-K93A TIME DELAY RELAY FAILS TO FUNCTION
CSTILF1E12K93B	2.93E-004	1E12-K93B TIME DELAY RELAY FAILS TO FUNCTION
CTAVNO1E12F0300A	2.00E-003	1E12-F0300A AIR VALVE NC - FAILS TO OPEN
CTAVNO1E12F0300B	2.00E-003	1E12-F0300B AIR VALVE NC - FAILS TO OPEN
CTCLLF1P11C0001A	1.25E-004	1P11-C0001A CONTROL LOGIC FAILS TO FUNCTION
CTCLLF1P11C0001B	1.25E-004	1P11-C0001B CONTROL LOGIC FAILS TO FUNCTION
CTCTUM1P11C0001A	5.00E-002	CTS PUMP 1P11-C001A UNAVAILABLE DUE TO MAINTENANCE
CTCTUM1P11C0001B	5.00E-002	CTS PUMP 1P11-C001B UNAVAILABLE DUE TO MAINTENANCE
CTCVNO1E12F0063A	1.00E-004	1E12-F0063A CHECK VALVE NC - FAILS TO OPEN
CTCVNO1E12F0063B	1.00E-004	1E12-F0063B CHECK VALVE NC - FAILS TO OPEN
CTHICPPS4:4-ALT	1.00E-001	OPERATOR FAILS TO ALIGN CONDENSATE XFER ALT INJECTION
CTHICPPS4:4-ALT6	1.00E-003	OPERATOR FAILS TO ALIGN CONDENSATE TRANSFER IN 6 HOURS
CTHICPPS4:4-ALTC	3.00E-001	OPERATOR FAILS TO ALIGN CONDENSATE XFER ALT INJECTION
CTMPCC	2.93E-004	CTS PUMP COMMON CAUSE FAILURE
CTMPFR1P11C0001A	7.20E-004	1P11-C0001A MOTOR PUMP FAILS TO RUN
CTMPFR1P11C0001B	7.20E-004	1P11-C0001B MOTOR PUMP FAILS TO RUN
CTMPFS1P11C0001A	2.93E-003	1P11-C0001A MOTOR PUMP FAILS TO START
CTMPFS1P11C0001B	2.93E-003	1P11-C0001B MOTOR PUMP FAILS TO START
CTSVLC1E12F0301A	2.40E-006	1E12-F0301A SOLENOID VALVE FAILS TO FUNCTION
CTSVLC1E12F0301B	2.40E-006	1E12-F0301B SOLENOID VALVE FAILS TO FUNCTION
CTSVLF1E12F0301A	2.00E-003	1E12-F0301A SOLENOID VALVE FAILS TO FUNCTION
CTSVLF1E12F0301B	2.00E-003	1E12-F0301B SOLENOID VALVE FAILS TO FUNCTION
CV01	4.30E-001	CORE VULNERABLE HPCS OPERATING
CV02	2.60E-001	CORE VULNERABLE LOW PRESS INJ OPER
CV03	4.00E-002	CORE VULNERABLE FDW OPERATING
CV04	0.00E+000	CORE VULNERABLE INJ OUTSITE AB
CV05	1.40E-001	CORE VULNERABLE ANCHORAGE FAILURE
CVCLLF1E12S64A	1.25E-004	CONTAINMENT SPRAY A CONTROL LOGIC SEAL-IN RESET FAILS
CVCLLF1E12S64B	1.25E-004	CONTAINMENT SPRAY B CONTROL LOGIC SEAL-IN RESET FAILS
CVCLLF1G41F0140	1.25E-004	1G41-F0140 CONTROL LOGIC FAILS

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
CVCLLF1G41F0145	1.25E-004	1G41-F0145 CONTROL LOGIC FAILS
CVHICPEPC-COM	1.00E-003	FAILS TO INITIATE CNTNMNT PRESS CNTRL RHR AND VENT
CVHICPEPC-FPCC	1.00E-001	FAILS TO INITIATE CNTNMNT PRESS CNTRL VENTING
CVHICPEPC-RHR	1.00E-002	FAILS TO INITIATE CNTNMNT PRESS CNTRL RHR
CVHICPEPC-RHR-E	1.00E-001	FAILS TO INITIATE CNTNMNT PRESS CNTRL RHR - S.P. CLNG
CVHICPPS7-RHR	1.00E-005	OPERATOR FAILS TO INITIATE RHR CONTAINMENT VENTING
CVHICPPS7:1 P T	1.00E-010	OPERATOR FAILS TO PREPARE FOR RHR CNTNMNT VENT - TRAN
CVHICPPS7:[illegible]-T	9.99E-005	OPERATOR FAILS TO ALIGN FPCC FOR CNTNMNT VENT - TRAN
CVHICPPS7:4E12-T	9.99E-005	OPERATOR FAILS TO ALIGN RHR FOR CNTNMNT VENT - TRAN
CVHICRPS7:3G41-T	5.00E-002	OPERATOR IS UNABLE TO LOCALLY OPEN 1G41-F0145
CVMV1G41F0140FC	3.13E-001	FRACTION OF TIME IN SBO THAT DG STARTED BEFORE FAILING
CVMV1G41F0140FCB	6.87E-001	FRACTION OF TIME IN SBO THAT DG DID NOT START
CVMVNC1E12F0003A	2.93E-003	1E12-F0003A MOTOR VALVE NO - FAILS TO CLOSE
CVMVNC1E12F0003B	2.93E-003	1E12-F0003B MOTOR VALVE NO - FAILS TO CLOSE
CVMVNO1G41F0140	2.93E-003	1G41-F0140 MOTOR VALVE FAILS TO OPEN
CVMVNO1G41F0145	2.93E-003	1G41-F0145 MOTOR VALVE FAILS TO OPEN
DBCLLF1M43C0001A	1.25E-004	1M43-C0001A CONTROL LOGIC FAILS TO FUNCTION
DBCLLF1M43C0001B	1.25E-004	1M43-C0001B CONTROL LOGIC FAILS TO FUNCTION
DBCLLF1M43C0001C	1.25E-004	1M43-C0001C CONTROL LOGIC FAILS TO FUNCTION
DBCLLF1M43C0002B	1.25E-004	1M43-C0002B CONTROL LOGIC FAILS TO FUNCTION
DBCLLF1M43C0002C	1.25E-004	1M43-C0002C CONTROL LOGIC FAILS TO FUNCTION
DBCLLFDIV1TRAIN1	1.25E-004	DG BLDG VENT DIV 1 CONTROL LOGIC TRN 1 FAILS TO FUNC
DBCLLFDIV1TRAIN2	1.25E-004	DG BLDG VENT DIV 1 CONTROL LOGIC TRN 2 FAILS TO FUNC
DBCLLFDIV2TRAIN1	1.25E-004	DG BLDG VENT DIV 2 CONTROL LOGIC TRN 1 FAILS TO FUNC
DBCLLFDIV2TRAIN2	1.25E-004	DG BLDG VENT DIV 2 CONTROL LOGIC TRN 2 FAILS TO FUNC
DBCLLFDIV3TRAIN1	1.25E-004	DG BLDG VENT DIV 3 CONTROL LOGIC TRN 1 FAILS TO FUNC
DBCLLFDIV3TRAIN2	1.25E-004	DG BLDG VENT DIV 3 CONTROL LOGIC TRN 2 FAILS TO FUNC
DBLVCC	2.93E-005	DG BLDG VENT LOUVER COMMON CAUSE FAILURE
DBLVFC1M43F0071A	2.40E-006	1M43-F0071A LOUVER NO - FAILS CLOSED
DBLVFC1M43F0071B	2.40E-006	1M43-F0071B LOUVER NO - FAILS CLOSED
DBLVNO1M43F0070A	2.93E-003	1M43-F0070A LOUVER NC - FAILS TO OPEN
DBLVNO1M43F0070B	2.93E-003	1M43-F0070B LOUVER NC - FAILS TO OPEN
DBLVNO1M43F0070C	2.93E-003	1M43-F0070C LOUVER NC - FAILS TO OPEN
DBLVNO1M43F0071C	2.93E-003	1M43-F0071C LOUVER NO - FAILS CLOSED
DBLVNO1M43F0080A	2.93E-003	1M43-F0080A LOUVER NC - FAILS TO OPEN
DBLVNO1M43F0080B	2.93E-003	1M43-F0080B LOUVER NC - FAILS TO OPEN
DBLVNO1M43F0080C	2.93E-003	1M43-F0080C LOUVER NC - FAILS TO OPEN
DBLVNO1M43F0081A	2.93E-003	1M43-F0081A LOUVER NC - FAILS TO OPEN
DBLVNO1M43F0081B	2.93E-003	1M43-F0081B LOUVER NC - FAILS TO OPEN
DBLVNO1M43F0081C	2.93E-003	1M43-F0081C LOUVER NC - FAILS TO OPEN
DBMDCC	2.93E-005	DG BLDG VENT COMMON CAUSE MOTOR DAMPER FAILS
DBMDFC1M43F0020A	2.40E-006	1M43-F0020A MOTOR DAMPER FAILS CLOSED
DBMDFC1M43F0020B	2.40E-006	1M43-F0020B MOTOR DAMPER FAILS CLOSED
DBMDFC1M43F0020C	2.40E-006	1M43-F0020C MOTOR DAMPER FAILS CLOSED
DBMDFC1M43F0030A	2.40E-006	1M43-F0030A MOTOR DAMPER FAILS CLOSED
DBMDFC1M43F0030B	2.40E-006	1M43-F0030B MOTOR DAMPER FAILS CLOSED
DBMDFC1M43F0030C	2.40E-006	1M43-F0030C MOTOR DAMPER FAILS CLOSED
DBMDFC1M43F0031A	2.40E-006	1M43-F0031A MOTOR DAMPER FAILS CLOSED
DBMDFC1M43F0031B	2.40E-006	1M43-F0031B MOTOR DAMPER FAILS CLOSED
DBMDFC1M43F0031C	2.40E-006	1M43-F0031C MOTOR DAMPER FAILS CLOS

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
DBMDFC1M43F0220A	2.40E-006	1M43-F0220A MOTOR DAMPER FAILS CLOSED
DBMDFC1M43F0220B	2.40E-006	1M43-F0220B MOTOR DAMPER FAILS CLOSED
DBMDFC1M43F0220C	2.40E-006	1M43-F0220C MOTOR DAMPER FAILS CLOSED
DBMDFC1M43F0230A	2.40E-006	1M43-F0230A MOTOR DAMPER FAILS CLOSED
DBMDFC1M43F0230B	2.40E-006	1M43-F0230B MOTOR DAMPER FAILS CLOSED
DBMDFC1M43F0230C	2.40E-006	1M43-F0230C MOTOR DAMPER FAILS CLOSED
DBMDFC1M43F0231A	2.40E-006	1M43-F0231A MOTOR DAMPER FAILS CLOSED
DBMDFC1M43F0231B	2.40E-006	1M43-F0231B MOTOR DAMPER FAILS CLOSED
DBMDFC1M43F0231C	2.40E-006	1M43-F0231C MOTOR DAMPER FAILS CLOSED
DBMFCC	2.93E-005	DG BLDG VENT MOTOR FAN COMMON MODE FAILURE
DBMFFR1M43C0001A	1.20E-004	1M43-C0001A MOTOR FAN FAILS TO RUN
DBMFFR1M43C0001B	1.20E-004	1M43-C0001B MOTOR FAN FAILS TO RUN
DBMFFR1M43C0001C	1.20E-004	1M43-C0001C MOTOR FAN FAILS TO RUN
DBMFFR1M43C0002A	1.20E-004	1M43-C0002A MOTOR FAN FAILS TO RUN
DBMFFR1M43C0002B	1.20E-004	1M43-C0002B MOTOR FAN FAILS TO RUN
DBMFFR1M43C0002C	1.20E-004	1M43-C0002C MOTOR FAN FAILS TO RUN
DBMFFS1M43C0001A	2.93E-003	1M43-C0001A MOTOR FAN FAILS TO START
DBMFFS1M43C0001B	2.93E-003	1M43-C0001B MOTOR FAN FAILS TO START
DBMFFS1M43C0001C	2.93E-003	1M43-C0001C MOTOR FAN FAILS TO START
DBMFFS1M43C0002A	2.93E-003	1M43-C0002A MOTOR FAN FAILS TO START
DBMFFS1M43C0002B	2.93E-003	1M43-C0002B MOTOR FAN FAILS TO START
DBMFFS1M43C0002C	2.93E-003	1M43-C0002C MOTOR FAN FAILS TO START
DBMFUMDIV1	1.36E-003	1M43-C001A & C002A UNAVAILABLE DUE TO MAINTENANCE
DBMFUMDIV2	1.36E-003	1M43-C001B & C002B UNAVAILABLE DUE TO MAINTENANCE
DBMFUMDIV3	1.36E-003	1M43-C001C & C002C UNAVAILABLE DUE TO MAINTENANCE
DCBCLC1E22S0006	2.40E-005	EFD1C BATTERY CHARGER HARDWARE FAILURE
DCBCLC1R42S0006	2.40E-005	EFD1A BATTERY CHARGER HARDWARE FAILURE
DCBCLC1R42S0007	2.40E-005	ED12A RES BATTERY CHARGER HARDWARE FAILURE
DCBCLC1R42S0008	2.40E-005	EFD1B BATTERY CHARGER HARDWARE FAILURE
DCBCLC1R42S0009	2.40E-005	EDF12B RES BATTERY CHARGER HARDWARE FAILURE
DCBCLC1R42S0011	2.40E-005	EDF12C RES BATTERY CHARGER HARDWARE FAILURE
DCBCLC2R42S0006	2.40E-005	EDF2A BATTERY CHARGER HARDWARE FAILURE
DCBCLC2R42S00[illegible]	2.40E-005	EFD2B BATTERY CHARGER HARDWARE FAILURE
DCBDLC1R42S0024	3.12E-006	ED-1-A 125 V DC BUS HARDWARE FAILURE
DCBDLC1R42S0025	3.12E-006	ED-1-B 125 V DC BUS HARDWARE FAILURE
DCBDLC1R42S0037	3.12E-006	ED-1-C 125 V DC BUS HARDWARE FAILURE
DCBDLC2R42S0024	3.12E-006	ED-2-A 125 V DC BUS HARDWARE FAILURE
DCBDLC2R42S0025	3.12E-006	ED-2-B 125 V DC BUS HARDWARE FAILURE
DCBTCC	1.37E-005	BATTERY COMMON CAUSE FAILURE
DCBTLC1E22S0005	1.37E-003	FAILURE OF DIV 3 BATTERY 1E22-S0005
DCBTLC1R42S0002	1.37E-003	FAILURE OF DIV 1 BATTERY 1R42-S0002
DCBTLC1R42S0003	1.37E-003	FAILURE OF DIV 2 BATTERY 1R42-S0003
DCBTLC2E22S0005	1.37E-003	FAILURE OF UNIT 2 DIV 3 BATTERY 1E22-S0005
DCBTLC2R42S0002	1.37E-003	FAILURE OF UNIT 2 DIV 1 BATTERY 2R42-S0002
DCBTLC2R42S0003	1.37E-003	FAILURE OF UNIT 2 DIV 2 BATTERY 2R42-S0003
DCBTUM1E22S0005	4.38E-004	BATTERY 1E22-S0005 UNAVAILABLE DUE TO MAINTENANCE
DCBTUM1R42S0002	4.38E-004	BATTERY 1R42-S0002 UNAVAILABLE DUE TO MAINTENANCE
DCBTUM1R42S0003	4.38E-004	BATTERY 1R42-S0003 UNAVAILABLE DUE TO MAINTENANCE
DCBTUM2R42S0002	4.38E-004	BATTERY 2R42-S0002 UNAVAILABLE DUE TO MAINTENANCE
DCBTUM2R42S0003	4.38E-004	BATTERY 2R42-S0003 UNAVAILABLE DUE TO MAINTENANCE

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
DGBALC1R22S0006	9.60E-005	EH12 4160 V AC BUS HARDWARE FAILURE
DGBALC1R22S0007	9.60E-005	EH11 4160 V AC BUS HARDWARE FAILURE
DGBALC1R23S0009	9.60E-005	EF-1-A 480 V AC BUS HARDWARE FAILURE
DGBALC1R23S0010	9.60E-005	EF-1-B 480 V AC BUS HARDWARE FAILURE
DGBALC1R23S0011	9.60E-005	EF-1-C 480 V AC BUS HARDWARE FAILURE
DGBALC1R23S0012	9.60E-005	EF-1-D 480 V AC BUS HARDWARE FAILURE
DGBALC1R24S0029	9.60E-005	EF1E MCC HARDWARE FAILURE
DGBALC2R23S0010	9.60E-005	EF-2-B 480 V AC BUS HARDWARE FAILURE
DGBALC2R23S0012	9.60E-005	EF-2-D 480 V AC BUS HARDWARE FAILURE
DGCLDE1R43S0001A	1.60E-003	DIV 1 D/G ACTUATION LOGIC FAILURE
DGCLDE1R43S0001B	1.60E-003	DIV 2 D/G ACTUATION LOGIC FAILURE
DGDGCC	3.67E-004	DIESEL GENERATOR COMMON MODE FAILURE
DGDGFR1R43S0001A	7.86E-003	DIVISION 1 DIESEL GENERATOR FAILS TO RUN
DGDGFR1R43S0001B	7.86E-003	DIVISION 2 DIESEL GENERATOR FAILS TO RUN
DGDGFS1R43S0001A	3.00E-002	DIVISION 1 DIESEL GENERATOR FAILS TO START
DGDGFS1R43S0001B	3.00E-002	DIVISION 2 DIESEL GENERATOR FAILS TO START
DGDGUM1R43S0001A	3.08E-002	DIV 1 D/G UNAVAILABLE DUE TO MAINTENANCE
DGDGUM1R43S0001B	3.08E-002	DIV 2 D/G UNAVAILABLE DUE TO MAINTENANCE
DGHICPOS11-2:5	1.25E-003	OPERATOR FAILS TO INITIATE DIESEL GENERATORS
DGHICPOS11-2:5:A	1.25E-003	OPERATOR FAILS TO INITIATE DIV 1 D/G
DGHICPOS11-2:5:B	1.25E-003	OPERATOR FAILS TO INITIATE DIV 2 D/G
DGHIMASR43-4:1:A	0.00E+000	FAILURE TO RESTORE FOLLOWING MAINTENANCE
DGHIMASR43-4:1:B	0.00E+000	FAILURE TO RESTORE FOLLOWING MAINTENANCE
DGHXPL1R46B0002A	2.05E-003	1R46-B0002A HEAT EXCHANGER PLUGS
DGHXPL1R46B0002B	2.05E-003	1R46-B0002B HEAT EXCHANGER PLUGS
DHBALC1R22S0009	9.60E-005	EH13 4160 V AC BUS HARDWARE FAILURE
DHCLDE1E22S0001	1.60E-003	DIV 3 D/G ACTUATION LOGIC FAILURE
DHDGFR1E22S0001	7.86E-003	DIVISION 3 DIESEL GENERATOR FAILS TO RUN
DHDGFS1E22S0001	3.00E-002	DIVISION 3 DIESEL GENERATOR FAILS TO START
DHDGUM1E22S0001	2.43E-002	DIV 3 D/G UNAVAILABLE DUE TO MAINTENANCE
DHHICPOS11-2:5:C	1.25E-003	OPERATOR FAILS TO INITIATE DIV 3 D/G
DHHIMASE22B-4:1	0.00E+000	FAILURE TO RESTORE FOLLOWING MAINTENANCE
DHHXPL1E22S0001	2.05E-003	1E22-S0001 HEAT EXCHANGER PLUGS
DMCLLF1M43C0002A	1.25E-004	1M43-C0002A CONTROL LOGIC FAILS TO FUNCTION
DVMVCC	9.35E-005	M16 VACUUM BREAKERS COMMON CAUSE FAILURES
DVMVNO1M16F0010A	2.93E-003	1M16-F0010A MOTOR VALVE NC - FAILS TO OPEN
DVMVNO1M16F0010B	2.93E-003	1M16-F0010B MOTOR VALVE NC - FAILS TO OPEN
DVVBNO1M16F0020A	1.00E-004	1M16-F0020A VACUUM BREAKER NC - FAILS TO OPEN
DVVBNO1M16F0020B	1.00E-004	1M16-F0020B VACUUM BREAKER NC - FAILS TO OPEN
E1DGACEH1112	2.00E-001	FAILURE TO RESTORE DIV 1 DIESEL GEN IN 12 HOURS
E1DGACEH1124	2.00E-001	FAILURE TO RESTORE DIV 1 DIESEL GEN IN 24 HOURS
E1DGACEH116	6.00E-001	FAILURE TO RESTORE DIV 1 DIESEL GEN IN 6 HOURS
E1DGACEH118	5.00E-001	FAILURE TO RESTORE DIV 1 DIESEL GEN IN 8 HOURS
E2DGACEH1212	2.00E-001	FAILURE TO RESTORE DIV 2 DIESEL GEN IN 12 HOURS
E2DGACEH1224	2.00E-001	FAILURE TO RESTORE DIV 2 DIESEL GEN IN 24 HOURS
E2DGACEH126	6.00E-001	FAILURE TO RESTORE DIV 2 DIESEL GEN IN 6 HOURS
E2DGACEH128	5.00E-001	FAILURE TO RESTORE DIV 2 DIESEL GEN IN 8 HOURS
ECCLLF1P42C0001A	1.25E-004	1P42-C0001A CONTROL LOGIC FAILS TO FUNCTION
ECCLLF1P42C0001B	1.25E-004	1P42-C0001B CONTROL LOGIC FAILS TO FUNCTION
ECCVNO1P42F0519A	1.00E-004	1P42-F0519A CHECK VALVE NC - FAILS TO OPEN

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
ECCVN01P42F0519B	1.00E-004	1P42-F0519B CHECK VALVE NC - FAILS TO OPEN
ECECUMA	1.98E-002	ECC TRAIN A UNAVAILABLE DUE TO MAINTENANCE
ECECUMB	1.98E-002	ECC TRAIN B UNAVAILABLE DUE TO MAINTENANCE
ECHICPSP42-4:2	5.00E-002	OPERATOR FAILS TO CLOSE VALVE 0P42-F0150A(B)
ECHICPSP42-4:2A	5.00E-002	OPERATOR FAILS TO CLOSE VALVE 0P42-F0150A
ECHICPSP42-4:2B	5.00E-002	OPERATOR FAILS TO CLOSE VALVE 0P42-F0150B
ECHICPSP42-4PMP	5.00E-002	OPERATOR FAILS TO INITIATE PUMP 1P42-C0001A(B)
ECHICPSP42-4PMPA	5.00E-002	OPERATOR FAILS TO INITIATE PUMP 1P42-C0001A
ECHICPSP42-4PMPB	5.00E-002	OPERATOR FAILS TO INITIATE PUMP 1P42-C0001B
ECHIMASP42-4:1A	0.00E+000	ECC TRAIN A NOT RESTORED FOLLOWING MAINTENANCE
ECHIMASP42-4:1B	0.00E+000	ECC TRAIN B NOT RESTORED FOLLOWING MAINTENANCE
ECHXPL1P42B0001A	2.05E-003	1P42-B0001A HEAT EXCHANGER PLUGS
ECHXPL1P42B0001B	2.05E-003	1P42-B0001B HEAT EXCHANGER PLUGS
ECMPCC	1.19E-005	ECC MOTOR PUMP COMMON CAUSE FAILURE
ECMPFR1P42C0001A	7.20E-004	1P42-C0001A MOTOR PUMP FAILS TO RUN
ECMPFR1P42C0001B	7.20E-004	1P42-C0001B MOTOR PUMP FAILS TO RUN
ECMPFS1P42C0001A	2.93E-003	1P42-C0001A MOTOR PUMP FAILS TO START
ECMPFS1P42C0001B	2.93E-003	1P42-C0001B MOTOR PUMP FAILS TO START
ECMVCC	9.25E-005	ECC MOTOR VALVE COMMON CAUSE FAILURE
ECMVNC0P42F0150A	2.93E-003	0P42-F0150A MOTOR VALVE NO - FAILS TO CLOSE
ECMVNC0P42F0150B	2.93E-003	0P42-F0150B MOTOR VALVE NO - FAILS TO CLOSE
ECXVPL1P42F0515A	4.50E-005	1P42-F0515A MANUAL VALVE PLUGS
ECXVPL1P42F0515B	4.50E-005	1P42-F0515B MANUAL VALVE PLUGS
ECXVPL1P42F0520A	4.50E-005	1P42-F0520A MANUAL VALVE PLUGS
ECXVPL1P42F0520B	4.50E-005	1P42-F0520B MANUAL VALVE PLUGS
ECXVPL1P42F0527A	4.50E-005	1P42-F0527A MANUAL VALVE PLUGS
ECXVPL1P42F0527B	4.50E-005	1P42-F0527B MANUAL VALVE PLUGS
ECXVPL1P42F0555A	4.50E-005	1P42-F0555A MANUAL VALVE PLUGS
ECXVPL1P42F0555B	4.50E-005	1P42-F0555B MANUAL VALVE PLUGS
ECXVPL1P42F0555C	4.50E-005	1P42-F0555C MANUAL VALVE PLUGS
ECXVPL1P42F0558A	4.50E-005	1P42-F0558A MANUAL VALVE PLUGS
ECXVPL1P42F0558B	4.50E-005	1P42-F0558B MANUAL VALVE PLUGS
ECXVPL1P42F0558C	4.50E-005	1P42-F0558C MANUAL VALVE PLUGS
EPCLLF1M39B0001A	1.25E-004	1M39-B0001A CONTROL LOGIC FAILS TO FUNCTION
EPCLLF1M39B0001B	1.25E-004	1M39-B0001B CONTROL LOGIC FAILS TO FUNCTION
EPCLLF1M39B0002	1.25E-004	1M39-B0002 CONTROL LOGIC FAILS TO FUNCTION
EPCLLF1M39B0003	1.25E-004	1M39-B0003 CONTROL LOGIC FAILS TO FUNCTION
EPCLLF1M39B0004	1.25E-004	1M39-B0004 CONTROL LOGIC FAILS TO FUNCTION
EPCLLF1M39B0006	1.25E-004	1M39-B0006 CONTROL LOGIC FAILS TO FUNCTION
EPEPUMHPCS	1.63E-003	ECCS PMP RM CLNG FOR HPCS UNAVAIL DUE TO MAINTENANCE
EPEPUMLPCIA	1.63E-003	ECCS PMP RM CLNG FOR LPCI A UNAVAIL DUE TO MAINTENANCE
EPEPUMLPCIB	1.63E-003	ECCS PMP RM CLNG FOR LPCI B UNAVAIL DUE TO MAINTENANCE
EPEPUMLPCIC	1.63E-003	ECCS PMP RM CLNG FOR LPCI C UNAVAIL DUE TO MAINTENANCE
EPEPUMLPCS	1.63E-003	ECCS PMP RM CLNG FOR LPCS UNAVAIL DUE TO MAINTENANCE
EPEPUMRCIC	1.63E-003	ECCS PMP RM CLNG FOR RCIC UNAVAIL DUE TO MAINTENANCE
EPFACCEPRCS	3.75E-006	ECCS PMP RM CLNG COMMON MODE COOLER FAILURE
EPFAFR1M39B0001A	3.00E-004	1M39-B0001A FAN FAILS TO RUN
EPFAFR1M39B0001B	3.00E-004	1M39-B0001B FAN FAILS TO RUN
EPFAFR1M39B0002	3.00E-004	1M39-B0002 FAN FAILS TO RUN
EPFAFR1M39B0003	3.00E-004	1M39-B0003 FAN FAILS TO RUN

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
EPFAFR1M39B0004	3.00E-004	1M39-B0004 FAN FAILS TO RUN
EPFAFR1M39B0006	3.00E-004	1M39-B0006 FAN FAILS TO RUN
EPFAFS1M39B0001A	3.75E-004	1M39-B0001A FAN FAILS TO START
EPFAFS1M39B0001B	3.75E-004	1M39-B0001B FAN FAILS TO START
EPFAFS1M39B0002	3.75E-004	1M39-B0002 FAN FAILS TO START
EPFAFS1M39B0003	3.75E-004	1M39-B0003 FAN FAILS TO START
EPFAFS1M39B0004	3.75E-004	1M39-B0004 FAN FAILS TO START
EPFAFS1M39B0006	3.75E-004	1M39-B0006 FAN FAILS TO START
EPXVPL1P42F0537	4.50E-005	1P42-F0537 MANUAL VALVE PLUGS
EPXVPL1P42F0541	4.50E-005	1P42-F0541 MANUAL VALVE PLUGS
EPXVPL1P42F0563A	4.50E-005	1P42-F0563A MANUAL VALVE PLUGS
EPXVPL1P42F0563B	4.50E-005	1P42-F0563B MANUAL VALVE PLUGS
EPXVPL1P42F0563C	4.50E-005	1P42-F0563C MANUAL VALVE PLUGS
EPXVPL1P42F0567A	4.50E-005	1P42-F0567A MANUAL VALVE PLUGS
EPXVPL1P42F0567B	4.50E-005	1P42-F0567B MANUAL VALVE PLUGS
EPXVPL1P42F0567C	4.50E-005	1P42-F0567C MANUAL VALVE PLUGS
EPXVPL1P42F0568	4.50E-005	1P42-F0568 MANUAL VALVE PLUGS
EPXVPL1P42F0572	4.50E-005	1P42-F0572 MANUAL VALVE PLUGS
EPXVPL1P45F0514	4.50E-005	1P45-F0514 MANUAL VALVE PLUGS
EPXVPL1P45F0518	4.50E-005	1P45-F0518 MANUAL VALVE PLUGS
ERCVNO1P45F0575	1.00E-004	1P45-F0575 CHECK VALVE NC - FAILS TO OPEN
ERHICPPS4:2-ESW	1.00E-002	OPERATORS FAIL TO ALIGN 11 VLVS FOR RPV INJECTION
ERXVNO1P45F0572	1.00E-004	1P45-F0572 MANUAL VALVE NC - FAILS TO OPEN
ERXVNO1P45F0573	1.00E-004	1P45-F0573 MANUAL VALVE NC - FAILS TO OPEN
ESCLLF1P45C0001A	1.25E-004	1P45-C0001A CONTROL LOGIC FAILS
ESCLLF1P45C0001B	1.25E-004	1P45-C0001B CONTROL LOGIC FAILS
ESCLLF1P45C0002	1.25E-004	1P45-C0002 CONTROL LOGIC FAILS
ESCVNO1P45F0501A	1.00E-004	1P45-F0501A CHECK VALVE NC - FAILS TO OPEN
ESCVNO1P45F0501B	1.00E-004	1P45-F0501B CHECK VALVE NC - FAILS TO OPEN
ESCVNO1P45F0552	1.00E-004	1P45-F0552 CHECK VALVE NC - FAILS TO OPEN
ESESUMA	1.89E-002	ESW TRAIN A UNAVAILABLE DUE TO MAINTENANCE
ESESUMB	1.89E-002	ESW TRAIN B UNAVAILABLE DUE TO MAINTENANCE
ESESUMC	9.37E-003	ESW TRAIN C UNAVAILABLE DUE TO MAINTENANCE
ESFLPL1P45D0002A	9.60E-004	1P45-D0002A STRAINER PLUGS
ESFLPL1P45D0002B	9.60E-004	1P45-D0002B STRAINER PLUGS
ESFLPL1P45D0003	9.60E-004	1P45-D0003 STRAINER PLUGS
ESHIMASP45-4:1A	0.00E+000	ESW TRAIN A NOT RESTORED FOLLOWING MAINTENANCE
ESHIMASP45-4:1B	0.00E+000	ESW TRAIN B NOT RESTORED FOLLOWING MAINTENANCE
ESHIMASP45-4:1C	0.00E+000	ESW TRAIN C NOT RESTORED FOLLOWING MAINTENANCE
ESMPCC	3.42E-004	ESW PUMP COMMON MODE FAILURE
ESMPFR1P45C0001A	7.20E-004	1P45-C0001A MOTOR PUMP FAILS TO RUN
ESMPFR1P45C0001B	7.20E-004	1P45-C0001B MOTOR PUMP FAILS TO RUN
ESMPFR1P45C0002	7.20E-004	1P45-C0002 MOTOR PUMP FAILS TO RUN
ESMPFS1P45C0001A	2.93E-003	1P45-C0001A MOTOR PUMP FAILS TO START
ESMPFS1P45C0001B	2.93E-003	1P45-C0001B MOTOR PUMP FAILS TO START
ESMPFS1P45C0002	2.93E-003	1P45-C0002 MOTOR PUMP FAILS TO START
ESMVCC	9.25E-005	ESW MOTOR VALVE COMMON CAUSE FAILURE
ESMVFC1P45F0014A	2.40E-006	1P45-F0014A MOTOR VALVE NO - FAILS CLOSED
ESMVFC1P45F0014B	2.40E-006	1P45-F0014B MOTOR VALVE NO - FAILS CLOSED
ESMVFC1P45F0068A	2.40E-006	1P45-F0068A MOTOR VALVE NO - FAILS CLOSED

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
ESMVFC1P45F0068B	2.40E-006	1P45-F0068B MOTOR VALVE NO - FAILS CLOSED
ESMVNO1P45F0130A	2.93E-003	1P45-F0130A MOTOR VALVE NC - FAILS TO OPEN
ESMVNO1P45F0130B	2.93E-003	1P45-F0130B MOTOR VALVE NC - FAILS TO OPEN
ESMVNO1P45F0140	2.93E-003	1P45-F0140 MOTOR VALVE NC - FAILS TO OPEN
ESMVPL1P45F0014A	4.50E-005	1P45-F0014A MOTOR VALVE PLUGS
ESMVPL1P45F0014B	4.50E-005	1P45-F0014B MOTOR VALVE PLUGS
ESMVPL1P45F0068A	4.50E-005	1P45-F0068A MOTOR VALVE PLUGS
ESMVPL1P45F0068B	4.50E-005	1P45-F0068B MOTOR VALVE PLUGS
ESMVPL1P45F0130A	4.50E-005	1P45-F0130A MOTOR VALVE PLUGS
ESMVPL1P45F0130B	4.50E-005	1P45-F0130B MOTOR VALVE PLUGS
ESMVPL1P45F0140	4.50E-005	1P45-F0140 MOTOR VALVE PLUGS
ESSCPL1P49D0001A	9.60E-004	1P49-D0001A TRAVELING SCREEN PLUGS
ESSCPL1P49D0001B	9.60E-004	1P49-D0001B TRAVELING SCREEN PLUGS
ESXVPL1P45F0519	4.50E-005	1P45-F0519 MANUAL VALVE PLUGS
ESXVPL1P45F0523	4.50E-005	1P45-F0523 MANUAL VALVE PLUGS
ESXVPL1P45F0530A	4.50E-005	1P45-F0530A MANUAL VALVE PLUGS
ESXVPL1P45F0530B	4.50E-005	1P45-F0530B MANUAL VALVE PLUGS
ESXVPL1P45F0534A	4.50E-005	1P45-F0534A MANUAL VALVE PLUGS
ESXVPL1P45F0534B	4.50E-005	1P45-F0534B MANUAL VALVE PLUGS
ESXVPL1P45F0536A	4.50E-005	1P45-F0536A MANUAL VALVE PLUGS
ESXVPL1P45F0536B	4.50E-005	1P45-F0536B MANUAL VALVE PLUGS
ESXVPL1P45F0541A	4.50E-005	1P45-F0541A MANUAL VALVE PLUGS
ESXVPL1P45F0541B	4.50E-005	1P45-F0541B MANUAL VALVE PLUGS
ESXVPL1P45F0550A	4.50E-005	1P45-F0550A MANUAL VALVE PLUGS
ESXVPL1P45F0550B	4.50E-005	1P45-F0550B MANUAL VALVE PLUGS
FPDPFROP54C0001	1.75E-001	0P54-C0001 DIESEL PUMP FAILS TO RUN
FPDPFSOP54C0001	3.00E-002	0P54-C0001 DIESEL PUMP FAILS TO START
FPDPUM	2.14E-002	DIESEL DRIVEN FIRE PUMP UNAVAILABLE DUE TO MAITNENANCE
FPHICPPS4:2-DD-0	3.00E-002	OPERATORS FAIL TO MAINTAIN FUEL OIL FOR DIESEL FIRE PMP
FPHICPPS4:2FP-LE	5.00E-002	OPERATOR FAILS TO ALIGN VLVS FOR LATE FP ALT INJ < 3 HRS
FPHICPPS4:2FP-LL	5.00E-003	OPERATOR FAILS TO ALIGN VLVS FOR LATE FP ALT INJ > 3 HRS
FPHICPPS4:2RCIC1	3.00E-001	FAIL TO ALIGN FP AFTER RCIC FAILS DUE SUPP POOL TEMP
FPHICPPS4:2RCIC2	1.00E-001	FAIL TO ALIGN FP AFTER RHR FAILS DUE TO MCC TEMP
FPHICPPS4:2RCIC3	1.00E-002	FAIL TO ALIGN FP AFTER HPCS FAILS DUE TO MCC TEMP
FPHICPPS4:2RCIC4	1.00E-001	FAIL TO ALIGN FAST FIRE PROTECTION ALTERNATE INJECTION
FPOFFSITEPUMPER	6.00E-001	OFFSITE PUMPER FAILS TO ARRIVE & PROVIDE WATER
FPXVNC1E12F0027B	1.00E-004	1E12-F0027B MANUAL VALVE NO FAILS TO CLOSE
FPXVNC1P45F0014B	1.00E-004	1P45-F0014B MANUAL VALVE NO FAILS TO CLOSE
FPXVNC1P45F0530B	1.00E-004	1P45-F0530B MANUAL VALVE FROZEN OPEN
FPXVNC1P45F0536B	1.00E-004	1P45-F0536B MANUAL VALVE NO FAILS TO CLOSE
FPXVNC1P45F0578	1.00E-004	1P45-F0578 MANUAL VALVE NO FAILS TO CLOSE
FPXVNO1E12F0024B	1.00E-004	1E12-F0024B MANUAL VALVE NC FAILS TO OPEN
FPXVNO1E12F0053B	1.00E-004	1E12-F0053B MANUAL VALVE NC FAILS TO OPEN
FPXVPL1P45F0572	8.21E-004	1P45-F0572 MANUAL VALVE PLUGGED
FPXVPL1P45F0573	8.21E-004	1P45-F0573 MANUAL VALVE PLUGGED
FPXVPL1P45F0589	8.21E-004	1P45F0589 MANUAL VALVE PLUGGED
FPXVPL1P45F0593	8.21E-004	1P45-F0593 MANUAL VALVE PLUGGED
FPXVPL1P45F0631	8.21E-004	1P45-F0631 MANUAL VALVE PLUGGED
FPXVPL1P45F0632	8.21E-004	1P45-F0632 MANUAL VALVE PLUGGED
FWAVFC1N21F0230	2.40E-006	1N21-F0230 AIR VALVE NO - FAILS CLOSED

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
FWAVNO1N21F0220	2.00E-003	1N21-F0220 AIR VALVE NC - FAILS OPEN
FWCLDE1N27C0004	1.60E-003	1N27-C0004 ACTUATION LOGIC FAILS
FWCLLF1N21C0001C	1.25E-004	1N21-C0001C CONTROL LOGIC FAILS TO FUNCTION
FWCLLF1N27C0001D	1.25E-004	1N27-C0001D CONTROL LOGIC FAILS TO FUNCTION
FWCLLF1N27C0004	1.25E-004	1N27-C0004 CONTROL LOGIC FAILS TO FUNCTION
FWCVNO1N27F0559A	1.00E-004	1N27-F0559A CHECK VALVE NC - FAILS TO OPEN
FWCVNO1N27F0559B	1.00E-004	1N27-F0559B CHECK VALVE NC - FAILS TO OPEN
FWHICPEC5-2:3LCS	1.00E-002	OPER FAILS TO CNTRL RPV LEVEL AT TAF W/ FW DURING IORV
FWHICPEC5-3:2	1.00E-002	OPER FAILS TO CONTROL RPV LEVEL AT TAF
FWHICPEL-2-FDW-L	1.00E-002	OPERATOR FAILS TO REOPEN MFP CONTROL VALVES FOR T3C-C
FWHICPEL-2-FDW-V	5.00E-003	OPER FAILS TO REOPN MFP CNTRL VALVES OR DEPRESSURIZE RPV
FWHICPSN27-4:1IA	1.20E-001	OPER FAILS TO CNTRL RX FEED BOOSTER PMP DURING LOSS OF IA
FWHICPSN27-4:1IL	5.00E-003	OPER FAILS TO CNTRL RX FEED BOOSTER PMP LOSS OF IA >2 HRS
FWMPFR1N21C0001A	7.20E-004	1N21-C0001A MOTOR PUMP FAILS TO RUN
FWMPFR1N21C0001B	7.20E-004	1N21-C0001B MOTOR PUMP FAILS TO RUN
FWMPFR1N21C0001C	7.20E-004	1N21-C0001C MOTOR PUMP FAILS TO RUN
FWMPFR1N27C0001A	7.20E-004	1N27-C0001A MOTOR PUMP FAILS TO RUN
FWMPFR1N27C0001B	7.20E-004	1N27-C0001B MOTOR PUMP FAILS TO RUN
FWMPFR1N27C0001C	7.20E-004	1N27-C0001C MOTOR PUMP FAILS TO RUN
FWMPFR1N27C0001D	7.20E-004	1N27-C0001D MOTOR PUMP FAILS TO RUN
FWMPFR1N27C0004	7.20E-004	1N27-C0004 MOTOR PUMP FAILS TO RUN
FWMPFS1N21C0001C	2.93E-003	1N21-C0001C MOTOR PUMP FAILS TO START
FWMPFS1N27C0001D	2.93E-003	1N27-C0001D MOTOR PUMP FAILS TO START
FWMPFS1N27C0004	2.93E-003	1N27-C0004 MOTOR PUMP FAILS TO START
FWMPUM1N21C0001C	5.00E-001	1N21-C0001C MOTOR PUMP UNAVAIL DUE TO MAINT
FWMPUM1N27C0001D	5.00E-001	1N27-C0001D MOTOR PUMP UNAVAIL DUE TO MAINTENANCE
FWMPUM1N27C0004	5.00E-003	1N27-C0004 MOTOR PUMP UNAVAIL DUE TO MAINTENANCE
FWMVNO1N27F0200	2.93E-003	1N27-F0200 MOTOR VALVE FAILS TO OPEN - NC
FWTPFR1N27C0002A	1.13E-001	1N27-C0002A TURBINE PUMP FAILS TO RUN
FWTPFR1N27C0002B	1.13E-001	1N27-C0002B TURBINE PUMP FAILS TO RUN
FWXVPL1N27F0560A	4.50E-005	1N27-F0560A MANUAL VALVE PLUGS
FWXVPL1N27F0560B	4.50E-005	1N27-F0560B MANUAL VALVE PLUGS
HIHICPEC5-3:2-F	1.00E-003	FAILS TO RESTR RHR A/B/LPCS AND CONTRL AT TAF W/ FDW
HIHICPEC5-3:2-S	1.00E-003	FAILS TO RESTR RHR A/B/LPCS AND CNTRL AT TAF W/O FDW
HIHICPEC5-5-CRIT	2.00E-003	OPER FAILS TO CNTRL RPV LEVEL AND FLUSHES BORON
HIHICPOR10-4:0-I	5.00E-002	OPERATOR FAILS TO CLOSE FPCC OUTBOARD VALVE 1G41-F0145
HIHICPOR10-4:3-B	1.00E-002	OPER FAILS TO X-TIE UNIT 1 AND 2 BATT AND LOAD SHED
HIHICPOR10-4:3-D	2.00E-003	OPER FAILS TO OPEN DIV 3 SWITCHGEAR ROOM DOORS
HIHICPOR10-XTIE	5.00E-003	OPERATOR FAILS TO CROSSTIE DIV 3 BUS TO DIV 2 BUS
HPCLDECST	1.60E-003	CST LOW LEVEL ACTUATION LOGIC HARDWARE FAILURE
HPCLDEL2	1.60E-003	HPCS ACTUATION LOGIC FAILURE
HPCLDEMINFLOW	1.60E-003	MIN. FLOW MOV ACTUATION LOGIC FAILURE
HPCLLF1E22C0001	1.25E-004	1E22-C0001 CONTROL LOGIC FAILS TO ENERGIZE
HPCVFC1E22F0002	3.44E-003	1E22-F0002 CHECK VALVE NC - FAILS TO OPEN
HPCVNO1E22F0002	1.00E-004	1E22-F0002 CST SUCTION LINE CHK VL NC - FAILS TO OPEN
HPCVNO1E22F0005	1.00E-004	1E22-F0005 CHECK VALVE NC - FAILS TO OPEN
HPCVNO1E22F0016	1.00E-004	1E22-F0016 CHECK VALVE NC - FAILS TO OPEN
HPCVNO1E22F0024	1.00E-004	1E22-F0024 PUMP DISC CHECK VALVE NC - FAILS TO OPEN
HPHICPEL-1	1.25E-003	OPERATOR FAILS TO INITIATE HIGH PRESSURE INJECTION
HPHICPSE22-5:0	5.00E-002	OPERATOR FAILS TO CONTROL MIN FLOW VALVE 1E22-F0012

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
HPHICPSE22-5:2	5.00E-002	OPER FAILS TO XFER TO SUPR POOL WITH 1E22-F015
HPHIMASE22-4:1	0.00E+000	FAILURE TO RESTORE HPCS AFTER MAINTENANCE
HPSPUM	2.68E-003	HPCS UNAVAILABLE DUE TO MAINTENANCE
HPLXDE1E22N0654C	9.58E-004	1E22-N0654C LEVEL INSTRUMENT FAILS TO FUNCTION
HPLXDE1E22N0654G	9.58E-004	1E22-N0654G LEVEL INSTRUMENT FAILS TO FUNCTION
HPMPFR1E22C0001	7.20E-004	1E22-C0001 HPCS MOTOR DRIVEN PUMP FAILS TO RUN
HPMPFS1E22C0001	2.93E-003	1E22-C0001 HPCS MOTOR DRIVEN PUMP FAILS TO START
HPMVNC1E22F0001	2.93E-003	1E22-F0001 MOTOR VALVE NO - FAILS TO CLOSE
HPMVNO1E22F0004	2.93E-003	1E22-F0004 MOTOR VALVE NC - FAILS TO OPEN
HPMVNO1E22F0012	2.93E-003	1E22-F0012 MOTOR VALVE NC - FAILS TO OPEN
HPMVNO1E22F0015	2.93E-003	1E22-F0015 MOTOR VALVE NC - FAILS TO OPEN
HPXVPL1E22F0036	4.50E-005	1E22-F0036 MANUAL VALVE PLUGS
HPXVPL1E22F0518	4.50E-005	1E22-F0518 CST SUCTION MANUAL VLV PLUGS
HVFAFR1M41C0001A	3.00E-004	1M41-C0001A FAN FAILS TO RUN
HVFAFR1M41C0001B	3.00E-004	1M41-C0001B FAN FAILS TO RUN
HVFAFR1M41C0002A	3.00E-004	1M41-C0002A FAN FAILS TO RUN
HVFAFR1M41C0002B	3.00E-004	1M41-C0002B FAN FAILS TO RUN
IAACLF	0.00E+000	SA/IA AIR RECEIVER TANKS DO NOT HAVE SUFFICIENT AIR
IAAVFC1P52F0050	2.40E-006	1P52-F0050 PNEUMATIC VALVE NO - FAILS CLOSED
IAAVFC1P52F0210	2.40E-006	1P52-F0210 AIR VALVE NO - FAILS CLOSED
IAAVFC2P52F0050	2.40E-006	2P52-F0050 AIR VALVE NO - FAILS CLOSED
IAAVFC2P52F0210	2.40E-006	2P52-F0210 AIR VALVE NO - FAILS CLOSED
IAAVNC1P52F0050	2.00E-003	1P52-F0050 PNEUMATIC VALVE NO - FAILS TO CLOSE
IAAVNC2P52F0050	2.00E-003	2P52-F0050 PNEUMATIC VALVE NO - FAILS TO CLOSE
IACLLF1P52C0001	1.25E-004	1P52-C0001 CONTROL LOGIC HARDWARE FAILURE
IACLLF2P52C0001	1.25E-004	2P52-C0001 CONTROL LOGIC HARDWARE FAILURE
IACMCC	8.25E-003	IA/SA COMPRESSORS COMMON CAUSE FAILURES
IACMFR1P52C0001	3.83E-003	1P52-C0001 COMPRESSOR FAILS TO RUN
IACMFR2P52C0001	3.83E-003	2P52-C0001 COMPRESSOR FAILS TO RUN
IACMFS1P52C0001	8.25E-002	1P52-C0001 COMPRESSOR FAILS TO START
IACMFS2P52C0001	8.25E-002	2P52-C0001 COMPRESSOR FAILS TO START
IACMUM2P51C0001	5.00E-001	2P51-C0001 UNAVAILABLE DUE TO MAINTENANCE
IACMUM2P52C0001	5.00E-001	2P52-C0001 UNAVAILABLE DUE TO MAINTENANCE
IACVNO1P52F0532	1.00E-004	1P52-F0532 CHECK VALVE NC - FAILS TO OPEN
IACVNO1P52F0550	1.00E-004	1P52-F0550 CHECK VALVE NC - FAILS TO OPEN
IACVNO1P52F0639	1.00E-004	1P52-F0639 CHECK VALVE NC - FAILS TO OPEN
IACVNO1P52F1004	1.00E-004	1P52-F1004 CHECK VALVE NC - FAILS TO OPEN
IACVNO2P52F0532	1.00E-004	2P52-F0532 CHECK VALVE NC - FAILS TO OPEN
IACVNO2P52F1004	1.00E-004	2P52-F1004 CHECK VALVE NC - FAILS TO OPEN
IAHICPSP51-4:2	5.00E-002	OPERATOR FAILS TO REPOSITION NCC LUBE OIL CLR OUTLET VLV
IAHICRSP52-7:2	1.00E-001	OPERATOR FAILS TO OVERRIDE ISOLATION SIGNAL
IAHIMASP51-4:1:2	1.00E-003	FAILURE TO RESTORE FOLLOWING MAINTENANCE
IAMVFC1P52F0200	2.40E-006	1P52-F0200 MOTOR VALVE NO - FAILS CLOSED
IAMVFC1P52F0646	2.40E-006	1P52-F0646 MOTOR VALVE NO - FAILS CLOSED
IAPLLR1P52	2.71E-003	IA & SA FAIL DUE TO LINE BREAK
IAPXDE1P52N0060	9.59E-005	1P52-N0060 PRESSURE SENSOR FAILS TO FUNCTION
IAPXDE1P52N0185	9.59E-005	1P52-N0185 PRESSURE SENSOR FAILS TO FUNCTION
IAPXDE2P52N0060	9.59E-005	2P52-N0060 PRESSURE SENSOR FAILS TO FUNCTION
IAPXDE2P52N0185	9.59E-005	2P52-N0185 PRESSURE SENSOR FAILS TO FUNCTION
IF	1.00E+000	INTERNAL FLOODING

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
LCCLDE	1.60E-003	LPCI B AUTOMATIC ACTUATION LOGIC HARDWARE FAILURE
LCCLDEMINFLOW	1.60E-003	MINIMUM FLOW MOV ACTUATION LOGIC FAILURE
LCCLDEMINFLOWA	1.60E-003	MINIMUM FLOW MOV ACTUATION LOGIC FAILURE
LCCLDEMINFLOWB	1.60E-003	MINIMUM FLOW MOV ACTUATION LOGIC FAILURE
LCCLDEMINFLOWC	1.60E-003	MINIMUM FLOW MOV ACTUATION LOGIC FAILURE
LCCLLF1E12C0002A	1.25E-004	1E12-C0002A CONTROL LOGIC FAILS TO FUNCTION
LCCLLF1E12C0002B	1.25E-004	1E12-C0002B CONTROL LOGIC FAILS TO FUNCTION
LCCLLF1E12C0002C	1.25E-004	1E12-C0002C CONTROL LOGIC FAILS TO FUNCTION
LCCVN01E12F0031A	1.00E-004	1E12-F0031A CHECK VALVE NC - FAILS TO OPEN
LCCVN01E12F0031B	1.00E-004	1E12-F0031B CHECK VALVE NC - FAILS TO OPEN
LCCVN01E12F0031C	1.00E-004	1E12-F0031C CHECK VALVE NC - FAILS TO OPEN
LCCVN01E12F0041A	1.00E-004	1E12-F0041A CHECK VALVE NC - FAILS TO OPEN
LCCVN01E12F0041B	1.00E-004	1E12-F0041B CHECK VALVE NC - FAILS TO OPEN
LCCVN01E12F0041C	1.00E-004	1E12-F0041C CHECK VALVE NC - FAILS TO OPEN
LCCVN01E12F0046A	1.00E-004	1E12-F0046A CHECK VALVE NC - FAILS TO OPEN
LCCVN01E12F0046B	1.00E-004	1E12-F0046B CHECK VALVE NC - FAILS TO OPEN
LCCVN01E12F0046C	1.00E-004	1E12-F0046C CHECK VALVE NC - FAILS TO OPEN
LCCVN01E12F0050A	1.00E-004	1E12-F0050A CHECK VALVE NC - FAILS TO OPEN
LCCVN01E12F0050B	1.00E-004	1E12-F0050B CHECK VALVE NC - FAILS TO OPEN
LCHICPEL-1-LPCI	1.00E-003	OPERATOR FAILS TO INITIATE LPCI B
LCHICPSE12-5:1	5.00E-002	OPERATOR FAILS TO CONTROL MIN FLOW VALVE 1E12-F0064A
LCHIMASE12-4:1A	0.00E+000	FAILURE TO RESTORE TRAIN A LPCI FOLLOWING MAINT
LCHIMASE12-4:1B	0.00E+000	FAILURE TO RESTORE TRAIN B LPCI FOLLOWING MAINT
LCHIMASE12-4:1C	0.00E+000	FAILURE TO RESTORE TRAIN C LPCI FOLLOWING MAINT
LCHXPL1E12B0001A	2.05E-003	1E12-B0001A HEAT EXCHANGER PLUGS
LCHXPL1E12B0001B	2.05E-003	1E12-B0001B HEAT EXCHANGER PLUGS
LCHXPL1E12B0001C	2.05E-003	1E12-B0001C HEAT EXCHANGER PLUGS
LCHXPL1E12B0001D	2.05E-003	1E12-B0001D HEAT EXCHANGER PLUGS
LCLCUMA	1.93E-002	LPCI TRAIN A UNAVAILABLE DUE TO MAINTENANCE
LCLCUMB	1.81E-002	LPCI TRAIN B UNAVAILABLE DUE TO MAINTENANCE
LCLCUMC	1.03E-002	LPCI TRAIN C UNAVAILABLE DUE TO MAINTENANCE
LCLCUMLPCIALPCS	5.29E-003	LPCI A AND LPCS BOTH UNAVAILABLE DUE TO MAINT
LCLCUMLPCIBLPCIC	1.77E-002	LPCI B AND LPCI C BOTH UNAVAILABLE DUE TO MAINT
LCLCUMRHRALPRC	9.22E-003	LPCI A, LPCS & RCIC ALL UNAVAILABLE DUE TO MAINTENANCE
LCMPCC	2.93E-004	LPCI PUMP COMMON MODE FAILURE
LCMPFR1E12C0002A	7.20E-004	1E12-C0002A MOTOR PUMP FAILS TO RUN
LCMPFR1E12C0002B	7.20E-004	1E12-C0002B MOTOR PUMP FAILS TO RUN
LCMPFR1E12C0002C	7.20E-004	1E12-C0002C MOTOR PUMP FAILS TO RUN
LCMPFS1E12C0002A	2.93E-003	1E12-C0002A MOTOR PUMP FAILS TO START
LCMPFS1E12C0002B	2.93E-003	1E12-C0002B MOTOR PUMP FAILS TO START
LCMPFS1E12C0002C	2.93E-003	1E12-C0002C MOTOR PUMP FAILS TO START
LCMVCC	9.25E-005	INJECTION VALVE COMMON MODE FAILURE
LCMVFC1E12F0003A	2.40E-006	1E12-F0003A MOTOR VALVE NO - FAILS CLOSED
LCMVFC1E12F0003B	2.40E-006	1E12-F0003B MOTOR VALVE NO - FAILS CLOSED
LCMVFC1E12F0004A	2.40E-006	1E12-F0004A MOTOR VALVE NO - FAILS CLOSED
LCMVFC1E12F0004B	2.40E-006	1E12-F0004B MOTOR VALVE NO - FAILS CLOSED
LCMVFC1E12F0027A	2.40E-006	1E12-F0027A MOTOR VALVE NO - FAILS CLOSED
LCMVFC1E12F0027B	2.40E-006	1E12-F0027B MOTOR VALVE NO - FAILS CLOSED
LCMVFC1E12F0047A	2.40E-006	1E12-F0047A MOTOR VALVE NO - FAILS CLOSED
LCMVFC1E12F0047B	2.40E-006	1E12-F0047B MOTOR VALVE NO - FAILS CLOSED

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
LCMVFC1E12F0048A	2.40E-006	1E12-F0048A MOTOR VALVE NO - FAILS CLOSED
LCMVFC1E12F0048B	2.40E-006	1E12-F0048B MOTOR VALVE NO - FAILS CLOSED
LCMVFC1E12F0105	2.40E-006	1E12-F0105 MOTOR VALVE NO - FAILS CLOSED
LCMVFO1E12F0021	1.20E-005	1E12-F0021 MOTOR VALVE NC - FAILS OPEN
LCMVFO1E12F0024A	1.[illegible]-005	1E12-F0024A MOTOR VALVE NC - FAILS OPEN
LCMVFO1E12F0024B	1.20E-005	1E12-F0024B MOTOR VALVE NC - FAILS OPEN
LCMVFO1E12F0028A	1.20E-005	1E12-F0028A MOTOR VALVE NC - FAILS OPEN
LCMVFO1E12F0028B	1.20E-005	1E12-F0028B MOTOR VALVE NC - FAILS OPEN
LCMVFO1E12F0537A	1.20E-005	1E12-F0537A MOTOR VALVE NC - FAILS OPEN
LCMVFO1E12F0537B	1.20E-005	1E12-F0537B MOTOR VALVE NC - FAILS OPEN
LCMVNC1E12F0021	2.93E-003	1E12-F0021 MOTOR VALVE FAILS TO CLOSE
LCMVNO1E12F0042A	2.93E-003	1E12-F0042A MOTOR VALVE NC - FAILS TO OPEN
LCMVNO1E12F0042B	2.93E-003	1E12-F0042B MOTOR VALVE NC - FAILS TO OPEN
LCMVNO1E12F0042C	2.93E-003	1E12-F0042C MOTOR VALVE NC - FAILS TO OPEN
LCMVNO1E12F0064A	2.93E-003	1E12-F0064A MOTOR VALVE NC - FAILS TO OPEN
LCMVNO1E12F0064B	2.93E-003	1E12-F0064B MOTOR VALVE NC - FAILS TO OPEN
LCMVNO1E12F0064C	2.93E-003	1E12-F0064C MOTOR VALVE NC - FAILS TO OPEN
LCMVNO1E12F0105	2.93E-003	1E12-F0105 MOTOR VALVE FAILS TO OPEN
LCMVPL1E12F0003A	4.50E-005	1E12-F0003A MOTOR VALVE PLUGS
LCMVPL1E12F0003B	4.50E-005	1E12-F0003B MOTOR VALVE PLUGS
LCMVPL1E12F0004A	4.50E-005	1E12-F0004A MOTOR VALVE PLUGS
LCMVPL1E12F0004B	4.50E-005	1E12-F0004B MOTOR VALVE PLUGS
LCMVPL1E12F0027A	4.50E-005	1E12-F0027A MOTOR VALVE PLUGS
LCMVPL1E12F0027B	4.50E-005	1E12-F0027B MOTOR VALVE PLUGS
LCMVPL1E12F0047A	4.50E-005	1E12-F0047A MOTOR VALVE PLUGS
LCMVPL1E12F0047B	4.50E-005	1E12-F0047B MOTOR VALVE PLUGS
LCMVPL1E12F0048A	4.50E-005	1E12-F0048A MOTOR VALVE PLUGS
LCMVPL1E12F0048B	4.50E-005	1E12-F0048B MOTOR VALVE PLUGS
LCMVPL1E12F0105	4.50E-005	1E12-F0105 MOTOR VALVE PLUGS
LCPXCCLPCI	9.59E-006	LPCI PMP PERMISSIVE COMMON MODE FAILURE
LCPXDE1E12N0655A	9.59E-005	1E12-N0655A PRESSURE INSTRUMENT FAILS TO FUNCTION
LCPXDE1E12N0655B	9.59E-005	1E12-N0655B PRESSURE INSTRUMENT FAILS TO FUNCTION
LCPXDE1E12N0655C	9.59E-005	1E12-N0655C PRESSURE INSTRUMENT FAILS TO FUNCTION
LCPXDE1E12N0656A	9.59E-005	1E12-N0656A PRESSURE INSTRUMENT FAILS TO FUNCTION
LCPXDE1E12N0656B	9.59E-005	1E12-N0656B PRESSURE INSTRUMENT FAILS TO FUNCTION
LCPXDE1E12N0656C	9.59E-005	1E12-N0656C PRESSURE INSTRUMENT FAILS TO FUNCTION
LCXVPL1E12F0018A	4.50E-005	1E12-F0018A MANUAL VALVE PLUGS
LCXVPL1E12F0018B	4.50E-005	1E12-F0018B MANUAL VALVE PLUGS
LCXVPL1E12F0018C	4.50E-005	1E12-F0018C MANUAL VALVE PLUGS
LCXVPL1E12F0029A	4.50E-005	1E12-F0029A MANUAL VALVE PLUGS
LCXVPL1E12F0029B	4.50E-005	1E12-F0029B MANUAL VALVE PLUGS
LCXVPL1E12F0029C	4.50E-005	1E12-F0029C MANUAL VALVE PLUGS
LCXVPL1E12F0039A	4.50E-005	1E12-F0039A MANUAL VALVE PLUGS
LCXVPL1E12F0039B	4.50E-005	1E12-F0039B MANUAL VALVE PLUGS
LCXVPL1E12F0039C	4.50E-005	1E12-F0039C MANUAL VALVE PLUGS
LI19B	7.65E-001	COMPLEMENT TO LI19
LI26B	7.20E-001	COMPLEMENT TO LI26
LPCLDE	1.60E-003	LPCS AUTOMATIC ACTUATION LOGIC HARDWARE FAILURE
LPCLDEMINFLOW	1.60E-003	MIN. FLOW MOV ACTUATION LOGIC FAILURE
LPCLLF1E21C0001	1.25E-004	1E21-C0001 CONTROL LOGIC FAILS TO ENERGIZE

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
LPCVN01E21F0003	1.00E-004	1E21-F0003 CHECK VALVE NC - FAILS TO OPEN
LPCVN01E21F0006	1.00E-004	1E21-F0006 CHECK VALVE NC - FAILS TO OPEN
LPCVN01E21F0501	1.00E-004	1E21-F0501 CHECK VALVE NC - FAILS TO OPEN
LPHICPEL-1	1.00E-003	OPERATOR FAILS TO INITIATE LOW PRESSURE INJECTION
LPHICPEL-1-LPCS	1.00E-003	OPERATOR FAILS TO INITIATE LPCS
LPHICPSE21-5:1	5.00E-002	OPERATOR FAILS TO CONTROL MIN FLOW VALVE 1E21-F0011
LPHIMASE21-4:1	0.00E+000	FAILURE TO RESTORE LPCS AFTER MAINTENANCE
LPLPUM	2.03E-002	LPCS UNAVAILABLE DUE TO MAINTENANCE
LPMPFR1E21C0001	7.20E-004	1E21-C0001 LPCS MOTOR PUMP FAILS TO RUN
LPMPFS1E21C0001	2.93E-003	1E21-C0001 LPCS MOTOR PUMP FAILS TO START
LPMVNC1E21F0011	2.93E-003	1E21-F0011 MOTOR VALVE FAILS TO CLOSE
LPMVNO1E21F0005	2.93E-003	1E21-F0005 MOTOR VALVE NC - FAILS TO OPEN
LPMVNO1E21F0011	2.93E-003	1E21-F0011 MOTOR VALVE FAILS TO OPEN
LPMVPL1E21F0001	4.50E-005	1E21-F0001 MOTOR VALVE PLUGS
LPPXDE1E21N0650	9.59E-005	1E21-N0650 RX PRESS PRESSURE INSTRUMENT FAILS TO FUNCTION
LPPXDE1E21N0652	9.59E-005	1E21-N0652 PRESSURE INSTRUMENT FAILS TO FUNCTION
LPPXDE1E21N0653	9.59E-005	1E21-N0653 PRESSURE INSTRUMENT FAILS TO FUNCTION
LPXVPL1E21F0007	4.50E-005	1E21-F0007 MANUAL VALVE PLUGS
M	1.00E-004	AT LEAST ONE SRV FAILS TO OPEN AGAINST SPRING
MCHICRRECVRD4	0.00E+000	MCC, SWTCHGR, & MISC ELECT AREAS RECVRD IN 4 HOURS
MCHICRRECVRD9	0.00E+000	MCC, SWTCHGR, & MISC ELECT AREAS RECVRD IN 9 HOURS
MCMFFR1M23C0001A	3.00E-011	1M23-C0001A MOTOR FAN FAILS TO RUN
MCMFFR1M23C0001B	3.00E-011	1M23-C0001B MOTOR FAN FAILS TO RUN
MCMFFR1M23C0002A	3.00E-011	1M23-C0002A MOTOR RETURN FAN FAILS TO RUN
MCMFFR1M23C0002B	3.00E-011	1M23-C0002B MOTOR RETURN FAN FAILS TO RUN
MCMFFS1M23C0001B	2.93E-003	1M23-C0001B MOTOR FAN FAILS TO START
MCMFFS1M23C0002B	2.93E-003	1M23-C0002B MOTOR RETURN FAN FAILS TO START
MESA133	7.76E-003	NO FLOW TO ESW TRAIN A HEAT EXCHANGERS
MESB133	7.76E-003	NO FLOW TO ESW TRAIN B HEAT EXCHANGERS
MESC133	7.76E-003	NO FLOW TO ESW TRAIN C HEAT EXCHANGERS
MIAA174	1.05E-004	UNIT 2 F1004 AND F0210 VALVES FAIL CLOSED
MIAA241	8.66E-002	1P52-C0001 COMPRESSOR FAILS
MIAA331	8.66E-002	2P52-C0001 COMPRESSOR FAILS
MIAA353	1.05E-004	UNIT 1 F1004 AND F0210 VALVES FAIL CLOSED
MIAA431	8.66E-002	2P51-C0001 COMPRESSOR FAILS
MIAA584	3.77E-003	1P43-C0001C MOTOR PUMP FAILS
MIAA884	3.77E-003	1P41-C001D MOTOR PUMP FAILS
MLHICPE32	1.00E-002	OPERATOR FAILS TO INITIATE MSIV LEAKAGE CONTROL
MLMFFR1E32C0001	5.03E-003	1E32-C0001 MOTOR FAN FAILS TO RUN
MLMFFS1E32C0001	2.93E-003	1E32-C0001 MOTOR FAN FAILS TO START
MLMLUMA	1.00E-002	INBOARD MSIV LEAKGE CONTROL UNAVAILABLE DUE TO MAINTENANCE
MLMV1E32F0003ANC	2.93E-004	1E32-F0003A FAILS TO CLOSE AFTER BEING OPENED
MLMV1E32F0003ENC	2.93E-004	1E32-F0003E FAILS TO CLOSE AFTER BEING OPENED
MLMV1E32F0003JNC	2.93E-004	1E32-F0003J FAILS TO CLOSE AFTER BEING OPENED
MLMV1E32F0003NNC	2.93E-004	1E32-F0003N FAILS TO CLOSE AFTER BEING OPENED
MLMVCC	9.25E-005	1E32 MOTOR VALVE COMMON CAUSE FAILURES
MLMVNO1E32F0001A	2.93E-003	1E32-F0001A MOTOR VALVE NC - FAILS TO OPEN
MLMVNO1E32F0001E	2.93E-003	1E32-F0001E MOTOR VALVE NC - FAILS TO OPEN
MLMVNO1E32F0001J	2.93E-003	1E32-F0001J MOTOR VALVE NC - FAILS TO OPEN
MLMVNO1E32F0001N	2.93E-003	1E32-F0001N MOTOR VALVE NC - FAILS TO OPEN

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
MLMVN01E32F0002A	2.93E-003	1E32-F0002A MOTOR VALVE NC - FAILS TO OPEN
MLMVN01E32F0002E	2.93E-003	1E32-F0002E MOTOR VALVE NC - FAILS TO OPEN
MLMVN01E32F0002J	2.93E-003	1E32-F0002J MOTOR VALVE NC - FAILS TO OPEN
MLMVN01E32F0002N	2.93E-003	1E32-F0002N MOTOR VALVE NC - FAILS TO OPEN
MLMVN01E32F0003A	2.93E-003	1E32-F0003A MOTOR VALVE NC - FAILS TO OPEN
MLMVN01E32F0003E	2.93E-003	1E32-F0003E MOTOR VALVE NC - FAILS TO OPEN
MLMVN01E32F0003J	2.93E-003	1E32-F0003J MOTOR VALVE NC - FAILS TO OPEN
MLMVN01E32F0003N	2.93E-003	1E32-F0003N MOTOR VALVE NC - FAILS TO OPEN
NBLXCCL1	9.59E-005	RPV LEVEL 1 COMMON MODE MISCALIBRATION
NBLXCCL2	9.59E-005	RPV LEVEL 2 COMMON MODE MISCALIBRATION
NBLXCCL3	9.59E-005	RPV LEVEL 3 COMMON MODE MISCALIBRATION
NBLXDE1B21N0667C	9.58E-004	1B21-N0667C LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0667G	9.58E-004	1B21-N0667G LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0667L	9.58E-004	1B21-N0667L LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0667R	9.58E-004	1B21-N0667R LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0673C	9.58E-004	1B21-N0673C RPV L2 LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0673G	9.58E-004	1B21-N0673G RPV L2 LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0673L	9.58E-004	1B21-N0673L RPV L2 LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0673R	9.58E-004	1B21-N0673R RPV L2 LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0691A	9.58E-004	1B21-N0691A RPV L1 LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0691B	9.58E-004	1B21-N0691B RPV L1 LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0691E	9.58E-004	1B21-N0691E RPV L1 LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0691F	9.58E-004	1B21-N0691F RPV L1 LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0692A	9.58E-004	1B21-N0692A RPV L2 LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0692B	9.58E-004	1B21-N0692B RPV L2 LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0692E	9.58E-004	1B21-N0692E RPV L2 LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0692F	9.58E-004	1B21-N0692F RPV L2 LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0695A	9.58E-004	1B21-N0695A LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1B21N0695B	9.58E-004	1B21-N0695B LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1C34N0004A	9.58E-004	1C34-N0004A RPV L8 LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1C34N0004B	9.58E-004	1C34-N0004B RPV L8 LEVEL INSTRUMENT FAILS TO FUNCTION
NBLXDE1C34N0004C	9.58E-004	1C34-N0004C RPV L8 LEVEL INSTRUMENT FAILS TO FUNCTION
NBPXCCDW	9.59E-006	DRYWELL PRESSURE COMMON MODE MISCALIBRATION
NBPXDE1B21N0667C	9.59E-005	1B21-N0667C PRESSURE INSTRUMENT FAILS TO FUNCTION
NBPXDE1B21N0667G	9.59E-005	1B21-N0667G PRESSURE INSTRUMENT FAILS TO FUNCTION
NBPXDE1B21N0667L	9.59E-005	1B21-N0667L PRESSURE INSTRUMENT FAILS TO FUNCTION
NBPXDE1B21N0667R	9.59E-005	1B21-N0667R PRESSURE INSTRUMENT FAILS TO FUNCTION
NBPXDE1B21N0694A	9.59E-005	1B21-N0694A DW PRES PRESSURE INSTRUMENT FAILS TO FUNCTION
NBPXDE1B21N0694B	9.59E-005	1B21-N0694B DW PRES PRESSURE INSTRUMENT FAILS TO FUNCTION
NBPXDE1B21N0694E	9.59E-005	1B21-N0694E DW PRES PRESSURE INSTRUMENT FAILS TO FUNCTION
NBPXDE1B21N0694F	9.59E-005	1B21-N0694F DW PRES PRESSURE INSTRUMENT FAILS TO FUNCTION
NBTICCADS	2.93E-005	ADS TIMER RELAY COMMON MODE FAILURE
NBTILF1B21K5A	2.93E-004	1B21-K5A TIME DELAY RELAY FAILS TO FUNCTION
NBTILF1B21K5B	2.93E-004	1B21-K5B TIME DELAY RELAY FAILS TO FUNCTION
NCCLLF1P43C0001C	1.25E-004	1P43-C0001C CONTROL LOGIC FAILS
NCMPFR1P43C0001A	7.20E-004	1P43-C0001A MOTOR PUMP FAILS TO RUN
NCMPFR1P43C0001B	7.20E-004	1P43-C0001B MOTOR PUMP FAILS TO RUN
NCMPFR1P43C0001C	7.20E-004	1P43-C0001C MOTOR PUMP FAILS TO RUN
NCMPFS1P43C0001C	2.93E-003	1P43-C0001C MOTOR PUMP FAILS TO START
NCMPUM1P43C0001C	5.00E-001	NCC STANDBY PUMP UNAVAILABLE DUE TO MAINTENANCE

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
NSAVNC1B21F0022A	2.00E-003	1B21-F0022A (MSIV) AIR VALVE NO - FAILS TO CLOSE
NSAVNC1B21F0022B	2.00E-003	1B21-F0022B (MSIV) AIR VALVE NO - FAILS TO CLOSE
NSAVNC1B21F0022C	2.00E-003	1B21-F0022C (MSIV) AIR VALVE NO - FAILS TO CLOSE
NSAVNC1B21F0022D	2.00E-003	1B21-F0022D (MSIV) AIR VALVE NO - FAILS TO CLOSE
NSAVNC1B21F0028A	2.00E-003	1B21-F0028A (MSIV) AIR VALVE NO - FAILS TO CLOSE
NSAVNC1B21F0028B	2.00E-003	1B21-F0028B (MSIV) AIR VALVE NO - FAILS TO CLOSE
NSAVNC1B21F0028C	2.00E-003	1B21-F0028C (MSIV) AIR VALVE NO - FAILS TO CLOSE
NSAVNC1B21F0028D	2.00E-003	1B21-F0028D (MSIV) AIR VALVE NO - FAILS TO CLOSE
NSHICPEC5-2-L1	1.00E-001	OPERATOR FAILS TO BYPASS MSIV LEVEL 1 ISOLATION FOR T3C-C
NSHICPEC5-2-L1T3	1.00E+000	OPERATOR FAILS TO BYP MSIV LVL 1 ISOL FOR T3A-C OR T3B-C
NSNSCC	2.00E-004	MSIV COMMON CAUSE FAILURES
ONE	1.00E+000	BASIC EVENT SET TO 1.0
P1	1.60E-002	ONE SRV FAILS TO OPEN AND RECLOSE
P2	1.60E-003	TWO SRVS FAIL TO OPEN AND RECLOSE
PX	1.00E+000	PROBABILITY THAT SRV'S WILL OPEN FOR A GIVEN TRANSIENT
R1	5.73E-002	NONRECOVERY OF OFFSITE AC POWER IN 6 HOURS
R10	6.27E-001	OFFSITE AC PWR NOT RESTRD IN 12 HOURS IF NO PWR AT 10 HRS
R11	6.57E-002	NONRECOVERY OF OFFSITE AC POWER IN 5 HOURS
R12	3.24E-002	NONRECOVERY OF OFFSITE AC POWER IN 9 HOURS
R13	2.66E-001	NONRECOVERY OF OFFSITE AC POWER IN 1 HOUR
R13B	7.34E-001	RECOVERY OF OFFSITE AC POWER IN 1 HOUR
R14	1.74E-001	NONRECOVERY OF OFFSITE AC POWER IN 2 HOURS
R14B	8.26E-001	RECOVERY OF OFFSITE AC POWER IN 2 HOURS
R15	8.23E-002	NONRECOVERY OF OFFSITE AC POWER IN 3 HOURS
R15B	9.18E-001	RECOVERY OF OFFSITE AC POWER IN 3 HOURS
R16	5.73E-001	NONRECOVERY OF OFFSITE AC POWER IN 0.3 HOURS
R16B	4.27E-001	RECOVERY OF OFFSITE AC POWER IN 0.3 HOURS
R17	1.29E-001	OFFSITE AC PWR NOT RESTRD IN 4 HOURS IF NO PWR AT 0.3 HR
R17B	8.71E-001	OFFSITE AC PWR RESTRD IN 4 HOURS IF NO PWR AT 0.3 HR
R18	4.25E-001	OFFSITE AC PWR NOT RESTRD IN 4 HOURS IF NO PWR AT 2 HRS
R18B	5.75E-001	OFFSITE AC PWR RESTRD IN 4 HOURS IF NO PWR AT 2 HRS
R19	9.85E-003	NONRECOVERY OF OFFSITE AC POWER IN 14 HOURS
R2	4.07E-002	NONRECOVERY OF OFFSITE AC POWER IN 8 HOURS
R20	5.75E-003	NONRECOVERY OF OFFSITE AC POWER IN 17 HOURS
R23	3.81E-001	OFFSITE AC PWR NOT RESTRD IN 17 HOURS IF NO PWR AT 12 HRS
R23B	6.19E-001	OFFSITE AC PWR RESTRD IN 17 HOURS IF NO PWR AT 12 HRS
R24	3.78E-001	OFFSITE AC PWR NOT RESTRD IN 5 HOURS IF NO PWR AT 2 HRS
R24B	6.22E-001	OFFSITE AC PWR RESTRD IN 5 HOURS IF NO PWR AT 2 HRS
R25	1.96E-002	NONRECOVERY OF OFFSITE AC POWER IN 11 HOURS
R26	5.03E-001	OFFSITE AC PWR NOT RESTRD IN 14 HOURS IF NO PWR AT 11 HRS
R26B	4.97E-001	OFFSITE AC PWR RESTRD IN 14 HOURS IF NO PWR AT 11 HRS
R27	1.31E-003	NONRECOVERY OF OFFSITE AC POWER IN 24 HOURS
R28	8.21E-003	NONRECOVERY OF OFFSITE AC POWER IN 15 HOURS
R29	1.60E-001	OFFSITE AC PWR NOT RESTRD IN 24 HOURS IF NO PWR AT 15 HRS
R29B	8.40E-001	OFFSITE AC PWR RESTRD IN 24 HOURS IF NO PWR AT 15 HRS
R3	1.51E-002	NONRECOVERY OF OFFSITE AC POWER IN 12 HOURS
R30	2.67E-002	OFFSITE AC PWR NOT RESTRD IN 24 HOURS IF NO PWR AT 7 HRS
R31	4.91E-002	NONRECOVERY OF OFFSITE AC POWER IN 7 HOURS
R32	5.97E-001	OFFSITE AC PWR NOT RESTRD IN 7 HOURS IF NO PWR AT 3 HRS
R32B	4.03E-001	OFFSITE AC PWR RESTRD IN 7 HOURS IF NO PWR AT 3 HRS

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
R33	2.02E-001	NONRECOVERY OF OFFSITE AC POWER IN 1.7 HOURS
R33B	7.98E-001	RECOVERY OF OFFSITE AC POWER IN 1.7 HOURS
R34	6.49E-003	OFFSITE AC PWR NOT RESTRD IN 24 HOURS IF NO PWR AT 1.7 HR
R35	3.53E-001	OFFSITE AC PWR NOT RESTRD IN 1.7 HRS IF NO PWR AT 0.3 HR
R3	6.47E-001	OFFSITE AC PWR RESTRD IN 1.7 HOURS IF NO PWR AT 0.3 HR
R	5.29E-001	NONRECOVERY OF OFFSITE AC POWER IN 0.4 HOURS
R3	4.71E-001	RECOVERY OF OFFSITE AC POWER IN 0.4 HOURS
R	1.40E-001	OFFSITE AC PWR NOT RESTRD IN 4 HOURS IF NO PWR AT 0.4 HR
R3	8.60E-001	OFFSITE AC PWR RESTRD IN 4 HOURS IF NO PWR AT 0.4 HR
R	8.98E-001	OFFSITE AC PWR NOT RESTRD IN 4 HOURS IF NO PWR AT 3 HRS
R3	1.02E-001	OFFSITE AC PWR RESTRD IN 4 HOURS IF NO PWR AT 3 HRS
R	1.23E-002	NONRECOVERY OF OFFSITE AC POWER IN 13 HOURS
R4	2.04E-001	OFFSITE AC PWR NOT RESTRD IN 12 HOURS IF NO PWR AT 4 HRS
R40	6.16E-001	NONRECOVERY OF OFFSITE AC POWER IN 0.26 HOURS
R40B	3.84E-001	RECOVERY OF OFFSITE AC POWER IN 0.26 HOURS
R41	1.20E-001	OFFSITE AC PWR NOT RESTRD IN 4 HOURS IF NO PWR AT 0.26HR
R41B	8.80E-001	OFFSITE AC PWR RESTRD IN 4 HOURS IF NO PWR AT 0.26HR
R42	4.60E-003	NONRECOVERY OF OFFSITE AC POWER IN 18 HOURS
R43	6.22E-002	OFFSITE AC PWR NOT RESTRD IN 18 HOURS IF NO PWR AT 4 HRS
R44	5.44E-001	OFFSITE AC PWR NOT RESTRD IN 15 HOURS IF NO PWR AT 12 HRS
R44B	4.56E-001	OFFSITE AC PWR RESTRD IN 15 HOURS IF NO PWR AT 12 HRS
R45	5.59E-002	OFFSITE AC PWR NOT RESTRD IN 18 HOURS IF NO PWR AT 3 HRS
R46	2.29E-001	NONRECOVERY OF OFFSITE AC POWER IN 1.4 HOURS
R46B	7.71E-001	NONRECOVERY OF OFFSITE AC POWER IN 1.4 HOURS
R47	4.33E-001	OFFSITE AC PWR NOT RESTRD IN 1.4 HRS IF NO PWR AT .4 HRS
R47B	5.67E-001	OFFSITE AC PWR NOT RESTRD IN 1.4 HRS IF NO PWR AT .4 HRS
R48	4.32E-001	OFFSITE AC PWR NOT RESTRD IN 1 HOUR IF NO PWR AT .26 HR
R48B	5.68E-001	OFFSITE AC PWR RESTRD IN 1 HOUR IF NO PWR AT .26 HR
R49	2.46E-003	NONRECOVERY OF OFFSITE AC POWER IN 21 HOURS
R4B	7.96E-001	OFFSITE AC PWR RESTRD IN 12 HOURS IF NO PWR AT 4 HRS
R5	7.39E-002	NONRECOVERY OF OFFSITE AC POWER IN 4 HOURS
R50	2.00E-001	OFFSITE AC PWR NOT RESTRD IN 21 HOURS IF NO PWR AT 13 HRS
R50B	8.00E-001	OFFSITE AC PWR RESTRD IN 21 HOURS IF NO PWR AT 13 HRS
R51	4.19E-001	OFFSITE AC PWR NOT RESTRD IN 15 HOURS IF NO PWR AT 11 HRS
R51B	5.81E-001	OFFSITE AC PWR RESTRD IN 15 HOURS IF NO PWR AT 11 HRS
R52	4.67E-001	OFFSITE AC PWR NOT RESTRD IN 17 HOURS IF NO PWR AT 13 HRS
R52B	5.33E-001	OFFSITE AC PWR RESTRD IN 17 HOURS IF NO PWR AT 13 HRS
R53	6.96E-001	OFFSITE AC PWR NOT RESTRD IN 6 HOURS IF NO PWR AT 3 HRS
R53B	3.04E-001	OFFSITE AC PWR RESTRD IN 6 HOURS IF NO PWR AT 3 HRS
R54	2.29E-002	OFFSITE AC PWR NOT RESTRD IN 24 HOURS IF NO PWR AT 6 HRS
R55	2.38E-001	NONRECOVERY OF OFFSITE AC POWER IN 1.3 HOURS
R55B	7.62E-001	RECOVERY OF OFFSITE AC POWER IN 1.3 HOURS
R56	5.50E-003	OFFSITE AC PWR NOT RESTRD IN 24 HOURS IF NO PWR AT 1.3 HR
R57	4.50E-001	OFFSITE AC PWR NOT RESTRD IN 1.3 HRS IF NO PWR AT 0.4 HR
R57B	5.50E-001	OFFSITE AC PWR RESTRD IN 1.3 HOURS IF NO PWR AT 0.4 HR
R58	1.64E-003	NONRECOVERY OF OFFSITE AC POWER IN 23 HOURS
R59	8.37E-002	OFFSITE AC PWR NOT RESTORED IN 23 HRS IF NO PWR AT 11 HRS
R6	2.41E-002	NONRECOVERY OF OFFSITE AC POWER IN 10 HOURS
R60	3.29E-001	OFFSITE AC PWR NOT RESTORED IN 6 HOURS IF NO PWR AT 2 HRS
R60B	6.71E-001	OFFSITE AC PWR RESTORED IN 6 HOURS IF NO PWR AT 2 HRS

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
R61	4.92E-003	OFFSITE AC PWR NOT RESTORED IN 24 HRS IF NO PWR AT 1 HOUR
R62	1.09E-002	OFFSITE AC PWR NOT RESTORED IN 17 HRS IF NO PWR AT 0.4 HR
R7	7.75E-001	OFFSITE AC PWR NOT RESTRD IN 6 HOURS IF NO PWR AT 4 HR
R7B	2.25E-001	OFFSITE AC PWR RESTRD IN 6 HOURS IF NO PWR AT 4 HR
R8	3.26E-001	OFFSITE AC PWR NOT RESTRD IN 10 HOURS IF NO PWR AT 4 HR
R8B	6.74E-001	OFFSITE AC PWR RESTRD IN 10 HOURS IF NO PWR AT 4 HR
R9	3.11E-002	OFFSITE AC PWR NOT RESTRD IN 10 HOURS IF NO PWR AT 6 HR
RCCLDECST	1.60E-003	CST LOW LEVEL ACTUATION LOGIC HARDWARE FAILURE
RCCLDEL2	1.60E-003	RCIC AUTOMATIC L2 ACTUATION LOGIC HARDWARE FAILURE
RCCLDEL8	1.60E-003	RCIC LEVEL 8 ACTUATION LOGIC HARDWARE FAILURE
RCCLDEMINFLOW	1.60E-003	MIN FLOW ACTUATION LOGIC HARDWARE FAILURE
RCCLLF1E51C0001	1.25E-004	1E51-C0001 CONTROL LOGIC FAILS TO FUNCTION
RCCVN01E51F0011	1.00E-004	1E51-F0011 CHECK VALVE NC - FAILS TO OPEN
RCCVN01E51F0021	1.00E-004	1E51-F0021 CHECK VALVE NC - FAILS TO OPEN
RCCVN01E51F0030	1.00E-004	1E51-F0030 CHECK VALVE NC - FAILS TO OPEN
RCCVN01E51F0040	1.00E-004	1E51-F0040 CHECK VALVE NC - FAILS TO OPEN
RCCVN01E51F0065	1.00E-004	1E51-F0065 CHECK VALVE NC - FAILS TO OPEN
RCCVN01E51F0066	1.00E-004	1E51-F0066 CHECK VALVE NC - FAILS TO OPEN
RCCVN01E51F0577	1.00E-004	1E51-F0577 CHECK VALVE NC - FAILS TO OPEN
RCHICPEL-2-CST-S	5.00E-002	OPERATOR FAILS TO PREVENT SUCTION SHIFT TO SUPP POOL
RCHICPS51-LDTRIP	5.00E-002	FAILURE TO RECOVER ISOL SIGNAL ON HIGH STEAM TUNNEL TEMP
RCHICPSE51-5:1	5.00E-002	OPERATOR FAILS TO PERFORM RCIC SUCTION SHIFT
RCHIMASE51-4:1	0.00E+000	FAILURE TO RESTORE RCIC FOLLOWING MAINTENANCE
RCLXCCCSTLOW	9.59E-005	CST LOW LEVEL INSTR COMMON CAUSE MISCALIBRATION
RCLXDE1E51N0035A	9.58E-004	1E51-N0035A CST LEVEL INSTRUMENT FAILS TO FUNCTION
RCLXDE1E51N0035E	9.58E-004	1E51-N0035E CST LEVEL INSTRUMENT FAILS TO FUNCTION
RCMVFC1E51F0010	2.40E-006	1E51-F0010 MOTOR VALVE FAILS TO REMAIN OPN
RCMVFC1E51F0063	2.40E-006	1E51-F0063 MOTOR VALVE FAILS TO REMAIN OPN
RCMVFC1E51F0064	2.40E-006	1E51-F0064 MOTOR VALVE FAILS TO REMAIN OPN
RCMVFC1E51F0068	2.40E-006	1E51-F0068 MOTOR VALVE FAILS TO REMAIN OPN
RCMVNC1E51F0045	2.93E-003	1E51-F0045 MOTOR VALVE NO - FAILS TO CLOSE
RCMVN01E51F0013	2.93E-003	1E51-F0013 MOTOR VALVE NC - FAILS TO OPEN
RCMVN01E51F0019	2.93E-003	1E51-F0019 MOTOR VALVE NC - FAILS TO OPEN
RCMVN01E51F0031	2.93E-003	1E51-F0031 MOTOR VALVE NC - FAILS TO OPEN
RCMVN01E51F0045	2.93E-003	1E51F0045 MOTOR VALVE NC - FAILS TO OPEN
RCMVN01E51F0046	2.93E-003	1E51-F0046 MOTOR VALVE NC - FAILS TO OPEN
RCPRN01E51F0015	1.00E-003	1E51-F0015 PRESSURE REGULATOR FAILS TO OPEN
RCRCUM	1.81E-002	RCIC UNAVAILABLE DUE TO MAINTENANCE
RCTPFR1E51C0001	1.13E-001	1E51-C0001 TURBINE PUMP FAILS TO RUN
RCTPFR1E51C001	1.08E-002	1E51-C0001 TURBINE PUMP FAILS TO RUN
RCTPFS1E51C0001	2.93E-003	1E51-C0001 TURBINE PUMP FAILS TO START
RCXVPL1E51F0501	1.37E-004	1E51-F0501 MANUAL VALVE PLUGGED
RCXVPL1E51F0502	1.37E-004	1E51-F0502 MANUAL VALVE PLUGGED
RPHICPERC-1:Q-2	1.00E-004	OPERATOR FAILS TO SCRAM REACTOR
RPT	1.00E-004	[illegible] RECIRC PUMP FAILS TO TRIP
RPVLEVEL8TRIP	1.00E-001	PROB THAT RFP WILL NOT BE AVAIL DUE TO RPV LEVEL 8 TRIP
S1	3.00E-004	INTERMEDIATE LOSS OF COOLANT ACCIDENT (INTERMEDIATE LOCA)
S2	3.00E-003	SMALL LOSS OF COOLANT ACCIDNET (SMALL LOCA)
SACLLF2P51C0001	1.25E-004	2P51-C0001 CONTROL LOGIC HARDWARE FAILURE
SACMFR1P51C0001	3.83E-003	1P51-C0001 COMPRESSOR FAILS TO RUN

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
SACMFR2P51C0001	3.83E-003	2P51-C0001 COMPRESSOR FAILS TO RUN
SACMFS2P51C0001	8.25E-002	2P51-C0001 COMPRESSOR FAILS TO START
SAHICPSP51-4:2	5.00E-002	OPERATOR FAILS TO REPOSITION NCC LUBE OIL CLR OUTLET VLV
SAHIMASP51-4:1:2	1.00E-003	FAILURE TO RESTORE FOLLOWING MAINTENANCE
SAPXDE2P51N0185	9.59E-005	2P51-N0185 PRESSURE SENSOR FAILS TO FUNCTION
SCHICPSE12-5:3	1.09E-004	OPERATOR FAILS TO ALIGN RHR TO SUPP POOL COOLING
SCHICPSE12-5:3A	1.09E-004	OPERATOR FAILS TO ALIGN RHR TRAIN A TO SUPP POOL CLNG
SCHICPSE12-5:3B	1.09E-004	OPERATOR FAILS TO ALIGN RHR TRAIN B TO SUPP POOL CLNG
SCMVCC	9.25E-005	SUPP POOL RTRN VLVS COMMON CAUSE FAILURE
SCMVFO1E12F0053A	1.20E-005	1E12-F0053A MOTOR VALVE NC - FAILS OPEN
SCMVFO1E12F0053B	1.20E-005	1E12-F0053B MOTOR VALVE NC - FAILS OPEN
SCMVNC1E12F0027A	2.93E-003	1E12-F0027A MOTOR VALVE NO - FAILS TO CLOSE
SCMVNC1E12F0027B	2.93E-003	1E12-F0027B MOTOR VALVE NO - FAILS TO CLOSE
SCMVNC1E12F0048A	2.93E-003	1E12-F0048A MOTOR VALVE NO - FAILS TO CLOSE
SCMVNC1E12F0048B	2.93E-003	1E12-F0048B MOTOR VALVE NO - FAILS TO CLOSE
SCMVNO1E12F0024A	2.93E-003	1E12-F0024A MOTOR VALVE NC - FAILS TO OPEN
SCMVNO1E12F0024B	2.93E-003	1E12-F0024B MOTOR VALVE NC - FAILS TO OPEN
SICMFR1P57C0001	3.83E-003	1P57-C0001 COMPRESSOR FAILS TO RUN
SICVFO1P57F0555A	3.44E-003	1P57-F0555A CHECK VALVE FAILS OPEN
SICVFO1P57F0555B	3.44E-003	1P57-F0555B CHECK VALVE FAILS OPEN
SICVFO1P57F0556A	3.44E-003	1P57-F0556A CHECK VALVE FAILS OPEN
SICVFO1P57F0556B	3.44E-003	1P57-F0556B CHECK VALVE FAILS OPEN
SICVNO1P57F0524A	1.00E-004	1P57-F0524A CHECK VALVE NC - FAILS TO OPEN
SICVNO1P57F0524B	1.00E-004	1P57-F0524B CHECK VALVE NC - FAILS TO OPEN
SIHICPSP57-7:1	5.00E-002	OPERATORS FAIL TO CONNECT AIR CYLINDERS
SIHICPSP57-7:1:A	5.00E-003	OPERATORS FAIL TO CONNECT AIR CYLINDERS
SIHICPSP57-7:1:B	5.00E-003	OPERATORS FAIL TO CONNECT AIR CYLINDERS
SIHIMASP57-4:0:A	0.00E+000	FAILURE TO RESTORE FOLLOWING MAINTENANCE
SIHIMASP57-4:0:B	0.00E+000	FAILURE TO RESTORE FOLLOWING MAINTENANCE
SIMVFC1P57F0015A	2.40E-006	1P57-F0015A MOTOR VALVE NO - FAILS CLOSED
SIMVFC1P57F0015B	2.40E-006	1P57-F0015B MOTOR VALVE NO - FAILS CLOSED
SIMVFC1P57F0020A	2.40E-006	1P57-F0020A MOTOR VALVE NO - FAILS CLOSED
SIMVFC1P57F0020B	2.40E-006	1P57-F0020B MOTOR VALVE NO - FAILS CLOSED
SLCVNO1C41F0006	1.00E-004	1C41-F0006 CHECK VALVE NC - FAILS TO OPEN
SLCVNO1C41F0007	1.00E-004	1C41-F0007 CHECK VALVE NC - FAILS TO OPEN
SLCVNO1C41F0033A	1.00E-004	1C41-F0033A CHECK VALVE NC - FAILS TO OPEN
SLCVNO1C41F0033B	1.00E-004	1C41-F0033B CHECK VALVE NC - FAILS TO OPEN
SLEVCC	2.93E-004	SLC EXPLOSIVE VALVE COMMON MODE FAILURE
SLEVNO1C41F0004A	2.93E-003	1C41-F0004A EXPLOSIVE VALVE NC - FAILS TO OPEN
SLEVNO1C41F0004B	2.93E-003	1C41-F0004B EXPLOSIVE VALVE NC - FAILS TO OPEN
SLHICPEQ-6-RPVLC	5.00E-002	OPER FAILS TO CNTRL RPV LVL & MAINTAIN BORON INVENTORY
SLHICPEQ-6-SLC1	1.25E-003	OPERATOR FAILS TO INITIATE SLC 1 PUMP INJECTION
SLHICPEQ-6-SLCX	1.00E+000	OPERATOR FAILS TO INITIATE SLC LEVEL CONTROL FAILS
SLHICREQ-6-SLCR	1.00E-001	OPER FAILS TO INITIATE SLC GIVEN CORE DAMAGE
SLHIMA	0.00E+000	FAILURE TO RESTORE SLC FOLLOWING MAINTENANCE/TEST
SLMPCC	2.93E-004	SLC MOTOR PUMPS COMMON CAUSE FAILURE
SLMPFR1C41C0001A	1.20E-004	1C41-C0001A MOTOR PUMP FAILS TO RUN
SLMPFR1C41C0001B	1.20E-004	1C41-C0001B MOTOR PUMP FAILS TO RUN
SLMPFS1C41C0001A	2.93E-003	1C41-C0001A MOTOR PUMP FAILS TO START
SLMPFS1C41C0001B	2.93E-003	1C41-C0001B MOTOR PUMP FAILS TO START

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
SLMPUM1C41C0001A	1.65E-002	1C41-C0001A UNAVAILABLE DUE TO MAINTENANCE
SLMPUM1C41C0001B	1.65E-002	1C41-C0001B UNAVAILABLE DUE TO MAINTENANCE
SLMVCC	9.25E-005	SLC MOTOR VALVES COMMON CAUSE FAILURE
SLMVCC1G33	9.25E-005	RWCU VALVES COMMON CAUSE FAILURE
SLMVNC1G33F0001	2.93E-003	1G33-F0001 MOTOR VALVE NO - FAILS TO CLOSE
SLMVNC1G33F0004	2.93E-003	1G33-F0004 MOTOR VALVE NO - FAILS TO CLOSE
SLMVNO1C41F0001A	2.93E-003	1C41-F0001A MOTOR VALVE NC - FAILS TO OPEN
SLMVNO1C41F0001B	2.93E-003	1C41-F0001B MOTOR VALVE NC - FAILS TO OPEN
SLMVUM1C41F0001A	1.08E-002	1C41-F0001A UNAVAILABLE DUE TO MAINTENANCE
SLMVUM1C41F0001B	1.08E-002	1C41-F0001B UNAVAILABLE DUE TO MAINTENANCE
SLRVFO1C41F0029A	1.60E-005	1C41-F0029A RELIEF VALVE NC - FAILS OPEN
SLRVFO1C41F0029B	1.60E-005	1C41-F0029B RELIEF VALVE NC - FAILS OPEN
SLXVPL1C41F0002A	4.50E-005	1C41-F0002A MANUAL VALVE PLUGS
SLXVPL1C41F0002B	4.50E-005	1C41-F0002B MANUAL VALVE PLUGS
SLXVPL1C41F0003A	4.50E-005	1C41-F0003A MANUAL VALVE PLUGS
SLXVPL1C41F0003B	4.50E-005	1C41-F0003B MANUAL VALVE PLUGS
SLXVPL1C41F0036	4.50E-005	1C41-F0036 MANUAL VALVE PLUGS
SLXVPL1C41F0037A	4.50E-005	1C41-F0037A MANUAL VALVE PLUGS
SLXVPL1C41F0037B	4.50E-005	1C41-F0037B MANUAL VALVE NC - FAILS TO OPEN
SMCLLF1G43A	1.25E-004	SPMU TRAIN A CONTROL LOGIC FAILS
SMCLLF1G43B	1.25E-004	SPMU TRAIN B CONTROL LOGIC FAILS
SMMVCC	9.25E-005	SPMU MOTOR VALVE COMMON CAUSE FAILURE
SMMVNO1G43F0030A	2.93E-003	1G43-F0030A MOTOR VALVE FAILS TO OPEN
SMMVNO1G43F0030B	2.93E-003	1G43-F0030B MOTOR VALVE FAILS TO OPEN
SMMVNO1G43F0040A	2.93E-003	1G43-F0040A MOTOR VALVE FAILS TO OPEN
SMMVNO1G43F0040B	2.93E-003	1G43-F0040B MOTOR VALVE FAILS TO OPEN
SMRANDOM-SRVTPLL	5.00E-001	SMALL LOCA RANDOM SPMU DUMP LIFTS SP INTO UNSAFE ZONE
SPHICPPS4:5SPCU	1.00E+000	OPERATORS FAIL TO ALIGN SUPP POOL C/U ALTERNATE INJECTION
SPHICPPS4:5SPCUL	5.00E-002	OPERATORS FAIL TO ALIGN SUP POOL C/U ALT INJECTION(LATE)
SPMPFR1G42C0001	7.20E-004	1G42-C0001 MOTOR PUMP FAILS TO RUN
SPMPFS1G42C0001	2.93E-003	1G42-C0001 MOTOR PUMP FAILS TO START
SPMVNO1G42F0010	2.93E-003	1G42-F0010 MOTOR VALVE NC - FAILS TO OPEN
SPMVNO1G42F0020	2.93E-003	1G42-F0020 MOTOR VALVE NC - FAILS TO OPEN
SWCLLF1P41C001D	1.25E-004	1P41-C001D CONTROL LOGIC FAILS TO FUNCTION
SWMPFR1P41C001A	7.20E-004	1P41-C001A MOTOR PUMP FAILS TO RUN
SWMPFR1P41C001B	7.20E-004	1P41-C001B MOTOR PUMP FAILS TO RUN
SWMPFR1P41C001C	7.20E-004	1P41-C001C MOTOR PUMP FAILS TO RUN
SWMPFR1P41C001D	7.20E-004	1P41-C001D MOTOR PUMP FAILS TO RUN
SWMPFS1P41C001D	2.93E-003	1P41-C001D MOTOR PUMP FAILS TO START
SWMPUM1P41C001D	5.00E-001	STANDBY SERVICE WATER PUMP UNAVAIL DUE TO MAINTENANCE
T1	6.09E-002	LOSS OF OFFSITE POWER TRANSIENT (LOOP)
T2	1.62E+000	TRANSIENT WITH LOSS OF POWER CONVERSION SYSTEM
T3A	4.51E+000	TRANSIENTS WITH PCS INITIALLY AVAILABLE
T3B	7.60E-001	TRANSIENT W/ LOSS OF FEEDWATER BUT W/ PCS INITIALLY AVAIL
T3C	1.40E-001	TRANSIENT CAUSED BY INADVERTENT OPEN RELIEF VALVE ON RPV
TBFAFR1M35C0001A	3.00E-004	1M35-C0001A FAN FAILS TO RUN
TBFAFR1M35C0001B	3.00E-004	1M35-C0001B FAN FAILS TO RUN
TBFAFR1M35C0001C	3.00E-004	1M35-C0001C FAN FAILS TO RUN
TBFAFS1M35C0001C	3.75E-004	1M35-C0001C FAN FAILS TO START
TBHICPSM35	5.00E-002	OPERATOR FAILS TO START STANDBY FAN

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
TCW	8.51E-002	LOSS OF CONTROL COMPLEX CHILLED WATER
TIA	9.20E-002	LOSS OF INSTRUMENT AIR
TSW	1.00E-003	LOSS OF SERVICE WATER
TWCLLF1P44C0001C	1.25E-004	1P44-C0001C CONTROL LOGIC FAILS TO START
TWMPFR1P44C0001A	7.20E-004	1P44-C0001A MOTOR PUMP FAILS TO RUN
TWMPFR1P44C0001B	7.20E-004	1P44-C0001B MOTOR PUMP FAILS TO RUN
TWMPFR1P44C0001C	7.20E-004	1P44-C0001C MOTOR PUMP FAILS TO RUN
TWMPFS1P44C0001C	2.93E-003	1P44-C0001C MOTOR PUMP FAILS TO START
TWMPUM1P44C0001C	5.00E-001	1P44-C0001C MOTOR PUMP UNAVAIL DUE TO MAINTENANCE
U109B	8.95E-001	COMPLEMENT TO U109
U111B	8.96E-001	COMPLEMENT TO U111
U202B	7.82E-001	COMPLEMENT TO U202
U207B	8.75E-001	COMPLEMENT TO U207
U210B	8.34E-001	COMPLEMENT TO U210
VA03B	8.38E-001	COMPLEMENT TO VA03
VA04B	7.33E-001	COMPLEMENT TO VA04
VA05B	8.76E-001	COMPLEMENT TO VA05
VA08B	7.19E-001	COMPLEMENT TO VA08
VA09B	5.33E-001	COMPLEMENT TO VA09
VA10B	8.23E-001	COMPLEMENT TO VA10
VA15B	5.44E-001	COMPLEMENT TO VA15
VA16B	7.29E-001	COMPLEMENT TO VA16
VA18B	8.01E-001	COMPLEMENT TO VA18
VA20B	7.44E-001	COMPLEMENT TO VA20
X26B	1.00E+000	COMPLEMENT TO X26
X33B	9.00E-001	COMPLEMENT TO X33
XHOS-LO-DW-PRESS	0.00E+000	DW PRESSURE TOO LOW TO CAUSE ACTUATION YES=1.0 NO=0.0
XHOSACPOWER1	0.00E+000	RECOVERY OF AC PWR IN 1 HOUR 1 HOUR = 1
XHOSACPOWER10	0.00E+000	RECOVERY OF AC PWR IN 10 HOURS 10 = 1
XHOSACPOWER10:4	0.00E+000	RECOVERY OF AC PWR IN 10 HOURS GIVEN 4 10:4 = 1
XHOSACPOWER10:6	0.00E+000	RECOVERY OF AC PWR IN 10 HOURS GIVEN 6 10:6 = 1
XHOSACPOWER11	0.00E+000	RECOVERY OF AC PWR IN 11 HOURS 11 HOURS = 1
XHOSACPOWER12	0.00E+000	RECOVERY OF AC PWR IN 12 HOURS 12 HOURS = 1
XHOSACPOWER12:10	0.00E+000	RECOVERY OF A CPWR IN 12 HRS GIVEN 10 12:10 = 1
XHOSACPOWER12:4	0.00E+000	RECOVERY OF AC PWR IN 12 HOURS GIVEN 4 12:4 = 1
XHOSACPOWER13	0.00E+000	RECOVERY OF AC PWR IN 13 HOURS 13 HOURS = 1
XHOSACPOWER14	0.00E+000	RECOVERY OF AC PWR IN 14 HOURS 14 HOURS = 1
XHOSACPOWER14:11	0.00E+000	RECOVERY OF AC PWR IN 14 GIVEN 11 HRS 14:11 = 1
XHOSACPOWER15	0.00E+000	RECOVERY OF AC PWR IN 15 HOURS 15 HOURS = 1
XHOSACPOWER15:11	0.00E+000	RECOVERY OF AC PWR IN 15 GIVEN 11 HRS 15:11 = 1
XHOSACPOWER15:12	0.00E+000	RECOVERY OF AC PWR IN 15 GIVEN 12 HRS 15:12 = 1
XHOSACPOWER17	0.00E+000	RECOVERY OF AC PWR IN 17 HOURS 17 = 1
XHOSACPOWER17:12	0.00E+000	RECOVERY OF AC PWR IN 17 HRS GIVEN 12 17:12 = 1
XHOSACPOWER17:13	0.00E+000	RECOVERY OF AC PWR IN 17 HRS GIVEN 13 17:13 = 1
XHOSACPOWER17:P4	0.00E+000	RECOVERY OF AC PWR IN 17 GIVEN 0.4 HR 17:0.4 = 1
XHOSACPOWER18	0.00E+000	RECOVERY OF AC PWR IN 18 HOURS 18 HOURS = 1
XHOSACPOWER18:3	0.00E+000	RECOVERY OF AC PWR IN 18 GIVEN 3 HOURS 18:3 = 1
XHOSACPOWER18:4	0.00E+000	RECOVERY OF AC PWR IN 18 GIVEN 4 HOURS 18:4 = 1
XHOSACPOWER1:P26	0.00E+000	RECOVERY OF AC PWR IN 1 HOUR GIVN 0.26 1:P26 = 1
XHOSACPOWER1P3	0.00E+000	RECOVERY OF AC PWR IN 1.3 HOURS 1.3 HOURS = 1

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
XHOSACPOWER1P3P4	0.00E+000	RECOVERY OF AC PWR IN 1.3 GIVEN 0.4 HR 1.3:0.4 = 1
XHOSACPOWER1P4	0.00E+000	RECOVERY OF AC PWR IN 1.4 HOURS 1.4 HOURS = 1
XHOSACPOWER1P4P4	0.00E+000	RECOVERY OF AC PWR IN 1.4 GIVEN 0.4 HR 1.4:0.4 = 1
XHOSACPOWER1P7	0.00E+000	RECOVERY OF AC PWR IN 1.7 HOURS 1.7 HOURS = 1
XHOSACPOWER1P7P3	0.00E+000	RECOVERY OF AC PWR IN 1.7 GIVEN 0.3 HR 1.7:0.3 = 1
XHOSACPOWER2	0.00E+000	RECOVERY OF AC PWR IN 2 HOURS 2 HOURS = 1
XHOSACPOWER21	0.00E+000	RECOVERY OF AC PWR IN 21 HOURS 21 HOURS = 1
XHOSACPOWER21:13	0.00E+000	RECOVERY OF AC PWR IN 21 GIVEN 13 HRS 21:13 = 1
XHOSACPOWER23	0.00E+000	RECOVERY OF AC PWR IN 23 HOURS 23 HOURS = 1
XHOSACPOWER23:11	0.00E+000	RECOVERY OF AC PWR IN 23 GIVEN 11 HRS 23:11 = 1
XHOSACPOWER24	0.00E+000	RECOVERY OF AC PWR IN 24 HOURS 24 HOURS = 1
XHOSACPOWER241P3	0.00E+000	RECOVERY OF AC PWR IN 24 GIVEN 1.3 HRS 24:1.3 = 1
XHOSACPOWER241P7	0.00E+000	RECOVERY OF AC PWR IN 24 GIVEN 1.7 HRS 24:1.7 = 1
XHOSACPOWER24:1	0.00E+000	RECOVERY OF AC PWR IN 24 GIVEN 1 HOUR 24:1 = 1
XHOSACPOWER24:15	0.00E+000	RECOVERY OF AC PWR IN 24 GIVEN 15 HRS 24:15 = 1
XHOSACPOWER24:6	0.00E+000	RECOVERY OF AC PWR IN 24 GIVEN 6 HOURS 24:6 = 1
XHOSACPOWER24:7	0.00E+000	RECOVERY OF AC PWR IN 24 GIVEN 7 HRS 24:7 = 1
XHOSACPOWER3	0.00E+000	RECOVERY OF AC PWR IN 3 HOURS 3 HOURS = 1
XHOSACPOWER4	0.00E+000	RECOVERY OF AC PWR IN 4 HOURS 4 HOURS = 1
XHOSACPOWER4:2	0.00E+000	RECOVERY OF AC PWR IN 4 HOURS GIVEN 2 4:2 = 1
XHOSACPOWER4:3	0.00E+000	RECOVERY OF AC PWR IN 4 HOURS GIVEN 3 4:3 = 1
XHOSACPOWER4:P26	0.00E+000	RECOVERY OF AC PWR IN 4 HOURS GVN 0.26 4:P26 = 1
XHOSACPOWER4:P3	0.00E+000	RECOVERY OF AC PWR IN 4 HOURS GIVEN .3 4:P3 = 1
XHOSACPOWER4:P4	0.00E+000	RECOVERY OF AC PWR IN 4 HOURS GIVEN .4 4:P4 = 1
XHOSACPOWER5	0.00E+000	RECOVERY OF AC PWR IN 5 HOURS 5 HOURS = 1
XHOSACPOWER5:2	0.00E+000	RECOVERY OF AC PWR IN 5 HOURS GIVEN 2 5:2 = 1
XHOSACPOWER6	0.00E+000	RECOVERY OF AC PWR IN 6 HOURS 6 HOURS = 1
XHOSACPOWER6:2	0.00E+000	RECOVERY OF AC PWR IN 6 GIVEN 2 HOURS 6:2 = 1
XHOSACPOWER6:3	0.00E+000	RECOVERY OF AC PWR IN 6 GIVEN 3 HOURS 6:3 = 1
XHOSACPOWER6:4	0.00E+000	RECOVERY OF AC PWR IN 6 HOURS GIVEN 4 6:4 = 1
XHOSACPOWER7	0.00E+000	RECOVERY OF AC PWR IN 7 HOURS 7 HOURS = 1
XHOSACPOWER7:3	0.00E+000	RECOVERY OF AC PWR IN 7 GIVEN 3 HOURS 7:3 = 1
XHOSACPOWER8	0.00E+000	RECOVERY OF AC PWR IN 8 HOURS 8 HOURS = 1
XHOSACPOWER9	0.00E+000	RECOVERY OF AC PWR IN 9 HOURS 9 HOURS = 1
XHOSACPOWERP26	0.00E+000	RECOVERY OF AC PWR IN 0.26 HOURS 0.26 HOURS = 1
XHOSACPOWERP3	0.00E+000	RECOVERY OF AC PWR IN 0.3 HOURS 0.3 HOURS = 1
XHOSACPOWERP4	0.00E+000	RECOVERY OF AC PWR IN 0.4 HOURS 0.4 HOURS = 1
XHOSADSAC	0.00E+000	ADS ACCUMULATORS NOT CONSIDERED A SUCCESS
XHOSCNTNMNTSPRAY	0.00E+000	CONTAINMENT SPRAY SIGNAL PRESENT CS = 1; NO CS = 0
XHOSFPTTRIPSGNAL	0.00E+000	FEED PUMP TURBINE TRIP SIGNAL TRIP SIGNAL = 1
XHOSHIGHDRYWELL	1.00E+000	HIGH DRYWELL PRESSURE SIGNAL HDP = 1
XHOSLONGTERMEVNT	1.00E+000	LONG TERM EVENT LTE = 1
XHOSLOOP	1.00E+000	LOSS OF OFFSITE POWER
XHOSMFPSTARTSGNL	1.00E+000	MOTOR FEED PUMP START SIGNAL NO START SIGNAL = 1
XHOSNOAUTOADS	0.00E+000	ADS INHIBITED AUTO ADS = 1 NO AUTO ADS = 0
XHOSNOLOOP	0.00E+000	NO LOSS OF OFFSITE POWER
XHOSNONADSAC	0.00E+000	NON-ADS ACCUMUL NOT CONSIDERED A SUCCESS
XHOSNOSBO	0.00E+000	NO STATION BLACKOUT SBO = 0 NO SBO = 1
XHOSNOSTEAM	1.00E+000	POWER CONVERSION SYSTEM IS NOT AVAILABLE
XHOSRCICISOL	0.00E+000	STEAM SUPPLY FAILS DUE TO ISOLATION SIGNAL

Table 3.3.2-1 BASIC EVENT DATA BASE

Event	Point Est	Description
XHOSRCICROOMCLNG	0.00E+000	RCIC RM CLNG RQRD NO COOLING = 0 COOLING = 1
XHOSRPVL1	1.00E+000	PRV LOW WATER LEVEL 1 SIGNAL LEVEL 1 = 1
XHOSSBO	1.00E+000	STATION BLACKOUT SBO = 1 NO SBO = 0
Y17B	9.21E-001	COMPLEMENT TO Y17
ZERO	0.00E+000	BASIC EVENT SET TO 0.0

Table 3.3.3-1 Human Interaction Basic Events

Basic Event Id	Point Est	Description
ADHICPC5-1-ADS-A	3.79E-003	OPERATOR FAILS TO INHIBIT ADS ATWS W/ FEEDWATER
ADHICPC5-1-ADS-I	3.60E+000	OPERATOR FAILS TO INHIBIT ADS ATWS W/O FDW & IORV
ADHICPC5-1-ADS-L	3.60E-002	OPERATOR FAILS TO INHIBIT ADS ATWS W/ LOOP
ADHICPC5-1-ADS-O	7.20E+000	OPERATOR FAILS TO INHIBIT ADS ATWS W/O FEEDWATER
ADHICPEC2-ADS-LR	9.99E-005	FAILS TO RECOVER FROM RPV DEPRESS CD FAILURE
ADHICPEC2-ADS-R	1.00E-001	FAILS TO RECOVER FROM RPV DEPRESS CD FAILURE
ADHICPEC2-ADS-T	1.00E-003	FAILS TO EMERGENCY RPV DEPRESS TRANSIENT
ADHICPEC5-ADS-FL	7.00E-003	FAILS TO EMERGENCY RPV DEPRESS - ATWS W/ FDW & LVL CNTRL
ADHICPEC5-ADS-FX	1.40E-002	FAILS TO EMERGENCY RPV DEPRESS - ATWS W/ FDW & NO LVL CNT
CAHICPI008-4:2	1.00E-003	OPERATORS FAIL TO SHIFT CONDENSER AIR REMOVAL TO VP & AB
CCHICP	0.00E+000	OPERATOR FAILS TO BYPASS LOCA SIGNAL IN 4.5 HOURS
CCHICPSP47-5:4	0.00E+000	OPERATOR FAILS TO REALIGN CCCW LOOP C IN 4.5 HOURS
CDHICPPS2:1-XH11	1.00E-001	OPERATOR FAILS TO BYPASS THE RHR LOCA SIGNAL - XH11
CDHICPPS2:1-XH12	1.00E-001	OPERATOR FAILS TO BYPASS THE RHR LOCA SIGNAL - XH12
CDHICPPS2:1-XH1X	1.00E-003	OPERATOR FAILS TO BYPASS THE RHR LOCA SIGNAL - XH1X
CSHICPET-2:P-1	1.70E-002	OPERATOR FAILS TO INITIATE RHR CONTAINMENT SPRAY
CSHICPET-2:P-1-A	1.70E-002	OPERATOR FAILS TO INITIATE TRN A RHR CONTAINMENT SPRAY
CSHICPET-2:P-1-B	1.70E-002	OPERATOR FAILS TO INITIATE TRN B RHR CONTAINMENT SPRAY
CTHICPPS4:4-ALT	1.00E-001	OPERATOR FAILS TO ALIGN CONDENSATE XFER ALT INJECTION
CTHICPPS4:4-ALT6	1.00E-003	OPERATOR FAILS TO ALIGN CONDENSATE TRANSFER IN 6 HOURS
CTHICPPS4:4-ALTC	3.00E-001	OPERATOR FAILS TO ALIGN CONDENSATE XFER ALT INJECTION
CVHICPEPC-COM	1.00E-003	FAILS TO INITIATE CNTNMNT PRESS CNTRL RHR AND VENT
CVHICPEPC-FPCC	1.00E-001	FAILS TO INITIATE CNTNMNT PRESS CNTRL VENTING
CVHICPEPC-RHR	1.00E-002	FAILS TO INITIATE CNTNMNT PRESS CNTRL RHR
CVHICPEPC-RHR-E	1.00E-001	FAILS TO INITIATE CNTNMNT PRESS CNTRL RHR - S.P. CLNG
CVHICPPS7-RHR	1.00E-005	OPERATOR FAILS TO INITIATE RHR CONTAINMENT VENTING
CVHICPPS7:1-P-T	1.00E-010	OPERATOR FAILS TO PREPARE FOR RHR CNTNMNT VENT - TRAN
CVHICPPS7:3G41-T	9.99E-005	OPERATOR FAILS TO ALIGN FPCC FOR CNTNMNT VENT - TRAN
CVHICPPS7:4E12-T	9.99E-005	OPERATOR FAILS TO ALIGN RHR FOR CNTNMNT VENT - TRAN
CVHICRPS7:3G41-T	5.00E-002	OPERATOR IS UNABLE TO LOCALLY OPEN 1G41-F0145
DGHICPOS11-2:5	1.25E-003	OPERATOR FAILS TO INITIATE DIESEL GENERATORS
DGHICPOS11-2:5:A	1.25E-003	OPERATOR FAILS TO INITIATE DIV 1 D/G
DGHICPOS11-2:5:B	1.25E-003	OPERATOR FAILS TO INITIATE DIV 2 D/G
DGHIMASR43-4:1:A	0.00E+000	FAILURE TO RESTORE FOLLOWING MAINTENANCE
DGHIMASR43-4:1:B	0.00E+000	FAILURE TO RESTORE FOLLOWING MAINTENANCE
DHHICPOS11-2:5:C	1.25E-003	OPERATOR FAILS TO INITIATE DIV 3 D/G
DHHIMASE22B-4:1	0.00E+000	FAILURE TO RESTORE FOLLOWING MAINTENANCE
ECHICPSP42-4:2	5.00E-002	OPERATOR FAILS TO CLOSE VALVE OP42-F0150A(B)
ECHICPSP42-4:2A	5.00E-002	OPERATOR FAILS TO CLOSE VALVE OP42-F0150A
ECHICPSP42-4:2B	5.00E-002	OPERATOR FAILS TO CLOSE VALVE OP42-F0150B
ECHICPSP42-4PMP	5.00E-002	OPERATOR FAILS TO INITIATE PUMP 1P42-C0001A(B)
ECHICPSP42-4PMPA	5.00E-002	OPERATOR FAILS TO INITIATE PUMP 1P42-C0001A
ECHICPSP42-4PMPB	5.00E-002	OPERATOR FAILS TO INITIATE PUMP 1P42-C0001B
ECHIMASP42-4:1A	0.00E+000	ECC TRAIN A NOT RESTORED FOLLOWING MAINTENANCE
ECHIMASP42-4:1B	0.00E+000	ECC TRAIN B NOT RESTORED FOLLOWING MAINTENANCE
ERHICPPS4:2-ESW	1.00E-002	OPERATORS FAIL TO ALIGN 11 VLVS FOR RPV INJECTION
ESHIMASP45-4:1A	0.00E+000	ESW TRAIN A NOT RESTORED FOLLOWING MAINTENANCE
ESHIMASP45-4:1B	0.00E+000	ESW TRAIN B NOT RESTORED FOLLOWING MAINTENANCE
ESHIMASP45-4:1C	0.00E+000	ESW TRAIN C NOT RESTORED FOLLOWING MAINTENANCE
FPHICPPS4:2-DD-O	3.00E-002	OPERATORS FAIL TO MAINTAIN FUEL OIL FOR DIESEL FIRE PMP

Table 3.3.3-1 Human Interaction Basic Events (continued)

Basic Event Id	Point Est	Description
FPHICPPS4:2FP-LE	5.00E-002	OPERATOR FAILS TO ALIGN VLVS FOR LATE FP ALT INJ < 3 HRS
FPHICPPS4:2FP-LL	5.00E-003	OPERATOR FAILS TO ALIGN VLVS FOR LATE FP ALT INJ > 3 HRS
FPHICPPS4:2RCIC1	3.00E-001	FAIL TO ALIGN FP AFTER RCIC FAILS DUE SUPP POOL TEMP
FPHICPPS4:2RCIC2	1.00E-001	FAIL TO ALIGN FP AFTER RHR FAILS DUE TO MCC TEMP
FPHICPPS4:2RCIC3	1.00E-002	FAIL TO ALIGN FP AFTER HPCS FAILS DUE TO MCC TEMP
FPHICPPS4:2RCIC4	1.00E-001	FAIL TO ALIGN FAST FIRE PROTECTION ALTERNATE INJECTION
FWHICPEC5-2:3LCS	1.00E-002	OPER FAILS TO CNTRL RPV LEVEL AT TAF W/ FW DURING IORV
FWHICPEC5-3:2	1.00E-002	OPER FAILS TO CONTROL RPV LEVEL AT TAF
FWHICPEL-2-FDW-L	1.00E-002	OPERATOR FAILS TO REOPEN MFP CONTROL VALVES FOR T3C-C
FWHICPEL-2-FDW-V	5.00E-003	OPER FAILS TO REOPN MFP CNTRL VALVES OR DEPRESSURIZE RPV
FWHICPSN27-4:1IA	1.20E-001	OPER FAILS TO CNTRL RX FEED BOOSTER PMP DURING LOSS OF IA
FWHICPSN27-4:1IL	5.00E-003	OPER FAILS TO CNTRL RX FEED BOOSTER PMP LOSS OF IA > 2 HRS
HIHICPEC5-3:2-F	1.00E-003	FAILS TO RESTR RHR A/B/LPCS AND CONTRL AT TAF W/ FDW
HIHICPEC5-3:2-S	1.00E-003	FAILS TO RESTR RHR A/B/LPCS AND CNTRL AT TAF W/O FDW
HIHICPEC5-5-CRIT	2.00E-003	OPER FAILS TO CNTRL RPV LEVEL AND FLUSHES BORON
HIHICPOR10-4:0-I	5.00E-002	OPERATOR FAILS TO CLOSE FPCC OUTBOARD VALVE 1G41-F0145
HIHICPOR10-4:3-B	1.00E-002	OPER FAILS TO X-TIE UNIT 1 AND 2 BATT AND LOAD SHED
HIHICPOR10-4:3-D	2.00E-003	OPER FAILS TO OPEN DIV 3 SWITCHGEAR ROOM DOORS
HIHICPOR10-XTIE	5.00E-003	OPERATOR FAILS TO CROSSTIE DIV 3 BUS TO DIV 2 BUS
HPHICPEL-1	1.25E-003	OPERATOR FAILS TO INITIATE HIGH PRESSURE INJECTION
HPHICPSE22-5:0	5.00E-002	OPERATOR FAILS TO CONTROL MIN FLOW VALVE 1E22-F0012
HPHICPSE22-5:2	5.00E-002	OPER FAILS TO XFER TO SUPR POOL WITH 1E22-F015
HPHIMASE22-4:1	0.00E+000	FAILURE TO RESTORE HPCS AFTER MAINTENANCE
IAHICPSP51-4:2	5.00E-002	OPERATOR FAILS TO REPOSITION NCC LUBE OIL CLR OUTLET VLV
IAHICRSP52-7:2	1.00E-001	OPERATOR FAILS TO OVERRIDE ISOLATION SIGNAL
IAHIMASP51-4:1:2	1.00E-003	FAILURE TO RESTORE FOLLOWING MAINTENANCE
LCHICPEL-1-LPCI	1.00E-003	OPERATOR FAILS TO INITIATE LPCI B
LCHICPSE12-5:1	5.00E-002	OPERATOR FAILS TO CONTROL MIN FLOW VALVE 1E12-F0064A
LCHIMASE12-4:1A	0.00E+000	FAILURE TO RESTORE TRAIN A LPCI FOLLOWING MAINT
LCHIMASE12-4:1B	0.00E+000	FAILURE TO RESTORE TRAIN B LPCI FOLLOWING MAINT
LCHIMASE12-4:1C	0.00E+000	FAILURE TO RESTORE TRAIN C LPCI FOLLOWING MAINT
LPHICPEL-1	1.00E-003	OPERATOR FAILS TO INITIATE LOW PRESSURE INJECTION
LPHICPEL-1-LPCS	1.00E-003	OPERATOR FAILS TO INITIATE LPCS
LPHICPSE21-5:1	5.00E-002	OPERATOR FAILS TO CONTROL MIN FLOW VALVE 1E21-F0011
LPHIMASE21-4:1	0.00E+000	FAILURE TO RESTORE LPCS AFTER MAINTENANCE
MCHICRRECVRD4	0.00E+000	MCC, SWTCHGR, & MISC ELECT AREAS RECVRD IN 4 HOURS
MCHICRRECVRD9	0.00E+000	MCC, SWTCHGR, & MISC ELECT AREAS RECVRD IN 9 HOURS
MLHICPE32	1.00E-002	OPERATOR FAILS TO INITIATE MSIV LEAKAGE CONTROL
NSHICPEC5-2-L1	1.00E-001	OPERATOR FAILS TO BYPASS MSIV LEVEL 1 ISOLATION FOR T3C-C
NSHICPEC5-2-L1T3	1.00E+000	OPERATOR FAILS TO BYP MSIV LVL 1 ISOL FOR T3A-C OR T3B-C
RCHICPEL-2-CST-S	5.00E-002	OPERATOR FAILS TO PREVENT SUCTION SHIFT TO SUPP POOL
RCHICPS51-LDTRIP	5.00E-002	FAILURE TO RECOVER ISOL SIGNAL ON HIGH STEAM TUNNEL TEMP
RCHICPSE51-5:1	5.00E-002	OPERATOR FAILS TO PERFORM RCIC SUCTION SHIFT
RCHIMASE51-4:1	0.00E+000	FAILURE TO RESTORE RCIC FOLLOWING MAINTENANCE
RPHICPERC-1:Q-2	1.00E-00[illegible]	OPERATOR FAILS TO SCRAM REACTOR
SAHICPSP51-4:2	5.00E-002	OPERATOR FAILS TO REPOSITION NCC LUBE OIL CLR OUTLET VLV
SAHIMASP51-4:1:2	1.00E-003	FAILURE TO RESTORE FOLLOWING MAINTENANCE
SCHICPSE12-5:3	1.09E-004	OPERATOR FAILS TO ALIGN RHR TO SUPP POOL COOLING
SCHICPSE12-5:3A	1.09E-004	OPERATOR FAILS TO ALIGN RHR TRAIN A TO SUPP POOL CLNG
SCHICPSE12-5:3B	1.09E-004	OPERATOR FAILS TO ALIGN RHR TRAIN B TO SUPP POOL CLNG

Table 3.3.3-1 Human Interaction Basic Events (continued)

Basic Event Id	Point Est	Description
SIHICPSP57-7:1	5.00E-002	OPERATORS FAIL TO CONNECT AIR CYLINDERS
SIHICPSP57-7:1:A	5.00E-003	OPERATORS FAIL TO CONNECT AIR CYLINDERS
SIHICPSP57-7:1:B	5.00E-003	OPERATORS FAIL TO CONNECT AIR CYLINDERS
SIHIMASP57-4:0:A	0.00E+000	FAILURE TO RESTORE FOLLOWING MAINTENANCE
SIHIMASP57-4:0:B	0.00E+000	FAILURE TO RESTORE FOLLOWING MAINTENANCE
SLHICPEQ-6-RPVLC	5.00E-002	OPER FAILS TO CNTRL RPV LVL & MAINTAIN BORON INVENTORY
SLHICPEQ-6-SLC1	1.25E-003	OPERATOR FAILS TO INITIATE SLC 1 PUMP INJECTION
SLHICPEQ-6-SLCX	1.00E+000	OPERATOR FAILS TO INITIATE SLC LEVEL CONTROL FAILS
SLHICREQ-6-SLCR	1.00E-001	OPER FAILS TO INITIATE SLC GIVEN CORE DAMAGE
SLHIMA	0.00E+000	FAILURE TO RESTORE SLC FOLLOWING MAINTENANCE/TEST
SPHICPPS4:5SPCU	1.00E+000	OPERATORS FAIL TO ALIGN SUPP POOL C/U ALTERNATE INJECTION
SPHICPPS4:5SPCUL	5.00E-002	OPERATORS FAIL TO ALIGN SUP POOL C/U ALT INJECTION(LATE)
TBHICPSM35	5.00E-002	OPERATOR FAILS TO START STANDBY FAN

Table 3.3.4-1

CCF Probabilities for Various CCF Events

Component	Means	s	EF	Median	95%
MOVs	9.234×10^{-5}	1.82	20	1.762×10^{-5}	3.524×10^{-4}
ESW Pumps	3.413×10^{-4}	0.81	3.8	2.458×10^{-4}	9.340×10^{-4}
ECC Pumps	1.291×10^{-4}	1.15	6.6	6.664×10^{-5}	7.664×10^{-4}
DGs	3.670×10^{-4}	1.85	21	6.629×10^{-5}	1.392×10^{-3}

Table 3.3.5-1

Function Unavailabilities

Function Equation	Function Description	Function Probability
A	Large LOCA	1.00×10^{-4}
B101	Div 1 and 2 D/G fail	8.36×10^{-3}
C01	Rx scram - control rods in	1.00×10^{-5}
C101	SLC injects with 1 of 2 pumps - LOOP	2.35×10^{-2}
C105	SLC injects with 1 of 2 pumps	4.78×10^{-3}
C107	SLC injects with 1 of 2 pumps	4.78×10^{-3}
C108	SLC injects with 1 of 2 pumps	1.00
C109	SLC injects with 1 of 2 pumps	4.52×10^{-2}
C110	SLC injects with 1 of 2 pumps	1.03×10^{-1}
C111	SLC injects with 1 of 2 pumps	4.52×10^{-3}
C112	SLC injects with 1 of 2 pumps - LOOP	7.25×10^{-2}
C114	SLC injects with 1 of 2 pumps	5.46×10^{-2}
CC01	Control Complex Chilled Water recovery	0.00
CV01	Core not vulnerable to damage - HPCS	5.70×10^{-1}
CV02	Core not vulnerable to damage - Low Press ECCS	4.00×10^{-1}
CV03	Core not vulnerable to damage - Feedwater	1.80×10^{-1}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
CV04	Core not vulnerable to damage - Injctn outside aux bldng	1.40×10^{-1}
HI01	Operator actions taken to extend HPCS operation SBO	1.30×10^{-2}
HV01	MCC, Swtchgr, & Misc Elect Areas HVAC - LOOP	0.00
HV02	MCC, Swtchgr, & Misc Elect Areas HVAC - LOOP	0.00
HV03	MCC, Swtchgr, & Misc Elect Areas HVAC - LOOP	0.00
HV04	MCC, Swtchgr, & Misc Elect Areas HVAC - LOOP	0.00
I01	FPCC isolation shut by operator or diesel	3.44×10^{-2}
LC02	Level Control ATWS	1.00×10^{-2}
LI01	Late injection	1.40×10^{-1}
LI02	Late injection	2.85×10^{-5}
LI03	Late injection	2.85×10^{-5}
LI04	Late injection	1.40×10^{-5}
LI09	Late injection	1.60×10^{-1}
LI10	Late injection	1.44×10^{-1}
LI12	Late injection	1.95×10^{-2}
LI14	Late injection	1.99×10^{-1}
LI15	Late injection	5.43×10^{-1}
LI16	Late injection	1.60×10^{-1}
LI17	Late injection	1.12×10^{-1}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
LI19	Late injection	2.36×10^{-1}
LI21	Late injection	2.13×10^{-2}
LI22	Late injection	2.41×10^{-4}
LI23	Late injection	2.88×10^{-4}
LI24	Late injection	7.04×10^{-1}
LI25	Late injection	1.57×10^{-1}
LI26	Late injection	2.81×10^{-1}
LI27	Late injection	1.27×10^{-1}
LI29	Late injection	3.28×10^{-3}
LI30	Late injection	3.76×10^{-1}
LI31	Late injection	1.41×10^{-1}
LI32	Late injection	1.06×10^{-1}
LI33	Late injection	1.14×10^{-1}
LI34	Late injection	1.06×10^{-1}
LI35	Late injection	2.41×10^{-1}
LI36	Late Injection	6.54×10^{-3}
LI37	Late injection	2.16×10^{-2}
LI38	Late injection	7.94×10^{-2}
LI39	Late injection	1.84×10^{-1}
LI40	Late injection	1.52×10^{-1}
LI41	Late injection	4.14 X 10-1
LI42	Late Injection	4.65×10^{-1}
LI43	Late injection	2.07×10^{-1}
LI48	Late injection	1.80×10^{-1}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
LI49	Late injection	2.04×10^{-1}
LI50	Late injection	4.40×10^{-1}
M01	SRVs open	1.00×10^{-4}
P101	One SRV opens and recloses	1.60×10^{-2}
P102	One SRV opens and recloses w/ PCS avail	1.60×10^{-2}
P201	Two SRVs open and reclose	1.60×10^{-3}
P202	Two SRVs open and reclose w/PCS avail	1.60×10^{-3}
Q01	Power conversion system remains available	3.51×10^{-3}
Q03	Power conversion system remains available	1.00
Q04	Power conversion system remains available	1.00
Q05	Power conversion system remains available	1.00×10^{-1}
R01	Restoration of AC power before core damage	1.23×10^{-2}
R02	Restoration of AC power before core damage	1.96×10^{-2}
R03	Restoration of AC power before core damage	1.23×10^{-2}
R04	Restoration of AC power before core damage	8.23×10^{-2}
R05	Restoration of AC power before core damage	1.96×10^{-2}
R07	Restoration of AC power before core damage	5.29×10^{-1}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
R08	Restoration of AC power before core damage	1.96×10^{-2}
R09	Restoration of AC power before core damage	1.74×10^{-1}
R101	Restoration of AC power before core damage	8.23×10^{-2}
R102	Restoration of AC power before core damage	5.29×10^{-1}
R103	Restoration of AC power before core damage	6.16×10^{-1}
R201	Restoration of AC power before core damage	1.51×10^{-2}
R202	Restoration of AC power before core damage	8.98×10^{-1}
R203	Restoration of AC power before core damage	1.40×10^{-1}
R204	Restoration of AC power before core damage	1.20×10^{-1}
R301	Restoration of ac power before RPV failure	3.81×10^{-1}
R302	Restoration of ac power before RPV failure	3.26×10^{-1}
R303	Restoration of ac power before RPV failure	5.97×10^{-1}
R304	Restoration of ac power before RPV failure	4.19×10^{-1}
R305	Restoration of ac power before RPV failure	4.33×10^{-1}
R306	Restoration of ac power before RPV failure	2.66×10^{-1}
R307	Restoration of ac power before RPV failure	6.96×10^{-1}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
R308	Restoration of ac power before RPV failure	4.50×10^{-1}
R309	Restoration of ac power before RPV failure	5.75×10^{-3}
R310	Restoration of ac power before RPV failure	2.66×10^{-1}
R311	Restoration of ac power before RPV failure	2.00×10^{-1}
R312	Restoration of ac power before RPV failure	2.66×10^{-1}
R313	Restoration of ac power before RPV failure	2.66×10^{-1}
R314	Restoration of ac power before RPV failure	4.19×10^{-1}
R315	Restoration of ac power before RPV failure	5.03×10^{-1}
R316	Restoration of ac power before RPV failure	5.44×10^{-1}
R317	Restoration of ac power before RPV failure	8.37×10^{-2}
R318	Restoration of ac power before RPV failure	3.29×10^{-1}
R319	Restoration of ac power before RPV failure	6.22×10^{-2}
R320	Restoration of ac power before RPV failure	1.09×10^{-2}
R321	Restoration of ac power before RPV failure	5.59×10^{-2}
R323	Restoration of ac power before RPV failure	5.97×10^{-1}
R324	Restoration of ac power before RPV failure	4.32×10^{-1}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
R326	Restoration of ac power before RPV failure	4.67×10^{-1}
R329	Restoration of ac power before RPV failure	2.04×10^{-1}
R401	Restoration of ac power before cntnmnt failure	2.29×10^{-2}
R402	Restoration of ac power before cntnmnt failure	5.50×10^{-3}
R403	Restoration of ac power before cntnmnt failure	1.60×10^{-1}
R404	Restoration of ac power before cntnmnt failure	2.29×10^{-2}
R405	Restoration of ac power before cntnmnt failure	4.92×10^{-3}
RPT01	Recirculation pump trip	1.00×10^{-4}
S1	Intermediate LOCA	3.00×10^{-4}
S2	Small LOCA	3.00×10^{-3}
T1	Loss of Offiste Power	6.09×10^{-2}
T2	Transient with loss of PCS	1.62
T3A	Transient with PCS available	4.51
T3B	Loss of feedwater	$7.60 \times 10^{-[illegible]}$
T3C	Inadvertent open relief valve	1.40×10^{-1}
TCW	Loss of CCCW	8.51×10^{-2}
TIA	Loss of instrument	9.20×10^{-2}
TSW	Loss of Service Water	1.00×10^{-3}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
U101	High Pressure Core Spray	3.94×10^{-2}
U102	High Pressure Core Spray	3.91×10^{-2}
U104	High Pressure Core Spray	3.73×10^{-2}
U105	High Pressure Core Spray	3.73×10^{-2}
U106	High Pressure Core Spray	3.73×10^{-2}
U109	High Pressure Core Spray - SBO	1.05×10^{-1}
U110	High Pressure Core Spray	3.91×10^{-2}
U111	High Pressure Core Spray - LOOP & MCC HVAC not modeled	1.04×10^{-1}
U112	High Pressure Core Spray - TCW	3.73×10^{-2}
U113	High Pressure Core Spray - TCW	3.73×10^{-2}
U114	High Pressure Core Spray - Flooding	3.73×10^{-2}
U202	Reactor Core Isolation Cooling - LOOP	2.18×10^{-1}
U203	Reactor Core Isolation Cooling	1.74×10^{-1}
U204	Reactor Core Isolation Cooling	1.74×10^{-1}
U205	Reactor Core Isolation Cooling	1.74×10^{-1}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
U207	Reactor Core Isolation Cooling - SBO	1.25×10^{-1}
U208	Reactor Core Isolation Cooling - Loss of SW	2.24×10^{-1}
U209	Reactor Core Isolation Cooling - Small LOCA	6.74×10^{-1}
U210	Reactor Core Isolation Cooling - Flooding	1.66×10^{-1}
U301	Condensate Feedwater	1.46×10^{-3}
U302	Condensate Feedwater	9.53×10^{-3}
U303	Condensate Feedwater	1.82×10^{-3}
U304	Condensate Feedwater	9.53×10^{-3}
U305	Condensate Feedwater	9.45×10^{-4}
U306	Condensate Feedwater ATWS	1.45×10^{-2}
U307	Condensate Feedwater ATWS	5.95×10^{-3}
U308	Condensate Feedwater ATWS	1.09×10^{-2}
U309	Condensate Feedwater Loss of Instrument Air	1.45×10^{-2}
U310	Condensate Feedwater w/ MFP & RPV level 8 trip	9.53×10^{-3}
V01	Low Pressure Makeup	[illegible]3 $\times 10^{-3}$
V03	Low Pressure Makeup	2.93×10^{-3}
V04	Low Pressure Makeup ATWS - LOOP	1.85×10^{-2}
V05	Low Pressure Makeup ATWS w/Feedwater	5.06×10^{-3}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
V06	Low Pressure Makeup ATWS w/Feedwater	5.06×10^{-3}
V07	Low Pressure Makeup ATWS w/o Feedwater	5.06×10^{-3}
V08	Low Pressure Makeup ATWS w/o Feedwater	5.06×10^{-3}
V11	Low Pressure Makeup Station Blackout	8.55×10^{-2}
V12	Low Pressure Makeup Station Blackout	2.28×10^{-2}
V13	Low Pressure Makeup Station Blackout	1.77×10^{-1}
V15	Low Pressure Makeup LOOP & MCC HVAC not modeled	1.48×10^{-2}
V17	Low Pressure Makeup SBO & MCC HVAC not modeled	8.53×10^{-2}
VA01	Alternate Low Pressure Makeup	1.10×10^{-1}
VA02	Alternate Low Pressure Makeup	3.10×10^{-1}
VA03	Alternate Low Pressure Makeup	1.11×10^{-1}
VA04	Alternate Low Pressure Makeup - Fire Water	2.66×10^{-1}
VA05	Alternate Low Pressure Makeup - Rx Feed Bstr Pmps	1.24×10^{-1}
VA06	Alternate low pressure makeup - after HPCS success	1.58×10^{-2}
VA07	Alternate low pressure makeup - SBO	8.54×10^{-2}
VA08	Alternate low pressure makeup - LOOP	2.67×10^{-1}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
VA09	Alternate low pressure makeup - LOOP	4.67×10^{-1}
VA10	Alternate low pressure makeup - SBO	6.45×10^{-1}
VA11	Alternate low pressure makeup - LOOP	2.67×10^{-1}
VA12	Alternate low pressure makeup - LOOP	2.08×10^{-2}
VA13	Alternate low pressure makeup	7.31×10^{-2}
VA14	Alternate low pressure makeup	4.96×10^{-4}
VA15	Alternate low pressure makeup	4.56×10^{-1}
VA16	Alternate low pressure makeup	2.58×10^{-1}
VA17	Alternate low pressure makeup	6.40×10^{-1}
VA18	Alternate low pressure makeup	7.32×10^{-1}
VA19	Alternate low pressure makeup	1.10 X 10^-[illegible]
VA20	Alternate low pressure makeup	2.56×10^{-1}
VP01	Prevent vessel overfill ATWS	2.00×10^{-3}
W01	Long Term Containment Heat Removal w/RHR	6.94×10^{-3}
W03	Long Term Containment Heat Removal w/RHR Loss of offsite power	2.30×10^{-2}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
W04	Long Term Containment Heat Removal w/RHR Loss of offsite power	2.37×10^{-2}
W05	Long Term Containment Heat Removal w/RHR	6.53×10^{-3}
W06	Long Term Containment Heat Removal w/RHR	6.94×10^{-3}
W07	Long Term Containment Heat Removal w/RHR	6.53×10^{-3}
W08	Long Term Containment Heat Removal w/RHR LOOP/ATWS	2.57×10^{-2}
W10	Long Term Containment Heat Removal w/RHR ATWS	7.17×10^{-3}
W11	Long Term Containment Heat Removal w/RHR ATWS	7.17×10^{-3}
W12	Long Term Containment Heat Removal w/RHR Station Blackout	1.96×10^{-2}
W13	Long Term Containment Heat Removal w/RHR Station Blackout	8.96×10^{-2}
W14	Long Term Containment Heat Removal w/RHR Station Blackout	2.69×10^{-2}
W15	Long Term Containment Heat Removal w/RHR Station Blackout	1.96×10^{-2}
W18	Long Term Containment Heat Removal with RHR	6.38×10^{-3}
W19	Long Term Containment Heat Removal with RHR Station Blackout	2.69×10^{-2}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
W20	Long Term Containment Heat Removal with RHR Station Blackout	2.69×10^{-2}
W21	Long Term Containment Heat Removal with RHR Station Blackout	1.81×10^{-1}
W23	Long Term Containment Heat Removal with RHR SBO & MCC HVAC not modeled	2.18×10^{-2}
W24	Long Term Containment Heat Removal with RHR SBO & MCC HVAC not modeled	8.94×10^{-2}
W25	Long Term Containment Heat Removal with RHR SBO & MCC HVAC not modeled	9.05×10^{-1}
W26	Long Term Containment Heat Removal with RHR SBO & MCC HVAC not modeled	5.36×10^{-1}
W27	Long Term Containment Heat Removal with RHR SBO & MCC HVAC not modeled	1.47×10^{-1}
W28	Long Term Containment Heat Removal with RHR Station Blackout	5.36×10^{-1}
W40	Long Term Containment Heat Removal with RHR Station Blackout	5.23×10^{-1}
W41	Long Term Containment Heat Removal with RHR SBO & MCC HVAC not modeled	1.27×10^{-1}
WC03	Containment Spray	4.33×10^{-2}
WC04	Containment Spray	3.37×10^{-2}
WC05	Containment Spray	1.27×10^{-2}
WC06	Containment Spray	1.32×10^{-2}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
WC07	Containment Spray	1.27×10^{-2}
WC08	Containment Spray	4.60×10^{-2}
WC10	Containment Spray	2.62×10^{-2}
WC11	Containment Spray	2.62×10^{-2}
WC13	Containment Spray	1.09×10^{-1}
WC16	Containment Spray	2.58×10^{-2}
WC18	Containment Spray	1.27×10^{-2}
WC20	Containment Spray	4.60×10^{-2}
WC23	Containment Spray	2.81×10^{-2}
WC24	Containment Spray	9.58×10^{-2}
WC26	Containment Spray	5.55×10^{-1}
WC28	Containment Spray	5.55×10^{-1}
WC30	Containment Spray	4.45×10^{-1}
WC31	Containment Spray	7.22×10^{-1}
WC32	Containment Spray	4.93×10^{-2}
WC33	Containment Spray	4.76×10^{-1}
WC34	Containment Spray	3.19×10^{-2}
WC35	Containment Spray	2.92×10^{-1}
WC36	Containment Spray	2.92×10^{-1}
WC37	Containment Spray	4.93×10^{-1}
WC38	Containment Spray	5.29×10^{-1}
WC39	Containment Spray	1.09×10^{-1}
WC40	Containment Spray	6.42×10^{-1}
WC42	Containment Spray	3.55×10^{-1}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
WC43	Containment Spray	4.07×10^{-1}
WC44	Containment Spray	5.70×10^{-1}
WC45	Containment Spray	8.85×10^{-2}
WC46	Containment Spray	2.30×10^{-1}
WC47	Containment Spray	4.50×10^{-1}
WC48	Containment Spray	4.58×10^{-1}
WC49	Containment Spray	3.07×10^{-2}
WC50	Containment Spray	1.32×10^{-2}
WC51	Containment Spray	3.16×10^{-2}
WC52	Containment Spray	3.67×10^{-2}
WC53	Containment Spray	2.92×10^{-1}
WS01	Suppression Pool Cooling SBO & MCC HVAC not modeled	9.10×10^{-2}
WS02	Suppression Pool Cooling LOOP & MCC HVAC not modeled	2.50×10^{-2}
WS03	Suppression Pool Cooling	2.53×10^{-2}
WS04	Suppression Pool Cooling	2.60×10^{-2}
WS05	Suppression Pool Cooling	7.88×10^{-3}
WS06	Suppression Pool Cooling	8.31×10^{-3}
WS07	Suppression Pool Cooling	7.88×10^{-3}
WS18	Suppression Pool Cooling	7.88×10^{-3}
WS23	Suppression Pool Cooling	2.32×10^{-2}
WS29	Suppression Pool Cooling	2.44×10^{-2}
WS50	Suppression Pool Cooling	8.31×10^{-3}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
X01	Emergency RPV Depressurization	8.22×10^{-6}
X02	Emergency RPV Depressurization	2.39×10^{-5}
X03	Emergency RPV Depressurization	8.22×10^{-6}
X05	Emergency RPV Depressurization - ATWS	5.03×10^{-3}
X06	Emergency RPV Depressurization - ATWS	7.01×10^{-3}
X07	Emergency RPV Depressurization - ATWS	1.40×10^{-2}
X08	Emergency RPV Depressurization - ATWS	7.01×10^{-3}
X09	Emergency RPV Depressurization - ATWS	1.40×10^{-2}
X10	Emergency RPV Depressurization - SBO	2.52×10^{-5}
X13	Emergency RPV Depressurization - SBO	2.52×10^{-5}
X14	Emergency RPV Depressurization - ATWS	5.01×10^{-3}
X15	Emergency RPV Depressurization Loss of Inst Air	8.22×10^{-6}
X16	Emergency RPV Depressurization - ATWS Loss of Inst Air	5.01×10^{-3}
X18	Emergency RPV Depressurization	2.52×10^{-5}
X19	Emergency RPV Depressurization - SBO	2.52×10^{-5}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
X20	Emergency RPV Depressurization - LOOP	2.39×10^{-5}
X21	Emergency RPV Depressurization - SBO	2.52×10^{-5}
X22	Emergency RPV Depressurization - LOOP	1.00×10^{-2}
X23	Emergency RPV Depressurization - SBO	2.52×10^{-5}
X24	Emergency RPV Depressurization - SBO	1.00×10^{-2}
X25	Emergency RPV Depressurization - SBO	2.37×10^{-5}
X26	Emergency RPV Depressurization - SBO	5.26×10^{-5}
X27	Emergency RPV Depressurization - ATWS	1.40×10^{-2}
X29	Emergency RPV Depressurization - SBO	1.03×10^{-3}
X30	Emergency RPV Depressurization - SBO	2.52×10^{-5}
X31	Emergency RPV Depressurization - SBO	1.02×10^{-3}
X32	Emergency RPV Depressurization - SBO	1.00×10^{-2}
X33	Emergency RPV Depressurization - SBO	1.00×10^{-1}
X34	Emergency RPV Depressurization - SBO	1.24×10^{-4}
XP01	ADS inhibit - ATWS	3.60×10^{-2}
XP02	ADS inhibit - ATWS	3.64×10^{-2}
XP03	ADS inhibit - ATWS	3.79×10^{-3}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
XP04	ADS inhibit - ATWS	3.64×10^{-2}
XP05	ADS inhibit - ATWS	3.60
Y01	Long Term Containment Heat Removal w/Venting	1.08×10^{-4}
Y03	Long Term Containment Heat Removal w/Venting LOOP	1.36×10^{-3}
Y04	Long Term Containment Heat Removal w/Venting LOOP	1.36×10^{-3}
Y05	Long Term Containment Heat Removal w/Venting	1.08×10^{-4}
Y06	Long Term Containment Heat Removal w/Venting	1.08×10^{-4}
Y07	Long Term Containment Heat Removal w/Venting	1.08×10^{-4}
Y08	Long Term Containment Heat Removal w/Venting ATWS	1.82×10^{-2}
Y10	Long Term Containment Heat Removal w/Venting ATWS	6.27×10^{-4}
Y11	Long Term Containment Heat Removal w/Venting ATWS	6.27×10^{-4}
Y12	Long Term Containment Heat Removal w/Venting ATWS	6.27×10^{-4}
Y13	Long Term Containment Heat Removal w/Venting Station Blackout	1.24×10^{-2}
Y14	Long Term Containment Heat Removal w/Venting Station Blackout	8.24×10^{-2}
Y15	Long Term Containment Heat Removal w/Venting Station Blackout	1.96×10^{-2}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
Y16	Long Term Containment Heat Removal w/Venting Station Blackout	1.23×10^{-2}
Y17	Long Term Containment Heat Removal w/Venting Station Blackout	7.83×10^{-2}
Y18	Long Term Containment Heat Removal w/Venting Station Blackout	1.96×10^{-2}
Y19	Long Term Containment Heat Removal w/Venting	7.70×10^{-5}
Y20	Long Term Containment Heat Removal w/Venting Station Blackout	1.97×10^{-2}
Y21	Long Term containment Heat Removal w/venting Station Blackout	1.74×10^{-1}
Y22	Long Term Containment Heat Removal w/Venting SBO	1.51×10^{-2}
Y23	Long Term Containment Heat Removal w/Venting SBO	8.23×10^{-2}
Y24	Long Term Containment Heat Removal w/Venting SBO	8.58×10^{-1}
Y25	Long Term Containment Heat Removal w/Venting SBO	5.29×10^{-1}
[illegible]	Long Term Containment Heat Removal w/Venting SBO	1.40×10^{-1}
Y27	Long Term Containment Heat Removal w/Venting SBO	5.29×10^{-1}
Y28	Long term Containment Heat Removal w/Venting SBO	4.19×10^{-1}
Y29	Long Term Containment Heat Removal w/Venting SBO	6.96×10^{-1}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
Y30	Long Term Containment Heat Removal w/Venting SBO	2.30×10^{-2}
Y31	Long term Containment Heat Removal w/Venting SBO	4.50×10^{-1}
Y32	Long Term Containment Heat Removal w/Venting SBO	5.60×10^{-3}
Y34	Long Term Containment Heat Removal w/Venting SBO	2.66×10^{-1}
Y35	Long Term Containment Heat Removal w/Venting SBO	2.66×10^{-1}
Y36	Long Term Containment Heat Removal w/Venting SBO	4.6[illegible] X 10^-[illegible]
Y37	Long Term Containment Heat Removal w/Venting SBO	5.03×10^{-1}
Y38	Long Term Containment Heat Removal w/Venting SBO	8.38×10^{-2}
Y39	Long Term Containment Heat Removal w/Venting SBO	6.16×10^{-1}
Y40	Long term Containment Heat Removal w/Venting SBO	1.20×10^{-1}
Y41	Long term Containment Heat Removal w/Venting	3.29×10^{-1}
Y42	Long term Containment Heat Removal w/Venting	3.81×10^{-1}
Y43	Long term Containment Heat Removal w/Venting	5.44×10^{-1}
Y46	Long term Containment Heat Removal w/Venting	6.23×10^{-2}
Y47	Long term Containment Heat Removal w/Venting	2.04×10^{-1}
Y48	Long term Containment Heat Removal w/Venting	5.02×10^{-3}

Table 3.3.5-1 continued

Function Unavailabilities

Function Equation	Function Description	Function Probability
Y49	Long term Containment Heat Removal w/Venting	5.97×10^{-1}
Y51	Long term Containment Heat Removal w/Venting	1.10×10^{-2}
Y52	Long term Containment Heat Removal w/Venting	4.33×10^{-1}
Y53	Long term Containment Heat Removal w/Venting	4.32×10^{-1}
Y54	Long term Containment Heat Removal w/Venting	5.85×10^{-3}
Y56	Long term Containment Heat Removal w/Venting	2.66×10^{-1}

Table 3.3.7-1

RESULTS OF INTERNAL FLOODING LEVEL ONE
ANALYSIS FOR PERRY NUCLEAR POWER PLANT

Flood Zone	Flood Zone Description	Accident Mitigating Systems Disabled	Assumed Initial Plant Response	Initiating Event Type	Screening Flooding Frequency (1/yr)	Conditional Core Damage Probability	Core Damage Frequency (1/yr)	Discussion/Comments
1	Turbine Building, Heater Bay, Turbine Power Complex	Feedwater recovery with motor feedwater pump, condensate transfer system	Automatic SCRAM	T2 - Loss of PCS	6E-3	3.8E-03	2E-5	Result is greater than 3E-07/yr; more detailed evaluation required
1A	Steam Tunnel	Feedwater recovery with motor feedwater pump, RCIC	Automatic SCRAM	T2 - Loss of PCS	6E-4	5.6E-03	3E-6	Result is greater than 3E-07/yr; more detailed evaluation required
2	Aux. Bldg. HPCS Room	HPCS	Manual SCRAM	T3A - PCS Available	1.4E-3	9.7E-05	1E-7	Result is less than 3E-07/yr; no further evaluation required
3	Aux. Bldg. RHR B Room	RHR train B; train C supply for low pressure makeup	Manual SCRAM	T3A - PCS Available	1.4E-3	1.6E-05	2E-8	Result is less than 3E-07/yr; no further evaluation required
4	Aux. Bldg. RHR C Room	RHR train C supply for low pressure makeup	Automatic SCRAM	T3A - PCS Available	1.4E-3	1.6E-05	2E-8	Result is less than 3E-07/yr; no further evaluation required

Table 3.3.7-1 (continued)

RESULTS OF INTERNAL FLOODING LEVEL ONE
ANALYSIS FOR PERRY NUCLEAR POWER PLANT

Flood Zone	Flood Zone Description	Accident Mitigating Systems Disabled	Assumed Initial Plant Response	Initiating Event Type	Screening Flooding Frequency (1/yr)	Conditional Core Damage Probability	Core Damage Frequency (1/yr)	Discussion/Comments
5	Aux. Bldg. RCIC Room	RCIC	Manual SCRAM	T3A - PCS Available	1.4E-3	1.7E-05	2E-8	Result is less than 3E-07/yr; no further evaluation required
6	Aux. Bldg. RHR A Room	RCIC, LPCI train A, condensate transfer train A, RHR train A supply to suppression pool cooling and containment spray, and RHR train A supply for long term containment heat removal	Manual SCRAM	T3A - PCS Available	1.4E-3	1.7E-05	2E-8	Result is less than 3E-07/yr; no further evaluation required
7	Aux. Bldg. LPCS Room	LPCS	Manual SCRAM	T3A - PCS Available	1.4E-3	1.4E-05	2E-8	Result is less than 3E-07/yr; no further evaluation required
8	Aux. Bldg. Corridors	RCIC, HPCS, LPCS, RHR A, RHR B, RHR C, suppression pool cooling and containment spray	Manual SCRAM	T3A - PCS Available	1.1E-3	2.1E-03	2E-6	Result is greater than 3E-07/yr; more detailed evaluation required

Table 3.3.7-1 (continued)

RESULTS OF INTERNAL FLOODING LEVEL ONS
ANALYSIS FOR PERRY NUCLEAR POWER PLANT

Flood Zone	Flood Zone Description	Accident Mitigating Systems Disabled	Assumed Initial Plant Response	Initiating Event Type	Screening Flooding Frequency (1/yr)	Conditional Core Damage Probability	Core Damage Frequency (1/yr)	Discussion/Comments
9	Intermediate Building Elevation 574.8	None	Manual SCRAM	T3A - PCS Available	1.1E-3	1.0E-05	1E-8	Result is less than 3E-07/yr; no further evaluation required
10	Intermediate Building CRD Pump Room Elevation 574.8	None	Automatic SCRAM	T3A - PCS Available	7E-4	1.0E-05	7E-9	Result is less than 3E-07/yr; no further evaluation required
11	Drywell	None	—	—	—		—	Design features preclude losses
12	Containment	None	—	—	—		—	Design features preclude losses
13	Control Complex Elevation 574.8	HPCS lost after 10 hours; low pressure makeup, RHR train A and B supply for long term containment heat removal	Automatic SCRAM	T1A - Loss of Inst. Air	2.6E-3	3.0E-01	8E-4	Result is greater than 3E-07/yr; more detailed evaluation required

Table 3.3.7-1 (continued)

RESULTS OF INTERNAL FLOODING LEVEL ONE
ANALYSIS FOR PERRY NUCLEAR POWER PLANT

Flood Zone	Flood Zone Description	Accident Mitigating Systems Disabled	Assumed Initial Plant Response	Initiating Event Type	Screening Flooding Frequency (1/yr)	Conditional Core Damage Probability	Core Damage Frequency (1/yr)	Discussion/Comments
13 Alter-nate Run	Control Complex Elevation 574.8	Same as for above except suppression pool cooling also unavailable	Automatic SCRAM	T1A - Loss of Inst. Air	2.6E-3	1.0E+00	3E-3	Further evaluation is required to determine the like-lihood for failure of suppression pool cooling due to floods in zone 13
14	Radwaste Building	None	—	—	—		—	No further consid-eration required except that zone 14 may serve as a propagation pathway between flood zones
15	Intermediate Building Elevation 599	Train A of instrument air supply for ADS unavailable	Automatic SCRAM	T1A - Loss of PCS	7.5E-4	6.5E-05	5E-8	Result is less than 3E-07/yr; no further evaluation required

Table 3.3.7-1 (continued)

RESULTS OF INTERNAL FLOODING LEVEL ONE
ANALYSIS FOR PERRY NUCLEAR POWER PLANT

Flood Zone	Flood Zone Description	Accident Mitigating Systems Disabled	Assumed Initial Plant Response	Initiating Event Type	Screening Flooding Frequency (1/yr)	Conditional Core Damage Probability	Core Damage Frequency (1/yr)	Discussion/Comments
15 to 22	Intermediate Building Elevation 599 and 620.5	Train A of instrument air supply for ADS unavailable; FPCC vent path for long term containment heat removal with vent	Automatic SCRAM	T2 - Loss of PCS	7.5E-4	6.5E-05	5E-8	Result is less than 3E-07/yr; no further evaluation required
16	Aux. Bldg. Corridors Elevation 599	RCIC, low pressure makeup, RHR train A and B supply for long term containment heat removal and long term containment heat removal with vent	Automatic SCRAM	T2 - Loss of PCS	7.5E-4	6.4E-03	5E-6	Result is greater than 3E-07/yr; more detailed evaluation required
17	Control Complex Elevation 599	FPCC path for long term containment heat removal with vent, HPCS lost after 10 hours	Automatic SCRAM	T1A - Loss of Inst. Air	1.2E-3	1.8E-02	2E-5	Result is greater than 3E-07/yr; more detailed evaluation required

Table 3.3.7-1 (continued)

RESULTS OF INTERNAL FLOODING LEVEL ONE
ANALYSIS FOR PERRY NUCLEAR POWER PLANT

Flood Zone	Flood Zone Description	Accident Mitigating Systems Disabled	Assumed Initial Plant Response	Initiating Event Type	Screening Flooding Frequency (1/yr)	Conditional Core Damage Probability	Core Damage Frequency (1/yr)	Discussion/Comments
18, 19, 20	Diesel Generator Buildings	Emergency diesel generators	Controlled Shutdown	—	—		—	Flooding of the diesel generator buildings results in no other disabling of accident mitigation systems; loss of only the diesel generators results in a controlled shutdown and is therefore not considered further; the likelihood of a flood occurring in the diesel generator buildings (and loss of all emergency diesel power) with a coincident loss of offsite power (and therefore station blackout) is insignificant

Table 3.3.7-1 (continued)

RESULTS OF INTERNAL FLOODING LEVEL ONE
ANALYSIS FOR PERRY NUCLEAR POWER PLANT

Flood Zone	Flood Zone Description	Accident Mitigating Systems Disabled	Assumed Initial Plant Response	Initiating Event Type	Screening Flooding Frequency (1/yr)	Conditional Core Damage Probability	Core Damage Frequency (1/yr)	Discussion/Comments
21	Control Complex Switchgear Rooms Elevation 620.5	Emergency Switchgear	Automatic SCRAM	T3A - PCS Available	1E-5/yr (divisions 1 and 2); 5E-6/yr (divisions 1, 2 & 3)		--	
22	Intermediate Building West Side Elevation 620.5	FPCC vent path for long term containment heat removal with vent	Automatic SCRAM	T3A - PCS Available	6E-4	1.0E-05	6E-9	Result is less than 3E-07/yr; no further evaluation required
23	Aux. Bldg. Corridors West Side Elevation 620.5	HPCS	Automatic SCRAM	T3A - PCS Available	3E-4	1.0E-04	3E-8	Result is less than 3E-07/yr; no further evaluation required
23 to 22	Aux. Bldg. Corridor West Side to Intermediate Building West Side Elevation 620.5	HPCS; FPCC vent path for long term containment heat removal with vent	Automatic SCRAM	T3A - PCS Available	3E-4	1.0E-04	3E-8	Result is less than 3E-07/yr; no further evaluation required

Table 3.3.7-1 (continued)

RESULTS OF INTERNAL FLOODING LEVEL ONE
ANALYSIS FOR PERRY NUCLEAR POWER PLANT

Flood Zone	Flood Zone Description	Accident Mitigating Systems Disabled	Assumed Initial Plant Response	Initiating Event Type	Screening Flooding Frequency (1/yr)	Conditional Core Damage Probability	Core Damage Frequency (1/yr)	Discussion/Comments
24	Aux. Bldg. Corridors East Side Elevation 620.5	B train of P57 lost for emergency RPV depressurization	Automatic SCRAM	T3A - PCS Available	3E-4	1.0E-05	3E-9	Result is less than 3E-07/yr; no further evaluation required
25	Intermediate Building East Side Fuel Handling Area Elevation 620.5	None	Manual SCRAM	T3A - PCS Available	3E-4	1.0E-05	3E-9	Result is less than 3E-07/yr; no further evaluation required
26	Intermediate Building Elevation 654.5	None	Manual SCRAM	T2 - Loss of PCS	1.1E-4	6.6E-05	7E-9	Result is less than 3E-07/yr; no further evaluation required
27, 28, 29	Control Complex Unit 1, Div. 1/2 Battery Room/switchgear and cable spreading room Elevation 638.5	dc power supplied to division 1 and 2 switchgear	SCRAM	—	—		—	More detailed evaluation required

Table 3.3.7-1 (continued)

RESULTS OF INTERNAL FLOODING LEVEL ONE
ANALYSIS FOR PERRY NUCLEAR POWER PLANT

Flood Zone	Flood Zone Description	Accident Mitigating Systems Disabled	Assumed Initial Plant Response	Initiating Event Type	Screening Flooding Frequency (1/yr)	Conditional Core Damage Probability	Core Damage Frequency (1/yr)	Discussion/Comments
30, 30A, 31, 31A	Control Complex Unit 1 Control Room, control complex corridor, Unit 1 and Unit 2 HVAC rooms Elevation 679.5 and 654.5	Control room and potentially control complex HVAC	SCRAM	—	—		—	More detailed evaluation required
32	Intermediate Building Elevation 682.5	None	—	—	—		—	No further evaluation required
33	Emergency Service Water Pumphouse	Emergency Service Water System	Manual SCRAM	T3A - PCS Available	5E-3 (assumed)	1.7E-05	9E-8	The result is less than the 3E-07/yr screening criterion and therefore no further evaluation is required

Table 3.3.7-1 (continued)

RESULTS OF INTERNAL FLOODING LEVEL ONE
ANALYSIS FOR PERRY NUCLEAR POWER PLANT

Flood Zone	Flood Zone Description	Accident Mitigating Systems Disabled	Assumed Initial Plant Response	Initiating Event Type	Screening Flooding Frequency (1/yr)	Conditional Core Damage Probability	Core Damage Frequency (1/yr)	Discussion/Comments
33 Alter-nate Run	Emergency Service Water Pumphouse	Emergency Service Water System and RHR trains A, B, and C	[illegible]ual SCRAM	T3A - PCS Available	5E-3 (assumed)	2.5E-05	1E-7	The result is less than the 3E-07/yr screening criterion and therefore no further evaluation is required
34	Service Water Pumphouse	Service water system	Automatic SCRAM	TSW Loss of Service Water	5E-3 (assumed)	6.0E-03	3E-5	Result is greater than 3E-07/yr; more detailed evaluation required
35	Off-G[illegible] Building	None	—	—	—		—	No further consideration required except that zone 35 may serve as a propagation pathway between flood zones

Table 3.3.7-2

Perry Internal Flooding Analysis
Conditional Core Damage Probabilities

Flood State	Zones/ Areas Affected	Accident Sequence Assumed	Equipment Disabled	Conditional Core Damage Probability
13B	Zone 13 or Zone 13 & 17	T1A (modified for fldng)	Low pressure make-up, RHR long term cntnmnt heat removal, instrument air	1.5E-3
16A	Zone 16	T3A	RCIC	2.6E-9
16B	Zone 16	T2	RCIC	6.7E-7
16C	Zones 16 & 8	T3A	RCIC, Low pressure make-up	8.8E-6
16D	Zones 16 & 8	T2	RCIC, Low pressure make-up	3.7E-5
16E	Zones 16, 8 & RHR pmp rm A	T3A	RCIC, Low pressure make-up, & Cntnmnt Spray A	2.1E-5
16F	Zones 16, 8 & RHR pmp rm A	T2	RCIC, Low pressure make-up, & Cntnmnt Spray A	8.6E-5
16G	Zones 16, 8, 13 & RHR pmp rm A	T2	RCIC, Low pressure make-up, RHR long term cntnmnt heat removal, instrument air, & cntnmnt spray A	8.6E-5
16H	Zone 16	T2	RCIC, Low pressure make-up	3.7E-5
16I	Zone 16	T2	RCIC, Low pressure make-up	8.8E-6
1AA	Zone 1A	T2	RCIC	6.7E-7
1AB	Zone 1A	T2	Feedwater recovery w/ motor feed pump	4.9E-[illegible]
1AC	Zone 1	T2	RCIC and Cndnst Transfer	3.4E-5
F1A	Zone 1	T2	Cndnst Transfer, feedwater recovery w/ motor feed pmp	5.7E-5
F1B	Zone 1	T2	Feedwater recovery w/ motor feed pump, cndnst transfer, RCIC, low pressure make-up, & RHR long term cntnmnt heat removal	3.7E-2

Table 3.3.[illegible]-2 (continued)

Perry Internal Flooding Analysis
<u>Conditional Core Damage Probabilities</u>

Flood State	Zones/ Areas Affected	Accident Sequence Assumed	Equipment Disabled	Conditional Core Damage Probability
F1C	Zone 1	T2	Feedwater recover w/ motor feed pump	4.9E-6
F1D	Zone 1	T2	Feedwater recover w/ motor feed pump, RCIC, Low pressure make-up, & RHR long term cntnmnt heat removal	4.3E-3
F1E	Zone 1	T2	Feedwater recover w/ motor feed pump & RCIC	5.6E-6
F8A	Zone 8	T3A	RCIC, Low pressure make-up alternate low pressure make-up	6.8E-5
F8B	Zone 8	T1A	RCIC & Low pressure make-up	9.2E-3
T2F	T2 accident seq induced by internal flood	T2	None	1.0E-6
T3A	T3A only	T3A	None	1.0E-9
TIAF	Loss of inst air only	TIA	None	1.0E-5

Table 3.3.7-3

Results of Detailed Flooding Analysis

Flood zone & Flood magnitude	Core Damage Frequency	Zone Percent	Sub-Total
13 mod	4.5E-7	29	
13 small	2.7E-7	18	
13 severe	1.6E-7	10	57
17 mod	1.6E-7	10	
17 small	1.0E-7	6	
17 severe	5.6E-8	4	20
TPC N71E	4.4E-8	3	
TPC N71W	4.4E-8	3	6
1 mod	1.1E-8	1	
1 severe	7.2E-9	< 1	< 1
1 small	8.0E-12	< 1	< 1
1 Exp Joint Sev Leak	9.9E-9	1	1
1 Exp Joint Small Leak	1.6E-7	10	10
1A	1.0E-9	10	10
8	3.0E-8	2	2
16	1.2E-8	1	1
Total	1.5E-6		

Table 3.4.1-1

Summary of Core Damage Frequency by Initiating Event

	Core Damage Freq	Percent of CDF	
Loss of Offsite Power			
T1	1.80×10^{-7}	1.5	(Loss of Offsite Power)
R	7.19×10^{-7}	6.2	(Loss of Offsite Power and no Offsite Power Recovery at 3 hr)
U	4.14×10^{-7}	3.6	(Loss of Offsite Power w/no HPCS or RCIC)
T1P1	7.62×10^{-8}	0.7	(Loss of Offsite Power and 1 SORV)
T1P1U	1.53×10^{-8}	0.1	(Loss of Offsite Power and 1 SORV w/no HPCS or RCIC)
T1P2	3.89×10^{-8}	0.3	(Loss of Offsite Power and 2 SORVs)
Total	1.44×10^{-6}	12.4	
Station Blackout			
B	2.11×10^{-6}	18.1	(Station Blackout)
BP1	8.36×10^{-8}	0.7	(Station Blackout and 1 SORV)
BP2	5.91×10^{-8}	0.5	(Station Blackout and 2 SORVs)
Total	2.25×10^{-6}	19.3	
Transients			
T3A	$< 10^{-8}$	0.0	(Transient w/ PCS)
T3AP1	$< 10^{-8}$	0.0	
T3AP2	$< 10^{-8}$	0.1	
T3B	$< 10^{-8}$	0.0	(Loss of feedwater)
T3C	1.38×10^{-7}	1.2	(Inadvertent open SRV)
T2	1.64×10^{-6}	14.1	(Transient w/o PCS)
T2P1	2.47×10^{-8}	0.2	
T2P2	$< 10^{-8}$	0.0	
TIA	1.01×10^{-6}	8.7	(Loss of instrument air)
TIAP1	$< 10^{-8}$	0.1	
TIAP2	$< 10^{-8}$	0.0	
TSW	6.68×10^{-8}	0.6	(Loss of service water)
TSWP1	$< 10^{-8}$	0.0	
TSWP2	$< 10^{-8}$	0.0	
Total	2.90×10^{-6}	25.0	

Table 3.4.1-1 continued

		Core Damage Freq	Percent of CDF	
LOCAs				
	A	2.11×10^{-7}	1.8	(Large LOCA)
	S1	6.18×10^{-8}	0.5	(Intermediate LOCA)
	S2	3.34×10^{-8}	0.3	(Small LOCA)
	Total	3.06×10^{-7}	2.6	
ATWS				
	T1-C	3.61×10^{-8}	0.3	
	T3A-C	$< 10^{-8}$	0.1	
	T3B-C	5.42×10^{-7}	4.6	
	T3C-C	9.38×10^{-8}	0.8	
	T2-C	4.02×10^{-6}	34.5	
	TIAC	4.33×10^{-8}	0.4	
	Total	4.74×10^{-6}	40.7	

Total Core Damage Frequency (internal initiators) 1.17×10^{-5}

Table 3.4.1-2

Sequence Core Damage Frequencies Grouped by Initiator

T2-C Sum =	4.02E-006	34.5%	
T2-CS30	2.27E-006	19.5%	T2-C-U3-X'
T2-CS20	6.25E-007	5.4%	T2-C-Lc-C1
T2-CS28	3.12E-007	2.7%	T2-C-U3-X
T2-CS11	2.90E-007	2.5%	T2-C-C1
T2-CS12	2.37E-007	2.0%	T2-C-X'
T2-CS06	1.58E-007	1.4%	T2-C-V
T2-CS05	1.25E-007	1.1%	T2-C-V'
B Sum =	2.11E-006	18.1%	
BS24	7.71E-007	6.6%	B-U1-Va-R
BS34	5.26E-007	4.5%	B-U1-U2-R-Va1
BS17	3.36E-007	2.9%	B-U1-R
BS07	1.60E-007	1.4%	B-R-Y-Cv
BS12	1.04E-007	0.9%	B-HI-R
BS22	5.96E-008	0.5%	B-U1-Va-V
BS35	5.15E-008	0.4%	B-U1-U2-R-X
BS33	5.11E-008	0.4%	B-U1-U2-R-Y-Cv
BS29	3.37E-008	0.3%	B-U1-U2-Va1
BS30	1.81E-008	0.2%	B-U1-U2-X
T2 Sum =	1.64E-006	14.1%	
T2S04	1.62E-006	13.9%	T2-W-Y-Cv
T2S09	1.90E-008	0.2%	T2-U3-U2-W-Y-Cv
T2S18	9.56E-009	0.1%	T2-U3-U2-U1-V-Va
TIA Sum =	1.01E-006	8.7%	
TIAS14	7.53E-007	6.5%	TIA-U2-U1-V-Va
TIAS05	2.57E-007	2.2%	TIA-U2-W-Y-Cv
R Sum =	7.19E-007	6.2%	
RS20	6.04E-007	5.2%	R-Ws-V-Va
RS19	8.73E-008	0.7%	R-Ws-V-Cv
RS10	2.75E-008	0.2%	R-Ws-W-Y-Cv
T3B-C Sum =	5.42E-007	4.6%	
T3B-CS29	2.76E-007	2.4%	T3B-C-U3-X'
T3B-CS19	7.60E-008	0.7%	T3B-C-Q-Lc-C1
T3B-CS09	5.32E-008	0.5%	T3B-C-Q-X
T3B-CS27	3.80E-008	0.3%	T3B-C-U3-X
T3B-CS10	3.45E-008	0.3%	T3B-C-Q-C1
T3B-CS11	2.88E-008	0.2%	T3B-C-Q-X'
T3B-CS08	1.78E-008	0.2%	T3B-C-Q-V
T3B-CS07	1.52E-008	0.1%	T3B-C-Q-V'
U Sum =	4.14E-007	3.6%	
US29	3.34E-007	2.9%	U-R1-V-Va
US12	5.99E-008	0.5%	U-V-Va
US28	1.63E-008	0.1%	U-R1-V-W-Y-Cv

Table 3.4.1-2 continued

A Sum =	2.11E-007	1.8%	
AS09	2.10E-007	1.8%	A-U1-V
T1 Sum =	1.80E-007	1.5%	
T1S08	1.47E-007	1.3%	T1-R2-W-Y-Cv
T1S04	1.94E-008	0.2%	T1-W-Y-Cv
T1S35	1.33E-008	0.1%	T1-U1-Ws-V-Va
T3C Sum =	1.38E-007	1.2%	
T3CS04	1.38E-007	1.2%	T3C-W-Y-Cv
T3C-C Sum =	9.38E-008	0.8%	
T3C-CS27	5.49E-008	0.5%	T3C-C-U3-X'
T3C-CS17	1.40E-008	0.1%	T3C-C-Lc-C1
T3C-CS07	9.81E-009	0.1%	T3C-C-X
BP1 Sum =	8.36E-008	0.7%	
BP1S27	4.85E-008	0.4%	BP1-U1-U2
BP1S17	1.69E-008	0.1%	BP1-U1-V
BP1S26	1.13E-008	0.1%	BP1-U1-Va-R
T1P1 Sum =	7.62E-008	0.7%	
T1P1S31	7.24E-008	0.6%	T1P1-U1-Ws-V
TSW Sum =	6.68E-008	0.6%	
TSWS10	5.49E-008	0.5%	TSW-U2-U1-V
TSWS14	1.00E-008	0.1%	TSW-C
S1 Sum =	6.18E-008	0.5%	
S1S13	5.85E-008	0.5%	S1-U1-V-Va
BP2 Sum =	5.91E-008	0.5%	
BP2S13	5.91E-008	0.5%	BP2-U1
TIAC Sum =	4.33E-008	0.4%	
TIACS09	3.31E-008	0.3%	TIAC-X'
T1P2 Sum =	3.89E-008	0.3%	
T1P2S11	2.39E-008	0.2%	T1P2-U1-V
T1P2S04	1.49E-008	0.1%	T1P2-W-Y-Cv
T1-C Sum =	3.61E-008	0.3%	
T1-CS09	2.19E-008	0.2%	T1-C-X'
T1-CS08	8.44E-009	0.1%	T1-C-C1
S2 Sum =	3.24E-008	0.3%	
S2S20	3.00E-008	0.3%	S2-C
T2P1 Sum =	2.47E-008	0.2%	
T2P1S04	2.46E-008	0.2%	T2P1-W-Y-Cv
T1P1U Sum =	1.53E-008	0.1%	
T1P1US22	1.37E-008	0.1%	T1P1U-R1-V

Table 3.4.1-3

Sequence within the Upper 95% of Total Core Damage

T2-CS30	2.27E-006	19.5%	T2-C-U3-X'
T2S04	1.62E-006	13.9%	T2-W-Y-Cv
BS24	7.71E-007	6.6%	B-U1-Va-R
TIAS14	7.53E-007	6.5%	TIA-U2-U1-V-Va
T2-CS20	6.25E-007	5.4%	T2-C-Lc-C1
RS20	6.04E-007	5.2%	R-Ws-V-Va
BS34	5.26E-007	4.5%	B-U1-U2-R-Va1
BS17	3.36E-007	2.9%	B-U1-R
US29	3.34E-007	2.9%	U-R1-V-Va
T2-CS28	3.12E-007	2.7%	T2-C-U3-X
T2-CS11	2.90E-007	2.5%	T2-C-C1
T3B-CS29	2.76E-007	2.4%	T3B-C-U3-X'
TIAS05	2.57E-007	2.2%	TIA-U2-W-Y-Cv
T2-CS12	2.37E-007	2.0%	T2-C-X'
AS09	2.10E-007	1.8%	A-U1-V
BS07	1.60E-007	1.4%	B-R-Y-Cv
T2-CS06	1.58E-007	1.4%	T2-C-V
T1S08	1.47E-007	1.3%	T1-R2-W-Y-Cv
T3CS04	1.38E-007	1.2%	T3C-W-Y-Cv
T2-CS05	1.25E-007	1.1%	T2-C-V'
BS12	1.04E-007	0.9%	B-HI-R
RS19	8.73E-008	0.7%	R-Ws-V-Cv
T3B-CS19	7.60E-008	0.7%	T3B-C-Q-Lc-C1
T1P1S31	7.24E-008	0.6%	T1P1-U1-Ws-V
US12	5.99E-008	0.5%	U-V-Va
BS22	5.96E-008	0.5%	B-U1-Va-V
BP2S13	5.91E-008	0.5%	BP2-U1
S1S13	5.85E-008	0.5%	S1-U1-V-Va
TSWS10	5.49E-008	0.5%	TSW-U2-U1-V
T3C-CS27	5.49E-008	0.5%	T3C-C-U3-X'
T3B-CS09	5.32E-008	0.5%	T3B-C-Q-X
BS35	5.15E-008	0.4%	B-U1-U2-R-X
BS33	5.11E-008	0.4%	B-U1-U2-R-Y-Cv
BP1S27	4.85E-008	0.4%	BP1-U1-U2
T3B-CS27	3.80E-008	0.3%	T3B-C-U3-X
T3B-CS10	3.45E-008	0.3%	T3B-C-Q-C1
BS29	3.37E-008	0.3%	B-U1-U2-Va1
TIACS09	3.31E-008	0.3%	TIAC-X'
S2S20	3.00E-008	0.3%	S2-C
T3B-CS11	2.88E-008	0.2%	T3B-C-Q-X'
RS10	2.75E-008	0.2%	R-Ws-W-Y-Cv

Table 3.4.1-4

Sequences Leading to Containment Failure Prior to Core Damage

T2S04	1.62E-006	13.9%	T2-W-Y-Cv
TIAS05	2.57E-007	2.2%	TIA-U2-W-Y-Cv
BS07	1.60E-007	1.4%	B-R-Y-Cv
T1S08	1.47E-007	1.3%	T1-R2-W-Y-Cv
T3CS04	1.38E-007	1.2%	T3C-W-Y-Cv
RS19	8.73E-008	0.7%	R-Ws-V-Cv
BS33	5.11E-008	0.4%	B-U1-U2-R-Y-Cv
RS10	2.75E-008	0.2%	R-Ws-W-Y-Cv
T2P1S04	2.46E-008	0.2%	T2P1-W-Y-Cv
T1S04	1.94E-008	0.2%	T1-W-Y-Cv
T2S09	1.90E-008	0.2%	T2-U3-U2-W-Y-Cv
US28	1.63E-008	0.1%	U-R1-V-W-Y-Cv
T1P2S04	1.49E-008	0.1%	T1P2-W-Y-Cv
Event V	< 10E-008		
Total	2.58E-006	22%	

Table 3.4.1-5

Sequences Below 10^{-7} Because of Human Interaction Reliability

	Base Case	HI value rasied to 1.0
T3B-CS19	7.60E-8	7.60E-6
US12	6.00E-8	3.50E-7
BS22	6.00E-8	3.20E-7
S1S13	5.85E-8	8.15E-7
T3C-CS27	5.49E-8	1.40E-6
T3B-CS09	5.32E-8	7.60E-6
BS33	5.11E-8	1.28E-6
BP1S27	4.85E-8	5.09E-7
T3B-CS27	3.80E-8	7.60E-6
T3B-CS10	3.45E-8	7.63E-6
BS29	3.37E-8	5.52E-6
TIACS09	3.31E-8	9.20E-7
T3B-CS11	2.88E-8	7.60E-6
RS10	2.75E-8	3.72E-8
T2P1S04	2.46E-8	4.67E-3
T1-CS09	2.19E-8	6.09E-7
T3B-CS08	1.78E-8	7.61E-6
US28	1.63E-8	2.02E-8
T3B-CS07	1.52E-8	7.60E-6
T1P2S04	1.49E-8	2.49E-8
T3C-CS17	1.40E-8	1.40E-6
T1S35	1.33E-8	5.67E-7
BP1S26	1.13E-8	6.11E-8
T3C-CS07	9.81E-9	1.40E-6
T2S18	9.56E-9	9.56E-8
T1-CS08	8.44E-9	6.17E-7
T3AP2S04	6.28E-9	1.30E-3
T3C-CS09	5.31E-9	1.40E-6
T3C-CS08	4.69E-9	1.40E-6
TIACS07	4.60E-9	9.20E-7
T3BS12	4.48E-9	4.48E-8
TIAP1S14	4.01E-9	3.35E-8
US30	3.37E-9	2.09E-8
TI-CS07	3.05E-9	6.09E-7
T3C-CS05	2.80E-9	1.40E-6
T2-CS04	2.67E-9	2.67E-9
TIACS08	2.56E-9	9.21E-7
S2S04	2.35E-9	5.40E-4
T2-CS21	2.32E-9	6.13E-5
BP1S07	2.02E-9	4.03E-8
T2P2S04	1.94E-9	4.67E-4
T3C-CS06	1.89E-9	1.40E-6
TIACS05	1.84E-9	9.20E-7
T1P1US10	1.66E-9	7.74E-6

Table 3.4.1-5 continued

	Base Case	HI value rasied to 1.0
TI-CS06	1.43E-9	6.10E-7
T1P1S52	1.31E-9	3.57E-9
T1-CS05	1.22E-9	6.09E-7
T1ACS06	1.18E-9	9.20E-7
T2-CS29	9.32E-10	6.17E-5
T2-CS26	8.78E-10	4.39E-7
US13	8.46E-10	2.89E-9
BP1S24	8.44E-10	4.55D-9
BP1S12	6.63E-10	6.64E-8

Table 3.4.1-6

Summary of Core Damage Frequency by Internal Events Initiator

Internal Events Initiator	Frequency	Percentage
ATWS	$4.74E^{-6}$	40.7
Transients	$2.90E^{-6}$	25.0
Station Blackout	$2.25E^{-6}$	19.3
Loss of Offsite Power	$1.44E^{-6}$	12.4
LOCAs	$3.06E^{-7}$	2.6

Table 3.4.1-7

Importance Ranking of Components (Fussell-Vesely)

Rank	Event Name	Point Est	F-V Imp	Risk Ach	Risk Red
1	CM	1.000E-005	4.110E-001	41100.41	1.698
2	T1	6.090E-002	3.202E-001	5.94	1.471
3	NSHICPEC5-2-L1T3	1.000E+000	2.735E-001	1.00	1.377
4	T3A	4.510E+000	2.530E-001	0.80	1.339
5	FWHICPEL-2-FDW-V	5.003E-003	2.472E-001	50.17	1.328
6	T2	1.620E+000	2.314E-001	0.91	1.301
7	ADH.CPC5-1-ADS-0	7.200E+000	2.165E-001	0.81	1.276
8	CV05	1.400E-001	1.468E-001	1.90	1.172
9	R15	8.230E-002	1.279E-001	2.43	1.147
10	DGBALC1R22S0006	9.600E-005	1.171E-001	1220.42	1.133
11	DHDGFS1E22S0001	3.000E-002	1.164E-001	4.76	1.132
12	U207B	8.752E-001	1.028E-001	1.01	1.115
13	FPOFFSITEPUMPER	6.000E-001	1.003E-001	1.07	1.112
14	DGDGFS1R43S0001B	3.000E-002	9.750E-002	4.15	1.108
15	T1A	9.200E-002	9.134E-002	1.90	1.101
16	DGDGFS1R43S0001A	3.000E-002	8.504E-002	3.75	1.093
17	R36	5.290E-001	8.427E-002	1.08	1.092
18	FPDPFRO?54C0001	1.745E-001	7.890E-002	1.37	1.086
19	LCLCUMA	.930E-002	7.724E-002	4.92	1.084
20	U202B	7.823E-001	6.915E-002	1.02	1.074
21	LCLCUMRHRALPRC	9.220E-003	6.623E-002	8.12	1.071
22	FWHICPSN27-4:1IA	1.196E-001	6.408E-002	1.47	1.068
23	SPHICPPS4:5SPCU	1.000E+000	6.344E-002	1.00	1.068
24	FWHICPEC5-3:2	1.000E-002	6.151E-002	7.09	1.066
25	SLHICPEQ-6-SLCX	1.000E+000	6.128E-002	1.00	1.065
26	MESC133	7.760E-003	5.075E-002	7.49	1.053
27	T3B	7.600E-001	4.707E-002	1.01	1.049
28	MESA133	7.760E-003	4.603E-002	6.89	1.048
29	MESB133	7.760E-003	4.386E-002	6.61	1.046
30	R39	1.730E-002	4.258E-002	4.42	1.044
31	CV01	4.300E-001	4.132E-002	1.05	1.043
32	FPHICPPS4:2RCIC1	2.998E-001	3.813E-002	1.09	1.040
33	DGDGFR1R43S0001B	7.864E-003	3.627E-002	5.58	1.038
34	ESMPCC	3.417E-004	3.505E-002	103.55	1.036
35	CVHICPEPC-FPCC	1.000E-001	3.482E-002	1.31	1.036
36	CVHICPEPC-COM	1.001E-003	3.482E-002	35.76	1.036
37	FPHICPPS4:2RCIC4	1.000E-001	3.424E-002	1.31	1.035
38	CV03	4.000E-002	3.337E-002	1.80	1.035
39	ECECUMA	1.980E-002	3.322E-002	2.64	1.034
40	LCLCUMLPCIBLPCIC	1.770E-002	3.199E-002	2.78	1.033
41	ESESUMA	1.890E-002	3.131E-002	2.63	1.032
42	VA04B	7.327E-001	3.056E-002	1.01	1.032
43	ECECUMB	1.980E-002	3.023E-002	2.50	1.031
44	ESESUMB	1.890E-002	2.929E-002	2.52	1.030
45	FPHICPPS4:2RCIC2	1.000E-001	2.927E-002	1.26	1.030
46	DGDGUM1R43S0001B	3.030E-002	2.900E-002	1.91	1.030
47	CVHICPEPC-RHR	1.000E-002	2.879E-002	3.85	1.030
48	RCHICPS51-LDTRIP	4.999E-002	2.859E-002	1.54	1.029

Table 3.4.1-7 continued

Rank	Event Name	Point Est	F-V Imp	Risk Ach	Risk Red
49	DGDGUM1R43S0001A	3.080E-002	2.745E-002	1.86	1.028
50	LCLCUMLPCIALPCS	5.290E-003	2.622E-002	5.93	1.027
51	DHDGFR1E22S0001	7.864E-003	2.453E-002	4.10	1.025
52	ADHICPC5-1-ADS-A	3.793E-003	2.346E-002	7.16	1.024
53	DHDGUM1E22S0001	2.430E-002	2.331E-002	1.94	1.024
54	U109B	8.951E-001	2.290E-002	1.00	1.023
55	CVMVN01G41F0140	2.930E-003	2.212E-002	8.53	1.023
56	SCMVNC1E12F0048A	2.930E-003	2.210E-002	8.52	1.023
57	T3C	1.400E-001	2.0[illegible]2E-002	1.13	1.021
58	CVHICEPS7:3G41-T	5.003E-002	2.037E-002	1.39	1.021
59	ESESUMC	9.370E-003	1.876E-002	2.98	1.019
60	DGDGFR1R43S0001A	7.864E-003	1.837E-002	3.32	1.019
61	A	1.000E-004	1.814E-002	182.34	1.018
62	P1	1.600E-002	1.772E-002	2.09	1.018
63	U111B	8.956E-001	1.574E-002	1.00	1.016
64	ECMPFS1P42C0001A	2.930E-003	1.464E-002	5.98	1.015
65	HPMVN01E22F0015	2.930E-003	1.454E-002	5.95	1.015
66	HPMPFS1E22C0001	2.930E-003	1.453E-002	5.95	1.015
67	HPMVN01E22F0012	2.930E-003	1.453E-002	5.95	1.015
68	HPMVN01E22F0004	2.930E-003	1.453E-002	5.95	1.015
69	ECMPFS1P42C0001B	2.930E-003	1.338E-002	5.55	1.014
70	ECMVNC0P42F0150B	2.930E-003	1.338E-002	5.55	1.014
71	R3	1.510E-002	1.280E-002	1.83	1.013
72	HIHICPEC5-5-CRIT	2.001E-003	1.259E-002	7.28	1.013
73	R15B	9.180E-001	1.245E-002	1.00	1.013
74	RCTPFR1E51C0001	1.[illegible]31E-001	1.192E-002	1.09	1.012
75	SCMVNC1E12F0048B	2.930E-003	1.147E-002	4.90	1.012
76	LCLCUMB	1.810E-002	1.098E-002	1.60	1.011
77	DCBTLC1R42S0002	1.367E-003	1.071E-002	8.83	1.011
78	CTHICPPS4:4-ALT	1.000E-001	9.886E-003	1.09	1.010
79	HIHICPOR10-4:3-B	1.000E-002	9.852E-003	1.98	1.010
80	R36B	4.710E-001	9.634E-003	1.01	1.010
81	ECHXPL1P42B0001A	2.049E-003	9.6[illegible]7E-003	5.69	1.010
82	R25	1.960E-002	9.423E-003	1.47	1.010
83	ECMVNC0P42F015[illegible]	2.930E-003	9.157E-003	4.12	1.009
84	P2	1.600E-003	9.113E-003	6.69	1.009
85	ECHXPL1P42B0001B	2.049E-003	8.828E-003	5.30	1.009
86	DGDGCC	3.675E-004	8.743E-003	24.78	1.009
87	RCTPFR1E51C001	1.980E-002	8.740E-003	1.43	1.009
88	FPHICPPS4:2-DD-0	3.000E-002	8.276E-003	1.27	1.008
89	FPDPFS0P54C0001	3.000E-002	8.276E-003	1.27	1.008
90	DBMFCC	2.930E-005	8.087E-003	277.00	1.008
91	DBLVCC	2.930E-005	8.087E-003	277.00	1.003
92	DBMDCC	2.930E-005	8.087E-003	277.00	1.008
93	SLHICPEQ-6-SLC1	1.251E-003	7.903E-003	7.31	1.008
94	VA09B	5.327E-001	7.489E-003	1.01	1.008
95	ESMVCC	9.250E-005	7.284E-003	79.73	1.007
96	RCRCUM	1.810E-002	7.179E-003	1.39	1.007
97	HPCLDEL2	1.598E-003	6.987E-003	5.37	1.007
98	ADHICPC5-1-ADS-L	3.600E-002	3.726E-003	1.18	1.007

Table 3.4.1-8

Importance Ranking of Components (Risk Achievement Worth)

Rank	Event Name	Point Est	F-V Imp	Risk Ach	Risk Red
1	CM	1.000E-005	4.110E-001	41100.41	1.698
2	EFFACCEPRCS	3.750E-006	6.664E-003	1778.20	1.007
3	DGBALC1R22S0006	9.600E-005	1.171E-001	1220.42	1.133
4	DCB[illegible]C	1.370E-005	5.433E-003	397.55	1.005
5	DBMFCC	2.930E-005	8.087E-003	277.00	1.008
6	DBMDCC	2.930E-005	8.087E-003	277.00	1.008
7	DBLVCC	2.930E-005	8.087E-003	277.00	1.008
8	A	1.000E-004	1.814E-002	182.34	1.018
9	ESMPCC	3.417E-004	3.505E-002	103.55	1.036
10	ESMVCC	9.250E-005	7.284E-003	79.73	1.007
11	ADSRCCADS	8.000E-006	5.934E-004	75.18	1.001
12	ECMPCC	1.190E-005	7.606E-004	64.92	1.001
13	DGBALC1R22S0007	9.600E-005	5.953E-003	63.01	1.006
14	FWHICPEL-2-FDW-V	5.002E-003	2.472E-001	50.17	1.328
15	ECMVCC	9.250E-005	4.238E-003	46.81	1.004
16	CVHICPEPC-COM	1.001E-003	3.482E-002	35.76	1.036
17	DGDCCC	3.675E-004	8.743E-003	24.78	1.009
18	S1	3.000E-004	5.280E-003	18.60	1.005
19	DCBTLC1R42SC002	1.367E-003	1.071E-002	8.83	1.011
20	CVMVN01G41F0140	2.930E-003	2.212E-002	8.53	1.023
21	SCMVNC1E12F0048A	2.930E-003	2.210E-002	8.52	1.023
22	LCLCUMRHRALPRC	9.220E-003	6.623E-002	8.12	1.071
23	MESC133	7.760E-003	5.075E-002	7.49	1.053
24	SLHICPEQ-6-SLC1	1.251E-003	7.903E-003	7.31	1.008
25	HIHICPEC5-5-CRIT	2.001E-003	1.259E-002	7.28	1.013
26	SLEVCC	2.930E-004	1.826E-003	7.23	1.002
27	SLMPCC	2.930E-004	1.826E-003	7.23	1.002
28	ADHICPC5-1-ADS-A	3.793E-003	2.346E-002	7.16	1.024
29	HIHICPEC5-3:2-F	1.001E-003	6.127E-003	7.12	1.006
30	FWHICPEC5-3:2	1.000E-002	6.151E-002	7.09	1.066
31	SLMVCC1G33	9.250E-005	5.577E-004	7.03	1.001
32	SLMVCC	9.250E-005	5.577E-004	7.03	1.001
33	SLCVN01C41F0007	1.000E-004	6.028E-004	7.03	1.001
34	SLCVN01C41F0006	1.000E-004	6.028E-004	7.03	1.001
35	SLXVPL1C41F0036	4.499E-005	2.659E-004	6.91	1.000
36	MESA133	7.760E-003	4.603E-002	6.89	1.048
37	TSW	1.000E-003	5.731E-003	6.73	1.006
38	LCMPCC	2.930E-004	1.677E-003	6.72	1.002
39	P2	1.600E-003	9.113E-003	6.69	1.009
40	MESB133	7.760E-003	4.386E-002	6.61	1.046
41	ECMPFS1P42C0001A	2.930E-003	1.464E-002	5.98	1.015
42	HPMVN01E22F0015	2.930E-003	1.4[illegible]E-002	5.95	1.015
43	HPMVN01E22F0004	2.930E-003	1.453E-002	5.95	1.015
44	HPMPFS1E22C0001	2.930E-003	1.453E-002	5.95	1.015
45	HPMVN01E22F0012	2.930E-003	1.453E-002	5.95	1.015
46	T1	6.090E-002	3.202E-001	5.94	1.471
47	LCLCUMLPCIALPCS	5.290E-003	2.622E-002	5.93	1.027
48	ECHXPL1P42B0001A	2.049E-003	9.627E-003	5.69	1.010

Table 3.4.1-8 continued

Rank	Event Name	Point Est	F-V Imp	Risk Ach	Risk Red
49	M	1.000E-004	4.640E-004	5.64	1.000
50	RPT	1.000E-004	4.640E-004	5.64	1.000
51	DGDGFR1R43S0001B	7.864E-003	3.627E-002	5.58	1.038
52	ECMPFS1P42C0001B	2.930E-003	1.338E-002	5.55	1.014
53	ECMVNCOP42F0150B	2.930E-003	1.338E-002	5.55	1.014
54	HPCLDEL2	1.598E-003	6.987E-003	5.37	1.007
55	ECHXPL1P42B0001B	2.049E-003	8.828E-003	5.30	1.009
56	LCLCUMA	1.930E-002	7.724E-002	4.92	1.084
57	SCMVNC1E12F0048B	2.930E-003	1.147E-002	4.90	1.012
58	DHDGFS1E22S0001	3.000E-002	1.164E-001	4.76	1.132
59	ECMPFR1P42C0001A	7.195E-004	2.626E-003	4.65	1.003
60	R39	1.230E-002	4.256E-002	4.42	1.044
61	ECMPFR1P42C0001B	7.195E-004	2.362E-003	4.28	1.002
62	HPMPFR1E22C0001	7.195E-004	2.347E-003	4.26	1.002
63	DGDGFS1R43S0001B	3.000E-002	9.750E-002	4.15	1.108
64	ECMVNCOP42F0150A	2.930E-003	9.157E-003	4.12	1.009
65	DHDGFR1E22S0001	7.864E-003	2.453E-002	4.10	1.025
66	CVHICPEPC-RHR	1.000E-002	2.879E-002	3.85	1.030
67	DGDGFS1R43S0001A	3.000E-002	8.504E-002	3.75	1.093
68	EPFAFS1M39B0003	3.750E-004	9.446E-004	3.52	1.001
69	EPFAFR1M39B0003	2.999E-004	7.263E-004	3.42	1.001
70	DGDGFR1R43S0001A	7.864E-003	1.837E-002	3.32	1.019
71	SMMVCC	9.250E-005	2.102E-004	3.27	1.000
72	DHHXPL1E22S0001	2.049E-003	4.633E-003	3.26	1.005
73	ESESUMC	9.370E-003	1.876E-002	2.98	1.019
74	DCBTLC1E22S0005	1.367E-003	2.607E-003	2.91	1.003
75	DCBTLC1R42S0003	1.367E-003	2.505E-003	2.83	1.003
76	LCLCUMLPCIBLPCIC	1.770E-002	3.199E-002	2.78	1.033
77	ECECUMA	1.980E-002	3.322E-002	2.64	1.034
78	ESESUMA	1.890E-002	3.131E-002	2.63	1.032
79	DGHXPL1R46B0002B	2.049E-003	3.131E-003	2.53	1.003
80	ESESUMB	1.890E-002	2.929E-002	2.52	1.030
81	ECECUMB	1.980E-002	3.023E-002	2.50	1.031
82	DGHXPL1R46B0002A	2.049E-003	2.976E-003	2.45	1.003
83	R15	8.230E-002	1.279E-001	2.43	1.147
84	HPHPUM	2.680E-003	3.611E-003	2.34	1.004
85	ECCLLF1P42C0001A	1.250E-004	1.527E-004	2.22	1.000
86	ESSCPL1P49D0001B	9.595E-004	1.167E-003	2.22	1.001
87	ESSCPL1P49D0001A	9.595E-004	1.167E-003	2.22	1.001
88	EPEPUMHPCS	1.630E-003	1.953E-003	2.20	1.002
89	HPCLLF1E22C0001	1.250E-004	1.382E-004	2.11	1.000
90	EPCLLF1M39B0003	1.250E-004	1.382E-004	2.11	1.000
91	P1	1.600E-002	1.772E-002	2.09	1.018
92	ECCLLF1P42C0001B	1.250E-004	1.278E-004	2.02	1.000
93	DHBALC1R22S0009	9.600E-005	9.778E-005	2.02	1.000
94	HPCVNO1E22F0002	1.000E-004	1.018E-004	2.02	1.000
95	HPCVNO1E22F0016	1.000E-004	1.018E-004	2.02	1.000
96	HPCVNO1E22F0024	1.000E-004	1.018E-004	2.02	1.000
97	HPCVNO1E22F0005	1.000E-004	1.018E-004	2.02	1.000
98	HIHICPOR10-4:3-B	1.000E-002	9.852E-003	1.98	1.010

Table 3.4.1-9

Contribution of Functional Failures to Core Damage

Functional Failure	Contribution to CDF (%)
Loss of Injection (U1, U2, U3, V, VA)	68
Failure of Decay Heat Removal (Q, M)	2
Failure of Containment Heat Removal (W, Ws, Y)	28
Failure to Recover Offsite Power (R, R1, R2)	22
Failure to Depressurize (X, X2)	4
Containment Failure (Cv)	22
Failure to inhibit ADS following ATWS (X')	25

TABLE 3.4.1-10 APET CONTAINMENT PERFORMANCE BASE CASE RESULTS

		FREQUENCY	FRACTION OF CDF
No RPV Fail:	No Containment Failure	3.39E-6	32.6%
	Vent	2.45E-6	19.3%
	Containment Failure	6.18E-7	4.9%
Subtotal No RPV Failure Core Damage Frequency:		6.46E-6	50.8%
RPV Fail:	No Containment Failure	1.58E-6	12.4%
	Vent	1.27E-6	10.0%
	Late Containment Failure	9.38E-7	7.4%
	Early CF: No Pool Bypass	4.30E-7	3.4%
	Late PB Bypass	1.54E-6	12.1%
	Early PB, Spray	6.12E-8	0.5%
	Early PB, No Spray	4.45E-7	3.5%
Subtotal RPV Failure Core Damage Frequency:		6.27E-6	49.2%
TOTAL CORE DAMAGE FREQUENCY:		1.27E-5	100.0%
Subtotal Containment Venting Frequency:		3.72E-6	29.2%
Subtotal Cntmt Structural Failure Frequency:		4.03E-6	31.7%
TOTAL CONTAINMENT FAILURE & VENTING FREQUENCY:		7.76E-6	60.9%
RPV FAILURE AND EARLY CONTAINMENT FAILURE WITH POOL BYPASS FREQUENCY:		2.04E-6	16.1%

TABLE 3.4.1-11 DOMINANT CONTAINMENT FAILURE PLANT DAMAGE STATES

Rank	PDS CLASS	FREQUENCY	CNTMT FAILURE PERCENT	CDF PERCENT	DOMINANT SEQUENCES	CDF%
1	56	3.40E-6	43.9%	34.0%	13B-U1-U2-Va1 TIA-U2-U1-V-Va-Wc-Ws B6-V-Wc F1D-U3-U2-U1-V-Va A-U1-V-Wc-Ws T15-Va	8.9% 5.9% 3.0% 1.8% 1.6% 1.6%
2	73	1.22E-6	15.8%	9.6%	T2-Wc-Ws-Y-Cv-Li	8.2%
3	61	6.23E-7	8.0%	4.9%	T15-Va-R3 UR-V-Va-R3	2.6% 1.9%
4	71	5.59E-7	7.2%	4.4%	T2-Wc-Ws-Y-Cv TIA-U2-Wc-Ws-Y-Cv	2.2% 1.5%
5	65	2.93E-7	3.8%	2.3%	T2-CA-C1-Wc	1.1%
6	67	2.65E-7	3.4%	2.1%		
7	69	2.56E-7	3.3%	2.0%		
8	63	1.69E-7	2.1%	1.3%	T2-CA	1.2%
9	25	2.09E-7	1.8%	1.6%		
10	36	1.02E-7	1.3%	0.8%		
11	66	1.00E-7	1.3%	0.8%		
12	9	9.73E-8	0.8%	0.8%		
13	58	6.15E-8	0.8%	0.5%		
14	62	5.02E-8	0.7%	0.4%		
15	53	4.44E-6	0.6%	34.9%		
16	70	3.23E-8	0.4%	0.3%		

Table 3.4.1-12

Recovery Actions Included in the Model

Identification	Description	Function /System
HIHRCPORID-4:3-B	Operator fails to X-tie Unit 1 and Unit 2 batteries	HI
CTHICPPS4:4-ALT	Operator fails to align condensate transfer alternate injection	Va
FPHICPPS4:2RCIC1	Operator fails to align fire protection after RCIC fails due to suppression pool temperature	Va
FPHICPPS4:2RCIC2	Operator fails to align fire protection after RHR fails due to MCC HVAC failure	Va
FPHICPPS4:2RCIC3	Operator fails to align fire protection After HPCS fails	Va
FPHICPPS4:2RCIC4	Operator fails to align fast fire protection alternate injection	Va
FWHICPSN27-4:LIA	Operator fails to control reactor feed booster pump during a loss of instrument air transient	Va
FWHICPSN27-4:1IA	Operator fails to control the reactor feed booster pump following loss of Instrument Air in a time frame greater than 2 hrs.	Va
SPHICPPS4:5SPCU(L)	Operator fails to align suppression pool clean up alternateinjection (late injection)	Va (Va,Li)
CVHICRPS7:3G41T	Operator fails to locally open 1G41-F0145	Y
ADHICREC2-ADS-R	Operator fails to depressurize after core damage having failed to depressurize early	X3
SLHICREQ-6-SLCR	Operator fails to initiate SLC given early failure to initiate	C1'

Table 3.4.1-13

Inmpact of Recovery Actions on Core Damage Frequency

Recovery Action	Core Damage Frequencies With Recovery	Core Damage Frequencies Without Recovery
1. Use of Alternate Low Press Injection	1.2E-5	2.6E-5
2. Cross connect DC supplies from Unit 2 to Unit 1	1.2E-5	2.3E-5
3. Operator fails to open 1G41-F0145	1.2E-5	1.6E-5
4. All the above actions	1.2E-5	4.2E-5

Table 3.4.1-14

Sensitivity Analysis Results

Base Case CDF 1.2E-5*

	Assumption	CDF	% Change
1.	Initiating event frequencies based on Perry experience (with lower limit of 0.5)	7.0E-6	-40%
2.	Automatic ADS Inhibit following ATWS	8.8E-6	-23%
3.	Operator does not override RCIC leak detection temp trip	1.8E-5	+54%
4.	Switchgear room cooling is required	1.5E-5	+25%
5.	Containment failure does not lead to loss of injection or passive vent to prevent containment failure	9.0E-6	+22%
6.	Contribution of maintenance to core damage reduced to zero	6.0E-6	-45%
7.	Human reliability data increased by a factor of 10 or to a maximum of 1.0	4.3E-4	+3,584%
8.	Perfect operator performance	3.9E-6	-66%
9.	Manual initiation of ADS following loss of all high pressure injection	1.3E-5	+13%
10.	Increase common cause by a factor of 10	2.2E-5	+91%
11.	Passive vent	9.1E-6	-22%

* Excludes flooding

Table 3.4.1-15

Core Damage Frequency Based
on Revised Initiating Event Frequencies

Station Blackout		
B	2.11E-6	30.2%
BP1	8.36E-8	1.2
BP2	5.91E-8	0.8
Loss of Offsite Power		
R	7.19E-7	10.3
U	4.19E-7	5.9
T1	1.80E-7	2.6
T1P1/P2	1.15E-7	1.7
ATWS		
T2-C	7.07E-7	10.1
T3B-C	3.56E-7	5.1
T3C-C	9.38E-8	1.3
T1-C	3.61E-8	0.6
Transients		
T1A	1.01E-6	14.5
T2	5.05E-7	7.2
T3C	1.38E-7	2.0
TSW	6.68E-8	1.0
LOCA's		
A	2.11E-7	3.0
S1	6.18E-8	0.9
S2	3.24E-8	0.5
Total	6.99E-6	

Table 3.4.2-1

Accident Sequence Grouping Criteria

Functional Accident Sequence	Definition	CDF	Percent CDF
1A	Accident Sequences Involving Loss of Coolant Inventory Makeup in Which Reactor Pressure Remains High.	7.0E-8	<1
1B	Accident Sequence Involving a Loss of AC Power and Loss of Coolant Inventory Makeup.	1.5E-6	13
1C	Accident Sequence Involving a Loss of All AC Power and No Recovery of AC Power.	1.2E-6	10
1D	Accident Sequences Involving a Loss of Coolant Inventory Makeup and ATWS.	4.4E-7	4
1E	Accident Sequence Involving a Loss of Coolant Inventory Makeup in which Reactor Pressure has been successfully reduced.	9.7E-7	8
22	Accident Sequences Involving Loss of Containment Heat Removal Leading to Containment Failures and Subsequent Loss of Coolant Inventory Makeup.	2.5E-6	22
3A	Vessel Rupture Leading beyond makeup capability.	<1.0E-7	<1
3B	Accident Sequence Initiatied or resulting in a small or medium LOCA for which reactor cannot be depressurized and inventory makeup is inadequate.	<1.0E-7	<1
3C	Accident sequences initiatied or resulting in medium or large LOCA for which the reactor is at low pressure and inadequate coolant inventory makeup is available.	3.9E-7	3
3D	Accident sequences which are initiated by a LOCA or failure for which vapour suppression is inadequate.	<1.E-7	<1
4	Accident sequences involving an ATWS leading to containment failure due to high pressure and subsequent loss of inventory.	4.0E-6	34
5	Unisolated LOCA outside containment leading to loss of effective coolant inventory makeup.	<1E-7	<1

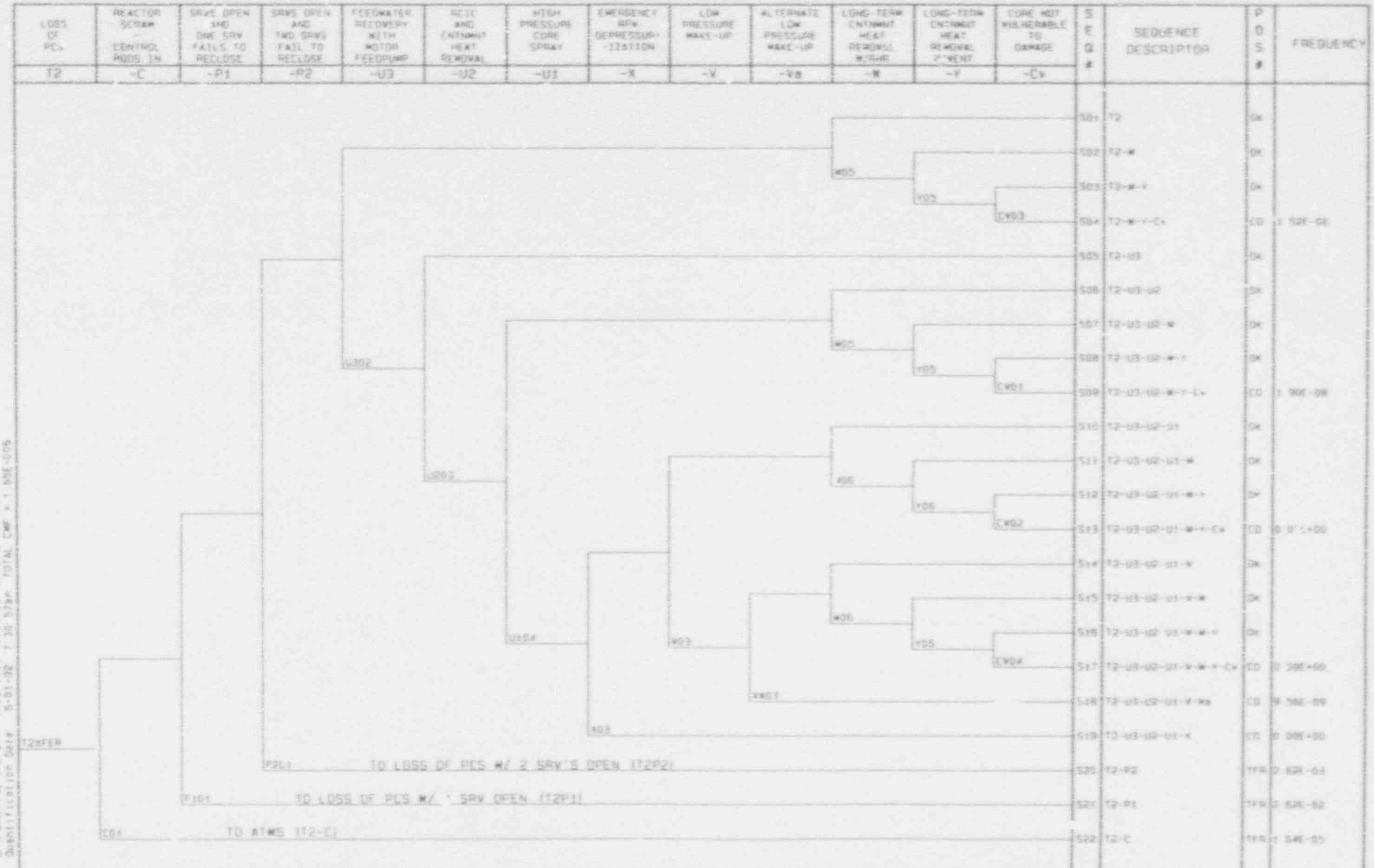
Table 3.4.2-2

Individual Core Damage Sequences Grouped by Accident Class

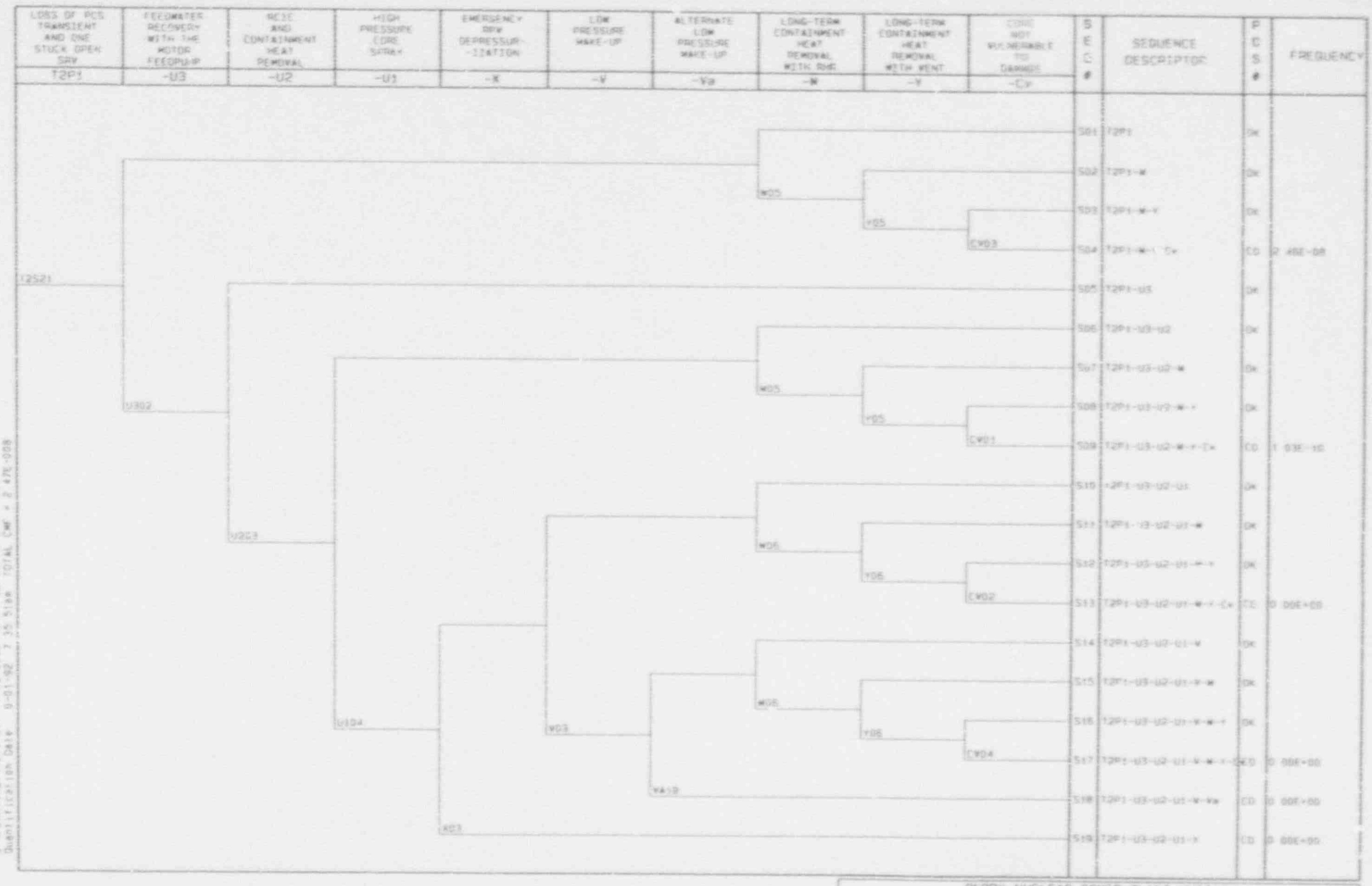
Group	Sequences	Frequency	Percent CDF
1A	BS35	5.15E-8	
	BS30	1.81E-8	
		6.96E-8	< 1
1B	BS34	5.26E-7	
	US29	3.34E-7	
	US12	5.99E-8	
	T1S35	1.33E-8	
	RS20	6.04E-7	
		1.54E-6	13
1C	BS24	7.71E-7	
	BS17	3.36E-7	
	BS12	1.04E-7	
	BP1S26	1.13E-8	
		1.22E-6	10
1D	T2-CS28	3.12E-7	
	T3B-CS09	5.32E-8	
	T3B-CS27	3.80E-8	
	T3E-CS08	1.78E-8	
	BS30	1.81E-8	
		4.[illegible]9E-7	4
1E	TIAS14	7.53E-7	
	US12	5.99E-8	
	BS22	5.96E-8	
	TSWS10	5.49E-8	
	T1S35	1.33E-8	
		9.41E-7	8
2	T2S04	1.62E-6	
	BS07	1.60E-7	
	T1AS05	2.57E-7	
	T1S08	1.47E-7	
	T3CS04	1.38E-7	
	[illegible]S19	[illegible]	
	BS33	[illegible]	
	RS10	[illegible]	
	T1S04	1.94E-8	
	T[illegible]S09	1.90E-3	
	US[illegible]8	1.63E-8	
		2.54E-6	22

Table 3.4.2-2 continued

3C	AS09	2.10E-7	
	S1S13	5.85E-8	
	T1P1S31	7.24E-8	
	T1P2S11	2.39E-8	
	BP1S17	1.69E-8	
	T1P1US22	1.37E-8	
		3.95E-7	3
4	T2-CS30	2.27E-6	
	T2-CS20	6.25E-7	
	T3B-CS29	2.76E-7	
	T2-CS11	2.90E-7	
	T2-CS12	2.37E-7	
	T3C-CS27	5.49E-8	
	T3B-CS19	7.60E-8	
	T3B-CS10	3.45E-8	
	TIACS09	3.31E-8	
	T3B-CS11	2.88E-8	
	T1-CS09	2.19E-8	
	T36-CS17	1.40E-8	
		3.96E-6	34

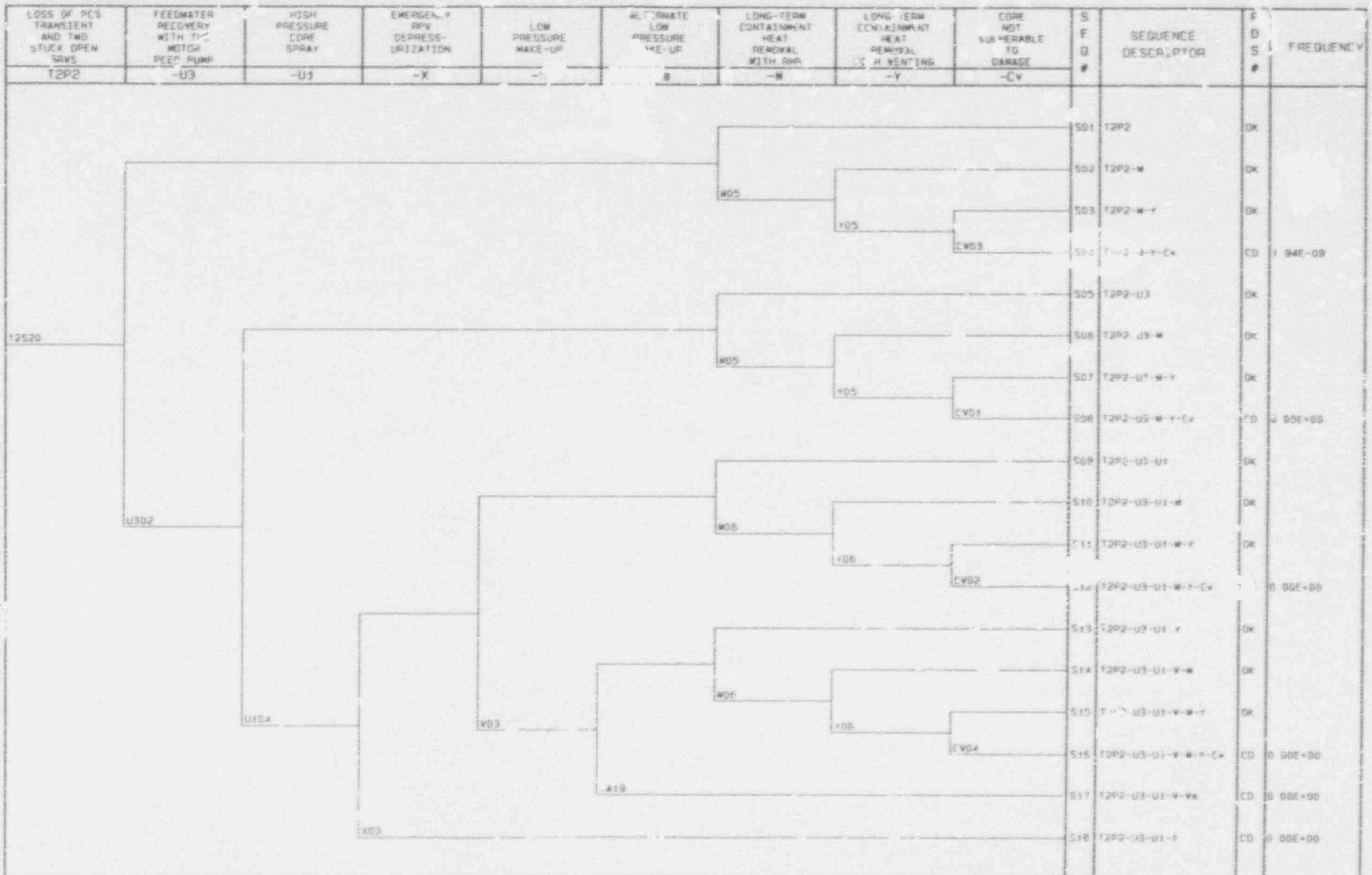


PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.2-1 SHEET 1 OF 3
LOSS OF PCS (T2) EVENT TREE
REVISION 2 (5/10/91)



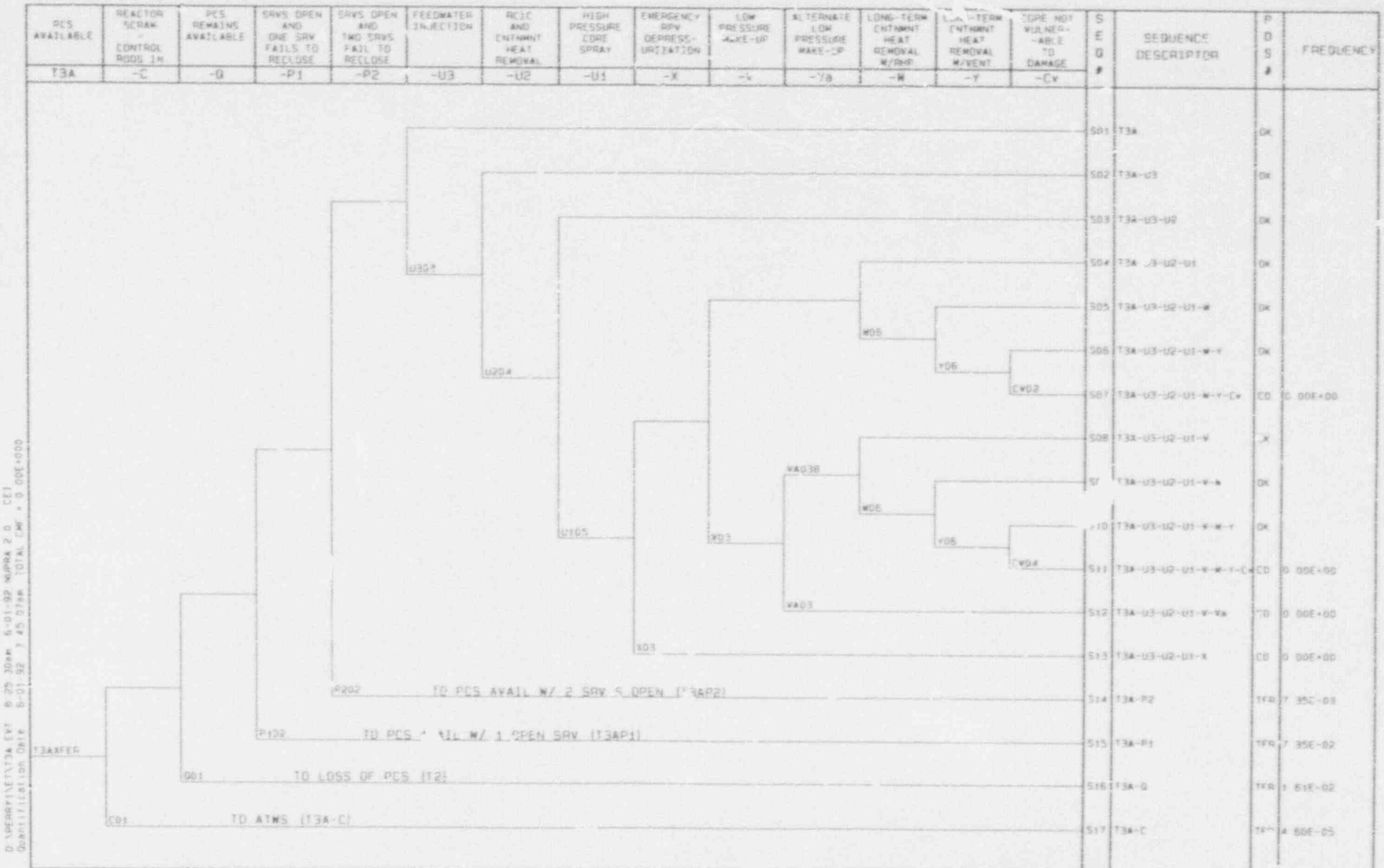
D:\PERRY\LE1\T2P1.EVT 7:42:50am 6-01-92 NUPRA 2.0 CEI
Quantification Date 6-01-92 7:35:51am TOTAL CMF = 2.47E-008

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.2-1 SHEET 2 OF 3
LOSS OF PCS & ONE STUCK OPEN SRV
REVISION 1 (5/15/91)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.2-1 SHEET 3 OF 3
LOSS OF PCS & TWO STUCK OPEN SRVs
REVISION 2 (03/12/92)



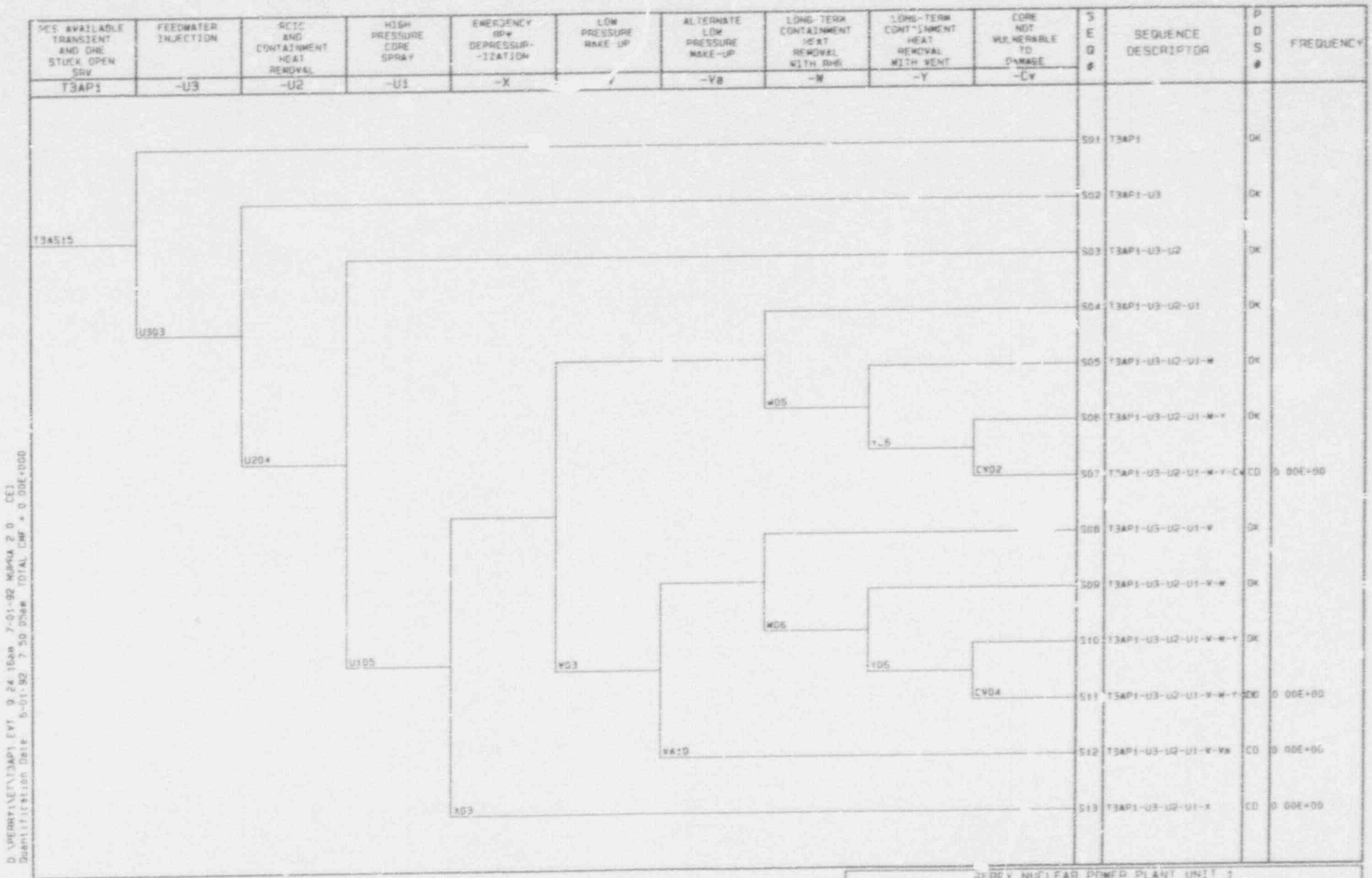
PERRY NUCLEAR POWER PLANT UNIT 1

INDIVIDUAL PLANT EXAMINATION

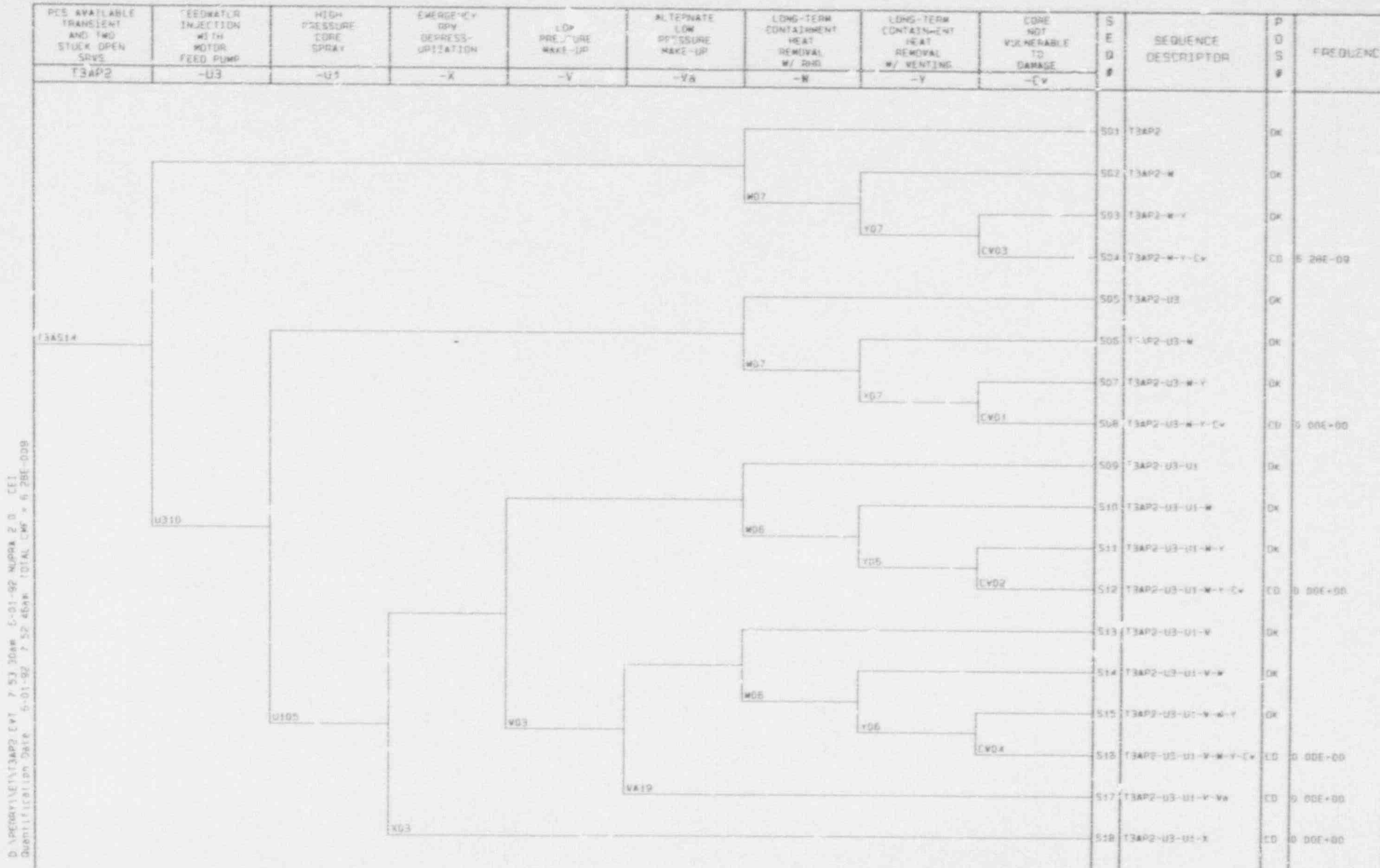
FIGURE 3 1 2-2 SHEET 1 OF 3

PCS AVAILABLE (T3A) EVENT TREE

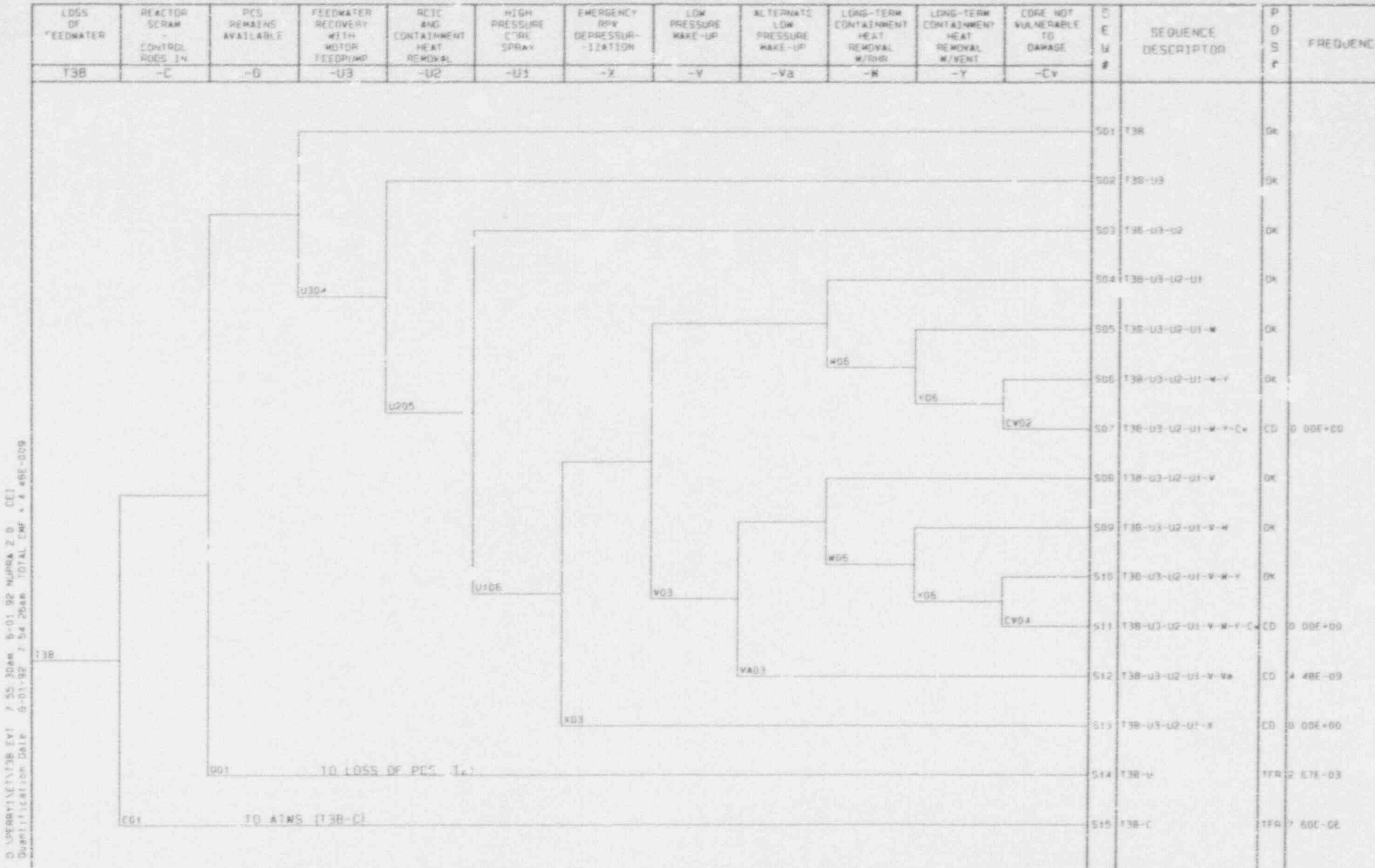
REVISION 2 (5/10/91)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.2-2 SHEET 2 OF 3
PCS AVAILABLE & ONE STUCK OPEN SRV
REVISION 1 (5/15/91)



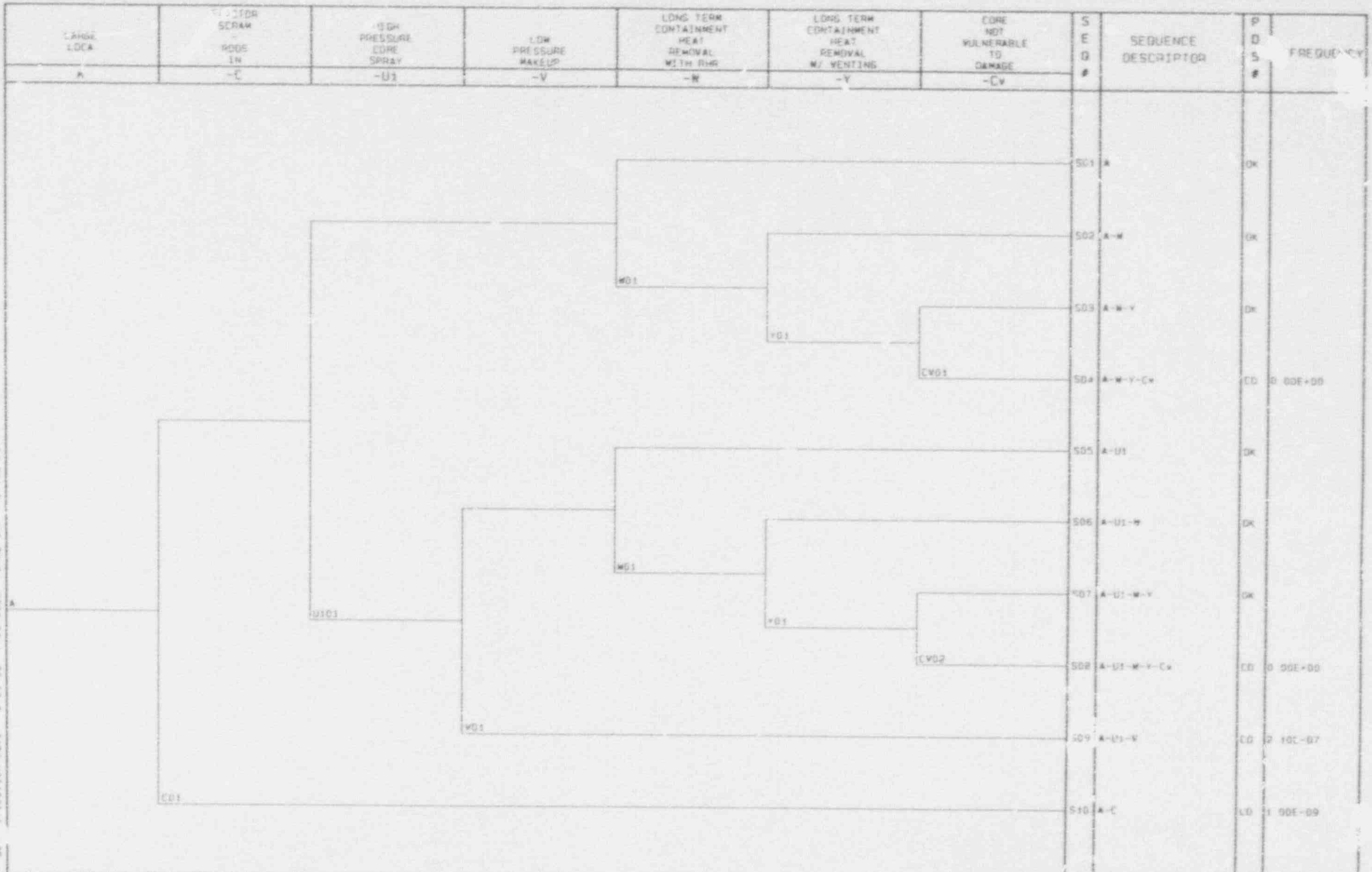
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.2-2 SHEET 3 OF 3
PCS AVAILABLE & TWO STUCK OPEN SRVs
REVISION 2 (12/06/91)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

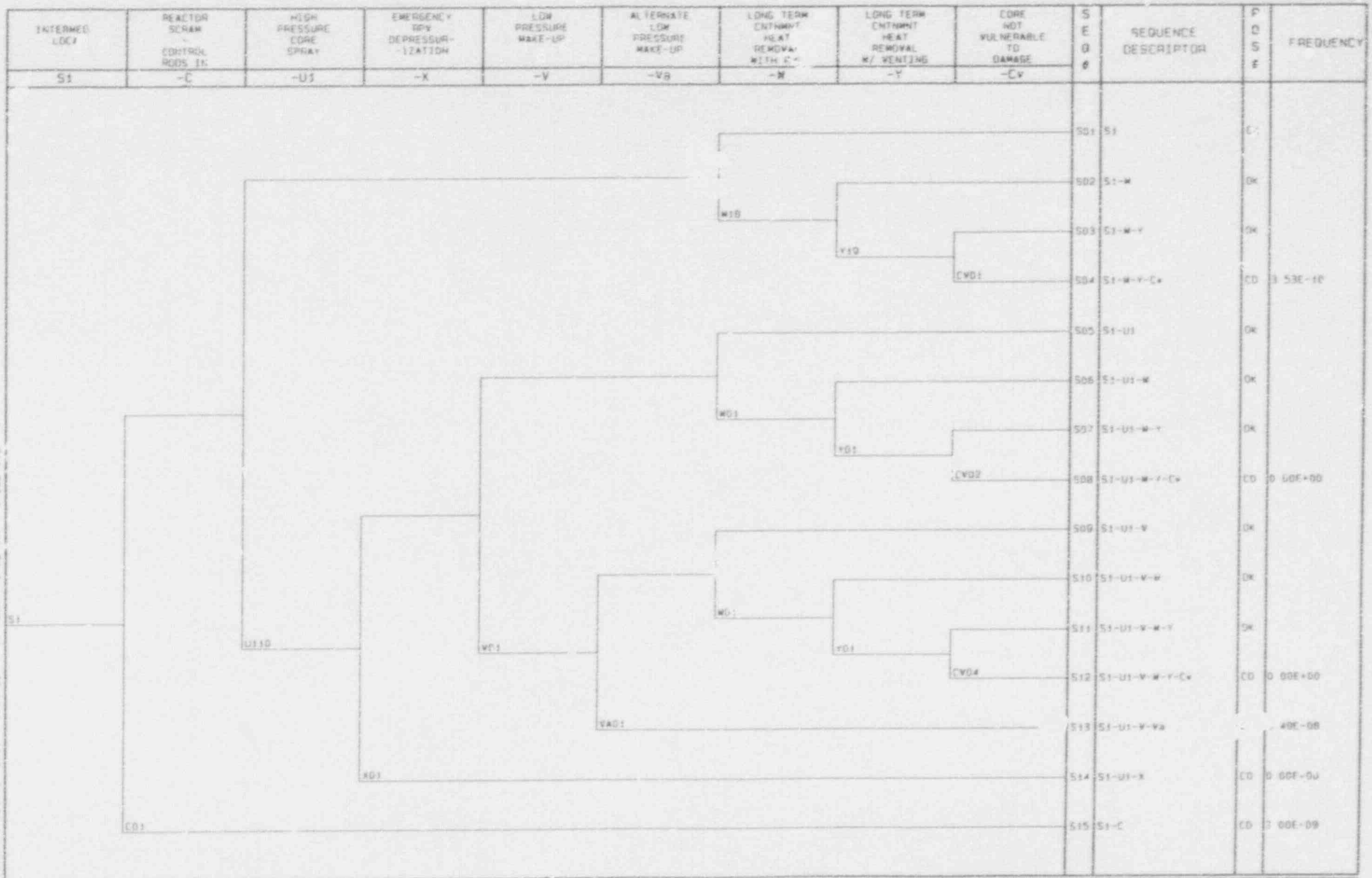
FIGURE 3.1.2-3
LOSS OF FEEDWATER (T3B) EVENT TREE
REVISION 2 (5/10/91)

D:\PERRY\ET\A.EVT 7.02.22am 6-01-92 NUPRA 2.0 CEI
Modification Date 6-01-92 7.02.20am TOTAL CMF = 2.11E-007



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

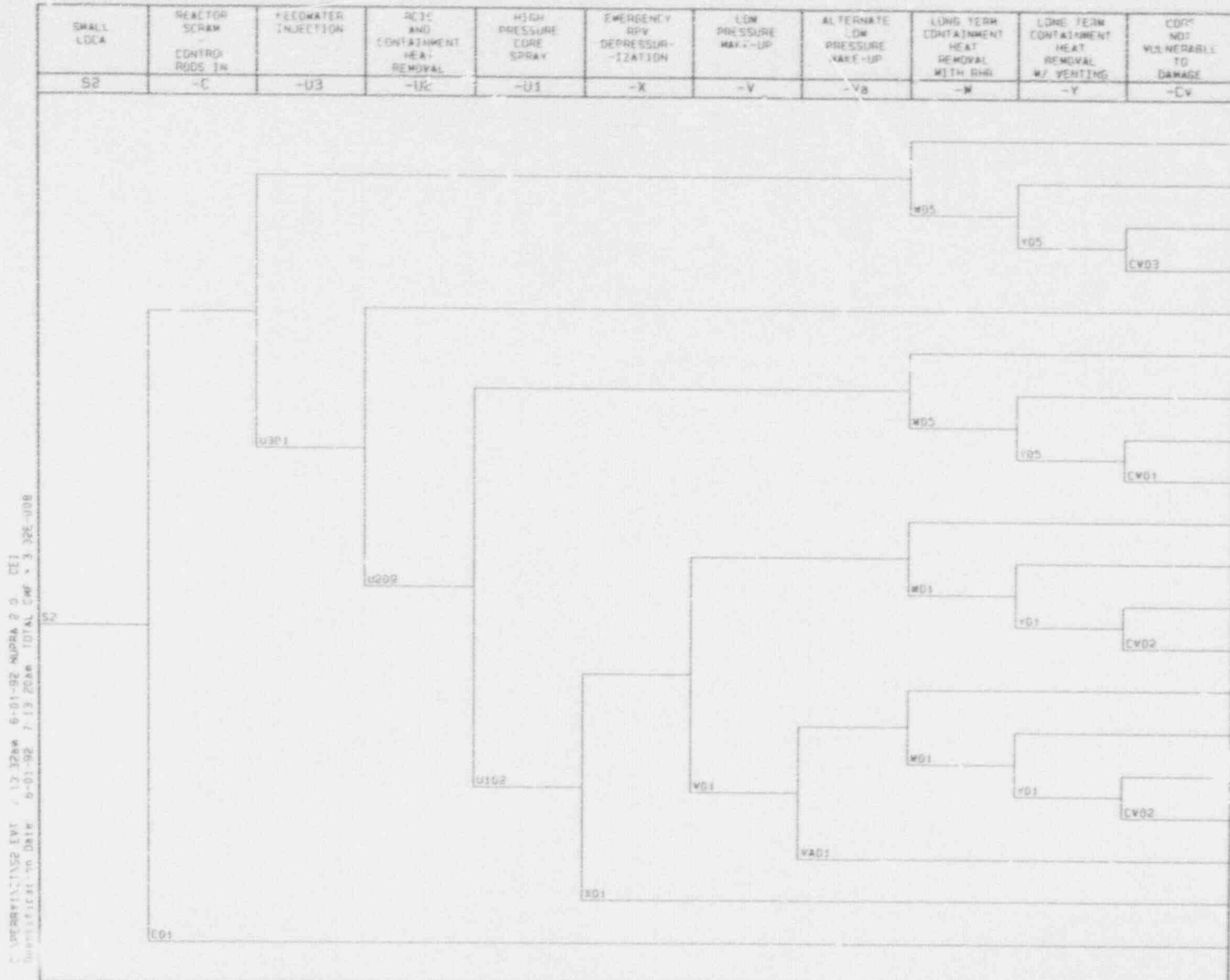
FIGURE 3.1.2-4
LARGE LOCA (A)
REVISION (5/1/91)



J \PERRY1\ET\S1 EVT 7 12 00am 6-01-92 NUPRA 2 0 CEI
Quantification Date 6-01-92 7 11 45am TOTAL CMF - 8 03E-008

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3 1 2-5
INTERMEDIATE LOCA (S1)
REVISION 1 (05/14/91)



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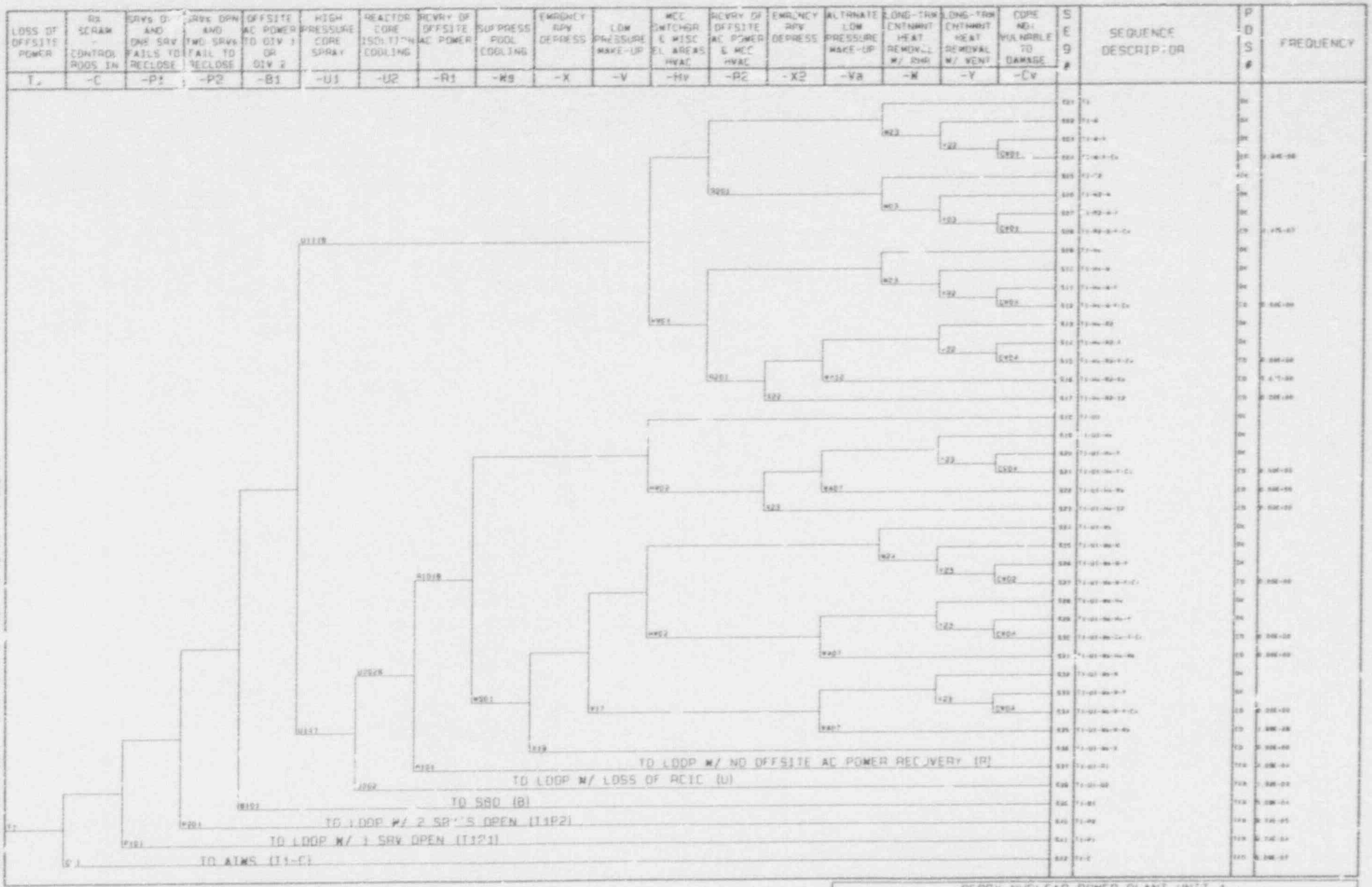
S02	S2-W	OK	
S03	S2-W-Y	OK	
S04	S2-W-Y-CV	CD	2 35E-09
S05	S2-U3	OK	
S06	S2-U3-U2	OK	
S07	S2-U3-U2-W	OK	
S08	S2-U3-U2-W-Y	OK	
S09	S2-U3-U2-W-Y-CV	CD	0 00E+00
S10	S2-U3-U2-U1	OK	
S11	S2-U3-U2-U1-W	OK	
S12	S2-U3-U2-U1-W-Y	OK	
S13	S2-U3-U2-U1-W-Y-CV	CD	0 00E+00
S14	S2-U3-U2-U1-V	OK	
S15	S2-U3-U2-U1-V-W	OK	
S16	S2-U3-U2-U1-V-W-Y	OK	
S17	S2-U3-U2-U1-V-W-Y-CV	CD	0 00E+00
S18	S2-U3-U2-U1-V-V8	CD	8 57E-10
S19	S2-U3-U2-U1-X	CD	0 00E+00
S20	S2-C	CD	3 00E-08

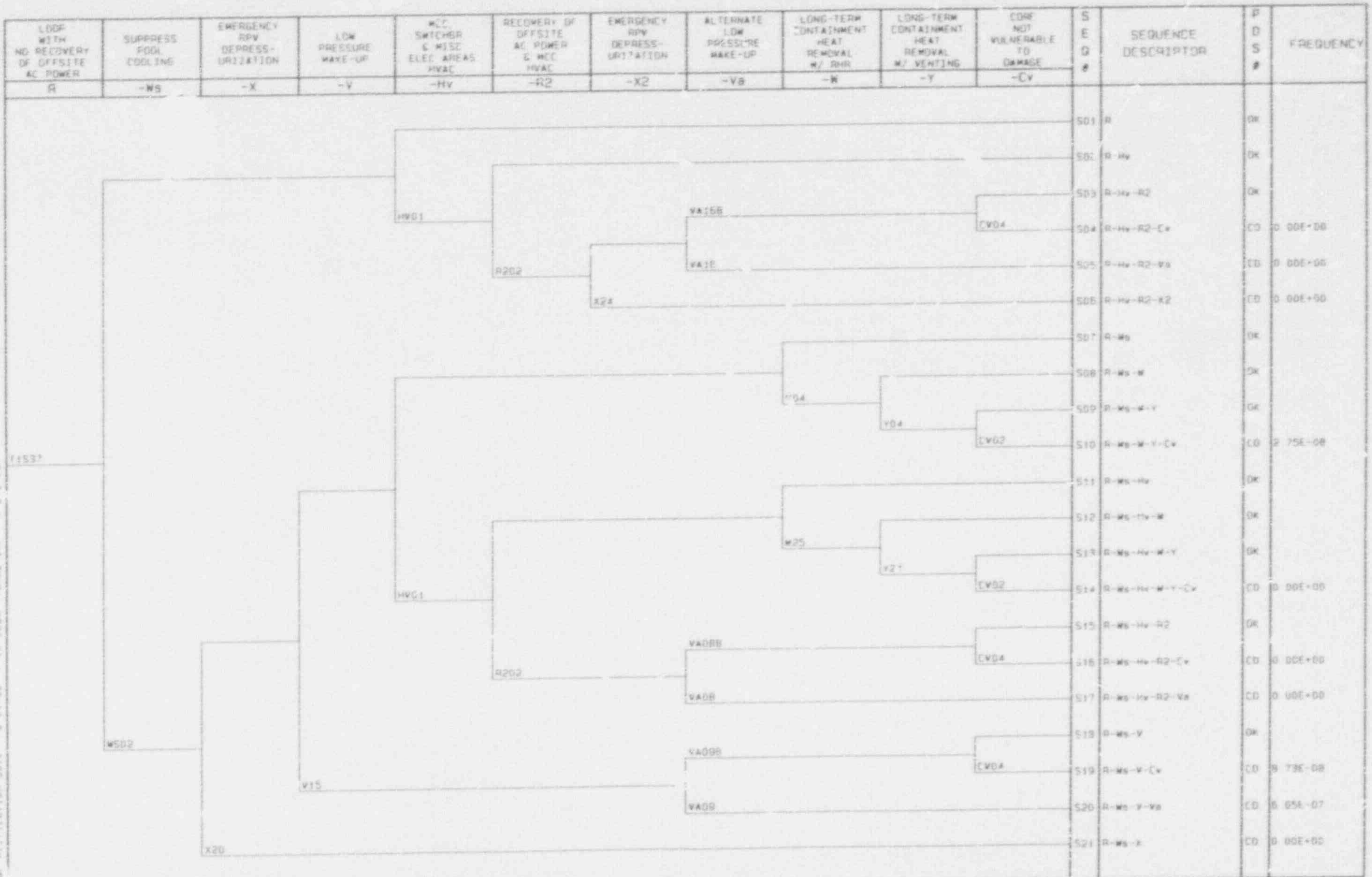
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3 1 2-6
SMALL LOCA (S2)
REVISION 0 (5/8/91)

D:\PERRY\ET\T1.EVT 7 17 10am 6-01-92 NUPRA 2.0 (CE)
Quantification Date 6-01-92 7 17 05am TOTAL CMF = 1.85E-007



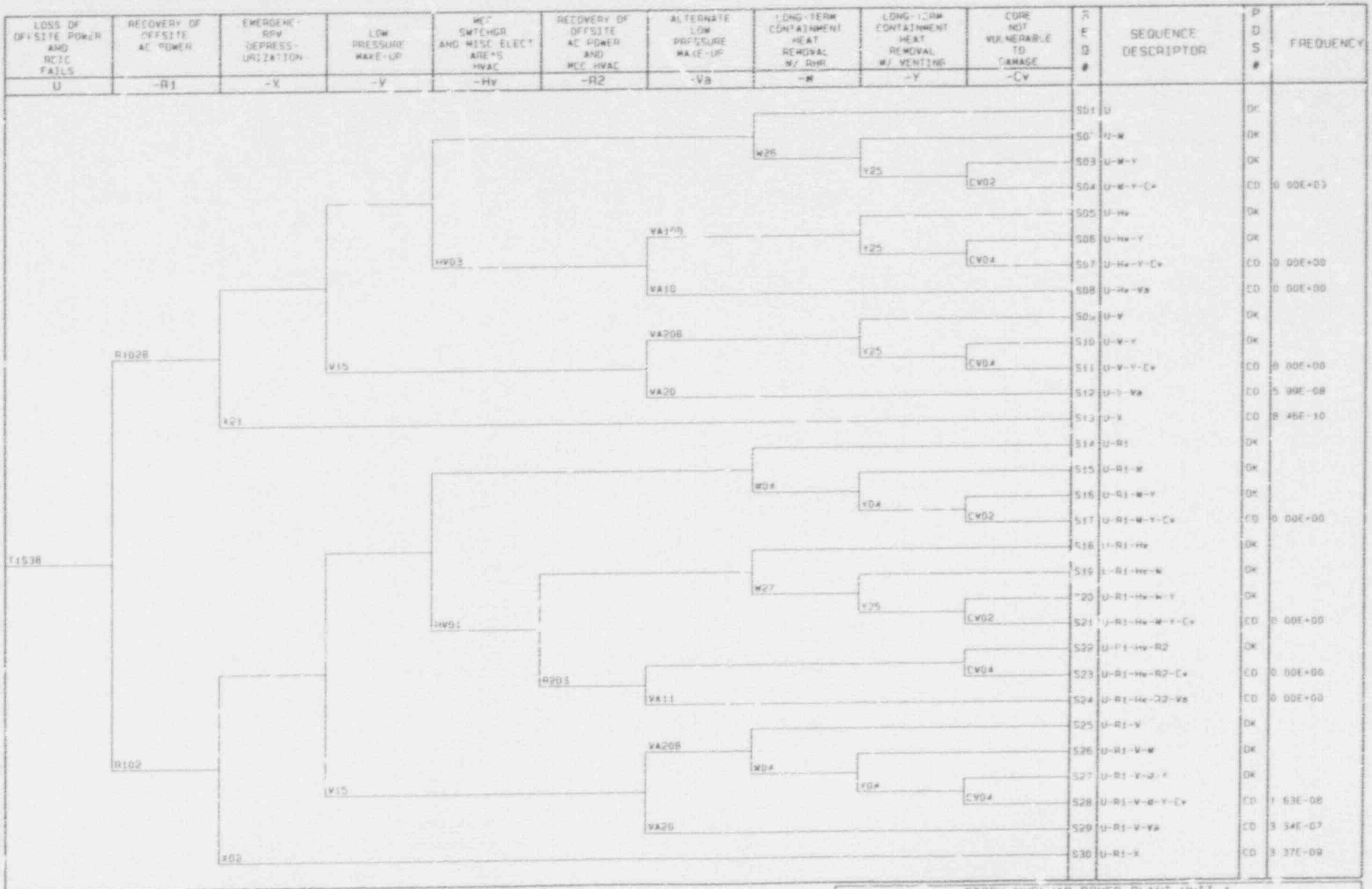
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.3-1 SHEET 1 OF 6
LOSS OF OFFSITE POWER (T1) EVENT TREE
REVISION 3 (05/11/92)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.3-1 SHEET 2 OF 6
LOOP W/O RECOVERY OF OFFSITE AC POWER
REVISION 0 (12/16/91)

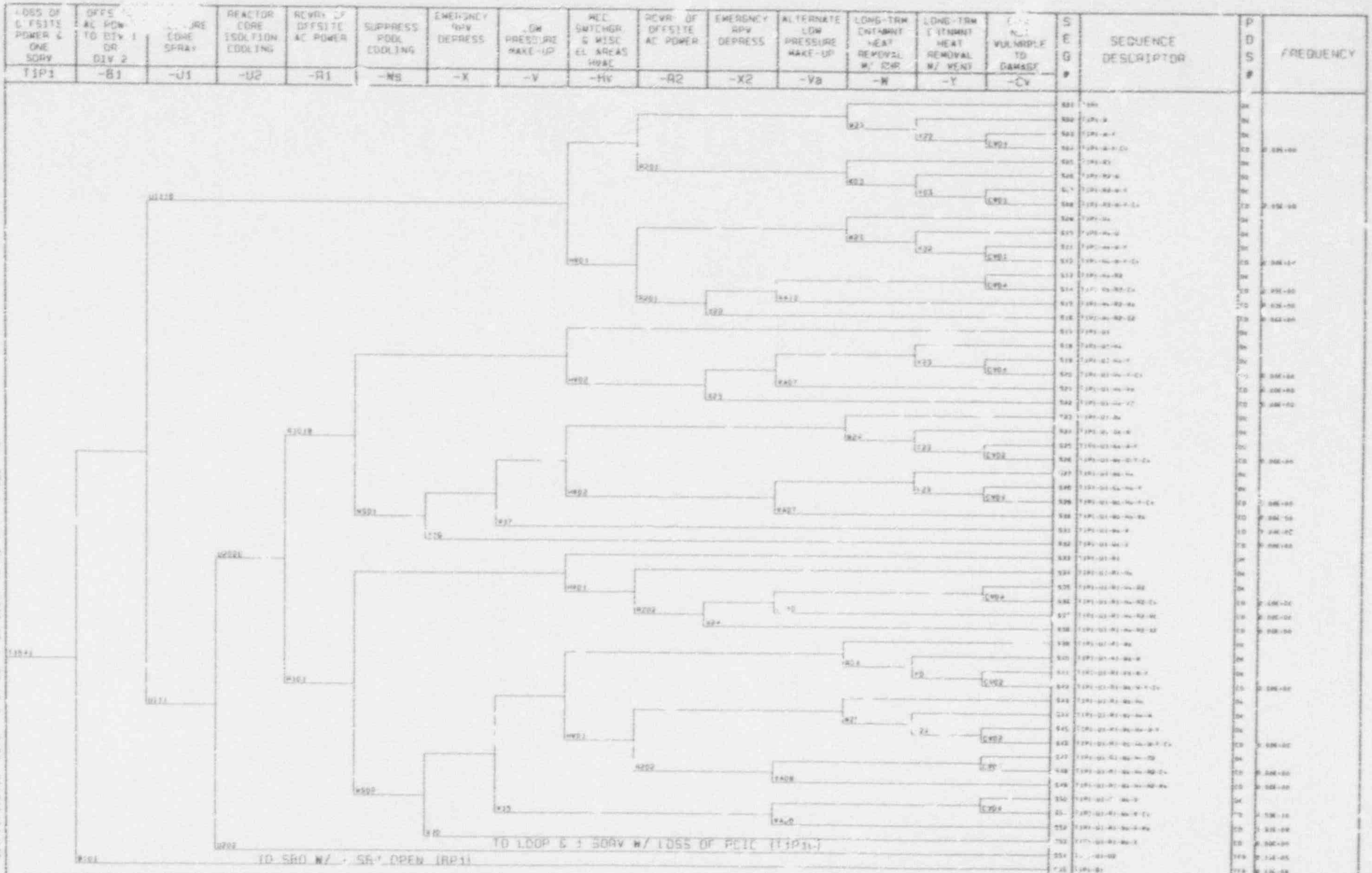
0 \PERRY\ET\R EXT 7-10 46am 6-01-92 NUPRA 2.0 CE1
Quantification Date 6-01-92 7:10 35am TOTAL CMF = 7.29E-007



D:\PERRY\ET\U EVT 8 13 42am 6-01-92 NUPRA 2.0 [E]
Quantification Date 6-01-92 8 13 05am TOTAL CMF = 4.14E-007

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

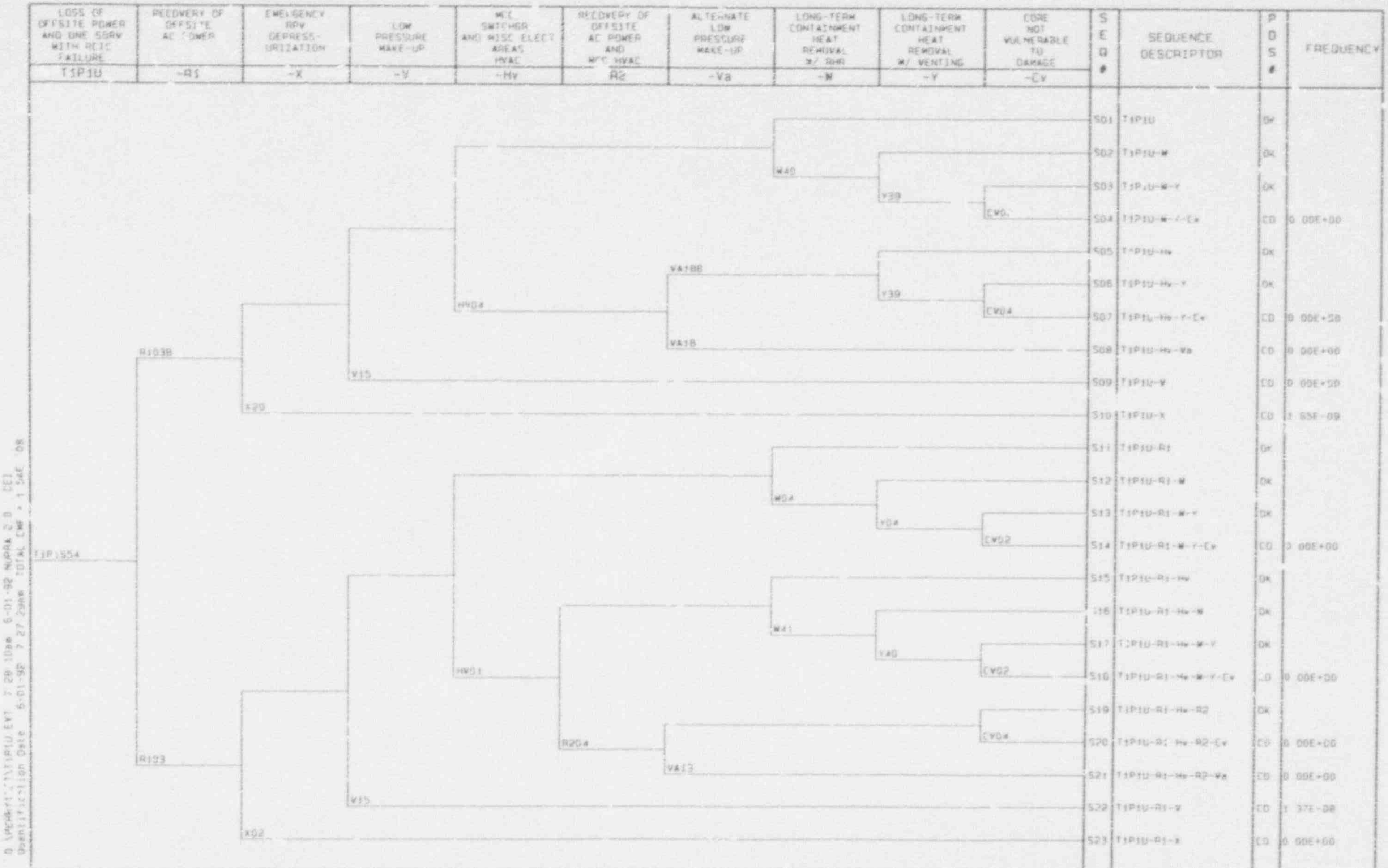
FIGURE 3.1.3-1 SHEET 3 OF 6
LOOP W/ RCIC FAILURE EVENT TREE
REVISION 1 (05/14/91)



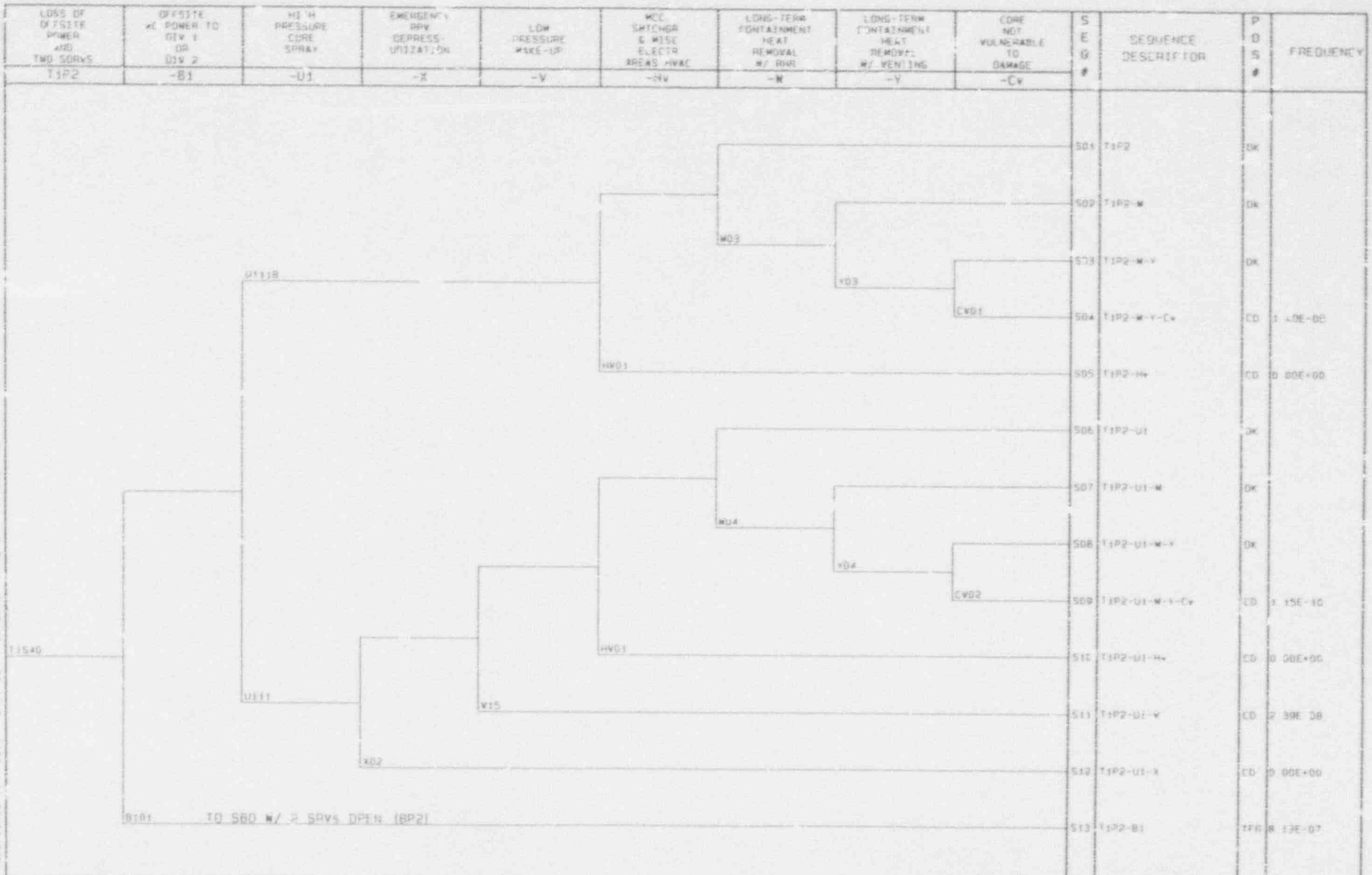
D:\PERRY\ET\T1P1.EVT 7:25 18am 6-01-92 NUPRA 2.0 CEI
Quantification Date 6-01-92 7:21 38am TOTAL CMF = 7.62E-08

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

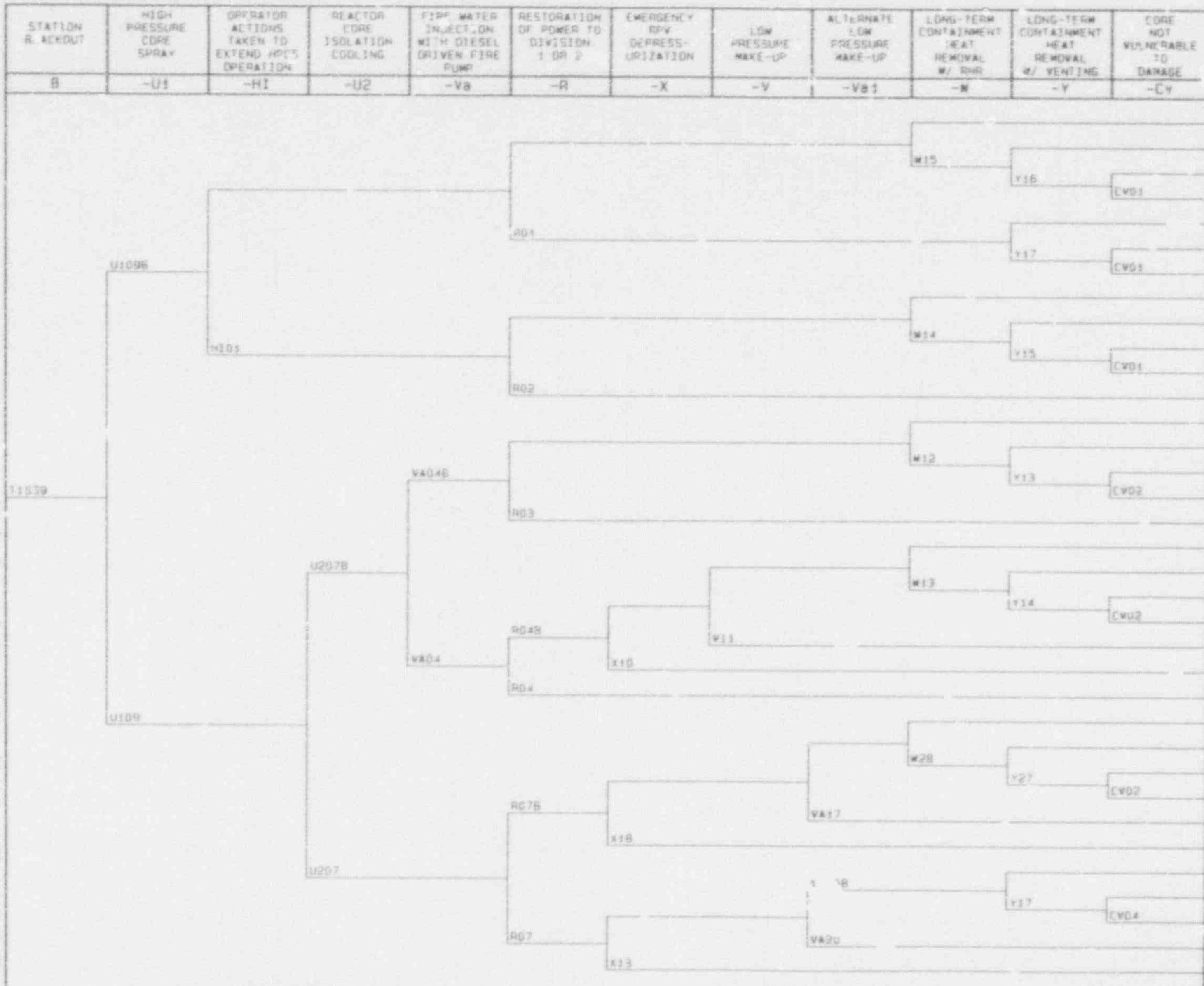
FIGURE 3.1.3-1 SHEET 4 OF 6
LOOP AND ONE STUCK OPEN SRV EVENT TREE
REVISION 3 (12/06/91)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.3-1 SHEET 5 OF 6
LOOP W/ 1 SORV & RCIC FAILURE E.T.
REVISION 0 (11/20/91)



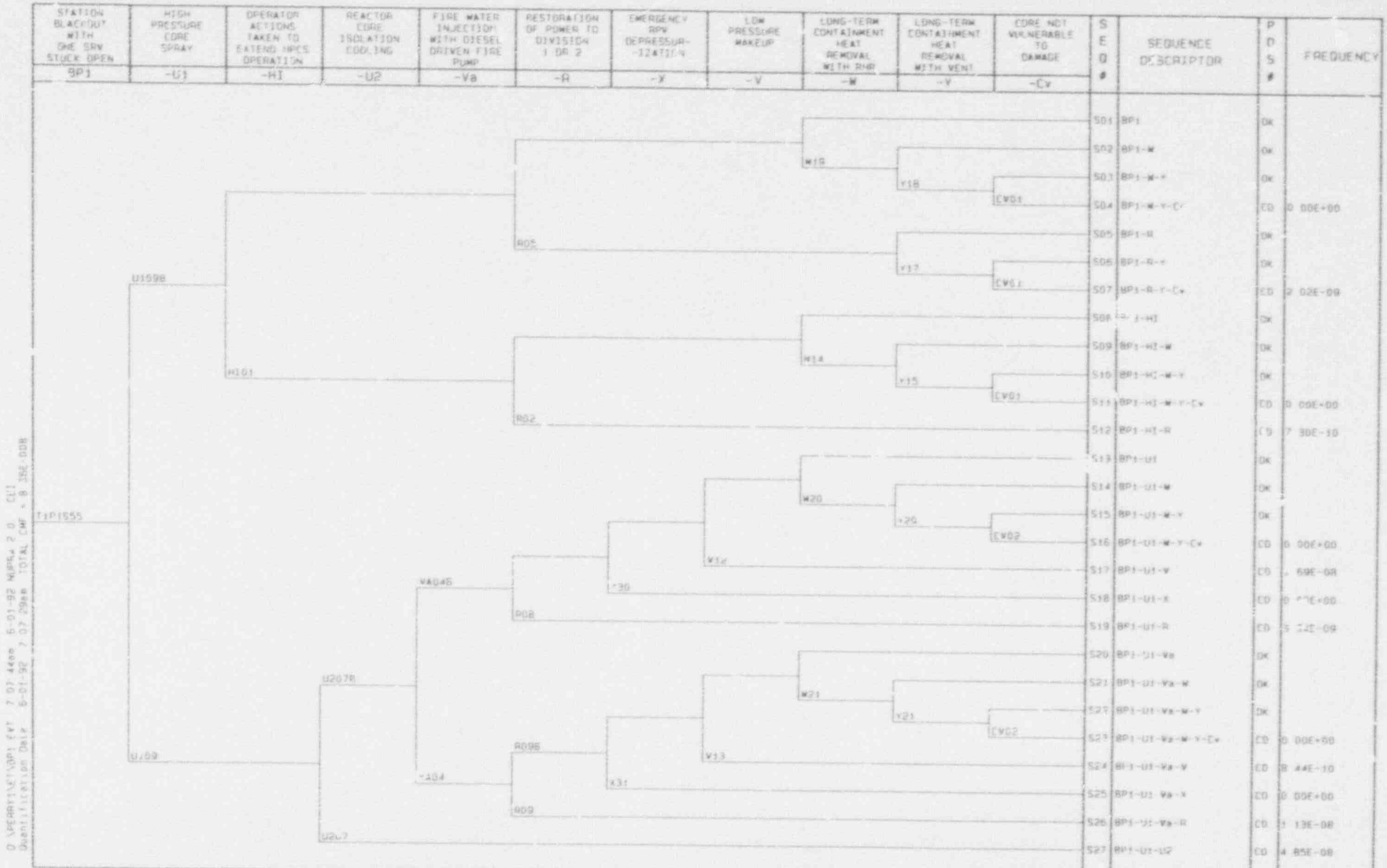
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.3-1 SHEET 6 OF 6
LOOP AND TWO STUCK OPEN SRVs
REVISION 2 (11/06/91)



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
S01	B	OK	
S02	B-W	OK	
S03	B-W-Y	OK	
S04	B-W-Y-Cv	CD	0 00E+00
S05	B-R	OK	
S06	B-R-Y	OK	
S07	B-R-Y-Cv	CD	1 60E-07
S08	B-HI	OK	
S09	B-HI-W	OK	
S10	B-HI-W-Y	OK	
S11	B-HI-W-Y-Cv	CD	0 00E+00
S12	B-HI-R	CD	1 13E-07
S13	B-U1	OK	
S14	B-U1-W	OK	
S15	B-U1-W-Y	OK	
S16	B-U1-W-Y-Cv	CD	0 00E+00
S17	B-U1-R	CD	3 36E-07
S18	B-U1-Va	OK	
S19	B-U1-Va-W	OK	
S20	B-U1-Va-W-Y	OK	
S21	B-U1-Va-W-Y-Cv	CD	0 00E+00
S22	B-U1-Va-V	CD	5 96E-08
S23	B-U1-Va-X	CD	0 00E+00
S24	B-U1-Va-R	CD	7 74E-07
S25	B-U1-U2	OK	
S26	B-U1-U2-W	OK	
S27	B-U1-U2-W-Y	OK	
S28	B-U1-U2-W-Y-Cv	CD	0 00E+00
S29	B-U1-U2-Va1	CD	4 73E-08
S30	B-U1-U2-X	CD	1 81E-08
S31	B-U1-U2-R	OK	
S32	B-U1-U2-R-Y	OK	
S33	B-U1-U2-R-Y-Cv	CD	5 11E-08
S34	B-U1-U2-R-Va1	CD	5 26E-07
S35	B-U1-U2-R-X	CD	5 15E-08

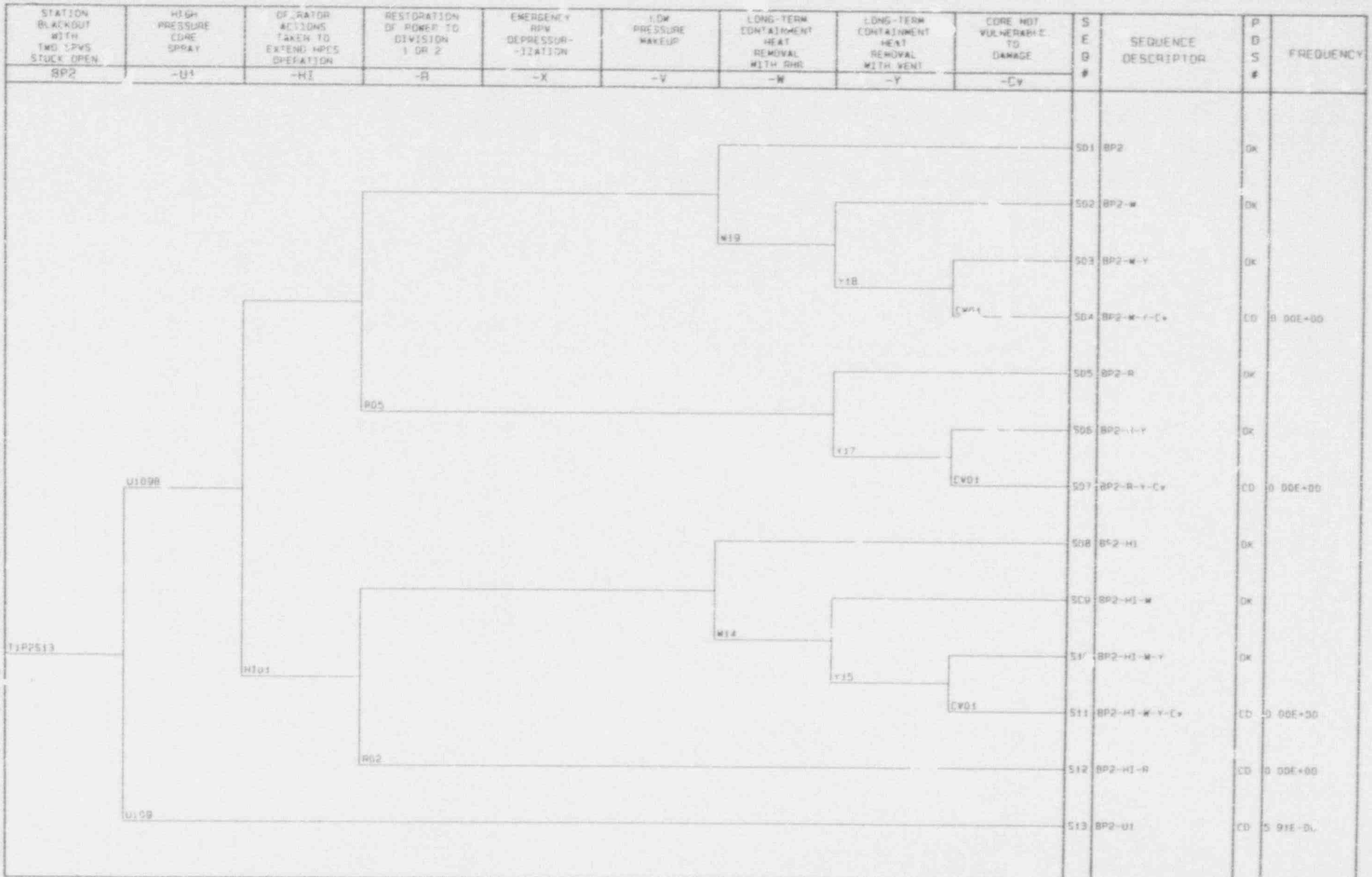
D:\PERRY\ET\B EVT 7 05 38am 5-01-92 NUPRA 2.0 CEI
Quantification Date 5-01-92 7 05 14am TOTAL CMF = 2 14E-006

PERRY NUCLEAR POWER PLANT
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.3-2 SHEET 1 OF 3
STATION BLACKOUT
REVISION 8 (05/14/92)



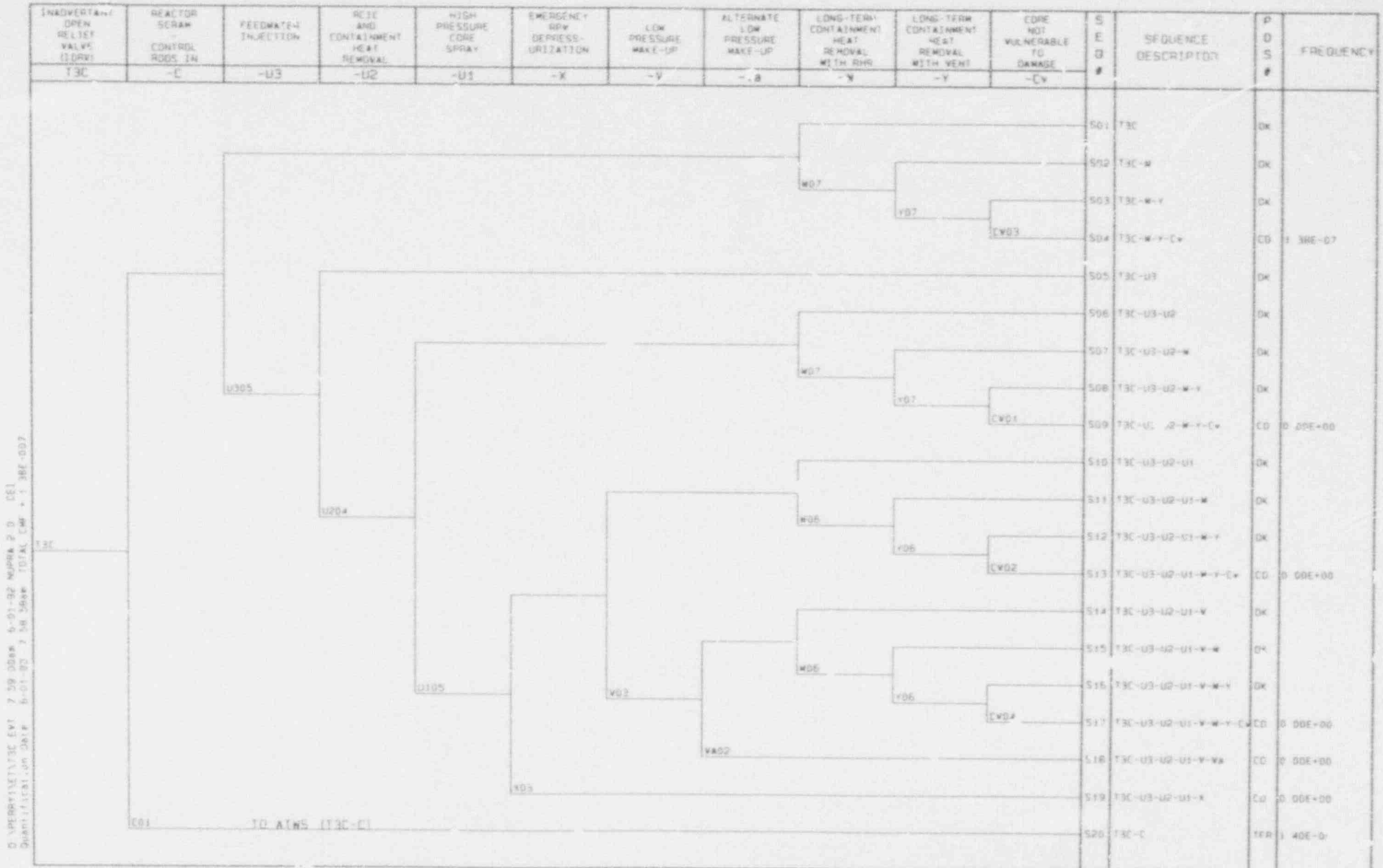
PERRY NUCLEAR POWER PLANT
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.3-2 SHEET 2 OF 3
STATION BLACKOUT WITH ONE SORV
REVISION 5 (03/11/92)



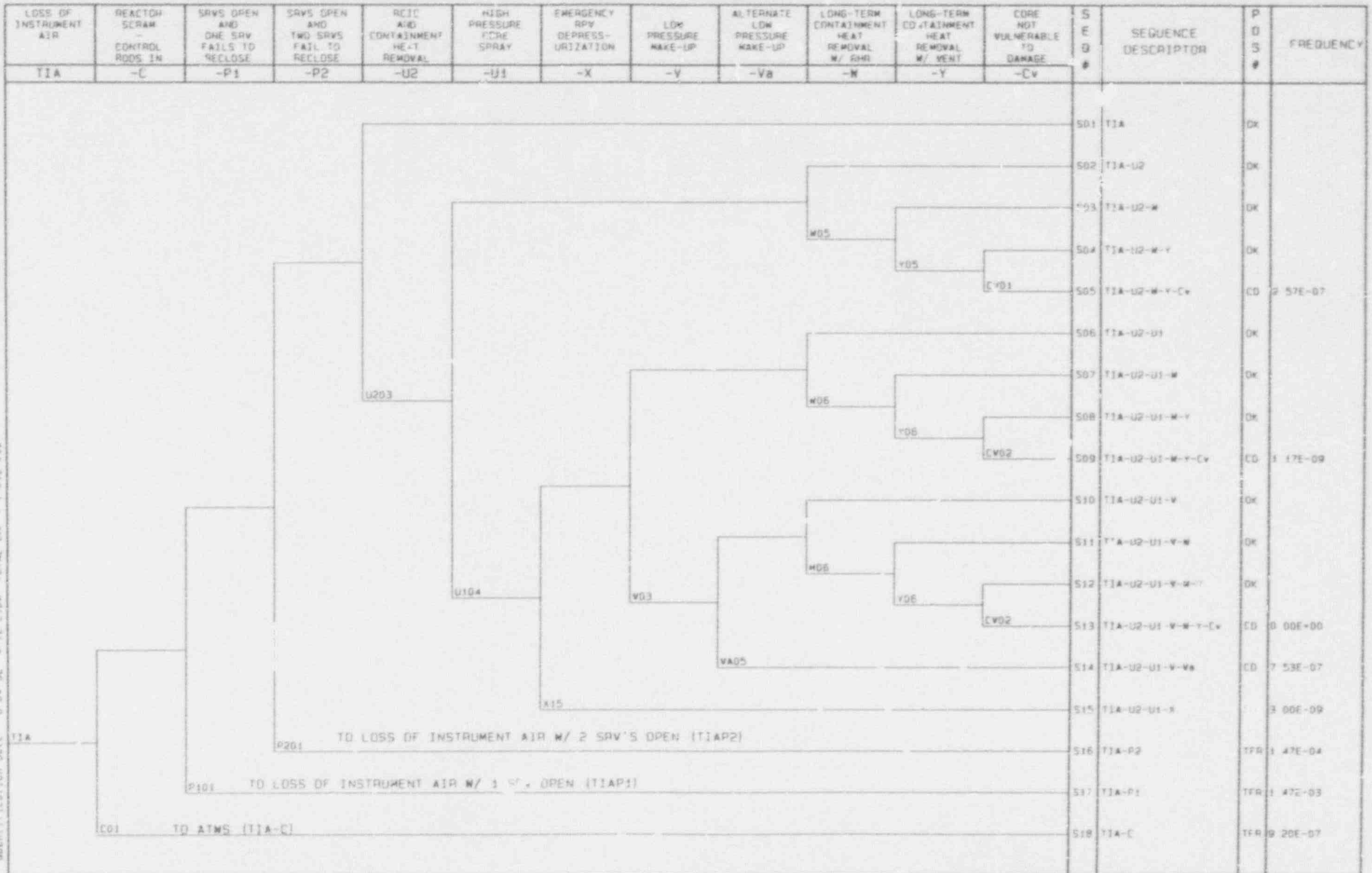
D:\PERRY1\ET\BP2.EVT 7:08:42am 6-01-92 NUPRA 2.0 CEI
Quantification Date: 6-01-92 7:08:33am TOTAL CMF = 5.91E-008

PERRY NUCLEAR POWER PLANT
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.3-2 SHEET 3 OF 3
STATION BLACKOUT WITH TWO SORVS
REVISION 5 (03/12/92)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

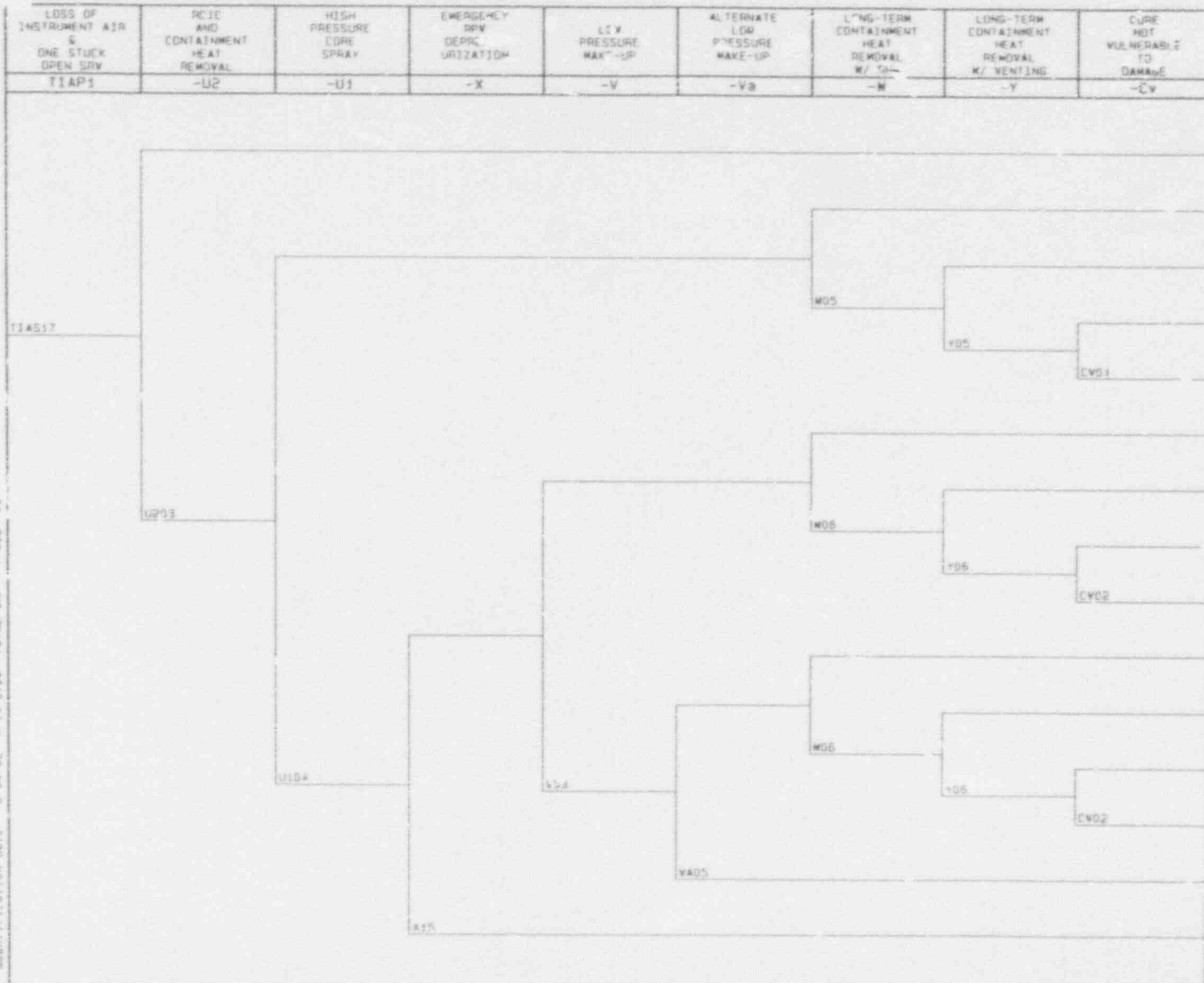
FIGURE 3.1.3-3
INADVERTENT OPEN RELIEF VALVE (T3C)
REVISION 2 (5/16/91)



D:\PERRY1\ET\TIA.EVT 4 12 22pm 6-04-92 NUPRA 2.0 CEI
Quantification Date 6-04-92 4 12 21pm TOTAL CMF = 1.01E-006

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.3-4 SHEET 1 OF 3
LOSS OF INSTRUMENT AIR EVENT TREE
REVISION 2 (11/19/91)



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
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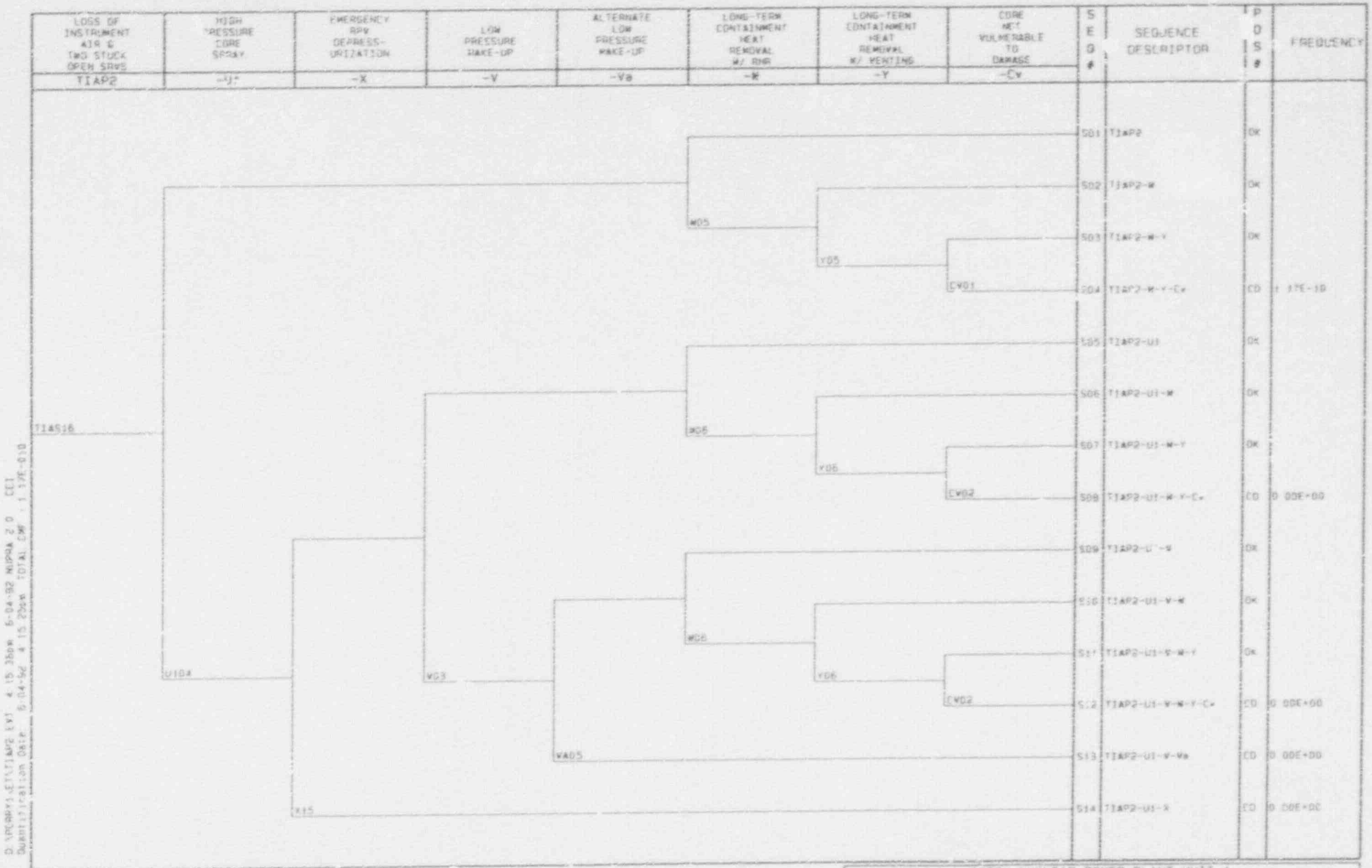
S02	TIAP1-U2	OK	
S03	TIAP1-U2-W	OK	
S04	TIAP1-U2-W-Y	OK	
S05	TIAP1-U2-W-Y-Cv	CD	3.01E-09
S06	TIAP1-U2-U1	OK	
S07	TIAP1-U2-U1-W	OK	
S08	TIAP1-U2-U1-W-Y	OK	
S09	TIAP1-U2-U1-W-Y-Cv	CD	0.00E+00
S10	TIAP1-U2-U1-V	OK	
S11	TIAP1-U2-U1-V-W	OK	
S12	TIAP1-U2-U1-V-W-Y	OK	
S13	TIAP1-U2-U1-V-W-Y-Cv	CD	0.00E+00
S14	TIAP1-U2-U1-V-Va	CD	4.01E-08
S15	TIAP1-U2-U1-X	CD	0.00E+00

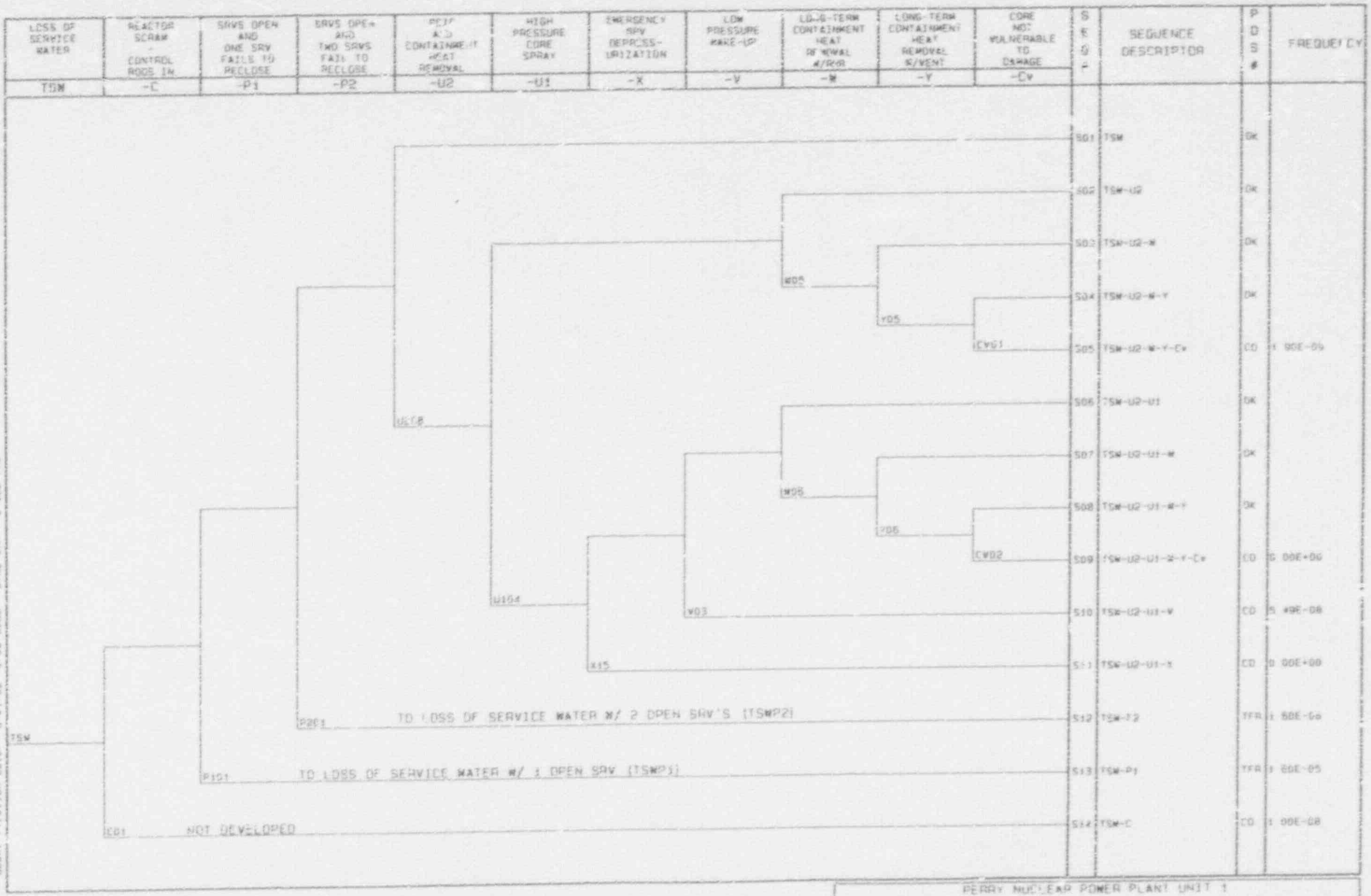
D:\PERRY1\EI\TIAP1.EVT 4:14 28pm 5-04-92 NUPRA 2.0 CEI
Quantification Date 5-04-92 4:14 27pm TOTAL CMF = 7.02E-09

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.3-4 SHEET 2 OF 3
LOSS OF INSTRUMENT AIR & ONE SORV
REVISION 2 (11/19/93)

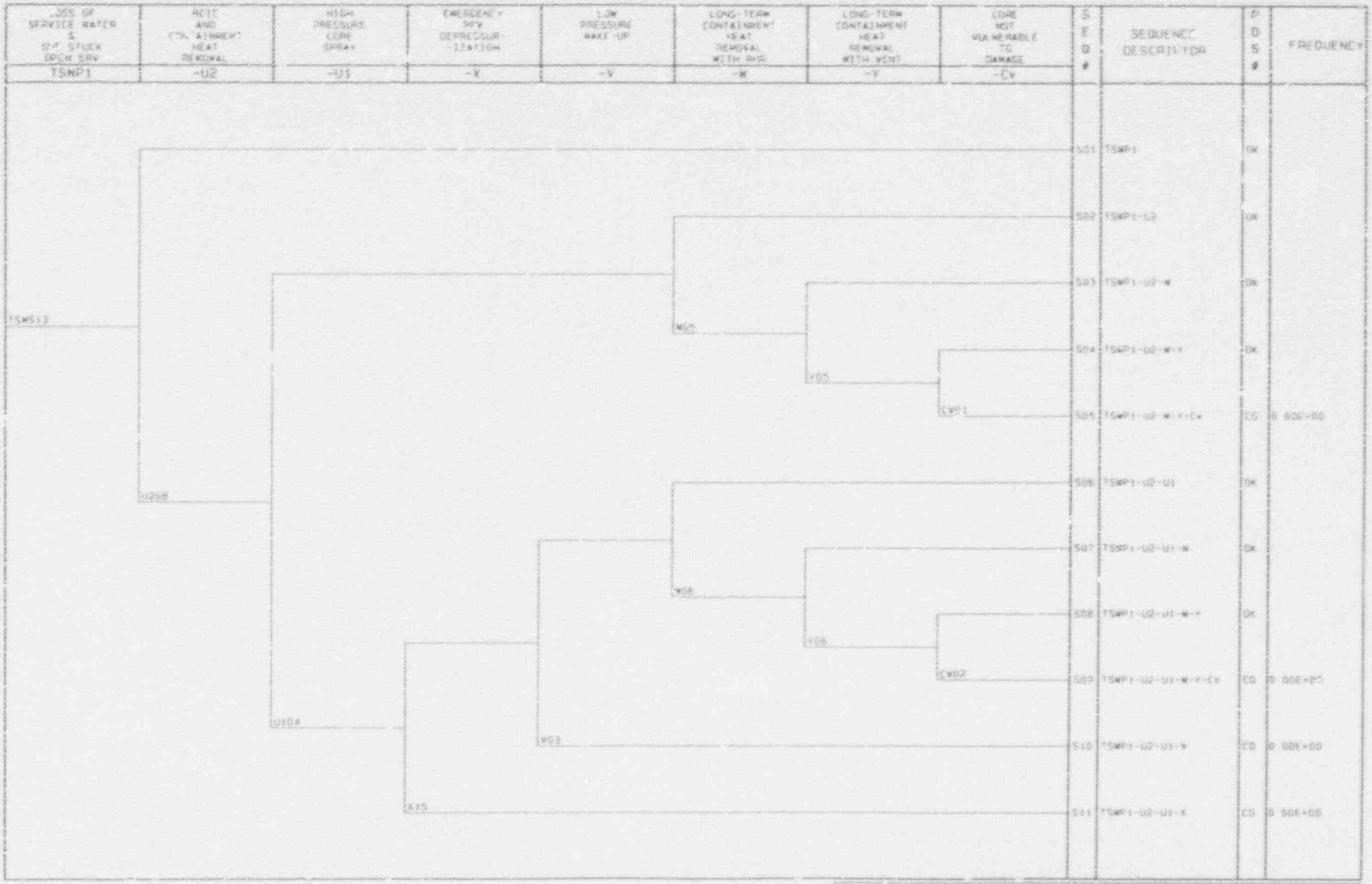


PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3 1 3-4 SHEET 3 OF 3
LOSS OF INSTRUMENT AI. & TWO SORVs
REVISION 2 (11/19/91)

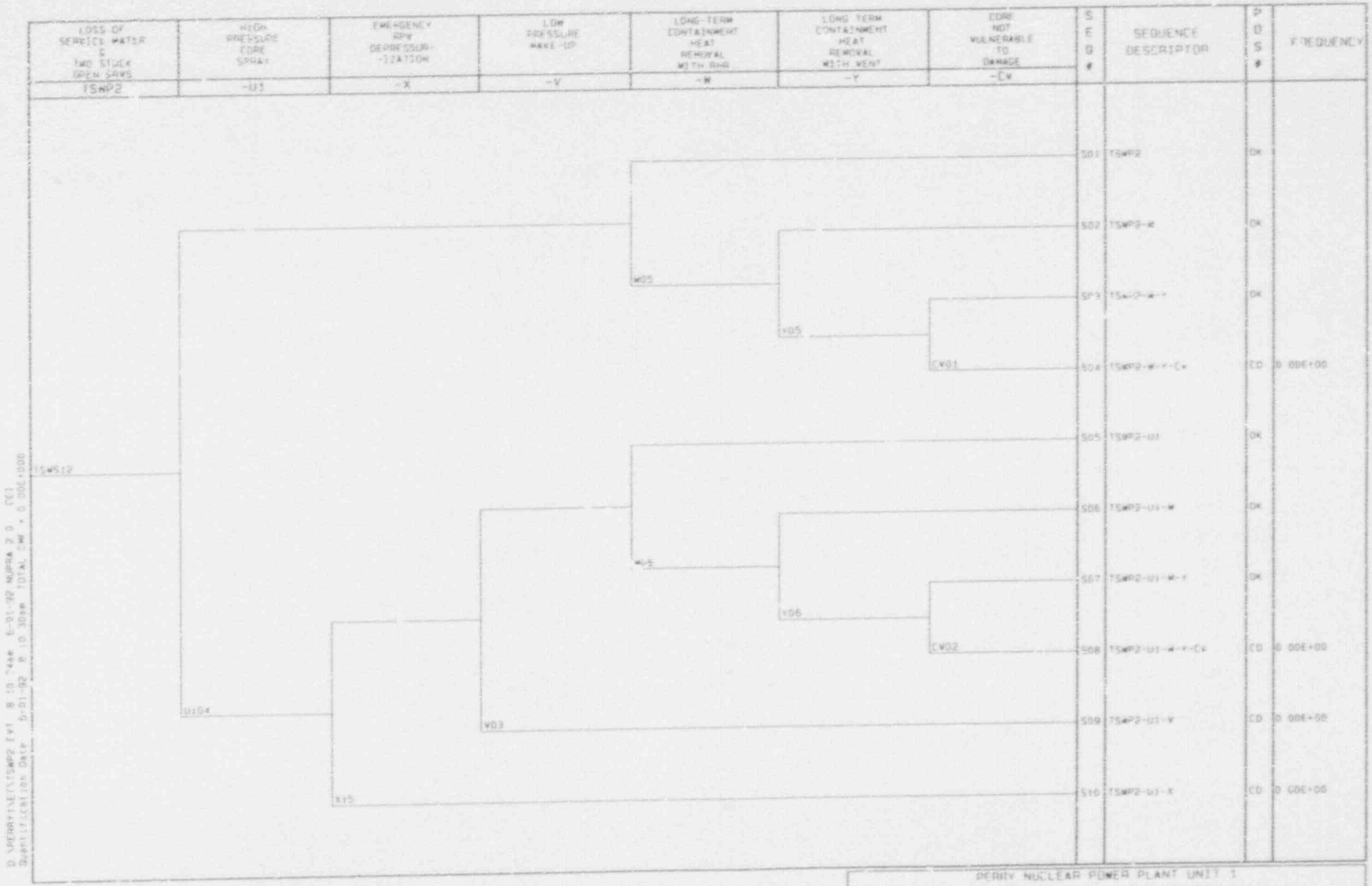


PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.3-5 SHEET 1 OF 2
LOSS OF SERVICE WATER EVENT TREE
REVISION 1 (5/16/91)

D:\PERRY\1\ET\TSW.EVT 9:41 02am 7-01-92 NUPRA 2.0 CEI
Quantification Date: 6-01-92 6:08:30am TOTAL CDF = 6.68E-008

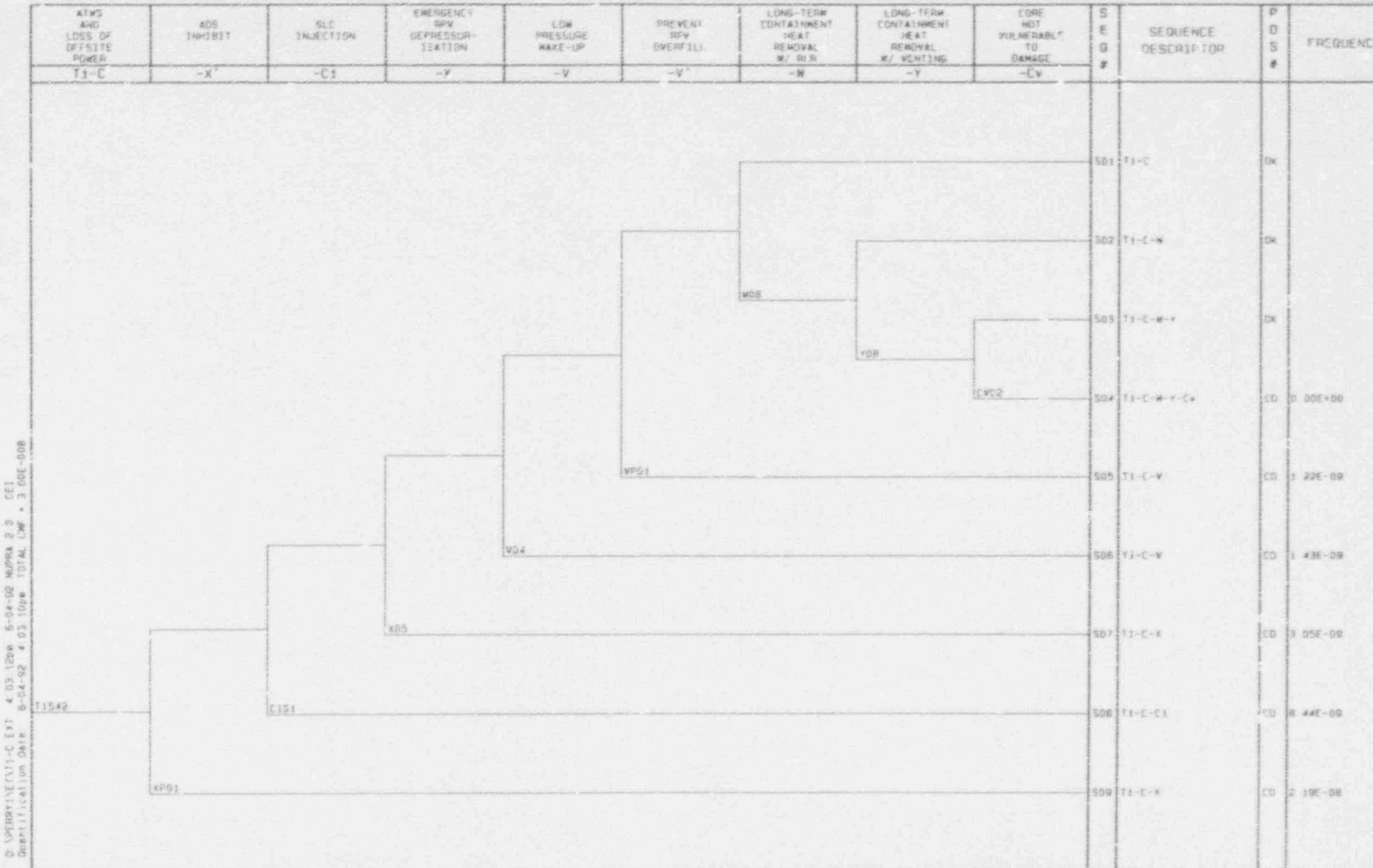


PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.3-5 SHEET 2 OF 3
LOSS OF SERVICE WATER & ONE SORV
REVISION 1 (5/16/91)



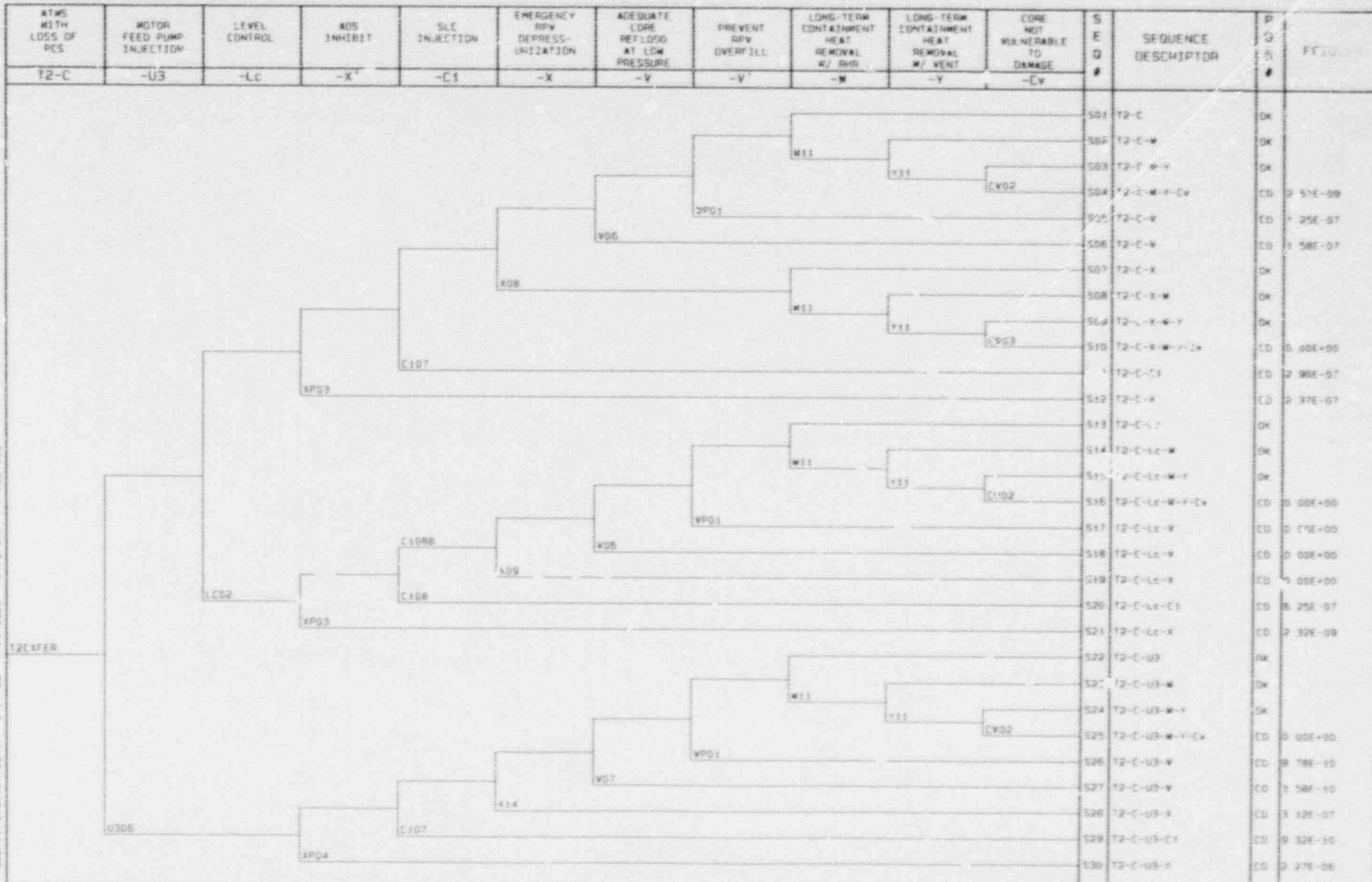
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.3-5 SHEET 3 OF 3
LOSS OF SERVICE WATER & TWO SORVs
REVISION 1 (5/16/91)



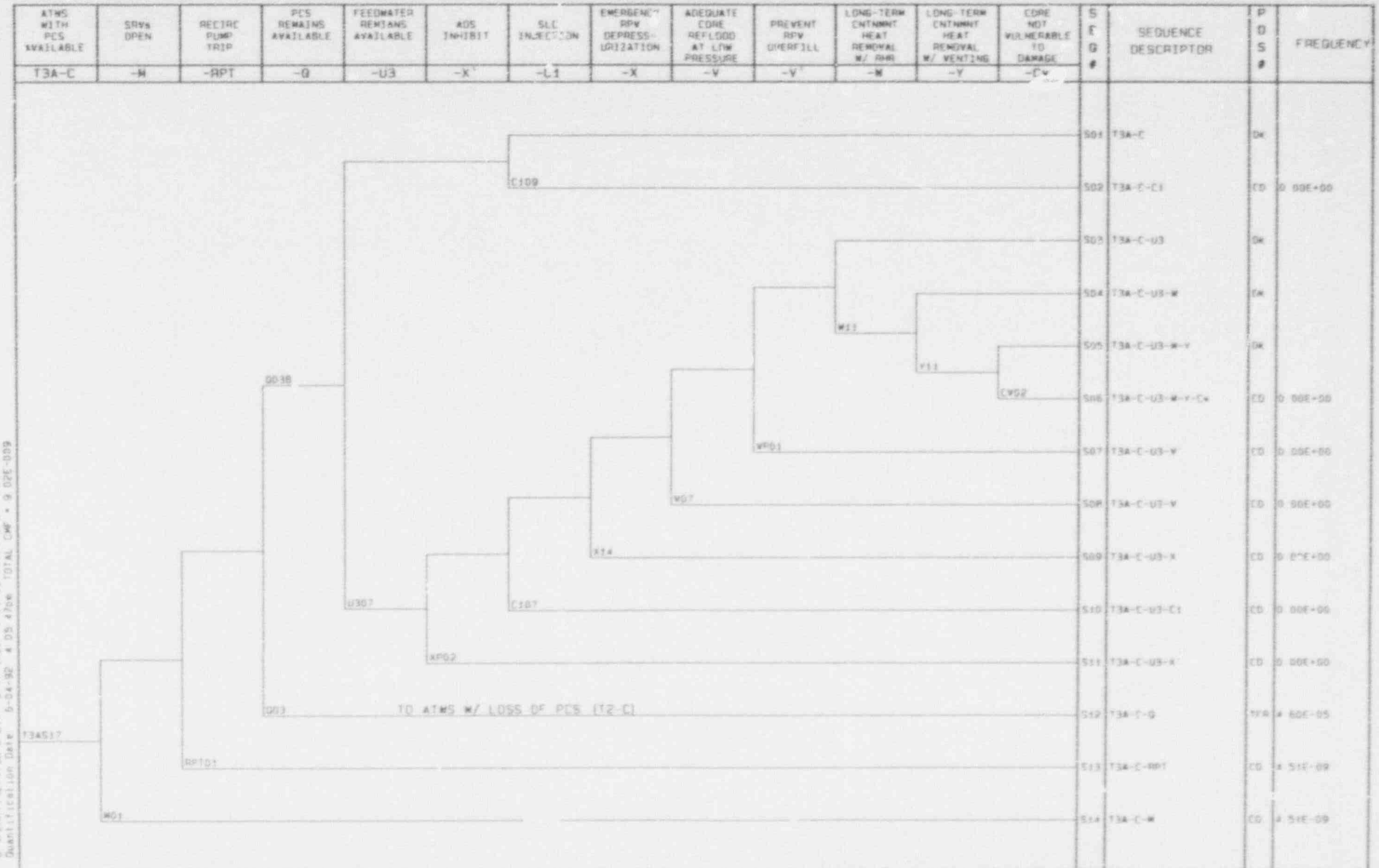
P RY NUCLEAR POWER PLANT UNIT 1
NDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.3-7
ATWS AND LOOP EVENT TREE
REVISION 3 (06/13/91)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.3-8
ATWS AND LOSS OF PCS EVENT TREE
REVISION 7 (04/02/92)

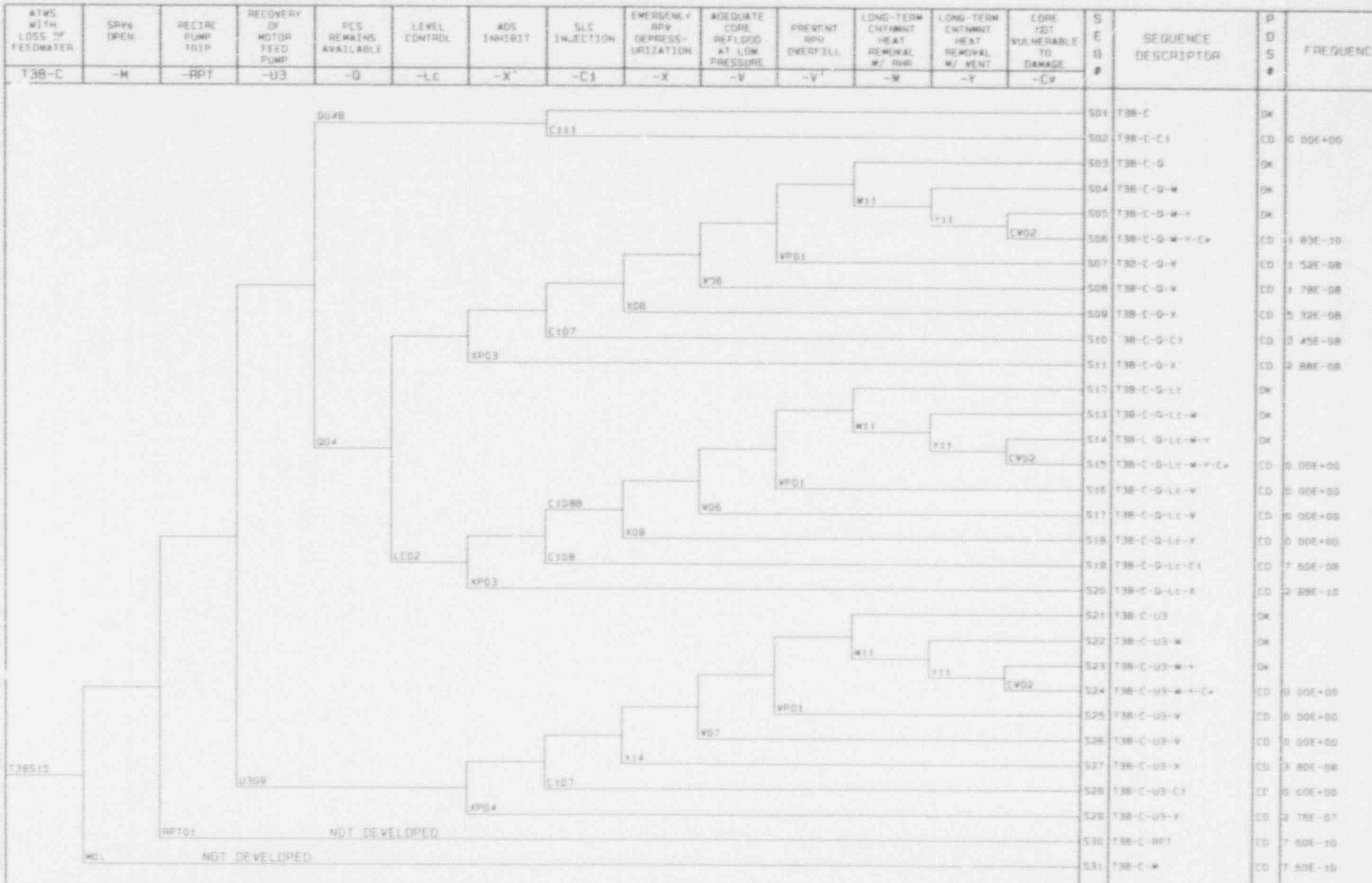
Q:\PERRY\1\ET\T2-C.EVT 4:05:00pm 6-04-92 HUPRA 2.0 CEI
Quantification Date 6-04-92 4:04:58pm TOTAL CMF = 4.02E-006



D:\PERRY\ET\T3A-C.EVT 9:47 (4am 7-01-92 NUPRA 2.0 CEI
Quantification Date 6-04-92 4:05 47pm TOTAL CMF = 9.02E-009

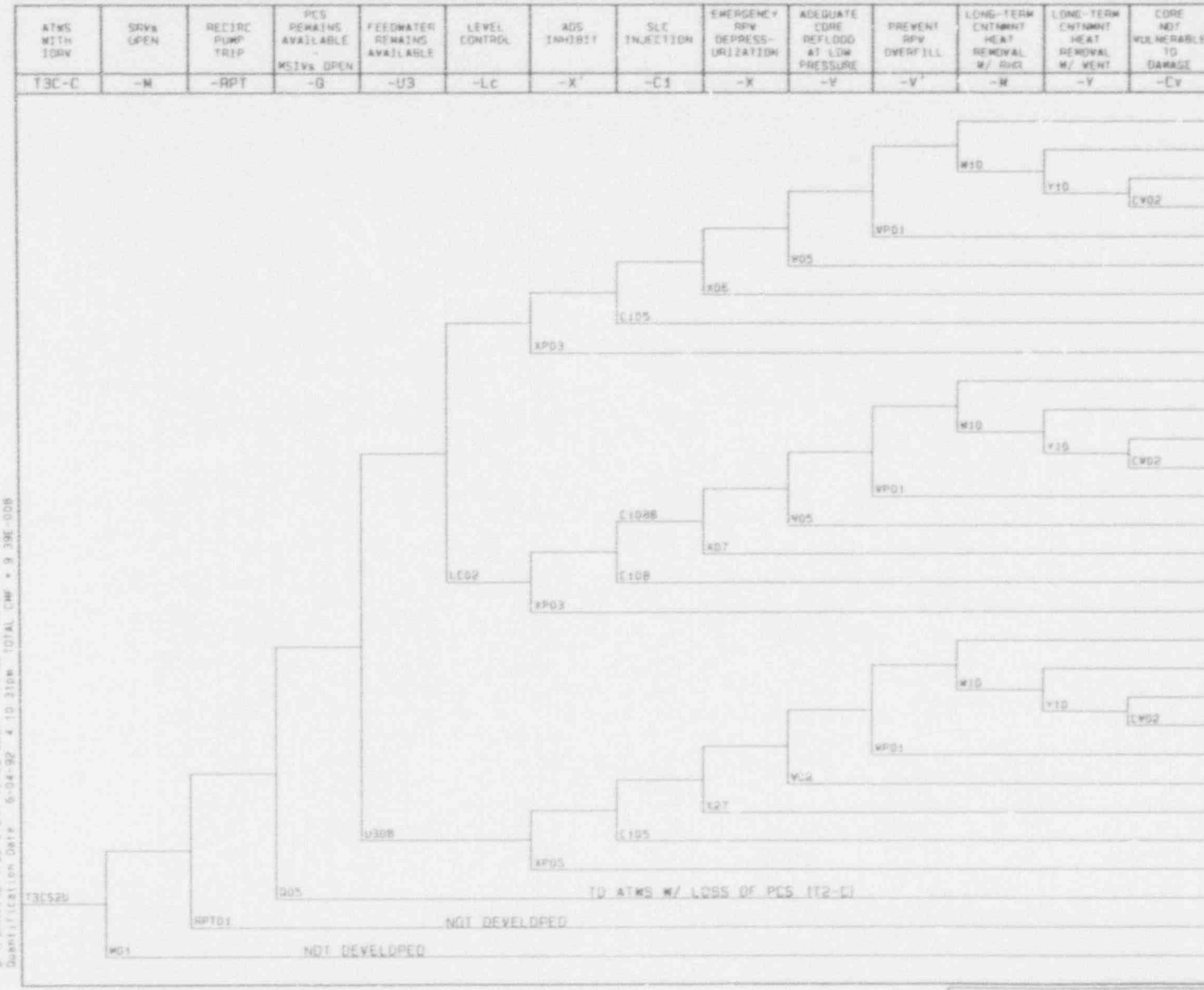
PERRY NUCLEAR POWERPLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.3-9
ATWS W/ PCS AVAILABLE
REVISION 1 (12/10/91)



PERRY NUCLEAR POWERPLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.3-10
ATWS W/ LOSS OF FEEDWATER
REVISION 4 (04/02/92)



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
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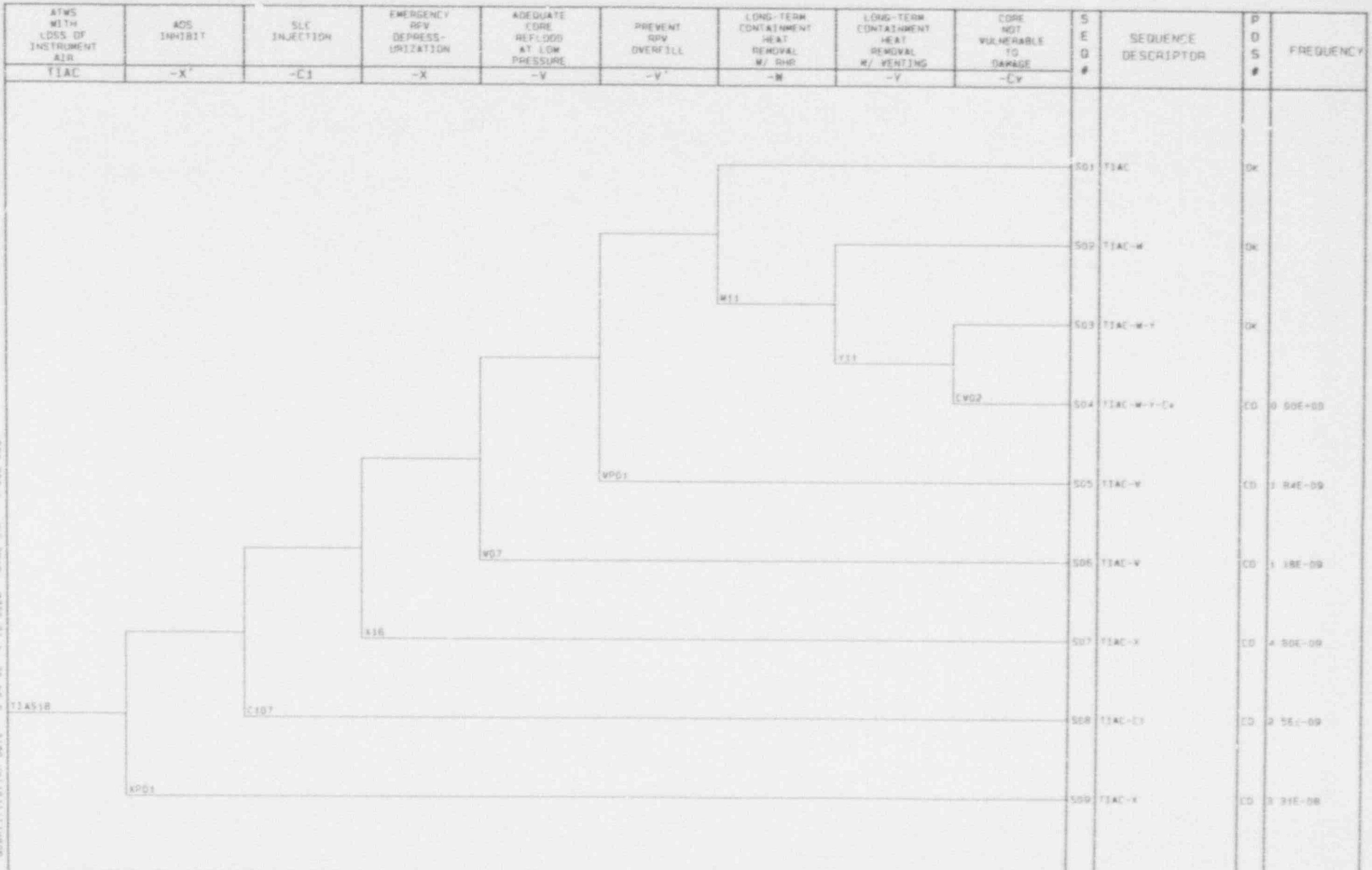
S02	T3C-C-W	OK	
S03	T3C-C-W-Y	OK	
S04	T3C-C-W-Y-Cv	CD	0 00E+00
S05	T3C-C-V	CD	2 80E-09
S06	T3C-C-V	CD	1 89E-09
S07	T3C-C-X	CD	9 81E-09
S08	T3C-C-C1	CD	[illegible] 68E-09
S09	T3C-C-X	CD	5 31E-09
S10	T3C-C-Lc	OK	
S11	T3C-C-Lc-W	OK	
S12	T3C-C-Lc-W-Y	OK	
S13	T3C-C-Lc-W-Y-Cv	CD	0 00E+00
S14	T3C-C-Lc-V	CD	0 00E+00
S15	T3C-C-Lc-V	CD	0 00E+00
S16	T3C-C-Lc-X	CD	0 00E+00
S17	T3C-C-Lc-C1	CD	1 40E-08
S18	T3C-C-Lc-X	CD	0 00E+00
S19	T3C-C-U3	OK	
S20	T3C-C-U3-W	OK	
S21	T3C-C-U3-W-Y	OK	
S22	T3C-C-U3-W-Y-Cv	CD	0 00E+00
S23	T3C-C-U3-V	CD	0 00E+00
S24	T3C-C-U3-V	CD	0 00E+00
S25	T3C-C-U3-X	CD	1 96E-10
S26	T3C-C-U3-C1	CD	0 00E+00
S27	T3C-C-U3-X	CD	5 40E-08
S28	T3C-C-G	TFR	1 40E-07
S29	T3C-C-RPT	CD	1 40E-10
S30	T3C-C-M	CD	1 40E-10

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

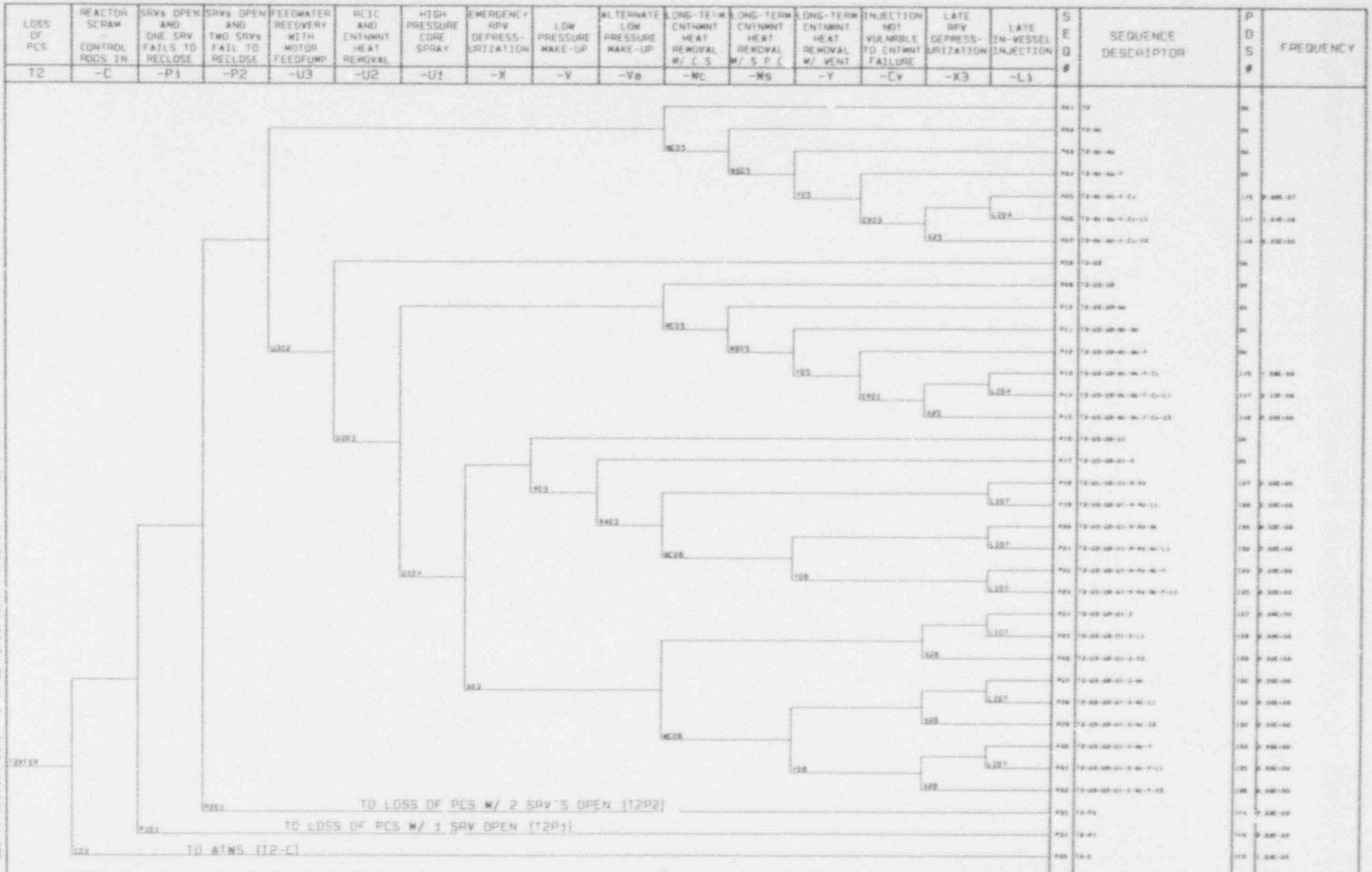
FIGURE 3.1.3-11
ATWS WITH IORV EVENT TREE
REVISION 7 (04/02/92)

D:\PERRY\ET\T3C-C EVT 10 08 22am 7-01-92 MURRA 2.0 CEI
Quantification Date 6-04-92 4 10 31pm TOTAL CHF = 9 39E-008



D:\PERRY\ET\TIAC.EVT 4:13:22pm 6-04-92 NUPRA 2.0 CEI
Quantification Date 6-04-92 4:12:59pm TOTAL CMF = 4.33E-008

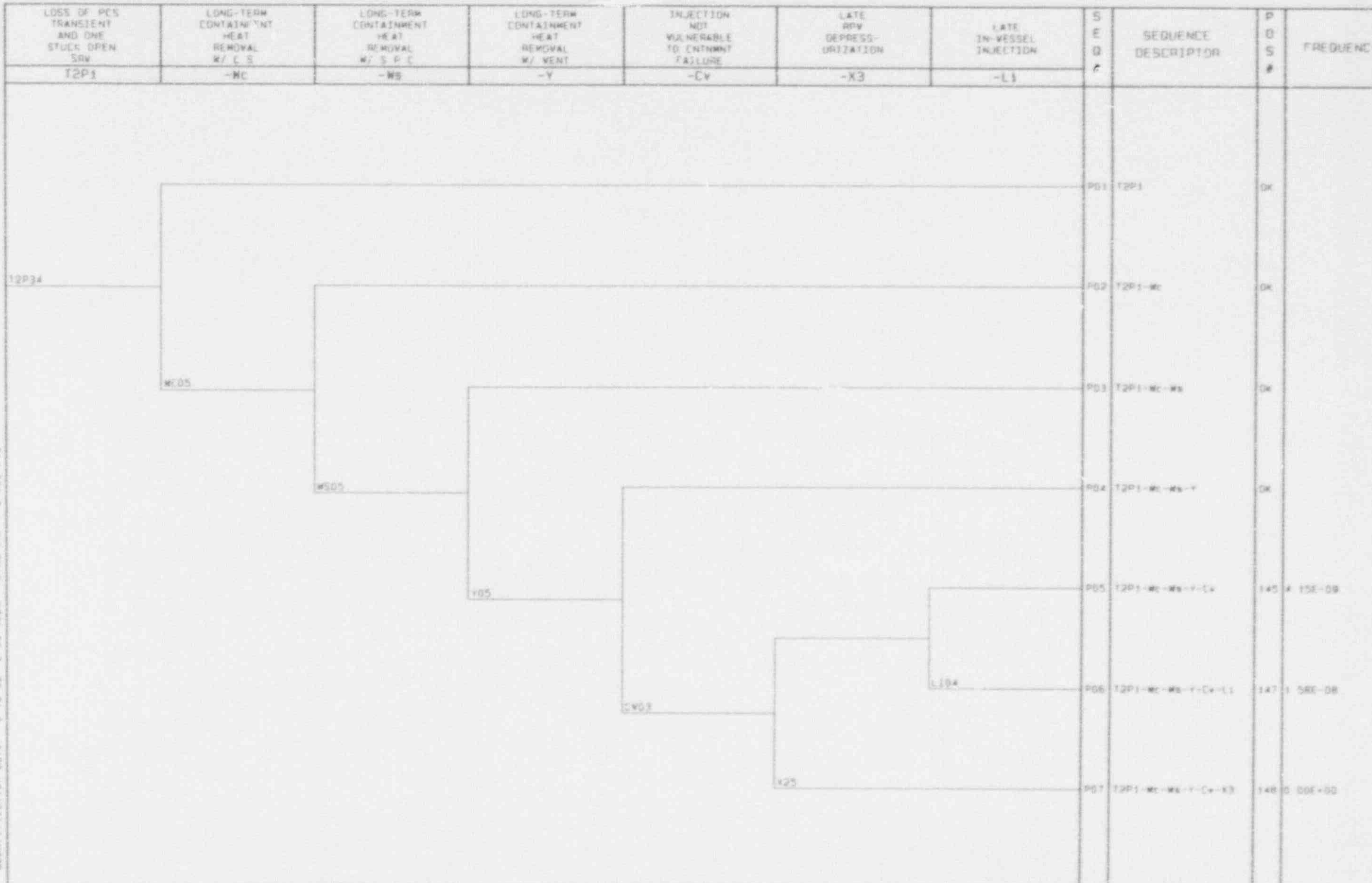
PERRY NUCLEAR POWERPLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.3-12
ATWS AND LOSS OF INSTRUMENT AIR
REVISION 2 (04/02/92)



D:\PDS\ET\T2.EVT 9:58:40am 6-17-92 NUPRA 2.0 CEI
Quantification Date 6-16-92 2:35:15pm TOTAL CMF = 1.35E-006

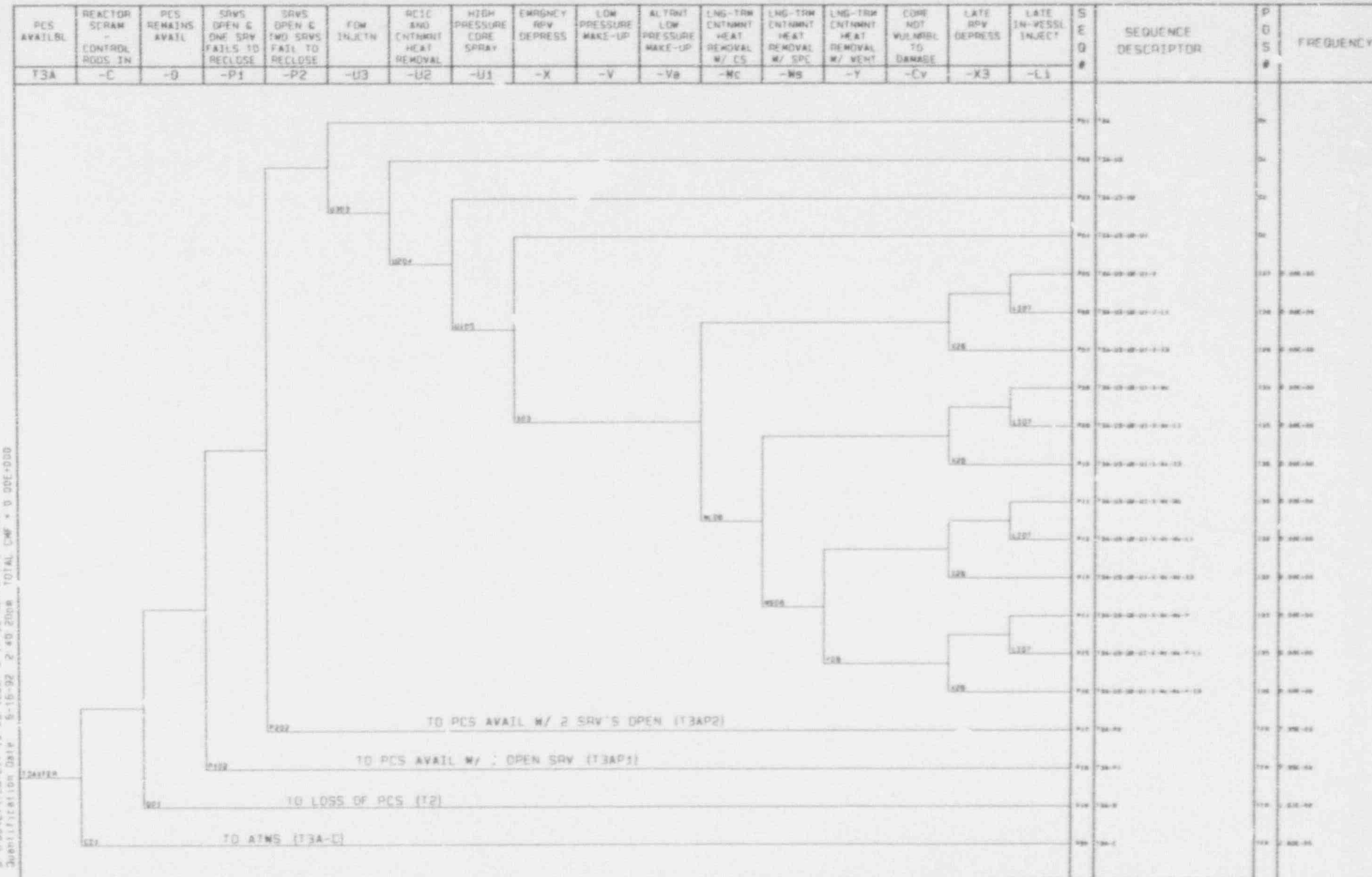
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-1 SHEET 1 OF 2
LOSS OF PCS - PDS EVENT TREE
REVISION 2 (03/27/92)



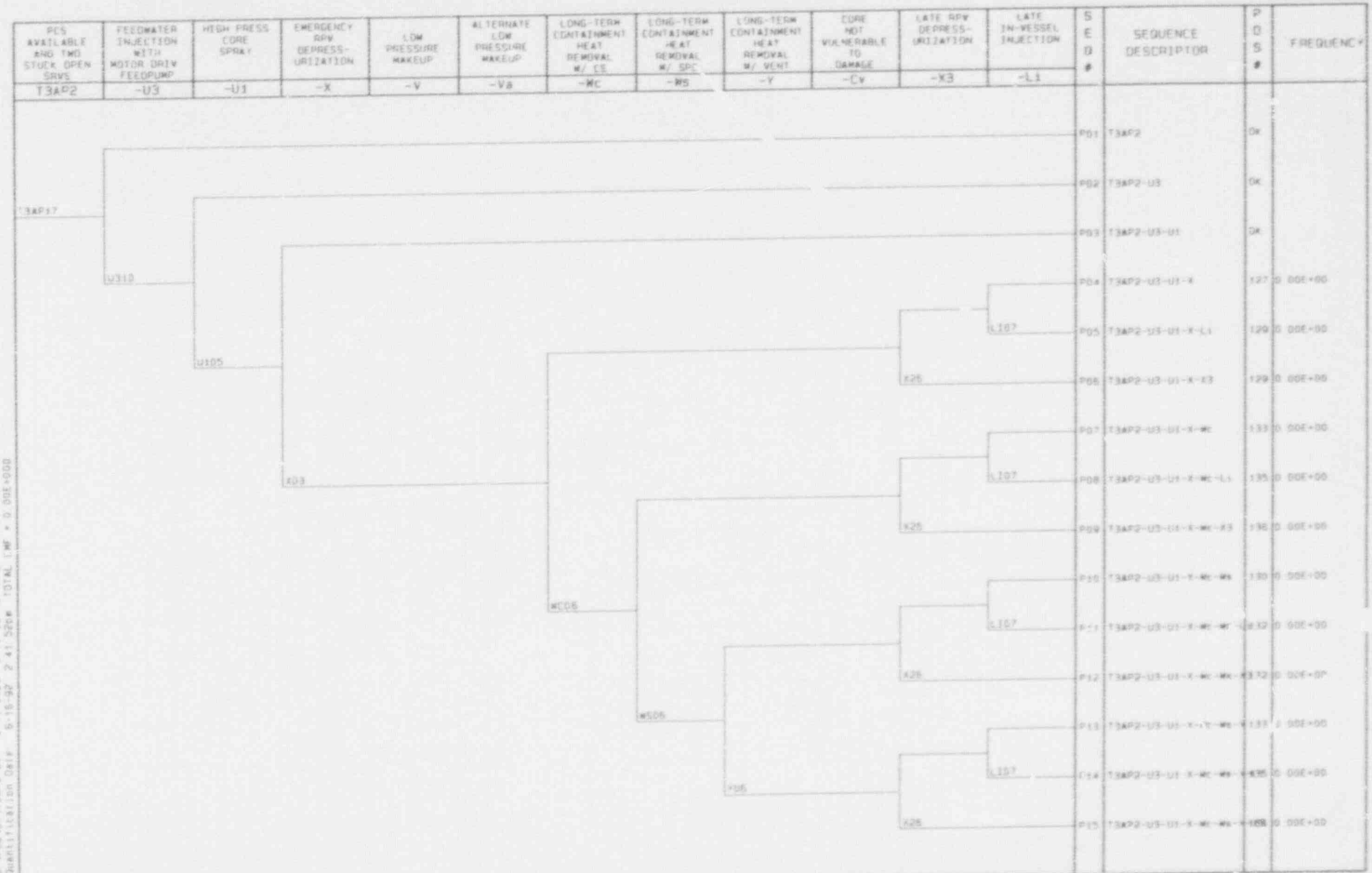
D:\PDS\ET\T2P1.EVT 2:39.48pm 6-16-92 NUPRA 2.0 LE1
Quantification Date: 6-16-92 2:39.12pm TOTAL CMF = 2.00E-008

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-1 SHEET 2 OF 2
LOSS OF PCS - PDS EVENT TREE
REVISION 1 (1/22/92)



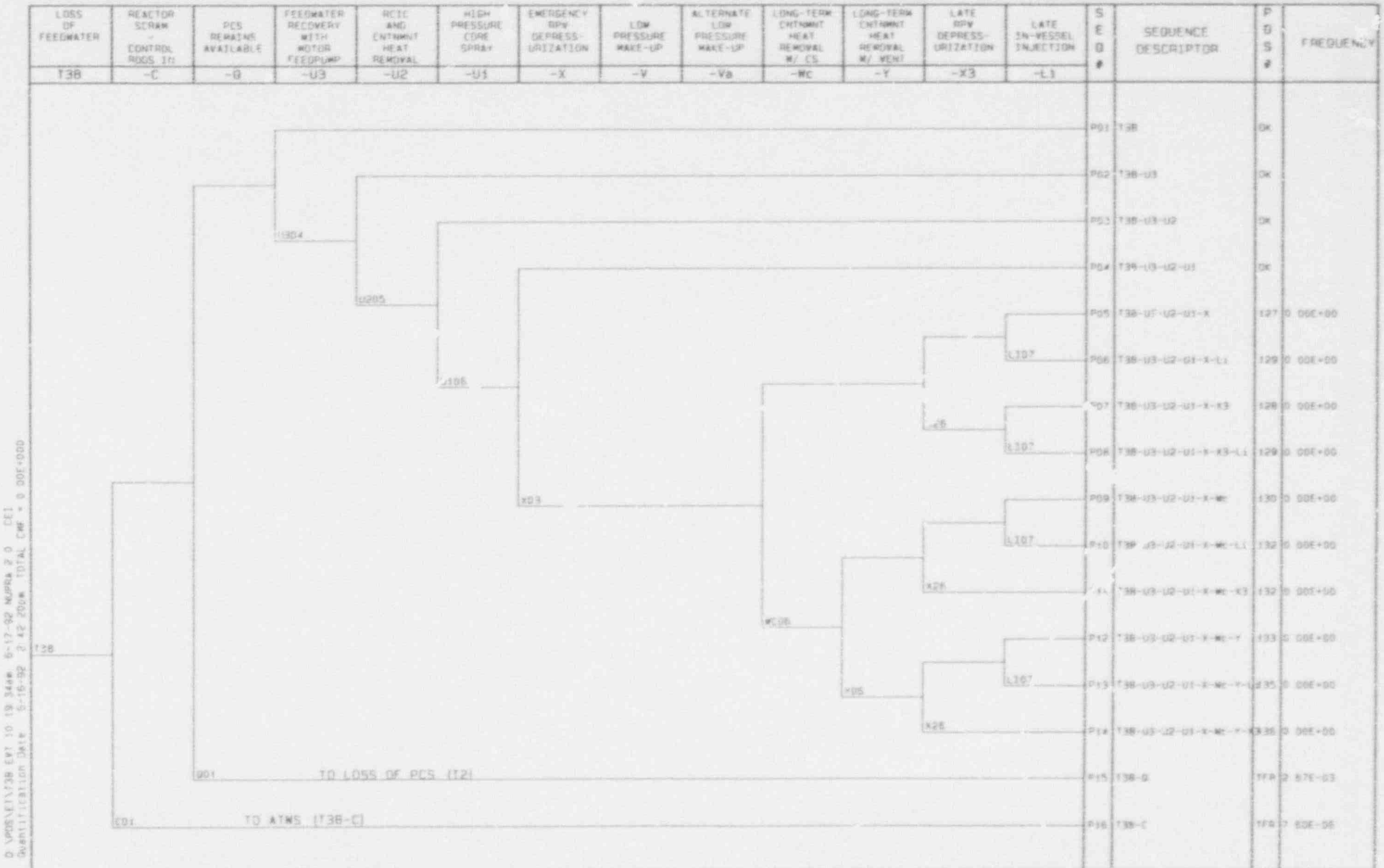
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-2 SHEET 1 OF 2
PCS AVAILABLE PDS EVENT TREE
REVISION 3 (1/22/92)



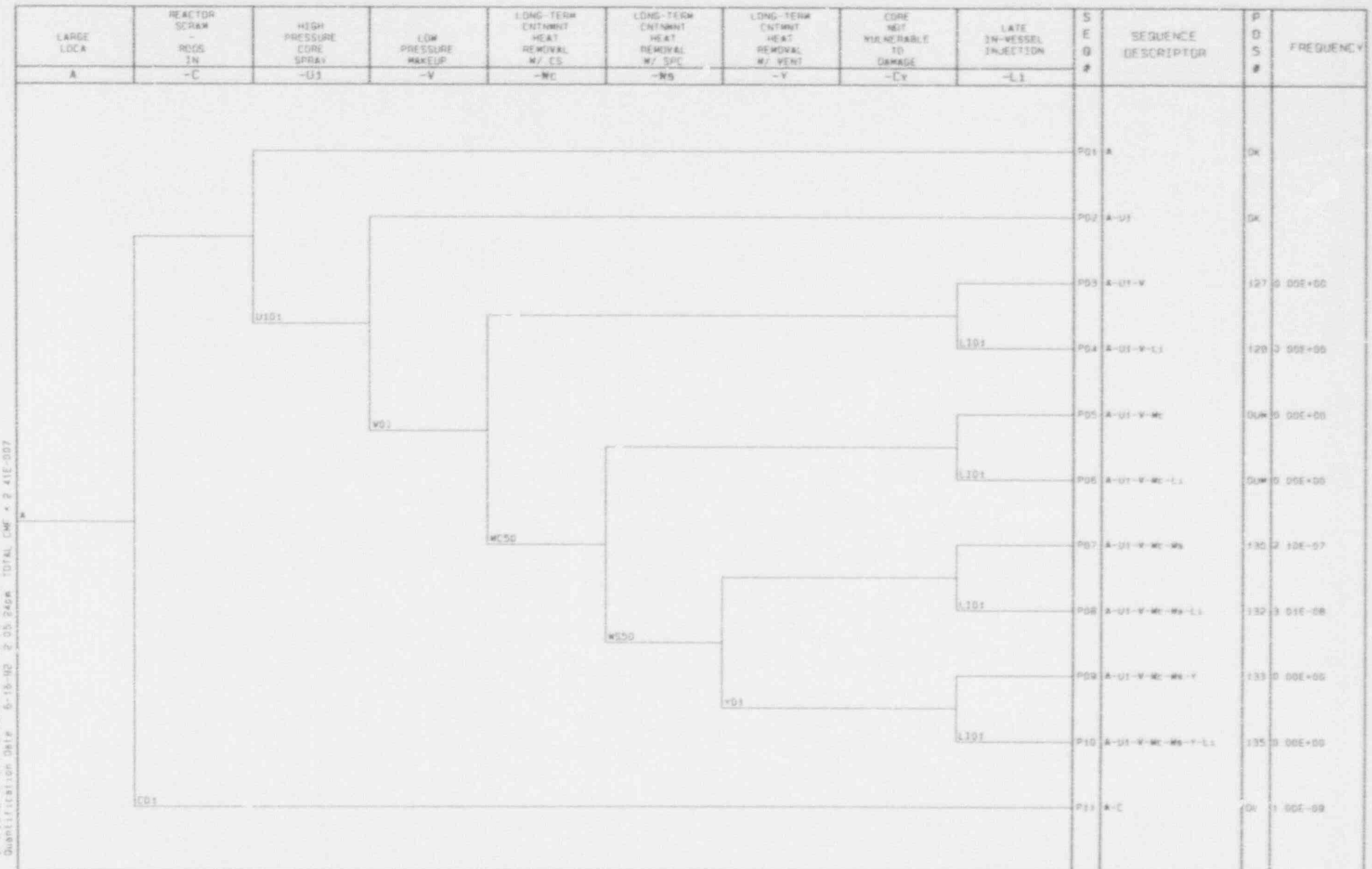
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-2 SHEET 2 OF 2
PCS AVAIL & TWO SORVS - PDS EVENT TREE
REVISION 1 (1/22/92)

D:\PDS\ET\T3AP2.EVT 2:41:54pm 6-16-92 NUPRA 2.0 CEI
Quantification Date 6-16-92 2:41:52pm TOTAL [illegible] = 0.00E+000



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-3
LOSS OF FEEDWATER - PDS EVENT TREE
REVISION 2 (03/20/92)

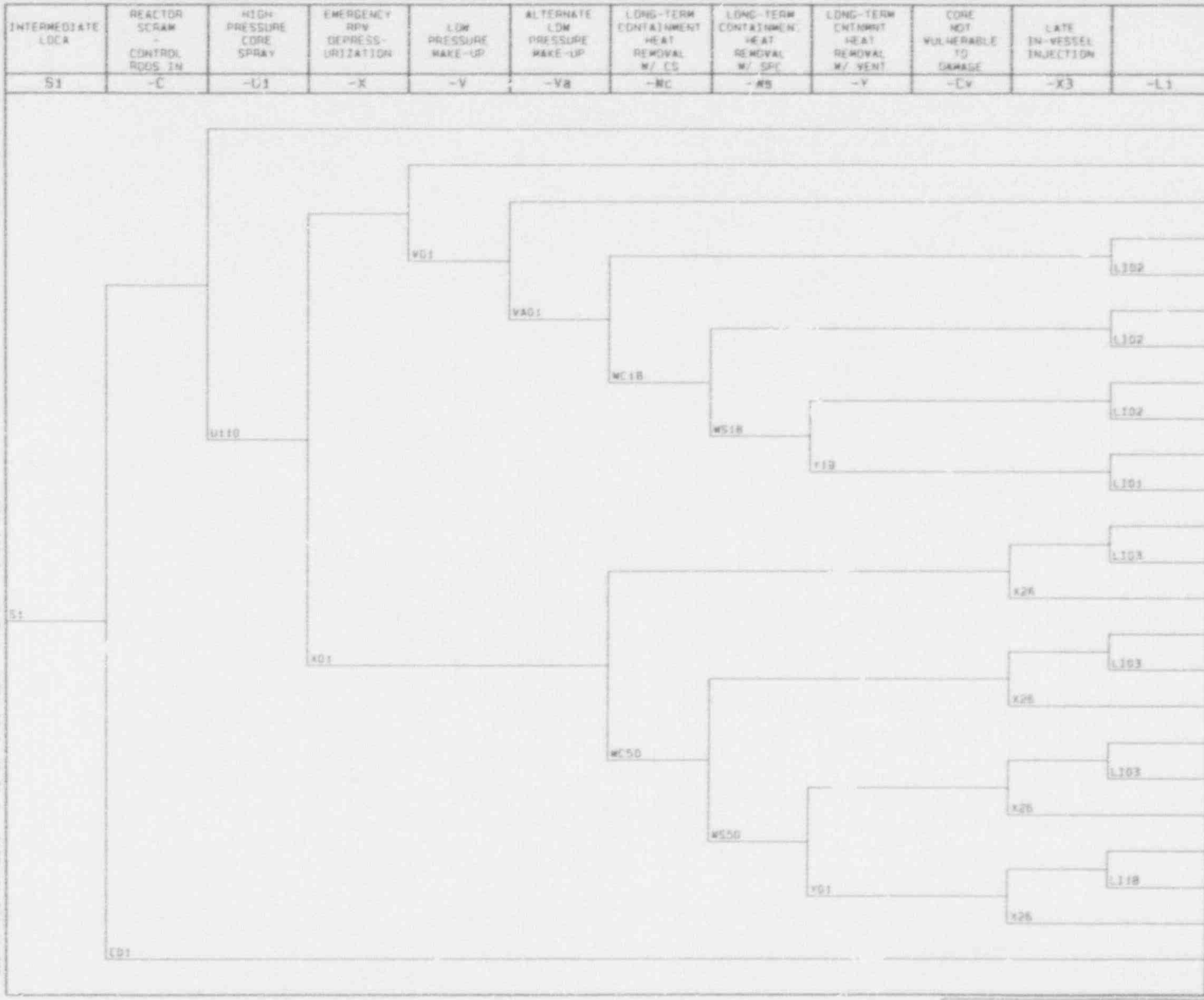


D:\PDS\ET\A.EVT 2.05 38am 6-16-92 NUPRA 2.0 CEI
Quantification Date 6-16-92 2.05 24pm TOTAL CMF = 2.41E-007

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.--4
LARGE LOCA - PDS EVENT TREE
REVISION 1 (1/23/92)

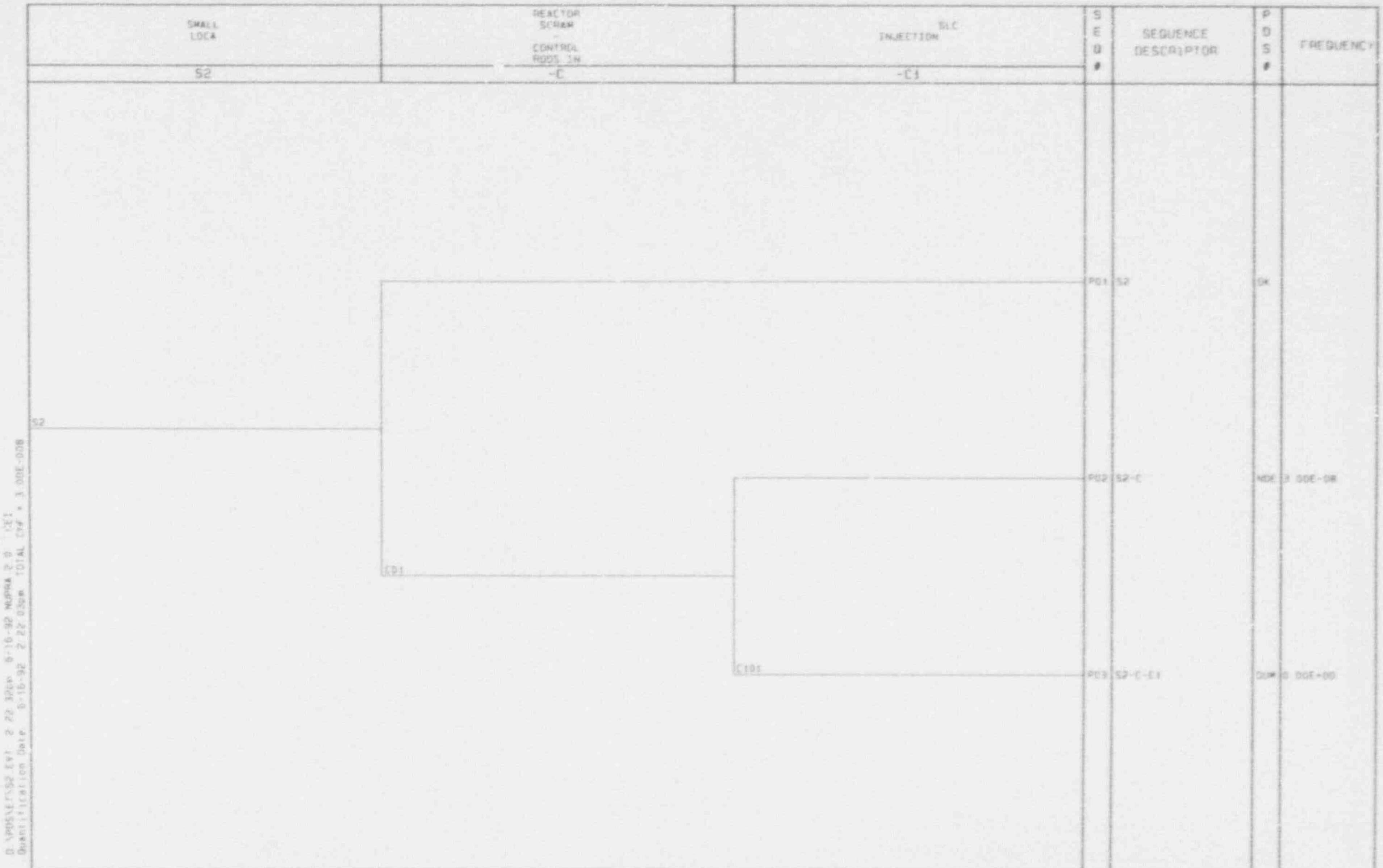
D:\PDS\ET\SI.EVT 2 21 54pm 5-16-92 NUPRA 2.0 CEI
Quantification Date 5-16-92 2 21 49pm TOTAL CMF = 6.82E-008



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	S1	OK	
P02	S1-U1	OK	
P03	S1-U1-V	OK	
P04	S1-U1-V-Va	127	0.00E+00
P05	S1-U1-V-Va-L1	129	0.00E+00
P06	S1-U1-V-Va-Wc	DUM	0.00E+00
P07	S1-U1-V-Va-Wc-L1	DUM	0.00E+00
P08	S1-U1-V-Va-Wc-Ws	130	5.85E-08
P09	S1-U1-V-Va-Wc-Ws-L1	132	6.65E-09
P10	S1-U1-V-Va-Wc-Ws-Y	133	0.00E+00
P11	S1-U1-V-Va-Wc-Ws-Y-L1	135	0.00E+00
P12	S1-U1-X	127	0.00E+00
P13	S1-U1-X-L1	129	0.00E+00
P14	S1-U1-X-X3	DUM	0.00E+00
P15	S1-U1-X-Wc	DUM	0.00E+00
P16	S1-U1-X-Wc-L1	DUM	0.00E+00
P17	S1-U1-X-Wc-X3	DUM	0.00E+00
P18	S1-U1-X-Wc-Ws	130	0.00E+00
P19	S1-U1-X-Wc-Ws-L1	132	0.00E+00
P20	S1-U1-X-Wc-Ws-X3	DUM	0.00E+00
P21	S1-U1-X-Wc-Ws-Y	133	0.00E+00
P22	S1-U1-X-Wc-Ws-Y-L1	135	0.00E+00
P23	S1-U1-X-Wc-Ws-Y-X3	DUM	0.00E+00
P24	S1-C	DUM	3.00E-09

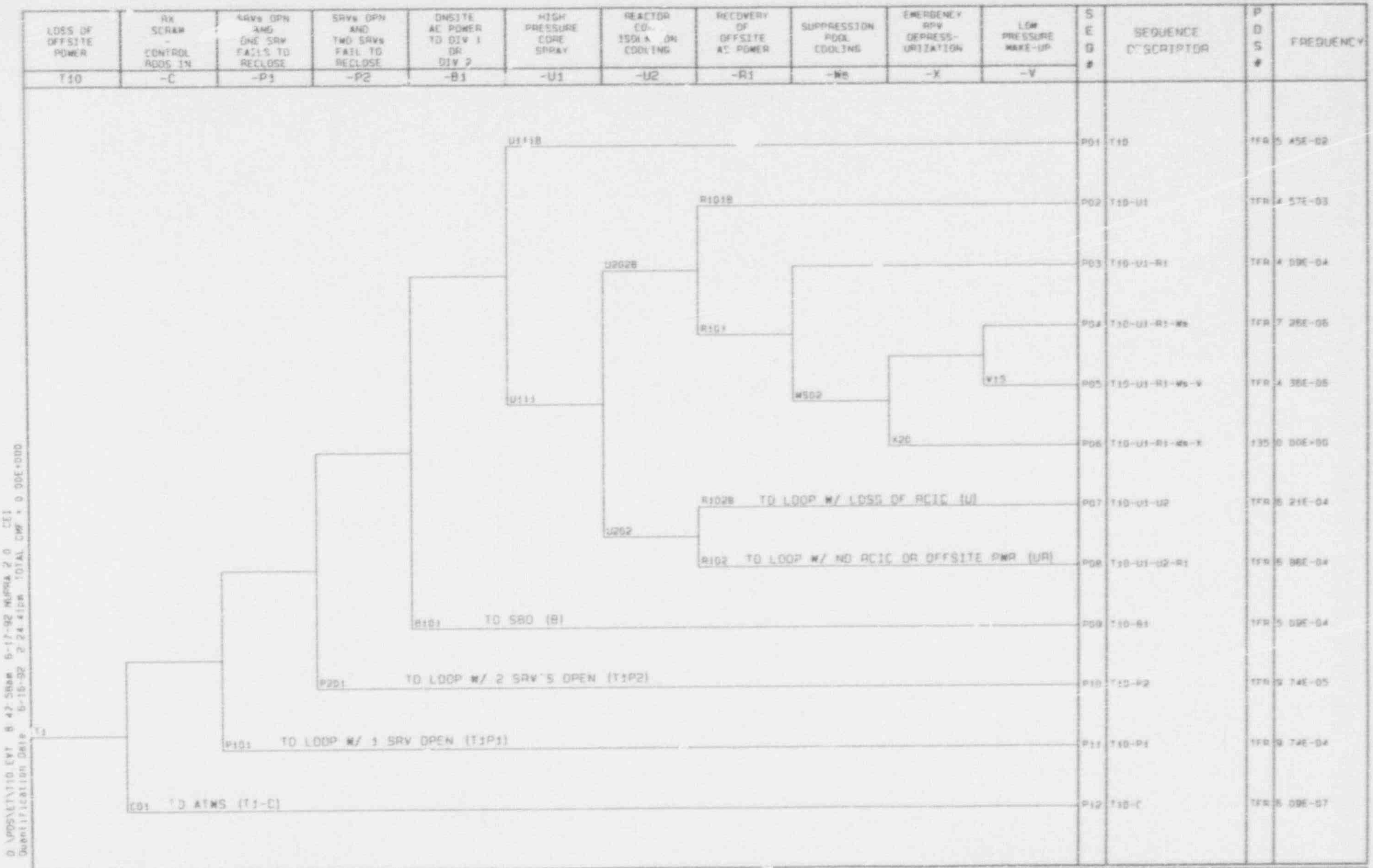
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-5
INTERMEDIATE LOCA - PDS EVENT TREE
REVISION 0 (10/29/91)



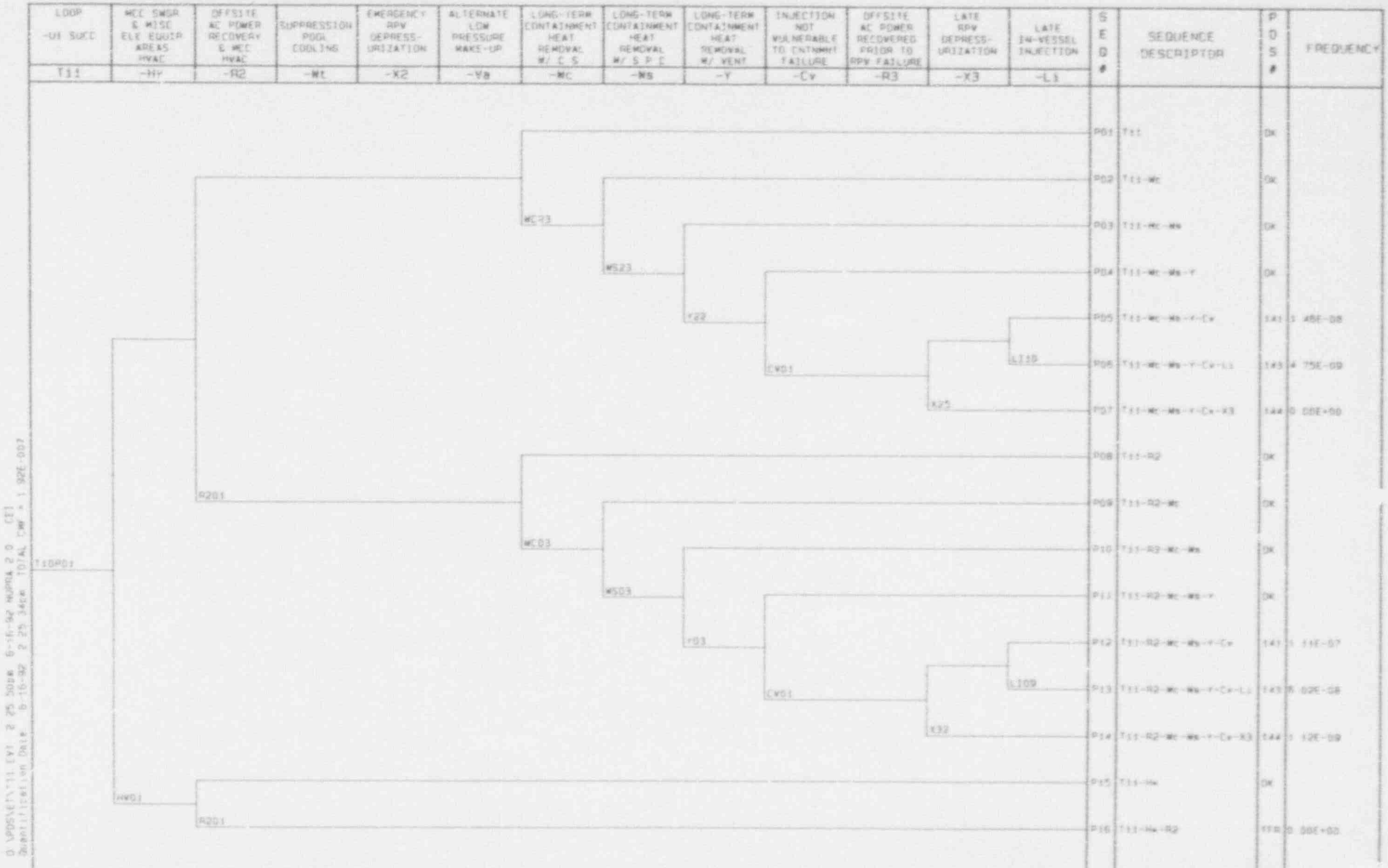
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-6
SMALL LOCA - PDS EVENT TREE
REVISION 0 (10/29/91)



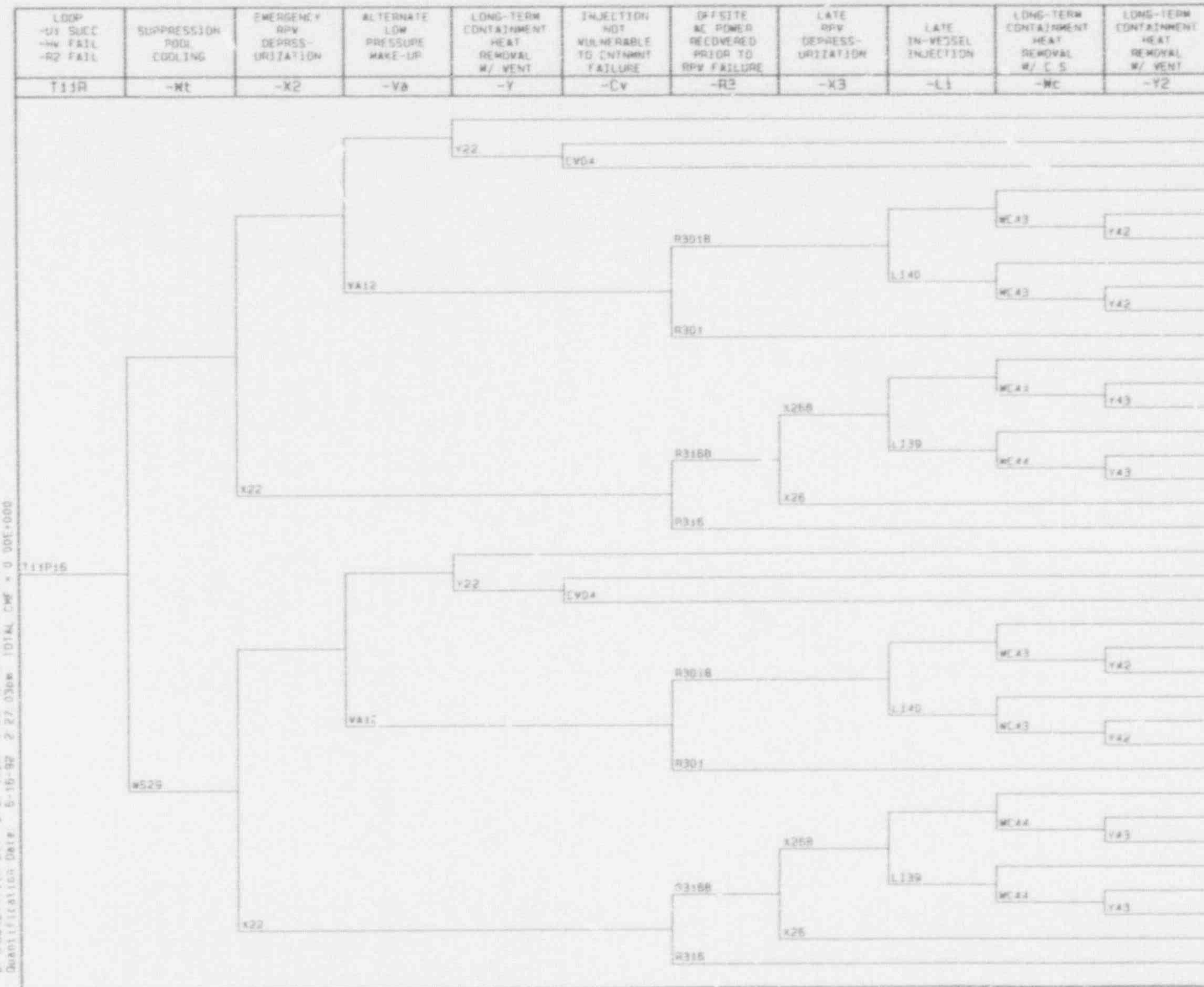
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-7 SHEET 1 OF 15
LOSS OF OFFSITE POWER PDS EVENT TREE
REVISION 1 (03/18/92)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

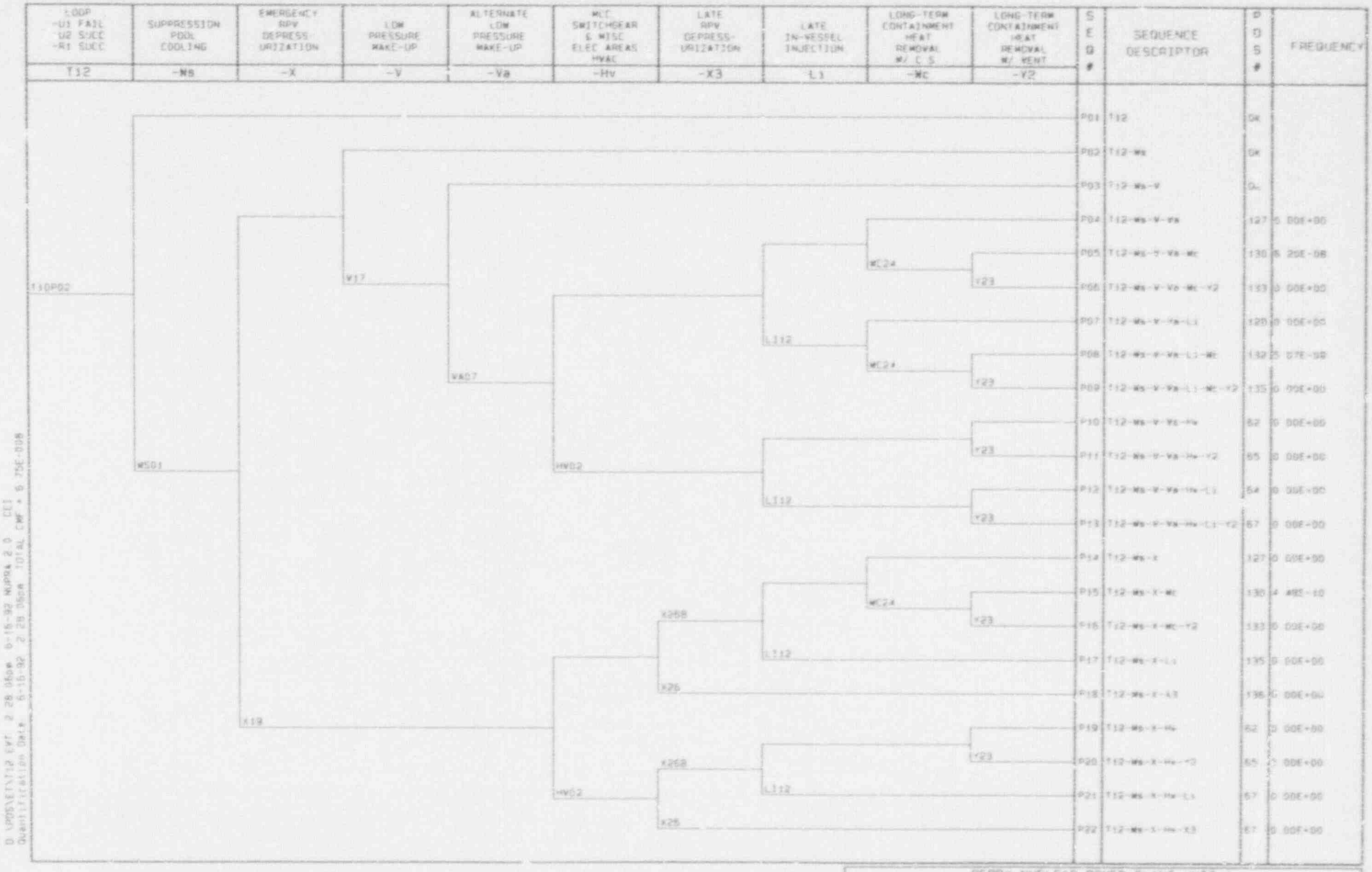
FIGURE 3.1.4-7 SHEET 2 OF 15
LOSS OF OFFSITE POWER PDS EVENT TREE
REVISION 1 (03/18/92)



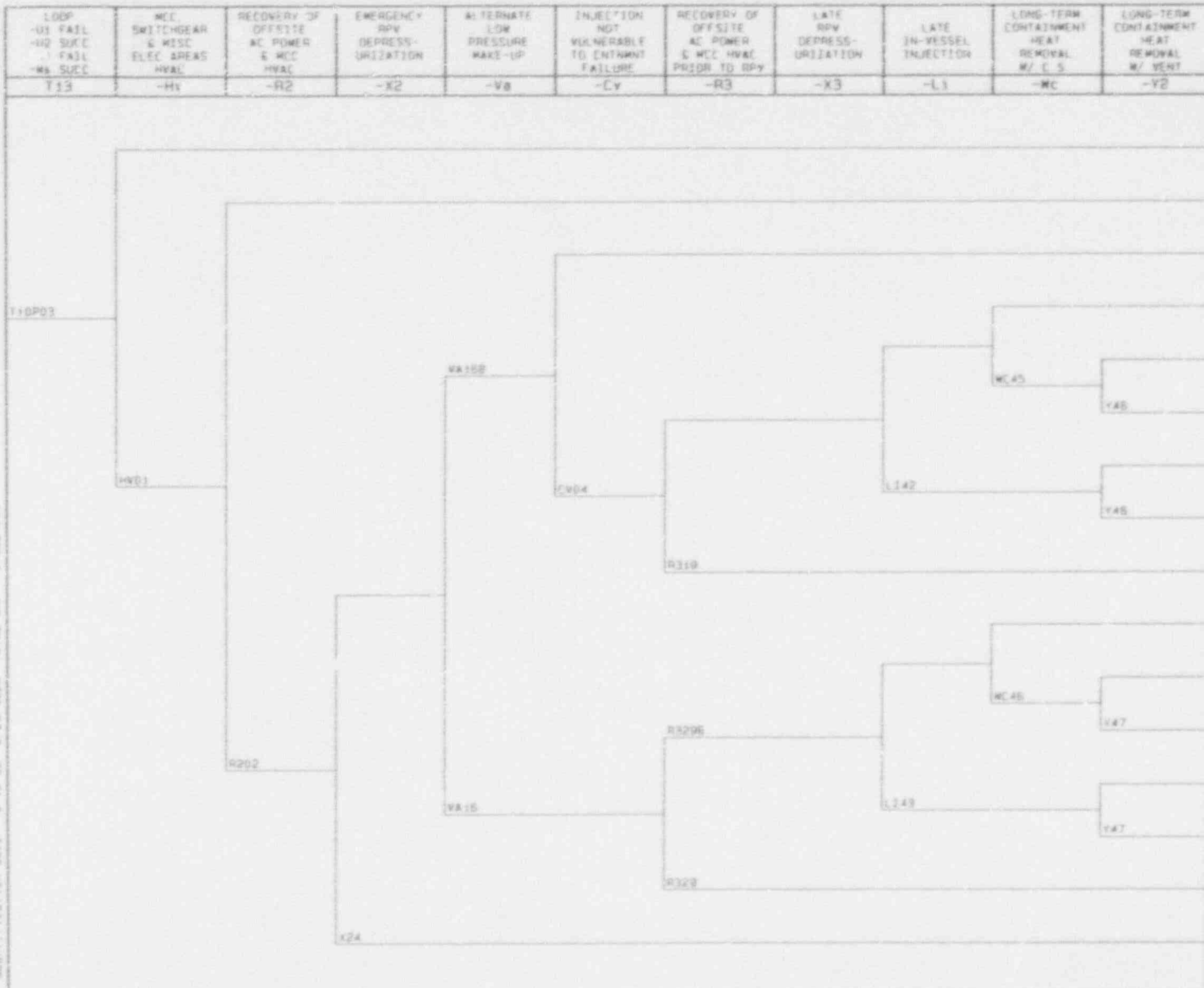
SEQ #	SEQUENCE DESCRIPTOR	POS #	FREQUENCY
P01	T11R	OK	
P02	T11R-Y	OK	
P03	T11R-Y-Cv	144	0.00E+00
P04	T11R-Va	111	0.00E+00
P05	T11R-Va-Wc	115	0.00E+00
P06	T11R-Va-Wc-Y2	119	0.00E+00
P07	T11R-Va-L1	113	0.00E+00
P08	T11R-Va-L1-Wc	117	0.00E+00
P09	T11R-Va-L1-Wc-Y2	121	0.00E+00
P10	T11R-Va-R3	125	0.00E+00
P11	T11R-X2	111	0.00E+00
P12	T11R-X2-Wc	115	0.00E+00
P13	T11R-X2-Wc-Y2	119	0.00E+00
P14	T11R-X2-L1	113	0.00E+00
P15	T11R-X2-L1-Wc	117	0.00E+00
P16	T11R-X2-L1-Wc-Y2	121	0.00E+00
P17	T11R-X2-X3	122	0.00E+00
P18	T11R-X2-R3	126	0.00E+00
P19	T11R-Wt	OK	
P20	T11R-Wt-Y	OK	
P21	T11R-Wt-Y-Cv	144	0.00E+00
P22	T11R-Wt-Va	78	0.00E+00
P23	T11R-Wt-Va-Wc	81	0.00E+00
P24	T11R-Wt-Va-Wc-Y2	81	0.00E+00
P25	T11R-Wt-Va-L1	80	0.00E+00
P26	T11R-Wt-Va-L1-Wc	83	0.00E+00
P27	T11R-Wt-Va-L1-Wc-Y2	83	0.00E+00
P28	T11R-Wt-Va-R3	84	0.00E+00
P29	T11R-Wt-X2	78	0.00E+00
P30	T11R-Wt-X2-Wc	81	0.00E+00
P31	T11R-Wt-X2-Wc-Y2	81	0.00E+00
P32	T11R-Wt-X2-L1	80	0.00E+00
P33	T11R-Wt-X2-L1-Wc	83	0.00E+00
P34	T11R-Wt-X2-L1-Wc-Y2	83	0.00E+00
P35	T11R-Wt-X2-X3	83	0.00E+00
P36	T11R-Wt-X2-R3	85	0.00E+00

D:\PDS\ET\T11R.EVT 2.27.10pm 6-16-92 NUPRA 2.0 CE1
Quantification Date 6-16-92 2.27.03pm TOTAL CMF = 0.00E+000

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-7 SHEET 3 OF 15
LOSS OF OFFSITE POWER PDS EVENT TREE
REVISION 0 (03/30/92)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-7 SHEET 4 OF 15
LOSS OF OFFSITE POWER PDS EVENT TREE
REVISION 1 (03/13/92)



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
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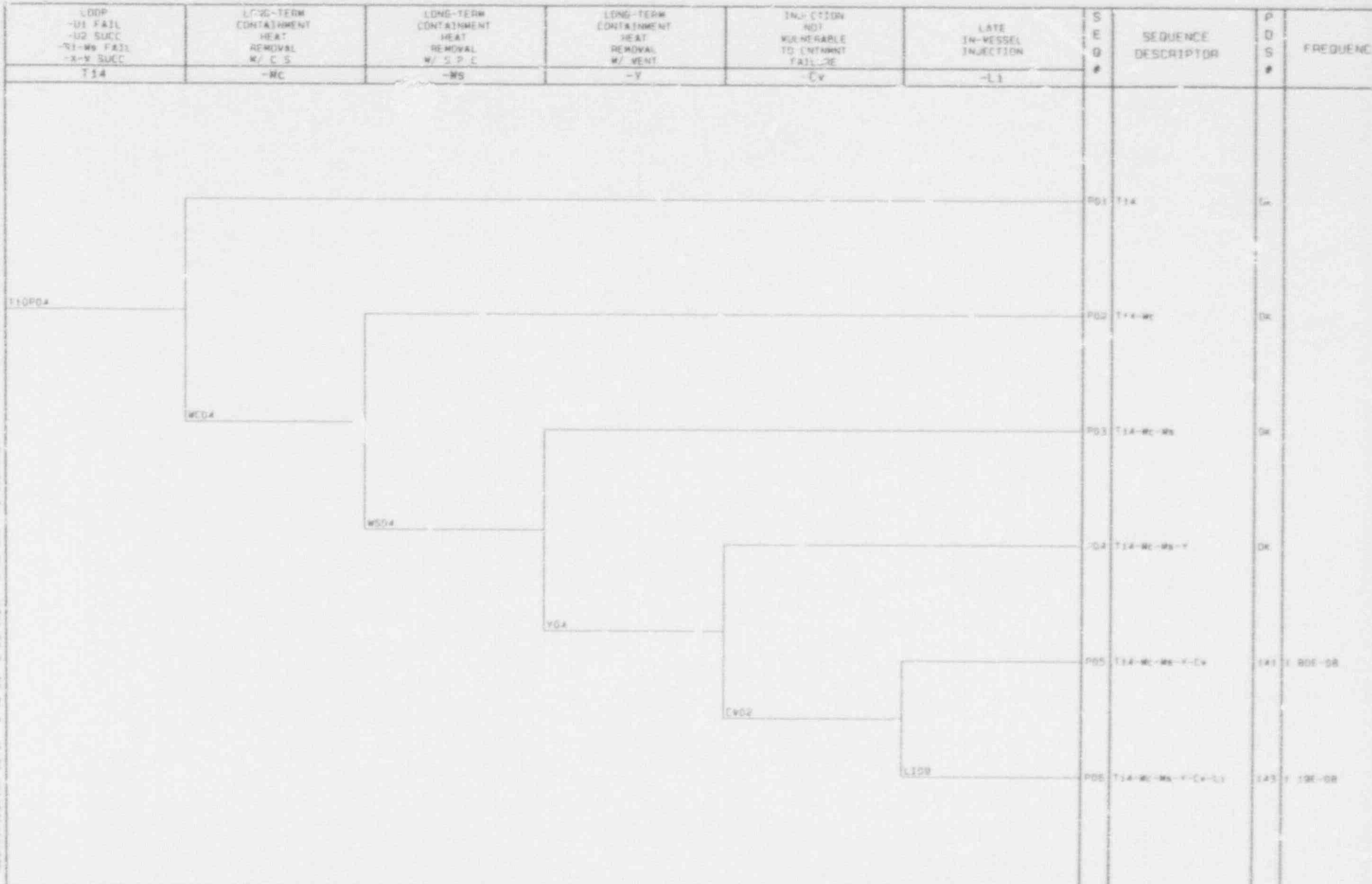
P02	T13-Hv	OK	
P03	T13-Hv-R2	OK	
P04	T13-Hv-R2-Cv	141	0.00E+00
P05	T13-Hv-R2-Cv-Wc	141	0.00E+00
P06	T13-Hv-R2-Cv-Wc-Y2	141	0.00E+00
P07	T13-Hv-R2-Cv-Li	143	0.00E+00
P08	T13-Hv-R2-Cv-Li-Y2	143	0.00E+00
P09	T13-Hv-R2-Cv-R3	144	0.00E+00
P10	T13-Hv-R2-Va	127	0.00E+00
P11	T13-Hv-R2-Va-Wc	130	0.00E+00
P12	T13-Hv-R2-Va-Wc-Y2	133	0.00E+00
P13	T13-Hv-R2-Va-Li	132	0.00E+00
P14	T13-Hv-R2-Va-Li-Y2	135	0.00E+00
P15	T13-Hv-R2-Va-R3	110	0.00E+00
P16	T13-Hv-R2-X2	110	0.00E+00

D:\PDS\ET\T13.EXT 2:28:42pm 6-16-92 MAPPA 2.0 CEI
Quantification Date 6-16-92 2:28:36pm TOTAL CMF = 0.00E+00

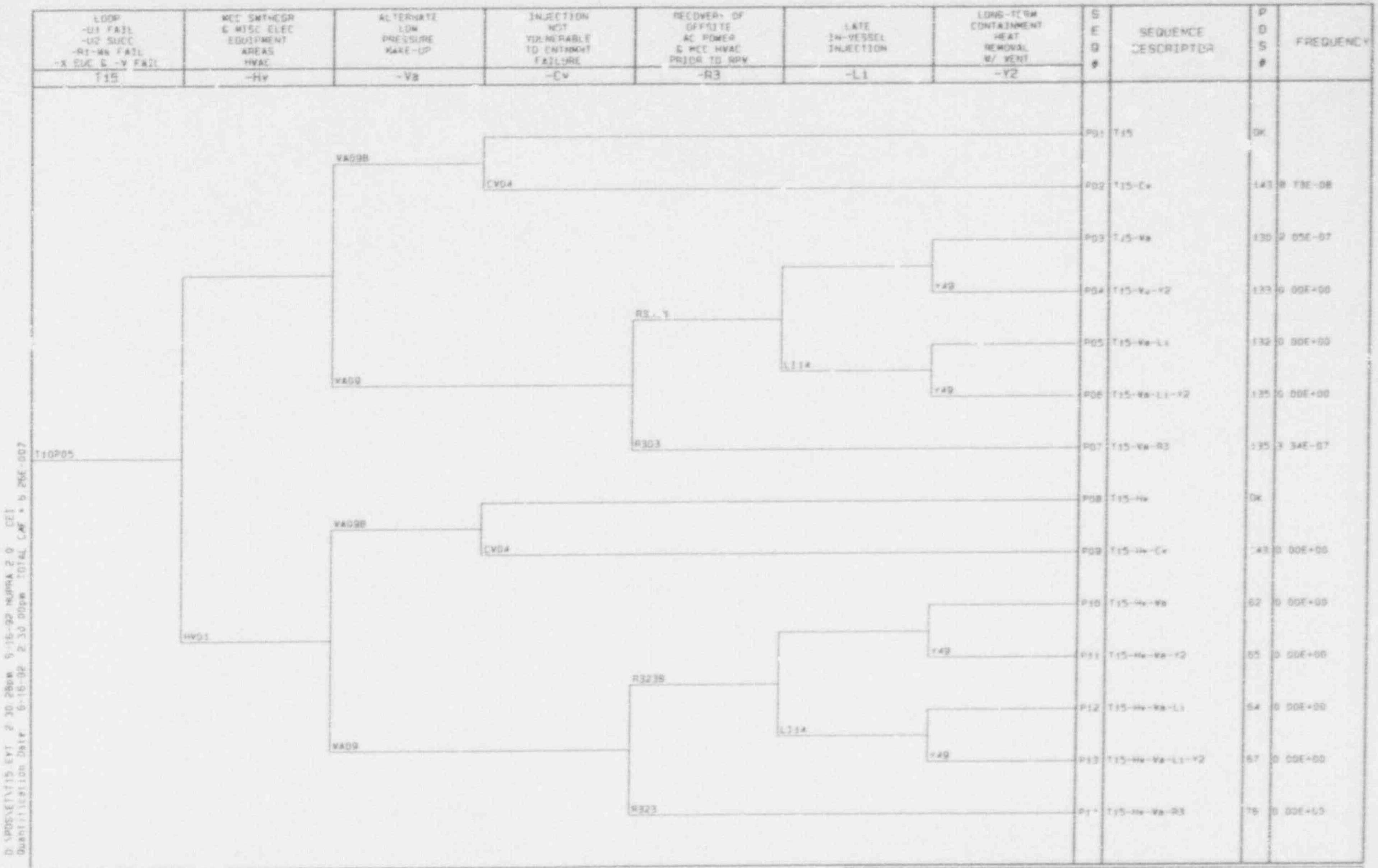
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-7 SHEET 5 OF 15
LOSS OF OFFSITE POWER PDS EVENT TREE
REVISION 1 (04/07/92)



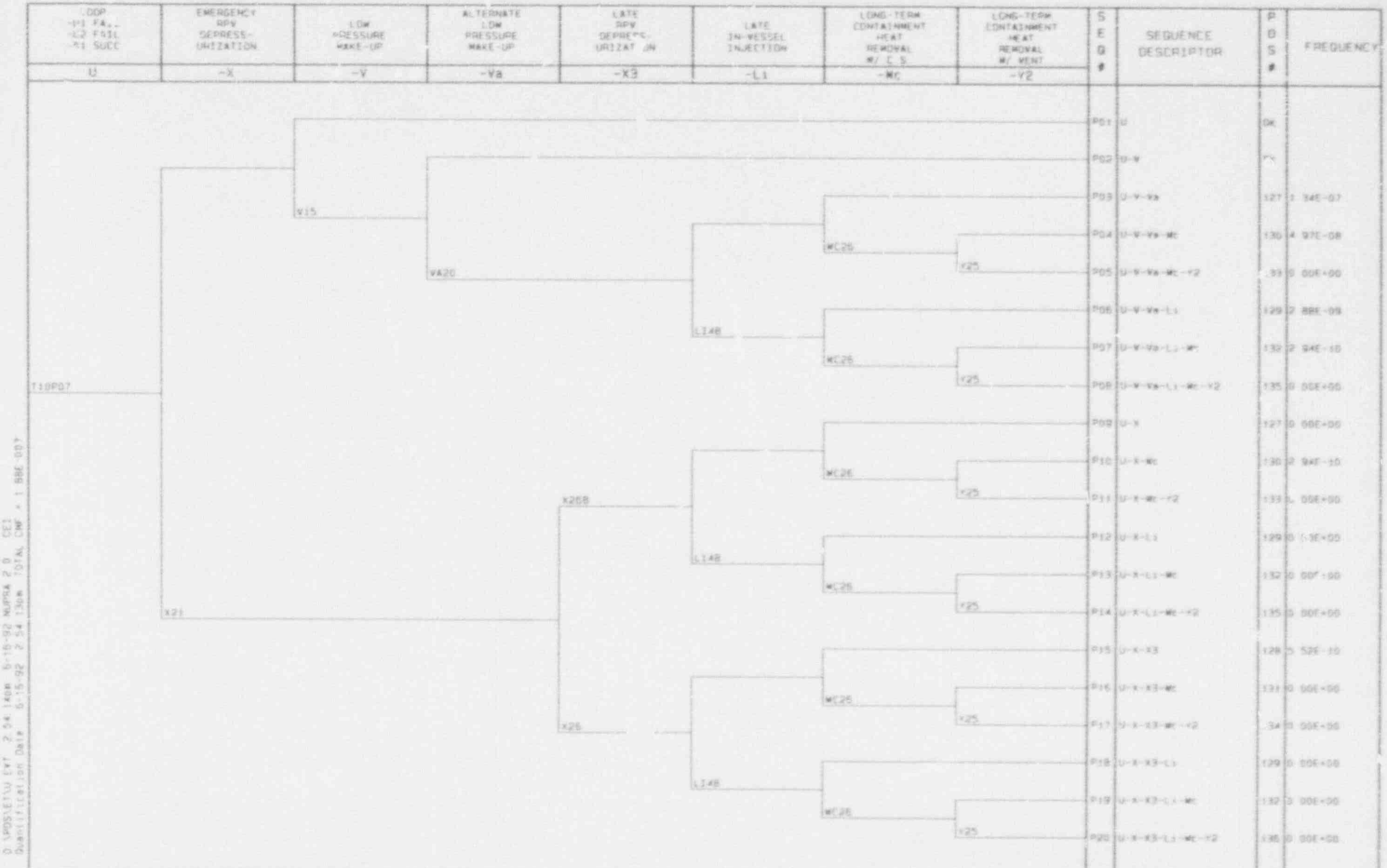
O:\PDS\ET\T14.EVT 2:29:24pm 5-16-92 NUPRA 2.0 CEI
Quantification Date 5-16-92 2:29:19pm TOTAL CMF = 2.99E-008

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

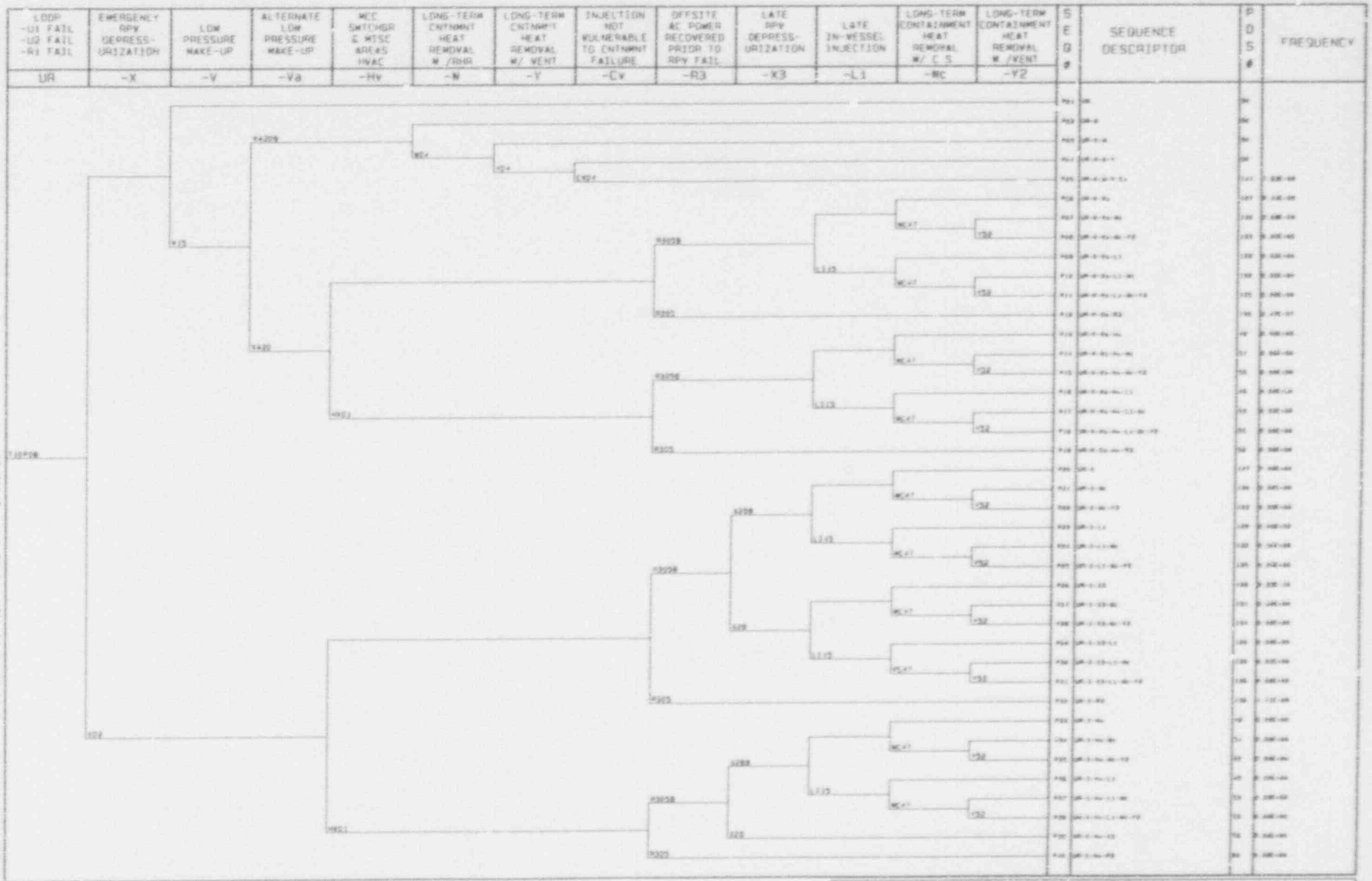
FIGURE 3.1.4-7 SHEET 6 OF 15
LOSS OF OFFSITE POWER PDS EVENT TREE
REVISION 0 (11/30/91)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-7 SHEET 7 OF 15
LOSS OF OFFSITE POWER PDS EVENT TREE
REVISION 1 (04/08/92)



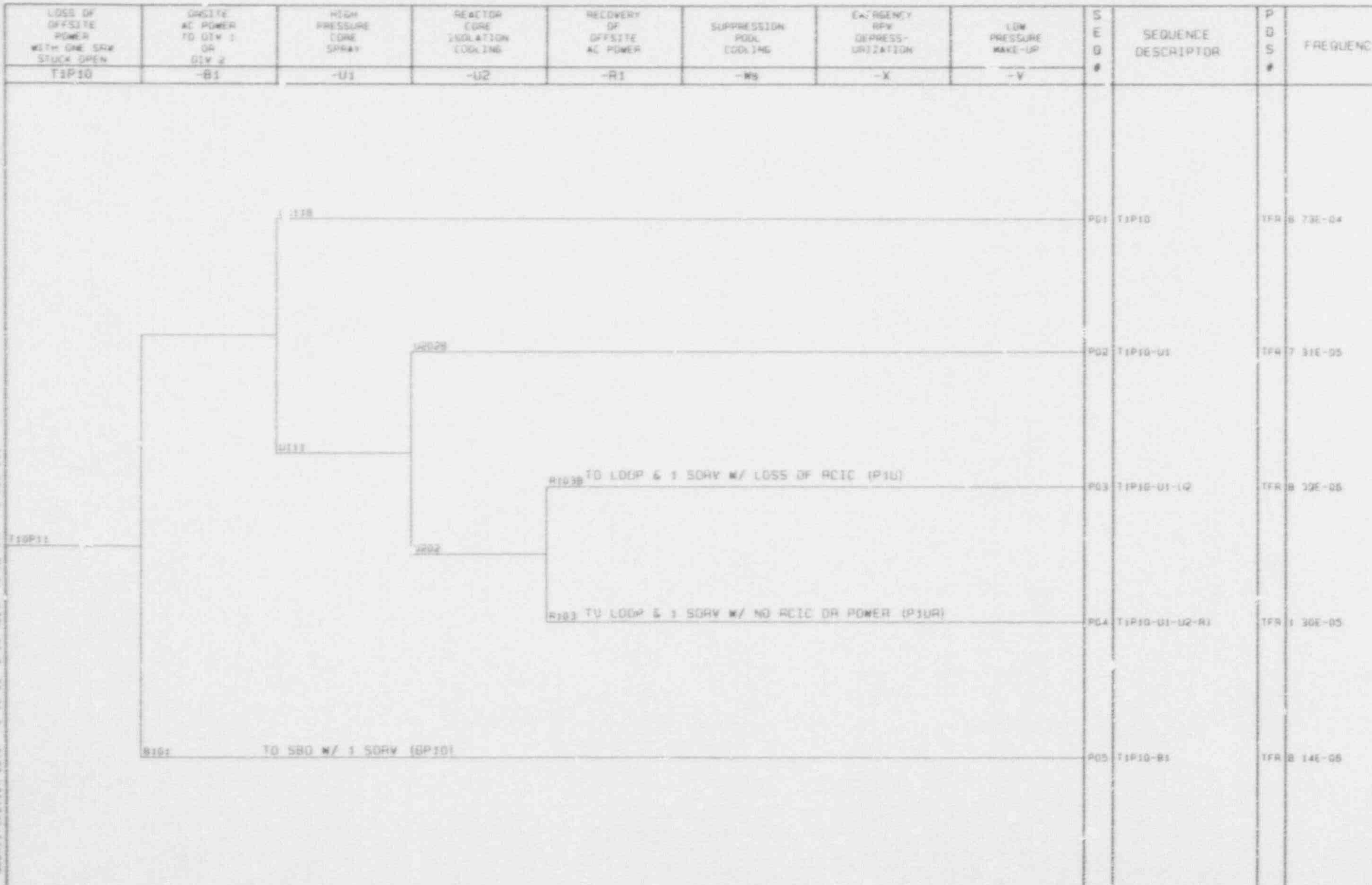
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-7 SHEET 8 OF 15
LOSS OF OFFSITE POWER PDS EVENT TREE
REVISION 1 (03/18/92)



D:\PDS\ET\UR.EXT 2:56:08pm 6-16-92 NUPRA 2.0 CEI
Quantification Date 6-16-92 2:55:45pm TOTAL CMF = 2.91E-007

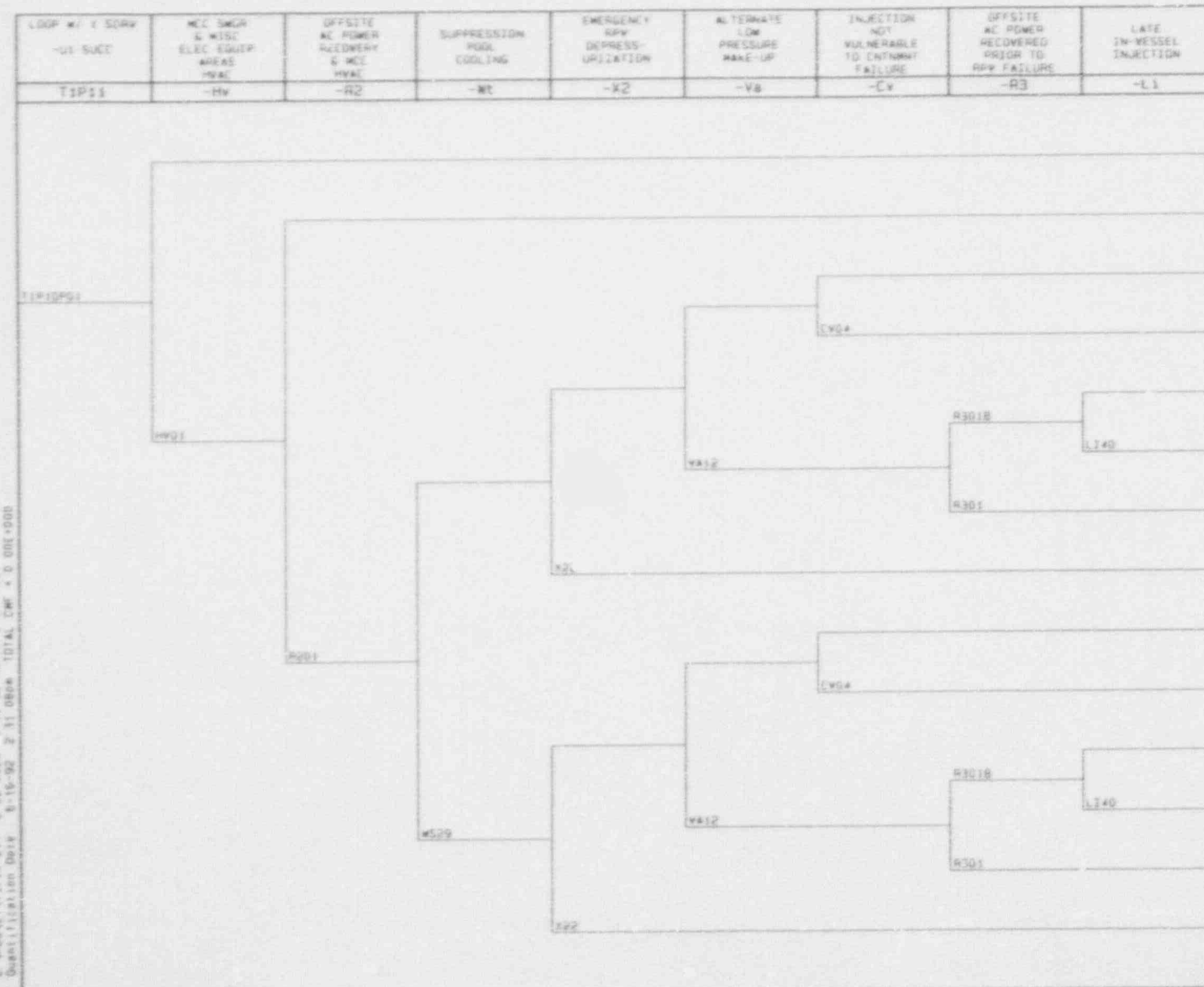
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-7 SHEET 9 OF 15
LOSS OF OFFSITE POWER PDS EVENT TREE
REVISION 1 (05/19/92)

D \PDS\ET\T1P10 EVT 10 34 12am 6-17-92 NUPRA 2 0 CEI
Quantification Date 6-16-92 2 30 42pm TOTAL CMF + 0 00E+000



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3 1 4-7 SHEET 10 OF 15
LOSS OF OFFSITE POWER PDS EVENT TREE
REVISION 1 (03/19/92)

D:\PDS\ET\T1P11.EVT 2:32.00pm 5-16-92 KJPRA 2.0 CEI
Quantification Date 5-16-92 2:31.00pm TOTAL CMF = 0.00E+000



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
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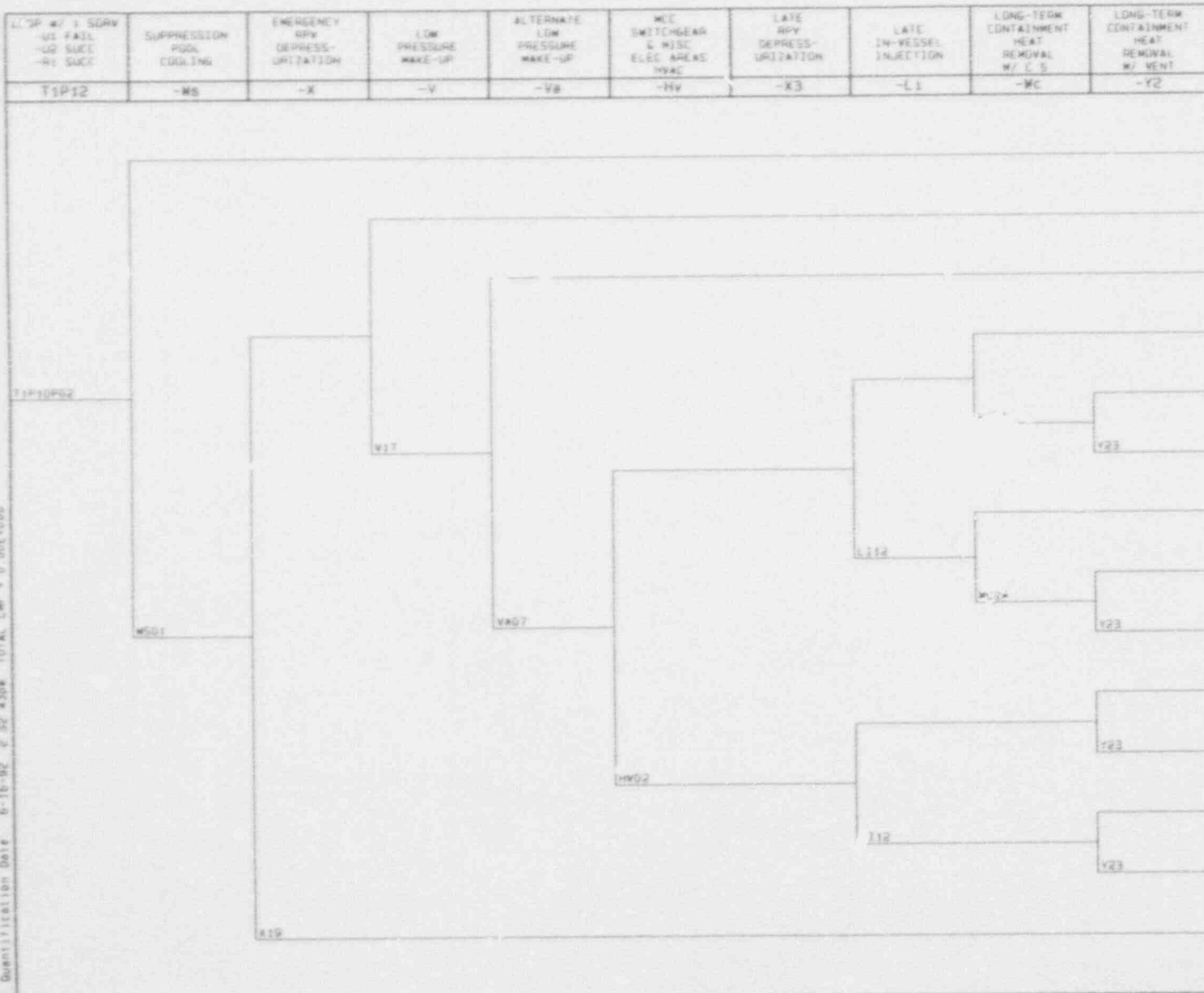
P02	T1P11-Hv	OK	
P03	T1P11-Hv-R2	OK	
P04	T1P11-Hv-R2-Cv	144	0.00E+00
P05	T1P11-Hv-R2-Va	119	0.00E+00
P06	T1P11-Hv-R2-Va-L1	121	0.00E+00
P07	T1P11-Hv-R2-Va-R3	125	0.00E+00
P08	T1P11-Hv-R2-X2	126	0.00E+00
P09	T1P11-Hv-R2-Wt	OK	
P10	T1P11-Hv-R2-Wt-Cv	144	0.00E+00
P11	T1P11-Hv-R2-Wt-Va	81	0.00E+00
P12	T1P11-Hv-R2-Wt-Va-L1	83	0.00E+00
P13	T1P11-Hv-R2-Wt-Va-R3	84	0.00E+00
P14	T1P11-Hv-R2-Wt-X2	85	0.00E+00

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

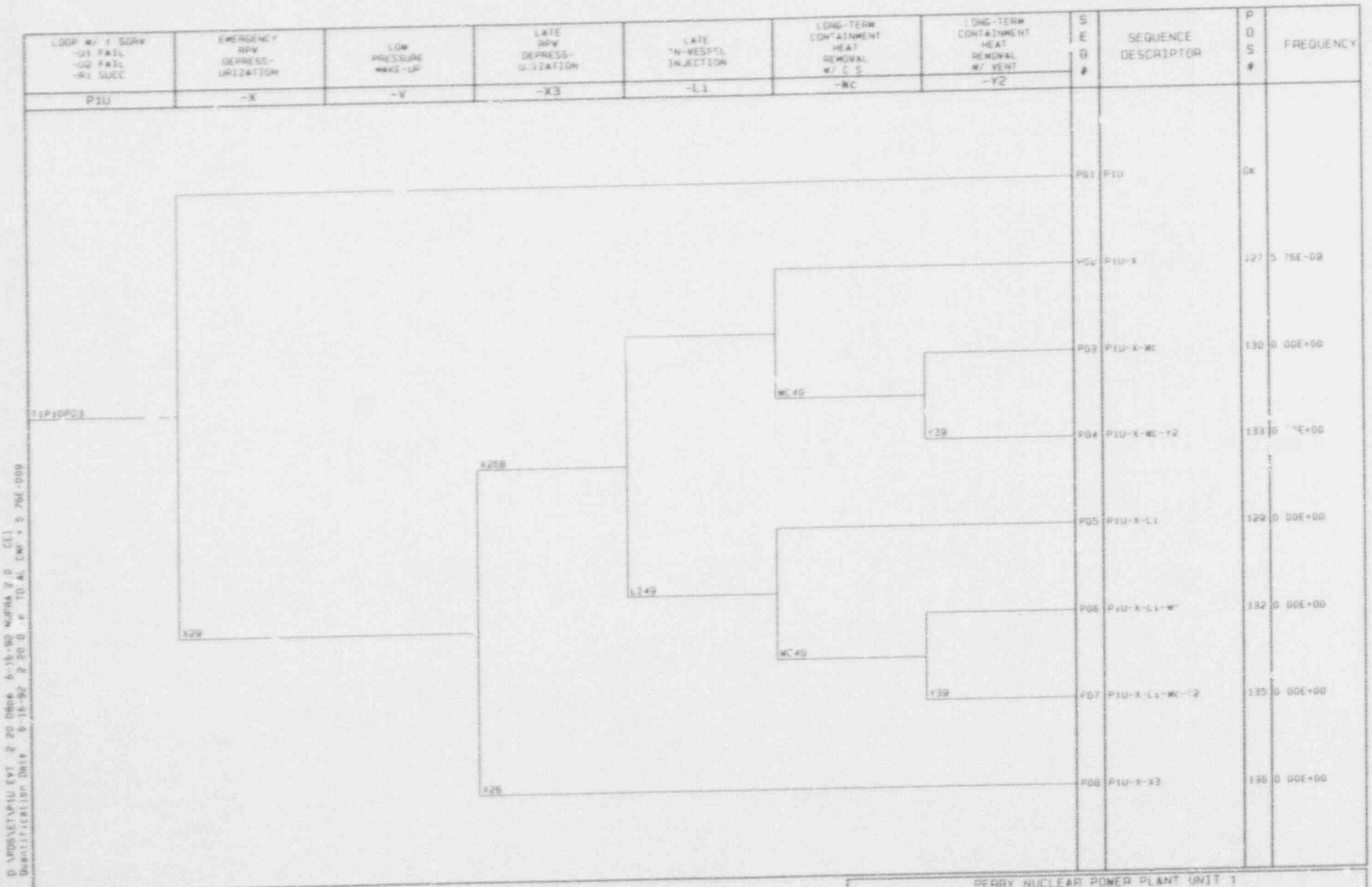
FIGURE 3.1.4-7 SHEET 11 OF 15
LOSS OF OFFSITE POWER PDS EVENT TREE
REVISION 1 (03/30/92)

D:\POS\ET\T1P12.EVT 2:33.08pm 6-16-92 NUPRA 2.0 CE1
Quantification Date 6-16-92 2:32:43pm TOTAL CMF = 0.00E+000



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T1P12	OK	
P02	T1P12-Ws	OK	
P03	T1P12-Ws-V	OK	
P04	T1P12-Ws-V-Va	127	0.00E+00
P05	T1P12-Ws-V-Va-Wc	130	0.00E+00
P06	T1P12-Ws-V-Va-Wc-Y2	133	0.00E+00
P07	T1P12-Ws-V-Va-L1	129	0.00E+00
P08	T1P12-Ws-V-Va-L1-Wc	132	0.00E+00
P09	T1P12-Ws-V-Va-L1-Wc-Y	135	0.00E+00
P10	T1P12-Ws-V-Va-Hv	62	0.00E+00
P11	T1P12-Ws-V-Va-Hv-Y2	65	0.00E+00
P12	T1P12-Ws-V-Va-Hv-L1	64	0.00E+00
P13	T1P12-Ws-V-Va-Hv-L1-Y	67	0.00E+00
P14	T1P12-Ws-X	67	0.00E+00

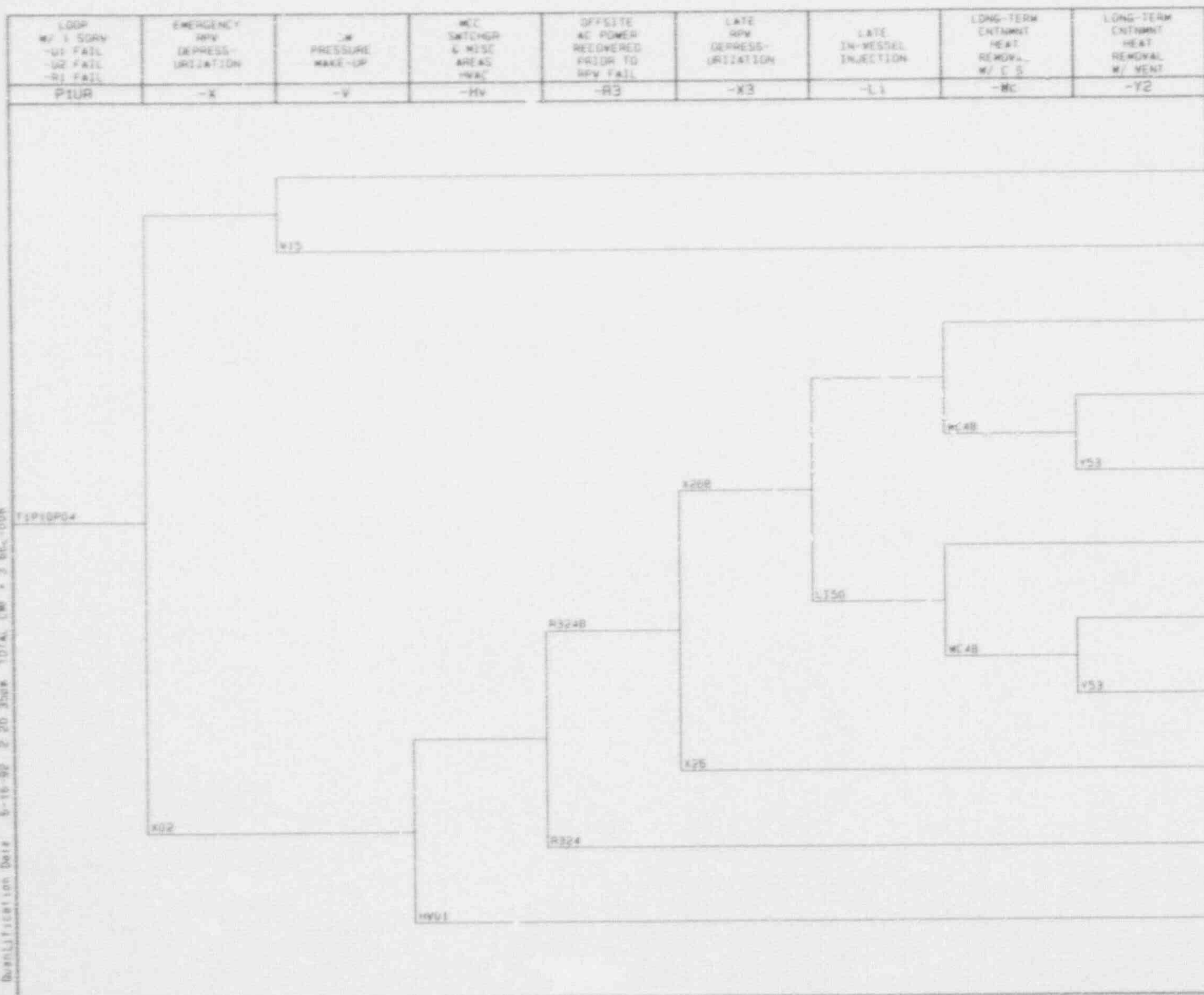
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-7 SHEET 12 OF 15
LOSS OF OFFSITE POWER PDS EVENT TREE
REVISION 0 (03/10/92)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-7 SHEET 15 OF 15
LOSS OF OFFSITE POWER PDS EVENT TREE
REVISION 0 (03/19/92)

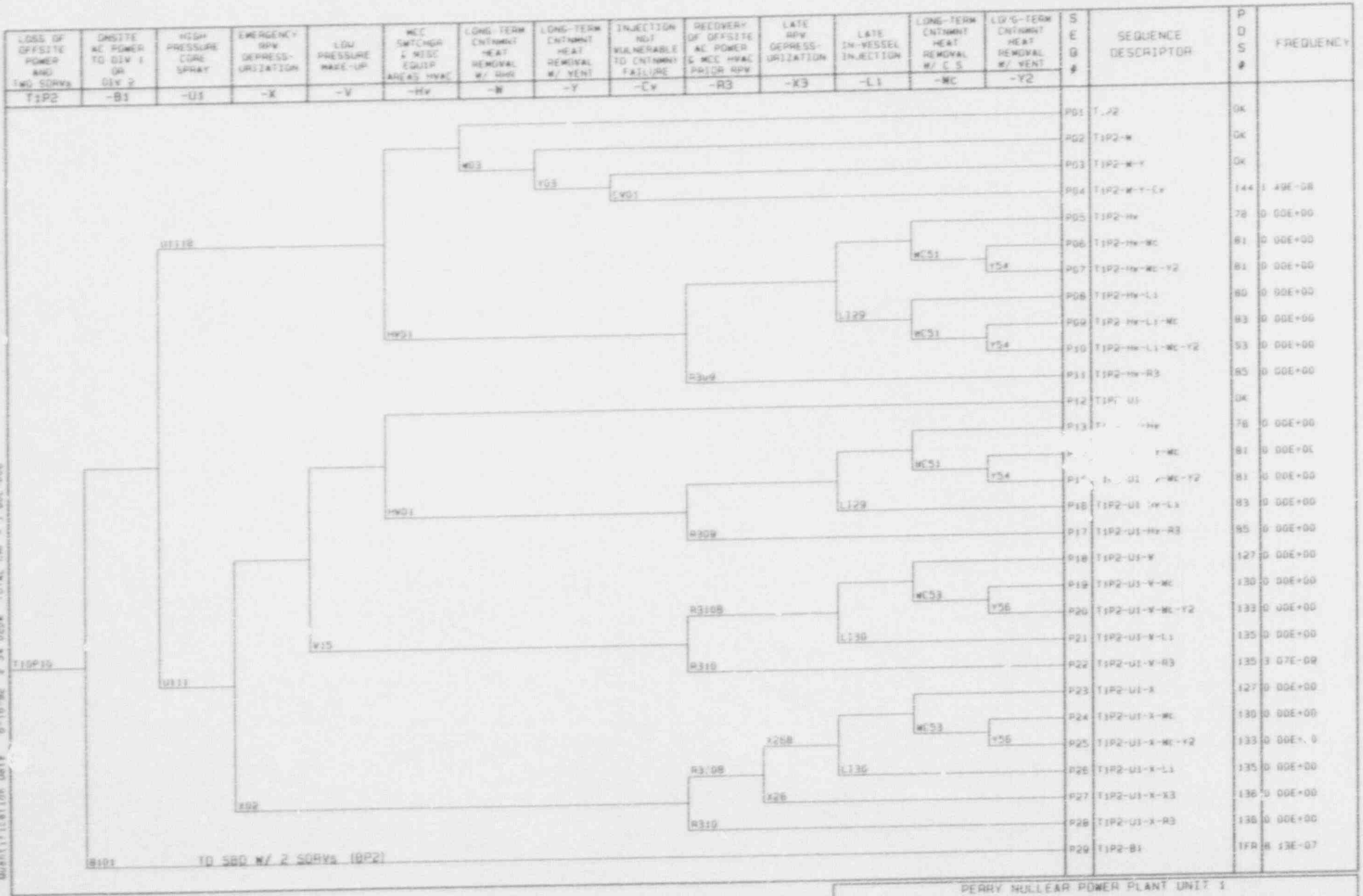
D:\PDS\ET\P1UR.EVT 2:20 56pm 6-16-92 NUPRA 2.0 CEI
Quantification Date 6-16-92 2:20 35pm TOTAL CMF = 3.66E-008



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	P1UR	OK	
P02	P1UR-V	136	3.66E-08
P03	P1UR-X	127	0.00E+00
P04	P1UR-X-Wc	130	0.00E+00
P05	P1UR-X-Wc-Y2	133	0.00E+00
P06	P1UR-X-L1	129	0.00E+00
P07	P1UR-X-L1-Wc	132	0.00E+00
P08	P1UR-X-L1-Wc-Y2	135	0.00E+00
P09	P1UR-X-X3	136	0.00E+00
P10	P1UR-X-R3	136	0.00E+00
P11	P1UR-X-Hv	60	0.00E+00

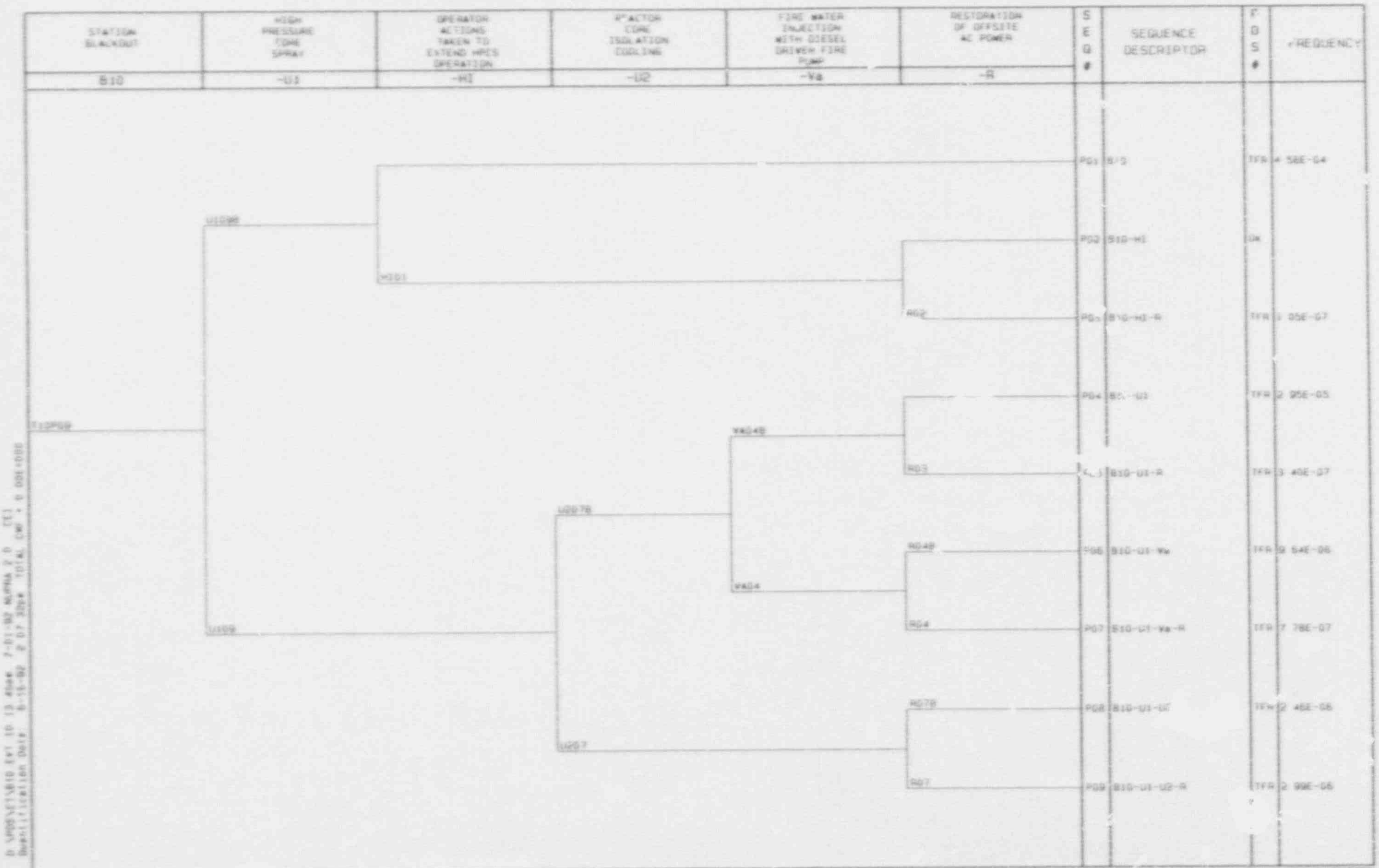
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-7 SHEET 14 OF 15
LOSS OF OFFSITE POWER PDS EVENT TREE
REVISION 0 (03/19/92)



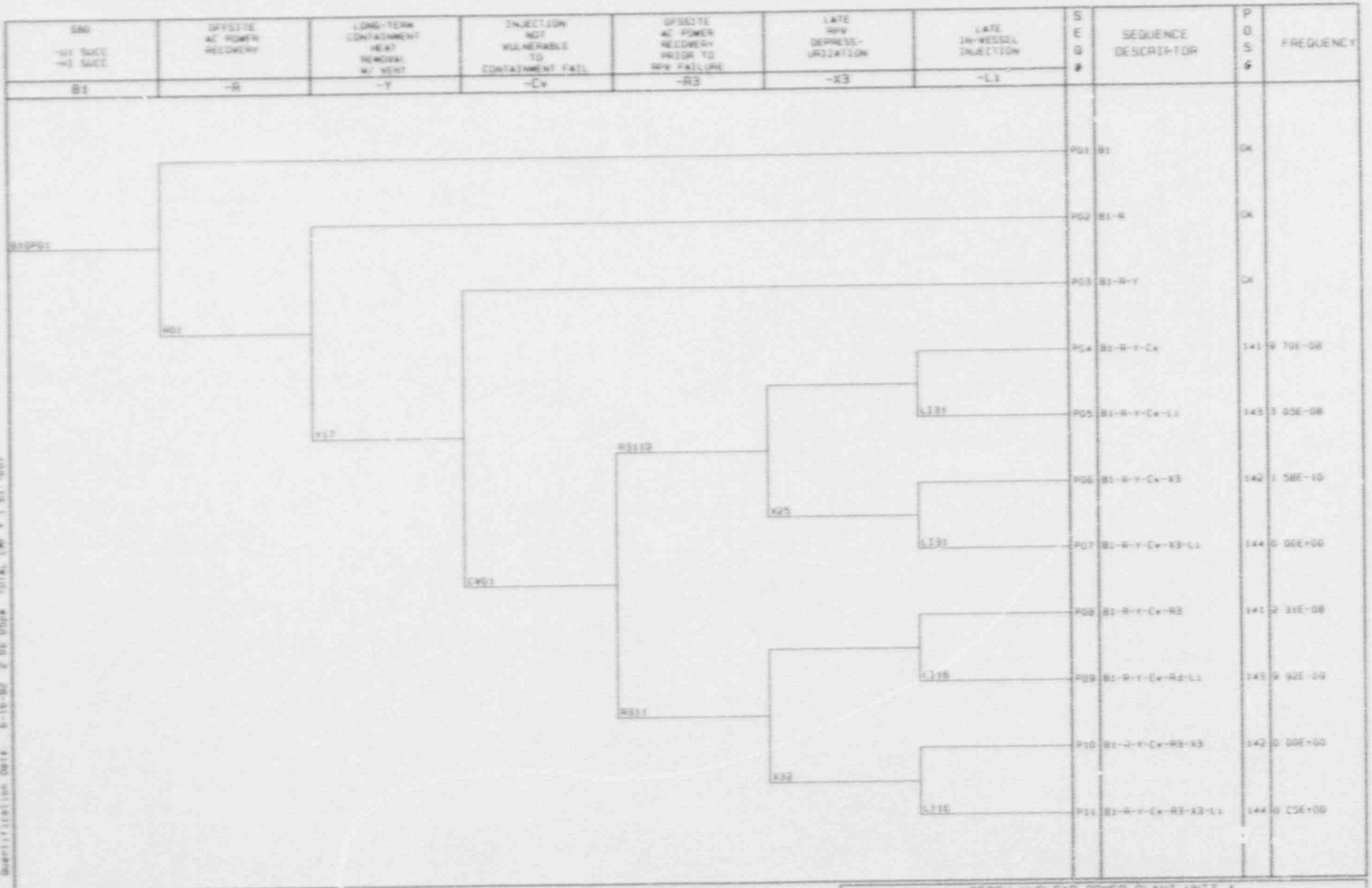
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-7 SHEET 15 OF 15
LOSS OF OFFSITE POWER PDS EVENT TREE
REVISION 1 (04/13/92)



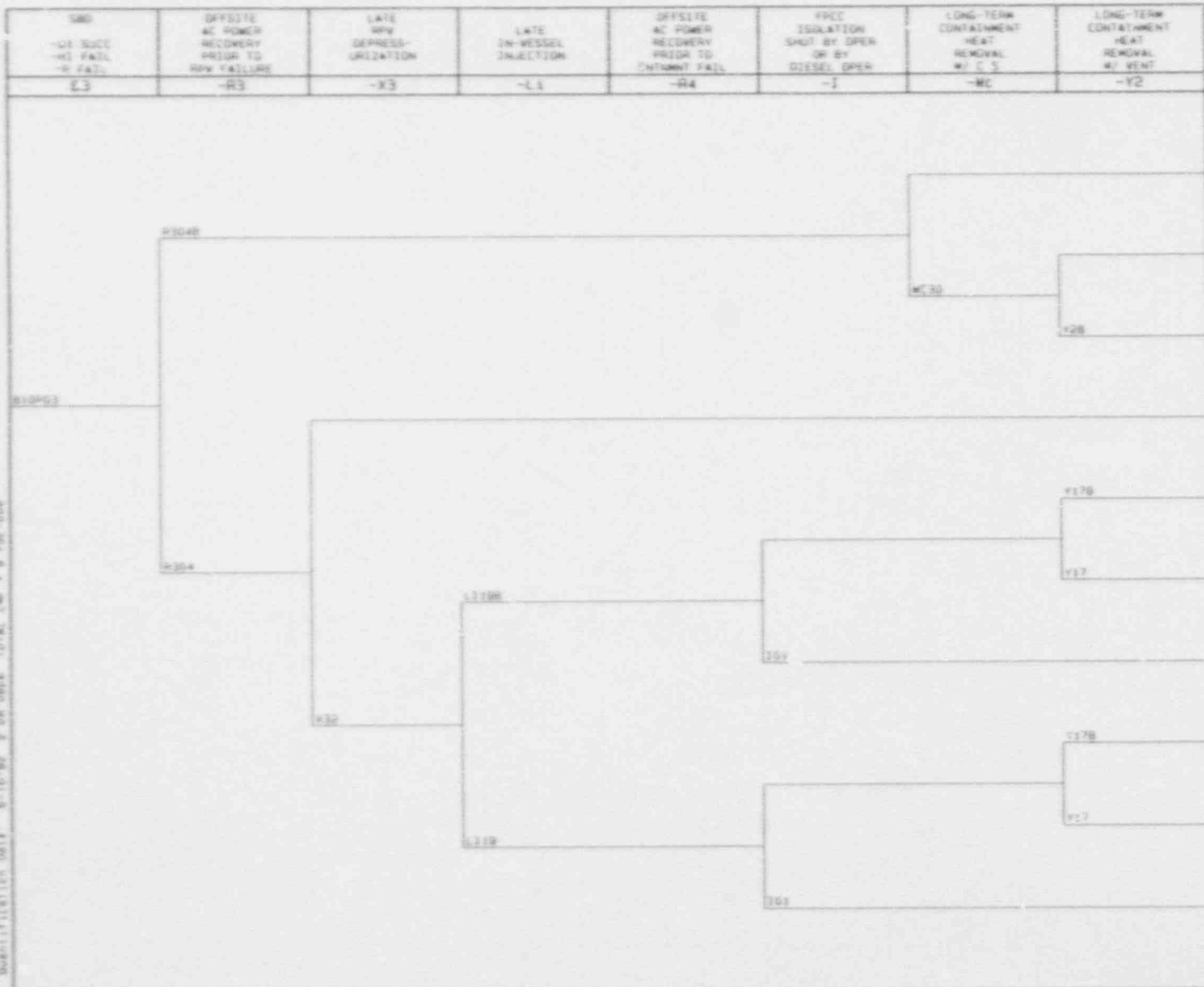
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-8 SHEET 1 OF 16
STATION BLACKOUT PDS EVENT TREE
REVISION 0 (02/13/92)

D:\PDS\ET\B1.EVT 10 14 24am 7-01-92 NUPRA 2.0 EE1
Quantification Date: 6-15-92 2:00 05pm TOTAL CMF = 1.61-007



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-B SHEET 2 OF 16
STATION BLACKOUT PDS EVENT TREE
REVISION 1 (03/14/92)

D:\PDS\ET\SB3.EVT 10:15:38am 7-01-92 NUPRA 2.0 CEI
Quantification Date: 6-16-92 2:08:08pm TOTAL CMF = 9.75E-008

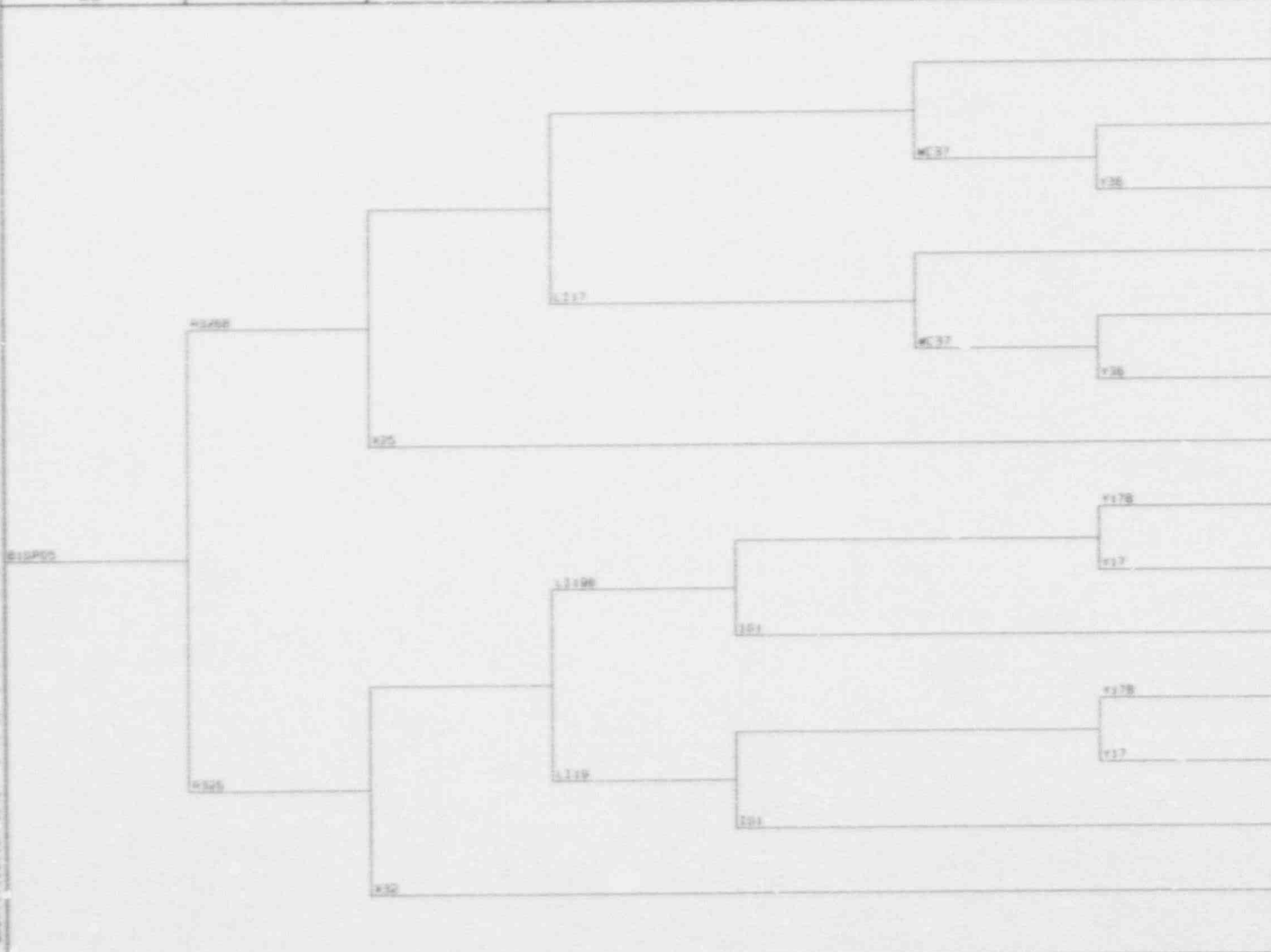


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	R3	32	5.31E-08
P02	R3-WC	33	6.40E-09
P03	R3-WC-Y2	35	0.00E+00
P04	R3-R3	42	5.34E-09
P05	R3-R3-X3	37	2.52E-08
P06	R3-R3-X3-Y2	41	6.64E-10
P07	R3-R3-X3-I	45	3.23E-10
P08	R3-R3-X3-L1	39	5.43E-09
P09	R3-R3-X3-L1-Y2	43	0.00E+00
P10	R3-R3-X3-L1-I	47	0.00E+00

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-8 SHEET 3 OF 16
STATION BLACKOUT PDS EVENT TREE
REVISION 1 (03/14/92)

D:\PDS\ET\B5.EVT 10 16 4### 7-01-92 NUPRA 2.0 CEI
Quantification Date: 6-15-92 2:08:58pm TOTAL CMF = 2.30E-007

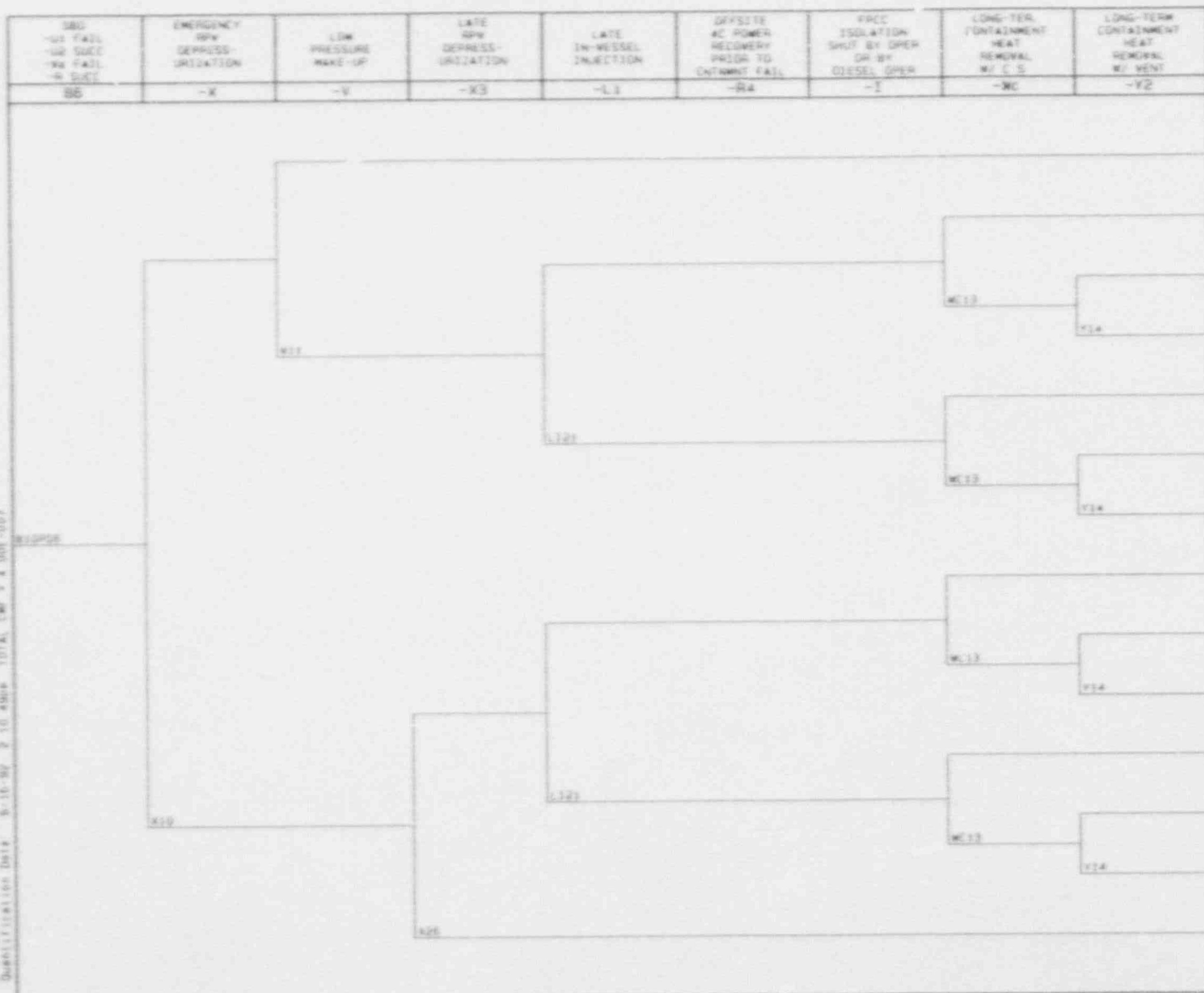
SBO -U1 FAIL -U2 SUCC -W# SUCC -R FAIL	OFFSITE AC POWER RECOVERY PRIOR TO RPV FAILURE	LATE RPV DEPRESS-URIZATION	LATE IN-VESSEL INJECTION	FPCC ISOLATION SHUT BY OPER OR BY DIESEL OPER	LONG-TERM CONTAINMENT HEAT REMOVAL W/ C.S.	LONG-TERM CONTAINMENT HEAT REMOVAL W/ VENT
B5	-R3	-X3	-L1	-I	-WC	-Y2



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	B5	32	7.15E-08
P02	B5-WC	33	6.21E-09
P03	B5-WC-Y2	35	0.00E+00
P04	B5-L1	DUM	0.00E+00
P05	B5-L1-WC	DUM	0.00E+00
P06	B5-L1-WC-Y2	DUM	0.00E+00
P07	B5-X3	DUM	0.00E+00
P08	B5-R3	36	1.[illegible]2E-07
P09	B5-R3-Y2	40	2.78E-08
P10	B5-R3-I	44	1.33E-09
P11	B5-R3-L1	38	1.83E-08
P12	B5-R3-L1-Y2	42	3.35E-09
P13	B5-R3-L1-I	[illegible]	0.00E+00
P14	B5-R3-X3	DUM	0.00E+00

PERRY NUCLEAR POWER PLANT
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-6 SHEET 4 OF 16
STATION BLACKOUT PDS EVENT TREE
REVISION 1 (03/17/92)

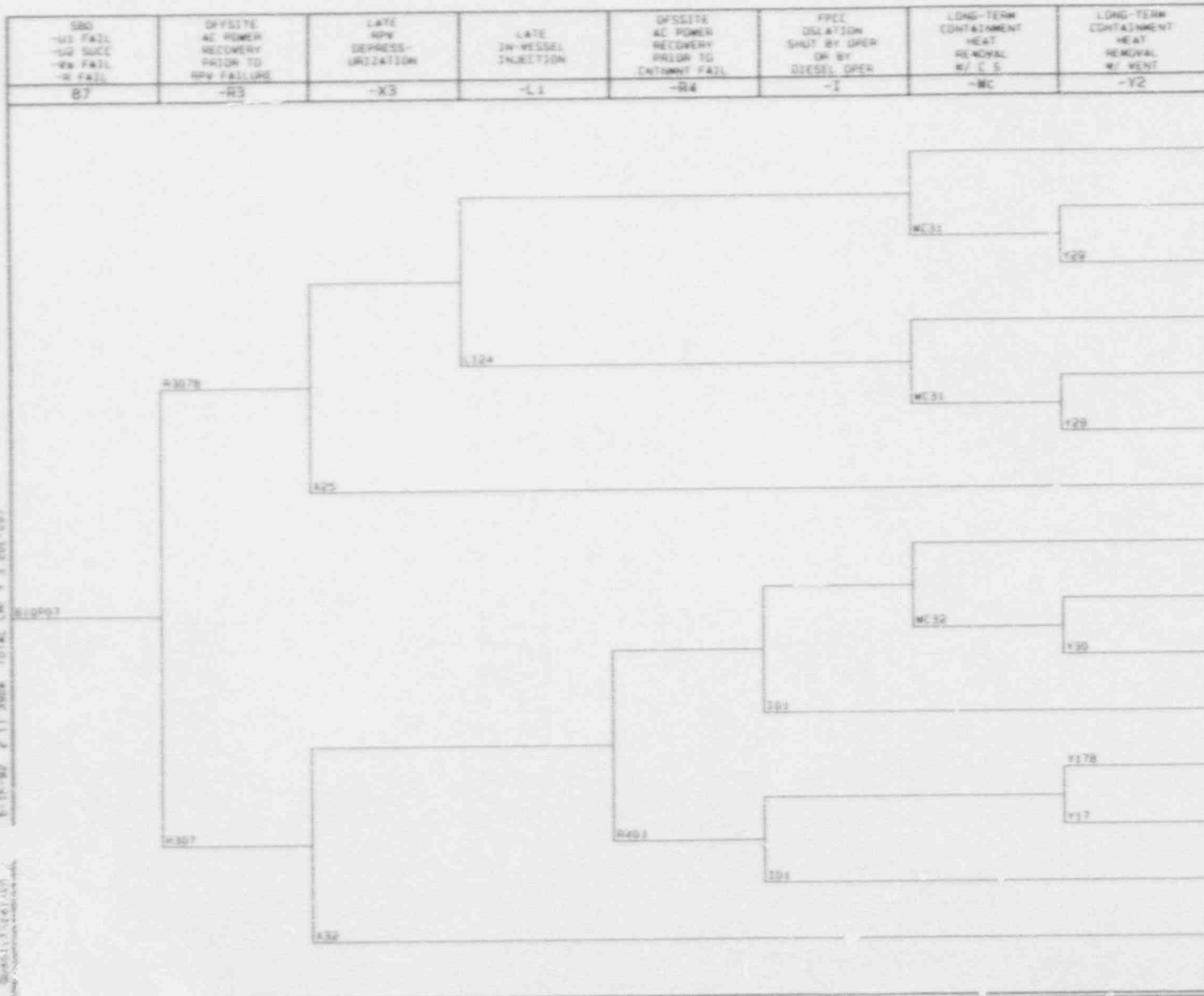
D:\POSSET\B6.EVT 10:17:28am 7-01-92 NUPRA 2.0 CEI
Quantification Date 9-16-92 2:10:49pm TOTAL CMF = 4.00E-007



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	B6	OK	
P02	B6-V	127	0.00E+00
P03	B6-V-WC	130	3.85E-07
P04	B6-V-WC-Y2	133	0.00E+00
P05	B6-V-L1	129	0.00E+00
P06	B6-V-L1-WC	132	1.46E-08
P07	B6-V-L1-WC-Y2	135	0.00E+00
P08	B6-X	127	0.00E+00
P09	B6-X-WC	DUM	0.00E+00
P10	B6-X-WC-Y2	DUM	0.00E+00
P11	B6-X-L1	DUM	0.00E+00
P12	B6-X-L1-WC	DUM	0.00E+00
P13	B6-X-L1-WC-Y2	DUM	0.00E+00
P14	B6-X-X3	DUM	0.00E+00

PERRY NUCLEAR POWER PLANT
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-8 SHEET 5 OF 16
STATION BLACKOUT PDS EVENT TREE
REVISION 0 (12/05/91)

1-RXJQ..., 4:16pm 7-01-92 NUPRA 2.0 CEI
QUANTIFICATION 6-18-92 2:11:39pm TOTAL CMF = 3.20E-007

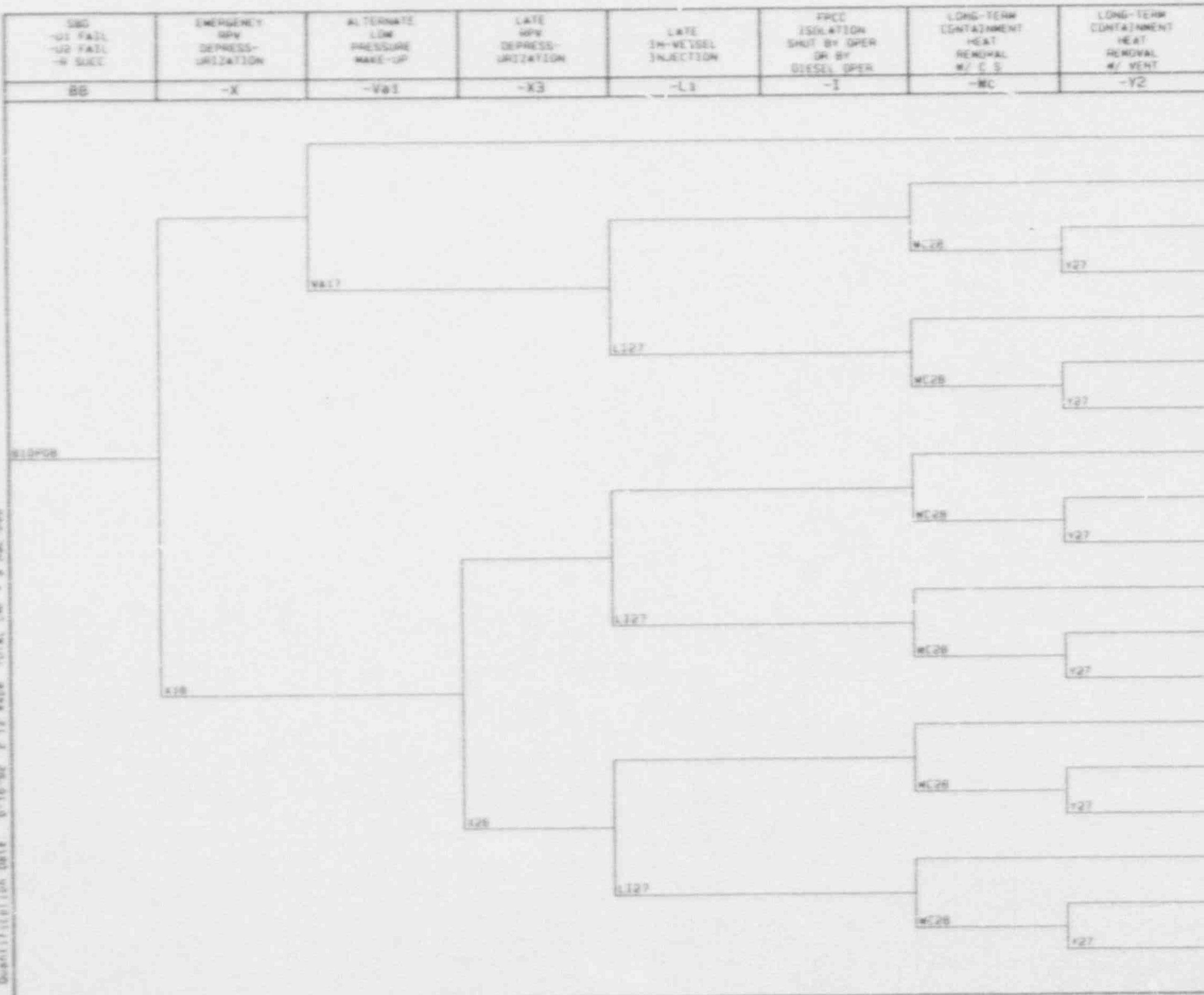


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	B7	20	7.37E-08
P02	B7-WC	22	4.15E-09
P03	B7-WC-Y2	24	0.00E+00
P04	B7-L1	21	0.00E+00
P05	B7-L1-WC	DUM	0.00E+00
P06	B7-L1-WC-Y2	DUM	0.00E+00
P07	B7-X3	DUM	0.00E+00
P08	B7-R3	25	2.08E-07
P09	B7-R3-WC	26	1.60E-08
P10	B7-R3-WC-Y2	DUM	0.00E+00
P11	B7-R3-I	45	1.89E-09
P12	B7-R3-R4	28	2.94E-09
P13	B7-R3-R4-Y2	30	0.00E+00
P14	B7-R3-R4-I	46	0.00E+00
P15	B7-R3-X3	31	1.34E-08

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-8 SHEET 5 OF 16
STATION BLACKOUT PDS EVENT TREE
REVISION 0 (12/05/91)

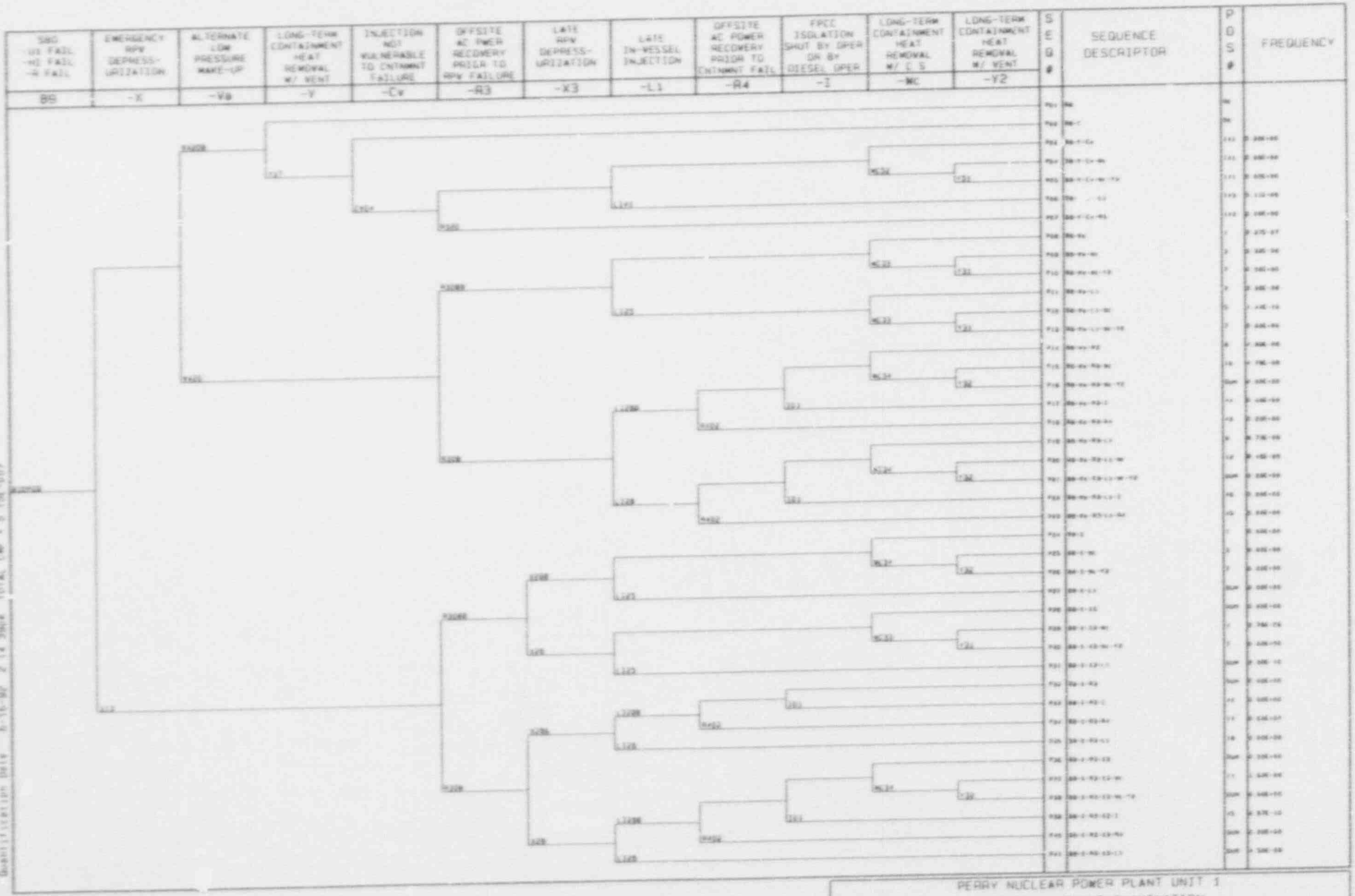
D:\PDS\E1\BB.EVT 10 18 58am 7-01-92 NUPRA 2.0 [E]
Quantification Date 0-16-92 2 12 44pm TOTAL CMF = 9 59E-008



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	BB	OK	
P02	BB-Va1	127	7.38E-08
P03	BB-Va1-WC	130	5.15E-09
P04	BB-Va1-WC-Y2	133	0.00E+00
P05	BB-Va1-L1	129	1.10E-10
P06	BB-Va1-L1-WC	132	1.00E-10
P07	BB-Va1-L1-WC-Y2	135	0.00E+00
P08	BB-X	127	0.00E+00
P09	BB-X-WC	130	0.00E+00
P10	BB-X-WC-Y2	133	0.00E+00
P11	BB-X-L1	129	0.00E+00
P12	BB-X-L1-WC	132	0.00E+00
P13	BB-X-L1-WC-Y2	135	0.00E+00
P14	BB-X-X3	128	0.00E+00
P15	BB-X-X3-WC	131	1.67E-08
P16	BB-X-X3-WC-Y2	133	0.00E+00
P17	BB-X-X3-L1	DUM	0.00E+00
P18	BB-X-X3-L1-WC	DUM	0.00E+00
P19	BB-X-X3-L1-WC-Y2	136	0.00E+00

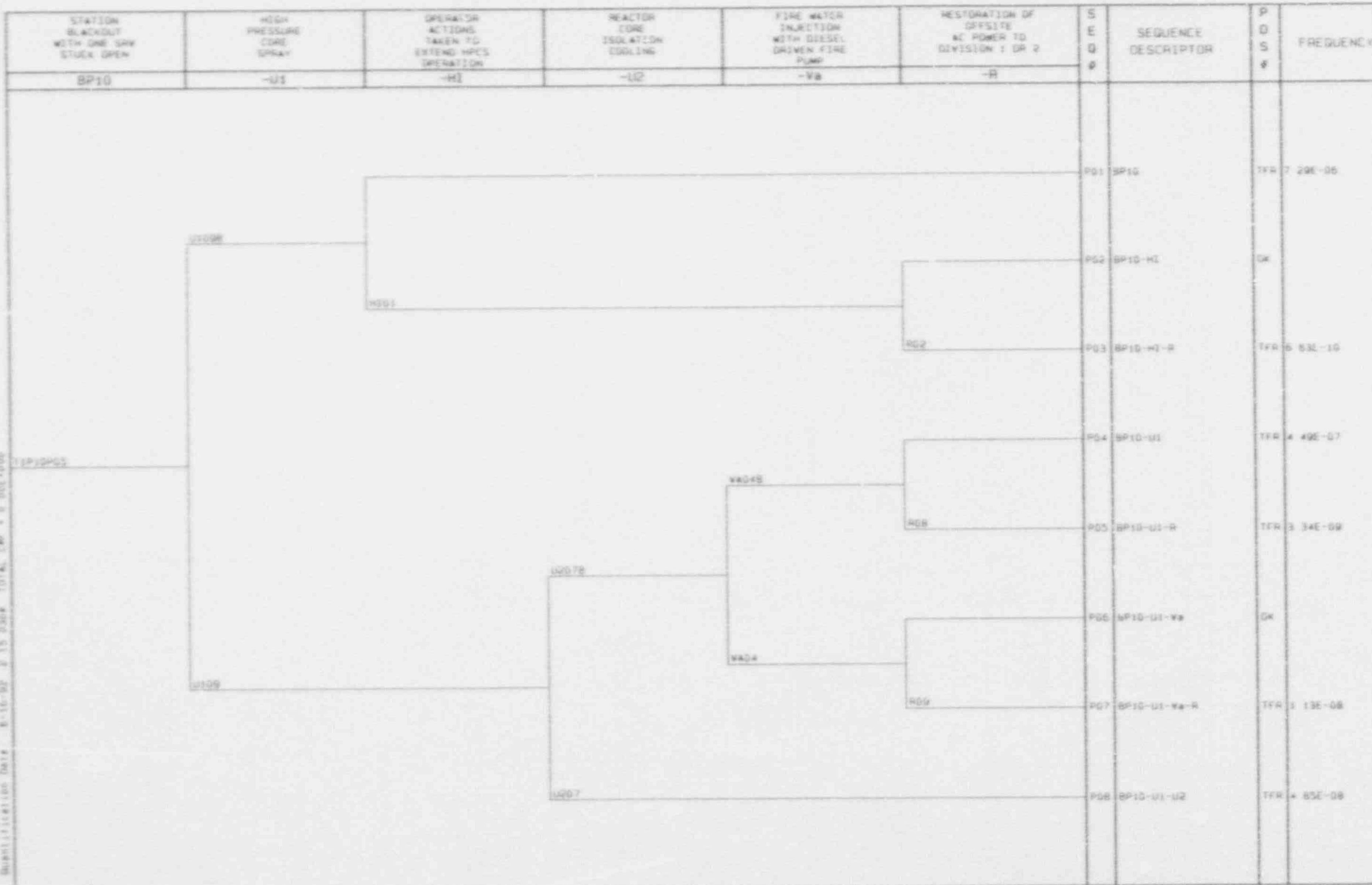
PERRY NUCLEAR POWER PLANT
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-B SHEET 7 OF 16
STATION BLACKOUT PDS EVENT TREE
REVISION 1 (03/17/92)

D:\PDS\ET\BB.EVT 10:23:40am 7-01-92 NUPRA 2.0 (C)
Quantification Date 6-16-92 2:14:39pm TOTAL CMF = 5.15E-007



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-B SHEET 8 of 16
STATION BLACKOUT PDS EVENT TREE
REVISION 2 (05/22/92)

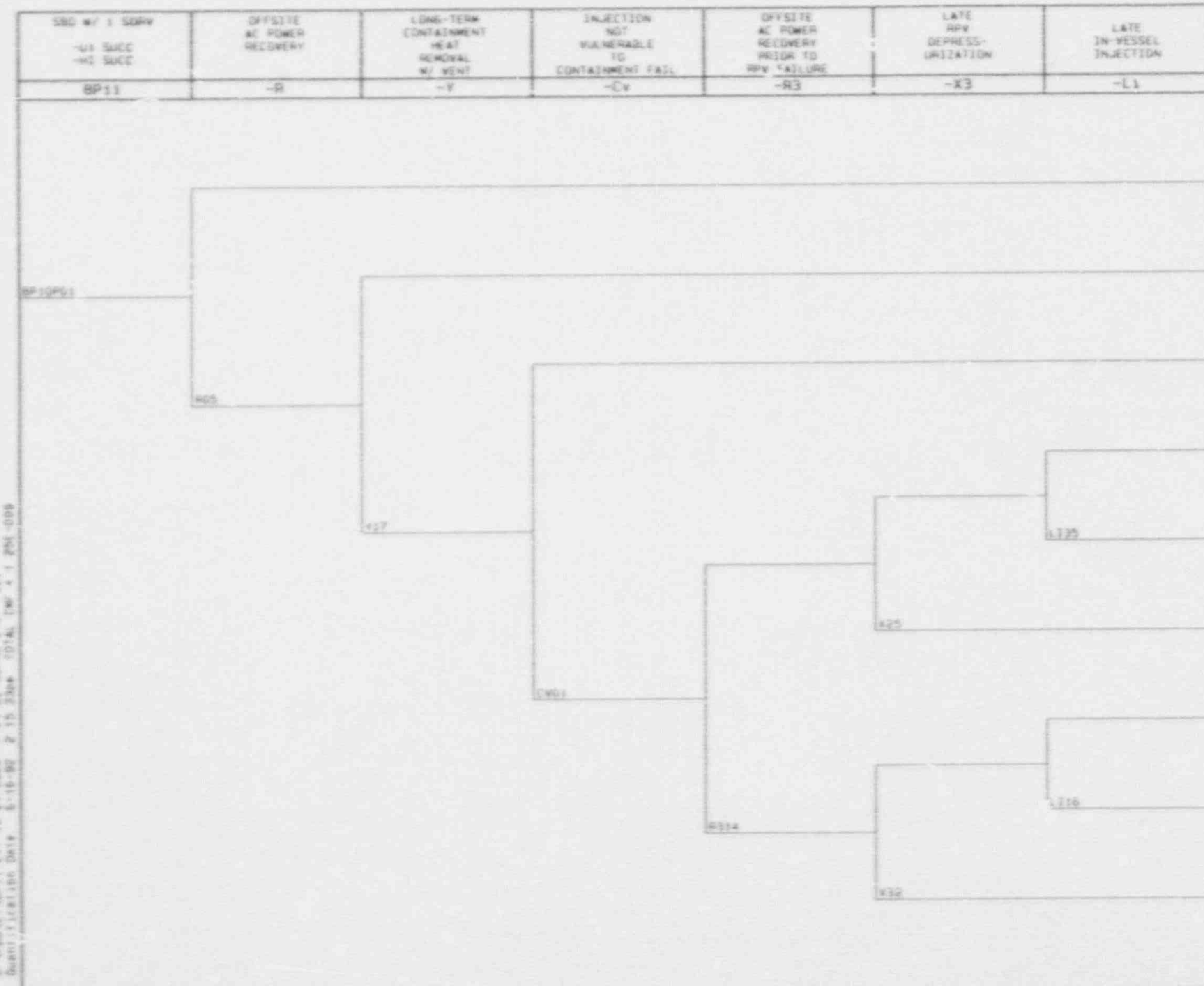
D:\PDS\E1\BP10.EVT 10:22 10am 7-01-92 NUPRA 2.0 CEI
Quantification Date: 8-16-92 2:15:03pm TOTAL CMF = 0.00E+000



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

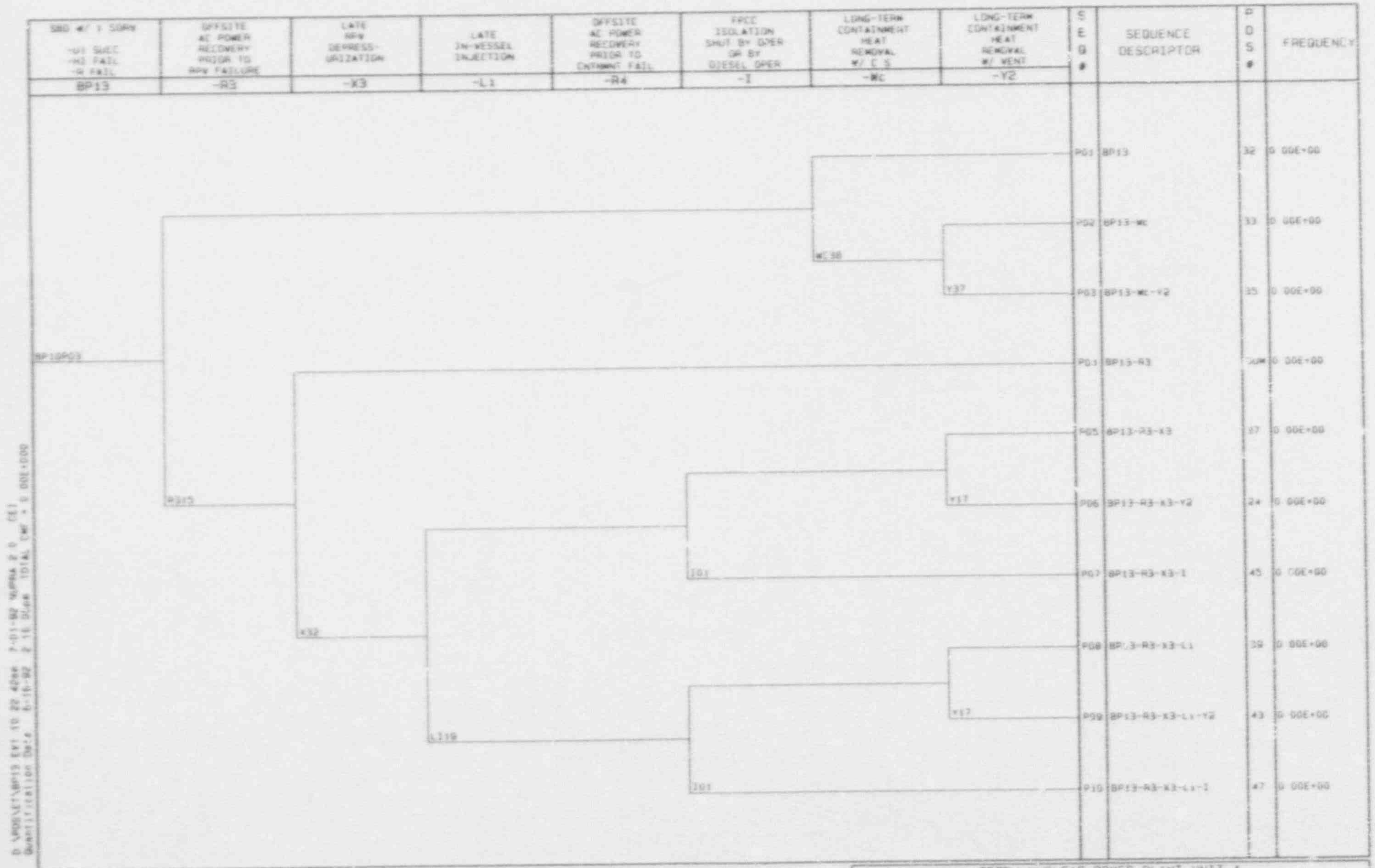
FIGURE 3.1.4-8 SHEET 9 of 16
STATION BLACKOUT PDS EVENT TREE
REVISION 0 (12/05/91)

D:\PSDS\E\BP11 EV1 10 21 28am 7-01-92 NUPRA 2.5 C11
Quantification Date 6-16-92 2 15 33pm TOTAL CMF = 1 25E-009

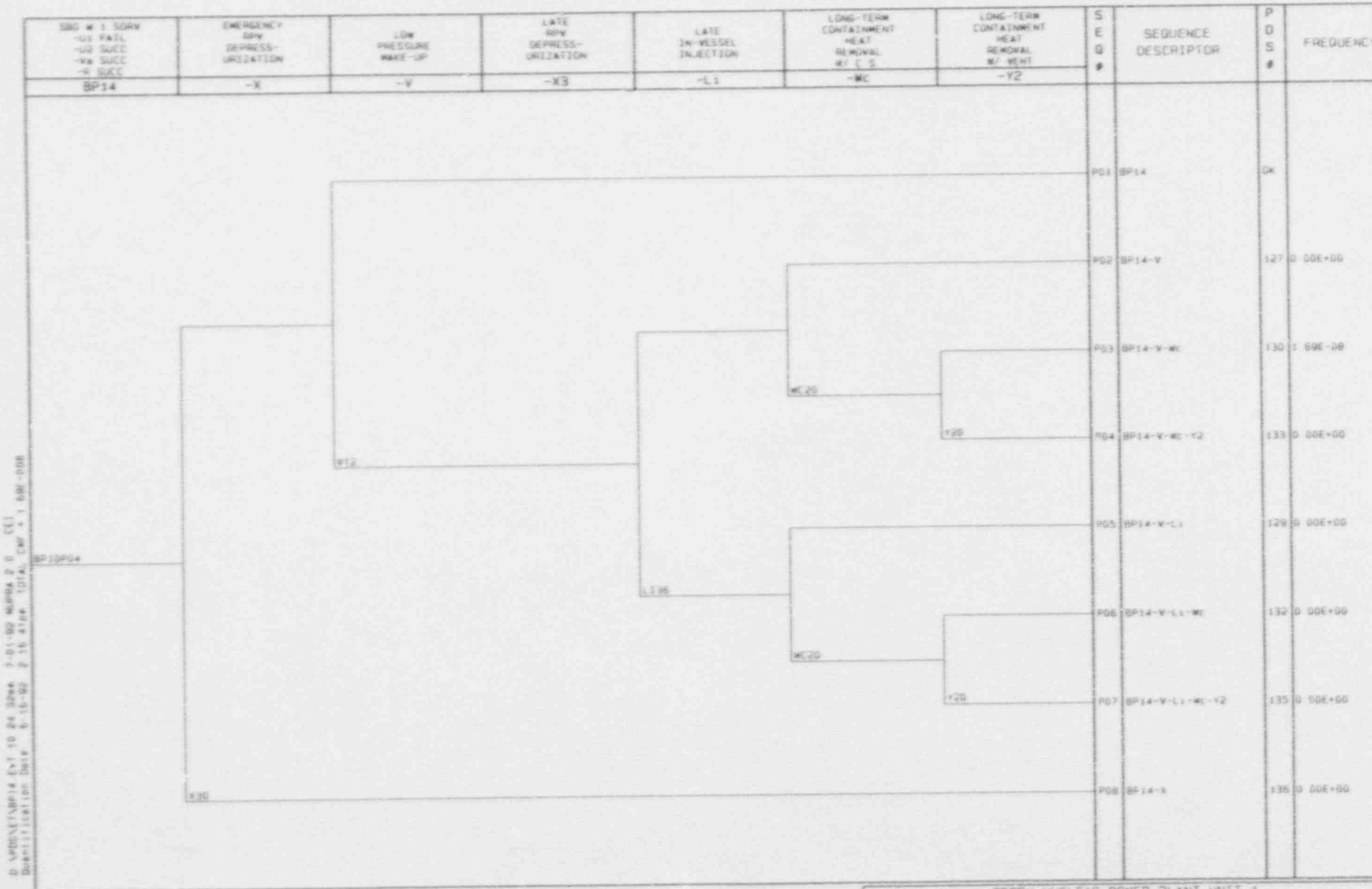


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	BP11	OK	
P02	BP11-R	OK	
P03	BP11-R-Y	OK	
P04	BP11-R-Y-Cv	141	8.29E-10
P05	BP11-R-Y-Cv-L1	143	0.00E+00
P06	BP11-R-Y-Cv-X3	144	0.00E+00
P07	BP11-R-Y-Cv-R3	141	4.23E-10
P08	BP11-R-Y-Cv-R3-L1	143	0.00E+00
P09	BP11-R-Y-Cv-R3-X3	144	0.00E+00

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.3.4-8 SHEET 10 OF 15
STATION BLACKOUT PDS EVENT TREE
REVISION 0 (12/05/91)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 1.1.4-8 SHEET 11 OF 16
STATION BLACKOUT PDS EVENT TREE
REVISION 1 (03/*8/92)

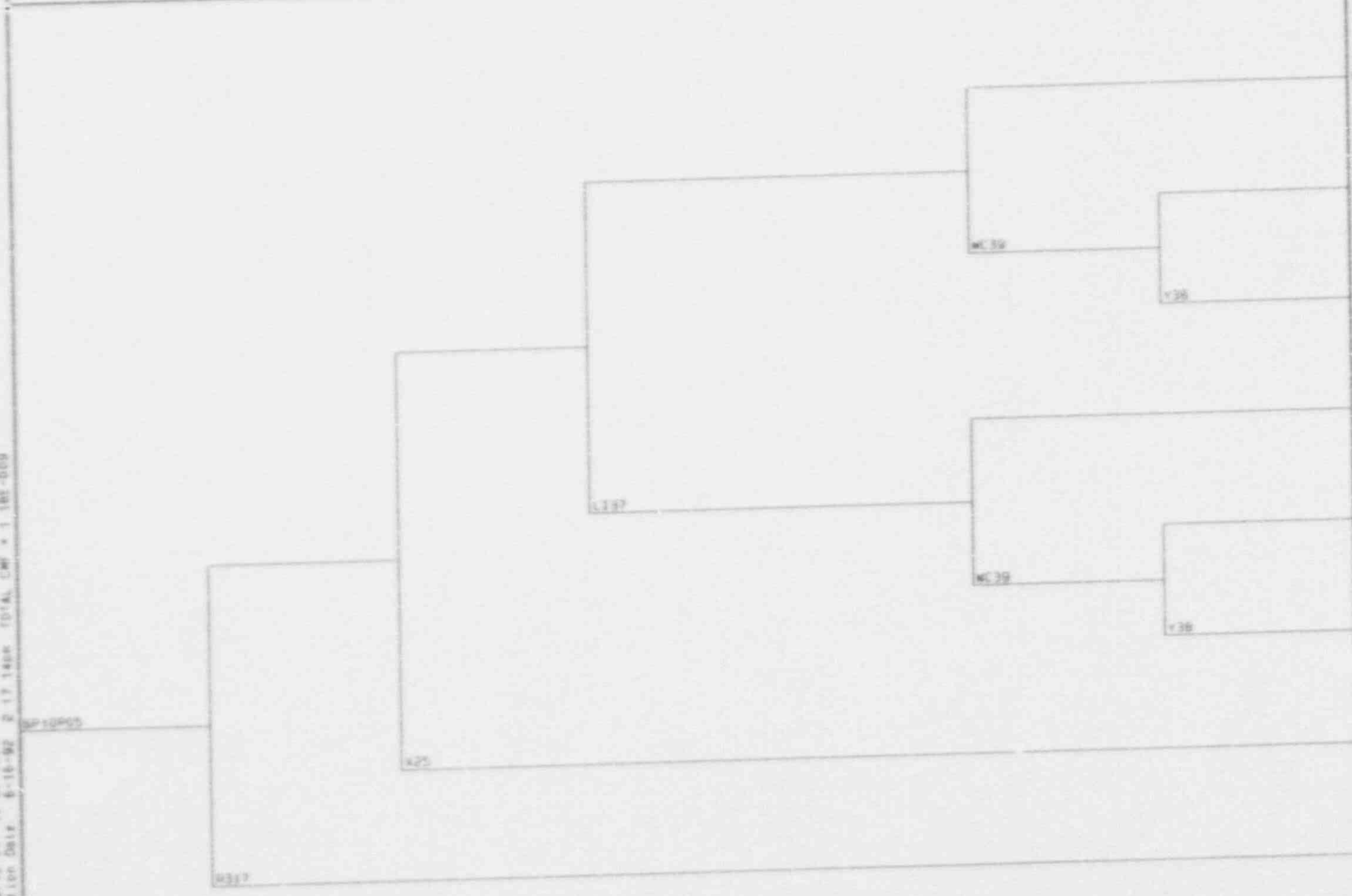


PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-8 SHEET 12 OF 16
STATION BLACKOUT PDS EVENT TREE
REVISION 1 (03/18/92)

D:\PDS\ET\BP15.EXT 10:25:12pm 7-01-92 NUPRA 2.0 CEI
Quantification Date 6-16-92 2:17:14pm TOTAL CMF = 1.18E-009

SBO W/ 1 SORV -U1 FAIL -U2 SUCC -Ha SUCC -S FAIL	OFFSITE AC POWER RECOVERY PRIOR TO RPV FAILURE	LATE RPV DEPRESS-URIZATION	LATE IN-VESSEL INJECTION	FPCC ISOLATION SHUT BY OPER OR BY DIESEL OPER	LONG-TERM CONTAINMENT HEAT REMOVAL W/ C.S	LONG-TERM CONTAINMENT HEAT REMOVAL W/ VENT
BP15	-R3	-X3	-L1	-I	-WC	-Y2

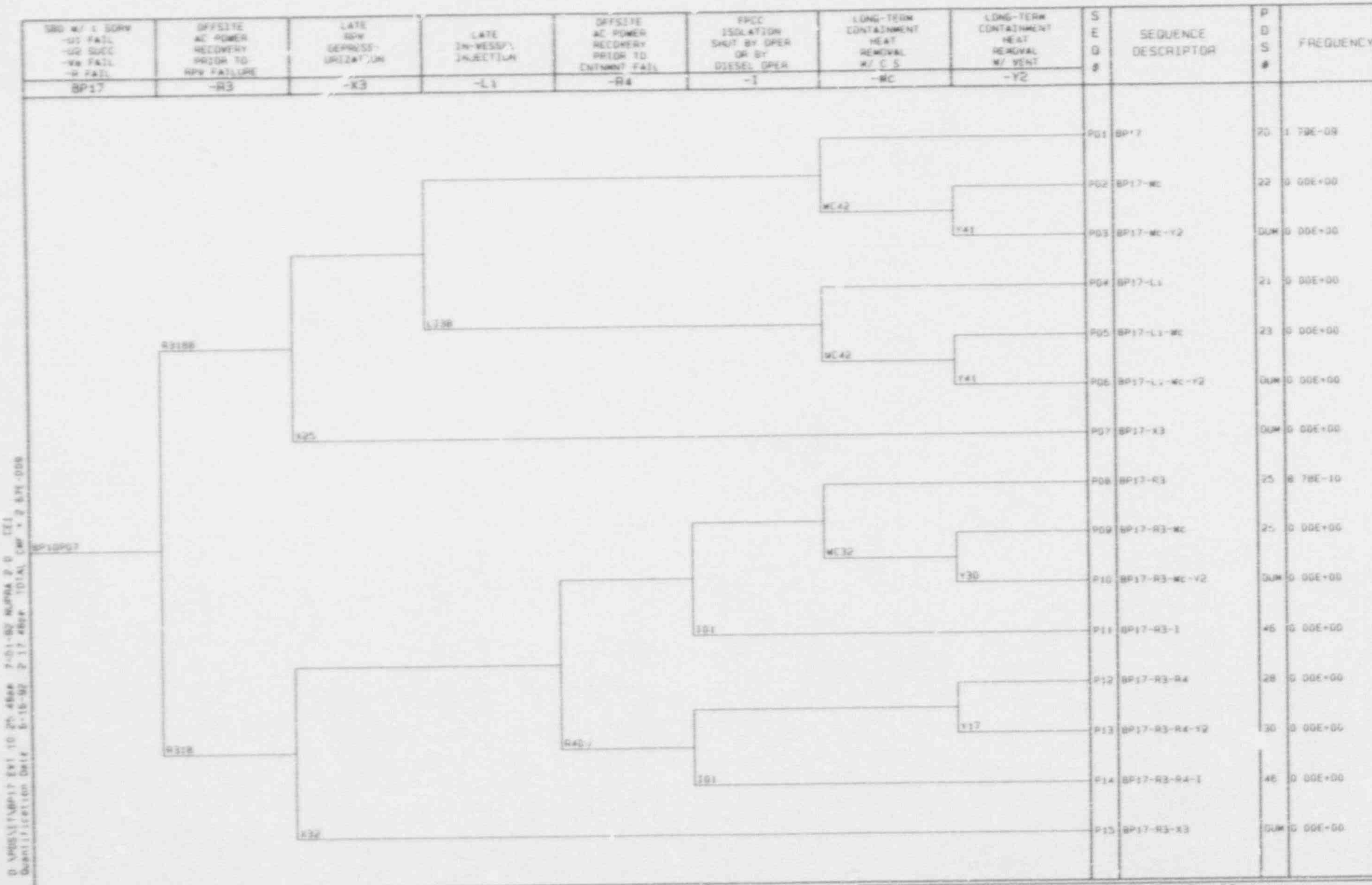


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	BP15	32	1.18E-09
P02	BP15-WC	33	0.00E+00
P03	BP15-WC-Y2	35	0.00E+00
P04	BP15-L1	32	0.00E+00
P05	BP15-L1-WC	33	0.00E+00
P06	BP15-L1-WC-Y2	35	0.00E+00
P07	BP15-X3	35	0.00E+00
P08	BP15-R3	43	0.00E+00

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-B SHEET 13 OF 16
STATION BLACKOUT PDS EVENT TREE
REVISION 1 (03/18/92)

D \PDS\ET\BP17 EVT 10 25 48am 7-01-92 NUPRA 2 0 CEI
Quantification Date 6-16-92 2 17 48pm TOTAL CMF = 2 67E-008



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3 1 4-B SHEET 14 OF 15
STATION BLACKOUT PDS EVENT TREE
REVISION 0 (12/05/91)

D \PDS\ET\BP18 EVT 10:27 14am 7-01-92 NUPRA 2.0 CEI
Quantification Date 6-16-92 2:18 50pm TOTAL CMF = 1.51E-008

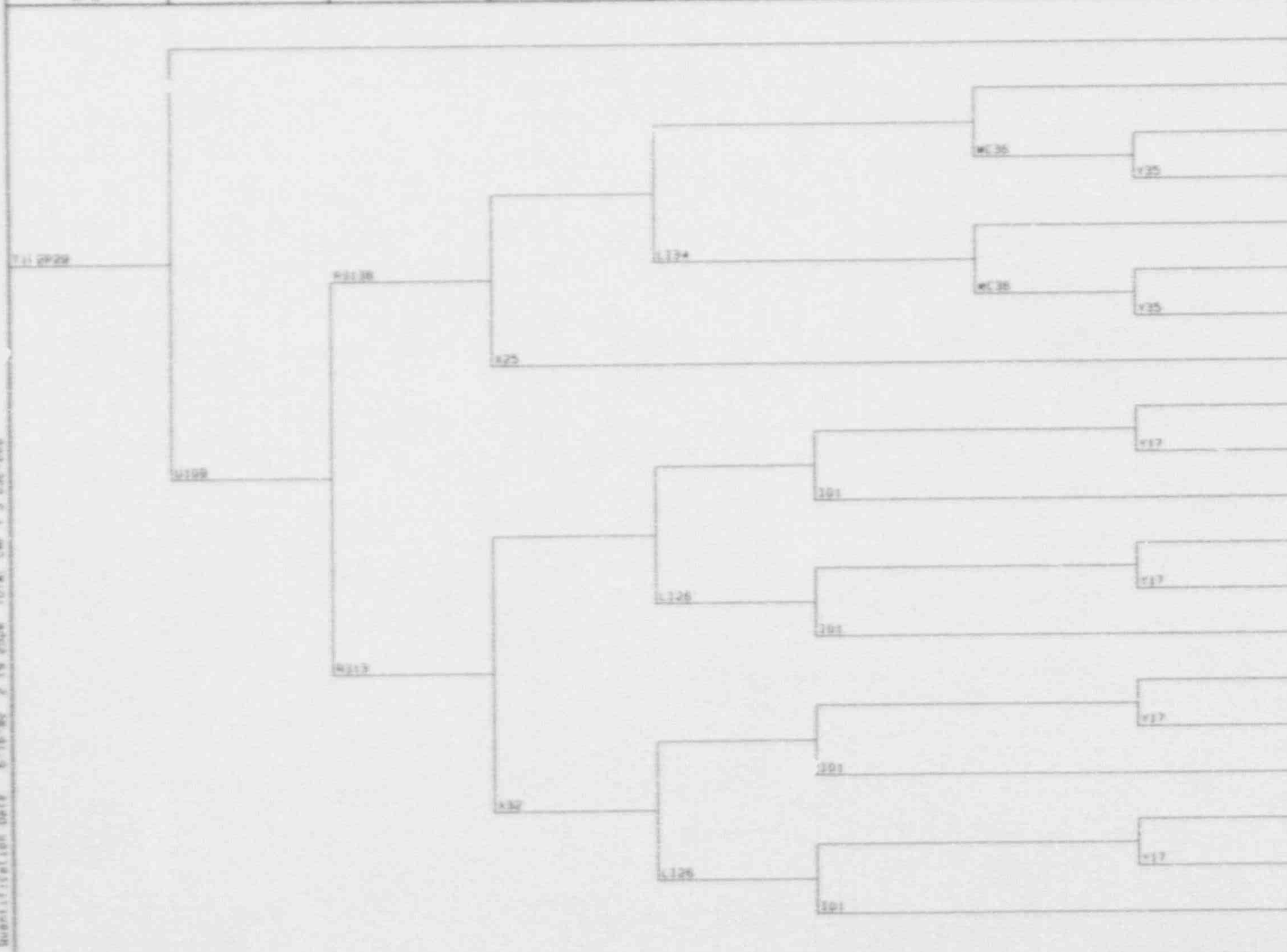
SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	BP18	1	8.77E-09
P02	BP18-WC	3	2.58E-10
P03	BP18-WC-Y2	7	0.00E+00
P04	BP18-L1	2	0.00E+00
P05	BP18-L1-WC	DUM	0.00E+00
P06	BP1? L1-WC-Y2	DUM	0.00E+00
P07	BP18-X3	DUM	0.00E+00
P08	BP18-X3-WC	DUM	0.00E+00
P09	BP18-X3-WC-Y2	DUM	0.00E+00
P10	BP18-X3-L1	DUM	0.00E+00
P11	BP18-X3-L1-WC	DUM	0.00E+00
P12	BP18-X3-L1-WC-Y2	DUM	0.00E+00
P13	BP18-R3	DUM	5.99E-09
P14	BP18-R3-WC	DUM	0.00E+00
P15	BP18-R3-WC-Y2	DUM	0.00E+00
P16	BP18-R3-I	44	0.00E+00
P17	BP18-R3-R4	13	0.00E+00
P18	BP18-R3-R4-Y2	17	0.00E+00
P19	BP18-R3-R4-I	44	0.00E+00
P20	BP18-R3-L1	9	0.00E+00
P21	BP18-R3-L1-WC	DUM	0.00E+00
P22	BP18-R3-L1-WC-Y2	DUM	0.00E+00
P23	BP18-R3-L1-I	DUM	0.00E+00
P24	BP18-R3-L1-R4	DUM	0.00E+00
P25	BP18-R3-L1-R4-Y2	DUM	0.00E+00
P26	BP18-R3-L1-R4-I	DUM	0.00E+00
P27	BP18-R3-X3	14	1.07E-10
P28	BP18-R3-X3-Y2	DUM	0.00E+00
P29	BP18-R3-X3-I	DUM	0.00E+00
P30	BP18-R3-X3-L1	DUM	0.00E+00
P31	BP18-R3-X3-L1-Y2	DUM	0.00E+00
P32	BP18-R3-X3-L1-I	DUM	0.00E+00

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-8 SHEET 15 OF 16
STATION BLACKOUT PDS EVENT TREE
REVISION 1 (03/18/92)

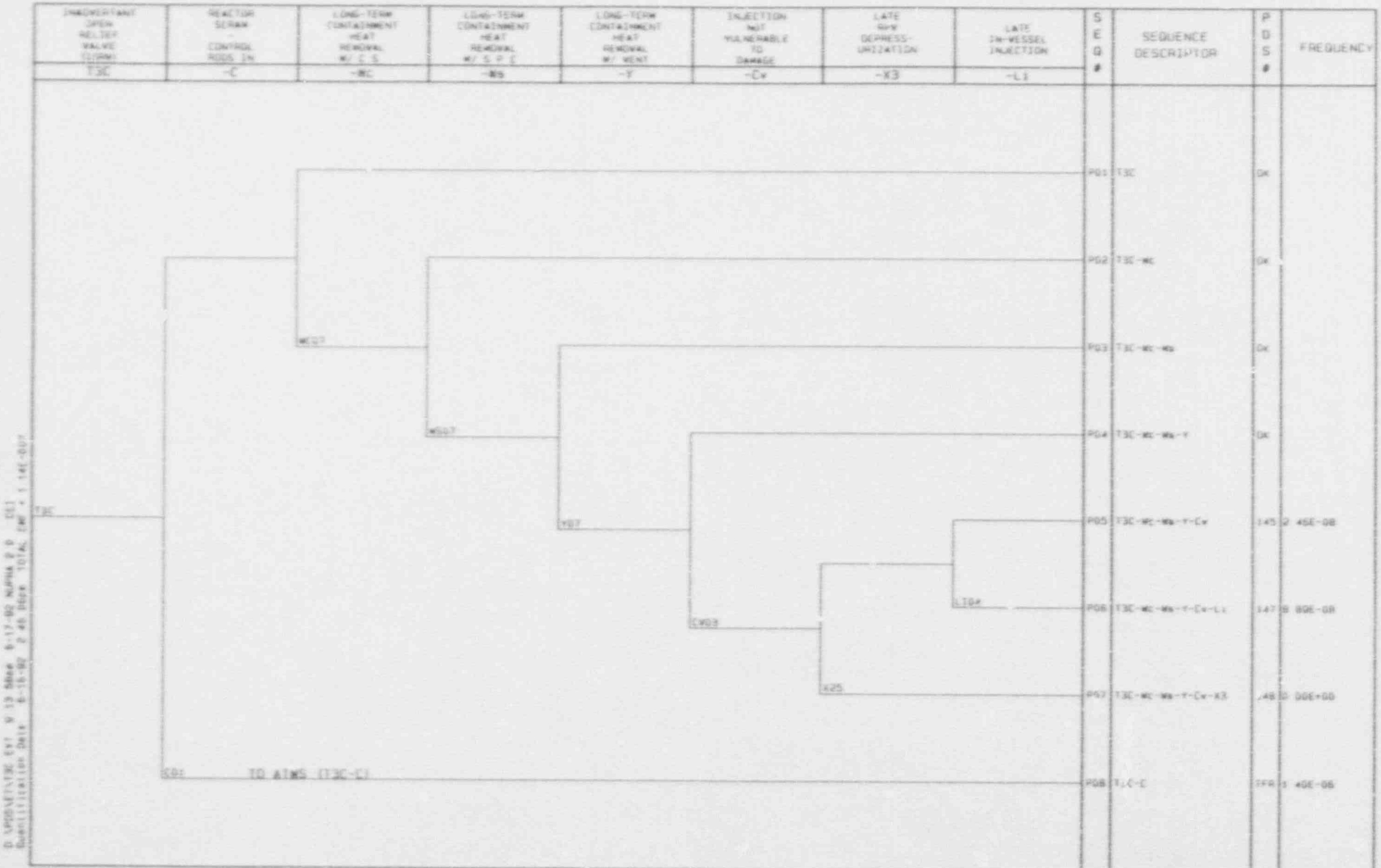
D \PDS\E1\BP2.EVT 10 27 46am 7-01-92 NUPRA 2.0 CEI
Quantification Date 6-16-92 2 19 25pm TOTAL CMF = 3.03E-006

STATION BLACKOUT WITH TWO SRVs STUCK OPEN	HIGH PRESSURE CORE SPRAY	OFFSITE AC POWER RECOVERY PRIOR TO RPV FAILURE	LATE RPV DEPRESS-URIZATION	LATE IN-VESSEL INJECTION	FPCC ISOLATION SHUT BY OPER OR BY DIESEL OPER	LONG-TERM CONTAINMENT HEAT REMOVAL W/ C.S	LONG-TERM CONTAINMENT HEAT REMOVAL W/ VENT
BP2	-U1	-R3	-X3	-L1	-I	-WC	[illegible]



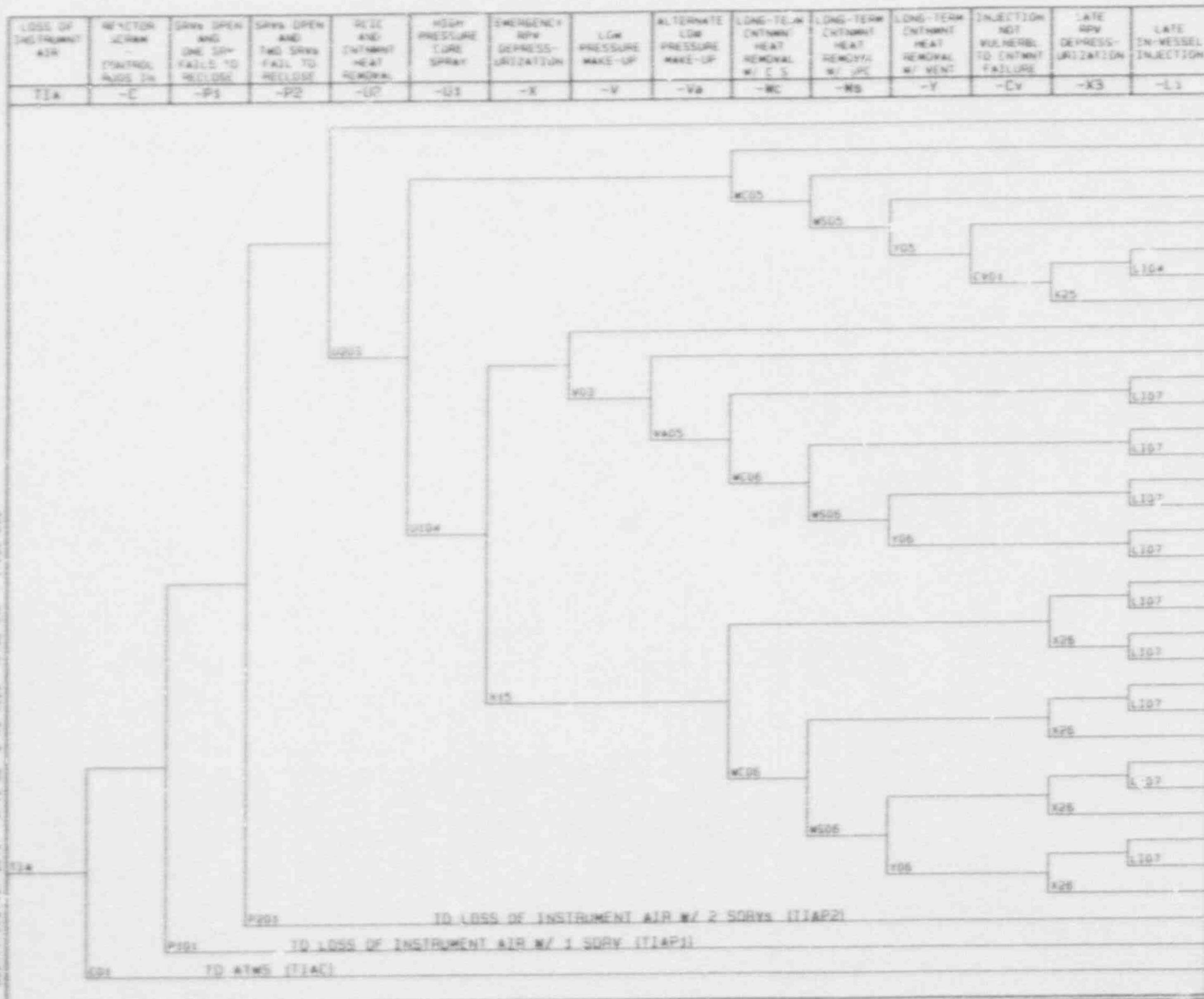
SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	BP2	OK	
P02	BP2-U1	1	1.57E-08
P03	BP2-U1-WC	3	5.13E-10
P04	BP2-U1-WC-Y2	7	0.00E+00
P05	BP2-U1-L1	2	0.00E+00
P06	BP2-U1-L1-WC	5	0.00E+00
P07	BP2-U1-L1-WC-Y2	7	0.00E+00
P08	BP2-U1-X3	DUM	0.00E+00
P09	BP2-U1-R3	13	1.14E-08
P10	BP2-U1-R3-Y2	17	1.67E-09
P11	BP2-U1-R3-I	44	0.00E+00
P12	BP2-U1-R3-L1	15	7.81E-10
P13	BP2-U1-R3-L1-Y2	19	0.00E+00
P14	BP2-U1-R3-L1-I	45	0.00E+00
P15	BP2-U1-R3-X3	DUM	0.00E+00
P16	BP2-U1-R3-X3-Y2	DUM	0.00E+00
P17	BP2-U1-R3-X3-I	DUM	0.00E+00
P18	BP2-U1-R3-X3-L1	DUM	0.00E+00
P19	BP2-U1-R3-X3-L1-Y2	DUM	0.00E+00
P20	BP2-U1-R3-X3-L1-I	DUM	0.00E+00

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-8 SHEET 16 OF 16
STATION BLACKOUT PDS EVENT TREE
REVISION 1 (03/18/92)



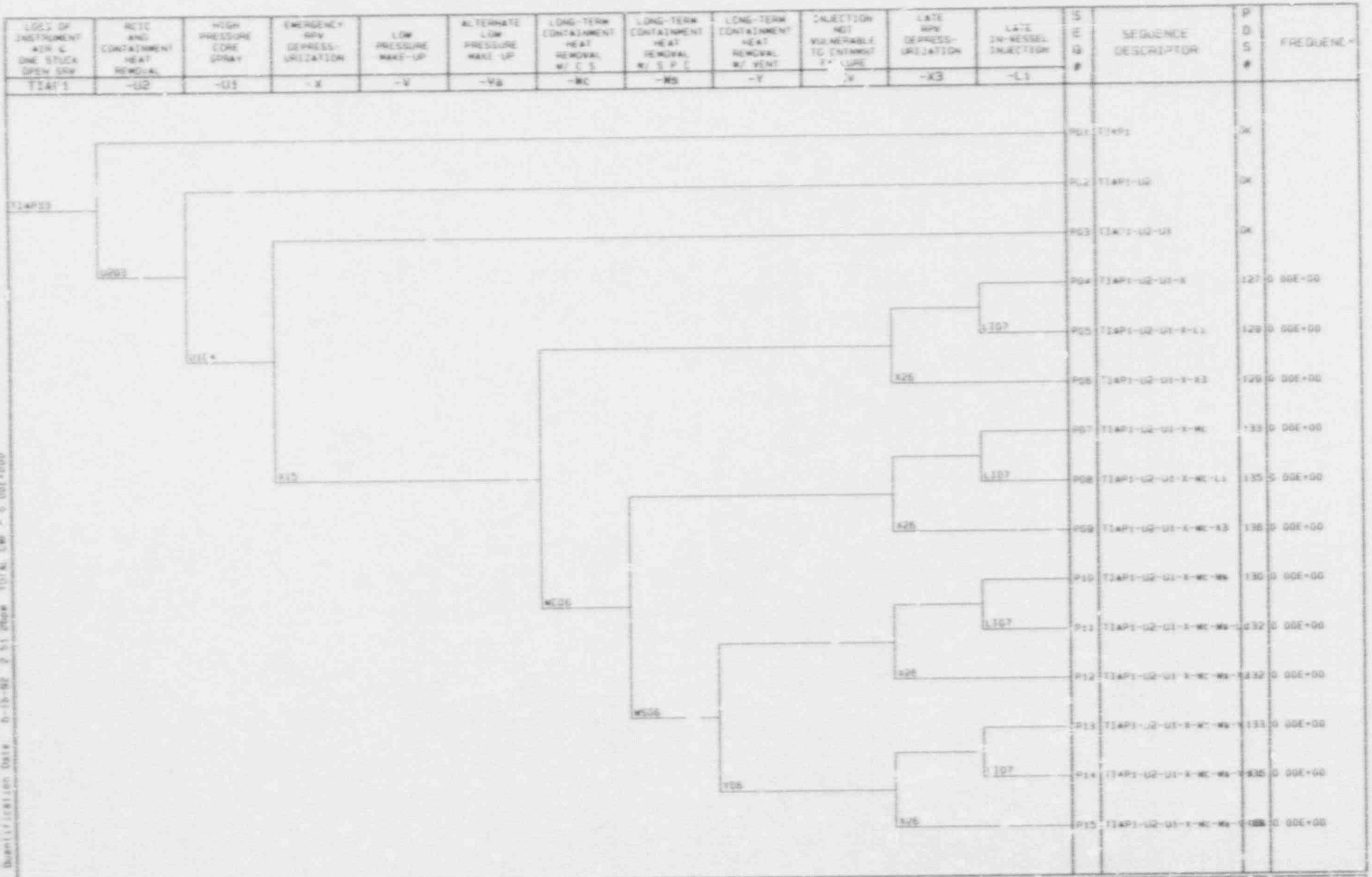
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-9
INDVRTNT OPN RELF VLV - PDS EVENT TREE
REVISION 1 (1/22/92)

D:\POS\ET\T1A.EVT 8:37 08am 5-17-92 NUPRA 2.0 (C)
Quantification Date 6-16-92 2:43 1/2m TOTAL CMF = 1.02E-006

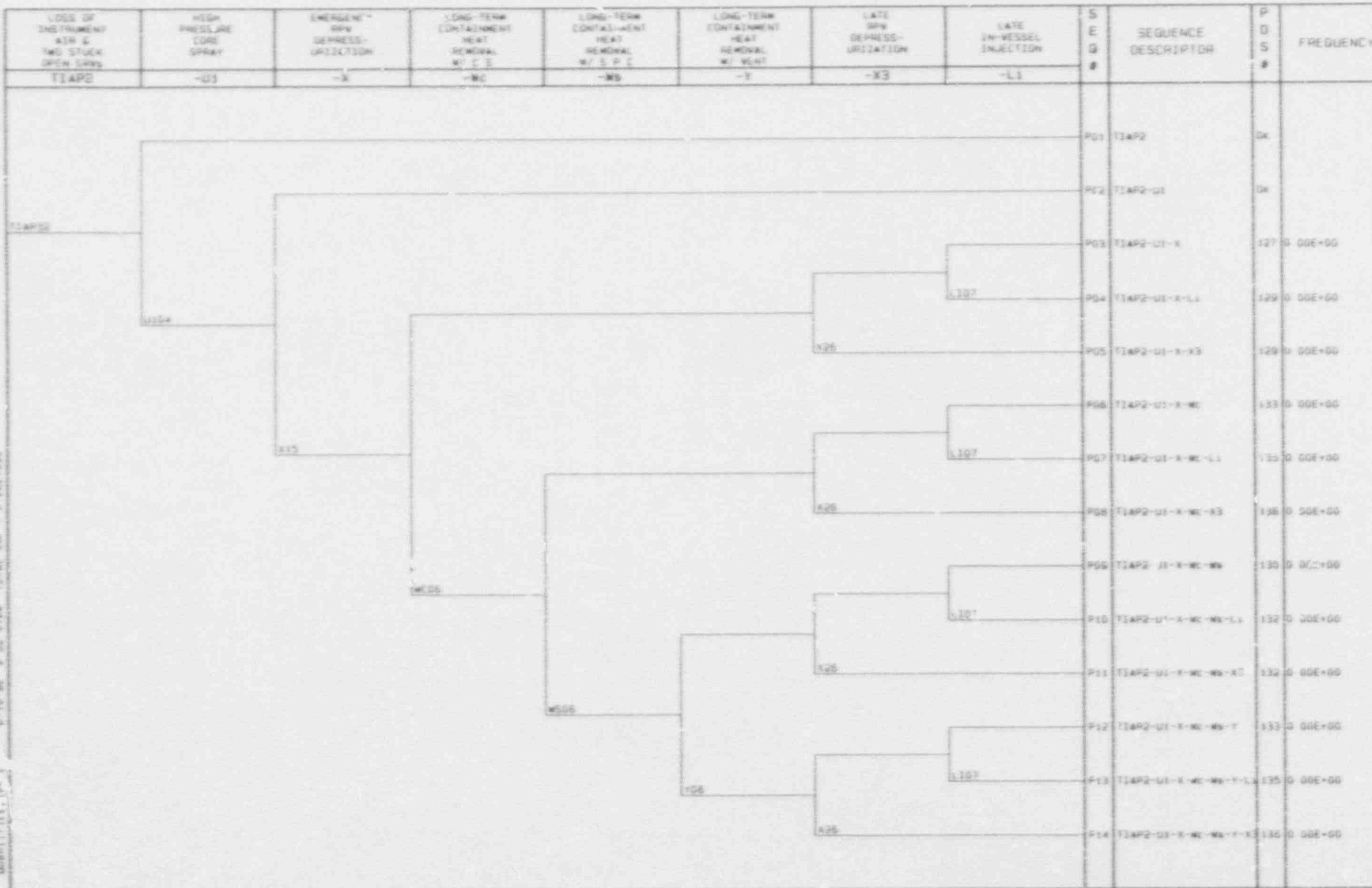


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T1A	OK	
P02	T1A-U2	OK	
P03	T1A-U2-WC	OK	
P04	T1A-U2-WC-WS	OK	
P05	T1A-U2-WC-WS-Y	OK	
P06	T1A-U2-WC-WS-Y-CV	145	1.91E-07
P07	T1A-U2-WC-WS-Y-CV-L1	147	8.51E-08
P08	T1A-U2-WC-WS-Y-CV-X3	148	0.00E+00
P09	T1A-U2-U1	OK	
P10	T1A-U2-U1-V	OK	
P11	T1A-U2-U1-V-VA	127	0.00E+00
P12	T1A-U2-U1-V-VA-L1	129	0.00E+00
P13	T1A-U2-U1-V-VA-WC	133	0.00E+00
P14	T1A-U2-U1-V-VA-WC-L1	135	0.00E+00
P15	T1A-U2-U1-V-VA-WC-WS	130	7.53E-07
P16	T1A-U2-U1-V-VA-WC-WS[illegible]	[illegible]	3.47E-08
P17	T1A-U2-U1-V-VA-WC-WS	133	0.00E+00
P18	T1A-U2-U1-V-VA-WC-WS[illegible]	[illegible]	0.00E+00
P19	T1A-U2-U1-X	127	0.00E+00
P20	T1A-U2-U1-X-L1	129	0.00E+00
P21	T1A-U2-U1-X-X3	128	3.00E-09
P22	T1A-U2-U1-X-X3-L1	129	0.00E+00
P23	T1A-U2-U1-X-WC	133	0.00E-10
P24	T1A-U2-U1-X-WC-L1	135	0.00E+00
P25	T1A-U2-U1-X-WC-X3	136	0.00E+00
P26	T1A-U2-U1-X-WC-WS	130	0.00E+00
P27	T1A-U2-U1-X-WC-WS-L1	132	0.00E+00
P28	T1A-U2-U1-X-WC-WS-X3	132	0.00E+00
P29	T1A-U2-U1-X-WC-WS-Y	133	0.00E+00
P30	T1A-U2-U1-X-WC-WS-Y-L[illegible]	[illegible]35	0.00E+00
P31	T1A-U2-U1-X-WC-WS-Y-X[illegible]	[illegible]37	0.00E+00
P32	T1A-P2	TFR	1.47E-04
P33	T1A-P1	TFR	1.47E-03
P34	T1A-C	TFR	9.20E-07

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-10 SHEET 1 OF 3
LOSS OF INST AIR POS EVENT TREE
REVISION 4 (03/23/92)



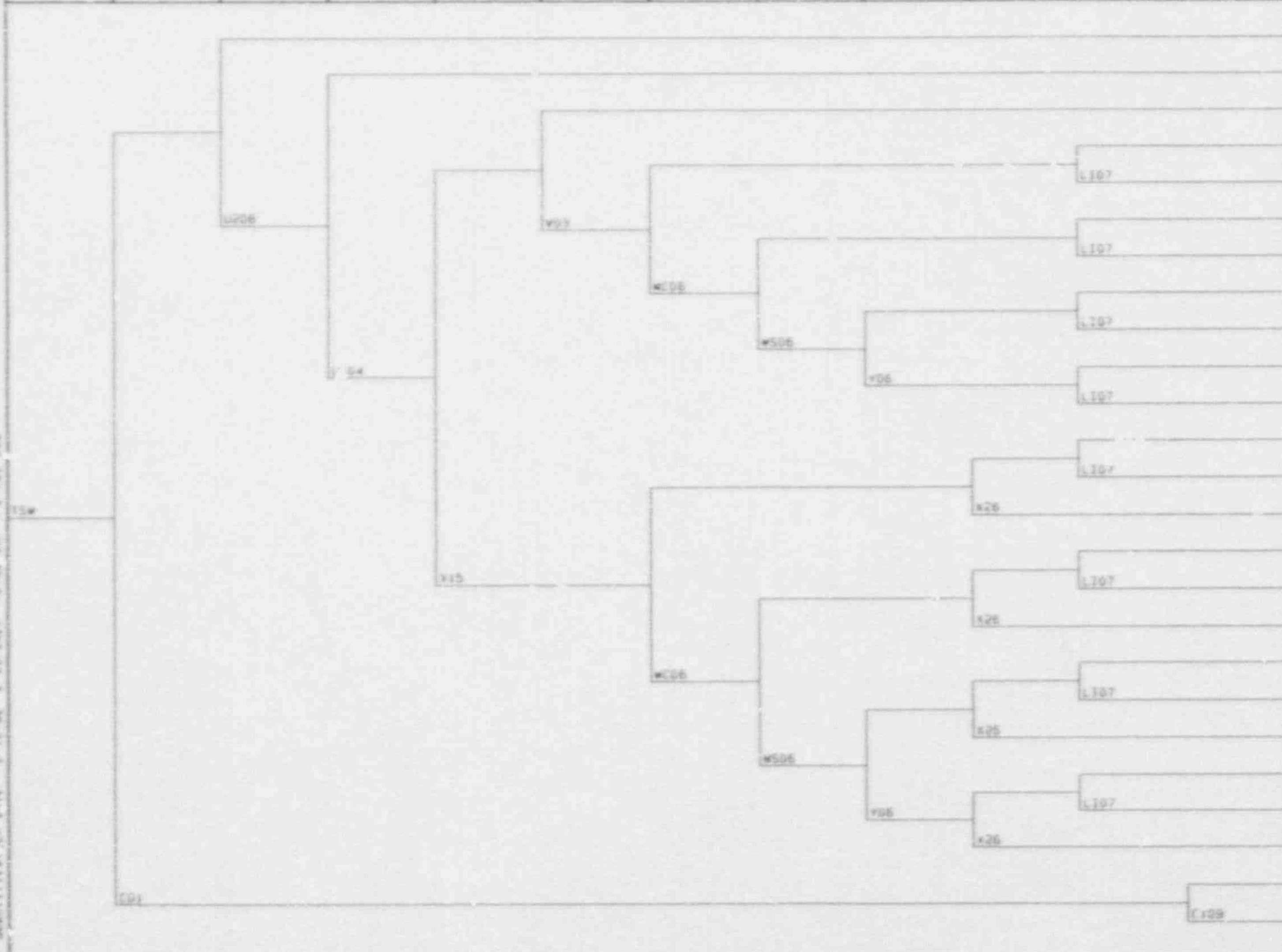
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-10 SHEET 2 OF 3
LOSS OF INST AIR PDS EVENT TREE
REVISION 3 (01/22/92)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-10 SHEET 3 OF 3
LOSS OF INST AIR PDS EVENT TREE
REVISION : (03/23/92)

L:\PDS\ET\TSW.EVT 2:53.06pm 6-16-92 NUPRA 2.0 (C)
Quantification Date 6-16-92 2:53.04pm TOTAL CMF = 5.49E-008

LOSS OF SERVICE WATER	REACTOR SCRAM - CONTROL RODS IN	RCIC AND CONTAINMENT HEAT REMOVAL	HIGH PRESSURE CORE SPRAY	EMERGENCY RPV DEPRESS-URIZATION	LOW PRESSURE MAKE-UP	LONG-TERM CONTAINMENT HEAT REMOVAL W/ C.S.	LONG-TERM CONTAINMENT HEAT REMOVAL W/ S.P.C.	LONG-TERM CONTAINMENT HEAT REMOVAL W/ VENT	LATE RPV DEPRESS-URIZATION	LATE IN-VESSEL INJECTION	
TSW	-C	-U2	-U1	-X	-V	-WC	-WS	-Y	-X3	-L1	-C1

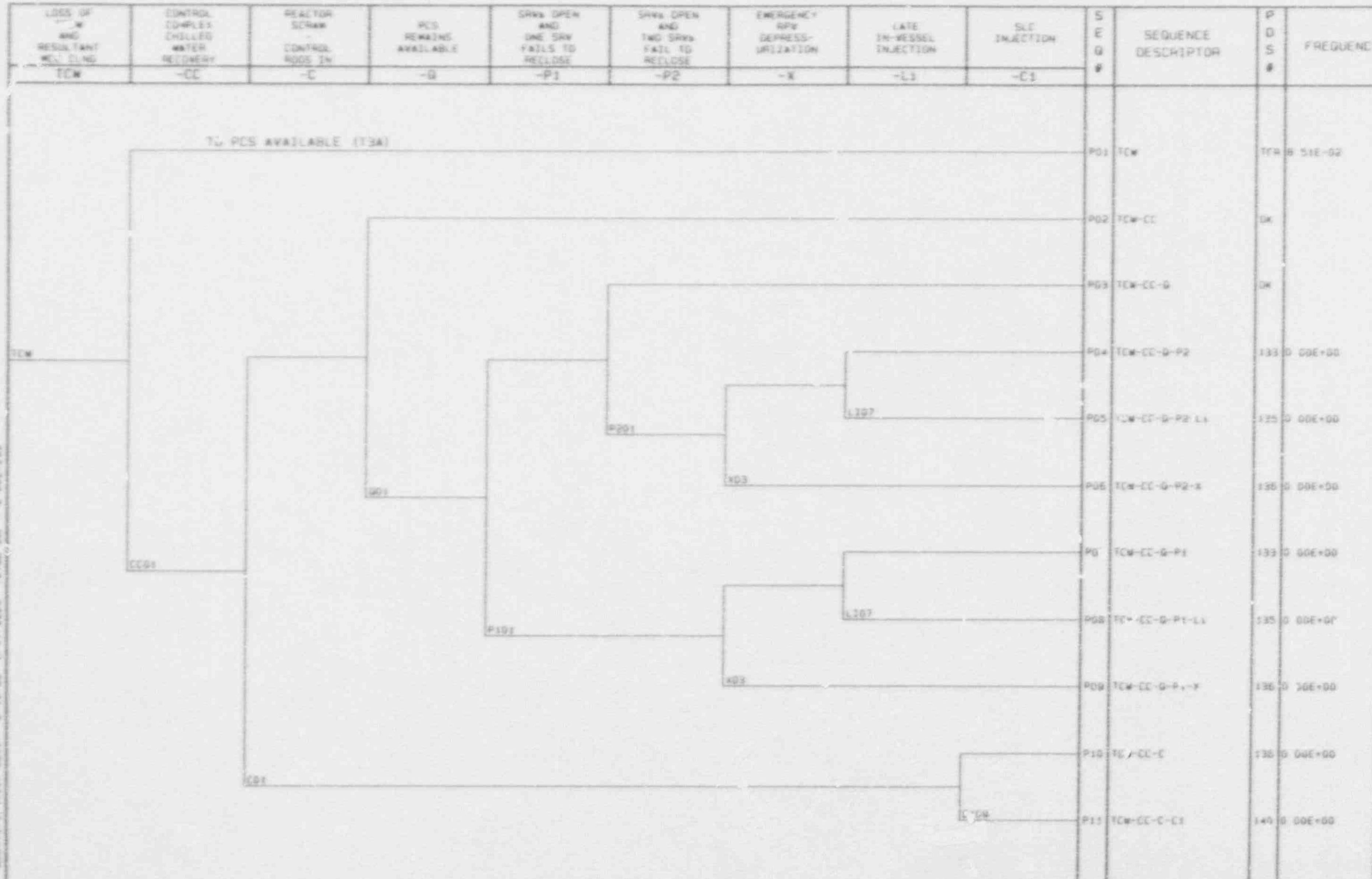


SEQ #	SEQUENCE DESCRIPTOR	POS #	FREQUENCY
P01	TSW	OK	
P02	TSW-U2	OK	
P03	TSW-U2-U1	OK	
P04	TSW-U2-U1-V	127	0.00E+00
P05	TSW-U2-U1-V-L1	129	0.00E+00
P06	TSW-U2-U1-V-WC	133	0.00E+00
P07	TSW-U2-U1-V-WC-L1	135	0.00E+00
P08	TSW-U2-U1-V-WC-WS	130	5.49E-08
P09	TSW-U2-U1-V-WC-WS-L1	132	0.00E+00
P10	TSW-U2-U1-V-WC-WS-Y	133	0.00E+00
P11	TSW-U2-U1-V-WC-WS-Y-L1	135	0.00E+00
P12	TSW-U2-U1-X	127	0.00E+00
P13	TSW-U2-U1-X-L1	129	0.00E+00
P14	TSW-U2-U1-X-X3	129	0.00E+00
P15	TSW-U2-U1-X-WC	133	0.00E+00
P16	TSW-U2-U1-X-WC-L1	135	0.00E+00
P17	TSW-U2-U1-X-WC-X3	136	0.00E+00
P18	TSW-U2-U1-X-WC-WS	130	0.00E+00
P19	TSW-U2-U1-X-WC-WS-L1	132	0.00E+00
P20	TSW-U2-U1-X-WC-WS-X3	132	0.00E+00
P21	TSW-U2-U1-X-WC-WS-Y	133	0.00E+00
P22	TSW-U2-U1-X-WC-WS-Y-L1	135	0.00E+00
P23	TSW-U2-U1-X-WC-WS-Y-X3	136	0.00E+00
P24	TSW-C	136	1.00E-[illegible]
P25	TSW-C-C1	140	0.00E+00

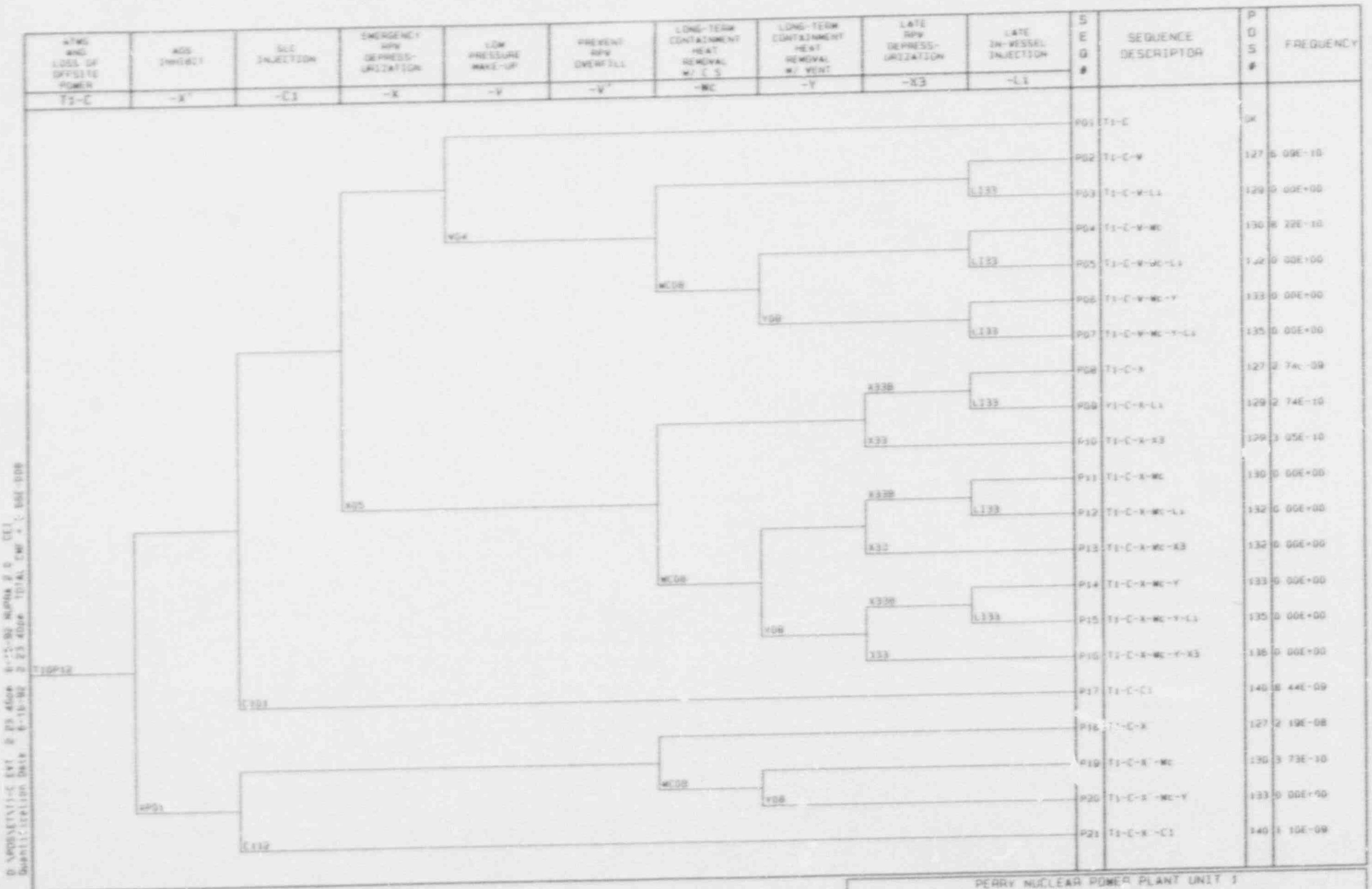
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-11
LOSS OF SERVICE WATER - POS EVENT TREE
REVISION (03/23/92)

D \PDS\ET\TCW.EXT 9 22 14am 6-17-92 NUPRA 2.0 CEI
Quantification Date 6-16-92 2 47 22pm TOTAL CMF = 0 00E+000

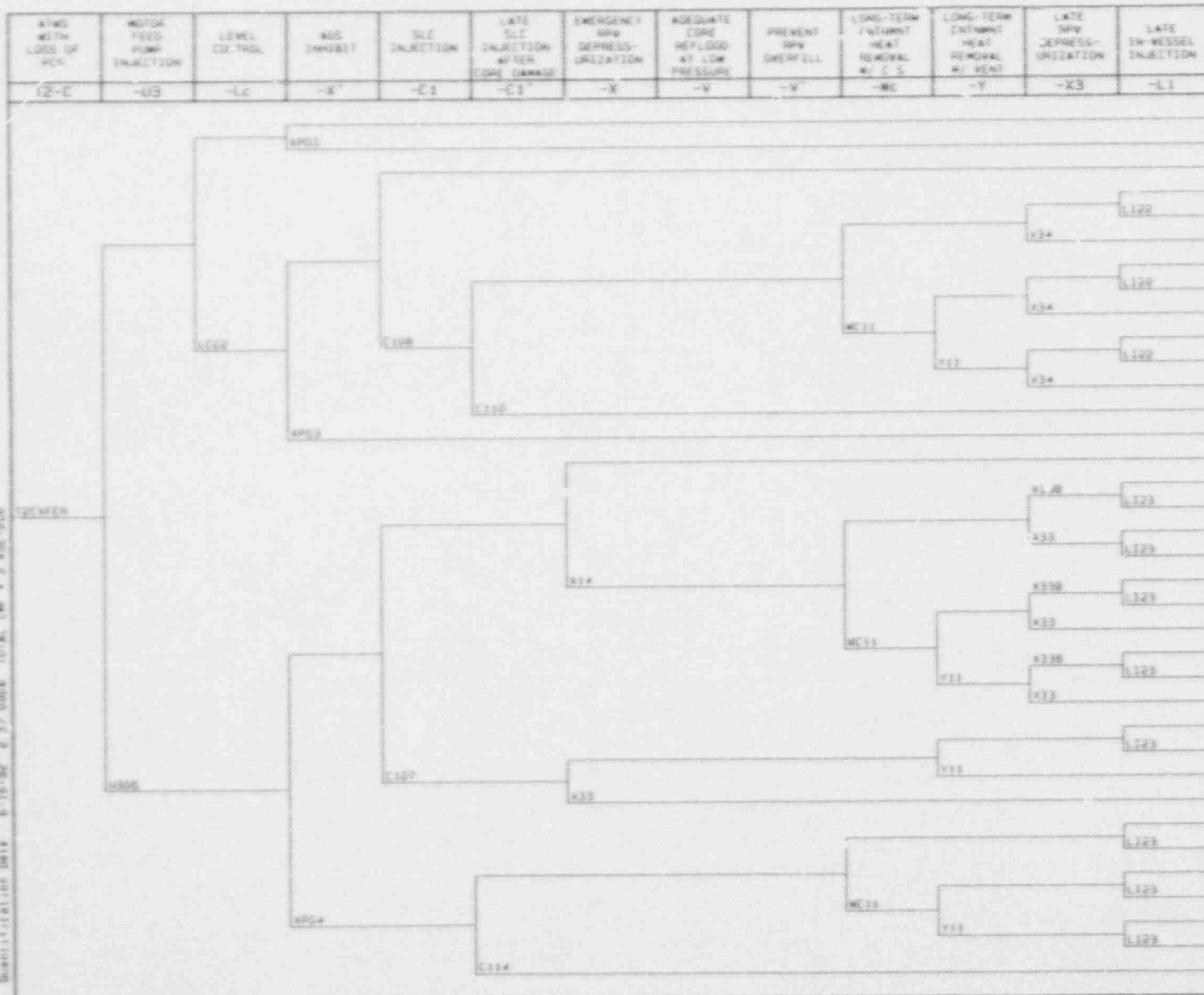


PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-12
LOSS OF CCCW - PDS EVENT TREE
REVISION 2 (03/20/92)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-13
ATWS AND LOOP PDS EVENT TREE
REVISION 2 (05/29/92)

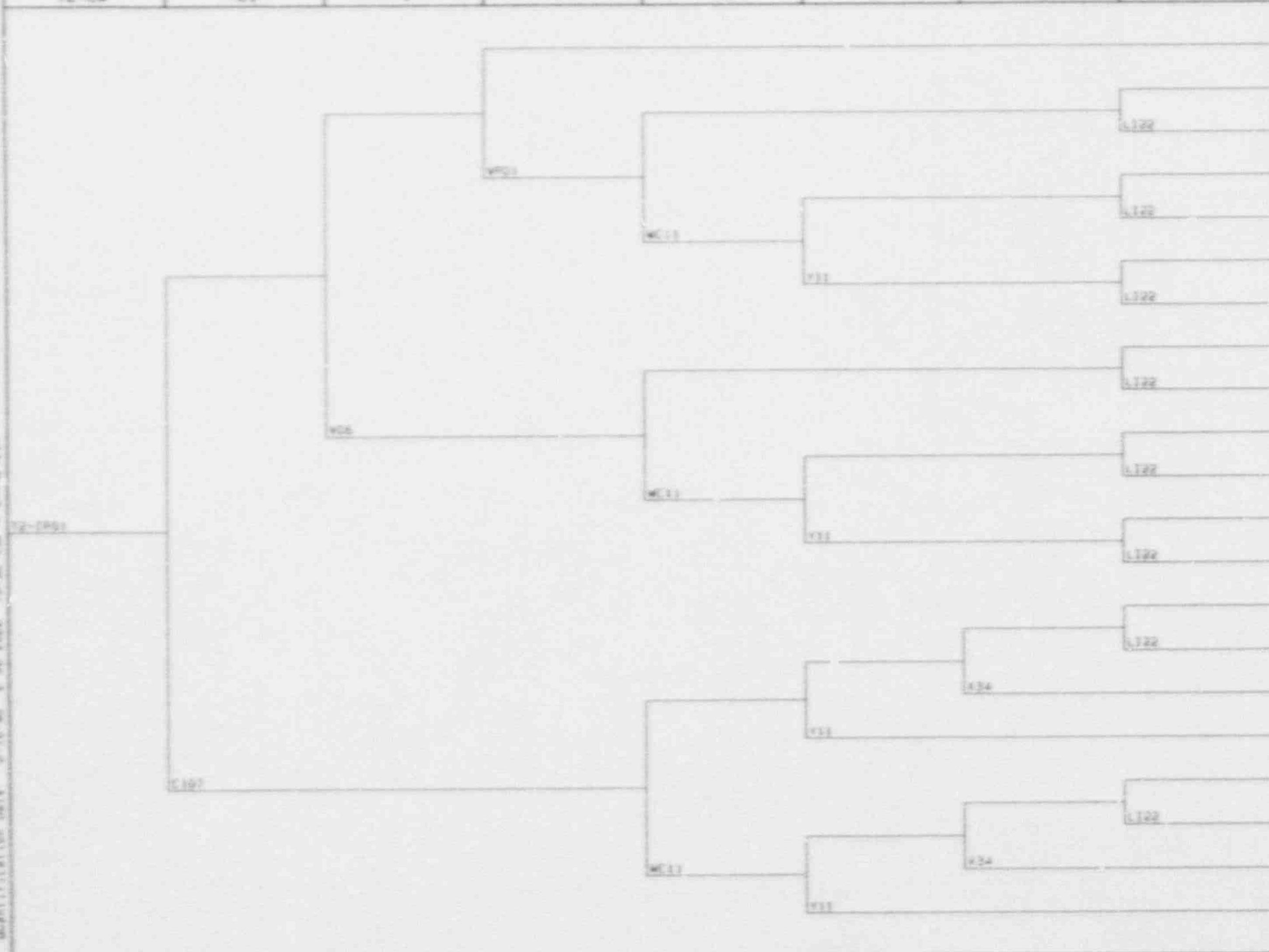


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T2-C	TFR	6.23E-05
P02	T2-C-X	TFR	2.36E-07
P03	T2-C-Lc	OK	
P04	T2-C-Lc-C1	127	6.23E-07
P05	T2-C-Lc-C1-Li	129	0.00E+00
P06	T2-C-Lc-C1-X3	129	0.00E+00
P07	T2-C-Lc-C1-Wc	130	1.67E-08
P08	T2-C-Lc-C1-Wc-Li	132	0.00E+00
P09	T2-C-Lc-C1-Wc-X3	132	0.[illegible]E+00
P10	T2-C-Lc-C1-Wc-Y	133	0.00E+00
P11	T2-C-Lc-C1-Wc-Y-Li	135	0.00E+00
P12	T2-C-Lc-C1-Wc-Y-X3	136	0.00E+00
P13	T2-C-Lc-C1-C1'	140	6.27E-08
P14	T2-C-Lc-X	139	2.32E-09
P15	T2-C-U3	OK	
P16	T2-C-U3-X	127	2.81E-07
P17	T2-C-U3-X-Li	129	0.00E+00
P18	T2-C-U3-X-X3	128	3.11E-08
P19	T2-C-U3-X-X3-Li	129	0.00E+00
P20	T2-C-U3-X-Wc	130	4.89E-09
P21	T2-C-U3-X-Wc-Li	132	0.00E+00
P22	T2-C-U3-X-Wc-X3	132	5.21E-10
P23	T2-C-U3-X-Wc-Y	133	0.00E+00
P24	T2-C-U3-X-Wc-Y-Li	135	0.00E+00
P25	T2-C-U3-X-Wc-Y-X3	136	0.00E+00
P26	T2-C-U3-C1	139	9.32E-10
P27	T2-C-U3-C1-Li	139	0.00E+00
P28	T2-C-U3-C1-Y	139	0.00E+[illegible]
P29	T2-C-U3-C1-X	140	0.00E+00
P30	T2-C-U3-X'	127	2.27E-08
P31	T2-C-U3-X'-Li	129	0.00E+00
P32	T2-C-U3-X'-Wc	130	3.99E-08
P33	T2-C-U3-X'-Wc-Li	132	0.00E+00
P34	T2-C-U3-X'-Wc-Y	133	1.50E-10
P35	T2-C-U3-X'-Wc-Y-Li	135	0.00E+00
P36	T2-C-U3-X'-C1	139	1.20E-07

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-14 SHEET 1 OF 3
ATWS & LOSS OF PCS PDS ET
REVISION 7 (05/29/92)

D:\PDS\ET\T2-CA.EVT 2:38 28pm 5-15-92 GoPRA 2.0 [E]
Quantification Date 5-15-92 2:38 28pm TOTAL CMF = 5.7RE-007

ATWS WITH LOSS OF PCS & ADS INHIBIT	SLC INJECTION	ADEQUATE CORE REFLOOD AT LOW PRESSURE	PREVENT RPV OVERFILL	LONG-TERM CONTAINMENT HEAT REMOVAL W/ C S	LONG-TERM CONTAINMENT HEAT REMOVAL W/ VENT	LATE RPV DEPRESS-URIZATION	LATE IN-VESSEL INJECTION
T2-CA	-C1	-V	-V'	-WC	-Y	-X3	-L1

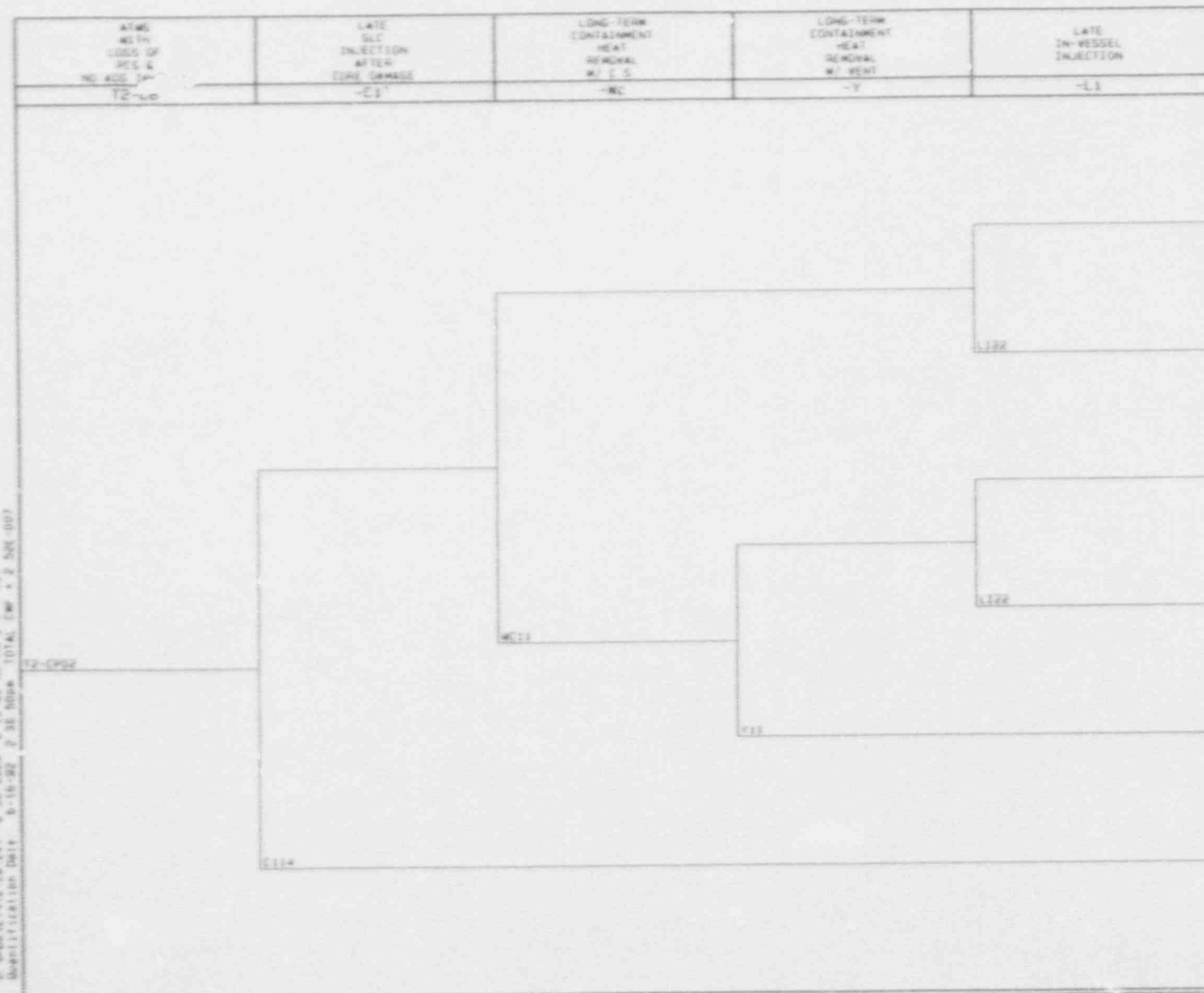


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T2-CA	OK	
P02	T2-CA-V	127	1.25E-07
P03	T2-CA-V -L1	129	0.00E+00
P04	T2-CA-V -WC	130	2.09E-09
P05	T2-CA-V -WC-L1	132	0.00E+00
P06	T2-CA-V -WC-Y	133	0.00E+00
P07	T2-CA-V -WC-Y-L1	135	0.00E+00
P08	T2-CA-V	127	7.46E-08
P09	T2-CA-V-L1	129	0.00E+00
P10	T2-CA-V-WC	130	8.40E-08
P11	T2-CA-V-WC-L1	132	0.00E+00
P12	T2-CA-V-WC-Y	137	0.00E+00
P13	T2-CA-V-WC-Y-L1	135	0.00E+00
P14	T2-CA-C1	137	1.53E-07
P15	T2-CA-C1-L1	137	0.00E+00
P16	T2-CA-C1-X3	138	0.00E+00
P17	T2-CA-C1-Y	138	0.00E+00
P18	T2-CA-C1-WC	139	1.38E-07
P19	T2-CA-C1-WC-L1	139	0.00E+00
P20	T2-CA-C1-WC-X3	140	0.00E+00
P21	T2-CA-C1-WC-Y	145	0.00E+00

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-14 SHEET 2 OF 3
ATWS & LOSS OF PCS PDS ET
REVISION 1 (05/30/92)

D:\PDS\E1\T2-CB.EVT 2:38:56pm 6-16-92 NUPRA 2.0 CE1
Quantification Date: 6-16-92 2:38:50pm TOTAL CMF = 2.52E-007

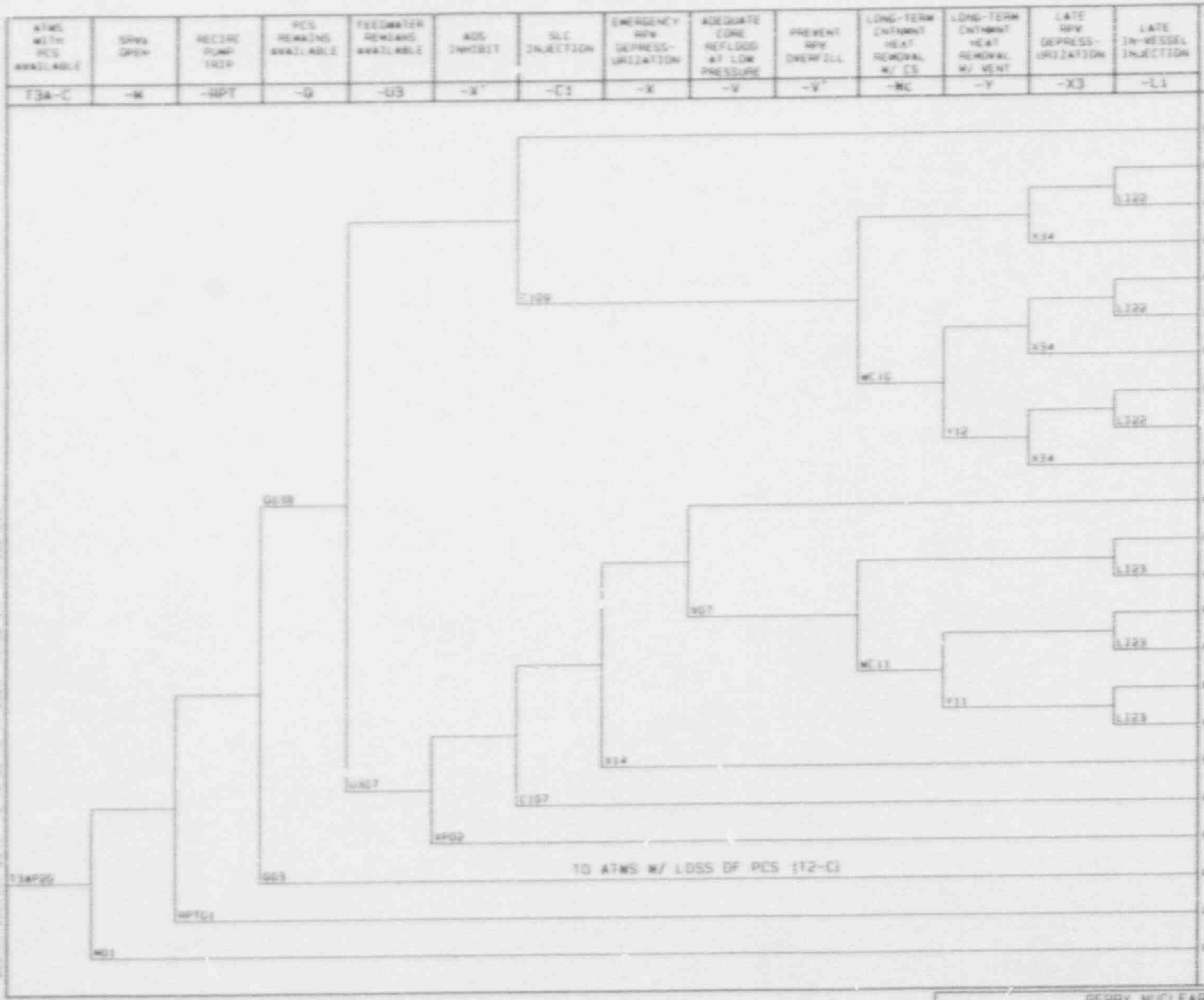


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T2-CB	127	2.36E-07
P02	T2-CB-L1	129	0.00E+00
P03	T2-CB-WC	130	3.95E-09
P04	T2-CB-WC-L1	132	0.00E+00
P05	T2-CB-WC-Y	135	0.00E+00
P06	T2-CB-C1	139	1.20E-08

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-14 SHEET 3 OF 3
ATWS & LOSS OF PCS POS CT
REVISION 2 (05/30/92)

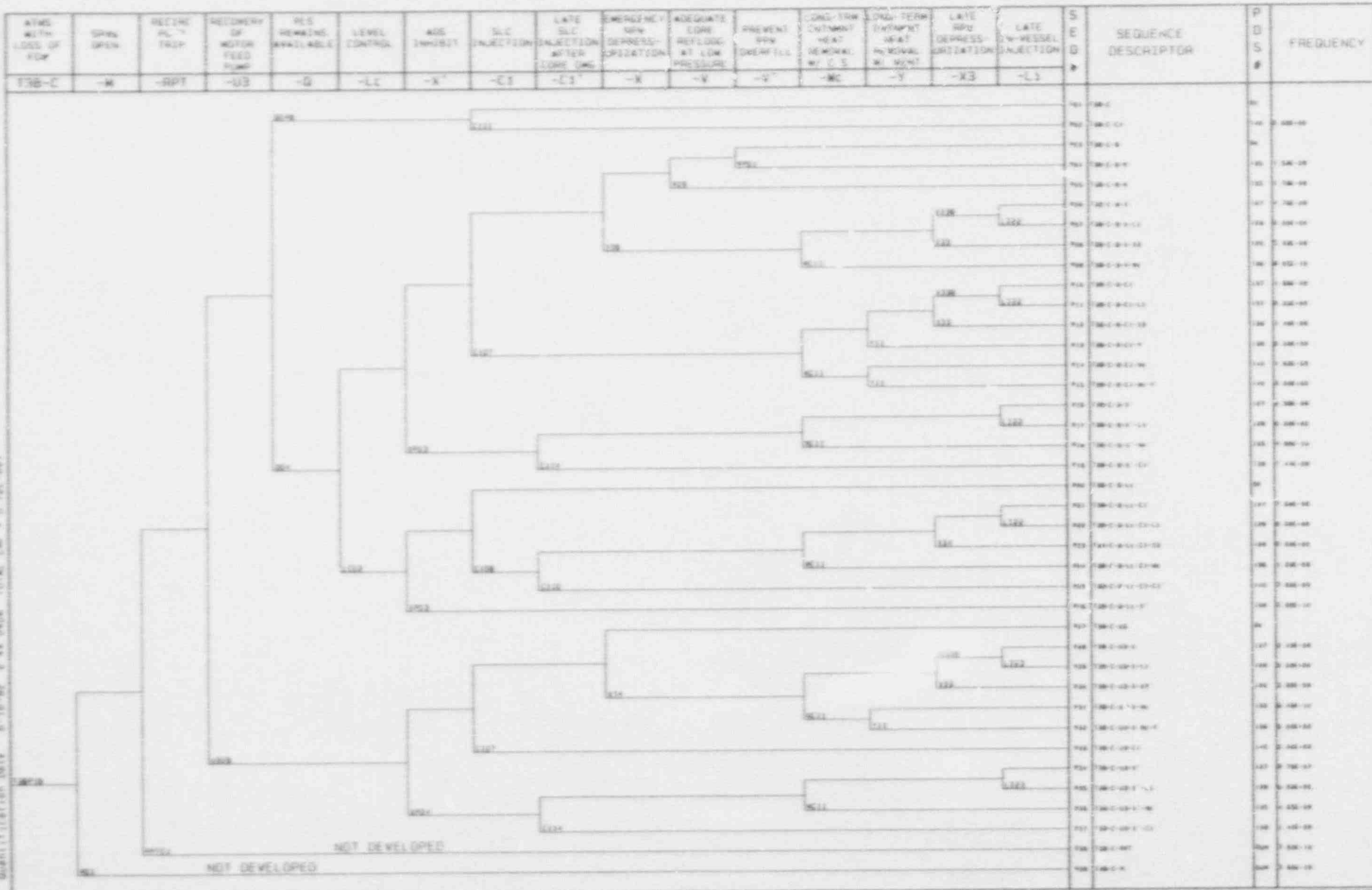
D:\PDS\ET\T3A-C.EVT 2:41 24pm 6-16-92 NUPRA 2.0 (E)
Quantification Date 6-16-92 2:41 22pm TOTAL CMF = 9.02E-009



S-Q #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T3A-C	OK	
P02	T3A-C-C1	137	0.00E+00
P03	T3A-C-C1-L1	137	0.00E+00
P04	T3A-C-C1-X3	138	0.00E+00
P05	T3A-C-C1-WC	139	0.00E+00
P06	T3A-C-C1-WC-L1	139	0.00E+00
P07	T3A-C-C1-WC-X3	140	0.00E+00
P08	T3A-C-C1-WC-Y	139	0.00E+00
P09	T3A-C-C1-WC-Y-L1	139	0.00E+00
P10	T3A-C-C1-WC-Y-X3	140	0.00E+00
P11	T3A-C-U3	OK	
P12	T3A-C-U3-V	127	0.00E+00
P13	T3A-C-U3-V-L1	129	0.00E+00
P14	T3A-C-U3-V-WC	130	0.00E+00
P15	T3A-C-U3-V-WC-L1	132	0.00E+00
P16	T3A-C-U3-V-WC-Y	133	0.00E+00
P17	T3A-C-U3-V-WC-Y-L1	135	0.00E+00
P18	T3A-C-U3-X	136	0.00E+00
P19	T3A-C-U3-C1	140	0.00E+00
P20	T3A-C-U3-X'	139	0.00E+00
P21	T3A-C-Q	TFR	4.60E-05
P22	T3A-C-RPT	DUM	4.51E-09
P23	T3A-C-W	DUM	4.51E-09

PERRY NUCLEAR POWERPLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.4-15
ATWS W/ PCS AVAILABLE PDS EVENT TREE
REVISION 3 (04/20/92)

D:\PDS\ET\T3B-C EVT 2:45 08pm 6-16-92 NUPRA 2.0 CEI
Quantification Date 6-16-92 2:44 54pm TOTAL CMF = 5.72E-007

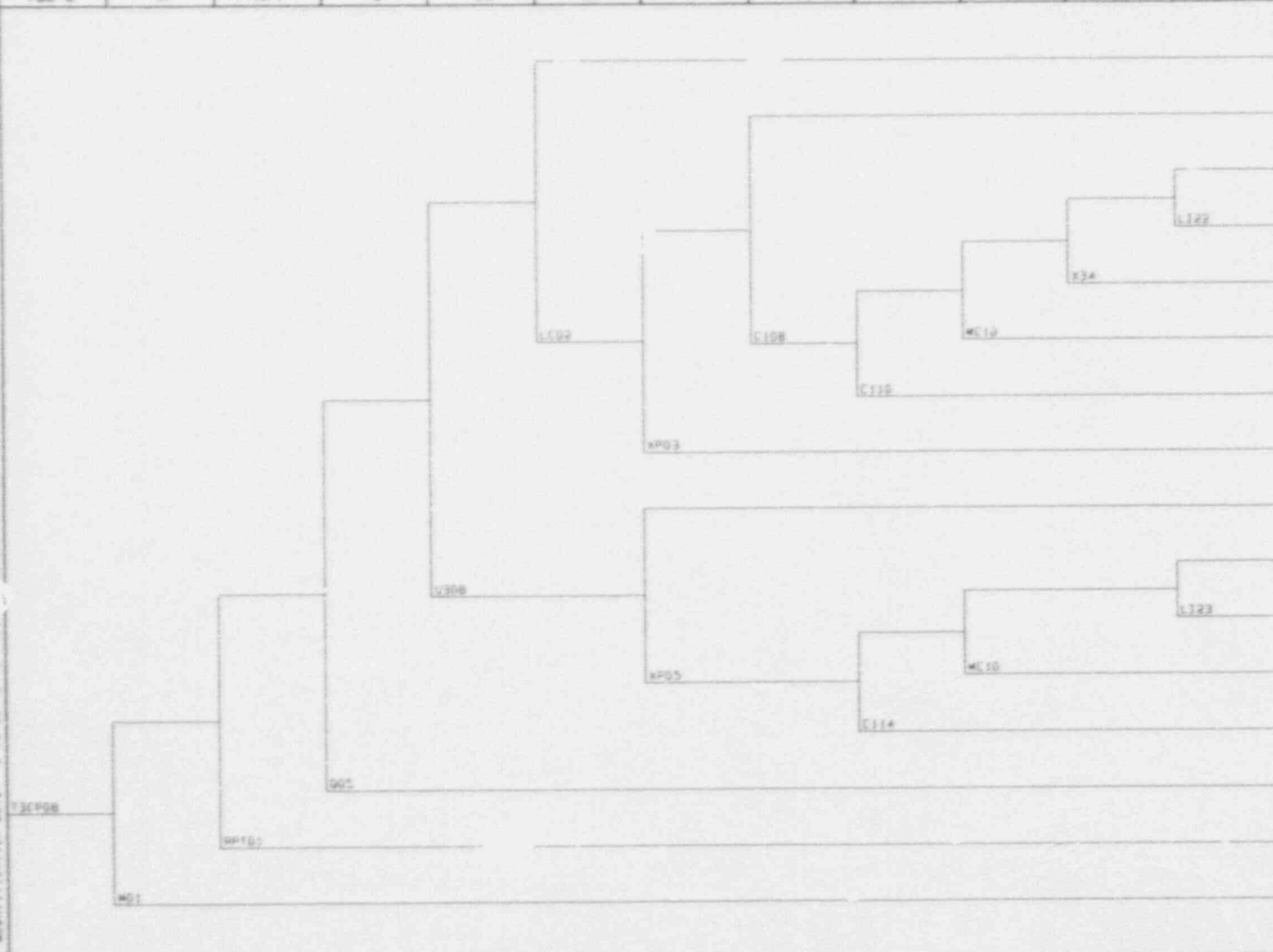


PERRY NUCLEAR POWERPLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-16
ATWS W/ LOSS OF FEEDWATER PDS ET
REVISION 5 (05/30/92)

D \PDS\E\T3C-C.EVT 2.46.48pm 6-16-92 N... 2.0 TEI
Quantification Date 6-16-92 2.46.47pm ...L CMF = 7.42E-008

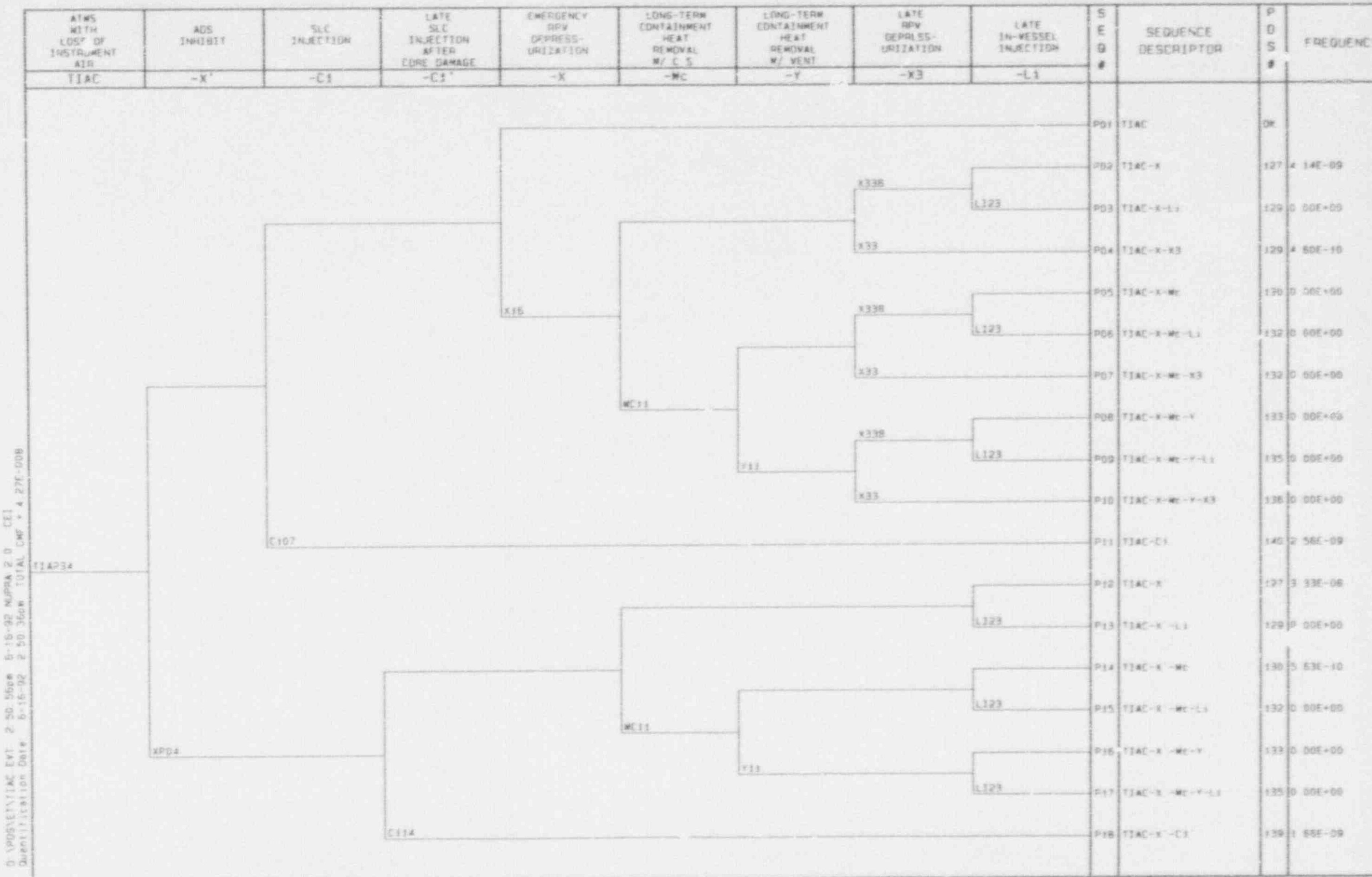
ATWS WITH IORV	SRVs OPEN	RECIRC PUMP TRIP	PCS REMAINS AVAILABLE - MSIVs OPEN	FEEDWATER REMAINS AVAILABLE	LEVEL CONTROL	ADS INHIBIT	SLC INJECTION	LATE SLC INJECTION AFTER CORE DAMAGE	LONG-TERM CONTAINMENT HEAT REMOVAL W/ C.S	LATE RPV DEPRESS-URIZATION	LATE IN-VESSEL INJECTION
T3C-C	-W	-RPT	-Q	-U3	-Lc	-X'	-C1	-C1'	-Wc	-X3	-L1



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T3C-C	OK	
P02	T3C-C-Lc	OK	
P03	T3C-C-Lc-C1	127	1.40E-08
P04	T3C-C-Lc-C1-L1	129	0.00E+00
P05	T3C-C-Lc-C1-X3	129	0.00E+00
P06	T3C-C-Lc-C1-Wc	136	2.38E-10
P07	T3C-C-Lc-C1-C1'	140	1.40E-09
P08	T3C-C-Lc-X'	139	0.00E+00
P09	T3C-C-U3	OK	
P10	T3C-C-U3-X'	127	5.49E-08
P11	T3C-C-U3-X'-L1	129	0.00E+00
P12	T3C-C-U3-X'-Wc	135	8.57E-10
P13	T3C-C-U3-X'-C1'	139	2.52E-09
P14	T3C-C-Q	TFR	1.40E-07
P15	T3C-C-RPT	DUM	1.40E-10
P16	T3C-C-W	DUM	1.40E-10

PERRY NUCLEAR POWERPLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-17
ATWS W/ IOR PDS EVENT TREE
REVISION 1 (04/20/92)



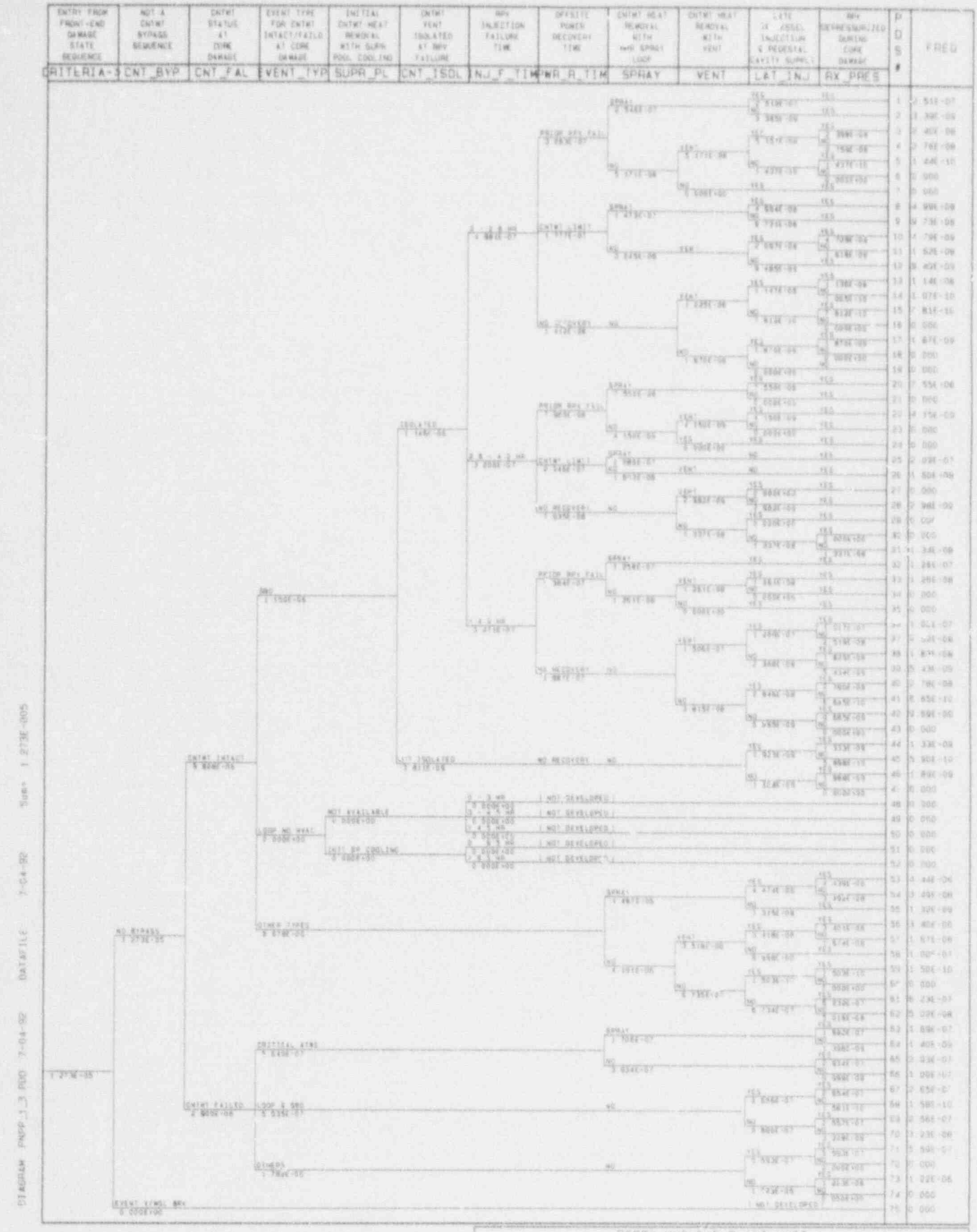
PERRY NUCLEAR POWERPLANT UNIT 1

INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-18

ATWS & LOSS OF IA PDS EVENT TREE

REVISION 3 (05/30/92)



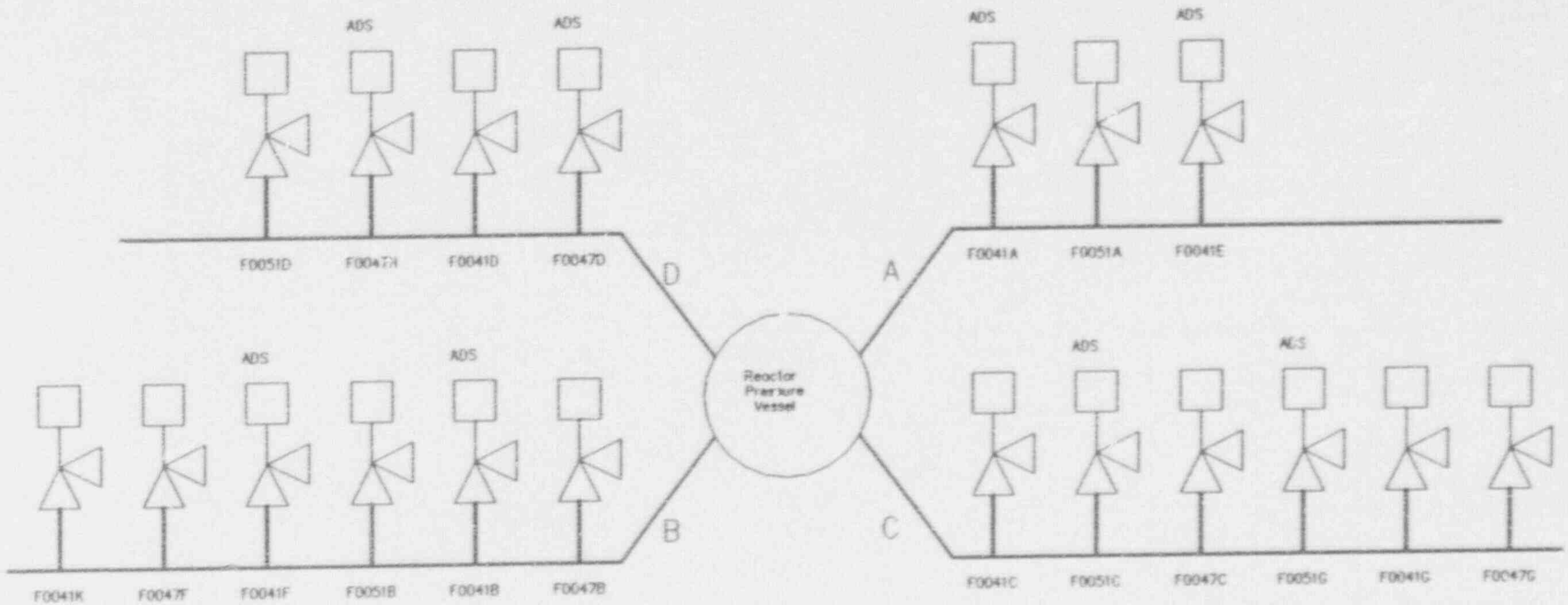
PERRY NUCLEAR PLANT UNIT 1

INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.4-19

PLANT DAMAGE STATE GROUPING LOGIC

REVISION 0

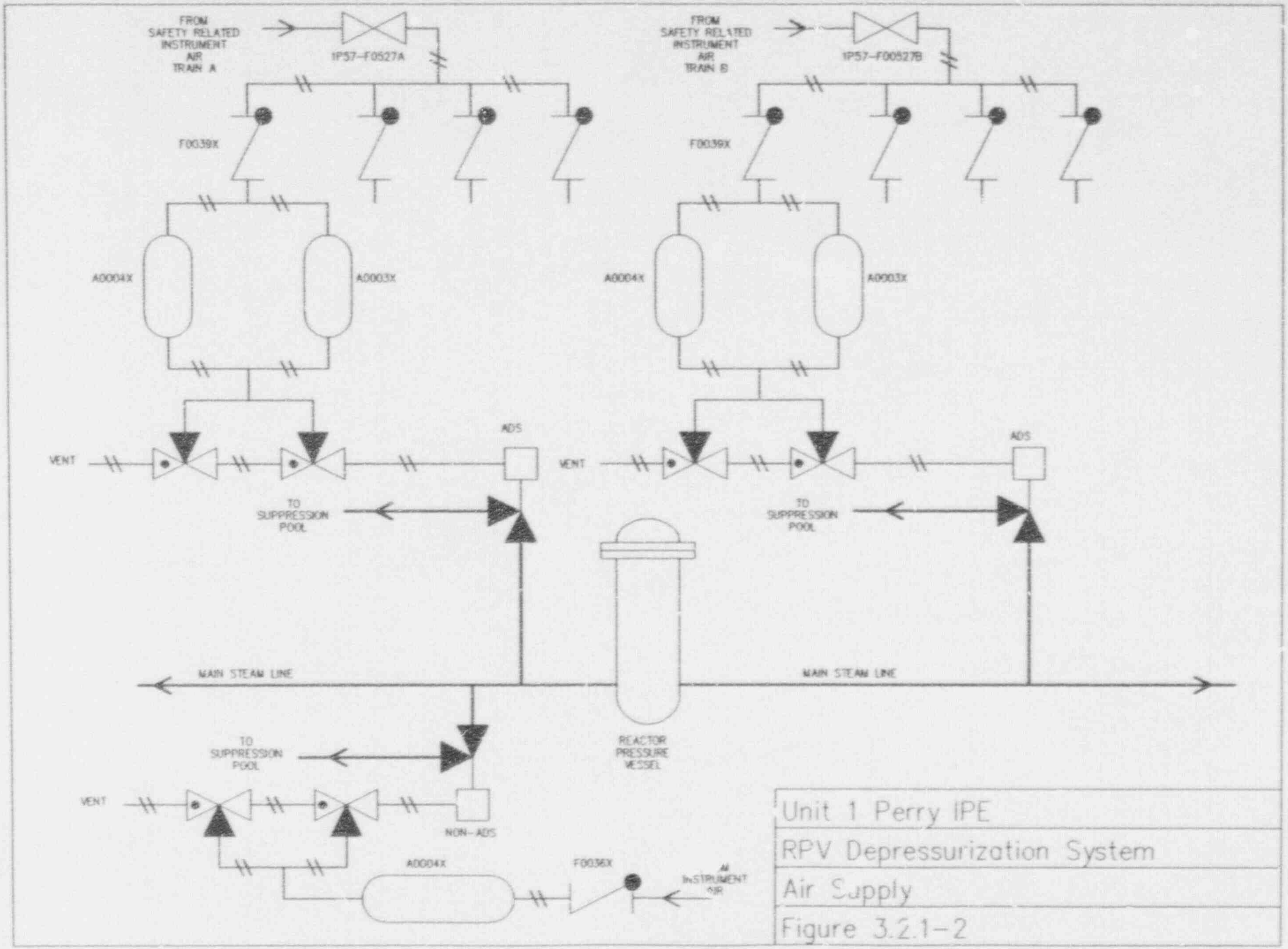


Perry Unit 1 IPE

RPV Depressurization System

ADS/SRV Orientation

Figure 3.2.1-1

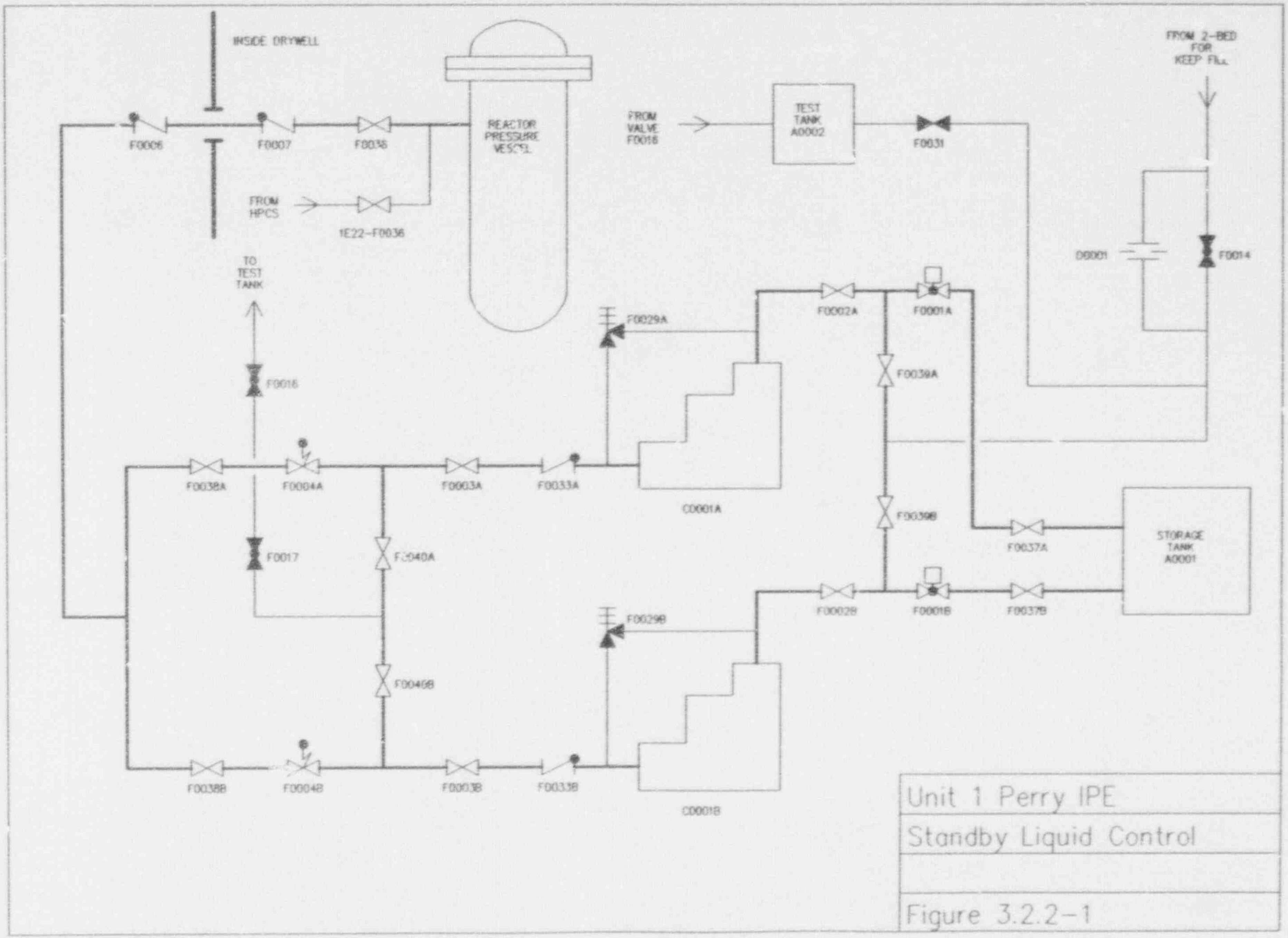


Unit 1 Perry IPE

RPV Depressurization System

Air Supply

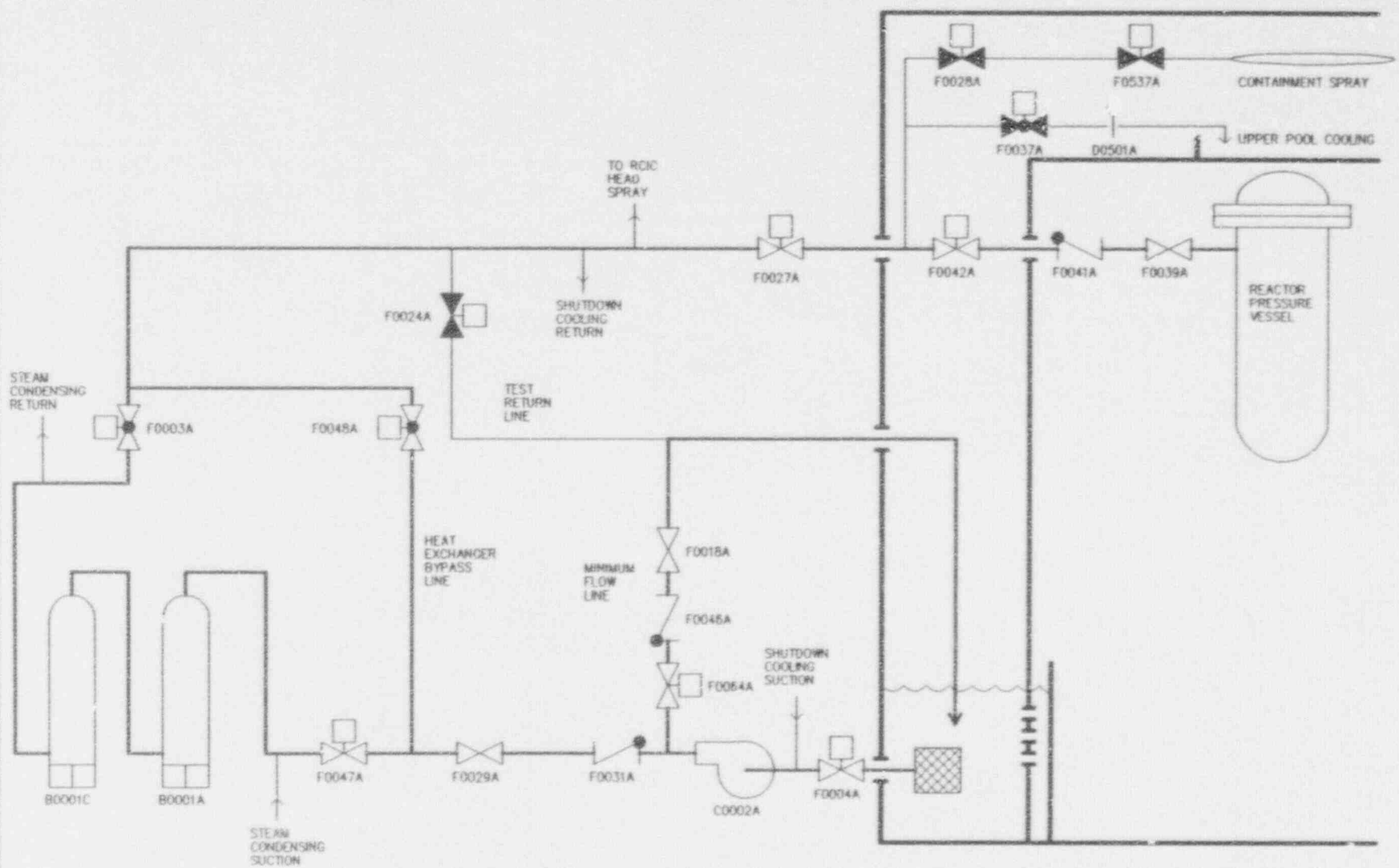
Figure 3.2.1-2



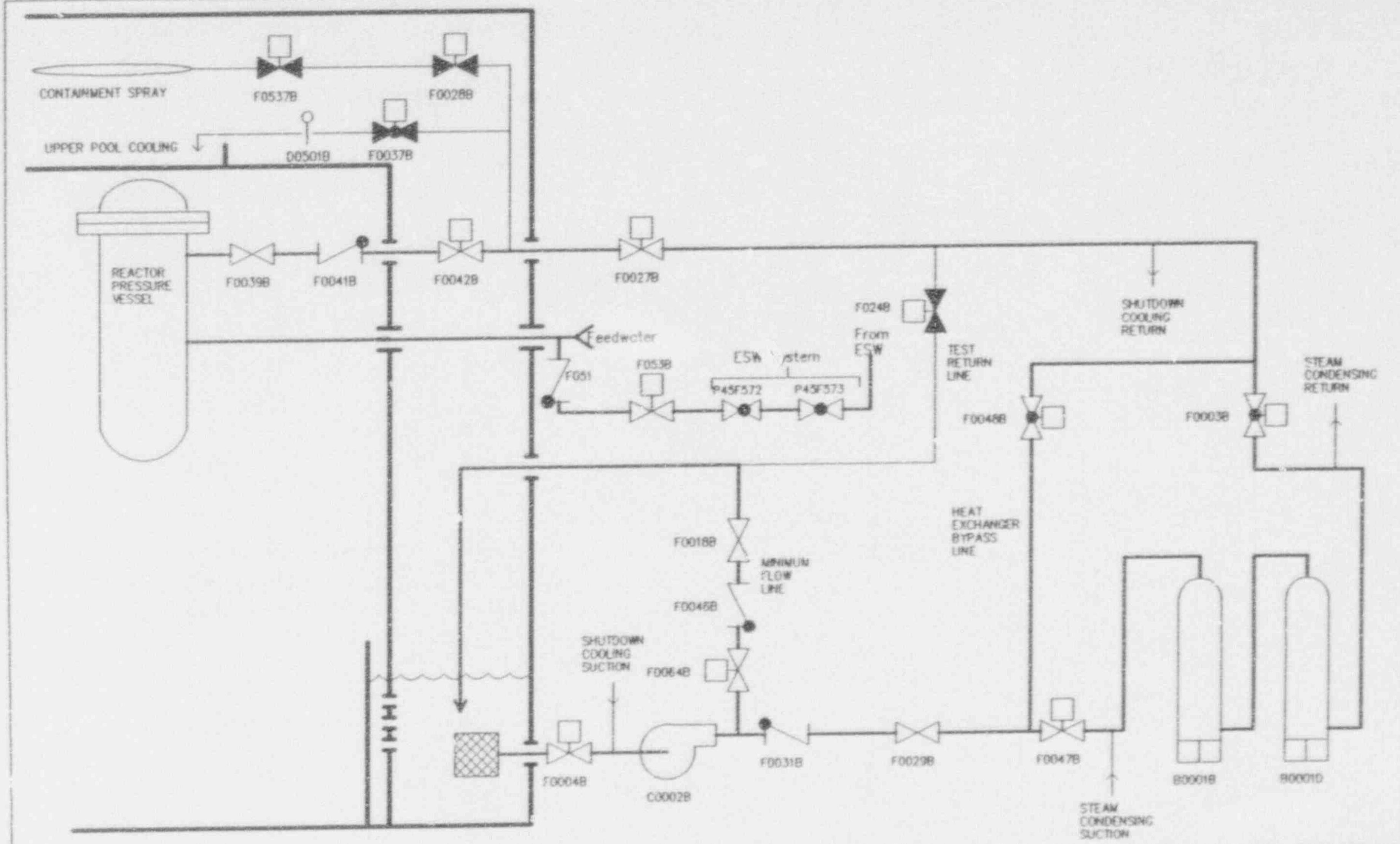
Unit 1 Perry IPE

Standby Liquid Control

Figure 3.2.2-1



Unit 1 Perry IPE
RHR – Low Press Clnt Injection
Train A
Figure 3.2.3-1

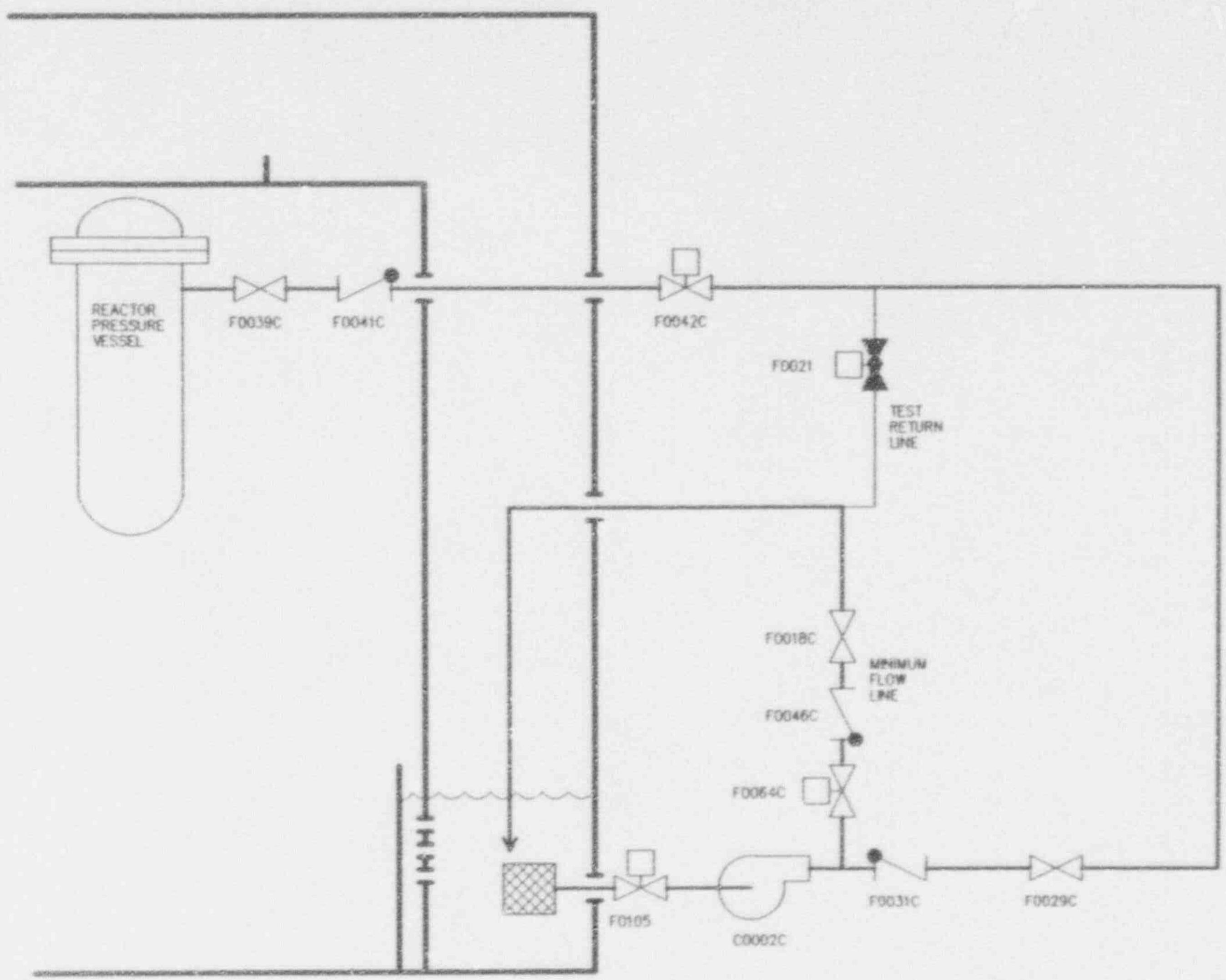


Unit 1 Perry IPE

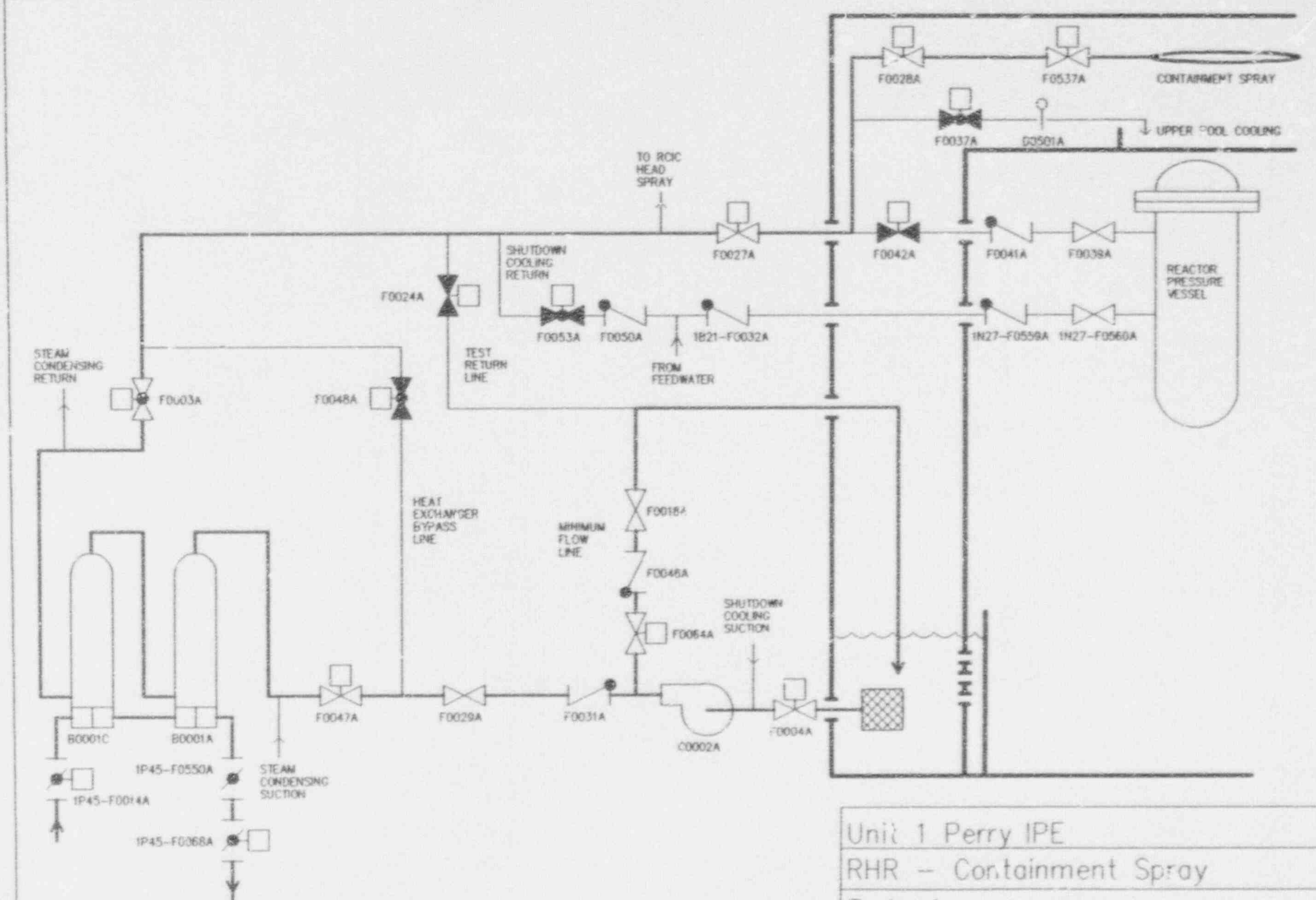
RHR – Low Press Clnt Injection

Train B

Figure 3.2.3-2



Unit 1 Perry IPE
RHR – Low Press Clnt Injection
Train C
Figure 3.2.3-3

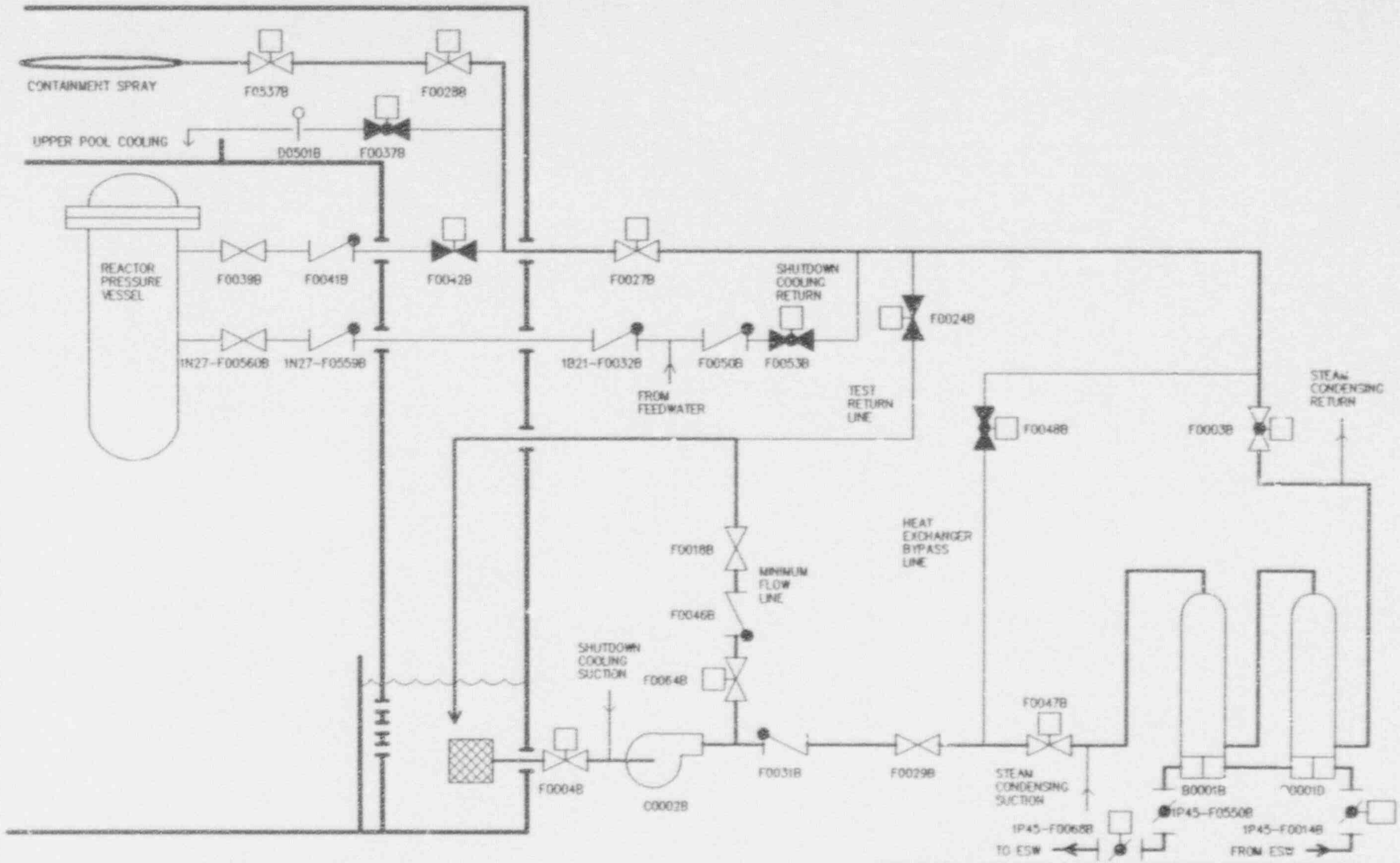


Unit 1 Perry IPE

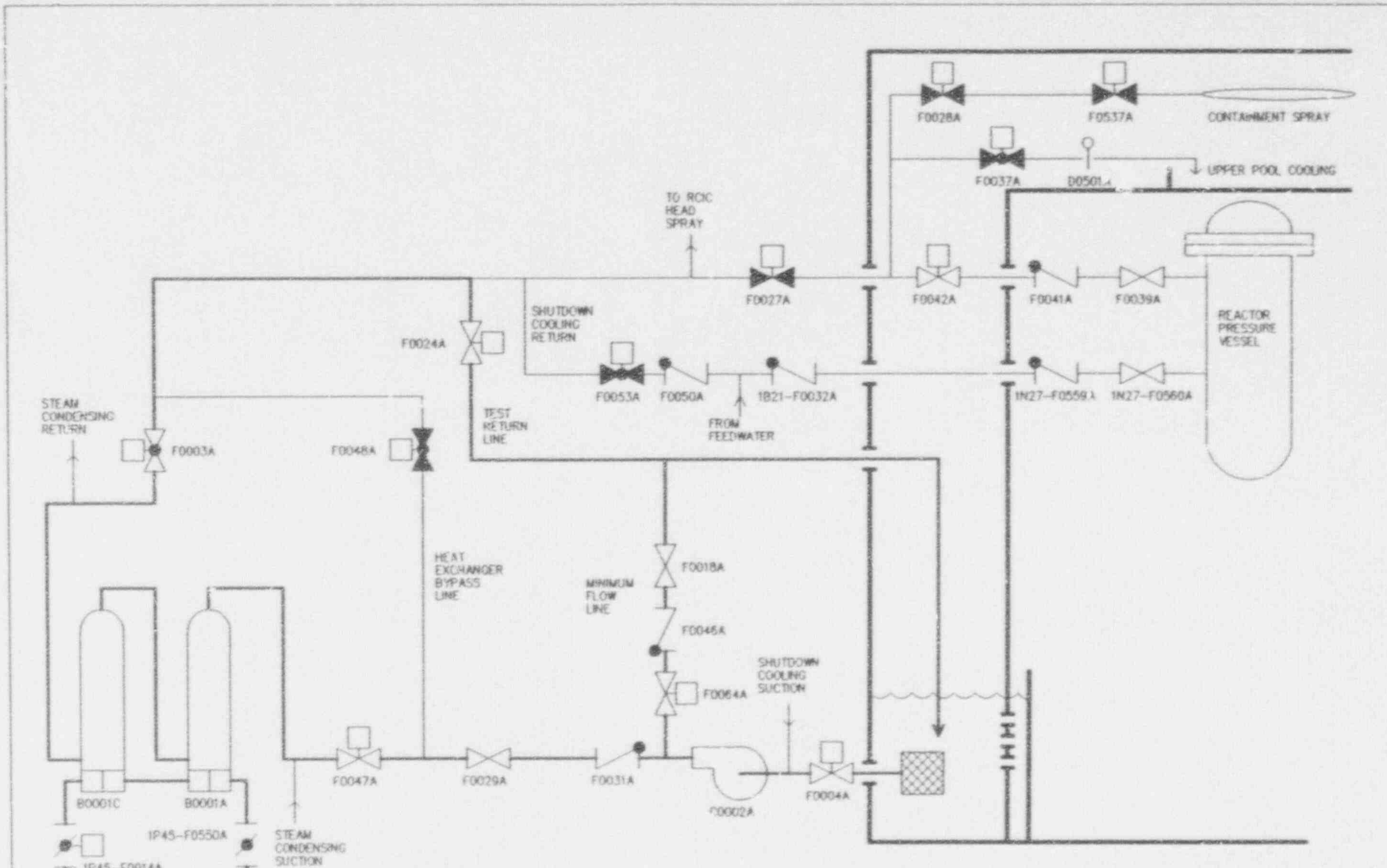
RHR – Containment Spray

Train A

Figure 3.2.3-4



Unit 1 Perry IPE
RHR – Containment Spray
Train B
Figure 3.2.3-5

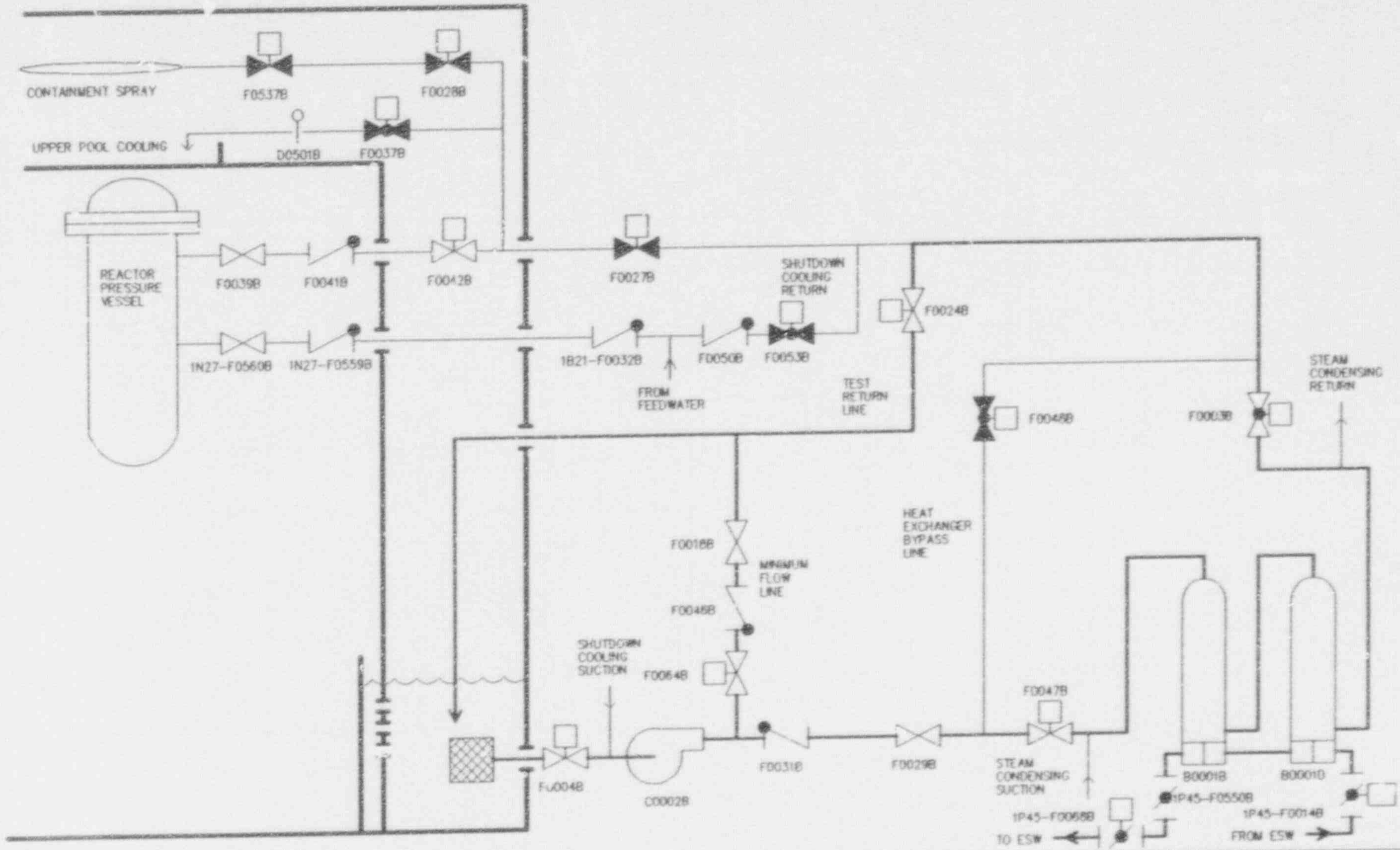


Perry Unit 1 IPE

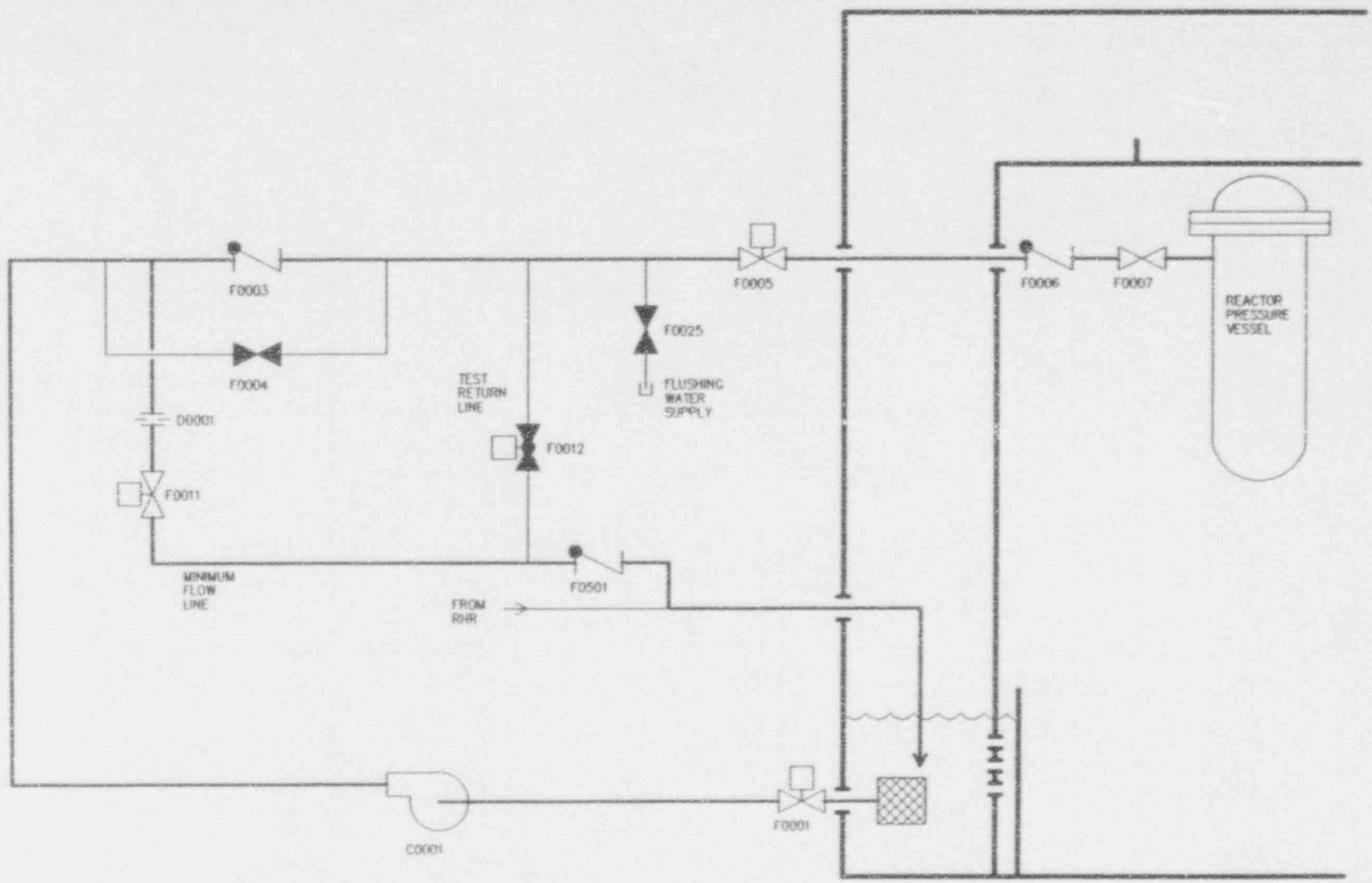
RHR - Suppression Pool Cooling

Train A

Figure 3.2.3-6



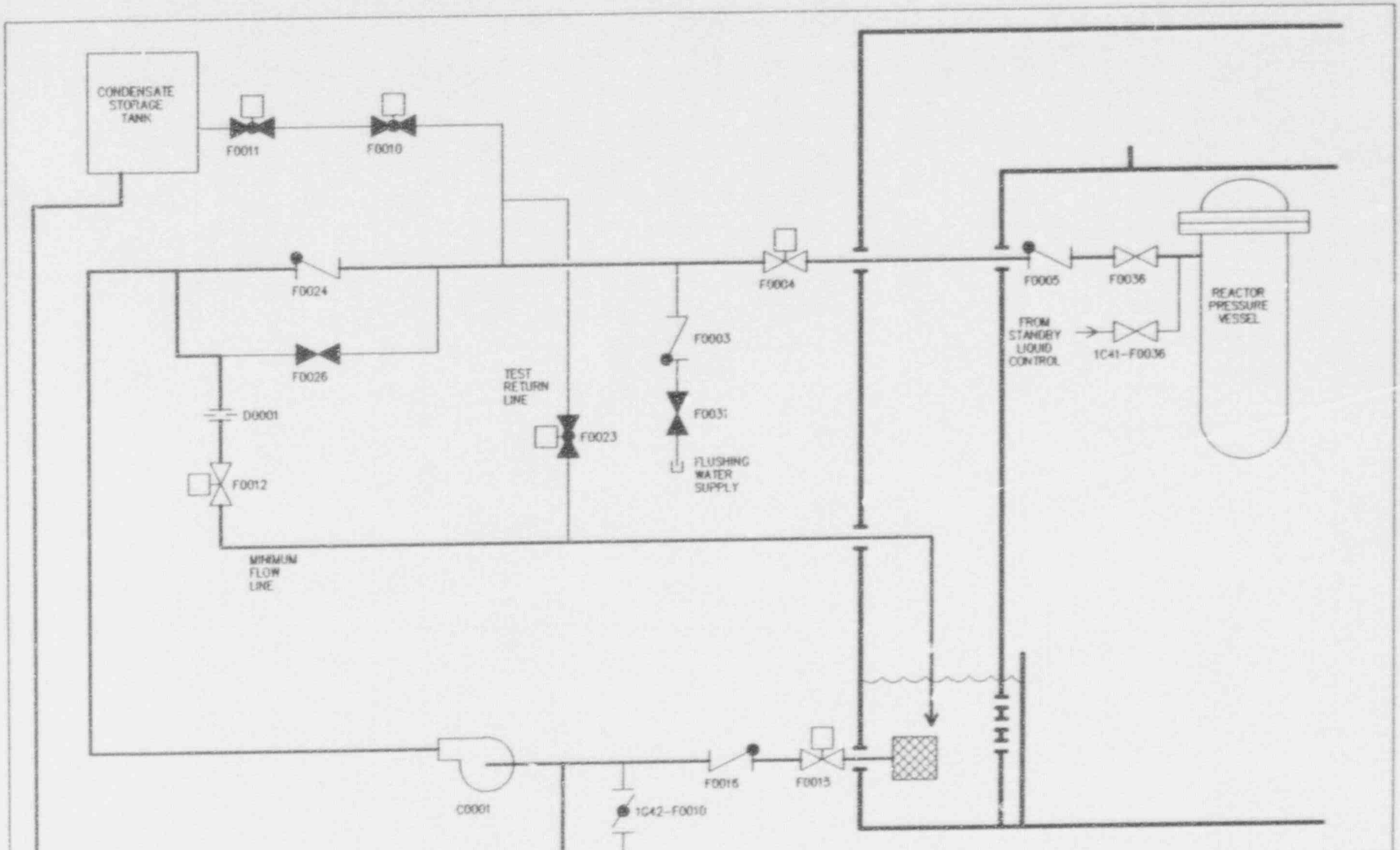
Perry Unit 1 IPE
RHR – Suppression Pool Cooling
Train B
Figure 3.2.3-7



Unit 1 Perry IPE

Low Pressure Core Spray

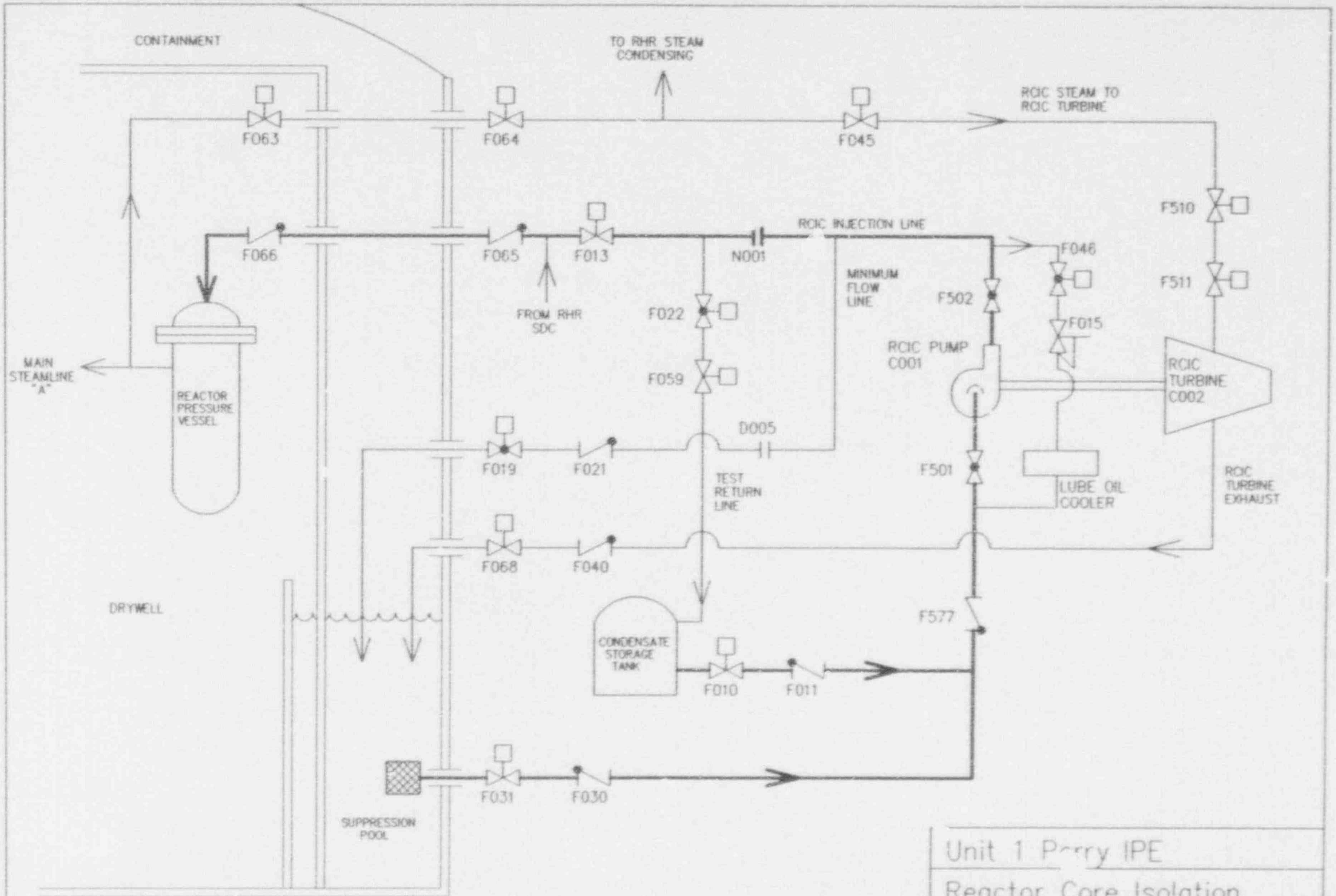
Figure 3.2.4-1



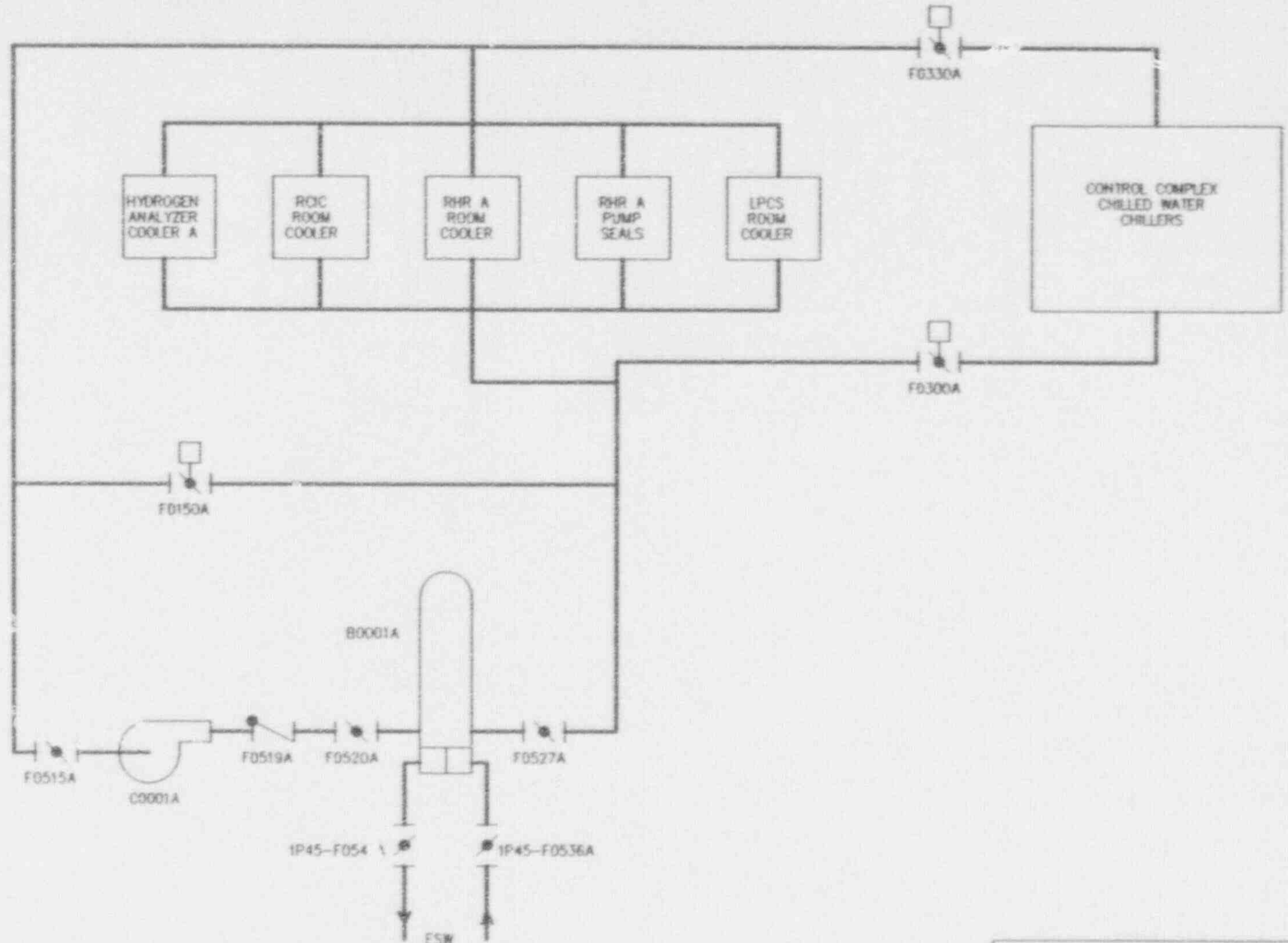
Unit 1 Perry IPE

High Pressure Core Spray

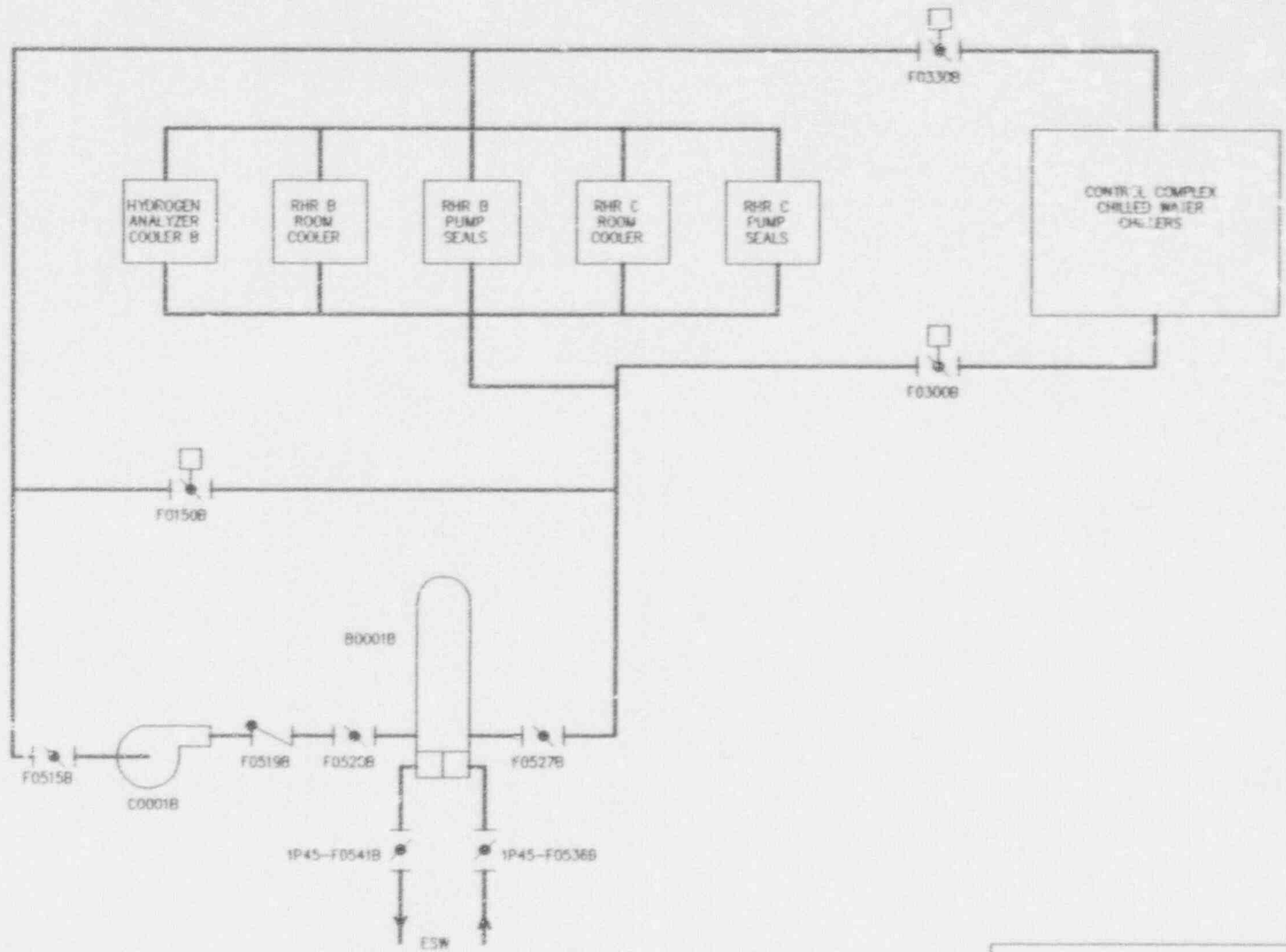
Figure 3.2.5-1



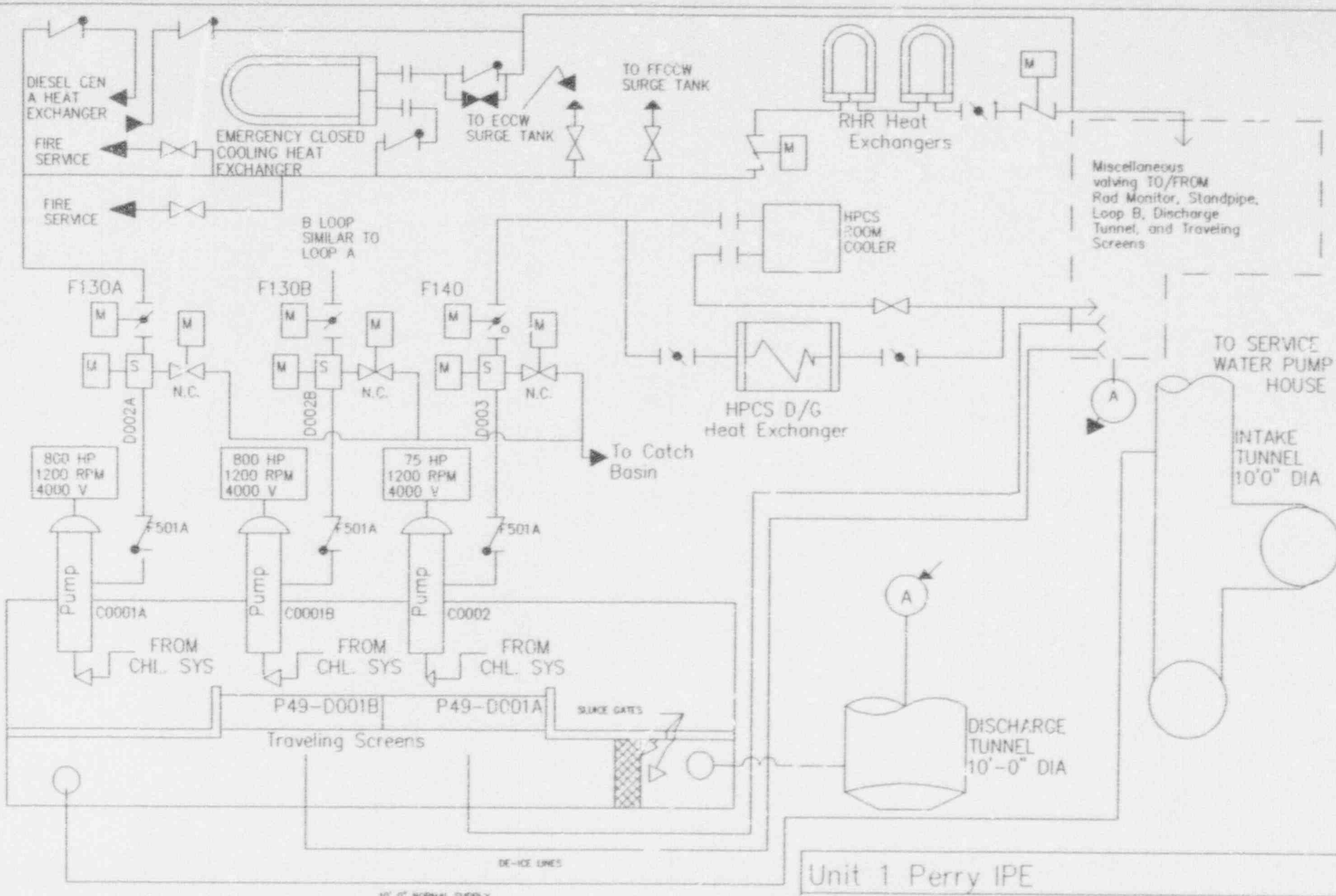
Unit 1 Perry IPE
Reactor Core Isolation
Cooling System (E51)
Figure 3.2.6-1



Unit 1 Perry IPE
Emergency Closed Cooling
Train A
Figure 3.2.7-1



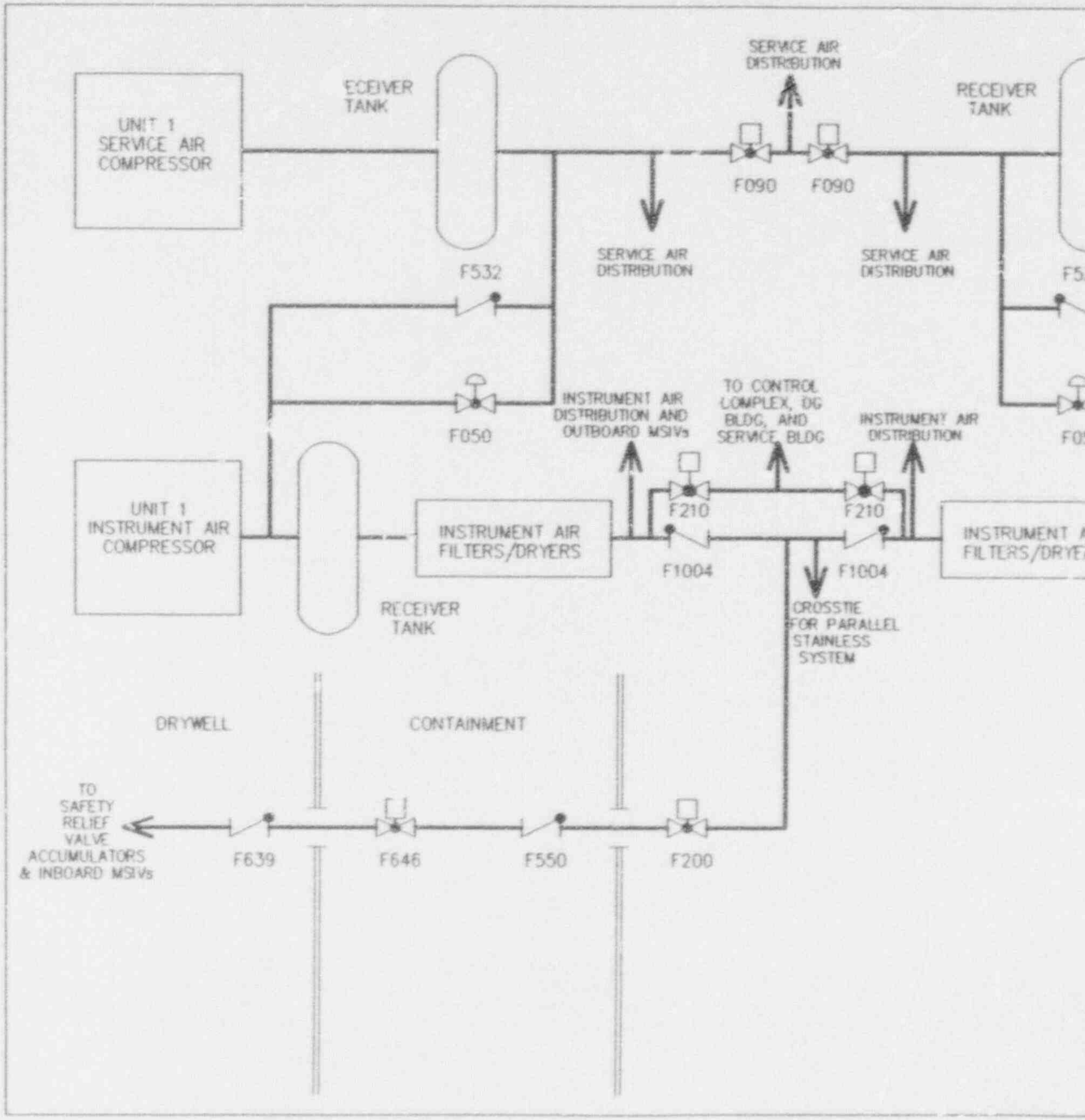
Unit 1 Perry IPE
Emergency Closed Cooling
Train B
Figure 3.2.7-2



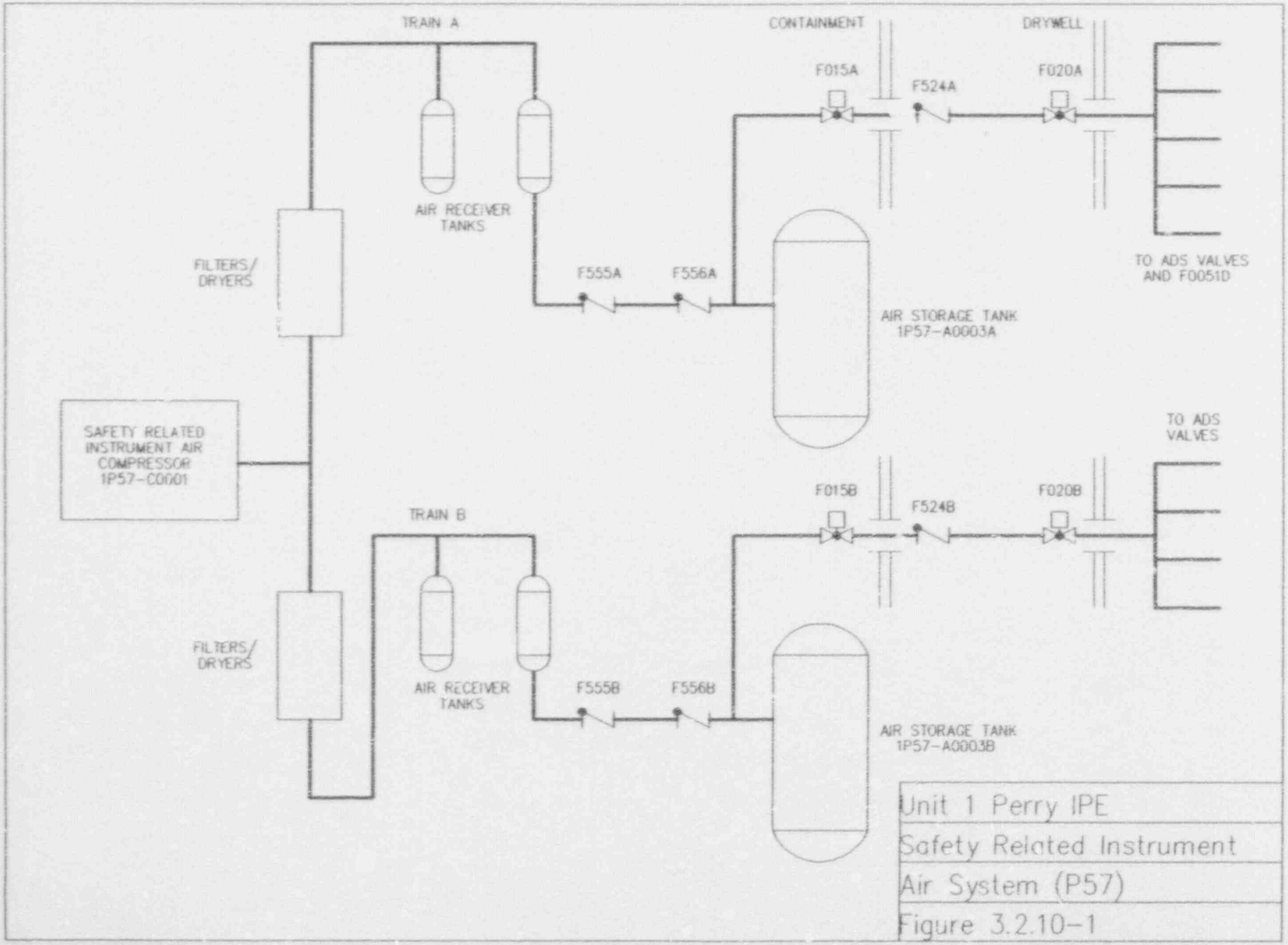
Unit 1 Perry IPE

Emergency Service Water

Figure 3.2.8-1



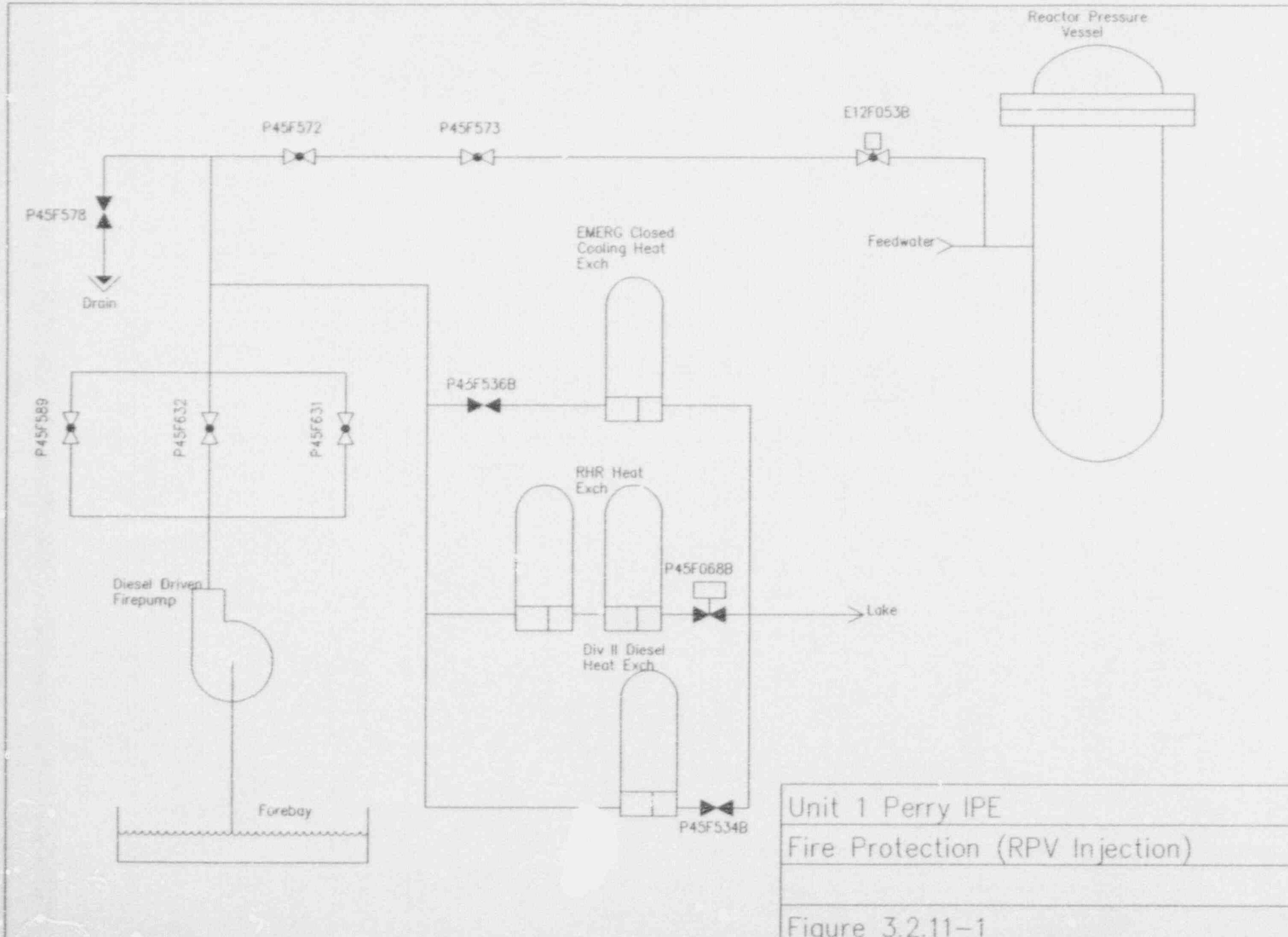
Unit 1 Perry IPE
Service/Instrument Air
System (P51/P52)
Figure 3.2.9-1



Unit 1 Perry IPE

Safety Related Instrument Air System (P57)

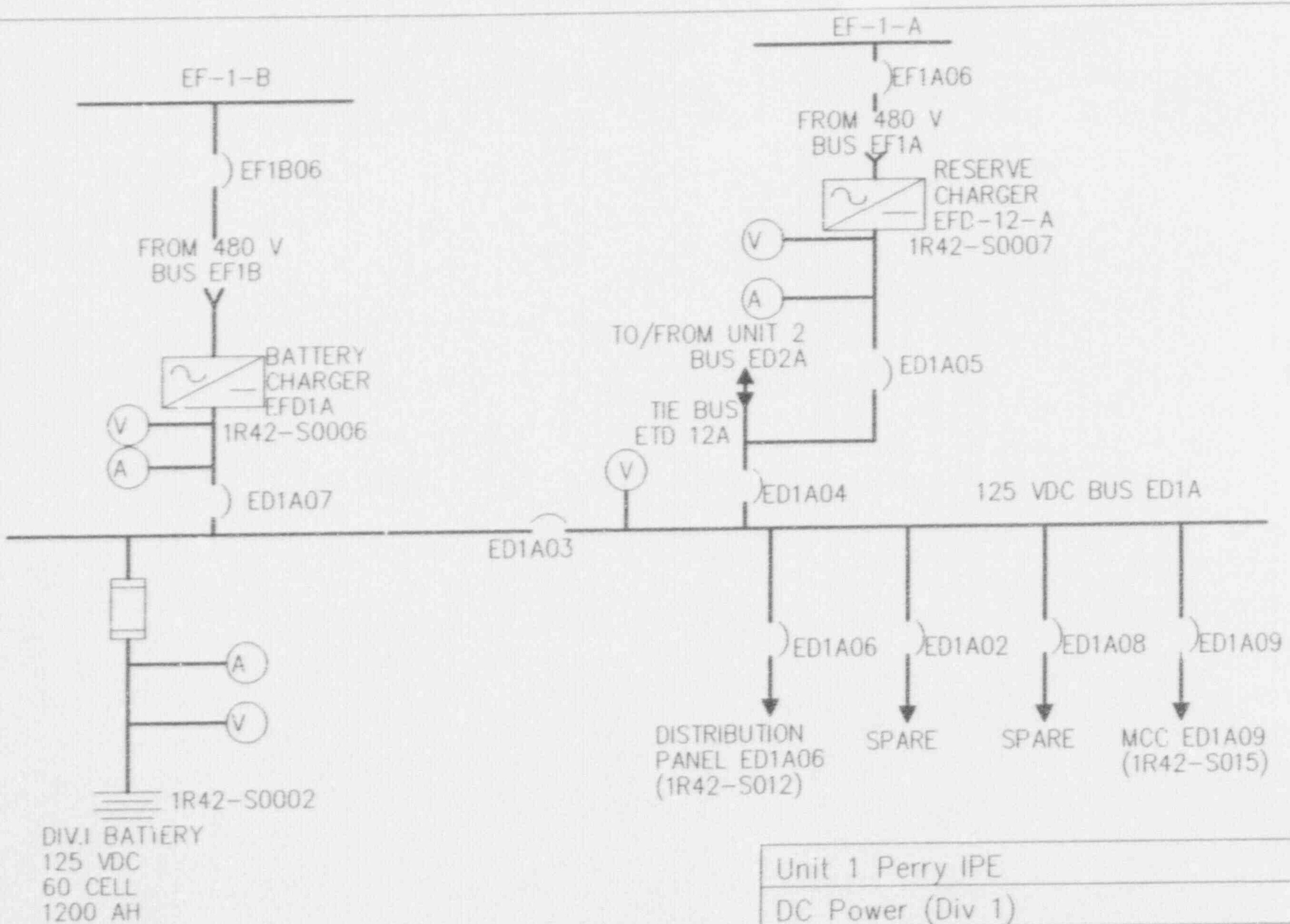
Figure 3.2.10-1



Unit 1 Perry IPE

Fire Protection (RPV Injection)

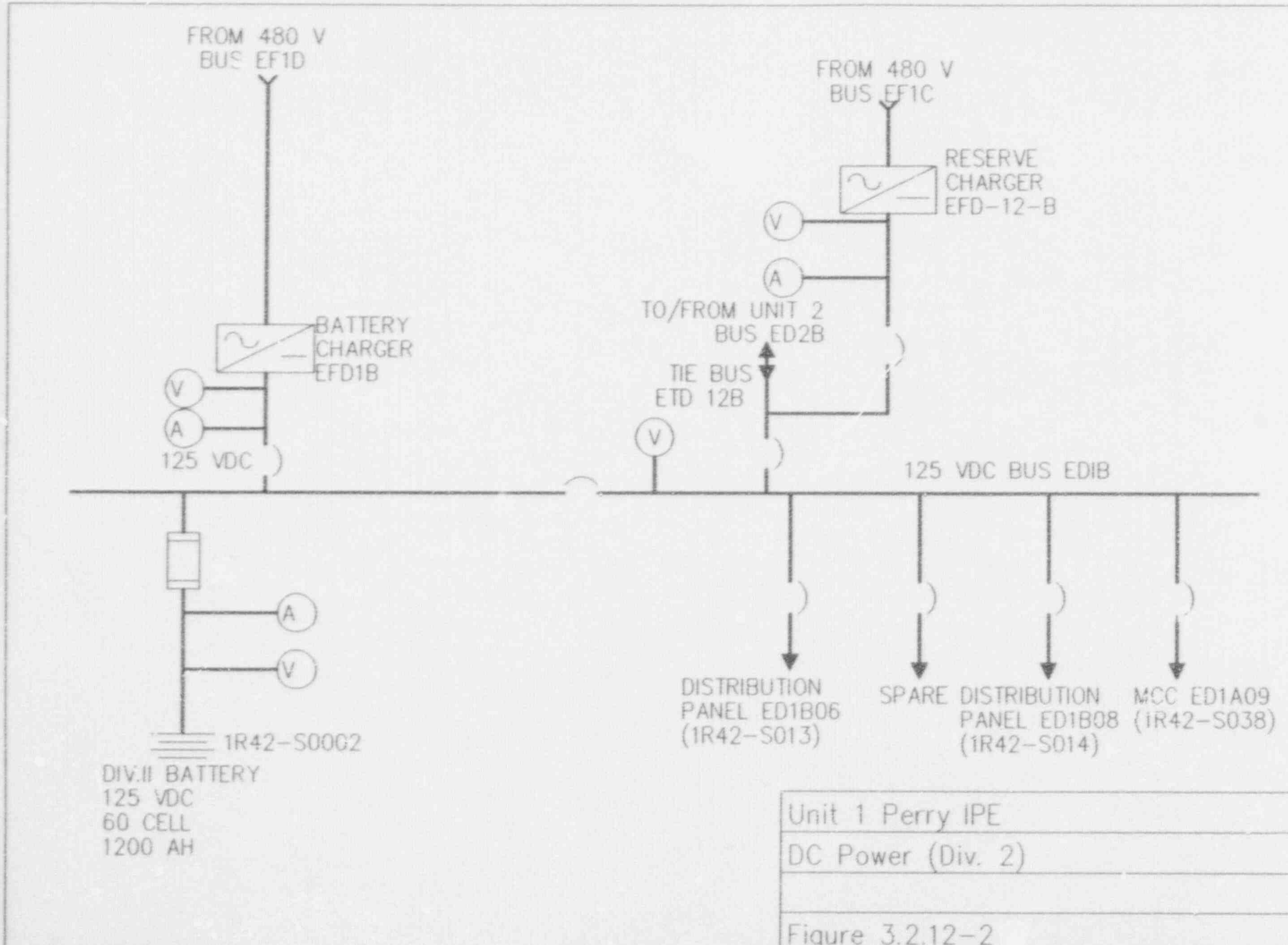
Figure 3.2.11-1



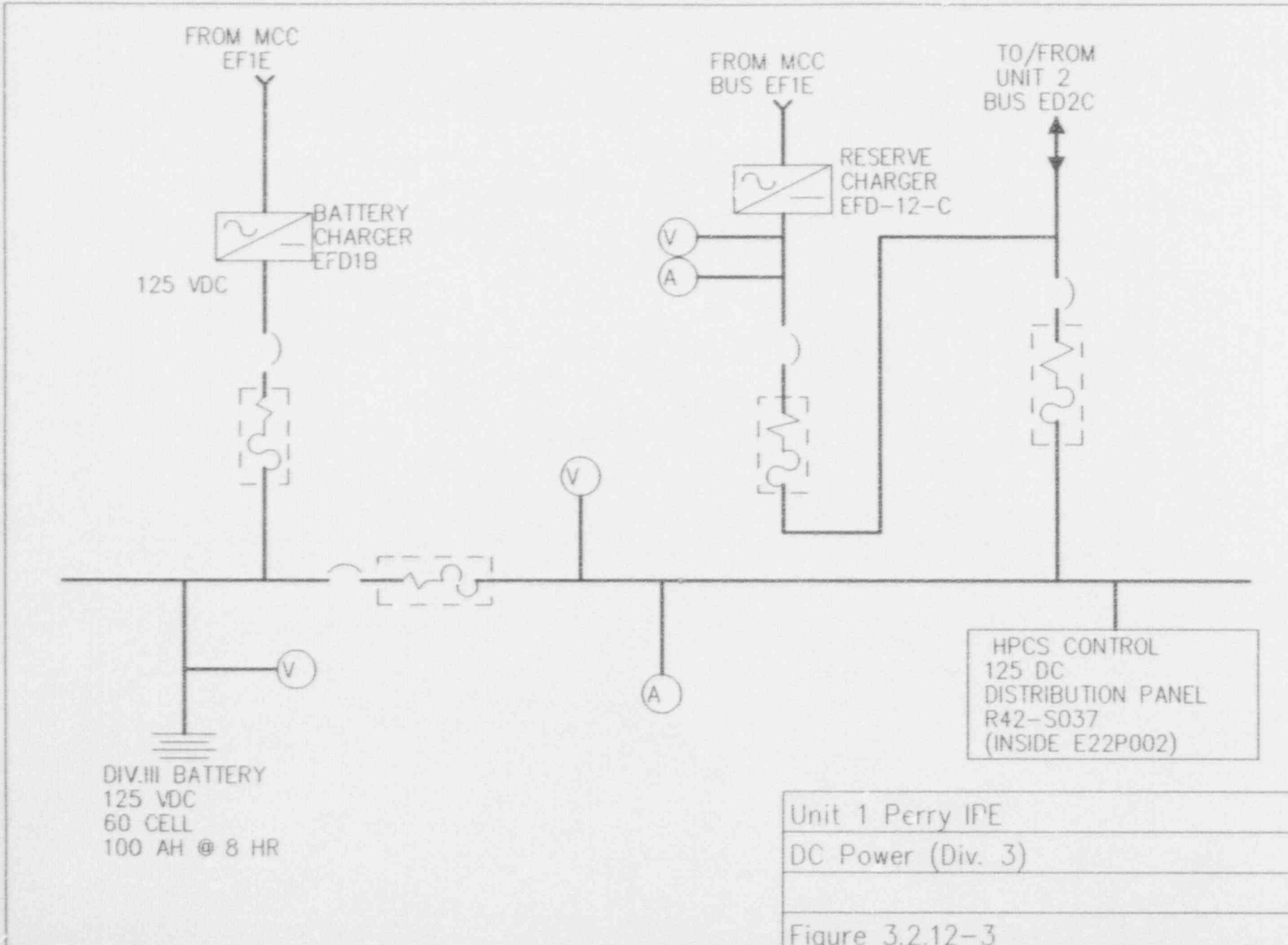
Unit 1 Perry IPE

DC Power (Div 1)

Figure 3.2.12-1



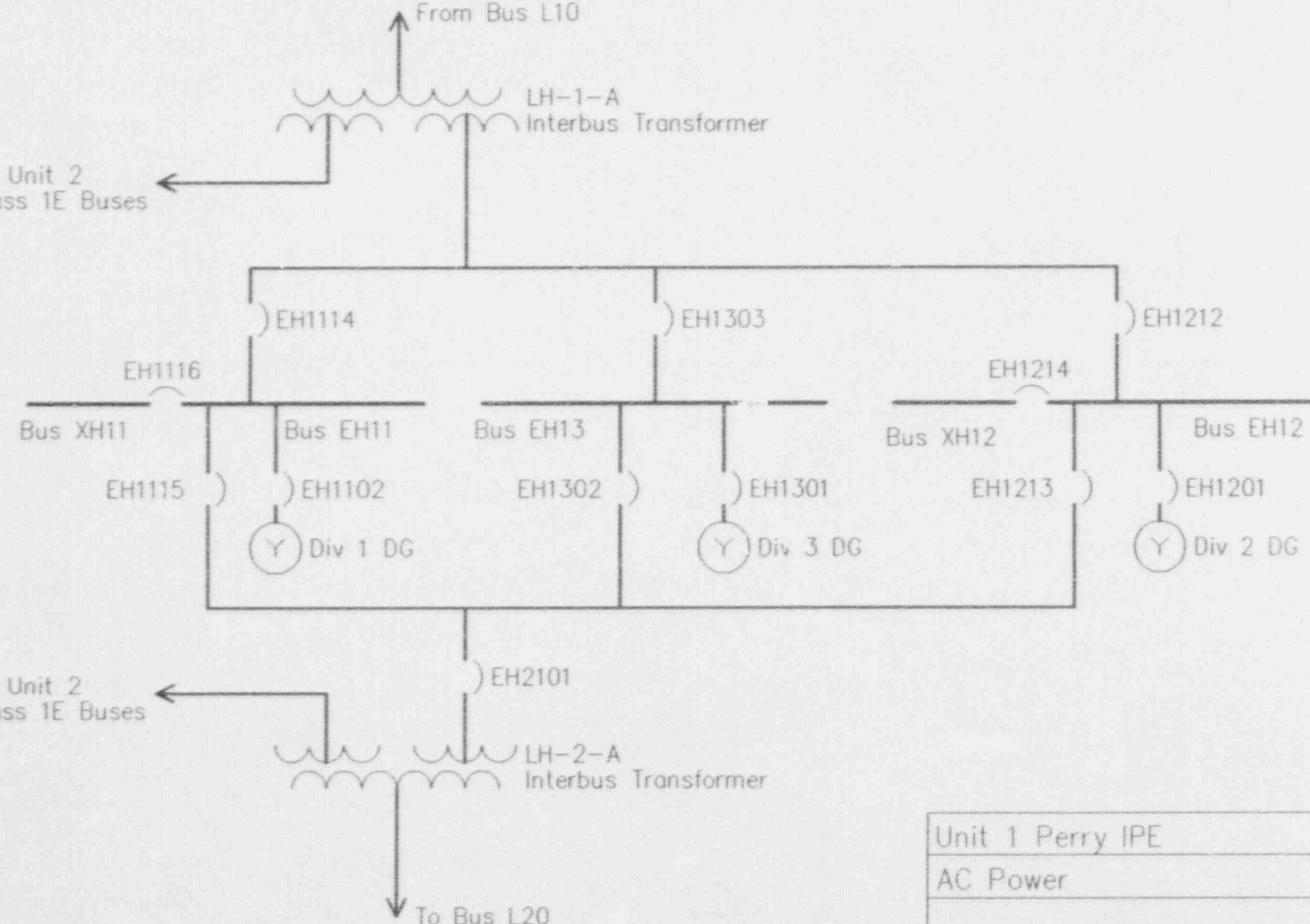
Unit 1 Perry IPE
DC Power (Div. 2)
Figure 3.2.12-2



Unit 1 Perry IPE

DC Power (Div. 3)

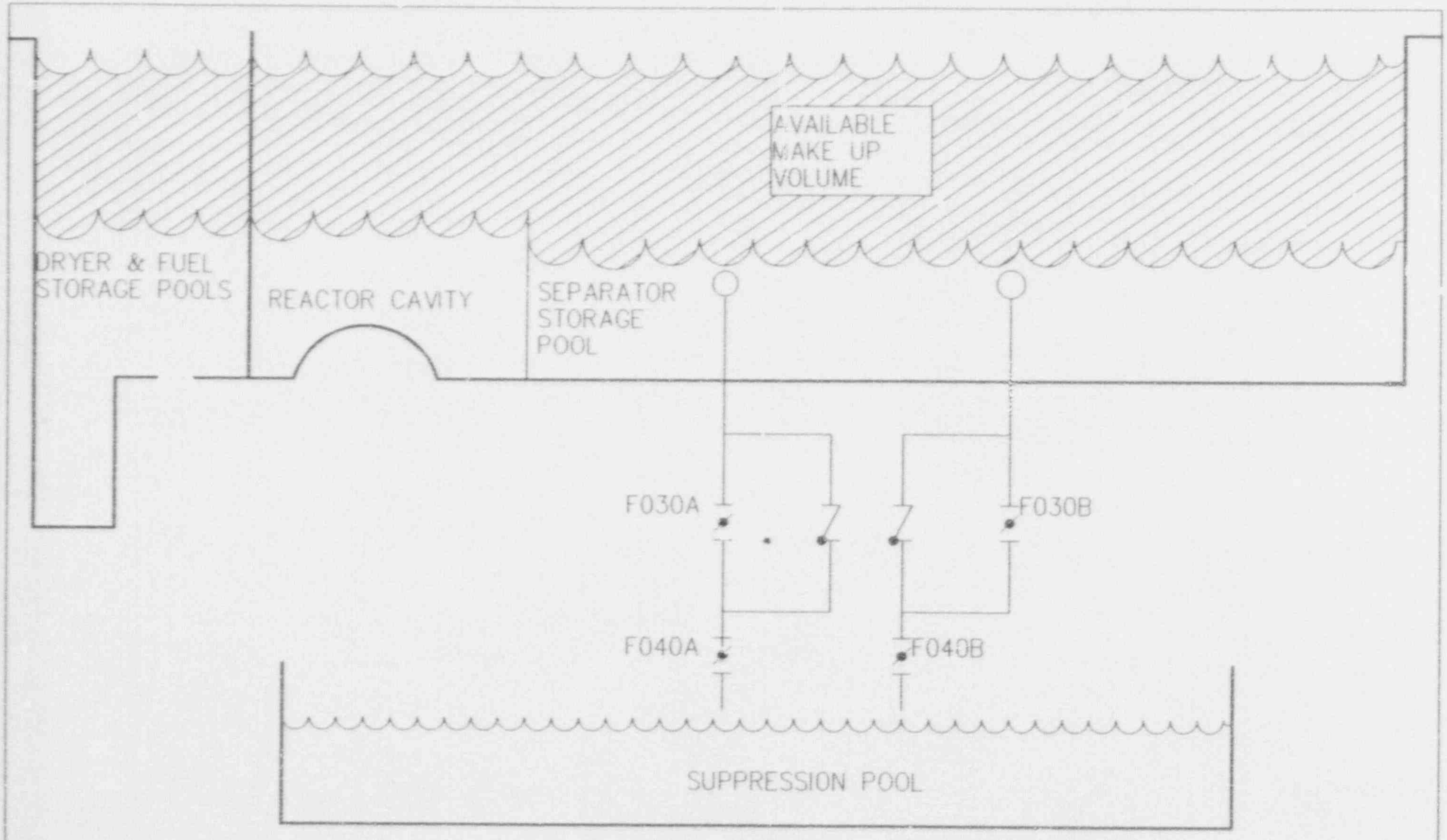
Figure 3.2.12-3



Unit 1 Perry IPE

AC Power

Figure 3.2.13-1

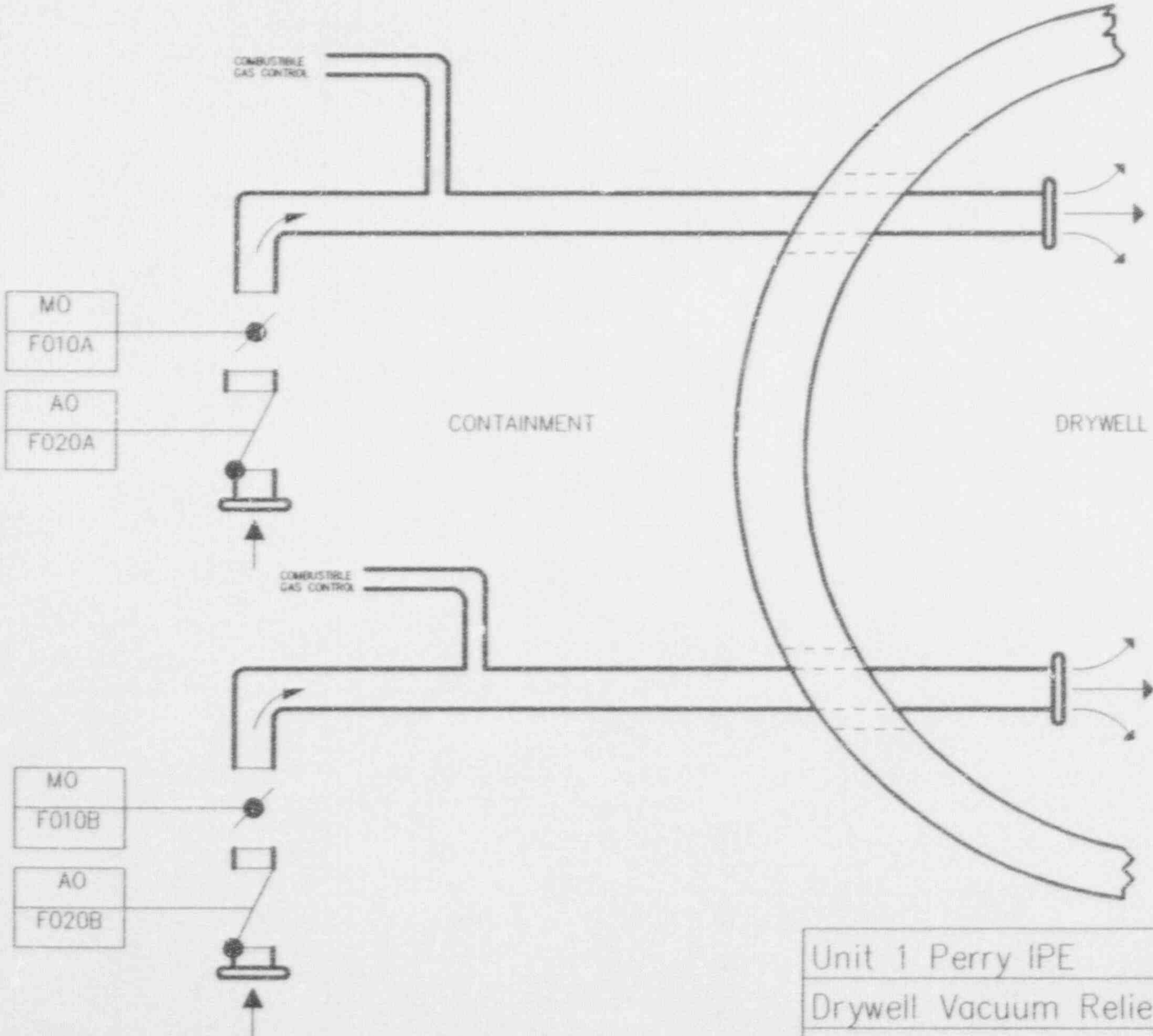


Unit 1 Perry IPE

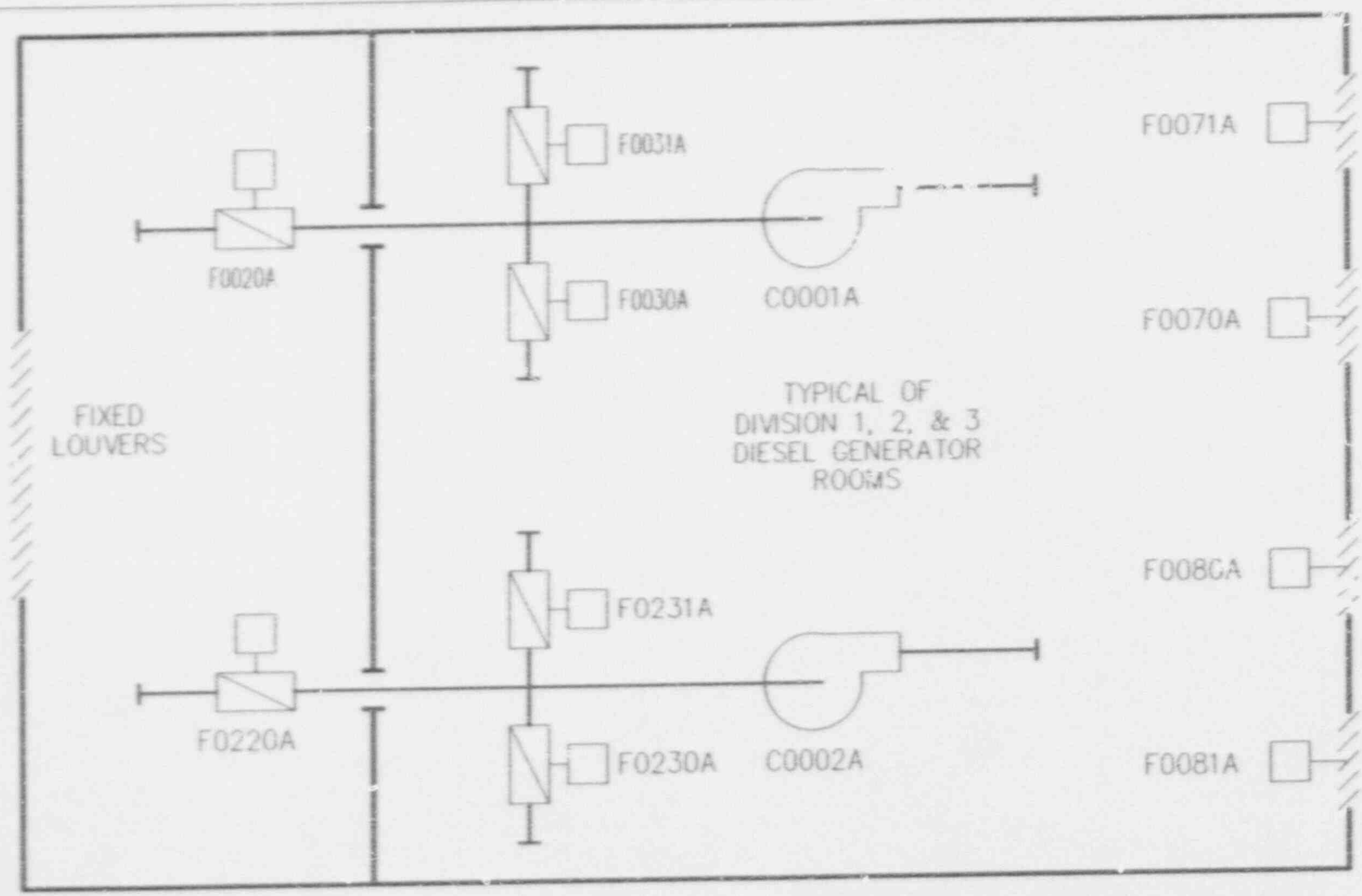
Suppression Pool Make Up

Trains A & B

Figure 3.2.14-1



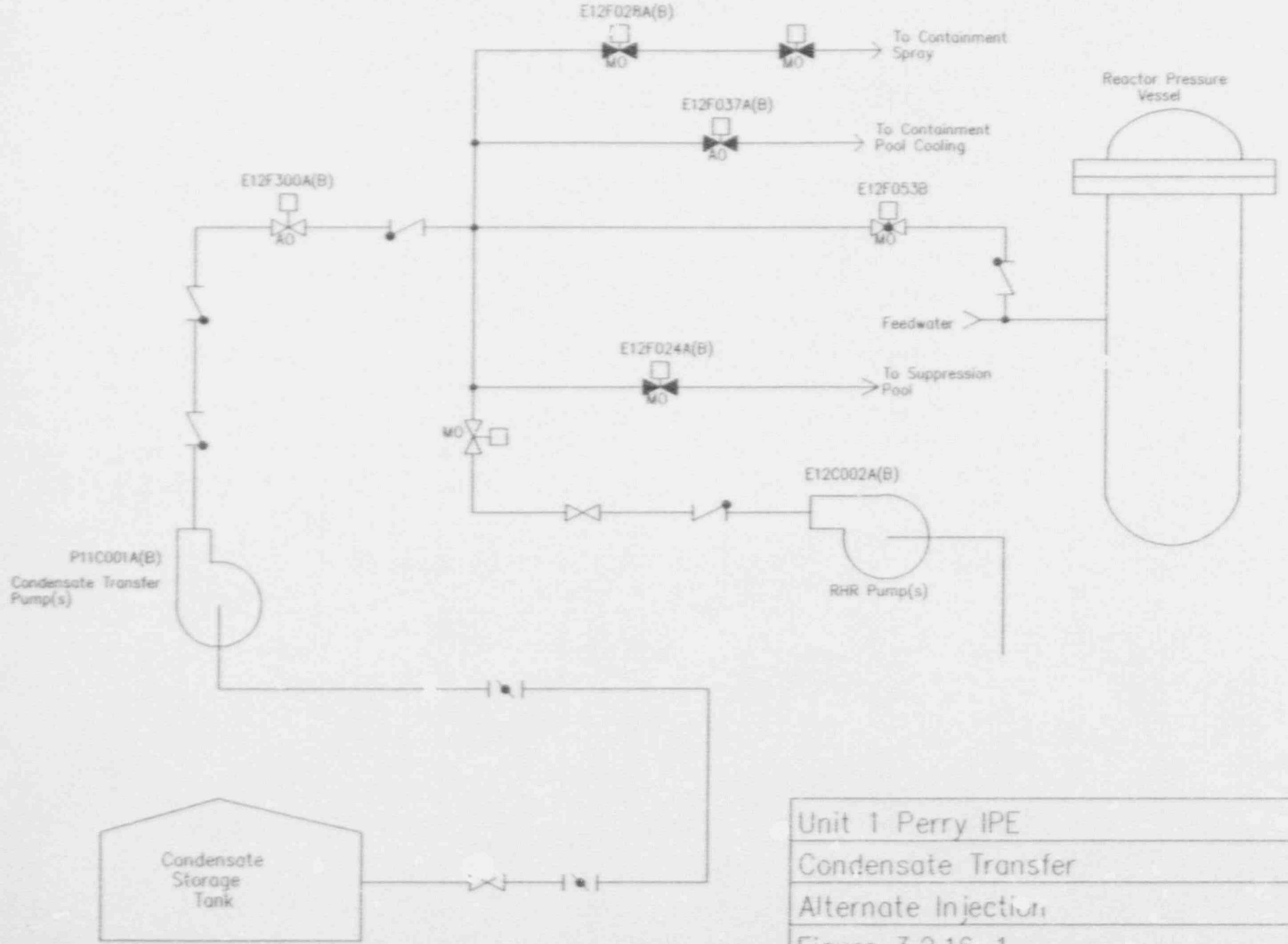
Unit 1 Perry IPE	
Drywell Vacuum Relief System	
Trains A & B	
Figure 3.2.14-2	



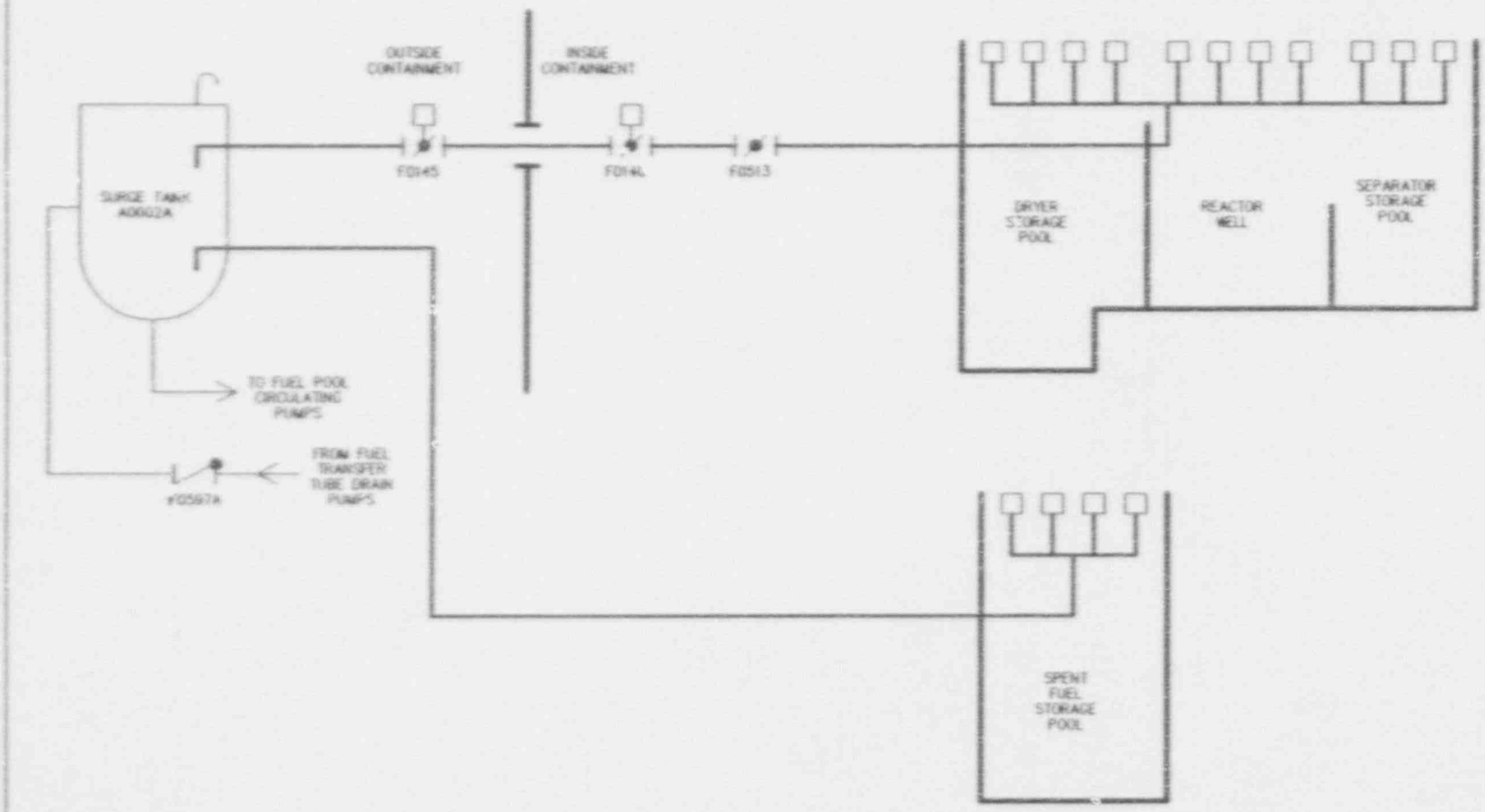
Unit 1 Perry IPE

Diesel Generator Room Ventilation

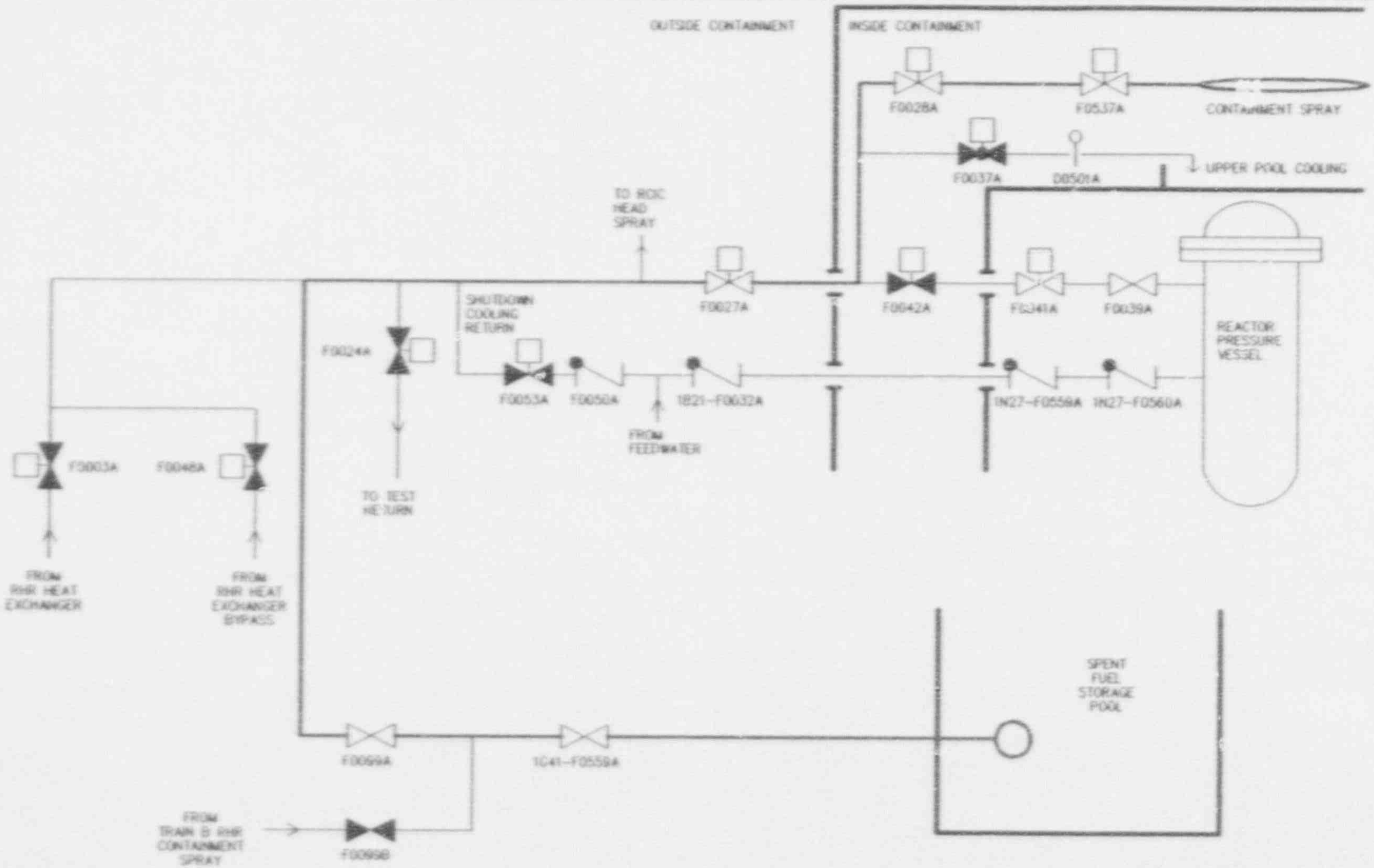
Figure 3.2.15-1



Unit 1 Perry IPE
Condensate Transfer
Alternate Injection
Figure 3.2.16-1



Unit 1 Perry IPE
Containment Venting
Fuel Pool Clng & Cleanup
Figure 3.2.17-1

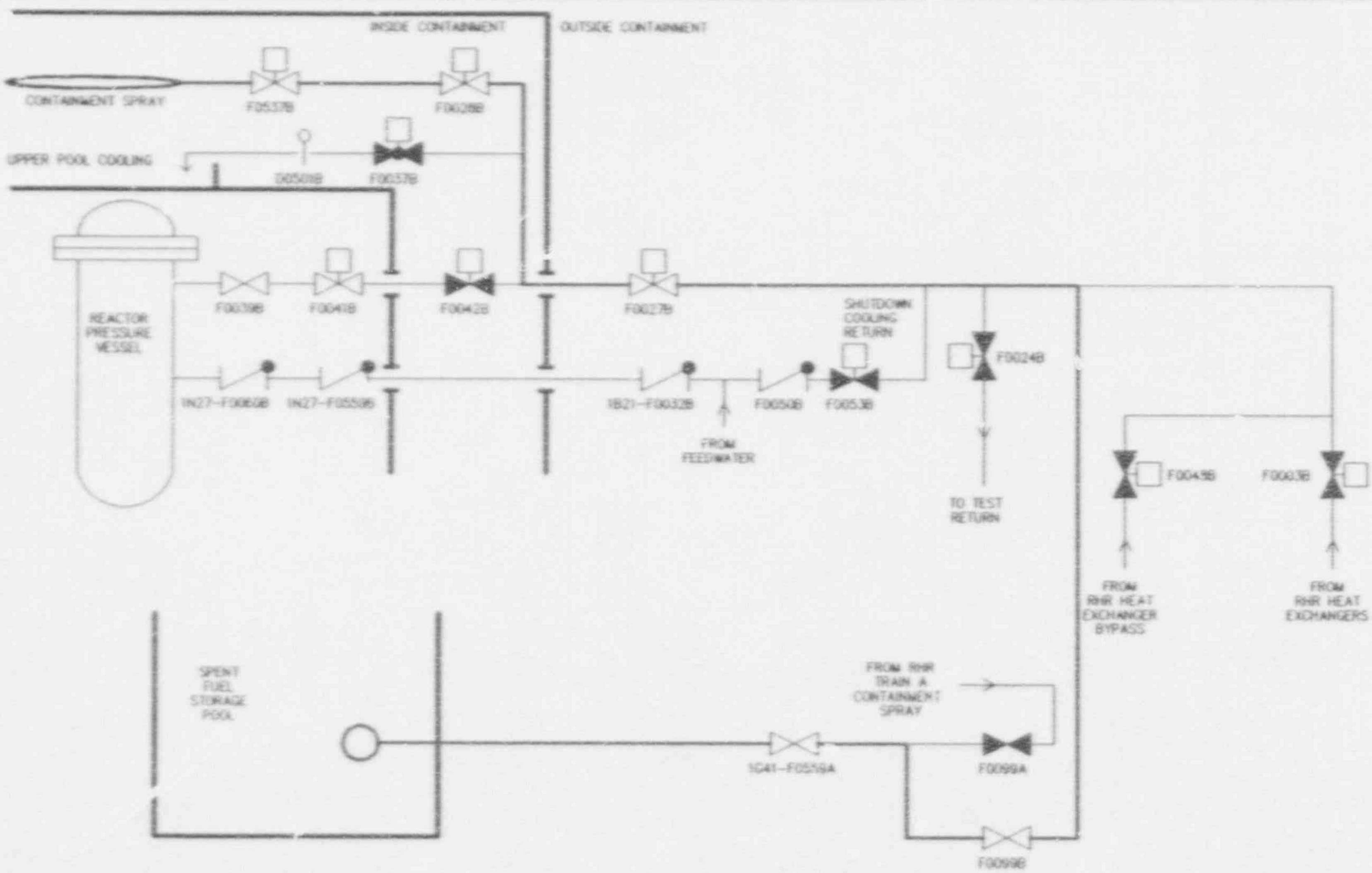


Unit 1 Perry IPE

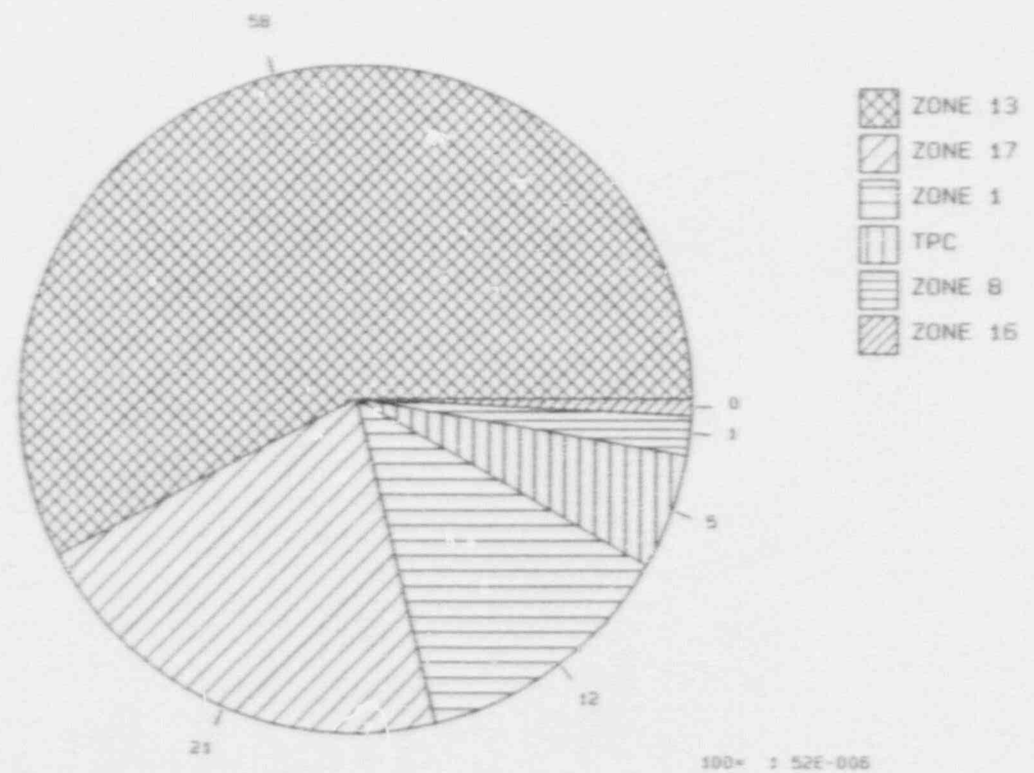
Containment Venting

RHR Containment Spray - Train A

Figure 3.2.18-1



Unit 1 Perry IPE
Containment Venting
RHR Containment Spray - Train B
Figure 3.2.18-2

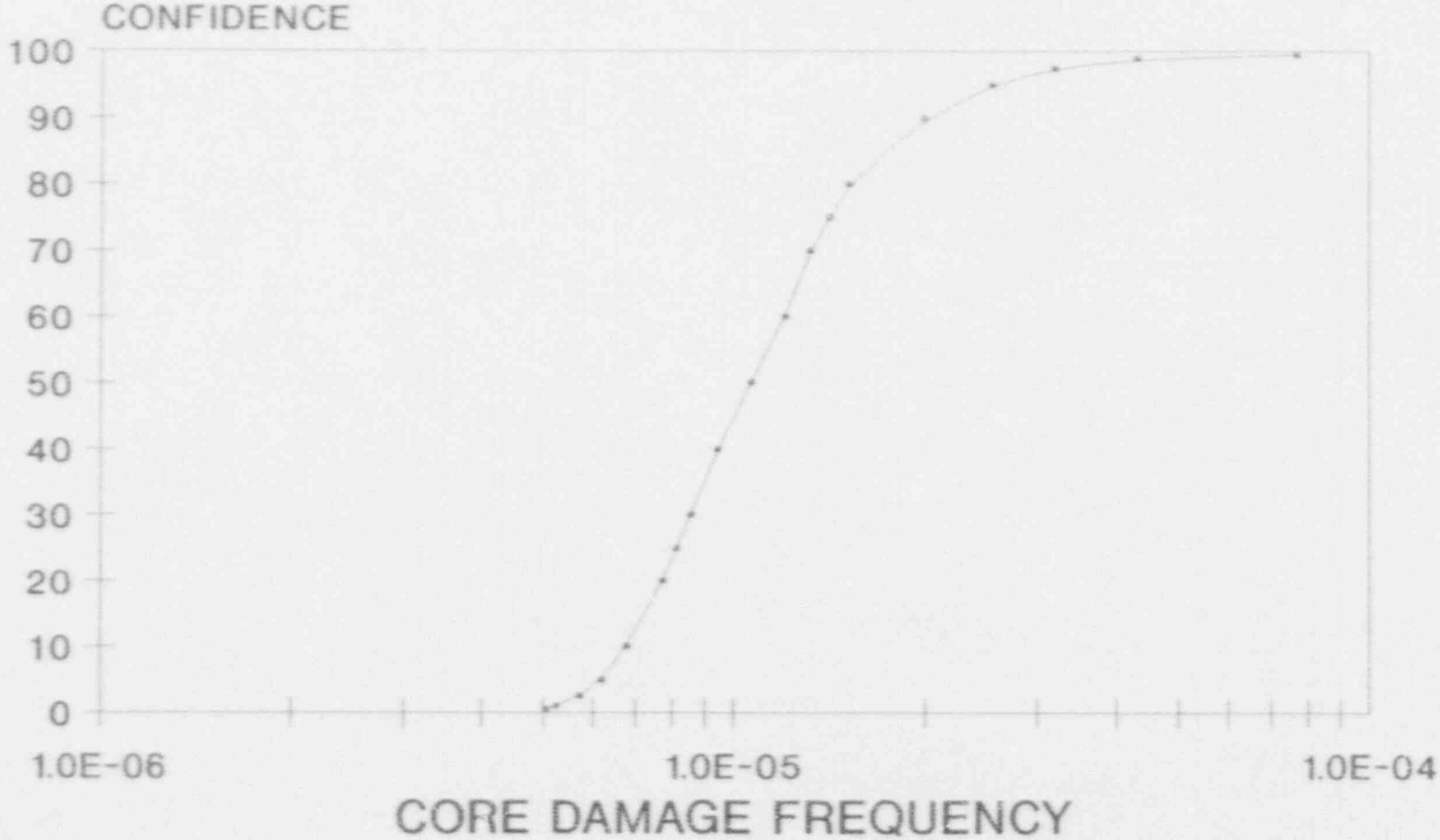


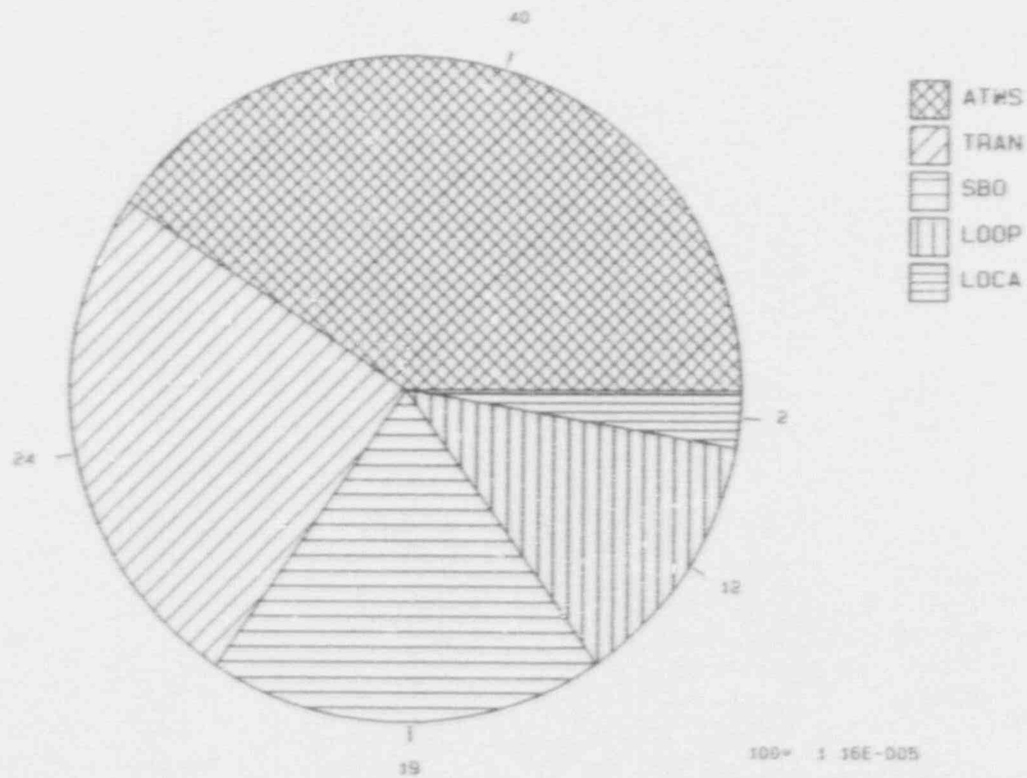
CONTRIBUTION TO CORE DAMAGE BY FLOOD ZONE

Figure 3.3.7-1

CORE DAMAGE FREQ DISTRIB

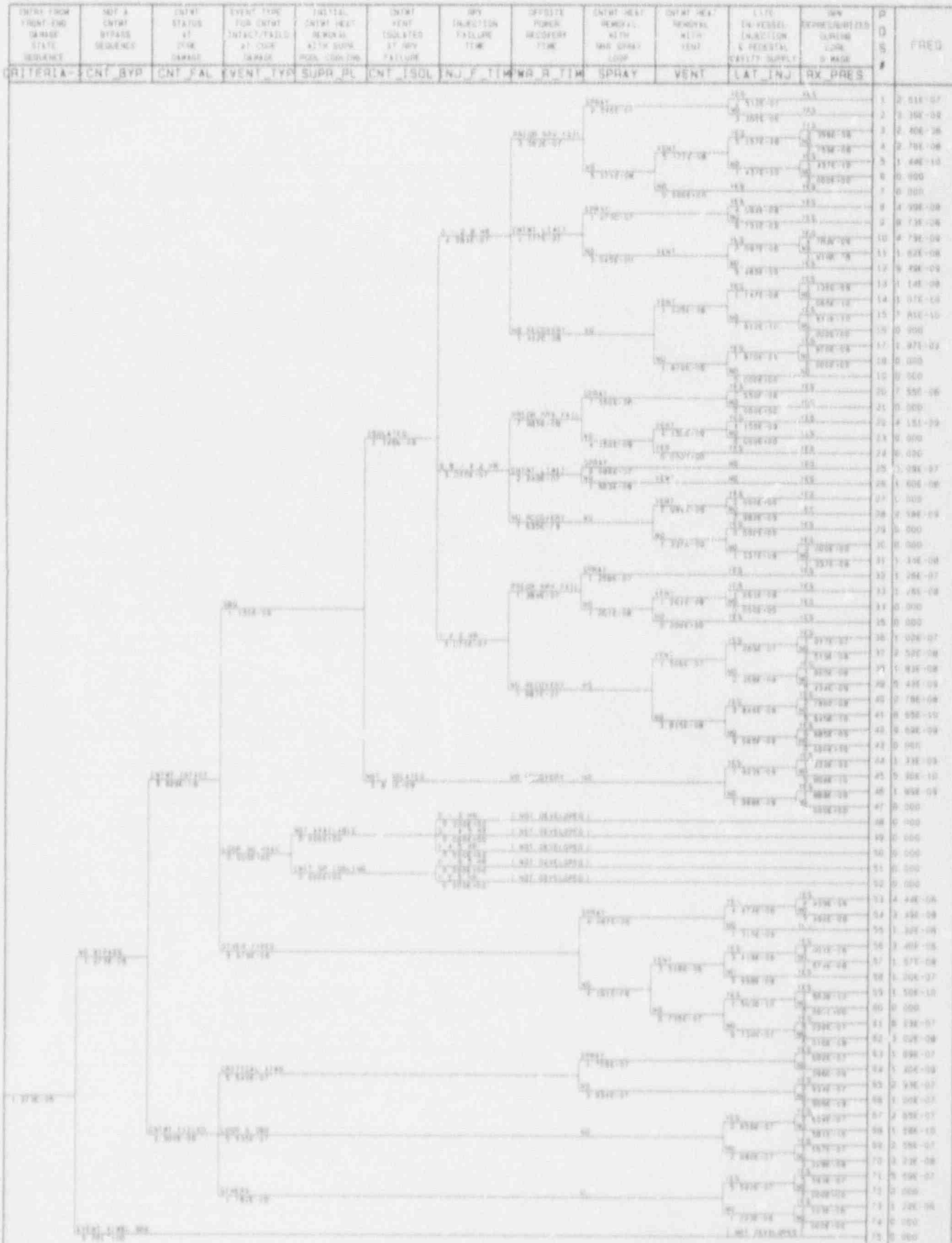
FIGURE 3.4.1-1





CONTRIBUTION OF INTERNAL INITIATORS TO CDF

Figure 3.4.1-2



PERRY NUCLEAR PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.4.1-3
PLANT DAMAGE STATE GROUPING LOGIC
REVISION 0

4.0 BACK-END ANALYSIS

4.1 PLANT DATA

4.1.1 Mark III Plant Features

The Perry Nuclear Power Plant (PNPP) Unit 1 has a General Electric BWR/6 reactor rated at 3579 MWt and a Mark III free standing steel containment. The Mark III containment system is designed to minimize the release of fission products to the environment by containing and suppressing a radioactive steam release caused by an accident.

The PNPP Mark III containment shown in Figure 4.1.1-1 consists of a concrete drywell structure that contains the reactor vessel and the connected piping systems. The drywell is enclosed within a steel containment vessel. A suppression pool is at the bottom of the containment. The suppression pool is made common to both the containment and the drywell air volumes by three rows of horizontal vents which are installed in the drywell wall below the normal level of the suppression pool. The suppression pool water volume serves to dissipate the energy released from opened reactor safety relief valves or from a line break in the drywell. A concrete shield building surrounds the containment vessel and forms an annulus which is maintained at a slight negative pressure.

The design pressure of the Perry Nuclear Power Plant containment is 15 psig, and the total containment volume is comparable with a large PWR containment (1.44 million cubic feet).

Containment heat is removed with the two trains of the Residual Heat Removal (RHR) system in either the suppression pool cooling mode or containment spray mode. In the event that the RHR system fails to suppress the pressure in the containment, the containment can be vented.

To reduce the potential of a severe hydrogen combustion event during an accident, the containment has a Hydrogen Ignition System (HIS). This system is designed to prevent the build-up of large quantities of hydrogen inside the containment. Igniters are located throughout the containment and drywell volumes.

A general comparison of design information for the Mark III containments at Perry and Grand Gulf (the NUREG-1150 plant) is presented in Table 4.1.1-1.

A detailed description of the Perry containment is provided in section 2.0 of Appendix H.1, PNPP IPE Containment Analysis.

4.1.2 Sources for Data

The Perry plant and containment specifications pertinent to the IPE Back-End analysis are included in the Modular Accident Analysis Program (MAAP) Parameter File. The entire PNPP IPE MAAP Parameter File is contained in Appendix H.5. The MAAP Parameter file includes plant and containment modeling information sections which are described in the MAAP Users Guide (EPRI 1991).

4.2 PLANT MODELS AND METHODS FOR PHYSICAL PROCESSES

The Electric Power Research Institute (EPRI) maintains a deterministic containment evaluation software code called the Modular Accident Analysis Program (MAAP). MAAP is used for the Perry Nuclear Power Plant IPE containment evaluation for the accident progression analysis, to assist in quantifying the Accident Progression Event Tree (APET), and for estimating source terms.

MAAP 3.0B BWR Version 7.02 is the basis for the Perry analysis. Two minor modifications were added to the code with the direction of Fauske and Associate, Inc. (FAI), the MAAP Maintenance Contractor to correct identified coding problems. The Perry MAAP code was validated to the FAI test cases.

4.2.1 MAAP Analysis Assumptions (Model Parameters)

The MAAP model parameters generally represent inputs to phenomenological models in which significant uncertainties exist. Variations in the values of these parameters can be made to assess the impact of uncertainties in important physical models. The best estimate values used in the Perry IPE are provided in the Model Parameters section of the PNPP IPE MAAP Parameter File, provided as Appendix H.5. These best estimate values were taken directly from the "Recommended Sensitivity Analyses For An Individual Plant Examination Using MAAP 3.0B" (EPRI 1990).

4.3 BINS AND PLANT DAMAGE STATES

4.3.1 Plant Damage State Grouping Parameters

The interface between the Front-End Analysis and the Back-End Analysis consists of a set of plant damage states (PDS). The plant damage states define a set of functional characteristics for system operation which are important to accident progression, containment failure and source term definition. Each Plant Damage State contains Front-End sequences with sufficient similarity of system functional characteristics that the containment accident progression for all sequences in the group can be considered to behave similarly in the period after core damage has begun. Each Plant Damage State defines a unique set of conditions regarding the state of the plant and containment systems, the physical state of the core, the primary coolant system, and the containment boundary at the time of core damage. The important functional characteristics for each Plant Damage State were determined by defining the critical parameters or system functions which impact key results. The sequence characteristics which are important are defined by the requirements of the containment accident progression analysis. These include the type of accident initiator, the operability of important systems, and the value of important state variables (e.g., reactor pressure) which are defined by system operation.

4.3.1.1 Identification of Important PDS Functional Characteristics

The accident initiators, plant systems and various possible states of the reactor system and containment (at the time of core damage), and plant operating instructions were reviewed to determine the potential impact on containment accident progression. The most important functional characteristics are listed below within functional categories.

Containment Status At Core Damage
- Containment Intact
- Containment Isolated
- Containment Bypassed
- Containment Failed

Event Type At Core DAmage
- SBO
- LOOP With No HVAC
- ATWS
- Other

Power Recovery For SBO Core Damage Sequences
- Before Reactor Vessel Failure
- Before Containment Vessel Failure
- No Recovery

Containment Mitigation Features
- Containment Heat Removal
- Containment Sprays
- Containment Venting
- * Pedestal Cavity Water Supply
- * Hydrogen Igniters
- * Annulus Exhaust Gas Treatment System

Reactor Pressure Vessel Status Prior To Vessel Failure
- Late In-Vessel Injection Available
- RPV Pressure

In the Perry IPE Plant Damage Stage Grouping Logic the Containment Heat Removal and Containment Spray functions were combined into one function, "Containment Heat Removal with RHR Spray Loop."

IPE preliminary evaluation of the core damage sequences indicated that a significant fraction of LOOP sequences involved the initial availability of the safe shutdown division 1 and 2 diesel generators, which later become unavailable due to the loss of MCC, Switchgear And Miscellaneous Electrical Equipment Area HVAC. Loss of HVAC sequences are conservatively modeled in the containment evaluation analysis as failing the HVAC system at time zero. The impact of tripping the thermal overload protection on Division 1 & 2 and 3 MCCs in 4.7 and 10 hours, respectively, is the loss of diesel ventilation and the postulated immediate loss of the associated diesel generator. LOOP with no HVAC sequences are similar to Station Blackout sequences (the Hydrogen Ignition System is unavailable). However, the Plant Damage State Logic Grouping distinguishes LOOP with No HVAC sequences, since the accident progression timing is substantially different. Preliminary accident progression analysis of LOOP with No HVAC sequences identified that timing of RPV Failure and Containment Failure were sensitive to whether suppression pool cooling was available for the first 4.7 hours. So, Initial Containment Heat Removal With Suppression Pool Cooling was added as a Plant Damage State functional characteristic. However, subsequent IPE examination of MCC thermal overload heating determined that this was not a contributor to core damage. Consequently, the LOOP with No HVAC core damage sequences have been deleted from the analysis. Therefore, these functional characteristics were initially modeled but are not currently used: LOOP with No

HVAC, and Initial Suppression Pool Cooling.

An examination of the above Plant Damage State functional characteristics list indicates that the following characteristics (designated above with an asterisk) were not explicitly required in the plant damage state grouping logic: (1) Pedestal Cavity Water Supply - since this is essentially the same as the Late In-Vessel Injection, except for suppression pool overflow into the drywell and pedestal; (2) Hydrogen Igniter availability, as well as (3) Annulus Exhaust Gas Treatment System availability which are dependent on AC power availability.

4.3.1.2 Identification of Key Event Timing

The timing of key events such as system failure and recovery, core cooling recovery, fuel damage, and operator actions was reviewed with MAAP computer runs and discussions with plant operators.

While both the NUREG/CR-4551 Grand Gulf report (Brown 1990) and the EPRI Generic Framework for Individual Plant Examination Backend Analysis (EPRI 1990) included one time related parameter in the plant damage state grouping criteria (core melt timing), the Perry IPE containment evaluation considered this information was more pertinent for use in offsite consequence analyses. Therefore, core melt timing is not included in the Perry Plant Damage State grouping criteria.

The timing related characteristics in the plant damage state grouping logic are: Vessel Injection Failure Time, and Offsite Power Recovery Time.

The Vessel Injection Failure Time windows for SBO are:

(1)	0 - 2.8 hours	Initial loss of all RPV injection
(2)	2.8 - 4.2	Short term RCIC injection loss
(3)	> 4.2	HPCS and Firewater pump injection loss

The Vessel Injection Failure Time windows for LOOP with No HVAC And With Suppression Pool Cooling Not Available are:

(1)	0 - 3 hours	Initial loss of all RPV injection
(2)	3 - 4.5	Short term RCIC injection loss
(3)	> 4.5	HPCS and Firewater pump injection loss

The Vessel Injection Failure Time windows for LOOP with No HVAC & With Suppression Pool Cooling Available are:

(1)	0 - 9.5 hours	Short term RCIC injection loss
(2)	> 9.5	HPCS and Firewater pump injection loss

Offsite Power Recovery Time during SBO sequences with no injection and with delayed RCIC injection loss is modeled with the three following time windows:

(1) After core damage and prior to RPV failure,
(2) After RPV failure and prior to slow overpressurization containment failure, and
(3) No power recovery.

Offsite Power Recovery Time during SBO sequences with delayed HPCS or

Firewater) injection loss is modeled with the two following time windows, since the time of RPV failure is near the time of containment failure (if containment heat removal has failed):

(1) After core damage and prior to RPV failure, and
(2) No power recovery.

4.3.2 PDS Grouping Logic and PDS Characteristics

The Plant Damage State Grouping Logic is shown in Figure 4.3.2-1. The Plant Damage State Grouping Logic defines 75 plant damage states which distinguish the important combinations of system states that can result in distinctly different accident progress pathways; and therefore, different containment failure and source term characteristics. This logic was developed using the important functional characteristics identified above in section 4.3.1.1.

The PDS logic tree was developed with the following guidelines:

1. The characteristics judged to be the most important, or on which subsequent decisions are dependent, are placed early (or left-most) in the diagram.

2. Decision points toward the end of the diagram are eliminated, where possible, based on preliminary information on sequence frequencies and consideration of whether a decision branch actually results in sufficient differences in system functional states to warrant differentiation.

4.3.2.1 PDS Grouping Logic Heading Definitions

The following PDS Grouping Logic Headings are used in Figure 4.3.2-1.

CNT_BYP Not A Containment Bypass Sequence

A containment bypass sequence with core damage results in a radiological release to the environment. Potential bypass sequences include the traditional Event V, Interfacing System LOCAs, and Main Steam Line Breaks.

CNT_FAL Containment Status At Core Damage

The success of containment at core damage maintains the radiological boundary. The failure of containment at, or before, core damage allows an early radiological release. The failure of the containment may result in the loss of RPV injection due to the physical interaction between the containment structure and the injection systems.

EVENT_TYP Event Type: For Containment Intact Or Failed At Core Damage

If the Containment is Intact At Core Damage, the event type is classified into one of the following: SBO, LOOP No HVAC, and OTHER TYPES. If the Containment is Not Intact At Core Damage, the event type is classified as: CRITICAL ATWS, LOOP & SBO, or OTHERS. Note that for SBO initiated core damage sequences where offsite power is recovered in the Plant Damage State event tree before core uncovery, the PDS grouping for the event type at core damage would be OTHER TYPES or OTHERS, as appropriate. Also, note that Critical ATWS defines ATWS sequences in which the core is not shut down by the standby liquid control system initially in the front-end event trees or after core damage by operator recovery action in the plant damage state event trees. Therefore, all the ATWS core damage sequences which are shutdown by early boron injection are classified as OTHERS.

SUPR_PL Initial Containment Heat Removal With Suppression Pool Cooling

LOOP With No HVAC sequences are subdivided into into those with initial containment heat removal and those without as: NOT AVAILABLE, and INIT SP COOLING. Initial Suppression Pool Cooling can impact the timing of RCIC failure, as well as the magnitude of the expected peak burn pressure if hydrogen generation or core concrete interaction subsequently occurs.

CNT_ISOL Containment Vent Isolated At RPV Failure

Containment isolation prior to fission product release maintains containment integrity and prevents the release of fission products following core damage. Failure of the containment isolation valves to isolate and maintain containment integrity is a function of the containment penetration valve isolation reliability, as well as of the sequence type.

At the onset of a SBO sequence the AC motor operated containment isolation valves may fail "as is", if the diesel generators do not load to the divisional safety buses. At Perry the only normally open system penetrating the containment boundary and interfacing with the containment atmosphere is the Fuel Pool Cooling and Cleanup discharge penetration. All other systems with motor operated containment isolation valves either have normally closed isolation valves or are closed systems which do not interface with the containment atmosphere in the normal operating lineup. The 9.5 inch equivalent diameter Containment Pools Return line provides a release path from the upper pool skimmers to the spent fuel pool skimmers and the fuel pool cooling surge tank vent. The backup hydrogen purge line is another containment penetration which may be open about half of the time during power operation. The 2 inch diameter backup hydrogen purge line routes drywell air from the drywell to the Annulus Exhaust Gas Treatment System in the intermediate building. Following a Loss of AC Power event. these open isolation valves will be manually closed locally in accordance with the off-normal instruction. When AC power is restored, the inboard and outboard isolation valves associated with each open penetration will close with high reliability with the re-energization of the safety related division 1 and 2 buses.

For SBO sequences where the containment is intact at core damage, the isolation status of the Fuel Pool Cooling and Cleanup penetration is modeled as ISOLATED or NOT ISOLATED. For the Not Isolated status, the containment is modeled as open throughout the remainder of the sequence.

For Non-SBO sequences when the containment is intact at core damage, the containment isolation valves are modeled as always isolated since the relative containment isolation failure probability is much lower than for SBO sequences. For Non-SBO sequences, AC power is available and the nuclear steam supply shutoff system is available to automatically isolate the containment. If an automatic isolation signal should fail to position the associated valve closed, the ERIS computer would alert the control room (as well as the Technical Support Center) of this abnormal alignment, and manual closure would occur.

For those sequences where the containment is not intact at core damage, the functional characteristic of Containment Isolation is not pertinent to the analysis and this is not used in the grouping logic.

INJ_F_TIM Reactor Pressure Vessel Injection Failure Time

If the core damage sequence Event Type is SBO or LOOP With No HVAC, the RPV Injection Failure Time is classified into the following time windows.

1) Initial Loss of All RPV Injection

2) Delayed RCIC Injection Loss

3) Delayed HPCS (or Firewater) Injection Loss

Time windows 1), 2) and 3) are utilized for SBO, and LOOP With No HVAC sequences.

Delayed loss of RCIC injection occurs when the suppression pool temperature approaches the Heat Capacity Temperature Limit - the RPV is depressurized and the RCIC turbine becomes unavailable due to low supply steam pressure. RCIC operation time varies for each sequence type.

SBO	2.8 - 4.2 hours
LOOP No HVAC - No Initial Supr Pl Clg	3.0 - 4.5 hours
LOOP No HVAC - Initial Supr Pool Clg	0 - 9.5 hours

PWR_R_TIM Offsite Power Recovery Time

Power restoration to either the division 1 or 2 safety-related bus from offsite power after core damage is classified into the following time windows.

1) PRIOR RPV FAIL After core damage and prior to RPV failure.

2) CNTMT LIMIT After RPV failure and prior to a containment capacity overpressure threshold.

3) NO RECOVERY No power recovery after core damage.

Generally recovery windows 1), 2) and 3) are utilized for the SBO sequences with the loss of all injection, and delayed RCIC injection failure sequences. Recovery windows 1) and 3) are used for SBO sequences with delayed HPCS (or Firewater) injection failure. If power is recovered before reactor vessel failure, then the AC powered injection systems can cool the core debris in-vessel and may prevent RPV failure. If power is recovered before containment failure due a gradual pressurization, then the RHR containment heat removal or venting motor-operated valves can be utilized to prevent containment failure.

SPRAY Containment Heat Removal With RHR Spray Loop

Containment heat removal ensures that the containment pressure will be maintained below the containment capacity overpressure threshold. Containment heat removal is accomplished with a RHR heat exchanger. Containment Sprays also remove aerosols and mitigate hydrogen combustion effects.

The Containment Heat Removal With RHR Spray Loop functional characteristic is included in the logic model whenever AC power is available or recovered at some time during the later part of the sequence progression. For those sequences where the containment is not intact or the containment is never isolated, this functional characteristic is not pertinent to the analysis and is not used in the grouping logic. In these cases the conservative assumption that RHR sprays are not available is made.

VENT Containment Heat Removal With Vent

Containment venting is accomplished with the Fuel Pool Cooling and Cleanup or Containment Spray Header vent pathways which can maintain containment pressure below the containment capacity threshold limit of 50 psig.

The Containment Vent functional characteristic is included in the logic model whenever Containment Heat Removal With Sprays is not successful and the containment is intact (and isolated) at core damage.

LAT_INJ Late In-Vessel Injection & Pedestal Cavity Supply

Successful late in-vessel injection can arrest core damage and can prevent RPV failure. In the Perry IPE Late In-vessel Injection is defined as those injection systems or alternate injection systems that are available and capable of injection when the RPV pressure is low and that initiate prior to core plate failure and continue to operate afterwards.

Conservatively those systems which be recovered after core plate failure are not considered in this analysis. Also it is postulated that if core damage occurs and the RPV is not depressurized, then late in-vessel injection is not successful. (However, these low pressure in-vessel injection systems may still be available to inject water into the RPV and into to the pedestal cavity through the breached vessel.) For the slow overpressure containment failure sequences in the PDS event trees, the late injection function includes the impact of containment failure when determining late injection system availability.

For those sequences where offsite power is recovered, Late In-Vessel Injection is modeled with the following systems: ECCS, Condensate Transfer Alternate Injection, Emergency Service Water Cross-Tie, the electric motor-driven firewater pump, the diesel-driven firewater pump and the Perry Township Fire Department Pumper.

For sequences where all AC power is lost, and offsite power is not recovered, Late In-Vessel Injection is modeled with the following systems: the diesel-driven firewater pump, and the Perry Township Fire Department Pumper.

For sequences where all AC power is available, Late In-Vessel Injection is modeled with the following systems: Condensate Transfer Alternate Injection, Emergency Service Water Cross-Tie, electric motor-driven firewater pump, diesel-driven firewater pump, and the Perry Township Fire Department Pumper.

RX_PRES RPV Depressurized During Core Damage

RPV depressurization during core damage and at the time of vessel failure effects in-vessel steam explosion phenomena and direct containment heating phenomena at vessel breach.

The RPV Depressurized During Core Damage functional characteristic is included on all Plant Damage State Grouping Logic branches.

4.3.2.3 Comparison With NUREG/CR-4551

The Perry IPE Plant Damage State grouping provides 11 questions regarding conditions prior to the initiation of core damage. The NUREG/CR-4551 Grand Gulf (Brown 1990) Accident Progression Event Tree includes 22 questions to describe conditions at the beginning of the accident. These include plant damage state grouping, evaluation of containment and drywell structural capabilities, whether the ignitors are operating and whether the containment is vented before core damage.

The Perry PDS grouping does not explicitly include a question about the availability of DC power, since this included in the PDS event tree for successful RPV Depressurization.

The Perry PDS grouping does not include a question on the number of SRVs which have failed to reclose, since core damage sequences with one or two open SRVs contribute about 3.3% of the total CDF. The Grand Gulf APET asks one question to determine if one or more SRVs fail to reclose.

The Perry PDS grouping provides one question for late in-vessel injection and pedestal cavity water supply, whereas the Grand Gulf APET includes seven questions regarding the status of injection systems and alternate injection systems.

The Perry PDS grouping provides one question regarding containment heat removal with an RHR spray loop that includes successful performance of the RHR heat exchanger. The Grand Gulf APET provides two questions: 1) Does RHR fail (heat exchangers not available), and 2) Are the containment sprays failed?

Both the Perry PDS grouping and the Grand Gulf APET include a question regarding the status of vessel depressurization.

The Perry PDS grouping indirectly includes the time of core damage for Station Blackout events events where the containment is intact at core damage by modeling the time of injection loss, and for those PDS groups for which the containment is failed at core damage; however this is not included to those PDS groups for which the containment intact at core damage and AC power is available. The Grand Gulf APET asks if core damage occurs in the short term (approximately 1 hour) or in the long term (approximately 12 hours).

The Perry PDS grouping includes a question to evaluate the isolation of the containment vent during Station Blackout sequences; if the vent is not isolated, then containment is modeled as not capable of pressurizing. The Grand Gulf APET includes two questions: 1) What is the level of pre-existing leakage of isolation failure, and 2) Is the containment not vented before core damage?

The Perry PDS grouping does not include the Grand Gulf APET question regarding the level of pre-existing suppression pool bypass through the drywell personnel hatch.

The Perry PDS grouping does not include the condition regarding the operator actuation of the Hydrogen Ignition System before core damage, but this action in included in the Accident Progression Event Tree logic structure as event 16 which is discussed in section 4.5.2.2.

4.3.2.4 Plant Damage State Event Trees

Plant Damage State Event Trees were developed by extending the Level 1 core damage event trees with the additional PDS functional characteristics defined above to fully evaluate all the "CD" core damage sequences, and pruning the "OK" no core damage branches. The Plant Damage Event Trees developed for Level 1/2 interface are included as Figures 3.1.4-1 through 19.

The Plant Damage State Event Trees for all initiating events include the following PDS functional headings: Containment Heat Removal With RHR Spray Loop, Containment Heat Removal With RHR Suppression Pool Cooling, Containment Heat Removal With Vent, Late RPV Depressurization, and Late In-Vessel Injection & Pedestal Cavity Supply. Late RPV Depressurization during core damage (and before RPV failure) is modeled on core damage sequences where this function has not been previously applied, and also applies a human interaction recovery to those sequences where the initial depressurization attempt has failed. The Late In-Vessel Injection function includes the dependency of containment failure on ECCS and alternate injection for those sequences where both RHR and venting containment heat removal fail which can result in failure of the operating injection system.

The Station Blackout Plant Damage State Event Trees include the following unique functional characteristics: Offsite AC Power Recovery Prior To RPV Failure, Offsite AC Power Recovery Prior To Containment Failure, and FPCC Isolation Shut By Operator Or By Diesel Operation. The time of RPV failure is determined by MAAP analysis when the core plate failure occurs. Similarly, the time of containment failure is determined by MAAP analysis when the containment pressure reaches the containment overpressure threshold limit of 50 psig. Offsite power is modeled as being restored to the plant by adjusting the RPV failure and containment failure times by the time required to align the electrical system from the control room, and to align the RPV injection or RHR containment heat removal system.

The ATWS Plant Damage State Event Trees include the human interaction recovery Late SLC Injection After Core Damage to account for successful reactor shutdown after nominal core damage has occurred when RPV injection is available.

4.3.3 PDS Frequencies and Dominant Sequences

Seventy five plant damage states are used to group the Level 1 (Plant Damage State) sequences. The Level 1 plant damage state event tree sequences were assigned to plant damage states using the Plant Damage State Grouping Logic diagram (Figure 4.3.2-1). Plant damage states with frequencies less than 10^{-8} were eliminated from the back-end containment analysis. The total back-end core damage frequency is 1.27×10^{-5}. The back-end core damage frequency 97.2% of the front-end core damage frequency of 1.31×10^{-5} due to binning cutoffs.

The Dominant Plant Damage States are listed in Table 4.3.3-1, by frequency with the individual sequences contributing above 1% listed. Fifteen PDS groups represent 95% of the core damage frequency. The most dominant PDS groups are 53, 56 and 73 which represent 71% of the core damage. A general discussion of these three PDS groups is provided below.

PDS group 53, represents Non-SBO sequences with the Containment Intact At Core Damage and successful Containment Heat Removal with the RHR Spray Loop, Late In-Vessel Injection, and RPV Depressurization During Core Damage. PDS group 53 has a core damage frequency of 4.44×10^{-6} (35%). This group has 7 dominant sequences above 1% of the core damage frequency. The most dominant sequence (18%) is an ATWS (successfully shutdown with the Standby Liquid Control) Loss of Power Conversion System, failure of the Motor Feedpump, and failure to inhibit ADS. This group also includes four other (shutdown) ATWS with Loss of PCS sequences with each contributing from 1 to 5%. A (shutdown) ATWS with Loss of Feedwater and failure to inhibit ADS contributes 2%. The other dominant sequence is a LOOP with loss of all injection which contributes 1%.

PDS group 56, represents Non-SBO sequences with Containment Intact At Core Damage and successful Containment Heat Removal with the Vent, Late Injection, and RPV Depressurization During Core Damage. PDS group 56 has a core damage frequency of 3.40×10^{-6} (27%). This group has 6 dominant sequences above 1% of the core damage frequency. The most dominant sequence (9%) is a Flood in Zone 13B with failure of all injection. The second sequence (6%) is a Loss Of Instrument Air with Loss of all injection. The third sequence (3%) is a Station Blackout sequence with successful recovery of offsite power before core uncovery and subsequent loss of all injection. The fourth sequence (2%) is a Flood in Zone F1D with loss of all injection. The remaining two sequences (1.6% each) are a Large LOCA and LOOP with failure of injection.

PDS group 73, represents sequences other than critical ATWS or LOOP & SBO with Containment Failed At Core Damage, unsuccessful Late Injection, and successful RPV Depressurization During Core Damage. PDS group 56 has a frequency of 1.22 x 10^{-6} (10%) This PDS has just 1 dominant sequence (8%), Loss of PCS and failure of containment heat removal with subsequent failure of all late injection.

7 of the 15 dominant PDS groups (63, 65, 66, 67, 69, 71 and 73) represent Containment Failed Before Core Damage sequences with a frequency of 2.86 x10^{-6} (22.5%). All the PDS groups associated with Containment Failed Before Core Damage contribute to a frequency of 2.90 x 10^{-6} (22.8%). This 22.8% Containment Failed Before Core Damage is comprised of: Critical ATWS (4.4%) with dominant PDS groups 63, 65 and 66, LOOP & SBO (4.3%) with dominant PDS groups 67 and, and OTHERS (14%) with dominant PDS groups 71 and 73.

5 of the 15 dominant PDS groups (1, 25, 32, 36 and 9) represent Station Blackout core damage sequences. The total core damage frequency in these five SBO PDS groups is 5.57 x 10^{-7} (4.4%). All the SBO PDS groups (1 through 47) comprise 9% of the core damage frequency.

4.4 CONTAINMENT AND DRYWELL FAILURE CHARACTERISTICS

4.4.1 Containment Failure Modes

The Perry Mark III steel containment is designed to withstand an internal pressure of 15 psig and an external differential pressure of 0.8 psi. The potential containment and drywell failure modes under severe accident loading conditions were analyzed by the Perry containment architect for the IPE containment evaluation (Gilbert/Commonwealth 1992).

The Perry Mark III steel containment failure modes are the following:

1. Dome Knuckle
2. Dome Apex
3. Cylinder
4. Personnel Air Lock
5. Equipment Hatch (Bolts)
6. Penetration P123
7. Penetration P205
8. Penetration P414
9. Anchorage, Steel
10. Anchorage, Concrete

Penetration failures (at P414, P205 and P123) are most likely and are expected to commence progressively in size with increasing pressure after leakage initiation. Failure at all containment failure modes, other than the steel and concrete anchorage, results in a release into the shield building through a failure located above the suppression pool. The less likely failure of the containment anchorage can result in gross failure of the containment vessel at the basemat foundation which can impact RPV injection line integrity and suppression pool inventory, as well as provide a radiological release pathway from the drywell to the environment with no pool scrubbing.

The results of the Gilbert/Commonwealth containment failure modes analysis are provided in Table 4.4.1-1 and are summarized below. For a more complete discussion refer to the Perry IPE Containment Capacity Analysis included as Appendix H.1.

The Dome Knuckle and Dome Apex failure modes result in rupture of the steel shell and consequential loss of pressure through the rupture opening. The Cylinder failure mode would result in gross rupture with a rapidly progressing failure area and an associated rapid depressurization of the containment.

The Personnel Air Lock failure mode results in leakage through the inflatable seals for failures occurring below the median failure pressure and rupture through the blown out seals for failures occurring above the median failure pressure.

The Equipment Hatch failure mode results in leakage through the seals for failures occurring below the median failure pressure and gross rupture (from hatch bolt failure) for failures occurring above the median failure pressure.

Penetration failure modes (associated with penetration P123 - RCIC Pump Discharge and RHR Spray, penetration P205 - Fuel Transfer Tube, and penetration P414 - Feedwater) commence with a small leakage area of about 5 square inches which increases with containment pressure to an area of about 30 inches.

Anchorage failure modes (associated with the 288 steel anchors and the adjacent basemat foundation concrete wedge) are modeled simplistically as resulting in gross failure of the containment vessel. This failure would be sudden with no yielding and could be progressive around the containment.

Failure of the Perry Mark III containment would result in pressurization of the shield building annulus which has silicone foam seals around pipe penetrations and neoprene expansion joints between building gaps. Failure of the low pressure rated silicone foam seals could route the steam release and radionuclides into the LPCS and HPCS pump rooms.

The Containment Capacity Analysis characterized the expected type of failure for each failure mode as: leakage, rupture, or gross rupture. Leakage was defined as an area of approximately 0.1 square feet which results in slow depressurization. Rupture was defined as an area from 0.1 to 7.0 square feet. Gross rupture was defined as an area of greater than 7.0 square feet. The Expected Types of Containment Failure are summarized in Table 4.4.1-2.

The [illegible] containment evaluation identified the anchorage failure modes as [illegible] and generally classified the containment failure type as either [illegible] or penetration. In the Perry level 2 IPE, penetration containment failure includes piping penetrations, the personnel airlock, and the equipment hatch. Also, because of similar failure characteristics (i.e., failure occurs in the containment gas space above the suppression pool) the dome knuckle, dome apex and cylinder failure modes are also combined with the penetration modes.

4.4.2 Drywell Failure Modes

The Perry [illegible] III drywell is designed to withstand an internal pressure of 30 psid and and external pressure of 21 psid.

The drywell failure modes are:

1. Drywell wall
2. Drywell roof
3. Drywell head
4. Drywell equipment hatches
5. Drywell personnel airlock

The analysis of the capacities of the above failure modes determined that the most significant is the drywell head. Internal pressure impacts all 5 failure modes; whereas, external pressure impacts failure modes 3 thru 5.

The results of the Gilbert/Commonwealth drywell failure modes analysis are provided in Table 4.4.2-1.

4.4.3 Containment Overpressure Fragility

A composite containment fragility curve was calculated using the methodology described in IDCOR Technical Report 10.1 Appendix B, Methodology Used to Generate Fragility Curves to Describe Containment Overpressure Failure Modes Suitable for Use in Probability Risk Assessment (IDCOR 1983).

The Containment Overpressure Fragility curve is provided as Figure 4.4.3-1. The Containment Overpressure Fragility curve exhibits a 1% probability of failure at 50 psig, a 5% probability of failure at 53.5 psig, a 50% probability of failure at 64.3 psig, and a 99% probability of failure at 79 psig. The containment pressure of 50 psig (with a 1% probability of failure) is defined to be the containment capacity overpressure threshold limit and is used to conservatively estimate the time of containment gradual overpressure failure in event trees and plant damage state event trees.

4.4.3.1 Slow Overpressure Conditional Probability of Anchorage Failure

The conditional probability for anchorage failure given slow overpressurization containment failure is 0.15. The Containment Capacity Analysis characterization of penetration failure states that early containment penetration failures which occur below the median failure pressure of each failure mode would initiate as a small leak that would breach the containment boundary but would not terminate the slow pressure increase. Containment failure leakage size is expected to increase to a larger leakage size in proportion to the increase in containment pressure toward the median failure pressure until the leakage from containment balances the steam generation rate. The Perry IPE calculated a best-estimate weighted average of the conditional probability of anchorage failure over the pressure range for the most likely containment failure modes (the 3 piping penetrations) of 70 to 80 psig where the expected leakage size of about 0.1 square foot was adequate to vent decay heat.

4.4.3.2 Fast Overpressure Conditional Probability of Anchorage Failure

The fast overpressure conditional probability of anchorage failure was examined to address the additional likelihood of anchorage failure due to fast pressure rises from uncontrolled hydrogen combustion for peak pressures above 79 psig, where 99% containment failure is expected to occur. When the hydrogen ignition system is unavailable and uncontrolled hydrogen combustion overpressurization may occur, containment failures up to the size of a gross rupture (characterized with a lower bound of 7 square feet) do not significantly reduce the expected peak pressure rise. Expected burn pressures from hydrogen combustion would continue to individually challenge the concrete anchorage (with a 92 psig median failure pressure) and the steel anchorage (with a 135 psig median failure pressure) even if containment failure occurs at other failure locations. The expected hydrogen combustion peak pressures during a severe accident (without continuously available hydrogen ignitors) can range upward to 200 psig. The Fast Overpressurization Failure Mode is applied to hydrogen combustion containment failures to determine the conditional probability of anchorage containment failure as a function of the peak burn pressure. For a complete discussion of hydrogen combustion analysis refer to sections H.3.3, H.3.6 and H.3.10 in Appendix H.3, PNPP IPE Accident Progression Event Tree Description.

The Fast Overpressure Conditional Probability of Anchorage Failure curve is provided as Figure 4.4.3-2. The Fast Overpressure Conditional Probability of Anchorage Failure curve exhibits a 26% conditional probability of anchorage failure at 80 psig, a 50% conditional probability of anchorage failure at 92 psig, and a 95% conditional probability of anchorage failure at 125 psig.

4.4.4 Containment Overpressure Failure Impact on RPV Injection

The containment overpressure failure impact on RPV injection is modeled in the front-end event trees and in the back-end accident progression event tree. The front-end event trees include the functional heading, Core Not Vulnerable To Damage, after the functional headings: Long-Term Containment Heat Removal - RHR, and Long-Term Containment Heat Removal - Venting. Given failure of Long-Term Containment Heat Removal, the functional event tree heading, Core Not Vulnerable to Damage, defines the susceptibility of the core to damage as a result of containment slow overpressure failure degrading the RPV injection system which was initially modeled as successfully maintaining adequate core cooling. The accident progression event tree similarly evaluates the impact of containment hydrogen combustion fast overpressure failures prior to RPV failure, and at RPV failure on the RHR containment spray and RPV injection systems.

The Perry IPE assessed the impact of containment failure from slow overpressurization before core damage on the operating RPV injection system, by accounting for the following factors that can degrade RPV injection sources: containment failure mode, injection system piping disruption, and injection system degradation due to environmental conditions. Special decomposition event trees are used to quantify the event tree heading, Core Not Vulnerable To Damage, for the following four RPV injection cases (shown in Figures 4.4.4-1 through 4.4.4-4): HPCS, low pressure ECCS, Injection Through the Feedwater Line, and Injection (from sources) Outside the Auxiliary Building.

The event tree for the Quantification of Core Not Vulnerable To Damage starts with the Containment slow overpressure conditional probability of 0.15 for anchorage failure and 0.85 for penetration failure.

The likelihood of injection system piping being disrupted due to anchorage failure was assigned a probability of 0.9 since there is a small likelihood that anchorage failure may occur without gross containment movement. The likelihood of injection system piping being disrupted due to containment penetration failure for the case of Injection Through the Feedwater Line was assigned a probability of 0.05 since the failure of the feedwater penetration P414 has a small likelihood of causing structural disruption.

The likelihood of RPV injection system degradation due to environmental conditions was assessed for each of the four cases. HPCS failure due to the steam release through a penetration failure was assigned a probability of 0.5. Low Pressure ECCS failure due to steam release through a containment penetration failure was characterized as somewhat less likely with a probability of 0.3, since only the LPCS pump room has penetrations that lead directly to the shield building annulus. However, low pressure ECCS failure due to steam release from anchorage failure was characterized as having a higher failure potential and assigned a probability of 0.5. Failure of Injection Through The Feedwater Line was not considered credible due to the remote location of the associated condensate and feedwater pumps and assigned a probability of 0.0. Also, to account for structural failure of Injection Through The Feedwater Line associated with penetration failures in close proximity a probability of 0.05 was assigned. Failure of Injection Outside the Auxiliary Building was assigned a probability of 0.0 due to the remote location of the firewater pumps.

The Cv, Core Not Vulnerable To Damage failure probabilities were calculated for the following RPV injection failure cases given failure of containment.

Cv - HPCS	=	0.57
Cv - Low Pressure ECCS	=	0.40
Cv - Alt Injection Thru the Feedwater Line	=	0.18
Cv - Alt Injection Outside the Auxiliary Bldg	=	0.14

4.4.5 Drywell Overpressure Fragility

A Drywell Overpressure Fragility curve was developed to support the evaluation of the impact of wetwell hydrogen combustion overpressurization on drywell integrity. This curve was developed using the Perry estimated mean failure pressures for external loading and the drywell failure mode uncertainties applied in the 1985 Kuosheng Nuclear Power Station Probabilistic Risk Assessment (ROC AEC).

The Drywell External Overpressure Fragility curve is provided as Figure 4.4.5-1. The Drywell External Overpressure Fragility curve exhibits a 5% probability of failure at 47.5 psid, a 50% probability of failure at 70.8 psid, and a 95% probability of failure at 95 psid.

4.5 CONTAINMENT EVENT TREES

4.5.1 Containment Severe Accident Analysis

The Perry IPE back-end analysis of a Mark III containment is developed to address the recommendations provided in the Generic Letter 88-20, Appendix 1, "Guidance of the Examination of Containment System Performance." To best transpose the severe accident analysis phenomenological framework and the associated quantification data of the Grand Gulf NUREG/CR-4551 evaluation (Brown 1990) and model the many dependencies associated with containment loading mechanisms such as steam gen ration, hydrogen generation and combustion, and the subsequent variations in pr sure and temperature within the containment, the generalized event tree proce or, EVNTRE was selected. EVNTRE (Griesmeyer 1989) was developed at Sandia National Laboratories for use in probabilistic risk analyses of severe accident progression for nuclear power plants, and was used in the Grand Gulf evaluation and which supported the final NUREG-1150 report (NRC 1989).

The Perry Accident Progression Event Tree (APET) is a concise version of the Grand Gulf NUREG/CR-4551 APET. The Perry APET consists of 68 questions or events that address four general time periods of severe accident analysis.

1) Plant Damage State - conditions prior to the initiation of core damage. 11 events identify important plant damage state functional characteristics. The Plant Damage State Grouping Logic developed in section 4.3.2 is applied in the APET as the initial plant conditions.

2) Early - accident progression from the beginning of core damage to just before vessel breach or Reactor Pressure Vessel (RPV) failure. 18 events analyze accident progression with regard to in-vessel cooling, hydrogen combustion before RPV Failure, and the impact of early containment failure on injection systems and RHR spray piping.

3) Intermediate - accident progression from immediately before RPV failure to the time of significant Core Concrete Interaction (CCI). 24 events analyze accident progression from just before RPV failure with regard to drywell failure, hydrogen combustion at or near the time of RPV failure, pool bypass before or near the time of RPV failure, and the impact of containment failure on injection systems and RHR spray piping.

4) Late - accident progression during CCI. 15 events analyze accident progression after RPV Failure with regard to pedestal failure from concrete erosion, late hydrogen combustion and slow overpressurization (from steam and non-condensible gases) containment failure, and late pool bypass from hydrogen combustion and other processes.

The 68 questions or events included in the Perry APET are listed in Table 4.5.1-1. These events are discussed in Section 4.5.2.2 along with the branch definitions. The Perry APET program input data file is provided in Appendix H.2, PNPP IPE APET Program Input Data File. A detailed description of the how the branch probabilities were assigned to the APET events is presented in Appendix H.3, PNPP IPE APET Description.

4.5.1.1 Grand Gulf NUREG/CR-4551 Comparison

The Grand Gulf NUREG/CR-4551 APET consists of 125 questions covering the same four general time periods:

1) Initial. 22 events describe the conditions at the beginning of the accident including plant damage state grouping, evaluation of containment and drywell structural capacities, whether the ignitors are operating and whether the containment is vented.

2) Early. 25 events for this period consider the status of important systems (coolant injection, AC power, hydrogen ignition system, etc.), the composition of the containment and drywell atmosphere, hydrogen burn phenomena (ignition and loads), and the containment and drywell response to the containment loads.

3) Intermediate. 41 events evaluate this intermediate period. The potential for in-vessel core damage arrest (no vessel failure) is addressed in this time period. The majority of these questions address the loads accompanying vessel breach and the containment and drywell structural response to these loads. Hydrogen combustion is considered at the time of vessel breach and during the time period before significant CCI commences.

4) Late. 27 events evaluate containment failure from hydr n combustion and late over-pressurization, as well as drywell fail rom hydrogen combustion and reactor pedestal failure.

4.5.2 Summary of Event Tree Structure

4.5.2.1 Perry Summary Containment Evaluation Tree

The Perry APET analysis structure is graphically described with two figures: the previously described the Plant Damage State Grouping Logic, Figure 4.3.2-1, and by the Summary Containment Event Tree, Figure 4.5.2-1. These figures provide a transparent graphical summary of the entire containment evaluation analysis framework. The Summary Containment Event Tree provides a general summary of the APET analysis framework over the three accident time periods: early, intermediate and late. The Summary Containment Event Tree is constructed to allow the various accident progression pathways to be easily understood since the detailed APET, because of its size, cannot be graphically presented. The Perry Summary CET is a graphical enhancement aid only, and is not used for quantification.

The Summary CET contains the most important events considered in the Level 2 analysis and summarizes possible paths that an accident sequence may progress along. The Summary CET headings are the most important "events" which can lead to significantly different outcomes in the sequence progression where the major outcomes of interest relate to ti ng and mode of containment failure and the atmospheric release of radionuclide The Summary CET events were chosen to:

1) represent the uncertainties in physical phenomena (e.g. hydrogen combustion containment loading)

2) assess operator recovery, and mitigation actions and

3) assess consequential failure of important systems given the occurrence of specific physical phenomena (e.g. hydrogen burns) or as a result of the general severe accident environment.

The 10 Perry Summary Containment Event Tree Headings are described below in section 4.5.2.2, Accident Progression Event Description. These events are also contained as distinct headings in the Perry IPE APET. The number for these events in the Perry APET is shown along with the APET events which these events are dependent.

Each Summary Containment Event Tree Heading is evaluated by a Decomposition Event Tree (DET) that includes the important APET events. (Reference DET Figures 4.5.2.2.1-1 through 4.5.2.2.10-1.) The DET logic structure depicts a general characterization of the APET and is provided to enhance the severe accident technological transfer.

4.5.2.2 Accident Progression Event Description

APET Events 1 through 11 are the Plant Damage State Grouping Logic parameters. The Plant Damage State parameter APET Events are fully described in section 4.3.2, PDS Grouping Logic and PDS Characteristics.

APET Events 12 through 68 are presented with a summary discussion below. A complete description of the APET Events is provided as Appendix H.3, PNPP IPE APET Description.

4.5.2.2.1 Debris Cooled In-Vessel (APET Event 15)

This event assesses the probability that the damaged core can be cooled in-vessel and vessel failure prevented. To be coolable, a source of in-vessel injection with flow in excess of that required to remove decay heat must be available. If the reactor has not been shutdown by Standby Liquid Control during ATWS sequences, the current Plant Emergency Instruction (PEI) does not ensure containment integrity will be maintained, and late in-vessel cooling is assumed not possible. When the debris is cooled in-vessel the threat to containment integrity will be reduced since: 1) the potentially large containment and drywell loads resulting from mechanisms at vessel failure (fuel coolant interaction, and vessel blowdown) will not be present, and 2) the production of combustible gases (H_2 and CO) and other non-combustible gases (CO_2) resulting from debris/concrete attack will be avoided.

Figure 4.5.2.2.1-1, Debris Cooled In-Vessel Decomposition Event Tree (DET) shows the general characterization of this portion of the APET.

This APET event, Debris Cooled In-Vessel, is dependent on the following Events in the Perry APET.

APET Event 12. Late RPV Low Pressure Injection Available

Three Branches: Summary Branches:

1) Water Injection Available to the RPV — WATER INJECTION

2) No Injection Available — NO INJECTION
3) Critical Reactivity Condition — CRITICAL ATWS

This event assesses if low pressure water injection to the reactor vessel can be established subsequent to core damage, but prior to reactor vessel failure. It should be noted, that CRD injection is not credited as a source of late injection. A separate branch is developed for the Critical (non-shutdown) ATWS sequences with failure of Standby Liquid Control.

APET Event 13. RPV Depressurized During Core Damage

Two Branches:	Summary Branches:
1) RPV Depressurized During Core Damage	LOW PRESSURE
2) RPV Not Depressurized	HIGH PRESSURE

In-vessel cooling requires that the reactor pressure vessel be depressurized to permit the low pressure Late-Injection systems to provide flow to the reactor core.

APET Event 14. Debris Mass Molten at RPV Breach

Two Branches:	Summary Branches:
1) Large Mass Molten Debris in Lower RPV	LARGE DEBRIS
2) Small Mass Molten Debris in Lower RPV	SMALL DEBRIS

The mass of molten debris in the reactor vessel lower head is a key factor in determining the probability of successful in-vessel cooling.

APET Event 15. Debris Cooled In-Vessel

Two Branches:	Summary Branches:
1) Debris Cooled In-Vessel	COOLED INVESSEL
2) Debris Not Cooled	NOT COOLED

The Debris Cooled In-Vessel event determines the probability that the debris is cooled in-vessel (and reactor vessel failure is prevented). The In-Vessel coolability probability is determined as a function of debris mass, and late injection availability. If the reactor has not been shutdown during ATWS sequences and containment is failed at core damage, then late in-vessel cooling is assumed not possible.

A special case is [illegible]ded to consider the impact of an ATWS Alternate Shutdown modification [illegible] the following two elements.

1) A PEI change to direct RPV Power/Level be controlled just above the Minimum Steam Cooling Water Level when containment pressure control is challenged. The reduced reactor power will enable successful containment heat removal with venting, and reactor steady state operation when the RPV injection is through the feedwater spargers. RPV injection through the feedwater spargers when the RPV water level is deliberately lowered will provide a high rate of heat

transfer in the shroud annulus, and reduce the subcooling of the core inlet water such that limit cycle reactor power oscillations will be prevented.

2) A plant modification to add alternate shutdown capability.

4.5.2.2.2 Mode Of Containment Failure Before RPV Failure (APET Events 24,25)

These events assess the probability that gradual steam overpressure or early hydrogen combustion results in one of several possible modes of containment failure.

A description of the Perry containment failure modes is provided in Appendix H.1, Perry Nuclear Power Plant Individual Plant Examination Containment Capacity Analysis. This containment capacity analysis is fully discussed in section 4.4, Containment and Drywell Failure Characteristics. The failure modes which make the largest contribution to the probability of failure (i.e., those with the lowest strengths) are as follows:

1) Several small/intermediate size penetrations
2) Containment equipment hatch
3) Basemat anchorage of the cylinder containment shell

The most important failure modes identified in the containment capacity analysis were the failure of the fuel transfer tube penetration and failure of the containment shell anchorage in the basemat. The penetration failures are characterized as being a leak type failure (nominal leakage area of 0.1 square foot or less) up to pressures of about 64 psig, and of rupture type (nominal leakage area of between 0.1 and 7.0 square feet) for pressures in excess of 64 psig. The failure of the containment shell anchorage would involve gross failure and the opening of a gross rupture area (greater than 7 square feet).

The equipment hatch and penetration containment failure modes are combined together, since for these (relatively non-energetic) failure modes, the releases are into the containment shield building annulus gas space above the annulus concrete. The anchorage failure mode may be quite energetic with the failure location below the suppression pool water level and with the possibility of significant gross movement of the containment shell accompanied by loss of suppression pool water and/or disruption of ECCS suction and injection lines which penetrate the containment steel liner pressure boundary. Additionally, failure of the anchorage would result in flooding of the annulus with the potential to flood the LPCS and HPCS pump rooms.

For a Mark III containment, the possibility of hydrogen (deflagration and detonation) combustion occurring prior to reactor vessel failure with a sufficiently high pressure to fail the containment (or drywell) must be examined for sequences where the hydrogen ignitors are either not available during Station Blackout sequences, and for recovery of AC power sequences where the plant emergency instruction inhibits ignitor operation when the hydrogen concentration cannot be determined. The human interaction, failure to initiate the Hydrogen Ignition System, is included in sequences when AC power is available. Very early hydrogen burns with sufficiently high intensities to seriously challenge containment integrity are considered not credible for sequences with the Hydrogen Ignition System (HIS) continually on based on test data.

Figure 4.5.2.2.2-1, Mode of Containment Failure Before RPV Failure DET, shows the general characterization of this portion of the APET. This DET provides additional information regarding hydrogen combustion. The containment hydrogen concentration is provided as a function of containment steam concentration and fraction of zirconium inventory reacted in-vessel. Also, this DET uses three informational headings after the Large H2 Burn to show the theoretical Adiabatic Isochoric Complete Combustion Pressure, the Burn Efficiency Factor, and the summary Expected Peak Burn Pressure.

The Mode of Containment Failure Before RPV Failure is determined by APET events 24 and 25, which are dependent on the following Events in the Perry APET.

APET Event 16. Hydrogen Ignition System Available

Two Branches:	Summary Branches:
1) Hydrogen Ignition System Off	HIS OFF
2) Hydrogen Ignition System On	HIS ON

With the hydrogen ignition system in operation it is assumed that controlled hydrogen combustion will preclude the build-up of hydrogen concentrations whose combustion would threaten containment integrity. Since the hydrogen ignition system requires AC power, station blackout results in loss of the ignitor system. When AC power is available, the human interaction, operator fails to initiate hydrogen ignition system, is modeled to evaluate this action which is initiated when the RPV water level decreases below Level 1, 16.5 inches above the top of the active fuel.

APET Event 17. Containment Vent Isolated Before RPV Failure

Two Branches:	Summary Branches:
1) Containment Vent Isolated Before RPV Failure	ISOLATED
2) Containment Vent Not Isolated	NOT ISOLATED

This event summarizes whether the containment vent path is isolated before RPV failure for SBO sequences. The most likely mechanism for loss of isolation at the Perry plant is for the normally open Fuel Pool Cooling and Cleanup (FPCC) vent path to fail to isolate following a station blackout. The FPCC path is utilized in the Plant Emergency Instruction for containment venting. AC power is required to automatically isolate this containment penetration. If both motor operated isolation valves fail to close, manual isolation is estimated to be performed within 90 minutes. This human interaction is modeled as operator fails to close FPCC Outboard Isolation - G41-F145.

APET Event 18. Mode Of RHR Spray Operation Early

Three Branches:	Summary Branches:
1) Controlled Spray Operation	CONTROLLED
2) Normal Spray Operation	DESIGN COOLING

3) Sprays Not Available NO SPRAY

This event summarizes whether the RHR sprays are available for containment heat removal when hydrogen combustion may occur. This event determines when RHR spray is available from the Plant Damage State initial condition, containment heat removal with RHR Spray Loop, and further considers whether offsite power is available during SBO sequences.

This event further allows for the assessment of a Plant Emergency Instruction enhancement regarding controlled cooling spray operation. Controlled cooling spray operation mode is directed at controlling the containment spray cooling rate (by regulating the RHR heat exchanger bypass flow and/or the RHR spray flow) such that the containment atmosphere steam concentration would be optimally maintained during the absence of hydrogen igniter operation to prevent or mitigate hydrogen deflagrations and detonations. When containment pressure is above 30 psig, controlled spray can prevent deflagrations by maintaining the containment steam concentration above the steam inerting limit for hydrogen combustion (55 volume percent) and controlling containment pressure below the emergency procedure pressure limit of 40 psig. When containment pressure is < 30 psig, controlled spray can mitigate the expected deflagration burn pressure. The controlled cooling spray operation mode is not evaluated in the base case APET.

APET Event 19. Containment Steam Concentration Before RPV Failure

Six Branches:	Summary Branches:
1) 0-15 Volume Percent Steam	0-15%
2) 15-25 Volume Percent Steam	15-25%
3) 25-35 Volume Percent Steam	25-35%
4) 35-45 Volume Percent Steam	35-45%
5) 45-55 Volume Percent Steam	45-55%
6) > 55 Volume Percent Steam	> 55%

This branch assesses the containment steam concentration during core damage before RPV failure. The probability of hydrogen burn ignition and the efficiency of the burn (considered in subsequent events) are dependent upon the branch taken under this heading. The containment steam concentration is a function of the mode of spray operation, the sequence type, the time of injection failure and whether the containment is intact at core damage. The steam concentration regime probabilities used for the various cases are estimated using MAAP results for each case and considering the variability of suppression pool temperature (since assuming a high initial temperature may be non-conservative for hydrogen combustion during SBO).

APET Event 20. Fraction Zirconium Inventory Reacted In-Vessel

Three Branches:	Summary Branches:
1) 33% Core Inventory Zirconium Oxidized	33%
2) 22% Core Inventory Zirconium Oxidized	22%
3) 11% Core Inventory Zirconium Oxidized	11%

The fraction of zirconium reacted in-vessel is used to determine the concentration of hydrogen in containment during core damage before RPV failure. Three discrete regimes are used to represent the range of in-vessel zirconium oxidation. These regimes are representative of the amounts estimated in the NUREG/CR-4551 Grand Gulf analysis (Brown 1990). The Perry IPE fraction of zirconium inventory oxidized probabilities are estimated using specific MAAP calculations for a spectrum of sequences (considering local blockage and no channel blockage).

APET Event 21. Small Burns At Low H2 Concentration

Two Branches:	Summary Branches:
1) No Small Burns Occur	NO SMALL BURNS
2) Small Burns Occur	SMALL BURNS

Burns which are ignited at low hydrogen concentrations (< 8%) will not threaten the containment integrity. For sequences with the hydrogen ignitors available it is assumed that the hydrogen will always burn at low concentrations or as diffusion flames. Furthermore, it is assumed that once one burn has ignited that all subsequent burns (through the time of vessel failure) will be small. This latter assumption, based on results from Grand Gulf (Brown - 1990), assumes that a burn will ignite transient combustibles in containment which will serve as an ignition source for future burn events without the HIS operable. (Since the Perry containment is maintained in accordance with a strict cleanliness standard, the impact of this transient combustible assumption is examined in the APET parameter sensitivity analysis in section 4.6.2.) For sequences without the HIS the probability of igniting a small burn is a function of the sequence type (i.e., SBO or not SBO), the time of injection failure, and whether AC power recovery occurs prior to RPV failure for SBO sequences. In addition, a steam concentration in excess of 55 volume percent is assumed to inhibit ignition of small burns.

APET Event 22. Large H2 Burn During Core Damage

Two Branches:	Summary Branches:
1) No Large Burns Ignited	NO BURN IGNITED
2) Large Burn Ignited	LARGE BURN IGNITED

This event assesses whether a large burn is ignited in the containment during core damage. Ignition of a large burn is assumed to be precluded by operation of the HIS, by ignition of a small burn or by a steam concentration in excess of 55 volume percent. The probability of igniting a large burn is a function of the containment steam concentration and hydrogen concentration. In addition to evaluating the probability of igniting a large burn this event sets the values of two EVNTRE parameters. Parameter 1 is the peak containment pressure for the burn and Parameter 2 is the pre-existing pressure in containment prior to the burn. These parameters are used in subsequent events to estimate the probabilities of containment and drywell failure given that a large burn has occurred. The values of these parameters, like the ignition probability, are a function of the containment steam and hydrogen concentrations.

APET Event 23. Hydrogen Detonation Containment Failure Before RPV Failure

Two Branches:	Summary Branches:
1) H2 Detonation Containment Failure	DET CF
2) No Detonation Failure	NO

Given that a large hydrogen burn is ignited, this event assesses the probability the burn transitions to a detonation and the detonation results in containment failure. If a large burn was not ignited then a detonation cannot occur. Based on Grand Gulf (Brown - 1990) it is assumed that the containment is inert to detonations if the steam concentrations is above 35 volume percent or if the containment hydrogen concentration is less than 12 volume percent. The probability of a hydrogen burn failing containment is a function of the containment hydrogen and steam concentrations, whether AC power recovery occurs, and whether sprays are available.

APET Event 24. Containment Failure Before RPV Failure

Two Branches:	Summary Branches:
1) Containment Failure Before RPV Failure	FAILURE
2) No Containment Failure	NO FAILURE

This event assess whether containment failure occurs before core damage due to gradual steam overpressure or during core damage as a result of a hydrogen combustion. If the containment is failed at core damage, then this is sorted as containment failure. If a hydrogen detonation containment failure occurred, then this is sorted as containment failure. If a large hydrogen burn occurred, then this event compares the expected peak containment pressure for the burn (Parameter 1) with the containment fragility curve and determines the split fraction of containment failure.

APET Event 25. Mode of Containment Failure Before RPV Failure

Two Branches:	Summary Branches:
1) Anchorage Containment Failure	ANCHORAGE
2) Penetration-Dome Containment Failure or No Containment Failure	PENET-DOME/NO CF

This event determines the probability of anchorage containment failure given containment failure due to gradual overpressure or deflagration, and assigns detonation containment failure sequences and no containment failure sequences to the second branch.

The IPE evaluation of containment failure modes analysis determined that two appropriate classifications of failure mode for the APET are: anchorage and penetration. Since detonation failures are expected to result in containment shell failure in the upper containment dome, this suffix has been added to the general penetration containment failure class.

Note that the above two categories of containment failure modes can be expanded into three containment failure modes (anchorage, penetration-dome, and no containment failure) by including the previous event, Containment Failure, in an event case logic structure. Thus, APET Event 24 (Containment Failure Before RPV Failure) and Event 25 (Mode of Containment Failure Before RPV Failure) are referenced as assessing the probability of the possible modes of containment failure in the introduction of this section.

For gradual steam overpressure containment failure before core damage, the estimated conditional probability of 0.15 is assigned for anchorage failure. All detonation failures are considered most likely to occur in the dome region and are sorted to the penetration-dome or no containment failure category. For sequences where the containment has failed by a hydrogen burn the expected peak containment pressure from the burn is used to estimate split fraction of anchorage failure. The individual fragility curves for the dominant failure modes are used to determine the conditional probabilities for each failure mode as a function of the failure pressure.

4.5.2.2.3 <u>Injection & Spray Failure Due To Containment Failure Before RPV Failure</u> (APET Event 29)

This event assesses the probability that containment failure causes the loss of all in-vessel injection and the failure of the RHR containment spray system for sequences where hydrogen combustion and ATWS gradual steam overpressure occur. The PDS Event Trees include the impact of gradual overpressure containment failure on Late Injection for all event types except Critical ATWS.

The containment sprays represent an effective fission product mitigation feature which can significantly limit atmospheric releases of radionuclides. Containment sprays will be particularly important for sequences where suppression pool bypass has occurred, since the sprays then represent the only remaining major engineered safety system which can significantly mitigate radionuclide releases.

The mechanisms associated with containment failure which may cause failure of in-vessel injection and/or the containment spray system include:

1) Movement of the containment shell and failure of in-vessel injection lines at/near penetrations in the shell containment.

2) Release of hot containment gases and radiation into regions of the auxiliary and intermediate buildings, or to outside the plant with critical injection components which may impair equipment and prevent operator access to align and maintain running equipment.

3) Loss of suppression pool water (as result of anchorage failure).

Figure 4.5.2.2.3-1, Injection & Spray Failure Due To Containment Failure Before RPV Failure DET shows the general characterization of this portion of the APET.

The APET event, Injection & Spray Failure, is dependent on the following Events in the Perry APET.

APET Event 26. Containment Failure Before RPV Failure

Impact on ECCS Injection & Spray Piping

Two Branches:	Summary Branches:
1) No Piping Failure	NO FAILURE
2) Failure of ECCS Injection & Spray Piping	FAILURE

This event assesses the probability that either the dynamic forces or movement of the containment which occur at containment failure are sufficient to disrupt the injection and spray system piping. Disruption of this piping is expected to be a serious threat for containment anchorage failure.

APET Event 27. Containment Failure Before RPV Failure
Impact On ECCS Injection & RHR Spray Motors

Two Branches:	Summary Branches:
1) No Failure	NO FAILURE
2) ECCS Injection & Spray Motor Failure	FAILURE

This event assesses the probability that leakage of water, steam or hot gases from containment into the auxiliary building which occur at containment failure cause failure of the injection and/or spray system motors.

APET Event 28. Containment Failure Before RPV Failure
Steam or Radiation Release
Impact On Firewater Injection

Two Branches:	Summary Branches:
1) No Failure	NO FAILURE
2) Failure of Firewater Alternate Injection	FAILURE

This event assesses the probability that leakage of steam or radionuclides from containment failure or containment venting into the plant or outside the plant will limit personnel access to the firewater system and result in failure to perform required local manual actions to initiate, or to assure continued operation of firewater alternate injection.

APET Event 29. Injection & Spray Failure Due To Containment Failure
Before RPV Failure

Two Branches:	Summary Branches:
1) Injection & RHR Spray Failure	INJ & SPRAY FAILURE
2) No Failure	NO FAILURE

Event 29 event summarizes the results of the prior three events in the APET.

4.5.2.2.4 Drywell Failure At/Near RPV Failure (APET Event 39)

A number of possible mechanisms have been identified which may lead to drywell failure at the time of RPV failure. These include alpha mode steam explosion failures (which by definition fail the drywell and containment), in-vessel steam explosions which fail the lower RPV head, overpressure failure of the pedestal wall, ex-vessel steam explosion in the pedestal cavity, drywell overpressure failure, and a large hydrogen burn in containment.

Loss of drywell integrity results in bypass of the suppression pool and removes the suppression pool from the radionuclide release pathway to the environment.

Figure 4.5.2.2.4-1, Drywell Failure At/Near RPV Failure DET, shows the general characterization of this portion of the APET.

This APET event, Drywell Failure At/Near RPV Failure, is dependent on the following Events in the Perry APET.

APET Event 30. Alpha Mode Steam Explosion Drywell and Containment Failure

Two Branches:	Summary Branches:
1) Alpha Mode Failure	ALPHA
2) No Alpha	NO ALPHA

This event assesses the probability that an in-vessel steam explosion occurs with sufficient energy to rupture the upper head of the RPV and create a missile with sufficient energy to fail the drywell and containment. The probability of an in-vessel steam explosion being triggered has been shown to be dependent on whether the RPV has been depressurized.

APET Event 31. Mode of In-vessel Steam Explosion Bottom Head RPV Failure

Four Branches:	Summary Branches:
1) Alpha Mode Failure	ALPHA
2) No RPV Failure In-Vessel Steam Explosion	NO FAILURE
3) Large Bottom Head Breach RPV Failure	LARGE VF
4) Small Lower Head Breach RPV Failure	SMALL VF

This event assesses the probability that an in-vessel steam explosion occurs with sufficient energy to rupture the lower head of the RPV. This event further differentiates between large (2 square meters) and small (0.1 square meter) vessel failure sizes. As noted previously the probability of an in-vessel steam explosion being triggered has been shown to be dependent on the RPV pressure.

APET Event 32. RPV Failure Mode & Failure Size

Four Branches:	Summary Branches:
1) Alpha Mode Failure	ALPHA
2) No RPV Failure - Debris Cooled In-Vessel	NO FAILURE
3) Large Bottom Head Breach RPV Failure	LARGE VF
4) Small Lower Head Breach RPV Failure	SMALL VF

Given that the debris is not cooled in-vessel, and that the RPV has not been previously failed by an alpha mode failure or by a steam explosion induced lower head failure, this event determines the probability of large (2. square meters) and small (0.1 square meters) vessel failure sizes due to thermal attach on the lower head.

APET Event 33. Water in Pedestal at RPV Failure

Four Branches: Summary Branches:

1) Flooding + Continuing Injection	FLD + INJ
2) Residual RPV Water + Continuing Injection	RPV + INJ
3) Flooding	FLOODING
4) Residual RPV Water Only	RPV WATER

Water in the reactor pedestal cavity at the time of RPV failure can impact the accident progression in several ways. With water in the pedestal cavity there is an increased potential for steam explosions (or rapid steam generation) which may threaten the integrity of the pedestal. Water in the cavity early also enhances the possibility that the debris will be coolable.

The branch definitions are summarized above. Pedestal cavity flooding occurs as a result of pressurization of the wetwell (such as by boiling of the suppression pool or by a hydrogen burn), depression of the pool level on the wetwell side of the suppression pool and overflow of the suppression pool into the drywell. Continuous injection to the pedestal cavity results from the addition of injection to the vessel following vessel breach. Residual RPV water refers to the water remaining in the RPV which is discharged from the RPV coincidentally (or following) expulsion of the core debris in the lower RPV head.

APET Event 34. Pedestal Failure Due to Overpressure at RPV Failure

Two Branches: Summary Branches:

1) Pedestal Failure Due to Overpressurization	PEDESTAL FAIL
2) No Pedestal Failure	NO FAILURE

Given that the debris is not cooled in-vessel, and that the RPV has not been previously failed by an alpha mode failure, this event determines the probability of overpressurization of the pedestal resulting from RPV depressurization and quenching of the debris. Pedestal structural failure would occur if: 1) the RPV is pressurized and a large RPV breach size occurs, 2) the RPV is pressurized and the pedestal cavity is flooded, and 3) the RPV is depressurized, the pedestal cavity is flooded and a large RPV breach size occurs.

APET Event 35. Pedestal Cavity Steam Explosion

Two Branches: Summary Branches:

1) Large Ex-vessel Steam Explosion	STM EXPLOSION

2) No Large Ex-vessel Steam Explosion NO FAILURE

This event assesses the probability of a large steam explosion occurring in the pedestal cavity following vessel failure. For sequences where the debris is cooled in-vessel and lower head failure does not occur, then an ex-vessel steam explosion will not occur. For sequences where an in-vessel steam explosion has failed the vessel, it is assumed that a large ex-vessel steam explosion cannot occur.

APET Event 36. Pedestal Failure Due To Steam Explosion

Two Branches:	Summary Branches:
1) Pedestal Failure Due to Steam Explosion	PEDESTAL FAIL
2) No Pedestal Failure Due To Steam Explosion	NO FAILURE

This event assesses the probability of a steam explosion failing the pedestal. Two failure mechanisms are considered. If an in-vessel steam explosion has caused a large breach in the lower reactor vessel head then it is considered possible that a large missile could be created (from part of the vessel lower head) which could cause failure in the pedestal wall. The second failure mechanism involves a steam explosion in the lower pedestal cavity which generates a shock wave which exceeds the impulse load capacity of the pedestal wall.

APET Event 37. Drywell Failure Due To Pedestal Failure

Two Branches:	Summary Branches:
1) Drywell Failure Due To Pedestal Failure	DW FAILURE
2) No Drywell Failure Due To Pedestal Failure	NO FAILURE

Given that pedestal failure has occurred this event assesses the probability that pedestal failure causes loss of drywell integrity.

APET Event 38. Drywell Overpressure Failure at RPV Failure

Two Branches:	Summary Branches:
1) Drywell Overpressure Failure	DW FAILURE
2) No Drywell Overpressure Failure	NO FAILURE

This event assess the probability that drywell overpressure failure will occur following RPV failure. In order for drywell pressurization to challenge drywell integrity, the RPV must be at high pressure at vessel failure.

APET Event 39. Drywell Failure At/Near RPV Failure

Two Branches:	Summary Branches:
1) Drywell Failure	DW FAILURE
2) No Drywell Failure	NO FAILURE

This summary event assesses the probability that drywell failure will occur following RPV failure. This event considers drywell failure resulting from alpha mode steam explosions, pedestal failure and overpressure failure.

4.5.2.2.5 Mode Of Containment Failure At/Near RPV Failure APET Events (44,45)

These events assess the probability of containment failure and the mode of containment failure at or within an hour of reactor pressure vessel failure. Containment failure at RPV failure can potentially result from a combination of energetic processes and events which may occur at reactor vessel breach. These processes and events include hydrogen combustion, and a large in-vessel steam explosion causing an alpha mode containment failure.

For the Perry APET, steam explosion induced (alpha mode) containment failures are also considered to result in a catastrophic rupture of the containment. Postulated alpha mode containment failures result from large coherent in-vessel steam explosions which fail the reactor vessel and generate a missile (from part of the reactor vessel upper head) with sufficient mass and energy to fail (the drywell and) containment. There is a substantial body of evidence to suggest that in-vessel steam explosions do not represent a credible threat to early containment failure (i.e., the probability of early containment failure from in-vessel steam explosions is negligibly small). This opinion appears to be shared by the authors of Appendix 1 to Generic Letter 88-20. However, if this event should occur, it can result in a large and early environmental releases. Therefore, this event is included in the Perry IPE APET.

Experimental evidence and calculations have shown that steam explosions are unlikely at elevated pressure; therefore, the probability of an alpha mode containment failure should be significantly less for high pressure sequences than for low pressure sequences.

Figure 4.5.2.2.5-1, Mode of Containment Failure At/Near RPV Failure DET, shows the general characterization of this portion of the APET. This DET provides addition information regarding hydrogen combustion. The containment hydrogen concentration is provided as a function of containment steam concentration and fraction of zirconium inventory reacted in-vessel. Following this, three informational headings show the theoretical Adiabatic Isochoric Complete Combustion Pressure, the Burn Efficiency Factor, and the summary Expected Peak Burn Pressure.

The Mode of Containment Failure At/Near RPV Failure is determined by APET events 44 and 45, which are dependent on the following Events in the Perry APET.

APET Event 40. Containment Steam Concentration At/Near RPV Failure

Six Branches:	Summary Branches:
1) 0-15 Volume Percent Steam	0-15%
2) 15-25 Volume Percent Steam	15-25%
3) 25-35 Volume Percent Steam	25-35%
4) 35-45 Volume Percent Steam	35-45%
5) 45-55 Volume Percent Steam	45-55%
6) > 55 Volume Percent Steam	> 55%

This branch assesses the containment steam concentration at or near RPV failure. The containment steam concentration is a function of the mode of spray operation, the event type type, the time of injection failure and whether the containment is intact at core damage.

APET Event 41. Fraction Zirconium Inventory Reacted At/Near RPV Failure

Three Branches:	Summary Branches:
1) 33% Core Inventory Zirconium Oxidized	33%
2) 22% Core Inventory Zirconium Oxidized	22%
3) 11% Core Inventory Zirconium Oxidized	11%

The fraction of zirconium reacted in-vessel is used to determine the concentration of hydrogen in containment at or near RPV failure. Three discrete regimes are used to represent the range of in-vessel zirconium oxidation. These regimes are representative of the amounts estimated in the NUREG/CR-4551 Grand Gulf analysis (Brown 1990). The fraction of zirconium oxidized probabilities are estimated using specific MAAP calculations for a spectrum of sequences (considering local blockage and no channel blockage).

APET Event 42. Hydrogen Ignition Sources Available at RPV Failure

Two Branches:	Summary Branches:
1) No Hydrogen Ignition Sources	NO IGN SOURCE
2) Hydrogen Ignition Sources Available	IGNITION SOURCE

With a continuous ignition source available in containment (at the time of RPV failure) it is assumed that controlled hydrogen combustion (or small hydrogen burns) will preclude the build-up of hydrogen concentrations whose combustion would threaten containment integrity. A continuous ignition source is assumed to be available if the hydrogen ignition system is operating or if a burn in containment has already occurred during core damage. In the latter situation it is assumed that the prior burn which has occurred will result in ignition of combustible materials in containment which will act as ignition sources. Additionally, this event includes the recovery human interaction to place the hydrogen ignition system in service late.

APET Event 42. High Pressure Melt Ejection

Two Branches:	Summary Branches:
1) High Pressure Melt Ejection Occurs	HPME
2) No High Pressure Melt Ejection Occurs	NO HPME

This event assesses whether a high pressure melt ejection occurs from the reactor pedestal cavity following vessel failure. For HPME to occur the RPV pressure must be elevated (above several hundred psi) at the time of vessel failure. HPME involves the entrainment and fragmentation of the debris in the pedestal cavity and transport of the debris throughout the drywell. If the drywell has failed then an HPME event can provide an ignition source for hydrogen in the containment.

APET Event 43. Large Hydrogen Burn At/Near RPV Failure

Two Branches: Summary Branches:

1) No Large Hydrogen Burn Ignited — NO LRG BURN IGN
2) Large Hydrogen Burn Ignited — LG BURN IGN

This event assess whether a large hydrogen burn is ignited in containment following RPV failure. If a continuous ignition source is available, then a large hydrogen burn is assumed to be prevented. Also if the steam concentration is above 55%, then the containment atmosphere is inert to hydrogen burns. The probability that a hydrogen burn is ignited is a function of the following parameters: containment steam concentration, containment hydrogen concentration (which is dependent on the fraction of zirconium reacted), Drywell failure, High Pressure Melt Ejection, and AC Power Recovery.

Forty eight cases (various combinations of the parameters listed above were identified to perform the quantification). In addition to assessing the probability of ignition this event also sets two EVNTRE parameter values which are used by subsequent events: peak hydrogen burn pressure (parameter 2) and containment base pressure prior to the burn (parameter 4).

APET Event 45. Hydrogen Detonation Containment Failure At/Near RPV Failure

Two Branches: Summary Branches:

1) Detonation Containment Failure — DET CF
2) No Containment Failure — NO

Given that a large burn was ignited in containment following RPV failure this event assesses whether the burn transitioned to a detonation and whether containment failure resulted from the detonation impulse loading. If no large burn was ignited or if the containment atmosphere was inert to detonations (> 55% steam concentration) then no detonation was assumed to occur. The probability of a hydrogen detonation occurring is taken to be a function of the steam concentration, the hydrogen concentration, whether power recovery occurs and whether sprays are initiated during this time period.

APET Event 46. Containment Failure At/Near RPV Failure

Two Branches: Summary Branches:

1) Containment Failure — FAILURE
2) No Containment Failure — NO FAILURE

This event assesses whether containment failure occurs following RPV failure as a result of a hydrogen burn or alpha mode steam explosion. If a hydrogen detonation has occurred which fails the containment then containment failure has occurred. If a large hydrogen burn was ignited then this event compares the peak containment pressure for the burn

(Parameter 2) with the containment fragility curve and determines the probability of containment failure.

APET Event 47. Mode of Containment Failure At/Near RPV Failure

Two Branches: Summary Branches:

1) Anchorage Containment Failure — ANCHORAGE
2) Penetration-Dome Containment Failure or No Containment Failure — PENET-DOME/NO CF

This event determines the probability of anchorage containment failure given containment failure due to hydrogen deflagration, and assigns alpha mode and detonation failure sequences and no containment failure sequences to the second branch. For sequences where the containment has failed by a hydrogen burn the peak containment pressure from the burn is used to estimate the probability of anchorage failure. The individual fragility curves for the dominant failure modes are used to determine the conditional probabilities for each failure mode as a function of the failure pressure.

4.5.2.2.6 Pool Bypass Before/Near RPV Failure (APET Event 49)

Fission product scrubbing in the suppression pool is an effective fission product mitigation mechanism. However, if the release pathway bypasses the suppression pool this mechanism is not effective. Pool bypass may result from a number of causes. These include: 1) structural failure of the drywell, 2) Drywell vacuum breaker failure, 3) loss of suppression pool water below the level of the horizontal vents or the SRV quenchers, and 4) other failure processes.

Drywell structural failure may result from transient over-pressurization of the drywell or wetwell resulting in a sufficiently high drywell/wetwell differential pressure to cause failure of the drywell head, ceiling or walls. Failure of Drywell vacuum breaker may occur during hydrogen combustion. Loss of suppression pool water may result from containment anchorage failure in the pool region.

Figure 4.5.2.2.6-1, Pool Bypass Before/Near RPV Failure DET, shows the general characterization of this portion of the APET.

This APET event, Pool Bypass Before/Near RPV Failure, is dependent on the following Events in the Perry APET.

APET Event 48. Drywell Failure Due to Containment Hydrogen Burn Before/Near RPV Failure

Two Branches: Summary Branches:

1) Drywell Failure Due To Hydrogen Burn — DRYWELL FAILURE
2) No Drywell Failure — NO DW FAILURE

Given that a large hydrogen burn has occurred during core damage or at RPV failure this event assesses whether drywell failure results from excessive differential pressure across the drywell boundary. For cases where a large hydrogen burn has occurred two parameters have been set which give the peak burn pressure in containment during the burn and the containment pressure prior to the burn. It is assumed that the drywell pressure remains constant during the burn in containment. Given the value of these two parameters the peak drywell differential pressure is calculated and compared against the drywell fragility curve in a user function to estimate the probability of drywell structural failure.

APET Event 49. Pool Bypass Before/Near RPV Failure

Two Branches:	Summary Branches:
1) Pool Bypass	POOL BYPASS
2) No Pool Bypass	NO POOL BYPASS

This event assess the probability that pool bypass will occur prior to, or at, RPV failure. This event considers pool bypass resulting from drywell failure resulting from processes occurring within the drywell structure, drywell failure form hydrogen combustion in the containment, pool bypass due to containment anchorage failure, drywell vacuum breaker failure due to large hydrogen burns, and other bypass failures.

4.5.2.2.7 Inject & Spray Failure Due To Containment Failure At/Near RPV Failure (APET Event 53)

This event assesses the probability that containment failure causes the loss of all in-vessel injection (assuming in-vessel injection has not previously failed) and failure of the RHR containment spray system. The containment sprays represent an effective fission product mitigation feature which can significantly limit atmospheric releases of radionuclides. Containment sprays will be particularly important for sequences where suppression pool bypass has occurred, since the sprays then represent the only remaining major engineered safety system which can significantly mitigate radionuclide releases.

The mechanisms associated with containment failure at/near RPV failure which may cause failure of in-vessel injection and/or the containment spray system are the same as those discussed above in section 4.5.2.2.3, Injection & Spray Failure Due To Containment Failure Before RPV Failure.

Figure 4.5.2.2.7-1, Injection & Spray Failure Due To Containment Failure At/Near RPV Failure DET, shows the general characterization of this portion of the APET.

This APET event, Injection & Spray Failure Due To Containment Failure At/Near RPV Failure, is dependent on the following Events in the Perry APET.

APET Event 50. Containment Failure At/Near RPV Failure Impact On ECCS Injection And Spray Piping

Two Branches:	Summary Branches:
1) No Piping Failure	NO FAILURE
2) Failure of ECCS Injection & Spray Piping	FAILURE

This event assesses the probability that either the dynamic forces or movement of the containment which occur at containment failure are sufficient to disrupt the injection and spray system piping. Disruption of this piping is expected to be a serious threat for containment anchorage failure.

APET Event 51. Containment Failure At/Near RPV Failure
Impact On ECCS Injection and Spray Motors

Two Branches:	Summary Branches:
1) No Motor Failure	NO FAILURE
2) ECCS Injection and Spray Motor Failure	FAILURE

This event assesses the probability that leakage of water, steam or hot gases from containment into the auxiliary building which occur at containment failure cause failure of the injection and/or spray system motors and related components.

APET Event 52. Containment Failure At/Near RPV Failure
Steam and Radiation Impact on Firewater Injection

Two Branches:	Summary Branches:
1) No Failure	NO FAILURE
2) Failure of Firewater Alternate Injection	FAILURE

This event assesses the probability that leakage of radionuclides from containment into the plant or outside the plant will limit personnel access to the firewater system and result in failure to perform required local manual actions to initiate, or to assure continued operation of the firewater alternate injection.

APET Event 53. Injection & Spray Failure Due To Containment Failure
Before/Near RPV Failure

Two Branches:	Summary Branches:
1) No Injection & Spray System Failure	NO FAILURE
2) Injection Or Spray System Failure	FAILURE

Event 53 event summarizes the results of the prior three events in the APET.

4.5.4.8 Pedestal Failure Due To Core Debris Concrete Interaction (APET Event 55)

This event assesses whether pedestal failure occurs as a result of sidewards core concrete attack in the pedestal cavity eroding the pedestal wall to a sufficient depth that the structural integrity of the pedestal wall is compromised. Failure of the pedestal wall may result in loss of support to the RPV and result in gross motion of the RPV. This motion may result in damage to the drywell or to containment penetrations.

The amount of radial erosion will be a function of the type and extent of core concrete attack that occurs, the ratio of radial to downwards concrete attack and the failure depth for the pedestal wall.

Figure 4.5.2.2.8-1, Pedestal Failure Due To Core Concrete Interaction DET, shows the general characterization of this portion of the APET.

This APET event, Pedestal Failure Due To Core Concrete Interaction, is dependent on the following Events in the Perry APET.

APET Event 54. Type of Core Debris Concrete Interaction

Four Branches:	Summary Branches:
1) CCI In A Dry Pedestal Cavity	DRY-CCI
2) Rapid CCI With an Overlying Water Layer	FAST-WET
3) Slow CCI With an Overlying Water Layer	SLOW-WET
4) NO CCI With Debris Cooled	NO-CCI

This event determines the probability of various types of CCI which may occur in the pedestal cavity following RPV failure. If no water is in the pedestal cavity prior to RPV failure and water is not supplied following RPV failure then dry CCI will occur. If a small amount of water is in the pedestal cavity prior to RPV failure but a continuing supply of water is not available then the debris/concrete attack may be delayed until the water pool is boiled away. (This case has been conservatively combined with DRY CCI). For cases where a large pool of water has entered the drywell prior to RPV failure or where a continuous supply of water is available to the pedestal following vessel failure then a pool of water will cover the debris. Depending upon the surface area of the debris, the debris particle size and the effective upward heat transfer rate the debris may be cooled or CCI may occur. The last three branches assess the rate of CCI given a debris pool which is covered by water. The probabilities for the types of CCI were estimated using MAAP for a series of cases.

APET Event 55. Pedestal Failure Due to Core Debris Concrete Interaction

Three Branches:	Summary Branches:
1) Pedestal Failure At RPV Failure	VESSEL BREACH
2) Pedestal Failure After RPV Failure	AFTER VB
3) No Pedestal Failure	NO FAILURE

Given a type of CCI determined in the previous event, this event assesses whether pedestal structural failure from CCI radial erosion of the pedestal wall will occur.

4.5.2.2.9 Mode Of Late Burn And Overpressure Containment Failure
(APET Events 64,65,66)

These events assess the probability (and mode) for containment failures which occur late in the accident sequences. These events assess containment failures resulting from hydrogen combustion and detonation and from gradual overpressurization due to steam and non-condensible gas production.

Figure 4.5.2.2.9-1, Mode of Late Burn & Overpressure Containment Failure DET, shows the general characterization of this portion of the APET. This DET provides addition information regarding the Expected Peak Burn Pressure during hydrogen combustion.

The Mode of Late Burn & Overpressure Containment Failure is determined by APET events 64, 65 and 66; which are dependent on the following Events in the Perry APET.

This event is dependent on the following Events in the Perry APET.

APET Event 56. Mode Of RHR Spray Operation Late

Three Branches:	Summary Branches:
1) Controlled Spray Operation	CONTROLLED
2) Normal Spray Operation	DESIGN COOLING
3) Sprays Not Available	NO SPRAY

This event assesses whether the RHR sprays are available for containment heat removal when hydrogen combustion may occur late in the accident sequence. This event determines when RHR spray is available from the Plant Damage State initial condition, containment heat removal with RHR spray loop, and further considers whether offsite power is available during SBO sequences. It further allows for the assessment of a Plant Emergency Instruction enhancement for controlled cooling spray operation previously discussed in APET Event 18.

APET Event 57. Hydrogen Ignition Sources Available Late

Two Branches:	Summary Branches:
1) No Hydrogen Ignition Sources	NO SOURCE
2) Hydrogen Ignition Sources Available	IGN SOURCE

With a continuous ignition source available in containment it is assumed that controlled hydrogen combustion (or small hydrogen burns) will preclude the build-up of hydrogen concentrations whose combustion would threaten containment integrity. A continuous ignition source is assumed to be available if the hydrogen ignition system is operating. Since there may have been a substantial time period between RPV failure and the time when late combustion may occur, earlier burns are not considered to provide a reliable ignition source for late burns (and thus ensure that only small burns would occur late).

APET Event 58. Containment Steam Concentration Late

Six Branches:	Summary Branches:
1) 0-15 Volume Percent Steam	0-15%
2) 15-25 Volume Percent Steam	15-25%
3) 25-35 Volume Percent Steam	25-35%
4) 35-45 Volume Percent Steam	35-45%
5) 45-55 Volume Percent Steam	45-55%

6) > 55 Volume Percent Steam > 55%

This branch assesses the containment steam concentration late in the accident. The probability of hydrogen burn ignition and the efficiency of the burn are dependent upon the branch taken under this heading. The containment steam concentration is a function of the mode of spray operation, the sequence type, the time of injection failure and whether the containment is intact at core damage. The steam concentration regimes probabilities for the various cases are estimated using MAAP results for each case.

APET Event 59. Hydrogen Combustion Before/At RPV Failure

Two Branches:	Summary Branches:
1) Early Burn Before/At RPV Failure	EARLY BURN
2) No Early Burn	NO EARLY BURN

This event summarizes whether an earlier burn in containment has occurred. This information is used to assess the late hydrogen concentration in containment.

APET Event 60. Containment Effective Hydrogen Concentration Late

Six Branches:	Summary Branches:
1) < 4 Volume Percent Hydrogen	< 4%
2) 4- 8 Volume Percent Hydrogen	4- 8%
3) 8-12 Volume Percent Hydrogen	8-12%
4) 12-16 Volume Percent Hydrogen	12-16%
5) 16-20 Volume Percent Hydrogen	16-20%
6) > 20 Volume Percent Hydrogen	> 20%

This branch assesses the containment effective hydrogen concentration late in the accident sequence (which includes the carbon monoxide produced during CCI). The probability of hydrogen burn ignition and the efficiency of the burn are dependent upon the branch taken under this heading. The containment hydrogen concentration is a function of the mode of CCI, and whether a hydrogen burn occurred early in the sequence. The hydrogen concentration probabilities were estimated using MAAP calculations for a spectrum of sequences (considering local blockage and no channel blockage).

APET Event 61. AC Power Available Late

Two Branches:	Summary Branches:
1) AC Power Available Late	AC LATE
2) No AC Power Available Late	NO AC LATE

This event summarizes whether AC power is available late in the sequence. AC power will be available late if AC power was never lost or if AC power was initially lost but was recovered. This information is used to assess the potential for hydrogen ignition late in the sequence.

APET Event 62. Large Hydrogen Burn Late

Two Branches:	Summary Branches:
1) No Large Hydrogen Burn Ignited	NO BURN
2) Large Hydrogen Burn Ignited	LARGE BURN

This event assess whether a large hydrogen burn is ignited in containment late in the accident sequence. If a continuous ignition source is available, then a large hydrogen burn is assumed to be prevented. Also if the steam concentration is above 55% then the containment atmosphere is inert to hydrogen burns. The probability that a large hydrogen burn is ignited is a function of the following parameters: containment steam concentration, containment hydrogen concentration, and AC power availability.

Fifty four cases (various combinations of the parameters listed above were identified to perform the quantification). In addition to assessing the probability of ignition this event also sets two EVNTRE parameter values which are used by subsequent events; peak hydrogen burn pressure (parameter 5) and containment base pressure prior to the burn (parameter 6).

APET Event 63. Hydrogen Detonation Late Containment Failure

Two Branches:	Summary Branches:
1) Hydrogen Detonation Containment Failure	DET CF
2) No Failure	NO

Given that a large burn was ignited in containment late in the sequence this event assesses whether the burn transitioned to a detonation and whether containment failure resulted from the detonation impulse loading. If no large burn was ignited or if the containment atmosphere was inert to detonations (> 35% steam concentration) then no detonation was assumed to occur. The probability of a hydrogen detonation occurring is taken to be a function of the steam concentration, the effective hydrogen concentration, whether AC power recovery occurs, and whether sprays are available.

APET Event 64. Hydrogen Burn Late Containment Failure

Two Branches:	Summary Branches:
1) Containment Failure Due to Late Hydrogen Burn	FAILURE
2) No Containment Failure	NO FAILURE

This event assesses whether containment failure occurs late as a result of a hydrogen burn. If a hydrogen detonation containment failure occurred then this is sorted as containment failure. If a large hydrogen burn was ignited then this event compares the peak containment pressure for the burn (Parameter 5) with the containment fragility curve and determines the probability of containment failure.

APET Event 65. Containment Status At Accident Progression Completion

Four Branches: Summary Branches:

1) Early Containment Failure — EARLY CF
2) Late Overpres ure Failure — LATE CF
3) Containment Vent — VENT
4) No Containment Failure (i.e., no containment failure or vent) — NO CF

This event assesses whether containment failure occurs late in the sequence progression due gradual overpressurization from steam and non-condensible gas generation. This event also assess whether the containment is vented to prevent overpressure failure. The event pathway probabilities are a function of whether containment failure has already occurred earlier in the accident, whether containment heat removal is available, whether the pool is bypasses, whether the containment is vented, and the event type.

APET Event 66. Mode of Late Hydrogen and Overpressure Containment Failure

Two Branches: Summary Branches:

1) Anchorage Containment Failure — ANCHORAGE
2) Penetration-Dome Containment Failure or No Containment Failure — PENET-DOM/NO CF

This event determines the probability of anchorage containment failure given containment failure due to late overpressure or deflagration, and assigns detonation containment failure sequences and no containment failure sequences to the second branch. For late gradual steam overpressure containment failure, the estimated conditionally probability of 0.15 is assigned for anchorage failure. All detonation failures are considered most likely to occur in the dome region and are sorted to the penetration-dome or no containment failure category. For sequences where the containment has failed by a late burn the expected peak containment pressure from the burn is used to estimate the probability of anchorage failure.

APET Event 64 (Hydrogen Burn Late Containment Failure), APET Event 65 (Containment Status At Accident Progression Completion) and APET 66 (Mode Of Late Hydrogen & Overpressure Containment Failure) can be referenced by the EVNTRE binner to determine the probability of the possible modes of containment failure, as indicated in the introduction of this section.

4.5.2.2.10 Pool Bypass Late (APET Event 68)

Fission product scrubbing in the suppression pool is an effective fission product mitigation mechanism unless the suppression pool is saturated. However, if the release pathway bypasses the suppression pool this mechanism is not effective. Pool bypass may result from a number of causes. These include: 1) structural failure of the drywell, 2) drywell vacuum breaker failure, 3) excessive leakage through drywell penetrations, and 4)loss of suppression pool water below the level of the horizontal vents or the SRV quenchers.

Drywell structural failure may result from transient over-pressurization of the drywell or wetwell resulting in a sufficiently high drywell/wetwell differential pressure to cause failure of the drywell head, ceiling or walls. Failure the drywell vacuum breakers may occur due to hydrogen burns. Loss of suppression pool water may result from containment anchorage failure. Dry CCI leads to high drywell temperature and failure of the drywell penetrations.

Figure 4.5.2.2.10-1, Late Pool Bypass DET, shows the general characterization of this portion of the APET.

This APET event, Late Pool Bypass, is dependent on the following Events in the Perry APET.

This event is dependent on the following Events in the Perry APET.

APET Event 67. Drywell Failure Due To Late Hydrogen Burn In Containment

Two Branches:	Summary Branches:
1) Drywell Failure Due To Late Hydrogen Burn	DW FAILURE
2) No Drywell Failure	NO FAILURE

Given that a large hydrogen burn has occurred late in the sequence this event assesses whether drywell failure results from excessive differential pressure across the drywell boundary. For cases where a large hydrogen burn has occurred two parameters have been set which give the peak burn pressure in containment during the burn and the containment pressure prior to the burn. It is assumed that the drywell pressure remains constant during the burn in containment. Given the value of these two parameters the peak drywell differential pressure is calculated and compared against the drywell fragility curve in a user function to estimate the probability of drywell structural failure.

APET Event 68. Pool Bypass Late

Two Branches:	Summary Branches:
1) Late Pool Bypass	LATE POOL BYP
2) No Late Bypass	NO LATE BYPASS

This event assess the probability that pool bypass will occur late in the accident sequence. This event considers pool bypass resulting from: drywell failure from late hydrogen combustion in the containment, containment anchorage failure, pedestal failure caused by core concrete interaction pedestal erosion, drywell penetration failures associated with high temperature during dry CCI, drywell vacuum breaker failures due to large hydrogen burns, and other failure processes.

4.5.3 APET Quantification

The Accident Progression Event Tree summary results for the Perry Mark III containment performance are: no containment failure - represents a 39% conditional probability given core damage and a frequency of 5.0×10^{-6}, containment venting - represents a 29% conditional probability and a frequency of 3.7×10^{-6}, and containment structural failure - represents a 32% conditional probability and a frequency of 4.0×10^{-6}. A summary of the quantified APET results for all sequences is provided in Table 4.5.3-1, APET Base Case Results.

Table 4.5.3-1 presents the containment failure modes in the format of the NUREG/CR-4551 Grand Gulf evaluation (Brown 1990). When the core debris is cooled in-vessel no RPV failure occurs and three modes of containment failure are shown: no containment failure, vent and containment (structural) failure. When the core is not cooled in-vessel RPV failure occurs and seven modes of containment failure are shown: no containment failure, vent, late containment (structural) failure, early containment failure with no pool bypass, early containment failure with late pool bypass, early containment failure with early pool bypass and containment spray, and early containment failure with early pool bypass and no containment spray. The containment failure frequency is the combination of both containment structural failure frequency and venting frequency (and includes both in-vessel cooling and no in-vessel cooling sequences). Early containment failure with pool bypass is the combination of early containment failure with: late pool bypass, early pool bypass with containment spray, and early pool bypass with no containment spray. The conditional probability estimates of the detailed failure modes evaluation are: 50.8% - in-vessel cooling and no RPV failure; and the balance of the sequences with RPV failure (49.2%) is composed of: 12.4% - no containment failure, 10% - venting with a damaged core, 7.4% - late containment failure, 3.4% - early containment failure with no pool bypass, and 16.1% containment failure with pool bypass.

Detailed APET base case quantification results for all sequences are provided in Appendix H.4, PNPP IPE APET Program Frequency Output File.

Table 4.5.3-2, APET Containment Performance Base Case Results For Dominant PDS Groups, shows the ranked results of the 16 plant damage states that contribute 95% of the total containment failure probability. The containment performance results are shown in the same format as that described above. In addition, the containment failure and venting composite identifies the probability of structural containment failure in the penetration or anchorage mode, as well as the probability of venting. The containment failure and venting composite probabilities when summed equal the total failure and venting probability for each plant damage sequence and for All PDS sequences (which is listed last in this table). The All PDS sequences results show the following conditional probabilities for the containment structural failure modes: 26.7% - penetration failures, and 5.0% - containment anchorage failures.

4.5.3.1 Containment Failure And Dominant PDS Groups

The APET Containment Performance Base Case Results for each of the 16 dominant Plant Damage State groups representing 95% of the total containment failure are shown in Table 4.5.3-2. The individual core damage sequences contributing above 1% core damage listed with the dominant containment failure PDS groups in Table 4.5.3-3.

Sixteen PDS groups represent 95% of the core damage frequency. The two most dominant PDS groups are 56 and 73 which represent 60% of the total containment failure frequency. A general discussion of these two PDS groups is provided below.

PDS group 56, represents Non-SBO sequences with the Containment Intact At Core Damage and successful Containment Heat Removal With the Vent, Late Injection, and RPV Depressurization During Core Damage. PDS group 56 represents 43.9% of the total combined containment failure and venting frequency. PDS group 56 has an early containment failure percentage of 1% and a venting percentage of 99%. RPV Failure and Early Containment Failure with Early Bypass is estimated to occur 1% of the time for sequences in this plant damage state. PDS group 56 has 6 dominant sequences above 1% of the core damage frequency. The most dominant sequence (9% CDF) is a Flood in Zone 13B with failure of all injection. The second sequence (6% CDF) is a Loss Of Instrument Air with Loss of all injection. The third sequence (3% CDF) is a Station Blackout sequence with successful recovery of offsite power before core uncovery and subsequent loss of all injection. The fourth sequence (2% CDF) is a Flood in Zone F1D with loss of all injection. The remaining two sequences (1.6% CDF each) are a Large LOCA and LOOP with failure of injection.

PDS group 73, represents the OTHERS sequences (i.e., other than Critical ATWS or LOOP & SBO) with Containment Failed At Core Damage, unsuccessful Late Injection, and successful RPV Depressurization During Core Damage. PDS group 73 represents 16% of the total combined containment failure and venting frequency. RPV Failure and Early Containment Failure with Pool Bypass are predicted for all plant damage state sequences (with 16% early pool bypass and 84% late pool bypass). Early Pool Bypass is associated with the containment anchorage failure mode which results in suppression pool loss. Late Pool Bypass occurs due to the high drywell temperatures which challenge the drywell penetration seals and result in suppression pool bypass. PDS group 73 has just 1 dominant core damage sequence (8% CDF), Loss of PCS and failure of containment heat removal with subsequent failure of all late injection.

Half of the 16 dominant containment failure PDS groups are in the general category of containment failed at core damage. These are PDS groups: 73, 71, 65, 67, 69, 63, 66 and 70 (which are ranked 2, 4, 5, 6, 7, 8, 11 and 16). These groups contribute a total containment failure frequency of 2.9×10^{-6} or 37% of the total containment failure and venting frequency. Critical ATWS contributes 7.3% of the total containment failure and venting frequency with the 3 dominant PDS groups: 65, 63 and 66. LOOP & SBO contributes 7.1% of the containment failure frequency with the 3 dominant PDS groups: 67, 69 and 70. OTHERS contributes 23% of the total containment failure and venting frequency with the PDS groups: 71 and 73.

Station Blackout containment failure is represented by three dominant PDS groups: 25, 36 and 9 (which are ranked 9, 10 and 12). These 3 SBO PDS groups represent 2.6% containment structural failure and 1.3% containment venting which is 3.9% of the total containment failure and venting frequency. These 3 dominate SBO groups represent 3.2% of the core damage frequency.

All Station Blackout PDS groups represent 9.0% of the total core damage frequency. All the Station Blackout PDS sequences contribute 7.9% of the total containment failure and venting frequency. The total Station Blackout containment structural failure frequency is 6.0×10^{-6}, which represents 4.7% of the total containment failure and venting frequency.

The containment structural failure conditional probability of 31.7% has contributors from the following PDS grouping categories: 22.8% - where the containment is failed at core damage (4.4% - from Critical ATWS PDS groups, 4.3% - from LOOP & SBO PDS groups, and 14% - from the other remaining PDS groups), and 8.9% - where the containment is intact at core damage and failure occurs later (4.7% - from SBO PDS groups, and 4.2% - from other PDS groups with AC power available).

4.6 ACCIDENT PROGRESSION

4.6.1 Summary of Sequences Analyzed

To support the development and quantification of the Accident Progression Event Tree (APET) an assessment of the physical progression of a spectrum of accident sequences was performed using the MAAP 3.0B code. This effort provided information regarding: the timing of key events, containment loads, debris relocation and cooling, mitigation effectiveness of injection systems, the generation and combustion of hydrogen and carbon monoxide, and core concrete interaction in the pedestal cavity. Over 200 MAAP computer runs support the assessment of accident progression.

The Perry accident progression analyses scope included the performance of all the recommended sensitivities applicable to a Mark III containment as described in the "Recommended Sensitivity Analyses For An Individual Plant Examination Using MAAP 3.0B" (EPRI 1990). This sensitivity evaluation scope included: the coefficient for critical heat flux used in debris coolability (FCHF), containment failure area (AOVPR), the impact of core geometry on cladding oxidation (FCRBLK), delayed hydrogen combustion (DXHIG), and diffusion flame modeling during hydrogen ignitor operation (FLPHI). The MAAP cladding oxidation (FCRBLK) sensitivity key results for Station Blackout sequences without the hydrogen ignition system available are provided in Tables 4.6.1-1 and 4.6.1-2. The EPRI recommended sensitivities and the IPE insights gained guided the accident progression assessment and the development of the APET.

To derive the release fractions for the various source term categories, the Back-End Analysis also used the MAAP 3.0B code. The eleven source term MAAP runs are discussed in section 4.7.3.

4.6.2 APET Sensitivity Analysis

A sensitivity analysis was performed to identify significant contributors to containment performance in the quantification of the APET. The sensitivity analysis benchmarked the APET sensitivity results to the containment performance characteristics of: No RPV Failure, Containment Integrity (No Failure, Vent, Failure), and No Pool Bypass. The following subsections summarize the sensitivity analysis of 14 APET parameters of which the first 13 are related to accident progression phenomena, and the last is the human interaction, failure to initiate the hydrogen ignition system.

The results of the APET sensitivity analysis determined the the following APET parameters are significant: (1) assured in-vessel debris cooling would increase No RPV Failure with a 34% change, and (2) in-vessel steam explosion bottom head failure estimated with the Grand Gulf NUREG/CR-4551 (Brown 1990) probabilities decrease No RPV Failure with a 58% change and decrease No Pool Bypass with a 10% change. The significant sensitivity impact of in-vessel debris cooling is realized and will be fur[illegible] examined by the industry during accident management using MAAP 4.0. The significant sensitivity impact of in-vessel steam explosion bottom head failure demonstrates the profound effect of variations in the estimate of this phenomena which is not well understood.

4.6.2.1 APET Event 13. RPV Depressurized During Core Damage

The impact of increased RPV depressurization was measured by setting the probability of successful depressurization to 1. The measured sensitivity changes on the four benchmark parameters described in 4.6.2 were on the order of 1 percent change, demonstrating that the current RPV Depressurization capability is sufficient.

4.6.2.2 APET Event 15. Debris Cooled In-Vessel

The impact of in-vessel recovery was measured by setting the probability of cooling the debris in-vessel to 1 and 0 (given the recovery of low pressure injection before core plate failure). The sensitivity results of assured in-vessel coolability show that No RPV Failure would increase with a 34% change, and No Pool Bypass would increase with a 6% change. The sensitivity results of no in-vessel coolability show that No RPV Failure would decrease with a 100% change and No Pool Bypass would decrease with a 16% change.

4.6.2.3 APET Event 18 and 56. Mode of RHR Spray Operation

The impact of controlled cooling spray operation was measured by changing the assignment of RHR (Design) Spray to Controlled (Spray). The sensitivity results show a 2% decrease in No Containment Failure and an increase of 2.5% in Containment Structural Failure. The increase in Containment Failure was not anticipated from the perspective of engineering judgment and examination of the SBO sequences, where this enhancement should be effective, found the APET model to be simplistic. Additional APET modeling to evaluate controlled RHR spray cooling during SBO recovery with hydrogen generation should be considered during Accident Management.

4.6.2.4 APET Event 20. Fraction of Zirconium Reacted In-Vessel

The impact of setting all the sequences to the maximum hydrogen production case, 33% fraction of zirconium reacted in-vessel, was measured by case assignment changes. The sensitivity results show No RPV Failure and No Pool Bypass decrease with about a 3% change and Containment (structural) Failure increases with a 8% change.

4.6.2.5 APET Event 21. Small Burns At Low Hydrogen Concentrations

The impact of transient combustibles maintaining a low hydrogen concentration by providing an ignition source for continuing small burns was measured by setting the probability of small burns to 0, except when the hydrogen ignition system is operating. The sensitivity results show essentially no change in the key parameters.

4.6.2.6 APET Event 24. Containment Failure Fragility Model

The impact of changing the containment failure fragility curve to a point estimate of the median (64.3 psig) and the 5th percentile (53.5 psig) was measured by revising the EVNTRE user function. Both sensitivity case results show essentially no change.

4.6.2.7 APET Event 25, 47 and 66. Mode of Containment Failure

The impact of changing the probability of the mode of containment failure from the fragility curve assignments for anchorage and penetration failures was measured by assigning all containment failure cases to anchorage failure. The sensitivity results show no change in No RPV Failure and Containment Integrity, and found an 11% increase in No Pool Bypass.

4.6.2.8 APET Event 31. Mode of In-Vessel Steam Explosion Bottom Head Failure

The impact of changing the probability of the estimate of in-vessel steam explosion bottom head failure by a factor of 10 to the value used in Grand Gulf NUREG/CR-4551 (Brown 1990) was measured by a case assignment change. The sensitivity results show a 58% decrease in No RPV Failure, a 10% decrease in No Pool Bypass, and a 1% increase in Containment Failure.

4.6.2.9 APET Event 32. RPV Lower Head Failure Size

The impact of changing the probability of a large vessel failure size to 1. was measured by case assignment. The sensitivity results indicated essentially no change in the overall results.

4.6.2.10 APET Event 33. Water In Pedestal At RPV Failure

The impact of water in the pedestal at RPV Failure was measured by setting the probability of pedestal flooding to 1. The sensitivity results show no notable change.

4.6.2.11 APET Event 36. Pedestal Failure Due To Steam Explosion

The impact of changing the probability of the estimate of pedestal failure due to steam explosion by a factor of 10 to the value used in Grand Gulf NUREG/CR-4551 (Brown 1990) was measured by a case assignment change. The sensitivity results show no notable change.

4.6.2.12 APET Event 37. Drywell Failure Due To Pedestal Failure

The impact of pedestal failure on drywell failure was measured by changing the conditional probability of drywell failure given pedestal failure to 1 by case assignment. The sensitivity results show a 1% decrease in No Pool Bypass.

4.6.2.13 APET Event 54. Type of Core Debris Concrete Interaction

The impact of ex-vessel debris coolability was assessed with two sensitivity cases. The sensitivity of assuming the ex-vessel core debris is never coolable was measured by case assignment with an equal probability of slow wet and fast wet core concrete interaction. The sensitivity results of non-coolable debris show a 2% decrease in No Pool Bypass. The sensitivity of assuming the debris is always cooling was measured by case assignment with a probability of 1, if late injection water to the pedestal cavity was available. The sensitivity results show a 8% increase in No Pool Bypass, and a 1% decrease in Containment Failure.

4.6.2.14 APET Events 16, 42 and 57. Hydrogen Ignitor Human Interaction

The impact of the human interaction, failure to initiate the hydrogen ignition system, on hydrogen combustion was assessed with two sensitivity cases. In the first case, the recovery human interaction failure probability was set to 1. The sensitivity results for no recovery of the hydrogen ignitors show less than a 0.4% change in the results of the four benchmark parameters. For the second case, the probabilities for failure to initiate and failure to recovery were both set to 1. The sensitivity results for complete failure to initiate the hydrogen ignitors show essentially no change in the results.

4.6.3 Containment Performance Improvement Sensitivity Analysis

Sensitivity analyses were performed to consider the improvement in containment performance with the following design change considerations: passive plant vent, ATWS automatic ADS inhibit and alternate shutdown system, and a secure power supply for the hydrogen ignition system. The sensitivity analysis results for these design considerations are shown in Table 4.6.3-1, Impact of Design Changes On Containment Failure Frequency.

4.6.3.1 Passive Containment Vent Design Consideration

The passive vent sensitivity was considered since 22.8% of the plant damage state sequences have a failed containment before core damage, of which 18.4% are non-Critical ATWS sequences. The front-end sensitivity analysis included this impact of containment failure on the loss of injection in section 3.4.1.6 and identified the core damage frequency would be reduced by about 22% with the design change consideration of a passive vent.

The containment vent pathways include the normally open Fuel Pool Cooling & Cleanup (FPCC) return line and normally closed RHR Containment Spray alignment. The Fuel Pool Cooling And Cleanup is generally open during normal operations. When the containment is isolated and containment heat removal with an RHR heat exchanger is not available, motor operated valves must be opened to align either the FPCC vent or the RHR Spray vent. If motor operated valves associated with a venting pathway fail to open remotely from the control room, local manual opening of the motor operated valves located inside is containment is not possible. A passive vent design change would provide a passive rupture disk in an alternate vent path to ensure that the containment pressure would remain below the containment overpressure threshold limit.

The results of the passive vent design change to ensure the opening of a vent release path before the containment overpressure threshold limit are shown in Table 4.6.3-1. The favorable containment performance results from the passive vent are: increased arrest of core damage in-vessel with no RPV failure changing from 50.8% of core damage frequency to 61.0% (20% change), decreased containment structural failure frequency from 4.03×10^{-6} to 9.92×10^{-7} (75% change), and decreased RPV failure & early containment failure with pool bypass frequency from 2.04×10^{-6} to 4.48×10^{-7} (78% change).

4.6.3.2 ATWS Alternate Shutdown & ADS Inhibit Design Consideration

The ATWS alternate shutdown & ADS inhibit sensitivity was considered to address the critical ATWS sequences with represent 4.4% of the core damage frequency and have containment failure before core damage. The front-end sensitivity analysis in section 3.4.1.6 included the impact of a design consideration to provide automatic ADS inhibit which would reduce the internal and flooding core damage frequency 19%. The containment performance improvement would provide an enhanced Plant Emergency Instruction to control the RPV Power/Level as a function of containment pressure with an RPV water level control band just above the Minimum Steam Cooling Water Level. This new water level control band just above the Minimum Steam Cooling Water Level (at 30 inches below the Top of the Active Fuel) would reduce the reactor power within the containment vent heat removal capability. This quasi steady-state RPV Power/Level control would provide a reasonable time for reactor shutdown recovery using an alternate boron injection system. A factor of 90% recovery from the Critical ATWS sequences was modeled in the sensitivity analysis.

The containment performance results of ATWS Alternate Shutdown & ADS Inhibit are shown in Table 4.6.3-1. The favorable containment performance results with the ATWS improvements are frequency reductions for RPV failure and containment failure: the containment structural failure frequency decreases from 4.03 x 10^{-6} to 3.49 x 10^{-6} (-13% change), and the RPV failure & early containment failure with pool bypass frequency decreases from 2.04 x 10^{-6} to 1.75 x 10^{-6} (14% change).

4.6.3.3 Hydrogen Ignition System Design Change Consideration

Supplement No. 3 of Generic Letter 88-20 directed that Mark III containments are expected to evaluate the vulnerability to interruption of power to the hydrogen ignitors. This staff recommendation originated from NUREG-1417 (NRC 1990) entitled Safety Evaluation Report Related To Hydrogen Control Owners Group Assessment of Mark III Containments. On the basis of its evaluation, the staff found the Hydrogen Control Owners Group typical report "Generic Hydrogen Control Information for BWR-6 Mark III Containments" provided an acceptable basis for the technical resolution of the Mark III containment degraded-core hydrogen control issue. Additional plant-specific analysis will confirm that the equipment necessary to establish and maintain safe shutdown and to maintain containment integrity will be capable of performing its functions during and after exposure to the environmental conditions resulting from hydrogen generation in all recoverable degraded-core severe-accident scenarios. The staff's assessment identified that a plant specific evaluation of the adequacy of the hydrogen ignition system (HIS) alternate power supply should be included as part of the individual plant examination of the plants with Mark III containments. NUREG-1417 noted that an important factor in this decision process is the level of risk associated with an SBO event leading to core damage. The Grand Gulf Mark III plant risk study included in NUREG-1150 (NRC 1989) show the overall core melt frequency very low (i.e., 1 x 10^{-6}) and that SBO conditions dominate the residual risk from severe accidents.

The modification of the electrical power supply to ensure the availability of the hydrogen ignition system (HIS) during SBO would remove the possibility of high containment loads from hydrogen deflagrations and detonations. The HIS backup power supply sensitivity assumes 100% HIS availability (and no unavailability due to human interaction error).

The containment performance results of the HIS backup power supply are shown in

Table 4.6.3-1. The favorable containment performance results with the HIS backup power supply are frequency reductions for containment failure: the containment structural failure frequency decreases from 4.03×10^{-6} to 3.76×10^{-6} (-6.7% change), and the RPV failure & early containment failure with pool bypass frequency decreases from 2.04×10^{-6} to 2.00×10^{-6} (-2% change).

4.6.3.4 Combined Passive Vent And ATWS Alternate Shutdown & ADS Inhibit Design Change Consideration

The combination of these two design consideration changes discussed above would reduce the core damage frequency from 1.27×10^{-5} to 8.01×10^{-6} (-37% change), and also reduce containment failure before core damage frequency from 2.90×10^{-6} to 5.3×10^{-8} (-98% change).

The containment performance results of this combined passive vent and ATWS modification are shown in Table 4.6.3-1 The favorable containment performance results with comparison to the base case are: increased arrest of core damage in vessel from 50.8% of core damage to 64.1% (26% change), decreased containment structural failure from 4.03×10^{-6} to 4.60×10^{-7} (-89% change), and decreased RPV failure and early containment failure with pool bypass frequency from 2.04×10^{-6} to 1.57×10^{-7} (-92% change).

4.6.3.5 Combined Passive Vent, ATWS Modifications And HIS Backup Power Supply Design Change Consideration

The combination of these three changes discussed above further examines the impact of backup power to the hydrogen ignitors with the passive vent and ATWS modifications.

The containment performance results of this combined passive vent, ATWS modification and HIS backup power supply are shown in Table 4.6.3-1. The favorable containment performance results with comparison to the base case are: increased arrest of core damage in vessel from 50.8% of core damage to 64.1% (26% change), decreased containment structural failure from 4.03×10^{-6} to 1.75×10^{-7} (-96% change), and decreased RPV failure and early containment failure with pool bypass frequency from 2.04×10^{-6} to 1.14×10^{-7}.

4.6.3.6 Updated Initiating Event Frequency

The front-end sensitivity discussion of initiating event frequency in section 3.4.1.6 identified that using updated initiating event frequencies would change the composition of dominate initiating events and reduce the core damage frequency. The updated initiator composition ranking is: SBO at 32% followed by LOOP at 20%, combined transient at 23%, and ATWS at 17%. The reduction in core damage frequency for internal events and flooding is 1.27×10^{-5} to 8.05×10^{-6} (-37% change).

The containment performance results of the updated initiator frequency are shown in Table 4.6.3-2, Impact of Design Changes On Containment Failure Frequency Using Updated Initiator Frequency. Included in this Table are sensitivity cases using the updated initiating event frequency with the four design change considerations discussed above: updated passive vent; updated ATWS Alternate Shutdown and ADS Inhibit; Updated Passive Vent and ATWS Modification; and Updated Passive Vent, ATWS Modifications and HIS Backup Power Supply.

The results of the updated initiating event frequency sensitivity analysis are compared with the results of the previous generic initiating event sensitivity analysis in Table 4.6.3-3. The impact of the updated initiating event frequency on the containment performance results of the design change considerations is significant.

4.7 RADIONUCLIDE RELEASE CHARACTERIZATION

The binned end points of the Accident Progression Event Tree (APET) represent the outcomes of possible in-containment accident progression sequences. These endpoints represent complete severe accident sequences from initiating event to release of radionuclides to the environment. The Level 1 system information is passed through to the Containment evaluation in discrete plant damage states. An atmospheric source term may be associated with each of these Containment sequences. Because of the large number of APET sequences and because of similarities in the sequence characteristics, however, it is neither necessary nor practical to develop a source term estimate for each Containment sequence. Sequences with similar characteristics are therefore grouped into release categories to reduce the required source term assessment effort.

4.7.1 Release Category Grouping Parameters and Grouping Logic

The first step in source term assessment effort is to identify the sequence characteristics which are most important to definition of the source term. These characteristics are identifiable from the Plant Damage State (PDS) characteristics and from the APET sequence characteristics since one of the primary objectives in the PDS grouping and APET evaluation has been to define those events and conditions most important to source term assessment. This select set of sequence characteristics important to source term assessment are used as grouping criteria to define the release categories and the associated source term magnitude, composition and timing.

The Containment sequence characteristics selected for use in definition of the Perry source term release categories are:

Containment Bypass (Event V or Main steam Line Break Outside Containment)
Debris Cooled In-Vessel
Containment Status At Core Damage
Time of Containment Failure (relative to core damage)
Mode/Location of Containment Failure
Suppression Pool Bypass
Spray Operation
Type of Core Concrete Interactions

The approach to the definition of release categories in similar to that discussed in Section 4.3 for plant damage state definition. It consisted of construction of a logic diagram with the grouping criteria defined above, as headings. The end points on the logic diagram represent unique release (source term) categories with their individual characteristi fined by the pathway through the logic diagram.

The goal of the grouping process is to develop the minimum number of release categories necessary to distinguish the important combinations of sequence characteristics that can result in distinctly different atmospheric source terms. The Source Term Grouping Logic diagram developed for the Perry IPE is shown in Figure 4.7-1. It defines 25 release categories. It is applicable for both the internal initiators and for internal flooding initiators.

The reasons for the selection of these parameters for definition of the Perry release categories and specific branch assignment decisions used in the logic diagram under each decision heading are discussed below.

Containment Bypass

Containment natural and engineered mitigation features (including scrubbing in the suppression pool) are ineffective in reducing fission product releases if the accident causes opening of a leakage path directly from the reactor vessel to a point outside of the Containment boundary which bypasses the drywell and containment gas volumes.

The two main ways that this can occur are if an unisolable steam line break outside of containment occurs or an interfacing system LOCA (Event V) occurs. These are both defined by the plant damage state attribute "Containment Bypass".

For interfacing system LOCAs the failures occur into the Auxiliary Building because of the failure of the check valves between the reactor vessel and the low pressure injection systems (or low pressure portions of high pressure injection systems), and subsequent failure of the low pressure piping outside of the containment. The factors of interest to source term assessment is whether or not the release point is above or below water in the Auxiliary Building at the time fission product releases are occurring and whether there is substantial deposition of radionuclides on Auxiliary Building surfaces. For unisolated steam line breaks the important considerations are the extent of radionuclide deposition in the Steam Tunnel and/or Turbine Building.

For containment bypass sequences, the phenomena that occur in the major containment volumes and containment ESF operation are largely irrelevant (at least until vessel failure), and the containment bypass sequences are assigned to a single source term category without differentiating other containment parameters considered for other sequence types.

Debris Cooled In-Vessel

This characteristic is important (for non-bypass sequences) since there is a significant probability of arresting the core-melt process in-vessel, thus preventing vessel failure, ex-vessel release (core concrete interaction) processes. (possibly) containment failure and reducing the magnitude of fission product release. This characteristic is only considered if Containment is not bypassed. Interfacing system LOCAs (Event V) and main steam line breaks outside containment generally preclude long-term cooling as reactor vessel and containment water inventory is lost from containment.

If the debris is cooled in-vessel (and reactor vessel failure prevented) then those processes involving ex-vessel debris interactions will be absent (ex-vessel debris coolability and debris concrete attack) and need not be considered in defining the source term categories.

However, sufficient hydrogen may be produced to threaten containment integrity. In addition, for loss of containment heat removal sequences - long term containment overpressure failure may still occur. Containment failure (by either of the above mechanisms) may also result in pool bypass.

Containment Status At Core Damage

This attribute is considered important because any fission products in the containment atmosphere are released to the Intermediate Building/environment early (i.e., near the time of core melt) and continuously, if the containment is not isolated or has failed prior to core damage initiation.

In this case, the effective available time for fission product deposition and possible spray washout in containment is reduced. The size of the most likely isolation failure path (Fuel Pool Cooling Return with an effective area of .492 square feet) is large enough so that even if a later larger area containment failure were to occur, it would not significantly increase the radionuclide release magnitude. As discussed in [illegible] 3.2.1, the Backup Hydrogen Purge line may be open initially during [illegible] the radionuclide release is limited by the maximum potential [illegible] this system (50 scfm). Sequences with successful in-vess[illegible] but without containment heat removal and with failure of venting will result in overpressurize failure of containment prior to core damage initiation. For sequences with the containment failed before core damage, it is assumed that the leak path is directly to the atmosphere, or, if to the Intermediate Building, to a location where further attenuation is not effective. This assumption is partly based on conservatism and partially on a review of the available release pathways.

Time Of Containment Failure

This release category attribute is considered important because it affects the time available for fission product release mitigation by natural removal processes and spray washout. It applies to all core damage sequences that do not involve containment bypass, or loss of isolation.

The times selected as significant are: At (or Before) Reactor Vessel Failure, and Late Failure (many hours after vessel failure). The possibility of no containment failure exists and is assigned to its own unique source term category.

Mode Of Containment Failure

This attribute is important because it governs the rate at which fission products are released to the atmosphere. It also affects the magnitude of the release by governing the time available for effective fission product attenuation in containment.

The three "failure" modes considered significant are: penetration failures, anchorage failures, and containment venting.

This attribute is only considered for those sequences evaluated to have an early or late containment failure. This attribute is clearly not a discriminant for sequences with no containment failure. This attribute is not relevant, (or at least not significant) for containment bypass sequences, as most of the fission products escape through the bypass. The mode of containment failure is also considered not relevant for other sequences that have containment failure before core damage since the containment will have already been depressurized prior to core damage.

Suppression Pool Bypass

Scrubbing in the suppression pool can be an effective mitigation mechanism for in-vessel radionuclide releases or from debris/concrete attack in the reactor pedestal cavity. If the pool is bypassed during periods of fission product release, then this attenuation process will not be effective. Pool bypass can occur as the result of containment failure (e.g., containment anchorage failure which allows the pool to drain down below the level of the SRV quenchers and/or the horizontal vents, as a result of drywell failure (e.g., a pedestal steam explosion causes pedestal failure which then fails the drywell) or due to failures of systems interfacing the drywell and containment gas space (e.g., a drywell vacuum breaker sticks open). Pool bypass will not have a significant impact on the source term if the containment does not fail (and is isolated). If the containment fails via the anchorage failure mode, it is assumed that pool bypass occurs.

Containment Spray Operation

This attribute is considered significant because it determines whether or not fission product washout by sprays occurs in the containment. This attribute also affects the energy level (i.e., temperature) of the release.

Spray operation is only used as a grouping parameter for sequences with pool bypass since if the suppression pool is in the radionuclide release pathway, an effective engineered safety feature radionuclide removal mechanism is already present. Consequently, for sequences with the suppression pool not bypassed - no branching is done and it is assumed (for calculating source term magnitudes) that the sprays are not operating. For sequences with the pool bypassed, the sprays must be operational over the entire time period when radionuclide release is occurring in order to follow the successful spray operation branch.

This characteristic is not considered for containment bypass sequences, the sprays do not attenuate the important in-vessel releases. It is irrelevant for all sequences in which containment failure does not occur as no significant release would occur.

Type Of Core Concrete Interaction

Three types of core debris are considered. No CCI - where the debris is cooled in the drywell and little or no CCI occurs. Dry CCI - where water is not supplied to the drywell following vessel failure or where pool overflow into the drywell has not occurred. Wet CCI - where a source of water to the drywell is present, however the debris is not coolable and CCI occurs in the presence of an overlying water layer.

Dry CCI cases and wet CCI cases result in significantly different fission product source terms since for CCI occurring with an overlying water pool the aerosols and radionuclides will be scrubbed by the overlying water pool. This can be particularly important for sequences where the suppression pool is bypassed.

4.7.2 Release Category Grouping Logic Tree and Release Category Characteristics

The Source Term Category Grouping Logic figure developed for the Perry IPE is shown as Figure 4.7.1-1. It defines 25 release categories.

4.7.2.1 STC Grouping Logic Heading Definitions

The following STC Grouping Logic Headings are used in Figure 4.7.1-1.

CNTMT_BYP Containment Bypass

A containment bypass sequence with core damage results in a radiological release to the environment. Potential bypass sequences include the traditional Event V, Interfacing LOCAs, and Main Steam Line Breaks. The STC grouping logic branches are labeled No and Yes.

RPV_FAIL Debris Cooled In-Vessel

If the debris is cooled in-vessel (and reactor failure is prevented) then those processes involving ex-vessel debris interactions will be absent. The STC grouping logic branches are labeled RPV Failure and No RPV Failure.

CNTMT_INT Containment Status At Core Damage

The success of containment at core damage maintains the radiological boundary. The failure of containment at, or before, core damage allows an early radiological release. The failure of the containment may result in the loss of injection due to the physical interaction between the containment structure and the injection systems. The STC grouping logic branches are labeled Containment Failed Before Core Damage and Intact.

TIME_CF Time Of Containment Failure

The time of containment failure is characterized for the sequences that do not involve containment bypass or containment failed before core damage. The STC grouping logic branches are labeled: Early Containment Failure, Late (many hours after RPV failure) Containment Failure, and No Containment Failure.

MODE_CF Mode Of Containment Failure

The modes of containment failure are: anchorage failure, penetration failure, and containment vent. The modes of containment failure are applied to the Early Containment Failure and Late Containment Failure branches.

POOL_BYP Suppression Pool Bypass

Suppression Pool Bypass is evaluated for all sequences with containment failure. The STC grouping logic branches are labeled Early or Late Pool Bypass and No Pool Bypass.

SPRAYS Containment Spray Operation

Containment spray operation can be applied to sequences with suppression pool bypass. The STC grouping logic branches are labeled Late or No spray. [It should be noted that the spray operation assessment would be most effective on dominate STC branches. During the course of the IPE assessment the contribution of SBO and ATWS initiators shifted and the grouping logic was not enhanced in this area due to time constraints. The application of this attribute demonstrates the potential benefit of this Engineered Safety System.]

TYPE_CCI Type of Core Concrete Interaction

Three types of Core Concrete Interaction are considered in section 4.7.1. However, to simply and conservatively model the source term release fraction for the IPE only two types of CCI are shown on the STC grouping logic: Dry CCI and No CCI. Wet CCI sequences are conservatively combined with Dry CCI sequences. Source term release assessment using all three categories for RPV Failure would develop a model which would require a more exhaustive quantification effort.

4.7.3 Release Category Source Term Characteristics

The determination of the source term magnitude, composition, and timing for each release category were performed with the MAAP 3.0B code. MAAP calculations were performed for eleven Source Term Category groups: 1, 2, 3, 5, 8, 10, 11, 13, 14, 21 and 23. The release fractions for the other categories were then characterized by similarity to one of the calculated MAAP source terms.

The representative sequences used for the MAAP release category calculations were selected on the basis of having the required characteristics as specified in the release category definitions, having a significant frequency of occurrence (at the Plant Damage State level), and having a significant containment failure probability. The sequences modeled for the source term release categories are listed in Table 4.7.3-1. The calculated times of occurrence of events important to radionuclide release are listed in Table 4.7.3.2. The calculated radionuclide release fractions for the analyzed release categories are shown in Table 4.7.3-3. The release fractions in this table are listed by MAAP "species" which are described below. The bases for characterizing the unanalyzed source term release categories are primarily the containment failure time and mode, and containment spray operation.

Fission product release in the MAAP code from the core in-vessel or core debris ex-vessel is modeled with "frozen" chemical states defined by the twelve species listed in Table 4.7.3-5. The chemical state is important in determining the transition between vapor and aerosol forms which affects the deposition and retention of fission products.

Each of the 12 MAAP fission product species can exist in up to four states in each region of the containment and reactor vessel. These states are "vapor", "aerosol", "deposited", and "contained in the core or corium". These states and the 12 species are used to characterized the MAAP calculated source term release.

4.7.4 Release Category Point Estimate Frequencies and Dominant Sequences

Twenty five source term release categories were developed for the containment evaluation sequences. The nine dominant source term categories which contribute 95% of the source term release are listed in Table 4.7.4-1. Table 4.7.4-2 presents the contribution of the notable individual Plant Damage State groups (those PDS groups that contribute greater that 0.1% of the core damage) to each source term category. The dominant core damage sequences in the PDS groups shown as being important to a STC group are included in Table 4.3.3-1.

Dominant Source Term Categories 24 and 12 (ranked 1 and 4, respectfully) represent No Containment Failure with No RPV Failure and with RPV Failure and contribute 27.7% and 14.4%, respectfully, of the source term category release fraction.

Dominant Source Term Category 23, ranked 2, represents No RPV Failure, Containment Intact At Core Damage and Late Venting and No Pool Bypass. This venting category contributes 19.2% of the release fraction.

Dominant Source Term Category 1, ranked 3, represents RPV Failure with Containment Failed at Core Damage and Pool Bypass. This source term category contributes 15.0% of the release fraction.

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Table 4.1.1-1

COMPARISON OF MARK III CONTAINMENT DESIGN CHARACTERISTICS

	Perry	Grand Gulf
Total Free Volume (Mft^3)	1.44	1.67
Pool Volume (Mft^3)	0.12	0.14
Containment Volume/Thermal Power (ft^3/kW)	0.40	0.44
Containment Pool Volume/Thermal Power (ft^3/kW)	0.032	0.037
Containment Design Pressure		
Internal (psig)	15	15
External (psi)	0.8	—
Drywell Design Pressure		
Internal (psi)	30	30
External (psi)	21	21

TABLE 4.3.3-1 DOMINANT PLANT DAMAGE STATES

Rank	PDS GROUP	FREQUENCY	CDF%	DOMINANT SEQUENCES	CDF%
1	53	4.44E-6	34.9%	T2-C-U3-X'	17.8%
				T2-C-Lc-C1	4.9%
				T2-C-U3-X	2.2%
				T3B-C-U3-X'	2.2%
				T2-CB	1.9%
				U-V-Va	1.1%
				T2-CA-V'	1.0%
2	56	3.40E-6	26.7%	13B-U1-U2-Va1	8.9%
				TIA-U2-U1-V-Va-Wc-Ws	5.9%
				B6-V-Wc	3.0%
				F1D-U3-U2-U1-V-Va	1.8%
				A-U1-V-Wc-Ws	1.6%
				T15-Va	1.6%
3	73	1.22E-6	9.6%	T2-Wc-Ws-Y-Cv-Li	8.2%
4	61	6.23E-7	4.9%	T15-Va-R3	2.6%
				UR-V-Va-R3	1.9%
5	71	5.59E-7	4.4%	T2-Wc-Ws-Y-Cv	2.2%
				TIA-U2-Wc-Ws-Y-Cv	1.5%
6	65	2.93E-7	2.3%	T2-CA-C1-Wc	1.1%
7	67	2.65E-7	2.1%		
8	69	2.56E-7	2.0%		
9	1	2.51E-7	2.0%	B9-Va	1.8%
10	25	2.09E-7	1.6%	B7-R3	1.6%
11	63	1.69E-7	1.3%	T2-CA	1.2%
12	32	1.26E-7	1.0%		
13	36	1.02E-7	0.8%		
14	66	1.00E-7	0.8%		
15	9	9.73E-8	0.8%		

TABLE 4.4.1-1 ULTIMATE CONTAINMENT FAILURE MODES

FAILURE MODE	FAILURE PRESSURE 5th Percentile	FAILURE PRESSURE Median
1. Dome Knuckle	65.1 psig	86.4 psig
2. Dome Apex	86.7	115.1
3. Cylinder	86.6	111.7
4. Personnel Airlock	94.0	119.2
5. Equipment Hatch (Bolts)	60.8	77.1
6. Penetration P123	60.0	77.3
7. Penetration P205	58.9	75.5
8. Penetration P414	57.4	74.0
9. Anchorage Steel	104.9	135.3
10. Anchorage Concrete	64.9	92.0
CONTAINMENT COMPOSITE FAILURE	53.5	64.3

TABLE 4.4.1-2 EXPECTED TYPE OF CONTAINMENT FAILURE

FAILURE MODE	LEAKAGE	RUPTURE	GROSS RUPTURE
1. Dome Knuckle		X	
2. Dome Apex		X	
3. Cylinder			X
4. Personnel Airlock	< Median	> Median	
5. Equipment Hatch (Bolts)	< Median		> Median
6. Penetration P123	< Median	> Median	
7. Penetration P205	< Median	> Median	
8. Penetration P414	< Median	> Median	
9. Anchorage Steel			X
10. Anchorage Concrete			X

NOTE: Median designates the median failure pressure for the associated containment failure mode. Leakage failure is expected to commence below the median failure pressure, and when the median failure pressure is exceed then transition occurs to a larger failure type, rupture or gross rupture.

TABLE 4.4.2-1 ULTIMATE DRYWELL FAILURE MODES

FAILURE MODE	MEAN FAILURE PRESSURE	
1. Drywell Wall	98.5 psid	
2. Drywell Roof	67.0	
3. Drywell Head	77.8 83.3	(internal) (external)
4. Equipment Hatches	84.7	
5. Personnel Airlocks	107.2	

TABLE 4.5.1-1 PERRY IPE ACCIDENT PROGRESSION EVENT TREE QUESTIONS

Question Number

PLANT DAMAGE STATE GROUPING LOGIC

1. Not A Containment Bypass Sequence?

No Bypass, Event V or Main Steam Line Break

2. Containment Status At Core Damage?

Containment Intact, Containment Failed.

3. Event Type?

For Containment Intact At Core Damage:

SBO, LOOP With No HVAC, Other Types

For Containment Failed At Core Damage:

Critical ATWS, LOOP & SBO, Others

4. Initial Containment Heat Removal With Suppression Pool Cooling?

For LOOP With No HVAC:

Not Available, Initial Suppression Pool Cooling

5. Containment Vent Isolated At RPV Failure?

For SBO Sequences: Isolated, Not Isolated

6. RPV Injection Failure Time?

For Containment Intact At Core Damage and SBO Sequences:

0 - 2.8 hrs (characterizes no injection)
2.8 - 4.2 hrs (characterizes RCIC availability)
> 4.2 hrs (characterizes HPCS & Firewater availability)

7. Offsite Power Recovery Time?

Prior To RPV Failure, Prior To Containment Limit, No Recovery

8. Containment Heat Removal With RHR Spray Loop?

RHR Spray, RHR Suppression Pool Cooling, No RHR

9. Containment Heat Removal With Vent?

Vent, No Vent

10. Late In-Vessel Injection & Pedestal Cavity Supply?

    Yes, No

11. RPV Depressurized During Core Damage?

    Yes, No

CET EVENT 1. DEBRIS COOLED IN-VESSEL

12. Late RPV Low Pressure Injection Available?

    Water Injection, No Injection, Critical ATWS

13. RPV Depressurized During Core Damage?

    Low Pressure, High Pressure

14. Debris Mass Molten At RPV Failure?

    Large Debris, Small Debris

15. Debris Cooled In-Vessel?

    Cooled In-Vessel, Not Cooled In-Vessel

CET EVENT 2. MODE OF CONTAINMENT FAILURE BEFORE RPV FAILURE

16. Hydrogen Ignition System Available?

    HIS On, HIS Off

17. Containment Vent Isolated Before RPV Failure?

    Isolated, Not Isolated

18. Mode Of RHR Spray Operation Early?

    Controlled, Design Cooling, No Spray

19. Containment Steam Concentration Before RPV Failure?

    0-15%, 15-25%, 25-35%, 35-45%, 45-55%, >55%

20. Fraction Zirconium Inventory Reacted In-Vessel?

11%, 22%, 33%

21. Small H2 Burns At Low H2 Concentration?

No Small Burns, Small Burns

22. Large H2 Burn During Core Damage?

No Burn Ignited, Large Burn Ignited

23. H2 Detonation Containment Failure Before RPV Failure?

Detonation Containment Failure, No

24. Containment Failure Before RPV Failure?

Failure, No Failure

25. Mode Of Containment Failure Before RPV Failure?

Anchorage, Penetration-Dome or No Failure

CET EVENT 3. INJECTION & SPRAY FAILURE DUE TO CONTAINMENT FAILURE BEFORE RPV FAILURE

26. Containment Failure Before RPV Failure Impact on ECCS Injection and Spray Piping?

No Failure, Failure

27. Containment Failure Before RPV Failure Impact on ECCS Injection and Spray Motors?

No Failure, Failure

28. Containment Failure Before RPV Failure Steam and Radiation Impact On Firewater Injection?

No Failure, Failure

29. Injection & Spray Failure Due To Containment Failure Before RPV Failure?

No Failure, Injection & Spray Failure

CET EVENT 4. DRYWELL FAILURE AT/NEAR RPV FAILURE

30. Alpha Mode Steam Explosion Drywell and Containment Failure?

Alpha, No Alpha

31. Mode Of In-Vessel Steam Explosion Bottom Head RPV Failure?

    Alpha, Large, Small, No In-Vessel Steam Failure

32. RPV Failure Mode & Failure Size?

    Alpha mode, No RPV Failure Debris Cooled In-Vessel, Small Size Bottom Head RPV Failure, Large Size Bottom Head Failure

33. Water In Pedestal At RPV Failure?

    Flooded + Injection, Residual Water + Injection, Flooding, Residual Water Only

34. Pedestal Failure Due To Overpressure At RPV Failure?

    Pedestal Failure, No Failure

35. Pedestal Cavity Steam Explosion?

    Steam Explosion, No Explosion

36. Pedestal Failure Due To Steam Explosion?

    Pedestal Failure, No Failure

37. Drywell Failure Due To Pedestal Failure?

    Drywell Failure, No Failure

38. Drywell Overpressure Failure At RPV Failure?

    Drywell Failure, No Failure

39. Drywell Failure At/Near RPV Failure?

    Drywell Failure, No Failure

CET EVENT 5. MODE OF CONTAINMENT FAILURE AT/NEAR RPV FAILURE

40. Containment Steam Concentration At/Near RPV Failure?

    0-15%, 15-25%, 25-35%, 35-45%, 45-55%, >55%

41. Fraction Zirconium Inventory Reacted At/Near RPV Failure?

    11%. 22%, 33%

42. Hydrogen Ignition Sources Available At RPV Failure?

No Ignition Source, Ignition Source

43. High Pressure Melt Ejection?

HPME, No HPME

44. Large H2 Burn At/Near RPV Failure?

No Large Burn Ignited, Large Burn Ignited

45. Hydrogen Detonation Containment Failure At/Near RPV Failure?

Detonation Containment Failure, No Failure

46. Containment Failure At/Near RPV Failure?

Failure, No Failure

47. Mode Of Containment Failure At/Near RPV Failure?

Anchorage, Penetration-Dome or No Failure

CET EVENT 6. POOL BYPASS BEFORE/NEAR RPV FAILURE

48. Drywell Failure Due To Containment H2 Burn Before/Near RPV Failure?

Drywell Failure, No Drywell Failure

49. Pool Bypass Before/Near RPV Failure?

Pool Bypass, No Pool Bypasss

CET EVENT 7. INJECTION & SPRAY FAILURE DUE TO CONTAINMENT FAILURE AT/NEAR RPV FAILURE

50. Containment Failure At/Near RPV Failure Impact On ECCS Injection And Spray Piping?

No Failure, Failure

51. Containment Failure At/Near RPV Failure Impact on ECCS Injection And Spray Motors?

No Failure, Failure

52. Containment Failure At/Near RPV Failure Steam & Radiation Impact

On Firewater System?

No Failure, Failure

53. Injection & Spray Failure Due To Containment Failure Before/Near RPV Failure?

No Failure, Injection & Spray Failure

CET EVENT 8. PEDESTAL FAILURE DUE TO CORE CONCRETE INTERACTION

54. Type Of Core Debris Concrete Interactions?

Dry CCI, Fast-Wet, Slow-Wet, No CCI

55. Pedestal Failure Due To Core Debris Concrete Interaction?

At Vessel Breach, After Vessel Breach, No Failure

CET EVENT 9. MODE OF LATE BURN & OVERPRESSURE CONTAINMENT FAILURE

56. Mode Of RHR Spray Operation Late?

Controlled, Design Cooling, No Spray

57. Hydrogen Ignition Sources Available Late?

No Source, Hydrogen Ignition Source

58. Containment Steam Concentration Late?

0-15%, 15-25%, 25-35%, 35-45%, 45-55%, > 55%

59. H2 Combustion Before/At RPV Failure?

Early Burn, No Early Burn

60. Containment Effective Hydrogen Concentration Late?

< 4%, 4-8%, 8-12%, 12-16%, 16-20%, > 20%

61. AC Power Available Late?

AC Late, No AC Late

62. Large H2 Burn Late?

No Burn Ignited, Large Burn Ignited

63. Hydrogen Detonation Late Cont[illegible]nment Failure?

Detonation Containment Failure, No

64. Hydrogen Burn Late Containment Failure?

Failure, No Failure

65. Containment Status At Accident Progression Progression?

Early Containment Failure, Late Containment Failure, Vent, No Containment Failure

66. Mode Of Late Hydrogen & Overpressure Containment Failure?

Anchorage, Penetration-Dome or No Failure

CET EVENT 10. LATE POOL BYPASS

67. Drywell Failure Due To Late Hydrogen Burn In Containment?

Drywell Failure, No Failure

68. Pool Bypass Late?

Late Pool Bypass, No Late Bypass

TABLE 4.5.3-1 APET CONTAINMENT PERFORMANCE BASE CASE RESULTS

		FREQUENCY	FRACTION OF CDF
No RPV Fail:	No Containment Failure	3.39E-6	26.7%
	Vent	2.45E-6	19.3%
	Containment Failure	6.18E-7	4.9%
Subtotal No RPV Failure Core Damage Frequency:		6.46E-6	50.8%
RPV Fail:	No Containment Failure	1.58E-6	12.4%
	Vent	1.27E-6	10.0%
	Late Containment Failure	9.38E-7	7.4%
	Early CF: No Pool Bypass	4.30E-7	3.4%
	Late Pool Bypass	1.54E-6	12.1%
	Early PB, Spray	6.12E-8	0.5%
	Early PB, No Spray	4.45E-7	3.5%
Subtotal RPV Failure Core Damage Frequency:		6.27E-6	49.2%
TOTAL CORE DAMAGE FREQUENCY:		1.27E-5	100.0%
Subtotal Containment Venting Frequency:		3.72E-6	29.2%
Subtotal Cntmt Structural Failure Frequency:		4.03E-6	31.7%
TOTAL CONTAINMENT FAILURE & VENTING FREQUENCY:		7.76E-6	60.9%
RPV FAILURE AND EARLY CONTAINMENT FAILURE WITH POOL BYPASS FREQUENCY:		2.04E-6	16.1%

TABLE 4.5.3-2 APET CONTAINMENT PERFORMANCE BASE CASE RESULTS FOR DOMINATE PDS GROUPS

Rank	PDS No.	FRACTION OF TOTAL CNTMT FAILURE AND VENTING FREQ (%)	CORE DAMAGE FREQUENCY (per yr)	CORE DAMAGE FREQUENCY (%)	CONTAINMENT PERFORMANCE: NO RPV FAILURE: No CF	VENT	CF	Total	RPV FAILURE: No CF	Vent	Late CF	Early Containment Failure: No Pool Byp	Late PB	Early PB, Spray	Early PB, No Spy	Total	CONTAINMENT FAILURE AND VENTING COMPOSITE: CONTAINMENT STRUCTURAL FAILURE: Penet	Anchor	VENTING
1	56	43.9%	3.40E-6	34.0%	00.0%	68.9%	00.0%	68.9%	00.0%	30.1%	00.0%	00.0%	00.0%	0.00%	1.00%	31.1%	1.0%	0.0%	99.0%
2	73	15.8	1.22E-6	9.6	00.0	00.0	00.0	00.0	00.0	00.0	00.0	00.0	84.1	0.00	15.9	100.0	85.0	15.0	0.0
3	61	8.0	6.23E-7	4.9	00.0	00.0	00.0	00.0	00.0	00.0	99.0	00.0	0.00	0.00%	1.00	100.0	85.1	14.9	0.0
4	71	7.2	5.59E-7	4.4	00.0	00.0	69.2	69.2	00.0	00.0	00.0	19.8	5.48	0.00	5.49	30.8	85.0	15.0	0.0
5	65	3.8	2.93E-7	2.3	00.0	00.0	00.0	00.0	00.0	00.0	00.0	46.9	37.2	0.00	15.9	100.0	85.0	15.0	0.0
6	67	3.4	2.65E-7	2.1	00.0	00.0	69.2	69.2	00.0	00.0	00.0	19.8	5.48	0.00	5.50	30.8	85.0	15.0	0.0
7	69	3.1	2.56E-7	2.0	00.0	00.0	00.0	00.0	00.0	00.0	00.0	00.0	84.1	0.00	15.9	100.0	85.0	15.0	0.0
8	63	[illegible]	1.69E-7	1.3	00.0	00.0	00.0	00.0	00.0	00.0	00.0	46.9	37.2	1.29	14.7	100.0	85.0	15.0	0.0
9	25	1.8	2.09E-7	1.6	00.0	00.0	00.0	00.0	33.7	00.0	52.5	0.26	4.11	0.00	9.37	100.0	31.0	35.3	0.0
10	[illegible]	1.3	1.02E-7	0.8	00.0	68.9	00.0	68.9	00.0	30.1	00.0	00.0	0.00	0.00	1.00	31.1	1.0	0.0	99.0
11	66	1.3	1.90E-7	0.8	00.0	00.0	00.0	00.0	00.0	00.0	00.0	46.8	36.5	0.00	16.7	100.0	85.0	15.0	0.0
12	9	0.8	9.73E-8	0.8	00.0	00.0	00.0	00.0	35.8	00.0	63.2	00.0	0.00	0.00	1.00	100.0	20.6	79.4	0.0
13	58	0.8	6.15E-8	0.5	00.0	00.0	00.0	00.0	00.0	98.9	0.05	00.0	0.00	0.00	1.01	100.0	1.0	0.1	98.9
14	62	0.7	5.02E-8	0.4	00.0	00.0	00.0	00.0	00.0	00.0	99.9	00.0	0.00	0.00	0.11	100.0	85.0	15.0	0.0
15	53	0.6	4.44E-6	34.9	69.2	00.0	0.01	69.2	29.8	00.0	00.0	00.0	0.00	1.00	0.00	30.8	1.0	0.0	0.0
16	70	0.4	3.23E-8	0.3	00.0	00.0	00.0	00.0	00.0	00.0	00.0	00.0	83.3	0.00	16.7	00.0	85.0	15.0	0.0
All PDSs		100%	1.27E-5	100%	26.7%	19.3%	4.9%	50.8%	12.4%	10.0%	7.4%	3.4%	12.1%	0.5%	3.5%	49.2%	26.7%	5.0%	29.2%

TABLE 4.5.3-3 DOMINANT CONTAINMENT FAILURE PLANT DAMAGE STATES

Rank	PDS CLASS	FREQUENCY	CNTMT FAILURE PERCENT	CDF PERCENT	DOMINANT SEQUENCES	CDF%
1	56	3.40E-6	43.9%	34.0%	13B-U1-U2-Va1	8.9%
					T1A-U2-U1-V-Va-Wc-Ws	5.9%
					B6-V-Wc	3.0%
					F1D-U3-U2-U1-V-Va	1.8%
					A-U1-V-Wc-Ws	1.6%
					T15-Va	[illegible].6%
2	73	1.22E-6	15.8%	9.6%	T2-Wc-Ws-Y-Cv-Li	8.2%
3	61	6.23E-7	8.0%	4.9%	T15-Va-R3	2.6%
					UR-V-Va-R3	1.9%
4	71	5.59E-7	7.2%	4.4%	T2-Wc-Ws-Y-Cv	2.2%
					T1[illegible]-U2-Wc-Ws-Y-Cv	1.5%
5	65	2.93E-7	3.8%	2.3%	T2-CA-C1-Wc	1.1%
6	67	2.65E-7	3.4%	2.1%		
7	69	2.56E-7	3.3%	2.0%		
8	63	1.69E-7	2.1%	1.3%	T2-CA	1.2%
9	25	2.09E-7	1.8%	1.6%		
10	36	1.02E-7	1.3%	0.8%		
11	66	1.00E-7	1.3%	0.8%		
12	9	9.73E-8	0.8%	0.8%		
13	58	6.15E-8	0.8%	0.5%		
14	62	5.02E-8	0.7%	0.4%		
15	53	4.44E-6	0.6%	34.9%		
16	70	3.23E-8	0.4%	0.3%		

TABLE 4.6.3-1 IMPACT OF DESIGN CHANGES ON CONTAINMENT FAILURE FREQUENCY

		BASE CASE	(1) PASSIVE VENT	(2) ATWS ALTRNATE SHUTDOWN & ADS INHIBIT	(3) HIS BACKUP POWER SUPPLY	(4) PASSIVE VENT, ALT S/D & ADS INHIBIT	(5) PASSIVE VENT, ATWS ALT S/D & ADS INHIBIT, HIS BACKUP POWER SUPPY
<u>No RPV Failure</u>:	No Containment Failure	3.39E-6 (26.7%)	3.39E-6 (32.6%)	1.82E-6 (17.6%)	3.42E-6 (26.9%)	1.82E-6 (22.7%)	1.85E-6 (23.1%)
	Vent	2.45E-6 (19.3%)	2.92E-6 (28.1%)	2.81E-6 (27.2%)	2.45E-6 (19.3%)	3.28E-6 (41.0%)	3.28E-6 (41.0%)
	Containment Failure	6.18E-7 (4.9%)	2.87E-8 (0.3%)	6.18E-7 (6.0%)	5.93E-7 (4.7%)	2.88E-8 (.4%)	3.58E-9 (.04%)
Subtotal No RPV Failure Core Damage Freq:		6.46E-6 (50.8%)	6.34E-6 (61.0%)	5.25E-6 (50.9%)	6.46E-6 (50.8%)	5.13E-6 (64.1%)	5.13E-6 (64.1%)
<u>RPV Failure</u>:	No Containment Failure	1.58E-6 (12.4%)	1.58E-6 (15.2%)	9.06E-7 (8.8%)	1.79E-6 (14.1%)	9.06E-7 (11.3%)	1.12E-6 (14.0%)
	Vent	1.27E-6 (10.0%)	1.52E-6 (14.6%)	1.30E-6 (12.5%)	1.31E-6 (10.3%)	1.54E-6 (19.3%)	1.59E-6 (19.8%)
	Late Containment Failure	9.38E-7 (7.4%)	2.46E-7 (2.4%)	9.39E-7 (9.1%)	7.31E-7 (5.7%)	2.46E-7 (3.1%)	3.25E-8 (0.4%)
	Early CF: No Pool Bypass	4.30E-7 (3.4%)	2.67E-7 (2.6%)	1.85E-7 (1.8%)	4.30E-7 (3.4%)	2.53E-8 (0.3%)	2.50E-8 (0.3%)
	Late Pool Bypass	1.54E-6 (12.1%)	2.26E-7 (2.2%)	1.34E-6 (13.0%)	1.53E-6 (12.0%)	3.58E-8 (0.4%)	2.58E-8 (0.3%)
	Early PB, Spray	6.12E-8 (0.5%)	6.12E-8 (0.6%)	3.74E-8 (0.4%)	5.12E-8 (0.4%)	3.74E-8 (0.5%)	2.74E-8 (0.3%)
	Early PB, No Spray	4.45E-7 (3.5%)	1.61E-7 (1.5%)	3.67E-7 (3.6%)	4.22E-7 (3.3%)	8.36E-8 (1.0%)	6.05E-8 (0.8%)
Subtotal RPV Failure Core Damage Freq:		6.27E-6 (49.2%)	4.06E-6 (39.0%)	5.08E-6 (49.1%)	6.27E-6 (49.2%)	2.87E-6 (35.9%)	2.88E-6 (35.9%)
TOTAL CORE DAMAGE FREQUENCY:		1.27E-5 (100%)	1.04E-5 (100%)	1.03E-5 (100%)	1.27E-5 (100%)	8.01E-6 (100%)	8.01E-6 (100%)
Subtotal Containment Venting Frequency:		3.72E-6 (29.2%)	4.43E-6 (42.6%)	4.11E-6 (39.8%)	3.76E-6 (29.5%)	4.82E-6 (60.2%)	4.87E-6 (60.8%)
Subtotal Cntmt Structural Failure Freq:		4.03E-6 (31.7%)	9.92E-7 (9.5%)	3.49E-6 (33.8%)	3.76E-6 (29.5%)	4.60E-7 (5.7%)	1.75E-7 (2.2%)
TOTAL CONTAINMENT FAILURE & VENTING FREQ:		7.76E-6 (60.9%)	5.43E-6 (52.2%)	7.60E-6 (73.6%)	7.52E-6 (59.1%)	5.28E-6 (65.9%)	5.04E-6 (63.0%)
RPV FAILURE & EARLY CONTAINMENT FAILURE WITH POOL BYPASS FREQUENCY:		2.04E-6 (16.1%)	4.48E-7 (4.3%)	1.75E-6 (16.9%)	[illegible] (15.8%)	1.57E-7 (2.0%)	1.14E-7 (1.4%)

TABLE 4.6.3-2 IMPACT OF DESIGN CHANGES ON CONTAINMENT FAILURE FREQUENCY USING UPDATED INITIATOR FREQUENCY

		BASE CASE	UPDATED INITIATOR FREQUENCY	(1) UPDATED PASSIVE VENT	(2) UPDATED ATWS ALT SHUTDOWN & ADS	(3) UPDATED PASSIVE VENT & ATWS MODS INHIBIT	(4) UPDATED PASSIVE VENT, ATWS MODS & RIS BACKUP POWER SUPPLY
No RPV Failure:	No Containment Failure	3.39E-6 (26.7%)	1.22E-6 (15.2%)	1.22E-6 (18.3%)	7.34E-7 (10.0%)	7.34E-7 (12.4%)	7.60E-7 (12.8%)
	Vent	2.45E-6 (19.3%)	2.36E-6 (29.4%)	2.82E-6 (42.3%)	2.45E-6 (33.5%)	2.91E-6 (49.0°)	2.91E-6 (49.0%)
	Containment Failure	6.18E-7 (4.9%)	4.70E-7 (5.8%)	6.74E-8 (1.0%)	4.70E-7 (6.4%)	2.90E-8 (0.5%)	3.79E-9 (0.06%)
Subtotal No RPV failure Core Damage Freq:		6.46E-6 (50.8%)	4.06E-6 (50.3%)	4.11E-6 (61.6%)	3.65E-6 (50.0%)	3.67E-6 (61.9%)	3.67E-6 (61.9%)
RPV Failure:	No Containment Failure	1.58E-6 (12.4%)	6.22E-7 (7.7%)	6.22E-7 (9.3%)	4.09E-7 (5.6%)	4.09E-7 (6.9%)	6.21E-7 (10.5%)
	Vent	1.27E-6 (10.0%)	1.23E-6 (15.3%)	1.47E-6 (22.1%)	1.24E-6 (16.9%)	1.48E-6 (25.0%)	1.53E-6 (25.7%)
	Late Containment Failure	9.38E-7 (7.4%)	9.25E-7 (11.5%)	2.45E-7 (3.7%)	9.25E-7 (12.7%)	2.45E-7 (4.1%)	3.20E-8 (0.5%)
	Early CF: No Pool Bypass	4.30E-7 (3.4%)	1.88E-7 (2.3%)	2.00E-8 (0.3%)	1.28E-7 (1.7%)	7.37E-9 (0.1%)	7.00E-9 (0.1%)
	Late Pool Bypass	1.54E-6 (12.1%)	7.51E-7 (9.3%)	8.40E-8 (1.3%)	7.04E-7 (9.6%)	2.05E-8 (0.3%)	1.04E-8 (0.2%)
	Early PB, Spray	6.12E-8 (0.5%)	2.82E-8 (0.3%)	2.78E-8 (0.4%)	2.09E-8 (0.3%)	2.09E-8 (0.4%)	1.08E-8 (0.2%)
	Early PB, No Spray	4.45E-7 (3.5%)	2.49E-7 (3.1%)	8.76E-8 (1.3%)	2.30E-7 (3.1%)	7.36E-8 (1.2%)	5.04E-8 (0.8%)
Subtotal RPV Failure Core Damage Freq:		6.27E-6 (49.2%)	4.00E-6 (49.7%)	2.56E-6 (38.4%)	3.66E-6 (50.0%)	2.26E-6 (38.1%)	2.26E-6 (38.1%)
TOTAL CORE DAMAGE FREQUENCY:		1.27E-5 (100%)	8.05E-6 (100%)	6.67E-6 (100%)	7.31E-6 (100%)	5.93E-6 (100%)	5.93E-6 (100%)
Subtotal Containment Venting Frequency:		3.72E-6 (29.2%)	3.59E-6 (44.7%)	4.29E-6 (64.4%)	3.69E-6 (50.4%)	4.39E-6 (74.0%)	4.43E-6 (74.8%)
Subtotal Cntmt Structural Failure		.03E-6 (31.7%)	2.61E-6 (32.5%)	5.34E-7 (8.0%)	2.48E-6 (33.9%)	3.99E-7 (6.7%)	1.15E-7 (1.9%)
TOTAL CONTAINMENT FAILURE & VENTING FREQ:		7.76E-6 (60.9%)	6.21E-6 (77.1%)	4.83E-6 (72.4%)	6.17E-6 (84.4%)	4.79E-6 (80.7%)	4.55E-6 (76.7%)
RPV FAILURE & EARLY CONTAINMENT FAILURE WITH POOL BYPASS FREQUENCY:		2.04E-6 (16.1%)	1.03E-6 (12.8%)	1.99E-7 (3.0%)	9.54E-7 (13.1%)	1.15E-7 (1.9%)	7.16E-8 (1.2%)

TABLE 4.6.3-3 CONTAINMENT PERFORMANCE COMPARISON OF DESIGN CHANGE CONSIDERATIONS

Bypass	Generic Initiating Event Frequency: Core Damage	Generic Initiating Event Frequency: Containment Structural Failure	Generic Initiating Event Frequency: RPV Failure And Early Containment Failure With Pool Bypass	Updated Initiating Event Frequency: Core Damage	Updated Initiating Event Frequency: Containment Structural Failure	Updated Initiating Event Frequency: RPV Failure And Early Containment Failure With Pool
Base Case	1.3E-5	4.0E-6	2.0E-6	8.1E-6	2.6E-6	1.0E-6
Passive Vent	1.0E-5 (-18% CHG)	9.9E-7 (-75% CHG)	4.5E-7 (-78% CHG)	6.7E-6 (-17% CHG)	5.3E-7 (-80% CHG)	2.0E-7 (-81% CHG)
ATWS Mods: Alt Shutdown & ADS Inhibit	1.0E-5 (-19% CHG)	3.5E-6 (-13% CHG)	1.8E-6 (-14% CHG)	7.3E-6 (- 9% CHG)	2.5E-6 (- 5% CHG)	9.5E-7 (- 7% CHG)
Passive Vent & ATWS Mods	8.0E-6 (-37% CHG)	4.6E-7 (-89% CHG)	1.6E-7 (-92% CHG)	5.9E-6 (-26% CHG)	4.0E-7 (-85% CHG)	1.2E-7 (-89% CHG)
Passive Vent, ATWS Mods & Ignitor Power	8.0E-6 (-37% CHG)	1.8E-7 (-96% CHG)	1.1E-7 (-94% CHG)	5.9E-6 (-26% CHG)	1.2E-7 (-96% CHG)	7.2E-8 (-93% CHG)

NOTE: These results are based on an analysis of the core damage sequences included in the plant damage state trees. Thus there are small differences in the impact of changes reported in this table compared with those reported for internal event core damage sequences in section 3.4. These differences do not change the overall conclusions.

TABLE 4.7.3-1 REPRESENTATIVE SEQUENCES FOR RELEASE FRACTION ANALYSIS

STC	MAAP Run No.	Description
1	ST_01	Transient with loss of containment heat removal. Containment overpressure failure results in loss of injection and suppression pool bypass.
2	ST_02	Transient with loss of containment heat removal. Containment overpressure failure results in loss of all injection.
3	ST_03	Transient with loss of injection at 12 hours. RPV failure occurs near containment failure. Containment overpressure failure results in a anchorage failure and suppression pool bypass.
5	ST_05	Transient with loss of injection at 12 hours. RPV failure occurs near containment failure. Containment overpressure failure results in a penetration failure. Suppression pool bypass occurs near the time of containment failure.
8	ST_08	Transient with initial loss of injection. Containment overpressure results in penetration failure late in the accident sequence. Suppression pool bypass occurs near the time of containment failure.
10	ST_10	Transient with initial loss of all injection. RPV failure occurs before the containment vent is opened. Suppression pool bypass occurs near the time of containment venting.
11	ST_11	Transient with initial loss of all injection. RPV failure occurs before the containment vent is opened.
13	ST_13	Transient with loss of containment heat removal. Containment overpressure failure results in loss of injection and suppression pool bypass. Recovery of injection cools the debris in-vessel.
14	ST_14	Transient with loss of containment heat removal. Containment overpressure failure results in loss of injection. Recovery of injection cools the debris in-vessel.
21	ST_21	Transient with initial loss of all injection. Recovery of injection cools the debris in-vessel. Containment overpressure results in penetration failure.
23	ST_23	Transient with initial loss of all injection. Recovery of injection cools the debris in-vessel. Containment overpressure failure results in penetration failure.

TABLE 4.7.3-2 TIME OF OCCURRENCE FOR IMPORTANT EVENTS IN RELEASE CATEGORY CASES

Time in Hours

STC No.	Initial Injection Failure	Core Uncovery	Injection Recovery	RPV Failure	Containment Failure	Containment Vent Open	Suppression Pool Bypass	End Of Source Release Run
1	19.7	22.2	--	27.8	19.7	--	19.7	80.
2	19.7	22.2	--	27.8	19.7	--	--	100.
3	12.0	15.8	--	20.8	20.8		--	100.
5	12.0	15.8	--	20.8	20.8	--	--	100.
8	00.0	0.55	--	1.8	59.0	--	59.0	100.
10	00.0	0.55	--	1.8	--	26.7	26.7	100.
11	00.0	0.55	--	1.8	--	26.7	--	100.
13	19.6	22.1	25.5	--	19.6	--	19.6	60.
14	19.6	22.1	25.5	--	19.6	--	--	60.
21	0.	0.55	1.5	--	50.3	--	--	100.
23	0.	0.55	1.5	--	--	30.0	--	100.

TABLE 4.7.3-3 COMPOSITE SOURCE TERM CATEGORY RELEASE FRACTIONS

Source Term Category	Basis	NOBLES	MAAP Specie Release Fraction CSI	TEO2	SRO	MOO2	CSOH	BAO	LA2O3	CEO2	SB	TE2	UO2
1	MAAP	1.0	2.7E-2	2.4E-2	1.1E-3	3.4E-4	2.9E-2	6.1E-4	4.6E-5	3.6E-4	7.7E-2	4.9E-3	8.4E-7
2	MAAP	.93	3.6E-3	2.0E-3	3.0E-5	4.5E-4	3.6E-3	5.7E-5	9.0E-7	6.3E-6	1.3E-3	8.5E-5	1.6E-6
3	MAAP	1.0	6.0E-2	4.8E-2	1.9E-4	3.1. -4	9.5E-2	1.2E-4	9.0E-6	8.5E-5	1.1E-1	5.9E-3	2.7E-7
4	Rec	(Use STC 5)											
5	MAAP	1.0	8.7E-2	8.0E-2	7.8E-4	7.8E-4	1.3E-1	5.6E-4	2.4E-5	2.4E-4	1.2E-1	4.3E-3	5.4E-7
6	Rec	(Urs STC 5)											
7	Rec	(Use STC 8)											
8	MAAP	.85	6.2E-3	1.6E-2	5.8E-7	5.2E-8	6.5E-3	3.9E-6	2.2E-9	6.6E-9	5.0E-3	3.4E-6	..1E-8
9	Rec	(Use STC 11)											
10	MAAP	.99	1.4E-1	1.0E-1	2.4E-6	1.1E-6	1.1E-1	3.3E-6	3.3E-8	2.2E-7	1.6E-2	8.6E-4	9.3E-8
11	MAAP	.82	3.1E-2	8.6E-3	2.9E-7	2.5E-7	2.9E-2	3.7E-7	1.1E-8	8.3E-8	1.3E-3	5.7E-4	1.0E-8
12	Rec	(Zero Release)											
13	MAAP	.92	9.9E-2	7.6E-3	1.8E-4	9.2E-3	1.0E-1	1.3E-3	1.3E-6	1.9E-6	5.1E-2	< E-8	< E-8
14	MAAP	.89	1.2E-3	1.6E-4	6.5E-6	3.3E-4	1.1E-3	5.5E-5	4.8E-8	5.9E-8	1.9E-3	< E-8	< E-8
15	Rec	(Use STC 13)											
16	Rec	(Use STC 13)											
17	Rec	(Use STC 13)											
18	Rec	(Use STC 13)											
19	Rec	(Use STC 13)											
20	Rec	(Use STC 13)											
21	MAAP	.96	1.4E-6	1.6E-7	< E-8	< E-8	9.1E-6	< E-10	< E-10	< E-10	1.2E-6	< E-8	< E-8
22	Rec	(Use STC 13)											
23	MAAP	.96	2.3E-7	1.4E-8	< E-3	< E-8	8.8E-6	< E-8	< E-8	< E-8	2.0E-7	< E-8	< E-8
24	Rec	(Zero Release)											

TABLE 4.7.3-4 RECOMMENDATIONS FOR RELEASE FRACTIONS FOR UNANALYZED SOURCE TERM CATEGORIES

STC No.	Containment Failure Time	Sprays	Recommended Alternate STC No.	Justification Bases
12,24	No CF		N/A	Assume Zero Release.
4	Early CF	Late	5	Conservatively ignore Late Sprays.
6	Early CF	No	5	Conservatively ignore the No pool bypass case.
7	Late CF	No	8	The rupture site associated with the penetration failure (3.5 sq ft) relatively large and is sufficient to model the anchorage failure (20 sq ft).
9	Late CF	No	11	Conservatively use the venting release which would occur earlier than penetration failure.
15 - 18	Early CF	Late/No	13	These Early CF categories which contribute less than 0.002 of the total release fraction can be conservatively modeled as having the containment failed at core damage with early pool bypass.
19,20,22	Late CF	No	13	The Late CF categories which contribute about 0.0004 of the total release fraction can be conservatively modeled as having the containment failed at core damage with early pool bypass.

TABLE 4.7.3-5 MAAP FISSION PRODUCT SPECIES

Specie Number	Specie I.D.	Composition
1	NOBLES	Noble Gases and Radioactively Inert Aerosols
2	CSI	CSI + RbI
3	TEO2	TeO_2
4	SRO	SrO
5	MOO2	MoO_2
6	CSOH	CsOH + RbOH
7	BAO	BaO
8	LA203	$La_2O_3 + Pr_2O_3 + Nd_2O_3 + Sm_2O_3 + Y_2O_3$
9	CEO3	CeO_2
10	SB	Sb
11	TE2	Te_2
12	UO2	$UO_2 + NpO_2 + PuO_2$

Notes: see following pages

Explanation of Species Notes

Specie (1): The Specie (1) vapors represent the noble gases. The Specie(1) aerosols are used to represent all non-radioactive aerosols (except for water droplets which are tracked separately in the thermal-hydraulic routines). The aerosol and deposited masses represent the core structural materials along with any concrete aerosols generated ex-vessel. The Specie (1) solid aerosols are assumed to have negligible vapor pressure at the temperatures of interest except in the core or core debris. The vapor pressure assumption used in the core and core debris are discussed in the write-ups for subroutines FPRATP and METOXA.

Specie (2): This specie represents the compounds CsI and RbI. all of the iodine is assumed to combine with the alkali fission products since the molar ratio is about 10 to 1 in favor of cesium and rubidium. Due to the dominance of cesium, CsI properties are chosen.

Specie (3): This specie represents tellurium that is oxidized to TeO_2. Tellurium released in-core is assumed to form TeO_2 directly. Tellurium released ex-vessel is assumed to be elemental; it is allowed to oxidize to TeO_2 in the cavity if steam or oxygen are present (see subroutine METOXA).

Specie (4): Strontium is primarily released in elemental form ex-vessel and is assumed to oxidize to SrO in Containment. In-vessel release is also assumed to lead to SrO formation.

Specie (5): this specie is MoO_2. This chemical state is assumed since molybdenum is thought to be mainly released during concrete attack.

Specie (6): this specie includes CsOH and RbOH. It represents any cesium and rubidium that is left over after combination with iodine.

Specie (7): this specie is BaO. Barium behaves similarly to strontium due to its chemical periodicity.

Specie (8): This specie represents the lanthanides. All sesquioxides in the lanthanide series are grouped together due to similar chemical behavior. These are rather nonvolatile, but in-vessel release is allowed. They are believed primarily to be released ex-vessel as monoxides, which are further oxidized in Containment.

TABLE 4.7.3-5 (Continued) MAAP FISSION PRODUCT SPECIES

Specie (9): Cerium behavior is similar to lanthanide behavior but stoichiometry and vapor pressure differ enough to warrant a separate group.

Specie (10): Antimony is released in-vessel and ex-vessel in elemental form.

Specie (11): Tellurium released ex-vessel which doesn't oxidize in the cavity remains "frozen" as Te_2.

Specie (12): Uranium and the transuranics are grouped separately from the other fission products such as cesium because of their different radiological characteristics. These are only released ex-vessel, and are assumed to oxidize (or reduce) to the dioxide form in Containment.

TABLE 4.7.4-1 DOMINANT SOURCE TERM CATEGORIES

Rank	Source Term Category	Frequency	Percent of Total CDF
1	24	3.39E-6	26.65
2	23	2.45E-6	19.23
3	1	1.90E-6	14.96
4	12	1.58E-6	12.42
5	11	8.92E-7	7.00
6	8	6.67E-7	5.26
7	14	4.86E-7	3.82
8	2	4.29E-7	3.37
9	10	3.79E-7	2.97

TABLE 4.7.4-2 IMPORTANCE OF PLANT DAMAGE STATES TO RELEASE CATEGORIES

Total Frequency = 1.27×10^{-5}

Source Term Category	PDS No.	Percent of STC Frequency	STC Frequency	Percent of Total Frequency
STC 1			1.90E-6	14.96%
	PDS 73	64	1.22E-6	
	PDS 69	13	2.56E-7	
	PDS 65	8	1.56E-7	
	PDS 63	5	8.98E-8	
	PDS 71	3	6.14E-8	
	PDS 66	3	5.32E-8	
	PDS 67	2	2.91E-8	
	PDS 70	2	3.23E-8	
STC 2			4.29E-7	3.37%
	PDS 65	32	1.38E-7	
	PDS 71	26	1.11E-7	
	PDS 63	18	7.94E-8	
	PDS 67	12	5.26E-8	
	PDS 66	11	4.68E-8	
STC 3			2.[illegible]6E-8	0.23%
	PDS 25	53	1.56E-8	
	PDS 20	21	6.25E-9	
	PDS 32	14	4.11E-9	
	PDS 9	2	5.10E-10	
	PDS 1	1	3.46E-10	
	PDS 53	1	1.88E-10	
	PDS 8	1	1.55E-10	
STC 4			4.89E-8	0.38%
	PDS 53	68	3.33E-8	
	PDS 25	20	9.71E-9	
	PDS 32	4	1.88E-9	
	PDS 20	3	1.71E-9	
	PDS 1	3	1.65E-9	
	PDS 9	1	3.47E-10	
	PDS 8	1	2.75E-10	

Source Term Category	PDS No.	Percent of STC Frequency	STC Frequency	Percent of Total Frequency
STC 5			6.16E-8	0.48%
	PDS 56	55	3.40E-8	
	PDS 53	18	1.11E-8	
	PDS 61	10	6.26E-9	
	PDS 25	5	2.87E-9	
	PDS 36	2	1.02E-9	
	PDS 20	1	8.51E-10	
	PDS 32	1	6.28E-10	
	PDS 58	1	6.18E-10	
	PDS 1	1	5.15E-10	
STC 6			7.90E-10	0.0062%
	PDS 25	70	5.50E-10	
	PDS 20	10	8.14E-11	
	PDS 32	9	6.95E-11	
STC 7			2.55E-7	2.00%
	PDS 61	36	9.25E-8	
	PDS 25	23	5.80E-8	
	PDS 9	19	4.91E-8	
	PDS 1	4	9.25E-9	
	PDS 62	3	7.52E-9	
	PDS 8	2	5.99E-9	
	PDS 32	1	3.66E-9	
	PDS 4	1	3.20E-9	
	PDS 20	1	2.02E-9	
STC 8			6.69E-7	5.26%
	PDS 61	78	5.24E-7	
	PDS 25	8	5.15E-8	
	PDS 62	6	4.26E-8	
	PDS 9	2	1.24E-8	
	PDS 1	1	4.96E-9	
STC 9			1.45E-8	0.11%
	PDS 40	37	5.40E-9	
	PDS 1	5	7.24E-10	
	PDS 32	2	2.85E-10	
	PDS 8	2	2.55E-10	
	PDS 4	2	2.44E-10	
	PDS 20	1	1.58E-10	

Source Term Category	PDS No.	Percent of STC Frequency	STC Frequency	Percent of Total Frequency
STC 10			3.79E-7	2.97%
	PDS 56	62	2 36E-7	
	PDS 58	16	6.08E-8	
	PDS 37	2	8.13E-9	
	PDS 36	2	7.15E-9	
	PDS 4	2	6.53E-9	
	PDS 38	1	4.06E-9	
STC 11			8.92E-7	7.01%
	PDS 56	88	7.88E-7	
	PDS 36	3	2.36E-8	
	PDS 37	2	1.70E-8	
	PDS 4	2	1.59E-8	
	PDS 38	2	1.41E-8	
STC 12			1.58E-6	12.42%
	PDS 53	84	1.32E-6	
	PDS 25	4	7.04E-8	
	PDS 1	4	5.99E-8	
	PDS 54	2	3.49E-8	
	PDS 9	2	3.48E-8	
	PDS 32	2	2.61E-8	
	PDS 20	1	1.32E-8	
STC 13			8.58E-8	0.67%
	PDS 71	68	5.81E-8	
	PDS 67	32	2.76E-8	
STC 14			4.86E-7	3.82%
	PDS 71	68	3.29E-7	
	PDS 67	32	1.56E-7	
STC 15			1.68E-8	0.13%
	PDS 32	52	8.78E-9	
	PDS 20	45	7.61E-9	
	PDS 53	1	1.14E-10	
	PDS 56	1	8.52E-11	

Source Term Category	PDS No.	Percent of STC Frequency	STC Frequency	Percent of Total Frequency
STC 16			6.06E-9	0.048%
	PDS 20	53	3.22E-9	
	PDS 32	45	2.71E-9	
	PDS 53	2	1.28E-10	
STC 17			2.00E-9	0.016%
	PDS 32	45	9.02E-10	
	PDS 20	38	7.68E-10	
	PDS 56	5	1.07E-10	
	PDS 53	1	1.54E-11	
STC 18			7.97E-10	0.0063%
	PDS 20	34	2.73E-10	
	PDS 32	32	2.54E-10	
	PDS 53	1	1.18E-11	
	PDS 56	1	7.99E-12	
STC 19			3.46E-9	0.027%
	PDS 40	83	2.89E-9	
	PDS 53	6	2.05E-10	
	PDS 56	5	1.56E-10	
STC 20			9.48E-11	0.00074%
	PDS 53	55	5.18E-11	
	PDS 56	41	3.92E-11	
	PDS 40	3	3.27E-12	
STC 21			1.76E-8	0.14%
	PDS 40	93	1.64E-8	
STC 22			1.79E-9	0.014%
	PDS 3	55	9.91E-10	
	PDS 56	30	5.35E-10	
	PDS 36	1	1.40E-11	

Source Term Category	PDS No.	Percent of STC Frequency	STC Frequency	Percent of Total Frequency
STC 23			2.45E-6	19.23%
	PDS 56	96	2.34E-6	
	PDS 36	3	7.02E-8	
	PDS 3	1	1.55E-8	
STC 24			3.39E-6	26.65%
	PDS 53	91	3.07E-6	
	PDS 1	5	1.74E-7	
	PDS 32	2	7.44E-8	
	PDS 20	1	3.82E-8	
	PDS 8	1	3.45E-8	

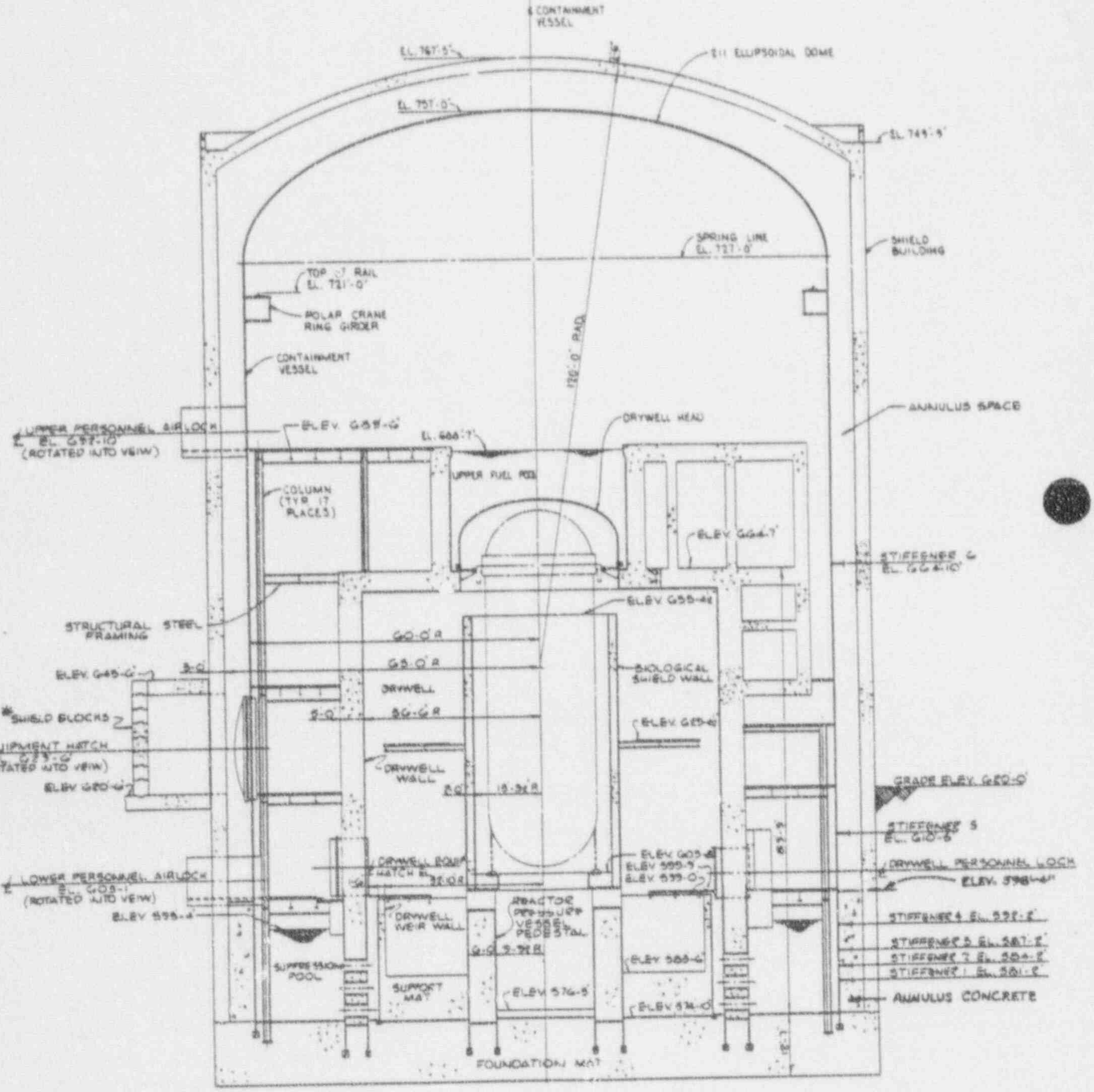
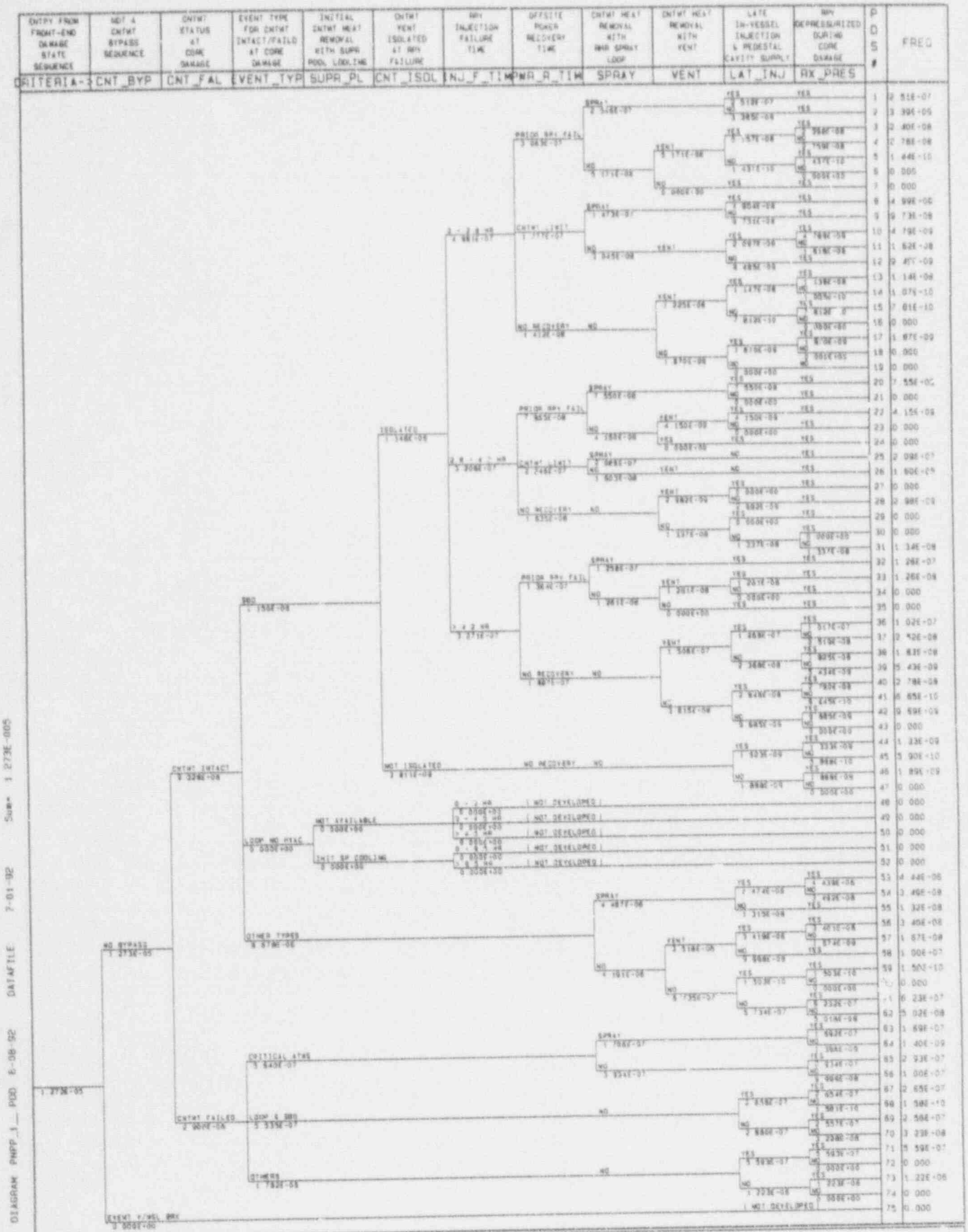
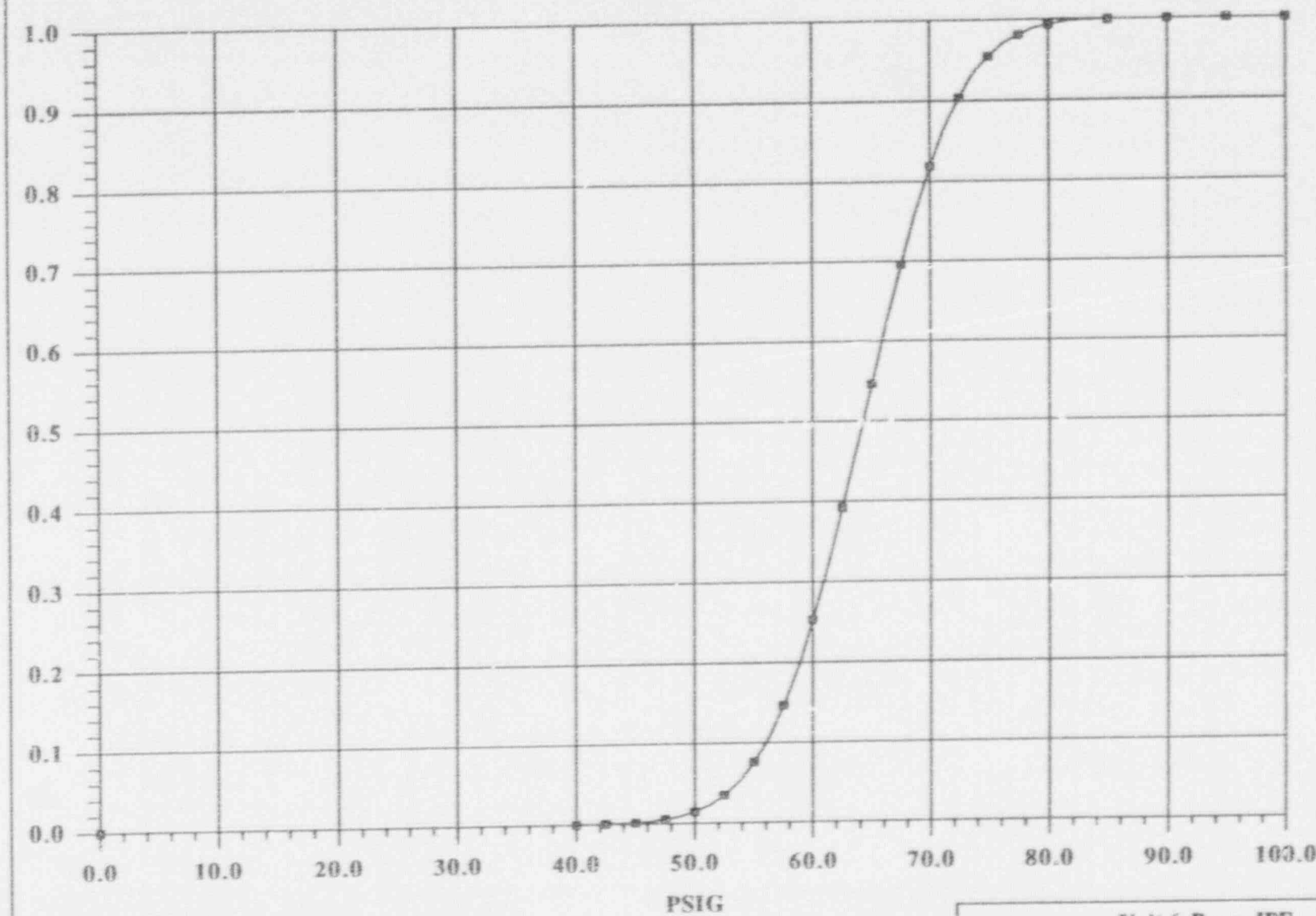


Figure 4.1.1-1 PNPP Mark III Containment



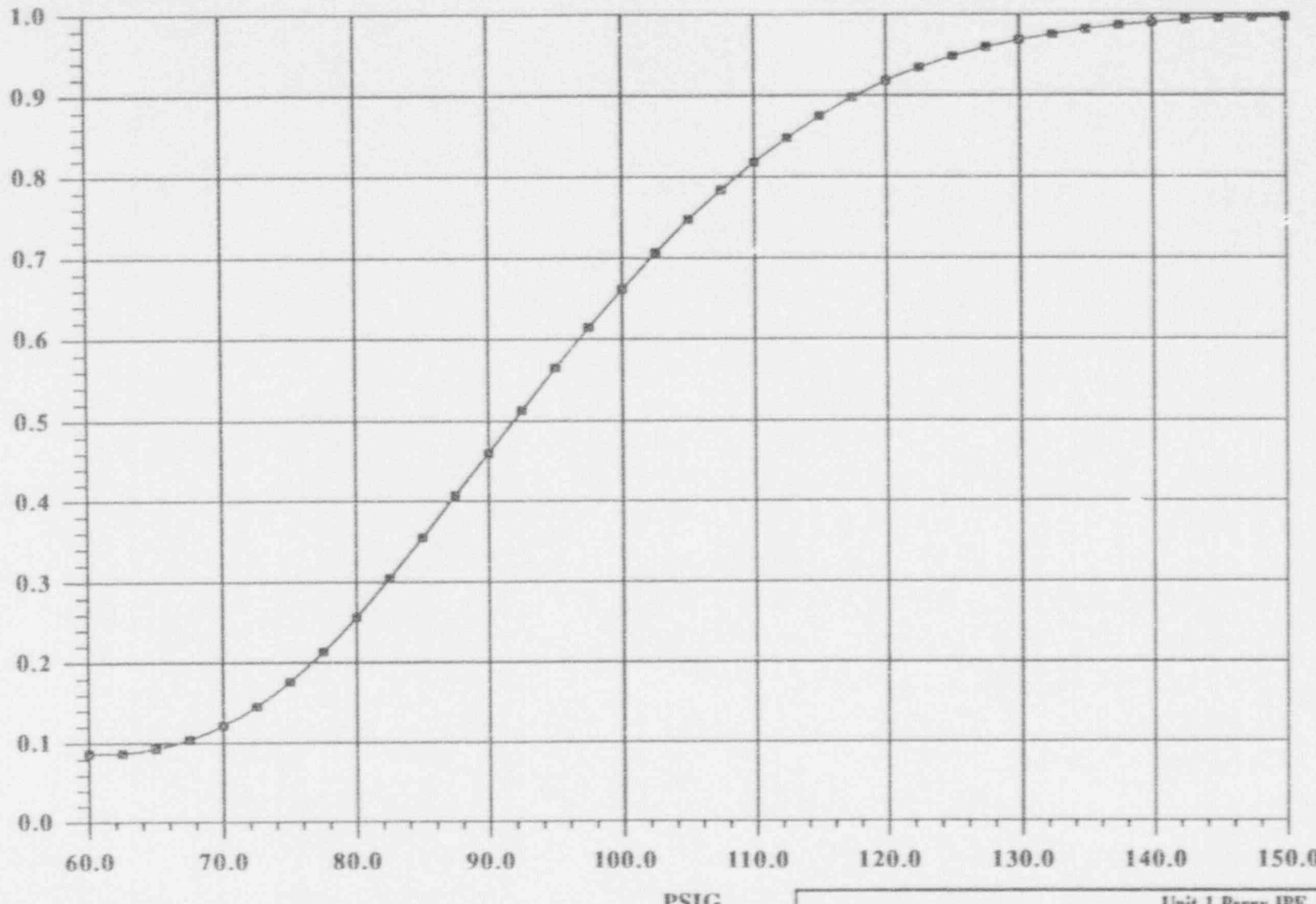
PERRY NUCLEAR PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 4.3.2-1
PLANT DAMAGE STATE GROUPING LOGIC
REVISION 0



Unit 1 Perry IPE

Cntmt Overpressure Fragility

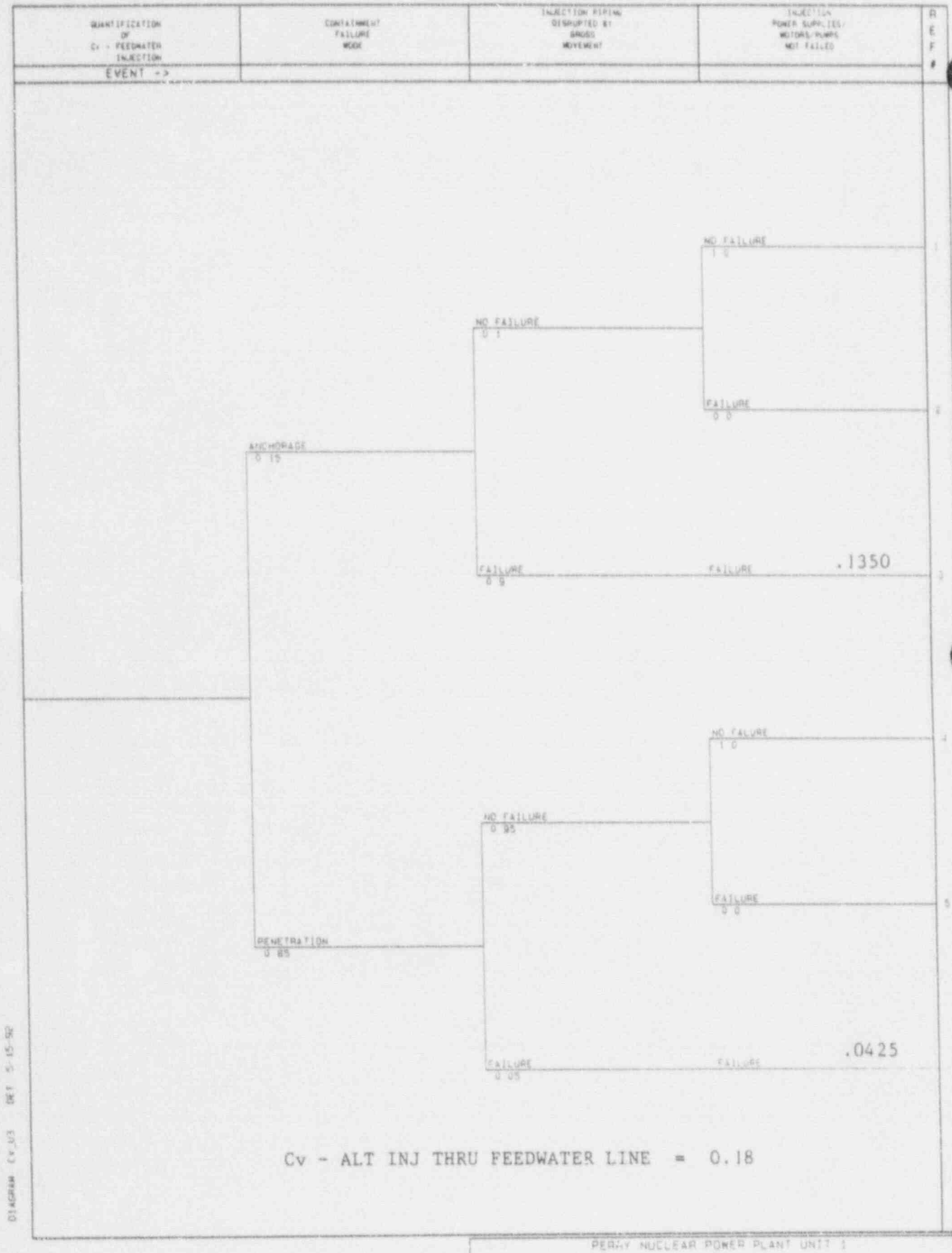
Figure 4.4.3-1



Unit 1 Perry IPE

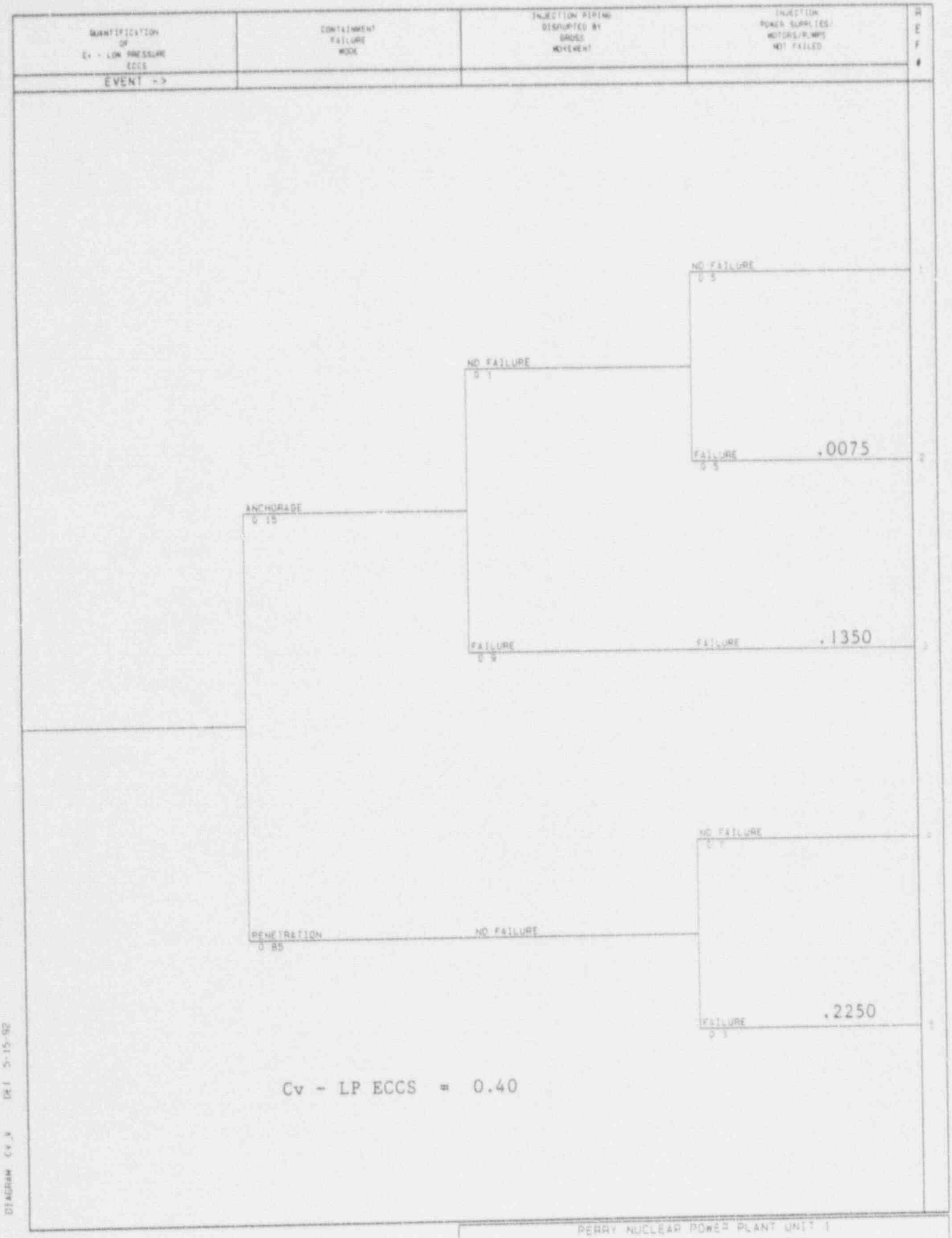
Fast Overpressure Conditional Prob of Anchorage Failure

Figure 4.4.3-2



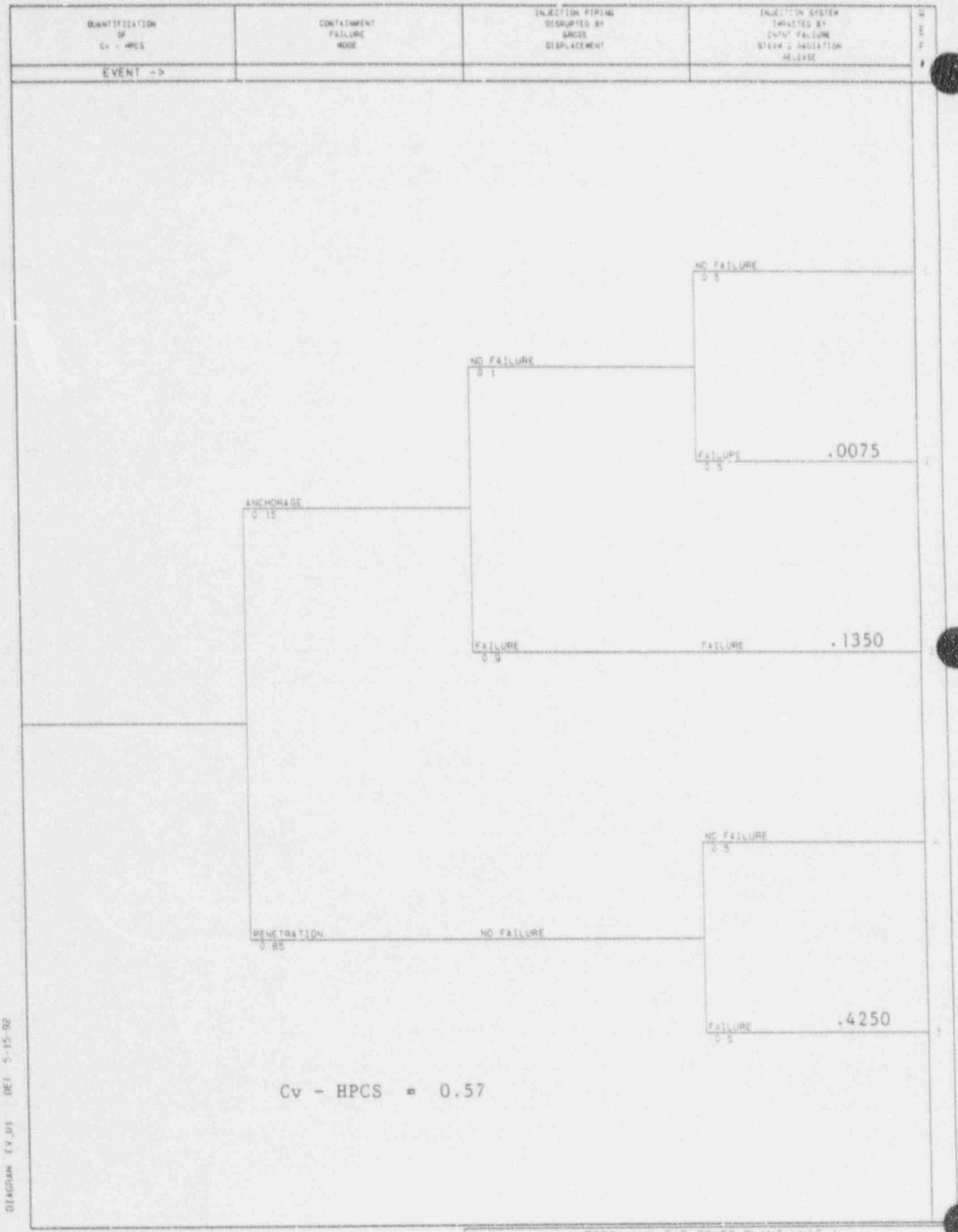
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 4.4.4-3
QUANTIFICATION Cv FOR INJECTION THRU FEEDWTR INJECTION LINE
REVISION 0



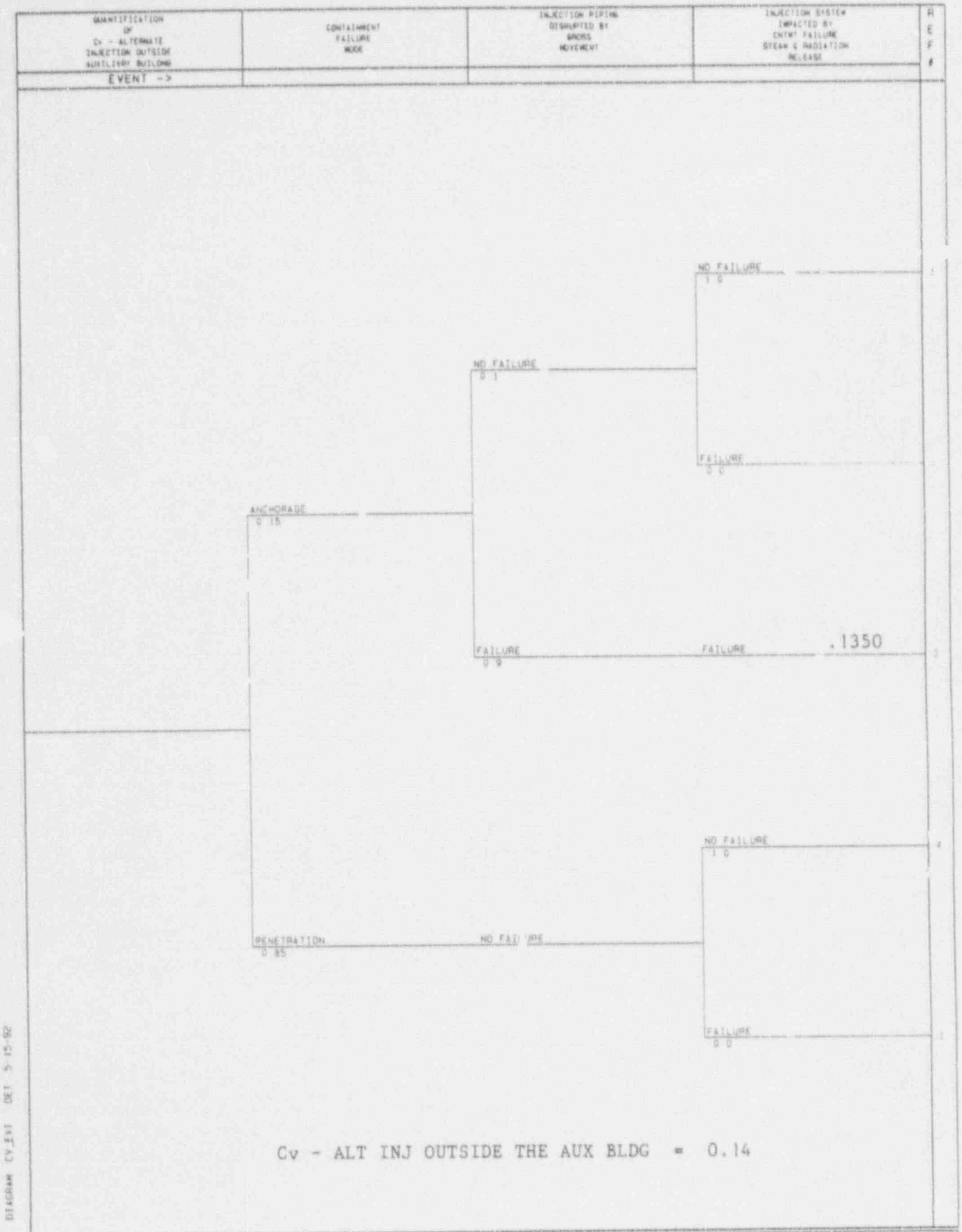
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 4.4.4-2
QUANTIFICATION OF Cv FOR LOW PRESSURE ECCS (LPCS & A-C)
REVISION 0

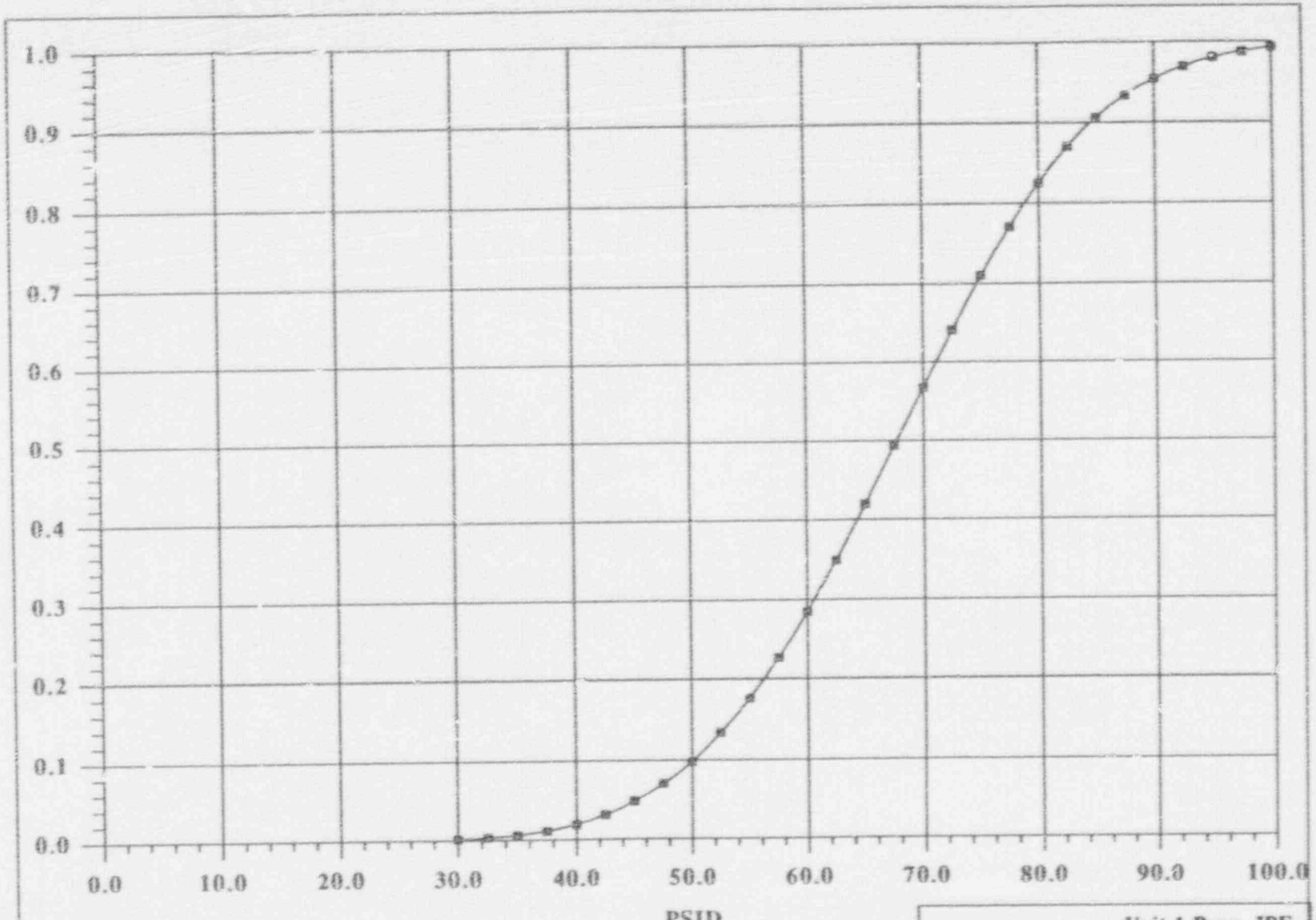


PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

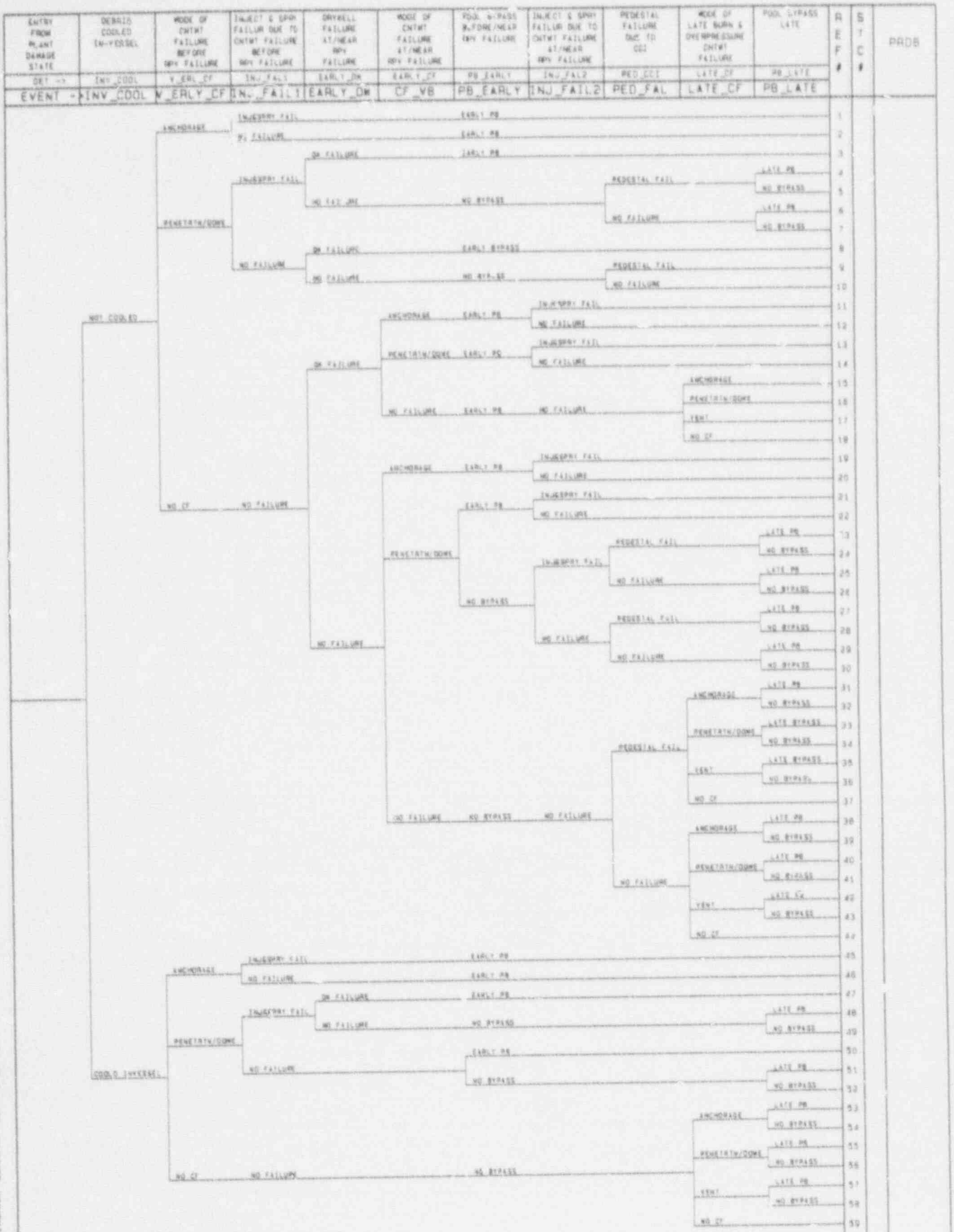
FIGURE 4.4.4-1
QUANTIFICATION OF Cv FOR HPCS
REVISION 0



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 4.4.4-4
QUANTIFICATION Cv - ALT INJECTION OUTSIDE THE AUX BUILDING
REVISION 0

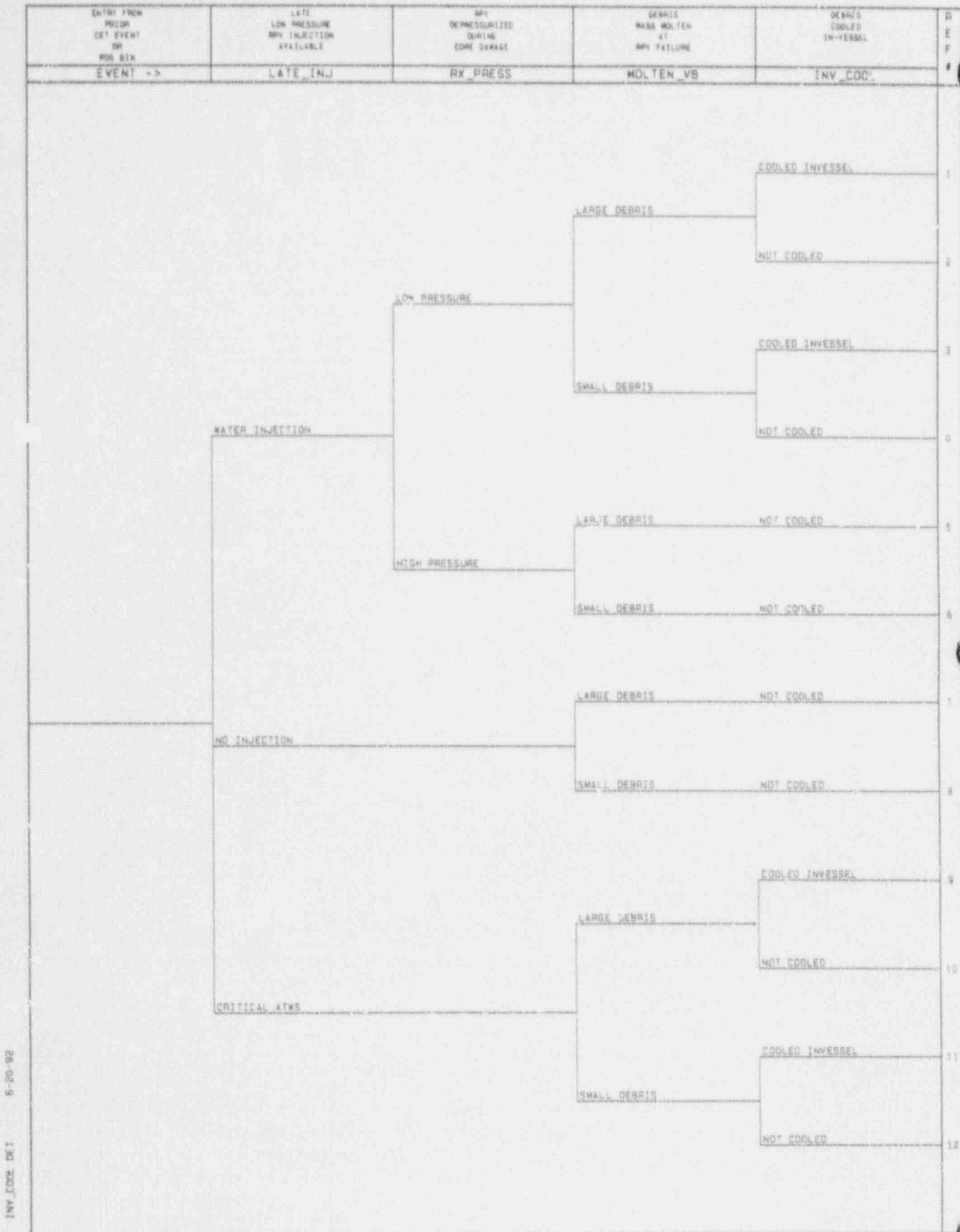


Unit 1 Perry IPE
External Pressure Drywell Fragility
Figure 4.4.5-1

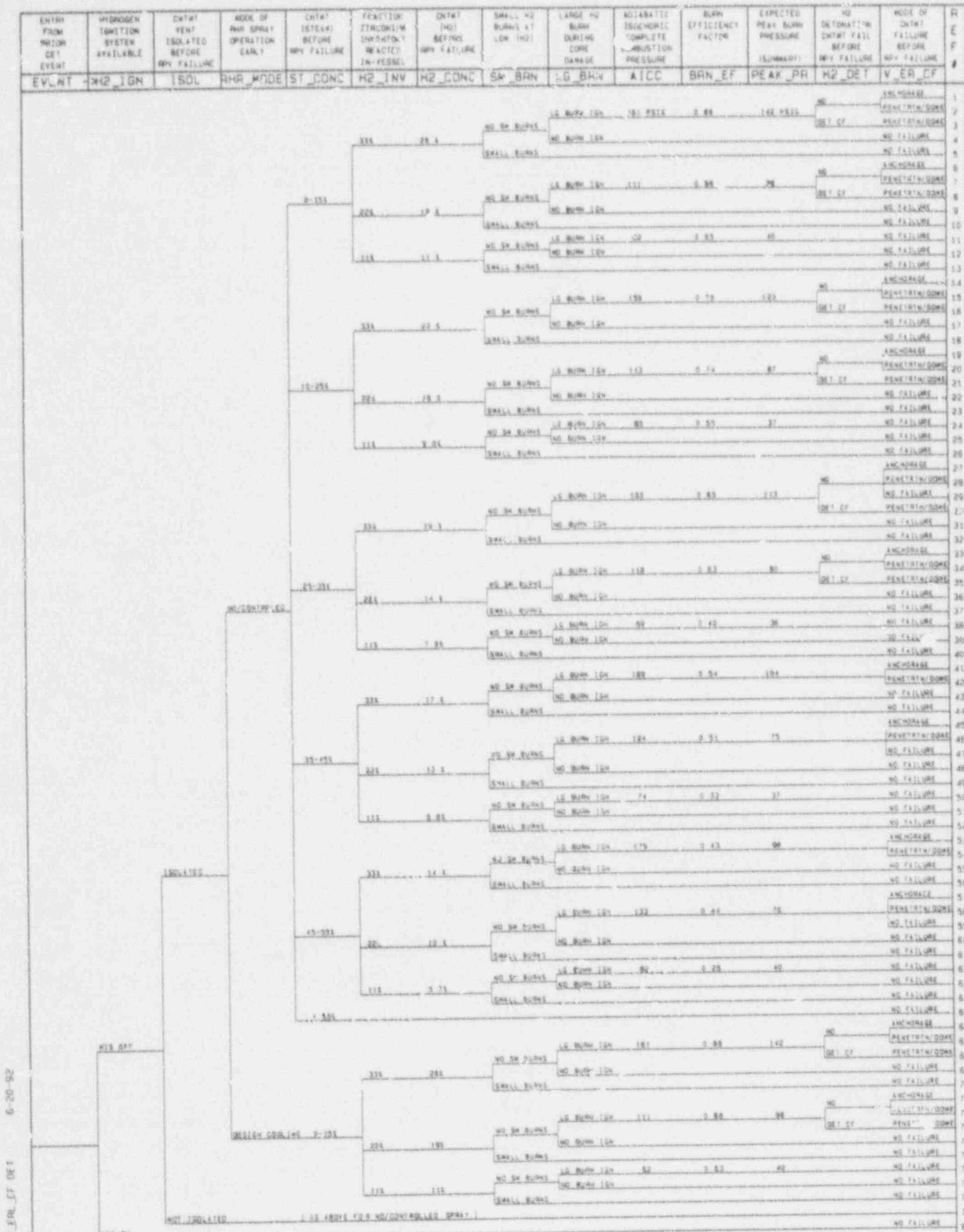


PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 4.5.2.1-1
SUMMARY CONTAINMENT EVENT TREE
REVISION 0

DIAGRAM SUMMARY CET 6-20-92

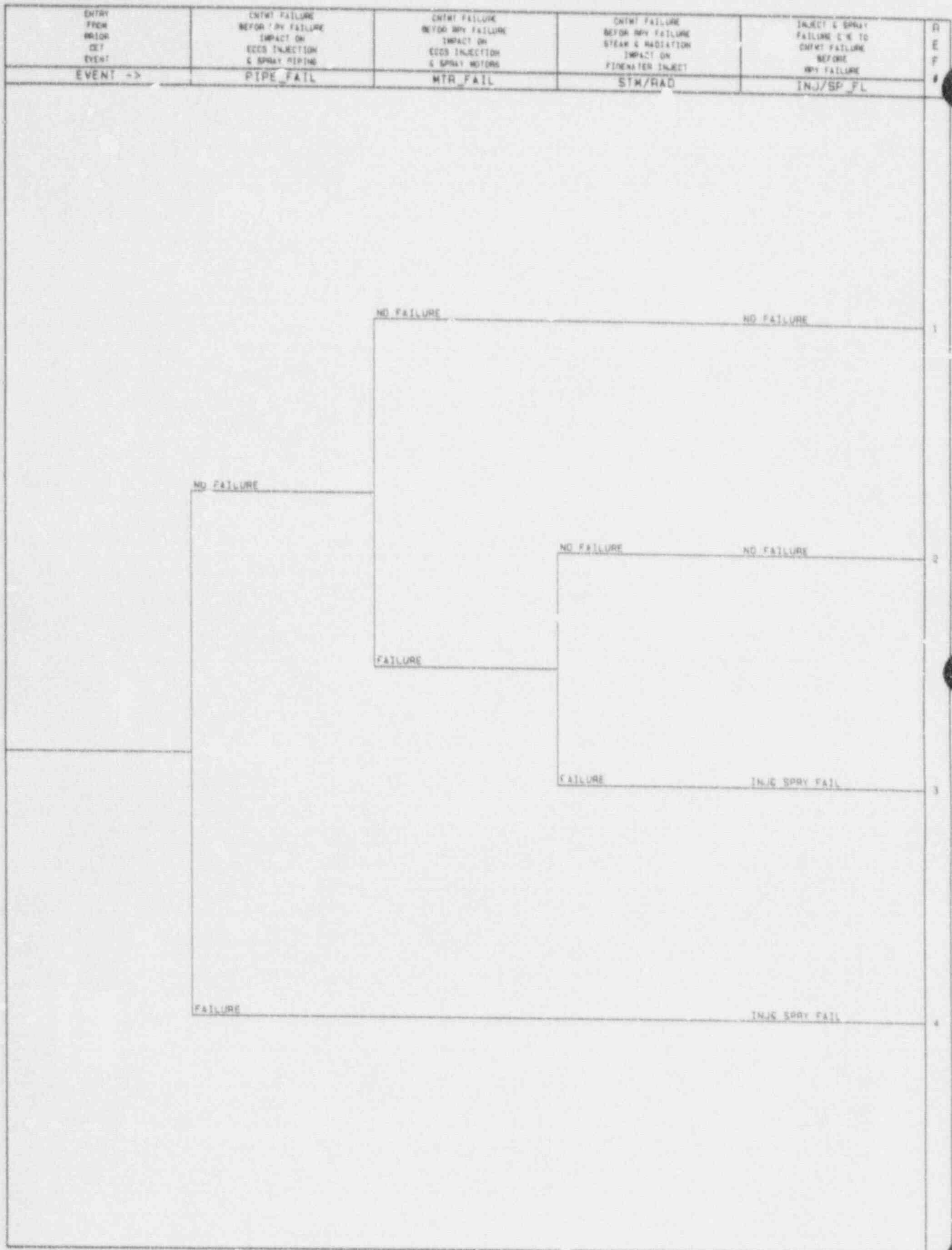


PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 4.5.2.2.1-1
DEBRIS COOLED IN-VESSEL DET
REVISION 0



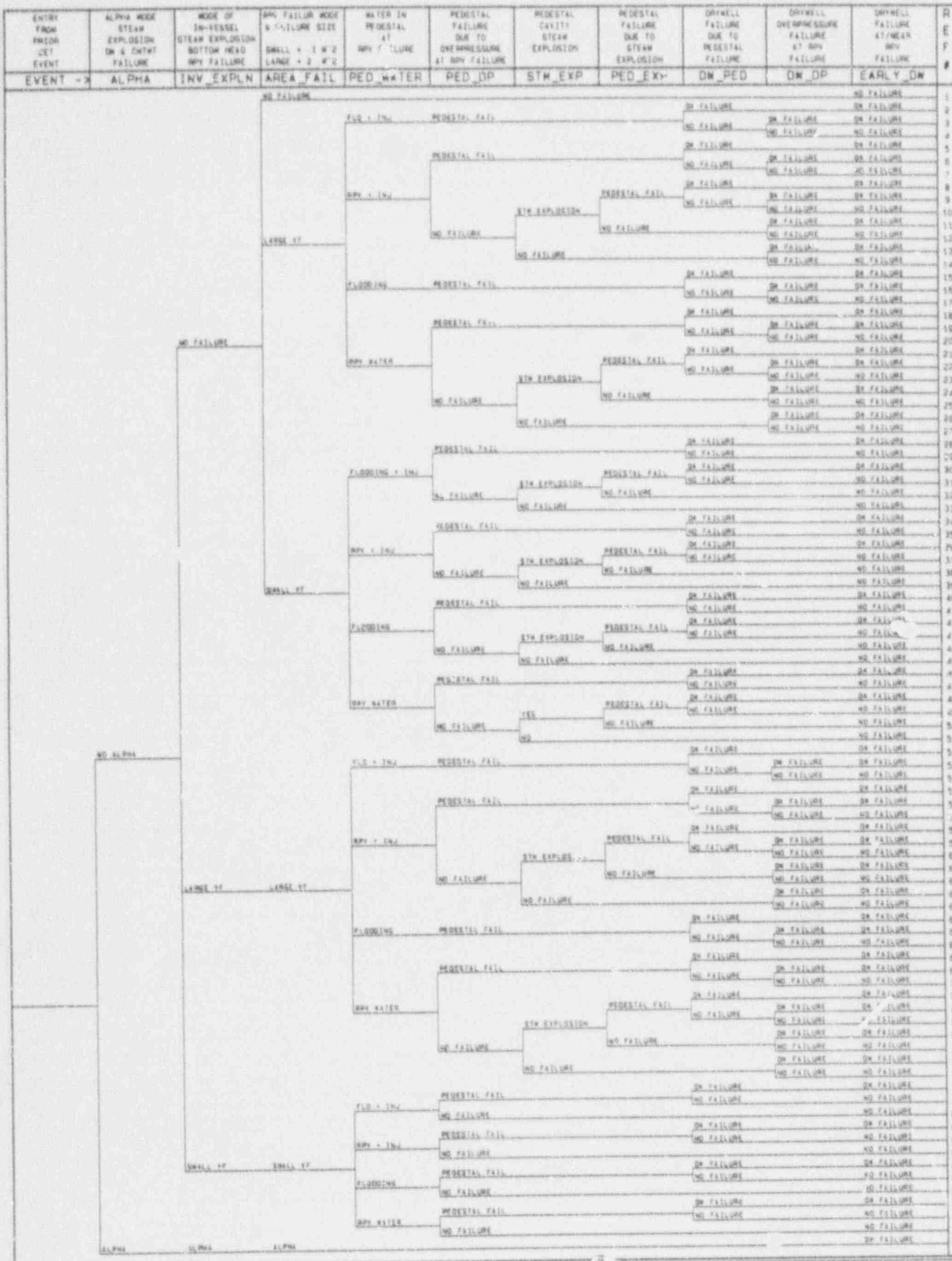
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 4.5.2.2.2-1
MODE OF CONTAINMENT FAILURE BEFORE RPV FAILURE DET
REVISION 0



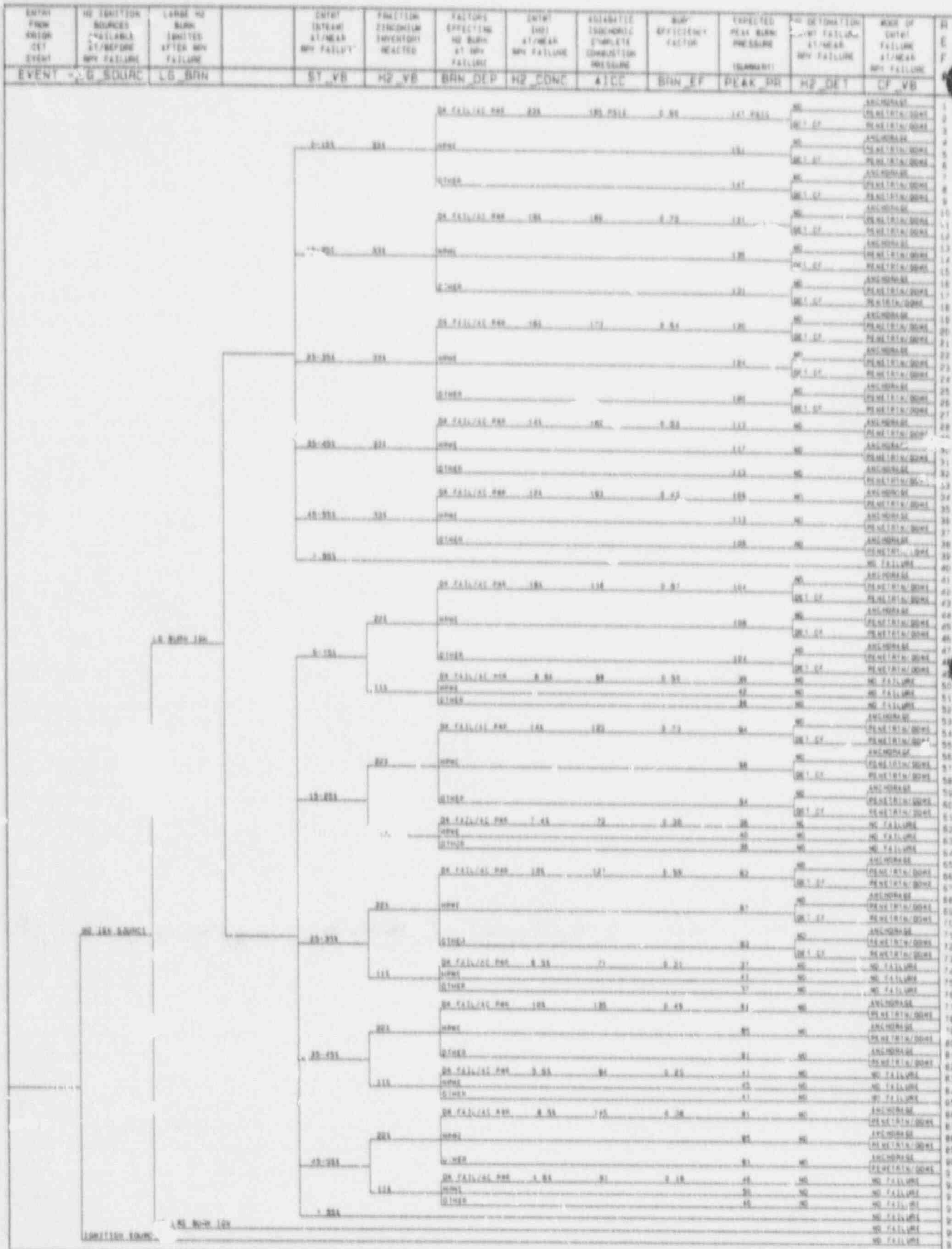
INJ_FAIL DET 6-26-92

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 4.5.2.2.3-1
INJECT & SPRAY FAILURE DUE TO CNTMT FAIL BEFOR RPV FAIL DET
REVISION 0



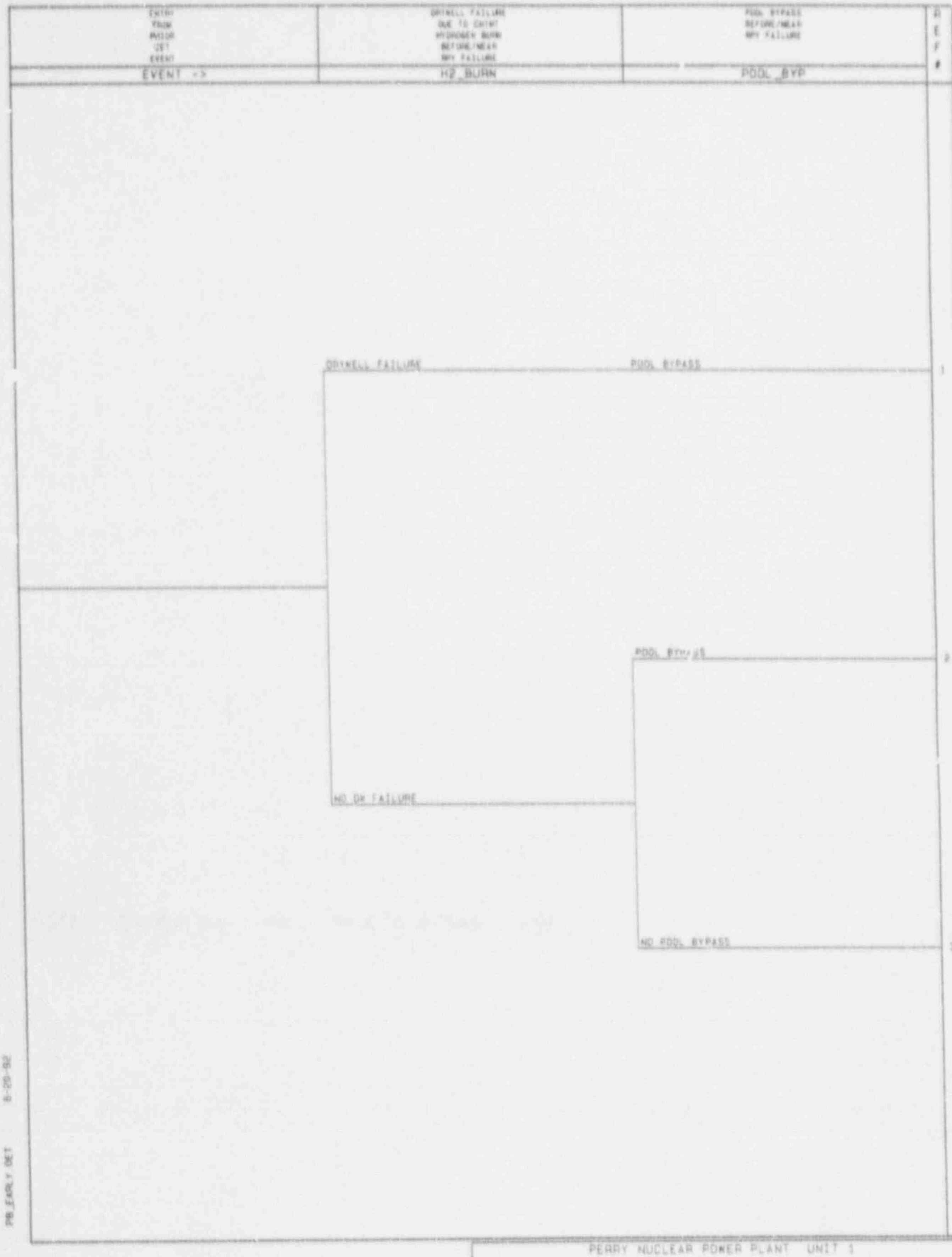
NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 4.5.2.2.4-1
DRYWELL FAILURE AT/NEAR RPV FAILURE DET
REVISION 0

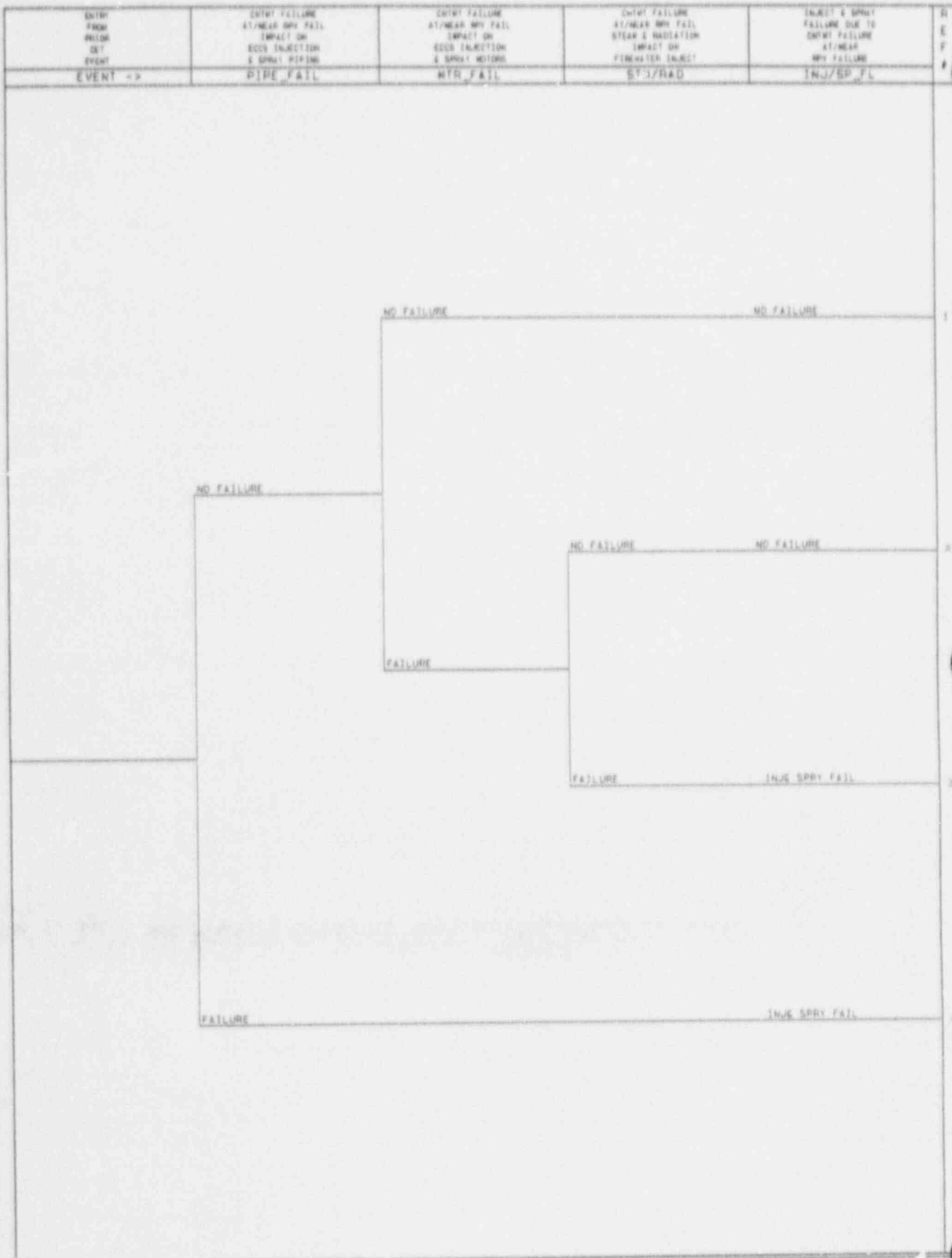


PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

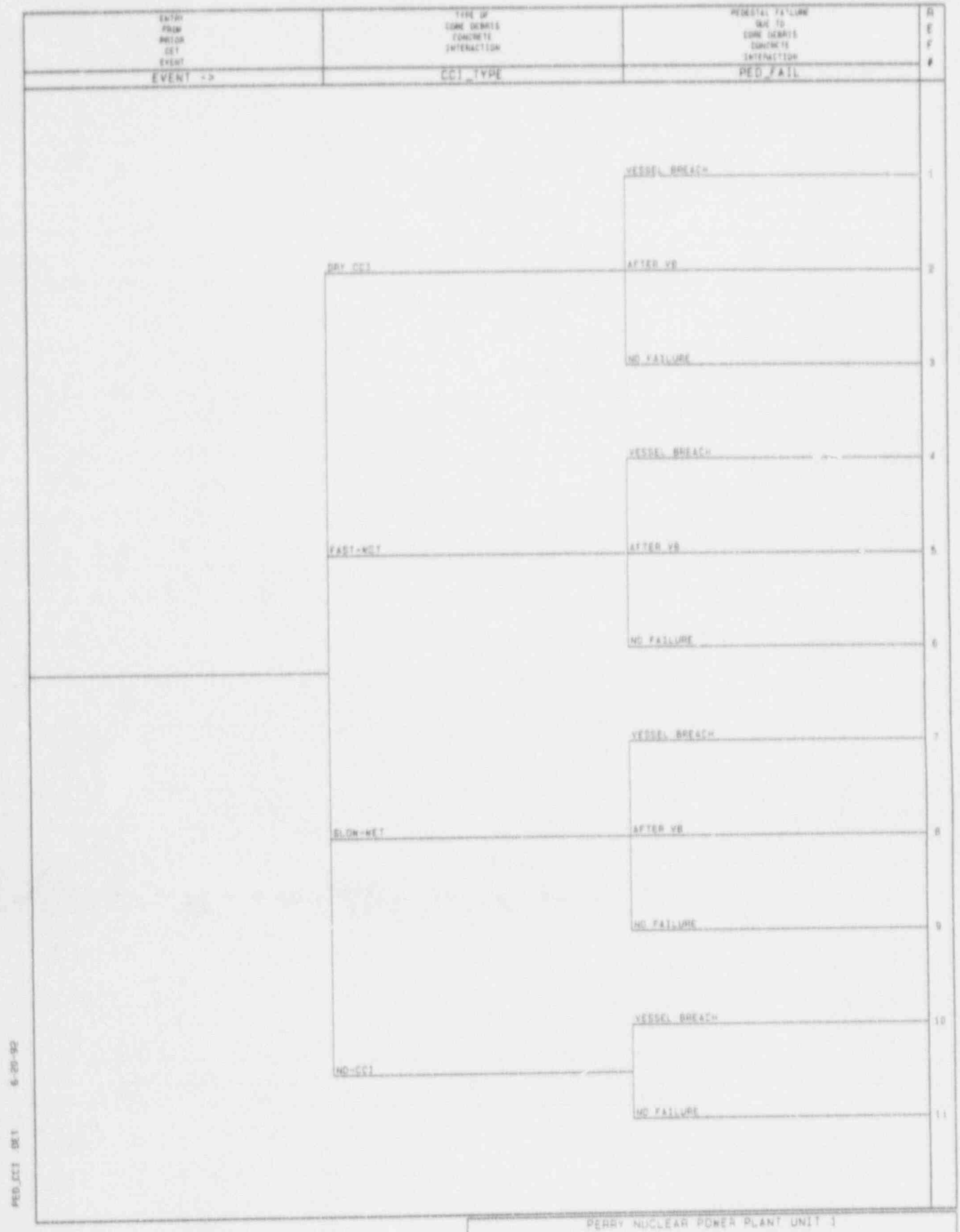
FIGURE 4.5.2.2.5-1
MODE OF CONTAINMENT FAILURE AT/NEAR RPV FAILURE DET
REVISION 0



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 4.5.2.2.6-1
POOL BYPASS BEFORE/NEAR RPV FAILURE DET
REVISION 0

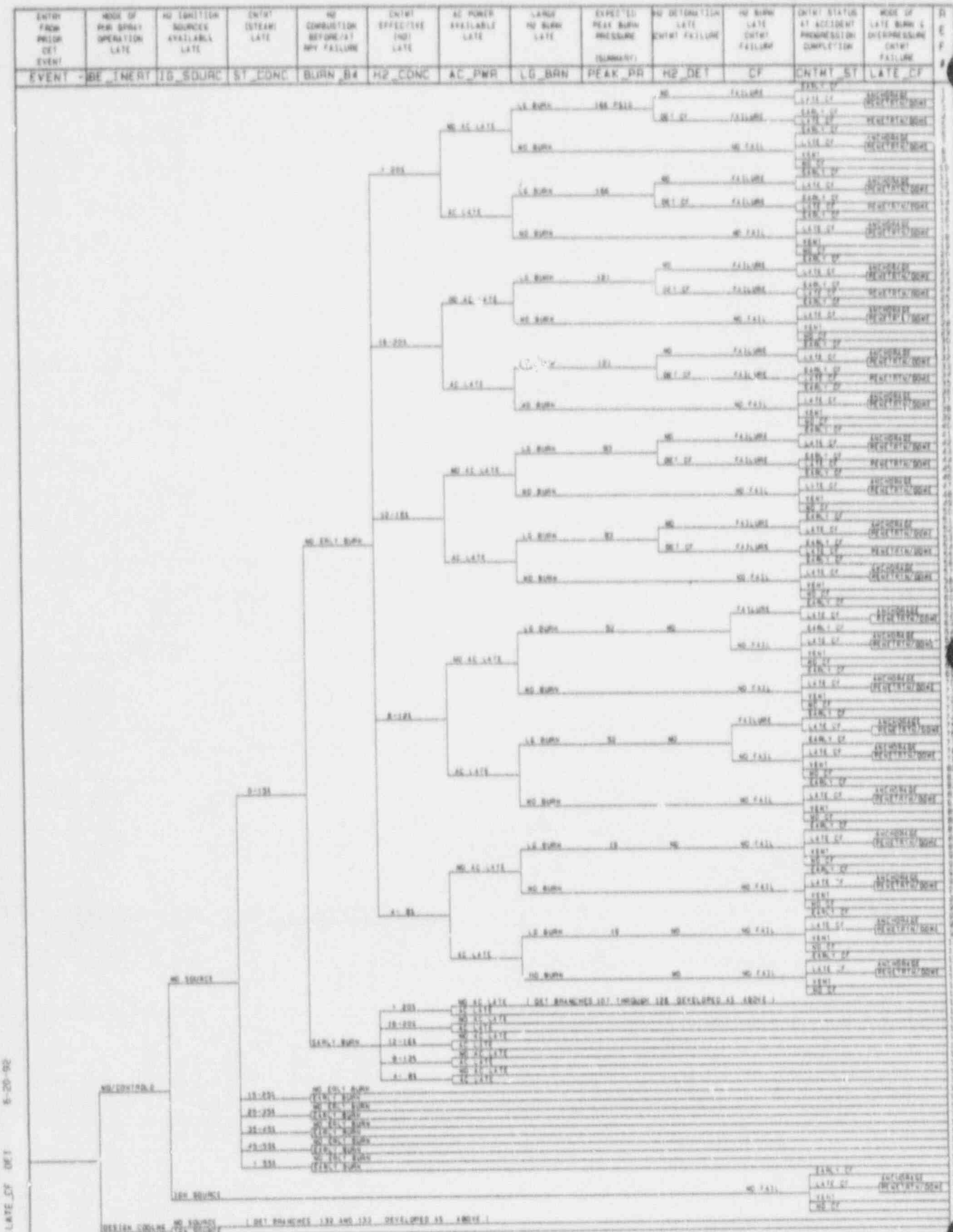


PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 4.5.2.2.7-1
INJECT & SPRY FAILUR DUE TO CMTNT FAIL AT/NEAR RPV FAIL DET
REVISION 0

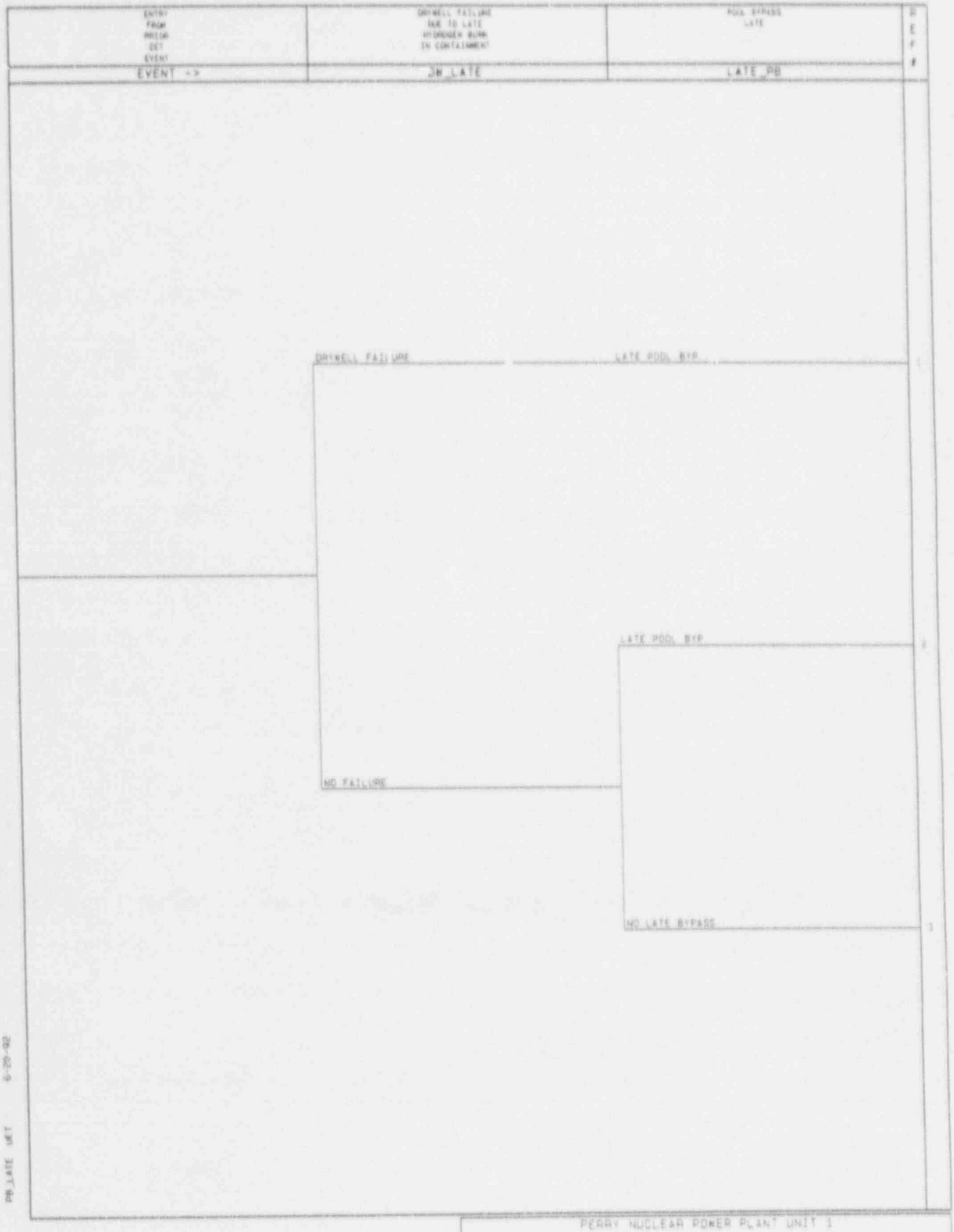


PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 4 5 2 2 8-1
PEDESTAL FAILUR DUE TO CORE DEBRIS CONCRETE INTERACTION DET
REVISION 0

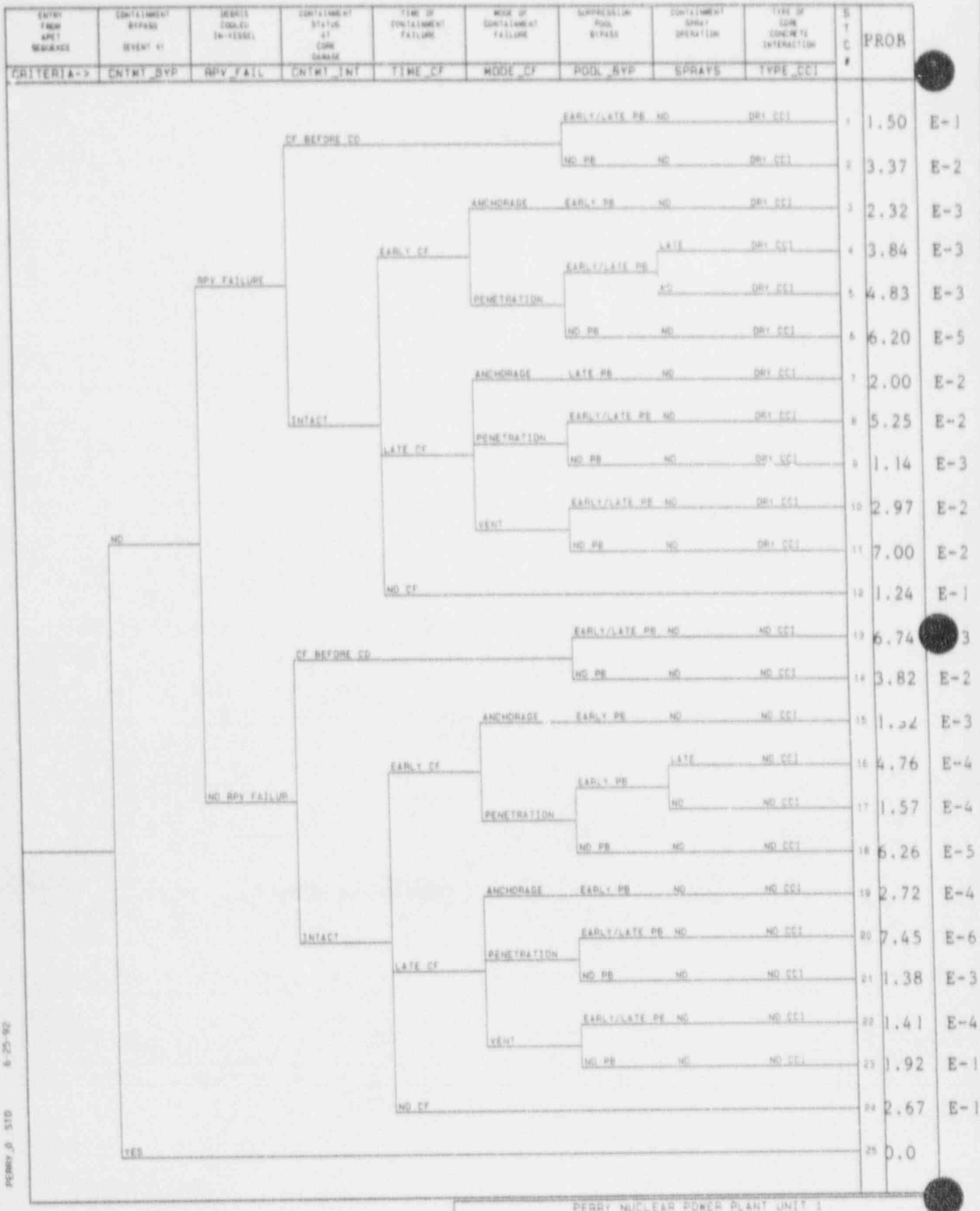


PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 4.5.2.2.9-1
MODE OF LATE BURN & OVERPRESSURE CNTMT FAILURE DET
REVISION 0



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 4.5.2.2.10-1
POOL BYPASS LATE DET
REVISION 0



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 4.7.1-1
SOURCE TERM CATEGORY GROUPING LOGIC
REVISION 0

5.0 UTILITY PARTICIPATION AND INDEPENDENT REVIEW TEAM

5.1 IPE ORGANIZATION

The organizational structure for the IPE is shown in Figure 5.1-1. The team was put together to optimize the CEI resources while meeting the requirements of Generic Letter 88-20. A consultant was retained to provide probabilistic risk assessment technical management and technology transfer, in order to produce the results within the three year time frame. Up to five engineers from the Perry staff were assigned to be team members, three of whom were dedicated full time to the project. However, the consultant, Halliburton NUS Corporation, retained overall responsibilities for the technical aspects of the work. This approach helped to optimize the IPE process by ensuring that personnel with in depth plant knowledge performed the detailed analysis under the guidance of personnel with wide experience of PRA performance and application.

A breakdown of the tasks and the participation of the CEI and Halliburton NUS personnel is shown in Figure 5.1-2. It can be seen from the table that CEI personnel participated in all tasks enabling significant technology transfer in all areas.

5.2 INDEPENDENT REVIEW

As shown in Figure 5.1-1 the independent review and peer review was performed by station and corporate personnel and consultants. The consultants were retained for two reasons. First, among the CEI staff at Perry there was little prior experience with either core damage or accident progression analysis, therefore it was necessary to find experienced personnel outside CEI. Second, in the case of the Level 1 review station engineering and operations staff were able to provide good insights into the details of system modeling and personnel response, but again had little experience in PRA and modeling techniques. The personnel who performed the independent review of the Level 1 (RAPA) and Level 2 (ERIN contracted and EPRI peer review) are listed in Figure 5.1-2.

The review by the corporate staff involved a detailed review of the system models by design engineers and a detailed review of the accident sequence models by operators. Each of the system notebooks were reviewed by responsible design engineers. Before the review effort, design engineers were given training in probabilistic risk assessment methodology. The reviews took place as the system notebooks were completed between July 1990 and January 1991. Comments were documented and addressed.

The initial operations review took place in August 1990. Two licensed operators were assigned full time to the IPE project for a two week period to review the accident sequence models. As the operators worked directly with the IPE staff any comments were immediately addressed and incorporated into the models. Following quantification, the dominant sequences were reviewed by the licensed operators responsible for the Plant Emergency Instructions (PEIs) taken from the Emergency Plant Guidelines (EPGs). The coordinator of this effort was a Shift Supervisor.

The primary reviewer from RAPA has 11 years experience in performing and reviewing probabilistic risk assessments. Two independent reviews by RAPA have taken place at appropriate intervals during the IPE effort. Phase 1 covered initiating events, accident sequence analysis, and system modeling and was completed in March 1991. Phase 2 covered common cause and dependency analysis, data base, human interactions, internal flooding, and sequence quantification and was completed in April 1992.

The Level 2 Independent review by ERIN was directed by the General Manager of BWR Technology, Dr. Edward Burns, who has over 18 years of experience in the field of probabilistic risk assessment, severe accident analysis, and emergency procedure examination. The Level 2 independent review commenced with a onsite meeting in June 1991 to review the initial work completed on the plant model development with the MAAP parameter file and on the containment capacity analysis. A second meeting in March of 1992 reviewed the Level 2 analysis scope from the plant damage state grouping to the development of the Accident Progression Event Tree and on to the source term category release model. During this March meeting, Dr. Edward Fuller, the EPRI Modular Accident Analysis Program (MAAP) Program Director, graciously participated as an peer reviewer. The final ERIN review scope included the MAAP accident progression analysis and the Level 2 IPE submittal. The Level 2 Independent Review was completed in July 1992.

5.3 AREAS OF REVIEW AND MAJOR COMMENTS

5.3.1 Level 1 Review

The general areas of review for each of the reviewing organizations is listed above. Each review was performed on the analysis files available at the time the review took place. The majority of the comments by the design engineers resulted in changes to the descriptions of system design and operation in the system notebooks. Only a few minor modeling changes resulted from this review. Changes to the sequence models from the operations review were discussed and made at the time the review was being performed.

RAPA provided a report for each of the review phases. In addition to individual comments summary comments were provided. Summary comments from Phase 1 are listed below:

1) The assumption not to model electric power into the support systems must be adequately defended. The logical means of doing this is through the use of dependency matrices.

2) More than one special initiator, Loss of Control Complex Chilled Water, will have to be modeled as initiators. Some of the cooling water systems and the air systems should also be modeled as initiators.

3) The level of detail and the breadth of the station blackout event trees is impressive. These are some of the best representations of this complicated event for a BWR that have been seen.

The summary comments from the second phase of the review are listed below.

In general, no major deficiencies were identified in the process used in the Perry PRA. Some of the comments question specific values used in the quantification process. However, it is not felt that any of the comments call into question the overall process or results of the PRA to date. The most significant issues that were identified in the Phase 2 review are:

1) The conclusion that no coupled human errors survived truncation is somewhat surprising. Although the use of BWR EPG based procedures does justify the decoupling of many actions, some coupling would still be expected to occur just because of time coincidence of actions.

2) Some of the specific data values, such as those for RCIC and the diesel generators, are different than those generally quoted in the literature.

3) The overall importance of plant specific maintenance unavailability on core damage frequency was startling. This was particularly true of events where an entire division of ECCS was taken out for maintenance at one time. This may in fact identify a plant specific vulnerability.

The remainder of the comments are minor and when resolved, will provide a good basis for documenting the development process of the PRA. Many of the comments may be resolved by the addition of documentation. Therefore, resolution of the comments will provide documentation of the quantification and assembly process.

5.3.2 Level 2 Review

ERIN provided a report for each of the review phases after the initial site visit which are summarized below. Also included is the EPRI MAAP Program Director's peer review participation in the March 1992 meeting.

ERIN IPE Senior Review - March 1992 Meeting

The following aspects of the PNPP IPE process were effectively presented at the review meeting:

1) A summary presentation

2) Responses to questions

3) A discussion of applications

Based on the lucid presentation, an appreciation of the breadth and depth of the Perry IPE was obtained. The major written report section included in the review was the containment structural analysis. The review of this report, draft sections 4.3 and 4.7 of the back-end analysis report and the presentation by the Perry Level 2 analyst resulted in the following conclusions.

- o It was clear from the review that PNPP has accomplished the objectives of the NRC Severe Accident Policy Statement Generic Letter 88-20 and the IPE Guidance Document NUREG-1335.

- o The basic assumptions and methods are state-of-the-technology.

- o The IPE is very useful and it is expected that the report will reflect the thoroughness and rigor of the analysis upon which it is based.

A list of summary comments was provided which was judged to be minor by the reviewers with respect to the goals of the IPE, but may be useful in finalizing the current version of the IPE and updating the IPE in the future.

It was observed that the MAAP analysis is among the most comprehensive and complete evaluations performed by any BWR utility, and the insights gained from these analyses will be valuable in establishing the accident management actions appropriate for Perry.

EPRI IPE Senior Review - March 1992 Meeting

Dr. Edward Fuller, the EPRI Modular Accident Analysis Program (MAAP) Program Director and Manager of Nuclear Reactor Safety, also attended the March 1992 meeting with ERIN. In addition, five station blackout MAAP runs were reviewed offsite to investigate the effects of hydrogen combustion that may cause containment failure early. Dr. Fuller's meeting report noted the thoroughness of the back-end analysis and the insights identified.

ERIN Review of MAAP Analysis To Support The Perry IPE - May 1992

The independent review of the Perry IPE MAAP analysis was performed by Jeff Gabor of Gabor, Kenton and Associates. The review included the following items.

1. Validation of MAAP 3.0B Revision 7.02

2. Perry Parameter File and other related I/O files

3. Results of sequences without the hydrogen ignition system

4. MAAP prediction for peak pedestal pressure

5. Sample MAAP output

6. MAAP sensitivity analysis scope

The comments from this MAAP review identified opportunities for additional accident progression analysis, but were judged not to materially affect the basis conclusions of the analysis.

ERIN Review of the Perry Level 2 IPE Submittal - June & July 1992

The Perry IPE submittal is a thorough and technically sound analysis. The IPE framework will be very useful to Cleveland Electric in the future as the basis for the "living" PRA. Because the Perry IPE is a living PRA, it is judged appropriate to submit the results when CEI is ready to support the IPE commitments. The comments provided previously and enclosed can be resolved in the future, and are judged not to materially affect the basic conclusions of the analysis. Several suggestions and comments are included to enhance the presentation to the NRC.

5.4 RESOLUTION OF COMMENTS

The comments discussed above have been resolved. The process of resolving the comments consisted of evaluation by the IPE team, communication with the reviewing organization to clarify any issues and subsequent decision by the IPE team on comment resolution. The resolution was documented on the comment forms.

5.4.1 Resolution of Comments on Level 1

For the summary comments noted above the resolutions are noted below:

Phase 1

1) Electric power to each of the components modeled in the support systems was traced back through the 120 V and 480 V circuits to the 4,160 V buses. Dependency matrices were developed as the tracing was performed. The switchgear and MCCs for these are not crosstied but divisionally separated. They are also housed in the same room by division. Therefore, any common event affecting one bus would affect them all.

2) Special initiators were also developed for the Loss of Instrument Air and Loss of Service Water events. The loss of other cooling water systems are bounded by these initiators or by the other initiating events such as loss of feedwater.

3) No response necessary.

Phase 2

1. At the time of the review the human reliability analysis was undergoing internal review. This resulted in the identification of major coupling of HIs in some of the ATWS sequences and in the initiation of some systems. This has resulted in modification to the human reliability modeling and quantification.

2. The values used for the failure to start and run for RCIC and HPCS are those used in the Grand Gulf study done on behalf of the NRC. Plant specific data for these failure modes was not available at the time the IPE started due to insufficient operating history. This is in line with the declared intention in the IPE of using data from the

Grand Gulf study.

3. Due to the importance of this contributor, plant-specific data for the most recent operating cycle has been used for the HPCS and RCIC systems in the final quantification of the core damage frequency. Now that Perry has achieved sufficient operating experience it is expected that more plant-specific data will be used in future modifications to the plant model. Note that maintenance did not appear as a vulnerability although it is important.

5.4.2 Resolution of Comments on Level 2

Since the Level 2 comments did not materially affect the basic conclusions of the back-end analysis, no resolution response was required.

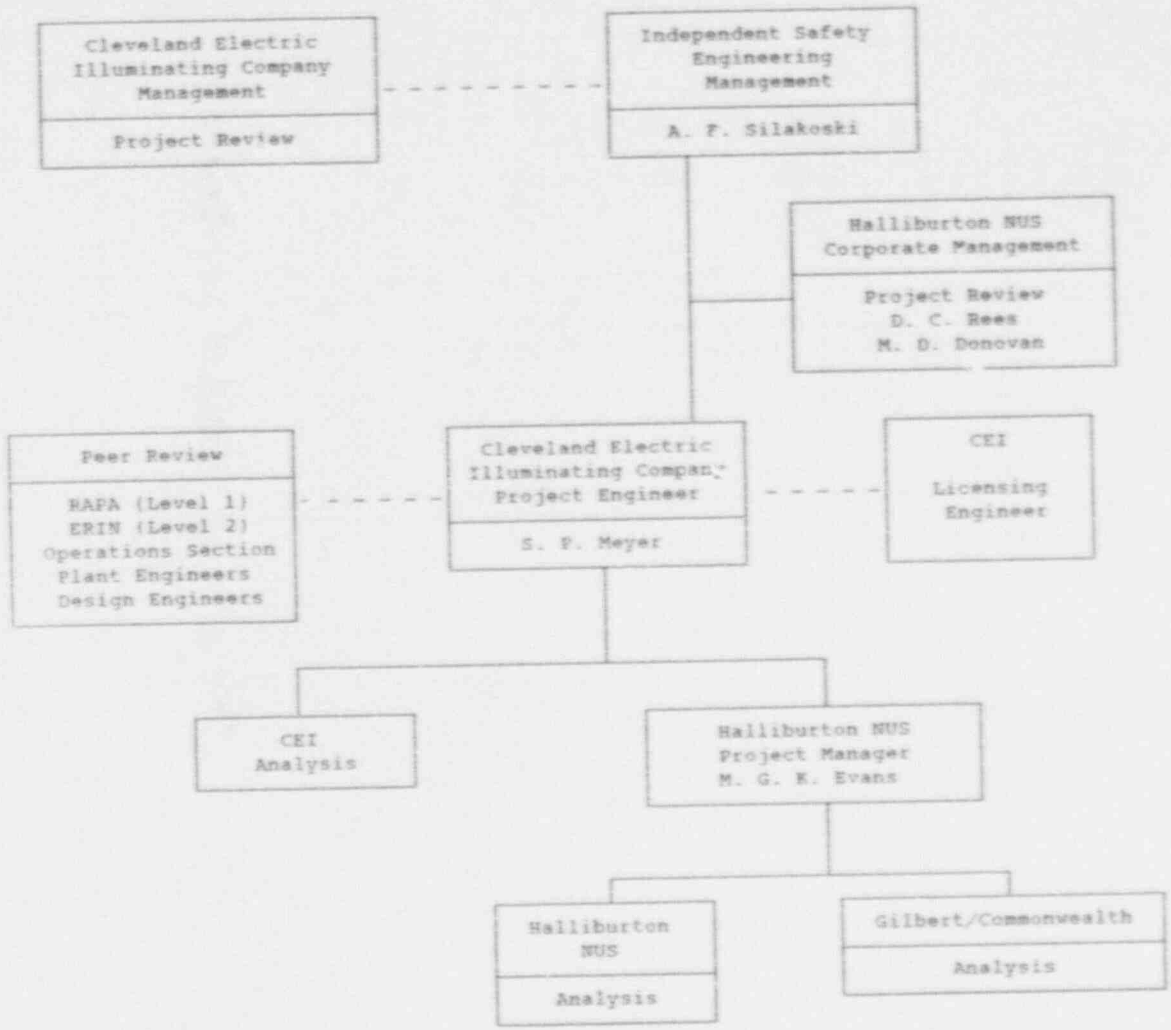


Figure 5.1-1 Project Organization Chart

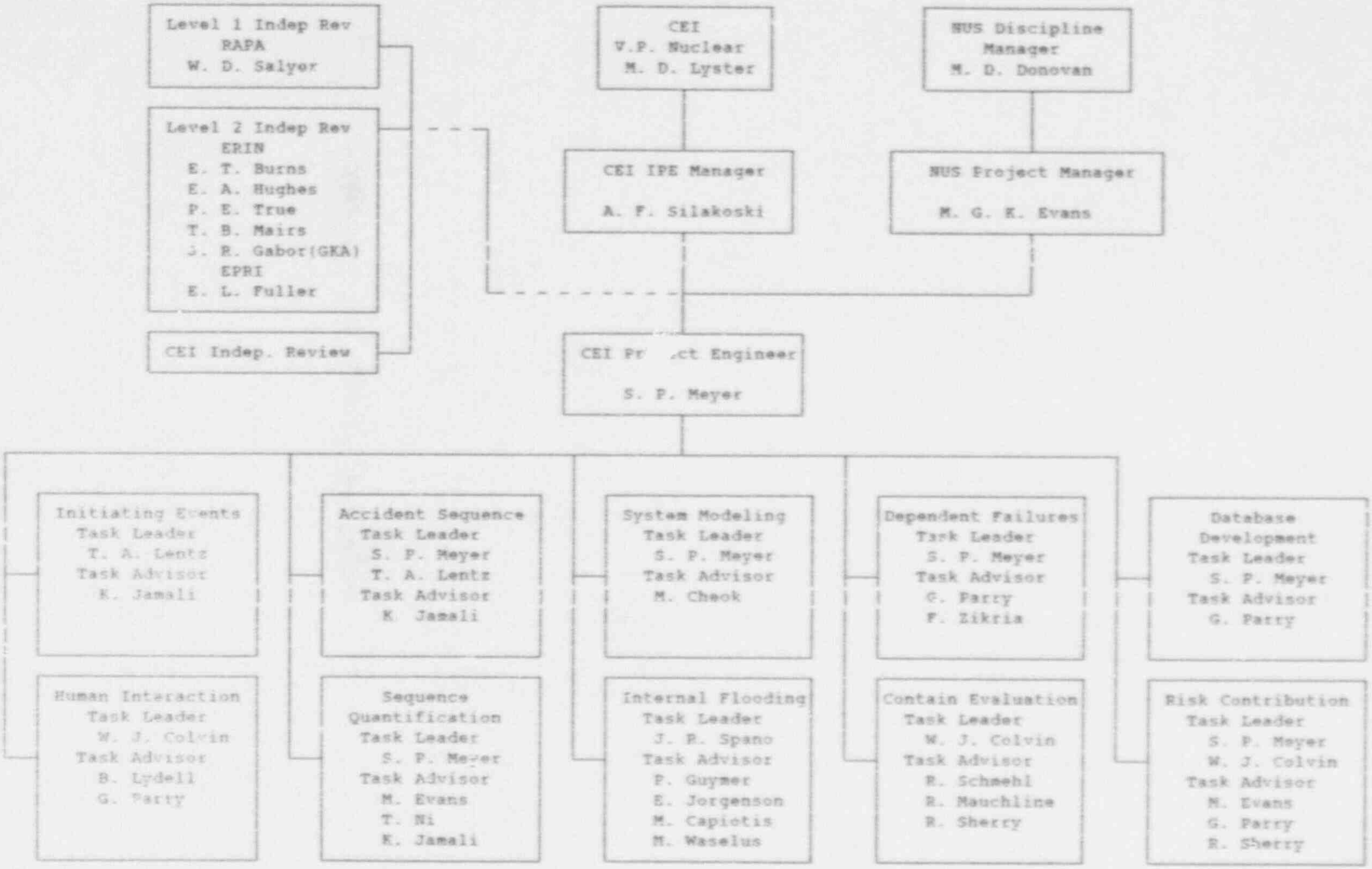


Figure 5.1-2 Project Organization

6.0 POTENTIAL PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

The analysis of the Perry plant is the fourth Probabilistic Risk Assessment of a BWR6 and therefore it is possible to compare features of the various designs and the way in which such features either enhance the safety of the plant or represent a vulnerability in terms of other core damage or fission product release. The presentation of the results in section 3.4 identifies the individual contributions to core damage, and in section 4.5.3 identifies the contributions to containment failure.

6.1 UNIQUE SAFETY FEATURES

The safety features associated with the current design and operation of Perry which contribute to the base case core damage frequency or prevention of containment failure or bypass are discussed in this section.

6.1.1 Core Damage Safety Features

There are a number of unique features at Perry which help to minimize the frequency of core damage from certain initiating events.

There is a motor driven feed pump which is normally in standby and will start on receipt of an automatic signal at level 2, following failure of the turbine driven pumps. Thus there are effectively three high pressure injection systems available to respond to loss of power conversion system or main steam isolation events. This has the effect of reducing the core damage frequency from such events.

The original design for the Perry site was for two units. Although only one unit has been completed, many of the support systems for the second unit were completed. For example, there are two sets of batteries for the safety-related DC buses. In the event of a loss of offsite power or station blackout the operators are instructed, by procedures, to use both sets of batteries to support Unit 1 operation thus considerably extending the availability of dc in this situation. The impact of this is to reduce the contribution from station blackout in relation to loss of offsite power. If it was not possible to use the second battery, the core damage from station blackout would be dependent on earlier recovery of offsite power which would raise the core damage frequency.

The diesel generator which supplies the HPCS system is not the same size or design as the two diesels supplying the emergency bus bars so common cause failure of all three diesel generators is less likely to occur. Similarly the differences in design between the emergency service water train for HPCS and the other emergency service water trains will lessen the probability of common cause failure of all three trains.

The facility exists to cross connect the supply from the HPCS diesel generator to the Division 2 emergency bus, which enables the containment vent valves (and potentially hydrogen ignitors) to be powered in the event of loss of offsite power and failure of both Division 1 and 2 emergency diesel generators. This reduces the frequency of core damage sequences as the result of station blackout.

6.1.2 Unique and/or Important Containment Features

The most significant feature in the design of the containment are the expected failure modes following a slow overpressurization following loss of containment heat removal (where RHR suppression pool cooling or containment spray, and containment venting have failed). The evaluation of containment overpressure in section 4.4.3 determined the gradual overpressure conditional probability of anchorage failure given containment failure is 0.15. Anchorage failure will result in loss of the suppression pool and most likely will fail all other injection sources. Thus anchorage containment failure has the effect of resulting in core damage as well as suppression pool bypass. Additionally, any failure of the containment is likely to pressurize the silicone foam seal on the adjacent auxiliary building and cause failure of the ECCS pumps.

6.2 PLANT DESIGN CONSIDERATIONS

The sensitivities and vulnerabilities analysis discussed in section 3.4.2 showed that according to the definition in the NUMARC document there are no vulnerabilities associated with core damage. However a number of items were identified which would lead to a reduction in the core damage frequency.

Similarly the containment analysis discussed in section 4.5.3 identified the dominant contributors to containment failure. Some potential design considerations which would reduce the core damage frequency and containment failure frequency are discussed in the following sections. It must be stressed that the potential design considerations are strictly conceptual. The feasibility of implementation (e.g., code compliance, materials availability, regulatory acceptance, cost justification, etc.) has not been evaluated.

6.2.1 Plant Improvements Made Due to IPE Insights

During the course of the IPE a number of areas were identified where changes to procedures or plant design would result in a reduction in core damage frequency. The following changes were implemented:

Loss of Offsite Power Instruction

- Retention of RCIC isolation bypass for high steam tunnel temperature
- Enhanced process for crosstieing Unit 1 and Unit 2 batteries
- Enhanced process for offsite power recovery to HPCS and alternate injection system buses

Flooding Instruction

- Enhanced response instructions for flooding scenarios

System Modifications

- ADS automatic initiation

- "Fast Firewater" tie between Fire Protection and HPCS

- Permanent Division 3 to Division 2 "quick" connect

- Reduction of Out-of-Service Time for certain critical components

6.2.2 Design Considerations for Reduction in Core Damage Frequency

Section 3.4 provided a discussion of the sensitivity analysis performed for the Perry IPE. In addition to the safety features discussed above, an analysis of several other potential design considerations were provided. These changes to the plant are described below. The impact of making each of the changes is summarized in Table 6-1.

Passive Containment Vent Path

One of the sensitivity analyses performed was to assess the impact of containment failure on loss of RPV injection and subsequent core damage. The addition of a passive containment vent path that does not depend on AC power would reduce the core damage frequency from internal and flooding events by 18 percent or from 1.3×10^{-5} to 1.1×10^{-5}.

Automatic ADS Inhibit for ATWS

One of the contributors to core damage frequency for ATWS is manually inhibiting ADS. By installing an automatic inhibit of ADS, those ATWS sequences in which manual inhibit fails would drop out. The overall core damage frequency is reduced by 19 percent from 1.3×10^{-5} to 1.0×10^{-5}. The sequences resulting from this failure result in an uncontrolled flow to the RPV from the low pressure injection systems with subsequent core damage and containment failure. The addition of the auto inhibit would reduce the frequency of this set of sequences.

6.2.3 Design Considerations for Reduction in Containment Failure

The sensitivity of the frequency of containment failure to various assumptions and potential design considerations is discussed in section 4.6.3. The design considerations are discussed below and the impact of performing them is summarized in Table 6-2 and shown in pie charts in Figures 6.2.2-1 through 6.2.2-6.

Passive Containment Vent

This consideration is described in section 6.2.2. As well as reducing the frequencies of containment failure before core damage sequences from 2.9×10^{-6} to 5.7×10^{-7} it will impact the fission product release with pool bypass. The overall impact is to reduce: the RPV failure and early containment failure with pool bypass frequency from 2.0×10^{-6} to 4.5×10^{-7} (-78% change), and the containment structural failure frequency from 4.0×10^{-6} to 9.9×10^{-7} (-75% change).

ATWS Automatic ADS Inhibit & Alternate Shutdown

Automatic ADS Inhibit for ATWS is described in section 6.2.2. The automatic

ADS inhibit feature would directly reduce the core damage frequency. Additionally, providing an enhanced instruction to control RPV Power/Level as a function of containment pressure with a RPV water level control band just above the Minimum Steam Cooling Water Level would reduce reactor power within the venting heat removal capability. This quasi steady-state RPV Power/Level control would provide a reasonable time for reactor shutdown recovery using an alternate boron injection system. This potential change is conservatively assumed to reduce the containment failure frequency from 5.6×10^{-7} to 4.3×10^{-8}. This impacts the potential for containment failure and fission product release with pool bypass. The overall impact is to reduce: the RPV failure and early containment failure with pool bypass frequency from 2.0×10^{-6} to 1.8×10^{-6} (-14% change), and the containment structural failure frequency from 4.0×10^{-6} to 3.5×10^{-6} (-13% change).

Secure Electrical Supply to Hydrogen Ignitors

Supplement No. 3 of Generic Letter 88-20 identified that Mark III containments are expected to evaluate the vulnerability to interruption of power to the hydrogen ignitors. The modification of the electrical supply to the hydrogen ignitors to ensure availability during station blackout would remove the possibility of high containment loads from hydrogen deflagrations and detonations. The overall impact is to slightly reduce: the RPV failure and early containment failure with pool bypass frequency from 2.0 x 10-6 to 2.0 x 10-6 (-2% change), and the containment structural failure frequency from 4.0×10^{-6} to 3.8×10^{-6} (-6.7% change).

Combined Passive Vent & ATWS Automatic ADS Inhibit & Alternate Shutdown

The combination of the two considerations discussed above would reduce significant contributors to core damage as well as containment failure. The overall impact is to reduce: the core damage frequency from 1.3×10^{-5} to 7.7×10^{-6} (-41% change), the RPV failure and early containment failure with pool bypass frequency from 2.0×10^{-6} to 1.6×10^{-7} (-92% change), and the containment structural failure frequency from 4.0×10^{-6} to 4.6×10^{-7} (-89% change).

Combined Passive Vent, ATWS Modifications and Hydrogen Ignition Power

The combination of these three changes together, further examines the impact of backup power to the hydrogen ignitors in with the passive vent and the ATWS modifications discussed above. The overall impact when compared to the base case is to reduce: the RPV failure and early containment failure with pool bypass frequency from $2.0 \text{ X } 10^{-6}$ to 1.1×10^{-7} (-94% change), and the containment structural failure frequency from 4.0×10^{-6} to 1.8×10^{-7} (-62% change).

6.3 PROGRAM FOR INCLUSION OF PLANT DESIGN CHANGES

As stated earlier the core damage frequency from all initiators is 1.3×10^{-5} and there is no class of sequences contributing above 1.0×10^{-5} which indicates there are no vulnerabilities at the Perry Nuclear Power Plant. The study has been performed using industry generic data for the majority of data (the major exceptions being the maintenance unavailabilities and a detailed analysis of the human interaction). The indication from the last two operating cycles is that the plant-specific initiating event frequencies are lower than those used in the study. As it is the intention to use the results of this study as a living PRA it is expected the study will be updated using plant-specific data, when available, starting in 1994, after development of the procedures for collection of data and the management of the living PRA have been developed. The use of the plant-specific data will enable a reassessment to be made of the design changes identified in the current base case study.

Table 6-1

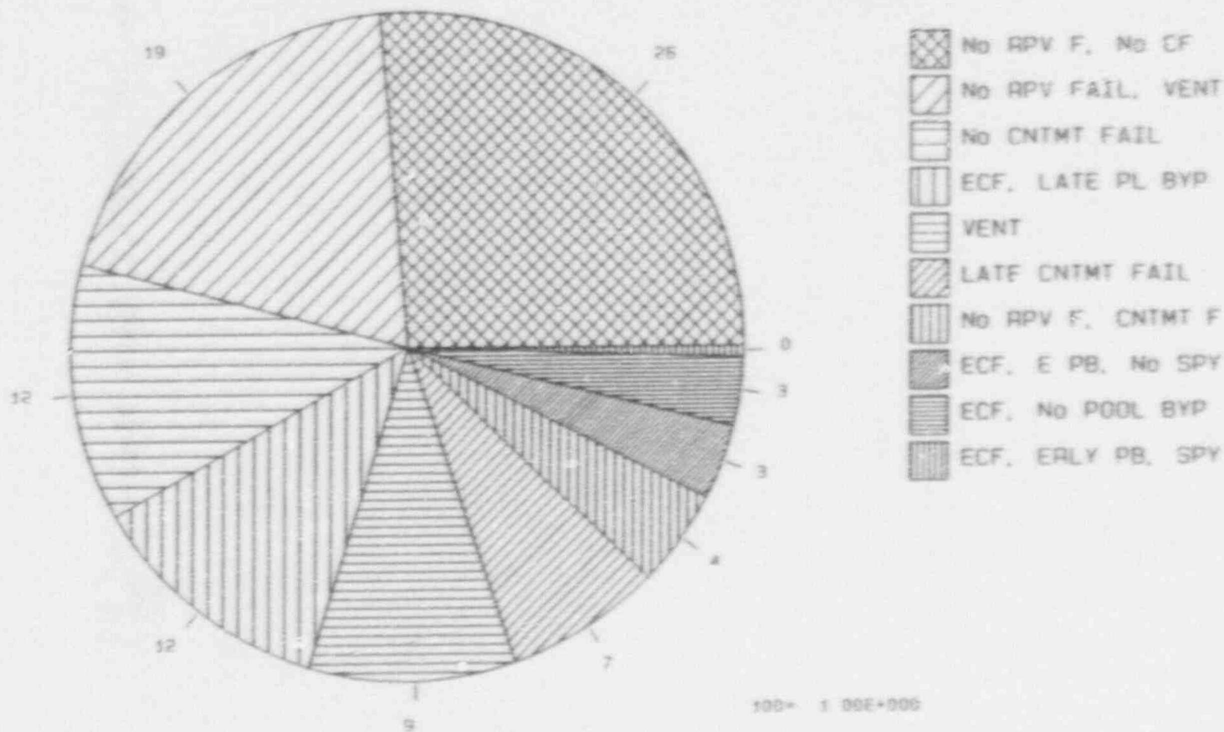
Impact of Design Change Considerations on Total Core Damage Frequency

Modification	Approx CDF After	Percent Change
Passive Vent Path	1.1E-5	-20
Automatic ADS Inhibit for ATWS	1.0E-5	-22
Combined	7.7E-6	-41

Note: The impact of the passive vent path will have an equivalent effect in reducing the core damage frequency due to flooding.

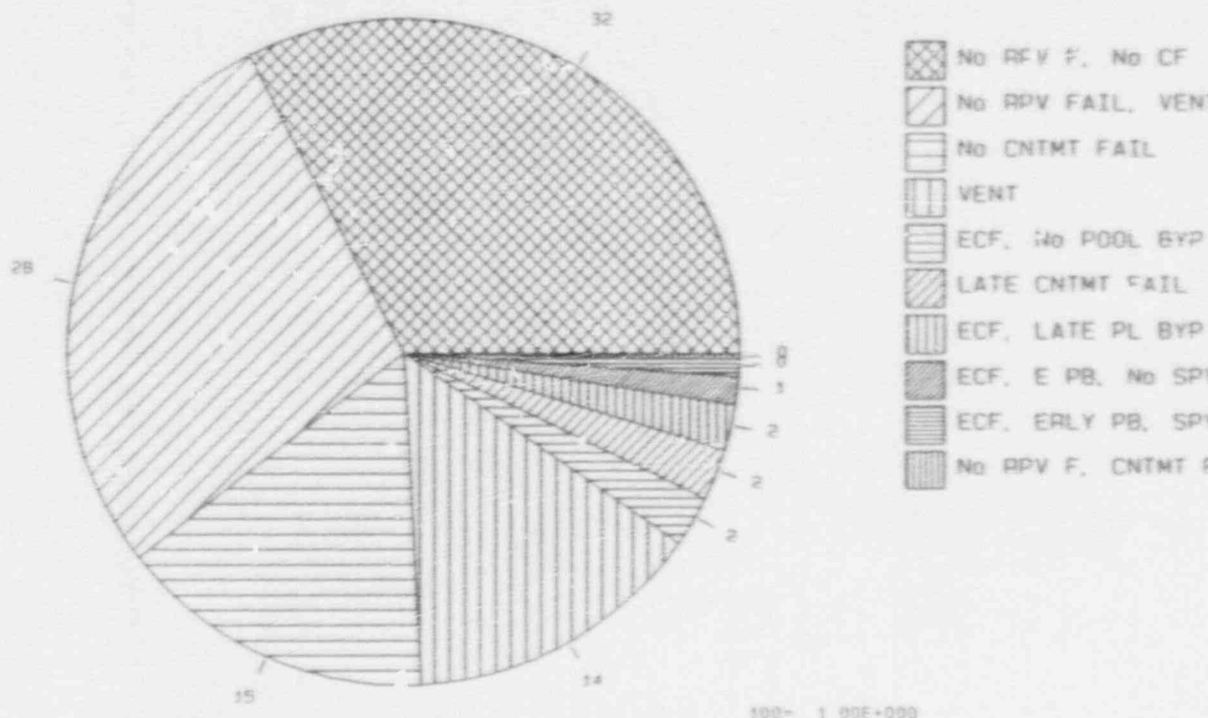
TABLE 6-2 IMPACT OF DESIGN CHANGE CONSIDERATIONS ON CONTAINMENT FAILURE FREQUENCY

		BASE CASE	(1) PASSIVE VENT	(2) ATWS ALTRNATE SHUTDOWN & ADS INHIBIT	(3) HIS BACKUP POWER SUPPLY	(4) PASSIVE VENT, ALT S/D & ADS INHIBIT	(5) PASSIVE VENT, ATWS ALT S/D & ADS INHIBIT, HIS BACKUP POWER SUPPY
No RPV Failure:	No Containment Failure	3.39E-6 (26.7%)	3.39E-6 (32.6%)	1.82E-6 (17.6%)	3.42E-6 (26.9%)	1.82E-6 (22.7%)	1.85E-6 (23.1%)
	Vent	2.45E-6 (19.3%)	2.92E-6 (28.1%)	2.81E-6 (27.2%)	2.45E-6 (19.3%)	3.28E-6 (41.0%)	3.28E-6 (41.0%)
	Containment F.	6.18E-7 (4.9%)	2.87E-8 (0.3%)	6.18E-7 (6.0%)	5.93E-7 (4.7%)	2.88E-8 (.4%)	3.58E-9 (.04%)
Subtotal No RPV Failure Core Damage Freq:		6.46E-6 (50.8%)	6.34E-6 (61.0%)	5.25E-6 (50.9%)	6.46E-6 (50.8%)	5.13E-6 (64.1%)	5.13E-6 (64.1%)
RPV Failure:	No Containment Failure	1.58E-6 (12.4%)	1.58E-6 (15.2%)	9.06E-7 (8.8%)	1.79E-6 (14.1%)	9.06E-7 (11.3%)	1.12E-6 (14.0%)
	Vent	1.27E-6 (10.0%)	1.52E-6 (14.6%)	1.30E-6 (12.5%)	1.31E-6 (10.3%)	1.54E-6 (19.3%)	1.59E-6 (19.8%)
	Late Containment Failure	9.38E-7 (7.4%)	2.46E-7 (2.4%)	9.39E-7 (9.1%)	7.31E-7 (5.7%)	2.46E-7 (3.1%)	3.25E-8 (0.4%)
	Early CF: No Pool Bypass	4.30E-7 (3.4%)	2.67E-7 (2.6%)	1.85E-7 (1.8%)	4.30E-7 (3.4%)	2.53E-8 (0.3%)	2.50E-8 (0.3%)
	Late Pool Bypass	1.54E-6 (12.1%)	2.26E-7 (2.2%)	1.34E-6 (13.0%)	1.53E-6 (12.0%)	3.58E-8 (0.4%)	2.58E-8 (0.3%)
	Early PB, Spray	6.12E-8 (0.5%)	6.12E-8 (0.6%)	3.74E-8 (0.4%)	5.12E-8 (0.4%)	3.74E-8 (0.5%)	2.74E-8 (0.3%)
	Early PB, No Spray	4.45E-7 (3.5%)	1.61E-7 (1.5%)	3.67E-7 (3.6%)	4.22E-7 (3.3%)	8.36E-8 (1.0%)	6.05E-8 (0.8%)
Subtotal RPV Failure Core Damage Freq:		6.27E-6 (49.2%)	4.06E-6 (39.0%)	5.08E-6 (49.1%)	6.27E-6 (49.2%)	2.87E-6 (35.9%)	2.88E-6 (35.9%)
TOTAL CORE DAMAGE FREQUENCY:		1.27E-5 (100%)	1.04E-5 (100%)	1.03E-5 (100%)	1.27E-5 (100%)	8.01E-6 (100%)	8.01E-6 (100%)
Subtotal Containment Venting Frequency:		3.72E-6 (29.2%)	4.43E-6 (42.6%)	4.11E-6 (39.8%)	3.76E-6 (29.5%)	4.82E-6 (60.2%)	4.87E-6 (60.8%)
Subtotal Cntmt Structural Failure Freq:		4.03E-6 (31.7%)	9.92E-7 (9.5%)	3.49E-6 (33.8%)	3.76E-6 (29.5%)	4.60E-7 (5.7%)	1.75E-7 (2.2%)
TOTAL CONTAINMENT FAILURE & VENTING FREQ:		7.76E-6 (60.9%)	5.43E-6 (52.2%)	7.60E-6 (73.6%)	7.52E-6 (59.1%)	5.28E-6 (65.9%)	5.04E-6 (63.0%)
RPV FAILURE & EARLY CONTAINMENT FAILURE WITH POOL BYPASS FREQUENCY:		2.04E-6 (16.1%)	4.48E-7 (4.3%)	1.75E-6 (16.9%)	2.00E-6 (15.8%)	1.57E-7 (2.0%)	1.14E-7 (1.4%)



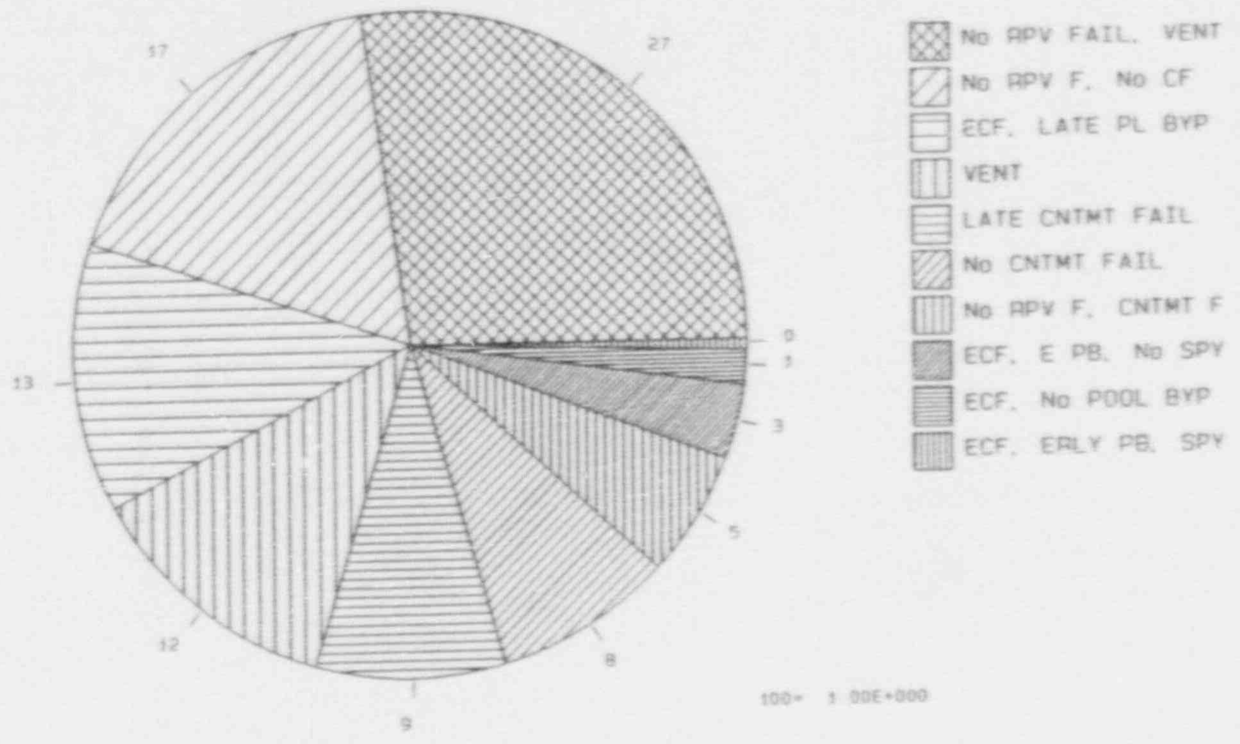
CONTAINMENT PERFORMANCE: BASE CASE

Figure 6.2.2-1



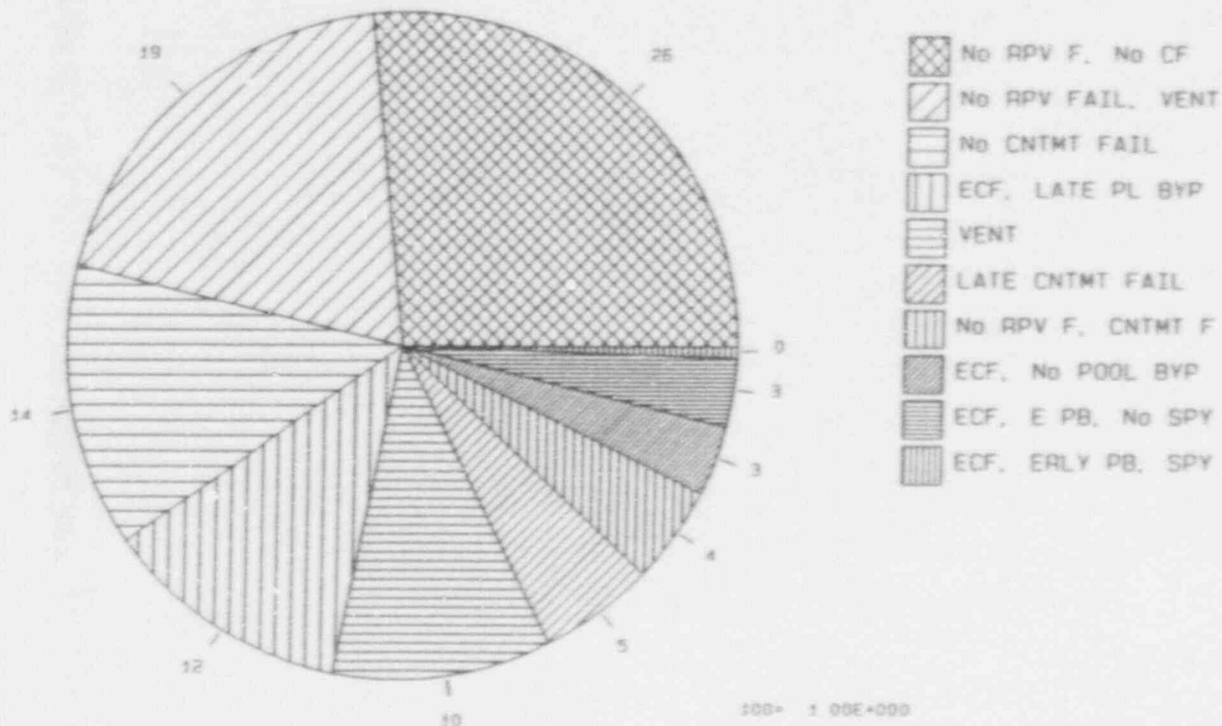
CONTAINMENT PERFORMANCE: PASSIVE VENT

Figure 6.2.2-2



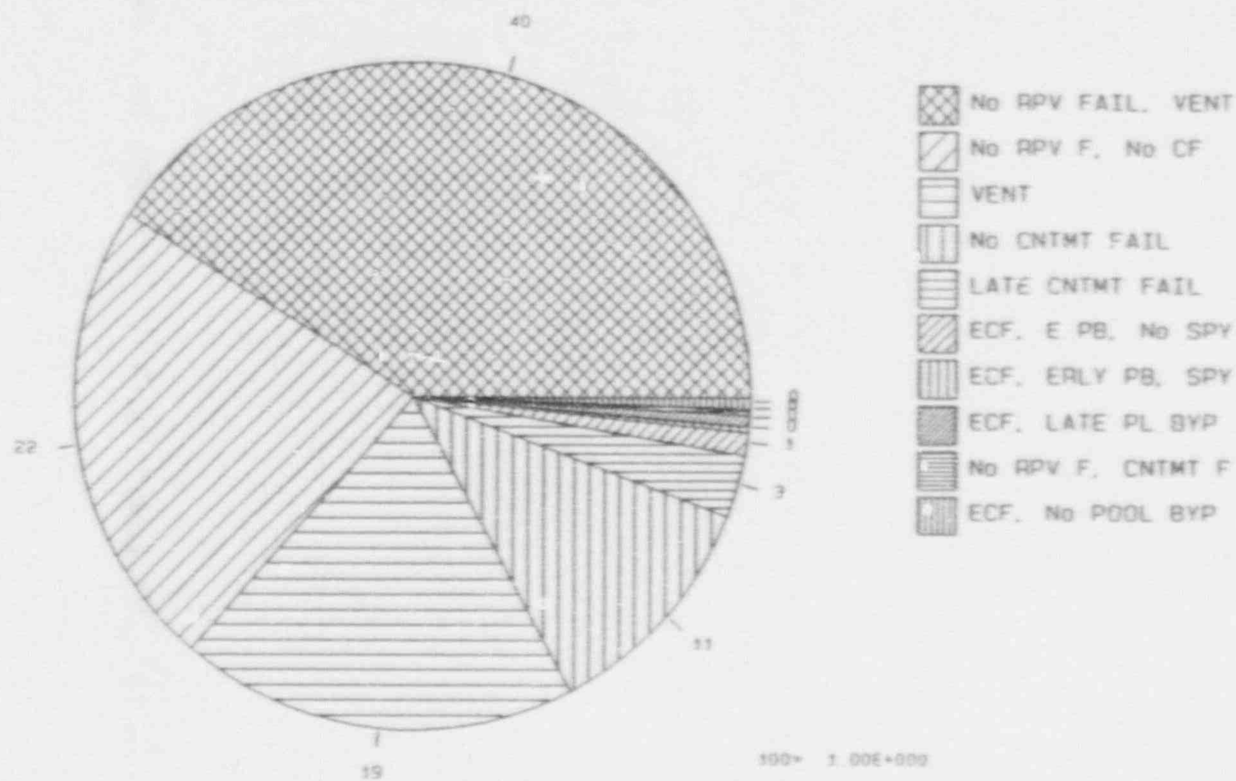
CNTMT PERFORMANCE: AUTO ADS INHIBIT & ALT SHUTDOWN

Figure 6.2.2-3



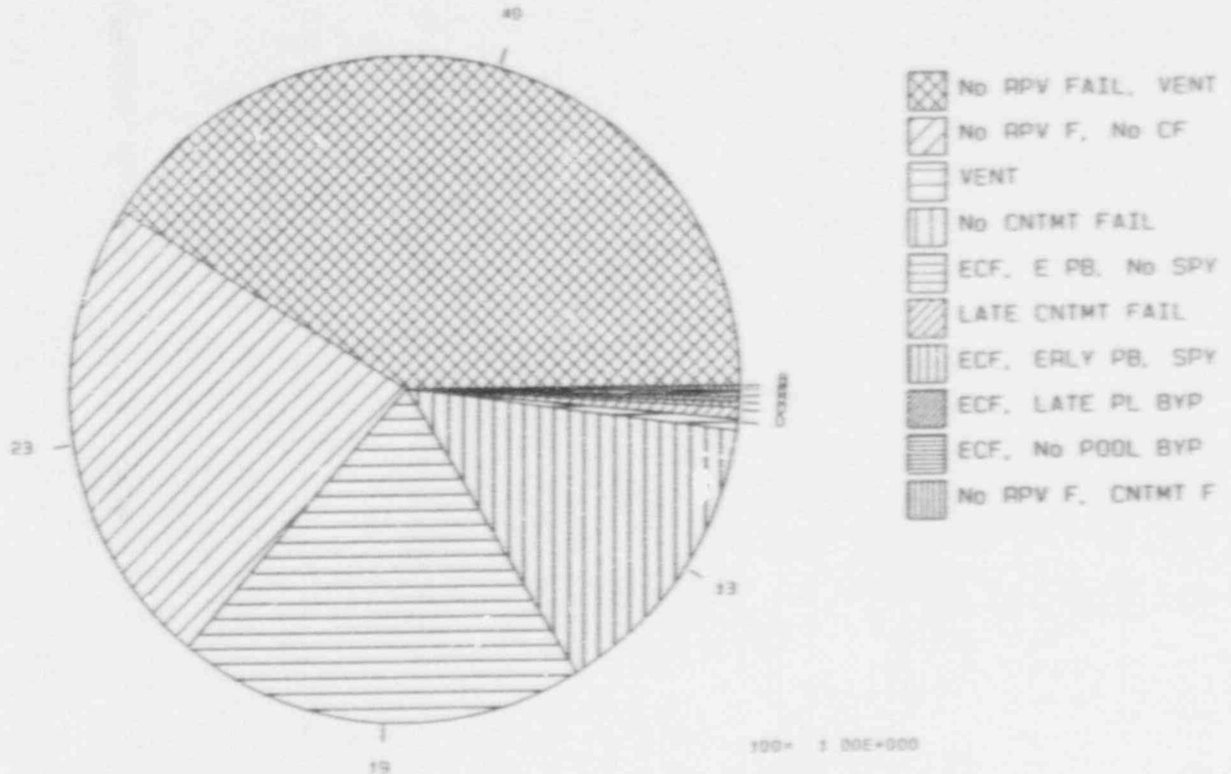
CNTMT PERFORMANCE: H2 IGNITION SYSTEM BACKUP POWER

Figure 6.2.2-4



CNTMT PERFORMANCE: PASSIVE VENT & ATWS MODS

Figure 6.2.2-5



CNTMT PERFORMANCE: VENT, ATWS MODS & HIS POWER

Figure 6.2.2-6

7.0 SUMMARY AND CONCLUSION

7.1 CORE DAMAGE FREQUENCY

The point estimate of core damage frequency from internal initiating events is 1.2×10^{-5} per year and from internal flooding 1.5×10^{-6} per year giving an overall core damage frequency of 1.3×10^{-5} per year. Eighty-six percent of the contribution from internal events comes from 21 sequences with a core damage frequency greater than 10^{-7} per year and the contribution to internal flooding comes from seven areas in which floods will result in core damage frequencies greater than 10^{-7}.

The dominant accident initiating event type is anticipated transient without scram at 40.7 percent, station blackout contributes 19.3 percent and loss of offsite power contributes 12.4 percent. All other transients contribute 25.0 percent and LOCAs 2.6 percent.

For the internal floods analysis, flooding in Zone 13, contribute 57 percent and in Zone 17 contributes 21 percent.

The breakdown by class of initiator is shown in Figure 7-1, by internal initiator in Figure 7-2, and by flood area in Figure 7-3.

The plant-specific safety feature and design considerations are discussed at length in section 6.0 and will not be repeated in this section. However it is considered appropriate to compare the results of this study with other published studies of BWR/6 plants to identify variations in the results and possible reasons for these differences.

The Grand Gulf plant, a BWR/6 Mark III very similar to the Perry plant, was the subject of a recent analysis reported in NUREG/CR-4550 (Drouin, 1989). The NRC study was performed using similar methodology, that is fault tree linking, so it is possible to compare the results in terms of sequences contributing to core damage. As the Grand Gulf study did not include an internal flooding analysis, only the results from the Perry internal analysis are compared with the Grand Gulf results.

In addition to comparing the results with those of previous studies, the results are also evaluated against the Nuclear Management and Resources Council (NUMARC) severe accident closure guidelines (NUMARC, 1992).

7.1.1 COMPARISON WITH GRAND GULF RESULTS

The major purpose of this study was to ensure that the PRA model was developed and understood by the CEI staff at Perry and represented the as-built as-operated condition at Perry as far as possible. Guidance for performing the IPE indicated that heavy reliance could be placed on the results of previous studies for similar plants (e.g., the Grand Gulf study). However it was decided that if the completed PRA was to be used as a living PRA, the success criteria and plant models would have to be developed for the as-built condition of the Perry plant and incorporate the latest understanding of the BWR/6 response to ATWS events, and the thermal hydraulic performance of the core when the water level is below the top of the active level.

In addition a plant-specific human reliability analysis was performed in order to incorporate the impact of the current emergency procedures at Perry into the event and fault trees and to ensure that the dependencies between the various operator actions are correctly modeled.

It can be seen in Table 7-2 that as the result of incorporating the plant specific insights into the models and developing a completely new set of event and fault trees the core damage frequency from internal events at Perry is 1.2×10^{-5} compared with 4.0×10^{-6} in NUREG/CR-4550 for Grand Gulf. The results from the BWR6/s in Taiwan (Kuosheng) and Spain (Cofrentes) are also included in Table 7-2 for comparison purposes. It was not possible to compare the results with the utility developed IPEs for Grand Gulf, Clinton or River Bend as the IPE submissions were not available at the time of production of this report.

The only significant differences between Perry and Grand Gulf are in the results for anticipated transient without scram, transients without the power conversion system available and loss of instrument air. Some of the reasons for these differences are discussed in the following paragraphs.

It can be seen from Table 7-2 that there is a wide variation in the assessment of ATWS events in the four studies, ranging from less than 10^{-7} to 2.6×10^{-5}, a factor of over 250. The specific reasons for the differences between the Grand Gulf core damage frequency and the Perry frequency are the result of different success criteria being used in the two studies. In the Perry study the latest information from the BWR/6 owners group that the High Pressure Core Spray can not be used to maintain vessel level has been factored in. In the Grand Gulf study, it was assumed that if HPCS started successful injection would be achieved. When it is assumed that HPCS can not be used for high pressure injection, the feedwater system becomes important. Runback occurs following ATWS requiring the operators to take control to restore feedwater injection and maintain level. The net effect is to place a dependence on the operator to achieve the required plant status. The quantification of the operator actions based on a detailed evaluation of the PEI resulted in the core damage frequency of 4.74×10^{-6} for ATWS.

The only other two initiators which show a significant difference are the Transient without PCS (1.7×10^{-6} at Perry compared with 1.3×10^{-8} for Grand Gulf) and loss of Instrument Air (1.0×10^{-6} compared with less than 10^{-7}). In the case of loss of PCS the failure of recovery of PCS at Grand Gulf is two orders of magnitude lower than at Perry. This is not in line with the value generally used in other studies which has been used in the Perry study. Similarly, in the loss of instrument air analysis, it is considered that non-conservative recovery actions have been used in the Grand Gulf study.

7.1.2 SEVERE ACCIDENT CLOSURE ISSUES

A concise definition of vulnerability is not given in the documentation associated with the performance and reporting of the IPE. In the response to questions in Appendix C to the Submittal Guidance Document (NRC, 1989), mention is made of examining sequences that are above the screening criteria in order to determine if a weakness exists. Thus the word weakness replaces the word vulnerability neither of which is defined in numerical or comparative terms. In another response it is suggested that a vulnerability is an outlier. The NUMARC Severe Accident Issues Closure Guidelines (NUMARC, 1992) proposes a set of guidelines based on a combination of the core damage frequency for a group of sequences and the individual contribution from a sequence group (Table 7-4). If the contribution from a given initiator or system failure is greater than 50 percent of the total core damage frequency it is interpreted as a significant vulnerability, if it contributes 20 to 50% it is interpreted as a potential vulnerability to be investigated. Similarly, contributions from sequence groups between core damage frequencies of 10^{-5} and 10^{-4} are reviewed to determine if there is an effective plant procedure or hardware change which would reduce the frequency of the sequences.

In [illegible]dy Importance and Sensitivity measures have been used to determine t[illegible]nificant contributions to the core damage frequency, containment system [illegible]ance, and decay heat removal functions.

[illegible] l accident sequence groups and the definition of each group and [illegible] of sequences in each group are shown in Table 7-3. It can be [illegible] table that there are no significant vulnerabilities as defined [illegible]ous section as all the accident sequence groups have a frequency [illegible] and no group contributes more than 50% to the overall core damage frequency. However there are two groups of accident sequences that contribute between 20 and 50 percent: Group 4 which is made up of accident sequences involving Anticipated Transient Without Scram and Group 2 which is made up of accident sequences involving loss of containment heat removal leading to containment failure and subsequent failure of coolant inventory make-up.

The contribution to core damage from sequences in group 4 comes primarily from ATWS sequences in which the motor feed pump has failed to inject water and ADS has not been inhibited resulting in rapid depressurization of the RPV and injection of low pressure ECCS. This leads to a series of reactivity oscillations resulting in generation of large quantities of steam and ultimately containment failure and core damage. However, it should be noted from the sensitivity analysis that the use of the plant operating data for cycles 2 and 3 will reduce the frequency of the initiators which contribute to this group and thus the contribution to core damage frequency of these sequences from 34% to approximately 15% which is no longer a potential vulnerability in terms of the NUMARC criteria.

The contribution to core damage from sequences in Group 2 arises from the failure of containment heat removal leading to containment failure and subsequent loss of injection. One of the reasons that this is a significant contributor is that the containment design is such that in approximately 15% of cases containment failure leads to injection failure. Thus if a passive vent was fitted, the core damage frequency of these sequences would be reduced. This also has an impact on source term magnitude and is further discussed in section 3.4.2.3 under containment vulnerabilities.

7.1.2.1 Closure Issue Sensitivity Analysis

A review of the plant operations during the past two operating cycles has shown that the frequency of scrams is much lower than the generic frequency used for the Grand Gulf study and this study. The results of using a revised set of initiating event frequencies based on the past two cycles results in a core damage frequency reduction from 1.2×10^{-5} to 7.0×10^{-6}. The contribution from the various initiators thus becomes 30% from station blackout, 23% from transients, 20% from loss of offsite power and 17% from ATWS (see Table 3.4.1-16). More significantly, the accident sequence groups which are now between 20 and 50 percent are Groups 1C and Group 2. Group 4 is no longer important. The contribution, in frequency terms of any group is now barely above 2.0×10^{-6}.

If this operational trend continues and the importance of the various accident sequences in Group 4 remains low, it is clear that the updating of the PRA using plant-specific data should be done before deciding on the necessity of performing any design changes.

7.2 CONTAINMENT ANALYSIS

7.2.1 SEVERE ACCIDENT CLOSURE ISSUES (NUMARC 91-04)

The Level 2 containment analysis assessed the performance of the Perry Mark III containment in mitigating severe accidents by defining the accident progression in containment, estimating the timing and mode of containment failure and estimating the source terms for the spectrum of accident sequences. The low values for the plant damage state frequency can be attributed to good plant emergency instructions and to prioritized alignment of alternate injection systems to recover a damaged core in-vessel.

The Modular Accident Analysis Program (MAAP) was used to analyze the dominant accident sequences. The ATWS and Station Blackout sequences were closely examined to determine key figures of merit (e.g., RPV core uncovery, RPV failure time and containment failure time). MAAP source term analyses were used to evaluate the fission product transport for the dominant source term categories from the reactor core to the drywell, through the suppression pool to the containment, and to the environment in accordance with the modes of containment failure.

The Perry Mark III steel containment was analyzed by the Perry plant architect to identify the potential failure modes under severe accident loading conditions. The analysis determined the expected failures from containment penetrations and other containment structural components as well as the less likely failure of the containment anchorage. Penetration failures would commence with a small leakage area and increase with containment pressure. Concrete and steel anchorage failure would result in gross failure of the containment vessel from the mat foundation. The IPE found that the drywell head is susceptible to external overpressure failure if a hydrogen burn should occur in containment. The containment capacity analysis characterized the expected type of failure for each failure mode into three classes (leakage, rupture or gross rupture) which is modeled in source term analyses.

The Perry IPE level 2 containment analysis used the Event Progression Analysis (EVNTRE) code which was utilized in the Grand Gulf NUREG/CR-4551 study (Brown, 1990). The Accident Progression Event Tree analyzed the progression of the accident from the onset of core damage through ex-vessel core-concrete interaction to containment failure and fission product release. To enable the complex structure of the Accident Progression Event Tree to be more clearly understood by the plant staff, the Perry APET is graphically described by a "Summary" Containment Event Tree. The Perry Accident Progression Event Tree generally references the Grand Gulf template study event data. However, the primary reference to modeling phenomenological parameters is with the MAAP 3.0B code.

The MAAP code was applied to determine values for key parameters in the Perry Accident Progression Event Tree; e.g., best-estimate hydrogen generation, pedestal overpressurization, and pedestal core-concrete interaction. All the sensitivity analyses recommended by the EPRI report "Recommended Sensitivity Analyses For An Individual Plant Examination Using MAAP 3.0B" were performed.

The plant damage state profile from the level 1/2 interface is: 0% containment bypass; 77% containment intact at core damage (9% - Station Blackout, and 68% - transients and other event types with AC power available), 23% - containment failed at core damage (4.4% - critical (not shutdown) ATWS sequences, 4.3% - Loss Of Offsite Power and Station Blackout, and 14% - other events).

The Accident Progression Event Tree summary results for containment performance are: no containment failure - represents a 39% conditional probability given core damage and a frequency of 5.0×10^{-6}, containment venting - represents a 29% conditional probability and a frequency of 3.7×10^{-6}, and containment structural failure - represents a 32% conditional probability and a frequency of 4.0×10^{-6}. The conditional probability estimates of the detailed containment failure modes evaluation are: 50.8% - in-vessel cooling and no RPV failure; and the balance of the sequences with RPV failure (49.2%) is composed of: 12.4% - no containment failure, 10% - venting with a damaged core, 7.4% - late containment failure, 3.4% - early containment failure with no pool bypass, and 16.1% containment failure with pool bypass.

7.2.2 COMPARISON WITH GRAND GULF RESULTS

The Perry IPE level 2 containment analysis used the same Event Progression Analysis (EVNTRE) code applied in the Grand Gulf NUREG/CR-4551 study (Brown, 1990) to transpose the severe accident analysis phenomenological framework of the template study and to model the many dependencies associated with containment loading mechanisms such as steam generation, hydrogen generation and combustion, and the resultant changes in containment pressure and temperature. The Perry IPE Accident Progression Event Tree consists of 68 questions which addresses the four general time frames of accident progression: initial, early, intermediate and late. The more extensive NUREG/CR-4551 Grand Gulf evaluation consists of 125 questions.

To evaluate the vulnerability to interruption of power to the hydrogen ignitors, the Perry APET addressed hydrogen combustion phenomena in a manner similar to the NUREG/CR-4551 Grand Gulf APET. The Perry APET models the recovery of offsite power both before RPV failure as well as after RPV failure, and possible variations in hydrogen concentration at the time of power restoration were evaluated including the possibility of detonable concentrations.

The Perry IPE use of the same event progression analysis code as the Grand Gulf template study enabled a good transfer of phenomenological modeling as well as of the quantification bases. The Perry APET routinely referenced the Grand Gulf template study and transferred many of the event models, such as the bounding model of alpha steam explosions. However, the primary reference to modeling phenomenological parameters was the EPRI maintained MAAP 3.0B code.

The IPE sensitivity analysis of parametric values used to quantify the Accident Progression Event Tree found one parameter important to containment performance: large in-vessel steam explosion bottom head failure. Bottom head steam explosion phenomena is not modeled in the MAAP code nor addressed in the EPRI Recommended Sensitivity Analyses discussed above in section 7.2. The probabilities applied in the Grand Gulf template plant study appeared to be overly conservative and were reduced by a factor of 10 for the Perry APET. The sensitivity of inputting the Grand Gulf NUREG/CR-4551 values for large in-vessel steam explosion failure resulted in a 58% decrease for successful in-vessel cooling and No RPV Failure (from the base case estimate of 51% to the sensitivity estimate of 21%).

The results of the Perry IPE containment performance differ from those in the NUREG/CR-4551 Grand Gulf Study, due to differences in containment failure modes, phenomenological assumptions, and plant damage state group frequencies. The Grand Gulf study was dominated by Station Blackout (97% plant damage state frequency) and determined that hydrogen combustion was the dominant cause of containment failure. The Perry IPE was dominated by shutdown ATWS sequences (ATWS with successful SLC) and other transients with Station Blackout only accounting for 9%. The modeling for debris cooled in-vessel is similar with the one variance noted above regarding the estimated value for large in-vessel steam explosions. When the Perry APET event for large in-vessel steam explosion was set to the Grand Gulf value the estimated results for No RPV Failure compare closely with 21% for the Perry APET and 18% for the Grand Gulf APET. The Perry APET estimates for the conditional probability of Containment Performance are shown below with the 4551 Grand Gulf APET estimates:

	Perry	NUREG/CR-4551 Grand Gulf
No RPV Failure	51%	18%
RPV Failure and No Containment Failure	12%	5%
RPV Failure and Venting	10%	4%
RPV Failure and Late Containment Failure	7%	28%
RPV Failure and Early Containment Failure With No Pool Bypass	3%	22%

RPV Failure and Early Containment Failure With Pool Bypass	16%	21%

7.2.3 SEVERE ACCIDENT CLOSURE ISSUES

The containment bypass (unisolated breach of the primary containment outside the containment) frequency is determined to be less than 1×10^{-8} and therefore does not require any action at Perry.

Supplement No. 3 of Generic Letter 88-20 identified that Mark III containments are expected to evaluate the vulnerability to interruption of power to the hydrogen ignitors. The modification of the electrical supply to the hydrogen ignitors to ensure availability during SBO would remove the possibility of high containment loads from hydrogen deflagrations and detonations. The overall reduction of this change on the base case assessment is very minor: 1) the RPV failure and early containment failure with pool bypass frequency changes from 2.04×10^{-6} to 2.00×10^{-6} (-2% change), 2) the containment structural failure frequency changes from 4.03×10^{-6} to 3.76×10^{-6} (-6.7% change). Thus, a hardware upgrade to provide uninterrupted electrical supply to the hydrogen ignitors is not warranted by this improvement.

Containment Performance base case results with the generic initiating event frequency (provided in Table 6-2) showed that the frequency of RPV failure and early containment failure with pool bypass was 2.0×10^{-6} or 16% of the core damage frequency. IPE Level 2 engineering insights suggest the greatest opportunity for containment performance improvement is with the following two design considerations: 1) passive vent, and 2) ATWS modifications (ADS Inhibit and Alternate Shutdown). The passive vent impact is to reduce the core damage from 1.3×10^{-5} to 1.0×10^{-5} (-18% change), and to reduce the RPV failure and early containment failure with pool bypass from 2.0×10^{-6} to 4.5×10^{-7} (-78% change). The ATWS modifications impact is to reduce the core damage frequency from 1.3×10^{-5} to 1.0×10^{-5} (-19% change), and to reduce the RPV failure and early containment failure with pool bypass from 2.0×10^{-6} to 1.8×10^{-6} (-14% change).

7.2.4 SEVERE ACCIDENT CLOSURE SENSITIVITY ANALYSIS

A review of containment performance design considerations using updated initiator frequencies (similar to that performed in section 7.1.2.1) showed that the associated reduction in core damage frequency and plant damage state frequency impacts the results of the containment performance analysis (Table 7-7). A summary comparison of design considerations using the the base case and updated initiator base case results provided in Table 7-5.

7.3 CONCLUSIONS

The performance of the level 2 PRA in response to the NRC's request in Generic Letter 88-20 for an Individual Plant Examination of the Perry Plant has resulted in the CEI gaining a number of insights into the contribution to risk at the plant. The outcome was the performance of the following improvements during the course of the study.

Loss of Offsite Power Instruction

- Retention of RCIC isolation bypass for high steam tunnel temperature
- Enhanced process for crosstieing Unit 1 and 2 batteries
- Enhanced process for offsite power recovery to HPCS and alternate injection system bus bars.

Flooding Instruction

- Enhanced response instructions to flooding scenarios.

In addition, the following improvements are expected to be made in the near future.

- Implementation of automatic RPV depressurization for non-ATWS events (following USNRC review).
- "Fast Firewater" tie between Fire Protection and HPCS
- Permanent Division 3 to Division 2 "quick" connect
- Reduction of Out of Service Time for certain critical components (already achieved for HPCS and RCIC)

The result of the improvements is an overall core damage frequency for the internal and internal flooding events of 1.3×10^{-5} per year and an RPV failure with early containment fail with pool bypass of 2×10^{-6} per year. Beyond the base case, a number of further enhancements discussed in the previous section have been identified and are being evaluated further. However careful analysis is required before any further improvements beyond those identified above are made.

When performing a major analysis of this type, it is necessary to fix the date for collection of design and operational information, in this case January 1, 1990. At this time Perry had only completed one full cycle of operation and therefore little or no operational data was available. Since that time, it has been possible to include data for maintenance outages on a small number of components. It is clear from the experience in operating cycles 2 and 3 that the initiating event frequencies are significantly lower than the generic values used in the NRC Grand Gulf study and therefore in the Perry study. Therefore, before any further decisions are made concerning plant improvements the first step will be to update the living PRA to include this data, and any design changes made since the freeze date of January 1, 1990. In the case of the latter, a brief review of the work performed during the first two cycles indicate that there have been no major design changes to the ECCS.

The enhancements discussed earlier improve the decay heat removal capabilities following an initiating event. It is considered that the current core damage frequency, as the result of decay heat removal failures is within the current guidelines and therefore the results of this study represent satisfactory resolution of Unresolved Safety Issue A-45 for the Perry Nuclear Power Plant.

There were no specific vulnerabilities identified with regard to containment performance in the Perry IPE. The Perry backend containment analysis indicates that the Perry containment response to severe core damage accidents is generally similar to that for other BWR/6 Mark III plants (e.g., NUREG-1150 Grand Gulf [NRC, 1989]).

The containment performance summary results for the Perry Mark III containment were: no containment failure - represents a 39% conditional probability given core damage and a frequency of 5.0×10^{-6}, containment venting - represents a 29% conditional probability and a frequency of 3.7×10^{-6}, and containment structural failure - represents a 32% conditional probability and a frequency of 4.0×10^{-6}. The conditional probability of RPV failure and early containment failure with pool bypass was estimated to be 0.16 in the Perry IPE (compared to 0.21 for Grand Gulf in NUREG-1150). The difference in these results between the Perry IPE and NUREG-1150 mainly result from significant difference in the sequences contributing to core damage, from different containment failure modes, and from the phenomenological assumption made in the Perry IPE regarding the probability of steam explosions failing the lower RPV head.

References

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Brown, T. D. et al 1990. Evaluation of Severe Accident Risks: Grand Gulf, Unit 1, NUREG/CR-4551 Vol 6 Rev 1 Parts 1 and 2. Sandia National Laboratories, Albuquerque, NM.

NRC, 1989a Severe Accident Risks: An assessment for Five U.S. Nuclear Power Plants NUREG-1150 Vol 1 & 2 (Second Draft), USNRC Washington, D.C.

NUMARC, 1992 Severe Accident Issue Closure Guidelines, NUMARC 91-04. Nuclear Management and Resources Council, Inc., Washington D.C.

Table 7-1

Summary of Core Damage Frequency by Initiating Event & Flood Zone

	Core Damage Free	Percent of CDF	
Loss of Offsite Power			
T1	1.80×10^{-7}	1.5	(Loss of Offsite Power)
R	7.19×10^{-7}	6.2	(Loss of Offsite Power and no Offsite Power Recover at 3 hr)
U	4.14×10^{-7}	3.6	(Loss of Offsite Power w/no HPCS or RCIC)
T1P1	7.62×10^{-8}	0.7	(Loss of Offsite Power and 1 SORV)
T1P1U	1.53×10^{-8}	0.1	(Loss of Offsite Power and 1 SORV w/no HPCS or RCIC)
T1P2	3.89×10^{-8}	0.3	(Loss of Offsite Power and 2 SORVs)
Total	1.44×10^{-6}	12.4	
Station Blackout			
B	2.11×10^{-6}	18.1	(Station Blackout)
BP1	8.36×10^{-8}	0.7	(Station Blackout and 1 SORV)
BP2	5.91×10^{-8}	0.5	(Station Blackout and 2 SORVs)
Total	2.25×10^{-6}	19.3	
Transients			
T3A	$< 10^{-8}$	0.0	(Transient w/PCS)
T3AP1	$< 10^{-8}$	0.0	
T3AP2	$< 10^{-8}$	0.1	
T3B	$< 10^{-8}$	0.0	(Loss of feedwater)
T3C	1.38×10^{-7}	1.2	(Inadvertent open SRV)
T2	1.64×10^{-6}	14.1	(Transient w/o PCS)
T2P1	2.47×10^{-8}	0.2	
T2P2	$< 10^{-8}$	0.0	
TIA	1.01×10^{-6}	8.7	(Loss of instrument air)
TIAP1	$< 10^{-8}$	0.1	
TIAP2	$< 10^{-8}$	0.0	
TSW	6.68×10^{-8}	0.6	(Loss of service water)
TSWP1	$< 10^{-8}$	0.0	
TSWP2	$< 10^{-8}$	0.0	
Total	2.90×10^{-6}	25.0	

Table 7-1 continued

		Core Damage Freq	Percent of CDF	
LOCAs				
	A	2.11×10^{-7}	1.8	(Large LOCA)
	S1	6.18×10^{-8}	0.5	(Intermediate LOCA)
	S2	3.34×10^{-8}	0.3	(Small LOCA)
	Total	3.06×10^{-7}	2.6	
ATWS				
	T1-C	3.61×10^{-8}	0.3	
	T3A-C	$< 10^{-8}$	0.1	
	T3B-C	5.42×10^{-7}	4.6	
	T3C-C	9.38×10^{-8}	0.8	
	T2-C	4.02×10^{-6}	34.5	
	TIAC	4.33×10^{-8}	0.4	
	Total	4.74×10^{-6}	40.7	

Total Core Damage Frequency (internal initiators) 1.17×10^{-5} (88%)

Flooding			
	Zone 13	8.84×10^{-7}	57
	Zone 17	3.22×10^{-7}	21
	TPC	8.70×10^{-8}	6
	Zone 1	1.93×10^{-7}	12
	Zone 1A	$< 10^{-8}$	< 1
	Zone 8	2.80×10^{-8}	2
	Zone 16	1.10×10^{-8}	1

Total Core Damage Frequency (flooding) 1.54×10^{-6} (12%)

Total Core Damage Frequency (internal initiators & flooding) 1.32×10^{-5}

Table 7-2

Comparison of Perry IPE with Kuosheng, Cofrentes, & NUREG/CR-4550 Results

<table>
<tr><th>Initiating Event</th><th>Perry</th><th>Kuosheng</th><th>Cofrentes</th><th>NUREG/CR-4550</th></tr>
<tr><td>Loss of Offsite Power</td><td>1.4×10^{-6}</td><td>1.0×10^{-6}</td><td rowspan="2">1.7×10^{-7}</td><td rowspan="2">3.9×10^{-6}</td></tr>
<tr><td>Station Blackout</td><td>2.2×10^{-6}</td><td>3.3×10^{-6}</td></tr>
<tr><td>Transient w/ PCS (T3A)</td><td>$< 10^{-8}$</td><td>1.1×10^{-6}</td><td rowspan="6">$< 10^{-7}$</td><td>$< 10^{-7}$</td></tr>
<tr><td>Loss of feedwater (T3B)</td><td>$< 10^{-8}$</td><td>3.0 X 10-7</td><td>$< 10^{-7}$</td></tr>
<tr><td>Inadvertent open SRV (T3C)</td><td>1.4×10^{-7}</td><td>3.0×10^{-8}</td><td>$< 10^{-7}$</td></tr>
<tr><td>Transient w/o PCS (T2)</td><td>1.7×10^{-6}</td><td>1.3×10^{-6}</td><td>1.3×10^{-8}</td></tr>
<tr><td>Loss of instrument air (TIA)</td><td>1.0×10^{-6}</td><td>-NA-</td><td>$< 10^{-7}$</td></tr>
<tr><td>Loss of service water (TSW)</td><td>6.7×10^{-8}</td><td>-NA-</td><td>-NA-</td></tr>
<tr><td>Large LOCA (A)</td><td>2.1×10^{-7}</td><td>$< 10^{-7}$</td><td rowspan="3">$< 10^{-7}$</td><td>$< 10^{-7}$</td></tr>
<tr><td>Intermediate LOCA (S1)</td><td>6.2×10^{-8}</td><td>4.1×10^{-7}</td><td>$< 10^{-7}$</td></tr>
<tr><td>Small LOCA (S2)</td><td>3.3×10^{-8}</td><td>$< 10^{-7}$</td><td>$< 10^{-7}$</td></tr>
<tr><td>ATWS</td><td>4.7×10^{-6}</td><td>2.6×10^{-5}</td><td>$< 10^{-7}$</td><td>1.1×10^{-7}</td></tr>
<tr><td>Vessel Rupture</td><td>$< 10^{-7}$</td><td>2.7×10^{-7}</td><td>$< 10^{-7}$</td><td>$< 10^{-7}$</td></tr>
<tr><td>Total Core Damage Freq*</td><td>1.2×10^{-5}</td><td>3.4×10^{-5}</td><td>2.6×10^{-7}</td><td>4.0×10^{-6}</td></tr>
<tr><td>Internal Flooding CDF</td><td>1.5×10^{-6}</td><td>5.7×10^{-7}</td><td>-NA-</td><td>-NA-</td></tr>
</table>

* Total CDF does not include flooding.

Table 7-3

Accident Sequence Grouping Criteria

Functional Accident Sequence	Definition	CDF	Percent CDF
1A	Accident Sequences Involving Loss of Coolant Inventory Makeup in Which Reactor Pressure Remains High.	7.0E-8	<1
1B	Accident Sequence Involving a Loss of All AC Power and Loss of Coolant Inventory Makeup.	1.5E-6	13
1C	Accident Sequence Involving a Loss of All AC Power and No Recovery of AC Power.	1.2E-6	10
1D	Accident Sequences Involving a Loss of Coolant Inventory Makeup and ATWS	5.8E-7	4
1E	Accident Sequence Involving a Loss of Coolant Inventory Makeup in which Reactor Pressure has been successfully resolved.	1.5E-6	8
2	Accident Sequences Involving Loss of Containment Heat Removal Leading to Containment Failures and Subsequent Loss of Coolant Inventory Makeup.	2.6E-6	22
3A	Vessel Rupture Leading beyond makeup capability.	<1.0E-7	<1
3B	Accident Sequence Initiatied or resulting in a small or medium LOCA for which reactor cannot be depressurized and inventory makeup is inadequate.	<1.0E-7	<1
3C	Accident sequences initiatied or resulting in medium or large LOCA for which the reactor is at low pressure and inadequate coolant inventory makeup is available.	3.9E-7	3
3D	Accident sequences which are initiated by a LOCA or failure for which vapour suppression is inadequate.	<1.0E-7	<1
4	Accident involving an ATWS leading to containment failure due to high pressure and subsequent loss of inventory.	4.0E-6	34
5	Unisolated LOCA outside containment leading to loss of effective coolant inventory makeup.	<1.0E-7	<1

Table 7-4

Primary IPE Core Damage Evaluation Process

Mean CDF Per Sequence Group (per reactor year)	Licensee Response
Greater than 1E-4 or greater than 50 percent of total CDF	1. Find a cost effective plant administrative, procedural or hardware modification with emphasis on reducing the likelihood of the source of the accident sequence initiator. 2. If unable to satisfy above response, treat in EOPs or other plant procedure with emphasis on prevention of core damage. 3. If unable to satisfy above responses, ensure SAMG is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure.
1E-4 to 1E-5 or 20 percent to 50 percent of total CDF	1. Find a cost effective treatment in EOPs or other plant procedure or minor hardware change with or emphasis on prevention of core damage. 2. If unable to satisfy above response, ensure SAMG is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure.
1E-5 to 1E-6	Ensure SAMG is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure.
Less than 1E-6	No specific action required.

TABLE 7-5 CONTAINMENT PERFORMANCE COMPARISON OF DESIGN CHANGE CONSIDERATIONS

Bypass	GENERIC INITIATING EVENT FREQUENCY			UPDATED INITIATING EVENT FREQUENCY		
	Core Damage	Containment Structural Failure	RPV Failure And Early Containment Failure With Pool Bypass	Core Damage	Containment Structural Failure	RPV Failure And Early Containment Failure With Pool
Base Case	1.3E-5	4.0E-6	2.0E-6	8.1E-6	2.6E-6	1.0E-6
Passive Vent	1.0E-5 (-18% CHG)	9.9E-7 (-75% CHG)	4.5E-7 (-78% CHG)	6.7E-6 (-17% CHG)	5.3E-7 (-80% CHG)	2.0E-7 (-81% CHG)
ATWS Mods: Alt Shutdown & ADS Inhibit	1.0E-5 (-19% CHG)	3.5E-6 (-13% CHG)	1.8E-6 (-14% CHG)	7.3E-6 (- 9% CHG)	2.5E-6 (- 5% CHG)	9.5E-7 (- 7% CHG)
Passive Vent & ATWS Mods	8.0E-6 (-37% CHG)	4.6E-7 (-89% CHG)	1.6E-7 (-92% CHG)	5.9E-6 (-26% CHG)	4.0E-7 (-85% CHG)	1.2E-7 (-89% CHG)
Passive Vent, ATWS Mods & Ignitor Power	8.0E-6 (-37% CHG)	1.8E-7 (-96% CHG)	1.1E-7 (-94% CHG)	5.9E-6 (-26% CHG)	1.2E-7 (-96% CHG)	7.2E-8 (-93% CHG)

NOTE: These results are based on an analysis of the core damage sequences included in the plant damage state trees. Thus there are small differences in the impact of changes reported in this table compared with those reported for internal event core damage sequences in section 3.4. These differences do not change the overall conclusions.

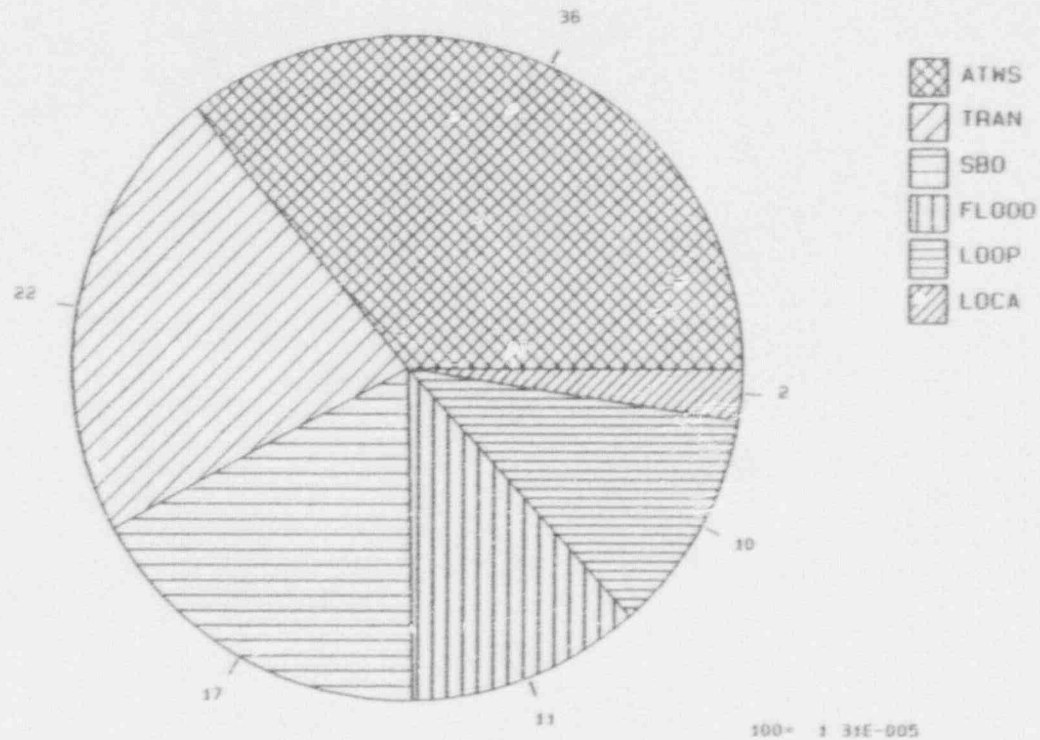
Table 7-6

IPE Containment Bypass Evaluation Process

Mean Containment Bypass Frequency (per reactor year)	Licensee Response
Greater than 1E-5 or greater than 20 percent of total CDF	1. Find a cost effective plant administrative, procedural or hardware modification with emphasis on eliminating or reducing the likelihood of the source of the accident sequence initiator. 2. If unable to satisfy above response, find cost effective treatment in EOPs or other plan procedure with emphasis on prevention of core damage. 3. If unable to satisfy above responses, ensure SAMG is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure.
1E-5 to 1E-6 or 5 to 20 percent of total CDF	1. Find a cost effective treatment in EOPs or other plant procedure or <u>minor</u> hardware change with emphasis on prevention of core damage. 2. If unable to satisfy above response, ensure SAMG is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure.
1E-6 to 1E-7	Ensure SAMG is in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failure.
Less than 1E-7	No specific action required

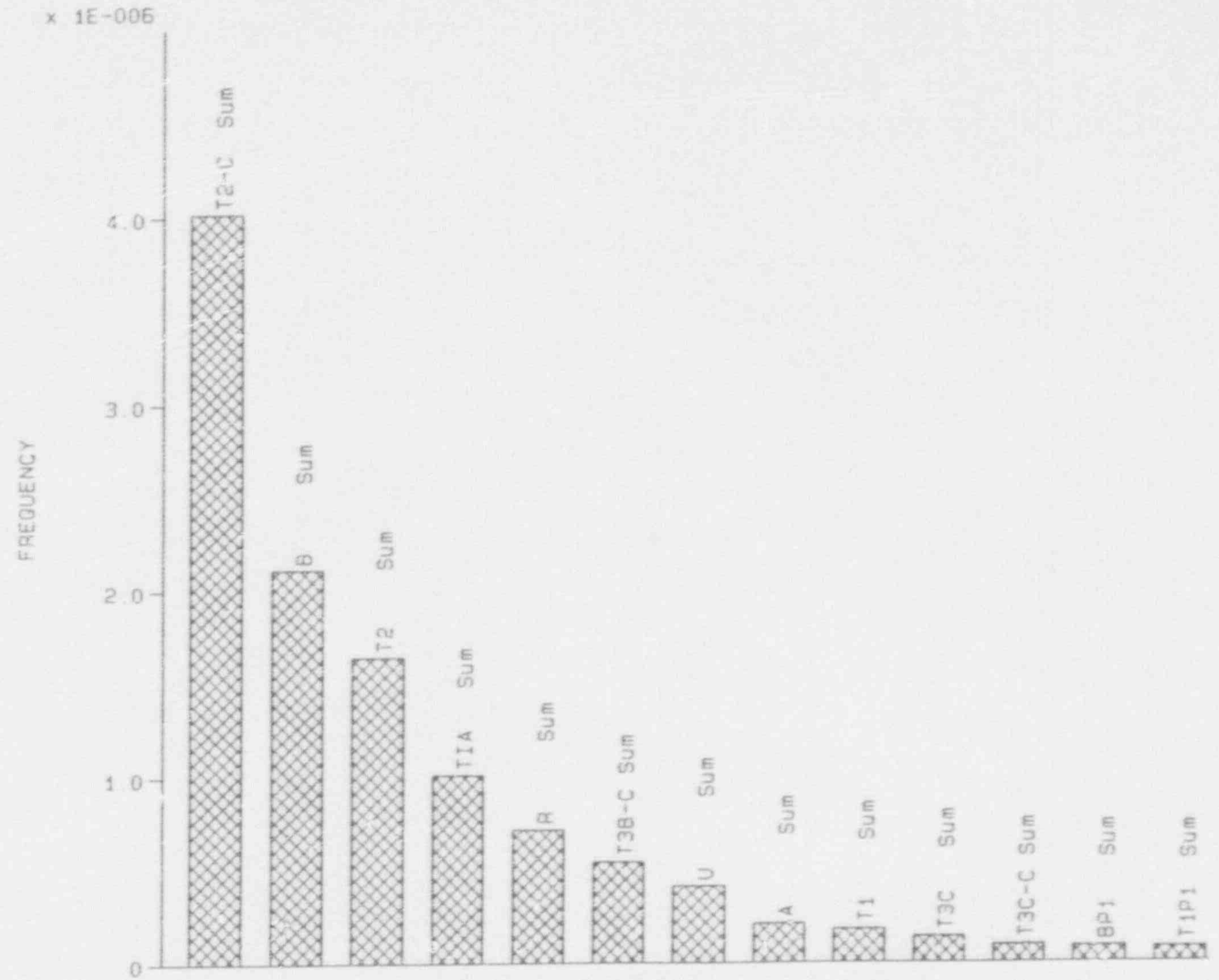
TABLE 7-7 IMPACT OF DESIGN CHANGES ON CONTAINMENT FAILURE FREQUENCY USING UPDATED INITIATOR FREQUENCY

		BASE CASE	UPDATED INITIATOR FREQUENCY	(1) UPDATED PASSIVE VENT	(2) UPDATED ATWS ALT SHUTDOWN & ADS	(3) UPDATED PASSIVE VENT & ATWS MODS INHIBIT	(4) UPDATED PASSIVE VENT, ATWS MODS & HIS BACKUP POWER SUPPLY
No RPV Failure:	No Containment Failure	3.39E-6 (26.7%)	1.22E-6 (15.2%)	1.22E-6 (18.3%)	7.34E-7 (10.0%)	7.34E-7 (12.4%)	7.60E-7 (12.8%)
	Vent	2.45E-6 (19.3%)	2.36E-6 (29.4%)	2.82E-6 (42.3%)	2.45E-6 (33.5%)	2.91E-6 (49.0%)	2.91E-6 (49.0%)
	Containment Failure	6.18E-7 (4.9%)	4.70E-7 (5.8%)	6.74E-8 (1.0%)	4.70E-7 (6.4%)	2.90E-8 (0.5%)	3.79E-9 (0.06%)
Subtotal No RPV Failure Core Damage Freq:		6.46E-6 (50.8%)	4.06E-6 (50.3%)	4.11E-6 (61.6%)	3.65E-6 (50.0%)	3.67E-6 (61.9%)	3.67E-6 (61.9%)
RPV Failure:	No Containment Failure	1.58E-6 (12.4%)	6.22E-7 (7.7%)	6.22E-7 (9.3%)	4.09E-7 (5.6%)	4.09E-7 (6.9%)	6.21E-7 (10.5%)
	Vent	1.27E-6 (10.0%)	1.23E-6 (15.3%)	1.47E-6 (22.1%)	1.24E-6 (16.9%)	1.48E-6 (25.0%)	1.53E-6 (25.7%)
	Late Containment Failure	9.38E-7 (7.4%)	9.25E-7 (11.5%)	2.45E-7 (3.7%)	9.25E-7 (12.7%)	2.45E-7 (4.1%)	3.20E-8 (0.5%)
	Early CF: No Pool Bypass	4.30E-7 (3.4%)	1.88E-7 (2.3%)	2.00E-8 (0.3%)	1.28E-7 (1.7%)	7.37E-9 (0.1%)	7.00E-9 (0.1%)
	Late Pool Bypass	1.54E-6 (12.1%)	7.51E-7 (9.3%)	8.40E-8 (1.3%)	7.04E-7 (9.6%)	2.05E-8 (0.3%)	1.04E-8 (0.2%)
	Early PB, Spray	6.12E-8 (0.5%)	2.82E-8 (0.3%)	2.78E-8 (0.4%)	2.09E-8 (0.3%)	2.09E-8 (0.4%)	1.08E-8 (0.2%)
	Early PB, No Spray	4.45E-7 (3.5%)	2.49E-7 (3.1%)	8.7[illegible] (1.3%)	2.30E-7 (3.1%)	7.36E-8 (1.2%)	5.04E-8 (0.8%)
Subtotal RPV Failure Core Damage Freq:		6.27E-6 (49.2%)	4.00E-6 (49.7%)	2.56E-6 (38.4%)	3.66E-6 (50.0%)	2.26E-6 (38.1%)	2.26E-6 (38.1%)
TOTAL CORE DAMAGE FREQUENCY:		1.27E-5 (100%)	8.05E-6 (100%)	6.67E-6 (100%)	7.31E-6 (100%)	5.93E-6 (100%	5.93E-6 (100%)
Subtotal Containment Venting Frequency:		3.72E-6 (29.2%)	3.59E-6 (44.7%)	4.29E-6 (64.4%)	3.69E-6 (50.4%)	4.39E-6 (74.0%)	4.43E-6 (74.8%)
Subtotal Cntmt Structural Failure Freq:		4.03E-6 (31.7%)	2.61E-6 (32.5%)	5.34E-7 (8.0%)	2.48E-6 (33.9%)	3.99E-7 (6.7%)	1.15E-7 (1.9%)
TOTAL CONTAINMENT FAILURE & VENTING FREQ:		7.76E-6 (60.9%)	6.21E-6 (77.1%)	4.83E-6 (72.4%)	6.17E-6 (84.4%)	4.79E-6 (80.7%)	4.55E-6 (76.7%)
RPV FAILURE & EARLY CONTAINMENT FAILURE WITH POOL BYPASS FREQUENCY:		2.04E-6 (16.1%)	1.03E-6 (12.8%)	1.99E-7 (3.0%)	9.54E-7 (13.1%)	1.15E-7 (1.9%)	7.16E-8 (1.2%)



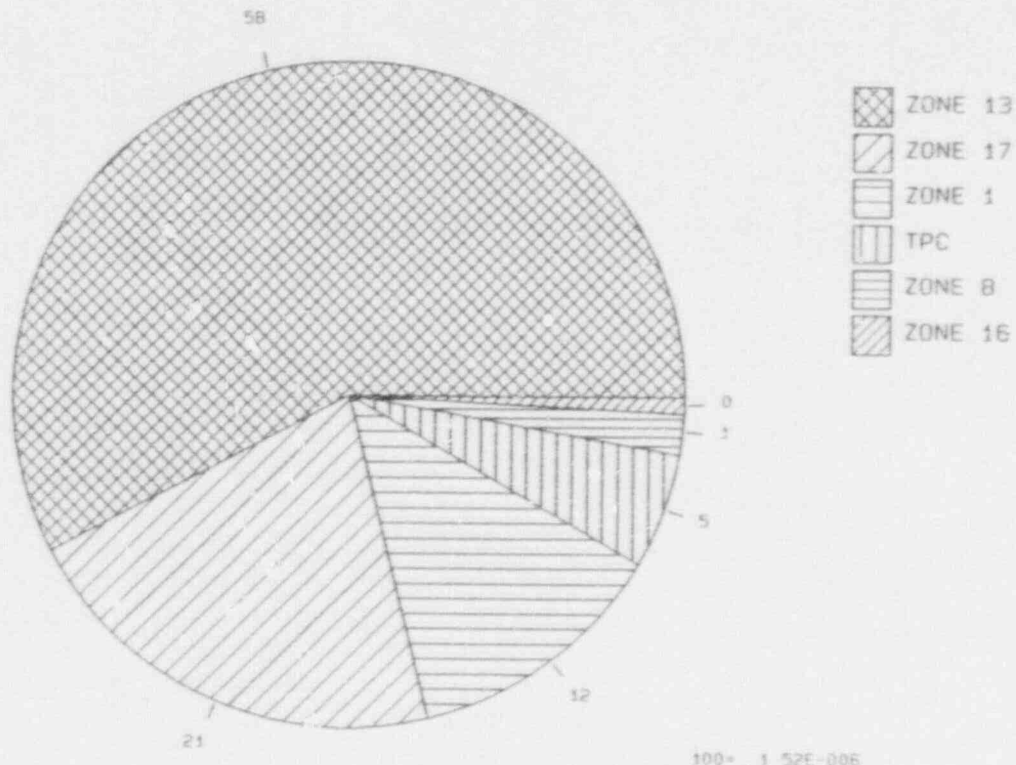
CONTRIBUTION TO CORE DAMAGE BY INITIAITNG EVENT

Figure 7-1



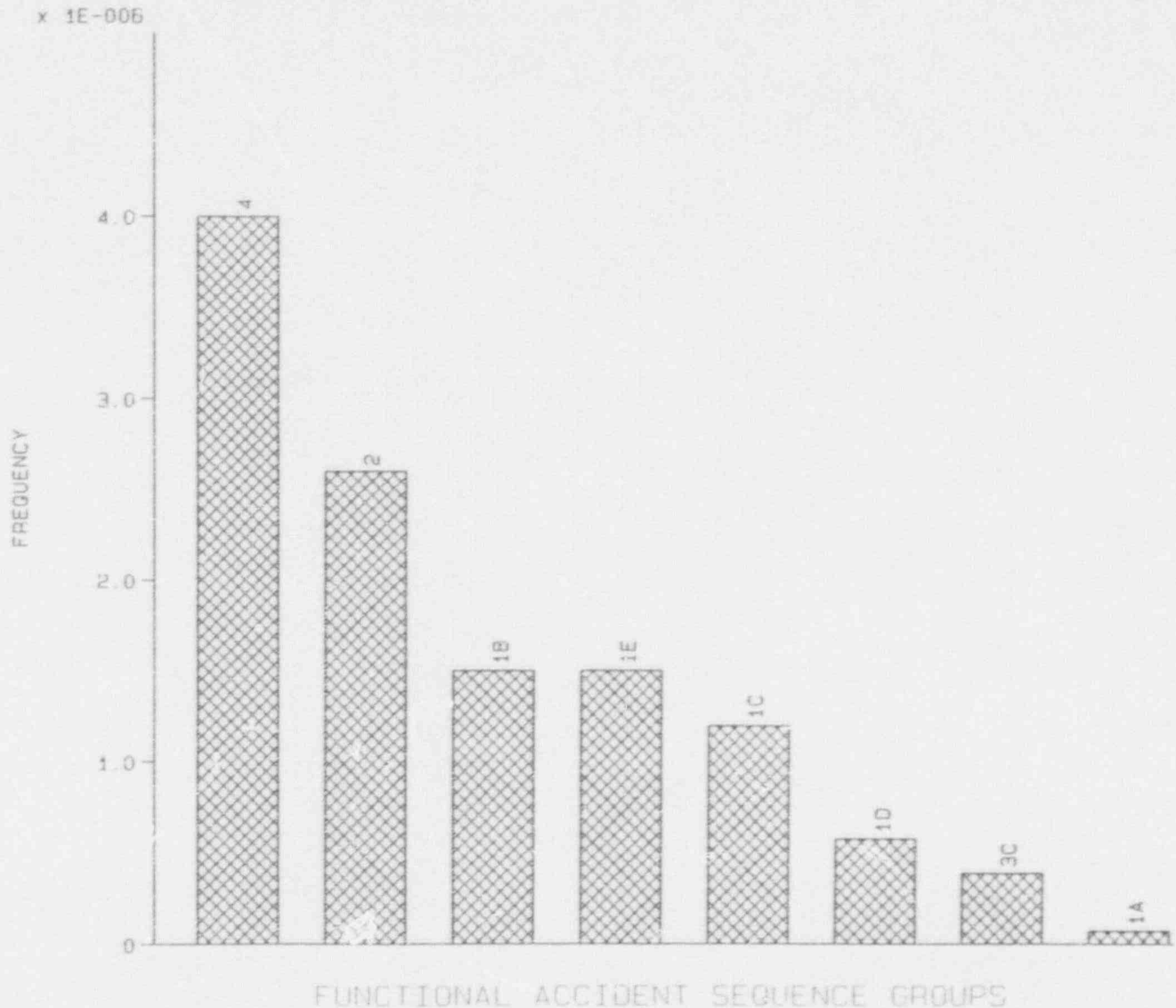
CONTRIBUTION TO CDF BY INITIATOR

Figure 7-2



CORE DAMAGE BY FLOOD ZONE

Figure 7-3


x 1E-006
FREQUENCY
4.0
3.0
2.0
1.0
0
4
2
1B
1E
1C
1D
3C
1A
FUNCTIONAL ACCIDENT SEQUENCE GROUPS

[illegible]ure 7-4

APPENDIX D

HUMAN RELIABILITY ANALYSIS FILE
FOR THE PERRY PLANT IPE

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FIGURES

?.0 INTRODUCTION

The objective of the human reliability analysis (HRA) is to provide estimates of the probabilities of the human interaction (HI) basic events included in the Perry Plant IPE logic model. In general, human interactions considered for inclusion in a PRA can be divided into three general classes according to the time phase in which they occur.

Type A HIs arise before an initiating event, when plant personnel can affect availability and safety of the plant by inadvertently leaving equipment disabled following test, maintenance and calibration activities.

Type B HIs are those HIs that result in, or contribute to, initiating events. Examples are: plant trips following mistakes during testing, failures to control feedwater leading to plant trip, etc. Type B events are almost invariably incorporated implicitly in the initiating event frequencies obtained from plant operating experience. No explicit consideration of Type B His has been included in this study.

Type C HIs cover a wide range of specific actions following an accident. There are two sub-categories of Type C: (1) operator action performed in response to an Emergency Operating Procedure (EOP), including manual backup on failure of automatic initiation of systems, and (2) recovery actions in response to unavailability of a safety function that failed because of equipment malfunction. Events in the first sub-category (Type CP) can either appear as headings in the event trees, or as basic events in system or functional fault trees, while recovery actions (Type CR) are addressed at the accident sequence cutset level.

This analysis file consists of two parts: Part A addresses type A HIs, and Part B, the Type C HIs.

Part A

Analysis of Type A Human Interactions

Type A HIs are those interactions that occur during normal plant operations and include testing, maintenance, and calibration activities. The adverse impact of these activities that is accounted for in PRA models is the potential for leaving components inadvertently available. Typically these include miscalibration of sensors or instrument channels, and leaving valves in an incorrect configuration.

Generally miscalibration events have not been found to be significant contributors to core damage, unless they are common cause failures, where the event affects more than one redundant train. In this study, the common cause error was not evaluated explicitly but was included with a value given by a tenth of the failure of a single sensor.

Because there are a large number of potential restoration error opportunities, a qualitative screening was performed using the following guidelines:

General Component Type A Guidelines

- o Restoration faults following maintenance will not be postulated if the system undergoes a full functional test following completion of maintenance.

- o Restoration faults will not be postulated if the component has an indication in the control room which is verified on a daily basis and is readily apparent to the operators if out of position or if power is disconnected.

- o Restoration faults will not be postulated if the components are included on a daily checklist.

Manual Valve Type A Guidelines

- Restoration faults will not be postulated if there is double (independent) verification of position following test/maintenance and the valve is also verified in the correct position between test events (e.g., on a checklist).

- Restoration or mispositioning faults will not be modeled if the valve is administratively controlled to be in its correct alignment as locked open, locked closed, or locked throttled and checked quarterly or more frequently.

Valves (Other than Manual) Type A Guidelines

- Restoration faults will not be postulated if the valve has an individual position indication in the control room and is included on a daily (or more frequent) checklist.

- Restoration faults will not be postulated if the valve receives a signal to go to the correct position and a position indication light shows if power is not connected.

- Restoration faults will not be postulated if there is double (independent) verification of position following test/maintenance and the valve is also verified in the correct position between test events (e.g., on a checklist).

- Restoration faults will not be modeled if the valve is administratively controlled to be in its correct alignment as locked open, locked closed, or locked throttled with motive power removed.

Only the non-safety systems remained after applying these screening guidelines. A value of 10^{-3} was used for restoration errors in the instrument air and system air systems.

PART B

ANALYSIS OF TYPE C HUMAN INTERACTIONS

1.0 INTRODUCTION

1.1 Overview

The plant logic model, the event trees and fault trees, were constructed to include human interaction basic events. To define the plant logic, these events are adequately defined in terms of the failure mode they represent e.g., operators fail to depressurize the reactor following a loss of high pressure injection. However, in order to quantify these events, i.e., estimate their probabilities, it is essential to define them in greater detail. For example, it is necessary to understand what cues and procedures the operators use to guide them to perform the required function, what they have to do to successfully accomplish that function, the time available, and other factors that might influence their probability of success or failure. These factors are all scenario specific. The first step in the HRA was, therefore, to define the events as clearly as possible in preparation for the quantification. This was done by studying the scenarios to which the human interaction events contributed, and understand, among other things, the time line of the events. This was documented in a series of calculation files (AS-09, AS-10, AS-11, etc.).

Another function of this step of the HRA task is an identification of potential dependence between the human interaction events that occur in the model. Functional dependencies of the type, "if event A occurs, event B cannot be successful," are handled in the overall structure of the model, i.e., they are hardwired into the event tree structure. What is principally of concern here is the influence of success or failure in a preceding event on the probability of success or failure of another event. There are a variety of reasons why the events may be probabilistically dependent; one important issue is that the cognitive processes needed to recognize the need for multiple actions may have common elements. To assist in the identification of such cognitively correlated HIs, the following groundrules are adopted:

(a) If two HI events are associated with responses to the same plant status (e.g., initiate HPCS pump, initiate RCIC on failure of auto initiation at Level 2), the cognitive part of the failure probabilities are considered to be totally dependent.

(b) As a corollary to this, if, in the chronological development of the scenarios, an HI failure event follow a successful HI, and the procedural instructions for both events are closely related, the cognitive failure probability of the second HI should be very small and can be neglected, since the success in the first event implies a successful recognition of the scenario.

(c) If human interactions are i) separated by a significant time (i.e., time between cues or required responses is long), or ii) separated chronologically in the sequence by a successful action, and/or iii) responses to different cues in different parts of the EOPs, they may be regarded as being independent.
However, when applying this would result in very low values for products of HEPs, this was not always used (see discussion on RHR and venting).

(d) In addition, the early memorized responses may be regarded as independent from those actions for which the procedures are expected to be providing the direction.

Other types of dependency, such as the fact that performing one function may take resources away from another is also considered by addressing, in the evaluation of the HEPs, the role of crew personnel, both in performing the actions called for, and in recovering from failure to execute correctly.

1.2 Quantification Approach

The model of human interactions used for the evaluation of a human error probabilities is the simple one that splits the response into two components, a detection, diagnosis and decision (DDD) base, and an execution phase. This is compatible with the ASEP methodology (1), the more recent EPRI proposed methodology (2), and the HRA Handbook (3), all of which were used in the quantification. Reference is made to these documents for details.

For the key time-critical human interactions, the time-reliability curve approach of Reference 2 was used to estimate the probability of failure in the DDD phase. The alternate approach of Reference 2 was used to evaluate the HEPs for those HIs considered dependent on the time critical HI, or which are not time critical. A simplified THERP or ASEP approach was used to estimate the HEP for the execution phase. The details can be found in the following sections.
The analysis is documented essentially on a function-by-function basis in the context of the sequences, rather than on an HI by HI basis, so that the relationship between the HI events and the scenario can be explored. Because of the large number of events included in the model, the detailed analysis has concentrated on these human interaction events which are direct contributors to system performance and appear in functional fault trees or as contributors to the top gates of system fault trees. Those that are responses to specific component or subsystem failures are handled by assigning fairly coarse screening values.

Because all the HRA methods available are to some extent arbitrary, the analysis should be regarded in some cases as providing plausibility arguments rather than accurate assessments. This is particularly true of the long-term actions such as initiation of RHR for which failures are believed to be of extremely low probability.

REFERENCES:

1. "Accident Sequence Evaluation Program Human Reliability Analysis Procedure", Alan D. Swain, NUREG/CR-4772, Feb 1987.

2. "An Approach to the Analysis of Operator Actions in Probabilistic Risk Assessment," EPRI-TR-100259, Dec 1991.

3. NUREG/CR-1278, August 1983, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Application", A.D. Swain and H.E. Guthman.

2.0 PNPP ATWS SEQUENCES

The purpose of this section is to document the human reliability analysis for postulated ATWS scenarios at PNPP.

2.1 MSIV Closure ATWS (T2-C):

2.1.1 Introduction

In the event of an MSIV-ATWS all the steam that is generated goes to the suppression pool via the SRVs. It is assumed that the MSIV closure leads to a loss of feedwater turbine-driven pumps and the PCS; i.e., no heat sink is available. During the initial few seconds the reactor responds to the transient as it would with control rod insertion, then it responds to the recirculation pump trip (RPT) which reduces core flow and therefore power to that corresponding to natural circulation. With no rods going in, the vessel level reaches the Level 2 set-point causing the HPCS and RCIC to automatically actuate and inject water into the vessel. The motor-driven feedwater pump (MFP) starts on L-2, but with the MFP feedwater regulation valves closed by the run-back of MFP, operator action is required to inject water into the vessel.

Operator actions play an important role in the progression of the MSIV closure ATWS sequences. There are four basic goals of the CR crew response to an ATWS: (i) reactivity control, (ii) reactor pressure vessel (RPV) level control, (iii) RPV pressure control, and (iv) suppression pool temperature control. The operators are expected to not only attempt to insert control rods, but to be aware of the limitations of the SLCS (i.e., relatively slow acting) and how void effects can be used to reduce reactor power; i.e., increasing voids by recirculation pump trip, depressurization, or lowering of the water level. During the postulated MSIV-ATWS scenarios, the CR operators will be guided by the PEIs and supporting documentation and the available control room aids (e.g., the Emergency Response Information System, ERIS). As indicated by the event tree and the supporting fault trees for the MSIV-ATWS initiating event, the following HIs are addressed:

- U3: manual start of the motor-driven feedwater pump (MFP) given failure of the auto-start, and opening of the feedwater regulation valves.
- LC; manual control of RPV level between the Minimum Steam Cooling Water level (MSCWL) top-of-active fuel (TAF) and L-1 (46" band).
- X'; inhibit the ADS to avoid an uncontrolled depressurization and overfilling the RPV with cold water from low pressure injection(LPI) systems.
- C1; manual initiation of SLC (1 pump).
- X: emergency depressurization of the RPV to maintain suppression pool temperature below the Heat Capacity Temperature Limit/ (HCTL).
- V; initiation of LPCI (or RHR A/B) to maintain vessel level.
- V'; prevention of inadvertent overfill of the vessel that causes boron dilution.
- W; manual initiation of long-term heat removal with RHR.
- Y; initiation of containment venting after

successful shutdown with boron, given that other means of containment heat removal have failed.

In performing the quantitative assessment of these HIs, the concept of the **cue-response structure** as a means of placing each HI in a context of key plant parameters applicable to the operator action of concern was utilized[2]. This also assists in assessing dependencies between HEPs. For the analysis of the MSIV-ATWS the following reference points are defined:

- Suppression pool temperature is assumed to be just below 90°F at the start of the transient. This temperature limit is alarmed in the control room and the operators are very familiar with the practical implications of it. The operating practice at Perry is such that, prior to reaching a pool temperature of 90°F, the suppression pool cooling is started and is kept operating until the temperature is brought to its lower level. Hence, at the start of the postulated transient the plant is at the high point of a pool cooldown-heatup cycle. The operators may be anticipating a need for suppression pool cooling at the time of the transient.

- Suppression pool temperature of 110°F. This is a limit identified in the technical specifications for the plant. If this limit is reached then a plant shutdown is required. The PEIs also use this temperature limit as a reference point for initiation of boron injection if the reactor cannot be shutdown with control rods, and is also used as one of the cues for initiation of the power-level control contingency.

The accident sequence analysis file, AS-11, provides the following times measured from the initiating event of MSIV closure:

- Time to L1 without HPCS or MFP: 1.9 minutes.
- Time to HCTL (185°F):
  - -- No level control: 8.8 minutes
  - -- With level control: 15.1 minutes
- Time to reach containment pressure of 15 psig (cue for initiation of containment venting): 5.5 hours
- Time to pressurize containment to 50 psig (IPE containment pressure threshold limit): 9 hours

2.1.2 Dependencies

The human interactions (HIs) in the ATWS model are highly dependent, which complicates the HRA. The dependencies and their implications are discussed below:

The transient starts with the reactor at full power and the plant transient response will be very rapid requiring timely operator response. The event tree functions U1 and U3 are associated with the auto-start of HPCS and MFP, respectively, upon reaching L-2. The HIs in these functions relate to manual back-up to the auto-start failure and should be regarded as totally dependent as they are responses to the same step (RC/L-2) in the EOPs. However, opening the MFP feedwater control valves is also an important action and is associated strongly with the level-power control actions.

Subsequent operator actions depend on whether HPCS and MFP run or not. If there is high pressure injection initially, there will be more time available to deal with the transient. Hence, the dependence between U1 and U3 and subsequent event tree functions are accounted with different time-windows, as well as different definitions of functions. For example, if the MFP is not initiated early enough, the level will drop to below -30" and the level/power control procedure instruct the operator to go to emergency depressurization. However, if the level is restored before -30", which can only be achieved with the MFP, the reason for depressurization is to maintain reactor pressure below the HCTL.

The level/power control (operator action to lower level) and power control (operator action to initiate SLC injection) are coupled by a requirement to control reactor pressure below the Heat Capacity Temperature Limit (HCTL), and/or to prevent containment failure. Per the PEI, the instructions for both these actions are associated with an instruction to inhibit the ADS. The purpose of the ADS inhibit is to prevent uncontrolled depressurization and LP-injection (which causes reactor power to go up again). This operator action is a simple action with the cue afforded by a procedural step. There are at least two prompts to inhibit ADS: (i) the first is in the PEI directs the operators to inhibit and (ii) the second results from actuation of the ADS timer (set at 105 seconds) which is alarmed.

Depressurization is called for when the suppression pool temperature exceeds 122°F and RPV pressure is greater than 1050 psig. If the suppression pool temperature cannot be maintained below the HCTL, then the operators are directed to perform Emergency RPV Depressurization. The initiation of RHR has a minor impact on the initial pool heatup rates, but on its own cannot remove enough heat to prevent containment failure without the other actions. To a large extent it can be considered independent of the other actions, since in any case it is only asked in the event tree when successful shutdown is accomplished.

2.1.3 Quantification of Non-Response Probabilities

As discussed earlier, non-response probabilities are expanded into two contributions, P_c - probabilities are failure of initiation (or of recognition of the need to act), and P_e - failure in execution.

2.1.4 ATWS With Loss Of PCS

Function C

The functional fault tree for event C contains the event RPHICPERC-1:Q-2 which represents failure of the operator to attempt to manually scram the reactor on the occurrence of a scram signal. This is an automatic instinctive response, and an HEP of 10^{-4} is considered appropriate. Swain [1] suggests that failure of the cognitive part of the response can be essentially neglected, and, since the controls are very clearly identified, the likelihood of an error of commission is negligible.

Function U3

One human interaction in function U3, FWHICPEL-2-FDW, relates to manual backup to failure of auto-initiation on Level 2. However, in the case of an ATWS another event FWHICPEL-2-FDW-V, is included in the U3 functional fault tree, to account for the fact that the operators have to manually open the feedwater regulation valve to allow the MFP to inject given it has started on level 2. The procedural instruction is given in the level/power control procedure to maintain level while pressurized between -30" and 217" with feedwater and RCIC, (but HPCS is used only above TAF). SLC will be having little effect at this time. The time to reach TAF with no injection is 2.? minutes, and the time to reach -30" is 2.5 minutes. There is a 55 second delay after the MSIV closure before the operators can take control of FW. This HI will be analyzed later, since chronologically the cues to initiate SLC will be occurring at about the same time (pool temperature reaches 110° F in about 40 seconds), and this is regarded as the principal concern of the operators. In the ATWS tree, the manual back-up event FWHICPEL-2-FDW is subsumed in this event and removed from the U306 function.

The functions that follow success in function U_3:

Function LC

The functional fault tree for LC02 is purely an HEP-FWHICPEC5:3-2. It follows an initial success in restoring the MFP injection before the level reaches -30". The value for this HEP is highly dependent on that used for the failure to initiate SLC in function C1. Chronologically it should occur after initiation of SLC. The operators are trained to initiate SLC before the BIIT temperature of 110°F is reached. Lowering level is only initiated when the BIIT is reached, and the other conditions, (a relief valve open, and power level above 4%) have also occurred.

Success in C1 implies that the operators have correctly diagnosed an ATWS and therefore are in the correct path through the procedures. From the above, therefore, success in LC implies that the operators have correctly diagnosed the need to initiate boron injection, since both actions involve a common cue (110°F pool temperature). It is also important to note that, feedwater has been restored successfully. Thus, what failure implies here is failure to pay heed to the instruction "maintain level between -30" and the level to which it was lowered to control power", and results in water level being maintained too high.

The value used for LCO2 is propagated through the event tree into sequences S-10 through S-16 as a conditional probability of failure to control level, <u>given</u> that SLC has been successfully initiated.

We will use the decision trees of EPRI TR-100259 as a basis for estimating this number.

Since this number is contingent upon the successful recognition of the need to inject SLC, and this is likely to occur early, given successful identification of the need for level control (P_c success), there should be ample chance to recover from an initial step in execution. (i.e., $P_e < 10^{-3}$, and the HEP is dominated by P_c (see attached worksheet).

FWHICPEC5:3-2 $P_c = 10^{-2}$ (Mean value)

An error factor of 3 is assigned (see attachment A)

ADS Inhibit

With successful level control and successful identification of the need to initiate SLC, the operators will have had two separate written instructions to inhibit ADS. In addition, when level is lowered, the ADS alarm will come on at Level 1 acting as a further prompt.

If the level is not lowered, which can only occur if the MFP is initiated before level 1 is reached, there will be no need to inhibit ADS.

WORKSHEET FOR CALCULATION OF p_c

Scenario: ______________________________

HI: 6C

Cue(s): ______________________________

Duration of time window available for action (T_W): ______ Seconds.

Approximate start time for T_W: ______________

Procedure and step governing HI: ______________

A. Initial Estimate of p_c

p_c Failure Mechanism		Branch	HEP	Reduce T_W by
p_{ca}:	Availability of information	NA	neg.	— min.
p_{cb}:	Failure of attention	NA		N/A min.
p_{cc}:	Misread/miscommunicate data	NA		N/A min.
p_{cd}:	Information misleading	NA		N/A min.
p_{ce}:	Skip a step in procedure	e or f.		N/A min.
p_{cf}:	Misinterpret instruction	NA		N/A min.
p_{cg}:	Misinterpret decision logic	.006		N/A min.
p_{ch}:	Deliberate violation	NA		N/A min.

.002 or .004

Sum of p_{ca} through p_{ch} = Initial p_c .01

Total reduction in T_W = ______ min.

Effective T_W = ______ min.

Check here if recovery credit claimed on page 2: ____

Notes:

However, this action is dependent on the initiation of SLC and level control. If these are successful, the only way to miss this action is by missing the two steps in the procedures, and by missing the alarm. This is extremely low probability and the failure to manipulate the switch, i.e., P_e will dominate.

This value is taken from NUREG/CR-1278, assuming item 2 Table 20-12. It is certainly arguable that items 3 or 4 may be more appropriate, but item 2 is chosen as a conservative option.

To convert a median value into a mean value the following formula is used:

$$\frac{\underline{\text{mean}}}{\text{Median}} = \exp\left[\frac{1}{2}\left[\frac{\ln EF}{1.645}\right]^2\right]$$

Summary: HEP for event ADHICPC51-ADS-A is given a mean value = 3.8E-3 and an EF = 3

The same value is used on both the success and failure of level control as the same failure mechanism and conditions apply. Even if level is restored and restored high, it is judged that level would dip below level 1 for a brief period. However, it should be noted that the timer will stop running if the level exceeds level 1 again, therefore, use of this value on the failure to control level branch is probably conservative.

Initiation of SLC

On the successful level control branch, as mentioned earlier, it is assumed that the operators have successfully recognized the need to initiate SLC. In this case, the remaining failure mechanism is failure to initiate.

P_t is taken from Table 20-12 of NUREG/CR-1278, using item no. 3 as a typical value.

median = 10^{-3}
EF = 3.
mean = 1.25×10^{-3}

for event SLHICPEQ-6-SLC1

It should be noted that with successful level control there is ample time to recover from the execution error and hence this value is considered conservative.

On the branch for failure to control level, it is conservatively assumed that the probability of failure to initiate SLC is unity. An alternate approach would have been to estimate an HEP based on the time available before containment failure using the HCR/ORE model.

Function X - Initiation of Depressurization

It is assumed that depressurization is always needed i.e., HCTL is always reached. Given successful level control and initiation of SLC, there is no failure even if depressurization is not performed when SP temp exceeds HCTL. Therefore, it is difficult to define a time window. Thus, the HCR method is inapplicable. Instead the alternate EPRI TR-100259 method is used as shown on the attached worksheet.

With failure to lower level, the HCTL is reached earlier and thus the probability of failure is expected to be a little higher. Even though not part of the EPRI method, this HEP will be evaluated by doubling the number for the previous case. This is somewhat arbitrary, but done to maintain relative significance.

Therefore:

ADHICPEC5-ADS-FL = .007 EF = 5
ADHICPEC5-ADS-FX = .014 EF = 3

The P_e contribution is negligible, since given the decision has been made to depressurize, there is obvious feedback in whether it has been successfully performed, and the action is simple, and as discussed earlier, failure to depressurize is not a failure.

WORKSHEET FOR CALCULATION OF p_c

Scenario: ____

____

HI: Failure to depressurize on HCTL

Cue(s): HCTL

Duration of time window available for action (T_W): N/A Seconds.

Approximate start time for T_W: ____________________

Procedure and step governing HI: ____________________

A. Initial Estimate of p_c

p_c Failure Mechanism		Branch	HEP	Reduce T_W by		
p_{ca}:	Availability of information	a	neg.	— min.	neg.	
p_{cb}:	Failure of attention	~~a~~ e	____	N/A min.	.008	– because everything else has been seen to bring this point. Can [illegible] low with load.
p_{cc}:	Misread/miscommunicate data	a	____	N/A min.	neg	
p_{cd}:	Information misleading	a	____	N/A min.	neg.	
p_{ce}:	Skip a step in procedure	e/f	____	N/A min.	.006/.004	
p_{cf}:	Misinterpret instruction	a	____	N/A min.	neg.	
p_{cg}:	Misinterpret decision logic	k	____	N/A min.	neg.	
p_{ch}:	Deliberate violation	N/A	____	N/A min.		

Sum of p_{ca} through p_{ch} = Initial p_c ~~.02~~ .007.

Total reduction in T_W = ______ min.

Effective T_W = ______ min.

Check here if recovery credit claimed on page 2: ____

Notes:

Event V

The initiation of low pressure systems is necessary following successful inhibit of ADS, and successful blowdown. When the emergency blowdown procedure is used, the procedure says to terminate and prevent injection from LPCS and LPCI.

This event is clearly coupled with event X cognitively, and therefore, the probability is estimated on the basis of the P_e term. There is no time to recover from errors of commission or of omission in this scenario.

The probability should be no different from a non-ATWS case since the reactor is being shutdown. Indeed, in the case of successful LC and SLC in a timely manner, HCTL is reached at 0.25 hrs (MAAP 11_03_48) and therefore the reactor is close to being shutdown.

Given that the decision has been made to blowdown and it has been performed correctly, the primary failure mechanism is assumed to be an error of commission. Using Table 20-12 of Ref. 1, item 3, and treating initiating all three pumps (RHR A, RHR B, and LPCS) as being totally correlated, a median value of .001 is used.

Summary: HEP for HIHICPEC5-3:2-S has a mean value of 0.001 and an EF = 3

Event V' - RPV Overfill

This event is dependent on successful initiation of low pressure injection, and is really a control action. None of the models is really applicable. The operators have be alert to level changes and control level if it gets too high.

However, one possible mechanism is to miss the step in the procedure that says (terminate and prevent). Using the EPRI TR-100259 approach, Mechanism (e), end point (e) gives an HEP = 0.002

Summary: HEP for HIHICPEC5-5-CRIT has a mean value of 0.002, and an EF = 3

<u>Initiation of RHR and Venting</u>

For the upper branch, with successful initiation of SLC and level control, the time window for successful initiation of RHR and venting is long (several hours) and therefore, the HEP is small. This is assumed to be the same as for the transients with scram (see discussion under the loss of offsite power event tree).

For the lower branch (ie., LC failed), the time scale is shorter, although not significantly so, and the same HEPs are used.

Functions following Failure of Feedwater Branch, U_3.

Chronologically, the first action required (or even possible, taking into account the 55 sec. delay for FW restoration) is initiation of SLC. However, even with successful SLC it is not clear that core damage can be avoided without restoring the level above -30". Therefore, the two connected actions of restoring MFP and/or depressurizing and establishing LPI before reaching -30" are regarded as key.

In the worst case, (no HPCS and no FW), Level 1 is reached in 1.9 minutes. Therefore, inhibiting ADS, must be achieved before 1 min 55 secs + 105 secs from the beginning of the transient, i.e., 220 secs. In addition, MAAP run 11_03_47 shows that there are an extra 33 secs before the RPV pressure reaches the LPCS shut-off head.

<u>Inhibiting ADS</u> is almost at the level of a memorized action, but even it if is not, there are two separate occasions that are called for in the procedures RP/Q, and Level/Power control.

These should be reached at about 40 seconds into the accident. Treating this as a type CPI action [2], assume 60 secs for $T_{1/2}$, and assume 15 secs to get the keys for the ADS inhibit switch, then Tw = 253 - 55 = ~ 203 secs.

$$P_c = 1 - \Phi\left(\frac{\ln(205/60)}{.7}\right) = 1 - \Phi(1.755) = 0.036$$

Given the switches are well indicated, the failure to perform the action is 10^{-3}. (no chance credited for recovery). Thus the failure to inhibit ADS is assessed as .036. Since this event appears in the event tree after failure to restore feedwater, it is included as a conditional probability given by:

$$\text{ADH1CPC5-1-ADS-0} = \frac{.036}{\text{FWHICPEL-2-FDW-V}} = \frac{.036}{.005} = 7.2$$

where AWHICPEL-2-FDW-V is the failure to restore feedwater - see below.

<u>Failure to restore feedwater</u>, FWHICPEL-2-FDW-V, has a time window greater than 12 minutes. The time window is measured up to the onset of core damage. (MAAP run 11_03_49)

The first procedural instruction encountered will be in level/power control procedure, which should be picked up in about 1 minute (power is >4%, and SRV is open and pool temp reaches 110°F in about 40 sec).

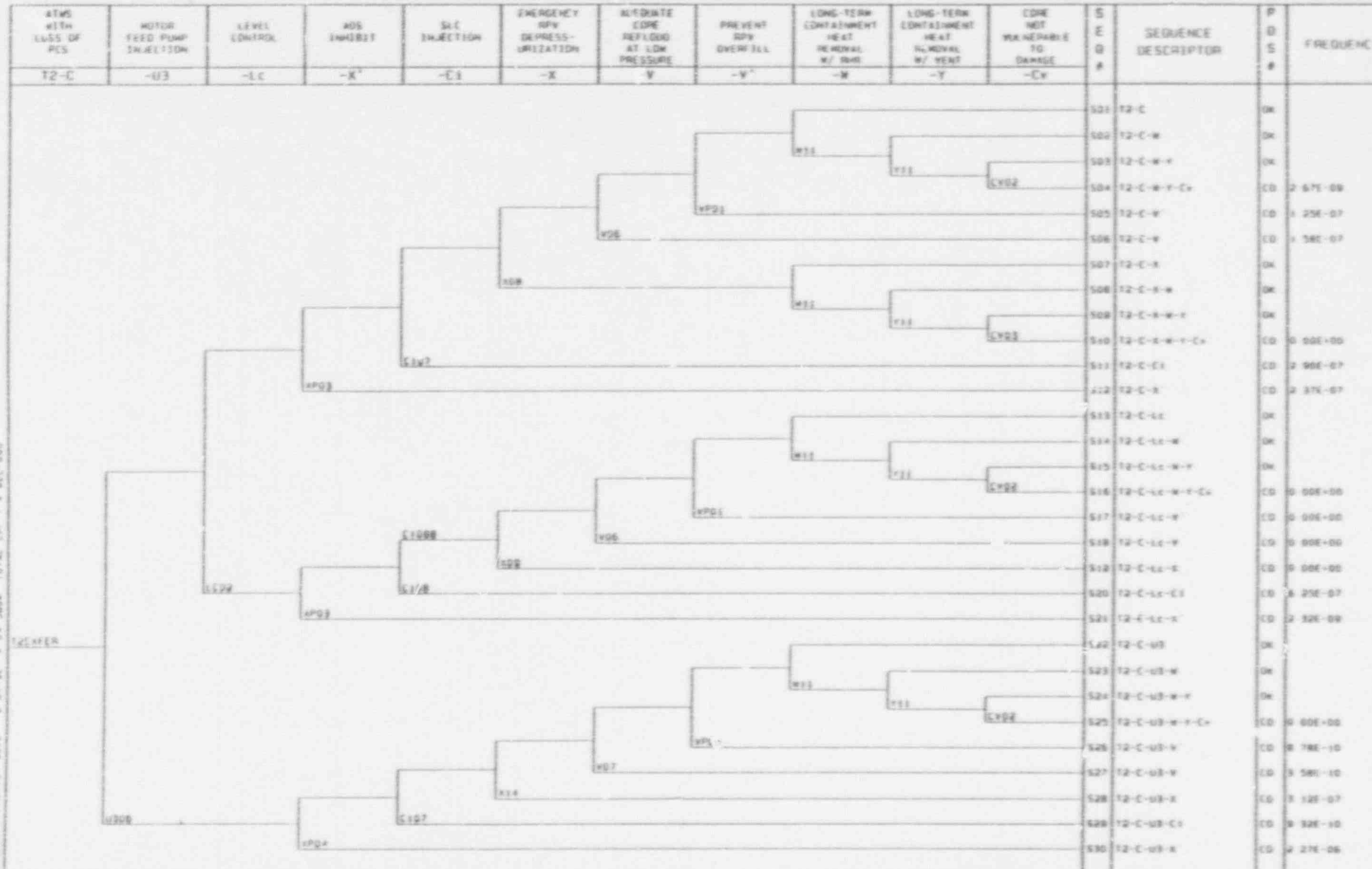
Restoring feedwater is a simple task - controlling MFP flow control valves - therefore the DDD contribution ought to be dominant. Given this is the case, and the actions to initiate FW and to blowdown are guided by the same symptom level and by the same part of the procedure, the cognitive parts of the HIs to initiate injection from MFP and to initiate blowdown are regarded as completely correlated.

If it is assumed that the operators take charge of FW as quickly as they can in accordance with procedures, then the median time is assumed to be about 2 minutes (the operator cannot succeed before 55 secs because of interlocks). With a time window of 12 mins, the HCR/ORE model gives a probability of failure for event FWHICPEC-2-FDW-V, of

$$1 - \Phi\left(\frac{\ln(12/2)}{.7}\right) \sim 0.005 \quad , \ EF = 5$$

Note that this same basic event is used in both the U3 FFT event and in the X14 FFT.

Given that SLC follows successful ADS inhibit on the tree, the same value for failure to initiate is used as before. The same HEPs are used in all the remaining functions.

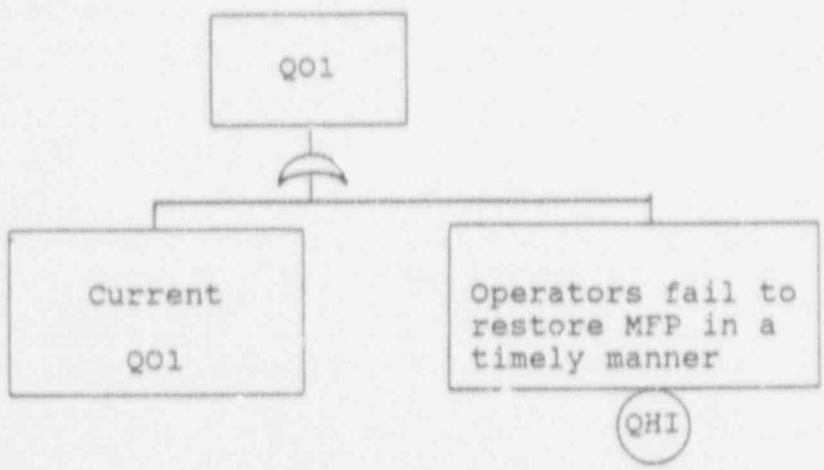


PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

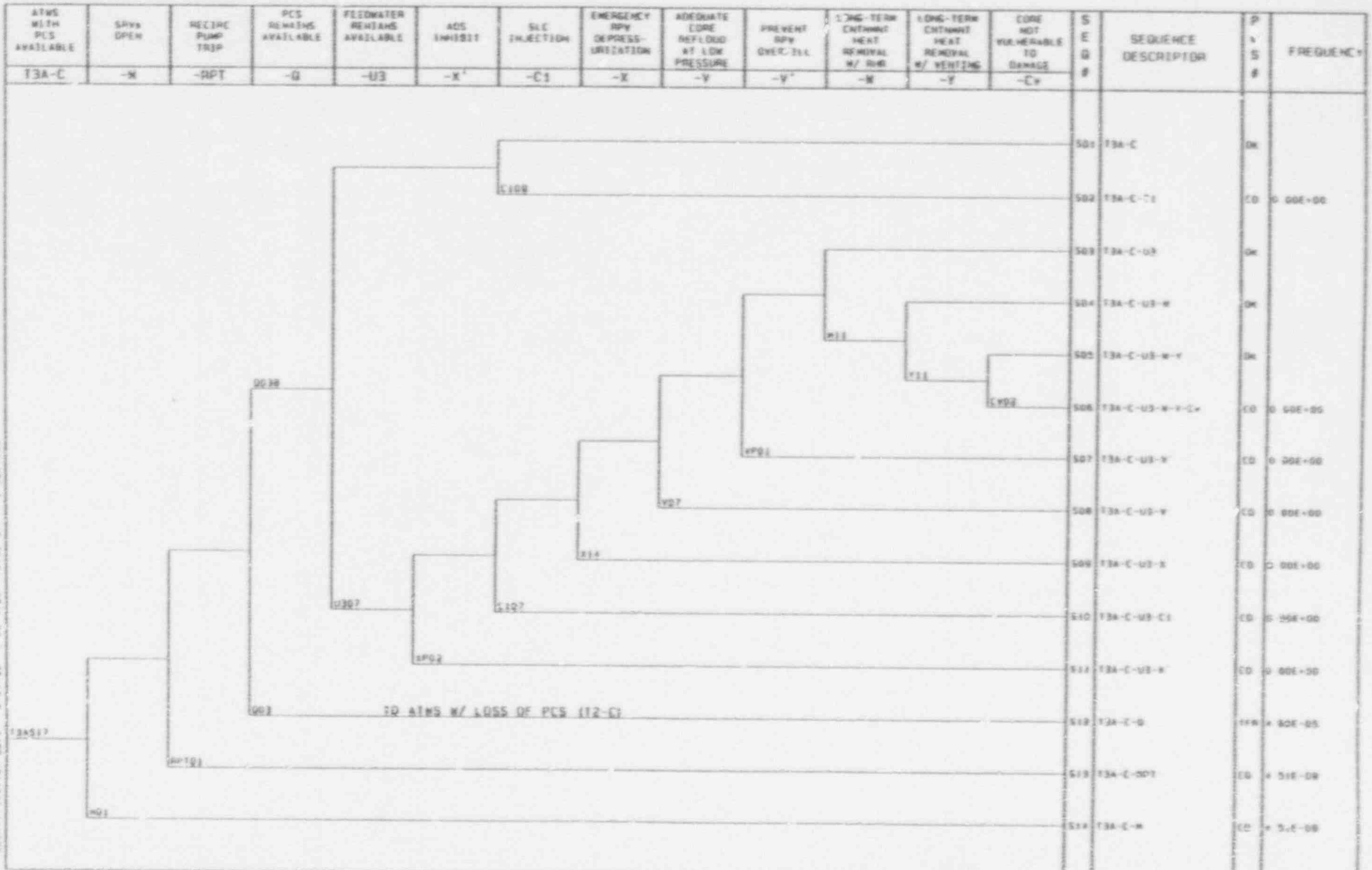
FIGURE 3.1.3-6
ATWS AND LOSS OF PCS EVENT TREE
REVISION 7 (04/02/92)

2.2 ATWS with PCS available (T3A-C):

In this case, the level will drop extremely quickly, to below level 1 in 1.9 minutes. Bypassing the MSIV isolation signal on level 1, although a simple action, does not appear to be high on the list of priorities of the operators. Therefore, the likelihood they will succeed is essentially zero. Therefore, HEP NSHICPEC5-L173 = 1.0. However, if the MFP is restored early enough, and level can be kept above level 1, the MSIVs will not close, so event Q in function Q01 could be modified as:

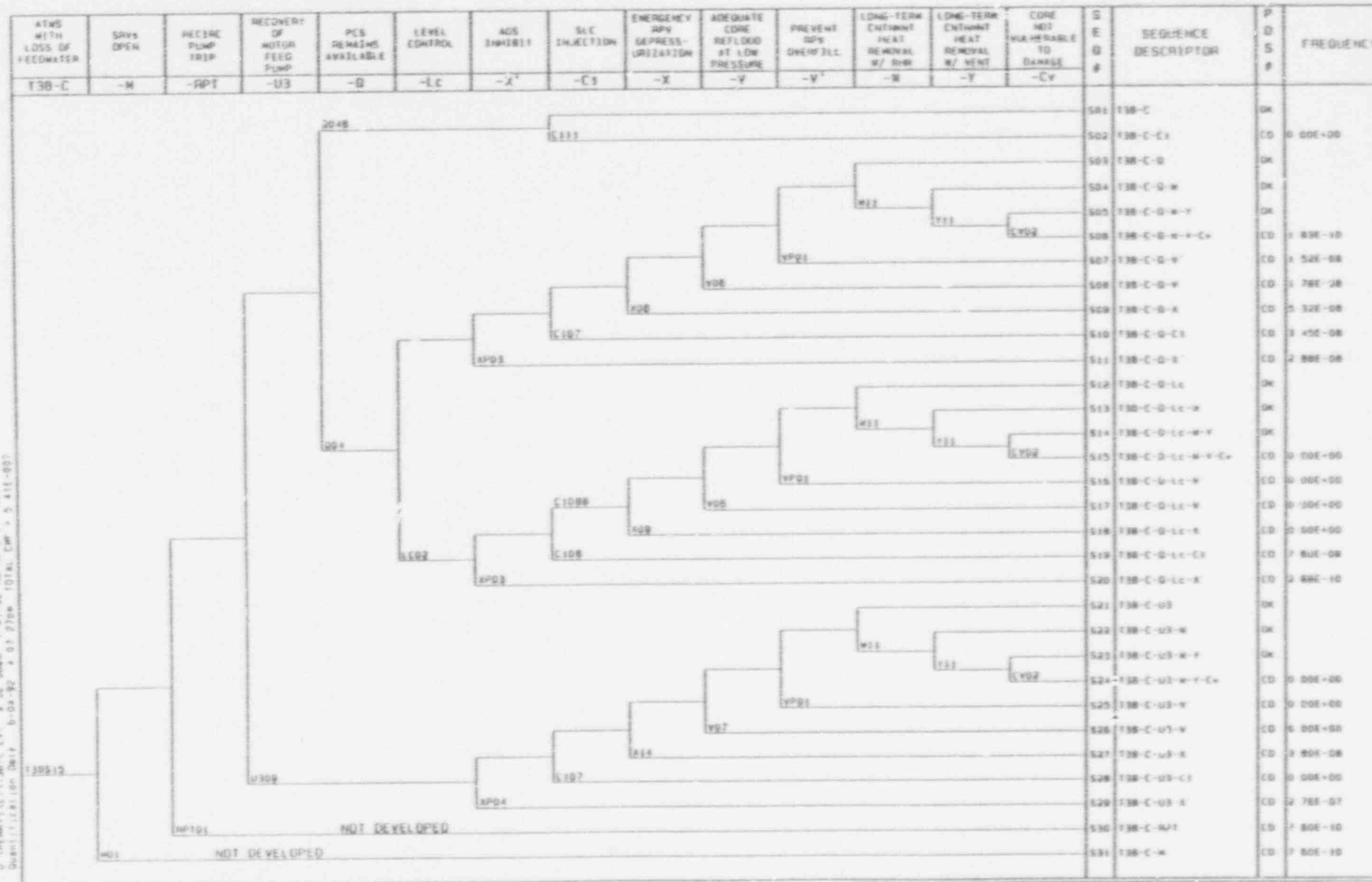


Since the time is so short, essentially no credit is given and Q01 is assumed to be 1.0. The tree then is developed exactly like the T2-C event tree, ie., MSIVs closed. If it could be shown that the SRVs would reclose with the level above Level 1, then it is possible that some chance of maintaining MSIVs open could be given.



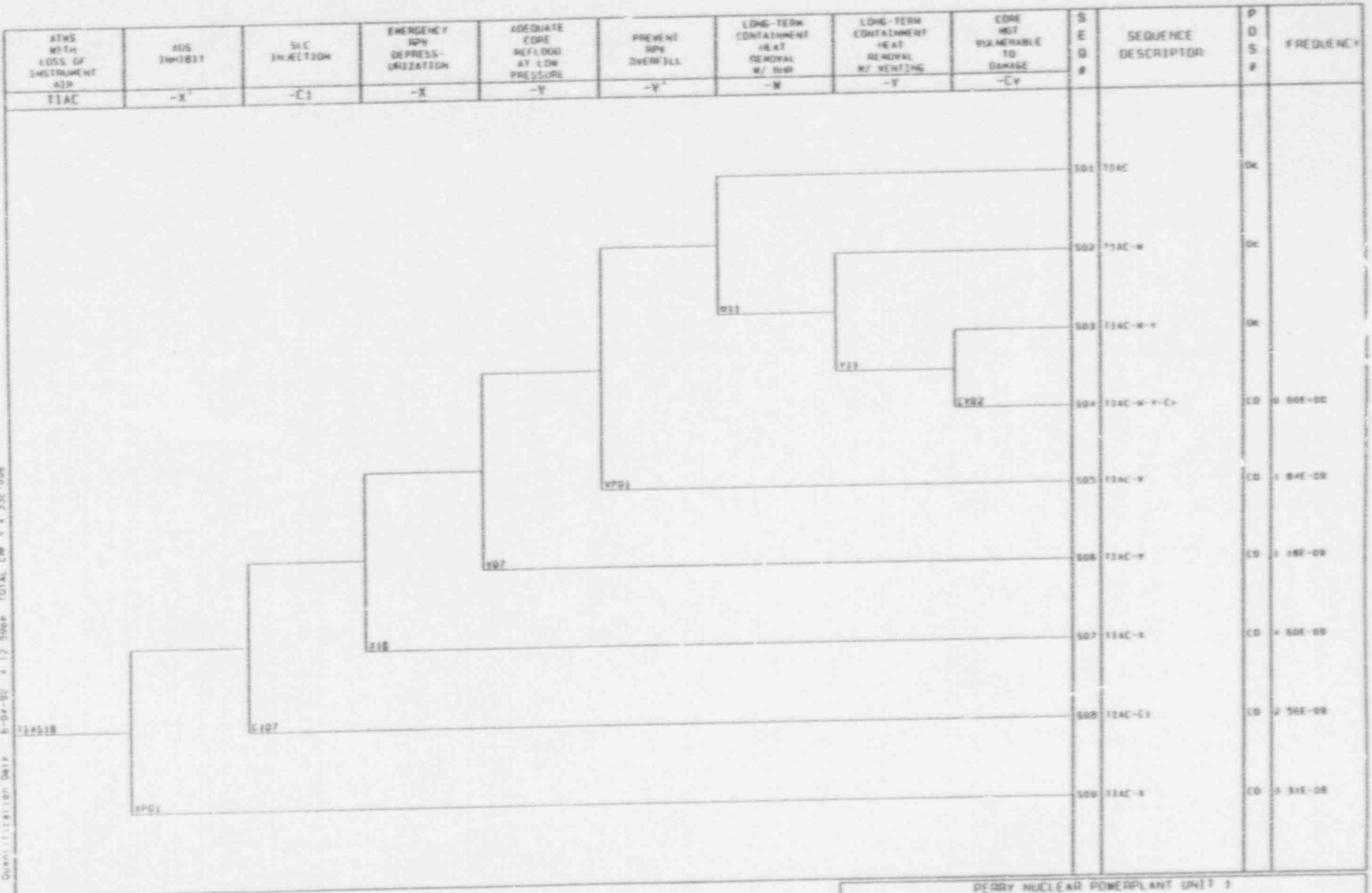
PERRY NUCLEAR POWERPLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.3-9
ATWS W/ PCS AVAILABLE
REVISION 1 (12/10/91)



PERRY NUCLEAR POWERPLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.3-10
ATWS W/ LOSS OF FEEDWATER
REVISION 4 (04/02/92)



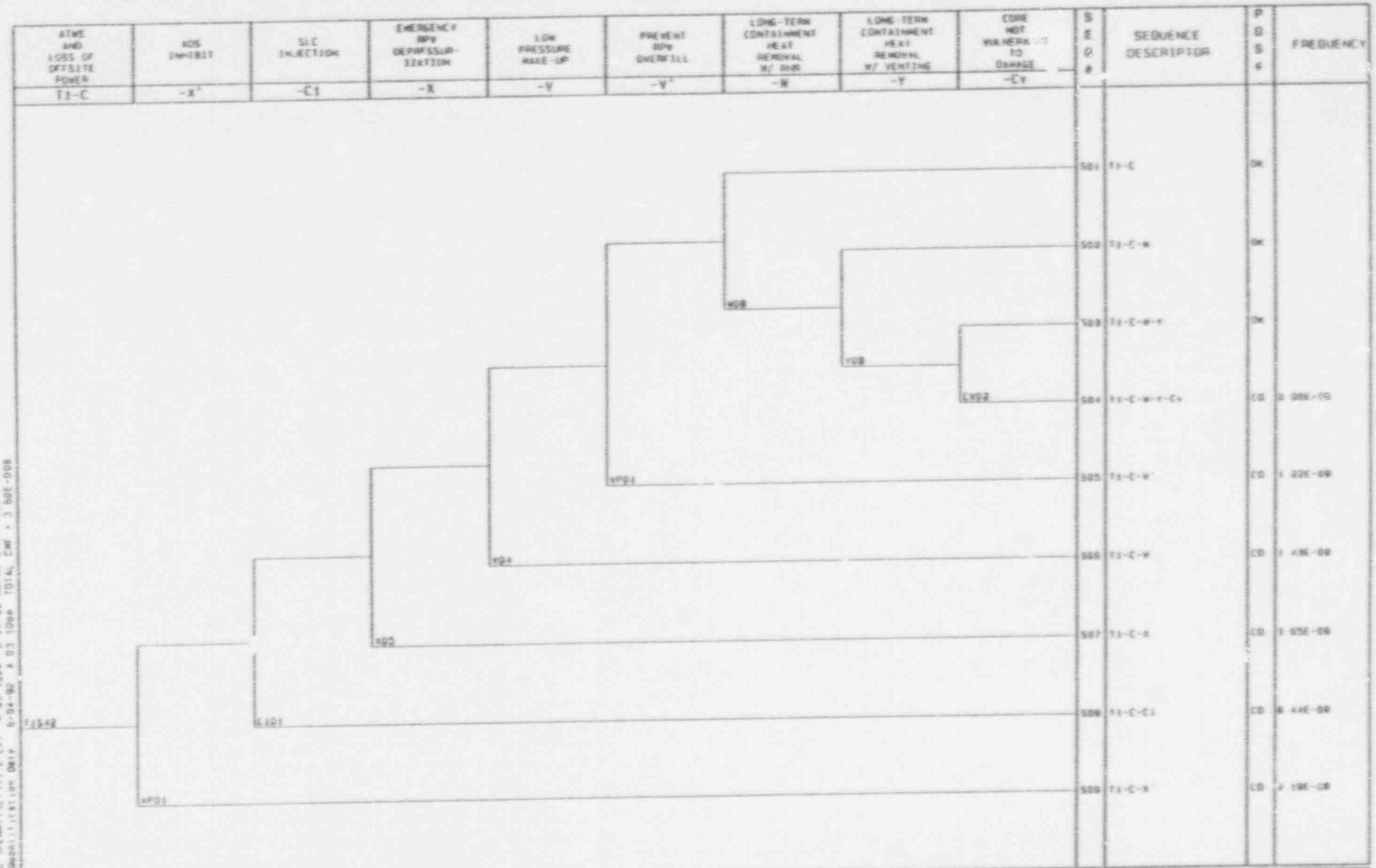
PERRY NUCLEAR POWERPLANT UNIT 1

INDIVIDUAL PLANT EXAMINATION

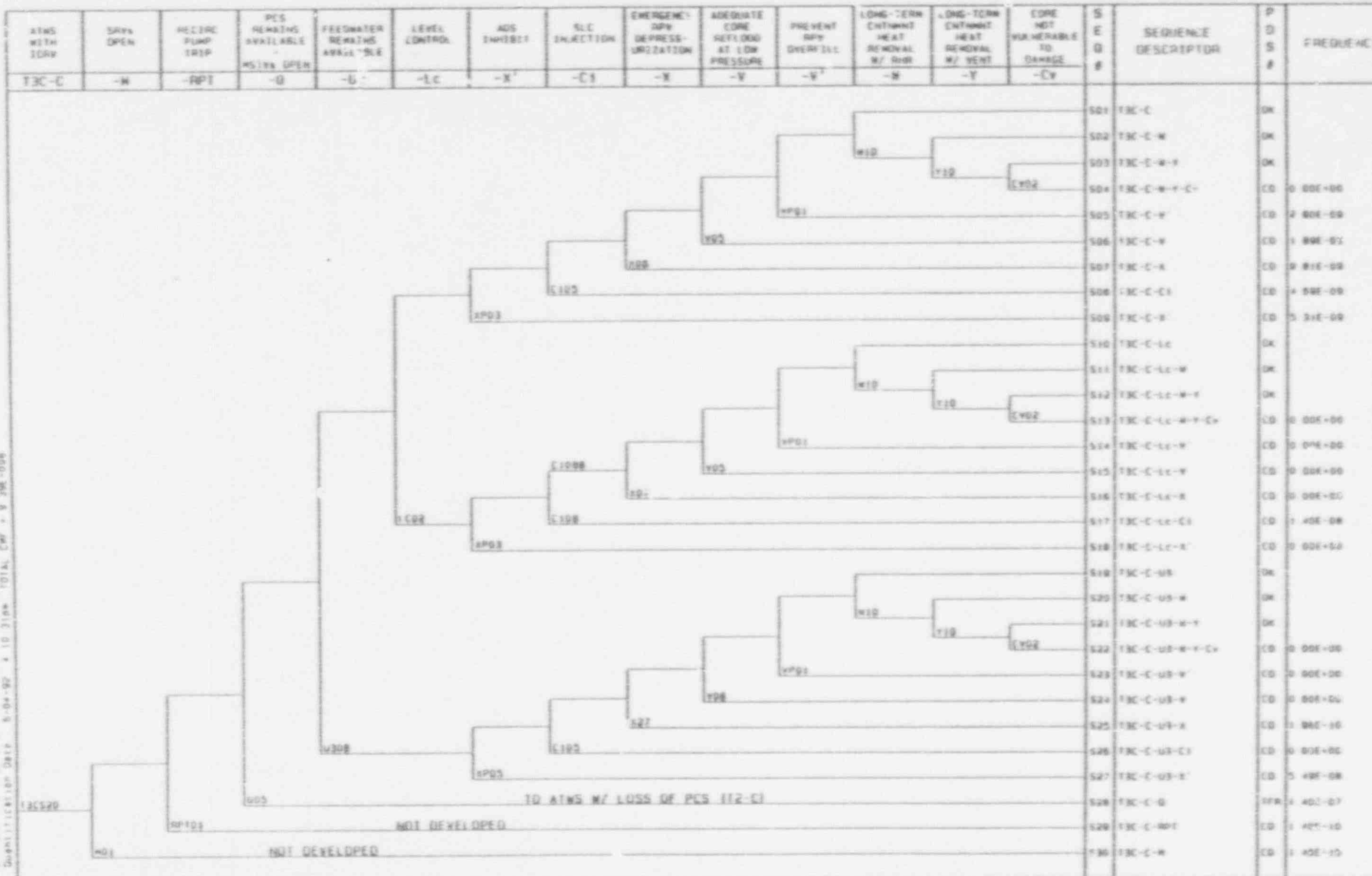
FIGURE 3.1.3-12

ATWS AND LOSS OF INSTRUMENT AIR

REVISION 2 (04/02/92)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.3-7
ATWS AND LOOP EVENT TREE
REVISION 3 (06/13/91)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.3-11
ATWS WITH IORV EVENT TREE
REVISION 7 (04/02/92)

2.3 Event Tree (T3B-C):

As for T3 A-C, function Q01 is 1.0 as above.

The other events may be analyzed as in the T2-C tree. However, in this case the function U3 = 1.0, and the value for ADS inhibit is the actual value calculated, not the ratio used previously, i.e., 0.036.

2.4 Event Tree (T1AC):

Model as for the T3 B-C event tree with U3 failed since feedwater is lost.

2.5 Event Tree (T1-C):

Model as for failure of U3 branch in T2-C, with ADS inhibit value being 0.036.

2.6 IORV ATWS (T3C-C):

The pool temperature reaches 110°F in about 20 minutes (analysis file AS-11). This is the cue to scram the reactor, and if it does not scram, to initiate boron injection, and since all these conditions (power >4%, pool temperature above 110°F, and an SRV open) exist, the operators will begin to lower level. This is done in a controlled manner, although the level/power procedure instructs the operators to terminate and prevent all injection. Since this is a controlled level reduction, the chances of restoring feedwater (motor-driven) at TAF are much greater.

The level will have to be dropped to TAF, since an SRV is open, otherwise if it reclosed, level may be maintained above level 1. Therefore bypassing the MSIV closure signal is of concern. This is addressed in the power/level procedure.

A screening value of 0.1 is considered adequate for maintaining SRVs open. This is event NSHICPEC5-2-4L1.

A screening value of 10^{-2} is proposed for establishing injection with the MFP, event FWHICPEL-2-FDW-L, since the transition can be performed in an orderly manner.

With the feedwater available, the other HEPs are as in the MSIV closure ATWS case (they are all conditional probabilities, not time dominated HEPs).

With MSIVs open, RHR is still required since heat is going to the pool through the SORV. However, failure to initiate RHR and venting could be lower. Using the values suggested for ATWS will be conservative, but won't contribute much to CDF.

With failure to establish feedwater, inhibit ADS still has to be achieved in the same time scale. Therefore, the same value as for T2-C with failure in U_3 is used. However, this is fed into the model as a conditional probability for event ADHICPC5-1-ADS-I equal to

$$\frac{\text{P(failure to inhibit)}}{\text{P(failure to maintain FW)}} = \frac{.036}{.01} \sim 3.6$$

3.0 PNPP LOSS OF OFFSITE POWER SEQUENCES

The purpose of this section is to document the human reliability analysis for postulated loss of offsite power sequences at PNPP. Again, the HI events are discussed in the order in which they appear in the event trees. The events are as follows:

Event Tree T1

Event C - Contains RPHICPERC-1:Q-2 - failure to manually scram the reactor. This is a skill-based action and is given an HEP of 10^{-4}.

Event U_1 - Contains HPHICPEL-1 - failure to initiate high pressure injection give auto-start failed. This is an instinctive type of annunciator response and is also covered by a step in the RPV- Level Control procedure. An HEP 1.25E-3 is used as a typical value.

Event U_2 - Contain[illegible] CPEL-1 - this is the same event as above but related to RCIC. It is included in the model with the Same identifier as above since the two actions are clearly correlated.

For both these actions there is a long time (24 mins from MAAP run 10_00_50) from Level 2 to TAF, the time within which the operators have to take note of th[illegible] failure to initiate.

Contains RCHICPSE51-LDTRIP - failure to recover from RCIC High Temperature Leak Detection trip during LOOP event. This trip is actuated after a 30 minute time delay. The alarm condition is clearly annunciated in the main control room. A value of 0.05 is considered conservative.

Event W_s - The functional fault tree contains the events CVHICPEPC-COM and CVHICPEPC-RHR-E, and the system fault tree for the suppression pool cooling function contains the HI basic event SCHICPSE12-5:3.

In function W_s, the success criterion is that suppression pool cooling should be started before the pool temperature gets to 185° (RCIC limitation), which occurs in 2.8 hours. In this model the first two events are associated with the cognitive part of the response, i.e., the product of the two events is the probability of not recognizing the need to initiate RHR (DDD phase). The reason for including two events is discussed later under event Y. The SCHICPSER-5:3 event is associated with the line-up of RHR for suppression pool cooling.

The failure probability in the DDD phase is estimated using the HCR/ORE approach. If a response time (T1/2) of 5 mins is assumed, since this is a type CP3 HI, with a value of 0.75 and an assumed 5 minutes required for lineup,

$$Pc = 1 - \Phi\left[\frac{\ln(151/5)}{.75}\right] \sim 1 - \Phi(4.5) \sim 10^{-4}$$

For reasons explained later this is partitioned between the CV events in the following way

CVHICPEPC-COM = 10^{-3}

CVHICPEPC-RHR-E = 10^{-1}

For suppression pool cooling only four manual actions (per train) that need to be performed in the control room (SOI-E12, p.9). Initiation of suppression pool cooling is an action that is performed relatively frequently and practiced in the simulator. In addition, there is ample time to correct for mistakes given that initial line-up has not succeeded, and suppression pool temperature is a key parameter that is attended to. Therefore, an HEP of 10^{-4} is considered adequate for this function.

This is based on the following argument. An error of omission of 10^{-3} per item and an error of commission of 10^{-4} per item is obtained per item 3 of Table 20-7, and item 4 of Table 20-12 (corrected by a factor of 2.7 per EF, all numbers rounded to nearest whole number) giving a base HEP of $5x10^{-3}$ per train. However, steps 9+11 require checking flow through the pump and act as a potent recovery mechanism. A recovery factor of 0.1 per step is applied and this reduces the HEP to $5x10^{-5}$. The value is conservatively taken as $1.0x10^{-4}$.

Function X, Emergency Depressurization

Even though, in this scenario, the operators may be depressurizing slowly by following the HCTL, the more demanding scenario that RCIC fails, and the reactor has to be depressurized to allow low pressure injection, will be assumed.

The cues are based on RPV level and are different from these associated with function Ws. Therefore, even if failure in function Ws is a result of operator inaction, this set of cues acts like a new set of stimuli and the event is treated as independent. This late into the accident, the time for boil-off is considerably longer than that available in the case of immediate loss of all injection. Therefore, the value used for transients with loss of high pressure injection is conservatively used in this case, and also for functions X22, and X23, which occur even later into the transient (see later). The value is conservatively taken as 1.00E-3.

Event V

Contains event LPHICPEL-1. This is the event representing failure to align low pressure injection given successful blowdown. The value of 1.25E-3 is used for event LPHICPEL-1. This is the same as used for HPHICPEL-1, even though this could be argued to be lower as it is a planned action, rather than a response to a sudden change in plant status.

Event V_8

Function tree Va12 contains the HI event FPHICPPS4:2RCIC3, which represents failure to align fire water after HPCS fails due to high MCC temp. The time window to the failure is on the order of 10 hours. At this time, the boil-off time ought to be on the order of at least 3-4 hours per (MAAP 10_01_01).

According to a walkdown performed by the operations department, it takes about 75 mins to perform the line-up, thus this allows at least a couple of attempts to line up.

It is judged that the execution probability will dominate the HEP, since the procedure clearly calls for alignment of RHR loop B containment flooding (PEI-SP1 section 4.2) if no more than one source of injection is lined up.

If it can be assumed that the operators would attempt to line up all the sources of LPI before lining up Firewater Alternate Injection, the time window could be much greater than 3-4 hours.

There about ten steps in the procedure, therefore conservatively assume a 0.1 value for a raw P_e. Since there is time for at least one more attempt to perform the alignment, we model the HI as $(.1)^2$ i.e., 0.01.

Also included in the function is event FPHICPPS4:2-DD-0, failure to maintain fuel oil for the diesel-driven fire pump. This is called out as step 9 in Section 4.3 of ONI-R10. Given that a loss of offsite power is obvious, the main failure mechanism will be missing the step in the procedure. Given that about 7.5 hours is available on 1/2 tank, the execution steps ought to be achievable. Assume a value of $3X10^{-2}$ since the procedure is somewhat long.

Function Tree Va07 contains the HI event FPHICPPS4:2RCIC1 - failure to align firewater given RCIC fails due to Emergency RPV Depressurization at the Heat Capacity Temperature Limit of 185°F. This failure occurs at about 3 hours, with a boil-off time of about 3 hours. We conservatively use an HEP = 0.3. However, this action is proceduralized and is emphasized in training which encourages the pre-lineup of an alternate injection system if no more than one injection system is available. On the other hand, the lineup is mimicked though not actually performed during simulator exercises. On balance, it is considered that this assessment may be somewhat conservative.

Function Tree Va04 contains the HI event FPHICPPS4:2RCIC2 - failure to align firewater given RHR fails due to MCC temp. This failure occurs at about 5 hours, with a boil-off time of about 2 hours. (See analysis file AS-10 under discussion for sequence B-U1). We conservatively use an HEP = 0.1, but take no credit for a second attempt as in the case of function Va12. Again, it is considered this assessment may be somewhat conservative.

Function Tree Va12 contains the HI event FPHICPPS4:2RCIC3 - failure to align firewater given HPCS fails due to MCC temperature. This failure is postulated at about 10 hours, with a boildown time of about 2.7 hours. Since the timewindow is extended from the beginning of the transient, an HEP of 0.01 is conservatively assumed.

Function Tree Va20 contains the HI event FPHICPPS4:2RCIC4 - failure to align fast fire protection alternate injection given a loss of all injection before the indicated minimum zero injection water level MZIWL is reached. The boildown time from Level 2 to the indicated MZIWL of 24 minutes will provide ample time for a fast connecting 3-valve lineup. Since the time is short an HEP of 0.1 is conservatively applied.

Function W - Containment Heat Removal with RHR

The functional fault tree has the same structure as for W_s discussed earlier. However, because of the longer time available, and the additional opportunities for recovery from an error in the DDD phase, afforded by a shift change, and input from the tech support center, the HEP in this phase is assessed to be lower than the 10^{-4} assessed for W_s. It is taken to be 10^{-5}. This is accommodated by giving event CVHICPEPC-RHR the value of 10^{-2}.

In addition, the containment spray mode of RHR has an event SCHICPSE12-5:3. This represents failure to execute the action of initiation, which is essentially to arm and depress the manual pushbutton (p. 9 of SOI-E12). The HEP is conservatively assessed as 10^{-4}, similar to RHR SP initiation. (Note this is manually much simpler, but the same HEP is used).

Function Y - Venting

The decision to vent is considered to be somewhat correlated with the decision to initiate RHR. This is a little conservative, since the cues are different, suppression pool cooling being cued off suppression pool temperature, and venting being cued off containment pressure. The HEPs are combined to give a total HEP of failure to initiate RHR and venting of 10^{-6}. Failure to decide to vent is assessed to have a moderately higher HEP than the decision to initiate RHR because of its implications for potential offsite releases. Therefore the independent value is assumed to be 10^{-4}. While the constraints that failure to initiate RHR is at 10^{-5}, the constraint imposed not to give an HEP lower than 10^{-6} was met by creating the event CVHIPEPC-COM with a value of 10^{-3}, and the event CVHIPEPC-RHR has a value of 10^{-2}, and CVHICPEPC-FPCC has a value of 10^{-1}.

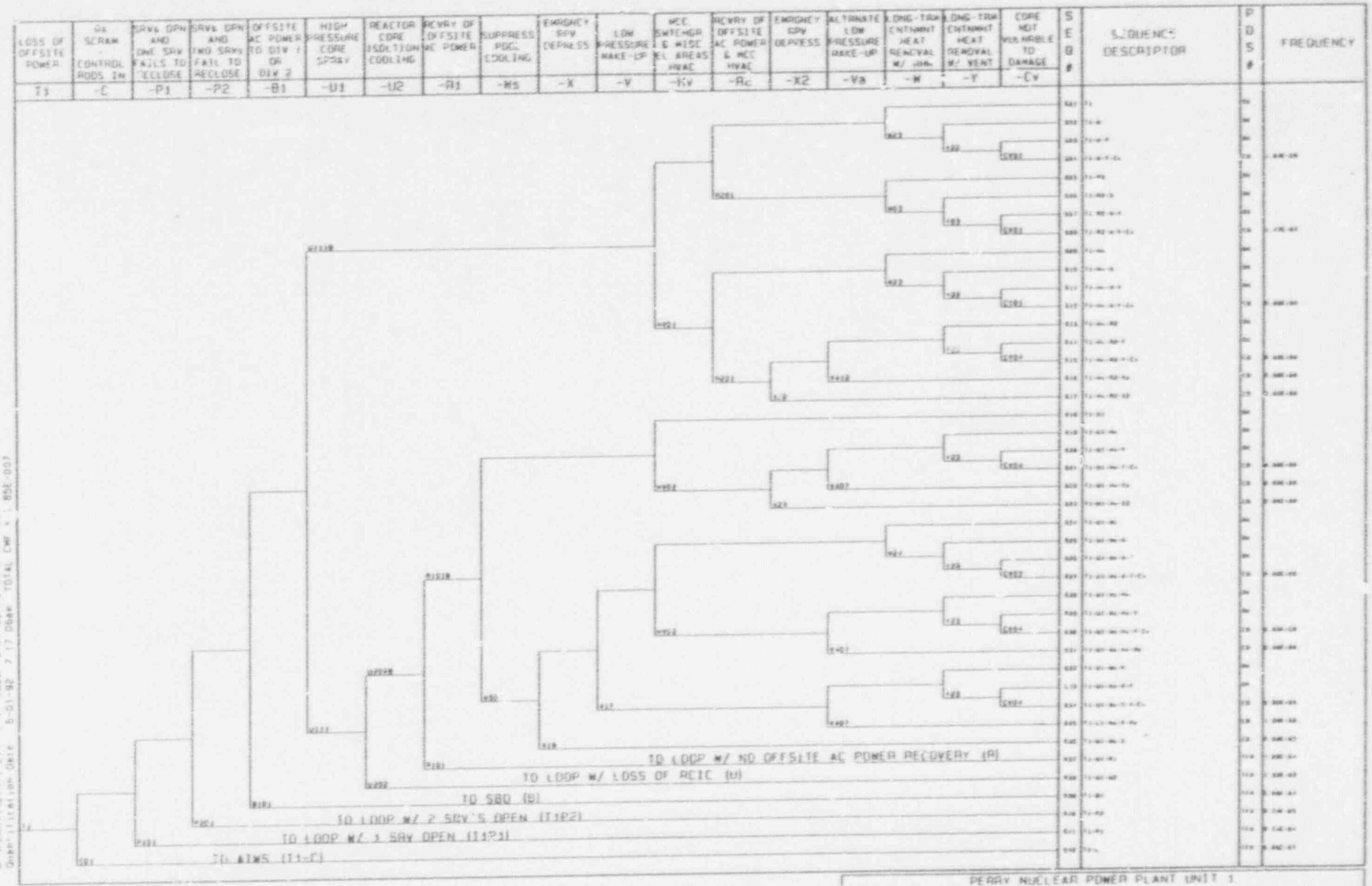
The system fault tree for venting also contains two events which represent the lineup of venting paths. These are called for in the procedures when the pressure reaches 15 psig. The events are CVHICPPS7:3G41-T representing failure to align the FPCC for vent, and CVHICPPS7:4E12-T, failure to align for RHR containment venting through the containment spray sparger.

The preparatory work for both paths is initiated when containment pressure is at 15 psig. Both involve opening a manual valve. For the FPCC lineup, in addition, a tank has to be drained (PEI-SPI, page 49). The time taken to achieve the lineup is on the order of 25 mins according to the operations staff. Since about 5 hours is available between 15 psig and the venting pressure (on the order of 35 psig) (MAAP run 05_01_01) there is ample time to do this. The containment spray initiation overrides are precautionary and failure to accomplish them would not defeat the venting function.

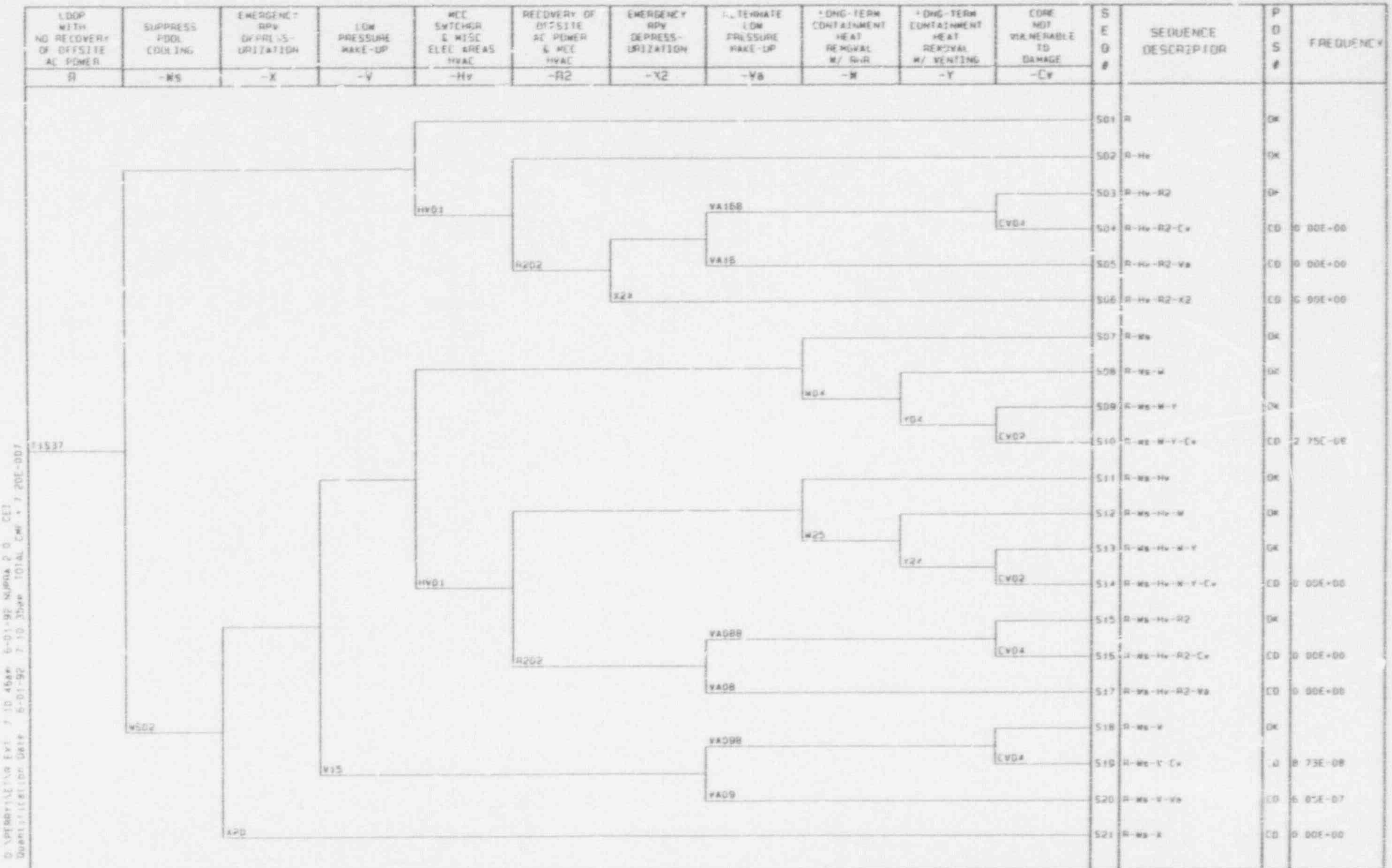
Venting through the FPCC is accomplished by opening two valves, (1G41-F145 and 1G41-F140). Since failure will not occur for some time after the initiation temperature and close attention will be paid to containment pressure, the HEP for execution is judged to be low and on the order of 10^{-4}.

The procedure for implementing the RHR vent path involves closing two valves, opening a manual valve (1 for each train), and contains precautionary steps to inhibit containment spray, and therefore is somewhat more complex than implementing the FPCC path. Nevertheless, given the slowly developing accident, the HEP of 10^{-4} is again used.

The system fault tree for Function Y also contains event CVHICRPS7:3G41-T which represents failure of a recovery action to open the outboard valve manually, given it has failed to open remotely. This is assessed as having an HEP of $5x10^{-2}$ on the basis of there being adequate time to identify the problem and carry out the recovery.

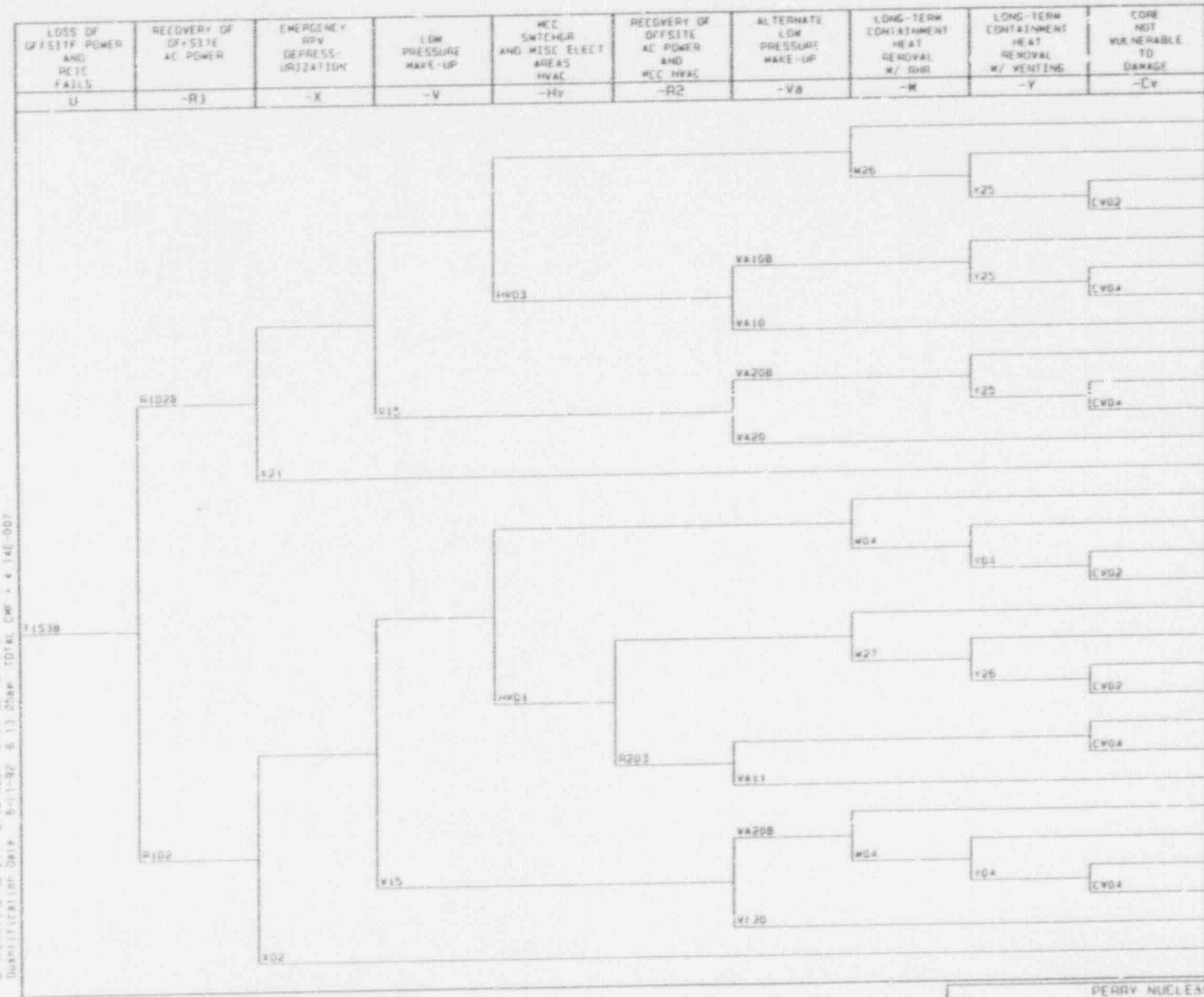


PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.3-1 SHEET 1 OF 6
LOSS OF OFFSITE POWER (T1) EVENT TREE
REVISION 3 (05/11/92)



PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.3-1 SHEET 2 OF 5
LOOP W/O RECOVERY OF OFFSITE AC POWER
REVISION 0 (12/16/91)

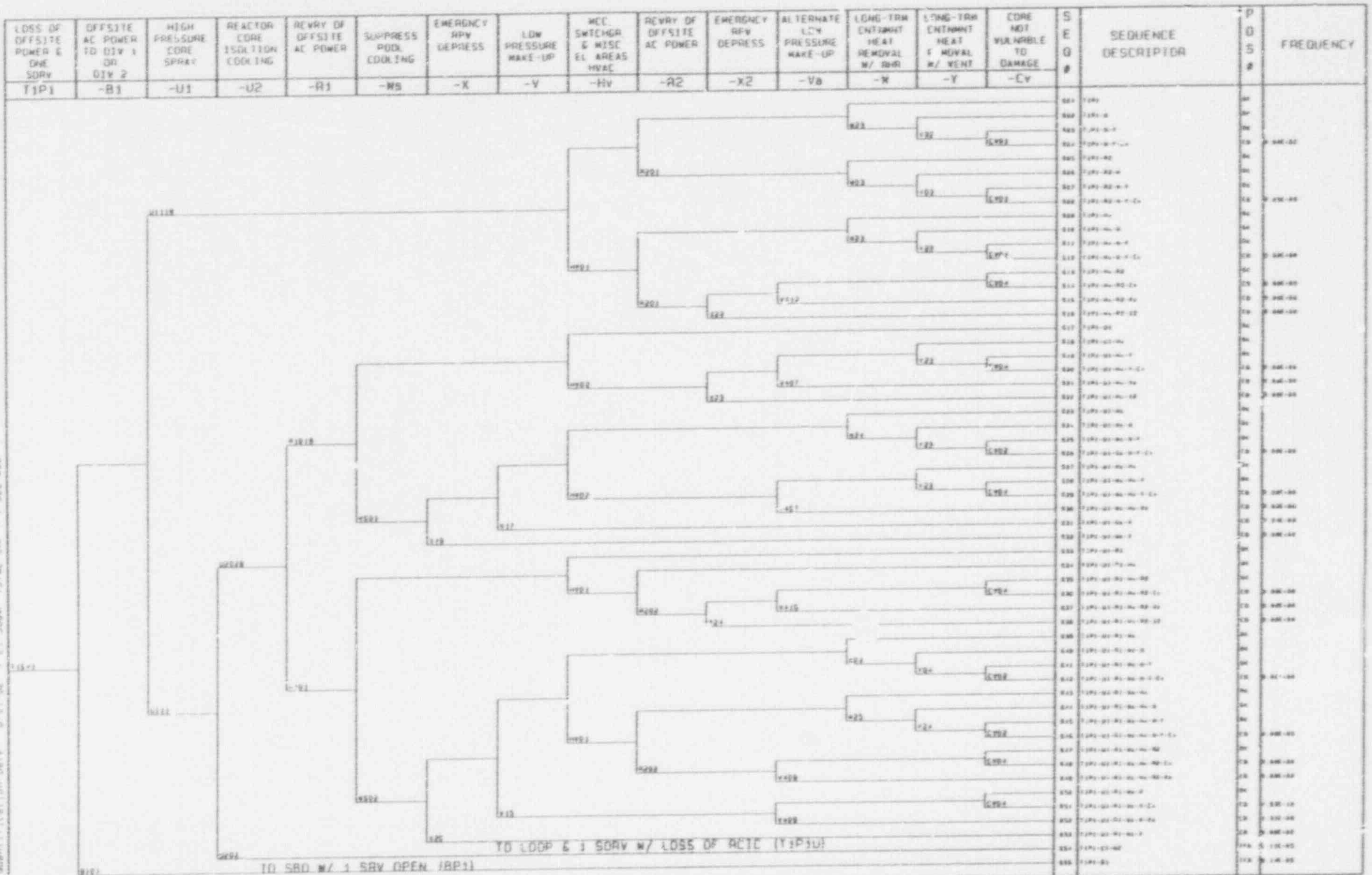


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
S01	U	OK	
S02	U-W	OK	
S03	U-W-Y	OK	
S04	U-W-Y-Cv	CD	0 00E+00
S05	U-Hv	OK	
S06	U-Hv-Y	OK	
S07	U-Hv-Y-Cv	CD	0 00E+00
S08	U-Hv-Va	CD	0 00E+00
S09	U-V	OK	
S10	U-V-Y	OK	
S11	U-V-Y-Cv	CD	0 00E+00
S12	U-V-Va	CD	5 99E-08
S13	U-X	CD	8 46E-10
S14	U-R1	OK	
S15	U-R1-W	OK	
S16	U-R1-W-Y	OK	
S17	U-R1-W-Y-Cv	CD	0 00E+00
S18	U-R1-Hv	OK	
S19	U-R1-Hv-W	OK	
[illegible]	U-R1-Hv-W-Y	OK	
S21	U-R1-Hv-W-Y-Cv	CD	0 00E+00
S22	U-R1-Hv-R2	OK	
S23	U-R1-Hv-R2-Cv	CD	0 00E+00
S24	U-R1-Hv-R2-Va	CD	0 00E+00
S25	U-R1-V	OK	
S26	U-R1-V-W	OK	
S27	U-R1-V-W-Y	OK	
S28	U-R1-V-W-Y-Cv	CD	1 63E-08
S29	U-R1-V-Va	CD	3 34E-07
S30	U-R1-X	CD	3 37E-09

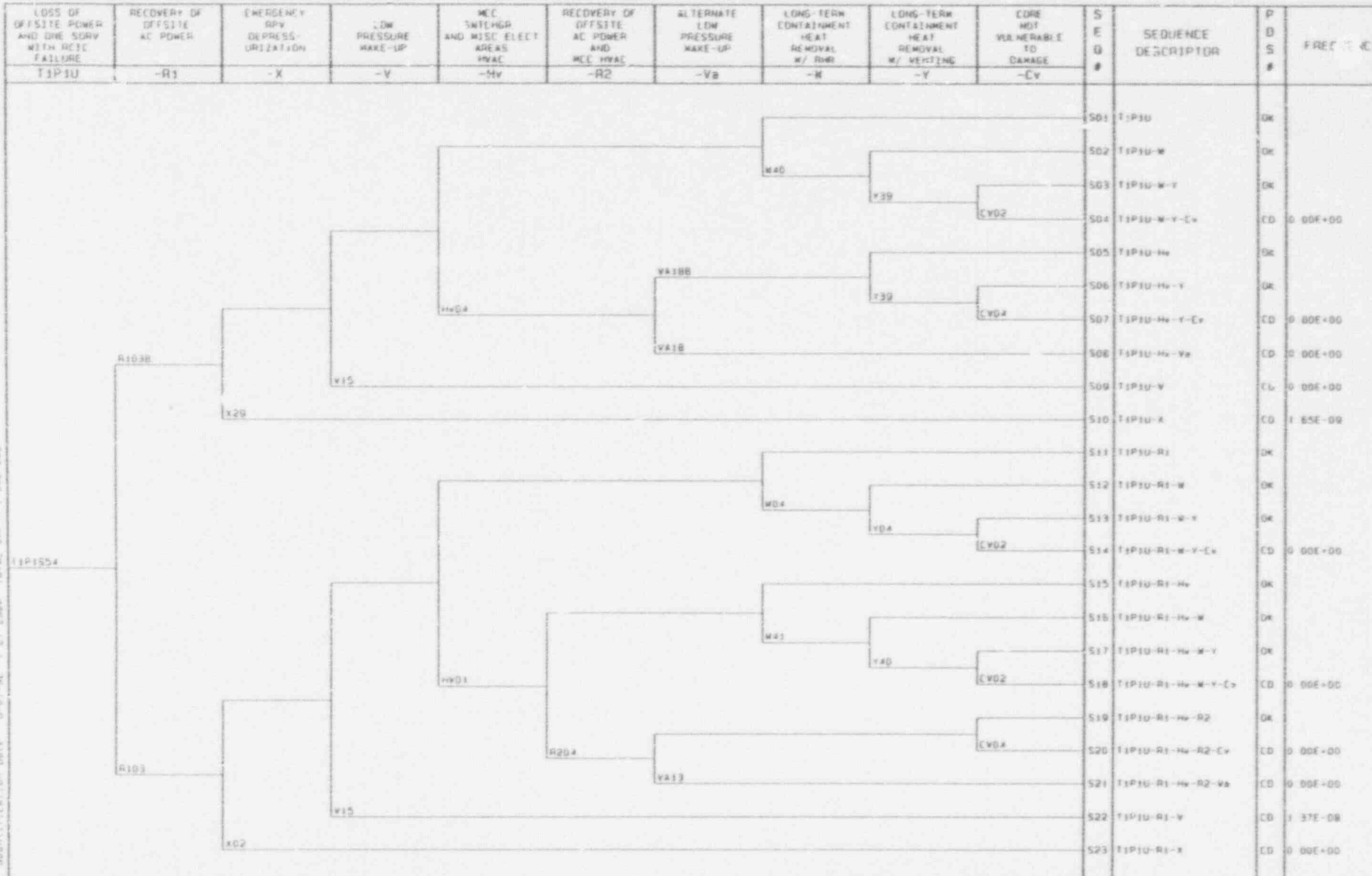
PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.3-1 SHEET 3 OF 6
LOOP W/ RCIC FAILURE EVENT TREE
REVISION 1 (05/14/91)

D:\PERRY1\ET\U.EVT 8 13 42am 6-01-92 NUPRA 2.0 CET
Quantification Date 6-01-92 8 13 05am TOTAL CMF = 4 14E-007

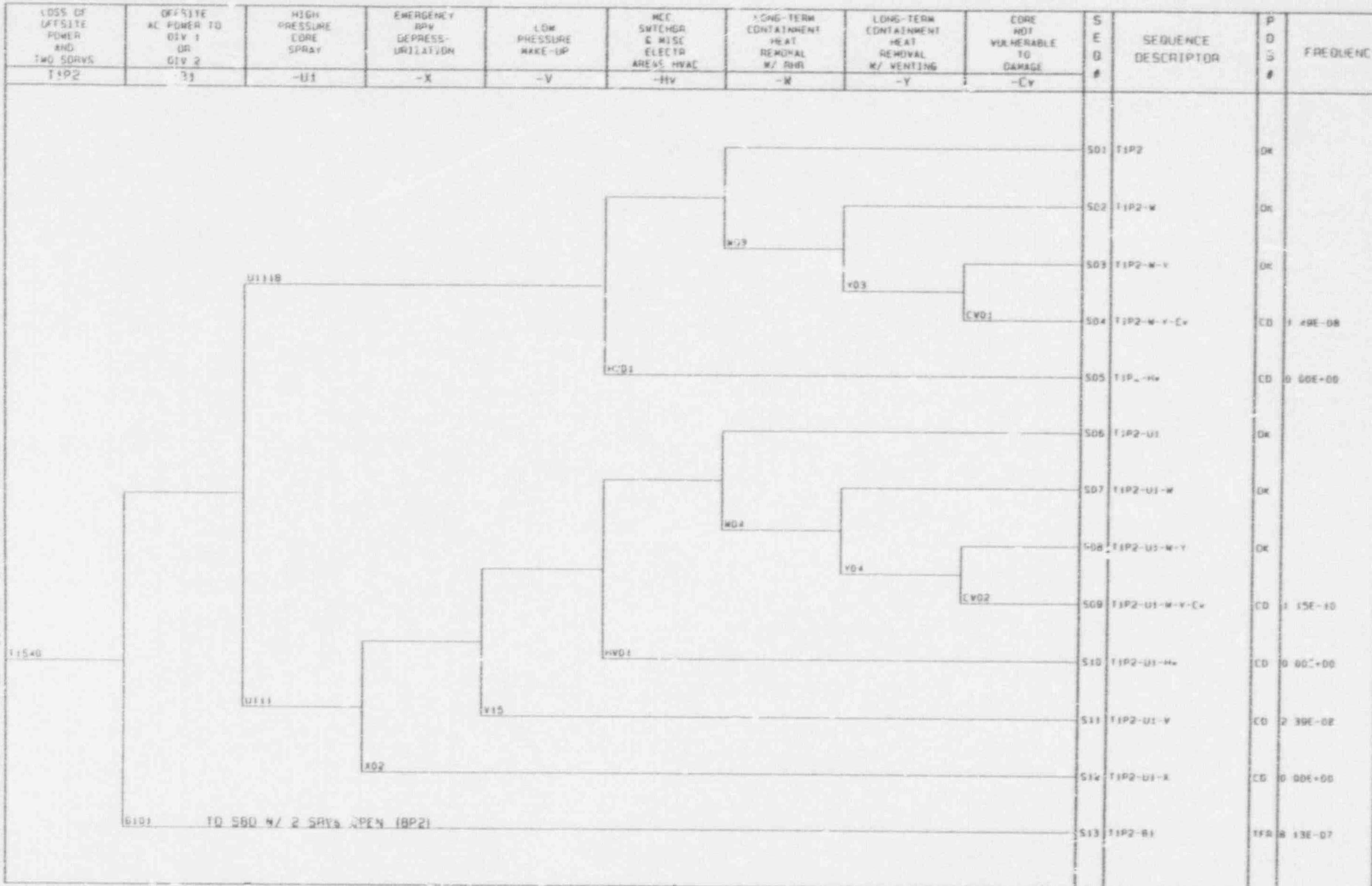


PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.3-1 SHEET 4 OF 6
LOOP AND ONE STUCK OPEN SRV EVENT TREE
REVISION 3 (12/06/91)



D:\PERRY\E\T1P1U.EV1 7.28.10am 6-01-92 NUPRA 2.0 (E)
Quantification Date 6-01-92 7.27.29am TOTAL CMF = 1.54E-008

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.3-1 SHEET 5 OF 6
LOOP W/ 1 SORV & RCIC FAILURE E.T.
REVISION 0 (11/20/91)



D:\PERRY\ET\T1P2.EVT 7:29 34am 6-01-92 NUPRA 2.0 (E)
Quantification Date 6-01-92 7:29 20am TOTAL CMF = 3.89E-008

PERRY NUCLEAR POWER PLANT UNIT 1
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.3-1 SHEET 6 OF 6
LOOP AND TWO STUCK OPEN SRVs
REVISION 2 (11/06/91)

Event Tree R

The HIs are the same as in event tree T1.

Event Tree U

The HIs are the same as in event tree T_1.

Event Trees T1P1, T1P1U, T1P2

The analysis file AS-09 does not differentiate between the success criteria for the HEP for these trees and the T1 tree.

4.0 PNPP STATION BLACKOUT SEQUENCES

The purpose of this section is to document the human reliability analysis for postulated station blackout scenarios at PNPP. The principal HI events are discussed as they occur in the scenarios.

Event Tree B

Event U1, High Pressure Core Spray

The event contains HI event HPHICPEL-1, representing failure to start HPCS given auto start fails on level 2. The time window is 28 minutes (given RCIC has also failed). This is an instinctive type of response, and the screening value of 1.25E-3 is used as in the case of loss of offsite power sequences.

Event HI - Operator Actions Taken to Extend HPCS Operation

This is conservatively modeled as an OR gate of two HIs. The first is HIHI CPOR10-4:3-B which represents failure to cross tie unit 1 and unit 2 batteries and perform load shedding. The action is directed by ONI-R10 (section 4.3). The time window to perform the cross-tie is 2 hours. A caution warns against cross-tying batteries after two hours. There is ample time to enter the procedure, therefore the P_c parameter should be small- use 10^{-3}.

The P_e parameter is a function of the complexity of the process for cross-tying the DC buses. A value of 10^{-2} is conservatively assumed as a screening value. The action, per train involves closing two breakers only, and should be simple. However, a relatively high value was assumed to reflect uncertainty in how much of the load shedding is effective and therefore how much margin for error there is. In fact, the shedding task has been enhanced by operator walkthrough and its effectiveness verified by engineering calculations.

The second event is HIHICPOR10-4:3-D, which represents failure to open the Division 3 switchgear room doors. The time window is 10 hours, and the applicable procedure is ONI-R10. The action is a much simpler one (i.e., open a door), thus assume a total HEP of $2x10^{-3}$. This event should be considered for a sensitivity analysis. Given success in HI, HPCS is assumed to be operable for 24 hours.

Event W - FFT WI5

Given restoration of power to division 1 or 2 before 13 hours, (success in R), RHR has to be initiated before containment pressure reaches 50 psig (14.4 hours). The alignment time for the RHR system is estimated to be .8 hrs. Therefore, in this case, the time window measured from power recovery is 1.4 hours. This is long enough to assume a low P_c ($\sim 10^{-4}$) given that the accident has been in progress for so long, and that concern for containment heat removal will be high.

Event Y - Function Y16

Since this is a long-term action, it is treated the same way as for the loss of offsite power or transient trees.

Function Y17

Event CVHICRPS7:3G41-T is a contributor to the top gate of this function. Since it relates to a valve called out explicitly in the vent procedure, and DC power is available to provide indication of position, and there is a long time to perform the recovery, a value of 0.05 is considered an appropriate HEP.

For sequences following failure of HPCS (U1), the analysis assumes failure of RCIC on depressurization as a result of high suppression pool temperature at 2.6 hours, with a 1.5 hour time window to initiate the diesel-driven fire pump before TAF is reached at 4.1 hours. Since the failure assumes depressurization there is no need to consider failure to depressurize, which would in any case prolong RCIC life.

For consistency with the evaluation of event Va in the T1 event tree, HI event FPHICPPS4:2RCIC1 will be assumed to have the same value, i.e., 0.3.

Other HEPs are the same as for success in HPCS.

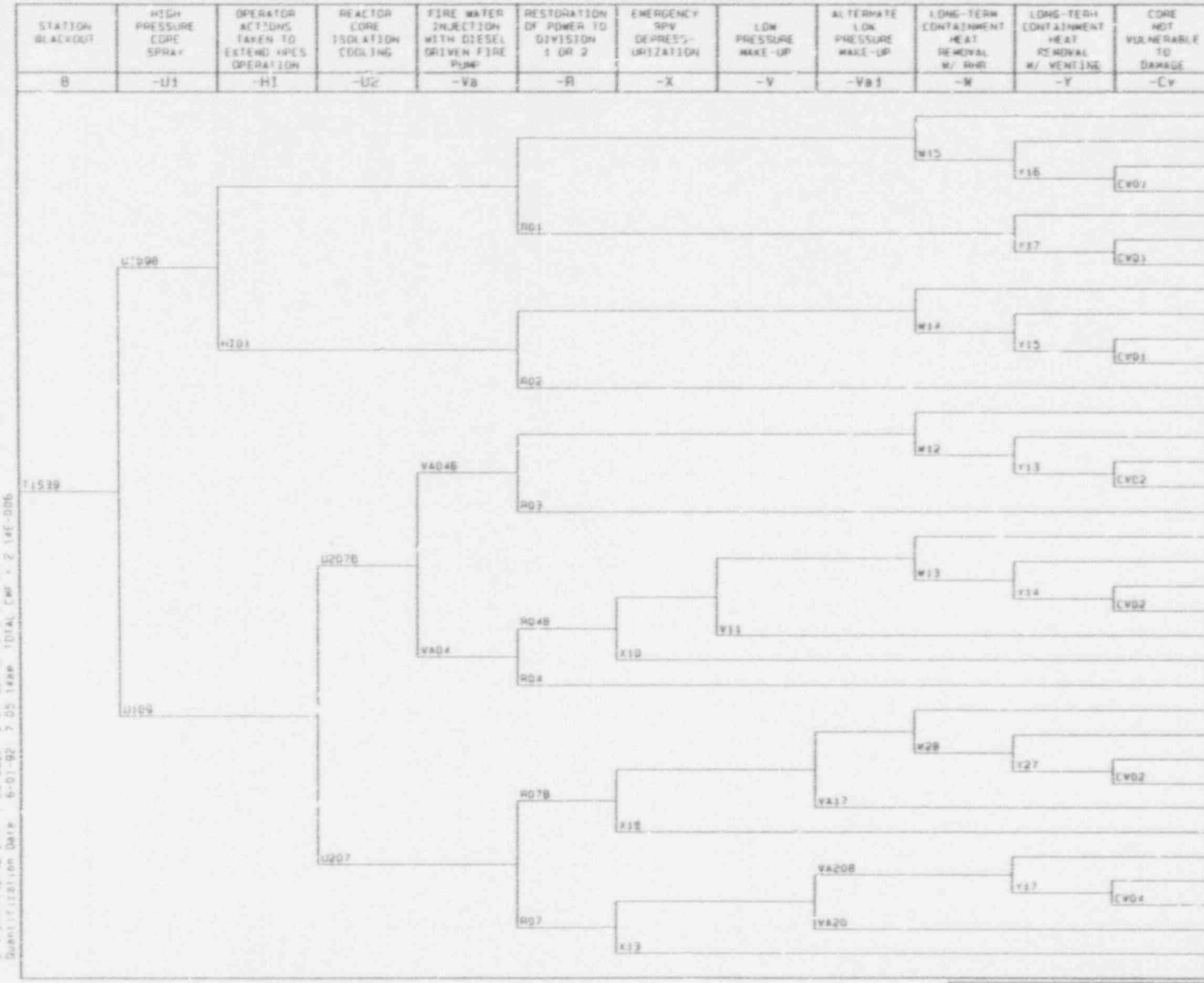
Event Tree BP1

The major difference in timing in this event tree is that, on failure of HPCS, but success of RCIC, the fire water system is required in a somewhat shorter time, about 3.8 hours, because RCIC will fail on depressurization when the suppression pool temperature of 185°F is reached at 2.3 hours. The likelihood of success in Va is similar for this situation as in the B event tree since the timing is almost the same.

All other HEPs are the same as for the B tree.

Event Tree BP2

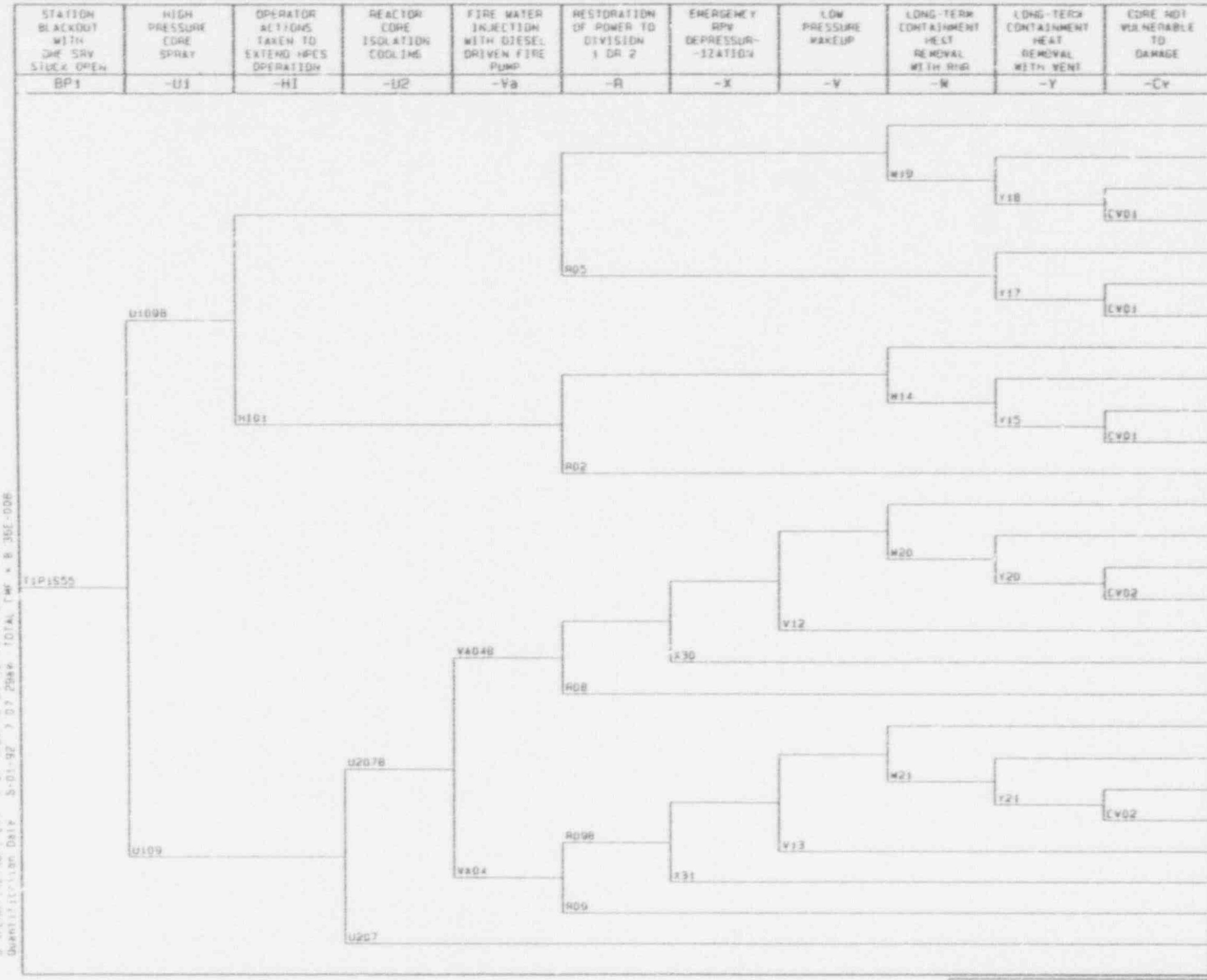
Use HEPs from B event tree.



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
S01	B	OK	
S02	B-W	OK	
S03	B-W-Y	OK	
S04	B-W-Y-Cv	CD	0 00E+00
S05	B-R	OK	
S06	B-R-Y	OK	
S07	B-R-Y-Cv	CL	1 80E-07
S08	B-HI	OK	
S09	B-HI-W	OK	
S10	B-HI-W-Y	OK	
S11	B-HI-W-Y-Cv	CD	0 00E+00
S12	B-HI-R	CA	1 13E-07
S13	B-U1	OK	
S14	B-U1-W	OK	
S15	B-U1-W-Y	OK	
S16	B-U1-W-Y-Cv	CD	0 00E+00
S17	B-U1-R	CD	2 36E-07
S18	B-U1-Va	OK	
S19	B-U1-Va-W	OK	
S20	B-U1-Va-W-Y	OK	
S21	B-U1-Va-W-Y-Cv	CD	0 00E+00
S22	B-U1-Va-V	CD	5 96E-0[illegible]
S23	B-U1-Va-X	CD	0 00E+00
S24	B-U1-Va-R	CG	7 7[illegible]-07
S25	B-U1-U2	OK	
S26	B-U1-U2-W	OK	
S27	B-U1-U2-W-Y	OK	
S28	B-U1-U2-W-Y-Cv	CD	0 00E+00
S29	B-U1-U2-Va1	CD	4 73E-08
S30	B-U1-U2-X	CD	1 81E-08
S31	B-U1-U2-R	OK	
S32	B-U1-U2-R-Y	OK	
S33	B-U1-U2-R-Y-Cv	CD	5 11E-08
S34	B-U1-U2-R-Va1	CD	5 28E-07
S35	B-U1-U2-R-X	CD	[illegible] 15E-08

PERRY NUCLEAR POWER PLANT
INDIVIDUAL PLANT EXAMINATION
FIGURE 3.1.3-2 SHEET 1 OF 3
STATION BLACKOUT
REVISION 8 (05/14/92)

D \PERRY\E\B EVT 7 05 38am 6-01-92 NUPRA 2.0 CEI
Quantification Date 6-01-92 7 05 14am TOTAL CMF = 2 14E-006



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
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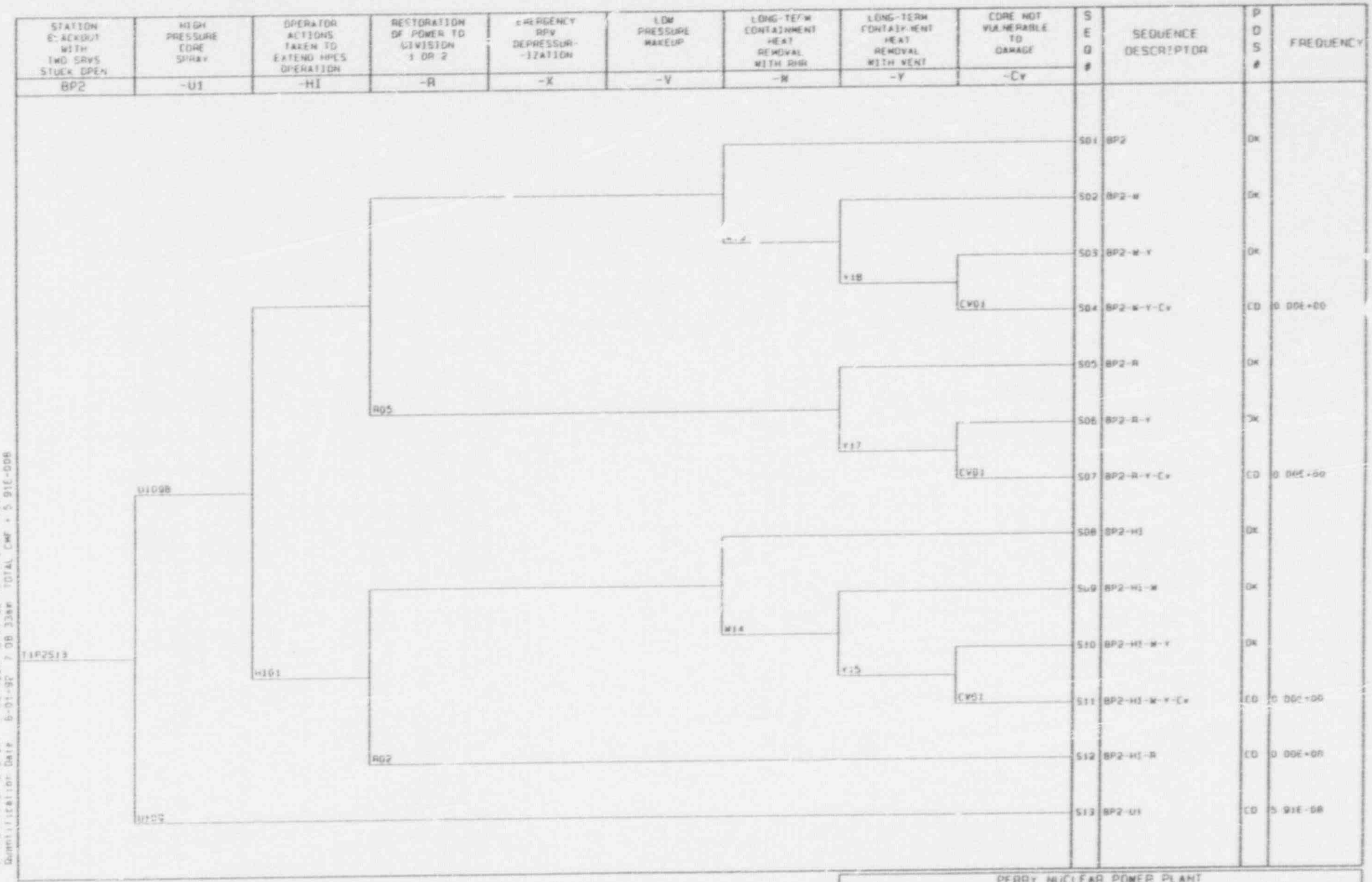
S02	BP1-W	OK	
S03	BP1-W-Y	OK	
S04	BP1-W-Y-Cv	CD	0.00E+00
S05	BP1-R	OK	
S06	BP1-R-Y	OK	
S07	BP1-R-Y-Cv	CD	2.02E-09
S08	BP1-HI	OK	
S09	BP1-HI-W	OK	
S10	BP1-HI-W-Y	OK	
S11	BP1-HI-W-Y-Cv	CD	0.00E+00
S12	BP1-HI-R	CD	7.30E-10
S13	BP1-U1	OK	
S14	BP1-U1-W	OK	
S15	BP1-U1-W-Y	OK	
S16	BP1-U1-W-Y-Cv	CD	0.00E+00
S17	BP1-U1-V	CD	1.69E-08
S18	BP1-U1-X	CD	0.00E+00
S19	BP1-U1-R	CD	3.34E-09
S20	BP1-U1-Va	OK	
S21	BP1-U1-Va-W	OK	
S22	BP1-U1-Va-W-Y	OK	
S23	BP1-U1-Va-W-Y-Cv	CD	0.00E+00
S24	BP1-U1-Va-V	CD	8.44E-10
S25	BP1-U1-Va-X	CD	0.00E+00
S26	BP1-U1-Va-R	CD	1.13E-08
S27	BP1-U1-U2	CD	4.85E-08

D:\PERRY\IE\BP1.EXT 7:07:44AM 6-01-92 NUPRA 2.0 CEI
Quantification Date 5-01-92 7:07:29AM TOTAL CMF = 8.35E-008

PERRY NUCLEAR POWER PLANT
INDIVIDUAL PLANT EXAMINATION

FIGURE 3.1.3-2 SHEET 2 OF 3
STATION BLACKOUT WITH ONE SORV
REVISION 5 (03/11/92)



PERRY NUCLEAR POWER PLANT
INDIVIDUAL PLANT EXAMINATION
FIGURE 3 1 3-2 SHEET 3 OF 3
STATION BLACKOUT WITH TWO SORVs
REVISION 5 (03/12/92)

D:\PERRY\ET\BP2 ET 7 08 42am 6-01-92 NUPRA 2 0 CEI
Quantification Date 6-01-92 7 08 33am TOTAL CMF = 5 91E-008

5.0 TRANSIENTS AND LOCAs

For most of the functions, the principal HIs are the same as for the loss of offsite power event tree. However, the analysis for event ADHICPEC2-ADS-T, the probability of failure to depressurize the reactor, is presented here.

It is assumed that, in accordance with the procedures, ADS is inhibited, and that it is necessary to initiate a blowdown.

The initial cue is RPV level reaching level 2, with HPCS and RCIC not operating. The final cue to initiate emergency blowdown is RPV level reaching TAF. There is some margin, as the level can drop to the MZIW level before core damage occurs. While the HCR/ORE methods could be used, the method is not particularly realistic for type CP2 actions with a long delay between the initial and secondary (or trigger) cue. Therefore, the decision tree approach will be used (see attached).

The action is a simple action, and there is ample indication to show whether it is being successful. Therefore, given that the correct decision has been made, the probability of incorrectly executing the blowdown is considered negligible, and the HEP is given by 1×10^{-3}.

WORKSHEET FOR CALCULATION OF p_c

Scenario: Transient, loss of high pressure injection

HI: ADHECPEC2-ADS-T

Cue(s): level at level 2 and decreasing, first cue at TAF

Duration of time window available for action (T_W): ______ Seconds.

Approximate start time for T_W: ____________________

Procedure and step governing HI: ____________________

A. Initial Estimate of p_c

p_c	Failure Mechanism	Branch	HEP	Reduce T_W by
p_ca:	Availability of information	NA	neg.	— min.
p_cb:	Failure of attention	(l)	.00075 ~~neg~~	N/A min.
p_cc:	Misread/miscommunicate data	(a)	neg	N/A min.
p_cd:	Information misleading	(a)	neg	N/A min.
p_ce:	Skip a step in procedure	(a)	.001	N/A min.
p_cf:	Misinterpret instruction	(a)	neg	N/A min.
p_cg:	Misinterpret decision logic	(k)	neg	N/A min.
p_ch:	Deliberate violation	(a)	neg	N/A min.

Sum of p_ca through p_ch = Initial p_c .00175

Total reduction in T_W = ______ min.

Effective T_W = ______ min.

Check here if recovery credit claimed on page 2: ✓

Notes:

WORKSHEET FOR CALCULATION OF p_c RECOVERY FACTORS

Scenario: ______________________________

HI: ______________________________

B. Recovery Factors Identified:

____
____
____
____
____
____
____
____
____
____

C. Recovery Factors Applied to p_c

p_c Failure Mechanism	Initial HEP	Recovery Factor	Multiply by:	Final Value	When Effective
p_{ca}					
p_{cb}	.00075	bell / alarm	.1	.000075	
p_{cc}					
p_{cd}					
p_{ce}	.001	—		.001	
p_{cf}					
p_{cg}					
p_{ch}					

Could apply recovery for STA, but for conservatism do n.

Sum of recovered p_{ca} through p_{ch} = Recovered p_c .001

Time at which all recovery factors effective ______

Given the minor differences in timing for TP1 and TP2 (reference MAAP runs 10_00_50, 10_00_51, 10_00_53) this value is equally good for all trees.

6.0 OTHER HI EVENTS

There are several other events included in the fault tree models, which represents failure of the operator to recover from specific failures. These human actions are generally governed by system operating procedures, or special plant instructions. Since they often appear in the fault tree models in an AND gate with hardware failures, they are generally not important contributors. Rather than perform a detailed analysis, the HEPs were assigned screening values judgmentally using the following guidelines. In addition, an attempt was made to characterize the relative likelihoods of the event failures by considering the scenario specific influences.

1. If an action is called out in an SPI or an SOI that is called for by the EOPs, then if the response time is limited with respect to the time taken to perform the required action an HEP of 0.1 is assumed. A higher value is used when the available time constraints are very tight.
2. If as above, but not time limited, then allow for recovery by assessing HEPs in the range 0.1 to 10^{-3}, particularly if these are alternate cues to lead to the required action, or if there are annunciators that direct attention.
3. If the actions are not called out specifically via the EOPs, a lower value than 0.1 is definitely not used.
4. For well practiced actions or those which are memorized responses, an HEP of 10^{-3} may be assumed.

6.1 OTHER TYPE CP HIs

Theses events are discussed in turn below:

ADHICPEC2-AD-LR Failure to recover from RPV depressurization (when core damage has occurred) in a plant damage state tree when depressurization has not yet been asked. The value of 0.0001 is considered appropriate since this is equal to the HEP for failure to Emergency RPV Depressurize during a Transient times a recovery factor of 0.1.

CAHICP1008-4:2 Failure to shift condenser air removal to vacuum pump and Auxiliary Boiler. This is routinely done as part of any shutdown in a controlled manner and is regarded as skill of the craft. It is also guided by the Integrated Operating Instruction. Therefore, the HEP is considered to be low and estimated as 10

CCHICPSP47-5:4 Failure to realign CCCW loop in 4.5 hours. There are many indications of loss of a CCCW pump train, and a long time to recover. Therefore, the HEP is considered to be low and estimated as 10^{-3}.

CDHICPPS2:1-XH1X Failure to bypass the RHR LOCA signal and reestablish power to stub buses XH11 and/or XH12. This is required to restore instrument air and nuclear closed cooling. This is a practiced response to a LOCA signal and the HEP is 10^{-1}.

CTHICPPS4:4-ALT Failure to align condensate transfer as an alternate injection system. This is essentially a simple task (PEI-SPI, section 4.4) directed by the EOPs. It does involve one ex-control room action at the RHR Flush Water Control panel (1H51-P275) on the 620′ elevation of the Aux. Building, and a verification step on the 599′

elevation at the RHR pump room. Because these might cause delays of several minutes, even though there are more than 20 minutes to complete the action, following the indication of no injection at Level 2, relative'. little credit is taken for this action in the short-term and an HEP of 10^{-1} is assumed.

CTHICPPS4:4-ALT6 Failure to align condensate transfer in 6 hours, following success in HPCS. Since the available time is now long, the boildown time is greater than 1 hour, and the action is proceduralized in the PEI-SPI, the value of 10^{-3} is assumed.

CTHICPPS4:4-ALTC This event is as above, but for the IORV transient. Since the boil-off time to TAF is shorter (~15 vs 20 min), the HEP is correspondingly hi_ner and is assumed to be $3x10^{-1}$.

DGHICP0011-2:5 Failure to initiate diesel generators. Attempting to start a system which should have started but didn't is an instinctive, learned response and in uniformally given a value of 1.25E-3. (see also HPHICPEC-1).

ECHICPSP42-4:2 Failure to close Emergency Closed Cooling to Control Complex Chiller A/B Bypass Valve following failure of the bypass valve to close during auto start. ECC is required for the RHR Room Cooler support of low pressure pump operation. The time window is considered to be several hours. A screening value of 0.05 is assumed.

ECHICPSP42-4PMP Failure to initiate Emergency Closed Cooling pump following auto start failure. It is considered that this failure to start would be identified with the normal control room verification of automatic actions. A value of 0.05 is considered conservative.

ERHICPPS4:2-ESW

Failure to establish ESW into the RPV for containment flooding per PEI-SPI, section 4.2. Although there may be a relatively long time available, the valve line-up is difficult so little credit is taken, and an HEP of 10^{-1} is assumed.

FPHICPPS4:2FP-LE
FPHICPPS4:2FP-IL

Failure to align the injection path for alternate injection, (given core damage has occurred), in accordance with PEI-SPI, section 4.2, in two time frames, less then 3 hours, and greater then 3 hours. The HEPs are given gradually lower values with increasing time frame of 5E-2 and 5E-3, respectively.

FWHICPSN27-4:1IA

Failure to control feedwater and condensate during a loss of instrument air. This is a skill of the craft action, not explicitly addressed in the procedures. An HEP of 10^{-1} is assumed.

FWHICPSN27-4:1IL

As for FWHICPSN27-4:1IA, failure to control reactor feed pump following a loss of instrument air, but later in the sequence. Since boil-off time is much larger, a lower HEP of 5×10^{-3} is used.

HIHICPOR10-4:0-I

Failure to close Fuel Pool Cooling & Cleanup Outboard Isolation valve 1G41 0F145, Containment Pools Return Outboard Isolation before RPV Failure during a Station blackout event. RPV failure occurs at about 1.8 hours with loss of all injection. The symptomatic Off-Normal Instruction ONI-R10, Loss of All AC, directs that this isolation by manually closed within 1.5 hours, and monitors the completion of this task within the defined time window with Attachment 6, Steps That Have Time Requirements. The HEP of 0.05 is considered conservative for the timewindow and the enhance ONI.

HIHICPOR10-XTIE

Failure to cross-tie division 3 to the division 2 MCC. The power transfer task with hardwired switchgear would open the closed inboard containment isolation to enhance containment venting reliability. Since the time to pressurize containment is on the order of 12 hours, a value of 0.005 is conservatively applied.

HPHICPSE22-5:0
HPHICPSE22-5:2

Failure to control HPCS min flow valve, given failure to auto control, and failure to transfer to suppression pool given failure to auto switchover. They are called out via verification steps in the system operating instruction and well practiced. Therefore an HEP of $5x10^{-2}$ is used for both.

IAHICPSP51-4:2

Failure to reposition NCC lube oil cooler outlet valve when an instrument air compressor auto starts. This is a well practiced task. A screening HEP of $5x10^{-2}$ is assumed.

IGHICPEH-1-H2IG

Operator fails to initiate Hydrogen Ignition System is modeled in the Level 2 Accident Progression Event Tree. The most limiting time window for hydrogen generation to commence is for a loss of all injection accident where the maximum core clad temperature reaches 2200°F after 51 minutes. The crew is trained to routinely monitor and control hydrogen during an accident in the Plant Emergency Instruction flowchart by placing the hydrogen analyzers on the third step of entering RPV Control. During a loss of all injection the reactor water level decreases below L1 (<16.5 inches above the TAF) at about 29 minutes. The RPV water level decrease below L1 is the entry condition to Hydrogen Control. The Unit Supervisor will typically direct action from the Hydrogen Control flow chart that the Hydrogen Ignitors be place on about 1 minute after entering the flow chart. The execution time to initiate this system at a nearby back panel is relatively short requiring about 30 seconds. Thus, the value of 0.005 is considered conservative.

LCHICPSE12-5:1
LPHICPSE21-5:1

Failure to control RHR and LPCS min flow values, given failure to auto control. They are called out in verification steps in the respective system operating instruction and well practiced. Therefore, an HEP of $5x10^{-2}$ is used for both.

RCHICPSE51-5:1
RCHICPEL-2-CST-S

Failure to control RCIC min flow valve, given failure to auto control; and failure to transfer to the suppression pool when the CST become depleted. They are called out via verification steps in the system operating instruction and plant emergency instruction flow chart, and well practiced. Therefore an HEP of $5x10^{-2}$ is used for both.

SIHICPSP57-7:1

Failure to connect air cylinders to safety related air system which backs up loss normal system pressure. Since the safety related air receiver tanks are large, it is considered that the need to replenish the air would be several hours after the start of the initiator. Safety Related Air is indicated and alarmed in the main control room. A value of 0.05 is conservatively assigned.

SLHICPEQ-6-RPVLV Failure during ATWS to control RPV level and maintain boron inventory given failure to ADS Inhibit. During a full power ATWS with failure of ADS Inhibit, studies like EPRI NP-5562, Analysis of Anticipated Transients Without Scram in Severe BWR Accidents (1987) have shown that extended periods of time are available for operator recovery action prior to postulated severe core degradation or containment failure. In the case of ADS Inhibit failure with auto low pressure LPCS injection, a Peach Bottom type reactor was determined to self-regulate the reactor power by maintaining RPV pressure slightly above the LPCS pump shutoff head and reactor power was maintained at a quasi-static power level of 2.5% of rated for several hours. Furthermore, no excessive fuel pellet cladding temperatures are predicted at any time during the accident. Since it is more likely that ADS Inhibit failure will occur during a full power ATWS due to the smaller time window, it is also more likely that SLC recovery will overfill the RPV with water injection due to the self-regulating ATWS characteristic. Therefore, an HEP of 0.05 is conservatively applied to define the probability that RPV level will be maintained and boron will not be flushed out the SRVs.

SPHICPPS4:5SPCU SPHICPPS4:5SPCUL	Failure to align suppression pool clean up system for alternate injection. Although addressed in PEI-SPI, section 4.5, no credit is taken in the pre-core damage phase (-SPCU). However, limited credit (5.E-2) is taken in the plant damage state trees for late injection (SPCUL).
TBHICPSM35	Failure to start the back-up fan in the turbine building. Failure of a fan is alarmed in the control room, and immediate action is not required, therefore this is not a time critical action and an intermediate HEP of $5x10^{-2}$ is assumed.

6.2 OTHER TYPE CR HIs

In addition, any HIs that relate to recovery in the post core-damage phase are estimated on the basis of multiplying the HEP evaluated for the phase up to core damage by a factor of 0.1. This is somewhat arbitrary, but given that the procedures are being followed, then there is some chance that the extra time available will allow success. The following events fall into this category:

ADHICREC2-ADS-R	Operator fails to recovery recover from RPV emergency depressurization core damage before RPV failure.
IAHICRSP52-7:2	Operator fails to override isolation signal to reestablish instrument air following a LOCA signal. Again, this is a practiced response to the LOCA signal and an HEP of 0.1 is assumed.
SLCHICREQ-6-SLCR	Operator fails to initiate SLC given core damage failure before containment pressure threshold limit.

ATTACHMENT A

HEP ERROR FACTOR BASIS

The EPRI methods used in this analysis do not address uncertainty. Consequently, the somewhat arbitrary approach of assigning error factors on the basis of the point estimate was adopted as shown below:

$0.1 < HEP < 1.0$	$EF = 1.2$
$HEP \sim 0.1$	$EF = 1.5$
$10^{-2} < HEP < 10^{-1}$	$EF = 3$
$10^{-3} < HEP < 10^{-2}$	$EF = 5$
$10^{-3} < HEP$	$EF = 10$

ATTACHMENT B
HEPs FOR FLOODING ANALYSIS

FLOODING TYPE CP HI MULTIPLIER FACTORS

Internal Flooding has the initiators of PCS Loss, IA Loss, SW Loss, and transient with PCS. Therefore, the HI associated with the following initiators are not included: LOCA, ATWS, LOOP and SBO.

Multiplier Factors for the HI basic event where determined as noted below during a meeting with Wallace Colvin, John Spano, Bengt Lydell, and Eric Jorgensen on 18 December 1991. The basis for assignment is provided on the following page.

HI BASIC EVENT	HI DESCRIPTION	FLOODING MULTIPLIER
ADHICPEC2-ADS_T	FAILS TO EMERG RPV DEPRESS - TRANSIENT	1
CAHICPI008-4:2	OPERS FAIL TO SHIFT CONDENSER AIR REMOVAL TO VP & AB	2
CCHICPSP47_5:4	OPER FAILS TO REALIGN CCCW LOOP C BEFORE 4.5 HOURS	2
CDHICPPS2:1-XH1X	OPER FAILS TO BYPASS RHR LOCA SIGNAL-XH1X	1
CSHICPET-2:P-1	OPER FAILS TO INITIATE CNTMT SPRAY	1
CTHICPPS4:4_ALT	OPERS FAIL TO ALIGN CONDENSATE TRANSFER	F
CTHICPPS4:4_ALT6	OPERS FAIL TO ALIGN CONDENSATE TRANSFER - AT 6 HOURS	F
CVHICPEPC-COM	OPER FAILS TO INITIATE CNTMT PRESS CNTL AND VENT	1
CVHICPEPC-FPCC	OPER FAILS TO INITIATE CNTMT PRESS CNTL VENTING	1
CVHICPEPC-RHR	OPER FAILS TO INITIATE CNTMT PRESS CNTL RHR	1
CVHICPEPC-RHR-E	OPER FAILS TO INITIATE RHR SPC EARLY	1
CVHICPPS7:3E12-T	OPER FAILS TO ALIGN RHR FOR CNTMT VENT	1
CVHICPPS7:3G41-T	OPER FAILS TO ALIGN FPCC FOR CNTMT VENT	1
ECHICPSP42-4:2	OPERFAILS TO CLOSE VALVE 0P42-F0150A(B)	1
FPHICPPS4:2RCIC1	FAIL TO ALIGN FP AFTER RCIC FAILS DUE TO SP TEMP	1
FPHICPPS4:2RCIC4	FAIL TO ALIGN FAST FIRE PROTECTION ALT INJECTION	2
ECHICPSP42-4PMP	OPER FAILS TO INITIATE PUMP 1P42-C001A(B)	1
FWHICPSN27-4:1IA	OPER FAILS TO CNTRL RX FEED BOOSTER PUMP DURING IA LOSS	2
HIHICPAH13870AB1	FAIL TO SECURE SW/ESW DURING 8-10 MIN FOR AB FLOOD	NA
HIHICPAH13870AB2	FAIL TO SECURE SW/ESW DURING 10-15 MIN FOR AB FLOOD	NA
HIHICPAH13870AB3	FAIL TO SECURE SW/ESW DURING 15-40 MIN FOR AB FLOOD	NA
HIHICPAH13870AB4	FAIL TO SECURE SW/ESW/CNDS XFR AFTER 40 MIN FOR AB FLD	NA
HIHICPAH13870TB1	FAIL TO SECURE CIRC WATER/SW BY 14 MIN FOR TB FLOOD	NA
HIHICPAH13870TB2	FAIL TO SECURE SW AFTER 30 MIN FOR TB FLOOD	NA
HIHICPAH13970CC1	FAIL TO SECURE SW/ESW DURING 8-10 MIN FOR CC FLOOD	NA
HIHICPAH13970CC5	FAIL TO SECURE SW/ESW DURING 10-12 MIN FOR CC FLOOD	NA
HIHICPAH13970CC2	FAIL TO SECURE SW/ESW DURING 10-15 MIN FOR CC FLOOD	NA
HIHICPAH13970CC3	FAIL TO SECURE SW/ESW DURING 15-40 MIN FOR CC FLOOD	NA
HIHICPAH13970CC4	FAIL TO SECURE SW/ESW AFTER 40 MIN DURING CC FLOOD	NA
HPHICPEL-1	OPER FAILS TO INITIATE HIGH PRESSURE INJECTION	1
HPHICPSE22-5:0	OPER FAILS TO CONTROL MIN FLOW VALVE 1E22-F012	1
HPHICPSE22-5:2	OPER FAILS TO XFR TO SUPR POOL WITH 1E22-F015	1
IAHICPSP51-4:2	OPER FAILS TO REPOSITION NCC LUBE OIL CLR OUTLET VLV	2
LCHICPSE12-5:1	OPER FAILS TO CONTROL MIN FLOW VALVE 1E12-F064A	1
LPHICPEL-1	OPER FAILS TO INITIATE LOW PRESSURE ECCS	1
LPCPSE21-5:1	OPER FAILS TO CONTROL MIN FLOW VALVE 1E21-F011	1

HI BASIC EVENT	HI DESCRIPTION	FLOODING MULTIPLIER
RCHICPEL-2_CST_S	OPER FAILS TO PREVENT RCIC SUCTION SHIFT TO SP	1
RCHICPS51-LDTRIP	OPER FAILS TO RECOVERY FROM RCIC HIGH TEMP LD TRIP	.5F
RCHICPSE51-5:1	OPER FAILS TO PERFORM RCIC SUCTION SHIFT	1
RPHICPERC-1:Q-2	OPER FAILS TO MANUALLY SCRAM REACTOR	1
SCHICPSE12-5:3	OPER FAILS TO ALIGN RHR SUPR POOL COOLING	1
SIHICPSP57-7:1	OPER FAILS TO CONNECT AIR CYLINDERS	1
SPHICPPS4:5SPCUL	OPERS FAIL TO ALIGN SUPR POOL C/U ALT INJECTION LATE	1
TBHICPSM35	OPER FAILS TO START STANDBY FAN	2

BASIS FOR MULTIPLIER FACTOR ASSIGNMENT

Multiplier Factors are applied to the HIs used in the initial IPE study of internal plant initiator to model the increased failure probability when performing flooding tasks to mitigate a plant transient. Increase human interaction error is due to: 1) increased stress as a result of the control room response to a low frequency event that challenges reactor safety, 2) increased stress as a result of mitigating a flooding event with less available operations staff to perform routine recovery task - due to need to man power to attend to the initiating flooding event, and/or 3) increased stress due to a high alarm and information processing during the unique flooding accident progression.

1. MULTIPLIER FACTOR = 1 No quantitative increase to the normal human interaction model of operator response. The basis is that these actions are the same or nearly the same as those previously modeled for non-flooding events. No significant impact of any flooding initiator was identified which would degrade the control room response or the local manual action.

2. MULTIPLIER FACTOR = 2 The normal human interaction model of operator response is increased by a factor of two. The basis is that these actions may be impacted by unique man power demands to mitigate the flooding event or by the increased stress associated with the response to a challenging low frequency event. This multiplier factor of 2 is considered to conservatively bound the human interactions identified as listed below. These human interaction events are not performed in critical time windows and thus a low, yet conservative factor of 2 is postulated.

CAHICPIOO8-4:2	OPERS FAIL TO SHIFT CONDENSER AIR REMOVAL TO VP & AB
CCHICPSP47_5:4	OPER FAILS TO REALIGN CCCW LOOP C BEFORE 4.5 HOURS
FWHICPSN27-4:1IA	OPER FAILS TO CNTRL RX FEED BOOSTER PUMP DURING IA LOSS
IAHICPSP51-4:2	OPER FAILS TO REPOSITION NCC LUBE OIL CLR OUTLET VLV
TBHICPSM35	OPER FAILS TO START STANDBY FAN

3. MULTIPLIER = .5F The normal human interaction model of operator response is conservatively set to a high screening value of 0.50 to account for the potential for this alarm to be masked by the high frequency of alarm/information processing during the the first half hour of a low frequency flooding accident.

   RCHICPS51-LDTRIP OPER FAILS TO RECOVERY FROM RCIC HIGH TEMP LD TRIP

4. MULTIPLIER = F The normal human interaction model of operator response is conservatively set to the failed state to conservatively model: the demand for plant operators to attend to the flooding event results in the immediate unavailability of personnel during the critical initial transient time window, or the potentially hazardous environment associated with the task of walking through a flooded hallway. A Multiplier Factor "F" will be calculated on a case by case basis which results in a system failure probability of between .99 and 1.00. The events classified into this type are the following.

   CTHICPPS4:4_ALT OPERS FAIL TO ALIGN CONDENSATE TRANSFER
   CTHICPPS4:4_ALT6 OPERS FAIL TO ALIGN CONDENSATE TRANSFER - AT 6 HOURS

5. MULTIPLIER = NA Not applicable, no multiplier factor is required with those human interaction which are used only for flooding.

ATTACHMENT C

[illegible]ARY IPE HI LIST

1. LEVEL 1 AND PLANT DAMAGE STATE TYPE CP HIs

HI ID	HI DESCRIPTION	HEP	ERROR FACTOR
ADHICPC5-1-ADS-A	OPER FAILS TO INHIBIT ADS - ATWS WITH FEEDWATER	3.80E-3	5
ADHICPC5-1-ADS-I	OPER FAILS TO INHIBIT ADS - ATWS WITH IORV & WITH FDW	3.6	-
ADHICPC5-1-ADS-L	OPER FAILS TO INHIBIT ADS - ATWS WITH LOOP	3.60E-2	3
ADHICPC5-1-ADS-O	OPER FAILS TO INHIBIT ADS - ATWS WITH NO FEEDWATER	7.2	-
ADHICPEC2-ADS-T	FAILS TO EMERG RPV DEPRESS - TRANSIENT	1.00E-3	5
ADHICPEC5-ADS-FL	FAILS TO EMERG RPV DEPRESS - ATWS, FDW & LEVEL CONTROL	7.00E-3	5
ADHICPEC5-ADS-FX	FAILS TO EMERG RPV DEPRESS - ATWS, FDW & NO LEVEL CNTL	1.40E-2	3
CAHICPIO08-4:2	OPERS FAIL TO SHIFT CONDENSER AIR REMOVAL TO VP & AB	1.00E-3	5
CCHICPSP47-5:4	OPER FAILS TO REALIGN CCCW LOOP C BEFORE 4.5 HOURS	1.00E-3	5
CDHICPPS2:1-XH1X	OPER FAILS TO BYPASS RHR LOCA SIGNAL-XH1X	1.00E-3	5
CSHICPET-2:P-1	OPER FAILS TO INITIATE CNTMT SPRAY	1.70E-2	3
CTHICPPS4:4-ALT	OPERS FAIL TO ALIGN CONDENSATE TRANSFER	1.00E-1	1.5
CTHICPPS4:4-ALT6	OPERS FAIL TO ALIGN CONDENSATE TRANSFER - AT 6 HOURS	1.00E-3	5
CTHICPPS4:4-ALTC	OPERS FAIL TO ALIGN CONDENSATE TRANSFER - IN T3C	3.00E-1	1.2
CVHICPEPC-COM	OPER FAILS TO INITIATE CNTMT PRESS CNTL AND VENT	1.00E-3	5
CVHICPEPC-FPCC	OPER FAILS TO INITIATE CNTMT PRESS CNTL VENTING	1.00E-1	1.5
CVHICPEPC-RHR	OPER FAILS TO INITIATE CNTMT PRESS CNTL RHR	1.00E-2	3
CVHICPEPC-RHR-E	OPER FAILS TO INITIATE RHR SPC EARLY	1.00E-1	1.5
CVHICPPS7:3E12-T	OPER FAILS TO ALIGN RHR FOR CNTMT VENT	1.00E-4	10
CVHICPPS7:3G41-T	OPER FAILS TO ALIGN FPCC FOR CNTMT VENT	1.00E-4	10
DGHICPOS11-2:5	OPER FAILS TO INITIATE DIV 1 D/G	1.25E-3	5
ECHICPSP42-4:2	OPER FAILS TO CLOSE VALVE OP42-F0150A(B)	5.00E-2	3
ECHICPSP42-4PMP	OPER FAILS TO INITIATE PUMP 1P42-C001A(B)	5.00E-2	3
ERHICPPS4:2-ESW	OPERS FAIL TO ALIGN 2 VLVS FOR RPV INJECTION	1.00E-2	3
FPHICPPS4:2-DD-O	OPERS FAIL TO MAINTAIN OIL FOR DIESEL DRIVEN FIRE PUMP	3.00E-2	3
FPHICPPS4:2RCIC1	FAIL TO ALIGN FP AFTER RCIC FAILS DUE TO SP TEMP	3.00E-1	1.2
FPHICPPS4:2RCIC2	FAIL TO ALIGN FP AFTER RHR FAILS DUE TO MCC TEMP	1.00E-1	1.5
FPHICPPS4:2RCIC3	FAIL TO ALIGN HPCS AFTER HPCS FAILS DUE TO MCC TEMP	1.00E-2	3
FPHICPPS4:2RCIC4	FAIL TO ALIGN FAST FIRE PROTECTION ALT INJECTION	1.00E-1	1.5
FWHICPEC5-2:3LCS	OPER FAILS TO CONTROL LVL AT TAF WITH FDW DURING IORV	1.00E-2	3
FWHICPEC5-3:2	FAILS TO CNTRL RPV LVL AT TAF	1.00E-2	3
FWHICPEL-2-FDW-L	OPER FAILS TO REOPEN MFP CONTROL VALVES FOR T3C-C	1.00E-2	3
FWHICPEL-2-FDW-V	OPER FAILS TO REOPEN MFP CONTROL VALVES	5.00E-3	1.2
FWHICPSN27-4:1IA	OPER FAILS TO CNTRL RX FEED BOOSTER PMP DURING IA LOSS	1.20E-1	1.5

FWEICPSN27-4:1IL	OPER FAILS TO CNTRL RFBP DURING IA LOSS > 2 HR	5.00E-3	5
HIHICPEC5-3:2-F	FAILS TO RESTR RHR A/B OR LPCS & CNTL AT TAF - W/ FDW	1.00E-3	5
HIHICPEC5-3:2-S	FAILS TO RESTR RHR A/B OR LPCS & CNTL AT TAF - W/O FDW	1.00E-3	5
HIHICPEC5-5-CRIT	OPER FAILS TO CNTRL RPV LVL & FLUSHES BORON	2.00E-3	5
HIHICPOR10-4:3-B	OPERS FAIL TO X-TIE UNIT 1 & 2 BATTERIES AND LOAD SHED	1.00E-2	3
HIHICPOR10-4:3-D	OPER FAILS TO OPEN DIV 3 SWITCHGEAR ROOM DOOR	2.00E-3	3
HIHICPOR10-XTIE	OPER FAILS TO X-TIE DIV 3 AND 2 MCCs	5.00E-3	3
HPHICPEL-1	OPER FAILS TO INITIATE HIGH PRESSURE INJECTION	1.25E-3	5
HPHICPSE22-5:0	OPER FAILS TO CONTROL MIN FLOW VALVE 1E22-F012	5.00E-2	3
HPHICPSE22-5:2	OPER FAILS TO XFR TO SUPR POOL WITH 1E22-F015	5.00E-2	3
IAHICPSP51-4:2	OPER FAILS TO REPOSITION NCC LUBE OIL CLR OUTLET VLV	5.00E-2	3
LCHICPSE12-5:1	OPER FAILS TO CONTROL MIN FLOW VALVE 1E12-F064A	5.00E-2	3
LPHICPEL-1	OPER FAILS TO INITIATE LOW PRESSURE ECCS	1.00E-3	5
LPHICPSE21-5:1	OPER FAILS TO CONTROL MIN FLOW VALVE 1E21-F011	5.00E-2	3
NSHICPEC5-2-L1	OPER FAILS TO BYPASS MSIV LEVEL 1 ISOLATION FOR T3C-C	1.00E-1	1.5
NSHICPEC5-2-L1T3	OPER FAILS TO BYP MSIV LVL 1 ISOL FOR T3C-A OR T3B-C	1.00E+0	-
RCHICPEL-2-CST-S	OPER FAILS TO PREVENT RCIC SUCTION SHIFT TO SP	5.00E-2	3
RCHICPS51-LDTRIP	OPER FAILS TO RECOVERY FROM RCIC HIGH TEMP LD TRIP	5.00E-2	3
RCHICPSE51-5:1	OPER FAILS TO PERFORM RCIC SUCTION SHIFT	5.00E-2	3
RPHICPERC-1:0-2	OPER FAILS TO MANUALLY SCRAM REACTOR	1.00E-4	10
SCHICPSE12-5:3	OPER FAILS TO ALIGN RHR SUPR POOL COOLING	1.09E-4	10
SIHICPSP57-7:1	OPER FAILS TO CONNECT AIR CYLINDERS	5.00E-2	3
SLHICPEQ-6-SLC1	OPER FAILS TO INITIATE SLC - 1 PUMP INJECTION	1.25E-3	5
SLHICPEQ-6-SLCX	OPER FAILS TO INITIATE SLC - LEVEL CONTROL FAILURE	1.00E+0	-
SPHICPPS4:5SPCU	OPERS FAIL TO ALIGN SUPR POOL C/U ALT INJECTION	1.00E+0	-
TBHICPSM35	OPER FAILS TO START STANDBY FAN	5.00E-2	3

2. FLOODING MULTIPLIER FACTORS AND ADJUSTED HEPs TYPE CP HIs

HI ID	HI DESCRIPTION	MULTIPLIER OR HEP
ADHICPEC2-ADS-T	FAILS TO EMERG RPV DEPRESS - TRANSIENT	1x
CAHICPI008-4:2	OPERS FAIL TO SHIFT CONDENSER AIR REMOVAL TO VP & AB	2x
CCHICPSP47-5:4	OPER FAILS TO REALIGN CCCW LOOP C BEFORE 4.5 HOURS	2x
CDHICPPS2:1-XH1X	OPER FAILS TO BYPASS RHR LOCA SIGNAL-XH1X	1x
CSHICPET-2:P-1	OPER FAILS TO INITIATE CNTMT SPRAY	1x
CTHICPPS4:4-ALT	OPERS FAIL TO ALIGN CONDENSATE TRANSFER	1.00E+0
CTHICPPS4:4-ALT6	OPERS FAIL TO ALIGN CONDENSATE TRANSFER - AT 6 HOURS	1.00E+0
CVHICPEPC-COM	OPER FAILS TO INITIATE CNTMT PRESS CNTL AND VENT	1x
CVHICPEPC-FPCC	OPER FAILS TO INITIATE CNTMT PRESS CNTL VENTING	1x
CVHICPEPC-RHR	OPER FAILS TO INITIATE CNTMT PRESS CNTL RHR	1x
CVHICPEPC-RHR-E	OPER FAILS TO INITIATE RHR S[illegible] EARLY	1x
CVHICPPS7:3G41-T	OPER FAILS TO ALIGN FPCC FOR CNTMT VENT	1x
CVHICPPS7:3E12-T	OPER FAILS TO ALIGN RHR FOR CNTMT VENT	1x
ECHICPSP42-4:2	OPER FAILS TO CLOSE VALVE 0P42-F0150	1x
ECHICPSP42-4PMP	OPER FAILS TO INITIATE PUMP 1P42-C001	1x
FPHICPPS4:2RCIC1	FAIL TO ALIGN FP AFTER RCIC FAILS DUE TO SP TEMP	1x
FPHICPPS4:2RCIC4	FAIL TO ALIGN FAST FIRE PROTECTION ALT INJECTION	2x
FWHICPSN27-4:1IA	OPER FAILS TO CNTRL RX FEED BOOSTER PUMP DURING IA LOSS	2x
HPHICPEL-1	OPER FAILS TO INITIATE HIGH PRESSURE INJECTION	1x
HPHICPSE22-5:0	OPER FAILS TO CONTROL MIN FLOW VALVE 1E22-F012	1x
HPHICPSE22-5:2	OPER FAILS TO XFR TO SUPR POOL WITH 1E22-F015	1x
IAHICPSP51-4:2	OPER FAILS TO REPOSITION NCC LUBE OIL CLR OUTLET VLV	2x
LCHICPSE12-5:1	OPER FAILS TO CONTROL MIN FLOW VALVE 1E12-F064A	1x
LPHICPEL-1	OPER FAILS TO INITIATE LOW PRESSURE ECCS	1x
LPCPSE21-5:1	OPER FAILS TO CONTROL MIN FLOW VALVE 1E21-F011	1x
RCHICPEL-2-CST-S	OPER FAILS TO PREVENT RCIC SUCTION SHIFT TO SP	1x
RCHICPS51-LDTRIP	OPER FAILS TO RECOVERY FROM RCIC HIGH TEMP LD TRIP	5.00E-1
RCHICPSE51-5:1	OPER FAILS TO PERFORM RCIC SUCTION SHIFT	1x
RPHICPERC-1:0-2	OPER FAILS TO MANUALLY SCRAM REACTOR	1x
SAHICPSP51-4:2	OPER FAILS TO REPOSIT NCC LO CLR OUTLET	2x
SCHICPSE12-5:3	OPER FAILS TO ALIGN RHR SUPR POOL COOLING	1x
SIHICPSP57-7:1	OPER FAILS TO CONNECT AIR CYLINDERS	1x
SPHICPPS4:5SPCUL	OPERS FAIL TO ALIGN SUPR POOL C/U ALT INJECTION LATE	1x
TBHICPSM35	OPER FAILS TO START STANDBY FAN	2x

3. FLOODING ONLY TYPE CP HIs

HI ID	HI DESCRIPTION	HEP	ERROR FACTOR
HIHICPAH13870AB1	FAIL TO SECURE SW/ESW DURING 8-10 MIN FOR AB FLOOD	5.00E-1	1.2
HIHICPAH13870AB2	FAIL TO SECURE SW/ESW DURING 10-15 MIN FOR AB FLOOD	1.00E-1	1.5
HIHICPAH13870AB3	FAIL TO SECURE SW/ESW DURING 15-40 MIN FOR AB FLOOD	5.00E-2	3
HIHICPAH13870AB4	FAIL TO SECURE SW/ESW/CNDS XFR AFTER 40 MIN FOR AB FLD	5.00E-3	5
HIHICPAH13870TB1	FAIL TO SECURE CIRC WATER/SW BY 14 MIN FOR TB FLOOD	5.00E-2	3
HIHICPAH13870TB2	FAIL TO SECURE SW AFTER 30 MIN FOR TB FLOOD	5.00E-3	5
HIHICPAH13970CC1	FAIL TO SECURE SW/ESW DURING 8-10 MIN FOR CC FLOOD	5.00E-2	3
HIHICPAH13970CC5	FAIL TO SECURE SW/ESW DURING 10-12 MIN FOR CC FLOOD	3.00E-1	1.2
HIHICPAH13970CC2	FAIL TO SECURE SW/ESW DURING 10-15 MIN FOR CC FLOOD	1.00E-1	1.5
HIHICPAH13970CC3	FAIL TO SECURE SW/ESW DURING 15-40 MIN FOR CC FLOOD	5.00E-2	3
HIHICPAH13970CC4	FAIL TO SECURE SW/ESW AFTER 40 MIN DURING CC FLOOD	5.00E-3	5

4. PLANT DAMAGE STATE TYPE CP HIs

HI ID	HI DESCRIPTION	HEP	ERROR FACTOR
ADHICPEC2-ADS-LR	FAILS TO RECOVER FROM RPV DEPRESS CD FAILURE	1.00E-4	10
FPHICPPS4:2FP-LE	OPERS FAIL TO ALIGN VLVS FOR FP LATE-INJ - BEFOR 3 HRS	5.00E-2	5
FPHICPPS4:2FP-LL	OPERS FAIL TO ALIGN VLVS FOR FP LATE-INJ - AFTER 3 HRS	5.00E-3	5
HIHICPOR10-4:0-I	OPER FAILS TO CLOSE FPCC OTBD ISOLATION - G41-F145	5.00E-2	5
SLHICPEQ-6-RPVLV	OPER FAILS TO CNTL RPV LEVEL & MAINTN BORON INVENTORY	5.00E-2	5
SPHICPPS4:5SPCUL	OPERS FAIL TO ALIGN SUPR POOL C/U ALT INJECTION LATE	5.00E-2	5

5. ACCIDENT PROGRESSION EVENT TREE TYPE CP HI

HI ID	HI DESCRIPTION	HEP	ERROR FACTOR
IGHICPEH-1-H2IG	OPER FAILS TO INITIATE HYDROGEN IGNITION SYSTEM	5.00E-3	5

6. LEVEL 1, FLOODING & PLANT DAMAGE STATE TYPE CR HIs

HI EVENT	HI DESCRIPTION	HEP	ERROR FACTOR
CVHICRPS7:3G41T	OPER IS UNABLE TO LOCALLY OPEN 1G41-F0145	5.00E-2	5
IAHICRSP52-7:2	OPER FAILS TO OVERRIDE ISOLATION SIGNAL	1.00E-1	1.5

7. PLANT DAMAGE STATE TYPE CR HIs

HI ID	HI DESCRIPTION	HEP	ERROR FACTOR
ADHICREC2-ADS-R	FAILS TO RECOVER FM RPV DEPRESS CD GIVEN EARLY FAILURE	1.00E-1	1.5
SLHICREQ-6-SLCR	OPER FAILS TO INITIATE SLC - GIVEN CD FAILURE	1.00E-1	1.5

8. TYPE MA HIs

HI ID	HI DESCRIPTION	HEP	ERROR FACTOR
DGHIMASR43-4:1:A	FAILURE TO RESTORE FOLLOWING MAINTENANCE	0.00E+0	
DGHIMASR43-4:1:B	FAILURE TO RESTORE FOLLOWING MAINTENANCE	0.00E+0	
DHHIMASE22B-4:1	FAILURE TO RESTORE FOLLOWING MAINTENANCE	0.00E+0	
ECHIMASP42-4:1A	ECC TRAIN A NOT RESTORED FOLLOWING MAINTENANCE	0.00E+0	
ECHIMASP42-4:1B	ECC TRAIN B NOT RESTORED FOLLOWING MAINTENANCE	0.00E+0	
ESHIMASP45-4:1A	ESW TRAIN A NOT RESTORED FOLLOWING MAINTENANCE	0.00E+0	
ESHIMASP45-4:1B	ESW TRAIN B NOT RESTORED FOLLOWING MAINTENANCE	0.00E+0	
ESHIMASP45-4:1C	ESW TRAIN C NOT RESTORED FOLLOWING MAINTENANCE	0.00E+0	
HPHIMASE22-4:1	FAILURE TO RESTORE HPCS AFTER MAINTENANCE	0.00E+0	
IAHIMASP51-4:1:2	FAILURE TO RESTORE FOLLOWING MAINTENANCE	1.00E-3	5
LCHIMASE12-4:1A	FAILURE TO RESTORE TRAIN A LPCI FOLLOWING MAINT	0.00E+0	
LCHIMASE12-4:1B	FAILURE TO RESTORE TRAIN B LPCI FOLLOWING MAINT	0.00E+0	
LCHIMASE12-4:1C	FAILURE TO RESTORE TRAIN C LPCI FOLLOWING MAINT	0.00E+0	
LPHIMASE21-4:1	FAILURE TO RESTORE LPCS AFTER MAINTENANCE	0.00E+0	
RCHIMASE51-4:1	FAILURE TO RESTORE RCIC FOLLOWING MAINTENANCE	0.00E+0	
SAHIMASP51-4:1:2	FAILURE TO RESTORE FOLLOWING MAINTENANCE	1.00E-3	5
SIHIMASP57-4:0:A	FAILURE TO RESTORE FOLLOWING MAINTENANCE	0.00E+0	
SIHIMASP57-4:0:B	FAILURE TO RESTORE FOLLOWING MAINTENANCE	0.00E+0	
SLHIMA	FAILURE TO RESTORE SLC FOLLOWING MAINTENANCE/TEST	0.00E+0	

APPENDIX H.1

CLEVELAND ELECTRIC ILLUMINATING COMPANY

PERRY NUCLEAR POWER PLANT

INDIVIDUAL PLANT EXAMINATION

CONTAINMENT CAPACITY ANALYSIS

by

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R. J. Schmehl

Submitted to

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February 17, 1992

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1.0 Purpose and Scope

An Individual Plant Examination is being performed for the Perry Nuclear Power Plant Unit I that meets the requirements of NRC Generic Letter 88-20. Gilbert/Commonwealth is under subcontract to NUS Corporation to define the capacity of the containment structure subjected to pressure resulting from a postulated accident. In this report, the probability of failure as a function of internal pressure is developed. Also included is the variability in the probability of failure which may be expected along with a description of potential leak paths and estimates of leak areas. The report also considers the effects of elevated accident temperatures on containment failure capacities.

2.0 Description of the Containment

The containment vessel for this plant is a free standing steel vessel which is located inside of a concrete shield building. The annulus concrete located between the containment and the shield building strengthens and stiffens the containment vessel. The drywell is located inside the containment vessel. All of these structures are supported on the foundation mat. A description of the structures that comprise the containment system is presented below.

2.1 Containment Vessel

The containment vessel is a pressure retaining structure composed of a free standing steel cylinder with an ellipsoidal dome, secured to a steel lined reinforced concrete foundation mat. The mat is the common foundation for the major structures of the reactor building complex. The free standing portion of the containment vessel is supported by and anchored into the foundation mat, and is designed, fabricated and erected in accordance with the requirements of ASME Code Section III for Class MC components.

The containment vessel is designed to contain radioactive material which might be released from the nuclear steam supply system following a loss-of-coolant accident. The steel containment vessel ensures a high degree of leak tightness during normal operating and design accident conditions. It is designed for a maximum internal pressure of 15 psig with a coincident temperature of 185°F at accident conditions, and a maximum external pressure differential of 0.8 psi due to accidental operation of the spray headers.

The basic dimensions of the containment vessel are:

a. Cylinder inside diameter - 120 feet
b. Cylinder height - 152 feet - 2 inches.
c. Ellipsoidal dome ratio - 2:1.

The containment vessel cylinder has six external stiffening rings at various elevations. Details of the containment vessel cylinder, dome and stiffeners are shown in Figure 1. Two personnel access airlocks and one equipment hatch are provided. Details are shown on Figures 2 and 3.

Details of typical penetrations are shown in Figures 4 and 5. The fuel transfer tube (penetration P205) is shown on Figure 4A. This penetration is oriented parallel to the 180° azimuth, 15 feet 2 inches east of the reactor building centerline. The penetration is skewed 32° 30′ from vertical at Elevation 636′-6-3/4". The lower 18 feet 6 inches of the containment vessel forms the outside boundary of the suppression pool; the inside boundary of the suppression pool is formed by the drywell weir wall. Corrosion of the lower 23 feet 6 inches of the containment vessel and exposed steel mat liner is minimized by the use of ASME SA 516, Grade 70, plate clad with stainless steel. At Elevation 721′-0" a 125 ton capacity polar crane is supported from the containment vessel.

Final design plate thickness of the containment vessel cylinder and dome is 1 1/2 inches. Local areas of the vessel, near large penetrations, have thicknesses greater than 1 1/2 inches.

2.2 Shield Building

The shield building is a reinforced concrete structure consisting of a flat foundation mat, a cylindrical wall and a shallow dome. The general configuration of the shield building and its relation to the other structures of the reactor building complex is shown in Figure 1. The foundation mat, common to the shield building, annulus concrete, containment vessel, and interior structure is circular in plan with a diameter of 136 feet and a thickness of 12 feet 6 inches. The foundation mat is founded on Chagrin shale at Elevation 562′-3", approximately 56 feet below grade.

The shield building cylindrical wall extends from the top of the foundation mat at Elevation 574'-10" to Elevation 749'-9" and has an outside diameter of 136 feet with a wall thickness of 3 feet 0 inches. The shallow dome has a radius of 120 feet 0 inches, with a thickness of 2 feet 6 inches. There is no thickened ring girder, but the elevation of the wall at the junction of the wall and dome was raised to provide a greater section to help resist the outward thrust of the dome. Details of the shield building and mat foundation are shown in Figures 1 and 6.

The concrete has a minimum 28 day cylinder compressive strength of 3,000 psi. The steel reinforcement is in accordance with the requirements of ASTM A615, Grade 60. A general reinforcing pattern of orthogonal bars arranged vertically and circumferentially in both faces of the wall was used in the shield building wall.

The reinforcing pattern for the dome is essentially radial and circumferential with the center section arranged orthogonally for ease of placing.

2.3 Annulus Concrete

The annulus concrete extends from the top of the foundation mat at Elevation 574'-10" to Elevation 598'-4" and has a radial thickness of 4 feet 10-1/2 inches. This annulus concrete provides stiffness to the steel containment vessel to reduce the dynamic response of the steel containment vessel due to the postulated safety relief valve discharge loading phenomena. Three inches of compressible material are also provided between the containment vessel and the annulus concrete below Stiffener #1 and above Stiffener #4 to reduce thermal compressive stresses. The concrete has a minimum 28 day cylinder compressive strength of 3,000 psi. The steel reinforcement is in accordance with the requirements of ASTM A615, Grade 60.

A general reinforcing pattern of orthogonal bars arranged vertically and circumferentially in both faces of the wall was used in the annulus concrete. The general reinforcement pattern is shown in Figure 7.

2.4 Drywell

The drywell wall is generally a right, vertical cylinder. It is 83 feet 0 inches outside diameter, 85 feet 9 inches high and 5 feet 0 inches thick. The drywell wall is subdivided into two regions which have different construction and design methods. The drywell is shown in Figure 1.

The lower 26 feet 2 inches of the drywell is the vent region. The main suppression pool area in the containment vessel is connected to the drywell by 120 vents. The vent sleeves are 28 inch outside diameter, 1/4 inch thick, stainless steel tubes located in 3 rows of 40 vents. Vents are constructed from ASTM A 240, Grade 304, stainless steel. The vent structure is a steel concrete composite construction which consists of two concentric cylinders fabricated from 1 inch thick ASTM A 516, Grade 70, steel with a ten percent Type 304 stainless steel cladding. The annulus between the cylinders is stiffened vertically by radial steel plates and is filled with 5,000 psi concrete. The steel plates are designed to carry all membrane tensile forces in this region. At Elevation 574-10", the vent region is anchored to the containment base mat by means of vertical tension ties in the form of anchor bars made of ASTM A 537 CL.2 steel for the transfer of any uplift forces to the base mat. The anchor plate and stiffeners at the bottom of the anchor bars are sized for the capacity of the anchor bars. There are a total of 144 anchor bars on each face of the drywell. Details of the vent region are shown in Figure 8.

The upper drywell wall is a reinforced concrete cylinder connected to the lower vent region by cadwelding all vertical and diagonal reinforcement to the ring girder at Elevation 600'-10". The inside face of the drywell is formed with a 1/4 inch thick steel plate of ASTM A 285, Grade A, material which is stiffened vertically by 2 inch x 3 inch x 1/4 inch angles spaced at approximately 1 foot 3 inches and horizontally by stiffener rings of 3 inch x 4 inch x 1/4 inch angles spaced at about 5 feet 0 inches center to center. The stiffener angles are ASTM A36 material. This steel liner is very conservatively not considered in the design as contributing to structural strength or leak tightness of the drywell.

The drywell top slab is a flat horizontal, circular, reinforced concrete slab. It contains a central circular opening of 31 feet 11 1/2 inch diameter which is closed by the drywell head. The configuration and seal details of the 14 feet 9 1/4 inch deep steel ellipsoidal drywell head are shown in Figure 10. The top slab is stiffened by two longitudinal reinforced concrete walls which are part of the upper pool water system. Details of the upper drywell wall and drywell slab are shown in Figure 9.

The personnel access air lock has an outside diameter of 9 feet 8 inches and is located at the 599'-9" elevation floor to provide access to the drywell. For large pieces of equipment, an 11 foot 0 inch square clear opening, bolted, double gasket sealed, equipment access hatch is provided at Elevation 599'-9". The personnel air lock and the equipment hatch are integrally connected by full penetration welds to steel frames designed to act as end anchorage for all of the drywell wall reinforcement in the vicinity of the lock and hatch. Details of the personnel air lock and the drywell equipment hatch are shown in Figures 11 and 12.

Penetrations through the drywell wall for piping and electrical systems are of the single barrier leak tight type. The main steam lines are anchored at the drywell and are provided with guard pipes through to the isolation valves outside containment.

3.0 Drywell Failure Modes

Stresses produced by pressure are the primary cause of postulated failure in the following components:

1. Drywell wall
2. Drywell roof
3. Drywell head
4. Drywell equipment hatches
5. Drywell personnel airlock

3.1 Failure Definitio

For the purposes of this report, failure pressures are defined as the pressures associated with mean yield stress in the structural components. Although higher pressures may be possible based on the ultimate strength of the materials, the load-deflection relationships of complex structures are difficult to predict for stresses above yield; therefore the pressure levels at yield represent a practical limit for quantitative description of failure.

3.2 Failure Pressures From Previous Studies

Previous studies of BWR Mark III containments with similar drywell structures are reported by References 1, 4, 8 and 12.

In Reference 1 failure of the drywell wall, equipment hatch and personnel airlocks for the Kuosheng Nuclear Plant are considered to be important failure modes from pressure induced stresses. From Table 1 and 2, Appendix F2 of Reference 1, failures from internal pressure are above 105 psi for all structural elements except the personnel airlock and upper equipment hatch. These "fail" at 53 psi and 60 psi respectively with an associated "minor" leakage. For external pressure the capacities are much the same as for internal except for the drywell head which could buckle at 70 psi with minor leakage and at 80 psi with major leakage.

In Reference 4, Volume 2, Appendix C, Table C.8.1, the failure pressure range for the Grand Gulf Nuclear Station is identified as 50-120 psi differential pressure for probabilities between 5 and 95 percentile.

In Reference 12 p. 9-15 the capacities of the Grand Gulf drywell structure and personnel hatch are identified as 67 psi and 72.9 psi respectively. These are termed "ultimate capacities" but in the context of Reference 12 they are based on yield, which we can conservatively assume to be mean yield.

In Reference 8 p. 16-17 the "ultimate capacity" of Grand Gulf drywell is identified at +67 and -89 psi differential pressure. Again, it is conservatively assumed that these pressures are based on mean yield material strengths.

3.3 Comparison of PNPP and Drywells from Previous Studies

A direct comparison of physical dimensions and structural details of the Perry and Kuosheng containments is unavailable. Both are BWR Mark III containments with similar loadings and both designs would be based on the same or very similar codes and acceptance criteria. From this it can be surmised that the capability of the overall designs are similar.

PNPP USAR (Reference 7) and Grand Gulf (Reference 10) show similar drywell structures. While exact dimensions and details for both structures are not available for a thorough comparison, it can be assumed that the overall size, design pressures, design codes and acceptance criteria for both drywells are the same; therefore, the overall design of the concrete wall and roof should be very similar.

3.4 PNPP Unique Data

The following PNPP unique data has been obtained:

1. Drywell Head:

From Reference 26 the following estimates are made: Stress level at design pressure of 30 psi = 15.12 ksi (radial in flange). Flange material is stainless steel SA 240 Type 304.

Therefore,

Pressure at minimum yield stress of 30 ksi =

$$30 \text{ psi} \times \frac{30}{15.12} = 59.5 \text{ psi}$$

Assuming that the ratio of mean yield to minimum yield is the same as for SA 516 Grade 70 (see Section 4.0), the mean yield pressure, based on the flange capability is,

$$59.5 \times \frac{49.7}{38.0} = 77.8 \text{ psi mean yield internal pressure}$$

For external pressure the head has a design allowable of 33 psi from Reference 26. It is difficult to accurately determine buckling failure pressure because buckling theory has difficulty accounting for all the variables like tolerances in curvature and thickness. The smallest estimate of critical buckling pressure from Reference 26 is 210 psi. Note also that the PNPP drywell head is 1 1/2" thick, compared to 1 3/8" for the Kuosheng head, for which Reference 1 reports a failure pressure of 70 psi. Assuming that buckling is proportional to t^2, the estimated failure buckling pressure for the PNPP head by comparison to the Kuosheng head is,

$$\left(\frac{1.5}{1.375}\right)^2 \times 70 \text{ psi} = 83.3 \text{ psi external buckling failure pressure}$$

2. Equipment Hatch

The equipment hatch is a stiffened flat plate structure and would have similar stress levels for internal or external pressure except that bolts and bolting flange are not challenged by external pressure. The material is SA 516 Grade 70. Reference 27 p. 98 shows a membrane stress in the coverplate of only 8.9 ksi at 30 psi pressure (thermal stresses not included). On p. 101 of Reference 27, maximum stress in the stiffener is identified as 18,000 psi under 30 psi pressure and thermal load. Further investigation of Reference 27 computer output could be performed to separate out the bending stresses from the total stress in the stiffener. Also there would be additional capacity in the stiffened door based on plastic strength of stiffeners and redistribution of moments. Conservatively using the 18,000 psi as a measure of bending stress level at 30.69 psi pressure (includes 0.69 psi "live load"), the projected failure pressure for a mean yield of 49.7 ksi equals,

$$30.69 \text{ psi} \times \frac{49.7}{18} = 84.7 \text{ psi}$$ mean yield pressure based on stiffened plate strength

From Reference 27 p. 104, comparison of bolt stress to allowable bolt stress and of flange stress to allowable flange stress under 30.69 psi pressure is,

$$\frac{\text{bolt allowable}}{\text{bolt stress}} = \frac{27.5}{17.63} = 1.56$$

and,

$$\frac{\text{flange allowable}}{\text{flange stress}} = \frac{28.59}{18.845} = 1.52$$

Conservatively, assuming that the bolt or flange yield stresses are at least 2 times their respective allowable stresses would give a failure capacity at yield of, 30.69 x 1.52 x 2.0 = 93 psi — mean yield pressure based on bolt strength

In summary 84.7 psi can conservatively be used as an estimate of mean failure capacity of the drywell hatch.

3. Personnel Access Airlock

The drywell airlock is nearly identical to the containment airlocks, except for a slightly smaller door size. The evaluation and mean yield pressure from the containment evaluation (Section 4.0) apply to the drywell airlock, i.e. mean yield pressure of 107.2 psi.

4. Drywell Wall

The circumferential reinforcement in the drywell wall is #18 bars @ 12", each face. Using the nominal bar area of 4.0 in^2, reinforcement per inch = 0.667 in^2. Using a mean yield stress for ASTM A615 Grade 60 reinforcing steel of 71,900 psi (Reference 12, p. 9-13), a Ø of 0.9, and an inside drywell radius of 36.5 ft x 12 = 438 in, the mean failure pressure based on hoop capacity is,

$$\frac{0.667 \times 71{,}900}{438''} \times 0.9 = 98.5 \text{ psi}$$ mean yield pressure based on hoop capacity

The vertical reinforcement in the drywell wall is #18 bars at approximately 18" spacing on each face, or

$0.44\ in^2/in$. Mean failure pressure based on longitudinal membrane capacity is therefore,

$$2 \times \frac{0.44 \times 71{,}900}{438"} \times 0.9 = 130 \text{ psi}$$

mean yield pressure based on longitudinal (vertical) capacity

The hoop capacity is controlling, giving a drywell concrete structure capacity of 98.5 psi. Since reinforcement at discontinuities and penetrations is typically designed to at least continue the basic strength of the shell, 98.5 psi is a reasonable estimate of mean yield failure pressure for the concrete portion of the drywell above the suppression pool.

The lower portion of the drywell is actually a double wall steel shell (ASTM A516 Grade 70) with connecting stiffeners and concrete fill. Hoop steel area can be found by subtracting the vent holes (ID = 27.5 in.) from the plate area,

$$\frac{(54 - 27.5)}{54} \times 2" = 0.98 \frac{in2}{in}$$

where 54" is the vertical spacing between vent holes and 2" is the combined thickness of inner and outer shells.

Using $.9 \times 0.98\ in^2/in$ to account for 10% stainless steel cladding on the exposed plate surface, the steel area becomes $0.88\ in^2$. Using 49.7 ksi (see "Drywell Head" this section) as the mean yield strength, the estimate of mean failure pressure capacity of the lower steel plate portion of the drywell is,

$$\frac{0.88 \text{ in}^2 \times 49700 \text{ psi}}{438} = 99.9 \text{ psi}$$ mean yield pressure based on steel hoop capacity

3.5 Selection of PNPP Mean Failure Pressures

The above summaries of previous drywell evaluations (Kuosheng and Grand Gulf) and the examination of some specific PNPP calculations and details, show that the drywell structure has higher pressure retaining capability than the steel containment. This is not surprising considering that the 30 psi internal design pressure for the drywell is twice that of the steel containment. A lower bound mean failure pressure (internal or external) for the drywell can be chosen based on the information presented above. More specifically, Table 1 gives an estimate of the pressure capacities of the various structural components of the drywell. For the cylindrical shell and roof the mean failure pressure of 67 psi from the Kuosheng analysis is used rather than the 98.5 psi calculated from PNPP specific data. This conservative pressure is used because the PNPP specific data did not include a detailed examination of all drywell shell discontinuities.

The mean failure pressures are considered to be proportional to the mean yield of the materials. Statistical descriptions of concrete strength and SA 516 Grade 70 plate strength are established in Section 4.0, Containment Failure Modes. ASTM A516 Grade 70 is considered to be identical to SA516 Grade 70. SA 240 Type 304 flange material on the drywell head is assumed to have the same ratio between mean yield and minimum yield as SA 516 Grade 70, i.e. 1.31. Reference 12 p. 9-13 gives the following statistical data on ASTM A615 Grade 60 reinforcing steel:

Specified minimum yield	=	60 ksi
Actual lower bound yield	=	67.3 ksi
Actual mean yield	=	71.9 ksi
Actual upper bound yield	=	75.1 ksi

For reinforced concrete, the strength of a particular element is generally more dependent upon reinforcing steel than on concrete strength. There are two reasons for this: (1) the steel is considered to carry all the tension loads (2) nearly all elements in bending are under-reinforced for both economical and design reasons. Concrete strength can be controlling in some cases such as in development of reinforcing steel or anchor bolts, and in members with high shear loads. For the statistical variation in strength of the upper Drywell wall and roof it should be sufficient to consider the properties of the reinforcing steel.

4.0 Containment Failure Modes

The following potential failure modes have been considered for the containment vessel: membrane failure, failure of the mat foundation, failure of the air locks and equipment hatch, thermal buckling of the suppression pool liner, failure at pipe penetrations, failure of the containment anchorage into the mat foundation, fracture due to existing weld defects, thermal buckling of the lower containment vessel, and thermal forces on stiffener #4 as a result of embedment into the annulus concrete. Each potential failure mode is discussed below.

Some components may be subjected to hydrostatic pressure in addition to gas pressure loadings. The hydrostatic pressure results from flooding of the containment vessel. Hydrostatic loading in addition to gas pressure loading will be evaluated on a sequence specific basis during the IPE.

Table 2 provides a summary of the mean value of yield and ultimate strength and their standard deviations for the dome and cylinder regions of the containment vessel. The mean value for the mat concrete ultimate compressive strength along with the standard deviation is also presented in Table 2 (References 17 and 18). For steel components, failure pressure is established by calculating the pressure producing the ASME Level D limit stress in the component. This pressure is set equal to a 5% probablility of failure. Pressures at any probability of failure level can then be determined by using the statistical equations explained in Item 7.0 of this report. The basic ASME Level D limit (Reference 9) for membrane stress intensities is 0.595 times the ultimate tensile stress of the material (0.595*Fu). This level of stress obviously has considerable factor of safety as compared to the the ultimate stress capability of the material; therefore it is reasonable to assign a 5% probability of failure to pressures producing ASME Level D limit stresses.

In the case of the concrete anchor the failure mechanism used to calculate capacity is based on the ultimate tensile strength of concrete which in turn is a function of the ultimate compressive tests of concrete specimens. In contrast to reinforced concrete behavior as a general case (see item 3.5) the strength of the concrete anchors are only indirectly affected by any reinforcing steel in the concrete, so that the best measure of strength is the ultimate strength of the concrete itself.

A summary of the pressures at 5%, 50%, and 95% probability of failure for the various potential failure modes is provided in Table 4.

4.1 Membrane Failure

The containment pressure/stress relationship is based upon a KSHEL axisymmetric computer analysis (Reference 18). The containment internal pressure required to cause a membrane stress equal to the ASME Level D limit stress is 65.10 psi for the dome knuckle, 86.77 psi for the dome apex, and 86.68 psi for the cylinder. The failure modes for the dome apex or knuckle would result in rupture of the steel plate and consequential loss of pressure through the rupture opening. The failure mode of the cylinder would result in rupture, progressing rapidly in size with rapid depressurization of the containment. Section 7.0 of this report compares expected opening sizes for the identified locations/modes of failure. See also Appendix B for more calculation details of the above pressures. The above pressures are set equal to a 5% probability of failure.

4.2 Mat Foundation Failure

The Kuosheng PRA considers both the shear failure and the flexural failure of the mat foundation. These failure modes are generally applicable to the Perry containment vessel because the concrete mat foundation and base liner provide the bottom closure to the hybrid containment vessel. The conclusion documented in the Kuosheng PRA are considered to be applicable to the Perry containment on the basis of the following comparison.

The design compressive strength for the Kuosheng containment concrete is 5000 psi at 28 days while the median compressive strength at 90 days is 5760 psi with a logarithmic standard deviation of 0.10. The design compressive strength for the Perry mat foundation was originally 3000 psi at 28 days and subsequently qualified as a 4000 psi 28-day mix and as a 5000 psi-90 day mix. However, further evaluation of the data available for the mat foundation concrete shows that by using the mix qualification defined by ACI 318 and ACI 214, an allowable concrete compressive strength of 5840 psi at 28 days may be calculated. The mean value for the mat foundation compressive strength is 6442 psi with a standard deviation of 448 psi.

The reinforcement for the Kuosheng containment vessel conforms to ASTM A615 Grade 60 for which the minimum specified yield stress is 60 ksi. The reinforcement for the Perry mat foundation also conforms to ASTM A615 Grade 60. The thickness and radius of the Kuosheng mat foundation is approximately 10 feet 0 inches and 70 feet 0 inches respectively based upon Figure 3-3 of Reference 1. The thickness and radius of the Perry mat foundation is 12 feet 6 inches and 68 feet 0 inches respectively.

The Kuosheng PRA concluded that shear and flexural failure of the mat was not determined to be an important failure mode. Based upon the similarities between the Kuosheng mat foundations and the Perry mat foundation, it is concluded that a shear or flexural failure of the Perry mat foundation is not a significant failure mode and additional strength evaluations are not required.

4.3 Equipment Hatch

The various components that comprise the equipment hatch are evaluated in Reference 14 for 45 psi internal pressure and dead load. To obtain the pressure at the ASME Level D limit stress, the 45 psi pressure used in Reference 14 was multiplied by the ratio of (stress at Level D limit stress at 45 psi pressure) to obtain 60.80 psi. See Appendix B for more detailed explanation of this calculation. The 60.80 psi pressure is set equal to a 5% probability of failure. The equipment hatch can experience hydrostatic pressure under post LOCA flooding.

4.4 Personnel Airlock

The various components that comprise the personnel airlock are evaluated in Reference 19. The acceptance criteria is ASME Code Service Level D limit stresses. The maximum pressure that the personnel airlock can resist was calculated to be 94.0 psi at the Service Level D limit stress allowable of 41.65 ksi. See Appendix B for additional explanation of the basis of the 94 psi pressure.

4.5 Liner Buckling

The PNPP suppression pool liner is SA 516, Grade 70 plate with a 10% thickness of stainless steel cladding conforming to SA 240, Type 304. The cladding meets the requirements of SA 264. The tee sections used for the base liner anchorage are SA 36. The minimum specified yield strength for the plate is 38 ksi for the SA 516 Grade 70 and 30 ksi for the SA 240, Type 304.

The Kuosheng liner conforms to SA 285 Grade A or C. The specified yield strengths for these materials are 24 ksi for Grade A and 30 ksi for Grade C. The probability of a failure resulting from the liner buckling was found to be negligible compared to that for other Kuosheng failure modes.

The following discussion from the Kuosheng PRA is considered to be applicable to the PNPP analysis and along with the fact that the PNPP liner has a greater yield strength than the liner for Kuosheng forms the basis for concluding that a failure resulting from liner buckling is also negligible for PNPP.

"The liner system consists of steel plate welded to plates or structural steel shapes embedded in the containment vessel concrete. Buckling of the liner could occur as a result of severe thermal gradients being developed between the liner and the concrete during an accident. The membrane compressive stresses in the liner result from the fact that the liner has a somewhat higher coefficient of thermal expansion than the concrete, and also heats up much more rapidly during the early transient portion of the accident. At the same time, the pressure increases in the containment, which tends to induce tension in the liner. Although the stress in the liner could be determined for given temperature and pressure time-histories, this was not considered necessary because even if the liner does buckle, the buckled configuration of the liner is deformation controlled by the thermal gradients and does not increase without bound as in load controlled buckling. Also the liner is in compression, which tends to limit any crack formation, and the internal pressure somewhat reduces the deformation expected. If the containment pressure rises quickly compared to the liner temperature, the liner may always be in tension, although possibly at proportionately lower stresses than at the ultimate load condition near failure. Even if the liner does buckle, this is not expected to lead to loss of liner integrity."

4.6 Pipe Penetrations

The pressure capacities for the pipe penetrations are summarized in Reference 18 for ASME Code Service Level C limits. The three most critical penetrations have been selected on the basis of the least pressure capacity. These are P123 - RCIC Pump Discharge and RHR Spray, P414 - Feedwater, and P205 - Fuel Transfer Tube.

Penetration P123 was previously investigated for a 45 psi pressure at which the calculated maximum stress was 40.2 ksi (Reference 18). To obtain the pressure at the ASME Level D limit stress, the 45 psi was multiplied by the ratio of (stress at Level D limit /stress at 45 psi pressure) to obtain 60 psi. See Appendix B for the calculation. This

60.0 psi pressure is set equal to a 5% probability of failure.

Penetration P414 was previously investigated for ultimate pressure capability corresponding to ASME Level D limit in Reference 18. The pressure was determined to be 54.7 psi using the ultimate material strength based on material certifications, and considering the increased stresses from the effects of a nearby penetration. See Appendix B for additional explanation. This 54.7 psi is set equal to a 5% probability of failure.

Penetration P205 was previously investigated in Reference 18 for ultimate pressure capability corresponding to the ASME Level D limit stress. This pressure was determined to be 58.9 psi using an elastic/plastic analysis. See Appendix B for additional explanation. This 58.9 psi pressure is set equal to the 5% probability of failure.

The leakage area associated with failure of each penetration is estimated to be 5.0 square inches and will increase with pressure. The maximum leakage area is estimated to be 30.0 square inches for each penetration. It should be noted that a failure of P205 could not result in the upper pool draining into the annulus between the shield building and containment vessel unless there is also a failure of the transfer tube or manual gate valve located below the pipe sleeve embedded in the drywell concrete at Elevation 647'-5 3/8". P205 is located at Elevation 636'-6 3/4" which is below Elevation 641'-0", the maximum water elevation during post LOCA flooding of the containment vessel.

4.7 Containment Vessel Anchorage into the Mat Foundation The containment vessel anchorage evaluation consists of an evaluation for the steel component of the anch rage (containment vessel) and of an evaluation for the concrete capacity of the anchorage.

The steel portion of the anchorage consists of 288 anchors. Each anchor is a 9.25" wide vertical extension of the containment vessel shell with bearing elements at the bott m of each anchor. Appendix B shows that the vertical extensions are critical and that the pressure correspond ng to ASME Level D limit stress is 104.98 psi. See Appendix B for additional explanation. The 104.98 psi pressu e is set equal to 5% probability of failure.

The concrete capacity for the containment vessel anchorage is based upon the methods defined in Reference 16 using the mean concrete ultimate compressive strength of 6442 psi. The resulting mean failure pressure capacity is 94.10 psi. A failure of the containment anchorage into the mat foundation could result in a gross failure of the containment vessel because the failure mechanism is the tension overstress of a concrete wedge. This failure would be sudden with no yielding and could be progressive around the containment as adjacent areas attempt to carry additional load not carried by the failed segment. The model used to calculate the 94.10 psi failure pressure and to predict the failure behavior is shown in Appendix A. As discussed there, this model is simplistic and conservative. A more accurate model might well show that the mean failure pressure is somewhat higher and that the failure mechanism involves the Shield Building structure. This more complex behavior might not be sudden and gross. The pressure at any probabiltiy of failure can be derived from the mean failure pressure of 94.10 psi using the statistical methods described in Section 7.0.

4.8 Fracture

A review of radiographics for the Perry Nuclear Power Plant found certain welds that contained potentially rejectable indications when evaluated according to the requirements of the ASME Boiler and Pressure Vessel Code Section III, Subsection NE-5320. References 21, 22, and 23 provide evaluations that document that failure will not occur during the plant design lifetime and a means to evaluate severe accident type loadings which were not a part of the plant design basis.

The containment meridional stresses are applicable to the weld defect problem (Reference 21). Based upon the limiting stress of 27000 ksi from Reference 23, the pressure capacity is 112.4 psi. The pressure capacity neglects dead load and is based upon the containment nominal 1 1/2 inch thickness. As a result, it is concluded that weld fracture does not significantly influence the results and conclusions of the report.

4.9 Thermal Buckling of the Lower Containment Vessel

The lower containment vessel is bounded on the outside surface by the annulus concrete. The shield building is located adjacent to the annulus concrete. The top of the annulus concrete is at Elevation 594′-4". As a result of this geometry and the fact that the steel containment vessel heats more quickly than the concrete, thermal circumferential compressive stresses in the containment vessel are caused by the constraint provided by the annulus concrete and shield building.

The existing design is based upon uncracked concrete (Reference 25). The safety factor for buckling is 2.03 for the accident condition which includes 15.0 psig internal pressure, 185°F design accident temperature, suppression pool hydrostatic pressure, and dynamic loads. The thermal loading is the most dominant loading for this design condition at this area of the vessel. Based upon this information and a thermal stress free temperature of 70°F, the estimated temperature at which buckling will occur is approximately 300°F.

Distinction is made between load-controlled buckling and strain-controlled buckling. Load-controlled buckling is characterized by continued application of an applied load in the post-buckling regime, leading to failure. Strain-controlled buckling is characterized by the immediate reduction of strain-induced load upon initiation of buckling, and by the self-limiting nature of the resulting deformations. Even though it is self-limiting, strain-controlled buckling must be avoided to guard against failure by fatigue, excessive strain, and interaction with load-controlled instability.

The thermal buckling limit for the lower containment vessel as described above is strain-controlled buckling and is a conservative estimate of the buckling capacity since it is based upon a stress limit instead of a strain limit. In addition, fatigue and interaction with load-controlled instability are not problems because the accident loading is not going to occur multiple times and because there are no significant loads that cause compression other than thermal.

4.10 Thermal Forces on Stiffener #4

Reference 20 defines a temperature and pressure limit based upon the stresses in stiffener #4 (embedded in the annulus concrete). The magnitude of the temperature and pressure is 280.4°F and 35.1 psi. The annulus concrete prevents displacement of this stiffener which results primarily from the thermal loads. The acceptance criteria for evaluation that was performed is based upon the ASME Service Level D limits which are a function of the minimum specified tensile strength of the steel.

Inelastic behavior could be considered as described in Reference 20 in order to obtain additional pressure capacity. For this reason it is concluded that the thermal stresses in the stiffener would not be a limiting condition for determining containment capacity.

5.0 Fragility Curves

As discussed in Section 4.0, the following failure locations have been identified for PNPP:

1. Dome Knuckle
2. Dome Apex
3. Cylinder

4. Personnel Air Lock

5. Equipment Hatch (Bolts)

6. Penetration P123

7. Penetration P205

8. Penetration P414

9. Anchorage, Steel

10. Anchorage, Concrete

For each steel component (Items 1-9), a 5% probability of failure pressure was established to the ASME Level D stress intensity limit. For anchorage, concrete, the mean failure pressure was calculated based on the mean ultimate strength of the containment mat concrete.

The fragility calculations are based on the assumption that the containment strength has a log-normal distribution, since this distribution, which does not permit negative values and has a positive skewness, has been shown to be descriptive of variation of material properties.

The cumulative log-normal distribution is described by the equation (Reference 1):

$$Pf = \text{Prob}\ (P \leq p) = \emptyset[\ln(p/p^{v})/\beta] \qquad (1)$$

in which

Pf probability that failure occurs at a pressure $P < p$

β standard deviation of the natural logarithm of the pressure capacity p, or logarithmic standard deviation of p

p^{v} median pressure capacity

Ø(.) cumulative distribution function for a standard normal random variable

βc the log standard deviation of capacity, is a combination of the log standard deviation of material properties βs and the log standard deviation due to modeling uncertainty, βm and is calculated from the standard relationship

$$\beta c = \{\beta_s^{\ 2} + \beta_m 2\}^{0.5} \qquad (2)$$

βs is calculated from the coefficient of variation δp of material properties using the relationship

$$\beta s = \{\ln (\delta_p^{\ 2} + 1)\}^{0.5} \qquad (3)$$

for the log-normal distribution (Reference 12)

The median pressure capacity p^{v} is calculated from the mean pressure capacity p(mean) and the log standard deviation bc using the relationship

$$p^{v}(\text{median}) = p(\text{mean}) * e(-1/2 + \beta c^{2}) \qquad (4)$$

for the log-normal distribution (Reference 12). It should be noted that the median of a log-normally distributed variable is always less than the mean.

βm, the log standard deviation for modeling uncertainty was estimated on the basis of similarity of PNPP failure modes/locations to failures considered in the Kuosheng PRA (Reference 1).

Table 4 lists, for each failure mode/location, the pressure corresponding to 5%, 50% and 95% probability of the failure as well as the log standard deviation of capacity used in calculating the pressures. The log standard deviation of strength and the log standard deviation of modeling, which form the basis for the log standard deviation of capacity, are also listed.

Tables 5 through 14 document the calculations for the fragility curves for the ten failure modes/locations. Table 15 shows the calculation of the containment composite fragility curve, based on the relationship,

$$P_{fc} = 1 - \Pi_i (1 - P_{fi}) \qquad (5)$$

in which

P_{fc} = probability of containment failure

P_{fi} = probability of failure mode/location i

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

Equation (5) is based on the assumption of statistical independence of the individual failure modes. While it is highly probable that the failure modes are dependent because of commonalty of material vendor, location and fabricator, the assumption of independence is conservative, i.e., it gives higher failure probabilities than the assumption of dependence.

Fragility curves for each of the ten failure modes/locations and the containment composite fragility curve are provided as Figures 13 through 23. It should be noted that while the fragility curves depict the expected behavior of each component and the containment as a composite over a wide range of containment pressures, the curves should be considered valid only over a limited range around the median capacity (i.e., the lower bound of failure pressure is in the range of the 1st - 5th percentile).

6.0 Leakage

6.1 Leakage Concerns

Leakage at pressures below mean yield pressures is a concern for penetrations of the containment vessel and the drywell where the pressure boundary integrity is dependent upon special details and materials such as elastomer seals or convoluted bellows. Components within this general description are 1) personnel airlocks, 2) equipment hatches, 3) drywell head, 4) mechanical (piping) penetrations, and 5) electrical penetrations. In addition to high pressures, high temperatures can be a contributing factor to leakage. Degradation of materials under elevated temperatures is discussed in section 8.

Deflections and stresses due to temperature and pressure induced relative movements is also a concern, particularly at piping penetrations which physically connect shield building, containment vessel and the drywell and must therefore accommodate the relative movements of these structures.

Per Reference 7, the design leakage rate of the containment vessel is 0.2% of contained volume over 24 hours at a pressure of 11.31 psi. The allowable test leakage rate for the containment vessel is 0.75 times this amount. The design leakage rate of the Drywell is given by:

$A/(K)^{0.5} = 1.68\ ft^2$

where A = flow area of leakage path

and K = geometric friction loss coefficient

The allowable test leakage rate for the drywell is one tenth of this or $0.168\ ft^2$.

6.2 Personnel Airlocks

The personnel airlocks for the containment vessel and drywell are shown in

Figures 2 and 11. The drywell has one personnel airlock and the containment vessel has two. The potential leakage path is around the moveable doors. Each airlock has two doors which seal by inflatable seals. Details of door size, door design, and seals for PNPP airlocks are very similar to the Grand Gulf personnel airlock which is analyzed for leakage in Reference 2, Appendix F to Appendix B. This Reference assumes that the seals fail to function or are deflated due to temperature degradation giving a leakage area for each airlock of 47.14 in^2. The same Reference also calculates a leak area of 125.70 in^2 if seals blow out. This seems like an extremely remote possibility for a number of reasons: 1) there are double seals on each end of the airlock (total = 4). 2) the seals are tested by introducing 30 psi pressure between seals, and again by a 34.5 psi test of theairlock itself. 3) the long sides of the rectangular doors have door stops which would prevent seals from blowing out even if they were blown loose. 4) if there were a pressure spike capable of blowing out the twin seals on one end of the airlock, the pressure build-up would be relatively slow within the airlock (as compared to the spike) with time for the initiating pressure to be dissipated by other means such as clearing of vent holes in the suppression pool.

In conclusion the potential leakage area for personnel airlocks is estimated at 47 in^2 based on non-functioning seals on both ends of the airlock.

6.3 Equipment Hatches

Containment vessel and drywell equipment hatches are shown in Figures 3 and 12. The removable hatch doors are attached by bolted flanged joints with o-ring seals. The bolts are preloaded to compress the seals and to limit flange separation under the design pressures. Even if the bolts had no preload and assuming no seals, leakage area due to flange separation would be very small as illustrated below:

flange separation assuming 6" long bolts and pressure of 87 psi giving a bolt stress of 49 psi,

$$\text{elongation} = \frac{49 \text{ ksi}}{29 \times 10^3 \text{ ksi}} \times 6" = 0.01 \text{ in}$$

In practice flanges rotate so that the gap is smaller than the bolt deflection. Also, the compressible seals provide effective sealing under such small flange deflections. Reference 1, Table 9-2 and Reference 2, Appendix F to Appendix B also indicate very small leak areas for equipment hatches under mean yield pressures.

In conclusion we can say that the potential leakage area for the equipment hatches is negligible.

6.4 Drywell Head

The drywell head is shown in Figure 10. All of the above general discussion on equipment hatches as to the improbability of leakage applies to the drywell head. We can conclude that the drywell head has insignificant leakage under mean yield pressures.

6.5 Electrical Penetrations

Figure 5 shows electrical penetrations through the containment vessel and the drywell.

The electrical penetrations used in the containment vessel were subjected to a number of environment qualification tests including a simulated accident test and a simulated 40 year aging test. (References 32, 33) After each test the penetrations were tested for leakage. The results of these tests are shown in Table 16. These tests indicate that leakage through electrical penetrations in the containment vessel is very unlikely.

The drywell penetrations were also tested for simulated aging and accident conditions. (Reference 34). Very minor leakage occurred after the tests as shown on Table 16. See item 8.4 for a discussion of the effects of temperature on electrical penetrations and the potential leak areas.

6.6 Mechanical (Pipe) Penetrations

Figure 4 shows pipe penetrations through the drywell, containment vessel and shield building. For the purposes of leakage considerations we can categorize these as follows.

1. No Leakage

These penetrations pass through the containment vessel and shield building only, with the guard pipe anchored to the containment vessel and the process pipe anchored to the guard pipe nearby. These anchor configurations are steel-to-steel attachments forming a pressure boundary. Moveable components are at the shield building and are not part of the containment pressure boundary. The majority of penetrations are of this type.

2. Leakage at Containment Vessel only:

   These penetrations pass through the drywell, containment vessel and shield building. The guard pipe is anchored at the drywell, while the process pipe is anchored to the guard pipe outside the shield building. A potential leakage path exists between the guard pipe and the guide rings at the containment vessel. Under design conditions the pressure boundary is maintained at the containment vessel by convoluted bellows, inside and outside the containment vessel. Should both bellows fracture, leakage would occur at the gap between the guard pipe and the guide rings. There are 9 penetrations of this type. The data below show the penetrations, the size of the guard pipe and the leak area if bellows were to completely rupture.

Penetration Data - Leakage at Containment Vessel Only

Potential Penetration No.	Designation	O.D. of Guard Pipe (in)	Gap Around Guard Pipe (in)	Leak Area (in2)
P122	Main Steam	38	1"	121
P123	RCIC Pump Discharge	18	0.25"	14
P124	Main Steam	38	1"	121
P131	RWCU Pump Suction	18	0.25"	14
P415	Main Steam	38	1"	121
P416	Main Steam	38	1"	121
P421	RHR Shutdown Suction Supply	32	1"	102
P422	RHR and RCIC Steam Supply	22	1"	71
P423	Main Steam Drain	14	0.25"	11

3. Leakage at Containment Vessel and Drywell:

These penetrations are through drywell, containment vessel and shield building. The guard pipe and process pipe are anchored at the shield building, with sliding details at the containment vessel and drywell. A potential leak path exists at the containment vessel and at drywell. Under design conditions, convoluted bellows maintain the pressure boundary at both drywell (single bellows) and containment vessel (bellows inside and outside). Leakage would occur if bellows fracture. There are 2 penetrations of this type. The data below shows the penetrations, the size of the guard pipe and the leak area if the bellows were to completely rupture.

Penetration Data - Leakage at Containment Vessel and Drywell

Penetration No.	Designation	O.D. of Guard Pipe (in)	Gap Around Guard Pipe (in)	Leak Area (in^2)
P121	Feedwater	32	1"	102
P414	Feedwater	32	1"	102

Two critical Containment Vessel bellows were investigated under the Hydrogen Ignition Evaluation (Reference 8). These two bellows were shown to have a minimum factor of safety on critical buckling of 5.3 under 45 psi pressure. It should be noted that buckling would only be possible for the bellows inside the containment vessel. If this bellows were to fail, the pressure boundary would still be maintained by the bellows outside the tainment vessel. The pressure direction on this outside bellows would not produce buckling. We can conclude that bellows cracking and consequential leakage at the containment vessel wall is extremely unlikely at the mean pressure capacities identified in section 4.0.

The two penetrations through the drywell with potential leakage in the event of bellows failure were not analyzed for ultimate pressure capability. In general they should have similar pressure retaining capacity to the bellows at the containment vessel and therefore failure at the mean drywell pressure of 67 psi is extremely unlikely.

6.7 Summary of Leakage Concerns

Table 21, "Leakage From Other Than Structural Failure" lists potential leak areas, cause of leakage, effects of temperature and a judgmental estimate of probability of leakage. At temperatures below 300°F the probability of leakage is very small. At temperatures above 300°F the probability of leakage for some components becomes difficult to judge because of either degradation of seal materials or effects of differential thermal expansion (mechanical penetrations).

The effects of pressure are not included in Table 21. In general we might expect some increase in probability of leakage with higher pressures due to greater probability of seal blow-out or of bellows fracture in the case of mechanical penetrations. The pressure effects are not considered to be very significant in comparison to temperature and therefore are not included in the judgmental probabilities of Table 21.

6.8 Attenuation of Radionuclides by the Shield Building

The shield building provides another barrier which may attenuate the radionuclides released subsequent to the initiation of leakage from the containment vessel. The shield building provides a relatively leak tight structure, for HVAC purposes, so that the annulus exhaust gas treatment system can be used to minimize the escape of the radionuclides to the environment by maintaining the annulus air space at a slight negative pressure. The design value for the negative pressure is approximately 0.01 psig.

Should a positive pressure develop in the annulus region, the pressure boundary would include the concrete shield building and the Bisco silicone foam used to seal the annular space between the pipe sleeve penetrations embedded in the concrete and the pipes through the penetrations. If leakage is initiated through the penetrations, the escaping gases could enter the 3 inch gap that is located between the shield building and adjacent structures. The waterproofing membrane is located at the bottom of this gap and is the upper boundary to the porous concrete and the groundwater. An architectural detail consisting of neoprene expansion joint material forms the top boundary of the gap. The auxiliary building steam tunnel will also be a potential leakage path from the gap located between the buildings. The following list of shield building penetrations provides potential paths that the radionuclides could use to enter the adjacent structures:

MPL	ELEVATION	AZIMUTH	FROM	TO	ROOM
1PRB2003	603'-6"	62	RB	AB	LPCS
1PRB4001	647'	58	RB	AB	LPCS
1PRB4002	647'	62	RB	AB	LPCS
1PRB4003	647'	66	RB	AB	LPCS
1PRB4005	647'	298	RB	AB	HPCS
1PRB4006	647'	302	RB	AB	HPCS
1PRB2005	603'-6"	254	RB	IB	
1ERB4019	631'-6"	223	RB	IB	
1PRB3091	641'-6"	150	RB	IB	

1ERB3006	641'-6"	235	RB	IB
1ERB4005	643'-3"	240	RB	IB
1PRB6002	678'	153	RB	IB
1PRB6003	678'	150	RB	IB
1PRB6005	678'	262	RB	IB

7.0 Conditional Probability Matrix

Failure locations are shown in Table 4. For failure locations 3, 9 and 10, the expected type of failure is gross rupture. For failure locations 1 and 2 the expected type of failure is rupture. For failure locations 4 through 8 the expected type of failure is leakage at lower pressures and either rupture or gross rupture at higher pressures. "Higher pressures" can be defined as any pressure above median failure pressure for that component; since, at these pressures, large areas of the structure are at high stress level tending to cause the rapid propogation of any fracture or deformation initiating the failure. Summarizing in tabular form:

	Expected Type of Failure(3)		
Mode / Location	Leakage	Rupture	Gross Rupture
1. Dome knuckle		X	
2. Dome apex		X	
3. Cylinder			X
4. Personnel airlock	(1)	(2)	
5. Equipment hatch	(1)		(2)
6. Penetration P123	(1)	(2)	
7. Penetration P205	(1)	(2)	

8. Penetration P414	(1)	(2)	
9. Anchorage, steel			X
10. Anchorage, concrete			X

(1) Expected type of failure at median failure pressure or below.

(2) Expected type of failure above median failure pressure.

(3) Leakage is defined as an area of approximately 0.1 ft^2. which results in slow depressurization. Rupture is defined as an area from 0.1 to 7.0 ft^2. Gross rupture is defined as an area of >7.0 ft^2.

Based on the above data for individual modes/locations, the composite conditional probability matrix of pressure and expected type of failure can be made:

Conditional Probability Matrix

Pressure, Psi	Leakage	Rupture	Gross Rupture
60	.85	.07	.08
65	.83	.08	.09
70	.79	.11	.10
75	.40	.50	.10
80	0	.58	.42

The conditional probability matrix was developed by first separating, at each pressure, the failure locations that would exhibit leakage, rupture and gross rupture. If the probability of failure is represented by P_{fL},

P_{fr} and P_{fGR} respectively,

$$P_f = 1 - (1-P_{fL})(1-P_{fR})(1-P_{fGR}) \ldots\ldots\ldots\ldots\ldots (1)$$

$$P_{fL} = 1 - \Pi(1-P_{fi}) \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (2)$$

$$P_{fR} = 1 - \Pi(1-P_{fj}) \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (3)$$

$$P_{fGR} = 1 - \Pi(1-P_{fk}) \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (4)$$

where i, j, and k are the failure locations at which leakage, rupture and gross rupture cause failure respectively at the presure being evaluated. For instance, at 40 psi, locations 4, 5, 6, 7, and 8 exhibit failure by leakage, locations 1 and 2 by rupture, and locations 3, 9, and 10 by gross rupture.

The conditional probability of leakage, rupture, and gross rupture are then obtained by the application of Bayes' theorem.

For leakage,

$$P(L/f) = \frac{P(f/L)(P_{fL})}{P(f/L)(P_{fL}) + P(f/R)(P_{fR}) + P(f/GR)(P_{fGR})} \quad (5)$$

Since the probability of containment failure given leakage, rupture or gross rupture is one by definition,

$$P(f/L) = P(f/R) = P(f/GR) = 1.0$$

$$P(L/f) = \frac{P_{fL}}{P_{fL} + P_{fR} + P_{fGR}} \qquad (6)$$

Similarly,

$$P(R/f) = \frac{P_{fR}}{P_{fL} + P_{fR} + P_{fGR}} \qquad (7)$$

and $$P(GR/f) = \frac{P_{fGR}}{P_{fL} + P_{fR} + P_{fGR}} \qquad (8)$$

8.0 Effects of High Temperatures

The effects of high temperatures on concrete, reinforcing steel, steel plate and seal materials are discussed below.

8.1 Effects on Concrete

Reference 28 addresses effects of temperatures to 1112°F (600°C) on concretes with limestone and dolomite aggregates. Reference 11 addresses effects of heating to 1600°F (871°C) on a concrete with a "carbonate" aggregate. There are a number of differences in the testing methods between Reference 28 and 11. Reference 11 investigates specimens that are stressed to various levels during heating and then loaded to failure, as well as heated specimens that are stored at room temperature for 7 days before being loaded to failure. Reference 28 only investigates unstressed specimens that have been heated and allowed to cool before testing. Reference 28 also looks at various exposure times: 48 hours, 1 month, and 4 months. There are significant differences in residual concrete strengths that result from the testing methods and also from the different aggregates. Based on short term exposure only (48 hours or less) and taking the lowest values from both references (unstressed specimen during heating, then cooled and tested to failure) some estimates of concrete strength at high temperatures are presented in Table 17. The strength of concrete exposed to these temperatures for longer periods of time could be lower than the values in Table 17.

8.2 Effects on Reinforcing Steel

Data on the effects of elevated temperature specifically on ASTM A 615 Grade 60 reinforcing steel was not obtained; however, some general information on carbon steels is provided by References 29 and 30. Table 18 shows effects of elevated temperatures on a number of steels including a high strength steel SAE 1040 (Reference 29). This steel has a high carbon (C = 0.41) and manganese (M_n = 0.77) content similar to what might be expected for reinforcing steel.Reference 30 provides information on common structural steel, ASTM A 36 (C = 0.26 max.), for temperatures to 1800°F. The PNPP values are based on the values for SAE 1040 except that above 1000°F the curve is extrapolated to follow the shape of the curve for ASTM A36.

Reinforced concrete is a composite material which could be affected by loss of strength of reinforcing steel or concrete. In general, the strength of reinforcing steel is more significant than concrete because all tension loads are assumed to be carried by the steel and because nearly all bending components (beams, walls, slabs, etc.) are under-reinforced for both economical and design reasons. Concrete strength is significant as affecting development of reinforcing steel stresses at bar ends and splices.

8.3 Effects on Steel Plate

Containment/drywell hatches, airlocks and the drywell head are mainly composed of SA 516 Grade 70. Reference 9 Table I-1.1 gives allowable stresses for this steel for temperatures from 100 to 700°F. It is reasonable to assume that the % loss of yield stress is very similar to the % loss of allowable stress. For higher temperatures than 700°F we can use References 29 and 30 which were previously discussed under effects on Reinforcing Steel. Estimated strengths of SA 516 Grade 70 are presented in Table 19.

The PNPP values follow Reference 9 up to 700°F since this is data specifically for ASTM A 516 Grade 70. Above 700°F the values are generally based upon the lower of References 29 and 30.

8.4 Effects on Seal Materials

Airlocks and Equipment Hatches

The seals for personnel airlocks and equipment hatches for both the containment and drywell are ethylene propylene diamene monomer (EPDM). We do not have specific chemical and physical properties for these seals, however the project specifications required that the drywell seals be able to withstand 330°F for the airlocks and 285° for the hatch. The design accident temperatures for the containment seals was not as high (185°F); however it seems likely that the same seal material was used for containment and drywell airlocks/hatches because these components are by the same manufacturers and are nearly identical as far as sealing details.

Reference 2 Figure 2.14 shows "seal life" in hours for a number of seal materials including ethylene propylene. Data from the graph of ethylene propolene is shown in Table 20.

Reference 2, Appendix A to Appendix B reports testing of two models of sealing configurations: neoprene o-rings in trapezoidal grooves and silicone compound in tongue and groove. The neoprene o-ring, which has the same seal life curve as EPDM, was tested at both room temperature and at 420°F. Some small leakage was detected at 420°F indicating some seal distress. The seals were also deformed by the high temperature test even though the leakage was very small.

In summary, for PNPP seal materials, we might expect the seals to significantly change in physical characteristics above 400°F. Whether this change results in seal leakage or seal blow-out cannot be determined from the information just presented. Even with specific information about the characteristics of seal material at temperatures above 400°F, the function of each seal detail could still be highly dependent on the magnitude of the pressure, the deflection of the flanges and the particulars of the configuration itself. For example the behavior of the o-ring seals on the equipment hatches would not necessarily be the same as the inflatable seals or the personnel airlocks even if the temperatures, pressures and flange deflections were identical.

Electrical Penetration Seals

There are 40 electrical penetrations through the containment vessel (see Figure 5). As indicated in Table 16, these have been tested at high temperature and pressure with negligible subsequent leakage. The penetrations have four o-ring seals, two of EPR material and two to silicon. These seals should perform well up to 300°F. Even if the seals were degraded, the potential leakage area would be small and is estimated as 200 in^2 for all 40 penetrations.

The drywell has 52 electrical penetrations. See Figure 5. These penetrations are sealed on the outside of the drywell wall by neoprene material under clamping pressure. These seals are designed as fire barriers and have been tested to withstand a 3 hour fire per U.L. rating and a 5 hour fire per American Nuclear Insurers. In addition, fire proofing material has been added within the electrical boxes between the neoprene seals and the drywell wall. The sealing material is actually designed to swell under increasing temperature up to 600°F to compensate for the possibility of degraded cable insulation. The potential leak area can be conservatively taken as the inside area of a 5" conduit, 19.6 in^2/penetration, or a total about 1000 in^2.

9.0 Summary and Conclusions

The composite fragility curve for the containment indicates a pressure capacity of 53.5 psi at a 5% probability of failure and a pressure capacity of 64.3 psi at a 50% probability of failure. The most critical failure mode was found to be a failure of the containment vessel plate adjacent to penetration P414. A pressure capacity of 57.4 psi at a 5% probability of failure with a logarithmic standard deviation of 0.151 was estimated for this failure mode. The next most likely failure mode is expected to be a failure in the containment vessel plate adjacent to penetration P205 with a pressure capacity of 58.1 psi at a 5% probability of failure with a logarithmic standard deviation of 0.135.

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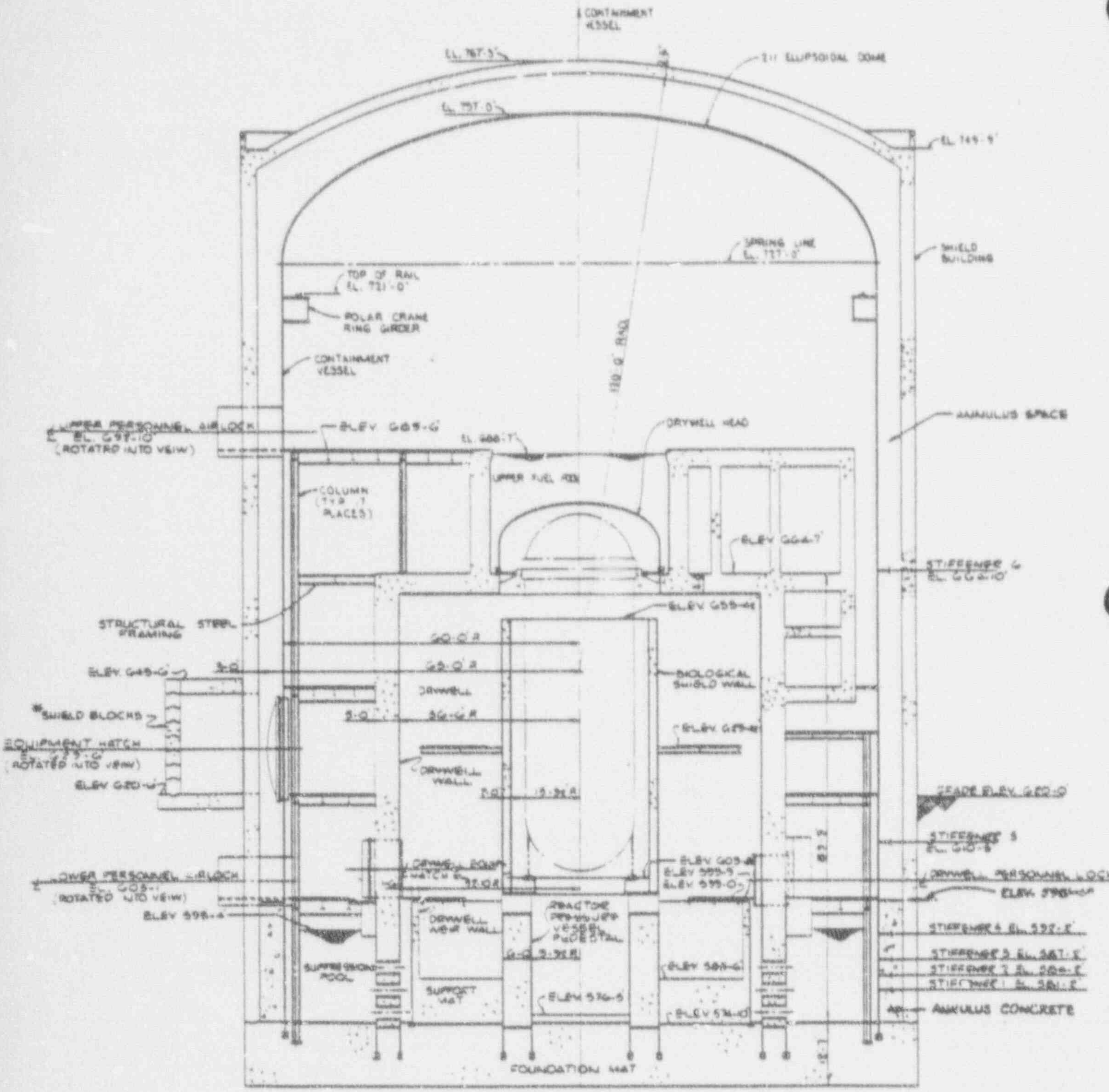
Figures

Tables

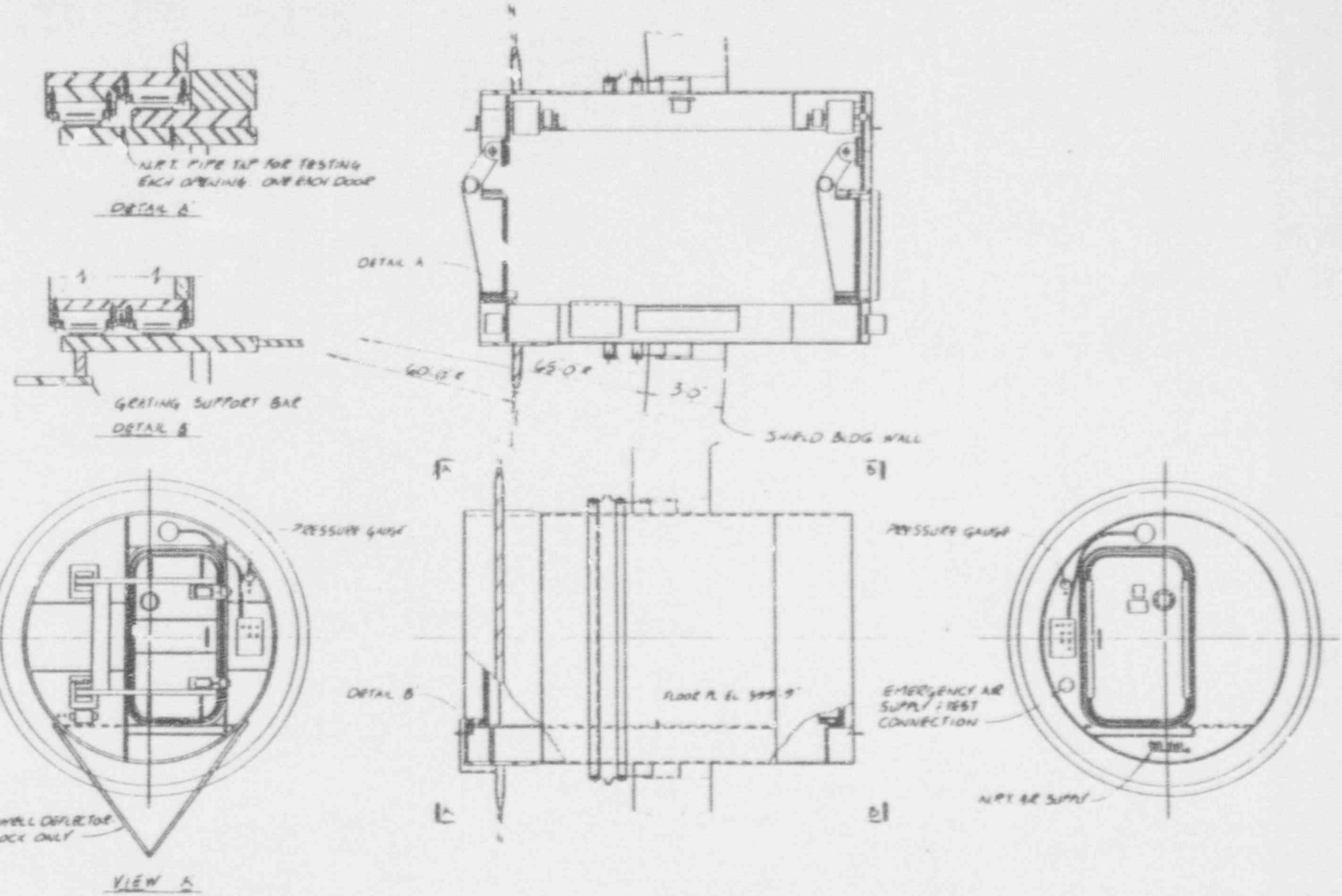
Appendix A

Appendix B



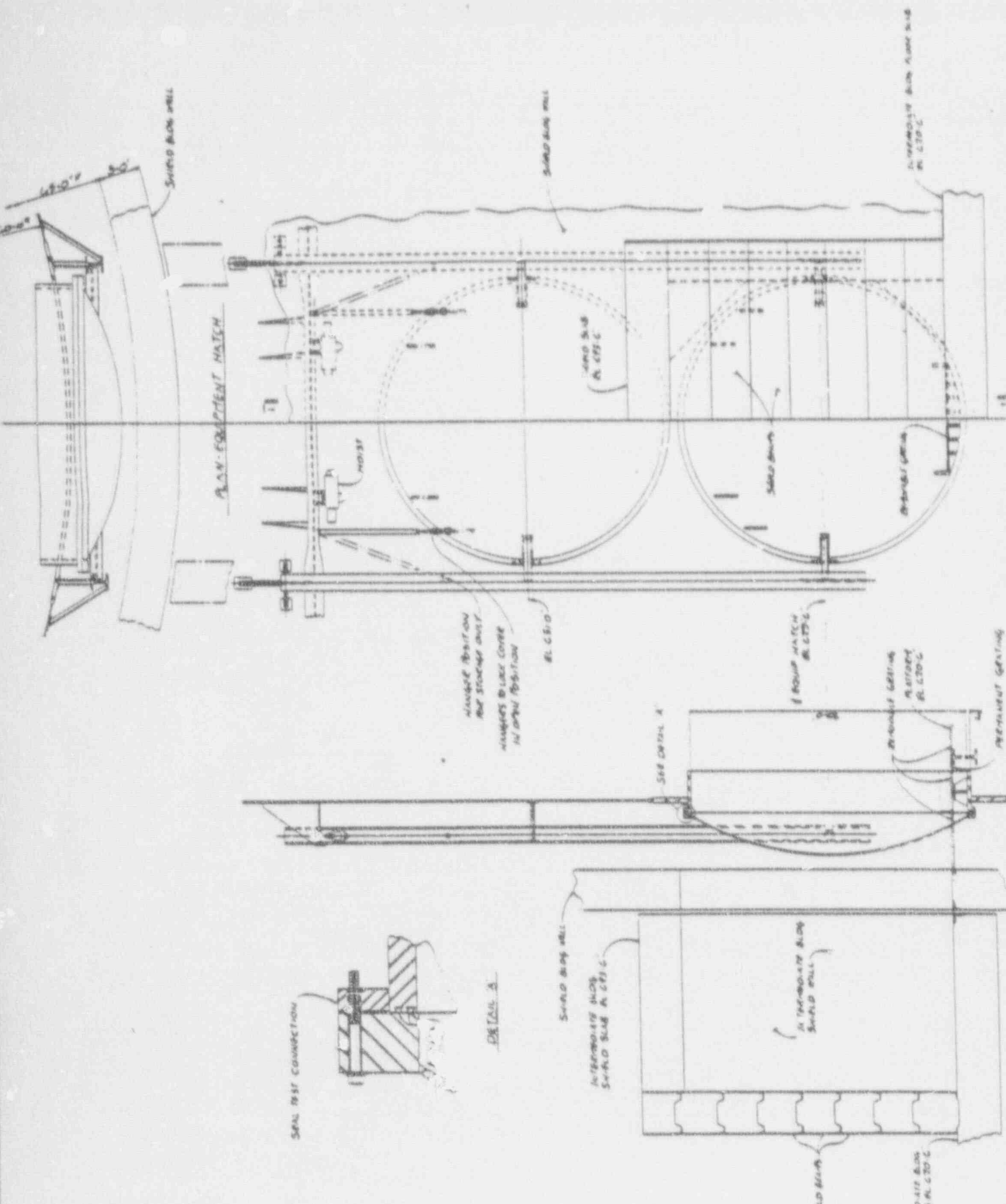
PNPP REACTOR BUILDING

FIGURE 1

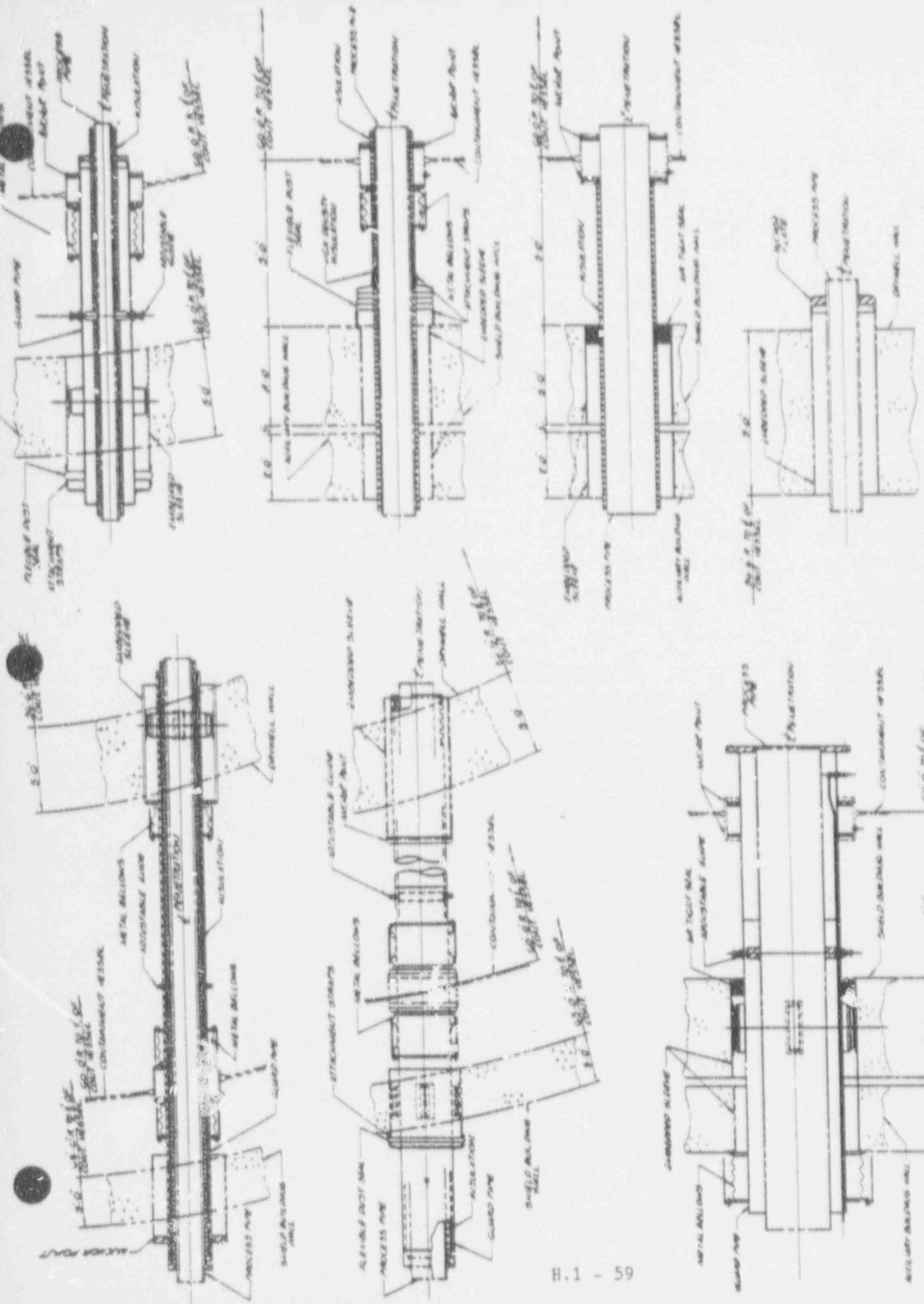


CONTAINMENT PERSONNEL AIRLOCK

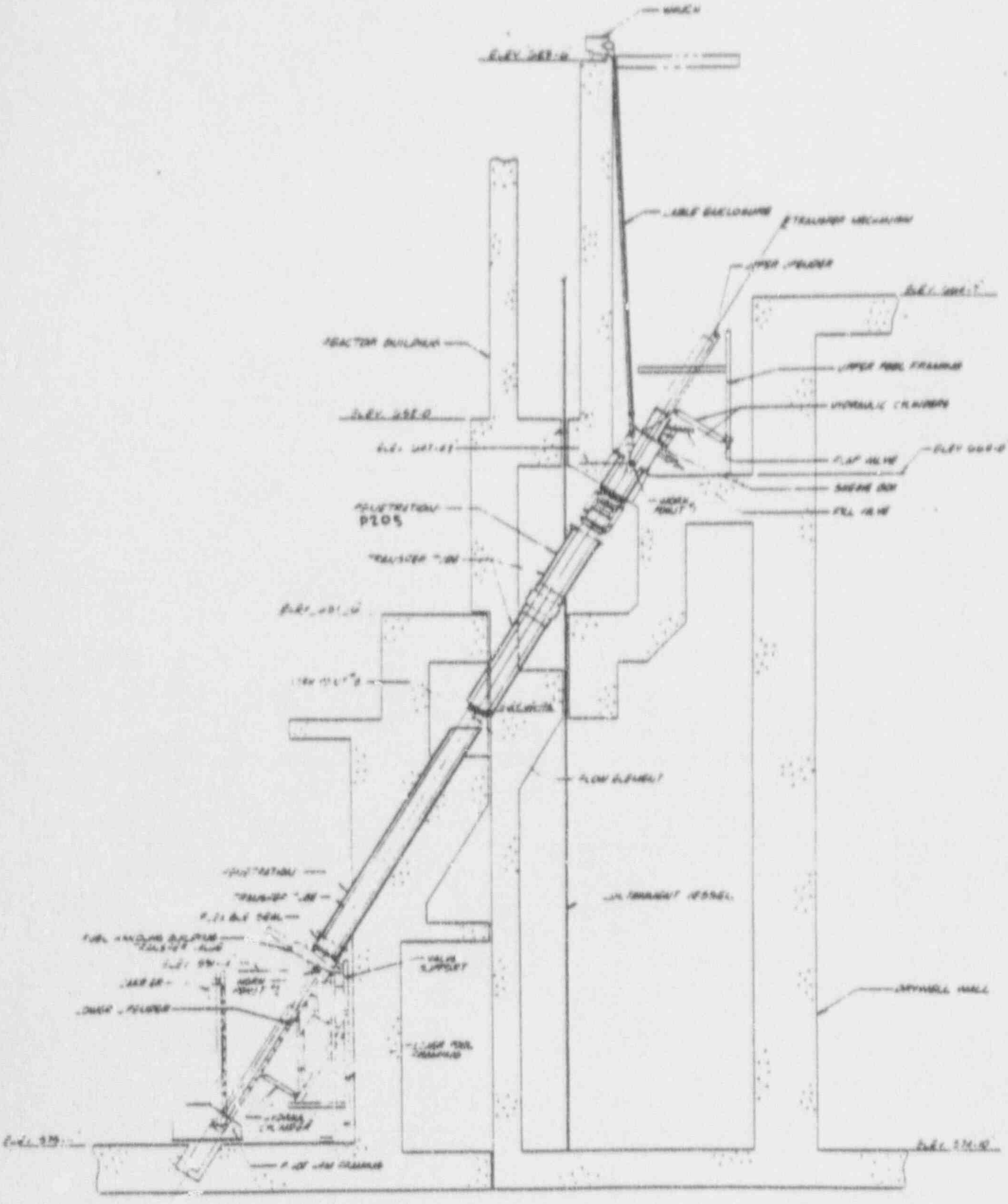
FIGURE 2



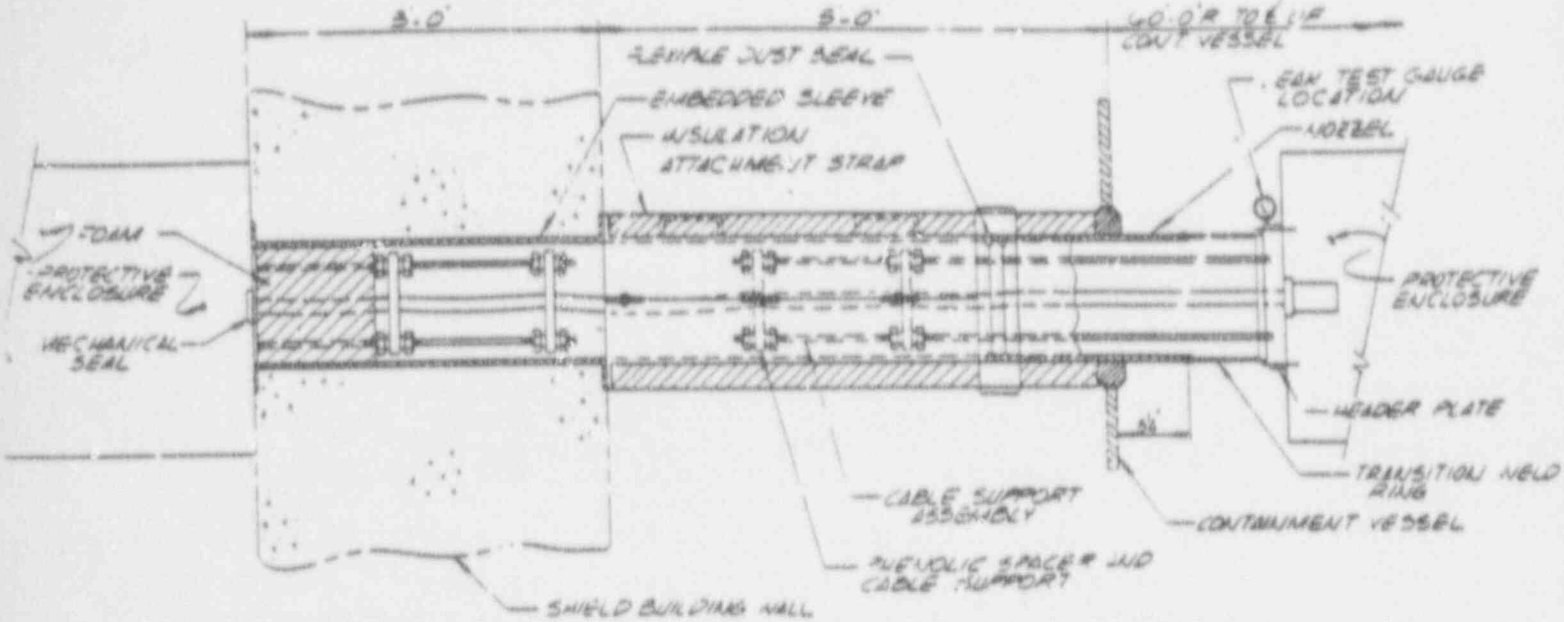
CONTAINMENT EQUIPMENT HATCH
FIGURE

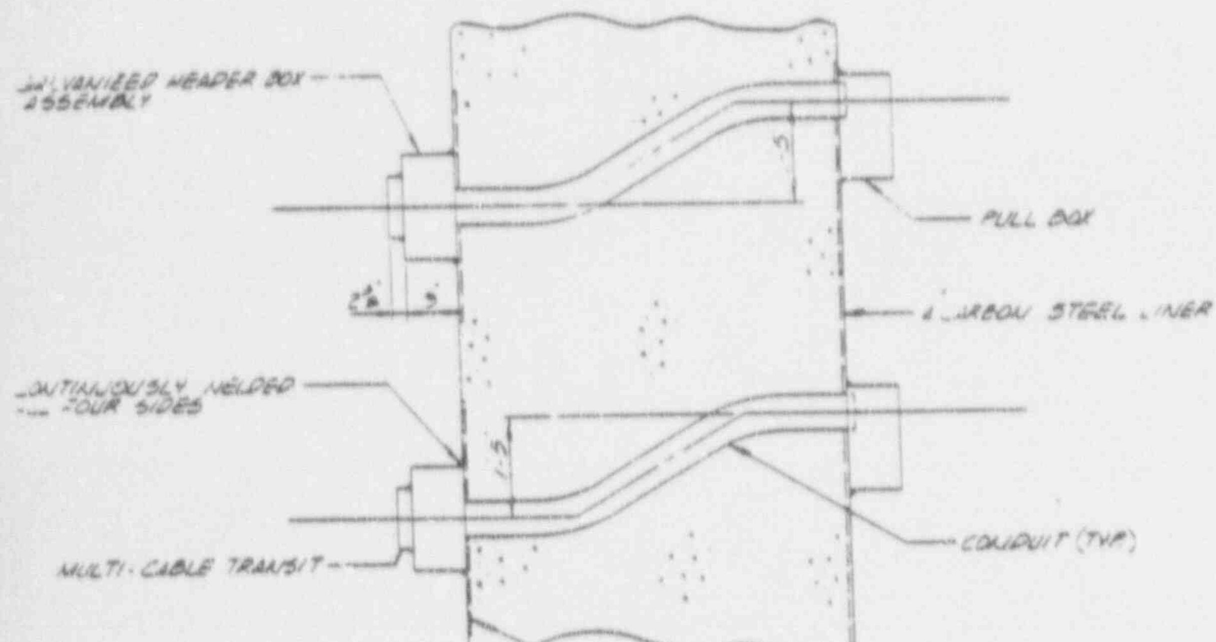


TYPICAL MECHANICAL PENETRATION
FIGURE 4

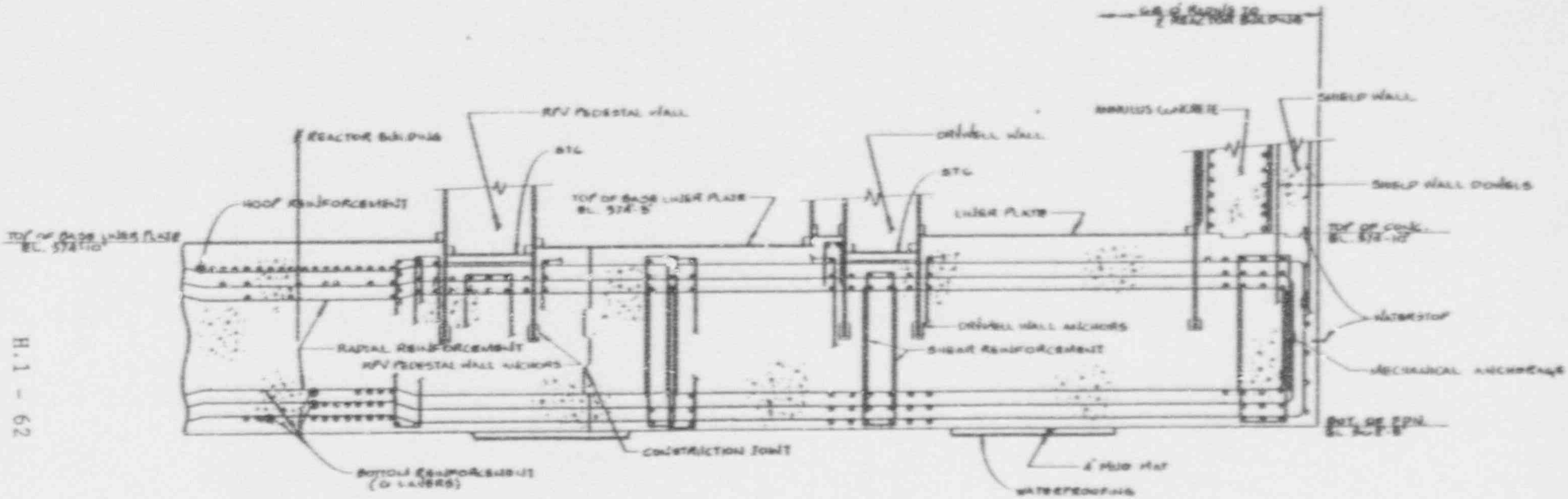


FUEL TRANSFER TUBE PENETRATION
FIGURE 4A



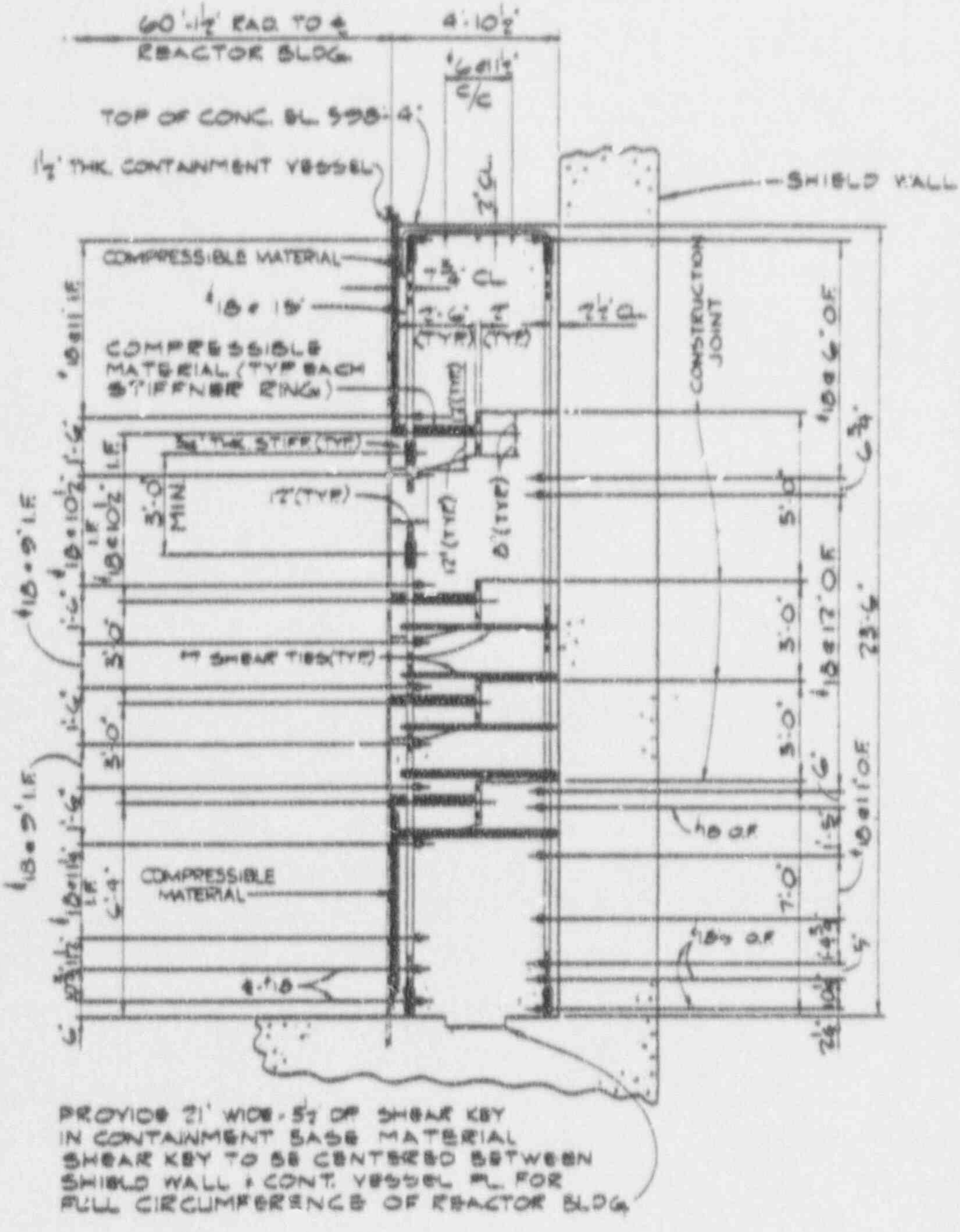


ELECTRICAL PENETRATIONS
FIGURE 5

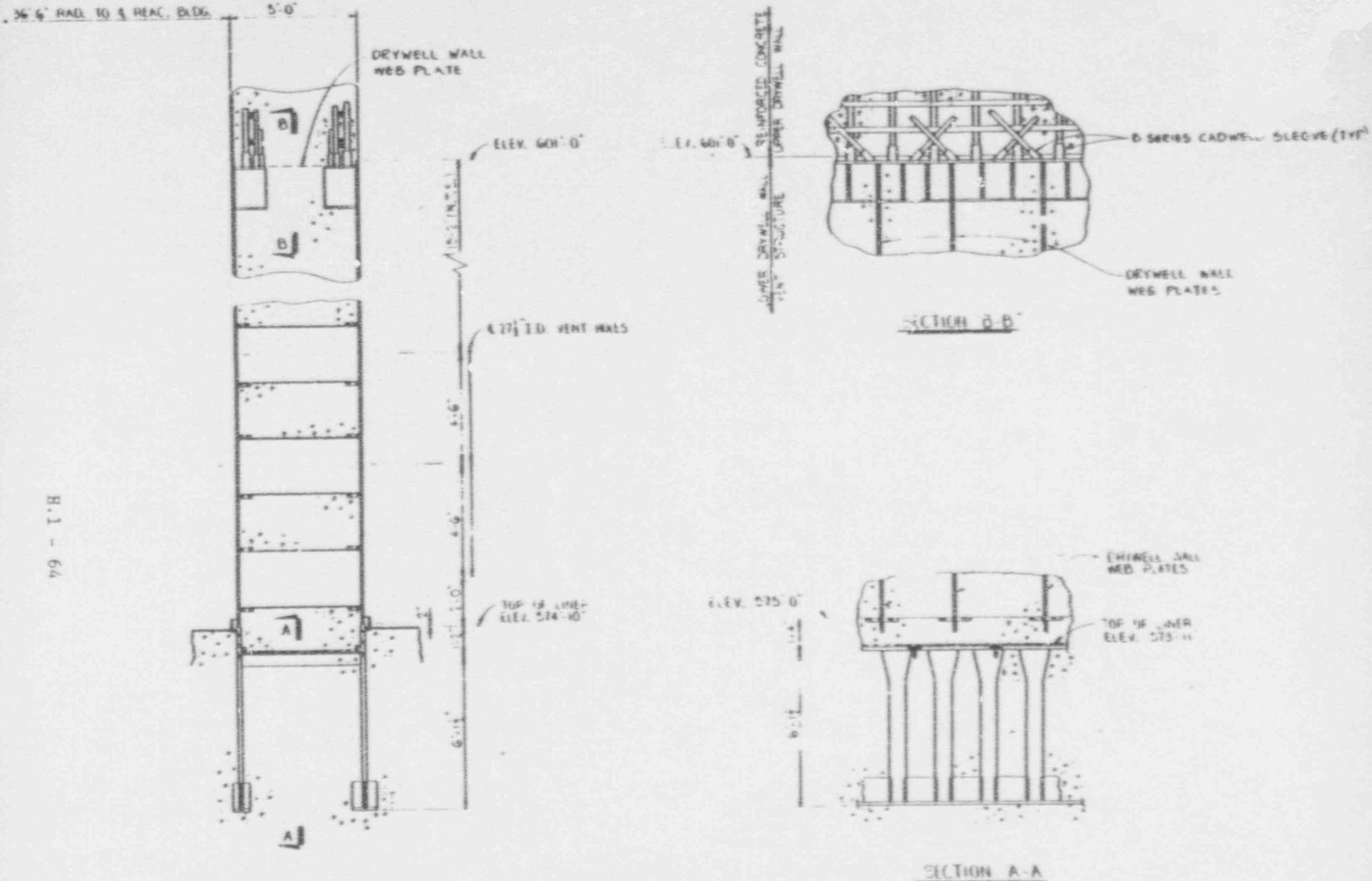


FOUNDATION MAT

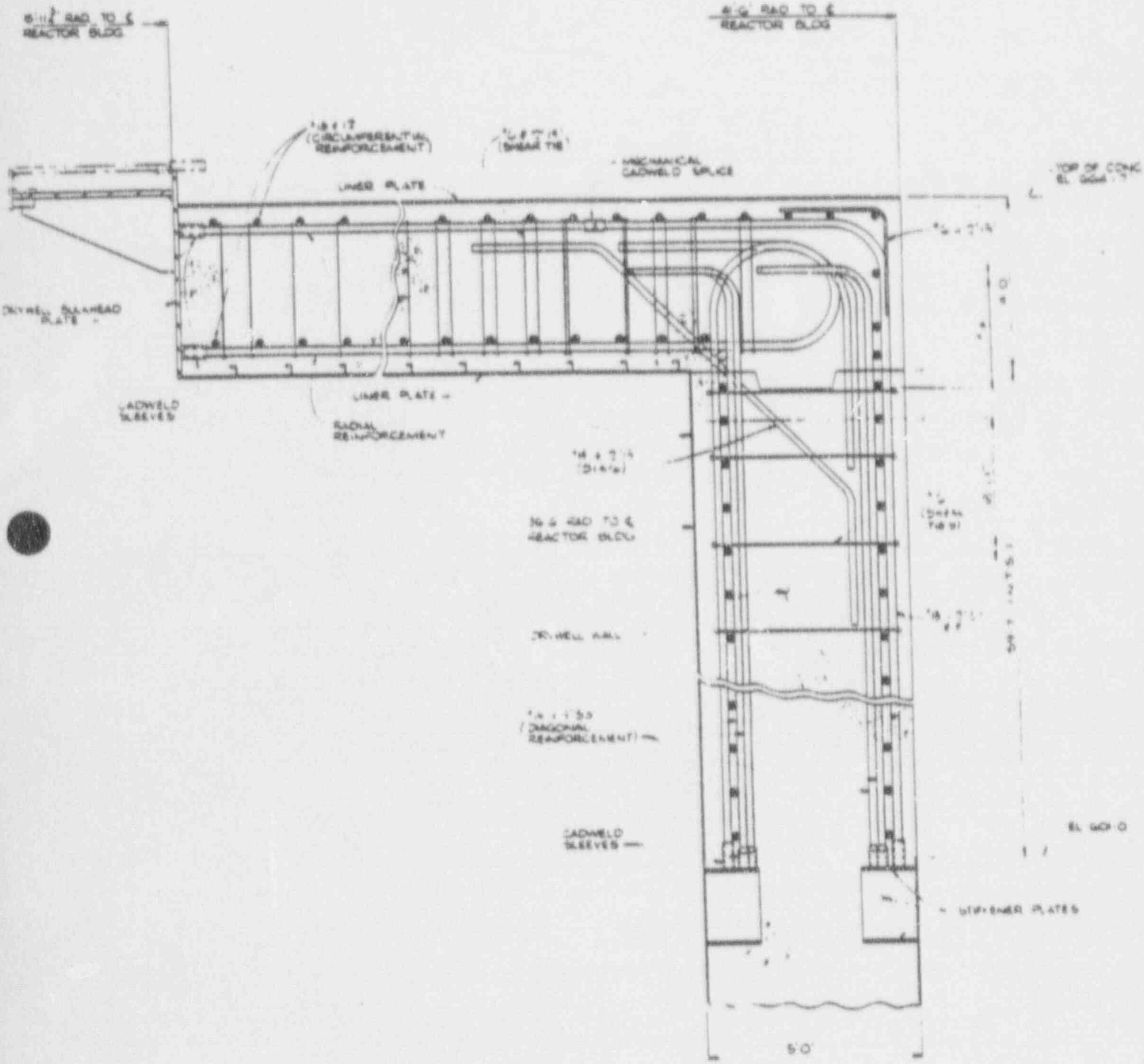
FIGURE 6



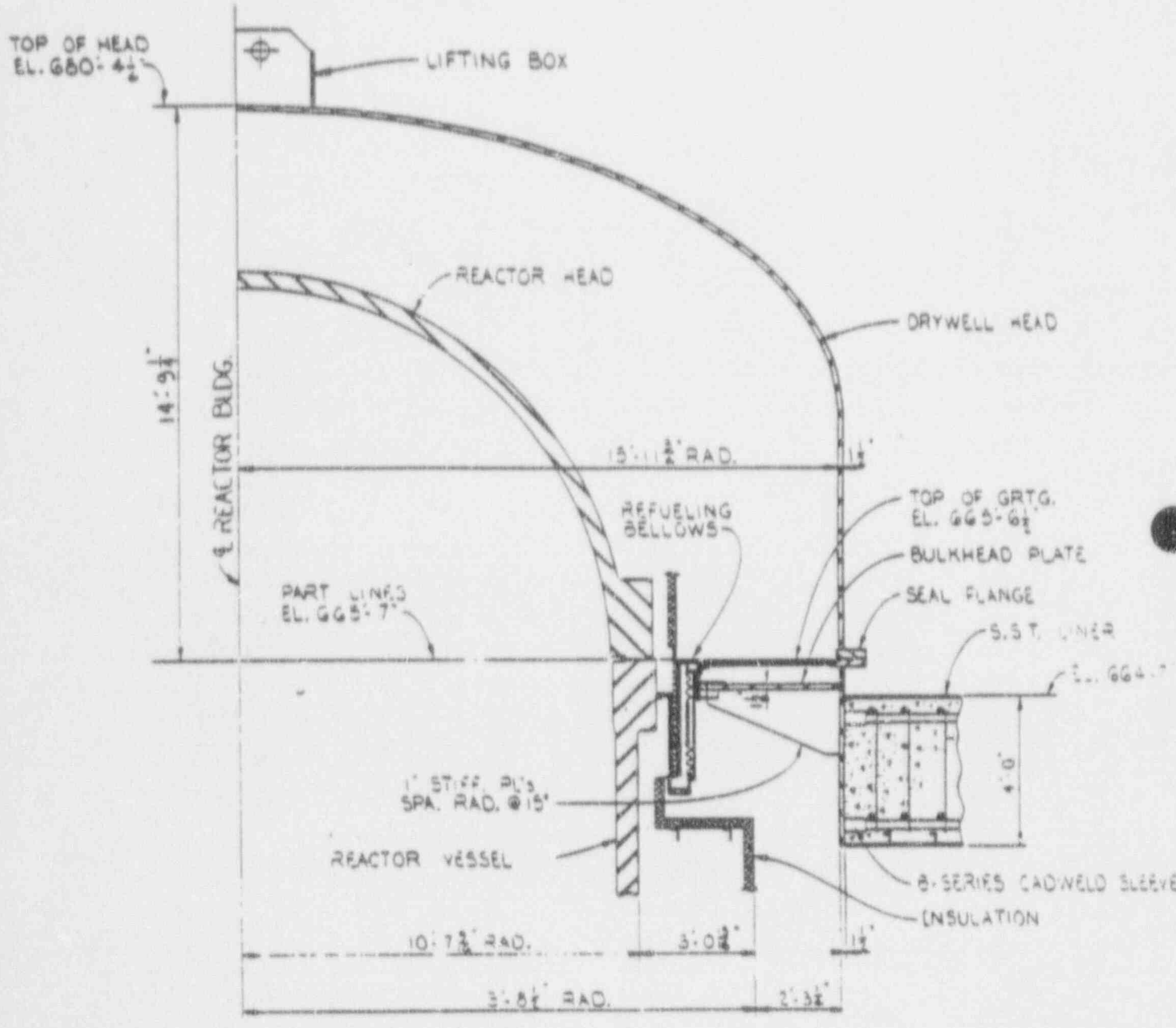
ANNULUS CONCRETE
FIGURE 7



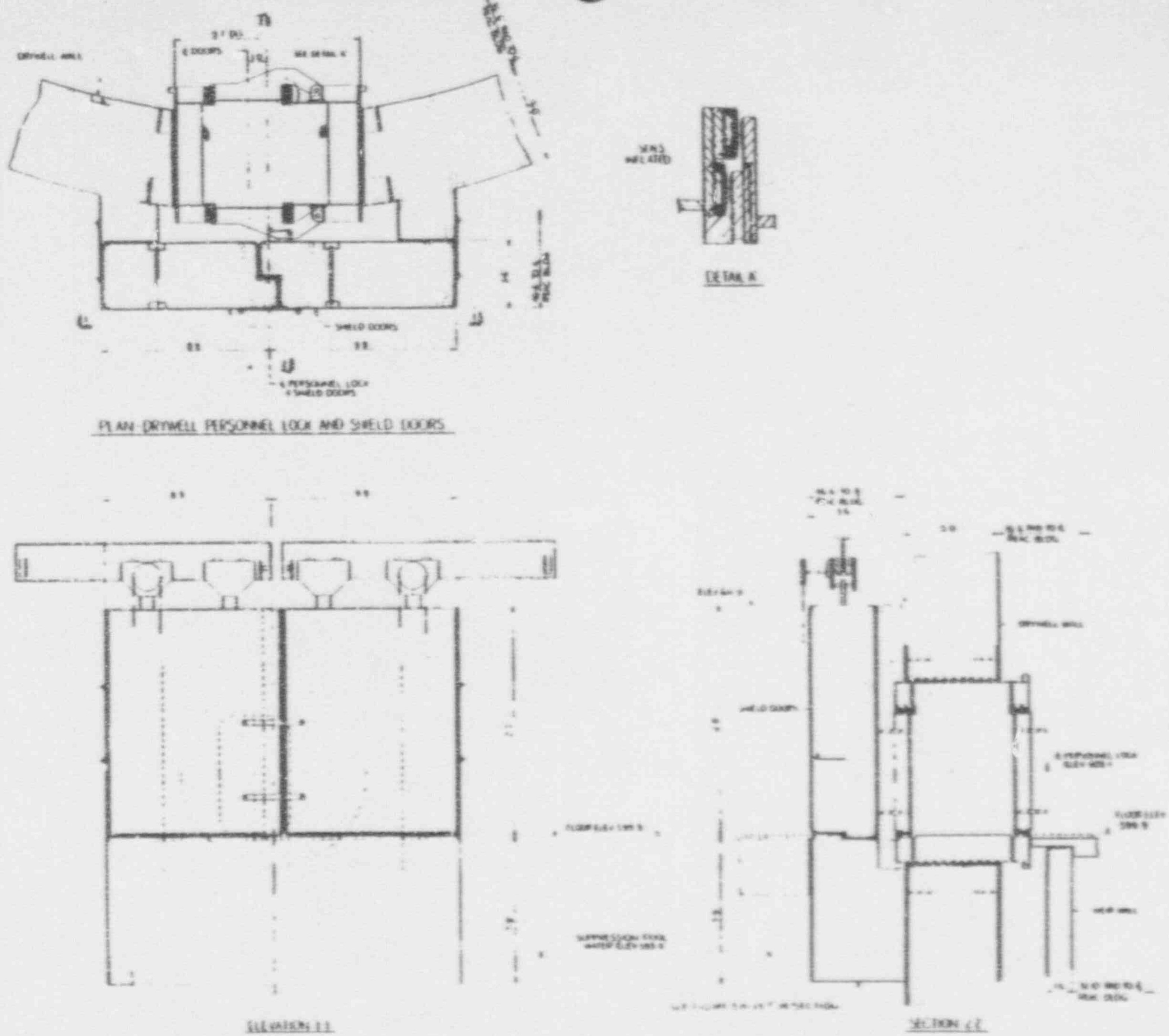
DRYWELL VENT STRUCTURE
FIGURE



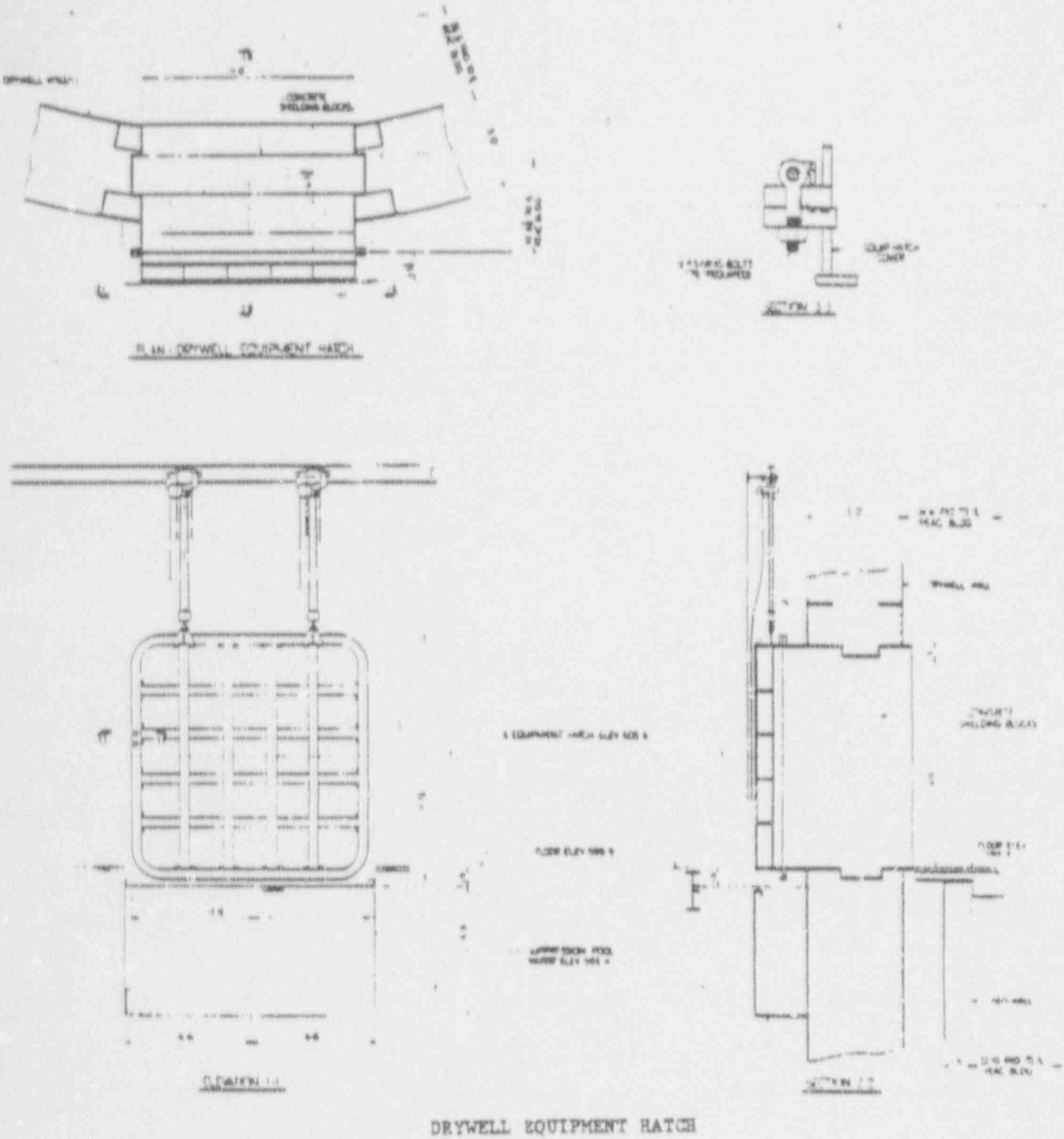
UPPER DRYWELL WALL AND SLAB
FIGURE 9



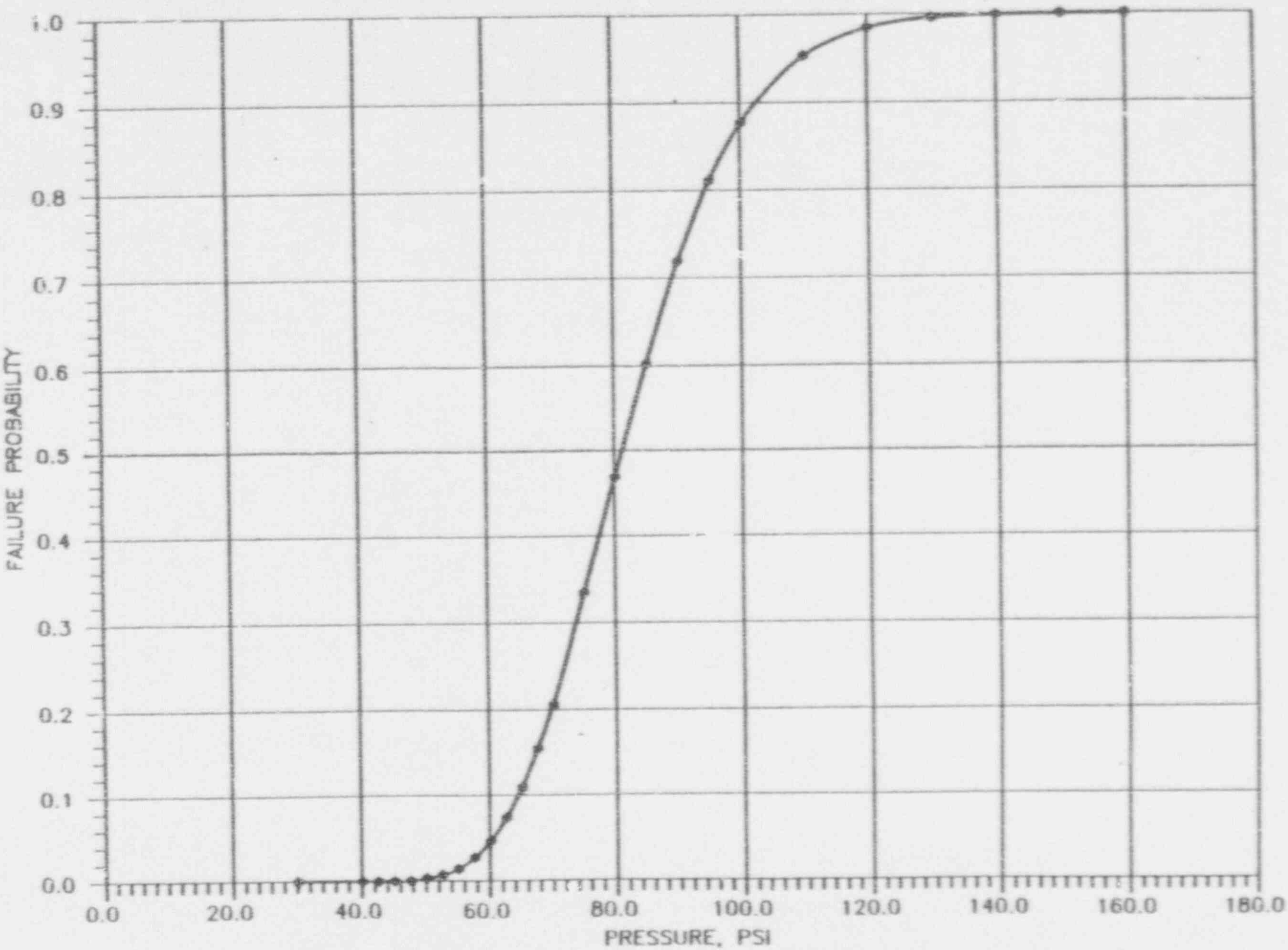
DRYWELL HEAD
FIGURE 10



DRYWELL PERSONNEL LOCK
FIGURE 11

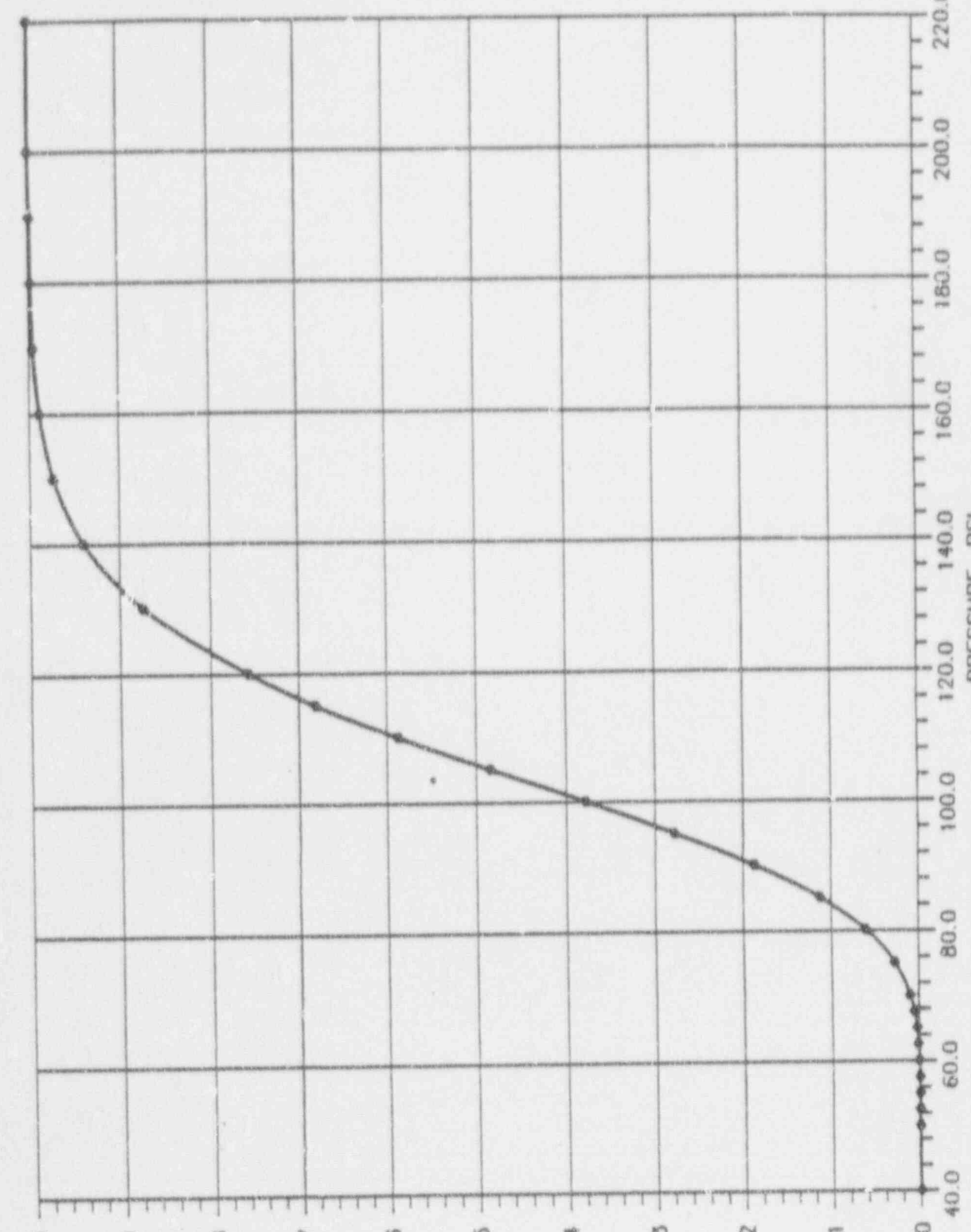


DRYWELL EQUIPMENT HATCH
FIGURE 12



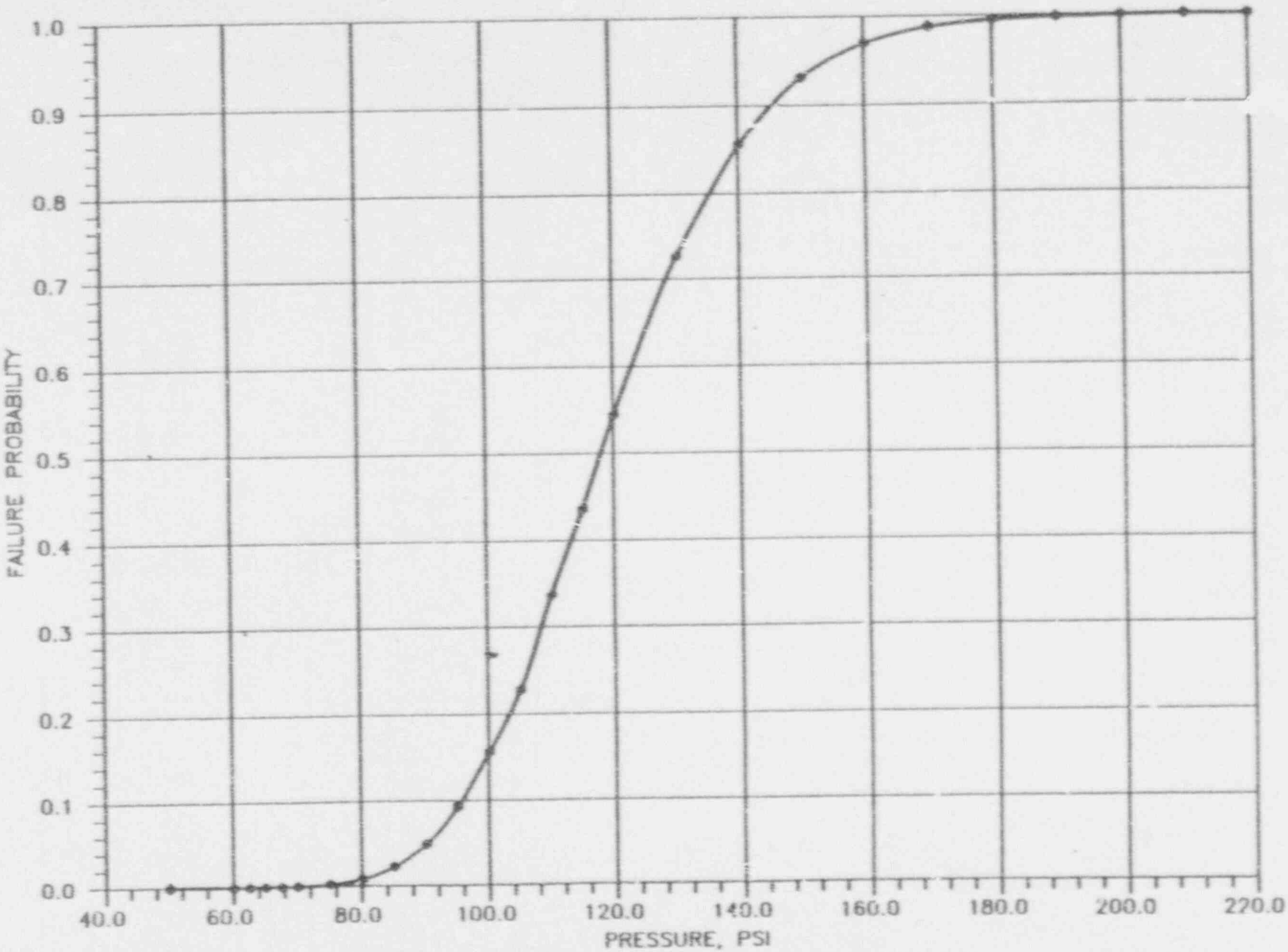
CEI – PERRY NUCLEAR POWER PLANT UNIT #1
DOME KNUCKLE FRAGILITY CURVE (Pf1)

FIGURE 13



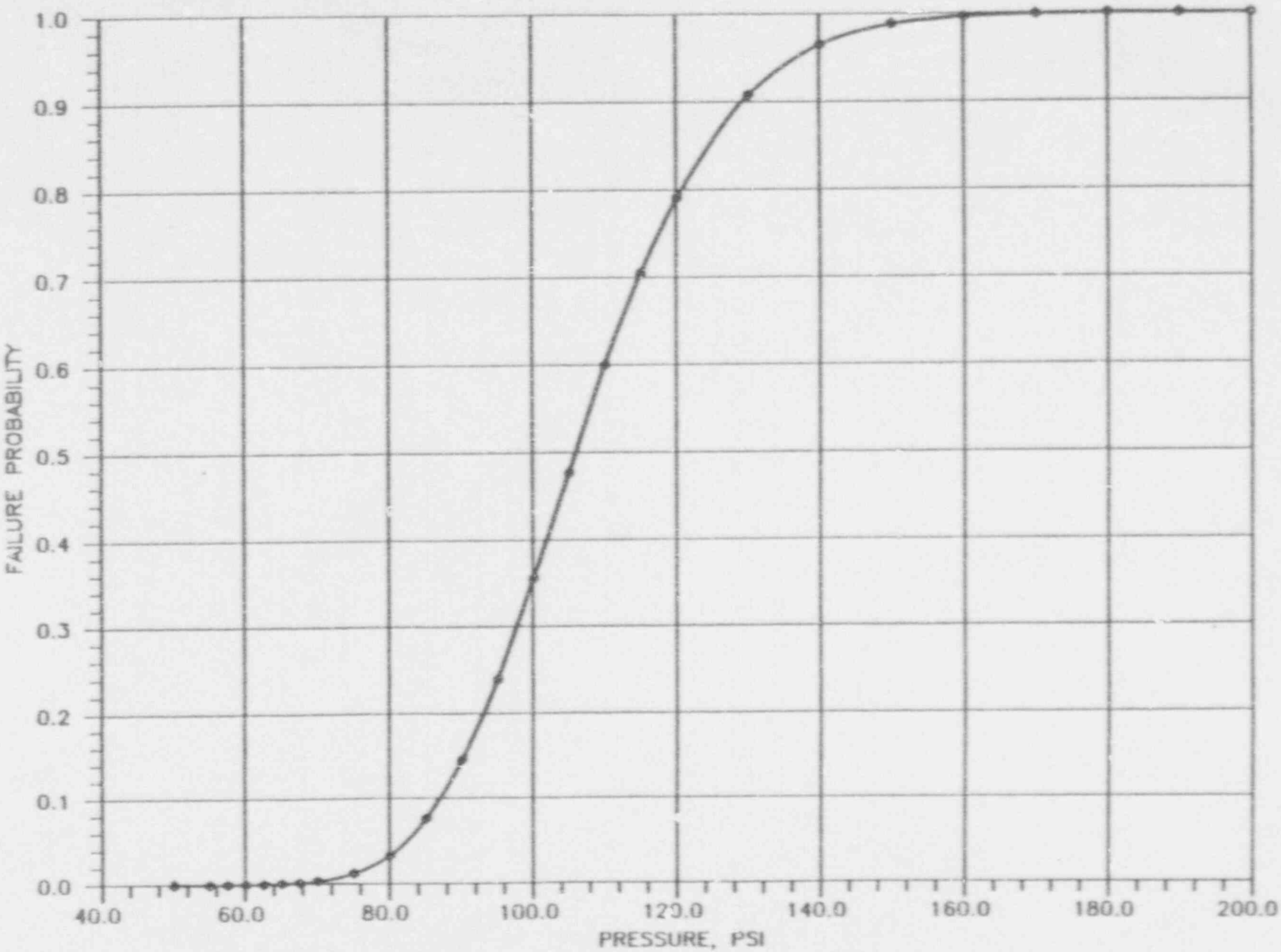
CEI – PERRY NUCLEAR POWER PLANT UNIT #1
DOME APEX FRAGILITY CURVE (Pf2)

FIGURE 14



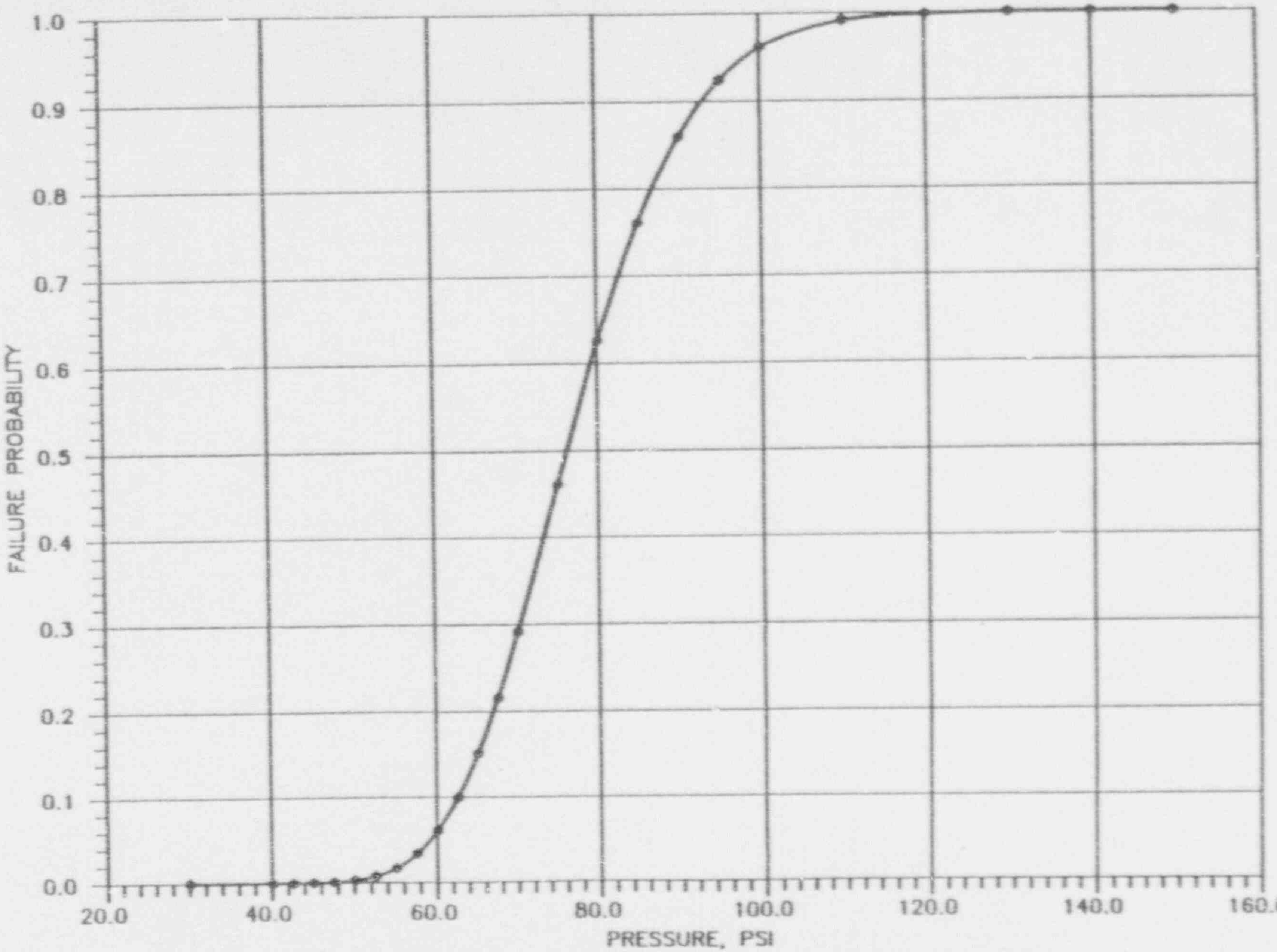
CEI – PERRY NUCLEAR POWER PLANT UNIT #1
CYLINDER FRAGILITY CURVE (Pf3)

FIGURE 15



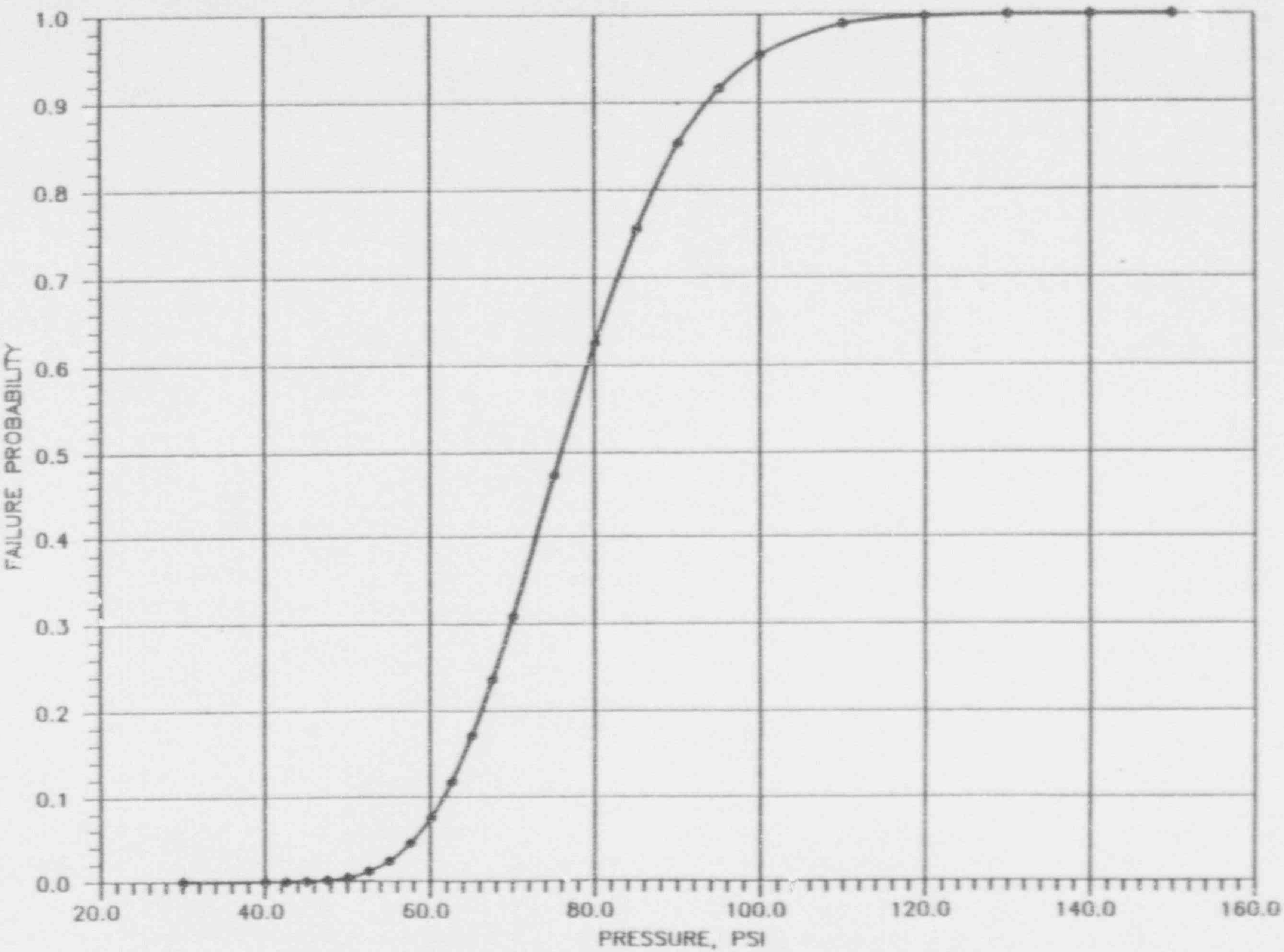
CEI - PERRY NUCLEAR POWER PLANT UNIT #1
PERSONNEL AIRLOCK FRAGILITY CURVE (Pf4)

FIGURE 16



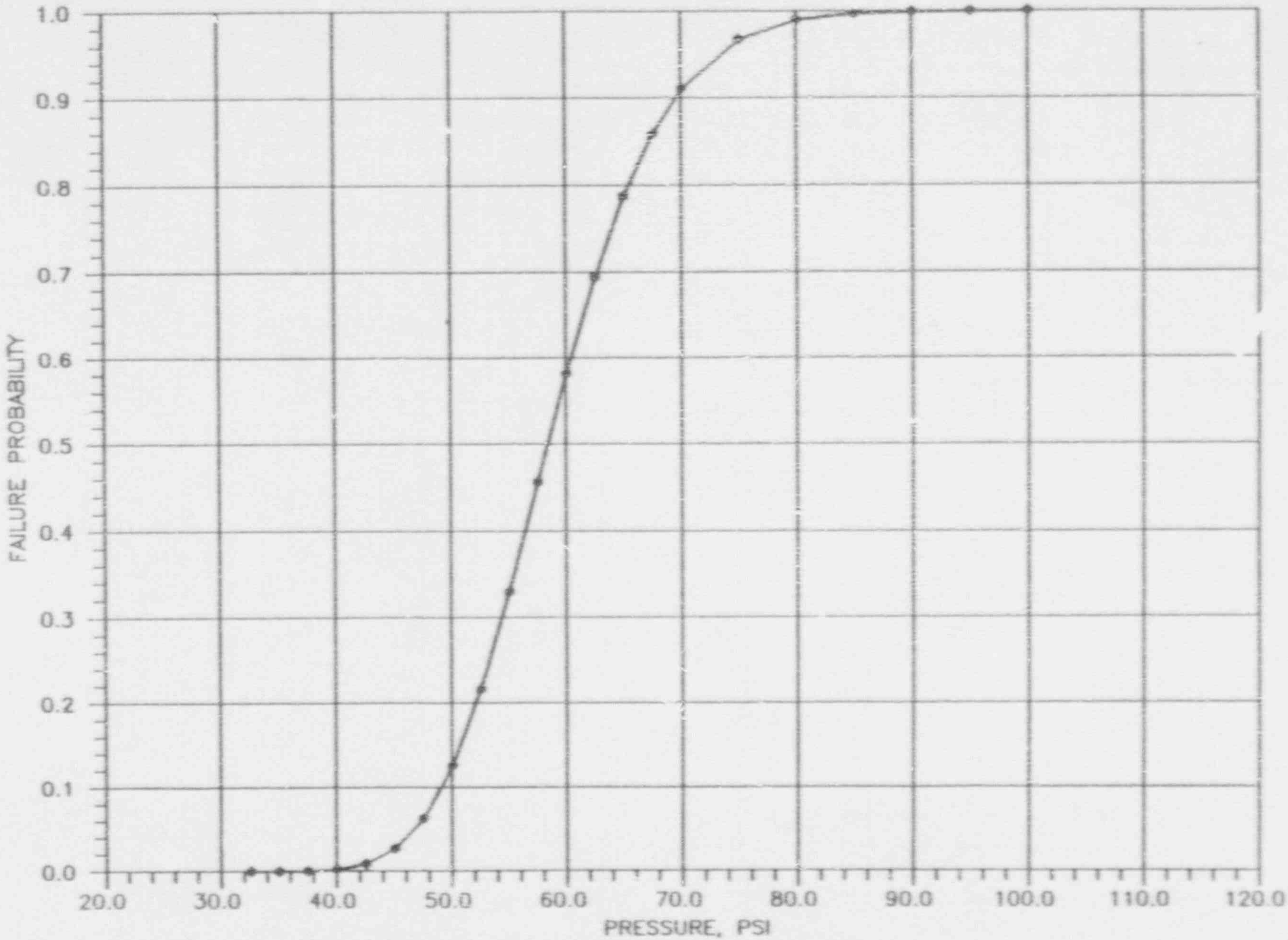
CEI – PERRY NUCLEAR POWER PLANT UNIT #1
EQUIPMENT HATCH(BOLTS) FRAGILITY CURVE (Pf5)

FIGURE 17



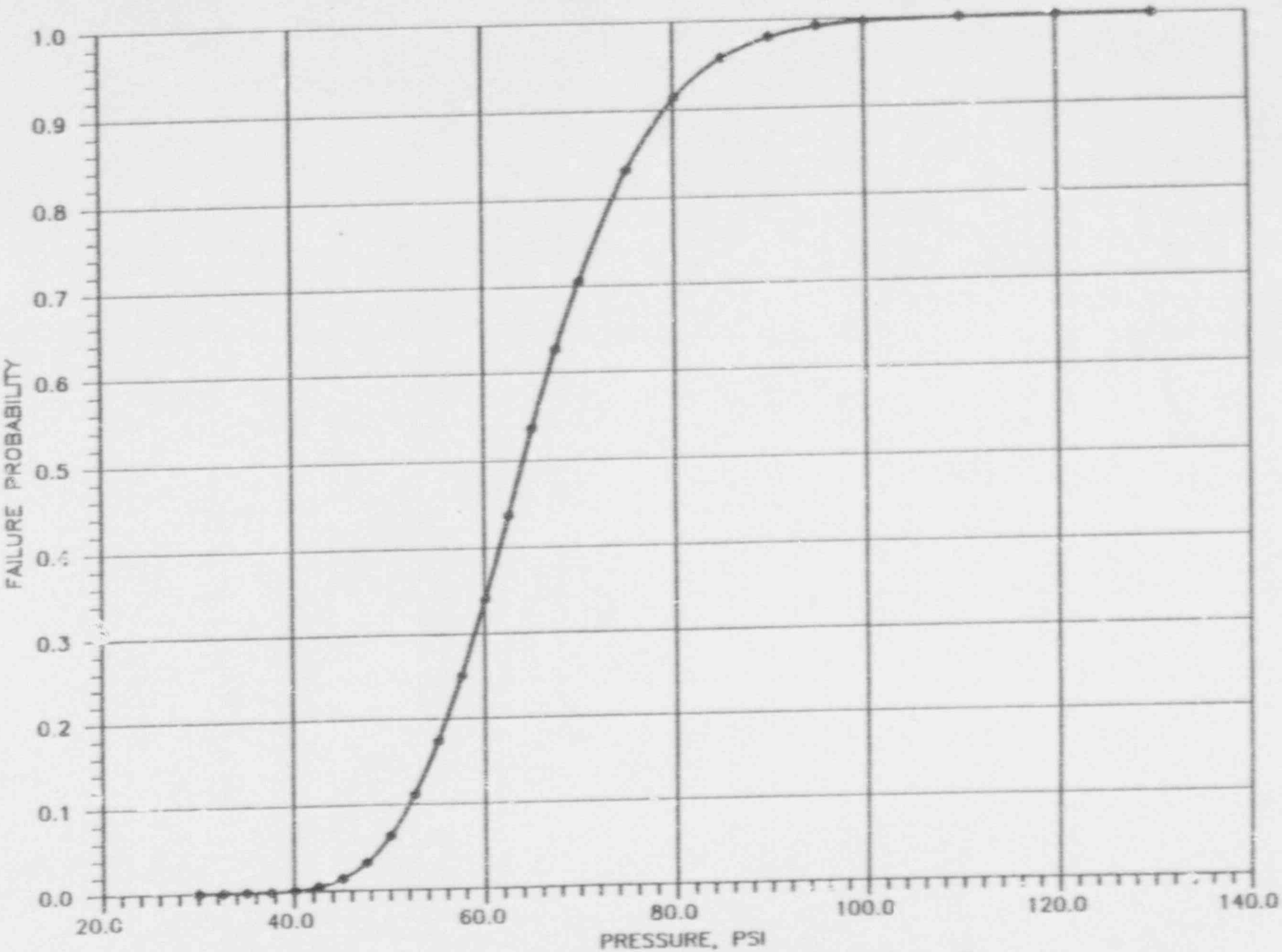
CEI – PERRY NUCLEAR POWER PLANT UNIT #1
PENETRATION P123 FRAGILITY CURVE (Pf6)

FIGURE 18



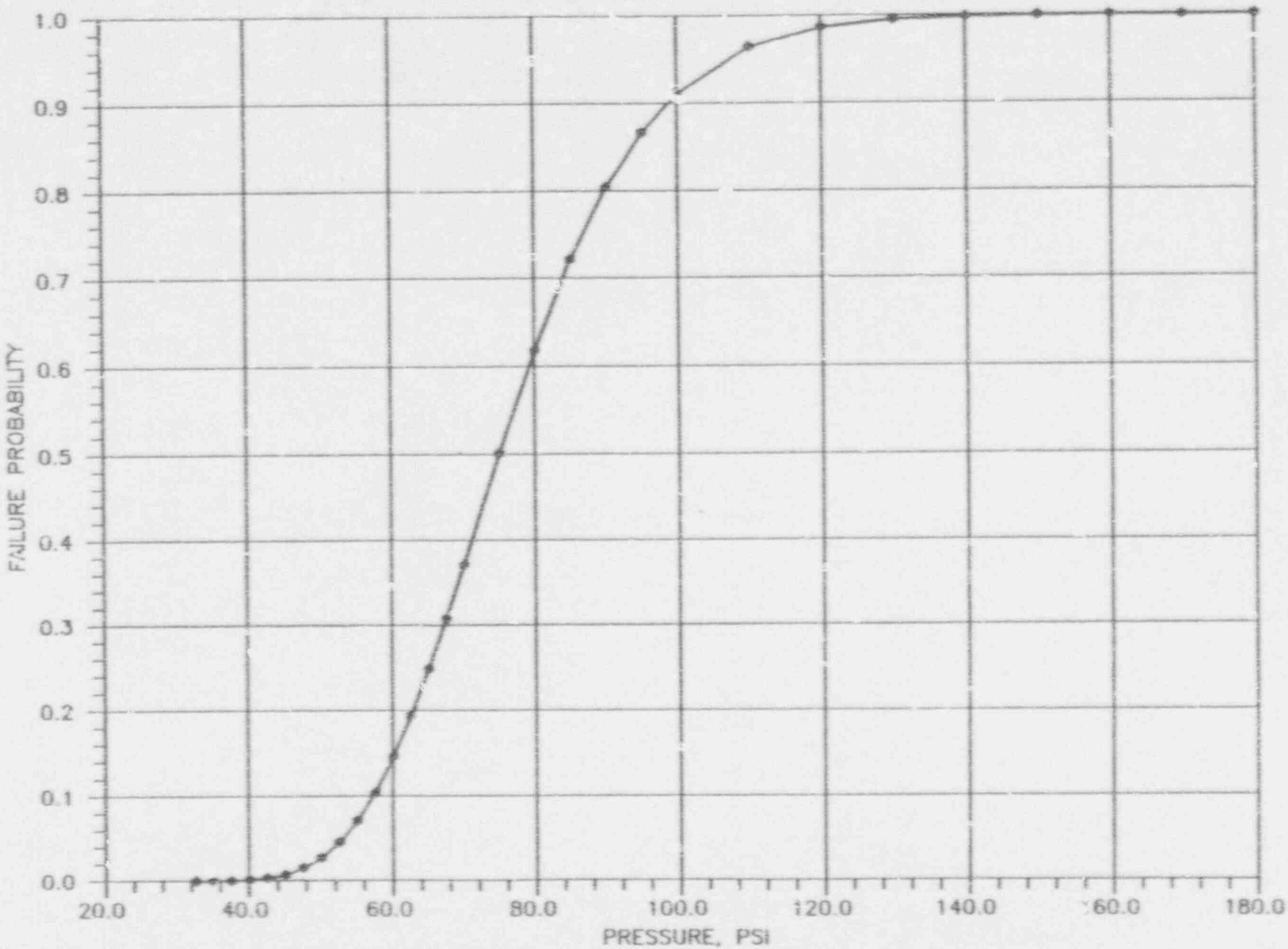
CEI – PERRY NUCLEAR POWER PLANT UNIT #1
PENETRATION P205 FRAGILITY CURVE (Pf7)

FIGURE 19



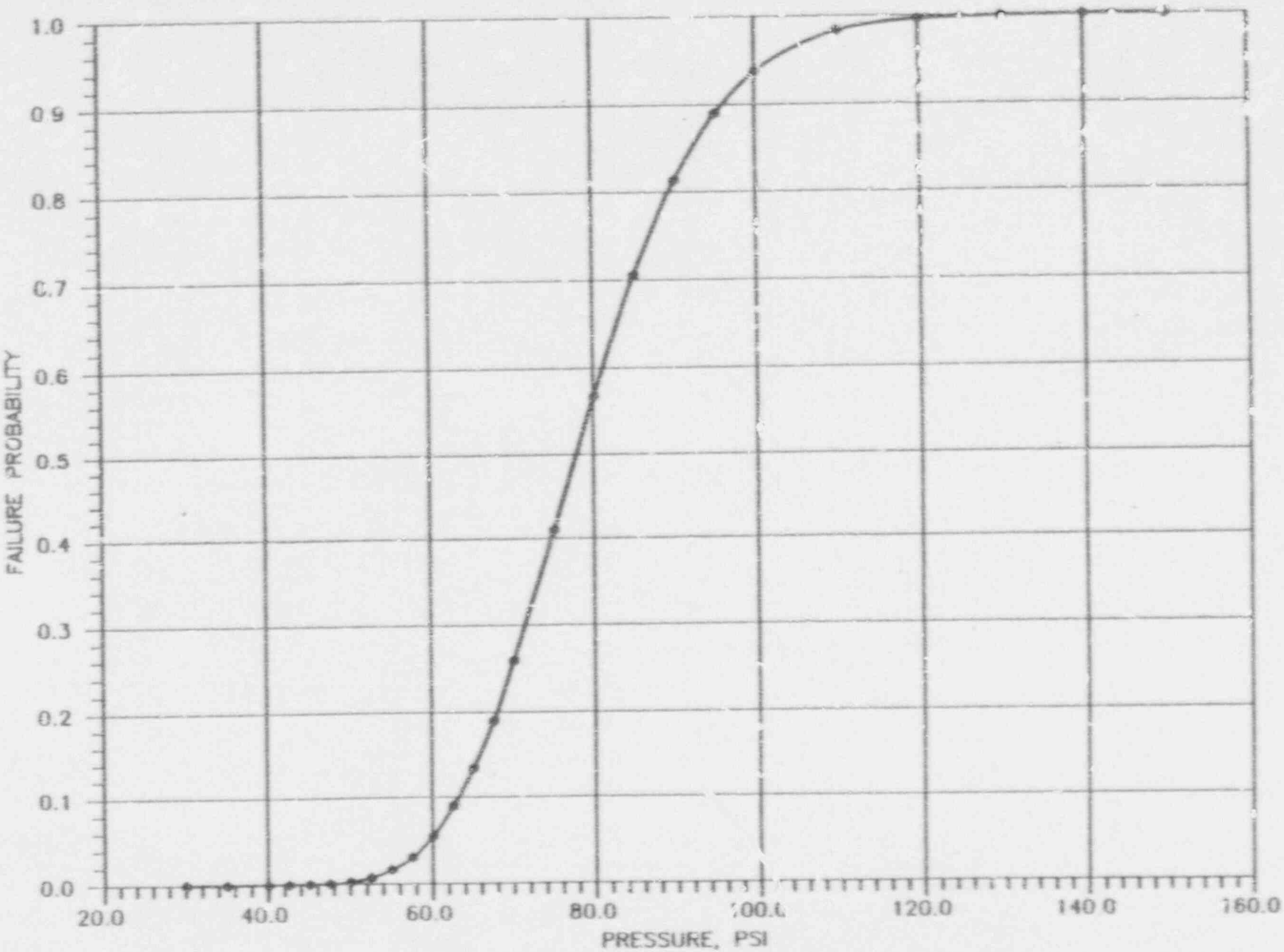
CEI – PERRY NUCLEAR POWER PLANT UNIT #1
PENETRATION P414 FRAGILITY CURVE (Pf8)

FIGURE 20



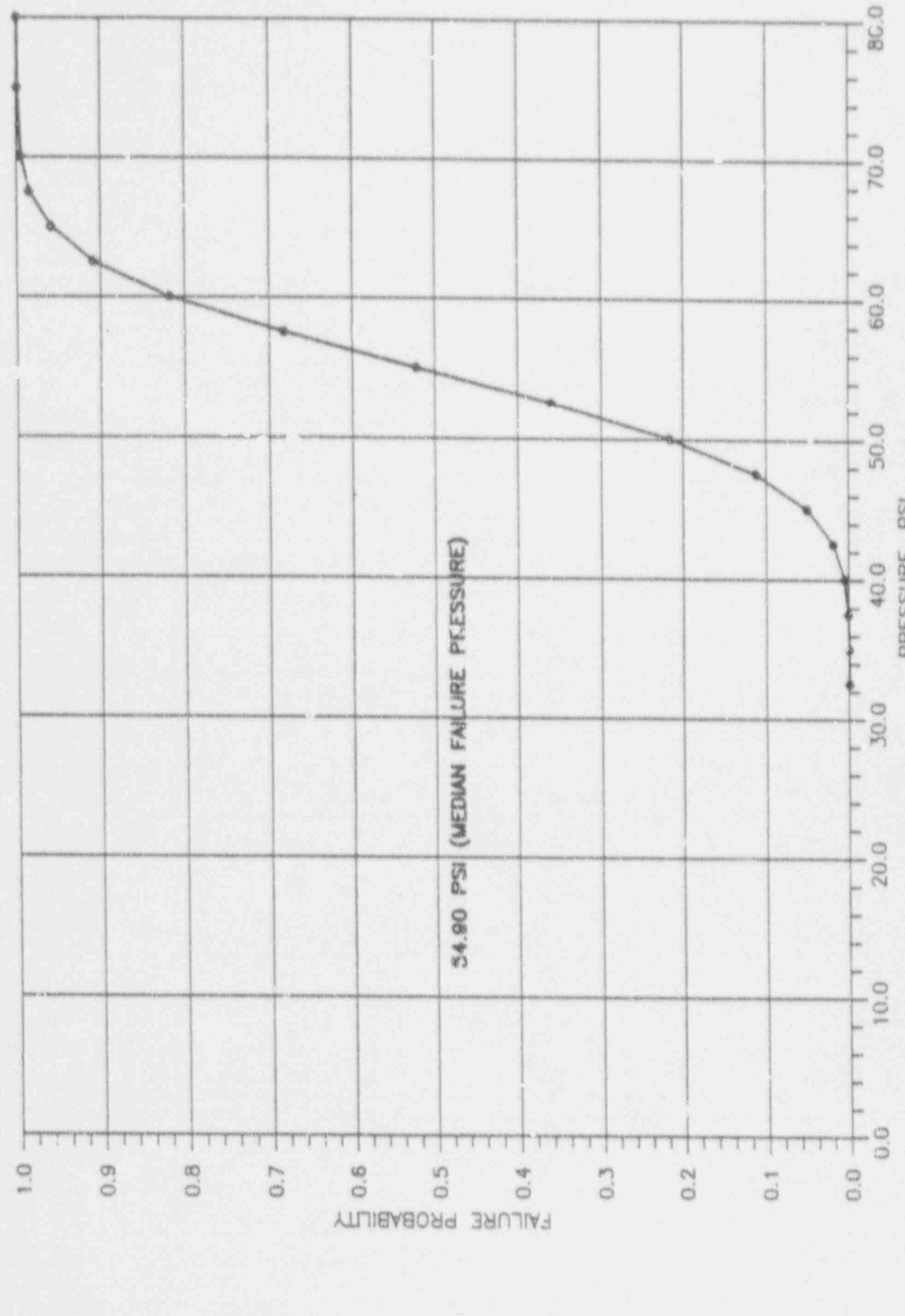
CEI – PERRY NUCLEAR POWER PLANT UNIT #1
CONCRETE ANCHORAGE FRAGILITY CURVE (Pf9)

FIGURE 21



CEI – PERRY NUCLEAR POWER PLANT UNIT #1
STEEL ANCHORAGE FRAGILITY CURVE (Pf10)

FIGURE 22



CEI – PERRY NUCLEAR POWER PLANT UNIT #1
CONTAINMENT COMPOSITE FRAGILITY CURVE

FIGURE 23

TABLE 1

ESTIMATE OF PNPP DRYWELL
MEAN FAILURE PRESSURE

Component	Material	Mean Failure Pressure (psi)
Lower wall	SA 516 - Grade 70	99.9
Upper wall	Reinforced concrete	67.0 (1)
Roof	Reinforced concrete	67.0 (1)
Head	SA 516 - Grade 70	77.8
Equipment hatches	SA 516 - Grade 70	84.7
Personnel airlocks	SA 516 - Grade 70	107.2

(1) This pressure is conservatively based on the Kuosheng analysis (Reference 1). Sections 3.4 and 3.5 of this PNPP IPE Containment analysis indicate that mean failure pressures based on actual PNPP structural details may be higher than 67 psi.

TABLE 2

MATERIAL PROPERTIES

Component	Mean Yield Strength (ksi)	Mean Tensile Strength (ksi)	Yield Strength Standard Deviation (ksi)	Tensile Strength Standard Deviation (ksi)
Dome	51.3	77.2	2.97	2.02
Cylinder	49.7	74.9	3.22	2.80

Component	Concrete Mean Compressive Strength (psi)	Standard Deviation (psi)
Mat	6442.0	448.0

TABLE 3

CONTAINMENT FAILURE MODES

Containment Failure Mode	Material Properties	Mean Failure Pressure (psig)	1st Percentile Failure Pressure (psig)	5th Percentile Failure Pressure (psig)
1. DOME KNUCKLE	dome	82.40	53.35	60.32
2. DOME APEX	dome	107.40	69.54	78.61
3. CYLINDER	cylinder	119.50	80.62	90.10
4. PERSONNEL AIRLOCK	cylinder	107.20	74.01	82.20
5. EQUIPMENT HATCH (BOLTS)	cylinder	77.10	53.26	59.15
6. PENETRATION P123	cylinder	76.90	51.95	58.05
7. PENETRATION P205	cylinder	[illegible].90	42.63	46.74
8. PENETRATION P414	cylinder	65.00	43.87	49.02
9. ANCHORAGE CONCRETE	concrete	76.75	45.83	52.95
10. ANCHORAGE STEEL	cylinder	78.90	53.32	59.58
11. COMPOSITE	-	-	40.73	44.95

TABLE 4

CONTAINMENT FAILURE MODE DATA

FAILURE MODE/LOCATION	MEAN FAILURE PRESSURE (PSI)	COEFFICIENT OF VARIATION	LOG. STANDARD DEVIATION OF STRENGTH, Bs	LOG. STANDARD DEVIATION FOR MODELING UNCERTAINTY, Bm	LOG. STANDARD DEVIATION OF CAPACITY, Bc	MEDIAN FAILURE PRESSURE (PSI)
1. DOME KNUCKLE	82.40	.058	.058	.17	.180	81.1
2. DOME APEX	107.40	.058	.058	.17	.180	105.7
3. CYLINDER	119.50	.065	.065	.15	.163	117.8
4. PERSONNEL AIRLOCK	107.20	.065	.065	.14	.154	105.9
5. EQUIPMENT HATCH (BOLTS)	77.10	.065	.065	.14	.154	76.2
6. PENETRATION P123	76.90	.065	.065	.15	.163	75.9
7. PENETRATION P208	58.90	.037	.037	.13	.135	58.4
8. PENETRATION P[illegible]14	65.00	.065	.065	.15	.163	64.1
9. ANCHORAGE, CONCRETE	76.75	.070	.070	.20	.211	75.1
10. ANCHORAGE, STEEL	78.9	.065	.065	.15	.163	77.9

TABLE 5

Failure Mode: Dome Knuckle (Pf1)
Median Pressure, Pm: 81.1 psi
Bc : 0.18

Pressure P,psi	u=ln(P/Pm)/Bc	Pf
30.00	-5.52	0.0000
40.00	-3.93	0.0000
42.50	-3.59	0.0002
45.00	-3.27	0.0005
47.50	-2.97	0.0010
50.00	-2.69	0.0040
52.50	-2.42	0.0080
55.00	-2.16	0.0150
57.50	-1.91	0.0280
60.00	-1.67	0.0470
62.50	-1.45	0.0740
65.00	-1.23	0.1090
67.50	-1.02	0.1540
70.00	-0.82	0.2060
75.00	-0.43	0.3340
80.00	-0.08	0.4680
85.00	0.26	0.6030
90.00	0.58	0.7190
95.00	0.88	0.8110
100.00	1.16	0.8770
110.00	1.69	0.9540
120.00	2.18	0.9850
130.00	2.62	0.9960
140.00	3.03	0.9990
150.00	3.42	1.0000
160.00	3.77	1.0000

TABLE 6

Failure Mode: Dome Apex (Pf2)
Median Pressure, Pm: 105.7 psi
Bc : 0.18

Pressure P,psi	u=ln(P/Pm)/Bc	Pf
40.00	-5.40	0.0000
50.00	-4.16	0.0000
52.50	-3.89	0.0000
55.00	-3.63	0.0001
57.50	-3.38	0.0004
60.00	-3.15	0.0008
62.50	-2.92	0.0020
65.00	-2.70	0.0030
67.50	-2.49	0.0060
70.00	-2.29	0.0110
75.00	-1.91	0.0280
80.00	-1.55	0.0610
85.00	-1.21	0.1130
90.00	-0.89	0.1870
95.00	-0.59	0.2780
100.00	-0.31	0.3780
105.00	-0.04	0.4840
110.00	0.22	0.5870
115.00	0.47	0.6810
120.00	0.70	0.7580
130.00	1.15	0.8750
140.00	1.56	0.9410
150.00	1.94	0.9740
160.00	2.30	0.9890
170.00	2.64	0.9960
180.00	2.96	0.9980

TABLE 7

Failure Mode: Cylinder (Pf3)
Median Pressure, Pm: 117.8 psi
Bc : 0.163

Pressure P,psi	u=ln(P/Pm)/Bc	Pf
50.00	-5.26	0.0000
[illegible]	-4.16	0.0000
62.50	-3.89	0.0000
65.00	-3.65	0.0001
67.50	-3.42	0.0003
70.00	-3.19	0.0007
75.00	-2.77	0.0030
80.00	-2.37	0.0090
85.00	-2.00	0.0230
90.00	-1.65	0.0490
95.00	-1.32	0.0930
100.00	-1.01	0.1560
105.00	-0.75	0.2270
110.00	-0.42	0.3370
115.00	-0.16	0.4360
120.00	0.11	0.5440
130.00	0.60	0.7260
140.00	1.06	0.8550
150.00	1.48	0.9310
160.00	1.88	0.9700
170.00	2.25	0.9880
180.00	2.60	0.9950
190.00	2.93	0.9980
200.00	3.25	0.9990
210.00	3.55	1.0000
220.00	3.83	1.0000

TABLE 8

Failure Mode: Personnel Airlock (Pf4)
Median Pressure Pm: 105.9 p
Bc : 0.154

Pressure P,psi	u=ln(P/Pm)/Bc	Pf
50.00	-4.87	0.0000
55.00	-4.25	0.0000
57.50	-3.97	0.0000
60.00	-3.69	0.0001
62.50	-3.42	0.0003
65.00	-3.17	0.0008
67.50	-2.92	0.0020
70.00	-2.69	0.0040
75.00	-2.24	0.0130
80.00	-1.82	0.0340
85.00	-1.43	0.0760
90.00	-1.06	0.1450
95.00	-0.71	0.2390
100.00	-0.37	0.3560
105.00	-0.06	0.4760
110.00	0.25	0.6000
115.00	0.54	0.7050
120.00	0.81	0.7910
130.00	1.33	0.9080
140.00	1.81	0.9650
150.00	2.26	0.9880
160.00	2.68	0.9960
170.00	3.07	0.9990
180.00	3.44	1.0000
190.00	3.80	1.0000
200.00	4.13	1.0000

TABLE 9

Failure Mode: Equipment Hatch (Bolts) (Pf5)
Median Pressure, Pm: 76.2 psi
Bc : 0.154

Pressure P,psi	u=ln(P/Pm)/Bc	Pf
30.00	-6.05	0.0000
40.00	-4.18	0.0000
42.50	-3.79	0.0000
45.00	-3.42	0.0003
47.50	-3.07	0.0010
50.00	-2.74	0.0030
52.50	-2.42	0.0080
55.00	-2.12	0.0170
57.50	-1.83	0.0340
60.00	-1.55	0.0610
62.50	-1.29	0.0990
65.00	-1.03	0.1510
67.50	-0.79	0.2150
70.00	-0.55	0.2910
75.00	-0.10	0.4600
80.00	0.32	0.6260
85.00	0.71	0.7610
90.00	1.08	0.8600
95.00	1.43	0.9240
100.00	1.76	0.9610
110.00	2.38	0.9910
120.00	2.95	0.9980
130.00	3.47	1.0000
140.00	3.95	1.0000
150.00	4.40	1.0000

TABLE 10

Failure Mode: Penetration P123 (Pf6)
Median Pressure, Pm: 75.9 psi
Bc : 0.163

Pressure P,psi	u=ln(P/Pm)/Bc	Pf
30.00	-5.69	0.0000
40.00	-3.93	0.0000
42.50	-3.56	0.0002
45.00	-3.21	0.0007
47.50	-2.88	0.0020
50.00	-2.56	0.0050
52.50	-2.26	0.0120
55.00	-1.98	0.0240
57.50	-1.70	0.0450
60.00	-1.44	0.0750
62.50	-1.19	0.1170
65.00	-0.95	0.1710
67.50	-0.72	0.2360
70.00	-0.50	0.3080
75.00	-0.07	0.4720
80.00	0.32	0.6260
85.00	0.69	0.7550
90.00	1.05	0.8530
95.00	1.38	0.9160
100.00	1.69	0.9540
110.00	2.28	0.9890
120.00	2.81	0.9980
130.00	3.30	1.0000
140.00	3.76	1.0000
150.00	4.18	1.0000

TABLE 11

Failure Mode: Penetration P205 (Pf7)
Median Pressure, Pm: 58.4 psi
Bc : 0.135

Pressure P,psi	u=ln(P/Pm)/Bc	Pf
32.50	-4.336	0.0000
35.00	-3.788	0.0001
37.50	-3.276	0.0005
40.00	-2.798	0.0026
42.50	-2.349	0.0094
45.00	-1.926	0.0270
47.50	-1.526	0.0635
50.00	-1.146	0.1259
52.50	-0.784	0.2165
55.00	-0.440	0.3300
57.50	-0.110	0.4562
60.00	0.205	0.5812
62.50	0.507	0.6940
65.00	0.798	0.7975
67.50	1.077	0.8592
70.00	1.346	0.9109
75.00	1.858	0.9684
80.00	2.336	0.9903
85.00	2.785	0.9973
90.00	3.208	0.9993
95.00	3.609	0.9998
100.00	3.989	1.0000

TABLE 12

Failure Mode: Penetration P414 (Pf8)
Median Pressure, Pm: 64.1 psi
Bc : 0.163

Pressure P,psi	u=ln(P/Pm)/Bc	Pf
30.00	-4.66	0.0000
32.50	-4.17	0.0000
35.00	-3.71	0.0001
37.50	-3.29	0.0005
40.00	-2.89	0.0020
42.50	-2.52	0.0060
45.00	-2.17	0.0150
47.50	-1.84	0.0330
50.00	-1.52	0.0640
52.50	-1.22	0.1110
55.00	-0.94	0.1740
57.50	-0.67	0.2510
60.00	-0.41	0.3410
62.50	-0.16	0.4360
65.00	0.09	0.5360
67.50	0.32	0.6260
70.00	0.54	0.7050
75.00	0.96	0.8320
80.00	1.36	0.9130
85.00	1.73	0.9580
90.00	2.08	0.9810
95.00	2.41	0.9920
100.00	2.73	0.9970
110.00	3.31	1.0000
120.00	3.85	1.0000
130.00	4.34	1.0000

TABLE 13

Failure Mode: Anchorage, Concrete (Pf9)
Median Pressure, Pm: 75.0 psi
Bc : 0.212

Pressure P,psi	u=ln(P/Pm)/Bc	Pf
32.50	-3.947	0.0000
35.00	-3.598	0.0002
37.50	-3.272	0.0005
40.00	-2.968	0.0015
42.50	-2.682	0.0037
45.00	-2.412	0.0079
47.50	-2.157	0.0155
50.00	-1.915	0.0277
52.50	-1.685	0.0460
55.00	-1.466	0.0714
57.50	-1.256	0.1045
60.00	-1.055	0.1458
62.50	-0.863	0.1940
65.00	-0.678	0.2489
67.50	-0.500	0.3085
70.00	-0.328	0.3715
75.00	0.003	0.5012
80.00	0.301	0.6182
85.00	0.587	0.7214
90.00	0.857	0.8043
95.00	1.112	0.8669
100.00	1.354	0.9121
110.00	1.804	0.9644
120.00	2.214	0.9866
130.00	2.592	0.9952
140.00	2.941	0.9984
150.00	3.267	0.9995
160.00	3.571	0.9998
170.00	3.857	0.9999
180.00	4.126	1.0000

TABLE 14

Failure Mode: Anchorage, Steel (Pf10)
Median Pressure, Pm: 77.9 psi
Bc : 0.163

Pressure P,psi	u=ln(P/Pm)/Bc	Pf
30.00	-5.85	0.0000
35.00	-4.91	0.0000
40.00	-4.09	0.0000
42.50	-3.71	0.0001
45.00	-3.36	0.0004
47.50	-3.03	0.0010
50.00	-2.72	0.0030
52.50	-2.42	0.0080
55.00	-2.13	0.0170
57.50	-1.86	0.0310
60.00	-1.60	0.0550
62.50	-1.35	0.0890
65.00	-1.11	0.1340
67.50	-0.88	0.1890
70.00	-0.65	0.2580
75.00	-0.23	0.4090
80.00	0.17	0.5680
85.00	0.54	0.7050
90.00	0.89	0.8130
95.00	1.22	0.8890
100.00	1.54	0.9380
110.00	2.12	0.9830
120.00	2.65	0.9960
130.00	3.15	0.9990
140.00	3.60	0.9998
150.00	4.02	1.0000

TABLE 15

CONTAINMENT COMPOSITE FRAGILITY CURVE

Pressure (psi)	1-Pf1	1-Pf2	1-Pf3	1-Pf4	1-Pf5	1-Pf6	1-Pf7	1-Pf8	1-Pf9	1-Pf10	PI(1-Pfi)	Composite Failure Probability 1-PI(1-Pfi)
32.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0000	0.0000
35.0	1.0	1.0	1.0	1.0	1.0	1.0	0.9999	0.9999	0.9998	1.0	0.9996	0.0004
37.5	1.0	1.0	1.0	1.0	1.0	1.0	0.9995	0.9995	0.9995	1.0	0.9985	0.0015
40.0	1.0	1.0	1.0	1.0	1.0	1.0	0.9974	0.9980	0.9985	1.0	0.9939	0.0061
42.5	0.9998	1.0	1.0	1.0	1.0	0.9998	0.9906	0.9940	0.9963	0.9999	0.9805	0.0195
45.0	0.9995	1.0	1.0	1.0	0.9997	0.9993	0.9730	0.9850	0.9921	0.9996	0.9490	0.0510
47.5	0.9990	1.0	1.0	1.0	0.9990	0.9980	0.9365	0.9670	0.9845	0.9990	0.8871	0.1129
50.0	0.9960	1.0	1.0	1.0	0.9970	0.9950	0.8741	0.9360	0.9723	0.9970	0.7836	0.2164
52.5	0.9920	1.0	1.0	1.0	0.9920	0.9880	0.7835	0.8890	0.9540	0.9920	0.6409	0.3599
55.0	0.9850	0.9999	1.0	1.0	0.9830	0.9760	0.6700	0.8260	0.9286	0.9830	0.4773	0.5226
57.5	0.9720	0.9996	1.0	1.0	0.9660	0.9550	0.5438	0.7490	0.8[illegible]55	0.9690	0.3167	0.6833
60.0	0.9530	0.9992	1.0	0.9999	0.9390	0.9250	0.4188	0.6590	0.8542	0.9450	0.1842	0.8185
62.5	0.9260	0.9980	1.0	0.9997	0.9010	0.8830	0.3060	0.5640	0.8060	0.9110	0.0931	0.9069
65.0	0.8910	0.9970	0.9999	0.9992	0.8490	0.8290	0.2125	0.4640	0.7511	0.8660	0.0401	0.9599
67.5	0.8460	0.9940	0.9997	0.9980	0.7850	0.7640	0.1408	0.3740	0.6915	0.8110	0.0147	0.9853
70.0	0.7940	0.9890	0.9993	0.9950	0.7090	0.6920	0.0891	0.2950	0.6285	0.7420	0.0047	0.9953
75.0	0.6640	0.9720	0.9970	0.9870	0.5400	0.5280	0.0316	0.1680	0.4988	0.5910	0.0003	0.9997
80.0	0.5320	0.9390	0.9910	0.9660	0.3740	0.3740	0.0097	0.0870	0.3818	0.4320	0.0000	1.0

TABLE 16

QUALIFICATION TESTS FOR ELECTRICAL PENETRATION

Penetration Type	Conditions for aging Test	Conditions for Accident Test	Leakage After Tests (per penetration)
Low Voltage through C.V.	302 Degree(F) for 100 hours	340 degree(F) and 108 psi for 3 hours	< 6E-4 in**3/sec at 75 psi
Medium Voltage through C.V.	302 Degree(F) for 100 hours	343 degree(F) and 111 psi for 3 hours	< 6E-4 in**3/sec at 75 psi
Drywell(MCT)	185 degree for 852 hours		259 in**3/sec at 5 psi
Drywell(MCT)		> 250 degree(F) and > 10psi for 2 days	190 in**3/sec at 5 psi

TABLE 17

EFFECTS OF ELEVATED TEMPERATURES ON CONCRETE

Temp Degree(F)	% of compressive strength at room temperature for 48 hours exposure		
	ASTM Publication STP 858 (Ref. 28)	ACI Publication SP 25 (Ref. 11)	PNPP estimate
200	85	90	85
300	85	85	85
400	80	80	80
600	55	60	55
800	40	50	40
1,000	33	45	30
1,200	20	40	20
1,400	-	30	10
1,600	-	7	5

TABLE 18

EFFECTS OF ELEVATED TEMPERATURES ON REINFORCING STEEL

Temp Degree(F)	% of yield strength at room temperature		
	ASTM Publication STP 180 (Ref. 29)	United State Steel Manual (Ref. 30)	PNPP estimate for ASTM A615 Grade 60
200	98	90	98
400	90	95	90
600	78	96	78
800	52	88	52
1,000	20	73	20
1,200	-	40	10
1,400	-	20	05

TABLE 19

EFFECTS OF ELEVATED TEMPERATURES ON SA 516 GRADE 70 PLATE

Temp Degree(F)	% of yield strength at room temperature			
	ASME B. & P.V. CODE (Ref. 9)	ASTM STP 180 (Ref. 29)	U.S.S. STEEL DESIGN MANUAL (Ref. 30)	PNPP estimate
100	100			100
200	91			91
300	89			89
400	86			86
500	81			81
600	74			74
700	72			72
800		52	88	50
1,000		20	73	25
1,200			40	20
1,400			20	10
1,600			12	5
1,800			8	0

TABLE 20

SEAL LIFE OF ETHELENE PROPOLENE

Temperature (Degree F)	Seal Life (hours)
300	1,000
400	5
500	0.75
600	0.1 to 0.2

TABLE 21

LEAKAGE FROM OTHER THAN STRUCTURAL FAILURE (1)

Item	Cause of Leakage	Potential Leak Area (in^2) and Probability of Leakage (2)				
		200 Deg.(F)	300 Deg.(F)	400 Deg.(F)	500 Deg.(F)	600 Deg.(F)
Airlocks	Temperature Degrades Seal	47 A	47 A	47 B	47 C	47 D
Airlocks	Seal Looses Inflation Pressure	47 B	47 B	47 C	N/A (Seal Degrades)	N/A (Seal Degrades)
Equipment Hatch	Bolts Stretch and Seal Looses Compression	2 A	2 B	2 C	2 D	2 D
Mechanical Penetrations (Drywell)	Fracture of Bellows	200 (3,5) A	200 (3,5) B	200 (3,5) C	200 (3,5) D	200 (3,5) D
Mechanical Penetrations (Containment)	Fracture of Bellows	900 (4,5) A	900 (4,5) B	900 (4,5) C	900 (4,5) D	900 (4,5) D
Electrical Penetrations (Drywell)	Temperature Degrades Seals	1000 (6) A	1000 (6) A	1000 (6) A	1000 (6) A	1000 (6) A
Electrical Penetrations (Containment)	Temperature Degrades Seals	200 (7) A	200 (7) B	200 (7) C	200 (7) D	200 (7) D

Notes:

(1) Effects of temperature on materials and components is discussed in item 8.4

(2) Key to estimated probabilities:

A - Highly unlikely : Probability of occurrence of .001 or less

B - Very unlikely : Probability of occurrence of .01 or less

C - Ulikely : Probability of occurrence of .10 or less

D - Indeterminent : Probability higher than 0.10 but not further defined

(3) Based on 2 penetrations with potential leakage of 100 in^2 each

(4) Based on 11 penetrations including 4 penetrations with 121 in^2 each

(5) Potential leakage area is a maximum that should never be reached. We should expect that some penetrations would fail before others because of different bellows configurations and relative movements. In addition a cracked bellows could give a much smaller leak area than the "potential" which is calculated on the basis of bellows being totally ineffective in holding pressure

(6) Based on 52 penetrations. See item 8.4

(7) Based on 40 penetrations. See item 8.4

APPENDIX A

CONTAINMENT VESSEL ANCHORAGE, CONCRETE

1. Calculation of mean failure pressure based on concrete strength

2. Selection of log standard deviations for materials, B_R, and for modeling, B_U.

CALCULATION OF MEAN FAILURE PRESSURE

Capacity of Containment Anchorage - Concrete

Find ultimate capacity of Containment Anchorage
Ref 16 using ACI 349 Appendix B with the following assumptions

1. $\phi = 1.00$ No reduction in strength due to workmanship, materials and modeling uncertainties. See discussion below for justification

Ref 17 Concrete strength $f_c' = 6442$ psi; mean strength established by 28 day cylinder tests of foundation mat concrete. See additional comments below

Justification of $\phi = 1.00$ for this calculation

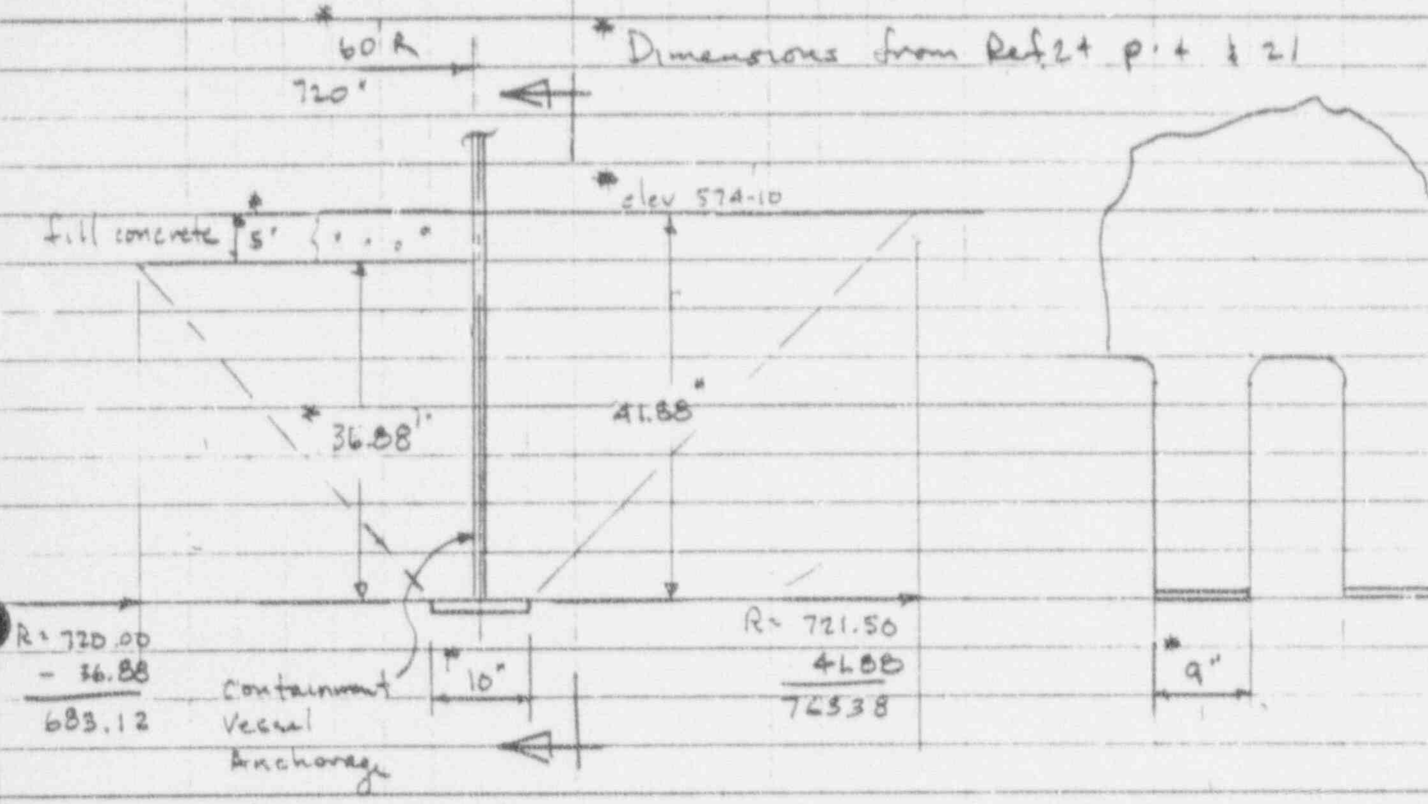
For the IPE analysis, the variations in materials and workmanship and the modeling uncertainties are accounted for by estimated values β_c and β_m in the probability analysis. To use a ϕ factor of < 1.00 would be to include these uncertainties twice which is is misleading and unnecessarily conservative.

Concrete strength

The $f_c' = 6442$ psi is the mean concrete strength based on 145 samples at 28 days. The expected mean strength of the in-place concrete would be significantly higher because of strength gain with age.

CALCULATION OF MEAN FAILURE PRESSURE

Capacity of Containment Anchorage - Concrete (continued)



Capacity/anchor = Projected Area × 4 ϕ $(f_c')^{1/2}$, (ϕ = 1.0)

$$= \left\{ \left[\frac{\pi \times (763.38^2 - 683.12^2)}{288 \text{ anchors}} \right] - (9 \times 10) \right\} \times 4 \times (6442)^{1/2}$$

$$= \{1176 \text{ in}^2\} \times 321 \text{ psi}$$

= 377553 * = 377.55 K/anchor

CALCULATION OF MEAN FAILURE PRESSURE
Anchorage Capacity, Concrete

Pressure Capacity of Containment based on Anchorage Capacity

Ref. 36 From SP660 Volume 3, (pages B4 and C5)

Stress (compression) meridional in 1½" thick plate due to D.L. = 520 psi × $\frac{2.25}{1.5}$ = 780 psi

Stress (tension) meridional in 1½" thick plate due to 15 psi internal pressure = 2187 psi × $\frac{2.25}{1.5}$ = 3280.5 psi *

$\frac{2.25"}{1.5"}$ is ratio of thickness at doubler (2.25") to basic thickness of vessel (1.5")

D.L./anchor = $\frac{780 \text{ psi} \times 1.5" \times 2\pi \times 720.75"}{288 \text{ anchors}}$ = 18.40^k comp.

Uplift/anchor (from 15 psi) = $\frac{3280.5 \times 1.5 \times 2\pi \times 720.75}{288}$ = 77.38^k tension

Pressure capacity based on 377.55^k anchor strength

P (mean failure) = $\frac{(377.55 + 18.4)}{77.38}$ × 15 psi = 76.75 psi

* The calculated tension stress in the C.V. below the stiffeners is conservative (high) because of a conservative assumption in modeling the fixity of the stiffeners in the annulus concrete. See Reference 36, Page A4

CALCULATION OF MEAN FAILURE PRESSURE

Anchorage Capacity, Concrete

Enhancement of Anchorage Capacity By Reinforcing Steel

1. Reinforcing steel ⊥ to direction of anchors.
Most designers feel that reinforcing steel ⊥ to direction of anchors (parallel to surface of concrete) helps to limit tensile strain in the concrete and thus enhances the capability of the concrete to withstand the tensile stresses from anchor pull-out. The code (Ref. 16, B.4) does not provide any means to calculate this enhancement. It does provide a rule (Ref. 16, B.4.2) whereby the phi factor (factor to account for uncertainties) can be revised from 0.65 to 0.85, if the tension stress in the concrete surface based on the uncracked section, is less than $5\sqrt{f'_c}$. Whether the calculated tension stresses in the concrete are this low or not is questionable and would require a fairly complex model to calculate. In any case if we were to count this reinforcing steel as enhancing the anchor capability, it would appear in the estimate of β_m, modeling uncertainty. See discussion and selection of this β_m as discussed later in this Appendix.

2. Reinforcing steel parallel to the anchor
The code does allow the calculation of anchor capacity based on reinforcing steel that is parallel to the anchor and intersecting the assumed pull-out block of the anchor. (Ref 16, B.4.4 and Commentary) The reinforcing steel should be within a distance of anchor depth ÷ 3 of the anchor itself or in our case ≈ 19". The code allows the anchor

Gilbert/Commonwealth ENGINEERS/CONSULTANTS CALCULATION	SUBJECT APPENDIX A C.V. ANCHORAGE - CONCRETE		IDENTIFIER		PAGE 5 OF PAGES
	REV. 0	1	2	3	
	MICROFILMED				
	ORIGINATOR				
	DATE				

CALCULATION OF MEAN FAILURE PRESSURE

Anchorage Capacity, Concrete

2. Reinforcing Steel parallel ... (continued)

capacity to be based on either the concrete tensile strength or the reinforcing steel strength but not the addition of the two. This is because the concrete is assumed to crack at low stress i.e. $4\sqrt{f_c}$ at which the strain is also low and the reinforcement is only stressed to a fraction of its strength = $\frac{E_c}{E_s} \times 4 \times \sqrt{f_c'} \approx 3000$ psi. After the concrete cracks the reinforcing steel must carry the entire load.

Calculating the anchor strength based on 11 reinforcing within a distance of approx. 24"
⇒ 2 ties (4 bars) @ 2'-30' spacing
or 2 ties @ every other anchor

$A_s = 4 \times 1.00 \text{ in}^2 = 4.00 \text{ in}^2$

Tie capacity = $A_s \times$ mean yield of rebar

= 4.00 × 71.9 ksi = 288^k or 144^k/anchor

This is less than the tension capacity of the the concrete anchor = 377^k/anchor.

Use 377^k/anchor based on concrete pull-out, previously calculated.

Mean failure pressure remains at 76.25 psi

APPENDIX A - CONTAINMENT VESSEL ANCHORAGE, CONCRETE

SELECTION OF LOG STANDARD DEVIATIONS FOR CONCRETE ANCHORAGE

B_M. Variation in Material Properties

Reference 17 shows that the mean cylinder test strength (28 days) of the concrete foundation mat is 6442 psi, with a standard deviation of 448 psi. The coefficient of variation is 448/6442 = 0.0695. Therefore,

$$B_M = [\ln(0.0695^2 + 1)]^{0.5} = 0.07$$

B_m. Variation in Modeling

A number of factors can cause the strength of a concrete anchor to vary from the theoretical predicted on the basis of material test strength:

1. concrete in tension in the anchor zone
2. local variations in concrete strength due to placing and curing techniques
3. tolerances in anchor position and concrete surfaces
4. modeling inaccuracy

1. Concrete in Tension:

Tension stresses in the concrete from other loads can reduce the concrete capacity for anchor pull-out. In particular, internal pressures could cause significant bending moments in the concrete mat at the anchor locations. Although these stresses may be small, we can conservatively assign a coefficient of deviation of 0.20 to the concrete anchor capacity to allow for the presence of these tension stresses.

2. Local Variations in Concrete:

Strength of the same concrete mix can vary locally due to differences in placing and curing techniques. These differences are significant for relatively shallow anchors that could be affected by local variations. The depth of the Containment Vessel anchors means that a number of pours is involved in the concrete associated with any one anchor and that any variation over a small thickness or area of the concrete would not significantly affect the strength of the concrete ring holding the 3+ feet deep anchors. Therefore no reduction in strength need be considered for local variations in concrete due to placing and curing techniques

APPENDIX A - CONTAINMENT VESSEL ANCHORAGE, CONCRETE

3. Tolerances on Anchor Position and Concrete Surfaces :

Typical construction tolerances of from 1/16" to 1/2" on anchors position and concrete surfaces can be cumulative and significant for shallow anchors. Tolerances on the Containment Vessel anchors are small, especially when compared to the anchor depth of 36 -. In addition these tolerances have already been considered in the calculation for the mean yield capacity of the concrete ancho We should not assign any additional reduction in concrete streng for construction tolerances.

4. Modeling Inaccuracies:

Loads on anchorages and distribution of loads to individual anchors can vary from the calculated values due to simplifying assumptions in the model. This is certainly true for the Containment Vessel anchor analysis where the load path at the junction of the Containment Vessel with the foundation mat is made complicated and redundant by the presence of the annulus concrete. In effect this concrete transfers a large part of the meridional loads in the Containment Vessel to the Shield Building. This means that the weight of the annulus concrete, the weight of the Shield Building, and the vertical reinforcing steel of the Shield Building are available to help resist uplift from internal pressure in the Containment Vessel. The meridional stresses in Reference 36 (Appendix C) from which the mean failure pressure of the containment anchors is calculated, are based on a model which is conservative in assigning the load transferred through the annulus concrete. In addition, temperature loads on the steel Containment Vessel actually produce downward forces in the containment vessel steel due to the effect of the annulus concrete in restraining growth of the containment vessel.

The significance of the annulus concrete in reducing the loads on the Containment Vessel anchors can be seen in Reference 37, where the anchors are actually in compression when the Containment Vessel experiences 31.82 psi. pressure and 275.8 degree F. temperature. It is apparent that the modeling assumptions used in calculating the mean yield pressure for anchors are all conservative, therefore we should not consider any strength reduction for modeling uncertainties.

APPENDIX A - CONTAINMENT VESSEL ANCHORAGE, CONCRETE

Summary of Log Standard Deviation for Modeling

From the above discussion it is apparent that the only uncertainty tending to reduce anchor capacity is the presence of tension stresses in the concrete foundation mat. The coefficient of variation for modeling is therefore estimated at 0.20 and the log standard deviation for modeling is,

$$B_M = [\ln (0.20^{2} + 1)]^{0.5} = 0.198 \quad \text{Use } 0.20$$

(file pnppcva)

APPENDIX H.2

PNPP IPE APET PROGRAM INPUT DATA FILE

The following PNPP IPE Accident Progression Event Tree (APET) Program Input Data File is developed for the Event Progression Analysis (EVNTRE) code developed at Sandia National Laboratories. Note that the EVNTRE code examines all cases sequentially, and once it finds a "true" outcome for a dependency it stops processing the case dependency logic. A complete discussion of the EVNTRE code is provided in the SAIC NUREG/CR--5174 reference manual (Griesmeyer 1989).

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PERRY IPE APET                              REVISION 0 19JUN1992
68
EVNTRE PROGRAM:  PY_APETO.DAT
1   1.0                    NO. OF SEQUENCES AND SEQUENCE FREQUENCIES
  'ALL SEQUENCES    ' SEQUENCE NAME
$
$
$ PLANT DAMAGE STATE GROUPING LOGIC PARAMETERS *************************** PDS
$
$
1  PDS PARAMETER 1  NOT A CONTAINMENT BYPASS SEQUENCE                  - CNT_BYP
    2  'NoBYPASS'  'EVENT V'
    1     1           2
          1.          0.

2  PDS PARAMETER 2  CONTAINMENT STATUS AT CORE DAMAGE                  - CNT_FAL
    2  'INTACT'    'FAILED'
    2     1           2
  2
    1   1                              $ IF NOT A BYPASS SEQUENCE
        1
      NOBYPASS
$          .9942        .0058          $ SENSITIVITY FOR PASSIVE VENT & ALT S/D
$          .8123        .1877          $ SENSITIVITY FOR ALT S/D ATWS
$          .9449        .0551          $ SENSITIVITY FOR PASSIVE VENT
           .7720        .2280          $ BASECASE
      OTHERWISE
        0.            1.

3  PDS PARAMETER 3  EVENT TYPE: CNTMT INTACT/FAILED AT CORE DAMAGE   - EVENT_TYP
   6 'SBO''LOOP NO H''OTHER TYPE' 'CRIT ATWS ''LOOP & SBO' 'OTHERS'
   2  1        2          3           4           5           6
  2
    1   2                              $ IF CNTMT INTACT AT CORE DAMAGE
        1
      INTACT
$          .1116  0.         .8393       .0491       0.          0.     $ ALT S/D
           .1170  0.         .8830       0.          0.          0.     $ BASECASE
      OTHERWISE                        $ DEFAULT TO NO BRANCH
        0.        0.         0.          .1945       .1910       .6145  $ BASECASE
$       0.        0.         0.          .9826       .0042       .0132  $ PSV VENT
$       0.        0.         0.          .0236       .2314       .7450  $ ALT S/D
$       0.        0.         0.          .8494       .0361       .1145  $ PV+A_S/D

4  PDS PARAMETER 4  INITIAL CNTMT HEAT REMOVAL WITH SUPR POOL COOLING - SUPR_PL
    2  'IN_PL'      'NO_IN_PL'
    2     1              2
  2
     1  3                              $ LOOP NO HVAC WITH CNTMT INTACT
        2
       LOOP_NO_H
        0.            1.
      OTHERWISE                        $ NOT A LOOP NO HVAC
        0.            1.
```

```
5  PDS PARAMETER 5  CONTAINMENT VENT ISOLATED AT RPV FAILURE           - CNT_ISOL
    2  'ISOLATED'  'NOT ISOL'
    2    1           2
  2
    1   3                              $ IF AN SBO SEQUENCE
        1
      SBO
        .9965        .0035
      OTHERWISE                        $ DEFAULT TO ISOLATED
        1.          0.

6  PDS PARAMETER 6  RPV INJECTION FAILURE TIME                         - INJ_F_TIM
    4  'NO INJECT'   'RCIC '   'HPCS '   'NO BRANCH'
    2    1             2          3          4
  5
    2   3             5                $ SBO AND ISOLATED
        1      *      1
      SBO          ISOLATED
        .4347        .2798        .2854       0.

    1   3                              $ SBC (NOT ISOLATED)
        1                              $ (ASSUME SAME SPLIT FRACTION AS ABOVE)
      SBO
        .4347        .2798        .2854       0.

    2   3             4                $ LOOP WITH NO HVAC
        2      *      2                $ NO INITIAL POOL COOLING
      LOOP_NO_H     NO_IN_PL           $ TOKEN VALUES - NOT DEVELOPED
        1.           0.         0.          0.

    2   3             4                $ LOOP WITH NO HVAC
        2      *      1                $ WITH INITIAL POOL COOLING
      LOOP_NO_H     IN_PL              $ TOKEN VALUES - NOT DEVELOPED
        1.           0.          0.          0.
      OTHERWISE                        $ DEFAULT
        1.           0.          0.          0.

7  PDS PARAMETER 7  OFFSITE POWER RECOVERY TIME                        - PWR_R_TIM
    4  'PRIOR RV'  'CNTMT LIM'  'NO RECOV'  'NO BRANCH'
    2    1            2            3           4
  4
    2   3             6                $ SBO WITH INJECTION FAILURE NO INJECTION
        1      *      1
      SBO          NO_INJECT
        .6148         .3567        .0285       0.

    2   3             6                $ SBO WITH INJECTION FAILURE RCIC
        1      *      2
      SBO           RCIC
        .2484         .7006        .0510       0.

    2   3             6                $ SBO WITH INJECTION FAILURE HPCS
        1      *      3
```

```
      SBO           HPCS
         .4231         0.              .5769         0.
   OTHERWISE
         0.            0.            1.              0.

8  PDS PARAMETER 8  CONTAINMENT HEAT REMOVAL WITH RHR SPRAY LOOP          - SPRAY
     3  'RHR_SPRY'    'RHR_POOL'     'NO_RHR'
     2    1             2              3
  10
     2    3             5                 $ SBO WITH LOSS ISOLATION
          1      *      2
        SBO          NOT_ISOL
          0.            0.              1.

     3    3             6              7  $ SBO WITH INJECTION FAILURE NO INJECT,
          1      *      1      *       1  $ POWER RECOVERY PRIOR TO RPV FAILURE
        SBO      NO_INJECT     PRIOR_RV
          .8312         0.             .1688

     3    3             6              7  $ SBO WITH INJECTION FAILURE NO INJECT,
          1      *      1      *       2  $ POWER RECOVERY PRIOR TO CNTMT LIMIT
        SBO      NO_INJECT     CNTMT_LMT
          .8284         0.             .1716

     3    3             6              7  $ SBO WITH INJECTION FAILURE RCIC,
          1      *      2      *       1  $ POWER RECOVERY PRIOR TO RPV FAILURE
        SBO           RCIC      PRIOR_RV
          .9479         0.             .0521

     3    3             6              7  $ SBO WITH INJECTION FAILURE RCIC,
          1      *      2      *       2  $ POWER RECOVERY PRIOR TO CNTMT LIMIT
        SBO           RCIC     CNTMT_LMT
          .9283         0.             .0717

     3    3             6              7  $ SBO WITH INJECTION FAILURE HPCS,
          1      *      3      *       1  $ POWER RECOVERY PRIOR TO RPV FAILURE
        SBO           HPCS      PRIOR_RV
          .9090         0.             .0910

     1    3                               $ OTHER TYPES (NON SBO/LOOP)
          3
        OTHER_TYPES
          .5171         0.             .4829

     2    3             2                 $ ATWS WITH CNTMT FAILURE PRIOR TO CD
          4      *      2
        CRIT_ATWS    FAILED
          .3025         0.             .6975

                                          $ ATWS NO CNTMT FAILURE PRIOR TO CD
                                          $ SENSIT  TY ALT S/D ATWS
                                          $ (ASS    PLIT FRACTION .1 LOWER THAN
                                          $ THE     ABOVE.  HOWEVER, HAS NO
                                          $ IMPAC. ON SOURCE TERM RELEASE)
```

```
      2    3          2
           4    *     1
         CRIT_ATWS   INTACT
           .2         0.           .8
         OTHERWISE
           0.         0.          1.

9  PDS PARAMETER 9  CONTAINMENT HEAT REMOVAL WITH VENT                    - VENT
   2 'VENT'        'NO VENT'
   2    1            2
 11
                                        $ SBO WITH INJECTION FAILURE NO INJECT,
                                        $ POWER RECOVERY PRIOR TO RPV FAILURE,
                                        $ NO RHR CNTNT HEAT REMOVAL
      4    3          6          7          8
           1    *     1    *     1    *     3
          SBO      NO_INJECT   PRIOR_RV    NO_RHR
           1.         0.
                                        $ SBO WITH INJECTION FAILURE NO INJECT,
                                        $ POWER RECOVERY PRIOR TO CNTMT LIMIT,
                                        $ NO RHR CNTMT HEAT REMOVAL
      4    3          6          7          8
           1    *     1    *     2    *     3
          SBO      NO_INJECT   CNTMT_LIM   NO_RHR
           1.         0.

      3    3          6          7  $ SBO WITH INJECTION FAILURE NO INJECT,
           1    *     1    *     3  $ NO POWER RECOVERY
          SBO      NO_INJECT  NO_RECOV
            .8676      .1324
$          1.         0.            $ SENSITIVITY FOR PASSIVE VENT

                                        $ SBO WITH INJECTION FAILURE RCIC,
                                        $ POWER RECOVERY PRIOR RV FAILURE,
                                        $ NO RHR CNTMT HEAT REMOVAL
      4    3          6          7          8
           1    *     2    *     1    *     3
          SBO        RCIC      PRIOR_RV    NO_RHR
           1.         0.

                                        $ SBO WITH INJECTION FAILURE RCIC,
                                        $ POWER RECOVERY PRIOR CNTMT LIM,
                                        $ NO RHR CNTMT HEAT REMOVAL
      4    3          6          7          8
           1    *     2    *     2    *     3
          SBO        RCIC     CNTMT_LIM    NO_RHR
           1.         0.

      3    3          6          7  $ SBO WITH INJECTION FAILURE RCIC,
           1    *     2    *     3  $ NO POWER RECOVERY
          SBO        RCIC     NO_RECOV
            .1824      .8176
$          1.         0.            $ SENSITIVITY FOR PASSIVE VENT
```

```
                                      $ SBO WITH INJECTION FAILURE HPCS,
                                      $ POWER RECOVERY PRIOR RV FAILURE,
                                      $ NO RHR CNTMT HEAT REMOVAL
    4   3           6           7          8
        1     *     3     *     1     *    3
       SBO         HPCS       PRIOR_RV     NO_RHR
        1.          0.

    3   3           6           7  $ SBO WITH INJECTION FAILURE HPCS,
        1     *     3     *     3  $ NO POWER RECOVERY
       SBO         HPCS     NO_RECOV
        .7976       .2024
$       1.          0.             $ SENSITIVITY FOR PASSIVE VENT

    2   3           8              $ OTHER TYPES (NON SBO/LOOP),
        3     *     3              $ NO RHR CNTMT HEAT REMOVAL
       OTHER_TYPES  NO_RHR
$       1.          0.             $ SENSITIVITY FOR PASSIVE VENT
        .8394       .1606

    2   3           2              $ CRIT ATWS WITH CONT INTACT AT CD
        4     *     1              $ ALTERNATE SHUTDOWN ATWS
       CRIT_ATWS   INTACT          $ SENSITIVITY ALT S/D ATWS
        1.          0.
       OTHERWISE                   $ VENTING UNNECESSARY OR IRRELAVENT
        0.          1.

10 PDS PARAMETER 10  LATE IN-VESSEL INJECT & PEDESTAL CAVITY SUPPLY    - LAT_INJ
    2 'LAT INJ'    'NO LT INJ'
    2   1           2
 24
                                      $ SBO WITH INJECTION FAILURE NO INJECT,
                                      $ POWER RECOVERY PRIOR TO RPV FAILURE
                                      $ CNTMT HEAT REMOVAL WITH RHR SPRAY
    4   3           6           7          8
        1     *     1     *     1     *    1
       SBO        NO_INJECT     PRIOR_RV    RHR_SPRY
        .9866       .0134
                                      $ SBO WITH INJECTION FAILURE NO INJECT,
                                      $ POWER RECOVERY PRIOR TO RPV FAILURE
                                      $ CNTMT HEAT REMOVAL WITH VENT
    4   3           6           7          9
        1     *     1     *     1     *    1
       SBO        NO_INJECT     PRIOR_RV     VENT
        .9973       .0027
                                      $ SBO WITH INJECTION FAILURE NO INJECT,
                                      $ POWER RECOVERY PRIOR TO CNTMT LIMIT,
                                      $ RHR CNTMT HEAT REMOVAL WITH SPRAYS
    4   3           6           7          8
        1     *     1     *     2     *    1
       SBO        NO_INJECT    CNTMT_LIM    RHR_SPRY
        .3393       .6607
```

```
                                          $ SBO WITH INJECTION FAILURE NO INJECT,
                                          $ POWER RECOVERY PRIOR TO CNTMT LIMIT,
                                          $ NO RHR CNTMT HEAT REMOVAL
 4   3             6             7             8
     1     *       1     *       2      *      3
    SBO        NO_INJECT    CNTMT_LIM      NO_RHR
     .6887         .3113

                                          $ SBO WITH INJECTION FAILURE NO INJECT,
                                          $ NO POWER RECOVERY,
                                          $ CNTMT VENT
 4   3             6             7             9
     1     *       1     *       3      *      1
    SBO        NO_INJECT    NO_RECOV         VENT
     .9363         .0637

                                          $ SBO WITH INJECTION FAILURE NO INJECT,
                                          $ NO POWER RECOVERY,
                                          $ NO CNTMT VENT
 4   3             6             7             9
     1     *       1     *       3      *      2
    SBO        NO_INJECT    NO_RECOV       NO_VENT
     1.            0.

                                          $ SBO WITH INJECTION FAILURE RCIC
                                          $ POWER RECOVERY PRIOR RV FAILURE
                                          $ RHR CNTMT HEAT REMOVAL SPRAY
 4   3             6             7             8
     1     *       2     *       1      *      1
    SBO           RCIC        PRIOR_RV     RHR_SPRY
     1.            0.

                                          $ SBO WITH INJECTION FAILURE RCIC
                                          $ POWER RECOVERY PRIOR RV FAILURE
                                          $ CNTMT VENT
 4   3             6             7             9
     1     *       2     *       1      *      1
    SBO           RCIC        PRIOR_RV       VENT
     1.            0.

 3   3             6             7        $ SBO WITH INJECTION FAILURE RCIC,
     1     *       2     *       2        $ POWER RECOVERY PRIOR TO CNTMT LIMIT
    SBO           RCIC     CNTMT_LIM
     C.            1.

                                          $ SBO WITH INJECTION FAILURE RCIC,
                                          $ NO POWER RECOVERY, CNTMT VENT
 4   3             6             7             9
     1     *       2     *       3      *      1
    SBO           RCIC      NO_RECOV         VENT
     0.            1.

                                          $ SBO WITH INJECTION FAILURE RCIC,
```

```
                                         $ NO POWER RECOVERY, NO CNTMT VENT
4   3              6              7              9
    1      *       2      *       3      *       2
   SBO            RCIC         NO_RECOV       NO_VENT
    0.             1.

                                         $ SBO WITH INJECTION FAILURE HPCS,
                                         $ POWER RECOVERY PRIOR TO RPV FAILURE
                                         $ CNTMT HEAT REMOVAL WITH RHR SPRAY
4   3              6              7              8
    1      *       3      *       1      *       1
   SBO            HPCS         PRIOR_RV       RHR_SPRY
    1.             0.

                                         $ SBO WITH INJECTION FAILURE HPCS,
                                         $ POWER RECOVERY PRIOR TO RPV FAILURE
                                         $ CNTMT VENT
4   3              6              7              9
    1      *       3      *       1      *       1
   SBO            HPCS         PRIOR_RV        VENT
    1.             0.

                                         $ SBO WITH INJECTION FAILURE HPCS,
                                         $ NO POWER RECOVERY, CNTMT VENT
4   3              6              7              9
    1      *       3      *       3      *       1
   SBO            HPCS         NO_RECOV        VENT
     .8432          .1568

                                         $ SBO WITH INJECTION FAILURE HPCS,
                                         $ NO POWER RECOVERY, NO CNTMT VENT
4   3              6              7              9
    1      *       3      *       3      *       2
   SBO            HPCS         NO_RECOV       NO_VENT
     .7460          .2540

2   3              5                     $ SBO, CNTMT NOT ISOLATED
    1      *       2
   SBO          NOT_ISOL
     .5046          .4954

2   3              8                     $ OTHER TYPES (NON SBO/LOOP),
    3      *       1                     $ AND RHR CNTMT HEAT REMOVAL WITH SPRAY
 OTHER_TYPES   RHR_SPRY
     .9969         .0031

2   3              9                     $ OTHER TYPES (NON SBO/LOOP),
    3      *       1                     $ CNTMT HEAT REMOVAL WITH VENT
 OTHER_TYPES     VENT
     .9716          .0284

2   3              9                     $ OTHER TYPES (NON SBO/LOOP),
    3      *       2                     $ NO CNTMT HT REMOVAL, NO VENT
 OTHER_TYPES   NO_VENT
```

```
      .00022         .99978

   2   3              2                $ CRIT ATWS WITH CONT INTACT AT CD
       4      *       1                $ ALTERNATE SHUTDOWN ATWS
     CRIT_ATWS     INTACT              $ SENSITIVITY ALT S/D ATWS
       1.             0.

   2   3              2                $ CRIT ATWS WITH CONT FAILED PRIOR CD
       4      *       2                $ (ASSUMED AVAILABILITY OF LATE INJECT
     CRIT_ATWS     FAILED              $  PRIOR TO CNTMT FAILURE IMPACT)
       .95            .05

   2   2              3                $ CONTAINMENT FAILED PRIOR CORE DAMAGE,
       2      *       5                $ SBO OR LOOP EVENT TYPE
     FAILED       SBO_&_LOOP
       .4797          .5203

   2   2              3                $ CONTAINMENT FAILED PRIOR CORE DAMAGE,
       2      *       6                $ ALL OTHERS TYPE SEQUENCES
     FAILED        OTHERS
       .3137          .6863
    OTHERWISE
       0.             1.

11 PDS PARAMETER 11  RPV DEPRESSURIZED DURING CORE DAMAGE            - RX_PRES
   2 'LOW PRES'  'HI PRES'
   2   1              2
 25
                                       $ SBO WITH INJECTION FAIL NO INJECT,
                                       $ POWER RECOVERY PRIOR RV FAILURE,
                                       $ VENT AND LATE INJECTION
   5   3              6              7              9              10
       1      *       1      *       1      *       1      *       1
     SBO           NO_INJECT     PRIOR_RV         VENT           LAT_INJ
       .4650          .5350

                                       $ SBO WITH INJECTION FAIL NO INJECT,
                                       $ POWER RECOVERY PRIOR RV FAILURE,
                                       $ VENT AND NO LATE INJECTION
   5   3              6              7              9              10
       1      *       1      *       1      *       1      *       2
     SBO          NO_INJECT      PRIOR_RV         VENT          NO_LT_INJ
       1.             0.

                                       $ SBO WITH INJECTION FAIL NO INJECT,
                                       $ POWER RECOVERY BEFORE CNTMT LIMIT,
                                       $ VENT AND LATE INJECTION
   5   3              6              7              9              10
       1      *       1      *       2      *       1      *       1
     SBO          NO_INJECT      CNTMT_LMT        VENT          LAT_INJ
       .2284          .7716

                                       $ SBO WITH INJECTION FAIL NO INJECT,
                                       $ NO POWER RECOVERY, VENT,
```

```
                              $ LATE INJECTION
5   3            6          7          9          10
    1     *      1    *     3     *    1     *    1
   SBO       NO_INJECT   NO_RECOV     VENT      LAT_INJ
    .9904       .0096

                              $ SBO WITH INJECTION FAIL NO INJECT,
                              $ NO POWER RECOVERY, VENT,
                              $ NO LATE INJECTION
5   3            6          7          9          10
    1     *      1    *     3     *    1     *    2
   SBO       NO_INJECT   NO_RECOV     VENT      NO_LT_INJ
   1.            0.

                              $ SBO WITH INJECTION FAIL NO INJECT,
                              $ NO POWER RECOVERY, NO VENT,
                              $ NO LATE INJECTION
5   3            6          7          9          10
    1     *      1    *     3     *    2     *    1
   SBO       NO_INJECT   NO_RECOV    NO_VENT    LAT_INJ
   1.            0.

                              $ SBO WITH INJECTION FAILURE RCIC,
                              $ NO POWER RECOVERY, NO VENT
                              $ NO LATE INJECTION
5   3            6          7          9          10
    1     *      2    *     3     *    2     *    2
   SBO          RCIC     NO_RECOV    NO_VENT    NO_LT_INJ
   0.            1.

                              $ SBO WITH INJECTION FAILURE HPCS
                              $ NO POWER RECOVERY, VENT
                              $ LATE INJECTION
5   3            6          7          9          10
    1     *      3    *     3     *    1     *    1
   SBO          HPCS     NO_RECOV     VENT      LAT_INJ
    .8014       .1986

                              $ SBO WITH INJECTION FAILURE HPCS
                              $ NO POWER RECOVERY, VENT
                              $ NO LATE INJECTION
5   3            6          7          9          10
    1     *      3    *     3     *    1     *    2
   SBO          HPCS     NO_RECOV     VENT      NO_LT_INJ
    .7704       .2296

                              $ SBO WITH INJECTION FAILURE HPCS
                              $ NO POWER RECOVERY, NO VENT
                              $ LATE INJECTION
5   3            6          7          9          10
    1     *      3    *     3     *    2     *    1
   SBO          HPCS     NO_RECOV    NO_VENT    LAT_INJ
    .9768       .0232
```

```
                                          $ SBO WITH INJECTION FAILURE HPCS
                                          $ NO POWER RECOVERY, NO VENT
                                          $ NO LATE INJECTION
5  3           6          7           9          10
   1     *     3     *    3     *     2     *     2
  SBO         HPCS      NO_RECOV     NO_VENT    NO_LT_INJ
   1.          0.

3  3           5          10  $ SBO WITH LOSS ISOLATION,
   1     *     2     *     1  $ LATE INJECTION AVAILABLE
  SBO       NOT_ISOL   LAT_INJ
   .6932       .3068

3  3           5          10  $ SBO WITH LOSS ISOLATION,
   1     *     2     *     2  $ NO LATE INJECTION AVAILABLE
  SBO       NOT_ISOL  NO_LT_INJ
   1.          0.

1  3                          $ ALL OTHER SBOs ARE DEPRESSURIZED
   1
  SBO
   1.          0.

3  3           8          10  $ OTHER TYPES (NON SBO/LOOP),
   3     *     1     *     1  $ RHR HT REMOVAL W/ SPRAY, LATE INJECT
  OTHER_TYPES  RHR_SPRY  LAT_INJ
   .9924       .0076

                              $ OTHER TYPES (NON SBO/LOOP)
                              $ NO RHR, VENTING, AND LATE INJECTION
4  3           8          9           10
   3     *     3     *    1     *     1
  OTHER_TYPES  NO_RHR     VENT        LAT_INJ
   .9950       .0050

                              $ OTHER TYPES (NON SBO/LOOP),
                              $ NO RHR, NO VENTING, AND LATE INJECTION
4  3           8          9           10
   3     *     3     *    2     *     1
  OTHER_TYPES  NO_RHR    NO_VENT      LAT_INJ
   1.          0.

                              $ OTHER TYPES (NON SBO/LOOP),
                              $ NO RHR, NO VENTING, AND NO LAT INJECT
4  3           8          9           10
   3     *     3     *    2     *     2
  OTHER_TYPES  NO_RHR    NO_VENT    NO_LT_INJ
   .9255       .0745

2  3           2              $ CONTAINMENT INTACT, CRITICAL ATWS
   4     *     1              $ SENSITIVITY ALT S/D
  CRIT_ATWS    INTACT
   1.          0.
```

```
   3   2            3            8  $ CNTMT FAILED PRIOR TO CORE DAMAGE
       2     *      4     *      1  $ CRITICAL ATWS WITH RHR SPRAY
     FAILED      CRIT_ATWS    RHR_SPRY
      .9918          .0082

   3   2            3            8  $ CNTMT FAILED PRIOR TO CORE  DAMAGE
       2     *      4     *      3  $ CRITICAL ATWS WITH NO RHR
     FAILED      CRIT_ATWS     NO_RHR
      .7458          .2542

   3   2            3           10  $ CNTMT FAILED PRIOR TO CORE DAMAGE
       2     *      5     *      1  $ LOOP & SBO, AND LATE INJECTION
     FAILED     LOOP_&_SBO    LAT_INJ
      .99940         .00060

   3   2            3           10  $ CNTMT FAILED PRIOR TO CORE DAMAGE
       2     *      5     *      2  $ LOOP & SBO, AND NO LATE INJECTION
     FAILED    LOOP_&_SBO    NO_LT_INJ
      .8882          .1118

   2   2            2               $ ALL REMAINING PDS ARE LOW PRESSURE
       2     +      1               $ WITH THE CNTMT FAILED/INTACT ARE
     FAILED       INTACT            $ LOW PRESSURE SEQUENCES
      1.             0.
    OTHERWISE
      1.             0.


$
$
$ CET EVENT 1  DEBRIS COOLED IN-VESSEL ****************************** INV_COOL
$
$
12   LATE LOW PRESSURE RPV INJECTION AVAILABLE                          - LATE_INJ
   3  'WATER_INJ'  'NO_INJECT'    'CRITICAL'
   2    1            2               3
  4
   1   10                           $ LATE IN-VESSEL INJECTION FAILED
        2
    LAT_INJ
        0.            1.            0.

   1    3                           $ NON-SHUTDOWN ATWS SEQUENCES
        4
    CRIT_ATWS
        0.            0.            1.

   1   10                           $ LATE IN-VESSEL INJECTION AVAILABLE
        1
    LAT_INJ
        1.            0.            0.
    OTHERWISE                       $ SHOULD NEVER REACH THIS CASE
        0.            1.            0.
```

```
13   RPV DEPRESSURIZED DURING CORE DAMAGE                                   - RX_PRESS
   2 'LOW_PRES'    'HI_PRES'
   2    1            2
  2
   1  11                              $ RPV HAS BEEN DEPRESSURIZED
      1
    LOW_PRES
      1.            0.
   OTHERWISE                          $ RPV HAS NOT BEEN DEPRESSURIZED
      0.            1.

14   DEBRIS MASS MOLTEN AT RPV FAILURE                                     - MOLTEN_VB
   2 'LG_DEB'      'SM_DEB'
   2    1            2
  2
   2  12            13                $ IF NO WATER INJECTION AVAILABLE, OR
      2      +       2                $ (WATER INJ AVAILABLE, BUT)
    NO_INJECT      HI_PRES            $ RPV NOT DEPRESSURIZED
     0.1           0.9
   OTHERWISE                          $ WATER INJECTION AVAILABLE
     .025          .975

15   DEBRIS COOLED IN-VESSEL - INV_COOL
   2 'COOL_INV'  'NCOOL_INV'
   2    1            2
  5
   2  12             2                $ CRITICAL REACTOR (NOT SHUTDOWN)
      3      *       2                $ CONTAINMENT FAILED PRIOR CORE DAMAGE
    CRITICAL     FAILED
      0.             1.

   1  12                              $ CRITICAL REACTOR(NOT SHUTDOWN),
      3                               $ ALTERNATE SHUTDOWN ATWS
     CRITICAL
      1.             0.

   3  12            13           14   $ LATE INJECTION, RPV DEPRESSURIZED,
      1      *       1      *     1   $ LARGE MOLTEN DEBRIS MASS IN LOWER HEAD
     LAT_INJ       LOW_PRES    LG_DEB
       .5            .5

   3  12            13           14   $ LATE INJECTION, RPV DEPRESSURIZED,
      1      *       1      *     2   $ SMALL MOLTEN DEBRIS MASS IN LOWER HEAD
     LAT_INJ       LOW_PRES    SM_DEB
      .75           .25
    OTHERWISE                         $ ALL OTHER CASES
      0.             1.


$
$
$ CET EVENT 3  MODE OF CONTAINMENT FAILURE BEFORE RPV FAILURE ******* V_ERLY_CF
$
```

```
$
16   HYDROGEN IGNITION SYSTEM AVAILABLE                                       - H2_IGN
   2  'HIS_OFF'   'HIS_ON'
   2     1           2
  2
   2   3            3                    $ IF NO LOSS OF AC POWER
     /1      *     /5
    not SBO    not LOOP_&_SBO
       .005          .995
    OTHERWISE                            $ LOSS OF AC POWER
       1.            0.
$      0.            1.                  $ SENSITIVITY FOR HIS BACKUP POWER

17   CONTAINMENT VENT ISOLATED BEFORE RPV FAILURE                               - ISOL
   2  'ISOLATED'  'NOT_ISOL'
   2    1           2
  2
   1   5                                 $ CNTMT VENT ISOLATED
       1                                 $ BEFORE VESSEL FAILURE
     ISOLATED
       1.
         0.
    OTHERWISE                            $ SBO WITH CNTMT VENT NOT ISOLATED
       0.            1.

18   MODE OF RHR SPRAY OPERATION EARLY                                      - RHR_MODE
   3  'CONTROLD'  'SPRAY   '  'NO_SPRAY'
   2    1           2            3
  5
   1    8                                $ RHR SPRAYS NOT AVAILABLE
      /1
     nRHR_SPRY
        0.           0.          1.

   2    3                                $ SBO EVENT TYPE,
        1     *     /1                   $ POWER NOT RECOVERD PRIOR TO RPV FAILURE
      SBO      not PRIOR_RV
        0.           0.          1.

   2    3           8                    $ NON-SBO EVENT TYPES,
      /1      *     1                    $ RHR SPRAYS ARE AVAILABLE
     not SBO      RHR_SPRY
        0.           1.          0.

   3    3           7           8        $ SBO EVENTS,
        1     *     1     *     1        $ POWER RECOVERED PRIOR TO VESSEL FAILURE
      SBO         PRIOR_RV    RHR_SPRY   $ SPRAYS ARE AVAILABLE
        0.           1.          0.
     OTHERWISE                           $ SHOULD NEVER GO THIS PATH
        0.           0.          1.

19   CONTAINMENT STEAM CONCENTRATION BEFORE RPV FAILURE                     - ST_CONC
   6  '0-15%'  '15-25%'  '25-35%'  '35-45%'  '45-55%'  '> 55%'
   2    1         2         3         4         5         6
```

```
6
1   18                               $ DESIGN SPRAY OPERATION
    2
  SPRAY
    1.        0.        0.        0.        0.        0.

2    3         6                     $ SBO EVENT,
     1    *    1                     $ INJECTION FAILURE TIME: NO INJECTION
   SBO        NO_INJECT
     1.       0.        0.        0.        0.        0.

2    3         6                     $ SBO EVENT,
     1    *    2                     $ INJECTION FAILURE TIME: RCIC
   SBO         RCIC
     0.       .51       .49       0.        0.        0.

2    3         6                     $ SBO EVENT,
     1    *    3                     $ INJECTION FAILURE TIME: HPCS
   SBO         HPCS
     0.       0.        0.        0.        0.        1.

1    2                               $ CONTAINMENT NOT INTACT AT CORE
     2                               $ DAMAGE
   FAILED
     0.       0.        0.        0.        0.        1.
   OTHERWISE                         $ FOR ALL OTHER EVENTS
     1.       0.        0.        0.        0.        0.

20   FRACTION ZIRCONIUM INVENTORY REACTED IN-VESSEL              - H2_INV
  3  '33%'     '22%'     '11%'
  2   1         2         3
 4
  2    3         6                   $ SBO EVENT
       1    *    1                   $ INJECTION FAILURE TIME: NO INJECTION
     SBO        NO_INJECT
       0.       0.        1.

  2    3         6                   $ SBO EVENT
       1    *    2                   $ INJECTION FAILURE TIME: RCIC
     SBO         RCIC
       0.       .13       .87

  2    3         6                   $ SBO EVENT
       1    *    3                   $ INJECTION FAILURE TIME: HPCS
     SBO         HPCS
       0.       0.        1.
     OTHERWISE                       $ BOUND OTHERS WITH DISTRIBUTION
       0.       .13       .87        $ FOR SBO WITH RCIC INJECTION FAILURE

21   SMALL HYDROGEN BURNS AT LOW H2 CONCENTRATION                - SM_BURN
  2  'NO_SMALL'  'SMALL_BRN'
  2   1            2
 6
```

```
1  19                                  $ IF > 55% STEAM, THEN INERT
    6
   > 55%
    1.          0.

1  16                                  $ HYDROGEN IGNITION SYSTEM AVAILABLE
    2
   HIS_ON
     0.           1.

2   3           16                     $ NON-SBO EVENT,
   /1      *     1                     $ (AC POWER NEVER LOST),
   not SBO      HIS_OFF                $ HYDROGEN IGNITION SYSTEM INOP
     .25          .75

2   3            7                     $ SBO EVENT,
    1      *     1                     $ POWER RECOVERED PRIOR TO RPV FAILURE
   SBO         PRIOR_RV                $ IGNITERS ASSUMED NOT AVAILABLE
     .96          .04

2   3            7                     $ SBO EVENT,
    1      *    /1                     $ NO POWER RECOVERY PRIOR TO RPV FAILURE
   SBO      not PRIOR_RV
    1.           0.
  OTHERWISE                            $ SHOULD NEVER GO THIS PATH
    1.           0.

22   LARGE H2 BURN DURING CORE DAMAGE                                - LG_BRN
   2  'NO_BURN'  'LG_BURN'
   4    1           2
 19
   1   2                               $ CONTAINMENT FAILED PRIOR TO
       2                               $ CORE DAMAGE
     FAILED
       1.           0.
    2
       1      .15     .15              $ CONTAINS COND PROB OF ANCHORAGE FAILURE
       3     0.      0.

   1  21                               $ SMALL BURN IGNITED
       2
    SMALL_BRN
       1.           0.
    2
       1     0.      0.
       3     0.      0.

   1   19                              $ IF > 55% STEAM, THEN INERT
        6
       > 55%
       1.           0.
    2
       1      .15     .15              $ CONTAINS COND PROB OF ANCHORAGE FAILURE
       3     0.      0.
```

```
2  19            20                $ 0-15% STEAM AND
    1      *      1                $ 33% ZIRC OXIDATION = 25.6% H2
  0-15%          33%
     .5             .5
 2
    1      0.     142.
    3      0.       9.

2  19            20                $ 0-15% STEAM AND
    1      *      2                $ 22% ZIRC OXIDATION = 18.8% H2
  0-15%          22%
     .5             .5
 2
    1      0.      98.
    3      0.       6.

2  19            20                $ 0-15% STEAM AND
    1      *      3                $ 11% ZIRC OXIDATION = 10.5% H2
  0-15%          11%
     .72            .28
 2
    1      0.      42.
    3      0.       4.

2  19            20                $ 15-25% STEAM AND
    2      *      1                $ 33% ZIRC OXIDATION = 22.1% H2
 15-25%          33%
     .5             .5
 2
    1      0.     123.
    3      0.      14.

2  19            20                $ 15-25% STEAM AND
    2      *      2                $ 22% ZIRC OXIDATION = 16.2% H2
 15-25%          22%
     .5             .5
 2
    1      0.      87.
    3      0.      11.

2  19            20                $ 15-25% STEAM AND
    2      *      3                $ 11% ZIRC OXIDATION = 9.0% H2
 15-25%          11%
     .72            .28
 2
    1      0.      37.
    3      0.       9.

2  19            20                $ 25-35% STEAM AND
    3      *      1                $ 33% ZIRC OXIDATION = 19.4% H2
 25-35%          33%
     .5             .5
 2
```

```
    1       0.       113.
    3       0.        20.

2  19              20                   $ 25-35% STEAM AND
    3      *        2                   $ 22% ZIRC OXIDATION = 14.2% H2
  25-35%           22%
     .61             .39
 2
    1       0.        80.
    3       0.        16.

2  19              20                   $ 25-35% STEAM AND
    3      *        3                   $ 11% ZIRC OXIDATION = 7.9% H2
  25-35%           11%
     .75             .25
 2
    1       0.        36.
    3       0.        13.

2  19              20                   $ 35-45% STEAM AND
    4      *        1                   $ 33% ZIRC OXIDATION = 16.6% H2
  35-45%           33%
     .56             .44
 2
    1       0.       104.
    3       0.        27.

2  19              20                   $ 35-45% STEAM AND
    4      *        2                   $ 22% ZIRC OXIDATION = 12.2% H2
  35-45%           22%
     .67             .33
 2
    1       0.        75.
    3       0.        23.

2  19              20                   $ 35-45% STEAM AND
    4      *        3                   $ 11% ZIRC OXIDATION = 6.8% H2
  35-45%           11%
     .77             .23
 2
    1       0.        37.
    3       0.        19.

2  19              20                   $ 45-55% STEAM AND
    5      *        1                   $ 33% ZIRC OXIDATION    = 13.8% H2
  45-55%           33%
     .61             .39
 2
    1       0.        98.
    3       0.        37.

2  19              20                   $ 45-55% STEAM AND
    5      *        2                   $ 22% ZIRC OXIDATION    = 10.2% H2
  45-55%           22%
```

```
     .72           .28
  2
     1     0.      76.
     3     0.      32.

 2  19            20                 $ 45-55% STEAM AND
     5      *      3                 $ 11% ZIRC OXIDATION    = 5.7% H2
   45-55%         11%
     .77           .23
  2
     1     0.      41.
     3     0.      27.
   OTHERWISE                         $ SHOULD NEVER GO THIS PATH
     1.           0.
  2
     1     0.      0.
     3     0.      0.


23   H2 DETONATION CONTAINMENT FAILURE                                - H2_DET
   2   'DET_CF'       'NO'
   2      1            2
 10
   1  22                             $ NO LARGE BURN IGNITED
      1
    NO_BURN
      0.           1.

   3  19           19          19    $ IF > 35% STEAM,
      4     +      5    +       6    $ THEN INERT TO DETONATIONS
    35-45%       45-55%       > 55%
      0.           1.

   1  20                             $ IF 11% ZIRC OXIDATION, THEN [H2] < 12%;
      3                              $ DETONATIONS HAVE NEGLIGIBLE PROBABILTY
     11 %
      0.           1.

                                     $ STEAM CONC > 35%  (HIGH),
                                     $ H2 CONC 12 - 16 %,
                                     $ POWER RECOVERED PRIOR TO RPV FAILURE,
                                     $ RHR IS AVAILABLE IN SPRAY MODE,
                                     $ NO EARLY INJECTION FAILURE SINCE THE
                                     $ STEAM CONC IS LOW FOR THAT SEQUENCE
   6  19      20      3      7       18       6
      1   *    2   *  1   *  1   *    2   *   /1
     0-15%    22%    SBO   PRIOR_RV  SPRAY  not NO_INJECT
       .022           .978

   2  19           20                $ STEAM CONC  < 35 %  (LOW)
      3     *      2                 $ H2 CONC 12 - 16 %
    25-35%        22%
      0.           1.
```

```
  3  19           19           20  $ STEAM CONC  < 35 %  (LOW),
     (1      +     2)     *     2  $ H2 CONC 16 - 20 %
     0-15%        15-25%       22%
       .16          .84

                                   $ STEAM CONC  < 35 %   HIGH
                                   $ H2 CONC > 20%
                                   $ IF AC POWER LOST AND RECOVERED
                                   $ PRIOR TO VESSEL FAILURE
                                   $ AND RHR IS AVAILAELE IN SPRAY MODE
                                   $ NO EARLY INJECTION FAILURE SINCE THE
                                   $ STEAM CONC IS LOW FOR THAT SEQUENCE
  6  19      20       3       7       18       6
      1   *   1   *   1   *   1   *    2   *   /1
     0-15%   33%     SBO   PRIOR_RV   SPRAY  not NO_INJECT
       .025            .975

  3  19           19           20  $ STEAM CONC  < 35%  (LOW)
     (1      +     2)     *     1  $ H2 CONC > 20%
     0-15%        15-25%       33%
       .27          .73

  2  19           20               $ STEAM CONC  < 35 %  (LOW)
      3      *     1               $ H2 CONC 16 - 20 %
    25-35%        33%
      .16           .84
    OTHERWISE                      $ SHOULD NEVER REACH THIS CASE
      0.            1.

24   CONTAINMENT FAILURE BEFORE RPV FAILURE
   2   'FAILURE'   'NO_FAILUR'
   6       1            2
 3
   1   2                           $ CONTAINMENT NOT INTACT AT CORE DAMAGE
       2                           $ IMPLIES STEAM OVER PRESSURE
     FAILED                        $ PRIOR TO CORE DAMAGE
       1
          BURNPRES
             MAX
             GETHRESH  1 -1.
      SET PROB OF CNTMT FAILURE FOR FOR SLOW OP (FORCE FIRST BRANCH)

   1  23                           $ H2 DETONATION CONTAINMENT FAILURE
       1
      DET_CF
       1    1
         BURNPRES
         MAX
         GETHRESH 1 -1.
       H2 DETONATION   -- FORCE FIRST BRANCH
     OTHERWISE                     $ CONTAINMENT INTACT PRIOR CORE DAMAGE
       1    1
          BURNPRES
             FUN-F_BURN
```

```
              EQUAL  0
        CALCULATE PROB OF CNTMT FAILURE (ANY MODE) GIVEN EXPECTED BURN PRESSURE

25   MODE OF CONTAINMENT FAILURE BEFORE RPV FAILURE                     - V_ER_CF
   2   'ANCHORAGE'  'PN-D/NoCF'
   6        4           2
  4


   1   2                              $ CNTMT FAILED AT CORE DAMAGE INITIATION
       2                              $ EARLY SLOW OVER PRESSURE
     FAILED                           $ IN THIS CASE BURNPRES CONTAINS COND
       1   1                          $ PROB OF ANCHORAGE FAILURE (0.15)
         BURNPRES
         MAX
         EQUAL 0
       CNTMT NOT INTACT AT CORE DAMAGE INITIATION

   1  23                              $ H2 DETONATION CONTAINMENT FAILURE
       1
      DET_CF
       1   1
         BURNPRES
         MAX
         GETHRESH 1 1.E20
       H2 DETONATION   -- FORCE SECOND BRANCH

   1  24                              $ CONTAINMENT NOT FAILED BY H2 BURN
       2
    NO FAILUR
       1   1
         BURNPRES
         MAX
         GETHRESH 1 1.E20
       NO FAILURE CASE  -- FORCE SECOND BRANCH
    OTHERWISE                         $ CONTAINMENT FAILED BY H2 BURN
       1     1
         BURNPRES
         FUN-F_MODE
         EQUAL 0
       GET MODE OF CONTAINMENT FAILURE

$
$
$ CET EVENT 3  INJECT & SPRAY FAILURE DUE TO CNTMT FAILURE BEFORE RPV FAILURE ***
$
$                                                                     INJ_FAL1
$
$
26   CNTMT FAILURE BEFORE RPV FAILURE IMPACT ON ECCS INJECTION AND SPRAY PIPING
$                                                                    - PIPE_FAIL
   2  'NO_FAILUR'  'FAILUR'
   2   1           2
  2
```

```
                                        $ ANCHORAGE CNTMT FAILURE MODE, AND
                                        $ EXCLUDE THE SEQs WHERE CNTMT IS FAILED
                                        $ AT CORE DAMAGE FOR SBO & LOOP & OTHERS
                                        $ SINCE THE PDS ET -Li INCLUDES THIS
   3  25            2           3
      1      *    ( 1     +     4 )
    ANCHORAGE      INTACT   CRIT_ATWS
       .1           .9
    OTHERWISE                            $ - NOT ANCHORAGE CNTMT FAILURE MODE
       1.          0.

27   CNTMT FAILURE BEFORE RPV FAILURE IMPACT ON ECCS INJECTION AND SPRAY MOTORS
$                                                                    - MTR-FAIL
   2  'NO_FAILUR'  'FAILUR'
   2    1            2
  3
                                        $ CNTMT FAILURE, AND NO PIPING FAILURE
                                        $ EXCLUDE THE SEQs WHERE CNTMT IS FAILED
                                        $ AT CD SINCE THE Li PDS FT INCLUDES THIS
                                        $ FOR LOOP & SBO AND OTHERS.
   4  24           26            2          3
      1      *     1      *    ( 1     +    4 )
     FAILURE   NO_FAILUR      INTACT     CRIT_ATWS
       .5           .5

   2   9           5                    $ CONTAINMENT VENT (OR NOT ISOLATED)
       1     +     2
     VENT       NOT_ISOL
       .95          .05
    OTHERWISE                           $ NO CF, NO VENTING AND NO LOSS ISOLATION
       1.          0.

28   CNTMT FAIL BEFORE RPV FAILURE STEAM & RADIATION RELEASE IMPACT ON FIREWATER
$                                                                     - STM/RAD
   2  'NO_FAILUR'  'FAILUR'
   2    1            2
  3
                                        $ CNTMT FAILURE & NO PIPING FAILURE
                                        $ EXCLUDE THE SEQs WHERE CNTMT IS FAILED
                                        $ AT CD SINCE THE Li PDS FT INCLUDES THIS
                                        $ FOR LOOP & SBO AND OTHERS.
   4  24           26            2          3
      1      *     1      *    ( 1     +    4 )
     FAILURE    NO_FAILUR      INTACT   CRIT_ATWS
       .5           .5

   2   9           5                    $ CONTAINMENT VENTED (OR NOT ISOLATED)
       1     +     2
     VENTED     NOT_ISOL
       .9           .1
    OTHERWISE                           $ NO CF, NO VENTING AND NO LOSS ISOLATION
       1.          0.

29   INJECT & SPRAY FAILURE DUE TO CONTAINMENT FAILURE BEFORE RPV FAILURE
```

```
$                                                                    - INJ/SP_FL
    2  'NO_FAILUR'   'INJ&SPY_F'
    2    1             2
  3
    1  26                                  $ PIPING FAILED
        2
      FAILUR
        0.            1.

    2  27            28                    $ ECCS INJECTION MOTORS FAILURE,
        2      *      2                    $ ALTERNATE FIREWATER INJECT FAILURE
      FAILUR        FAILUR
        0.            1.
      OTHERWISE                            $ ALL INJECTION NOT FAILED
        1.            0.

$
$
$ CET EVENT 4  DRYWELL FAILURE AT/NEAR TIME RPV FAILURE ************** EARLY_DW
$
$
30   ALPHA MODE STEAM EXPLOSION DRYWELL AND CONTAINMENT FAILURE          - ALPHA
    2  'ALPHA'     'NO_ALPHA'
    2    1             2
  2
    1  13                                  $ REACTOR VESSEL DEPRESSURIZED
        1
       LOW_PRES
         .01           .99
      OTHERWISE                            $ REA TOR VESSEL NOT DEPRESSURIZED
         .001          .999

31   MODE OF IN-VESSEL STEAM EXPLOSION BOTTOM HEAD FAILURE          - INV_EXPLN
    4  'ALPHA'     'NO_FAIL'     'LARGE_VF'    'SMALL_VF'
    2    1             2             3              4
  3
    1  30                                  $ ALPHA MODE FALURE H  OCCURED
        1
       ALPHA
        1.           0.            0.             0.

    1  13                                  $ REACTOR VESSEL DEPRESSURIZED
        1
       LOW_PRES
$       0.            .398          .344           .258     $ 4551 GRAND GULF #'S
        0.            .94           .034           .026
      OTHERWISE                            $ NO ALPHA FAILURE & RPV NOT DEPRESSURIZD
$       0.            .93           .04            .03      $ 4551 GRAND GULF #'S
        0.            .993          .004           .003

32   RPV FAILURE MODE AND SIZE OF RPV FAILURE                          - AREA_FAIL
    4  'ALPHA'     'NO_FAIL'     'LARGE_VF'    'SMALL_VF'
    2    1             2             3              4
  5
```

```
    1  30                                   $ ALPhA MODE FALURE HAS OCCURED
        1
       ALPHA
        1.          0.          0.          0.

    1  31                                   $ STEAM EXPLOSION CAUSES LARGE FAILURE
        3
       LARGE_VF
        0.          0.          1.          0.

    1  31                                   $ STEAM EXPLOSION CAUSES SMALL FAILURE
        4
       SMALL_VF
        0.          0.          0.          1.

    2  15          29                       $ CORE DEBRIS COOLED IN VESSEL, &
        1     *     1                       $ IN-VESSEL INJECTION NOT FAILED;
       COOL_INV   NO_FAILUR                 $ THEREFORE NO RPV FAILURE
        0.          1.          0.          0.
      OTHERWISE                             $ CORE DEBRIS CAUSES LOWER RPV FAILURE
        0.          .0          .1          .9

33   WATER IN PEDESTAL AT RPV FAILURE                                  - PED_WATER
    4  'FLD+INJ'  'RPV+INJ'   'FLD'    'RPV_WTR'
    2   1           2           3           4
  4
                                            $ LATE INJECTION/CAVITY WATER SUPPLY,
                                            $ INJECTION NOT FAILED BY EARLY CF,
                                            $ LARGE CONTAINMENT H2 BURN OR SBO,
                                            $ LATE LOSS OF NORM INJECTION CAUSES
                                            $ WATER OVERFLOW INTO DW
    5  12          29          22          3          6
       /2     *     1     *  (  2     +   ( 1     *     3 ) )
     not NO_INJECT  NO_FAILUR    LG_BURN      SBO        HPCS
        1.          0.        0.          0.

                                            $ LATE INJECT/CAVITY WATER SUPPLY FAILURE
                                            $ AND LARGE CONTAINMENT H2 BURN, OR SBO
                                            $ LOSS OF NORM INJECTION CAUSES
                                            $ WATER OVERFLOW INTO DW
    4  29          22          3           6
        2     *     2     +   ( 1     *     3 ) )
      INJ&SPR_F    LG_BURN     SBO        HPCS
        0.          0.        1.          0.

                                            $ LATE INJECTION/CAVITY WATER SUPPLY,
                                            $ AND INJECTION NOT FAILED BY EARLY CF
                                            $ - AND NO WATER OVERFLOW INTO DW
    2  12          29
       /2     *     1
      not NO_INJECT  NO_FAILUR
        0.          1.        0.          0.
     OTHERWISE                              $ RESIDUAL RV WATER ONLY
        0.          0.        0.          1.
```

```
34   PEDESTAL FAILURE DUE TO OVERPRESSURE AT RPV FAILURE                 - PED_OP
   2  'PED_FAIL'   'NO'
   2    1          2
 7
   1  32                                    $ CORE DEBRIS COOLED IN-VESSEL
       2                                    $ THEREFORE NO RPV FAILURE OR PEDESTAL
     NO_FAIL                                $ FAILURE
       0.            1.

   3  13            33              33      $ RPV NOT DEPRESSURIZED AT VESSEL FAILURE
       2      *    ( 1      +        3)     $ AND WATER IN PEDESTAL FROM FLOODING
     HI_PRES        FLD+INJ         FLD
       1.            0.

                                            $ RPV NOT DEPRESSURIZED AT VESSEL FAILURE
                                            $ NO WATER IN PEDESTAL FROM FLOODING
                                            $ AND LARGE RPV FAILURE
   4  13            33              33             32
       2      *    ( 2      +        4 )     *      3
     HI_PRES        RPV+INJ    RV_WTR              LG_VF
       1.            0.

                                            $ RPV NOT DEPRESSURIZED AT VESSEL FAILURE
                                            $ NO WATER IN PEDESTAL FROM FLOODING
                                            $ AND SMALL RPV FAILURE
   4  13            33              33             32
       2      *    ( 2      +        4 )     *      4
     HI_PRES        RPV+INJ   RPV_WTR              SM_VF
       0.            1.

                                            $ RPV DEPRESSURIZED AT VESSEL FAILURE
                                            $ WATER IN PEDESTAL FROM FLOODING
                                            $ AND LARGE RPV FAILURE
   4  13            33              33             32
       1      *    ( 1      +        3 )     *      3
     LO_PRES        FLD+INJ         FLD            LG_VF
       1.            0.

   1  13                                    $ RPV DEPRESSURIZED AT VESSEL FAILURE
       1                                    $ - ADDRESSES ALL OTHER SEQUENCES
     LO_PRES
       0.            1.
     OTHERWISE                              $ SHOULD NEVER GO THIS PATHWAY
       0.            1.

35   PEDESTAL CAVITY STEAM EXPLOSTION                                    - STM_EXP
   2 'STM_EXP'     'NO_EXP'
   2    1            2
 4
   1  32                                    $ DEBRIS COOLED IN-VESSEL
       2                                    $ THEREFORE NO STM EXPLOSION
     NO_FAIL
       0.            1.
```

```
    2  33             33                  $ NO WATER IN CAVITY PRIOR TO RPV FAILURE
        2      +       4                  $ THEREFORE NO STM EXPLOSION
     RPV+INJ       RPV_WTR
        0.            1.

    3  31             31            31    $ IN-VESSEL STEAM EXPLOSION FAILED VESSEL
        1      +       3      +      4    $ - THEN ASSUME LARGE EX-VESSEL STEAM
      ALPHA         LARGE_VF  SMALL_VF    $    EXPLOSION CANNOT OCCUR
        0.            1.
    OTHERWISE                             $ WATER IN CAVITY
        .86            .14

36   PEDESTAL FAILURE DUE TO STEAM EXPLOSION                               - PED_EXP
    2 'PED_FAIL'     'NO'
    2    1            2
  3
    1  31                                 $ IN-VESSEL STEAM EXPLOSION CAUSED
        3                                 $ LARGE BOTTOM HEAD FAILURE
      LARGE_VF
$          .5            .5               $ GRAND GULF 4551 #'S
          .05            .95

    1  35                                 $ NO EX-VESSEL STEAM EXPLOSION
        2
      NO_EXP
        0.            1.
    OTHERWISE                             $ EX-VESSEL STEAM EXPLOSION OCCURRED
$          .5            .5
          .05            .95

37   DRYWELL FAILURE DUE TO PEDESTAL FAILURE                                - DW_PED
    2 'DW_FAIL'      'NO'
    2    1            2
  2
    2  34             36                  $ PEDESTAL FAILURE HAS RESULTED FROM
        1      +       1                  $ OTHER THAN AN ALPHA MODE FAILURE
      PED_FAIL      PED_FAIL
        .175          .825
     OTHERWISE                            $ PEDESTAL FAILURE HAS NOT OCCURRED
        0.            1.

38   DRYWELL OVERPRESSURE FAILURE AT RPV FAILURE                            - DW_OP
    2 'DW_FAIL'      'NO'
    2    1            2
  3
    1  32                                 $ DEBRIS COOLED IN-VESSEL
        2                                 $ THEREFORE NO DW FAILURE
      NO_FAIL
        0.            1.

    2  13             32                  $ HIGH PRESSURE SEQUENCE,
        2      *       3                  $ LARGE RPV FAILURE SIZE
      HI_PRES       LG_VF
```

```
      .01          .99
    OTHERWISE
      0.          1.

39   DRYWELL FAILS AT/NEAR TIME OF RPV FAILURE                    - EARLY_DW
    2  'DW_FAIL'    'NO'
    2               2
  3
    1  32                          $ DEBRIS COOLED IN-VESSEL
      2                            $ THEREFORE NO VESSEL FAILURE
     NO_FAIL
      0.          1.

    3  30          37         38   $ ALPHA FAILURE, PEDESTAL FAILURE OR
      1     +      1      +     1  $ OVERPRESSURE HAS FAILED THE DRYWELL
      ALPHA      DW_FAIL      DW_FAIL
      1.          0.
     OTHERWISE                     $ NO DRYWELL FAILURE
      0.          1.

$
$
$ CET EVENT 5  MODE OF CONTAINMENT FAILURE AT/NEAR RPV FAILURE ********** CF_VB
$
$
40   CONTAINMENT STEAM CONCENTRATION AT/NEAR RPV FAILURE               - ST_VB
    6  '0-15%'  '15-25%'  '25-35%'  '35-45%'  '45-55%'  '> 55%'
    2    1         2         3         4         5         6
  6
    1  18                          $ DESIGN SPRAY COOLING OPERATION
      2
      SPRAY
      1.        0.        0.        0.        0.        0.

    2   3          6               $ SBO EVENT,
      1     *      1               $ INJECTION FAILURE TIME: NO INJECTION
      SBO       NO_INJECT
      1.        0.        0.        0.        0.        0.

    2   3          6               $ SBO EVENT,
      1     *      2               $ INJECTION FAILURE TIME: RCIC
      SBO         RCIC
      0.        .44       .56       0.        0.        0.

    2   3          6               $ SBO EVENT,
      1     *      3               $ INJECTION FAILURE TIME: HPCS
      SBO         HPCS
      0.        0.        0.        0.        0.        1.

    1   2                          $ CON  INMENT FAILED AT CORE DAMAGE
      2
      FAILED
      0.        0.        0.        0.        0.        1.
     OTHERWISE                     $ OTHER NON-SBO TYPE EVENTS
```

```
        1.        0.        0.        0.        0.        0.

41   FRACTION ZIRCONIUM INVENTORY REACTED AT/NEAR RPV FAILURE            - H2_VB
    3  '33%'     '22%'     '11%'
    4    1         2         3
  4
    2   3            6                  $ SBO EVENT
        1     *      1                  $ INJECTION FAILURE TIME: NO INJECTION
      SBO          NO_INJECT
        0.         0.         1.

    2   3            6                  $ SBO EVENT
        1     *      2                  $ INJECTION FAILURE TIME: RCIC
      SBO           RCIC
        0.          .31        .69

    2   3            6                  $ SBO EVENT
        1     *      3                  $ INJECTION FAILURE TIME: HPCS
      SBO           HPCS
        0.          .21        .79
     OTHERWISE                          $ BOUND OTHERS WITH DISTRIBUTION
        0.          .31        .69      $ FOR SBO WITH RCIC INJECTION FAILURE

42   HYDROGEN IGNITION SOURCES AVAILABLE AT/BEFORE RPV FAILURE         - IG_SOURC
    2  'NO_IG_SRC' 'IGNIT_SRC'
    2   1            2
  4
    1  16                               $ H2 IGNITORS AVAILABLE EARLY
        2
      HIS_ON
        0.           1.

                                        $ HIS NOT AVAILABLE EARLY DUE TO HI
                                        $ ( OPERATOR FAILS TO INITIATE HIS )
                                        $ NO LOSS OF AC POWER, AND RECOVERY
    3  16            3           3      $ OF HIS BEFORE CNTMT [H2] EXCEEDS
        1     *     /1     *    /5      $ H2 DEFLAGRATION OVERPRESSURE LIMIT
      HIS_OFF  not SBO  not SBO_&_LOOP
        .1           .9

    2  21            22                 $ H2 BURN BEFORE RPV FAILURE
        2     +      2
      SMALL_BRN   LG_BURN
        0.           1.
     OTHERWISE                          $ NO CONTINUOUS IGNITION SOURCE AVAILABLE
        1.           0.

43   HIGH PRESSURE MELT EJECTION AT RPV FAILURE
    2  'HPME'     'NO_HPME'
    2    1           2
  3
    1  32                               $ NO VESSEL FAILURE
        2
     NO_FAIL
```

```
      0.              1.

   1  13                              $ LOW RPV PRESSURE AT RPV FAILURE
       1
     LOW_PRES
      0.              1.
    OTHERWISE                         $ RPV FAILURE AT HIGH PRESSURE
       .8              .2

44   LARGE H2 BURN IGNITED AT/NEAR RPV FAILURE                         - LG_BRN
   2  'NO_BRN_IG'  'LG_BRN_IG'
   4   1              2
 48
   1  42                              $ CONTINUOUS IGNITION SOURCE AVAILABLE
       2
    IGNIT_SRC
      1.              0.
    2
      2      0.       0.
      4      0.       0.

   1  40                              $ IF > 55% STEAM THEN INERT
       6
    > [illegible]
      [illegible]     0.
    2
      2      0.       0.
      4      0.       0.

                                      $ 0-15% STEAM AND
                                      $ 33% ZIRC OXIDATION AT RPV FAIL
                                      $ AND DW FAILED OR AC POWER RECOVERED
                                      $ [H2] = 21.7%
   4  40             41           39            7
       1      *       1      *   (1      +      1)
     0-15%           33%          DW_FAIL  PRIOR_RV
      0.              1.
    2
      2      0.      147.
      4      0.       13.

   3  40             41           43  $ 0-15% STEAM AND
       1      *       1      *     1  $ 33% ZIRC OXIDATION AT RPV FAIL
     0-15%           33%         HPME $ AND HPME
       .37             .63            $ [H2] = 21.7%
    2
      2      0.      151.
      4      0.       17.

   2  40             41               $ 0-15% STEAM AND
       1      *       1               $ 33% ZIRC OXIDATION AT RPV FAIL
     0-15%           33%              $ NO DW FAILURE, NO AC POWER, NO HPME
       .51             .49            $ [H2] = 21.7%
    2
```

```
     2       0.      147.
     4       0.       13.

                                          $ 15-25% STEAM AND
                                          $ 33% ZIRC OXIDATION AT RPV FAIL
                                          $ AND DW FAILED OR AC POWER RECOVERED
                                          $ [H2] = 18.8%
 4  40             41             39            7
     2       *      1       *    ( 1      +      1 )
   15-25%          33%           DW_FAIL  PRIOR RV
    0.              1.
  2
     2       0.      131.
     4       0.       20.

 3  40             41             43  $ 15-25% STEAM AND
     2       *      1       *      1  $ 33% ZIRC OXIDATION AT RPV FAIL
   15-25%          33%           HPME $ AND HPME
     .37            .63               $ [H2] = 18.8%
  2
     2       0.      135.
     4       0.       24.

 2  40             41                 $ 15-25% STEAM AND
     2       *      1                 $ 33% ZIRC OXIDATION AT RPV FAIL
   15-25%          33%                $ NO DW FAILURE, NO AC POWER, NO HPME
     .51            .49               $ [H2] = 18.8%
  2
     2       0.      131.
     4       0.       20.

                                      $ 25-35% STEAM AND
                                      $ 33% ZIRC OXIDATION AT RPV FAIL
                                      $ AND DW FAILED OR AC POWER RECOVERED
                                      $ [H2] = 16.4%
 4  40             41             39            7
     3       *      1       *    ( 1      +      1 )
   25-35%          33%            DW_FAIL PRIOR_RV
    0.              1.
  2
     2       0.      120.
     4       0.       26.

 3  40             41             43  $ 25-35% STEAM AND
     3       *      1       *      1  $ 33% ZIRC OXIDATION AT RPV FAIL
   25-35%          33%           HPME $ AND HPME
     .37            .63               $ [H2] = 16.4%
  2
     2       0.      124.
     4       0.       30.

 2  40              .                 $ 25-35% STEAM AND
     3       *      1                 $ 33% ZIRC OXIDATION AT RPV FAIL
   25-35%          33%                $ NO DW FAILURE, NO AC POWER, NO HPME
```

```
      .51             .49                $ [H2] = 16.4%
  2
     2       0.      120.
     4       0.       26.

                                         $ 35-45% STEAM AND
                                         $ 33% ZIRC OXIDATION AT RPV FAIL
                                         $ AND DW FAILED OR AC POWER RECOVERED
                                         $ [H2] = 14.1%
  4  40              41             39             7
     4       *        1       *    ( 1       +     1 )
   35-45%            33%            DW_FAIL  PRIOR_RV
     0.               1.
  2
     2       0.      113.
     4       0.       35.

  3  40              41             43   $ 35-45% STEAM AND
     4       *        1       *      1   $ 33% ZIRC OXIDATION AT RPV FAIL
   35-45%            33%           HPME  $ AND HPME
      .44             .56                $ [H2] = 14.1%
  2
     2       0.      117.
     4       0.       39.

  2  40              41                  $ 35-45% STEAM AND
     4       *        1                  $ 33% ZIRC OXIDATION AT RPV FAIL
   35-45%            33%                 $ NO DW FAILURE, NO AC POWER, NO HPME
      .62             .38                $ [H2] = 14.1%
  2
     2       0.      113.
     4       0.       35.

                                         $ 45-55% STEAM AND
                                         $ 33% ZIRC OXIDATION AT RPV FAIL
                                         $ AND DW FAILED OR AC POWER RECOVERED
                                         $ [H2] = 11.7%
  4  40              41             39             7
     5       *        1       *    ( 1       +     1 )
   45-55%            33%            DW_FAIL  PRIOR_RV
     0.               1.
  2
     2       0.      109.
     4       0.       46.

  3  40              41             43   $ 45-55% STEAM AND
     5       *        1       *      1   $ 33% ZIRC OXIDATION AT RPV FAIL
   45-55%            33%           HPME  $ AND HPME
      .57             .43                $ [H2] = 11.7%
  2
     2       0.      113.
     4       0.       50.

  2  40              41                  $ 45-55% STEAM AND
```

```
    5       *      1                $ 33% ZIRC OXIDATION AT RPV FAIL
  45-55%          33%               $ NO DW FAILURE, NO AC POWER, NO HPME
     .72            .28             $ [H2] = 11.7%
 2
    2       0.     109.
    4       0.      46.

                                    $ 0-15% STEAM AND
                                    $ 22% ZIRC OXIDATION AT RPV FAIL
                                    $ AND DW FAILED OR AC POWER RECOVERED
                                    $ [H2] = 15.7%
4  40             41            39           7
    1       *      2      *    ( 1      +     1 )
  0-15%           22%          DW_FAIL  PRIOR_RV
    0.             1.
 2
    2       0.     104.
    4       0.      11.

3  40             41            43  $ 0-15% STEAM AND
    1       *      2      *      1  $ 22% ZIRC OXIDATION AT RPV FAIL
  0-15%           22%         HPME  $ AND HPME
     .44            .56             $ [H2] = 15.7%
 2
    2       0.     108.
    4       0.      15.

2  40             41                $ 0-15% STEAM AND
    1       *      2                $ 22% ZIRC OXIDATION AT RPV FAIL
  0-15%           22%               $ NO DW FAILURE, NO AC POWER, NO HPME $
     .62            .38             $ [H2] = 15.7%
 2
    2       0.     104.
    4       0.      11.

                                    $ 15-25% STEAM AND
                                    $ 22% ZIRC OXIDATION AT RPV FAIL
                                    $ AND DW FAILED OR AC POWER RECOVERED
                                    $ [H2] = 13.6%
4  40             41            39           7
    2       *      2      *    ( 1      +     1 )
  15-25%          22%          DW_FAIL  PRIOR_RV
    0.             1.
 2
    2       0.      94.3
    4       0.      17.

3  40             41            43  $ 15-25% STEAM AND
    2       *      2      *      1  $ 22% ZIRC OXIDATION AT RPV FAIL
  15-25%          22%         HPME  $ AND HPME
     .44            .56             $ [H2] = 13.6%
 2
    2       0.      98.3
    4       0.      21.
```

```
2  40            41                 $ 15-25% STEAM AND
    2      *      2                 $ 22% ZIRC OXIDATION AT RPV FAIL
  15-25%         22%                $ NO DW FAILURE, NO AC POWER, NO HPME $
      .62           .38             $ [H2] = 13.6%
 2
     2      0.      94.3
     4      0.      17.

                                    $ 25 35% STEAM AND
                                    $ 22% ZIRC OXIDATION AT RPV FAIL
                                    $ AND DW FAILED OR AC POWER RECOVERED
                                    $ [H2] = 11.9%
4  40            41          39            7
    3      *      2      *   ( 1      +      1 )
  25-35%         22%        DW_FAIL  PRICR_RV
    0.            1.
 2
     2      0.      83.3
     4      0.      23.

3  40            41          43  $ 25-35% STEAM AND
    3      *      2      *    1  $ 22% ZIRC OXIDATION AT RPV FAIL
  25-35%         22%        HPME $ AND HPME
      .57           .43          $ [H2] = 11.9%
 2
     2      0.      87.3
     4      0.      27.

2  40            41                 $ 25-35% STEAM AND
    3      *      2                 $ 22% ZIRC OXIDATION AT RPV FAIL
  25-35%         22%                $ NO DW FAILURE, NO AC POWER, NO HPME
      .72           .28             $ [H2] = 11.9%
 2
     2      0.      83.3
     4      0.      23.

                                    $ 35-45% STEAM AND
                                    $ 22% ZIRC OXIDATION AT RPV FAIL
                                    $ AND DW FAILED OR AC POWER RECOVERED
                                    $ [H2] = 10.2%
4  40            41          39            7
    4      *             *   ( 1      +      1 )
  35-45%                    DW_FAIL  PRIOR_RV
    0.            1.
 2
     2      0.      80.8
     4      0.      31.

3  40            41          43  $ 35-45% STEAM AND
    4      *      2      *    1  $ 22% ZIRC OXIDATION AT RPV FAIL
  35-45%         22%        HPME $ AND HPME
      .57           .43          $ [H2] = 10.2%
 2
```

```
     2       0.       84.8
     4       0.       35.

 2  40           41                 $ 35-45% STEAM AND
     4       *    2                 $ 22% ZIRC OXIDATION AT RPV FAIL
   35-45%        22%                $ NO DW FAILURE, NO AC POWER, NO HPME
      .72            .28            $ [H2] = 10.2%
  2
     2       0.       80.8
     4       0.       31.

                                    $ 45-55% STEAM AND
                                    $ 22% ZIRC OXIDATION AT RPV FAIL
                                    $ AND DW FAILED OR AC POWER RECOVERED
                                    $ [H2] =  8.5%
 4  40           41             39          7
     5       *    2      *    ( 1     +     1 )
   45-55%        22%           DW_FAIL PRIOR_RV
     0.              1.
  2
     2       0.       80.8
     4       0.       41.

 3  40           41             43  $ 45-55% STEAM AND
     5       *    2      *       1  $ 22% ZIRC OXIDATION AT RPV FAIL
   45-55%        22%           HPME $ AND HPME
     .57         .43                $ [H2] =  8.5%
  2
     2       0.       84.8
     4       0.       45.

 2  40           41                 $ 45-55% STEAM AND
     5       *    2                 $ 22% ZIRC OXIDATION AT RPV FAIL
   45-55%        22%                $ NO DW FAILURE, NO AC POWER, NO HPME
      .72            .28            $ [H2] =  8.5%
  2
     2       0.       80.8
     4       0.       41.

                                    $ 0-15% STEAM AND
                                    $ 11% ZIRC OXIDATION AT RPV FAIL
                                    $ AND DW FAILED OR AC POWER RECOVERED
                                    $ [H2] =  8.6%
 4  40           41             39          7
     1       *    3      *    ( 1     +     1 )
   0-1.5%        11%           DW_FAIL  PRIOR_RV
     0.              1.
  2
     2       0.       38.2
     4       0.        8.4

 3  40           41             43  $ 0-15% STEAM AND
     1       *    3      *       1  $ 11% ZIRC OXIDATION AT RPV FAIL
   0-15%         11%           HPME $ AND HPME
```

```
      .57           .43              $ [H2] =  8.6%
 2
      2      0.     42.2
      4      0.     12.4

2  40              41                $ 0-15% STEAM AND
    1      *        3                $ 11% ZIRC OXIDATION AT RPV FAIL
  0-15%            11%               $ NO DW FAILURE, NO AC POWER, NO HPME
      .72           .28              $ [H2] =  8.6%
 2
      2      0.     38.2
      4      0.      8.4

                                     $ 15-25% STEAM AND
                                     $ 11% ZIRC OXIDATION AT RPV FAIL
                                     $ AND DW FAILED OR AC POWER RECOVERED
                                     $ [H2] =  7.4%
4  40              41            39             7
    2      *        3      *    ( 1      +      1 )
  15-25%           11%          DW_FAIL  PRIOR_RV
    0.              1.
 2
      2      0.     36.2
      4      0.     14.

3  40              41            43  $ 15-25% STEAM AND
    2      *        3      *      1  $ 11% ZIRC OXIDATION AT RPV FAIL
  15-25%           11%          HPME $ AND HPME
      .71           .29              $ [H2] =  7.4%
 2
      2      0.     40.2
      4      0.     18.

2  40              41                $ 15-25% STEAM AND
    2      *        3                $ 11% ZIRC OXIDATION AT RPV FAIL
  15-25%           11%               $ NO DW FAILURE, NO AC POWER, NO HPME
      .79           .21                  $ [H2] =  7.4%
 2
      2      0.     36.2
      4      0.     14.

                                     $ 25-35% STEAM AND
                                     $ 11% ZIRC OXIDATION AT RPV FAIL
                                     $ AND DW FAILED OR AC POWER RECOVERED
                                     $ [H2] =  6.5%
4  40              41            39             7
    3      *        3      *    ( 1      +      1 )
  25-35%           11%          DW_FAIL  PRIOR_RV
    0.              1.
 2
      2      0.     37.3
      4      0.     19.

3  40              41            43  $ 25-35% STEAM AND
```

```
   3        *       3        *       1  $ 11% ZIRC OXIDATION AT RPV FAIL
 25-35%            11%            HPME  $ AND HPME
     .71               .29              $ [H2] =  6.5%
 2
     2       0.        41.3
     4       0.        23.

 2  40                41                $ 25-35% STEAM AND
     3        *        3                $ 11% ZIRC OXIDATION AT RPV FAIL
  25-35%              11%               $ NO DW FAILURE, NO AC POWER, NO HPME
      .79                .21            $ [H2] =  6.5%
 2
     2       0.        37.3
     4       0.        19.

                                        $ 35-45% STEAM AND
                                        $ 11% ZIRC OXIDATION AT RPV FAIL
                                        $ AND DW FAILED OR AC POWER RECOVERED
                                        $ [H2] =  5.6%
 4  40                41              39              7
     4        *        3        *     ( 1      +       1 )
  35-45%              11%             DW_FAIL  PRIOR_RV
     0.                1.
 2
     2       0.        40.8
     4       0.        27.

 3  40                41              43  $ 35-45% STEAM AND
     4        *        3        *      1  $ 11% ZIRC OXIDATION AT RPV FAIL
  35-45%              11%            HPME $ AND HPME
      .71                .29              $ [H2] =  5.6%
 2
     2       0.        44.8
     4       0.        31.

 2  40                41                  $ 35-45% STEAM AND
     4        *        3                  $ 11% ZIRC OXIDATION AT RPV FAIL
  35-45%              11%                 $ NO DW FAILURE, NO AC POWER, NO HPME
      .79                .21              $ [H2] =  5.6%
 2
     2       0.        40.8
     4       0.        27.

                                          $ 45-55% STEAM AND
                                          $ 11% ZIRC OXIDATION AT RPV FAIL
                                          $ AND DW FAILED OR AC POWER RECOVERED
                                          $ [H2] =  4.6%
 4  40                41              39              7
     5        *        3        *     ( 1      +       1 )
  45-55%              11%             DW_FAIL  PRIOR_RV
     0.                1.
 2
     2       0.        46.1
     4       0.        36.
```

```
 3  40           41             43  $ 45-55% STEAM AND
    5     *       3      *       1  $ 11% ZIRC OXIDATION AT RPV FAIL
  45-55%         11%          HPME  $ AND HPME
     .71            .29             $ [H2] =  4.6%
  2
    2      0.      50.1
    4      0.      40.

 2  40           41                 $ 45-55% STEAM AND
    5     *       3                 $ 11% ZIRC OXIDATION AT RPV FAIL
  45-55%         11%                $ NO DW FAILURE, NO AC POWER, NO HPME
     .79            .21             $ [H2] =  4.6%
  2
    2      0.      46.1
    4      0.      36.
  OTHERWISE                         $ SHOULD NEVER GO THIS PATH
    1.             0.
  2
    2      0.      0.
    4      0.      0.

45   CONTAINMENT FAILURE DUE TO H2 DETONATION AT/NEAR RPV FAILURE       - H2_DET
  2    'DET_CF'       'NO'
  2      1             2
 10
  1  44                             $ NO LARGE BURN IGNITED
     1
    NO_BRN_IG
     0.            1.

  3  40           40             40  $ IF > 35% STEAM THEN INERT TO
     4     +       5     +        6  $ DETONATIONS
   35-45%        45-55%        > 55%
     0.            1.

                                     $ > 33% ZIRC OXIDATION
                                     $ (> 20% H2 CONC) AND
                                     $ HIGH STEAM (POWER RECOV + SPRAY)
                                     $ NO EARLY INJECTION FAILURE SINCE THE
                                     $ STEAM CONC IS LOW FOR THAT SEQUENCE
  5  41           3              7           18          6
     1     *      1      *       1     *      2     *    /1
    33%          SBO          PRIOR_RV      SPRAY    not NO_INJECT
      .025          .975

  2  40           41                 $ > 33% ZIRC OXIDATION
     1     *      1                  $ (> 20% H2 CONC) AND
    0-15%        33%                 $ LOW STEAM
      .27           .73

  3  40           40             41  $ > 33% ZIRC OXIDATION
    ( 2     +      3 )    *       1  $ ( 16-20% H2 CONC) AND
    15-25%       25-35%         33%  $ LOW STEAM
```

```
     .16           .84

  1  41                              $ 11% ZIRC OXIDATION
      3                              $ (<12%  H2 CONC)
     11%
      0.            1.

                                     $ 22% ZIRC OXIDATION
                                     $ ( 12 - 16% [H2] ), AND HI STEAM
                                     $ (POWER WITH SPRAY MODE) NO EARLY INJECT
                                     $ FAILURE SINCE [ST] IS LOW
  5  41            3           7            18            6
      2     *      1     *     1     *      2      *     /1
     22%          SBO       PRIOR_RV       SPRAY    not NO_INJECT
       .022          .978

  3  40            40           41   $ 22% ZIRC OXIDATION
    ( 2     +      3 )    *      2   $ (12-16% [H2]), AND LOW STEAM
    15-25%        25-35%        22%
      0.            1.

  2  40            41                $ 22% ZIRC OXIDATION
      1     *      2                 $ (16-20% [H2]), AND LOW STEAM
     0-15%        22%
       .16           .84
    OTHERWISE                        $ SHOULD NEVER REACH THIS CASE
      0.            1.0

46   CONTAIMENT FAILURE AT/NEAR RPV FAILURE
   2   'FAILURE'   'NO_FAILUR'
   6       1           2
 3
  1  30                              $ ALPHA MODE FAILURE
      1
     ALPHA
      1    2
        BURNPRES
        MAX
        GETHRESH 1 -1.
      ALPHA                            (FORCE FIRST BRANCH)

  1  45                              $ DETONATION FAILURE
      1
     DET CF
      1    2
        BURNPRES
        MAX
        GETHRESH 1 -1.
     DETONATION     (FORCE FIRST BRANCH)
    OTHERWISE
      1    2
         BURNPRES
            FUN-F_BURN
            EQUAL  0
```

```
        CALCULATE PROB OF CONT FAILURE (ANY MODE) GIVEN BURN PRESSURE

47   MODE OF CONTAIMENT FAILURE AT/NEAR RPV FAILURE                      - CF_VB
   2  'ANCHORAGE'   'PN-D/NoCF'
   6      1             2
  3
   1  46                            $ CNTMT NOT FAILED BY ALPHA OR H2 BURN
       2
     NO_FAILUR
       1    2
         BURNPRES
         MAX
         GETHRESH 1 1.E20
       NO FAILURE CASE

   2  45         30                 $ DETONATION OR ALPHA MODE
       1    +     1                 $ FAILURE OF CONTAINMENT
     DET CF       ALPHA
       1    2
         BURNPRES
         MAX
         GETHRESH 1 1.E20
       H2 DETONATION OR ALPHA MODE FAILURE  -- FORCE SECOND BRANCH
    OTHERWISE
       1      2
         BURNPRES
         FUN-F_MODE
         EQUAL 0
       GET MODE OF CONTAINMENT FAILURE


$
$
$ CET EVENT 6  POOL BYPASS BEFORE/NEAR RPV FAILURE ******************* PB_EARLY
$
$
48   DRYWELL FAILURE DUE TO CONTAINMENT H2 BURN BEFORE/NEAR RPV FAILURE - H2BURN
   2  'DW_FAIL'    'NO_DW_FL'
   6      1             2
  3
   1  22                            $ LARGE BURN IN CNTMT BEFORE RPV FAILURE
       2
     LG_BURN
       2    1          3
         BURNPRES  CNTNT PRESSURE
         FUN-DWDELP
         EQUAL 0
       PROB OF CNTMT BURN FAILING DRYWELL

   1  44                            $ LARGE BURN IN CNTMT AT RPV FAILURE
       2
     LG_BRN_IG
       2    2          4
         BURNPRES  CONT PRESSURE
```

```
        FUN-DWDELP
        EQUAL 0
      PROB OF CNTMT BURN FAILING DRYWELL
    OTHERWISE                          $ NO LARGE BURNS IN CNTMT BEFORE/AT/
      2     2        4                 $ NEAR RPV FAILURE
        BURNPRES    CNTMT PRESSURE
        MAX
        GETHRESH  1 1.E20
      NO CNTMT H2 BURN    (FORCE SECOND BRANCH)

49   POOL BYPASS BEFORE/NEAR RPV FAILURE                             - POOL_BYP
   2  'POOL_BP'   'NO_PL_BP'
   2       1          2
  5
   1  39                               $ DRYWELL FAILURE BY PROCESSES INSIDE DW
       1
     DW_FAIL
       1.          0.

   1  48                               $ DRYWELL FAILURE BY CNTMT H2 BURN
       1
     DW_FAIL
       1.          0.

   2  25           47                  $ POOL BYPASS BY CONTAINMENT
       1     +      1                  $ ANCHORAGE FAILURE
     ANCHORAGE    ANCHORAGE
       1.          0.

                                       $ VACUUM BREAKERS FAILING OPEN FOR
                                       $ SEQUENCES WITH AC POWER (HENCE
                                       $ VB ISO VALVES ARE OPEN
                                       $ AND LARGE H2 BURN OCCURS IN CNTMT
   8  22      44        3        3        3        3        3        7
     ( 2   +   2)  *  (( 3   +   4    +   6)  +  (1    +   5)     *  1)
    LG_BURN LG_BRN   OTHER_TYPES CRIT_ATWS OTHERS    SBO LOOP_&_SBO PRIOR_RV
       .05            .95
    OTHERWISE                          $ BYPASS FOR SEQUENCES WITH VB ISOLATED
       .0001          .9999

$
$
$ CET EVENT 7  INJECT & SPRAY FAILURE DUE TO CNTMT FAIL BEFORE/NEAR RPV FAILURE
$
$                                                                  -INJ/FAL2
$
$
50   CNTMT FAILURE AT/NEAR RPV FAILURE IMPACT ON ECCS INJECTION AND SPRAY PIPING
$                                                                  - PIPE_FAL
   2  'NO_FAILUR'  'FAILUR'
   2     1            2
  2
   1  47                               $ ANCHORAGE CNTMT FAILURE
       1
```

```
      ANCHORAGE
         .1            .9
     OTHERWISE                          $ NOT ANCHORAGE
        1.          0.

51   CNTMT FAILURE AT/NEAR RPV FAILURE IMPACT ON ECCS INJECTION AND SPRAY MOTORS
$                                                                  - MTR_FAIL
    2  'NO_FAILUR'  'FAILUR'
    2      1             2
  2
    2  46           50                  $ CNTMT FAILURE AND NO PIPING FAILURE
        1      *     1
      FAILURE     NO_FAILUR
         .5          .5
      OTHERWISE                         $ NO CNTMT FAILURE
         1.          0.

52   CNTMT FAIL AT/NEAR RPV FAILURE STEAM/RADIATION RELEASE IMPACT ON FIREWATER
$                                                                   - STM/RAD
    2  'NO_FAILUR'  'FAILUR'
    2      1             2
  2
    2  46           50                  $ CNTMT FAILURE AND NO PIPING FAILURE
        1      *     1
      FAILURE     NO_FAILUR
        .5           .5
     OTHERWISE                          $ NO CNTMT FAILURE
        1.          0.

53   INJECT & SPRAY FAILURE DUE TO CNTMT FAILURE AT/NEAR RPV FAILURE
$                                                                - INJ/SP_FL
    2  'NO_FAILUR'  'INJ&SPY_F'
    2      1             2
  3
    1  50                               $ ECCS INJ & SPRAY PIPING FAILURE DUE TO
        2                               $ CNTMT FAILURE AT/NEAR RPV FAILURE
      FAILUR
        0.          1.

    2  51           52                  $ ECCS INJECTION & SPRAY MOTORS FAILURE,
        2      *     2                  $ AND FIREWATER INJECTION FAILURE
      FAILUR      FAILUR
        0            1.
     OTHERWISE                          $ ALL INJECT NOT FAILED AT/NEAR
        1.          0.                  $ RPV FAILURE

$
$
$ CET EVENT 8  PEDESTAL FAILURE DUE TO CCI **************************** PED_FAL
$
$
54   TYPE OF DEBRIS CONCRETE INTERACTIONS                               - CCI_TYPE
$
    4  'DRY-CCI'  'FAST-WET'  'SLOW-WET'   'NO-CCI'
```

```
  2    1           2           3           4
11
 1  32                                  $ NO VESSEL FAILURE
     2
   NO_FAIL
     0.          0.          0.          1.

 3  33           33          53  $ NO WATER IN CAVITY NEAR TO RPV FAILURE
     4     +     ( 2     *     2) $ DUE TO ONLY RPV WATER OR SUBSEQUENT
   RPV_WTR       RPV+INJ  INJ&SPY_F $ FAILURE OF RPV+INJ DUE TO CNTMT FAILUR
     1.          0.          0.          0.

                                 $ WATER IN CAVITY AT RPV FAILURE
                                 $ NO INJECTION TO CAVITY OR INJ FAILURE
                                 $ NEAR RPV FAILURE DUE TO CNTMT FAILURE
                                 $ HPME
 4   33          33          53              43
    ( 3    +    ( 1    *      2 ) )   *       1
     FLD         FLD+INJ  INJ&SPR_F          HPME
     0.          .175        .4875           .3375

                                 $ WATER IN CAVITY AT RPV FAILURE
                                 $ NO INJECTION TO CAVITY OR INJ FAILURE
                                 $ NEAR RPV FAILURE DUE TO CNTMT FAILURE
                                 $ NO HPME - LARGE MOLTEN MASS
 5   33          33          53              43           14
    ( 3    +    ( 1    *      2 ) )   *       2     *      1
     FLD         FLD+INJ  INJ&SPR_F        NO_HPME      LG_DEB
     0.          .28         .48             .24

                                 $ WATER IN CAVITY PRIOR RPV FAILURE
                                 $ NO INJECTION TO CAVITY OR INJ FAILS
                                 $ AT RPV FAILURE DUE TO CONT FAILURE
                                 $ NO HPME - SMALL MOLTEN MASS
 5   33          33          53              43           14
    ( 3 +       ( 1    *      2 ) )   *       2     *      2
     FLD         FLD+INJ  INJ&SPR_F        NO_HPME      SM_DEB
     0.          .28         .48             .24

 3   33          53          43  $ NO WATER IN CAVITY AT RPV FAILURE
    ( 2    *      1 )   *     1  $ CONTINUOUS INJECTION TO CAVITY
     RPV+INJ   NO_FAILUR    HPME $ AND HPME
     0.          .315       .4775            .2075

 3   33          53          43  $ NO WATER IN CAVITY AT RPV FAILURE
    ( 2    *      1 )   *     2  $ CONTINUOUS INJECTION TO CAVITY
     RPV+INJ   NO_FAILUR  NO_HPME $ AND NO HPME
     0.          .3375      .475             .1875

 3   33          53          43  $ WATER IN CAVITY AT RPV FAILURE
    ( 1    *      1 )   *     1  $ CONTINUOUS INJECTION TO CAVITY
     FLD+INJ   NO_FAILUR    HPME $ AND HPME OCCURS
     0.          .175       .4875            .3375
```

```
                                        $ WATER IN CAVITY AT RPV FAILURE
                                        $ CONTINUOUS INJECTION TO CAVITY
                                        $ NO HPME & LARGE MOLTEN MASS
   4  33         53          43          14
     ( 1    *     1 )   *     2     *     1
      FLD+INJ  NO_FAILUR    NO_HPME      LG_DEB
      0.          .28         .48         .24

                                        $ WATER IN CAVITY AT RPV FAILURE
                                        $ CONTINUOUS INJECTION TO CAVITY
                                        $ NO HPME & SMALL MOLTEN MASS
   4  33         53          43          14
     ( 1    *     1 )   *     2     *     2
      FLD+INJ  NO_FAILUR    NO_HPME      SM_DEB
      0.          .28         .48         .24
    OTHERWISE                           $ SHOULD NEVER TAKE THIS PATH
      1.        0.          0.          0.

55   PEDESTAL FAILURE DUE TO CORE DEBRIS CONCRETE INTERACTION        - PED_FAIL
   3 'AT_VB'  'AFTER_VB   'NO_FAILUR'
   2    1          2          3
 7
   2  39          48                    $ DRYWELL FAILURE BY PROCESSES INSIDE DW
      1     +      1                    $ OR BY H2 BURN IN CONTAINMENT
    DW_FAIL       DW_FAIL
      1.        0.          0.

   1  32                                $ NO VESSEL FAILURE
      2
    NO_FAIL
      0.        0.          1.

   1  54                                $ DRY CCI
      1
    DRY-CCI
      0.        .43         .57

   1  54                                $ FAST WET CCI
      2
    FAST-WET
      0.        .29         .71

   1  54                                $ SLOW WET CCI
      3
    SLOW_WET
      0.        .25         .75

   1  54                                $ NO CCI
      4
    NO_CCI
      0.        0.          1.
    OTHERWISE                           $ SHOULD NEVER GO THIS PATH
      0.        0.          1.
```

```
$
$
$ CET EVENT 9  MODE OF LATE CONTAINMENT FAILURE ********************** CF-LATE
$
$
56   MODE OF RHR SPRAY OPERATION LATE                                  - DE_INERT
   3  'CONTROLD'  'SPRAY  '  'NO_SPRAY'
   2    1          2          3
  5
   1   8                                 $ RHR SPRAYS NOT AVAILABLE
     /1
    not RHR_SPRY
      0.           0.          1.

   2   3           8                     $ IF AC POWER NEVER LOST AND
     /1      *     1                     $ RHR SPRAY AVAIABLE
    not SBO      RHR_SPRY
      .0           1.          0

   3   3           7             8       $ IF AC POWER LOST AND RECOVERED
       1     *     1      *      1       $ BEFORE RPV FAILURE AND
     SBO        PRIOR_RV     RHR_SPRY    $ RHR SPRAY AVAILBLE
      0.           1.          0.

   3   3           7             8       $ IF AC POWER LOST AND RECOVERED
       1     *     2      *      1       $ BEFORE CNTMT THRESHOLD LIMIT AND
     SBO        CNTMT_LIM    RHR_SPRY    $ RHR SPRAY AVAILABLE
      0.           1.          0.
    OTHERWISE                            $ SHOULD NEVER GO THIS WAY
      0.           0.          1.

57   HYDROGEN IGNITION SOURCES AVAILABLE LATE                          - IG_SOURC
   2  'NO_SOURCE'  'IGN_SOURC'
   2    1          2
  5
   1  16                                 $ IGNITORS AVAILABLE EARLY
       2
     HIS_ON
      0.           1.

                                         $ HIS NOT AVAILABLE EARLY DUE TO HI
                                         $ ( OPERATOR FAILS TO INITIATE HIS )
                                         $ NO LOSS OF AC POWER, AND RECOVERY
   3  16           3             3       $ OF HIS BEFORE CNTMT [H2] EXCEEDS
       1     *    /1      *     /5       $ H2 DEFLAGRATION OVERPRESSURE LIMIT
     HIS_OFF  not SBO  not SBO_&_LOOP
        .1          .9

   2   3           7                     $ LOSS OF AC POWER AND
       1     *     1                     $ POWER RECOVERY PRIOR PRIOR RPV FAILURE
     SBO        PRIOR_RV
        .5          .5
$       0.          1.                   $ SENSITIVITY FOR HIS BACKUP POWER
```

```
    2   3            7                  $ LOSS OF AC POWER AND
        1     *      2                  $ POWER RECOVERY PRIOR CONTAINMENT LIMIT
      SBO        CNTMT LIM
$       0.           1.                 $ SENSITIVITY FOR HIS BACKUP POWER
        1.           0.
    OTHERWISE                           $ LOSS OF AC POWER NO RECOVERY
        1.           0.
$       0.           1.                 $ SENSITIVITY FOR HIS BACKUP POWER

58   CONTAINMENT STEAM CONCENTRATION LATE                          - ST_CONC
    6  '0-15%'  '15-25%'  '25-35%'  '35-45%'  '45-55%'  '> 55%'
    2   1         2         3         4         5         6
$ WJC REV 21APR1992 START
  7
    1  56                               $ DESIGN SPRAY OPERATION
        2
      SPRAY
        1.        0.        0.        0.        0.        0.

    1   5                               $ BOUNDED BY SBO WITH INJECT FAILURE
        2                               $ AND NO LATE INJECTION
      NOT_ISOL
        .25       .75       .00       .00       0.        0.

    2   3            6                  $ SBO EVENT
        1     *      1                  $ INJECTION FAILURE TIME: NO INJECTION
      SBO        NO_INJECT
        .99       .01       0.        0.        0.        0.

    2   3            6                  $ SBO EVENT
        1     *      2                  $ INJECTION FAILURE TIME: RCIC
      SBO           RCIC
        0.        0.        .29       .71       0.        0.

    2   3            6                  $ SBO EVENT
        1     *      3                  $ INJECTION FAILURE TIME: HPCS
      SBO           HPCS
        0.        0.        0.        0.        0.        1.

    1   2                               $ CONTAINMENT FAILED AT CORE DAMAGE
        2
      FAILED
        0.        0.        0.        0.        0.        1.
     OTHERWISE                          $ NON-SBO TRANSIENTS WITH INTACT CNTMT
        1.        0.        0.        0.        0.        0.

59   H2 COMBUSTION BEFORE/AT RPV FAILURE                           - BURN_B4
    2  'EARLY_BRN'  'NO_ERLY_B'
    2      1            2
  2
    3  44            22             21  $ SMALL OR LARGE BURN EARLY
        2      +      2      +      2
     LG_BRN_IG     LG_BURN       SMALL_BRN
        1.            0.
```

```
   OTHERWISE
     0.          1.

60   CONTAINMENT H2 CONCENTRATION LATE                                    - H2_CONC
   6  '< 4 %'  '4-8 %'  '8-12 %'  '12-16 %'  '16-20 %'  ' > 20 %'
   2   1        2        3         4          5          6
 24
   2  59          54                  $ H2 BURN EARLY
       1     *     4                  $ NO CCI
    EARLY_BRN   NO_CCI
      .5        .5       0.         0.         0.         0.

   2  59          54                  $ H2 BURN EARLY
       1     *     3                  $ SLOW_WET CCI
    EARLY_BRN   SLOW-WET
      .1        .25      .3         .25        .1         0.

   2  59          54                  $ H2 BURN EARLY
       1     *     2                  $ FAST_WET CCI
    EARLY_BRN   FAST-WET
      0.        .1       .2         .35        .35        0.

                                      $ H2 BURN EARLY, DRY CCI, NO SPRAY
                                      $ SBO WITH NO INJECTION
   5  59          54           56            3             6
       1     *     1     *      3      *     1      *      1
    EARLY_BRN   DRY-CCI      NO_SPRAY       SBO         NO_INJECT
      0.        .35      .07        .06        .06       .46

                                      $ H2 BURN EARLY, DRY CCI, NO SPRAY
                                      $ SBO WITH RCIC FAILURE
   5  59          54           56            3             6
       1     *     1     *      3      *     1      *      2
    EARLY_BRN   DRY-CCI      NO_SPRAY       SBO         RCIC
      0.        .04      .60        .36        0.        0.

                                      $ H2 BURN EARLY, DRY CCI, NO SPRAY
                                      $ SBO WITH HPCS FAILURE
   5  59          54           56            3             6
       1     *     1     *      3      *     1      *      3
    EARLY_BRN   DRY-CCI      NO_SPRAY       SBO         HPCS
      .64       .36      0.         0.         0.        0.

                                      $ H2 BURN EARLY, DRY CCI, SPRAY
                                      $ SBO WITH NO INJECTION
   5  59          54           56            3             6
       1     *     1     *      2      *     1      *      1
    EARLY_BRN   DRY-CCI      SPRAY          SBO         NO_INJECT
      0.        .36      .07        .06        .06       .45

                                      $ H2 BURN EARLY, DRY CCI, NO SPRAY
                                      $ SBO WITH RCIC FAILURE
   5  59          54           56            3             6
       1     *     1     *      2      *     1      *      2
```

```
  EARLY_BRN    DRY-CCI         SPRAY          SBO           RCIC
   0.          .05       .43         .37          .14       .01

                                   $ H2 BURN EARLY, DRY CCI, NO SPRAY
                                   $ SBO WITH HPCS FAILURE
5  59          54             56             3              6
   1     *      1      *       2      *      1      *       3
  EARLY_BRN    DRY-CCI         SPRAY          SBO           HPCS
    .38        .51       .11        0.           0.        0.

                                   $ NO H2 BURN EARLY, NO CCI, NO SPRAY
                                   $ SBO WITH NO INJECTION
5  59          54             56             3              6
   2     *      4      *       3      *      1      *       1
  NO_ERLY_B    NO-CCI          NO_SPRAY        SBO          NO_INJECT
   0.        1.          0.         0.           0.        0.

                                   $ NO H2 BURN EARLY, NO CCI, NO SPRAY
                                   $ SBO WITH RCIC FAILURE
5  59          54             56             3              6
   2     *      4      *       3      *      1      *       2
  NO_ERLY_B    NO-CCI          NO_SPRAY       SBO           RCIC
   0.        0.          .8         .2           0.        0.

                                   $ NO H2 BURN EARLY, NO CCI, NO SPRAY
                                   $ SBO WITH HPCS FAILURE
5  59          54             56             3              6
   2     *      4      *       3      *      1      *       3
  NO_ERLY_B    NO-CCI          NO_SPRAY       SBO           HPCS
    .59        .41       0.         0.           0.        0.

                                   $ NO H2 BURN EARLY, NO CCI, SPRAY
                                   $ SBO WITH NO INJECTION
5  59          54             56             3              6
   2     *      4      *       2      *      1      *       1
  NO_ERLY_B    NO-CCI          SPRAY          SBO          NO_INJECT
   0.        1.          0.         0.           0.        0.

                                   $ NO H2 BURN EARLY, NO CCI, SPRAY
                                   $ SBO WITH RCIC FAILURE
5  59          54             56             3              6
   2     *      4      *       2      *      1      *       2
  NO_ERLY_B    NO-CCI          SPRAY          SBO           RCIC
   0.        0.          .56        .38          .06       0.

                                   $ NO H2 BURN EARLY, NO CCI, SPRAY
                                   $ SBO WITH HPCS FAILURE
5  59          54             56             3              6
   2     *      4      *       2      *      1      *       3
  NO_ERLY_B    NO-CCI          SPRAY          SBO           HPCS
    .33        .51       .16        0.           0.        0.

2  59          54                  $ NO H2 BURN EARLY
   2     *      3                  $ SLOW_WET CCI
```

```
   NO_ERLY_B   SLOW-WET
     0.         .1         .15         .25         .5       0.

2  59            54                    $ NO H2 BURN EARLY
   2      *      2                     $ FAST_WET CCI
   NO_ERLY_B   FAST-WET
     0.        0.          .1          .15         .75      0.

                                       $ NO H2 BURN EARLY, DRY CCI, NO SPRAY
                                       $ SBO WITH NO INJECTION
5  59            54               56             3              6
   2      *      1        *       3       *      1       *      1
   NO_ERLY_B    DRY-CCI         NO_SPRAY        SBO          NO_INJECT
     0.          .30       .07         .06         .06      .51

                                       $ NO H2 BURN EARLY, DRY CCI, NO SPRAY
                                       $ SBO WITH RCIC FAILURE
5  59            54               56             3              6
   2      *      1        *       3       *      1       *      2
   NO_ERLY_B    DRY-CCI         NO_SPRAY        SBO            RCIC
     0.        0.          .60         .36         .04      0.

                                       $ NO H2 BURN EARLY, DRY CCI, NO SPRAY
                                       $ SBO WITH HPCS FAILURE
5  59            54               56             3              6
   2      *      1        *       3       *      1       *      3
   NO_ERLY_B    DRY-CCI         NO_SPRAY        SBO            HPCS
     .59         .41      0.          0.          0.        0.

                                       $ NO H2 BURN EARLY, DRY CCI, SPRAY
                                       $ SBO WITH NO INJECTION
5  59            54               56             3              6
   2      *      1        *       2       *      1       *      1
   NO_ERLY_B    DRY-CCI         SPRAY           SBO          NO_INJECT
     0.          .31       .07         .06         .06      .50

                                       $ NO H2 BURN EARLY, DRY CCI, SPRAY
                                       $ SBO WITH RCIC FAILURE
5  59            54               56             3              6
   2      *      1        *       2       *      1       *      2
   NO_ERLY_B    DRY-CCI         SPRAY           SBO            RCIC
     0.        0.          .43         .42         .14      .01

                                       $ NO H2 BURN EARLY, DRY CCI, SPRAY
                                       $ SBO WITH HPCS FAILURE
5  59            54               56             3              6
   2      *      1        *       2       *      1       *      3
   NO_ERLY_B    DRY-CCI         SPRAY           SBO            HPCS
     .33         .51       .16        0.          0.        0.
   OTHERWISE                           $ SHOULD NOT TAKE THIS PATH
     0.        0.          0.         0.          1.        0.

61   AC POWER AVAILABLE LATE                                      - AC_PWR
  2  'AC_LATE'  'NO_AC_LAT'
```

```
   2      1          2
 2
   2   3             7                       $ SBO EVENT
       1     *       3                       $ NO POWER RECOVERY
      SBO          NO_RECOV
       0.            1.
     OTHERWISE                               $ AC POWER AVAILABLE LATE
       1.            0.

62   LARGE H2 BURN IGNITED LATE                                                  - LG_BRN
   2  'NO_BURN'  'LG_BURN'
   4    1            2
 54
   1  57                                     $ CONTINUOUS IGNITION SOURCE AVAILABLE
       2
     IGN_SOURC
       1.            0.
    2
       2      0.      0.
       4      0.      0.

   1  58                                     $ IF > 55% STEAM THEN INERT
       6
     > 55%
       1.            0.
    2
       5      0.      0.
       6      0.      0.

   1  60                                     $ IF < 4 % H2 THEN NO BURN
       1
     < 4 %
       1.            0.
    2
       5      0.      0.
       6      0.      0.

   3  58             60             61  $ 0-15 % STEAM AND
       1     *       2      *        2  $ 4-8 % H2 IN CNTMT   =  6% H2
     0-15%          4-8%      NO_AC_LAT $ NO AC POWER LATE
       .71            .29
    2
       5      0.      19.4
       6      0.       7.6

   3  58             60             61  $ 0-15 % STEAM AND
       1     *       3      *        2  $ 8-12 % H2 IN CNTMT  = 10% H2
     0-15%          8-12%     NO_AC_LAT $ NO AC POWER LATE
       .67            .33
    2
       5      0.      51.6
       6      0.       8.8

   3  58             60             61  $ 0-15% STEAM AND
```

```
    1       *      4       *       2  $ 12-16 % H2 IN CNTMT = 14% H2
  0-15%         12-16%   NO_AC_LAT $ NO AC POWER LATE
     .58            .42
 2
    5      0.      92.6
    6      0.      10.

3  58            60             61  $ 0-15% STEAM AND
    1       *      5       *       2  $ 16-20% H2 IN CNTMT  = 18% H2
  0-15%         16-20%   NO_AC_LAT $ NO AC POWER LATE
     .49            .51
 2
    5      0.     121.
    6      0.      11.

3  58            60             61  $ 0-15% STEAM AND
    1       *      6       *       2  $ > 20 % H2 IN CNTMT  = 24% H2
  0-15%         > 20%    NO_AC_LAT $ NO AC POWER LATE
     .49            .51
 2
    5      0.     166.
    6      0.      14.

3  58            60             61  $ 15-25% STEAM AND
    2       *      2       *       2  $ 4-8 % H2 IN CNTMT   =  6% H2
  15-25%         4-8%    NO_AC_LAT $ NO AC POWER LATE
     .71            .29
 2
    5      0.      26.7
    6      0.      13.

3  58            60             61  $ 15-25% STEAM AND
    2       *      3       *       2  $ 8-12 % H2 IN CNTMT  = 10% H2
  15-25%         8-12%   NO_AC_LAT $ NO AC POWER LATE
     .67            .33
 2
    5      0.      59.7
    6      0.      15.

3  58            60             61  $ 15-25% STEAM AND
    2       *      4       *       2  $ 12-16 % H2 IN CNTMT = 14% H2
  15-25%        12-16%   NO_AC_LAT $ NO AC POWER LATE
     .58            .42
 2
    5      0.      96.8
    6      0.      17.

3  58            60             61  $ 15-25% STEAM AND
    2       *      5       *       2  $ 16-20 % H2 IN CNTMT = 18% H2
  15-25%        16-20%   NO_AC_LAT $ NO AC POWER LATE
     .49            .51
 2
    5      0.     126.
    6      0.      19.
```

```
3  58           60           61  $ 15-25% STEAM AND
    2      *     6      *     2  $  >20 % H2 IN CNTMT  = 24% H2
  15-25%       > 20%    NO_AC_LAT $ NO AC POWER LATE
      .49           .51
 2
    5      0.     170.
    6      0.      23.

3  58           60           61  $ 25-35% STEAM AND
    3      *     2      *     2  $ 4-8 % H2 IN CNTMT   =  6% H2
  25-35%        4-8%    NO_AC_LAT $ NO AC POWER LATE
      .71           .29
 2
    5      0.      34.1
    6      0.      19.

3  58           60           61  $ 25-35% STEAM AND
    3      *     3      *     2  $ 8-12 % H2 IN CNTMT  = 10% H2
  25-35%        8-12%   NO_AC_LAT $ NO AC POWER LATE
      .67           .33
 2
    5      0.      67.9
    6      0.      21.

3  58           60           61  $ 25-35% STEAM AND
    3      *     4      *     2  $ 12-16 % H2 IN CNTMT = 14% H2
  25-35%       12-16%   NO_AC_LAT $ NO AC POWER LATE
      .58           .42
 2
    5      0.     103.
    6      0.      24.

3  58           60           61  $ 25-35% STEAM AND
    3      *     5      *     2  $ 16-20 % H2 IN CNTMT = 18% H2
  25-35%       16-20%   NO_AC_LAT $ NO AC POWER LATE
      .49           .51
 2
    5      0.     134.
    6      0.      28.

3  58           60           61  $ 25-55% STEAM AND
    3      *     6      *     2  $  >20 % H2 IN CNTMT  = 24% H2
  25-35%       > 20%    NO_AC_LAT $ NO AC POWER LATE
      .49           .51
 2
    5      0.     161.
    6      0.      34.

3  58           60           61  $ 35-45% STEAM AND
    4      *     2      *     2  $ 4-8 % H2 IN CNTMT   =  6% H2
  35-45%        4-8%    NO_AC_LAT $ NO AC POWER LATE
      .71           .29
 2
```

```
    5      0.      44.
    6      0.      27.

3  58             60              61  $ 35-45% STEAM AND
    4      *       3      *        2  $ 8-12 % H2 IN CNTMT  = 10% H2
  35-45%         8-12%    NO_AC_LAT $ NO AC POWER LATE
      .67          .33
 2
    5      0.      79.6
    6      0.      30.

3  58             60              61  $ 35-45% STEAM AND
    4      *       4      *        2  $ 12-16 % H2 IN CNTMT = 14% H2
  35-45%        12-16%    NO_AC_LAT $ NO AC POWER LATE
      .58          .42
 2
    5      0.     112.
    6      0.      35.

3  58             60              61  $ 35-45% STEAM AND
    4      *       5      *        2  $ 16-20 % H2 IN CNTMT = 18% H2
  35-45%        16-20%    NO_AC_LAT $ NO AC POWER LATE
      .49          .51
 2
    5      0.     143.
    6      0.      40.

3  58             60              61  $ 35-45% STEAM AND
    4      *       6      *        2  $  >20 % H2 IN CNTMT  = 24% H2
  35-45%         > 20%    NO_AC_LAT $ NO AC POWER LATE
      .49          .51
 2
    5      0.     157.
    6      0.      50.

3  58             60              61  $ 45-55% STEAM AND
    5      *       2      *        2  $ 4-8 % H2 IN CNTMT   =  6% H2
  45-55%         4-8%     NO_AC_LAT $ NO AC POWER LATE
      .71          .29
 2
    5      0.      58.2
    6      0.      38.

3  58             60              61  $ 45-55% STEAM AND
    5      *       3      *        2  $ 8-12 % H2 IN CNTMT  = 10% H2
  45-55%         8-12%    NO_AC_LAT $ NO AC POWER LATE
      .67          .33
 2
    5      0.      97.9
    6      0.      44.

3  58             60              61  $ 45-55% STEAM AND
    5      *       4      *        2  $ 12-16 % H2 IN CNTMT = 14% H2
  45-55%        12-16%    NO_AC_LAT $ NO AC POWER LATE
```

```
     .58          .42
 2
    5      0.      128.
    6      0.       51.

 3  58            60             61  $ 45-55% STEAM AND
    5      *       5      *       2  $ > 16 % H2 IN CNTMT  = 18% H2
  45-55%      16-20%     NO_AC_LAT $ NO AC POWER LATE
     .49          .51
 2
    5      0.      146.
    6      0.       59.

 3  58            60             61  $ 45-55% STEAM AND
    5      *       6      *       2  $  >20 % H2 IN CNTMT  = 24% H2
  45-55%       > 20%     NO_AC_LAT $ NO AC POWER LATE
     .49          .51
 2
    5      0.      168.
    6      0.       78.

$ CASES FOR AC POWER AVAILABLE

 3  58            60             61  $ 0-15 % STEAM AND
    1      *       2      *       1  $ 4-8 % H2 IN CNTMT   =  6% H2
   0-15%        4-8%       AC_LATE  $ AC POWER AVL LATE
    0.            1.
 2
    5      0.       19.4
    6      0.        7.6

 3  58            60             61  $ 0-15 % STEAM AND
    1      *       3      *       1  $ 8-12 % H2 IN CNTMT  = 10% H2
   0-15%       8-12%       AC_LATE $ AC POWER AVL LATE
    0.            1.
 2
    5      0.       51.6
    6      0.        8.8

 3  58            60             61  $ 0-15% STEAM AND
    1      *       4      *       1  $ 12-16 % H2 IN CNTMT = 14% H2
   0-15%      12-16%       AC_LATE $ AC POWER AVL LATE
    0.            1.
 2
    5      0.       92.6
    6      0.       10.

 3  58            60             61  $ 0-15% STEAM AND
    1      *       5      *       1  $ 16-20% H2 IN CNTMT  = 18% H2
   0-15%      16-20%       AC_LATE $ AC POWER AVL LATE
    0.            1.
 2
    5      0.      121.
    6      0.       11.
```

```
3  58            60            61  $ 0-15% STEAM AND
    1      *      6      *      1  $ > 20 % H2 IN CNTMT   = 24% H2
  0-15%         > 20%      AC_LATE $ NO AC POWER LATE
    0.            1.
 2
    5      0.      166.
    6      0.       14.

3  58            60            61  $ 15-25% STEAM AND
    2      *      2      *      1  $ 4-8 % H2 IN CNTMT    =  6% H2
 15-25%          4-8%      AC_LATE $ AC POWER AVL LATE
    0.            1.
 2
    5      0.       26.7
    6      0.       13.

3  58            60            61  $ 15-25% STEAM AND
    2      *      3      *      1  $ 8-12 % H2 IN CNTMT   = 10% H2
 15-25%          8-12%     AC_LATE $ AC POWER AVL LATE
    0.            1.
 2
    5      0.       59.7
    6      0.       15.

3  58            60            61  $ 15-25% STEAM AND
    2      *      4      *      1  $ 12-16 % H2 IN CNTMT = 14% H2
 15-25%         12-16%     AC_LATE $ AC POWER AVL LATE
    0.            1.
 2
    5      0.       96.8
    6      0.       17.

3  58            60            61  $ 15-25% STEAM AND
    2      *      5      *      1  $ 16-20 % H2 IN CNTMT = 18% H2
 15-25%         16-20%     AC_LATE $ AC POWER AVL LATE
    0.            1.
 2
    5      0.      126.
    6      0.       19.

3  58            60            61  $ 15-25% STEAM AND
    2      *      6      *      1  $ >20 % H2 IN CNTMT    = 24% H2
 15-25%         > 20%      AC_LATE $ NO AC POWER LATE
    0.            1.
 2
    5      0.      170.
    6      0.       23.

3  58            60            61  $ 25-35% STEAM AND
    3      *      2      *      1  $ 4-8 % H2 IN CNTMT    =  6% H2
 25-35%          4-8%      AC_LATE $ AC POWER AVL LATE
    0.            1.
 2
```

```
    5      0.      34.1
    6      0.      19.

3  58          60            61  $ 25-35% STEAM AND
    3     *     3      *      1  $ 8-12 % H2 IN CNTMT  = 10% H2
  25-35%      8-12%     AC_LATE $ AC POWER AVL LATE
  0.          1.
 2
    5      0.      67.9
    6      0.      21.

3  58          60            61  $ 25-35% STEAM AND
    3     *     4      *      1  $ 12-16 % H2 IN CNTMT = 14% H2
  25-35%     12-16%     AC_LATE $ AC POWER AVL LATE
  0.          1.
 2
    5      0.      103.
    6      0.       24.

3  58          60            61  $ 25-35% STEAM AND
    3     *     5      *      1  $ 16-20% H2 IN CNTMT  = 18% H2
  25-35%     16-20%     AC_LATE $ AC POWER AVL LATE
    0.          1.
 2
    5      0.      134.
    6      0.       28.

3  58          60            61  $ 15-25% STEAM AND
    3     *     6      *      1  $ >20 % H2 IN CNTMT  = 24% H2
  25-35%      > 20%     AC_LATE $ NO AC POWER LATE
    0.          1.
 2
    5      0.      161.
    6      0.       34.

3  58          60            61  $ 35-45% STEAM AND
    4     *     2      *      1  $ 4-8 % H2 IN CNTMT   =  6% H2
  35-45%       4-8%     AC_LATE $ AC POWER AVL LATE
    0.          1.
 2
    5      0.       44.
    6      0.       27.

3  58          60            61  $ 35-45% STEAM AND
    4     *     3      *      1  $ 8-12 % H2 IN CNTMT  = 10% H2
  35-45%      8-12%     AC_LATE $ AC POWER AVL LATE
    0.          1.
 2
    5      0.      79.6
    6      0.      30.

3  58          60            61  $ 35-45% STEAM AND
    4     *     4      *      1  $ 12-16 % H2 IN CNTMT = 14% H2
  35-45%     12-16%     AC_LATE $ AC POWER AVL LATE
```

```
      0.             1.
  2
      5       0.      112.
      6       0.       35.

3   58            60              61  $ 35-45% STEAM AND
      4       *      5       *       1  $ 16-20% H2 IN CNTMT = 18% H2
   35-45%       16-20%       AC_LATE $ AC POWER AVL LATE
      0.             1.
  2
      5       0.      143.
      6       0.       40.

3   58            60              61  $ 35-45% STEAM AND
      4       *      6       *       1  $ >20 % H2 IN CNTMT  = 24% H2
   35-45%        > 20%       AC_LATE $ NO AC POWER LATE
      0.             1.
  2
      5       0.      157.
      6       0.       50.

3   58            60              61  $ 45-55% STEAM AND
      5    *         2       *       1  $ 4-8 % H2 IN CNTMT  =  6% H2
   45-55%        4-8%        AC_LATE $ AC POWER AVL LATE
      0.             1.
  2
      5       0.       58.2
      6       0.       38.

3   58            60              61  $ 45-55% STEAM AND
      5       *      3       *       1  $ 8-12 % H2 IN CNTMT = 10% H2
   45-55%        8-12%       AC_LATE $ AC POWER AVL LATE
      0.             1.
  2
      5       0.       97.9
      6       0.       44.

3   58            60              61  $ 45-55% STEAM AND
      5       *      4       *       1  $ 12-16 % H2 IN CNTMT= 14% H2
   45-55%       12-16%      AC_LATE  $ AC POWER AVL LATE
      0.             1.
  2
      5       0.      128.
      6       0.       51.

3   58            60              61  $ 45-55% STEAM AND
      5       *      5       *       1  $ 16-20 % H2 IN CNTMT= 18% H2
   45-55%       16-20%       AC_LATE $ AC POWER AVL LATE
      0.             1.
  2
      5       0.      146.
      6       0.       59.

3   58            60              61  $ 45-55% STEAM AND
```

```
      5        *       6       *        1  $ >20 % H2 IN CNTMT  = 24% H2
   15-25%        > 20%        AC_LATE $ NO AC POWER LATE
      0.               1.
   2
      5      0.      168.
      6      0.       78.
   OTHERWISE                           $ SHOULD NEVER GO THIS PATH
      1.               0.
   2
      5      0.       0.
      6      0.       0.

63  H2 DETONATION LATE CONTAINMENT FAILURE                              - H2_DET
   2    'DET_CF'      'NO'
   2      1             2
 9
   1  62                               $ NO LARGE BURN IGNITION LATE
      1
    NO_BURN
      0.               1.

   3  58              58             58  $ IF > 35% STEAM
      4      +         5      +       6  $ THEN INERT TO DETONATIONS
    35-45%          45-55%          > 55%
      0.               1.

   3  60              60             60  $ < 12 % H2 IN CONT ATM LATE
      1      +         2      +       3
    < 4%            4-12%          8-12%
      0.               1.

                                       $ STEAM > 35% HIGH
                                       $ 12 - 16% H2 IN CNTMT ATM LATE
                                       $ POWER RECOVERED PRIOR TO CNTMT LIMIT
                                       $ RHR IS AVAILABLE IN SPRAY MODE,
   4  60               3              7            56
      4      *         1      *       2      *      2
    12-16%           SBO          CNTMT_LMT       SPRAY
       .022             .978

   1  60                               $ 12 - 16% H2 IN CONT ATM LATE
      4                                $ < 35% STEAM LOW
    12-16%
      0.               1.

                                       $ STEAM > 35% HIGH
                                       $ 16-20% OR > 20% H2 IN CNTMT ATM LATE
                                       $ POWER RECOVERED PRIOR TO CNTMT LIMIT
                                       $ RHR IS AVAILABLE IN SPRAY MODE,
   5  60              60              3             7             56
    ( 5      +         6 )    *       1      *      2      *       2
     16-20%         > 20%            SBO         CNTMT_LMT        SPRAY
       .025             .975
```

```
   1  60                                 $ 16 - 20% H2 IN CNTMT ATM LATE
      5                                  $ < 35% STEAM LOW
    16-20%
       .16           .84

   1  60                                 $ > 20% H2 IN CNTMT ATM LATE
      6                                  $ < 35% STEAM
    > 20%
       .27           .73
    OTHERWISE                            $ SHOULD NEVER GO THIS PATH
      0.            1.

64   HYDROGEN BURN LATE CONTAINMENT FAILURE                          - CF
   2   'FAILURE'   'NO_FAILUR'
   6       1           2
  2
   1  63                                 $ LATE DETONATION FAILS CONTAINMENT
      1
     DET_CF
      1    5
         BURNPRES
         MAX
         GETHRESH 1 -1
       LATE DETONATION      (FORCE FIRST BRANCH)
    OTHERWISE
      1    5
         BURNPRES
           FUN-F_BURN
           EQUAL  0
     CALCULATE PROB OF CNTMT FAILURE (ANY MODE) GIVEN BURN PRESSURE

65   CONTAINMENT STATUS AT ACCIDENT PROGRESSION COMPLETION          - CNTMT_ST
   4   'EARLY_CF'  'LATE_CF'   'VENT'   'NO_LAT_CF'
   4       1           2          3         4
 14
   1   2                                 $ CONTAINMENT FAILED AT CORE DAMAGE
       2
     FAILED
       1.          0.          0           0.
    1
    7   .15        .15         0.          0.

   1   5                                 $ CONTAINMENT VENT NOT ISOLATED FOR SBO
       2                                 $ SEQUENCES
     NOT_ISOL
       1.          0.          0.          0.
    1
    7  0.          0.          0.          0.

   1  24                                  $ CNTMT FAILURE PRIOR TO RPV FAILURE
       1
    FAILURE
       1.          0.          0           0.
    1
```

```
 7  .15          .15        0.             0.

1  46                               $ CNTMT FAILURE AT RPV FAILURE
    1
   FAILURE
    1.          0.          0.           0.
 1
 7  .15          .15        0.             0.

3   2            3            9  $ CONTAINMENT INTACT AT CD AND
    1      *     4      *     1  $ CRITICAL ATWS - THIS COMBINATION
   INTACT     CRIT_ATWS     VENT $ IMPLIES THE ALTERATE SHUTDOWN
                                 $ ATWS SEQUENCES
    0.          0.          1.         0.
1
7   .15          .15        0.             0.

                                 $ CONTAINMENT HEAT REMOVAL WITH SPRAY
                                 $ OR RHR WITH POOL AND NO POOL BYPASS
4   8            8          49            54
   (1      +    (2      *     2))    *     1
   RHR_SPRY      RHR_POOL NO_PL_BP       DRY-CCI
    0.           .1          0.           .9
 1
 7  .15          .15        0.             0.

                                 $ CONTAINMENT HEAT REMOVAL WITH SPRAY
                                 $ OR RHR WITH POOL AND NO POOL BYPASS
4   8            8          49            54
   (1      +    (2      *     2))    *    /1
   RHR_SPRY     RHR_POOL NO_PL_BP     Not DRY-CCI
    0.          0.          0.           1.
 1
 7  .15          .15        0.             0.

3   8           49            9  $ CNTMT HEAT REMOVAL WITH POOL COOLING
    2      *     1      *     2  $ AND POOL BYPASS
  RHR_POOL    POOL_BP      NO_VENT
    0.           .75        0.            .25
 1
 7  .15          .15        0.             0.

1   9                            $ CONTAINMENT VENT OPENED
    1
   VENT
    0.          0.          1.         0.
 1
 7  .15          .15        0.             0.

                                 $ NO CONTAINMENT HEAT REMOVAL
                                 $ AND SBO WITH EARLY AND INTER LOSS INJ
4   8            3            6            6
    3      *     1      *    (1      +     2)
   NO_RHR       SBO           NO_INJECT  RCIC
```

```
      0.          1.          0.          0.
  1
  7  .15        .15        0.          0.

3   8          3          6  $ NO CONTAINMENT HEAT REMOVAL
    3    *     1    *     3  $ AND SBO WITH LATE LOSS INJ
   NO_RHR       SBO        HPCS
    0.         1.         0.          0.
  1
  7  .15        .15        0.          0.

3   8          6          6  $ NO CONTAINMENT HEAT REMOVAL
    3     *    (1     +   2) $ WITH EARLY/INTER LOSS OF INJECTION
   NO_RHR         NO_INJECT  RCIC
    0.         1.         0.          0.
  1
  7  .15        .15        0.          0.

1   8                        $ NO CONTAINMENT HEAT REMOVAL
    3                        $ ALL OTHER SEQUENCES
   NO_RHR
    0.         1.         0.          0.
  1
  7  .15        .15        0.          0.
  OTHERWISE                  $ SHOULD NEVER GO THIS PATH
    0.         1.         0.          0.
  1
  7  .15        .15        0.          0.

66   MODE OF LATE HYDROGEN AND OVERPRESSURE CONTAINMENT FAILURE       - LATE_CF
  2   'ANCHORAGE'     'PN-D/NoCF'
  6      1           2
 4
  1   65                        $ LATE STEAM OVERPRESSURE FAILURE
       2
     LAT_CF
       1    7
        OP ANC PR
        MAX
        EQUAL 0
      LATE STEAM OVERPRESSURE FAILURE CASE

  1   64                        $ CNTMT NOT FAIL.D BY H2 BURN LATE
       2
    NO_FAILUR
      1    5
         BURNPRES
         MAX
        GETHRESH 1 1.E20
     NO FAILURE CASE     -- FORCE SECOND BRANCH

  1   63                        $ DETONATION FAILURE OF CNTMT
       1
    DET_CF
```

```
      1    5
          BURNPRES
          MAX
         GETHRESH 1 1.E20
       NO FAILURE CASE      -- FORCE SECOND BRANCH
      OTHERWISE
         1      5
           BURNPRES
           FUN-F_MODE
           EQUAL 0
         GET MODE OF CONTAINMENT FAILURE

$
$
$ CET EVENT 10  LATE POOL BYPASS ************************************** PB_LATE
$
$
67   DRYWELL FAILURE DUE TO LATE HYDROGEN BURN IN CONTAINMENT          - DW_LATE
    2   'DW_FAIL'    'NO_DW_FAL'
    6       1            2
  2
   1  62                                $ LARGE BURN IGNITION IN CNTMT LATE
       2
      LG_BURN
       2    5           6
          BURNPRES  CNTMT PRESSURE
          FUN-DWDELP
          EQUAL 0
        PROB OF CONT BURN FAILING DRYWELL
      OTHERWISE                         $ NO LARGE BURNS IN CNTMT LATE
        2    5           6
          BURNPRES    CNTMT PRESSURE
          MAX
          GETHRESH  1 1.E20
        NO CNTMT H2 BURN

 68   POOL BYPASS LATE                                                    - LATE_PB
     2   'LAT_PL_BP'   'NO_LAT_BP'
     2       1            2
   6
    1  67                               $ DRYWELL FAILED BY LATE CNTMT H2 BURN
        1
      DW_FAIL
        1.          0.

    1  66                               $ POOL BYPASSED BY CONTAINMENT
        1                               $ ANCHORAGE FAILURE LATE
      ANCHORAGE
        1.          0.

    1  55                               $ POOL BYPASSED BY PEDESTAL
        2                               $ FAILURE DUE TO CCI EROSION
      AFTER_VB
        1.          0.
```

```
1  54                                  $ POOL BYPASSED BY OVERTEMPERATURE
   1                                   $ FAILURE OF DRYWELL PENETRATIONS
  DRY-CCI                              $ ASSUMED TO OCCUR WITH NO WATER IN
    1.          0.                     $ DRYWELL - SAME CONDITIONS WHICH CAUSE
                                       $ DRY CCI

                                       $ VACUUM BREAKERS FAILING OPEN FOR
                                       $ SEQUENCES WITH AC POWER (HENCE
                                       $ VB ISO VALVES ARE OPEN)
                                       $ AND LARGE H2 BURN OCCURS IN CONT
8  62       3       3       3       3       3       7      7
    2  * (( 3    +  4    +  6)  +  (1    +  5)  *  (1  +  2) )
   LG_BURN OTHER_TYPES CRIT_ATWS OTHERS   SBO LOOP_&_SBO PRIOR_RV CNTMT_LIM
    .05           .95
OTHERWISE                              $ BYPASS PROB FOR SEQUENCES WITH VB
     .0001          .9999              $ ISOLATED
```

APPENDIX H.3

PNPP IPE APET DESCRIPTION

The following description of the IPE Level 2 Accident Progression Event Tree (APET) provides a case by case discussion of each of the 68 events. The Perry IPE APET is processed by the Event Progression Analysis (EVNTRE) code developed at Sandia National Laboratories. Note that the EVNTRE code examines all cases sequentially, and once it finds a "true" outcome for a dependency it stops processing the case dependency logic. A complete discussion of the EVNTRE code is provided in the SAIC NUREG/CR--5174 reference manual (Griesmeyer 1989).

H.3.1. PLANT DAMAGE STATE GROUPING LOGIC

This APET Group consisting of events 1 thru 11 inputs the conditions prior to the initiation of core damage using the plant damage state grouping parameters and logic tree described in sections 4.3.1 and 4.3.2.

The outcomes of this APET group are Plant Damage States 1 thru 75. Reference Figure 4.3.2-1, Plant Damage State Grouping Logic. [It should be noted during the development of the Plant Damage State Event Trees, 150 Plant Damage States (or Classes) are model to include LOOP With No HVAC. When it was determined that LOOP With No HVAC is not a contributor to core damage, the Plant Damage States initially modeled were revised to 75.]

The PDS frequencies transferred into the NUS containment accident process code (NUPCAP+) and the frequencies are summed back through the Plant Damage State Grouping Logic to calculate the total Front-End Damage State Frequency and all the branch frequency. Manual calculations of the branch split fractions are then input into the depending sorting events in the APET Plant Damage State Grouping Logic to enable the Perry APET to determine the probability of each outcome with an input frequency of 1. The probability results can be transposed to frequency by multiplying through with the total Plant Damage State Core Damage Frequency.

EVENT 1. NOT A CONTAINMENT BYPASS FREQUENCY - CNT_BYP

Two branches:

NO BYPASS	Not A containment bypass sequence.
EVENT V	A containment bypass sequence.

EVNTRE Question Type: 1. (Independent sorting event)

PROBABILITIES:

CASE 1. No Containment Bypass Sequence

NO BYPASS	1.0
EVENT V	0.0

Event Dependencies: None

Quantification Basis: IPE Calculation PDS Branch Split Fractions.

EVENT 2. CONTAINMENT STATUS AT CORE DAMAGE - CNT_FAL

Two branches:

INTACT	Containment Intact At Core Damage.
FAILED	Containment Failed At Core Damage.

EVNTRE Question Type: 2. (Dependent split fraction)

PROBABILITIES:

CASE 1. Not A Containment Bypass Sequence

INTACT	0.7720
FAILED	0.2280

CASE 2. Ot[illegible]er reach this case.

Event Dependencies: Not A Containment Bypass Sequence (Event 1).

Quantification Basis: IPE Calculation PDS Branch Split Fractions.

EVENT 3. EVENT TYPE: FOR CONTAINMENT INTACT OR FAILED AT CORE DAMAGE

- EVENT_TYP

Six branches:

SBO	Station Blackout With Cntmt Intact At Core Damage
LOOP NO H	LOOP No HVAC With Containment Intact At Core Damage
OTHER TYPES	Other Event Types With Cntmt Intact At Core Damage
CRIT ATWS	Critical ATWS With Containment Failed at Core Damage
LOOP & SBO	LOOP & SBO With Containment Failed At Core Damage
OTHERS	All OTHERS With Containment Failed At Core Damage

EVNTRE Question Type: 2. (Dependent split fraction)

PROBABILITIES:

CASE 1. Containment Intact At Core Damage

SBO	0.1170
LOOP NO H	0.0
OTHER TYPES	0.8830
CRIT ATWS	0.0
LOOP & SBO	0.0
OTHERS	0.0

CASE 2. Otherwise, Containment Failed At Core Damage

SBO	0.0
LOOP NO H	0.0
OTHER TYPES	0.0
CRIT ATWS	0.1945
LOOP & SBO	0.1910
OTHERS	0.6145

Event Dependencies: Containment Status At Core Damage (Event 2)

Quantification Basis: IPE Calculation PDS Branch Split Fractions.

EVENT 4. INITIAL CONTAINMENT HEAT REMOVAL WITH SUPR POOL COOLING - SUPR_PL

Two branches:

NOT AVAILABLE	Suppression Pool Cooling Initially Not Available.
INIT SP COOLING	Initial Suppression Pool Cooling Available.

The Initial Containment Heat Removal With Suppression Pool Cooling functional characteristic was included to better characterize the containment for LOOP With No HVAC sequences. However, the IPE front-end review later determined that core damage from this sequence was not possible. The Back-End Plant Damage State set of functional characteristics are maintained in the interest of minimizing changes. The LOOP With No HVAC branching as shown in Figure 4.3.2-1 is not developed.

EVNTRE Question Type: 2. (Dependent split fraction)

PROBABILITIES:

CASE 1. LOOP With No HVAC Sequence

NOT AVAILABLE	1.
INIT SP COOLING	0.

CASE 2. Otherwise, Not A LOOP No HVAC Sequence.

NOT AVAILABLE	1.
INIT SP COOLING	0.

Event Dependencies: Event Type: Cntmt Intact/Failed At Core Damage (Event 3).

Quantification Basis: Not applicable for a token place holder value.

EVENT 5. CONTAINMENT VENT ISOLATED AT RPV FAILURE - CNT_ISOL

Two branches:

ISOLATED	Containment Isolated At RPV Failure.
NOT ISOLATED	Containment Not Isolated At RPV Failure.

EVNTRE Question Type: 2. (Dependent split fraction)

PROBABILITIES:

CASE 1. SBO Sequence

ISOLATED	0.9965
NOT ISOLATED	0.0035

CASE 2. Otherwise, Default to Isolated for other sequences.

ISOLATED	1.0
NOT ISOLATED	0.0

Event Dependencies: Event Type (Event 3).

Quantification Basis: IPE Calculation PDS Branch Split Fractions.

EVENT 6. RPV INJECTION FAILURE TIME - INJ_F_TIM

Three branches:

NO INJECT	SBO RPV Injection Failure Time: 0 - 2.8 Hours
RCIC	SBO RPV Injection Failure Time: 2.8 - 4.2 Hours
HPCS	SBO RPV Injection Failure Time: > 4.2 Hours
[No Branch	Verifies APET branching assignment.]

EVNTRE Question Type: 2. (Dependent split fraction)

PROBABILITIES:

CASE 1. SBO Event And Containment Isolated At Core Damage

NO INJECT 0.4347

RCIC	0.2798
HPCS	0.2854
No Branch	0.0000

CASE 2. SBO Event And Containment Not Isolated At Core Damage

NO INJECT	0.4347
RCIC	0.2798
HPCS	0.2854
No Branch	0.0000

CASE 3. LOOP With No HVAC And Initial Pool Cooling Not Available

NO INJECT	1.0
RCIC	0.0
HPCS	0.0
No Branch	0.0

CASE 4. LOOP With No HVAC And Initial Pool Cooling

NO INJECT	1.0
RCIC	0.0
HPCS	0.0
No Branch	0.0

CASE 5. Otherwise, Default to No Injection

NO INJECT	1.0
RCIC	0.0
HPCS	0.0
No Branch	0.0

Event Dependencies: Event Type (Event 3), Initial Containment Heat Removal With RHR Suppression Pool Heat Removal (Event 4).

Quantification Basis: IPE Calculation PDS Branch Split Fractions. Cases 3 and 4 are token place holders, so a basis is not applicable.

EVENT 7. OFFSITE POWER RECOVERY TIME - PWR_R_TIM

Three branches:

PRIOR RPV	Offsite power recovery prior to RPV failure.
CNTMT LIM	Offsite power recovery prior to containment limit.
NO RECOV	No recovery of offsite power.
[No Branch	Verifies APET branching assignment.]

EVNTRE Question Type: 2. (Dependent split fraction)

PROBABILITIES:

CASE 1. SBO And No Injection

PRIOR RPV	0.6148
CNTMT LIM	0.3567
NO RECOV	0.0285
No Branch	0.0000

CASE 2. SBO And RCIC Injection Failure

PRIOR RPV	0.2484
CNTMT LIM	0.7006
NO RECOV	0.0510
No Branch	0.0000

CASE 3. SBO And HPCS Injection Failure

PRIOR RPV	0.4231
CNTMT LIM	0.0000
NO RECOV	0.5769
No Branch	0.0000

CASE 4. Otherwise, Default to No Recovery

PRIOR RPV	0.0
CNTMT LIM	0.0
NO RECOV	1.0
No Branch	0.0

Event Dependencies: Event Type (Event 3), Offsite Power Recovery Time (Event 6).

Quantification Basis: IPE Calculation PDS Branch Split Fractions. Engineering judgement.

EVENT 8. CONTAINMENT HEAT REMOVAL WITH RHR SPRAY LOOP - SPRAY

Three branches:

RHR SPRAY	RHR containment spray loop available before containment overpressure limit threshold.
RHR POOL	RHR suppression pool cooling loop available before containment overpressure limit threshold.
NO RHR	No RHR loop available for containment heat removal.

It should be noted that RHR suppression pool cooling is currently not directly applied in the APET. Containment spray is the only RHR mode used for containment heat removal.

EVNTRE Question Type: 2. (Dependent split fraction)

PROBABILITIES:

CASE 1. SBO And Containment Not Isolated

RHR SPRAY	0.
RHR POOL	0.
NO RHR	1.

CASE 2. SBO, No Injection And Offsite Power Recovery Prior to RPV Failure

RHR SPRAY	0.8312
RHR POOL	0.0000
NO RHR	0.1688

CASE 3. SBO, No Injection And Offsite Power Recovery Prior to the Containment Limit

RHR SPRAY	0.8284
RHR POOL	0.0000
NO RHR	0.1716

CASE 4. SBO, RCIC Injection Failure And Offsite Power Recovery Prior to RPV Failure

RHR SPRAY	0.9479
RHR POOL	0.0000
NO RHR	0.0521

CASE 5. SBO, RCIC Injection Failure And Offsite Power Recovery Prior to Containment Limit

RHR SPRAY	0.9283
RHR POOL	0.0000
NO RHR	0.0717

CASE 6. SBO, HPCS Injection Failure And Offsite Power Recovery Prior to RPV Failure

RHR SPRAY	0.9090
RHR POOL	0.0000

NO RHR	0.0910

CASE 7. OTHER TYPES Sequences

RHR SPRAY	0.5171
RHR POOL	0.0000
NO RHR	0.4829

CASE 8. CRITICAL ATWS And Containment Failed At Core Damage

RHR SPRAY	0.3025
RHR POOL	0.0000
NO RHR	0.6975

CASE 9. CRITICAL ATWS And Containment Intact At Core Damage

This case is not included in the Plant Damage State Grouping Logic, but is included in the base case to fully characterize the APET framework for later sensitivity analysis on ATWS modifications.

RHR SPRAY	0.2
RHR POOL	0.0
NO RHR	0.8

CASE 10. Otherwise, Default to No RHR.

RHR SPRAY	0.
RHR POOL	0.
NO RHR	1.

Event Dependencies: Containment Status At Core Damage (Event 2), Event Type (Event 3), Containment Vent Isolated At Core Damage (Event 5), Offsite Power Recovery Time (Event 7).

Quantification Basis: IPE Calculation PDS Branch Split Fractions. Case 9 is applied to later sensitivity analysis of ATWS modification and the value is conservatively assigned by Engineering judgement.

EVENT 9. CONTAINMENT HEAT REMOVAL WITH VENT - VENT

Two branches:

VENT	Fuel Pool Cooling & Cleanup Vent or RHR Spray Herder Vent available before containment overpressure limit.
NO VENT	No vent available before containment overpressure limit.

EVNTRE Question Type: 2. (Dependent split fraction)

PROBABILITIES:

CASE 1. SBO, No Injection, Offsite Power Recovery Prior To RPV Failure, And RHR Containment Heat Removal Not Available

VENT	1.
NO VENT	0.

CASE 2. SBO, No Injection, Offsite Power Recovery Prior To Containment Limit, And RHR Containment Heat Removal Not Available

VENT	1.
NO VENT	0.

CASE 3. SBO, No Injection And No Offsite Power Recovery

VENT	0.8676
NO VENT	0.1324

CASE 4. SBO, RCIC Injection Failure, Offsite Power Recovery Prior To RPV Failure, And RHR Containment Heat Removal Not Available

VENT	1.
NO VENT	0.

CASE 5. SBO, RCIC Injection Failure, Offsite Power Recovery Prior To Containment Limit, And RHR Containment Heat Removal Not Available

VENT	1.
NO VENT	0.

CASE 6. SBO, RCIC Injection Failure And No Offsite Power Recovery

VENT	0.1824
NO VENT	0.8176

CASE 7. SBO, HPCS Injection Failure, Offsite Power Recovery Prior To RPV Failure, And RHR Containment Heat Removal Not Available

VENT	1.
NO VENT	0.

CASE 8. SBO, HPCS Injection Failure And No Offsite Power Recovery

VENT	0.7976
NO VENT	0.2024

CASE 9. OTHER TYPES of Event Sequences And RHR Containment Heat Removal Not Available

VENT	0.8394
NO VENT	0.1606

CASE 10. CRITICAL ATWS And Containment Intact At Core Damage

VENT	1.
NO VENT	0.

CASE 11. Otherwise, Venting Unnecessary Or Irrelevant - Default to No Vent.

VENT	0.
NO VENT	1.

Event Dependencies: Containment Status At Core Damage (Event 2), Event Type (Event 3), RPV Injection Failure Time (Event 6), Offsite Power Recovery Time (Event 7), Containment Heat Removal With RHR Loop (Event 8).

Quantification Basis: IPE Calculation PDS Branch Split Fractions. Case 10 is applied to later sensitivity analysis of ATWS modification and the value is assigned by Engineering judgement based on the success criteria.

EVENT 10. LATE IN-VESSEL INJECT & PEDESTAL CAVITY SUPPLY - LAT_INJ

Two branches:

LAT INJ	Late Injection Available Before RPV Failure.
NO LT INJ	No Late Injection.

EVNTRE Question Type: 2. (Dependent split fraction)

PROBABILITIES:

CASE 1. SBO, No Injection, Offsite Power Recovery Prior To RPV Failure, And RHR Spray Available

LAT INJ	0.9866
NO LT INJ	0.0134

CASE 2. SBO, No Injection, Offsite Power Recovery Prior To RPV Failure, And Vent Available

LAT INJ	0.9973
NO LT INJ	0.0027

CASE 3. SBO, No Injection, Offsite Power Recovery Prior To Containment Limit, And RHR Spray Available

LAT INJ	0.3393
NO LT INJ	0.6607

CASE 4. SBO, No Injection, Offsite Power Recovery Prior To Containment Limit, And RHR Spray Not Available

LAT INJ	0.6887
NO LT INJ	0.3113

CASE 5. SBO, No Injection, No Offsite Power Recovery And Vent Available

LAT INJ	0.9363
NO LT INJ	0.0637

CASE 6. SBO, No Injection, No Offsite Power Recovery And Vent Available

LAT INJ	1.
NO LT INJ	0.

CASE 7. SBO, RCIC Injection Failure, Offsite Power Recovery Prior To RPV Failure, And RHR Spray Available

LAT INJ	1.
NO LT INJ	0.

CASE 8. SBO, RCIC Injection Failure, Offsite Power Recovery Prior To RPV Failure, And Vent Available

LAT INJ	1.
NO LT INJ	0.

CASE 9. SBO, RCIC Injection Failure, Offsite Power Recovery Prior To Containment Limit

LAT INJ	0.
NO LT INJ	1.

CASE 10. SBO, RCIC Injection Failure, No Offsite Power Recovery, And Vent Available

LAT INJ	0.
NO LT INJ	1.

CASE 11. SBO, RCIC Injection Failure, No Offsite Power Recovery, And No Vent

LAT INJ	0.
NO LT INJ	1.

CASE 12. SBO, HPCS Injection Failure, Offsite Power Recovery Prior To RPV Failure, And RHR Spray Available

LAT INJ	1.
NO LT INJ	0.

CASE 13. SBO, HPCS Injection Failure, Offsite Power Recovery Prior To RPV Failure, And Vent Available

LAT INJ	1.
NO LT INJ	0.

CASE 14. SBO, HPCS Injection Failure, No Offsite Power Recovery, And Vent Available

LAT INJ	0.8432
NO LT INJ	0.1568

CASE 15. SBO, RCIC Injection Failure, No Offsite Power Recovery,

And No Vent

LAT INJ	0.7460
NO LT INJ	0.2540

CASE 16. SBO And Containment Vent Not Isolated At RPV Failure

LAT INJ	0.5046
NO LT INJ	0.4954

CASE 17. OTHER TYPES of Sequences And RHR Spray Available

LAT INJ	0.9969
NO LT INJ	0.0031

CASE 18. OTHER TYPES of Sequences And Vent Available

LAT INJ	0.9716
NO LT INJ	0.0284

CASE 19. OTHER TYPES of Sequences And No Vent

LAT INJ	0.00022
NO LT INJ	0.99978

CASE 20. CRITICAL ATWS And Containment Intact At Core Damage

LAT INJ	1.
NO LT INJ	0.

CASE 21. CRITICAL ATWS And Containment Failed At Core Damage

LAT INJ	0.95
NO LT INJ	0.05

CASE 22. SBO & LOOP Sequences And
Containment Failed At Core Damage

LAT INJ	0.4749
NO LT INJ	0.5203

CASE 23. All OTHERS With Containment Failed At Core Damage

LAT INJ	0.3137
NO LT INJ	0.6863

CASE 24. Otherwise, Should not reach this case.

LAT INJ	0.
NO LT INJ	1.

Event Dependencies: Containment Status At Core Damage (Event 2), Event Type (Event 3), RPV Injection Failure Time (Event 6), Offsite Power Recovery Time (Event 7), Containment Heat Removal With RHR Loop (Event 8), Containment Heat Removal With Vent (Event 9).

Quantification Basis: IPE Calculation PDS Branch Split Fractions.

Case 20 is applied to later sensitivity analysis of ATWS modification and the value is assigned by Engineering judgement based on the success criteria.

Case 21 is applied to the CRITICAL ATWS sequence where Late Injection Availability is not explicitly modeled in the PDS event trees and the value is selected based on examination of similar late injection sequences when AC power is available and containment is intact at core damage.

EVENT 11. RPV DEPRESSURIZED DURING CORE DAMAGE - RX_PRESS

Two branches:

LOW PRES	RPV Depressurized During Core Damage.
HI PRES	RPV Not Depressurized During Core Damage.

EVNTRE Question Type: 2. (Dependent split fraction)

PROBABILITIES:

CASE 1. SBO, No Injection, Offsite Power Recovery Prior To RPV Failure, Vent Available, And Late Injection Available

LOW PRES	0.4650
HI PRES	0.5350

CASE 2. SBO, No Injection, Offsite Power Recovery Prior To RPV Failure, Vent Available, And No Late Injection

LOW PRES	1.
HI PRES	0.

CASE 3. SBO, No Injection, Offsite Power Recovery Prior To Containment Limit, Vent Available, And Late Injection Available

LOW PRES	0.2284
HI PRES	0.7716

CASE 4. SBO, No Injection, No Offsite Power Recovery, Vent Available, And Late Injection Available

LOW PRES	0.9904
HI PRES	0.0096

CASE 5. SBO, No Injection, No Offsite Power Recovery, Vent Available, And No Late Injection

LOW PRES	1.
HI PRES	0.

CASE 6 SBO, No Injection, No Offsite Power Recovery, No Vent, And Late Injection Available

LOW PRES	1.
HI PRES	0.

CASE 7. SBO, RCIC Injection Failure, No Offsite Power Recovery, No Vent, And No Late Injection

LOW PRES	0.
HI PRES	1.

CASE 8. SBO, HPCS Injection Failure, No Offsite Power Recovery, Vent Available, And Late Injection Available

LOW PRES	0.8014
HI PRES	0.1986

CASE 9. SBO, HPCS Injection Failure, No Offsite Power Recovery, Vent Available, And No Late Injection

LOW PRES	0.7704
HI PRES	0.2296

CASE 10. SBO, HPCS Injection Failure, No Offsite Power Recovery, No Vent, And Late Injection Available

LOW PRES	0.9768
HI PRES	0.0232

CASE 11. SBO, HPCS Injection Failure, No Offsite Power Recovery, No Vent, And No Late Injection

LOW PRES	1.
HI PRES	0.

CASE 12. SBO, Containment Not Isolated At RPV Failure, And Late Injection Available

LOW PRES	0.6^32
HI PRES	0.3068

CASE 13. SBO, Containment Not Isola...J At RPV Failure, And No Late Injection

LOW PRES	1.
HI PRES	0.

CASE 14. All other SBO Sequences are Depressurized

LOW PRES	1.
HI PRES	0.

CASE 15. OTHER TYPES Sequences, RHR Spray Available And Late Injection Available

LOW PRES	0.9924
HI PRES	0.0076

CASE 16. OTHER TYPES Sequences, No RHR Available, Vent Available And Late Injection Available

LOW PRES	0.9950
HI PRES	0.0050

CASE 17. OTHER TYPES Sequences, No RHR Available, No Vent And Late Injection Available

LOW PRES	1.
HI PRES	0.

CASE 18. OTHER TYPES Sequences, No RHR Available,

No Vent And No Late Injection

LOW PRES	0.9255
HI PRES	0.0745

CASE 19. CRITICAL ATWS And Containment Intact At Core Damage

LOW PRES	1.
HI PRES	0.

CASE 20. CRITICAL ATWS, Containment Failed At Core Damage And RHR Spray Available

LOW PRES	0.9918
HI PRES	0.0082

CASE 21. CRITICAL ATWS, Containment Failed At Core Damage And No RHR

LOW PRES	0.7458
HI PRES	0.2542

CASE 22. SBO & LOOP Sequences, Containment Failed At Core Damage, And Late Injection Available

LOW PRES	0.99940
HI PRES	0.00060

CASE 23. SBO & LOOP Sequences, Containment Failed At Core Damage, And No Late Injection

LOW PRES	0.8882
HI PRES	0.1118

CASE 24. In All Remaining Sequences With the Containment Intact or Failed At Core Damage The RPV Is Depressurized During Core Damage

LOW PRES	1.
HI PRES	0.

CASE 25. Otherwise, Should Never Reach This Case.

LOW PRES	1.
HI PRES	0.

Event Dependencies: Containment Status At Core Damage (Event 2), Event Type (Event 3), RPV Injection Failure Time (Event 6), Offsite Power Recovery Time (Event 7), Containment Heat Removal With RHR Loop (Event 8), Containment Heat Removal With Vent (Event 9), Late In-Vessel Injection & Pedestal Cavity Supply (Event 10).

Quantification Basis: IPE Calculation PDS Branch Split Fractions.

Case 19 is applied to later sensitivity analysis of ATWS modification and the value is assigned by Engineering judgement based on the success criteria.

H.3.2. DEBRIS COOLED IN-VESSEL

This APET Group determines in Events 12 thru 15 the probability that the debris is cooled in-vessel (and reactor vessel failure is prevented).

The two possible outcomes from this APET group are:

Debris is Cooled In-Vessel	(and RPV Failure Prevented)
Debris is Not Cooled In-Vessel	(and RPV Failure Occurs)

EVENT 12. LATE RPV LOW PRESSURE INJECTION AVAILABLE - LATE_INJ

Three branches:

WATER INJECT	Late water injection into vessel.
NO INJECTION	No late water injection.
CRITICAL	ATWS and the reactor is not shutdown with recovery action.

This question assesses whether there is water injection to the reactor vessel established subsequent to core damage but prior to reactor failure in excess of that provided by the CRD system alone. In addition, there is a separate branch for the ATWS sequence where it is assumed that even if a water supply is provided to the vessel subsequent to core damage that in-vessel cooling is not possible.

EVNTRE Question Type: 2. (Dependent Sorting Event)

SORTING:

CASE 1. No Late In-Vessel Injection

WATER INJECT	0.
NO INJECTION	1.

CRITICAL 0.

CASE 2. CRITICAL ATWS

WATER INJECT 0.
NO INJECTION 0.
CRITICAL 1.

CASE 3. Late In-Vessel Injection Available

WATER INJECT 1.
NO INJECTION 0.
CRITICAL 0.

CASE 4. Otherwise, Should never reach this case.

WATER INJECT 0.
NO INJECTION 1.
CRITICAL 0.

Event Dependencies: Event Type (Event 3), and Late In-Vessel Injection and Cavity Supply (Event 10).

EVENT 13. RPV DEPRESSURIZED DURING CORE DAMAGE - RX_PRESS

Two Branches:

LOW PRESSURE RPV Depressurized before core plate failure
HI PRESSURE RPV Not Depressurized during core damage

In-vessel cooling requires the that the reactor pressure vessel be depressurized to permit the low pressure Late-Injection systems to provide flow to the reactor core.

EVNTRE Question Type: 2. (Dependent sorting event)

SORTING:

CASE 1. All Sequences Where RPV Is Depressurized During Core Damage

LOW PRESSUR 1.
HI PRESSURE 0.

CASE 2. Otherwise, Assign to High Pressure

LOW PRESSUR 0.
HI PRESSURE 1.

Event Dependencies: RPV Depressurized During Core Damage (Event 11)

EVENT 14. DEBRIS MASS MOLTEN AT VESSEL BREACH - MOLTEN_VB

Two branches:

LARGE DEBRIS	Large mass of molten debris in lower reactor vessel
SMALL DEBRIS	Small mass of molten debris in lower reactor vessel

The mass of molten debris in the reactor vessel lower head is a key factor in determining the probability of successful in-vessel cooling. Two discrete magnitudes were evaluated: Large (~40% of core mass), and Small (~10% of core mass). The Grand Gulf (Brown 1990) APET provided the following estimated probabilities.

EVNTRE Question Type: 2.

PROBABILITIES:

CASE 1. No Late Injection Or RPV Not Depressurized During Core Damage.

LARGE DEBRIS	0.1
SMALL DEBRIS	0.9

CASE 2. Otherwise, for Late Injection Available and the RPV Depressurized During Core Damage, or Critical reactor state.

LARGE DEBRIS	0.025
SMALL DEBRIS	0.975

Event Dependencies: Late RPV Low Pressure Injection Available (Event 12), and RPV Depressurized During Core Damage (Event 13).

Quantification Basis: Grand Gulf (Brown 1990) APET Event Number 61.

EVENT 15. DEBRIS COOLED IN-VESSEL - INV_COOL

Two Branches:

COOLED IN-VESSEL	Debris Cooled In-Vessel
NOT COOLED IN-VESSEL	Debris Not Cooled In-Vessel

In NUREG/CR-4551 the mass of molten material in the reactor vessel at vessel breach was used as a discriminant for the coolability of the debris in-vessel. For sequences with a large mass of molten debris (~40% of core mass) it was judged that the debris had a 0.5 probability of being cooled in-vessel. For sequences with a smaller amount of molten debris (~10% molten), the NUREG/CR-4551 authors judged that the probability of successfully cooling the debris in-vessel is approximately 0.75.

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. Critical Reactor, And Containment Failed Before Core Damage

COOLED INV	0.
NOT COOLED INV	1.

CASE 2. Critical Reactor And Alternate Shutdown With Containment Integrity Maintained

This case is designed to apply to the later sensitivity analysis of the impact of controlling RPV Power/Level during an unmitigated ATWS such that the containment is maintained intact at core damage. Therefore this case applies to recovered ATWS sequences with alternate shutdown, if late injection is available in Event 12, Late RPV Low Pressure Injection Available, then In-Vessel Core Cooling is fully credited.

COOLED INV	1.
NOT COOLED INV	0.

CASE 3. Water Injection Available, RPV Depressurized During Core Damage And Large Molten Debris Mass

COOLED INV	0.5
NOT COOLED INV	0.5

CASE 4. Water Injection Available, RPV Depressurized During Core Damage And Small Molten Debris Mass

COOLED INV	0.75
NOT COOLED INV	0.25

CASE 5. Otherwise, No Cooling Water Available For Other Sequences.

COOLED INV 0.

NOT COOL INV 1.

Event Dependencies: Late Water Injection to Reactor Vessel (Event 12), RPV Depressurized During Core Damage (Event 13), and Debris Mass Molten at Vessel Breach (Event 14).

Quantification Basis: Grand Gulf (Brown 1990) APET Event Number 63.

H.3.3. MODE OF CONTAINMENT FAILURE BEFORE RPV FAILURE

This APET group determines in Events 16 thru 25 the probability and the mode of containment failure following a hydrogen deflagration or a hydrogen detonation.

A hydrogen deflagration results in a sudden pressure rise that can fail the containment in either of two possible modes: penetration failure (which includes the equipment hatch), or anchorage failure. The penetration failure modes are grouped with the equipment hatch failure mode to minimize the complexity of the analysis.

The three possible outcomes from this DET group are:

ANCHORAGE	Concrete/steel anchorage failure
PENETRATION/DOME	Penetration, containment shell, or equipment hatch failure
NO FAILURE	No containment failure

EVENT 16. HYDROGEN IGNITION SYSTEM AVAILABLE - H2_IGN

Two Branches:

HIS OFF	Hydrogen Ignitor System Inoperable or In Off Status
HIS ON	Hydrogen Ignitor System Available and In On Status

EVNTRE Question: Type 2. (Dependent Split Fraction)

PROBABILITIES:

CASE 1. No Loss Of AC Power

For this case it is assumed the ignitors are very likely to be available through out the sequence. The likelihood of the HIS being maintained off when AC Power is available is associated with the human interaction, Operator Fails To Initiate Hydrogen Ignition System. Monitoring and controlling hydrogen with the hydrogen analyzers and hydrogen ignitors is integrated in the simulator training of the Plant Emergency Instruction (PEI). The most limiting time window for hydrogen generation to commence is for a loss of all injection accident where the maximum core clad temperature reaches 2200 F after 51 minutes. During licensed operator simulator training, the control room operators are trained to routinely monitor and control hydrogen in the PEI flow chart by placing the hydrogen analyzers in service in the third step of the RPV Control entry. During a loss of all injection the reactor water level decreases below Level 1 (< 16.5 inches above the Top of the Active Fuel) at about 29 minutes. The RPV water level decrease below Level 1 is the entry condition to Hydrogen Control. The Unit Supervisor will typically direct action from the Hydrogen Control flow chart that the Hydrogen ignitors be placed in service about 1 minute after entering the flow chart. The execution time to initiate this system at a nearby back panel is relatively short requiring about 30 seconds. Therefore, the human error probability of 0.005 is considered conservative.

HIS OFF	0.005
HIS ON	0.995

CASE 2. Otherwise, For Remaining Sequences
Default to AC Power Lost Early

For this case, it is assumed that the AC powered ignitors are not available at any time during the sequence. Even if power recovery occurs it is not clear that the ignitors would be turned on in sufficient time to mitigate the accident. The Perry Plant Emergency Instructions direct the operators not to turn on the ignitors until containment and drywell hydrogen measurement is available and the respective concentrations are determined to be less than the associated Hydrogen Deflagration Overpressure Limits of 9% and 6%. It is estimated that hydrogen measurement would require as much as two hours following power recovery. The hydrogen analyzer equipment is AC powered and requires at least an hour to reach steady state calibrated conditions following restoration of AC power.

HIS OFF	1.
HIS ON	0.

Event Dependencies: Event Type (Event 3).

Quantification Basis: IPE Human Interaction Technical Assignment File, Perry Plant Emergency Instruction, and Discussion with Operation and Chemistry Staff.

EVENT 17. CONTAINMENT VENT ISOLATED BEFORE RPV FAILURE - ISOL

Two branches:

ISOLATED	Containment Isolated
NOT ISOLATED	Containment Not isolated (Fuel Pool Cooling & Cleanup Isolation Open)

This event determines whether the containment is isolated, or not, at the time of RPV failure for SBO sequences. Branching under this event is determined uniquely by the branch taken under Plant Damage State Event 5.

EVNTRE Question: 2. (Dependent Sorting Event)

SORTING:

CASE 1. Containment Vent Isolated Before RPV Failure For SBO Sequences

ISOLATED	1.
NOT ISOLATED	0.

CASE 2. Otherwise, Postulate Not Isolated For SBO Sequences.

ISOLATED	0.
NOT ISOLATED	1.

Event Dependencies: Containment Vent Isolated At RPV Failure (Event 5).

Quantification Basis: Not Applicable.

EVENT 18. MODE OF SPRAY OPERATION EARLY - RHR_MODE

Three Branches:

CONTROLLED	Sprays are operating in a throttled cooling mode to mitigate or exclude hydrogen deflagrations and detonations.
SPRAY	Sprays are operating at full design cooling.
NO SPRAY	Sprays are not available.

This event defines the conditions of containment spray operation at the time of core damage or vessel failure. The sprays can either be available or not available. If a containment spray loop is available, the operators could operate the system in a throttled RHR heat exchanger flow control mode such that the containment atmosphere remains at elevated pressure (~30 psig) in order to assure that the containment remains steam-inert to hydrogen combustion or maintains the a[illegible]ound containment steam concentration to minimize the expected peak burn pressures.

EVNTRE Question Type: 2. (Dependent Sorting Event)

SORTING:

CASE 1. RHR Spray Not Available

RHR Spray is determined to be not available in the Plant Damage State Grouping Logic.

CONTROLLED	0.
SPRAY	0.
NO SPRAY	1.

CASE 2. AC Power Lost and Not Recovered Prior to RPV Failure

CONTROLLED	0.
SPRAY	0.
NO SPRAY	1.

CASE 3. AC Power Never Lost And RHR Spray Available

With AC Power available, it is almost certain that the hydrogen igniters are available to control hydrogen by combining the available oxygen with the hydrogen generated. Therefore, a controlled, steam-inerted containment atmosphere is not necessary to credit the mitigative impact of controlled spray operation.

CONTROLLED	0.
SPRAY	1.
NO SPRAY	0.

CASE 4. AC Power Lost and Recovered Prior to RPV Failure And RHR Spray Available

CONTROLLED	0.
SPRAY	1.
NO SPRAY	0.

Under these conditions it is probable that there will be significant quantities of hydrogen released into the containment and that the hydrogen ignition system will not be energized during the first hour or two following AC power recovery. (Reference Event 16 discussion.) Under these conditions it is possible to limit the threat from hydrogen combustion by throttling the RHR bypass flow to maintain the containment steam concentration in the steam-inert regime. However, since this mode of operation is not in the Perry Plant Emergency Instruction, this mode of operation has been assigned a zero probability for base case evaluation.

CASE 5. Otherwise, Should never reach this case.

CONTROLLED	0.
SPRAY	1.
NO SPRAY	1.

Event Dependencies: Event Type (Event 3), Offsite Power Recovery Time (Event 7), and Containment Heat Removal With RHR Spray Loop (Event 8).

Sorting Basis: Perry Plant Emergency Instruction and Engineering Judgement.

EVENT 19. CONTAINMENT STEAM CONCENTRATION BEFORE RPV FAILURE - ST_CONC

Six branches:

0-15 %	Containment Steam Volume Percent
15-25 %	
25-35 %	
35-45 %	
45-55 %	
>55 %	

The hydrogen burn ignition probability and burn efficiency is a function of the steam concentration in the containment. The steam concentration regime probabilities given for the cases below were estimated using MAAP results for each case and considering the variability of suppression pool temperature.

EVNTRE Question Type: 2. (Dependent Split Fraction)

PROBABILITIES:

CASE 1. Spray Loop Operation At Design Cooling

With a containment spray loop operating at design cooling with containment heat removal optimized with the Residual Heat Removal heat exchanger, the steam concentration would be low.

0-15 %	1.
15-25 %	0.
25-35 %	0.
35-45 %	0.
45-55 %	0.
>55 %	0.

CASE 2. SBO And No Injection Failure Early (0 - 2.8 Hrs)

0-15 %	1.
15-25 %	0.
25-35 %	0.
35-45 %	0.
45-55 %	0.
>55 %	0.

CASE 3. SBO And RCIC Injection Failure (2.8-4.2 Hrs)

0-15 %	0.
15-25 %	0.51
25-35 %	0.[illegible]9
35-45 %	0.
45-55 %	0.
>55 %	0.

CASE 4. SBO And HPCS Injection Failure (> 4.2 Hrs)

[illegible]-15 %	0.
15-25 %	0.
25-35 %	0.
35-45 %	0.
45-55 %	0.
>55 %	1.

CASE 5. Containment Failed At Core Damage

0-15 %	0.
15-25 %	0.
25-35 %	0.
35-45 %	0.
45-55 %	0.
>55 %	1.

CASE 6. Otherwise, Other Sequences Default To Low Steam Concentration

All other sequences are conservatively evaluated by assuming a low steam concentration (which implies fast core damage) and can produce a higher expected burn pressure.

0-15 %	1.
15-25 %	0.
25-35 %	0.
35-45 %	0.
45-55 %	0.
>55 %	0.

Event Dependencies: Containment Status At Core Damage (Event 2), Event Type (Event 3), RPV Injection Failure Time (Event 6), and Mode of RHR Spray Operation Early (Event 18).

Quantification Basis: IPE Engineering Calculation - Hydrogen Burns (see APET Event 20 for summary description of hydrogen generation modeling which is related to the steam concentration), and Engineering Judgement.

EVENT 20. FRACTION OF CORE INVENTORY OF ZIRCONIUM REACTED IN-VESSEL - H2_INV

Three Branches:

11 %	Percent In-Vessel Zirconium Reacted In-Vessel
22 %	
33 %	

The mean values for the fraction of core inventory reacted in-vessel ranged from 0.104 to 0.218 for the long term ATWS and short term SBO case described in NUREG/CR-4551. Perry MAAP calculations for a wide spectrum of sequence types show the fraction of zirconium reaction ranging from approximately 0.03 to 0.20 for the no injection SBO (modeled with Local Blockage), and for RCIC injection loss with a cool suppression pool (modeled by No Blockage - no local node cutoff). The Perry IPE MAAP calculations implemented the best-estimate values for the model parameters discussed in the "Recommended Sensitivity Analyses For An Individual Plant Examination Using MAAP 3.0B" (EPRI 1991). The IDCOR BWR "blockage" model of Local Blockage option was considered more likely and the No Blockage option less likely, and a probability of 0.8 and 0.2 was assigned to each option, respectively. These results were used to define the split fractions shown below.

EVNTRE Question: Type 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. SBO And No Injection Failure Early (0 - 2.8 Hrs)

11 %	1.
22 %	0.
33 %	0.

CASE 2. SBO And RCIC Injection Failure (2.8 - 4.2 Hrs)

11 %	0.87
22 %	0.13
33 %	0.

CASE 3. SBO And HPCS Injection Failure (> 4.2 Hrs)

11 %	1.
22 %	0.
33 %	0.

CASE 4. Otherwise, Other Sequences Are Assigned The Maximum Fraction Zirc Reacted

Used in Case 2, SBO With RCIC Injection Loss

11 %	0.87
22 %	0.13
33 %	0.

Event Dependencies: Event Type (Event 3), and RPV Injection Failure Time (Event 6).

Quantification Basis: Grand Gulf (Brown 1990) APET Event 35, "Recommended Sensitivity Analyses For An IPE Using MAAP 3.0B" (EPRI 1991), and Perry IPE Engineering Calculations using "Recommended Sensitivity Analyses For An IPE Using MAAP 3.0B" (EPRI 1991).

EVENT 21. SMALL BURNS AT LOW HYDROGEN CONCENTRATION - SM_BRN

Two Branches:

NO SMALL	No Small Burns Occur
SMALL BURN	Small Burns Occur

If small burns are ignited at low hydrogen concentrations, then the threat to containment integrity from a large hydrogen burn is effectively eliminated. Ignition of small burns at low hydrogen concentrations depletes the containment of hydrogen (and oxygen) and results in an ignition source for later hydrogen burns (from residual fires or smoldering transient "trash").

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. Inert Steam Atmosphere

For this case the occurrence of small burns is not possible.

NO SMALL	1.
SMALL BURN	0.

CASE 2. Hydrogen Ignitors On

For this case the occurrence of small burns are virtually assured.

NO SMALL	0.
SMALL BURN	1.

CASE 3. AC Power Available, But Ignitors Not Available

Under these conditions NUREG/CR-4551 indicated that it was likely that the ignition source available as a result of the operation of AC powered equipment inside containment would result in the ignition of a small burn at low hydrogen concentrations.

NO SMALL	0.25
SMALL BURN	0.75

CASE 4. AC Power Lost and Recovered Prior to Vessel Failure

NUREG/CR-4551 estimated that there was a small time window within which AC power had to be recovered following the initiation of core damage in order for a small burn to be ignited (prior to release of sufficient large amounts of hydrogen that a large burn would occur). The following probabilities are taken from NUREG/CR-4551 for the case without the igniters available.

NO SMALL	0.96
SMALL BURN	0.04

CASE 5. AC Power Lost and Not Recovered Before RPV Failure

For this case ignition of a small burn at low hydrogen concentrations was estimated to be negligible from random ignition sources.

NO SMALL	1.
SMALL BURN	0.

CASE 6. Otherwise, Should Never Reach This Case

NO SMALL	1.
SMALL BURN	0.

Event Dependencies: Containment Steam Concentration Before RPV Failure (Event 19), Hydrogen Ignitors Available (Event 16), Event Type (Event 3), RPV Injection Failure Time (Event 6) and Offsite Power Recovery (Event 7).

Quantification Basis: Grand Gulf (Brown 1990) APET Event 41.

EVENT 22. LARGE HYDROGEN BURN DURING CORE DAMAGE - LG_BURN

Two Branches:

NO BURN	No Large Hydrogen Burn Ignited
LG BURN	Large Hydrogen Burn Ignited

This event assess whether a large hydrogen burn is ignited in containment prior to reactor vessel failure. If a hydrogen burn is ignited this event sets the value of two parameters (1 and 3) used in subsequent events. Parameter 1 is the peak containment pressure (psig) expected from the hydrogen burn. Parameter 3 is the pre-existing containment pressure (psig) before the burn.

The expected peak containment pressure was determined as follows. First, the adiabatic isochoric combustion pressure (AICC) was calculated for each combination of hydrogen concentration and steam concentration resulting from the branching under Events 19 and 20. Then, the AICC pressure was multiplied by a burn efficiency factor taken from NUREG/CR-4551 for each steam and hydrogen concentration case, which relates the expected peak pressure to the maximum (AICC) burn pressure. Table H.3.2-1 shows the results of the hydrogen pressure calculation and the input for each case where a large hydrogen burn was predicted to occur. Table H.3.2-1 also shows the probability assigned for ignition of a large burn.

EVNTRE Question Type: 4. (Dependent Split Fractions with parameter values set for each case)

PROBABILITIES:

CASE 1. Containment Failed At Core Damage

The steam-inert containment atmosphere prevents hydrogen combustion. Note in this case, the EVNTRE parameter for the Peak Containment Pressure is used to carry the probability of anchorage failure for gradual steam overpressurization.

NO BURN	1.				
LG BURN	0.	BASE PRESS	0	PEAK PRESS	.15

CASE 2. Small Burn Occurs At Low Hydrogen Concentrations

The small burn is assumed to prohibit the occurrence of a subsequent large burn.

NO BURN	1.				
LG BURN	0.	BASE PRESS	0	PEAK PRESS	0

CASE 3. Containment Concentration Is Greater Than 55%

The steam-inert containment atmosphere prevents hydrogen combustion. Note in this case, the EVNTRE parameter for the Peak Containment Pressure is used to carry the probability of anchorage failure for gradual steam overpressurization.

NO BURN	1.				
LG BURN	0.	BASE PRESS	0	PEAK PRESS	.15

CASE 4. Cntmt Steam [0-15%] & 33% Zirc Reacted, [H2] = 25.6%

NO BURN	0.5				
LG BURN	0.5	BASE PRESS	9	PEAK PRESS	142

CASE 5. Cntmt Steam [0-15%] & 22% Zirc Reacted, [H2] = 18.8%

NO BURN	0.5				
LG BURN	0.5	BASE PRESS	6	PEAK PRESS	98

CASE 6. Cntmt Steam [0-15%] & 11% Zirc Reacted, [H2] = 10.5%

NO BURN	0.72				
LG BURN	0.28	BASE PRESS	4	PEAK PRESS	40

CASE 7. Cntmt Steam [15-25%] & 33% Zirc Reacted, [H2] = 22.1%

NO BURN	0.5				
LG BURN	0.5	BASE PRESS	14	PEAK PRESS	123

CASE 8. Cntmt Steam [15-25%] & 22% Zirc Reacted, [H2] = 16.2%

NO BURN	0.5				
LG BURN	0.5	BASE PRESS	11	PEAK PRESS	87

CASE 9. Cntmt Steam [15-25%] & 11% Zirc Reacted, [H2] = 9.0%

NO BURN	0.72				
LG BURN	0.28	BASE PRESS	9	PEAK PRESS	37

CASE 10. Cntmt Steam [25-35%] & 33% Zirc Reacted, [H2] = 19.4%

NO BURN	0.5				
LG BURN	0.5	BASE PRESS	20	PEAK PRESS	113

CASE 11. Cntmt Steam [25-35%] & 22% Zirc Reacted, [H2] = 14.2%

NO BURN	0.61				
LG BURN	0.39	BASE PRESS	16	PEAK PRESS	80

CASE 12. Cntmt Steam [25-35%] & 11% Zirc Reacted, [H2] = 7.9%

NO BURN	0.75				
LG BURN	0.25	BASE PRESS	13	PEAK PRESS	36

CASE 13.	Cntmt Steam [35-45%] & 33% Zirc Reacted, [H2] = 16.6%			
	NO BURN	0.56		
	LG BURN	0.44	BASE PRESS 27	PEAK PRESS 104
CASE 14.	Cntmt Steam [35-45%] & 22% Zirc Reacted, [H2] = 12.2%			
	NO BURN	0.67		
	LG BURN	0.33	BASE PRESS 23	PEAK PRESS 75
CASE 15.	Cntmt Steam [35-45%] & 11% Zirc Reacted, [H2] = 6.8%			
	NO BURN	0.77		
	LG BURN	0.23	BASE PRESS 19	PEAK PRESS 37
CASE 16.	Cntmt Steam [45-55%] & 33% Zirc Reacted, [H2] = 13.8%			
	NO BURN	0.61		
	LG BURN	0.39	BASE PRESS 37	PEAK PRESS 98
CASE 17.	Cntmt Steam [45-55%] & 22% Zirc Reacted, [H2] = 10.2%			
	NO BURN	0.72		
	LG BURN	0.28	BASE PRESS 32	PEAK PRESS 76
CASE 18.	Cntmt Steam [45-55%] & 11% Zirc Reacted, [H2] = 5.7%			
	NO BURN	0.77		
	LG BURN	0.23	BASE PRESS 27	PEAK PRESS 40
CASE 19.	Otherwise, Should Never Reach This Case			
	NO BURN	1.		
	LG BURN	0.	BASE PRESS 0	PEAK PRESS 0

Event Dependencies: Containment Steam Concentration (Event 19), Fraction of Zircalloy Reacted In-Vessel (Event 20), Small Burns At Low Hydrogen Concentration (Event 21).

Quantification Basis: Grand Gulf (Brown 1990) APET Events 43 and 46, and Perry IPE Engineering Calculation - Hydrogen Burns (reference intermediate results in TABLE H.3.2-1.)

EVENT 23. HYDROGEN DETONATION CONTAINMENT FAILURE BEFORE RPV FAILURE - H2_DET

Two Branches:

DET CF	Hydrogen Detonation Containment Failure
NO	No Hydrogen Detonation Containment Failure

Given that the containment hydrogen is ignited (as previously determined in the APET by taking the Large Burn Ignited During Core Damage branch) this event assesses whether a detonation occurs which fails the containment. It is assumed that if detonation does occur, that it occurs in the upper containment and results in a large (> 7 square feet) failure area in the dome or upper cylindrical wall of the containment.

The probabilities for containment failure estimated below are based on the results from the Grand Gulf NUREG/CR-4551 analyses for hydrogen detonation loading and the Sehqoyah steel containment impulse strength.

Based on NUREG/CR-4551 results detonations are assumed to have negligible probabilities for steam concentrations in excess of 35% or hydrogen concentrations less than 12%. The probabilities of hydrogen detonation and the mean impulse loading under other conditions are shown in Table H.3.2-2 below.

The mean value of the containment failure impulse for Grand Gulf given in NUREG/CR-4551 was 19.5 kPa-s. However, unlike Grand Gulf, Perry is a free standing steel containment. Hence, the nearest containment design to Perry of the NUREG-1150 plants is Sequoyah which also has a free standing steel containment. For the Sequoyah plant the mean value of the containment failure was estimated to be 12.1 kPa-s (for the upper containment region). Assuming this value is appropriate to assess the containment failure impulse loading at Perry the containment failure probabilities shown on Table H.3.2-2 were estimated.

As in NUREG/CR-4551, the "high" steam level case represents sequences which were initially inert to detonations and where the steam is condensed and brought into the detonation range by spray operation. The low steam level corresponds to cases where the detonatable mixture formed while the containment atmosphere remained in a flammable regime (non-inert). In the Perry IPE analyses, it is assumed that the only time that a "high" steam level case could occur would be for SBO sequences with power (and spray) recovery subsequent to core damage.

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. No Large Burn Ignited

With no large burn, no trigger exists for a hydrogen detonation.

DET CF	0.
NO	1.

CASE 2. Containment Steam Concentration Greater Than 35%

Steam concentrations greater than 35% prevent hydrogen detonation.

DET CF	0.
NO	1.

CASE 3. 11% Fraction of Zirconium Reacted

Hydrogen concentrations of < 12% are associated with 11% Zirconium Reacted sequences which do not support detonation.

DET CF	0.
NO	1.

CASE 4. Containment Steam Concentration High And Slowly Decreasing Less Than 35%, And Hydrogen Concentration Is 12-16%

This occurs when power is recovered before RPV failure and sprays are available. This range of hydrogen concentration can be associated with 22% Zirc Reacted sequences. This case is assigned to those sequences where the containment steam concentration is 0-15%, the Fraction of Zirc Reacted In-Vessel is 22%, the Event Type is SBO, Power is Recovered Before RPV Failure, the Mode of RHR Spray Operation Early is Spray, and no Loss of Injection Early.

DET CF	0.022
NO	0.978

CASE 5. Containment Steam Concentration Low And Hydrogen Concentration Is 12-16%

This occurs when the containment steam concentration is between 25-35% and the Fraction of Zirc Reacted is 33%.

DET CF	0.
NO	1.

CASE 6. Containment Steam Concentration Low And Hydrogen Concentration Is 16-20%

This occurs when the containment steam concentration is between 0-25% and the Fraction of Zirc Reacted is 22%.

DET CF	0.16
NO	0.84

CASE 7. Containment Steam Concentration High And Slowly Decreasing Less Than 35%,

And Hydrogen Concentration is Greater Than 20%

This occurs when power is recovered before RPV failure and sprays are available. This range of hydrogen concentration can be associated with 33% Zirc Reacted sequences. This case is assigned to those sequences where the containment [Steam] is 0-15%, the Fraction of Zirc Reacted In-Vessel is 33%, the Event Type is SBO, Power is Recovered Before RPV Failure, the Mode of RHR Spray Operation Early is Spray, and no Loss of Injection Early.

DET CF	0.025
NO	0.975

CASE 8. Containment Steam Concentration Low
And Hydrogen Concentration Is Greater Than 20%

This occurs when the containment steam concentration is between 0-25% and the Fraction of Zirc Reacted is 33%.

DET CF	0.27
NO	0.73

CASE 9. Containment Steam Concentration Low
And Hydrogen Concentration Is 16-20%

This occurs when the containment [Steam] is between 25-35% and the Fraction of Zirc Reacted is 33%.

DET CF	0.16
NO DET CF	0.84

CASE 10. Otherwise, Should Never Reach This Case

DET CF	0.
NO	1.

Event Dependencies: Event Type (Event 3), RPV Injection Failure Time (Event 6), Offsite Power Recovery Time (Event 7), Mode of RHR Spray Operation Early (Event 18), Containment Steam Concentration (Event 19), Fraction of Zirconium Inventory Reacted In-Vessel (Event 20), Large H2 Burn Ignited During Core Damage (Event 22).

Quantification Basis: Grand Gulf (Brown 1990) APET Events 18 and 44, and Sequoyah Analysis (Volume 5 Rev 1 Part 1 Page 2.2, Part 2 Table A.3.1-1) page 15 and 16.

EVENT 24. CONTAINMENT FAILURE BEFORE RPV FAILURE

Two Branches:

FAILURE	Containment Failure Before RPV Failure
NO FAILURE	No Containment Failure Before RPV Failure

This event assigns a probability of 1 for failure to sequences with overpressure failure prior to core damage and to sequences with hydrogen detonation failure. For the hydrogen burns sequences remaining, this event compares the expected peak hydrogen burn pressure (Parameter 1 in APET event 22) with the Perry Containment Fragility Curve (Reference section 4.4.3) using a user function and returns the associated containment failure probability.

EVNTRE Question Type: 6. (Dependent Event using previously defined parameters and a User Function)

PROBABILITIES:

CASE 1. Containment Failed At Core Damage

FAILURE	1.
NO FAILURE	0.

CASE 2. Hydrogen Detonation Containment Failure

FAILURE	1.
NO FAILURE	0.

CASE 3. Otherwise, For Those Sequences With the Containment Intact At Core Damage - Determine the Probability of Containment Failure Due To The Remaining Large Hydrogen Burns

USER FUNCTION:		FAILURE	NO FAILURE
If Peak Containment Pressure	> 80 Psig	1.	0.
	> 75	0.98	0.02
	> 70	0.90	0.10
	> 65	0.69	0.31
	> 60	0.39	0.61
	> 55	0.15	0.85
	> 50	0.034	0.966
	Else	0.	1.

Event Dependencies: Containment Status At Core Damage (Event 2), and Hydrogen Detonation Containment Failure (Event 23).

Quantification Basis: Perry Nuclear Power Plant IPE Containment Capacity Analysis, and IPE Engineering Calculation - Containment

Fragility.

EVENT 25. MODE OF EARLY CONTAINMENT FAILURE BEFORE VESSEL FAILURE - V_ER_CF

Two Branches:

ANCHORAGE	Containment Anchorage Failure Mode
PEN-DOM/NO CF	Containment Penetration or Dome Failure Mode, Or No Containment Failure Mode

Given that the containment has failed as a result of a hydrogen burn, this event assigns a conditional probability for the mode of containment failure. The conditional probability that is assigned is dependent on the peak burn pressure (Parameter 1) and is evaluated in a user function. The user function assessed the conditional probability for each Perry failure mode based on the the Perry Containment Capacity (Reference section 4.4.3.2) for rapid containment loading conditions.

Note that this succinct sorting of failure modes into just two categories can be used to characterize penetration/dome containment failure with a question asking sequence of: 1) No Containment Failure, 2) Anchorage Containment Failure, and then 3) Containment Failure (which would provide the remaining Penetration/Dome containment failures).

EVNTRE Question Type: 6. (Dependent Event using previously defined parameter and a user function)

PROBABILITIES:

CASE 1. Containment Failed At Core Damage

This case is for gradual steam overpressure failure prior to the initiation of core damage.

ANCHORAGE	0.15
PEN-DOM/NO CF	0.85

CASE 2. Hydrogen Detonation Containment Failure

All detonation failures are considered to most likely occur in the dome region and are assigned to penetration/dome failure or no containment failure.

ANCHORAGE	0.
PEN-DOM/NO CF	1.

CASE 3. No Containment Failure

This sorting case assigns the no containment failure sequences.

ANCHORAGE	0.
PEN-DOM/NO CF	1.

CASE 4. Otherwise, The remaining sequences are Hydrogen Deflagration Containment Failure

USER FUNCTION		ANCHORAGE	PENETRATION
If Peak Containment Pressure	> 140 Psig	1.	0.
	> 130	0.98	0.02
	> 120	0.95	0.05
	> 115	0.90	0.10
	> 110	0.85	0.15
	> 105	0.78	0.22
	> 100	0.71	0.29
	> 95	0.61	0.39
	> 90	0.51	0.49
	> 85	0.41	0.59
	> 80	0.30	0.70
	> 75	0.21	0.79
	Else	0.15	0.85

Event Dependencies: Containment Status At Core Damage (Event 2), Hydrogen Detonation Containment Failure (Event 23), and Containment Failure Before RPV Failure (Event 24).

Quantification Basis: Perry Nuclear Power Plant IPE Containment Capacity Analysis, and IPE Engineering Calculation - Containment Failure Modes Conditional Probability.

H.3.4. INJECTION & SPRAY FAILURE DUE TO CONTAINMENT FAILURE BEFORE RPV FAILURE

This APET Group determines in Events 26 thru 29 the probability that containment failure causes the loss of all RPV injection (assuming the RPV injection has not previously failed) and the failure of RHR containment spray.

The two possible outcomes from this APET group are:

Injection and Spray Failure
No Failure

EVENT 26. CONTAINMENT FAILURE BEFORE RPV FAILURE IMPACT ON ECCS INJECTION & SPRAY PIPING - PIPE_FAIL

Two branches:

FAILURE	Failure of ECCS Injection & Spray Piping
NO FAILURE	No Piping Failure.

This event assesses the probability that either the dynamic forces or movement of the containment which occur during gross failure are sufficient to disrupt the injection and spray system piping when the containment is intact at core damage. Disruption of this piping is expected to be a serious threat for containment anchorage failure.

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. Anchorage Failure Mode Containment Failure With the Cntmt Intact At Core Damage Or With Critical ATWS

Those sequences with the containment intact at core damage and those associated with Critical ATWS and the containment not intact at core damage are evaluated for impact to associated systems when anchorage failure occurs before RPV Failure.

Note that the Plant Damage State event trees include this impact of containment failure in the Late Injection Fault Trees for SBO & LOOP and OTHERS when the containment has failed due to overpressure prior to core damage initiation. Therefore these sequences are excluded from this case.

FAILURE	0.9
NO FAILURE	0.1

CASE 2. Otherwise, For The Remaining Sequences Default to No Failure

FAILURE	0.
NO FAILURE	1.

Event Dependencies: Containment Status At Core Damage (Event 2), Event Type (Event 3), Mode of Containment Failure Before RPV Failure (Event 25).

Quantification Basis: Containment Capacity Analysis (Gilbert/Commonwealth 1992) and engineering judgement.

EVENT 27. CONTAINMENT FAILURE BEFORE RPV FAILURE IMPACT ON ECCS INJECTION & SPRAY MOTORS - MTR_FAIL

Two branches:

FAILURE	Failure of ECCS Injection & Spray Motors
NO FAILURE	No Motor Failure.

This event assesses the probability that leakage of water, steam or hot gases from containment into the Auxiliary Building which may occur at containment failure cause failure of the ECCS injection and spray system motors.

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. Containment Failure
With the Containment Intact At Core Damage
Or With Critical ATWS
And No Piping Failure (Modeled in the Above Event)

Those sequences with the containment intact at core damage and those associated with Critical ATWS and the containment not intact at core damage are evaluated for impact to associated systems during anchorage or penetration/dome containment failure before RPV failure. Those sequences which are modeled above in Event 26 with piping failure are not evaluated in this case, since piping failure is considered a single failure cutset.

Note that the Plant Damage State event trees include this impact of containment failure in the Late Injection Fault Trees for SBO & LOOP and OTHERS when the containment has failed due to overpressure prior to core damage initiation. Therefore these sequences are excluded from this case.

FAILURE	0.5
NO FAILURE	0.5

CASE 2. Containment Heat Removal With Vent
Or Containment Not Isolated During SBO

Venting within the Fuel Handling Building may overpressurize low pressure rated BISCO seals in the Shield Building penetrations (and other plant walls) and provide a steam and radiation release path to the in plant areas with firewater alternate injection lineup valves or may release to the environment through doorways and building seals to the diesel driven firewater pump oil tank and the backup fire truck engine.

FAILURE	0.05
NO FAILURE	0.95

CASE 3. Otherwise, For No Containment Failure And No Loss

Of Isolation - Default to No Failure.

FAILURE	0.
NO FAILURE	1.

Event Dependencies: Containment Status At Core Damage (Event 2), Event Type (Event 3), Containment Isolated (Event 5), Containment Heat Removal With Vent (Event 9), Containment Failure Before RPV Failure (Event 24), Containment Failure Before RPV Failure Impact On ECCS Injection and Spray Piping (Event 26).

Quantification Basis: Containment Capacity Analysis (Gilbert/Commonwealth 1992) and engineering judgement.

EVENT 28. CONTAINMENT FAILURE BEFORE RPV FAILURE
STEAM OR RADIATION RELEASE
IMPACT ON FIREWATER INJECTION - STM/RAD

Two branches:

FAILURE	Failure of Firewater Injection
NO FAILURE	No Failure

This event assesses the probability that leakage of steam or radionuclides from containment will limit personnel access to the firewater system and result in failure to perform required local manual actions to initiate, or to assure continued operation of the firewater alternate injection system.

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. Containment Failure
With the Containment Intact At Core Damag
Or With Critical ATWS
And No Piping Failure

Those sequences with the containment intact at core damage and those associated with Critical ATWS and the containment not intact at core damage are evaluated for impact to associated systems during anchorage or penetration/dome containment failure before RPV failure. Those sequences which are modeled above in Event 26 with piping failure are not evaluated in this case, since piping failure is considered a single failure cutset. Failure of the ECCS motor and firewater alternate injection will be considered a two failure cutset in the following summary event.

FAILURE 0.5

NO FAILURE 0.5

CASE 2. Containment Heat Removal With Vent Or Containment Not Isolated During SBO

FAILURE	0.1
NO FAILURE	0.9

CASE 3. Otherwise, For Other Sequences

FAILURE	0.
NO FAILURE	1.

Event Dependencies: Containment Status At Core Damage (Event 2), Event Type (Event 3), Containment Isolated (Event 5), Containment Heat Removal With Vent (Event 9), Containment Failure Before RPV Failure (Event 24), Containment Failure Before RPV Failure Impact On ECCS Injection and Spray Piping (Event 26).

Quantification Basis: Containment Capacity Analysis (Gilbert/Commonwealth 1992) and engineering judgement.

EVENT 29. INJECTION & SPRAY FAILURE DUE TO CONTAINMENT FAILURE BEFORE RPV FAILURE - INJ/SP_FL

Two branches:

INJ & SPRY FAIL	Failure of Injection & Spray
NO FAILURE	No Failure.

This event summarizes the results of the three APET events

EVNTRE Question Type: 2. (Dependent Sorting Event)

SORTING:

CASE 1. ECCS Injection & Spray Piping Failure

Single failure modeled in Event 26, Containment Failure Before RPV FAilure Impact on ECCS Injection & Spray Piping.

INJ & SPRY FAIL	1.
NO FAILURE	0.

CASE 2. ECCS Injection And Alternate Injection Failure

For the loss of all late injection include a two failure cutset modeled in Event 28, ECCS Injection & Spray Motor Failure; and in Event 29, Firewater Injection Failure.

INJ & SPRY FAIL	1.
NO FAILURE	0.

CASE 3. Otherwise, For Other Remaining Sequences Where All Injection Is Not Failed Default to No Failure.

INJ & SPRY FAIL	0.
NO FAILURE	1.

Event Dependencies: Previous three events (Event 26, 27, and 28).

Quantification Basis: Not Applicable.

H.3.5. DRYWELL FAILURE AT/NEAR RPV FAILURE

This APET Group in Events 38 thru 39 determines the probability of drywell failure at or near the time of RPV failure. A number of possible mechanisms have been identified which may lead to drywell failure at this time period. These include alpha mode steam explosion failures (which by definition fails the drywell and containment), an in-vessel steam explosion which fails the lower RPV head, an ex-vessel steam explosion in the pedestal cavity, drywell overpressure failure, and a large hydrogen burn in containment.

The two possible outcomes from this APET group are:

Drywell Failure
No Drywell Failure

EVENT 30. ALPHA MODE STEAM EXPLOSION DRYWELL AND CONTAINMENT FAILURE - ALPHA

Two branches:

ALPHA	Alpha Mode Failure
NO ALPHA	No Alpha Mode Failure

This event assesses the probability that an in-vessel steam explosion occurs with sufficient energy to rupture the RPV upper head and create a missile with sufficient energy to fail the drywell and containment. The probability of an in-vessel steam explosion has been shown to be dependent on whether the RPV has been depressurized.

The probabilities for alpha mode containment failure were taken directly from the Grand Gulf (Brown 1990) APET. These values are judged to be bounding values rather than best estimates.

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. RPV Depressurized During Core Damage

ALPHA	0.01
NO ALPHA	0.99

CASE 2. Otherwise, Sequences With RPV Not Depressurized

ALPHA	0.001
NO ALPHA	0.999

Event Dependencies: RPV Depressurized During Core Damage (Event 13).

Quantification Basis: Grand Gulf (Brown 1990) APET Event 58.

EVENT 31. MODE OF IN-VESSEL STEAM EXPLOSION BOTTOM HEAD RPV FAILURE - INV_EXPLN

Four Branches:

ALPHA	ALPHA Mode In-Vessel Steam Explosion
NO FAILURE	No RPV In-Vessel Steam Explosion Failure
LARGE VF	Large Bottom Head RPV Failure
SMALL VF	Small Lower Head RPV Failure

This event assesses the probability that an in-vessel steam explosion occurs with sufficient energy to rupture the lower head of the RPV. This event further differentiates between large (2 square meters) and small (0.1 square meter) vessel failure sizes. As noted previously the probability of an in-vessel steam explosion being triggered has been shown to be dependent on the primary system pressure.

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. Alpha Mode Failure

ALPHA	1.
NO FAILURE	0.
LARGE VF	0.
SMALL VF	0.

CASE 2. Sequences With RPV Depressurized During Core Damage

A review of the NUREG/CR-4551 Grand Gulf APET (Brown 1990), the Steam Explosion Review Report NUREG-1116 (SERG 1985), and BWR Lower Head Failure Assessment For CSNI Comparison Exercises (Rempe 1991) has convinced the Perry IPE analysts that the probabilities for steam explosion induced RPV bottom head failure were significantly overestimated in NUREG/CR-4551. Consequently the Grand Gulf 4551 failure estimates were reduced by a factor of 10.

ALPHA	0.
NO FAILURE	0.94
LARGE VF	0.034
SMALL VF	0.026

CASE 3. Otherwise, Remaining Sequences With No Alpha Mode Failure And RPV Not Depressurized

Reference the above discussion for case 2.

ALPHA	0.
NO FAILURE	0.993
LARGE VF	0.004
SMALL VF	0.003

Event Dependencies: RPV Depressurized During Core Damage (Event 13) and Alpha Mode Steam Explosion Drywell and Containment Failure (Event 30).

Quantification Basis: Grand Gulf (Brown 1990) APET Events 60 and 62, Steam Explosion Review Group (SERG 1985), and BWR Lower Head Failure Assessment For CSNI Comparison Exercises (Rempe 1991); and engineering judgement.

EVENT 32. RPV FAILURE MODE AND FAILURE SIZE - AREA_FAIL

Four Branches:

ALPHA	ALPHA Mode In-Vessel Steam Explosion
NO FAILURE	No RPV Failure - Debris Cooled In-Vessel
LARGE VF	Large Size Bottom Head RPV Failure
SMALL VF	Small Size Bottom Lower Head RPV Failure

This event characterizes the RPV failure mode and the vessel breach failure size. The Alpha mode failure, the large lower head failure by in-vessel steam explosion, and the small lower head failure by in-vessel steam explosion determined in the last APET event are similarly categorized here with regard to

RPV failure size. If the debris is not cooled in-vessel, and the RPV has not been previously failed by an Alpha mode failure or by a steam explosion induced lower head failure, this event then determines the probability of large (2 square meters) and small (0.1 square meter) vessel failure sizes due to thermal attack on the lower head.

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. Alpha Mode Failure

ALPHA	1.
NO FAILURE	0.
LARGE VF	0.
SMALL VF	0.

CASE 2. In-Vessel Steam Explosion Large Bottom Head RPV Failure

ALPHA	0.
NO FAILURE	0.
LARGE VF	1.
SMALL VF	0.

CASE 3. In-Vessel Steam Explosion Small Bottom Head RPV Failure

ALPHA	0.
NO FAILURE	0.
LARGE VF	0.
SMALL VF	1.

CASE 4. Core Debris Cooled In-Vessel And RPV Late Injection Available

ALPHA	0.
NO FAILURE	1.
LARGE VF	0.
SMALL VF	0.

CASE 5. Otherwise, Core Debris Causes RPV Melt Thru Failure Assign RPV Failure Size

ALPHA	0.
NO FAILURE	0.
LARGE VF	0.1
SMALL VF	0.9

Event Dependencies: Cooled In-Vessel (Event 15), Injection & Spray Failure Due to Containment Failure (Event 29), Alpha Mode Steam

Explosion Drywell and Containment Failure (Event 30), and Mode Of In-Vessel Steam Explosion Bottom Head RPV Failure (Event 31).

Quantification Basis: Grand Gulf (Brown 1990) APET Event 63; engineering judgement based on a recent study of BWR lower head failures (Rempe 1991).

Event 33. WATER IN PEDESTAL AT RPV FAILURE - PED_WATER

Four Branches:

FLOOD + INJ	Flooded From Wier Flow and Continuing Injection
RPV + INJ	Residual RPV Water and Continuing Injection
FLOOD	Flooded From Suppression Pool Wier Overflow
RPV WATER	Residual RPV Water Only

Water in the reactor pedestal cavity at the time of RPV failure can impact the accident progression in several ways. With water in the pedestal cavity there is an increased potential for steam explosions (or rapid steam generation) which may threaten the integrity of the pedestal. Water in the cavity early also enhances the possibility that the debris will be coolable.

The branch definitions are summarized above. A flooded pedestal cavity occurs as a result of pressurization of the containment wetwell (such as evaporation or boiling of the suppression pool, or by a hydrogen burn), depression of the pool level on the wetwell side of the suppression pool and overflow of the suppression pool into the drywell. Continuous injection to the pedestal cavity results from the addition of injection to the vessel following vessel breach. Residual RPV water refers to the water remaining in the RPV which is discharged from the RPV coincidentally (or following) expulsion of the core debris in the lower RPV head.

EVNTRE Question Type: 2. (Dependent Sorting Event)

SORTING:

CASE 1. Late In-Vessel Injection & Cavity Supply Available And Cavity Flooded Due To SBO With Long Term Injection Or Large Hydrogen Burn

FLOOD + INJ	1.
RPV + INJ	0.
FLOOD	0.
RPV WATER	0.

CASE 2. Late In-Vessel Injection & Cavity Supply Not Available And Cavity Flooded By A Large Burn During Core Damage, Or Cavity Flooded Due To SBO With Long Term Injection

FLOOD + INJ	0.
RPV + INJ	0.
FLOOD	1.
RPV WATER	0.

CASE 3. Late In-Vessel Injection & Cavity Supply Remains Available And Cavity Not Flooded

FLOOD + INJ	0.
RPV + INJ	1.
FLOOD	0.
RPV WATER	0.

CASE 4. Otherwi For Remaining Sequences Residual RPV Water Only Is Available

FLOOD + INJ	0.
RPV + INJ	0.
FLOOD	0.
RPV WATER	1.

Event Dependencies: Event Type (Event 3), RPV Injection Failure Time (Event 6), Late In-Vessel Injection Available (Event 12), Large H2 Burn During Core Damage (Event 22), Injection & Spray Failure Due to Containment Failure Before RPV Failure (Event 29).

Quantification Basis: With no uncontrolled hydrogen combustion phenomena, flooding of the cavity due to a heated suppression pool will only occur during a SBO with long term injection (past 11 hours) due to the differential pressure between containment and the drywell since the Drywell vacuum breaker isolation valves are closed. All other transients will have the Drywell Vacuum Breaker Isolation Valves open.

EVENT 34. PEDESTAL FAILURE DUE TO OVERPRESSURE AT RPV FAILURE - PED_OP

TWo Branches:

PED FAIL	Pedestal Failure Due to Overpressurization
NO FAIL	No Pedestal Failure

Given that the debris is not cooled in-vessel, and that the RPV has not been previously failed by an ALPHA mode failure, this event determines the probability of of overpressurization of the pedestal resulting from the depressurization of the RPV and the quenching of the debris.

MAAP calculations indicate that the pedestal integrity may be challenged from overpressure at the time of RPV failure under the following conditions:

1) High RPV pressure and a large RPV breach size (on the order of 2 square meters).

2) High RPV pressure and any RPV breach size with water in the pedestal cavity at RPV failure.

3) Low RPV pressure, a large RPV breach and Water in the pedestal cavity at RPV failure.

The peak pedestal pressures which were observed in the MAAP calculations are summarized in Table H.3.3-1.

In NUREG/CR-4551 it was estimated that the median (differential) failure pressure for the Grand Gulf Mark III pedestal (which is similar in design to Perry) would be 188 psi. Considering the results shown on Table H.3.3-1 and assuming 188 psi as the failure pressure for quasi-static loading, the following cases were evaluated in the Perry APET for pedestal overpressurization.

EVNTRE Question Type: 2. (Dependent Sorting Events)

SORTING:

CASE 1. No RPV Failure With Debris Cooled In-Vessel

PEDESTAL FAIL	0.
NO FAILURE	1.

CASE 2. RPV Not Depressurized At RPV Failure And Pedestal Cavity Flooded

PEDESTAL FAIL	1.
NO FAILURE	0.

CASE 3. RPV Not Depressurized At RPV Failure And Large RPV Failure Size

PEDESTAL FAIL	1.
NO FAILURE	0.

CASE 4. RPV Not Depressurized At RPV Failure, No Water in Pedestal Cavity From Flooding, And Small RPV Failure Size

PEDESTAL FAIL	0.
NO FAILURE	1.

CASE 5. RPV Depressurized At RPV Failure
Pedestal Cavity Flooded
And Large RPV Failure Size

PEDESTAL FAIL	1.
NO FAILURE	0.

CASE 6. Remaining Sequences Where RPV Is Depressurized

PEDESTAL FAIL	0.
NO FAILURE	1.

CASE 7. Otherwise, Should Never Reach This Case.

PEDESTAL FAIL	0.
NO FAILURE	1.

Event Dependencies: RPV Depressurized During Core Damage (Event 13), RPV Failure Mode And Failure Size (Event 32), Water in Pedestal At RPV Failure (Event 33), .

Quantification Basis: Grand Gulf (Brown 1990) APET Questions 71 and 74. IPE Engineering Calculation - MAAP Analysis of Pedestal Overpressure (reference TABLE H.3.3-1).

EVENT 35. PEDESTAL CAVITY EX-VESSEL STEAM EXPLOSION - STM_EXP

Two Branches:

STM EXPLOSION	Large Ex-vessel Steam Explosion
NO EXPLOSION	No Large Ex-vessel Steam Explosion

This event assesses the probability of a large steam explosion occurring in the pedestal cavity following vessel failure. For sequences where the debris is cooled in-vessel and lower head failure does not occur, then an ex-vessel steam explosion will not occur. For sequences where an in-vessel steam explosion has failed the vessel, it is assumed that a large ex-vessel steam explosion cannot occur.

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. No RPV Failure With Debris Cooled In-Vessel

STM EXPLOSION 0.

NO FAILURE 1.

CASE 2. No Water In Pedestal Cavity Before RPV Failure

STM EXPLOSION 0.
NO EXPLOSION 1.

CASE 3. In-Vessel Steam Explosion Already Failed RPV, Therefore Assume Large Ex-Vessel Steam Explosion Cannot Occur

STM EXPLOSION 0.
NO EXPLOSION 1.

CASE 4. Otherwise, For All Remaining Sequences With Water In Pedestal Cavity At RPV Failure A Large Ex-Vessel Steam Explosion May Occur

STM EXPLOSION 0.86
NO EXPLOSION 0.14

Event Dependencies: Mode of In-Vessel Steam Explosion Bottom Head RPV Failure (Event 31), RPV Failure Mode And Failure Size (Event 32), Water in Pedestal At RPV Failure (Event 33).

Quantification Basis: Grand Gulf (Brown 1990) APET Question 67.

EVENT 36. PEDESTAL FAILURE DUE TO STEAM EXPLOSION - PED_EXP

Two Branches:

PEDESTAL FAIL Pedestal Failure Due to Steam Explosion
NO FAILURE No Pedestal Failure Due To Steam Explosion

This event assesses the probability of a steam explosion failing the pedestal. Two failure mechanisms are considered. If an in-vessel steam explosion has caused a large breach in the lower reactor vessel head then it is considered possible that a large missile could be created (from part of the vessel lower head) which could cause failure in the pedestal wall. The second failure mechanism involves a steam explosion in the lower pedestal cavity which generates a shock wave which exceeds the impulse load capacity of the pedestal wall.

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. In-Vessel Steam Explosion
Caused Large Bottom Head Failure

PEDESTAL FAIL	0.05
NO FAILURE	0.95

CASE 2. No Ex-Vessel Steam Explosion

PEDESTAL FAIL	0.
NO FAILURE	1.

CASE 3. Otherwise, All Remaining Sequences Are
Ex-Vessel Steam Explosion

Grand Gulf NUREG/CR-4551 (Brown 1990) assumed a conditional probability of 0.5 for pedestal failure given a steam explosion in the lower pedestal cavity. Based on a review of NUREG/CR-4551 and NUREG-1116, this estimate appears exceptionally high. The Perry IPE reduced the Grand Gulf NUREG/CR-4551 failure estimate by a factor of 10 for the Perry APET best-estimate.

PEDESTAL FAIL	0.05
NO FAILURE	0.95

Event Dependencies: Mode of In-Vessel Steam Explosion Bottom Head RPV Failure (Event 31), and Pedestal Cavity Ex-Vessel Steam Explosion (Event 35).

Quantification Basis: Grand Gulf (Brown 1990) APET Question 75. The Steam Explosion Review Group Report (SERG 1985), NUREG-1116.

EVENT 37. DRYWELL FAILURE DUE TO PEDESTAL FAILURE - DW_PED

Two Branches:

DRYWELL FAIL	Drywell Failure Due To Pedestal Failure
NO FAILURE	No Drywell Failure Due To Pedestal Failure

Given that pedestal failure has occurred this event assesses the probability that pedestal failure causes loss of drywell integrity.

EVNTRF Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. Pedestal Failure From Other Than Alpha Mode Failure

This case includes pedestal failure due to overpressure at RPV failure and pedestal failure due to steam explosions.

DRYWELL FAIL	0.175
NO FAILURE	0.825

CASE 2. Otherwise, For the Remaining No Pedestal Failure Sequences, the Default is No Drywell Failure

DRYWELL FAIL	0.
NO FAILURE	1.

Event Dependencies: Pedestal Failure Due to Overpressure At RPV Failure (Event 34), and Pedestal Failure Due To Steam Explosion (Event 36).

Quantification Basis: Grand Gulf (Brown 1990) APET Question 76.

EVENT 38. DRYWELL OVERPRESSURE FAILURE AT RPV FAILURE - DW_OP

Two Branches:

DRYWELL FAIL	Drywell Overpressure Failure
NO FAILURE	No Drywell Overpressure Failure

This event assess the probability that drywell overpressure failure will occur following RPV failure. In order for drywell pressurization to challenge drywell integrity the RPV must be at high pressure at vessel failure and the vessel breach size must be large.

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. No RPV Failure With Debris Cooled In-Vessel

DRYWELL FAIL	0.
NO FAILURE	1.

CASE 2. RPV Not Depressurized At RPV Failure And Large RPV Failure Size

DRYWELL FAIL	0.01
NO FAILURE	0.99

CASE 3. Otherwise, the Remaining Sequences Default to No Drywell Failure

DRYWELL FAIL	0.
NO FAILURE	1.

EVENT DEPENDENCIES: RPV Depressurized During Core Damage (Event 13), RPV Failure Mode And Failure Size (Event 32).

QUANTIFICATION BASIS: Grand Gulf (Brown 1990) APET Questions 19 and 70. IPE Containment Capacity Analysis (Gilbert/Commonwealth 1992).

EVENT 39. DRYWELL FAILURE AT/NEAR RPV FAILURE - EARLY_DW

Two Branches:

DRYWELL FAIL	Drywell Failure
NO FAILURE	No Drywell Failure

This summary event assess the probability that drywell failure will occur following RPV failure. This event considers drywell failure resulting from alpha mode steam explosions, pedestal failure, and overpressure failure.

EVNTRE Question Type: 2. (Dependent Sorting Event)

SORTING:

CASE 1. No RPV Failure With Debris Cooled In-Vessel

DRYWELL FAIL	0.
NO FAILURE	1.

CASE 2. Drywell Failure Due To Alpha Failure, Pedestal Failure, Or Drywell Overpressurization Failure

DRYWELL FAIL	1.
NO FAILURE	0.

CASE 3. Otherwise, the Remaining Sequences Default to No Drywell Failure

DRYWELL FAIL	0.
NO FAILURE	1.

EVENT DEPENDENCIES: Alpha Mode Steam Explosion Drywell and Containment Failure (Event 30), RPV Failure Mode and RPV Failure

Size (Event 32), Drywell Failure Due To Pedestal (Event 37), and Drywell Overpressure Failure At RPV Failure (Event 38).

H.3.6. MOᴾ ᴼ ᴄONTAINMENT FAILURE AT/NEAR RPV FAILURE

This APET Group dete: mines with Events 40 thru 45 the probability of containment failure, and for containment fuilure, assigns the mode of containment failure at (or within a few hours of) reactor pressure vessel failure. Containment failure at RPV failure can potentially result from a combination of energetic processes and events which may occur at reactor vessel breach. These processes and events include hydrogen combustion and a large in-vessel steam explosion causing an alpha mode containment failure.

For the Perry APET, steam explosion induced (alpha mode) contain ent failures are also considered to result in a catastrophic rupture of the containment. Postulated alpha mode co tain ent failures result from large coherent in-vessel steam explosions which fails the reactor vessel and generates a missile (from part of the reactor vessel upper head) with sufficient mass and energy to fail (the drywell and) containment. There is a substantial body of evidence to suggest that in-vessel steam explosions do not represent a credible threat to early containment failure (i.e., the probability of early containment failure from in-vessel steam explosions is negligibly small). This opinion appears to be shared by the authors of Appendix 1 to Generic Letter 88-20. However, it this event should occur, it can result in a large and early environmental releases. Therefore, this event is included in the Perry IPE APET.

Experimental evidence and calculations have shown that steam explosions are unlikely at elevated pressure, subsequently the probab ity of an alpha mode containment failure should be significantly less for h pressure sequences than for low pressure sequences.

This event is dependent on the following Events in the Perry APET.

EVENT 40. CONTAINMENT STEAM CONCENTRATION AT/NEAR RPV FAILURE - ST_VB

Six Branches:

0-15 %	Containment Steam Volume Percent
15-25 %	
25-35 %	
35-45 %	
45-55 %	
> 55 %	

This branch assesses the containment steam concentration at or near RPV failure (from the time of RPV failure to 1 hour afterwards). The probability of hydrogen burn ignition and the efficiency of the burn (considered in subsequent events) are dependent upon the branch taken under this heading. The containment steam concentration is a function of the mode of spray operation, the sequence type, the time of injection failure and whether the containment is intact at core damage. The steam concentrations regime probabilities given for the cases below were estimated using MAAP results for each case.

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. Spray Loop Operation At Design Cooling

With a containment spray loop operating at design cooling with containment heat removal optimized with the Residual Heat Removal heat exchanger, the steam concentration would be low.

0-15 %	1.
15-25 %	0.
25-35 %	0.
35-45 %	0.
45-55 %	0.
>55 %	0.

CASE 2. SBO And No Injection Failure Early (0- 2.8 Hrs)

0-15 %	1.
15-25 %	0.
25-35 %	0.
35-45 %	0.
45-55 %	0.
>55 %	0.

CASE 3. SBO And RCIC Injection Failure (2.8-4.2 Hrs)

0-15 %	0.
15-25 %	0.44
25-35 %	0.56
35-45 %	0.
45-55 %	0.
>55 %	0.

CASE 4. SBO And HPCS Injection Failure (> 4.2 Hrs)

0-15 %	0.
15-25 %	0.

25-35 %	0.
35-45 %	0.
45-55 %	0.
>55 %	1.

CASE 5. Containment Failed At Core Damage

0-15 %	0.
15-25 %	0.
25-35 %	0.
35-45 %	0.
45-55 %	0.
>55 %	1.

CASE 6. Otherwise, All Other Sequences
Default to Low Steam Concentration

All other sequences are conservatively evaluated by assuming a low steam concentration.

0-15 %	1.
15-25 %	0.
25-35 %	0.
35-45 %	0.
45-55 %	0.
>55 %	0.

Event Dependencies: Containment Status At Core Damage (Event 2), Event Type (Event 3), RPV Injection Failure Time (Event 6), and Mode of RHR Spray Operation Early (Event 18).

Quantification Basis: IPE Engineering Calculation - Hydrogen Burns [with MAAP 3.0B Rev 7.02 using "Recommended Sensitivity Analyses For An IPE Using MAAP 3.0B" (EPRI 1991)].

EVENT 41. FRACTION ZIRCONIUM INVENTORY REACTED AT/NEAR RPV FAILURE - H2_VB

Three Branches:

33 %	Core Inventory Zirconium Oxidized
22 %	
11 %	

The fraction of zirconium reacted in-vessel and during RPV failure blowdown is used to determine the concentration of hydrogen in containment at/near RPV failure (from the time of RPV failure to an hour afterwards). Three discrete regimes are used to represent the range of zirconium oxidation. These regimes are representative of the amounts estimated in the Grand Gulf Analysis (Brown

1990) and for Perry specific MAAP calculations. The Perry IPE MAAP calculations implemented the best-estimate values for the model parameters discussed in the "Recommended Sensitivity Analyses For An Individual Plant Examination Using MAAP 3.0B" (EPRI 1991). The IDCOR BWR "blockage" model of Local Blockage option was considered more likely and the No Blockage option less likely, and a probability of 0.8 and 0.2 was assigned to each option, respectively. These results were used to define the split fractions shown below.

EVNTRE Question: Type 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. SBO And No Injection Failure Early (0 - 2.8 Hrs)

11 %	1.
22 %	0.
33 %	0.

CASE 2. SBO And RCIC Injection Failure (2.8 - 4.2 Hrs)

11 %	0.69
22 %	0.31
33 %	0.

CASE 3. SBO And HPCS Injection Failure (> 4.2 Hrs)

11 %	0.79
22 %	0.21
33 %	0.

CASE 4. Otherwise, All Other Remaining Sequences Conservatively Default to SBO And RCIC Injection Failure

11 %	0.69
22 %	0.31
33 %	0.

Event Dependencies: Event Type (Event 3), and RPV Injection Failure Time (Event 6).

Quantification Basis: IPE Engineering Calculation - Hydrogen Burns [with MAAP 3.0B Rev 7.02 using "Recommended Sensitivity Analyses For An IPE Using MAAP 3.0B" (EPRI 1991)].

For comparison reference Grand Gulf (Brown 1990) APET Event 35.

EVENT 42. HYDROGEN IGNITION SOURCES AVAILABLE AT/BEFORE RPV FAILURE - IG_SOURC

Two Branches:

NO IGN SOURCE No Hydrogen Ignition Sources
IGNITN SOURCE Hydrogen Ignition Sources Available

With a continuous ignition source available in containment (at the time of RPV failure) it is assumed that controlled hydrogen combustion (or small hydrogen burns) will preclude the build-up of hydrogen concentrations whose combustion would threaten containment integrity. A continuous ignition source is assumed to be available if the hydrogen ignition system is operating or if a burn in containment has already occurred during core damage. In the latter situation it is assumed that the prior burn which has occurred will result in ignition of combustible materials in containment which will act as ignition sources. This event also includes a recovery of the human interaction, Operator fails to initiate Hydrogen Ignition System.

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. Hydrogen Ignition System Available Before RPV Failure

NO IGN SOURCE	0.
IGNITN SOURCE	1.

CASE 2. No Loss Of AC Power,
Operator Failure To Initiate Hydrogen Ignition System,
And Recovery of Hydrogen Ignitors

When AC power is never lost, a nominal screening value is assigned to the recovery of the human interaction, Operator Fails To Initiate Hydrogen Ignition System, included in Event 16, Hydrogen Ignition System Available. For the loss of all injection sequence the expected containment hydrogen concentration at vessel failure would be < 4% (which is the SAFE region of the PEI Hydrogen Deflagration Overpressure Limit) and the recovery of Hydrogen Ignitors at RPV failure (1.8 hours into the sequence and 1.3 hours after the Level 1 cue) is very likely. For a sequence like loss of RCIC at 2.8 hours, when hydrogen generation commences a rapid increase in hydrogen concentration to above the Hydrogen Deflagration Overpressure Limit may occur in a half hour interval. However, recovery of Hydrogen Ignitors for a loss of RCIC transient with AC power is supported by the relatively long period of time from the Level 1 cue at about 4 hours to 2200 F maximum fuel clad temperature when hydrogen generation commences actively at 1.5 hours.

Since time windows greater than an hour generally exist for Perry IPE sequences for recovery of Hydrogen Ignitors and hydrogen analyzers are available for hydrogen concentration information, a screening value human error probability of 0.1 is conservatively assigned for this case.

NO IGN SOURCE	0.1
IGNITN SOURCE	0.9

CASE 3. Hydrogen Burns Before RPV Failure

Both large and small hydrogen burns are included.

NO IGN SOURCE	0.
IGNITN SOURCE	1.

CASE 4. Otherwise, For Remaining Sequences
No Continuous Ignition Source Is Available

NO IGN SOURCE	1.
IGNITN SOURCE	0.

EVENT DEPENDENCIES: Event Type (Event 3), Hydrogen Ignition System Available (Event 16), Small H2 Burns At Low H2 Concentrations (Event 21), and Large H2 Burn During Core Damage (Event 22).

QUANTIFICATION BASIS: IPE Human Interaction Technical Assignment File, IPE Engineering Calculation - MAAP Accident Progression Analyses, Perry Plant Emergency Instruction, and Discussion with Operation and Chemistry Staff.

Event 43. HIGH PRESSURE MELT EJECTION AT RPV FAILURE

Two Branches:

HPME	High Pressure Melt Ejection
NO HPME	No High Pressure Melt Ejection

This event assesses whether a high pressure melt ejection occurs from the reactor pedestal cavity following vessel failure. For HPME to occur the RPV pressure must be elevated (above several hundred psi) at the time of vessel failure. HPME involves the entrainment and fragmentation of the debris in the pedestal cavity and transport of the debris throughout the drywell. If the drywell has failed, then an HPME event can provide an ignition source for hydrogen in the containment.

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. No RPV Failure With Debris Cooled In-Vessel

No HPME occur when the debris is cooled in-vessel with no subsequent RPV injection failure.

HPME	0.
NO HPME	1.

CASE 2. RPV Depressurized During Core Damage

HPME	0.
NO HPME	1.

CASE 3. Otherwise, For Remaining Sequences
RPV Failure At High Pressure
And HPME May Occur

HPME	0.8
NO HPME	0.2

Event Dependencies: RPV Depressurized During Core Damage (Event 13), RPV Failure Mode and Failure Size (Event 32).

Quantification Basis: Grand Gulf (Brown 1990) APET Event 64.

Event 44. LARGE HYDROGEN BURN IGNITED AFTER RPV FAILURE - LG_BRN

Two Branches:

NO BURN IGN	No Large Hydrogen Burn
LG BURN IGN	Large Hydrogen Burn Ignited

This event assess whether a large hydrogen burn is ignited in containment following RPV failure. If a continuous ignition source is available (as defined in event 42) then a large hydrogen burn is assumed to be prevented. Also if the steam concentration is above 55% then the containment atmosphere is inert to hydrogen burns. The probability that a hydrogen burn is ignited is a function of the following parameters:

1. Containment Steam Concentration,
2. Containment Hydrogen Concentration
   (or the fraction of zirconium reacted)
3. Drywell Failure,
4. High Pressure Melt Ejection,
5. AC Power Recovery.

Forty eight cases (various combinations of the parameters listed above are identified to perform the quantification). In addition to assessing the probability of ignition this event also sets two parameter values which are used by subsequent events; peak hydrogen burn pressure (parameter 2), and containment base pressure before the burn (parameter 4). Table H.3.5-1 shows the results of the hydrogen pressure calculation and the input for each case where a large hydrogen burn was predicted to occur, and probability assigned for ignition of a large burn.

EVNTRE Question Type: 4. (Dependent Split Fraction with parameter values set for each case.)

PROBABILITIES:

CASE 1. Hydrogen Ignition Sources Available Early At RPV Failure

NO BURN IGN	1.				
LG BURN IGN	0.	BASE PRESS	0	PEAK PRESS	0

CASE 2. Containment Steam Concentration Is Greater Than 55%

NO BURN IGN	1.				
LG BURN IGN	0.	BASE PRESS	0	PEAK PRESS	0

CASE 3. Containment Steam [0-15%], 33% Zirc Reacted, And Drywell Failure Or AC Power Recovery. H2 = [21.7%]

NO BURN IGN	0.				
LG BURN IGN	1.	BASE PRESS	13	PEAK PRESS	147

CASE 4. Containment Steam [0-15%], 33% Zirc Reacted, And High Pressure Melt Ejection. H2 = [21.7%]

NO BURN IGN	0.37				
LG BURN IGN	0.63	BASE PRESS	17	PEAK PRESS	151

CASE 5. Containment Steam [0-15%], 33% Zirc Reacted, And No DW Failure, or No AC Power, or No HPME. H2 = [21.7%]

NO BURN IGN	0.51				
LG BURN IGN	0.49	BASE PRESS	13	PEAK PRESS	147

CASE 6. Containment Steam [15-25%], 33% Zirc Reacted, And Drywell Failure Or AC Power Recovery. H2 = [18.8%]

NO BURN IGN	0.				
LG BURN IGN	1.	BASE PRESS	20	PEAK PRESS	131

CASE 7. Containment Steam [15-25%], 33% Zirc Reacted, And High Pressure Melt Ejection. H2 = [18.8%]

NO BURN IGN 0.37
LG BURN IGN 0.63 BASE PRESS 24 PEAK PRESS 135

CASE 8. Containment Steam [15-25%], 33% Zirc Reacted, And No DW Failure or No AC Power or No HPME. H2 = [18.8%]

NO BURN IGN 0.51
LG BURN IGN 0.49 BASE PRESS 20 PEAK PRESS 131

CASE 9. Containment Steam [25-35%], 33% Zirc Reacted, And Drywell Failure Or AC Power Recovery. H2 = [16.4%]

NO BURN IGN 0.
LG BURN IGN 1. BASE PRESS 26 PEAK PRESS 120

CASE 10. Containment Steam [25-35%], 33% Zirc Reacted, And High Pressure Melt Ejection. H2 = [16.4%]

NO BURN IGN 0.37
LG BURN IGN 0.63 BASE PRESS 30 PEAK PRESS 124

CASE 11. Containment Steam [25-35%], 33% Zirc Reacted, And No DW Failure or No AC Power or No HPME. H2 = [16.4%]

NO BURN IGN 0.51
LG BURN IGN 0.49 BASE PRESS 26 PEAK PRESS 120

CASE 12. Containment Steam [35-45%], 33% Zirc Reacted, And Drywell Failure Or AC Power Recovery. H2 = [14.1%]

NO BURN IGN 0.
LG BURN IGN 1. BASE PRESS 35 PEAK PRESS 113

CASE 13. Containment Steam [35-45%], 33% Zirc Reacted, And High Pressure Melt Ejection. H2 = [14.1%]

NO BURN IGN 0.44
LG BURN IGN 0.56 BASE PRESS 39 PEAK PRESS 117

CASE 14. Containment Steam [35-45%], 33% Zirc Reacted, And No DW Failure or No AC Power or No HPME. H2 = [14.1%]

NO BURN IGN .62

LG BURN IGN .38 BASE PRESS 35 PEAK PRESS 113

CASE 15. Containment Steam [45-55%], 33% Zirc Reacted, And Drywell Failure Or AC Power Recovery. H2 = [11.7%]

NO BURN IGN 0.
LG BURN IGN 1. BASE PRESS 46 PEAK PRESS 109

CASE 16. Containment Steam [45-55%], 33% Zirc Reacted, And High Pressure Melt Ejection. H2 = [11.7%]

NO BURN IGN 0.57
LG BURN IGN 0.43 BASE PRESS 50 PEAK PRESS 113

CASE 17. Containment Steam [45-55%], 33% Zirc Reacted, And No DW Failure or No AC Power or No HPME. H2 = [11.7%]

NO BURN IGN 0.72
LG BURN IGN 0.28 BASE PRESS 46 PEAK PRESS 109

CASE 18. Containment Steam [0-15%], 22% Zirc Reacted, And Drywell Failure Or AC Power Recovery. H2 = [15.7%]

NO BURN IGN 0.
LG BURN IGN 1. BASE PRESS 11 PEAK PRESS 104

CASE 19. Containment Steam [0-15%], 22% Zirc Reacted, And High Pressure Melt Ejection. H2 = [15.7%]

NO BURN IGN 0.44
LG BURN IGN 0.56 BASE PRESS 15 PEAK PRESS 108

CASE 20. Containment Steam [0-15%], 22% Zirc Reacted, And No DW Failure or No AC Power or No HPME. H2 = [15.7%]

NO BURN IGN 0.62
LG BURN IGN 0.38 BASE PRESS 11 PEAK PRESS 104

CASE 21. Containment Steam [15-25%], 22% Zirc Reacted And Drywell Failure Or AC Power Recovery. H2 = [13.6%]

NO BURN IGN 0.
LG BURN IGN 1. BASE PRESS 17 PEAK PRESS 94.3

CASE 22. Containment Steam [15-25%], 22% Zirc Reacted And High Pressure Melt Ejection. H2 = [13.6%]

NO BURN IGN   0.44
LG BURN IGN   0.56   BASE PRESS 21   PEAK PRESS   98.3

CASE 23.   Containment Steam [15-25%], 22% Zirc Reacted, And
No DW Failure or No AC Power or No HPME. H2 = [13.6%]

NO BURN IGN   0.62
LG BURN IGN   0.38   BASE PRESS 17   PEAK PRESS   94.3

CASE 24.   Containment Steam [25-35%], 22% Zirc Reacted, And
Drywell Failure Or AC Power Recovery.   H2 = [11.9%]

NO BURN IGN   0.
LG BURN IGN   1.   BASE PRESS 23   PEAK PRESS   83.3

CASE 25.   Containment Steam [25-35%], 22% Zirc Reacted, And
High Pressure Melt Ejection.   H2 = [11.9%]

NO BURN IGN   0.57
LG BURN IGN   0.43   BASE PRESS 27   PEAK PRESS   87.3

CASE 26.   Containment Steam [25-35%], 22% Zirc Reacted, And
No DW Failure or No AC Power or No HPME. H2 = [11.9%]

NO BURN IGN   0.72
LG BURN IGN   0.28   BASE PRESS 23   PEAK PRESS   83.3

CASE 27.   Containment Steam [35-45%], 22% Zirc Reacted, And
Drywell Failure Or AC Power Recovery.   H2 = [10.2%]

NO BURN IGN   0.
LG BURN IGN   1.   BASE PRESS 31   PEAK PRESS   80.8

CASE 28.   Containment Steam [35-45%], 22% Zirc Reacted, And
High Pressure Melt Ejection.   H2 = [10.2%]

NO BURN IGN   0.57
LG BURN IGN   0.43   BASE PRESS 35   PEAK PRESS   84.8

CASE 29.   Containment Steam [35-45%], 22% Zirc Reacted, And
No DW Failure or No AC Power or No HPME. H2 = [10.2%]

NO BURN IGN   0.72
LG BURN IGN   0.28   BASE PRESS 31   PEAK PRESS   80.4

CASE 30. Containment Steam [45-55%], 22% Zirc Reacted, And Drywell Failure Or AC Power Recovery. H2 = [8.5%]

NO BURN IGN 0.
LG BURN IGN 1. BASE PRESS 41 PEAK PRESS 80.8

CASE 31. Containment Steam [45-55%], 22% Zirc Reacted, And High Pressure Melt Ejection. H2 = [8.5%]

NO BURN IGN 0.57
LG BURN IGN 0.43 BASE PRESS 45 PEAK PRESS 84.8

CASE 32. Containment Steam [45-55%], 22% Zirc Reacted, And No DW Failure or No AC Power or No HPME. H2 = [8.5%]

NO BURN IGN 0.72
LG BURN IGN 0.28 BASE PRESS 41 PEAK PRESS 80.8

CASE 33. Containment Steam [0-15%], 11% Zirc Reacted, And Drywell Failure Or AC Power Recovery. H2 = [8.6%]

NO BURN IGN 0.
LG BURN IGN 1. BASE PRESS 8.4 PEAK PRESS 38.2

CASE 34. Containment Steam [0-15%], 11% Zirc Reacted, And High Pressure Melt Ejection. H2 = [8.6%]

NO BURN IGN 0.57
LG BURN IGN 0.43 BASE PRESS 12.4 PEAK PRESS 42.2

CASE 35. Containment Steam [0-15%], 11% Zirc Reacted, And No DW Failure or No AC Power or No HPME. H2 = [8.6%]

NO BURN IGN 0.72
LG BURN IGN 0.28 BASE PRESS 8.4 PEAK PRESS 38.2

CASE 36. Containment Steam [15-25%], 11% Zirc Reacted, And Drywell Failure Or AC Power Recovery. H2 = [7.4%]

NO BURN IGN 0.
LG BURN IGN 1. BASE PRESS 14 PEAK PRESS 36.2

CASE 37. Containment Steam [15-25%], 11% Zirc Reacted, And High Pressure Melt Ejection. H2 = [7.4%]

NO BURN IGN 0.71
LG BURN IGN 0.39 BASE PRESS 18 PEAK PRESS 40.2

CASE 38. Containment Steam [15-25%], 11% Zirc Reacted, And No DW Failure or No AC Power or No HPME. H2 = [7.4%]

NO BURN IGN 0.79
LG BURN IGN 0.21 BASE PRESS 14 PEAK PRESS 36.2

CASE 39. Containment Steam [25-35%], 11% Zirc Reacted, And Drywell Failure Or AC Power Recovery. H2 = [6.5%]

NO BURN IGN 0.
LG BURN IGN 1. BASE PRESS 19 PEAK PRESS 37.3

CASE 40. Containment Steam [25-35%], 11% Zirc Reacted, And High Pressure Melt Ejection. H2 = [6.5%]

NO BURN IGN 0.71
LG BURN IGN 0.29 BASE PRESS 23 PEAK PRESS 41.3

CASE 41. Containment Steam [25-35%], 11% Zirc Reacted, And No DW Failure or No AC Power or No HPME. H2 = [6.5%]

NO BURN IGN 0.79
LG BURN IGN 0.21 BASE PRESS 19 PEAK PRESS 37.3

CASE 42. Containment Steam [35-45%], 11% Zirc Reacted, And Drywell Failure Or AC Power Recovery. H2 = [5.6%]

NO BURN IGN 0.
LG BURN IGN 1. BASE PRESS 27 PEAK PRESS 40.8

CASE 43. Containment Steam [35-45%], 11% Zirc Reacted, And High Pressure Melt Ejection. H2 = [5.6%]

NO BURN IGN 0.71
LG BURN IGN 0.29 BASE PRESS 31 PEAK PRESS 44.8

CASE 44. Containment Steam [35-45%], 11% Zirc Reacted, And No DW Failure or No AC Power or No HPME. H2 = [5.6%]

NO BURN IGN 0.79
LG BURN IGN 0.21 BASE PRESS 27 PEAK PRESS 40.8

CASE 45. Containment Steam [45-55%], 11% Zirc Reacted, And Drywell Failure Or AC Power Recovery. H2 = [4.6%]

NO BURN IGN 0.
LG BURN IGN 1. BASE PRESS 36 PEAK PRESS 46.1

CASE 46. Containment Steam [45-55%], 11% Zirc Reacted, And High Pressure Melt Ejection. H2 = [4.6%]

NO BURN IGN 0.71
LG BURN IGN 0.29 BASE PRESS 40 PEAK PRESS 50.1

CASE 47. Containment Steam [45-55%], 11% Zirc Reacted, And No DW Failure or No AC Power or No HPME. H2 = [4.6%]

NO BURN IGN 0.79
LG BURN IGN 0.21 BASE PRESS 36 PEAK PRESS 46.1

CASE 48. Otherwise, Should Never Reach This Case.

NO BURN IGN 1.
LG BURN IGN 0. BASE PRESS 0 PEAK PRESS 0

Event Dependencies: Offsite Power Recovery Time (Event 7), Drywell Failure At/Near RPV Failure (Event 39), Containment Steam Concentration At/Near RPV Failure (Event 40), Fraction Zirconium Reacted At/Near RPV Failure (Event 41), Hydrogen Ignition Sources Available At RPV Failure (Event 42), and High Pressure Melt Ejection (Event 43).

Quantification Basis: IPE Engineering Calculation - Hydrogen Burns [with MAAP 3.0B Rev 7.02 using "Recommended Sensitivity Analyses For An IPE Using MAAP 3.0B" (EPRI 1991)]. Grand Gulf (Brown 1990) APET Events 84 and 85. (see Table H.3.5-1.)

Event 45. H2 DETONATION CONTAINMENT FAILURE AT/NEAR RPV FAILURE - H2_DET

Two Branches:

DET CF	H2 Detonation Containment Failure At/Near RPV Failure
NO	No H2 Detonation Containment Failure

Given that a large burn was ignited in containment following RPV failure this event assesses whether the burn transitions to a detonation and whether containment failure results from the detonation impulse loading. If no large burn was ignited or if the containment atmosphere is inert to detonations (i.e., > 35% steam concentration), then no detonation is assumed to occur.

The probability of a hydrogen detonation occurring is taken to be a function of the steam concentration, the hydrogen concentration, whether power recovery occurs, and whether sprays are initiated during this time period.

EVNTRE Question Type: 2. (Dependent Split Fr .ions)

PROBABILITIES:

CASE 1. No Large Burn At/Near RPV Failure

With no large burn, no trigger exists for a hydrogen detonation.

DET CF	0.
NO	1.

CASE 2. Cont.inment Steam Concentration Greater Than 35%

Steam concentrations greater than 35% prevent hydrogen detonation.

DET CF	0.
NO	1.

CASE 3. Containment Steam Concentration High And Slowly Decreasing Less Than 35%, And Hydrogen Concentration Greater Than 20%

This occurs when offsite power is recovered before RPV Failure and sprays are available for those SBO sequences where high steam concentrations are possible (i.e., not early failure of all injection). This range of hydrogen concentration can be associated with 33% fraction of zirconium inventory reacted at/near RPV failure.

DET CF	0.025
NO	0.975

CASE 4. Containment Steam Concentration Low And Hydrogen Concentration Greater Than 20%

This occurs when the steam concentration is 0-15% and the fraction of zirconium reacted is 33%.

DET CF	0.27
NO	0.73

CASE 5. Containment Steam Concentration Low And Hydrogen Concentration Greater Than 20%

This occurs when the steam concentration is 15-35% and the fraction of zirconium reacted is 33%.

DET CF	0.16
NO	0.84

CASE 6. Hydrogen Concentration Less Than 12%

This occurs when the fraction of zirconium reacted is 11%.

DET CF	0.
NO	1.

CASE 7. Containment Steam Concentration High And Slowly Decreasing Less Than 35%, And Hydrogen Concentration 12-16%

This occurs when offsite power is recovered before RPV Failure and sprays are available for those SBO sequences where high steam concentrations are possible (i.e., not early failure of all injection). This range of hydrogen concentration can be associated with 22% fraction of zirconium inventory reacted at/near RPV failure.

DET CF	0.022
NO	0.978

CASE 8. Containment Steam Concentration Low And Hydrogen Concentration 12-16%

This occurs when the steam concentration is 15-35% and the fraction of zirconium reacted is 22%.

DET CF	0.
NO	1.

CASE 9. Containment Steam Concentration Low And Hydrogen Concentration 16-20%

This occurs when the steam concentration is 0-15% and the fraction of zirconium reacted is 22%.

DET CF	0.16
NO	0.84

CASE 10. Otherwise, Should Never Reach This Case.

DET CF	0
NO	1

Event Dependencies: Event Type (Event 3), RPV Injection Failure Time (Event 6), Offsite Power Recovery Time (Event 7), Mode of RHR Spray Operation Early (Event 18), Containment Steam Concentration At/Near RPV Failure (Event 40), Fraction

	Zirconium Inventory Reacted At/Near RPV Failure (Event 41), Large H2 Burn After RPV Failure (Event 44).
Quantification Basis:	Grand Gulf (Brown 1990) APET Events 18, 44 and 86; and Sequoyah Analysis (Volume 5 Rev 1 Part 1 Page 2.2, Part 2 Table A.3.1-1) page 15 and 16. Reference Table H.3.5-2.

EVENT 46. CONTAINMENT FAILURE AT/NEAR RPV FAILURE

Two Branches:

FAILURE	Containment Failure
NO FAILURE	No Containment Failure

This event assesses whether containment failure occurs following RPV failure as a result of an alpha mode steam explosion or a hydrogen burn. The alpha mode in-vessel steam explosion has sufficient energy to rupture the upper head of the RPV and create a missile with sufficient energy to fail the drywell and containment. If a hydrogen detonation has occurred which fails the containment then containment failure has occurred. If a large hydrogen burn was ignited then this event compares the peak containment pressure for the burn (Parameter 2) with the containment fragility curve and determines the probability of failing the containment.

EVNTRE Question Type: 6. (Dependent event using previously defined parameters and a user function)

PROBABILITIES:

CASE 1. Alpha Mode Steam Explosion Containment Failure

FAILURE	1.
NO FAILURE	0.

CASE 2. Hydrogen Detonation Containment Failure At/Near RPV Failure

FAILURE	1.
NO FAILURE	0.

CASE 3. Otherwise, For Those Remaining Sequences Determine the Probability of Containment Failure Due To Large Hydrogen Burns

USER FUNCTION		FAILURE	NO FAILURE
If Peak Containment Pressure	> 80 Psig	1.	0.
	> 75	0.98	0.02
	> 70	0.90	0.10

> 65	0.69	0.31
> 60	0.39	0.59
> 55	0.15	0.85
> 50	0.034	0.966
Else	0.	1.

Event Dependencies: Alpha Mode Steam Explosion Drywell and Containment Failure (Event 30), Hydrogen Detonation Containment Failure At/Near RPV Failure (Event 45).

Quantification Basis: Perry Nuclear Power Plant IPE Containment Capacity Analysis, and IPE Engineering Calculation - Containment Fragility.

EVENT 47. MODE OF CONTAINMENT FAILURE AT/NEAR RPV FAILURE - CF_VB

Two Branches:

ANCHORAGE	Containment Anchorage Failure Mode
PEN-DOM/NO CF	Containment Penetration or Dome Failure Mode, Or No Containment Failure Mode

Given that containment failure has occurred then the probability of various modes of containment failure are determined by this event. For sequences where the containment has failed by a hydrogen burn the peak containment pressure from the burn is used to estimate the probability of each failure mode. The individual fragility curves for the dominant failure modes were used to determine the conditional probabilities for each failure mode as a function of the failure pressure. The peak burn pressure is then used to determine the probability of each failure mode.

Note that this succinct sorting of failure modes into just two categories can be used to characterize penetration/dome containment failure with a question asking sequence of: 1) No Containment Failure, 2) Anchorage Containment Failure, and then 3) Containment Failure (which would provide the remaining Penetration/Dome containment failures).

EVNTRE Question Type: 6. (Dependent event using previously defined parameters and a user function)

PROBABILITIES:

CASE 1. No Containment Failure At/Near RPV Failure

This sorting case assigns the no containment failure sequences.

ANCHORAGE	0.
PEN-DOM/NO CF	1.

CASE 2. Alpha Mode Containment Failure

And Detonation Containment Failure At/Near RPV Failure

The Alpha mode and detonation failures are considered most likely to occur in the dome region and are assigned to penetration/dome failure or no containment failure.

ANCHORAGE	0.
PEN-DOM/NO CF	1.

CASE 3 Otherwise, The Remaining Sequences Are Hydrogen Deflagration Containment Failure

..E FUNCTION		ANCHORAGE	PENETRATION
If Peak Containment Pressure	> 140 Psig	1.	0.
	> 130	0.98	0.02
	> 120	0.95	0.05
	> 115	0.90	0.10
	> 110	0.85	0.15
	> 105	0.78	0.22
	> 100	0.71	0.29
	> 95	0.61	0.39
	> 90	0.51	0.49
	> 85	0.41	0.59
	> 80	0.30	0.70
	> 75	0.21	0.79
	Else	0.15	0.85

Event Dependencies: Alpha Mode Steam Explosion Drywell and Containment Failure (Event 30), Hydrogen Detonation Containment Failure At/Near RPV Failure (Event 45), and Containment Failure At/Near RPV Failure (Event 46).

Quantification Basis: Perry Nuclear Power Plant IPE Containment Capacity Analysis, and IPE Engineering Calculation - Containment Failure Modes Conditional Probability.

H.3.7. POOL BYPASS BEFORE/NEAR RPV FAILURE

This APET Group of Events 48 and 49 determines the probability of pool bypass early, before or near the time of RPV failure. Fission product scrubbing in the suppression pool is an effective fission product mitigation mechanism. However, if the release pathway bypasses the suppression pool this mechanism is not effective. Pool bypass may result from a number of causes. These include: 1) structural failure of the drywell, 2) Drywell vacuum breaker failure, 3) loss of suppression pool water below the level of the horizontal vents or the SRV quenchers, and 4) other failure processes.

Drywell structural failure may result from transient over-pressurization of the drywell or wetwell resulting in a sufficiently high drywell/wetwell differential pressure to cause failure of the drywell head, ceiling or walls. Failure of Drywell vacuum breakers may occur during large hydrogen burns. Loss of suppression pool water may result from containment anchorage failure in the pool region.

Event 48. DRYWELL FAILURE DUE TO CONTAINMENT HYDROGEN BURN BEFORE/NEAR RPV FAILURE - H2_BURN

Two Branches:

DW FAILURE	Drywell Failure Due To Hydrogen Burn
NO DW FAILURE	No Drywell Failure Due to Hydrogen Burn

Given that a large hydrogen burn has occurred during core damage or at RPV failure this event assesses whether drywell failure results from excessive differential pressure across the drywell boundary. If a large burn has not occurred in containment then drywell integrity is not challenged (by hydrogen combustion). For cases where a large hydrogen burn has occurred two parameters have been set which give the peak burn pressure in containment during the burn and the containment pressure prior to the burn. It is assumed that the drywell pressure remains constant during the burn in containment. Given the value of these two parameters the peak drywell differential pressure is calculated and compared against the drywell fragility curve in a user function to estimate the probability of drywell structural failure.

EVNTRE Question Type: 6. (Dependent event using previously defined parameters and a user function)

PROBABILITIES:

CASE 1. Large Hydrogen Burn During Core Damage

USER FUNCTION		DW FAILURE	NO FAILURE
If Cntmt/DW Differential Pressure	< 40 Psid	0.	1.
	< 55	0.05	0.95
	< 60	0.17	0.83
	< 65	0.27	0.73
	< 70	0.41	0.59
	< 75	0.55	0.45
	< 80	0.68	0.32
	< 85	0.79	0.21
	< 90	0.88	0.12
	< 95	0.93	0.07
	Else	1.	0.

CASE 2. Large Hydrogen Burn At/Near RPV Failure

Same probability assignment as above for Case 1.

CASE 3. Otherwise, For Remaining Sequences With No Large Burns Default To No Drywell Failure

DW FAILURE	0
NO DW FAIL	1

Event Dependencies: Large H2 Burn During Core Damage (Event 22), and Large H2 Burn At/Near RPV Failure (Event 44).

Quantification Basis: Perry Nuclear Power Plant IPE Containment Capacity Analysis, and IPE Engineering Calculation - Drywell Fragility.

EVENT 49. POOL BYPASS BEFORE/NEAR RPV FAILURE - POOL_BYP

Two Branches:

POOL BYPASS	Pool Bypass
NO POOL BYP	No Pool Bypass

This summary event assess the probability that pool bypass will occur before or near RPV failure. This event considers pool bypass resulting from drywell failure resulting from processes occurring within the drywell structure, drywell failure from hydrogen combustion in the containment, pool bypass due to containment anchorage failure, and miscellaneous pool bypass mechanisms (e.g. failed open vacuum breakers).

EVNTRE Question Type: 2. (Dependent split fractions)

PROBABILITIES:

CASE 1. Drywell Failure At/Near RPV Failure

Event 30, Drywell Failure At/Near RPV Failure, summarizes the evaluation of Drywell failure by processes inside the Drywell which are assessed with events 30 - 38.

POOL BYPASS	1.
NO POOL BYP	0.

CASE 2. Drywell Failure Due To Containment Hydrogen Burn Before/Near RPV Failure

POOL BYPASS	1.
NO POOL BYP	0.

CASE 3. Suppression Pool Bypass Due To Anchorage Containment Failure Before/Near RPV Failure

POOL BYPASS	1.
NO POOL BYP	0.

CASE 4. Drywell Vacuum Breaker Failure Due To Large Burn When AC Power Is Available

When AC power is available, the drywell vacuum breaker isolation valves will automatically open on drywell/containment differential pressure. If the isolation valve is open on a drywell vacuum breaker line, then large burns may fail the check valve open.

POOL BYPASS	0.05
NO POOL BYP	0.95

CASE 5. Otherwise, For The Remaining Sequences Pool Bypassed By Other Failures

POOL BYPASS	0.0001
NO POOL BYP	0.9999

Event Dependencies: Event Type (Event 3), Offsite Power Recovery Time (Event 7), Large H2 Burn During Core Damage (Event 22), Mode of Containment Failure Before RPV Failure (Event 25), Drywell Failure At/Near RPV Failure (Event 39), Large H2 Burn At/Near RPV Failure (Event 44), Mode of Containment Failure At/Near RPV Failure (Event 47), Drywell Failure Due To Containment Hydrogen Burn Before/Near RPV Failure (Event 48).

Quantification Basis: Grand Gulf (Brown 1990) APET 95, and engineering judgement.

H.3.8 CONTAINMENT FAILURE BEFORE/NEAR RPV FAILURE FAILS INJECTION & SPRAY

This APET Group of Events 50 thru 53 determines the probability that containment failure causes the loss of all in-vessel injection (assuming in-vessel injection has not previously failed) and failure of the RHR containment spray system. The wetwell sprays represent an effective fission product mitigation feature which can significantly limit atmospheric releases of radionuclides. Containment sprays will be particularly important for sequences where suppression pool bypass has occurred, since the sprays then represent the only remaining major engineered safety system which can significantly mitigate radionuclide releases.

The mechanisms associated with containment failure which may cause failure of in-vessel injection and/or the containment spray system are discussed above in H.3.4.

Event 50. CONTAINMENT FAILURE AT/NEAR RPV FAILURE
IMPACT ON ECCS INJECTION AND SPRAY PIPING - PIPE_FAIL

Two Branches:

FAILURE	Failure of ECCS Injection & Spray Piping
NO FAILURE	No Piping Failure

This event assesses the probability that either the dynamic forces or movement of the containment which occur at containment failure are sufficient to disrupt the injection and spray system piping. Disruption of this piping is expected to be a serious threat for containment anchorage failure.

EVNTRE Question Type: 2. (Dependent split fractions)

PROBABILITIES:

CASE 1. Anchorage Failure Mode Containment Failure At/Near RPV Failure

FAILURE	0.9
NO FAILURE	0.1

CASE 2. Otherwise, For Other Non Anchorage Failure Sequences Default to No Failure.

FAILURE	0.
NO FAILURE	1.

Event Dependencies: Mode of Containment Failure At/Near RPV Failure (Event 47).

Quantification Basis: Containment Capacity Analysis (Gilbert/Commonwealth 1992) and engineering judgment.

Event 51. CONTAINMENT FAILURE AT/NEAR RPV FAILURE
IMPACT ON ECCS INJECTION AND SPRAY MOTORS - MTR_FAIL

Two Branches:

FAILURE   Failure of ECCS Injection & Spray Motors
NO FAILURE   No Motor Failure

This event assesses the probability that leakage of water, steam or hot gases from containment into the auxiliary building which may occur at containment failure cause failure of the injection and/or spray system motors and related components.

EVNTRE Question Type: 2. (Dependent split fractions)

PROBABILITIES:

CASE 1. Containment Failure At/Near RPV Failure And No Anchorage Piping Failure

FAILURE	0.5
NO FAILURE	0.5

CASE 2. Otherwise, For Other Non Containment Failure Sequences Default to No Failure.

FAILURE	0.
NO FAILURE	1.

Event Dependencies: Containment Failure At/Near RPV Failure (Event 46), and Containment Failure At/Near RPV Failure Impact on ECCS Injection and Spray Piping.

Quantification Basis: Containment Capacity Analysis (Gilbert/Commonwealth 1992) and engineering judgment.

Event 52. CONTAINMENT FAILURE AT/NEAR RPV FAILURE STEAM AND RADIATION IMPACT ON FIREWATER INJECTION - STM/RAD

Two Branches:

FAILURE   Failure of Firewater Injection
NO FAILURE   No Failure

This event assesses the probability that leakage of radionuclides from containment will limit personnel access to the firewater lineup or components (e.g., Diesel Driven Firewater Pump Oil Tank, or Perry Pumper) and result in failure to perform required local manual actions to initiate, or to assure continued operation of the firewater injection.

EVNTRE Question Type: 2. (Dependent split fractions)

PROBABILITIES:

CASE 1. Containment Failure At/Near RPV Failure And No Anchorage Piping Failure

FAILURE	0.5
NO FAILURE	0.5

CASE 2. Otherwise, For Other Non Containment Failure Sequences

FAILURE	0.
NO FAILURE	1.

Event Dependencies: Containment Failure At/Near RPV Failure (Event 46), Containment Failure At/Near RPV Failure Impact on ECCS Injection and Spray Piping.

Quantification Basis: Containment Capacity Analysis (Gilbert/Commonwealth 1992) and engineering judgment.

Event 53. INJECTION & SPRAY FAILURE DUE TO CONTAINMENT FAILURE AT/NEAR RPV FAILURE - INJ/SP_FL

Two Branches:

INJ & SPRY FAIL	Injection & Spray Failure
NO FAILURE	No Failure

This event summarizes the results of the prior three events in the APET.

EVNTRE Question Type: 2. (Dependent Sorting Event)

SORTING:

CASE 1. ECCS Injection & Spray Piping Failure Due To Containment Failure At/Near RPV Failure

INJ & SPRY FAIL	1.
NO FAILURE	0.

CASE 2. ECCS Injection & Spray Motor Failure, And Firewater Injection Failure Due To Containment Failure At/Near RPV Failure

INJ& SPRY FAIL	1.
NO FAILURE	0.

CASE 3. Otherwise, For Other Remaining Sequences Where All Injection Is Not Failed Default to No Failure.

INJ& SPRY FAIL	0.
NO FAILURE	1.

Event Dependencies: Containment Failure At/Near RPV Failure Impact on ECCS Injection and Spray Piping (Event 50), Containment Failure At/Near RPV Failure Impact on ECCS Injection and Spray Motors (Event 51), Containment Failure At/Near RPV Failure Steam & Radiation Impact on Firewater (Event 52).

Quantification Basis: Not Applicable for summary event.

H.3.9 PEDESTAL FAILURE DUE TO CORE DEBRIS CONCRETE INTERACTION

This APET Group of Events 54 thru 55 determines the probability of pedestal failure as a result of sidewards core concrete attack in the pedestal cavity eroding the pedestal wall to a sufficient depth that the structural integrity of the pedestal wall is compromised. Failure of the pedestal wall may result in loss of support to the RPV and result in gross motion of the RPV. This motion may result in damage to the drywell or to containment penetrations.

The amount of radial erosion will be a function of the type and extent of core concrete attack that occurs, the ratio of radial to downwards concrete attack and the failure depth for the pedestal wall.

This event is dependent on the following Events in the Perry APET.

Event 54. TYPE OF CORE DEBRIS CONCRETE INTERACTION - CCI_TYPE

Four Branches:

DRY-CCI	CCI In A Dry Pedestal Cavity
FAST-WET	Rapid CCI With an Overlying Water Layer
SLOW-WET	Slow CCI With an Overlying Water Layer
NO-CCI	NO CCI With Debris Cooled

This event determines the probability of various types of CCI which may occur in the pedestal cavity following RPV failure. If no water is in the pedestal cavity prior to RPV failure and water is not supplied following RPV failure then dry CCI will occur. If a small amount of water is in the pedestal cavity prior to RPV failure but a continuing supply of water is not available then the debris/concrete attack may be delayed until the water pool is boiled away. (This case has been conservatively combined with DRY CCI).

For cases where a large pool of water has entered the drywell prior to RPV failure or where a continuous supply of water is available to the pedestal following vessel failure then a pool of water will cover the debris. Depending upon the surface area of the debris, the debris particle size and the effective upward heat transfer rate the debris may be cooled or CCI may occur. The last three branches assess the rate of CCI given a debris pool which is covered by water.

EVNTRE Question Type: 2. (Dependent split fractions)

PROBABILITIES:

CASE 1. No RPV Failure With Debris Cooled In-Vessel

DRY-CCI	0.
FAST-WET	0.
SLOW-WET	0.
NO-CCI	1.

CASE 2. No Water In Pedestal Cavity At RPV Failure, And No Continued Injection to the Pedestal Cavity

DRY-CCI	1.
FAST-WET	0.
SLOW-WET	0.
NO-CCI	0.

CASE 3. Water in Pedestal Cavity At RPV Failure From Flooding, No Continued Injection To The Pedestal Cavity, And HPME

DRY-CCI	0.
FAST-WET	0.175
SLOW-WET	0.4875
NO-CCI	0.3375

CASE 4. Water in Pedestal Cavity At RPV Failure From Flooding, No Continued Injection To The Pedestal Cavity, No HPME, And Large Molten Mass of Debris in Lower Head At RPV Failure

DRY-CCI	0.
FAST-WET	0.28
SLOW-WET	0.48
NO-CCI	0.24

CASE 5. Water in Pedestal Cavity At RPV Failure From Flooding, No Continued Injection To The Pedestal Cavity, No HPME, And Small Molten Mass of Debris in Lower Head

At RPV Failure

DRY-CCI	0.
FAST-WET	0.28
SLOW-WET	0.48
NO-CCI	0.24

CASE 6. No Water in Pedestal Cavity At RPV Failure, Continued Injection To The Pedestal Cavity, And HPME

DRY-CCI	0.
FAST-WET	0.315
SLOW-WET	0.4775
NO-CCI	0.2075

CASE 7. No Water in Pedestal Cavity At RPV Failure, Continued Injection To The Pedestal Cavity, And No HPME

DRY-CCI	0.
FAST-WET	0.3375
SLOW-WET	0.475
NO-CCI	0.1875

CASE 8. Water in Pedestal Cavity At RPV Failure, Continued Injection To The Pedestal Cavity, And HPME

DRY-CCI	0.
FAST-WET	0.175
SLOW-WET	0.4875
NO-CCI	0.3375

CASE 9. Water in Pedestal Cavity At RPV Failure, Continued Injection To The Pedestal Cavity, No HPME and Large Molten Mass

DRY-CCI	0.
FAST-WET	0.28
SLOW-WET	0.48
NO-CCI	0.24

CASE 10. Water in Pedestal Cavity At RPV Failure, Continued Injection To The Pedestal Cavity, No HPME and Small Molten Mass

DRY-CCI	0.
FAST-WET	0.28

SLOW-WET	0.48
NO-CCI	0.24

CASE 11. Otherwise, Should never reach this case.

DRY-CCI	1.
FAST-WET	0.
SLOW-WET	0.
NO-CCI	0.

Event Dependencies: Debris Molten Mass At RPV Failure (Event 14), RPV Failure Mode & Failure Size (Event 32), Water In Pedestal At RPV Failure (Event 33), HPME (Event 43), Injection & Spray Failure Due to Containment Failure Before/Near RPV Failure (Event 53).

Quantification Basis: Grand Gulf (Brown 1990) APET 99, IPE Engineering Calculation - CCI And Pedestal Ablation.

Event 55. PEDESTAL FAILURE DUE TO CORE DEBRIS CONCRETE INTERACTION - PED_FAIL

Three Branches:

AT VB	Pedestal Failure At RPV Failure
FAIL AFTER VB	Pedestal Failure After RPV Failure
NO FAILURE	No Pedestal Failure

Given a type of CCI determined in APET Event 54 this event assesses whether pedestal structural failure from CCI radial erosion of the pedestal wall will occur. The mean erosion depth for failure was taken as 3.6 feet from the Grand Gulf Pedestal Erosion expert's determination (Harper 1991).

EVNTRE Question Type: 2. (Dependent split fractions)

PROBABILITIES:

CASE 1. Drywell Failure At/Near RPV Failure
By Processes Inside The Drywell
Or Drywell Failure Due To Containment Hydrogen Burn
Before/Near RPV Failure

AT VB	1.
AFTER VB	0.
NO FAILURE	0.

CASE 2. No RPV Failure With Debris Cooled In-Vessel

AT VB	0.

AFTER VB	0.
NO FAILURE	1.

CASE 3. Dry CCI

AT VB	0.
AFTER VB	0.43
NO FAILURE	0.57

CASE 4. Fast Wet CCI

AT VB	0.
AFTER VB	0.29
NO FAILURE	0.71

CASE 5. Slow Wet CCI

AT VB	0.
AFTER VB	0.25
NO FAILURE	0.75

CASE 6. No CCI

AT VB	0.
AFTER VB	0.
NO FAILURE	1.

CASE 7. Otherwise, Should never reach this case.

AT VB	0.
AFTER VB	0.
NO FAILURE	1.

Event Dependencies: RPV Failure Mode and Failure Size (Event 32), Drywell Failure At/Near RPV Failure (Event 39), Drywell Failure Due to Containment H2 Burn Before/Near RPV Failure (Event 48), and Type of Core Debris Concrete Interaction (Event 54).

Quantification Basis: NUREG/CR-4551 Volume 2 Part 2, Molten Core Containment Interaction Issues (Harper 1991) Grand Gulf Pedestal Erosion Section 6.2. Radial erosion depths for the dry CCI cases were based on a weighted average of the results from Table 6-3 (Groups 5, 6 and 7). The fast-wet CCI cases were based on an average of Groups 1 and 2. The slow-wet CCI cases were based on an average of Groups 2 and 3.

H.3.10 MODE OF LATE BURN AND OVERPRESSURE CONTAINMENT FAILURE

This APET Group of Events 56 thru 66 determines the probability (and mode) for containment failures which occur late in the accident sequences. These events assess containment failures resulting from hydrogen combustion and detonation and from gradual overpressurization from steam and non condensible gas production.

This event is dependent on the following Events in the Perry APET.

EVENT 56. Mode Of RHR Spray Operation Late - DE_INERT

Three Branches:

CONTROLLED	Sprays are operating in a throttled cooling mode to mitigate or exclude hydrogen deflagrations and detonations
SPRAY	Sprays are operating at full design cooling
NO SPRAY	Sprays Not Available

This event summarizes whether the RHR sprays are available (and are operated) late in the accident sequence. It further allows for the assessment of controlled (throttled) spray operation or for normal design flow operation. The controlled spray operation mode is directed at limiting the spray flow rate such that the containment atmosphere steam concentration would remain elevated above the steam inerting limit for hydrogen combustion (55 volume percent) and below the emergency procedure pressure limit of 40 psig, or maintains the the as-found containment steam concentration to minimize the expected peak burn pressures. The controlled spray operation mode is not currently considered in the Perry Plant Emergency Instruction. Consequently, this mode of spray operation was not considered in the base case Perry analysis (the Controlled Spray Operation branch probability was set to zero). This branch was included for use in sensitivity analysis.

EVNTRE Question Type: 2. (Dependent Sorting Event)

SORTING:

CASE 1. RHR Spray Loop Not Available

RHR Spray is determined to be not available in the Plant Damage State Grouping Logic.

CONTROLLED	0.
SPRAY	0.
NO SPRAY	1.

CASE 2. AC Power Never Lost, And RHR Spray Available

With AC Power available, it is almost certain that the hydrogen igniters are available to control hydrogen. Therefore a controlled, steam-inerted containment atmosphere is not necessary to credit the mitigative impact of controlled spray operation.

CONTROLLED	0.
SPRAY	1.
NO SPRAY	0.

CASE 3. AC Power Lost, Power Recovered Prior to RPV Failure, And RHR Spray Available

Under these conditions it is probable that there will be a significant quantities of hydrogen released into the containment and that the hydrogen ignition system will not be energized during the first hour or two following AC power recovery. (Reference Event 16 discussion.) Under these conditions it is possible to limit the threat from hydrogen combustion by throttling the RHR bypass flow to maintain the containment steam concentration in the steam-inert regime. However, since this mode of operation is not in the Perry Plant Emergency Instruction, this mode of operation has been assigned a zero probability for base case evaluation.

CONTROLLED	0.
SPRAY	1.
NO SPRAY	0.

CASE 4. AC Power Lost, Power Recovered Prior to Containment Capacity Overpressure Threshold, And RHR Spray Available

Under these conditions it is probable that there will be a significant quantities of hydrogen released into the containment and that the hydrogen ignition system will not be energized during the first hour or two following AC power recovery. (Reference Event 16 discussion.) Under these conditions it is possible to limit the threat from hydrogen combustion by throttling the RHR bypass flow to maintain the containment steam concentration in the steam-inert regime. However, since this mode of operation is not in the Perry Plant Emergency Instruction, this mode of operation has been assigned a zero probability for base case evaluation.

CONTROLLED	0.
SPRAY	1.
NO SPRAY	0.

CASE 5. Otherwise, Should never reach this case.

CONTROLLED	0.

SPRAY 0.
NO SPRAY 1.

Event Dependencies: Event Type (Event 3), Offsite Power Recovery Time (Event 7), Containment Heat Removal With RHR Spray Loop (Event 8).

Quantification Basis: Not Applicable.

EVENT 57. HYDROGEN IGNITION SOURCES AVAILABLE LATE - IG_SOURC

Two Branches:

NO SOURCE	No Hydrogen Ignition Source
IGN SOURCE	Hydrogen Ignition Sources Available

With a continuous ignition source available in containment it is assumed that controlled hydrogen combustion (or small hydrogen burns) will preclude the build-up of hydrogen concentrations whose combustion would threaten containment integrity. A continuous ignition source is assumed to be available if the hydrogen ignition system is operating. Since there may have been a substantial time period between RPV failure and the time when late combustion may occur, earlier burns are not considered to provide a reliable ignition source for late burns (and thus ensure that only small burns would occur late).

EVNTRE Question Type: 2. (Dependent Split Fraction)

PROBABILITIES:

CASE 1 Hydrogen Ignition System Available Early

This is true is the HIS was placed in service early with no human interaction error.

NO SOURCE	0.
IGN SOURCE	1.

CASE 2. No Loss Of AC Power, Operator Failure To Initiate Hydrogen Ignition System, And Recovery Of Hydrogen Ignitors.

This recovery of ignitors before the Hydrogen Deflagration Overpressure Limit is reach is the same recovery modeled in Event 42., Hydrogen Ignition Sources Available At RPV Failure.

NO SOURCE	0.1
IGN SOURCE	0.9

CASE 3. SBO And Power Recovery Prior To RPV Failure

The HIS will not be placed on immediately after power recovery due to the hydrogen analyzers being out of calibration or due to the hydrogen concentration not determined. After the hydrogen analyzers warm up and reach steady state calibration, it is likely that the hydrogen concentration will be above the hydrogen deflagration overpressure limit, and the HIS will not be placed in service. Therefore, the availability of this ignition source is uncertain.

NO SOURCE	0.5
IGN SOURCE	0.5

CASE 4. SBO With Power Recovery Prior To Containment Limit

The HIS will not be placed on immediately after power recovery due to the hydrogen analyzers being out of calibration or due to the hydrogen concentration not determined. After the hydrogen analyzers warm up and reach steady state calibration, it is likely that the hydrogen concentration will be above the hydrogen deflagration overpressure limit, and the HIS will not be placed in service. Therefore, the availability of this ignition source is highly uncertain, and the availability of the ignition source is conservatively set to 0.

NO SOURCE	1.
IGN SOURCE	0.

CASE 4. Otherwise, For The Remaining Sequences
With Loss Of AC Power and No Power Recovery
Assign to No Hydrogen Ignition Source

NO SOURCE	1.
IGN SOURCE	0.

Event Dependencies: Event Type (Event 3), Offsite Power Recovery Time (Event 7), Hydrogen Ignition System Available (Event 16).

Quantification Basis: Review of the M51 Combustible Gas Control System and the M56 Hydrogen Ignition System, Perry Plant Emergency Instruction IPE Human Interaction Technical Assignment File, and engineering judgement.

Event 58. CONTAINMENT STEAM CONCENTRATION LATE - ST_CONC

Six branches:

0-15 % Containment Steam Volume Percent
15-25 %

25-35 %
35-45 %
45-55 %
>55 %

This branch assesses the containment steam concentration late in the accident. The probability of hydrogen burn ignition and the efficiency of the burn (considered in subsequent events) are dependent upon the branch taken under this heading. The containment steam concentration is a function of the mode of spray operation, the sequence type, the time of injection failure and whether the containment is intact at core damage. The steam concentration regime probabilities given for the cases below were estimated using MAAP results for each case.

EVNTRE Question Type: 2. (Dependent split fractions)

PROBABILITIES:

CASE 1. Spray Loop Operation At Design Cooling

With a containment spray loop operating at design cooling with containment heat removal optimized with the Residual Heat Removal heat exchanger, the steam concentration would be low.

0-15 %	1.
15-25 %	0.
25-35 %	0.
35-45 %	0.
45-55 %	0.
>55 %	0.

CASE 2. Containment Vent Not Isolated At RPV Failure

For the range of distributions in SBO sequences, the steam concentration is conservatively bounded by the case of SBO with no injection with the containment vent open.

0-15 %	0.25
15-25 %	0.75
25-35 %	0.
35-45 %	0.
45-55 %	0.
>55 %	0.

CASE 3. SBO And No Injection Failure Early (0 - 2.8 Hrs)

0-15 %	0.99
15-25 %	0.01
25-35 %	0.
35-45 %	0.
45-55 %	0.
>55 %	0.

CASE 4. SBO And RCIC Injection Failure (2.8-4.2 Hrs)

0-15 %	0.
15-25 %	0.
25-35 %	0.29
35-45 %	0.71
45-55 %	0.
>55 %	0.

CASE 5. SBO And HPCS Injection Failure (> 4.2 Hrs)

0-15 %	0.
15-25 %	0.
25-35 %	0.
35-45 %	0.
45-55 %	0.
>55 %	1.

CASE 6. Containment Failed At Core Damage

0-15 %	0.
15-25 %	0.
25-35 %	0.
35-45 %	0.
45-55 %	0.
>55 %	1.

CASE 7. Otherwise, All Other Sequences
Default to Low Steam Concentration

All other sequences are conservatively evaluated by assuming a low steam concentration.

0-15 %	1.
15-25 %	0.
25-35 %	0.
35-45 %	0.
45-55 %	0.
>55 %	0.
>55 %	0

Event Dependencies: Containment Status At Core Damage (Event 2), Event Type (Event 3), Containment Vent Isolated At RPV Failure (Event 5), RPV Injection Failure Time (Event 6), Mode of RHR Spray Operation Late (Event 56).

Quantification Basis: IPE Engineering Calculation - Hydrogen Burns [with MAAP 3.0B Rev 7.02 using "Recommended Sensitivity Analyses For An IPE Using MAAP 3.0B" (EPRI 1991)].

EVENT 59. HYDROGEN COMBUSTION BEFORE/AT RPV FAILURE - BURN_B4

Two Branches:

EARLY BURN	Early Burn Before/At RPV Failure
NO EARLY BURN	No Early Burn

This event summarizes whether an earlier burn in containment has occurred. This information is used to assess the late hydrogen concentration in containment.

EVNTRE Question Type: 2. (Dependent sorting event)

SORTING:

CASE 1. Early H2 Burn During Core Damage Or At/Near RPV Failure

Early burns include both small and large burns which occurred before RPV failure or at/near RPV failure.

EARLY BURN	1.
NO EARLY BURN	0.

CASE 2. Otherwise, No Early Burn

EARLY BURN	0.
NO EARLY BURN	1.

Event Dependencies: Small H2 Burns At Low H2 Concentrations (Event 21), Large H2 Burn During Core Damage (Event 22), and Large H2 Burn At/Near RPV Failure (Event 44).

Quantification Basis: Not Applicable.

EVENT 60. CONTAINMENT EFFECTIVE HYDROGEN CONCENTRATION LATE - H2_CONC

SIx Branches:

< 4%	Volume Percent Hydrogen
4- 8%	
8-12%	
12-16%	
16-20%	
> 20%	

This branch assesses the containment effective hydrogen concentration (including the carbon monoxide produced during core concrete interaction). The probability of hydrogen burn ignition and the efficiency of the burn (considered in subsequent events) are dependent upon the branch taken under this heading. The containment hydrogen concentration is a function of the mode of CCI (APET Event 54), and whether a hydrogen burn occurred early in the sequence (APET Event 59).

EVNTRE Question Type: 2. (Dependent split fractions)

PROBABILITIES:

CASE 1. Early Hydrogen Burn And No CCI

This case is estimated by judgement and comparison from the calculated distribution for SBO with Spray, No Burn, and No CCI. It is considered that an early burn would reduce the initial hydrogen concentration to a lower plateau of about 4.5% based on 1/4 testing of Mark III containment hydrogen combustion (EPRI - 1988).

< 4%	0.5
4- 8%	0.5
8-12%	0.0
12-16%	0.0
16-20%	0.0
> 20%	0.0

CASE 2. Early Hydrogen Burn And Slow Wet CCI

This case is estimated by judgement and comparison from the calculated distribution for SBO with Spray, No Burn, and Dry CCI. It is considered that an early burn would reduce the initial hydrogen concentration to a lower plateau of about 4.5% based on 1/4 testing of Mark III containment hydrogen combustion (EPRI - 1988).

< 4%	0.1
4- 8%	0.25
8-12%	0.3
12-16%	0.25
16-20%	0.1
> 20%	0.0

CASE 3. Early Hydrogen Burn And Fast Wet CCI

This case is conservatively estimated by judgement and comparison from the calculated distribution for SBO with Spray, No Burn, and Dry CCI. It is considered that an early burn would reduce the initial hydrogen concentration to a lower plateau of about 4.5% based on 1/4 testing of Mark III containment hydrogen combustion (EPRI - 1988).

< 4%	0.0
4- 8%	0.1
8-12%	0.2
12-16%	0.35
16-20%	0.35

> 20%	0.0

CASE 4. Early Hydrogen Burn, Dry CCI and No Spray During SBO And No Injection

This case is conservatively estimated by comparison from the calculated distribution for SBO with No Spray, No Burn, and Dry CCI with little credit for significant hydrogen reduction.

< 4%	0.0
4- 8%	0.35
8-12%	0.07
12-16%	0.06
16-20%	0.06
> 20%	0.46

CASE 5. Early Hydrogen Burn, Dry CCI and No Spray During SBO And RCIC Injection Failure

This case is conservatively estimated by comparison from the calculated distribution for SBO with No Spray, No Burn, and Dry CCI with little credit for significant hydrogen reduction.

< 4%	0.0
4- 8%	0.04
8-12%	0.60
12-16%	0.36
16-20%	0.04
> 20%	0.00

CASE 6. Early Hydrogen Burn, Dry CCI and No Spray During SBO And HPCS Failure

This case is conservatively estimated by comparison from the calculated distribution for SBO with No Spray, No Burn, and Dry CCI with little credit for significant hydrogen reduction.

< 4%	0.64
4- 8%	0.36
8-12%	0.00
12-16%	0.00
16-20%	0.00
> 20%	0.00

CASE 7. Early Hydrogen Burn, Dry CCI and Spray During SBO And No Injection

This case is conservatively estimated by comparison from the calculated distribution for SBO with Spray, No Burn, and Dry CCI with little credit for significant hydrogen reduction.

< 4%	0.0
4- 8%	0.36
8-12%	0.07
12-16%	0.06
16-20%	0.06
> 20%	0.45

CASE 8. Early Hydrogen Burn, Dry CCI and Spray During SBO And RCIC Injection Failure

This case is conservatively estimated by comparison from the calculated distribution for SBO with Spray, No Burn, and Dry CCI with little credit for significant hydrogen reduction.

< 4%	0.0
4- 8%	0.05
8-12%	0.43
12-16%	0.37
16-20%	0.14
> 20%	0.01

CASE 9. Early Hydrogen Burn, Dry CCI and Spray During SBO And HPCS Failure

This case is conservatively estimated by comparison from the calculated distribution for SBO with Spray, No Burn, and Dry CCI with little credit for significant hydrogen reduction.

< 4%	0.38
4- 8%	0.51
8-12%	0.11
12-16%	0.00
16-20%	0.00
> 20%	0 00

CASE 10. No Early Hydrogen Burn, No CCI and No Spray During SBO And No Injection

This case is estimated by directly using the calculated distribution for SBO with No Spray, No Burn, and No CCI.

< 4%	0.0
4- 8%	1.0
8-12%	0.0
12-16%	0.0
16-20%	0.0
> 20%	0.0

CASE 11. No Early Hydrogen Burn, No CCI and No Spray

During SBO And RCIC Injection Failure

This case is estimated by directly using the calculated distribution for SBO with No Spray, No Burn, and No CCI.

< 4%	0.0
4- 8%	0.0
8-12%	0.8
12-16%	0.2
16-20%	0.0
> 20%	0.0

CASE 12. No Early Hydrogen Burn, No CCI and No Spray During SBO And HPCS Failure

This case is estimated by directly using the calculated distribution for SBO with No Spray, No Burn, and No CCI.

< 4%	0.59
4- 8%	0 41
8-12%	0.00
12-16%	0.00
16-20%	0.00
> 20%	0.00

CASE 13. No Early Hydrogen Burn, No CCI and Spray During SBO And No Injection

This case is estimated by directly using the calculated distribution for SBO with Spray, No Burn, and No CCI.

< 4%	0.0
4- 8%	1.0
8-12%	0.0
12-16%	0.0
16-20%	0.0
> 20%	0.0

CASE 14. No Early Hydrogen Burn, No CCI and Spray During SBO And RCIC Injection Failure

This case is estimated by directly using the calculated distribution for SBO with Spray, No Burn, and No CCI.

< 4%	0.0
4- 8%	0.0
8-12%	0.56
12-16%	0.38
16-20%	0.06
> 20%	0.0

CASE 15. No Early Hydrogen Burn, No CCI and Spray During SBO And HPCS Failure

This case is estimated by directly using the calculated distribution for SBO with Spray, No Burn, and No CCI.

< 4%	0.33
4- 8%	0.51
8-12%	0.16
12-16%	0.00
16-20%	0.00
> 20%	0.00

CASE 16. No Early Hydrogen Burn And Slow Wet CCI

This case is conservatively estimated by comparison with the calculated distribution for SBO with Spray, No Burn, and Dry CCI.

< 4%	0.0
4- 8%	0.1
8-12%	0.15
12-16%	0.25
16-20%	0.5
> 20%	0.0

CASE 17. No Early Hydrogen Burn And Fast Wet CCI

This case is very conservatively estimated by with comparison from the calculated distribution for SBO with Spray, No Burn, and Dry CCI.

< 4%	0.0
4- 8%	0.0
8-12%	0.1
12-16%	0.15
16-20%	0.75
> 20%	0.0

CASE 18. No Early Hydrogen Burn, Dry CCI and No Spray During SBO And No Injection

This case is estimated by directly using the calculated distribution for SBO with No Spray, No Burn, and Dry CCI.

< 4%	0.0
4- 8%	0.30
8-12%	0.07
12-16%	0.06
16-20%	0.06
> 20%	0.51

CASE 19. No Early Hydrogen Burn, Dry CCI and No Spray During SBO And RCIC Injection Failure

This case is estimated by directly using the calculated distribution for SBO with No Spray, No Burn, and Dry CCI.

< 4%	0.0
4- 8%	0.0
8-12%	0.60
12-16%	0.36
16-20%	0.04
> 20%	0.0

CASE 20. No Early Hydrogen Burn, Dry CCI and No Spray During SBO And HPCS Failure

This case is estimated by directly using the calculated distribution for SBO with No Spray, No burn, and Dry CCI.

< 4%	0.59
4- 8%	0.41
8-12%	0.00
12-16%	0.00
16-20%	0.00
> 20%	0.00

CASE 21. No Early Hydrogen Burn, Dry CCI and Spray During SBO And No Injection

This case is estimated by directly using the calculated distribution for SBO with Spray, No Burn, and Dry CCI.

< 4%	0.00
4- 8%	0.31
8-12%	0.07
12-16%	0.06
16-20%	0.06
> 20%	0.50

CASE 22. No Early Hydrogen Burn, Dry CCI and Spray During SBO And RCIC Injection Failure

This case is estimated by directly using the calculated distribution for SBO with Spray, No Burn, and Dry CCI.

< 4%	0.00
4- 8%	0.00
8-12%	0.43
12-16%	0.42
16-20%	0.14

> 20%	0.01

CASE 23. No Early Hydrogen Burn, Dry CCI and Spray During SBO And HPCS Failure

This case is estimated by directly using the calculated distribution for SBO with Spray, No Burn, and Dry CCI.

< 4%	0.33
4- 8%	0.51
8-12%	0.16
12-16%	0.00
16-20%	0.00
> 20%	0.00

CASE 24. Otherwise, The Remaining Sequences

This case is estimated with judgement with comparison to the calculated distribution for SBO with Spray, No Burn, and Dry CCI.

< 4%	0.
4- 8%	0.
8-12%	0.
12-16%	0
16-20%	1.
> 20%	0.

Event Dependencies: Event Type (Event 3), RPV Injection Failure Time (Event 6), Type Of Core Debris Concrete Interactions (Event 54), Mode of RHR Spray Operation Late (Event 56), H2 Combustion Before/At RPV Failure (Event 59).

Quantification Basis: IPE Engineering Calculation - Hydrogen Burns [with MAAP 3.0B Rev 7.02 using "Recommended Sensitivity Analyses For An IPE Using MAAP 3.0B" (EPRI 1991)]. Hydrogen Combustion Experiments In a 1/4 Scale Model Of A Mark III Nuclear Reactor Containment (EPRI - 1988).

Event 61. AC POWER AVAILABLE LATE - AC_PWR

Two Branches:

AC LATE	AC Power Available Late
NO AC LATE	No AC Power Available Late

This event summarizes whether AC power is available late in the sequence. AC power will be available late if AC power was never lost or if AC power was initially lost but was recovered. This information is used to assess the potential for hydrogen ignition late in the sequence.

EVNTRE Question Type: 2. (Dependent sorting event)

SORTING:

CASE 1. SBO And No AC Power Recovery

AC LATE	0.
NO AC LATE	1.

CASE 2. Otherwise, AC Power Available

AC LATE	1.
NO AC LATE	0.

Event Dependencies: Event Type (Event 3), and Offsite Power Recovery Time (Event 7).

Quantification Basis: Not Applicable.

EVENT 62. LARGE HYDROGEN BURN LATE - LG_BURN

Two Branches:

1) No Large Hydrogen Burn Ignited NO BURN
2) Large Hydrogen Burn Ignited LARGE BURN

This event assess whether a large hydrogen burn is ignited in containment late in the accident sequence. If a continuous ignition source is available (as defined in Event 57) then a large hydrogen burn is assumed to be prevented. Also if the steam concentration is above 55% then the containment atmosphere is inert to hydrogen burns. The probability that a large hydrogen burn is ignited is a function of the following parameters;

1. Containment steam concentration,
2. Containment hydrogen concentration, and
3. AC power availability.

Forty four cases (various combinations of the parameters listed above were identified to perform the quantification). In addition to assessing the probability of ignition this event also sets two parameter values which are used by subsequent events; peak hydrogen burn pressure (parameter 5) and containment base pressure prior to the burn (parameter 6).

EVNTRE Question Type: 4. (Dependent Split Fraction with parameter values set for each case.)

PROBABILITIES:

CASE 1. Hydrogen Ignitions Sources Available Late

NO BURN	1.				
LG BURN	0.	BASE PRESS	0	PEAK PRESS	0

CASE 2. Containment Concentration Is Greater Than 55%

NO BURN	1.				
LG BURN	0.	BASE PRESS	0	PEAK PRESS	0

CASE 3. Containment Hydrogen Concentration Less Than 4%

No large burns are possible.

NO BURN	1.				
LG BURN	0.	BASE PRESS	0	PEAK PRESS	0

CASE 4. Cntmt Steam [0-15%], Cntmt Eff H2 [4-8%] [H2] = 6%
And No AC Power Available Late

NO BURN	0.71				
LG BURN	0.29	BASE PRESS	7.6	PEAK PRESS	19.4

CASE 5. Cntmt Steam [0-15%], Cntmt Eff H2 [8-12%] [H2] = 10%
And No AC Power Available Late

NO BURN	0.67				
LG BURN	0.33	BASE PRESS	8.8	PEAK PRESS	51.6

CASE 6. Cntmt Steam [0-15%], Cntmt Eff H2 [12-16%] [H2] = 14%
And No AC Power Available Late

NO BURN	0.58				
LG BURN	0.42	BASE PRESS	10	PEAK PRESS	92.6

CASE 7. Cntmt Steam [0-15%], Cntmt Eff H2 [16-20%] [H2] = 18%
And No AC Power Available Late

NO BURN	0.49				
LG BURN	0.51	BASE PRESS	11	PEAK PRESS	121

CASE 8. Cntmt Steam [0-15%], Cntmt Eff H2 [> 20%] [H2] = 24%
And No AC Power Available Late

NO BURN 0.49
LG BURN 0.51 BASE PRESS 14 PEAK PRESS 166

CASE 9. Cntmt Steam [15-25%], Cntmt Eff H2 [4-8%] [H2] = 6%
And No AC Power Available Late

NO BURN 0.71
LG BURN 0.29 BASE PRESS 13 PEAK PRESS 26.7

CASE 10. Cntmt Steam [15-25%], Cntmt Eff H2 [8-12%] [H2] = 10%
And No AC Power Available Late

NO BURN 0.67
LG BURN 0.33 BASE PRESS 15 PEAK PRESS 59.7

CASE 11. Cntmt Steam [15-25%], Cntmt Eff H2 [12-16%] [H2] = 14%
And No AC Power Available Late

NO BURN 0.58
LG BURN 0.42 BASE PRESS 17 PEAK PRESS 96.8

CASE 12. Cntmt Steam [15-25%], Cntmt Eff H2 [16-20%] [H2] = 18%
And No AC Power Available Late

NO BURN 0.49
LG BURN 0.51 BASE PRESS 19 PEAK PRESS 126

CASE 13. Cntmt Steam [15-25%], Cntmt Eff H2 [> 20%] [H2] = 24%
And No AC Power Available Late

NO BURN 0.49
LG BURN 0.51 BASE PRESS 23 PEAK PRESS 170

CASE 14. Cntmt Steam [25-35%], Cntmt Eff H2 [4-8%] [H2] = 6%
And No AC Power Available Late

NO BURN 0.71
LG BURN 0.29 BASE PRESS 19 PEAK PRESS 34.1

CASE 15. Cntmt Steam [25-35%], Cntmt Eff H2 [8-12%] [H2] = 10%
And No AC Power Available Late

NO BURN 0.67
LG BURN 0.33 BASE PRESS 21 PEAK PRESS 67.9

CASE 16. Cntmt Steam [25-35%], Cntmt Eff H2 [12-16%] [H2] = 14%

And No AC Power Available Late

NO BURN 0.58
LG BURN 0.42 BASE PRESS 24 PEAK PRESS 103

CASE 17. Cntmt Steam [25-35%], Cntmt Eff H2 [16-20%] [H2] = 18%
And No AC Power Available Late

NO BURN 0.49
LG BURN 0.51 BASE PRESS 28 PEAK PRESS 134

CASE 18. Cntmt Steam [25-35%], Cntmt Eff H2 [> 20%] [H2] = 24%
And No AC Power Available Late

NO BURN 0.49
LG BURN 0.51 BASE PRESS 34 PEAK PRESS 161

CASE 19. Cntmt Steam [35-45%], Cntmt Eff H2 [4-8%] [H2] = 6%
And No AC Power Available Late

NO BURN 0.71
LG BURN 0.29 BASE PRESS 27 PEAK PRESS 44

CASE 20. Cntmt Steam [35-45%], Cntmt Eff H2 [8-12%] [H2] = 10%
And No AC Power Available Late

NO BURN 0.67
LG BURN 0.33 BASE PRESS 30 PEAK PRESS 79.6

CASE 21. Cntmt Steam [35-45%], Cntmt Eff H2 [12-16%] [H2] = 14%
And No AC Power Available Late

NO BURN 0.58
LG BURN 0.42 BASE PRESS 35 PEAK PRESS 112

CASE 22. Cntmt Steam [35-45%], Cntmt Eff H2 [16-20%] [H2] = 18%
And No AC Power Available Late

NO BURN 0.49
LG BURN 0.51 BASE PRESS 40 PEAK PRESS 143

CASE 23. Cntmt Steam [35-45%], Cntmt Eff H2 [> 20%] [H2] = 24%
And No AC Power Available Late

NO BURN 0.49
LG BURN 0.51 BASE PRESS 50 PEAK PRESS 157

CASE 24. Cntmt Steam [45-55%], Cntmt Eff H2 [4-8%] [H2] = 6%
And No AC Power Available Late

NO BURN 0.71
LG BURN 0.29 BASE PRESS 38 PEAK PRESS 58.2

CASE 25. Cntmt Steam [45-55%], Cntmt Eff H2 [8-12%] [H2] = 10%
And No AC Power Available Late

NO BURN 0.67
LG BURN 0.63 BASE PRESS 44 PEAK PRESS 97.9

CASE 26. Cntmt Steam [45-55%], Cntmt Eff H2 [12-16%] [H2] = 14%
And No AC Power Available Late

NO BURN 0.58
LG BURN 0.42 BASE PRESS 51 PEAK PRESS 128

CASE 27. Cntmt Steam [45-55%], Cntmt Eff H2 [16-20%] [H2] = 18%
And No AC Power Available Late

NO BURN 0.49
LG BURN 0.51 BASE PRESS 59 PEAK PRESS 146

CASE 28. Cntmt Steam [45-55%], Cntmt Eff H2 [> 20%] [H2] = 24%
And No AC Power Available Late

NO BURN 0.49
LG BURN 0.51 BASE PRESS 78 PEAK PRESS 168

CASE 29. Cntmt Steam [0-15%], Cntmt Eff H2 [4-8%] [H2] = 6%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 7.6 PEAK PRESS 19.4

CASE 30. Cntmt Steam [0-15%], Cntmt Eff H2 [8-12%] [H2] = 10%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 8.8 PEAK PRESS 51.6

CASE 31. Cntmt Steam [0-15%], Cntmt Eff H2 [12-16%] [H2] = 14%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 10 PEAK PRESS 92.6

CASE 32. Cntmt Steam [0-15%], Cntmt Eff H2 [16-20%] [H2] = 18%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 11 PEAK PRESS 121

CASE 33. Cntmt Steam [0-15%], Cntmt Eff H2 [> 20%] [H2] = 24%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 14 PEAK PRESS 166

CASE 34. Cntmt Steam [15-25%], Cntmt Eff H2 [4-8%] [H2] = 6%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 13 PEAK PRESS 26.7

CASE 35. Cntmt Steam [15-25%], Cntmt Eff H2 [8-12%] [H2] = 10%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 15 PEAK PRESS 59.7

CASE 36. Cntmt Steam [15-25%], Cntmt Eff H2 [12-16%] [H2] = 14%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 17 PEAK PRESS 96.8

CASE 37. Cntmt Steam [15-25%], Cntmt Eff H2 [16-20%] [H2] = 18%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 19 PEAK PRESS 126

CASE 38. Cntmt Steam [15-25%], Cntmt Eff H2 [> 20%] [H2] = 24%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 23 PEAK PRESS 170

CASE 39. Cntmt Steam [25-35%], Cntmt Eff H2 [4-8%] [H2] = 6%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 19 PEAK PRESS 34.1

CASE 40. Cntmt Steam [25-35%], Cntmt Eff H2 [8-12%] [H2] = 10%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 21 PEAK PRESS 67.9

CASE 41. Cntmt Steam [25-35%], Cntmt Eff H2 [12-16%] [H2] = 14%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 24 PEAK PRESS 103

CASE 42. Cntmt Steam [25-35%], Cntmt Eff H2 [16-20%] [H2] = 18%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 28 PEAK PRESS 134

CASE 43. Cntmt Steam [25-35%], Cntmt Eff H2 [> 20%] [H2] = 24%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 34 PEAK PRESS 161

CASE 44. Cntmt Steam [35-45%], Cntmt Eff H2 [4-8%] [H2] = 6%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 27 PEAK PRESS 44

CASE 45. Cntmt Steam [35-45%], Cntmt Eff H2 [8-12%] [H2] = 10%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 30 PEAK PRESS 79.6

CASE 46. Cntmt Steam [35-45%], Cntmt Eff H2 [12-16%] [H2] = 14%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 35 PEAK PRESS 112

CASE 47. Cntmt Steam [35-45%], Cntmt Eff H2 [16-20%] [H2] = 18%
And AC Power Available Late

NO BURN 0.
LG BURN 1. BASE PRESS 40 PEAK PRESS 143

CASE 48. Cntmt Steam [35-45%], Cntmt Eff H2 [> 20%] [H2] = 24%
And AC Power Available Late

NO BURN	0.		
LG BURN	1.	BASE PRESS 50	PEAK PRESS 157

CASE 49. Cntmt Steam [45-55%], Cntmt Eff H2 [4-8%] [H2] = 6%
And AC Power Available Late

NO BURN	0.		
LG BURN	1.	BASE PRESS 38	PEAK PRESS 58.2

CASE 50. Cntmt Steam [45-55%], Cntmt Eff H2 [8-12%] [H2] = 10%
And AC Power Available Late

NO BURN	0.		
LG BURN	1.	BASE PRESS 44	PEAK PRESS 97.9

CASE 51. Cntmt Steam [45-55%], Cntmt Eff H2 [12-16%] [H2] = 14%
And AC Power Available Late

NO BURN	0.		
LG BURN	1.	BASE PRESS 51	PEAK PRESS 128

CASE 52. Cntmt Steam [45-55%], Cntmt Eff H2 [16-20%] [H2] = 18%
And AC Power Available Late

NO BURN	0.		
LG BURN	1.	BASE PRESS 59	PEAK PRESS 146

CASE 53. Cntmt Steam [45-55%], Cntmt Eff H2 [> 20%] [H2] = 24%
And AC Power Available Late

NO BURN	0.		
LG BURN	1.	BASE PRESS 78	PEAK PRESS 168

CASE 54. Otherwise, Should Never Reach This Case

NO BURN	1.		
LG BURN	0.	BASE PRESS 0	PEAK PRESS 0

Event Dependencies: Hydrogen Ignition Sources Available Late (Event 57), Containment Steam Concentration Late (Event 58), Containment Effective Hydrogen Concentration Late (Event 60), AC Power Available Late (Event 61).

Quantification Basis: IPE Engineering Calculation - Hydrogen Burns [with MAAP 3.0B Rev 7.02 using "Recommended Sensitivity Analyses For An IPE Using MAAP 3.0B" (EPRI 1991)]. Grand Gulf

(Brown 1990) APET Event 110. Reference Table H.3.10-1.

EVENT 63. HYDROGEN DETONATION LATE CONTAINMENT FAILURE - H2_DET

Two Branches:

DET CF	H2 Detonation Late Containment Failure
NO	No H2 Detonation Late Containment Failure

Given that a large burn was ignited in containment late in the sequence this event assesses whether the burn transitioned to a detonation and whether containment failure resulted from the detonation impulse loading. If no large burn was ignited or if the containment atmosphere was inert to detonations (> 35% steam concentration) then no detonation was assumed to occur.

The probability of a hydrogen detonation occurring is taken to be a function of the steam concentration and the effective hydrogen concentration in containment.

EVNTRE Question Type: 2. (Dependent Split Fraction)

PROBABILITIES:

CASE 1. Large Burn Late

With no large burn late, no trigger exists for a hydrogen detonation.

DET CF	0.
NO	1.

CASE 2. Containment Steam Concentration Greater Than 35%

Steam concentrations greater than 35% prevent hydrogen detonation.

DET CF	0.
NO	1.

CASE 3. Containment Effective Hydrogen Concentration Less Than 12%

DET CF	0.
NO	1.

CASE 4. Containment Steam Concentration High And Slowly Decreasing Less Than 35% And Effective Hydrogen Concentration 12-16%

This occurs when offsite power is recovered before the containment overpressure threshold limit and sprays are available during SBO sequences, and the effective hydrogen is 12-16%.

DET CF	0.022
NO	0.978

CASE 5. Containment Steam Concentration Low
And Effective Hydrogen Concentration 12-16%

DET CF	0.
NO	1.

CASE 6. Containment Steam Concentration High
And Slowly Decreasing Less Than 35%
And Effective Hydrogen Concentration > 16%

This occurs when offsite power is recovered before the containment overpressure threshold limit and sprays are available during SBO sequences, and the effective hydrogen is > 16%.

DET CF	0.025
NO	0.975

CASE 7. Containment Steam Concentration Low
And Effective Hydrogen Concentration 16-20%

DET CF	0.16
NO	0.84

CASE 8. Containment Steam Concentration Low
And Effective Hydrogen Concentration > 20%

DET CF	0.27
NO	0.73

CASE 9. Otherwise Should Never Reach This Case.

DET CF	0.
NO	1.

Event Dependencies: Event Type (Event 3), Offsite Power Recovery Time (Event 7), Mode of RHR Spray Operation Late (Event 56), Containment Steam Concentration Late (Event 58), Containment Effective Hydrogen Concentration Late (Event [illegible]), Large H2 Burn Late (Event 62).

Quantification Basis: Grand Gulf (Brown 1990) APET Events 18, 44, 86, and 111; and Sequoyah Analysis (Volume 5 Rev 1 Part 1 Page 2.2, Part 2 Table A.3.1-1) page 15 and 16. Reference Table 4.3.10-2.

EVENT 64. Hydrogen Burn Late Containment Failure - CF

Two Branches:

FAILURE	Containment Failure Due to Late Hydrogen Burn
NO FAILURE	Containment Not Failed

This event assesses whether containment failure occurs following RPV failure as a result of a hydrogen burn. If a hydrogen detonation has occurred which fails the containment then containment failure has occurred. If a large hydrogen burn was ignited then this event compares the peak containment pressure for the burn (Parameter 5) with the containment fragility curve and determines the probability of failing the containment.

EVNTRE Question Type: 6. (Dependent event using previously defined parameters and a user function)

PROBABILITIES:

CASE 1. Hydrogen Detonation Late Containment Failure

FAILURE	1.
NO FAILURE	0.

CASE 2. Otherwise, For Those Remaining Sequences Determine the Probability of Containment Failure Due To Large Hydrogen Burns

USER FUNCTION		FAILURE	NO FAILURE
If Peak Containment Pressure	> 80 Psig	1.	0.
	> 75	0.98	0.02
	> 70	0.90	0.10
	> 65	0.69	0.31
	> 60	0.39	0.59
	> 55	0.15	0.85
	> 50	0.034	0.966
	Else	0.	1.

Event Dependencies: Hydrogen Detonation Late Containment Failure (Event 63).

Quantification Basis: Perry Nuclear Power Plant IPE Containment Capacity Analysis, and IPE Engineering Calculation - Containment Fragility.

EVENT 65. CONTAINMENT STATUS AT ACCIDENT PROGRESSION COMPLETION - CNTMT_ST

Four Branches:

EARLY CF	Early Containment Failure
LATE CF	Late Overpressure Failure
VENT	Containment Vent
NO LATE CF	Containment Not Failed or Vented

This event assesses whether containment failure occurs late in the sequence progression due gradual overpressurization from steam and non-condensible gas generation. This event also assess whether the containment was vent[illegible] prevent overpressure failure. The event pathway probabilities are a [illegible]tion of whether containment failure has already occurred earlier in the acci[illegible] whether containment heat removal is available, whether the containment is vented, and the sequence type.

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. Containment Failed At Core Damage

EARLY CF	1.
LATE CF	0.
VENT	0.
NO LATE CF	0.

CASE 2. Containment Vent Not Isolated At RPV Failure During SBO Sequences

EARLY CF	1.
LATE CF	0.
VENT	0.
NO LATE CF	0.

CASE 3. Containment Failure Before RPV Failure

EARLY CF	1.
LATE CF	0.
VENT	0.
NO LATE CF	0.

CASE 4. Containment Failure At/Near RPV Failure

EARLY CF	1.
LATE CF	0.

VENT	0.
NO LATE CF	0.

CASE 5. Containment Intact At Core Damage And A Critical ATWS Sequence

This applies to the sensitivity case of improving the Plant Emergency Instructions to control reactor power such that containment heat removal can be successful with venting.

EARLY CF	0.
LATE CF	0.
VENT	1.
NO LATE CF	0.

CASE 6. Containment Heat Removal With RHR Spray or RHR Pool Cooling and No Pool Bypass, And Dry CCI

EARLY CF	0.0
LATE CF	0.1
VENT	0.0
NO LATE CF	0.9

CASE 7. Containment Heat Removal With RHR Spray or RHR Pool Cooling and No Pool Bypass, And No Dry CCI

EARLY CF	0.
LATE CF	0.
VENT	0.
NO LATE CF	1.

CASE 8. Containment Heat Removal With RHR Pool Cooling And No Pool Bypass, And No Vent

EARLY CF	0.00
LATE CF	0.75
VENT	0.00
NO LATE CF	0.25

CASE 9. Containment Heat Removal With Vent

EARLY CF	0.
LATE CF	0.
VENT	1.
NO LATE CF	0.

CASE 10. No Containment Heat Removal With RHR, And SBO Sequences With Early and Intermediate Loss Of Injection

EARLY CF	0.
LATE CF	1.
VENT	0.
NO LATE CF	0.

CASE 11. No Containment Heat Removal With RHR And SBO Sequences With Late Injection Loss

EARLY CF	0.
LATE CF	1.
VENT	0.
NO LATE CF	0.

CASE 12. No Containment Heat Removal With RHR And Early or Intermediate Loss of Injection

EARLY CF	0.
LATE CF	1.
VENT	0.
NO LATE CF	0.

CASE 13. No Containment Heat Removal With RHR For All Other Sequences

EARLY CF	0.
LATE CF	1.
VENT	0.
NO LATE CF	0.

CASE 14. Otherwise, Should Never Reach This Case

EARLY CF	
LATE CF	
VENT	
NO LATE CF	

Event Dependencies: Containment Status At Core Damage (Event 2), Event Type (Event 3), Containment Vent Isolated At RPV Failure (Event 5), RPV Injection Failure Time (Event 6), Offsite Power Recovery Time (Event 7), Containment Heat Removal With RHR Loop (Event 8), Containment Heat Removal With Vent (Event 9), Containment Failure Before RPV Failure (Event 24), Containment Failure At/Near RPV Failure (Event 46), Pool Bypass Before/Near RPV Failure (Event

49).

Quantification Basis: IPE Engineering Calculation - MAAP Accident Progression Analysis, and engineering judgement.

EVENT 66. MODE OF LATE HYDROGEN AND SLOW OVERPRESSURE CONTAINMENT FAILURE

- LATE_CF

Two Branches:

ANCHORAGE	Containment Anchorage Failure Mode
PEN-DOM/NO CF	Containment Penetration or Dome Failure Mode, Or No Containment Failure Mode

Given that containment failure has occurred then the probability of various modes of containment failure are determined by this event. For sequences where the containment has failed by a hydrogen burn the peak containment pressure from the burn is used to estimate the probability of each failure mode. The individual fragility curves for the dominant failure modes were used to determine the conditional probabilities for each failure mode as a function of the failure pressure. The peak burn pressure is then used to determine the probability of each failure mode.

Note that this succinct sorting of failure modes into just two categories can be used to characterize penetration/dome containment failure with a question asking sequence of: 1) No Containment Failure, 2) Anchorage Containment Failure, and then 3) Containment Failure (which would provide the remaining Penetration/Dome containment failures).

EVNTRE Question Type: 6. (Dependent event using previously defined parameters and a user function)

PROBABILITIES:

CASE 1. Late Containment Failure Due To Steam Overpressure

This case is for gradual steam overpressure late.

ANCHORAGE	0.15
PEN-DOM/NO CF	0.85

CASE 2. No Containment Failure Late Due to Hydrogen Burns

This sorting case assigns the no containment failure sequences.

ANCHORAGE	0.
PEN-DOM/NO CF	1.

CASE 3. Hydrogen Detonation Containment Failure.

All detonations are considered to most likely occur in the dome region and are assigned to the penetration-dome or no containment failure.

ANCHORAGE	0.
PEN-DOM/NO CF	1.

CASE 4. Otherwise, The Remaining Sequences Are Hydrogen Deflagration Containment Failure

USER FUNCTION		ANCHORAGE	PENETRATION
If Peak Containment Pressure	> 140 Psig	1.	0.
	> 130	0.98	0.02
	> 120	0.95	0.05
	> 115	0.90	0.10
	> 110	0.85	0.15
	> 105	0.78	0.22
	> 100	0.71	0.29
	> 95	0.61	0.39
	> 90	0.51	0.49
	> 85	0.41	0.59
	> 80	0.30	0.70
	> 75	0.21	0.79
	Else	0.15	0.85

Event Dependencies: Hydrogen Burn Late Containment Failure (Event 64), Containment Status At Accident Completion (Event 65).

Quantification Basis: Perry Nuclear Power Plant IPE Containment Capacity Analysis, and IPE Engineering Calculation - Containment Failure Modes Conditional Probability.

H.3.11 POOL BYPASS LATE

This APET Group of Events 67 and 68, determines the probability of pool bypass late in the accident progression. Fission product scrubbing in the suppression pool is an effective fission product mitigation mechanism. However, if the release pathway bypasses the suppression pool this mechanism is not effective. Pool bypass may result from a number of causes. These include: 1) structural failure of the drywell, 2) excessive leakage through drywell penetrations, or loss of suppression pool water below the level of the horizontal vents or the SRV quenchers.

Drywell structural failure may result from transient over-pressurization of the drywell or wetwell resulting in a sufficiently high drywell/wetwell differential pressure to cause failure of the drywell head, ceiling or walls. Loss of suppression pool water may result from containment failure modes (such as the containment anchorage) in the pool region or as a result of interfacing systems LOCAs.

This event is dependent on the following Events in the Perry APET.

EVENT 67. DRYWELL FAILURE LATE DUE TO LATE HYDROGEN BURN IN THE CONTAINMENT - DW_LATE

Two Branches:

DW FAILURE	Drywell Failure Due To Late Hydrogen Burn
NO DW FAIL	No Drywell Failure

Given that a large hydrogen burn has occurred late in the sequence this event assesses whether drywell failure results from excessive differential pressure across the drywell boundary. If a large burn has not occurred in containment then drywell integrity is not challenged (by hydrogen combustion). For cases where a large hydrogen burn has occurred two parameters have been set which give the peak burn pressure in containment during the burn and the containment pressure prior to the burn. It is assumed that the drywell pressure remains constant during the burn in containment. Given the value of these two parameters the peak drywell differential pressure is calculated and compared against the drywell fragility curve in a user function to estimate the probability of drywell structural failure.

EVNTRE Question Type: 6. (Dependent event using previously defined parameters and a user function)

PROBABILITIES:

CASE 1. Large Hydrogen Burn Late

USER FUNCTION		DW FAILURE	NO FAILURE
If Cntmt/DW Differential Pressure	< 40 Psid	0.	1.
	< 55	0.05	0.95
	< 60	0.17	0.83
	< 65	0.27	0.73
	< 70	0.41	0.59
	< 75	0.55	0.45
	< 80	0.68	0.32
	< 85	0.79	0.21
	< 90	0.88	0.12
	< 95	0.93	0.07
	Else	1.	0.

CASE 2. Otherwise, For Remaining Sequences With No Large Burns Default To No Drywell Failure

DW FAILURE 0
NO DW FAIL 1

Event Dependencies: Large Hydrogen Burn Late (Event 62).

Quantification Basis: Perry Nuclear Power Plant IPE Containment Capacity Analysis, and IPE Engineering Calculation - Drywell Fragility.

EVENT 68. POOL BYPASS LATE - LATE_PB

Two Branches:

LATE POOL BYP Late Pool Bypass
NO LATE BYP No Late Bypass

This summary event assess the probability that pool bypass will occur late in the accident sequence. This event considers pool bypass resulting from drywell failure form hydrogen combustion in the containment (APET Event 60), pool bypass due to containment anchorage failure (APET Event 64) and pool bypass resulting from pedestal failure caused by CCI (APET Event 53).

EVNTRE Question Type: 2. (Dependent Split Fractions)

PROBABILITIES:

CASE 1. Drywell Failure Due To Late Hydrogen Burn In Containment

LATE POOL BYP 1.
NO LATE BYP 0.

CASE 2. Pool Bypass By Containment Anchorage Failure Late

LATE POOL BYP 1.
NO LATE BYP 0.

CASE 3. Pool Bypass By Pedestal Failure Due To CCI Erosion

LATE POOL BYP 1.
NO LATE BYP 0.

CASE 4. Pool Bypass By Penetration Failure Associated With High Temperature During Dry CCI

LATE POOL BYP 1.

NO LATE BYP 0.

CASE 5. Drywell Vacuum Breaker Failure Due To Large Burn When AC Power Is Available

LATE POOL BYP 0.05
NO LATE BYP 0.95

CASE 6. Otherwise, For The Remaining Sequences Pool May Be Bypassed By Other Failures

LATE POOL BYP 0.0001
NO LATE BYP 0.9999

Event Dependencies: Event Type (Event 3), Offsite Power Recovery Time (Event 7), Type Of Core Debris Concrete Interactions (Event 54), Pedestal Failure Due To Core Debris Concrete Interaction (Event 55), Large Hydrogen Burn Late (Event 62), Mode of Late Hydrogen And Overpressure Containment Failure (Event 66), Drywell Failure Due To Late Hydrogen Burn In Containment (Event 67).

Quantification Basis: Grand Gulf (Brown 1990) APET 95, and engineering judgement.

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TABLES

TABLE H.3.3-1 HYDROGEN COMBUSTION PARAMETERS BEFORE RPV FAILURE

CASE	[H2]	[STEAM]	H2 BURN EFFICIENCY	BASE PRESSURE	AICC PRESSURE	PEAK PRESSURE	LARGE BURN PROBABILITY
1	Low	---	(Small Burn At Low Hydrogen Concentration)				.0
2	---	>55%	(Steam-Inert Containment Atmosphere)				.0
3	.256	.075	.88	9	161	142	.5
4	.188	.075	.88	6	111	98	.5
5	.105	.075	.62	4	62	40	.28
6	.221	.20	.75	14	159	123	.5
7	.162	.20	.74	11	113	87	.5
8	.090	.20	.50	9	65	37	.28
9	.194	.30	.65	20	163	113	.5
10	.142	.30	.63	16	118	80	.39
11	.079	.30	.40	13	69	36	.25
12	.166	.40	.54	27	169	104	.44
13	.122	.40	.51	23	124	75	.33
14	.068	.40	.33	19	74	37	.23
15	.138	.50	.43	37	179	98	.39
16	.102	.50	.44	32	133	76	.28
17	.057	.50	.25	27	82	40	.23

TABLE H.3.3-2 PROBABILITY OF CONTAINMENT FAILURE FROM A H2 DETONATION PRIOR TO RPV FAILURE

CASE	[H2]	[STEAM]	DETONATION PROBABILITY	MEAN IMPULSE LOADING (KPA-S)	CONTAINMENT CONDITIONAL FAILURE PROBABILITY (GIVEN A H2 DET)	CONTAINMENT FAILURE PROBABILITY
1	--	--			(NO LARGE BURN IGNITED	)
2	ALL	>35%	0.	0.	0.	0.
3	< 12%	ALL	0.	0.	0.	0.
4	12-16	HI & DECREASING	.22	5.8	0.1	0.022
5	12-16	LOW	0.	0.	0.	0.
7	> 20	HI & DECREASING	.25	5.8	0.1	0.025
8	> 20	LOW	.45	12.4	0.6	0.27
6,9	16-20	LOW	.26	12.4	0.6	0.16

TABLE H.3.5-1 MAAP PREDICTION FOR PEAK PEDESTAL PRESSURE AT RPV FAILURE FOR LOOP AND SBO SEQUENCE TYPES WITH LOSS OF ALL INJECTION

MAAP RUN NUMBER	WATER IN PEDESTAL FM SPMU (FEET)	RPV PRESS	INITIAL SIZE OF RPV LOWER HEAD FAILURE (SQUARE FEET)	PEAK PEDESTAL PRESSURE (PSID)
PPD_0_01	6	HIGH	INSTRUMENT TUBE PENETRATION .023	648
PPD_0_02	6	HIGH	CONTROL ROD DRIVE PENETRATION .20	289
PPD_0_03	6	HIGH	BOTTOM RPV HEAD LARGE FAILURE 21.5	1124
PPD_0_06	0	HIGH	INST TUBE PENETRATION	78
PPD_0_07	0	HIGH	CRD DRIVE PENETRATION	48
PPD_0_08	0	HIGH	BTM HEAD LARGE FAILURE	533
PPD_0_11	0	Low	Inst Tube Penetration	2.8
PPD_0_12	0	Low	CRD Drive Penetration	1.8
PPD_0_13	0	Low	Btm Head large Failure	15.9
PPD_0_16	3	Low	Inst Tube Penetration	59
PPD_0_17	3	Low	CRD Drive Penetration	46
PPD_0_18	3	Low	Btm Head large Failure	791

TABLE H.3.6-1 HYDROGEN COMBUSTION PARAMETERS AT/NEAR RPV FAILURE

CASE	[H2]	[STEAM]	H2 BURN EFFICIENCY	BASE PRESSURE	AICC PRESS	PEAK PRESS	BURN PARM	LARGE BURN PROBABILITY
1	low	---	(Continuous Ignition Source Available)					0.
2	---	>55%	(Steam-Inert Containment Atmosphere)					0.
3	.217	.075	.88	13	165	147	A,D	1.
4	.217	.075	.88	17	169	151	H	0.63
5	.217	.075	.88	13	165	147	O	0.49
6	.188	.20	.75	20	169	131	A,D	1.
7	.188	.20	.75	24	173	135	H	0.63
8	.188	.20	.75	20	169	131	O	0.49
9	.164	.30	.64	26	173	120	A,D	1.
10	.164	.30	.64	30	177	124	H	0.63
11	.164	.30	.64	26	173	120	O	0.49
12	.141	.40	.53	35	182	113	A,D	1.
13	.141	.40	.53	39	186	117	H	0.56
14	.141	.40	.53	35	182	113	O	0.38
15	.117	.50	.43	46	193	109	A,D	1.
16	.117	.50	.43	50	197	113	H	0.43
17	.117	.50	.43	46	193	109	O	0.28
18	.157	.075	.87	11	118	104	A,D	1.
19	.157	.075	.87	15	122	108	H	0.56
20	.157	.075	.87	11	118	104	O	0.38
21	.136	.20	.73	17	123	94	A,D	1.
22	.136	.20	.73	21	127	98	H	0.56
23	.136	.20	.73	17	123	94	O	0.38
24	.119	.30	.58	23	127	83	A,D	1.
25	.119	.30	.58	27	131	87	H	0.43
26	.119	.30	.58	23	127	83	O	0.28
27	.102	.40	.48	31	135	81	A,D	1.
28	.102	.40	.48	35	139	85	H	0.43
29	.102	.40	.48	31	135	81	O	0.28
30	.085	.50	.38	41	145	81	A,D	1.
31	.085	.50	.38	45	149	85	H	0.43
32	.085	.50	.38	41	145	81	O	0.28
33	.086	.075	.50	8	68	38	A,D	1.
34	.086	.075	.50	12	72	42	H	0.43
35	.086	.075	.50	8	68	38	O	0.28
36	.074	.20	.38	14	72	36	A,D	1.
37	.074	.20	.38	18	76	40	H	0.29
38	.074	.20	.38	14	72	36	O	0.21

BURN PARAMETER KEY: A - AC Power Recovered
D - Drywell Failure
H - High Pressure Melt Ejection
O - Other, No AC Power - No Drywell Failure - No HPME

TABLE H.3.6-1 (cont) HYDROGEN COMBUSTION PARAMETERS AT/NEAR RPV FAILURE

CASE	[H2]	[STEAM]	H2 BURN EFFICIENCY	BASE PRESSURE	AICC PRESS	PEAK PRESS	BURN PARM	LARGE BURN PROBABILITY
39	.065	.30	.31	19	77	37	A,D	1.
40	.065	.30	.31	23	81	41	H	0.29
41	.065	.30	.31	19	77	37	O	0.21
42	.056	.40	.25	27	84	41	A,D	1.
43	.056	.40	.25	31	88	45	H	0.29
44	.056	.40	.25	27	84	41	O	0.21
45	.046	.50	.18	36	91	46	A,D	1.
46	.046	.50	.18	40	95	50	H	0.29
47	.046	.50	.18	36	91	46	O	0.21

BURN PARAMETER KEY: A - AC Power Recovered
D - Drywell Failure
H - High Pressure Melt Ejection
O - Other, No AC Power - No Drywell Failure - No HPME

TABLE H.3.6-2 PROBABILITY OF CONTAINMENT FAILURE FROM A H2 DETONATION AT/NEAR TO RPV FAILURE

CASE	[H2]	[Steam]	Detonation Probability	an Impulse Loading (kPa-s)	Containment Conditional Failure Probability (Given a H2 Det)	Containment Failure Probability
1	--	--			(No Large Burn Ignited)	
2	All	>35%	0.	0.	0.	0.
6	< 12%	All	0.	0.	0.	0.
7	12-16	Hi & Decreasing	.22	5.8	0.1	0.022
8	12-16	Low	0.	0.	0.	0.
3	> 20	Hi & Decreasing	.25	5.8	0.1	0.025
4,6	> 20	Low	.45	12.4	0.6	0.27
5,9	16-20	Low	.26	12.4	0.6	0.16

TABLE H.3.10-1 HYDROGEN COMBUSTION PARAMETERS AT LATE CONTAINMENT FAILURE

CASE	[H2]	[STEAM]	H2 BURN EFFICIENCY	BASE PRESSURE	AICC PRESSURE	PEAK PRESSURE	LARGE BURN PROBABILITY
1	Low	---	(Continuous Ignition Source Available)				.0
2	---	>55%	(Steam-Inert Containment Atmosphere)				.0
3	< .04	---	(Large Deflagration Burns Not Possible)				.0
[Cases 4 - 28 are for the condition: No AC Power Late]							
4	.06	.075	.28	7.6	50	19.4	.29
5	.10	.075	.62	8.8	78	51.6	.33
6	.14	.075	.86	10	106	93	.42
7	.18	.075	.88	11	135	121	.51
8	.24	.075	.88	14	187	166	.51
9	.06	.20	.28	13	61	26.7	.29
10	.10	.20	.57	15	93	59.7	.33
11	.14	.20	.73	17	126	97	.42
12	.18	.20	.75	19	161	126	.51
13	.24	.20	.75	23	219	170	.51
14	.06	.30	.28	19	72	34.1	.29
15	.10	.30	.53	21	109	67.9	.33
16	.14	.30	.63	24	149	103	.42
17	.18	.30	.65	28	191	134	.51
18	.24	.30	.65	34	229	161	.51
19	.06	.40	.28	27	88	44	.29
20	.10	.40	.48	30	133	79.6	.33
21	.14	.40	.53	35	181	112	.42
22	.18	.40	.54	40	231	143	.51
23	.24	.40	.54	50	249	157	.51
24	.06	.50	.28	38	110	58.2	.29
25	.10	.50	.44	44	167	98	.33
26	.14	.50	.43	51	230	128	.42
27	.18	.50	.44	59	256	146	.51
28	.24	.50	.44	78	283	168	.51

TABLE H.3.10-1 continued

HYDROGEN COMBUSTION PARAMETERS AT LATE CONTAINMENT FAILURE

CASE	[H2]	[STEAM]	H2 BURN EFFICIENCY	BASE PRESSURE	AICC PRESSURE	PEAK PRESSURE	LARGE BURN PROBABILITY
[Cases 29 - 54 are for the condition: AC Power Available Late]							
29	.06	.075	.28	7.6	50	19.4	1.
30	.10	.075	.62	8.8	79	51.6	1.
31	.14	.075	.86	10	106	93	1.
32	.18	.075	.88	11	135	121	1.
33	.24	.075	.88	14	187	166	1.
34	.06	.20	.28	13	61	26.7	1.
35	.10	.20	.57	15	93	59.7	1.
36	.14	.20	.73	17	126	97	1.
37	.18	.20	.75	19	161	126	1.
38	.24	.20	.75	23	219	170	1.
39	.06	.30	.28	19	72	34.1	1.
40	.10	.30	.53	21	109	67.9	1.
41	.14	.30	.63	24	149	103	1.
42	.18	.30	.65	28	191	134	1.
43	.24	.30	.65	34	229	161	1.
44	.06	.40	.28	27	88	44	1.
45	.10	.40	[illegible]	30	133	79.6	1.
46	.14	.40	.53	35	191	112	1.
47	.18	.40	.54	40	231	143	1.
48	.24	.40	.54	50	249	157	1.
50	.06	50	.28	38	110	58.2	1.
51	.10	.50	.44	44	167	98	1.
52	.14	.50	.43	51	230	128	1.
53	.18	.50	.44	59	256	146	1.
54	.24	.50	.44	78	283	168	1.

TABLE H.3.10-2 PROBABILITY OF CONTAINMENT FAILURE FROM A H2 DETONATION LATE

CASE	[H2]	[Steam]	Detonation Probability	Mean Impulse Loading (kPa-s)	Containment Conditional Failure Probability (Given a H2 Det)	Containment Failure Probability
1	--	--			(No Large Burn Ignited	)
2	All	>35%	0.	0.	0.	0.
3	< 12%	All	0.	0.	0.	0.
4	12-16	Hi & Decreasing	.22	5.8	0.1	0.022
5	12-16	Low	0.	0.	0.	0.
6	> 16	Hi & Decreasing	.25	5.8	0.1	0.025
8	> 20	Low	.45	12.4	0.6	0.27
7	16-20	Low	.26	12.4	0.6	0.16

APPENDIX H.4

PNPP IPE APET PROGRAM FREQUENCY OUTPUT FILE

The PNPP IPE Accident Progression Event Tree (APET) Program Frequency Output File from the Event Progression Analysis (EVNTRE) code with the base case input data file provides a record of how the paths through the tree were propagated. It provides useful information concerning split fractions for event branches and cases.

```
1           TREE ID: PERPY IPE APET   BASE CASE           REVISION 0 19JUN1992
    # OF QUESTIONS:   68
  STATS FOR SEQUENCE:            1
       SEQUENCE ID:           ALL SEQUENCES
      INITIAL FREQ.:          1.000E+00


  ********  QUESTION:     1  PDS PARAMETER 1  NOT A CONTAINMENT BYPASS SEQUENCE
  Q-TYPE/TIMES ASKED:        INDEP. INPUT PROB.                               1
            BRANCHES:            NoBYPASS  EVENT V
                                   1          2
      REALIZED SPLIT:          1.000E+00 0.000E+00


  ********  QUESTION:     2  PDS PARAMETER 2  CONTAINMENT STATUS AT CORE DAMAGE
  Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                                 2
            BRANCHES:            INTACT    FAILED
                                   1          2
      REALIZED SPLIT:          7.720E-01 2.280E-01

                        SUMMARY BY CASE

    CASE NUMBER/SPLIT:    1      1.000E+00
        DEPENDENCIES:     1
       REQ. BRANCHES:     1
         DESCRIPTION:   NoBYPASS
    CASE/BRANCH SPLIT:           7.720E-01 2.280E-01

    CASE NUMBER/SPLIT:    2      0.000E+00
         DESCRIPTION:        OTHERWISE
    CASE/BRANCH SPLIT:           0.000E+00 0.000E+00


   ********  QUESTION:     3  PDS PARAMETER 3  EVENT TYPE: CNTMT INTACT/FAILED AT CORE D
   Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                               5
             BRANCHES:           SBO       LOOP NO H OTHER TYP CRIT ATWS LOOP & SB OTHE
                                   1         2         3         4         5         6
       REALIZED SPLIT:         9.032E-02 0.000E+00 6.817E-01 4.435E-02 4.355E-02 1.401

                        SUMMARY BY CASE

     CASE NUMBER/SPLIT:    1      7.720E-01
         DEPENDENCIES:     2
        REQ. BRANCHES:     1
          DESCRIPTION:   INTACT
     CASE/BRANCH SPLIT:           9.032E-02 0.000E+00 6.817E-01 0.000E+00 0.000E+00 0.000

     CASE NUMBER/SPLIT:    2      2.280E-01
          DESCRIPTION:        OTHERWISE                          $ DEFAULT TO NO BRANCH
     CASE/BRANCH SPLIT:           0.000E+00 0.000E+00 0.000E+00 4.435E-02 4.355E-02 1.401


    ********  QUESTION:     4  PDS PARAMETER 4  INITIAL CNTMT HEAT REMOVAL WITH SUPP[illegible]01
    Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                               5
```

```
           BRANCHES:            IN_PL     NO_IN_PL
                                  1          2
     REALIZED SPLIT:            0.000E+00 1.000E+00

                      SUMMARY BY CASE

  CASE NUMBER/SPLIT:     1      0.000E+00
       DEPENDENCIES:     3
      REQ. BRANCHES:     2
        DESCRIPTION:   LOOP NO H
 CASE/BRANCH SPLIT:             0.000E+00 0.000E+00

  CASE NUMBER/SPLIT:     2      1.000E+00
        DESCRIPTION:        OTHERWISE                        $ NOT A LOOP NO HVAC
  CASE/BRANCH SPLIT:            0.000E+00 1.000E+00


 ********  QUESTION:     5  PDS PARAMETER 5  CONTAINMENT VENT ISOLATED AT RPV FAILURE
 Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                                   6
           BRANCHES             ISOLATED  NOT ISOL
                                  1          2
     REALIZED SPLIT:            9.997E-01 3.161E-04

                      SUMMARY BY CASE

  CASE NUMBER/SPLIT:     1      9.032E-02
       DEPENDENCIES:     3
      REQ. BRANCHES:     1
        DESCRIPTION:   SBO
  CASE/BRANCH SPLIT:            9.001E-02 3.161E-04

  CASE NUMBER/SPLIT:     2      9.097E-01
        DESCRIPTION:        OTHERWISE                        $ DEFAULT TO ISOLATED
  CASE/BRANCH SPLIT:            9.097E-01 0.000E+00


 ********  QUESTION:     6  PDS PARAMETER 6  RPV INJECTION FAILURE TIME
 Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                                  10
           BRANCHES:            NO INJECT RCIC      HPCS      NO BRANCH
                                  1          2         3          4
     REALIZED SPLIT:            9.489E-01 2.527E-02 2.578E-02 0.000E+00

                      SUMMARY BY CASE

  CASE NUMBER/SPLIT:     1      9.000E-02
       DEPENDENCIES:     3          5
      REQ. BRANCHES:     1     *    1
        DESCRIPTION:   SBO        ISOLATED
  CASE/BRANCH SPLIT:            3.913E-02 2.518E-02 2.569E-02 0.000E+00

  CASE NUMBER/SPLIT:     2      3.161E-04
       DEPENDENCIES:     3
      REQ. BRANCHES:     1
        DESCRIPTION:   SBO
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CASE/BRANCH SPLIT:          1.374E-04 8.846E-05 9.023E-05 0.000E+00

CASE NUMBER/SPLIT:     3      0.000E+00
     DEPENDENCIES:     3          4
    REQ. BRANCHES:     2    *     2
      DESCRIPTION:   LOOP NO H NO .N PL
CASE/BRANCH SPLIT:            0.000E+00 0.000E+00 0.000E+00 0.000E+00

CASE NUMBER/SPLIT:     4      0.000E+00
     DEPENDENCIES:     3          4
    REQ. BRANCHES:     2    *     1
      DESCRIPTION:   LOOP NO H IN PL
CASE/BRANCH SPLIT:            0.000E+00 0.000E+00 0.000E+00 0.000E+00

CASE NUMBER/SPLIT:     5      9.097E-01
      DESCRIPTION:         OTHERWISE                   $ DEFAULT
CASE/BRANCH SPLIT:            9.097E-01 0.000E+00 0.000E+00 0.000E+00


********  QUESTION:     7  PDS PARAMETER 7  OFFSITE POWER RECOVERY TIME
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                                20
         BRANCHES:              PRIOR RV  CNTMT LIM NO RECOV  NO BRANCH
                                   1          2         3         4
   REALIZED SPLIT:              4.132E-02 3.171E-02 9.270E-01 0.000E+00

                    SUMMARY BY CASE

CASE NUMBER/SPLIT:     1      3.926E-02
     DEPENDENCIES:     3          6
    REQ. BRANCHES:     1    *     1
      DESCRIPTION:   SBO        NO INJECT
CASE/BRANCH SPLIT:            2.414E-02 1.401E-02 1.119E-03 0.000E+00

CASE NUMBER/SPLIT:     2      2.527E-02
     DEPENDENCIES:     3          6
    REQ. BRANCHES:     1    *     2
      DESCRIPTION:   SBO        RCIC
CASE/BRANCH SPLIT:            6.278E-03 1.771E-02 1.289E-03 0.000E+00

CASE NUMBER/SPLIT:     3      2.578E-02
     DEPENDENCIES:     3          6
    REQ. BRANCHES:     1    *     3
      DESCRIPTION:   SBO        HPCS
CASE/BRANCH SPLIT:            1.091E-02 0.000E+00 1.487E-02 0.000E+00

CASE NUMBER/SPLIT:     4      9.097E-01
      DESCRIPTION:         OTHERWISE
CASE/BRANCH SPLIT:            0.000E+00 0.000E+00 9.097E-01 0.000E+00


********  QUESTION:     8  PDS PARAMETER 8  CONTAINMENT HEAT REMOVAL WITH RHR SPRAY L
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                                 27
         BRANCHES:              RHR_SPRY  RHR_POOL  NO_RHR
                                   1          2         3
```

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     REALIZED SPLIT:          4.297E-01 0.000E+00 5.703E-01

                        SUMMARY BY CASE

CASE NUMBER/SPLIT:      1     3.161E-04
     DEPENDENCIES:      3         5
    REQ. BRANCHES:      1    *    2
      DESCRIPTION:   SBO        NOT ISOL
CASE/BRANCH SPLIT:            0.000E+00 0.000E+00 3.161E-04

CASE NUMBER/SPLIT:      2     2.406E-02
     DEPENDENCIES:      3         6           7
    REQ. BRANCHES:      1    *    1     *     1
      DESCRIPTION:   SBO        NO INJECT PRIOR RV
CASE/BRANCH SPLIT:            1.999E-02 0.000E+00 4.061E-03

CASE NUMBER/SPLIT:      3     1.396E-02
     DEPENDENCIES:      3         6           7
    REQ. BRANCHES:      1    *    1     *     2
      DESCRIPTION:   SBO        NO INJECT CNTMT LIM
CASE/BRANCH SPLIT:            1.156E-02 0.000E+00 2.395E-03

CASE NUMBER/SPLIT:      4     6.256E-03
     DEPENDENCIES:      3         6           7
    REQ. BRANCHES:      1    *    2     *     1
      DESCRIPTION:   SBO        RCIC        PRIOR RV
CASE/BRANCH SPLIT:            5.930E-03 0.000E+00 3.259E-04

CASE NUMBER/SPLIT:      5     1.764E-02
     DEPENDENCIES:      3         6           7
    REQ. BRANCHES:      1    *    2     *     2
      DESCRIPTION:   SBO        RCIC        CNTMT LIM
CASE/BRANCH SPLIT:            1.638E-02 0.000E+00 1.265E-03

CASE NUMBER/SPLIT:      6     1.087E-02
     DEPENDENCIES:      3         6           7
    REQ. BRANCHES:      1    *    3     *     1
      DESCRIPTION:   SBO        HPCS        PRIOR RV
CASE/BRANCH SPLIT:            9.880E-03 0.000E+00 9.891E-04

CASE NUMBER/SPLIT:      7     6.817E-01
     DEPENDENCIES:      3
    REQ. BRANCHES:      3
      DESCRIPTION:   OTHER TYP
CASE/BRANCH SPLIT:            3.525E-01 0.000E+00 3.292E-01

CASE NUMBER/SPLIT:      8     4.435E-02
     DEPENDENCIES:      3         2
    REQ. BRANCHES:      4    *    2
      DESCRIPTION:   CRIT ATWS FAILED
CASE/BRANCH SPLIT:            1.341E-02 0.000E+00 3.093E-02

CASE NUMBER/SPLIT:      9     0.000E+00
     DEPENDENCIES:      3         2
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```
    REQ. BRANCHES:   4    *    1
      DESCRIPTION:  CRIT ATWS INTACT
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   10      2.009E-01
      DESCRIPTION:         OTHERWISE
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00 2.009E-01


********  QUESTION:    9  PDS PARAMETER 9  CONTAINMENT HEAT REMOVAL WITH VENT
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                                  34
          BRANCHES:            VENT       NO VENT
                                1          2
    REALIZED SPLIT:          2.987E-01 7.013E-01

                  SUMMARY BY CASE

CASE NUMBER/SPLIT:    1      4.145E-03
     DEPENDENCIES:    3          6          7          8
    REQ. BRANCHES:    1    *     1    *     1    *     3
      DESCRIPTION:  SBO       NO INJECT PRIOR RV  NO_RHR
CASE/BRANCH SPLIT:           4.145E-03 0.000E+00

CASE NUMBER/SPLIT:    2      2.444E-03
     DEPENDENCIES:    3          6          7          8
    REQ. BRANCHES:    1    *     1    *     2    *     3
      DESCRIPTION:  SBO       NO INJECT CNTMT LIM NO_RHR
CASE/BRANCH SPLIT:           2.444E-03 0.000E+00

CASE NUMBER/SPLIT:    3      1.119E-03
     DEPENDENCIES:    3          6          7
    REQ. BRANCHES:    1    *     1    *     3
      DESCRIPTION:  SBO       NO INJECT NO RECOV
CASE/BRANCH SPLIT:           9.709E-04 1.482E-04

CASE NUMBER/SPLIT:    4      3.479E-04
     DEPENDENCIES:    3          6          7          8
    REQ. BRANCHES:    1    *     2    *     1    *     3
      DESCRIPTION:  SBO       RCIC      PRIOR RV  NO_RHR
CASE/BRANCH SPLIT:           3.479E-04 0.000E+00

CASE NUMBER/SPLIT:    5      1.327E-03
     DEPENDENCIES:    3          6          7          8
    REQ. BRANCHES:    1    *     2    *     2    *     3
      DESCRIPTION:  SBO       RCIC      CNTMT LIM NO_RHR
CASE/BRANCH SPLIT:           1.327E-03 0.000E+00

CASE NUMBER/SPLIT:    6      1.289E-03
     DEPENDENCIES:    3          6          7
    REQ. BRANCHES:    1    *     2    *     3
      DESCRIPTION:  SBO       RCIC      NO RECOV
CASE/BRANCH SPLIT:           2.351E-04 1.054E-03

CASE NUMBER/SPLIT:    7      1.027E-03
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      DEPENDENCIES:    3         6         7         8
     REQ. BRANCHES:    1    *    3    *    1    *    3
       DESCRIPTION:  SBO        HPCS      PRIOR RV  NO_RHR
CASE/BRANCH SPLIT:          1.027E-03 0.000E+00

CASE NUMBER/SPLIT:     8     1.487E-02
      DEPENDENCIES:    3         6         7
     REQ. BRANCHES:    1    *    3    *    3
       DESCRIPTION:  SBO        HPCS      NO RECOV
CASE/BRANCH SPLIT:          1.186E-02 3.010E-03

CASE NUMBER/SPLIT:     9     3.292E-01
      DEPENDENCIES:    3         8
     REQ. BRANCHES:    3    *    3
       DESCRIPTION:  OTHER TYP NO RHR
CASE/BRANCH SPLIT:          2.763E-01 5.287E-02

CASE NUMBER/SPLIT:    10     0.000E+00
      DEPENDENCIES:    3         2
     REQ. BRANCHES:    4    *    1
       DESCRIPTION:  CRIT ATWS INTACT
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00

CASE NUMBER/SPLIT:    11     6.442E-01
       DESCRIPTION:        OTHERWISE                        $ VENTING UNNECESSARY OR
CASE/BRANCH SPLIT:          0.000E+00 6.442E-01


********  QUESTION:    10  PDS PARAMETER 10  LATE IN-VESSEL INJECT & PEDESTAL CAVITY
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                              53
          BRANCHES:             LAT INJ   NO LT INJ
                                   1          2
    REALIZED SPLIT:             7.868E-01 2.132E-01

                     SUMMARY BY CASE

CASE NUMBER/SPLIT:     1     1.999E-02
      DEPENDENCIES:    3         6         7         8
     REQ. BRANCHES:    1    *    1    *    1    *    1
       DESCRIPTION:  SBO        NO INJECT PRIOR RV  RHR_SPRY
CASE/BRANCH SPLIT:          1.973E-02 2.679E-04

CASE NUMBER/SPLIT:     2     4.145E-03
      DEPENDENCIES:    3         6         7         9
     REQ. BRANCHES:    1    *    1    *    1    *    1
       DESCRIPTION:  SBO        NO INJECT PRIOR RV  VENT
CASE/BRANCH SPLIT:          4.134E-03 1.119E-05

CASE NUMBER/SPLIT:     3     1.156E-02
      DEPENDENCIES:    3         6         7         8
     REQ. BRANCHES:    1    *    1    *    2    *    1
       DESCRIPTION:  SBO        NO INJECT CNTMT LIM RHR_SPRY
CASE/BRANCH SPLIT:          3.923E-03 7.639E-03
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CASE NUMBER/SPLIT:    4      2.444E-03
     DEPENDENCIES:    3          6           7          8
    REQ. BRANCHES:    1     *    1     *     2     *    3
      DESCRIPTION:  SBO        NO INJECT CNTMT LIM NO_RHR
CASE/BRANCH SPLIT:           1.683E-03 7.608E-04

CASE NUMBER/SPLIT:    5      9.709E-04
     DEPENDENCIES:    3          6           7          9
    REQ. BRANCHES:    1     *    1     *     3     *    1
      DESCRIPTION:  SBO        NO INJECT NO RECOV  VENT
CASE/BRANCH SPLIT:           9.090E-04 6.184E-05

CASE NUMBER/SPLIT:    6      1.482E-04
     DEPENDENCIES:    3          6           7          9
    REQ. BRANCHES:    1     *  [illegible]   3     *    2
      DESCRIPTION:  SBO        [illegible]CT NO RECOV  NO VENT
CASE/BRANCH SPLIT:           1.[illegible]04 0.000E+00

CASE NUMBER/SPLIT:    7      5.930E-03
     DEPENDENCIES:    3          6           7          8
    REQ. BRANCHES:    1     *    2     *     1     *    1
      DESCRIPTION:  SBO          RCIC      PRIOR RV  RHR_SPRY
CASE/BRANCH SPLIT:           5.930E-03 0.000E+00

CASE NUMBER/SPLIT:    8      3.479E-04
     DEPENDENCIES:    3          6           7          9
    REQ. BRANCHES:    1     *    2     *     1     *    1
      DESCRIPTION:  SBO          RCIC      PRIOR RV  VENT
CASE/BRANCH SPLIT:           3.479E-04 0.000E+00

CASE NUMBER/SPLIT:    9      1.771E-02
     DEPENDENCIES:    3          6           7
    REQ. BRANCHES:    1     *    2     *     2
      DESCRIPTION:  SBO          RCIC      CNTMT LIM
CASE/BRANCH SPLIT:           0.000E+00 1.771E-02

CASE NUMBER/SPLIT:   10      2.351E-04
     DEPENDENCIES:    3          6           7          9
    REQ. BRANCHES:    1     *    2     *     3     *
      DESCRIPTION:  SBO          RCIC      NO RECOV  VE
CASE/BRANCH SPLIT:           0.000E+00 2.351E-04

CASE NUMBER/SPLIT:   11      1.054E-03
     DEPENDENCIES:    3          6           7          9
    REQ. BRANCHES:    1     *    2     *     3     *    2
      DESCRIPTION:  SBO          RCIC      NO RECOV  NO VENT
CASE/BRANCH SPLIT:           0.000E+00 1.054E-03

CASE NUMBER/SPLIT:   12      9.880E-03
     DEPENDENCIES:    3          6           7          8
    REQ. BRANCHES:    1     *    3     *     1     *    1
      DESCRIPTION:  SBO          HPCS      PRIOR RV  RHR_SPRY
CASE/BRANCH SPLIT:           9.880E-03 0.000E+00
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CASE NUMBER/SPLIT:   13     1.027E-03
     DEPENDENCIES:    3          6          7          9
    REQ. BRANCHES:    1    *     3    *     1    *     1
      DESCRIPTION:   SBO       HPCS       PRIOR RV   VENT
CASE/BRANCH SPLIT:          1.027E-03 0.000E+00

CASE NUMBER/SPLIT:   14     1.186E-02
     DEPENDENCIES:    3          6          7          9
    REQ. BRANCHES:    1    *     3    *     3    *     1
      DESCRIPTION:   SBO       HPCS       NO RECOV   VENT
CASE/BRANCH SPLIT:          1.000E-02 1.860E-03

CASE NUMBER/SPLIT:   15     3.010E-03
     DEPENDENCIES:    3          6          7          9
    REQ. BRANCHES:    1    *     3    *     3    *     2
      DESCRIPTION:   SBO       HPCS       NO RECOV   NO VENT
CASE/BRANCH SPLIT:          2.245E-03 7.645E-04

CASE NUMBER/SPLIT:   16     0.000E+00
     DEPENDENCIES:    3          5
    REQ. BRANCHES:    1    *     2
      DESCRIPTION:   SBO       NOT ISOL
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   17     3.525E-01
     DEPENDENCIES:    3          8
    REQ. BRANCHES:    3    *     1
      DESCRIPTION    [illegible]ER TYP RHR_SPRY
CASE/BRANCH SPLIT           3.514E-01 1.093E-03

CASE NUMBER/SPLIT:   18     2.763E-01
     DEPENDENCIES:    3          9
    REQ. BRANCHES:    3    *     1
      DESCRIPTION:   OTHER TYP VENT
CASE/BRANCH SPLIT:          2.685E-01 7.847E-03

CASE NUMBER/SPLIT:   19     5.287E-02
     DEPENDENCIES:    3          9
    REQ. BRANCHES:    3    *     2
      DESCRIPTION:   OTHER TYP NO VENT
CASE/BRANCH SPLIT:          1.163E-05 5.286E-02

CASE NUMBER/SPLIT:   20     0.000E+00
     DEPENDENCIES:    3          2
    REQ. BRANCHES:    4    *     1
      DESCRIPTION:   CRIT ATWS INTACT
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   21     4.435E-02
     DEPENDENCIES:    3          2
    REQ. BRANCHES:    4    *     2
      DESCRIPTION:   CRIT ATWS FAILED
CASE/BRANCH SPLIT:          4.213E-02 2.217E-03
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CASE NUMBER/SPLIT:   22      4.355E-02
     DEPENDENCIES:    2          3
    REQ. BRANCHES:    2    *     5
      DESCRIPTION:  FAILED     LOOP & SB
CASE/BRANCH SPLIT:           2.089E-02 2.266E-02

CASE NUMBER/SPLIT:   23      1.401E-01
     DEPENDENCIES:    2          3
    REQ. BRANCHES:    2    *     6
      DESCRIPTION:  FAILED     OTHERS
CASE/BRANCH SPLIT:           4.395E-02 9.616E-02

CASE NUMBER/SPLIT:   24      0.000E+00
      DESCRIPTION:         OTHERWISE
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00


******** QUESTION:   11   PDS PARAMETER 11  RPV DEPRESSURIZED DURING CORE DAMAGE
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                                   76
         BRANCHES:             LOW PRES  HI PRES
                                  1         2
   REALIZED SPLIT:           9.745E-01 2.553E-02

                    SUMMARY BY CASE

CASE NUMBER/SPLIT:    1      4.134E-03
     DEPENDENCIES:    3          6          7          9         10
    REQ. BRANCHES:    1    *     1    *     1    *     1    *     1
      DESCRIPTION:  SBO        NO INJECT PRIOR RV  VENT       LAT INJ
CASE/BRANCH SPLIT:           1.922E-03 2.212E-03

CASE NUMBER/SPLIT:    2      1.119E-05
     DEPENDENCIES:    3          6          7          9         10
    REQ. BRANCHES:    1    *     1    *     1    *     1    *     2
      DESCRIPTION:  SBO        NO INJECT PRIOR RV  VENT       NO LT INJ
CASE/BRANCH SPLIT:           1.119E-05 0.000E+00

CASE NUMBER/SPLIT:    3      1.683E-03
     DEPENDENCIES:    3          6          7          9         10
    REQ. BRANCHES:    1    *     1    *     2    *     1    *     1
      DESCRIPTION:  SBO        NO INJECT CNTMT LIM VENT       LAT INJ
CASE/BRANCH SPLIT:           3.844E-04 1.299E-03

CASE NUMBER/SPLIT:    4      9.090E-04
     DEPENDENCIES:    3          6          7          9         10
    REQ. BRANCHES:    1    *     1    *     3    *     1    *     1
      DESCRIPTION:  SBO        NO INJECT NO RECOV  VENT       LAT INJ
CASE/BRANCH SPLIT:           9.003E-04 8.727E-06

CASE NUMBER/SPLIT:    5      6.184E-05
     DEPENDENCIES:    3          6          7          9         10
    REQ. BRANCHES:    1    *     1    *     3    *     1    *     2
      DESCRIPTION:  SBO        NO INJECT NO RECOV  VENT       NO LT INJ
CASE/BRANCH SPLIT:           6.184E-05 0.000E+00
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CASE NUMBER/SPLIT:     6       1.482E-04
     DEPENDENCIES:     3          6          7          9          10
    REQ. BRANCHES:     1     *    1     *    3     *    2     *     1
      DESCRIPTION:   SBO        NO INJECT NO RECOV  NO VENT    LAT INJ
CASE/BRANCH SPLIT:             1.482E-04 0.000E+00

CASE NUMBER/SPLIT:     7       1.054E-03
     DEPENDENCIES:     3          6          7          9          10
    REQ. BRANCHES:     1     *    2     *    3     *    2     *     2
      DESCRIPTION:   SBO        RCIC       NO RECOV  NO VENT    NO LT INJ
CASE/BRANCH SPLIT:             0.000E+00 1.054E-03

CASE NUMBER/SPLIT:     8       1.000E-02
     DEPENDENCIES:     3          6          7          9          10
    REQ. BRANCHES:     1     *    3     *    3     *    1     *     1
      DESCRIPTION:   SBO        HPCS       NO RECOV  VENT       LAT INJ
CASE/BRANCH SPLIT:             8.015E-03 1.986E-03

CASE NUMBER/SPLIT:     9       1.860E-03
     DEPENDENCIES:     3          6          7          9          10
    REQ. BRANCHES:     1     *    3     *    3     *    1     *     2
      DESCRIPTION:   SBO        HPCS       NO RECOV  VENT       NO LT INJ
CASE/BRANCH SPLIT:             1.433E-03 4.270E-04

CASE NUMBER/SPLIT:    10       2.245E-03
     DEPENDENCIES:     3          6          7          9          10
    REQ. BRANCHES:     1     *    3     *    3     *    2     *     1
      DESCRIPTION:   SBO        HPCS       NO RECOV  NO VENT    LAT INJ
CASE/BRANCH SPLIT:             2.193E-03 5.210E-05

CASE NUMBER/SPLIT:    11       7.646E-04
     DEPENDENCIES:     3          6          7          9          10
    REQ. BRANCHES:     1     *    3     *    3     *    2     *     2
      DESCRIPTION:   SBO        HPCS       NO RECOV  NO VENT    NO LT INJ
CASE/BRANCH SPLIT:             7.646E-04 0.000E+00

CASE NUMBER/SPLIT:    12       6.015E-05
     DEPENDENCIES:     3          5          10
    REQ. BRANCHES:     1     *    2     *    1
      DESCRIPTION:   SBO        NOT ISOL  LAT INJ
CASE/BRANCH SPLIT:             4.169E-05 1.845E-05

CASE NUMBER/SPLIT:    13       7.805E-05
     DEPENDENCIES:     3          5          10
    REQ. BRANCHES:     1     *    2     *    2
      DESCRIPTION:   SBO        NOT ISOL  NO LT INJ
CASE/BRANCH SPLIT:             7.805E-05 0.000E+00

CASE NUMBER/SPLIT:    14       6.730E-02
     DEPENDENCIES:     3
    REQ. BRANCHES:     1
      DESCRIPTION:   SBO
CASE/BRANCH SPLIT:             6.730E-02 0.000E+00
```

CASE NUMBER/SPLIT:	15 3.514E-01
DEPENDENCIES:	3 8 10
REQ. BRANCHES:	3 * 1 * 1
DESCRIPTION:	OTHER TYP RHR SPRY LAT INJ
CASE/BRANCH SPLIT:	3.487E-01 2.671E-03

CASE NUMBER/SPLIT:	16 2.685E-01
DEPENDENCIES:	3 8 9 10
REQ. BRANCHES:	3 * 3 * 1 * 1
DESCRIPTION:	OTHER TYP NO RHR VENT LAT INJ
CASE/BRANCH SPLIT:	2.671E-01 1.342E-03

CASE NUMBER/SPLIT:	17 1.163E-05
DEPENDENCIES:	3 8 9 10
REQ. BRANCHES:	3 * 3 * 2 * 1
DESCRIPTION:	OTHER TYP NO RHR NO VENT LAT INJ
CASE/BRANCH SPLIT:	1.163E-05 0.000E+00

CASE NUMBER/SPLIT:	18 5.286E-02
DEPENDENCIES:	3 8 9 10
REQ. BRANCHES:	3 * 3 * 2 * 2
DESCRIPTION:	OTHER TYP NO RHR NO VENT NO LT INJ
CASE/BRANCH SPLIT:	4.892E-02 3.938E-03

CASE NUMBER/SPLIT:	19 0.000E+00
DEPENDENCIES:	3 2
REQ. BRANCHES:	4 * 1
DESCRIPTION:	CRIT ATWS INTACT
CASE/BRANCH SPLIT:	0.000E+00 0.000E+00

CASE NUMBER/SPLIT:	20 1.341E-02
DEPENDENCIES:	2 3 8
REQ. BRANCHES:	2 * 4 * 1
DESCRIPTION:	FAILED CRIT ATWS RHR SPRY
CASE/BRANCH SPLIT:	1.330E-02 1.100E-04

CASE NUMBER/SPLIT:	21 3.093E-02
DEPENDENCIES:	2 3 8
REQ. BRANCHES:	2 * 4 * 3
DESCRIPTION:	FAILED CRIT ATWS NO RHR
CASE/BRANCH SPLIT:	2.307E-02 7.863E-03

CASE NUMBER/SPLIT:	22 2.089E-02
DEPENDENCIES:	2 3 10
REQ. BRANCHES:	2 * 5 * 1
DESCRIPTION:	FAILED LOOP & SB LAT INJ
CASE/BRANCH SPLIT:	2.088E-02 1.253E-05

CASE NUMBER/SPLIT:	23 2.266E-02
DEPENDENCIES:	2 3 10
REQ. BRANCHES:	2 * 5 * 2
DESCRIPTION:	FAILED LOOP & SB NO LT INJ
CASE/BRANCH SPLIT:	2.013E-02 2.533E-03

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CASE NUMBER/SPLIT:   24     1.490E-01
     DEPENDENCIES:    2         2
    REQ. BRANCHES:    2    +    1
      DESCRIPTION:  FAILED    INTACT
CASE/BRANCH SPLIT:          1.490E-01 0.000E+00

CASE NUMBER/SPLIT:   25     0.000E+00
      DESCRIPTION:        OTHERWISE
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00


********  QUESTION:   12  LATE LOW PRESSURE RPV INJECTION AVAILABLE
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                            76
         BRANCHES:            WATER_INJ NO_INJECT CRITICAL
                                 1         2         3
   REALIZED SPLIT:          7.447E-01 2.132E-01 4.213E-02

                    SUMMARY BY CASE

CASE NUMBER/SPLIT:    1     2.132E-01
     DEPENDENCIES:   10
    REQ. BRANCHES:    2
      DESCRIPTION:  NO LT INJ
CASE/BRANCH SPLIT:          0.000E+00 2.132E-01 0.000E+00

CASE NUMBER/SPLIT:    2     4.213E-02
     DEPENDENCIES:    3
    REQ. BRANCHES:    4
      DESCRIPTION:  CRIT ATWS
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 4.213E-02

CASE NUMBER/SPLIT:    3     7.447E-01
     DEPENDENCIES:   10
    REQ. BRANCHES:    1
      DESCRIPTION:  LAT INJ
CASE/BRANCH SPLIT:          7.447E-01 0.000E+00 0.000E+00

CASE NUMBER/SPLIT:    4     0.000E+00
      DESCRIPTION:        OTHERWISE                         $ SHOULD NEVER REACH THIS
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 0.000E+00


********  QUESTION:   13  RPV DEPRESSURIZED DURING CORE DAMAGE
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                            76
         BRANCHES:            LOW_PRES  HI_PRES
                                 1         2
   REALIZED SPLIT:          9.745E-01 2.553E-02

                    SUMMARY BY CASE

CASE NUMBER/SPLIT:    1     9.745E-01
     DEPENDENCIES:   11
    REQ. BRANCHES:    1
```

```
         DESCRIPTION:  LOW PRES
   CASE/BRANCH SPLIT:            9.745E-C1 0.000E+C0

   CASE NUMBER/SPLIT:    2       2.553E-02
         DESCRIPTION:      OTHERWISE                          $ RPV HAS NOT BEEN DEPRES
   CASE/BRANCH SPLIT:            0.000E+00 2.553E-02


******** QUESTION:   14  DEBRIS MASS MOLTEN AT RPV FAILURE
 Q-TYPE/TIMES ASKED:      DEP. INPUT PROB.                          152
           BRANCHES:             LG_DEB    SM_DEB
                                   1         2
      REALIZED SPLIT:            4.228E-02 9.577E-01

                    SUMMARY BY CASE

   CASE NUMBER/SPLIT:    1       2.304E-01
        DEPENDENCIES:   12          13
       REQ. BRANCHES:    2   +      2
         DESCRIPTION:  NO_INJECT HI_PRES
   CASE/BRANCH SPLIT:            2.304E-02 2.073E-01

   CASE NUMBER/SPLIT:    2       7.696E-01
         DESCRIPTION:      OTHERWISE                          $ WATER INJECTION AVAILAB
   CASE/BRANCH SPLIT:            1.924E-02 7.504E-01


******** QUESTION:   15  DEBRIS COOLED IN-VESSEL - INV_COOL
 Q-TYPE/TIMES ASKED:      DEP. INPUT PROB.                          202
           BRANCHES:             COOL_INV  NCOOL_INV
                                   1         2
      REALIZED SPLIT:            5.467E-01 4.533E-01

                    SUMMARY BY CASE

   CASE NUMBER/SPLIT:    1       4.213E-02
        DEPENDENCIES:   12          2
       REQ. BRANCHES:    3   *      2
         DESCRIPTION:  CRITICAL  FAILED
   CASE/BRANCH SPLIT:            0.000E+00 4.213E-02

   CASE NUMBER/SPLIT:    2       0.000E+00
        DEPENDENCIES:   12
       REQ. BRANCHES:    3
         DESCRIPTION:  CRITICAL
   CASE/BRANCH SPLIT:            0.000E+00 0.000E+00

   CASE NUMBER/SPLIT:    3       1.838E-02
        DEPENDENCIES:   12          13          14
       REQ. BRANCHES:    1   *      1    *      1
         DESCRIPTION:  WATER_INJ LOW_PRES  LG_DEB
   CASE/BRANCH SPLIT:            9.189E-03 9.189E-03

   CASE NUMBER/SPLIT:    4       7.167E-01
```

```
      DEPENDENCIES:   12          13          14
     REQ. BRANCHES:    1    *      1    *      2
       DESCRIPTION:  WATER_INJ LOW_PRES  SM_DEB
 CASE/BRANCH SPLIT:            5.375E-01 1.792E-01

 CASE NUMBER/SPLIT:    5       2.228E-01
       DESCRIPTION:        OTHERWISE                        $ ALL OTHER CASES
 CASE/BRANCH SPLIT:            0.000E+00 2.228E-01


********  QUESTION:   16  HYDROGEN IGNITION SYSTEM AVAILABLE
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                          246
          BRANCHES:            HIS_OFF   HIS_ON
                                1         2
    REALIZED SPLIT:            1.382E-01 8.618E-01

                     SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1       8.661E-01
      DEPENDENCIES:    3            3
     REQ. BRANCHES:   /1    *      /5
       DESCRIPTION: /SBO         /LOOP & SB
 CASE/BRANCH SPLIT:            4.331E-03 8.618E-01

 CASE NUMBER/SPLIT:    2       1.339E-01
       DESCRIPTION:        OTHERWISE                        $ LOSS OF AC POWER
 CASE/BRANCH SPLIT:            1.339E-01 0.000E+00


********  QUESTION:   17  CONTAINMENT VENT ISOLATED BEFORE RPV FAILURE
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                          246
          BRANCHES:            ISOLATED  NOT_ISOL
                                1         2
    REALIZED SPLIT:            9.997E-01 3.161E-04

                     SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1       9.997E-01
      DEPENDENCIES:    5
     REQ. BRANCHES:    1
       DESCRIPTION:  ISOLATED
 CASE/BRANCH SPLIT:            9.997E-01 0.000E+00

 CASE NUMBER/SPLIT:    2       3.161E-04
       DESCRIPTION:        OTHERWISE                        $ SBO WITH CNTMT VENT NOT
 CASE/BRANCH SPLIT:            0.000E+00 3.161E-04


********  QUESTION:   18  MODE OF RHR SPRAY OPERATION EARLY
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                          246
          BRANCHES:            CONTROLD  SPRAY     NO_SPRAY
                                1         2         3
    REALIZED SPLIT:            0.000E+00 4.017E-01 5.983E-01
```

```
              SUMMARY BY CASE

CASE NUMBER/SPLIT:     1     5.703E-01
     DEPENDENCIES:     8
    REQ. BRANCHES:    /1
      DESCRIPTION: /RHR_SPRY
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00 5.703E-01

CASE NUMBER/SPLIT:     2     2.794E-02
     DEPENDENCIES:     3        7
    REQ. BRANCHES:     1    *   /1
      DESCRIPTION:   SBO       /PRIOR RV
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00 2.794E-02

CASE NUMBER/SPLIT:     3     3.659E-01
     DEPENDENCIES:     3        8
    REQ. BRANCHES:    /1    *    1
      DESCRIPTION: /SBO        RHR_SPRY
CASE/BRANCH SPLIT:           0.000E+00 3.659E-01 0.000E+00

CASE NUMBER/SPLIT:     4     3.580E-02
     DEPENDENCIES:     3        7          8
    REQ. BRANCHES:     1    *   1     *    1
      DESCRIPTION:   SBO       PRIOR RV  RHR_SPRY
CASE/BRANCH SPLIT:           0.000E+00 3.580E-02 0.000E+00

CASE NUMBER/SPLIT:     5     0.000E+00
      DESCRIPTION:         OTHERWISE                     $ SHOULD NEVER GO THIS FA
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00 0.000E+00


********  QUESTION:    19  CONTAINMENT STEAM CONCENTRATION BEFORE RPV FAILURE
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                              270
          BRANCHES:            0-15%     15-25%    25-35%    35-45%    45-55%    > 55
                                 1         2         3         4         5         6
    REALIZED SPLIT:           7.502E-01 9.865E-03 9.478E-03 0.000E+00 0.000E+00 2.305

              SUMMARY BY CASE

CASE NUMBER/SPLIT:     1     4.017E-01
     DEPENDENCIES:    18
    REQ. BRANCHES:     2
      DESCRIPTION:   SPRAY
CASE/BRANCH SPLIT:           4.017E-01 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000

CASE NUMBER/SPLIT:     2     1.927E-02
     DEPENDENCIES:     3        6
    REQ. BRANCHES:     1    *   1
      DESCRIPTION:   SBO       NO INJECT
CASE/BRANCH SPLIT:           1.927E-02 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000

CASE NUMBER/SPLIT:     3     1.934E-02
     DEPENDENCIES:     3        6
    REQ. BRANCHES:     1    *   2
```

```
        DESCRIPTION:  SBO       RCIC
  CASE/BRANCH SPLIT:            0.000E+00 9.865E-03 9.478E-03 0.000E+00 0.000E+00 0.000

  CASE NUMBER/SPLIT:   4       1.590E-02
       DEPENDENCIES:   3          6
      REQ. BRANCHES:   1    *     3
        DESCRIPTION:  SBO       HPCS
  CASE/BRANCH SPLIT:            0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.590

  CASE NUMBER/SPLIT:   5       2.146E-01
       DEPENDENCIES:   2
      REQ. BRANCHES:   2
        DESCRIPTION:  FAILED
  CASE/BRANCH SPLIT:            0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 2.146

  CASE NUMBER/SPLIT:   6       3.292E-01
        DESCRIPTION:        OTHERWISE                      $ FOR ALL OTHER EVENTS
  CASE/BRANCH SPLIT:            3.292E-01 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000

******** QUESTION:   20  FRACTION ZIRCONIUM INVENTORY REACTED IN-VESSEL
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                                   421
          BRANCHES:             33%       22%       11%
                                 1         2         3
     REALIZED SPLIT:            0.000E+00 1.215E-01 8.785E-01

                     SUMMARY BY CASE

  CASE NUMBER/SPLIT:   1       3.926E-02
       DEPENDENCIES:   3          6
      REQ. BRANCHES:   1    *     1
        DESCRIPTION:  SBO       NO INJECT
  CASE/BRANCH SPLIT:            0.000E+00 0.000E+00 3.926E-02

  CASE NUMBER/SPLIT:   2       2.527E-02
       DEPENDENCIES:   3          6
      REQ. BRANCHES:   1    *     2
        DESCRIPTION:  SBO       RCIC
  CASE/BRANCH SPLIT:            0.000E+00 3.285E-03 2.199E-02

  CASE NUMBER/SPLIT:   3       2.578E-02
       DEPENDENCIES:   3          6
      REQ. BRANCHES:   1    *     3
        DESCRIPTION:  SBO       HPCS
  CASE/BRANCH SPLIT:            0.000E+00 0.000E+00 2.578E-02

  CASE NUMBER/SPLIT:   4       9.097E-01
        DESCRIPTION:        OTHERWISE                      $ BOUND OTHERS WITH DISTR
  CASE/BRANCH SPLIT:            0.000E+00 1.183E-01 7.914E-01

******** QUESTION:   21  SMALL HYDROGEN BURNS AT LOW H2 CONCENTRATION
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                                   546
          BRANCHES:             NO_SMALL  SMALL_BRN
```

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                            1        2
    REALIZED SPLIT:           3.042E-01 6.958E-01

                     SUMMARY BY CASE

  CASE NUMBER/SPLIT:    1     2.305E-01
       DEPENDENCIES:   19
      REQ. BRANCHES:    6
        DESCRIPTION:  > 55%
  CASE/BRANCH SPLIT:          2.305E-01 0.000E+00

  CASE NUMBER/SPLIT:    2     6.916E-01
       DEPENDENCIES:   16
      REQ. BRANCHES:    2
        DESCRIPTION:  HIS_ON
  CASE/BRANCH SPLIT:          0.000E+00 6.916E-01

  CASE NUMBER/SPLIT:    3     3.475E-03
       DEPENDENCIES:    3        16
      REQ. BRANCHES:   /1    *    1
        DESCRIPTION: /SBO        HIS_OFF
  CASE/BRANCH SPLIT:          8.689E-04 2.607E-03

  CASE NUMBER/SPLIT:    4     4.030E-02
       DEPENDENCIES:    3         7
      REQ. BRANCHES:    1    *    1
        DESCRIPTION:  SBO        PRIOR RV
  CASE/BRANCH SPLIT:          3.869E-02 1.612E-03

  CASE NUMBER/SPLIT:    5     3.412E-02
       DEPENDENCIES:    3         7
      REQ. BRANCHES:    1    *   /1
        DESCRIPTION:  SBO       /PRIOR RV
  CASE/BRANCH SPLIT:          3.412E-02 0.000E+00

  CASE NUMBER/SPLIT:    6     0.000E+00
        DESCRIPTION:       OTHERWISE                          $ SHOULD NEVER GO THIS PA
  CASE/BRANCH SPLIT:          0.000E+00 0.000E+00


 ********  QUESTION:   22  LARGE H2 BURN DURING CORE DAMAGE
 Q-TYPE/TIMES ASKED:       DEP. INPUT PROB. INPUT PARM.                761
           BRANCHES:            NO_BURN   LG_BURN
                                 1         2
     REALIZED SPLIT:          9.790E-01 2.098E-02

                     SUMMARY BY CASE

  CASE NUMBER/SPLIT:    1     2.280E-01
       DEPENDENCIES:    2
      REQ. BRANCHES:    2
        DESCRIPTION:  FAILED
  CASE/BRANCH SPLIT:          2.280E-01 0.000E+00
```

```
CASE NUMBER/SPLIT:    2      6.824E-01
     DEPENDENCIES:   21
    REQ. BRANCHES:    2
      DESCRIPTION:  SMALL_BRN
CASE/BRANCH SPLIT:           6.824E-01 0.000E+00

CASE NUMBER/SPLIT:    3      1.590E-02
     DEPENDENCIES:   19
    REQ. BRANCHES:    6
      DESCRIPTION:  > 55%
CASE/BRANCH SPLIT:           1.590E-02 0.000E+00

CASE NUMBER/SPLIT:    4      0.000E+00
     DEPENDENCIES:   19         20
    REQ. BRANCHES:    1    *     1
      DESCRIPTION:  0-15%      33%
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

CASE NUMBER/SPLIT:    5      8.508E-04
     DEPENDENCIES:   19         20
    REQ. BRANCHES:    1    *     2
      DESCRIPTION:  0-15%      22%
CASE/BRANCH SPLIT:           4.254E-04 4.254E-04

CASE NUMBER/SPLIT:    6      5.348E-02
     DEPENDENCIES:   19         20
    REQ. BRANCHES:    1    *     3
      DESCRIPTION:  0-15%      11%
CASE/BRANCH SPLIT:           3.850E-02 1.497E-02

CASE NUMBER/SPLIT:    7      0.000E+00
     DEPENDENCIES:   19         20
    REQ. BRANCHES:    2    *     1
      DESCRIPTION:  15-25%     33%
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

CASE NUMBER/SPLIT:    8      1.282E-03
     DEPENDENCIES:   19         20
    REQ. BRANCHES:    2    *     2
      DESCRIPTION:  15-25%     22%
CASE/BRANCH SPLIT:           6.408E-04 6.408E-04

CASE NUMBER/SPLIT:    9      8.576E-03
     DEPENDENCIES:   19         20
    REQ. BRANCHES:    2    *     3
      DESCRIPTION:  15-25%     11%
CASE/BRANCH SPLIT:           6.175E-03 2.401E-03

CASE NUMBER/SPLIT:   10      0.000E+00
     DEPENDENCIES:   19         20
    REQ. BRANCHES:    3    *     1
      DESCRIPTION:  25-35%     33%
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00
```

```
CASE NUMBER/SPLIT:   11      1.231E-03
     DEPENDENCIES:   19         20
    REQ. BRANCHES:    3    *     2
      DESCRIPTION:  25-35%     22%
CASE/BRANCH SPLIT:           7.511E-04 4.802E-04

CASE NUMBER/SPLIT:   12      8.240E-03
     DEPENDENCIES:   19         20
    REQ. BRANCHES:    3    *     3
      DESCRIPTION:  25-35%     11%
CASE/BRANCH SPLIT:           6.180E-03 2.060E-03

CASE NUMBER/SPLIT:   13      0.000E+00
     DEPENDENCIES:   19         20
    REQ. BRANCHES:    4    *     1
      DESCRIPTION:  35-45%     33%
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   14      0.000E+00
     DEPENDENCIES:   19         20
    REQ. BRANCHES:    4    *     2
      DESCRIPTION:  35-45%     22%
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   15      0.000E+00
     DEPENDENCIES:   19         20
    REQ. BRANCHES:    4    *     3
      DESCRIPTION:  35-45%     11%
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   16      0.000E+00
     DEPENDENCIES:   19         20
    REQ. BRANCHES:    5    *     1
      DESCRIPTION:  45-55%     33%
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   17      0.000E+00
     DEPENDENCIES:   19         20
    REQ. BRANCHES:    5    *     2
      DESCRIPTION:  45-55%     22%
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   18      0.000E+00
     DEPENDENCIES:   19         20
    REQ. BRANCHES:    5    *     3
      DESCRIPTION:  45-55%     11%
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   19      0.000E+00
      DESCRIPTION:        OTHERWISE                        $ SHOULD NEVER GO THIS PA
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00


********  QUESTION:    23  H2 DETONATION CONTAINMENT FAILURE
```

```
Q-TYPE/TIMES ASKED:      DEP. INPUT PROB.
          BRANCHES:          DET_CF     NO
                               1         2
    REALIZED SPLIT:          1.195E-04 9.999E-01

                    SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1     9.790E-01
     DEPENDENCIES:    22
     REQ. BRANCHES:    1
       DESCRIPTION:  NO_BURN
 CASE/BRANCH SPLIT:          0.000E+00 9.790E-01

 CASE NUMBER/SPLIT:    2     0.000E+00
     DEPENDENCIES:    19        19        19
     REQ. BRANCHES:    4    +    5    +    6
       DESCRIPTION:  35-45%    45-55%    > 55%
 CASE/BRANCH SPLIT:          0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    3     1.943E-02
     DEPENDENCIES:    20
     REQ. BRANCHES:    3
       DESCRIPTION:  11%
 CASE/BRANCH SPLIT:          0.000E+00 1.943E-02

 CASE NUMBER/SPLIT:    4     3.700E-04
     DEPENDENCIES:    19        20        3         7        18        6
     REQ. BRANCHES:    1    *    2    *    1    *    1    *    2    *   /1
       DESCRIPTION:  0-15%      22%      SBO      PRIOR RV  SPRAY    /NO INJECT
 CASE/BRANCH SPLIT:          8.141E-06 3.619E-04

 CASE NUMBER/SPLIT:    5     4.802E-04
     DEPENDENCIES:    19        20
     REQ. BRANCHES:    3    *    2
       DESCRIPTION:  25-35%     22%
 CASE/BRANCH SPLIT:          0.000E+00 4.802E-04

 CASE NUMBER/SPLIT:    6     6.961E-04
     DEPENDENCIES:    19        19        20
     REQ. BRANCHES: (  1    +    2)   *    2
       DESCRIPTION:  0-15%     15-25%    22%
 CASE/BRANCH SPLIT:          1.114E-04 5.848E-04

 CASE NUMBER/SPLIT:    7     0.000E+00
     DEPENDENCIES:    19        20        3         7        18        6
     REQ. BRANCHES:    1    *    1    *    1    *    1    *    2    *   /1
       DESCRIPTION:  0-15%      33%      SBO      PRIOR RV  SPRAY    /NO INJECT
 CASE/BRANCH SPLIT:          0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    8     0.000E+00
     DEPENDENCIES:    19        19        20
     REQ. BRANCHES: (  1    +    2)   *    1
       DESCRIPTION:  0-15%     15-25%    33%
 CASE/BRANCH SPLIT:          0.000E+00 0.000E+00
```

```
CASE NUMBER/SPLIT:    9      0.000E+00
     DEPENDENCIES:   19         20
    REQ. BRANCHES:    3    *     1
      DESCRIPTION:  25-35%    33%
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   10      0.000E+00
      DESCRIPTION:        OTHERWISE                    $ SHOULD NEVER REACH THIS
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00


********  QUESTION:   24  CONTAINMENT FAILURE BEFORE RPV FAILURE
Q-TYPE/TIMES ASKED:        DEP. CALC. PROB.                              824
          BRANCHES:            FAILURE   NO_FAILUR
                                 1          2
   REALIZED SPLIT:           2.295E-01 7.705E-01

                    SUMMARY BY CASE

CASE NUMBER/SPLIT:    1      2.280E-01
     DEPENDENCIES:    2
    REQ. BRANCHES:    2
      DESCRIPTION:  FAILED
CASE/BRANCH SPLIT:           2.280E-01 0.000E+00

CASE NUMBER/SPLIT:    2      1.195E-04
     DEPENDENCIES:   23
    REQ. BRANCHES:    1
      DESCRIPTION:  DET_CF
CASE/BRANCH SPLIT:           1.195E-04 0.000E+00

CASE NUMBER/SPLIT:    3      7.719E-01
      DESCRIPTION:        OTHERWISE                    $ CONTAINMENT INTACT PRIO
CASE/BRANCH SPLIT:           1.417E-03 7.705E-01


********  QUESTION:   25  MODE OF CONTAINMENT FAILURE BEFORE RPV FAILURE
Q-TYPE/TIMES ASKED:        DEP. CALC. PROB.                             1007
          BRANCHES:            ANCHORAGE PN-D/NoCF
                                 1          2
   REALIZED SPLIT:           3.477E-02 9.652E-01

                    SUMMARY BY CASE

CASE NUMBER/SPLIT:    1      2.280E-01
     DEPENDENCIES:    2
    REQ. BRANCHES:    2
      DESCRIPTION:  FAILED
CASE/BRANCH SPLIT:           3.420E-02 1.938E-01

CASE NUMBER/SPLIT:    2      1.195E-04
     DEPENDENCIES:   23
    REQ. BRANCHES:    1
```

```
       DESCRIPTION:  DET_CF
 CASE/BRANCH SPLIT:            0.000E+00 1.195E-04

 CASE NUMBER/SPLIT:    3     7.705E-01
      DEPENDENCIES:   24
      REQ. BRANCHES:   2
       DESCRIPTION:  NO_FAILUR
 CASE/BRANCH SPLIT:            0.000E+00 7.705E-01

 CASE NUMBER/SPLIT:    4     1.417E-03
       DESCRIPTION:      OTHERWISE                       $ CONTAINMENT FAILED BY H
 CASE/BRANCH SPLIT:            5.686E-04 8.486E-04


******** QUESTION:   26  CNTMT FAILURE BEFORE RPV FAILURE IMPACT ON ECCS INJECTION
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                              1127
          BRANCHES:            NO_FAILUR FAILUR
                                 1         2
    REALIZED SPLIT:            9.935E-01 6.499E-03

                     SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1     7.221E-03
      DEPENDENCIES:   25          2          3
     REQ. BRANCHES:    1    *(    1     +    4)
       DESCRIPTION:  ANCHORAGE INTACT    CRIT ATWS
 CASE/BRANCH SPLIT:            7.221E-04 6.499E-03

 CASE NUMBER/SPLIT:    2     9.928E-01
       DESCRIPTION:      OTHERWISE                        $ -- NOT ANCHORAGE CNTMT
 CASE/BRANCH SPLIT:            9.928E-01 0.000E+00


******** QUESTION:   27  CNTMT FAILURE BEFORE RPV FAILURE IMPACT ON ECCS INJECTION
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                              1763
          BRANCHES:            NO_FAILUR FAILUR
                                 1         2
    REALIZED SPLIT:            9.654E-01 3.462E-02

                     SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1     3.938E-02
      DEPENDENCIES:   24          26          2          3
     REQ. BRANCHES:    1     *    1     *(    1     +    4)
       DESCRIPTION:  FAILURE   NO_FAILUR INTACT    CRIT ATWS
 CASE/BRANCH SPLIT:            1.969E-02 1.969E-02

 CASE NUMBER/SPLIT:    2     2.986E-01
      DEPENDENCIES:    9          5
     REQ. BRANCHES:    1    +     2
       DESCRIPTION:  VENT      NOT ISOL
 CASE/BRANCH SPLIT:            2.837E-01 1.493E-02

 CASE NUMBER/SPLIT:    3     6.620E-01
```

```
          DESCRIPTION:        OTHERWISE                          $ NO CF, NO VENTING AND N
    CASE/BRANCH SPLIT:           6.620E-01 0.000E+00

********  QUESTION:    28  CNTMT FAIL BEFORE RPV FAILURE STEAM & RADIATION RELEASE IM
  Q-TYPE/TIMES ASKED:         DEP. INPUT PROB.                                  2935
             BRANCHES:            NO_FAILUR FAILUR
                                     1         2
      REALIZED SPLIT:            9.504E-01 4.955E-02

                       SUMMARY BY CASE

    CASE NUMBER/SPLIT:     1      3.938E-02
        DEPENDENCIES:     24          26          2          3
        REQ. BRANCHES:     1    *     1     *(    1    +     4)
          DESCRIPTION:   FAILURE   NO FAILUR INTACT     CRIT ATWS
    CASE/BRANCH SPLIT:           1.969E-02 1.969E-02

    CASE NUMBER/SPLIT:     2      2.986E-01
        DEPENDENCIES:      9           5
        REQ. BRANCHES:     1    +      2
          DESCRIPTION:   VENT       NOT ISOL
    CASE/BRANCH SPLIT:           2.687E-01 2.986E-02

    CASE NUMBER/SPLIT:     3      6.620E-01
          DESCRIPTION:        OTHERWISE                          $ NO CF, NO VENTING AND N
    CASE/BRANCH SPLIT:           6.620E-01 0.000E+00

********  QUESTION:    29  INJECT & SPRAY FAILURE DUE TO CONTAINMENT FAILURE BEFORE R
  Q-TYPE/TIMES ASKED:         DEP. INPUT PROB.                                  2935
             BRANCHES:            NO_FAILUR INJ&SPY_F
                                     1         2
      REALIZED SPLIT:            9.822E-01 1.784E-02

                       SUMMARY BY CASE

    CASE NUMBER/SPLIT:     1      6.499E-03
        DEPENDENCIES:     26
        REQ. BRANCHES:     2
          DESCRIPTION:   FAILUR
    CASE/BRANCH SPLIT:           0.000E+00 6.499E-03

    CASE NUMBER/SPLIT:     2      1.134E-02
        DEPENDENCIES:     27          28
        REQ. BRANCHES:     2    *     2
          DESCRIPTION:   FAILUR     FAILUR
    CASE/BRANCH SPLIT:           0.000E+00 1.134E-02

    CASE NUMBER/SPLIT:     3      9.822E-01
          DESCRIPTION:        OTHERWISE                          $ ALL INJECTION NOT FAILE
    CASE/BRANCH SPLIT:           9.822E-01 0.000E+00
```

```
******** QUESTION:   30  ALPHA MODE STEAM EXPLOSION DRYWELL AND CONTAINMENT FAILURE
Q-TYPE/TIMES ASKED:      DEP. INPUT PROB.                                4379
         BRANCHES:           ALPHA      NO_ALPHA
                               1          2
   REALIZED SPLIT:           9.770E-03 9.902E-01

                      SUMMARY BY CASE

 CASE NUMBER/SPLIT:   1      9.745E-01
      DEPENDENCIES:  13
     REQ. BRANCHES:   1
       DESCRIPTION:  LOW_PRES
 CASE/BRANCH SPLIT:          9.745E-03 9.647E-01

 CASE NUMBER/SPLIT:   2      2.553E-02
       DESCRIPTION:      OTHERWISE                          $ REACTOR VESSEL NOT DEPR
 CASE/BRANCH SPLIT:          2.546E-05 2.550E-02

******** QUESTION:   31  MODE OF IN-VESSEL STEAM EXPLOSION BOTTOM HEAD FAILURE
Q-TYPE/TIMES ASKED:      DEP. INPUT PROB.                                7864
         BRANCHES:           ALPHA      NO_FAIL    LARGE_VF   SMALL_VF
                               1          2          3          4
   REALIZED SPLIT:           9.770E-03 9.322E-01 3.290E-02 2.516E-02

                      SUMMARY BY CASE

 CASE NUMBER/SPLIT:   1      9.770E-03
      DEPENDENCIES:  30
     REQ. BRANCHES:   1
       DESCRIPTION:  ALPHA
 CASE/BRANCH SPLIT:          9.770E-03 0.000E+00 0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:   2      9.647E-01
      DEPENDENCIES:  13
     REQ. BRANCHES:   1
       DESCRIPTION:  LOW_PRES
 CASE/BRANCH SPLIT:          0.000E+00 9.069E-01 3.280E-02 2.508E-02

 CASE NUMBER/SPLIT:   3      2.550E-02
       DESCRIPTION:      OTHERWISE                          $ NO ALPHA FAILURE & RPV
 CASE/BRANCH SPLIT:          0.000E+00 2.532E-02 1.019E-04 7.643E-05

******** QUESTION:   32  RPV FAILURE MODE AND SIZE OF RPV FAILURE
Q-TYPE/TIMES ASKED:      DEP. INPUT PROB.                                9753
         BRANCHES:           ALPHA      NO_FAIL    LARGE_VF   SMALL_VF
                               1          2          3
   REALIZED SPLIT:           9.770E-03 5.076E-01 7.536E-02 4.073E-01

                      SUMMARY BY CASE

 CASE NUMBER/SPLIT:   1      9.770E-03
      DEPENDENCIES:  30
```

```
     REQ. BRANCHES:    1
       DESCRIPTION:  ALPHA
 CASE/BRANCH SPLIT:          9.770E-03 0.000E+00 0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    2     3.290E-02
      DEPENDENCIES:   31
     REQ. BRANCHES:    3
       DESCRIPTION:  LARGE_VF
 CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 3.290E-02 0.000E+00

 CASE NUMBER/SPLIT:    3     2.516E-02
      DEPENDENCIES:   31
     REQ. BRANCHES:    4
       DESCRIPTION:  SMALL_VF
 CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 0.000E+00 2.516E-02

 CASE NUMBER/SPLIT:    4     5.076E-01
      DEPENDENCIES:   15          29
     REQ. BRANCHES:    1    *      1
       DESCRIPTION:  COOL_INV  NO_FAILUR
 CASE/BRANCH SPLIT:          0.000E+00 5.076E-01 0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    5     4.246E-01
       DESCRIPTION:        OTHERWISE                      $ CORE DEBRIS CAUSES LOWE
 CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 4.246E-02 3.821E-01


********  QUESTION:   33  WATER IN PEDESTAL AT RPV FAILURE
 Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                            9763
          BRANCHES:           FLD+INJ   RPV+INJ   FLD       RPV_WTR
                                 1         2        3          4
    REALIZED SPLIT:          3.318E-02 7.371E-01 3.460E-03 2.262E-01

                     SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1     3.318E-02
      DEPENDENCIES:   12         29         22        3         6
     REQ. BRANCHES:   /2    *     1    *(    2    +(   1    *    3))
       DESCRIPTION:  /NO_INJECT NO_FAILUR LG_BURN   SBO       HPCS
 CASE/BRANCH SPLIT:          3.318E-02 0.000E+00 0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    2     3.460E-03
      DEPENDENCIES:   29         22        3         6
     REQ. BRANCHES:    2    *     2    +(   1    *    3)
       DESCRIPTION:  INJ&SPY_F LG_BURN   SBO       HPCS
 CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 3.460E-03 0.000E+00

 CASE NUMBER/SPLIT:    3     7.371E-01
      DEPENDENCIES:   12         29
     REQ. BRANCHES:   /2    *     1
       DESCRIPTION:  /NO_INJECT NO_FAILUR
 CASE/BRANCH SPLIT:          0.000E+00 7.371E-01 0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    4     2.262E-01
```

```
        DESCRIPTION:       OTHERWISE                               $ RESIDUAL RV WATER ONLY
  CASE/BRANCH SPLIT:             0.000E+00 0.000E+00 0.000E+00 2.262E-01


********  QUESTION:    34  PEDESTAL FAILURE DUE TO OVERPRESSURE AT RPV FAILURE
 Q-TYPE/TIMES ASKED:         DEP. INPUT PROB.                             9763
           BRANCHES:             PED_FAIL  NO
                                    1        2
     REALIZED SPLIT:             7.859E-03 9.921E-01

                        SUMMARY BY CASE

   CASE NUMBER/SPLIT:     1      5.076E-01
        DEPENDENCIES:    32
       REQ. BRANCHES:     2
         DESCRIPTION:   NO_FAIL
   CASE/BRANCH SPLIT:            0.000E+00 5.076E-01

   CASE NUMBER/SPLIT:     2      3.468E-03
        DEPENDENCIES:    13          33         33
       REQ. BRANCHES:     2     *(    1    +     3)
         DESCRIPTION:   HI_PRES    FLD+INJ    FLD
   CASE/BRANCH SPLIT:            3.468E-03 0.000E+00

   CASE NUMBER/SPLIT:     3      2.276E-03
        DEPENDENCIES:    13          33         33          32
       REQ. BRANCHES:     2     *(    2    +     4)    *     3
         DESCRIPTION:   HI_PRES    RPV+INJ    RPV_WTR    LARGE_VF
   CASE/BRANCH SPLIT:            2.276E-03 0.000E+00

   CASE NUMBER/SPLIT:     4      1.976E-02
        DEPENDENCIES:    13          33         33          32
       REQ. BRANCHES:     2     *(    2    +     4)    *     4
         DESCRIPTION:   HI_PRES    RPV+INJ    RPV_WTR    SMALL_VF
   CASE/BRANCH SPLIT:            0.000E+00 1.976E-02

   CASE NUMBER/SPLIT:     5      2.115E-03
        DEPENDENCIES:    13          33         33          32
       REQ. BRANCHES:     1     *(    1    +     3)    *     3
         DESCRIPTION:   LOW_PRES   FLD+INJ    FLD         LARGE_VF
   CASE/BRANCH SPLIT:            2.115E-03 0.000E+00

   CASE NUMBER/SPLIT:     6      4.647E-01
        DEPENDENCIES:    13
       REQ. BRANCHES:     1
         DESCRIPTION:   LOW_PRES
   CASE/BRANCH SPLIT:            0.000E+00 4.647E-01

   CASE NUMBER/SPLIT:     7      2.201E-05
         DESCRIPTION:         OTHERWISE                            $ SHOULD NEVER GO THIS PA
   CASE/BRANCH SPLIT:            0.000E+00 2.201E-05


  ********  QUESTION:    35  PEDESTAL CAVITY STEAM EXPLOSTION
```

```
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                              10634
          BRANCHES:            STM_EXP   NO_EXP
                                  1         2
    REALIZED SPLIT:           1.154E-02 9.885E-01

                    SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1      5.076E-01
      DEPENDENCIES:   32
     REQ. BRANCHES:    2
       DESCRIPTION:  NO_FAIL
 CASE/BRANCH SPLIT:           0.000E+00 5.076E-01

 CASE NUMBER/SPLIT:    2      4.766E-01
      DEPENDENCIES:   33          33
     REQ. BRANCHES:    2    +      4
       DESCRIPTION:  RPV+INJ   RPV_WTR
 CASE/BRANCH SPLIT:           0.000E+00 4.766E-01

 CASE NUMBER/SPLIT:    3      2.330E-03
      DEPENDENCIES:   31          31          31
     REQ. BRANCHES:    1    +      3     +     4
       DESCRIPTION:  ALPHA      LARGE_VF  SMALL_VF
 CASE/BRANCH SPLIT:           0.000E+00 2.330E-03

 CASE NUMBER/SPLIT:    4      1.342E-02
       DESCRIPTION:      OTHERWISE                         $ WATER IN CAVITY
 CASE/BRANCH SPLIT:           1.154E-02 1.879E-03


******** QUESTION:   36  PEDESTAL FAILURE DUE TO STEAM EXPLOSION
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                              12283
          BRANCHES:            PED_FAIL  NO
                                  1         2
    REALIZED SPLIT:           2.222E-03 9.978E-01

                    SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1      3.290E-02
      DEPENDENCIES:   31
     REQ. BRANCHES:    3
       DESCRIPTION:  LARGE_VF
 CASE/BRANCH SPLIT:           1.645E-03 3.126E-02

 CASE NUMBER/SPLIT:    2      9.556E-01
      DEPENDENCIES:   35
     REQ. BRANCHES:    2
       DESCRIPTION:  NO_EXP
 CASE/BRANCH SPLIT:           0.000E+00 9.556E-01

 CASE NUMBER/SPLIT:    3      1.154E-02
       DESCRIPTION:      OTHERWISE                         $ EX-VESSEL STEAM EXPLOSI
 CASE/BRANCH SPLIT:           5.770E-04 1.097E-02
```

```
******** QUESTION:   37  DRYWELL FAILURE DUE TO PEDESTAL FAILURE
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                                 14794
          BRANCHES:           DW_FAIL   NO
                               1         2
    REALIZED SPLIT:          1.719E-03 9.983E-01

                    SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1      9.828E-03
     DEPENDENCIES:   34         36
     REQ. BRANCHES:    1    +    1
       DESCRIPTION:  PED_FAIL  PED_FAIL
 CASE/BRANCH SPLIT:          1.719E-03 8.109E-03

 CASE NUMBER/SPLIT:    2      9.902E-01
       DESCRIPTION:      OTHERWISE                        $ PEDESTAL FAILURE HAS NO
 CASE/BRANCH SPLIT:          0.000E+00 9.902E-01

******** QUESTION:   38  DRYWELL OVERPRESSURE FAILURE AT RPV FAILURE
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                                 15272
          BRANCHES:           DW_FAIL   NO
                               1         2
    REALIZED SPLIT:          2.612E-05 1.000E+00

                    SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1      5.076E-01
     DEPENDENCIES:   32
     REQ. BRANCHES:    2
       DESCRIPTION:  NO_FAIL
 CASE/BRANCH SPLIT:          0.000E+00 5.076E-01

 CASE NUMBER/SPLIT:    2      2.634E-03
     DEPENDENCIES:   13         32
     REQ. BRANCHES:    2    *    3
       DESCRIPTION:  HI_PRES   LARGE_VF
 CASE/BRANCH SPLIT:          2.612E-05 2.607E-03

 CASE NUMBER/SPLIT:    3      4.898E-01
       DESCRIPTION:     OTHERWISE
 CASE/BRANCH SPLIT:          0.000E+00 4.898E-01

******** QUESTION:   39  DRYWELL FAILS AT/NEAR TIME OF RPV FAILURE
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                                 15272
          BRANCHES:           DW_FAIL   NO
                               1         2
    REALIZED SPLIT:          1.151E-02 9.885E-01

                    SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1      5.076E-01
```

```
        DEPENDENCIES:   32
       REQ. BRANCHES:    2
         DESCRIPTION:   NO_FAIL
   CASE/BRANCH SPLIT:            0.000E+00 5.076E-01

  CASE NUMBER/SPLIT:     2      1.151E-02
        DEPENDENCIES:   30          37        38
       REQ. BRANCHES:    1    +      1    +    1
         DESCRIPTION:   ALPHA      DW_FAIL   DW_FAIL
   CASE/BRANCH SPLIT:            1.151E-02 0.000E+00

  CASE NUMBER/SPLIT:     3      4.809E-01
         DESCRIPTION:       OTHERWISE                          $ NO DRYWELL FAILURE
   CASE/BRANCH SPLIT:            0.000E+00 4.809E-01


******** QUESTION:   40  CONTAINMENT STEAM CONCENTRATION AT/NEAR RPV FAILURE
Q-TYPE/TIMES ASKED:      DEP. INPUT PROB.                            18163
          BRANCHES:          0-15%      15-25%     25-35%     35-45%     45-55%     > 55
                               1          2          3          4          5          6
    REALIZED SPLIT:          7.502E-01 8.510E-03 1.083E-02 0.000E+00 0.000E+00 2.305

                  SUMMARY BY CASE

  CASE NUMBER/SPLIT:     1      4.017E-01
        DEPENDENCIES:   18
       REQ. BRANCHES:    2
         DESCRIPTION:   SPRAY
   CASE/BRANCH SPLIT:            4.017E-01 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000

  CASE NUMBER/SPLIT:     2      1.927E-02
        DEPENDENCIES:    3           6
       REQ. BRANCHES:    1    *      1
         DESCRIPTION:   SBO        NO INJECT
   CASE/BRANCH SPLIT:            1.927E-02 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000

  CASE NUMBER/SPLIT:     3      1.934E-02
        DEPENDENCIES:    3           6
       REQ. BRANCHES:    1    *      2
         DESCRIPTION:   SBO        RCIC
   CASE/BRANCH SPLIT:            0.000E+00 8.510E-03 1.083E-02 0.000E+00 0.000E+00 0.000

  CASE NUMBER/SPLIT:     4      1.590E-02
        DEPENDENCIES:    3           6
       REQ. BRANCHES:    1    *      3
         DESCRIPTION:   SBO        HPCS
   CASE/BRANCH SPLIT:            0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.590

  CASE NUMBER/SPLIT:     5      2.146E-01
        DEPENDENCIES:    2
       REQ. BRANCHES:    2
         DESCRIPTION:   FAILED
   CASE/BRANCH SPLIT:            0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 2.146
```

```
CASE NUMBER/SPLIT:    6     3.292E-01
      DESCRIPTION:        OTHERWISE                          $ OTHER NON-SBO TYPE EVEN
CASE/BRANCH SPLIT:          3.292E-01 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000

******** QUESTION:   41  FRACTION ZIRCONIUM INVENTORY REACTED AT/NEAR RPV FAILURE
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                              28853
          BRANCHES:          33%       22%       11%
                              1         2         3
   REALIZED SPLIT:          0.000E+00 2.953E-01 7.048E-01

                   SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1     3.926E-02
      DEPENDENCIES:    3          6
     REQ. BRANCHES:    1    *     1
       DESCRIPTION:  SBO        NO INJECT
 CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 3.926E-02

 CASE NUMBER/SPLIT:    2     2.527E-02
      DEPENDENCIES:    3          6
     REQ. BRANCHES:    1    *     2
       DESCRIPTION:  SBO        RCIC
 CASE/BRANCH SPLIT:          0.000E+00 7.833E-03 1.744E-02

 CASE NUMBER/SPLIT:    3     2.578E-02
      DEPENDENCIES:    3          6
     REQ. BRANCHES:    1    *     3
       DESCRIPTION:  SBO        HPCS
 CASE/BRANCH SPLIT:          0.000E+00 5.4[illegible]3E-03 2.037E-02

 CASE NUMBER/SPLIT:    4     9.097E-01
       DESCRIPTION:        OTHERWISE                          $ BOUND OTHERS WITH DISTR
 CASE/BRANCH SPLIT:          0.000E+00 2.820E-01 6.277E-01

******** QUESTION:   42  HYDROGEN IGNITION SOURCES AVAILABLE AT/BEFORE RPV FAILURE
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                              31016
          BRANCHES:          NO_IG_SRC IGNIT_SRC
                              1         2
   REALIZED SPLIT:          1.120E-01 8.881E-01

                   SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1     8.618E-01
      DEPENDENCIES:   16
     REQ. BRANCHES:    2
       DESCRIPTION:  HIS_ON
 CASE/BRANCH SPLIT:          0.000E+00 8.618E-01

 CASE NUMBER/SPLIT:    2     4.328E-03
      DEPENDENCIES:   16          3          3
     REQ. BRANCHES:    1    *    /1    *    /5
       DESCRIPTION:  HIS_OFF  /SBO       /LOOP & SB
```

```
CASE/BRANCH SPLIT:            4.320E-04 3.896E-03

CASE NUMBER/SPLIT:     3      2.233E-02
     DEPENDENCIES:    21          22
    REQ. BRANCHES:     2    +      2
      DESCRIPTION:   SMALL_BRN LG_BURN
CASE/BRANCH SPLIT:            0.000E+00 2.233E-02

CASE NUMBER/SPLIT:     4      1.115E-01
      DESCRIPTION:       OTHERWISE                    $ NO CONTINUOUS IGNITION
CASE/BRANCH SPLIT:            1.115E-01 0.000E+00

********  QUESTION:   43  HIGH PRESSURE MELT EJECTION AT RPV FAILURE
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                          35093
         BRANCHES:             HPME      NO_HPME
                                 1         2
   REALIZED SPLIT:            2.042E-02 9.796E-01

                     SUMMARY BY CASE

CASE NUMBER/SPLIT:     1      5.076E-01
     DEPENDENCIES:    32
    REQ. BRANCHES:     2
      DESCRIPTION:   NO_FAIL
CASE/BRANCH SPLIT:            0.000E+00 5.076E-01

CASE NUMBER/SPLIT:     2      4.669E-01
     DEPENDENCIES:    13
    REQ. BRANCHES:     1
      DESCRIPTION:   LOW_PRES
CASE/BRANCH SPLIT:            0.000E+00 4.669E-01

CASE NUMBER/SPLIT:     3      2.552E-02
      DESCRIPTION:       OTHERWISE                    $ RPV FAILURE AT HIGH PRE
CASE/BRANCH SPLIT:            2.042E-02 5.104E-03

********  QUESTION:   44  LARGE H2 BURN IGNITED AT/NEAR RPV FAILURE
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB. INPUT PARM.              37679
         BRANCHES:             NO_BRN_IG LG_BRN_IG
                                 1         2
   REALIZED SPLIT:            9.655E-01 3.453E-02

                     SUMMARY BY CASE

CASE NUMBER/SPLIT:     1      8.880E-01
     DEPENDENCIES:    42
    REQ. BRANCHES:     2
      DESCRIPTION:   IGNIT_SRC
CASE/BRANCH SPLIT:            8.880E-01 0.000E+00

CASE NUMBER/SPLIT:     2      5.954E-02
     DEPENDENCIES:    40
```

REQ. BRANCHES: 6
DESCRIPTION: > 55%
CASE/BRANCH SPLIT: 5.954E-02 0.000E+00

CASE NUMBER/SPLIT: 3 0.000E+00
DEPENDENCIES: 40 41 39 7
REQ. BRANCHES: 1 * 1 *(1 + 1)
DESCRIPTION: 0-15% 33% DW_FAIL PRIOR RV
CASE/BRANCH SPLIT: 0.000E+00 0.000E+00

CASE NUMBER/SPLIT: 4 0.000E+00
DEPENDENCIES: 40 41 43
REQ. BRANCHES: 1 * 1 * 1
DESCRIPTION: 0-15% 33% HPME
CASE/BRANCH SPLIT: 0.000E+00 0.000E+00

CASE NUMBER/SPLIT: 5 0.000E+00
DEPENDENCIES: 40 41
REQ. BRANCHES: 1 * 1
DESCRIPTION: 0-15% 33%
CASE/BRANCH SPLIT: 0.000E+00 0.000E+00

CASE NUMBER/SPLIT: 6 0.000E+00
DEPENDENCIES: 40 41 39 7
REQ. BRANCHES: 2 * 1 *(1 + 1)
DESCRIPTION: 15-25% 33% DW_FAIL PRIOR RV
CASE/BRANCH SPLIT: 0.000E+00 0.000E+00

CASE NUMBER/SPLIT: 7 0.000E+00
DEPENDENCIES: 40 41 43
REQ. BRANCHES: 2 * 1 * 1
DESCRIPTION: 15-25% 33% HPME
CASE/BRANCH SPLIT: 0.000E+00 0.000E+00

CASE NUMBER/SPLIT: 8 0.000E+00
DEPENDENCIES: 40 41
REQ. BRANCHES: 2 * 1
DESCRIPTION: 15-25% 33%
CASE/BRANCH SPLIT: 0.000E+00 0.000E+00

CASE NUMBER/SPLIT: 9 0.000E+00
DEPENDENCIES: 40 41 39 7
REQ. BRANCHES: 3 * 1 *(1 + 1)
DESCRIPTION: 25-35% 33% DW_FAIL PRIOR RV
CASE/BRANCH SPLIT: 0.000E+00 0.000E+00

CASE NUMBER/SPLIT: 10 0.000E+00
DEPENDENCIES: 40 41 43
REQ. BRANCHES: 3 * 1 * 1
DESCRIPTION: 25-35% 33% HPME
CASE/BRANCH SPLIT: 0.000E+00 0.000E+00

CASE NUMBER/SPLIT: 11 0.000E+00
DEPENDENCIES: 40 41

```
    REQ. BRANCHES:    3     *     1
      DESCRIPTION:  25-35%      33%
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   12      0.000E+00
     DEPENDENCIES:   40          41         39          7
    REQ. BRANCHES:    4     *     1     *(   1     +    1)
      DESCRIPTION:  35-45%      33%        DW_FAIL   PRIOR RV
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   13      0.000E+00
     DEPENDENCIES:   40          41         43
    REQ. BRANCHES:    4     *     1     *    1
      DESCRIPTION:  35-45%      33%        HPME
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   14      0.000E+00
     DEPENDENCIES:   40          41
    REQ. BRANCHES:    4     *     1
      DESCRIPTION:  35-45%      33%
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   15      0.000E+00
     DEPENDENCIES:   40          41         39          7
    REQ. BRANCHES:    5     *     1     *(   1     +    1)
      DESCRIPTION:  45-55%      33%        DW_FAIL   PRIOR RV
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   16      0.000E+00
     DEPENDENCIES:   40          41         43
    REQ. BRANCHES:    5     *     1     *    1
      DESCRIPTION:  45-55%      33%        HPME
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   17      0.000E+00
     DEPENDENCIES:   40          41
    REQ. BRANCHES:    5     *     1
      DESCRIPTION:  45-55%      33%
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   18      2.655E-03
     DEPENDENCIES:   40          41         39          7
    REQ. BRANCHES:    1     *     2     *(   1     +    1)
      DESCRIPTION:  0-15%       22%        DW_FAIL   PRIOR RV
CASE/BRANCH SPLIT:           0.000E+00 2.655E-03

CASE NUMBER/SPLIT:   19      9.105E-07
     DEPENDENCIES:   40          41         43
    REQ. BRANCHES:    1     *     2     *    1
      DESCRIPTION:  0-15%       22%        HPME
CASE/BRANCH SPLIT:           3.977E-07 5.128E-07

CASE NUMBER/SPLIT:   20      1.050E-04
     DEPENDENCIES:   40          41
```

```
     REQ. BRANCHES:     1     *     2
       DESCRIPTION:  0-15%      22%
CASE/BRANCH SPLIT:            6.516E-05 3.988E-05

 CASE NUMBER/SPLIT:    21      5.221E-05
      DEPENDENCIES:    40         41          39          7
     REQ. BRANCHES:     2     *     2     *(    1     +     1)
       DESCRIPTION:  15-25%     22%         DW_FAIL    PRIOR RV
 CASE/BRANCH SPLIT:            0.000E+00 5.221E-05

 CASE NUMBER/SPLIT:    22      8.013E-05
      DEPENDENCIES:    40         41          43
     REQ. BRANCHES:     2     *     2     *     1
       DESCRIPTION:  15-25%     22%         HPME
 CASE/BRANCH SPLIT:            3.526E-05 4.487E-05

 CASE NUMBER/SPLIT:    23      1.743E-03
      DEPENDENCIES:    40         41
     REQ. BRANCHES:     2     *     2
       DESCRIPTION:  15-25%     22%
 CASE/BRANCH SPLIT:            1.080E-03 6.6[illegible]E-04

 CASE NUMBER/SPLIT:    24      6.648E-05
      DEPENDENCIES:    40         41          39          7
     REQ. BRANCHES:     3     *     2     *(    1     +     1)
       DESCRIPTION:  25-35%     22%         DW_FAIL    PRIOR RV
 CASE/BRANCH SPLIT:            0.000E+00 6.648E-05

 CASE NUMBER/SPLIT:    25      1.020E-04
      DEPENDENCIES:    40         41          43
     REQ. BRANCHES:     3     *     2     *     1
       DESCRIPTION:  25-35%     22%         HPME
 CASE/BRANCH SPLIT:            5.813E-05 4.385E-05

 CASE NUMBER/SPLIT:    26      2.218E-03
      DEPENDENCIES:    40         41
     REQ. BRANCHES:     3     *     2
       DESCRIPTION:  25-35%     22%
 CASE/BRANCH SPLIT:            1.597E-03 6.209E-04

 CASE NUMBER/SPLIT:    27      0.000E+00
      DEPENDENCIES:    40         41          39          7
     REQ. BRANCHES:     4     *     2     *(    1     +     1)
       DESCRIPTION:  35-45%     22%         DW_FAIL    PRIOR RV
 CASE/BRANCH SPLIT:            0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    28      0.000E+00
      DEPENDENCIES:    40         41          43
     REQ. BRANCHES:     4     *  [illegible]  *     1
       DESCRIPTION:  35-45%     22%         HPME
 CASE/BRANCH SPLIT:            0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    29      0.000E+00
      DEPENDENCIES:    40         41
```

REQ. BRANCHES:	4 *	2		
DESCRIPTION:	35-45%	22%		
CASE/BRANCH SPLIT:		0.000E+00	0.000E+00	
CASE NUMBER/SPLIT:	30	0.000E+00		
DEPENDENCIES:	40	41	39	7
REQ. BRANCHES:	5 *	2 *(	1 +	1)
DESCRIPTION:	45-55%	22%	DW_FAIL	PRIOR RV
CASE/BRANCH SPLIT:		0.000E+00	0.000E+00	
CASE NUMBER/SPLIT:	31	0.000E+00		
DEPENDENCIES:	40	41	43	
REQ. BRANCHES:	5 *	2 *	1	
DESCRIPTION:	45-55%	22%	HPME	
CASE/BRANCH SPLIT:		0.000E+00	0.000E+00	
CASE NUMBER/SPLIT:	32	0.000E+00		
DEPENDENCIES:	40	41		
REQ. BRANCHES:	5 *	2		
DESCRIPTION:	45-55%	22%		
CASE/BRANCH SPLIT:		0.000E+00	0.000E+00	
CASE NUMBER/SPLIT:	33	2.492E-02		
DEPENDENCIES:	40	41	39	7
REQ. BRANCHES:	1 *	3 *(	1 +	1)
DESCRIPTION:	0-15%	11%	DW_FAIL	PRIOR RV
CASE/BRANCH SPLIT:		0.000E+00	2.492E-02	
CASE NUMBER/SPLIT:	34	7.402E-04		
DEPENDENCIES:	40	41	43	
REQ. BRANCHES:	1 *	3 *	1	
DESCRIPTION:	0-15%	11%	HPME	
CASE/BRANCH SPLIT:		4.219E-04	3.183E-04	
CASE NUMBER/SPLIT:	35	1.027E-02		
DEPENDENCIES:	40	41		
REQ. BRANCHES:	1 *	3		
DESCRIPTION:	0-15%	11%		
CASE/BRANCH SPLIT:		7.391E-03	2.874E-03	
CASE NUMBER/SPLIT:	36	1.164E-04		
DEPENDENCIES:	40	41	39	7
REQ. BRANCHES:	2 *	3 *(	1 +	1)
DESCRIPTION:	15-25%	11%	DW_FAIL	PRIOR RV
CASE/BRANCH SPLIT:		0.000E+00	1.164E-04	
CASE NUMBER/SPLIT:	37	1.784E-04		
DEPENDENCIES:	40	41	43	
REQ. BRANCHES:	2 *	3 *	1	
DESCRIPTION:	15-25%	11%	HPME	
CASE/BRANCH SPLIT:		1.267E-04	5.172E-05	
CASE NUMBER/SPLIT:	38	3.879E-03		
DEPENDENCIES:	40	41		

REQ. BRANCHES:	2	*	3				
DESCRIPTION:	15-25%		11%				
CASE/BRANCH SPLIT:			3.064E-03		8.145E-04		

CASE NUMBER/SPLIT:	39		1.482E-04				
DEPENDENCIES:	40		41		39		7
REQ. BRANCHES:	3	*	3	*(	1	+	1)
DESCRIPTION:	25-35%		11%		DW_FAIL		PRIOR RV
CASE/BRANCH SPLIT:			0.000E+00		1.482E-04		

CASE NUMBER/SPLIT:	40		2.270E-04		
DEPENDENCIES:	40		41		43
REQ. BRANCHES:	3	*	3	*	1
DESCRIPTION:	25-35%		11%		HPME
CASE/BRANCH SPLIT:			1.612E-04		6.583E-05

CASE NUMBER/SPLIT:	41		4.937E-03	
DEPENDENCIES:	40		41	
REQ. BRANCHES:	3	*	3	
DESCRIPTION:	25-35%		11%	
CASE/BRANCH SPLIT:			3.900E-03	1.037E-03

CASE NUMBER/SPLIT:	42		0.000E+00				
DEPENDENCIES:	40		41		39		7
REQ. BRANCHES:	4	*	3	*(	1	+	1)
DESCRIPTION:	35-45%		11%		DW_FAIL		PRIOR RV
CASE/BRANCH SPLIT:			0.000E+00		0.000E+00		

CASE NUMBER/SPLIT:	43		0.000E+00		
DEPENDENCIES:	40		41		43
REQ. BRANCHES:	4	*	3	*	1
DESCRIPTION:	35-45%		11%		HPME
CASE/BRANCH SPLIT:			0.000E+00		0.000E+00

CASE NUMBER/SPLIT:	44		0.000E+00	
DEPENDENCIES:	40		41	
REQ. BRANCHES:	4	*	3	
DESCRIPTION:	35-45%		11%	
CASE/BRANCH SPLIT:			0.000E+00	0.000E+00

CASE NUMBER/SPLIT:	45		0.000E+00				
DEPENDENCIES:	40		41		39		7
REQ. BRANCHES:	5	*	3	*(	1	+	1)
DESCRIPTION:	45-55%		11%		DW_FAIL		PRIOR RV
CASE/BRANCH SPLIT:			0.000E+00		0.000E+00		

CASE NUMBER/SPLIT:	46		0.000E+00		
DEPENDENCIES:	40		41		43
REQ. BRANCHES:	5	*	3	*	1
DESCRIPTION:	45-55%		11%		HPME
CASE/BRANCH SPLIT:			0.000E+00		0.000E+00

CASE NUMBER/SPLIT:	47		0.000E+00
DEPENDENCIES:	40		41

```
     REQ. BRANCHES:    5    *    3
       DESCRIPTION:  45-55%     11%
 CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:   48      0.000E+00
       DESCRIPTION:        OTHERWISE                          $ SHOULD NEVER GO THIS PA
 CASE/BRANCH SPLIT:           0.000E+00 0.000E+00


******** QUESTION:   45  CONTAINMENT FAILURE DUE TO H2 DETONATION AT/NEAR RPV FAILU
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                               37950
          BRANCHES:           DET_CF    NO
                                1        2
    REALIZED SPLIT:           6.489E-05 9.999E-01

                    SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1      9.655E-01
      DEPENDENCIES:   44
     REQ. BRANCHES:    1
       DESCRIPTION:  NO_BRN_IG
 CASE/BRANCH SPLIT:           0.000E+00 9.655E-01

 CASE NUMBER/SPLIT:    2      0.000E+00
      DEPENDENCIES:   40         40         40
     REQ. BRANCHES:    4    +    5    +    6
       DESCRIPTION:  35-45%    45-55%     > 55%
 CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    3      0.000E+00
      DEPENDENCIES:   41          3          7         18          6
     REQ. BRANCHES:    1    *     1    *     1    *     2    *    /1
       DESCRIPTION:  33%         SBO        PRIOR RV  SPRAY     /NO INJECT
 CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    4      0.000E+00
      DEPENDENCIES:   40         41
     REQ. BRANCHES:    1    *     1
       DESCRIPTION:  0-15%       33%
 CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    5      0.000E+00
      DEPENDENCIES:   40         40         41
     REQ. BRANCHES: (  2    +     3)   *     1
       DESCRIPTION:  15-25%    25-35%      33%
 CASE/BRANCH SPLIT:           0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    6      3.035E-02
      DEPENDENCIES:   41
     REQ. BRANCHES:    3
       DESCRIPTION:  11%
 CASE/BRANCH SPLIT:           0.000E+00 3.035E-02

 CASE NUMBER/SPLIT:    7      2.654E-03
```

```
      DEPENDENCIES:  41         3        7       18        6
     REQ. BRANCHES:   2    *    1    *   1   *   2    *   /1
       DESCRIPTION:  22%       SBO      PRIOR RV  SPRAY    /NO INJECT
CASE/BRANCH SPLIT:            5.839E-05 2.596E-03

CASE NUMBER/SPLIT:    8      1.490E-03
      DEPENDENCIES:  40         40       41
     REQ. BRANCHES: (  2   +    3)   *    2
       DESCRIPTION:  15-25%    25-35%    22%
CASE/BRANCH SPLIT:            0.000E+00 1.490E-03

CASE NUMBER/SPLIT:    9      4.135E-05
      DEPENDENCIES:  40         41
     REQ. BRANCHES:   1    *     2
       DESCRIPTION:  0-15%      22%
CASE/BRANCH SPLIT:            6.506E-06 3.485E-05

CASE NUMBER/SPLIT:   10      0.000E+00
       DESCRIPTION:        OTHERWISE                           $ SHOULD NEVER REACH THIS
CASE/BRANCH SPLIT:            0.000E+00 0.000E+00


********  QUESTION:   46   CONTAIMENT FAILURE AT/NEAR RPV FAILURE
Q-TYPE/TIMES ASKED:        DEP. CALC. PROB.                                37950
          BRANCHES:            FAILURE   NO_FAILUR
                                 1         2
    REALIZED SPLIT:           1.389E-02 9.861E-01

                     SUMMARY BY CASE

CASE NUMBER/SPLIT:    1      9.770E-03
      DEPENDENCIES:  30
     REQ. BRANCHES:   1
       DESCRIPTION:  ALPHA
CASE/BRANCH SPLIT:            9.770E-03 0.000E+00

CASE NUMBER/SPLIT:    2      6.416E-05
      DEPENDENCIES:  45
     REQ. BRANCHES:   1
       DESCRIPTION:  DET_CF
CASE/BRANCH SPLIT:            6.416E-05 0.000E+00

CASE NUMBER/SPLIT:    3      9.902E-01
       DESCRIPTION:        OTHERWISE
CASE/BRANCH SPLIT:            4.054E-03 9.861E-01


********  QUESTION:   47   MODE OF CONTAIMENT FAILURE AT/NEAR RPV FAILURE
Q-TYPE/TIMES ASKED:        DEP. CALC. PROB.                                39093
          BRANCHES:            ANCHORAGE PN-D/NoCF
                                 1         2
    REALIZED SPLIT:           2.449E-03 9.976E-01

                     SUMMARY BY CASE
```

```
CASE NUMBER/SPLIT:    1      9.861E-01
     DEPENDENCIES:   46
    REQ. BRANCHES:    2
      DESCRIPTION:   NO_FAILUR
CASE/BRANCH SPLIT:           0.000E+00 9.861E-01

CASE NUMBER/SPLIT:    2      9.834E-03
     DEPENDENCIES:   45         30
    REQ. BRANCHES:    1    +     1
      DESCRIPTION:   DET_CF     ALPHA
CASE/BRANCH SPLIT:           0.000E+00 9.834E-03

CASE NUMBER/SPLIT:    3      4.054E-03
      DESCRIPTION:       OTHERWISE
CASE/BRANCH SPLIT:           2.449E-03 1.605E-03


********  QUESTION:   48  DRYWELL FAILURE DUE TO CONTAINMENT H2 BURN BEFORE/NEAR RPV
Q-TYPE/TIMES ASKED:       DEP. CALC. PROB.                                  44944
          BRANCHES:          DW_FAIL   NO_DW_FL
                               1         2
   REALIZED SPLIT:           4.175E-03 9.958E-01

                     SUMMARY BY CASE

CASE NUMBER/SPLIT:    1      2.098E-02
     DEPENDENCIES:   22
    REQ. BRANCHES:    2
      DESCRIPTION:   LG_BURN
CASE/BRANCH SPLIT:           9.573E-04 2.002E-02

CASE NUMBER/SPLIT:    2      3.452E-02
     DEPENDENCIES:   44
    REQ. BRANCHES:    2
      DESCRIPTION:   LG_BRN_IG
CASE/BRANCH SPLIT:           3.217E-03 3.131E-02

CASE NUMBER/SPLIT:    3      9.445E-01
      DESCRIPTION:       OTHERWISE                      $ NO LARGE BURNS IN CNTMT
CASE/BRANCH SPLIT:           0.000E+00 9.445E-01


********  QUESTION:   49  POOL BYPASS BEFORE/NEAR RPV FAILURE
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                                  47658
          BRANCHES:          POOL_BP   NO_PL_BP
                               1         2
   REALIZED SPLIT:           5.184E-02 9.482E-01

                     SUMMARY BY CASE

CASE NUMBER/SPLIT:    1      1.151E-02
     DEPENDENCIES:   39
    REQ. BRANCHES:    1
```

```
        DESCRIPTION:  DW_FAIL
  CASE/BRANCH SPLIT:          1.151E-02 0.000E+00

  CASE NUMBER/SPLIT:    2     4.086E-03
       DEPENDENCIES:   48
      REQ. BRANCHES:    1
        DESCRIPTION:  DW_FAIL
  CASE/BRANCH SPLIT:          4.086E-03 0.000E+00

  CASE NUMBER/SPLIT:    3     3.440E-02
       DEPENDENCIES:   25         47
      REQ. BRANCHES:    1    +     1
        DESCRIPTION:  ANCHORAGE ANCHORAGE
  CASE/BRANCH SPLIT:          3.440E-02 0.000E+00

  CASE NUMBER/SPLIT:    4     3.528E-02
       DEPENDENCIES:   22         44         3         3         3         3         3
      REQ. BRANCHES: (  2    +     2)   *((  3    +     4    +     6)   +(  1    +     5
        DESCRIPTION:  LG_BURN   LG_BRN_LG OTHER TYP CRIT ATWS OTHERS     SBO       LOO
  CASE/BRANCH SPLIT:          1.763E-03 3.351E-02

  CASE NUMBER/SPLIT:    5     9.148E-01
        DESCRIPTION:      OTHERWISE                          $ BYPASS FOR SEQUENCES WI
  CASE/BRANCH SPLIT:          9.081E-05 9.147E-01


******** QUESTION:   50  CNTMT FAILURE AT/NEAR RPV FAILURE IMPACT ON ECCS INJECTION
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                                 48566
           BRANCHES:            NO_FAILUR FAILUR
                                    1         2
     REALIZED SPLIT:          9.978E-01 2.203E-03

                     SUMMARY BY CASE

  CASE NUMBER/SPLIT:    1     2.448E-03
       DEPENDENCIES:   47
      REQ. BRANCHES:    1
        DESCRIPTION:  ANCHORAGE
  CASE/BRANCH SPLIT:          2.444E-04 2.203E-03

  CASE NUMBER/SPLIT:    2     9.976E-01
        DESCRIPTION:      OTHERWISE                          $ NOT ANCHORAGE
  CASE/BRANCH SPLIT:          9.976E-01 0.000E+00


******** QUESTION:   51  CNTMT FAILURE AT/NEAR RPV FAILURE IMPACT ON ECCS INJECTION
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                                 52777
           BRANCHES:            NO_FAILUR FAILUR
                                    1         2
     REALIZED SPLIT:          9.942E-01 5.840E-03

                     SUMMARY BY CASE

  CASE NUMBER/SPLIT:    1     1.168E-02
```

```
      DEPENDENCIES:   46        50
     REQ. BRANCHES:    1    *    1
       DESCRIPTION:  FAILURE   NO_FAILUR
CASE/BRANCH SPLIT:          5.840E-03 5.840E-03

CASE NUMBER/SPLIT:    2     9.883E-01
       DESCRIPTION:       OTHERWISE                           $ NO CNTMT FAILURE
CASE/BRANCH SPLIT:          9.883E-01 0.000E+00


********  QUESTION:   52  CNTMT FAIL AT/NEAR RPV FAILURE STEAM/RADIATION RELEASE IMP
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                                 59189
          BRANCHES:         NO_FAILUR FAILUR
                              1         2
    REALIZED SPLIT:         9.942E-01 5.839E-03

                  SUMMARY BY CASE

CASE NUMBER/SPLIT:    1     1.168E-02
      DEPENDENCIES:   46        50
     REQ. BRANCHES:    1    *    1
       DESCRIPTION:  FAILURE   NO_FAILUR
CASE/BRANCH SPLIT:          5.839E-03 5.839E-03

CASE NUMBER/SPLIT:    2     9.883E-01
       DESCRIPTION:      OTHERWISE                            $ NO CNTMT FAILURE
CASE/BRANCH SPLIT:          9.883E-01 0.000E+00


********  QUESTION:   53  INJECT & SPRAY FAILURE DUE TO CNTMT FAILURE AT/NEAR RPV FA
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                                 59189
          BRANCHES:         NO_FAILUR INJ&SPY_F
                              1         2
    REALIZED SPLIT:         9.949E-01 5.123E-03

                  SUMMARY BY CASE

CASE NUMBER/SPLIT:    1     2.203E-03
      DEPENDENCIES:   50
     REQ. BRANCHES:    2
       DESCRIPTION:  FAILUR
CASE/BRANCH SPLIT:          0.000E+00 2.203E-03

CASE NUMBER/SPLIT:    2     2.919E-03
      DEPENDENCIES:   51        52
     REQ. BRANCHES:    2    *    2
       DESCRIPTION:  FAILUR    FAILUR
CASE/BRANCH SPLIT:          0.000E+00 2.919E-03

CASE NUMBER/SPLIT:    3     9.949E-01
       DESCRIPTION:      OTHERWISE                            $ ALL INJECT NOT FAILED A
CASE/BRANCH SPLIT:          9.949E-01 0.000E+00
```

```
********  QUESTION:   54  TYPE OF DEBRIS CONCRETE INTERACTIONS
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                                  91191
         BRANCHES:            DRY-CCI   FAST-WET  SLOW-WET  NO-CCI
                                 1         2         3         4
   REALIZED SPLIT:            2.284E-01 8.772E-02 1.256E-01 5.585E-01

                    SUMMARY BY CASE

 CASE NUMBER/SPLIT:     1     5.077E-01
      DEPENDENCIES:    32
     REQ. BRANCHES:     2
       DESCRIPTION:  NO_FAIL
 CASE/BRANCH SPLIT:           0.000E+00 0.000E+00 0.000E+00 5.077E-01

 CASE NUMBER/SPLIT:     2     2.284E-01
      DEPENDENCIES:    33        33        53
     REQ. BRANCHES:     4    +(   2    *    2)
       DESCRIPTION:  RPV_WTR   RPV+INJ   INJ&SPY_F
 CASE/BRANCH SPLIT:           2.284E-01 0.000E+00 0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:     3     3.746E-04
      DEPENDENCIES:    33        33        53        43
     REQ. BRANCHES: (  3    +(   1    *    2))  *    1
       DESCRIPTION:  FLD       FLD+INJ   INJ&SPY_F HPME
 CASE/BRANCH SPLIT:           0.000E+00 6.540E-05 1.828E-04 1.264E-04

 CASE NUMBER/SPLIT:     4     2.965E-04
      DEPENDENCIES:    33        33        53        43        14
     REQ. BRANCHES: (  3    +(   1    *    2))  *    2    *    1
       DESCRIPTION:  FLD       FLD+INJ   INJ&SPY_F NO_HPME   LG_DEB
 CASE/BRANCH SPLIT:           0.000E+00 8.292E-05 1.426E-04 7.098E-05

 CASE NUMBER/SPLIT:     5     3.163E-03
      DEPENDENCIES:    33        33        53        43        14
     REQ. BRANCHES: (  3    +(   1    *    2))  *    2    *    2
       DESCRIPTION:  FLD       FLD+INJ   INJ&SPY_F NO_HPME   SM_DEB
 CASE/BRANCH SPLIT:           0.000E+00 8.853E-04 1.519E-03 7.587E-04

 CASE NUMBER/SPLIT:     6     9.182E-03
      DEPENDENCIES:    33        53        43
     REQ. BRANCHES: (  2    *    1)   *    1
       DESCRIPTION:  RPV+INJ   NO_FAILUR HPME
 CASE/BRANCH SPLIT:           0.000E+00 2.892E-03 4.384E-03 1.905E-03

 CASE NUMBER/SPLIT:     7     2.391E-01
      DEPENDENCIES:    33        53        43
     REQ. BRANCHES: (  2    *    1)   *    2
       DESCRIPTION:  RPV+INJ   NO_FAILUR NO_HPME
 CASE/BRANCH SPLIT:           0.000E+00 8.071E-02 1.136E-01 4.484E-02

 CASE NUMBER/SPLIT:     8     2.398E-[illegible]
      DEPENDENCIES:    33        53        43
     REQ. BRANCHES: (  1    *    1)   *    1
       DESCRIPTION:  FLD+INJ   NO_FAILUR HPME
```

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CASE/BRANCH SPLIT:          0.000E+00 4.194E-04 1.169E-03 8.092E-04

CASE NUMBER/SPLIT:    9     4.462E-04
     DEPENDENCIES:   33        53        43        14
    REQ. BRANCHES: (  1    *    1)   *    2    *    1
      DESCRIPTION:  FLD+INJ   NO_FAILUR NO_HPME   LG_DEB
CASE/BRANCH SPLIT:          0.000E+00 1.249E-04 2.144E-04 1.070E-04

CASE NUMBER/SPLIT:   10     9.057E-03
     DEPENDENCIES:   33        53        43        14
    REQ. BRANCHES: (  1    *    1)   *    2    *    2
      DESCRIPTION:  FLD+INJ   NO_FAILUR NO_HPME   SM_DEB
CASE/BRANCH SPLIT:          0.000E+00 2.536E-03 4.348E-03 2.173E-03

CASE NUMBER/SPLIT:   11     0.000E+00
      DESCRIPTION:       OTHERWISE                        $ SHOULD NEVER TAKE THIS
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 0.000E+00 0.000E+00


********  QUESTION:   55  PEDESTAL FAILURE DUE TO CORE DEBRIS CONCRETE INTERACTION
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                            114.90
          BRANCHES:            AT_VB     AFTER_VB  NO_FAILUR
                                 1          2         3
    REALIZED SPLIT:         1.558E-02 1.509E-01 8.336E-01

                   SUMMARY BY CASE

CASE NUMBER/SPLIT:    1     1.558E-02
     DEPENDENCIES:   39        48
    REQ. BRANCHES:    1    +    1
      DESCRIPTION:  DW_FAIL   DW_FAIL
CASE/BRANCH SPLIT:          1.558E-02 0.000E+00 0.000E+00

CASE NUMBER/SPLIT:    2     5.059E-01
     DEPENDENCIES:   32
    REQ. BRANCHES:    2
      DESCRIPTION:  NO_FAIL
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 5.059E-01

CASE NUMBER/SPLIT:    3     2.229E-01
     DEPENDENCIES:   54
    REQ. BRANCHES:    1
      DESCRIPTION:  DRY-CCI
CASE/BRANCH SPLIT:          0.000E+00 9.585E-02 1.271E-01

CASE NUMBER/SPLIT:    4     8.512E-02
     DEPENDENCIES:   54
    REQ. BRANCHES:    2
      DESCRIPTION:  FAST-WET
CASE/BRANCH SPLIT:          0.000E+00 2.468E-02 6.043E-02

CASE NUMBER/SPLIT:    5     1.216E-01
     DEPENDENCIES:   54
    REQ. BRANCHES:    3
```

```
      DESCRIPTION:  SLOW-WET
CASE/BRANCH SPLIT:          0.000E+00 3.040E-02 9.121E-02

CASE NUMBER/SPLIT:   6      4.907E-02
     DEPENDENCIES:  54
    REQ. BRANCHES:   4
      DESCRIPTION:  NO-CCI
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 4.907E-02

CASE NUMBER/SPLIT:   7      0.000E+00
      DESCRIPTION:      OTHERWISE                            $ SHOULD NEVER GO THIS PA
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 0.000E+00


******** QUESTION:  56  MODE OF RHR SPRAY OPERATION LATE
Q-TYPE/TIMES ASKED:      DEP. INPUT PROB.                           114790
         BRANCHES:            CONTROLD  SPRAY     NO_SPRAY
                                 1        2          3
   REALIZED SPLIT:          0.000E+00 4.297E-01 5.704E-01

                   SUMMARY BY CASE

CASE NUMBER/SPLIT:   1      5.704E-01
     DEPENDENCIES:   8
    REQ. BRANCHES:  /1
      DESCRIPTION: /RHR_SPRY
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 5.704E-01

CASE NUMBER/SPLIT:   2      3.660E-01
     DEPENDENCIES:   3          8
    REQ. BRANCHES:  /1    *     1
      DESCRIPTION: /SBO        RHR_SPRY
CASE/BRANCH SPLIT:          0.000E+00 3.660E-01 0.000E+00

CASE NUMBER/SPLIT:   3      3.581E-02
     DEPENDENCIES:   3          7         8
    REQ. BRANCHES:   1    *     1    *    1
      DESCRIPTION:  SBO        PRIOR RV  RHR_SPRY
CASE/BRANCH SPLIT:          0.000E+00 3.581E-02 0.000E+00

CASE NUMBER/SPLIT:   4      2.794E-02
     DEPENDENCIES:   3          7         8
    REQ. BRANCHES:   1    *     2    *    1
      DESCRIPTION:  SBO        CNTMT LIM RHR_SPRY
CASE/BRANCH SPLIT:          0.000E+00 2.794E-02 0.000E+00

CASE NUMBER/SPLIT:   5      0.000E+00
      DESCRIPTION:      OTHERWISE                            $ SHOULD NEVER GO THIS WA
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 0.000E+00


******** QUESTION:  57  HYDROGEN IGNITION SOURCES AVAILABLE LATE
Q-TYPE/TIMES ASKED:      DEP. INPUT PROB.                           136142
         BRANCHES:           NO_SOURCE IGN_SOURC
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```
                          1          2
   REALIZED SPLIT:        1.136E-01 8.865E-01

                SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1     8.619E-01
      DEPENDENCIES:   16
     REQ. BRANCHES:    2
       DESCRIPTION:  HIS_ON
 CASE/BRANCH SPLIT:          0.000E+00 8.619E-01

 CASE NUMBER/SPLIT:    2     4.314E-03
      DEPENDENCIES:   16        3          3
     REQ. BRANCHES:    1    *   /1    *   /5
       DESCRIPTION:  HIS_OFF  /SBO       /LOOP & SB
 CASE/BRANCH SPLIT:          4.276E-04 3.887E-03

 CASE NUMBER/SPLIT:    3     4.131E-02
      DEPENDENCIES:    3        7
     REQ. BRANCHES:    1    *   1
       DESCRIPTION:  SBO       PRIOR RV
 CASE/BRANCH SPLIT:          2.066E-02 2.066E-02

 CASE NUMBER/SPLIT:    4     3.171E-02
      DEPENDENCIES:    3        7
     REQ. BRANCHES:    1    *   2
       DESCRIPTION:  SBO       CNTMT LIM
 CASE/BRANCH SPLIT:          3.171E-02 0.000E+00

 CASE NUMBER/SPLIT:    5     6.083E-02
       DESCRIPTION:     OTHERWISE                        $ LOSS OF AC POWER NO REC
 CASE/BRANCH SPLIT:          6.083E-02 0.000E+00


******** QUESTION:    58  CONTAINMENT STEAM CONCENTRATION LATE
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                              153684
          BRANCHES:         0-15%     15-25%    25-35%    35-45%    45-55%    > 55
                              1         2         3         4         5         6
   REALIZED SPLIT:        7.665E-01 3.021E-04 8.217E-04 2.026E-03 0.000E+00 2.304

                SUMMARY BY CASE

 CASE NUMBER/SPLIT:    1     4.297E-01
      DEPENDENCIES:   56
     REQ. BRANCHES:    2
       DESCRIPTION:  SPRAY
 CASE/BRANCH SPLIT:          4.297E-01 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000

 CASE NUMBER/SPLIT:    2     3.013E-04
      DEPENDENCIES:    5
     REQ. BRANCHES:    2
       DESCRIPTION:  NOT ISOL
 CASE/BRANCH SPLIT:          7.333E-05 2.280E-04 0.000E+00 0.000E+00 0.000E+00 0.000
```

```
CASE NUMBER/SPLIT:    3     7.568E-03
     DEPENDENCIES:    3        6
    REQ. BRANCHES:    1    *   1
      DESCRIPTION:   SBO      NO INJECT
 CASE/BRANCH SPLIT:          7.494E-03 7.418E-05 0.000E+00 0.000E+00 0.000E+00 0.000

CASE NUMBER/SPLIT:    4     2.847E-03
     DEPENDENCIES:    3        6
    REQ. BRANCHES:    1    *   2
      DESCRIPTION:   SBO      RCIC
 CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 8.217E-04 2.026E-03 0.000E+00 0.000

CASE NUMBER/SPLIT:    5     1.581E-02
     DEPENDENCIES:    3        6
    REQ. BRANCHES:    1    *   3
      DESCRIPTION:   SBO      HPCS
 CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 1.581

CASE NUMBER/SPLIT:    6     2.146E-01
     DEPENDENCIES:    2
    REQ. BRANCHES:    2
      DESCRIPTION:   FAILED
 CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 2.146

CASE NUMBER/SPLIT:    7     3.292E-01
      DESCRIPTION:       OTHERWISE                       $ NON-SBO TRANSIENTS WITH
 CASE/BRANCH SPLIT:          3.292E-01 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000


********  QUESTION:   59  H2 COMBUSTION BEFORE/AT RPV FAILURE
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                          153684
         BRANCHES:            EARLY_BRN NO_ERLY_B
                                  1         2
   REALIZED SPLIT:           7.513E-01 2.488E-01

                    SUMMARY BY CASE

CASE NUMBER/SPLIT:    1     7.513E-01
     DEPENDENCIES:   44        22        21
    REQ. BRANCHES:    2    +    2    +    2
      DESCRIPTION:   LG_BRN_IG LG_BURN   SMALL_BRN
 CASE/BRANCH SPLIT:          7.513E-01 0.000E+00

CASE NUMBER/SPLIT:    2     2.488E-01
      DESCRIPTION:       OTHERWISE
 CASE/BRANCH SPLIT:          0.000E+00 2.488E-01


********  QUESTION:   60  CONTAINMENT H2 CONCENTRATION LATE
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB.                          302752
         BRANCHES:            < 4 %     4-8 %     8-12 %    12-16 %   16-20 %    > 2
                                  1         2         3         4         5         6
   REALIZED SPLIT:           2.630E-01 2.917E-01 5.930E-02 6.685E-02 3.151E-01 4.355
```

SUMMARY BY CASE

```
CASE NUMBER/SPLIT:   1      4.939E-01
     DEPENDENCIES:  59         54
    REQ. BRANCHES:   1    *     4
      DESCRIPTION:  EARLY_BRN NO-CCI
CASE/BRANCH SPLIT:          2.470E-01 2.470E-01 0.000E+00 0.000E+00 0.000E+00 0.000

CASE NUMBER/SPLIT:   2      1.024E-01
     DEPENDENCIES:  59         54
    REQ. BRANCHES:   1    *     3
      DESCRIPTION:  EARLY_BRN SLOW-WET
CASE/BRANCH SPLIT:          1.024E-02 2.561E-02 3.073E-02 2.561E-02 1.024E-02 0.000

CASE NUMBER/SPLIT:   3      7.210E-02
     DEPENDENCIES:  59         54
    REQ. BRANCHES:   1    *     2
      DESCRIPTION:  EARLY_BRN FAST-WET
CASE/BRANCH SPLIT:          0.000E+00 7.207E-03 1.442E-02 2.524E-02 2.524E-02 0.000

CASE NUMBER/SPLIT:   4      4.339E-04
     DEPENDENCIES:  59         54         56         3          6
    REQ. BRANCHES:   1    *     1    *     3    *    1    *     1
      DESCRIPTION:  EARLY_BRN DRY-CCI   NO_SPRAY  SBO        NO INJECT
CASE/BRANCH SPLIT:          0.000E+00 1.521E-04 3.018E-05 2.583E-05 2.583E-05 2.000

CASE NUMBER/SPLIT:   5      1.217E-03
     DEPENDENCIES:  59         54         56         3          6
    REQ. BRANCHES:   1    *     1    *     3    *    1    *     2
      DESCRIPTION:  EARLY_BRN DRY-CCI   NO_SPRAY  SBO        RCIC
CASE/BRANCH SPLIT:          0.000E+00 4.567E-05 7.336E-04 4.376E-04 0.000E+00 0.000

CASE NUMBER/SPLIT:   6      0.000E+00
     DEPENDENCIES:  59         54         56         3          6
    REQ. BRANCHES:   1    *     1    *     3    *    1    *     3
      DESCRIPTION:  EARLY_BRN DRY-CCI   NO_SPRAY  SBO        HPCS
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000

CASE NUMBER/SPLIT:   7      4.032E-03
     DEPENDENCIES:  59         54         56         3          6
    REQ. BRANCHES:   1    *     1    *     2    *    1    *     1
      DESCRIPTION:  EARLY_BRN DRY-CCI   SPRAY     SBO        NO INJECT
CASE/BRANCH SPLIT:          0.000E+00 1.452E-03 2.822E-04 2.419E-04 2.419E-04 1.815

CASE NUMBER/SPLIT:   8      7.532E-03
     DEPENDENCIES:  59         54         56         3          6
    REQ. BRANCHES:   1    *     1    *     2    *    1    *     2
      DESCRIPTION:  EARLY_BRN DRY-CCI   SPRAY     SBO        RCIC
CASE/BRANCH SPLIT:          0.000E+00 3.760E-04 3.239E-03 2.787E-03 1.054E-03 7.467

CASE NUMBER/SPLIT:   9      0.000E+00
     DEPENDENCIES:  59         54         56         3          6
    REQ. BRANCHES:   1    *     1    *     2    *    1    *     3
      DESCRIPTION:  EARLY_BRN DRY-CCI   SPRAY     SBO        HPCS
```

```
CASE/BRANCH SPLIT:         0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000

CASE NUMBER/SPLIT:   10     6.650E-04
     DEPENDENCIES:   59        54        56         3         6
    REQ. BRANCHES:    2    *    4    *    3    *    1    *    1
      DESCRIPTION:  NO_ERLY_B NO-CCI    NO_SPRAY  SBO       NO INJECT
CASE/BRANCH SPLIT:         0.000E+00 6.650E-04 0.000E+00 0.000E+00 0.000E+00 0.000

CASE NUMBER/SPLIT:   11     0.000E+00
     DEPENDENCIES:   59        54        56         3         6
    REQ. BRANCHES:    2    *    4    *    3    *    1    *    2
      DESCRIPTION:  NO_ERLY_B NO-CCI    NO_SPRAY  SBO       RCIC
CASE/BRANCH SPLIT:         0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000

CASE NUMBER/SPLIT:   12     9.892E-03
     DEPENDENCIES:   59        54        56         3         6
    REQ. BRANCHES:    2    *    4    *    3    *    1    *    3
      DESCRIPTION:  NO_ERLY_B NO-CCI    NO_SPRAY  SBO       HPCS
CASE/BRANCH SPLIT:         5.836E-03 4.056E-03 0.000E+00 0.000E+00 0.000E+00 0.000

CASE NUMBER/SPLIT:   13     1.522E-03
     DEPENDENCIES:   59        54        56         3         6
    REQ. BRANCHES:    2    *    4    *    2    *    1    *    1
      DESCRIPTION:  NO_ERLY_B NO-CCI    SPRAY     SBO       NO INJECT
CASE/BRANCH SPLIT:         0.000E+00 1.522E-03 0.000E+00 0.000E+00 0.000E+00 0.000

CASE NUMBER/SPLIT:   14     0.000E+00
     DEPENDENCIES:   59        54        56         3         6
    REQ. BRANCHES:    2    *    4    *    2    *    1    *    2
      DESCRIPTION:  NO_ERLY_B NO-CCI    SPRAY     SBO       RCIC
CASE/BRANCH SPLIT:         0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000

CASE NUMBER/SPLIT:   15     0.000E+00
     DEPENDENCIES:   59        54        56         3         6
    REQ. BRANCHES:    2    *    4    *    2    *    1    *    3
      DESCRIPTION:  NO_ERLY_B NO-CCI    SPRAY     SBO       HPCS
CASE/BRANCH SPLIT:         0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000

CASE NUMBER/SPLIT:   16     2.312E-02
     DEPENDENCIES:   59        54
    REQ. BRANCHES:    2    *    3
      DESCRIPTION:  NO_ERLY_B SLOW-WET
CASE/BRANCH SPLIT:         0.000E+00 2.311E-03 3.468E-03 5.781E-03 1.156E-02 0.000

CASE NUMBER/SPLIT:   17     1.561E-02
     DEPENDENCIES:   59        54
    REQ. BRANCHES:    2    *    2
      DESCRIPTION:  NO_ERLY_B FAST-WET
CASE/BRANCH SPLIT:         0.000E+00 0.000E+00 1.560E-03 2.340E-03 1.171E-02 0.000

CASE NUMBER/SPLIT:   18     4.279E-04
     DEPENDENCIES:   59        54        56         3         6
    REQ. BRANCHES:    2    *    1    *    3    *    1    *    1
      DESCRIPTION:  NO_ERLY_B DRY-CCI   NO_SPRAY  SBO       NO INJECT
```

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CASE/BRANCH SPLIT:          0.000E+00 1.284E-04 2.990E-05 2.563E-05 2.563E-05 2.

CASE NUMBER/SPLIT:    19    1.319E-03
     DEPENDENCIES:    59       54         56         3         6
     REQ. BRANCHES:    2   *    1    *     3    *    1    *    2
      DESCRIPTION:  NO_ERLY_B DRY-CCI   NO_SPRAY  SBO        RCIC
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 7.923E-04 4.749E-04 5.203E-05 0.000

CASE NUMBER/SPLIT:    20    0.000E+00
     DEPENDENCIES:    59       54         56         3         6
     REQ. BRANCHES:    2   *    1    *     3    *    1    *    3
      DESCRIPTION:  NO_ERLY_B DRY-CCI   NO_SPRAY  SBO        HPCS
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000

CASE NUMBER/SPLIT:    21    3.921E-03
     DEPENDENCIES:    59       54         56         3         6
     REQ. BRANCHES:    2   *    1    *     2    *    1    *    1
      DESCRIPTION:  NO_ERLY_B DRY-CCI    SPRAY    SBO        NO INJECT
CASE/BRANCH SPLIT:          0.000E+00 1.216E-03 2.745E-04 2.353E-04 2.353E-04 1.960

CASE NUMBER/SPLIT:    22    8.704E-03
     DEPENDENCIES:    59       54         56         3         6
     REQ. BRANCHES:    2   *    1    *     2    *    1    *    2
      DESCRIPTION:  NO_ERLY_B DRY-CCI    SPRAY    SBO        RCIC
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 3.743E-03 3.656E-03 1.219E-03 8.701

CASE NUMBER/SPLIT:    23    0.000E+00
     DEPENDENCIES:    59       54         56         3         6
     REQ. BRANCHES:    2   *    1    *     2    *    1    *    3
      DESCRIPTION:  NO_ERLY_B DRY-CCI    SPRAY    SBO        HPCS
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000E+00 0.000

CASE NUMBER/SPLIT:    24    2.535E-01
      DESCRIPTION:      OTHERWISE                           $ SHOULD NOT TAKE THIS PA
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00 0.000E+00 0.000E+00 2.535E-01 0.000


********  QUESTION:    61  AC POWER AVAILABLE LATE
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                        302752
         BRANCHES:            AC_LATE   NO_AC_LAT
                                 1         2
   REALIZED SPLIT:          9.828E-01 1.726E-02

                   SUMMARY BY CASE

CASE NUMBER/SPLIT:     1    1.726E-02
     DEPENDENCIES:     3       7
     REQ. BRANCHES:    1   *   3
      DESCRIPTION:   SBO       NO RECOV
CASE/BRANCH SPLIT:          0.000E+00 1.726E-02

CASE NUMBER/SPLIT:     2    9.828E-01
      DESCRIPTION:       OTHERWISE                          $ AC POWER AVAILABLE   E
CASE/BRANCH SPLIT:          9.828E-01 0.000E+00
```

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********  QUESTION:   62  LARGE H2 BURN IGNITED LATE
Q-TYPE/TIMES ASKED:       DEP. INPUT PROB. INPUT PARM.            315229
          BRANCHES:            NO_BURN   LG_BURN
                                  1         2
    REALIZED SPLIT:          9.559E-01 4.426E-02

                     SUMMARY BY CASE

  CASE NUMBER/SPLIT:    1     8.865E-01
       DEPENDENCIES:   57
      REQ. BRANCHES:    2
        DESCRIPTION:  IGN_SOURC
  CASE/BRANCH SPLIT:          8.865E-01 0.000E+00

  CASE NUMBER/SPLIT:    2     5.897E-02
       DEPENDENCIES:   58
      REQ. BRANCHES:    6
        DESCRIPTION:  > 55%
  CASE/BRANCH SPLIT:          5.897E-02 0.000E+00

  CASE NUMBER/SPLIT:    3     8.919E-03
       DEPENDENCIES:   60
      REQ. BRANCHES:    1
        DESCRIPTION:  < 4 %
  CASE/BRANCH SPLIT:          8.919E-03 0.000E+00

  CASE NUMBER/SPLIT:    4     6.463E-04
       DEPENDENCIES:   58         60         61
      REQ. BRANCHES:    1    *     2    *     2
        DESCRIPTION:  0-15%      4-8 %     NO_AC_LAT
  CASE/BRANCH SPLIT:          4.591E-04 1.873E-04

  CASE NUMBER/SPLIT:    5     5.494E-05
       DEPENDENCIES:   58         60         61
      REQ. BRANCHES:    1    *     3    *     2
        DESCRIPTION:  0-15%      8-12 %    NO_AC_LAT
  CASE/BRANCH SPLIT:          3.694E-05 1.800E-05

  CASE NUMBER/SPLIT:    6     6.844E-05
       DEPENDENCIES:   58         60         61
      REQ. BRANCHES:    1    *     4    *     2
        DESCRIPTION:  0-15%     12-16 %    NO_AC_LAT
  CASE/BRANCH SPLIT:          3.978E-05 2.866E-05

  CASE NUMBER/SPLIT:    7     1.104E-04
       DEPENDENCIES:   58         60         61
      REQ. BRANCHES:    1    *     5    *     2
        DESCRIPTION:  0-15%     16-20 %    NO_AC_LAT
  CASE/BRANCH SPLIT:          5.410E-05 5.634E-05

  CASE NUMBER/SPLIT:    8     3.196E-05
       DEPENDENCIES:   58         60         61
```

REQ. BRANCHES: 1 * 6 * 2
DESCRIPTION: 0-15% > 20 % NO_AC_LAT
CASE/BRANCH SPLIT: 1.566E-05 1.630E-05

CASE NUMBER/SPLIT: 9 1.808E-05
DEPENDENCIES: 58 60 61
REQ. BRANCHES: 2 * 2 * 2
DESCRIPTION: 15-25% 4-8 % NO_AC_LAT
CASE/BRANCH SPLIT: 1.294E-05 5.142E-06

CASE NUMBER/SPLIT: 10 3.213E-06
DEPENDENCIES: 58 60 61
REQ. BRANCHES: 2 * 3 * 2
DESCRIPTION: 15-25% 8-12 % NO_AC_LAT
CASE/BRANCH SPLIT: 2.256E-06 9.565E-07

CASE NUMBER/SPLIT: 11 3.858E-06
DEPENDENCIES: 58 60 61
REQ. BRANCHES: 2 * 4 * 2
DESCRIPTION: 15-25% 12-16 % NO_AC_LAT
CASE/BRANCH SPLIT: 2.295E-06 1.563E-06

CASE NUMBER/SPLIT: 12 8.940E-06
DEPENDENCIES: 58 60 61
REQ. BRANCHES: 2 * 5 * 2
DESCRIPTION: 15-25% 16-20 % NO_AC_LAT
CASE/BRANCH SPLIT: 4.373E-06 4.568E-06

CASE NUMBER/SPLIT: 13 3.227E-07
DEPENDENCIES: 58 60 61
REQ. BRANCHES: 2 * 6 * 2
DESCRIPTION: 15-25% > 20 % NO_AC_LAT
CASE/BRANCH SPLIT: 1.581E-07 1.646E-07

CASE NUMBER/SPLIT: 14 8.020E-06
DEPENDENCIES: 58 60 61
REQ. BRANCHES: 3 * 2 * 2
DESCRIPTION: 25-35% 4-8 % NO_AC_LAT
CASE/BRANCH SPLIT: 5.936E-06 2.084E-06

CASE NUMBER/SPLIT: 15 2.154E-04
DEPENDENCIES: 58 60 61
REQ. BRANCHES: 3 * 3 * 2
DESCRIPTION: 25-35% 8-12 % NO_AC_LAT
CASE/BRANCH SPLIT: 1.448E-04 7.063E-05

CASE NUMBER/SPLIT: 16 1.287E-04
DEPENDENCIES: 58 60 61
REQ. BRANCHES: 3 * 4 * 2
DESCRIPTION: 25-35% 12-16 % NO_AC_LAT
CASE/BRANCH SPLIT: 7.486E-05 5.380E-05

CASE NUMBER/SPLIT: 17 7.364E-06
DEPENDENCIES: 58 60 61

```
    REQ. BRANCHES:   3    *    5    *    2
      DESCRIPTION:  25-35%    16-20 %   NO AC LAT
CASE/BRANCH SPLIT:          3.608E-06 3.756E-06

CASE NUMBER/SPLIT:  18      0.000E+00
     DEPENDENCIES:  58         60         61
    REQ. BRANCHES:   3    *    6    *    2
      DESCRIPTION:  25-35%     > 20 %   NO AC LAT
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00

CASE NUMBER/SPLIT:  19      2.170E-05
     DEPENDENCIES:  58         60         61
    REQ. BRANCHES:   4    *    2    *    2
      DESCRIPTION:  35-45%     4-8 %    NO AC LAT
CASE/BRANCH SPLIT:          1.576E-05 5.936E-06

CASE NUMBER/SPLIT:  20      5.322E-04
     DEPENDENCIES:  58         60         61
    REQ. BRANCHES:   4    *    3    *    2
      DESCRIPTION:  35-45%    8-12 %    NO AC LAT
CASE/BRANCH SPLIT:          3.573E-04 1.749E-04

CASE NUMBER/SPLIT:  21      3.193E-04
     DEPENDENCIES:  58         60         61
    REQ. BRANCHES:   4    *    4    *    2
      DESCRIPTION:  35-45%    12-16 %   NO AC LAT
CASE/BRANCH SPLIT:          1.855E-04 1.338E-04

CASE NUMBER/SPLIT:  22      1.886E-05
     DEPENDENCIES:  58         60         61
    REQ. BRANCHES:   4    *    5    *    2
      DESCRIPTION:  35-45%    16-20 %   NO AC LAT
CASE/BRANCH SPLIT:          9.229E-06 9.629E-06

CASE NUMBER/SPLIT:  23      0.000E+00
     DEPENDENCIES:  58         60         61
    REQ. BRANCHES:   4    *    6    *    2
      DESCRIPTION:  35-45%     > 20 %   NO AC LAT
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00

CASE NUMBER/SPLIT:  24      0.000E+00
     DEPENDENCIES:  58         60         61
    REQ. BRANCHES:   5    *    2    *    2
      DESCRIPTION:  45-55%     4-8 %    NO AC LAT
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00

CASE NUMBER/SPLIT:  25      0.000E+00
     DEPENDENCIES:  58         60         61
    REQ. BRANCHES:   5    *    3    *    2
      DESCRIPTION:  45-55%    8-12 %    NO AC LAT
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00

CASE NUMBER/SPLIT:  26      0.000E+00
     DEPENDENCIES:  58         60         61
```

```
     REQ. BRANCHES:    5    *    4    *    2
       DESCRIPTION:  45-55%    12-16 %   NO_AC_LAT
CASE/BRANCH SPLIT:            0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   27       0.000E+00
      DEPENDENCIES:   58         60          61
     REQ. BRANCHES:    5    *    5    *    2
       DESCRIPTION:  45-55%    16-20 %   NO_AC_LAT
CASE/BRANCH SPLIT:            0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   28       0.000E+00
      DEPENDENCIES:   58         60          61
     REQ. BRANCHES:    5    *    6    *    2
       DESCRIPTION:  45-55%     > 20 %   NO_AC_LAT
CASE/BRANCH SPLIT:            0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   29       1.454E-02
      DEPENDENCIES:   58         60          61
     REQ. BRANCHES:    1    *    2    *    1
       DESCRIPTION:  0-15%     4-8 %     AC_LATE
CASE/BRANCH SPLIT:            0.000E+00 1.454E-02

CASE NUMBER/SPLIT:   30       9.415E-03
      DEPENDENCIES:   58         60          61
     REQ. BRANCHES:    1    *    3    *    1
       DESCRIPTION:  0-15%     8-12 %    AC_LATE
CASE/BRANCH SPLIT:            0.000E+00 9.415E-03

CASE NUMBER/SPLIT:   31       9.056E-03
      DEPENDENCIES:   58         60          61
     REQ. BRANCHES:    1    *    4    *    1
       DESCRIPTION:  0-15%     12-16 %   AC_LATE
CASE/BRANCH SPLIT:            0.000E+00 9.056E-03

CASE NUMBER/SPLIT:   32       4.750E-03
      DEPENDENCIES:   58         60          61
     REQ. BRANCHES:    1    *    5    *    1
       DESCRIPTION:  0-15%     16-20 %   AC_LATE
CASE/BRANCH SPLIT:            0.000E+00 4.750E-03

CASE NUMBER/SPLIT:   33       4.237E-03
      DEPENDENCIES:   58         60          61
     REQ. BRANCHES:    1    *    6    *    1
       DESCRIPTION:  0-15%      > 20 %   AC_LATE
CASE/BRANCH SPLIT:            0.000E+00 4.237E-03

CASE NUMBER/SPLIT:   34       4.381E-05
      DEPENDENCIES:   58         60          61
     REQ. BRANCHES:    2    *    2    *    1
       DESCRIPTION:  15-25%    4-8 %     AC_LATE
CASE/BRANCH SPLIT:            0.000E+00 4.381E-05

CASE NUMBER/SPLIT:   35       4.049E-05
      DEPENDENCIES:   58         60          61
```

REQ. BRANCHES:	2 *	3 *	1
DESCRIPTION:	15-25%	8-12 %	AC_LATE
CASE/BRANCH SPLIT:		0.000E+00	4.049E-05
CASE NUMBER/SPLIT:	36	3.188E-05	
DEPENDENCIES:	58	60	61
REQ. BRANCHES:	2 *	4 *	1
DESCRIPTION:	15-25%	12-16 %	AC_LATE
CASE/BRANCH SPLIT:		0.000E+00	3.188E-05
CASE NUMBER/SPLIT:	37	1.968E-05	
DEPENDENCIES:	58	60	61
REQ. BRANCHES:	2 *	5 *	1
DESCRIPTION:	15-25%	16-20 %	AC_LATE
CASE/BRANCH SPLIT:		0.000E+00	1.968E-05
CASE NUMBER/SPLIT:	38	9.276E-06	
DEPENDENCIES:	58	60	61
REQ. BRANCHES:	2 *	6 *	1
DESCRIPTION:	15-25%	> 20 %	AC_LATE
CASE/BRANCH SPLIT:		0.000E+00	9.276E-06
CASE NUMBER/SPLIT:	39	2.602E-05	
DEPENDENCIES:	58	60	61
REQ. BRANCHES:	3 *	2 *	1
DESCRIPTION:	25-35%	4-8 %	AC_LATE
CASE/BRANCH SPLIT:		0.000E+00	2.602E-05
CASE NUMBER/SPLIT:	40	2.172E-04	
DEPENDENCIES:	58	60	61
REQ. BRANCHES:	3 *	3 *	1
DESCRIPTION:	25-35%	8-12 %	AC_LATE
CASE/BRANCH SPLIT:		0.000E+00	2.172E-04
CASE NUMBER/SPLIT:	41	1.316E-04	
DEPENDENCIES:	58	60	61
REQ. BRANCHES:	3 *	4 *	1
DESCRIPTION:	25-35%	12-16 %	AC_LATE
CASE/BRANCH SPLIT:		0.000E+00	1.316E-04
CASE NUMBER/SPLIT:	42	1.008E-05	
DEPENDENCIES:	58	60	61
REQ. BRANCHES:	3 *	5 *	1
DESCRIPTION:	25-35%	16-20 %	AC_LATE
CASE/BRANCH SPLIT:		0.000E+00	1.008E-05
CASE NUMBER/SPLIT:	43	0.000E+00	
DEPENDENCIES:	58	60	61
REQ. BRANCHES:	3 *	6 *	1
DESCRIPTION:	25-35%	> 20 %	AC_LATE
CASE/BRANCH SPLIT:		0.000E+00	0.000E+00
CASE NUMBER/SPLIT:	44	6.606E-05	
DEPENDENCIES:	58	60	61

```
     REQ. BRANCHES:     4     *     2     *     1
       DESCRIPTION:   35-45%      4-8 %       AC_LATE
CASE/BRANCH SPLIT:                0.000E+00 6.606E-05

 CASE NUMBER/SPLIT:    45      5.351E-04
      DEPENDENCIES:    58          60             61
     REQ. BRANCHES:     4     *     3     *     1
       DESCRIPTION:   35-45%      8-12 %      AC_LATE
CASE/BRANCH SPLIT:                0.000E+00 5.351E-04

 CASE NUMBER/SPLIT:    46      3.251E-04
      DEPENDENCIES:    58          60             61
     REQ. BRANCHES:     4     *     4     *     1
       DESCRIPTION:   35-45%      12-16 %     AC_LATE
CASE/BRANCH SPLIT:                0.000E+00 3.251E-04

 CASE NUMBER/SPLIT:    47      2.583E-05
      DEPENDENCIES:    58          60             61
     REQ. BRANCHES:     4     *     5     *     1
       DESCRIPTION:   35-45%      16-20 %     AC_LATE
CASE/BRANCH SPLIT:                0.000E+00 2.583E-05

 CASE NUMBER/SPLIT:    48      0.000E+00
      DEPENDENCIES:    58          60             61
     REQ. BRANCHES:     4     *     6     *     1
       DESCRIPTION:   35-45%      > 20 %      AC_LATE
CASE/BRANCH SPLIT:                0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    49      0.000E+00
      DEPENDENCIES:    58          60             61
     REQ. BRANCHES:     5     *     2     *     1
       DESCRIPTION:   45-55%      4-8 %       AC_LATE
CASE/BRANCH SPLIT:                0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    50      0.000E+00
      DEPENDENCIES:    58          60             61
     REQ. BRANCHES:     5     *     3     *     1
       DESCRIPTION:   45-55%      8-12 %      AC_LATE
CASE/BRANCH SPLIT:                0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    51      0.000E+00
      DEPENDENCIES:    58          60             61
     REQ. BRANCHES:     5     *     4     *     1
       DESCRIPTION:   45-55%      12-16 %     AC_LATE
CASE/BRANCH SPLIT:                0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    52      0.000E+00
      DEPENDENCIES:    58          60             61
     REQ. BRANCHES:     5     *     5     *     1
       DESCRIPTION:   45-55%      16-20 %     AC_LATE
CASE/BRANCH SPLIT:                0.000E+00 0.000E+00

 CASE NUMBER/SPLIT:    53      0.000E+00
      DEPENDENCIES:    58          60             61
```

```
    REQ. BRANCHES:    5     *     6     *     1
      DESCRIPTION:   45-55%      > 20 %    AC_LATE
CASE/BRANCH SPLIT:            0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   54      0.000E+00
      DESCRIPTION:       OTHERWISE                         $ SHOULD NEVER GO THIS PA
CASE/BRANCH SPLIT:            0.000E+00 0.000E+00


********  QUESTION:   63  H2 DETONATION LATE CONTAINMENT FAILURE
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                              325281
         BRANCHES:              DET_CF    NO
                                  1        2
   REALIZED SPLIT:            7.223E-04 9.994E-01

                     SUMMARY BY CASE

CASE NUMBER/SPLIT:    1      9.558E-01
     DEPENDENCIES:   62
    REQ. BRANCHES:    1
      DESCRIPTION:   NO_BURN
CASE/BRANCH SPLIT:            0.000E+00 9.558E-01

CASE NUMBER/SPLIT:    2      1.276E-03
     DEPENDENCIES:   58         58         58
    REQ. BRANCHES:    4    +     5    +     6
      DESCRIPTION:   35-45%     45-55%     > 55%
CASE/BRANCH SPLIT:            0.000E+00 1.276E-03

CASE NUMBER/SPLIT:    3      2.457E-02
     DEPENDENCIES:   60         60         60
    REQ. BRANCHES:    1    +     2    +     3
      DESCRIPTION:   < 4 %      4-8 %      8-12 %
CASE/BRANCH SPLIT:            0.000E+00 2.457E-02

CASE NUMBER/SPLIT:    4      7.129E-03
     DEPENDENCIES:   60          3          7          56
    REQ. BRANCHES:    4    *     1    *     2    *     2
      DESCRIPTION:   12-16 %    SBO        CNTMT LIM SPRAY
CASE/BRANCH SPLIT:            1.558E-04 6.973E-03

CASE NUMBER/SPLIT:    5      2.174E-03
     DEPENDENCIES:   60
    REQ. BRANCHES:    4
      DESCRIPTION:   12-16 %
CASE/BRANCH SPLIT:            0.000E+00 2.174E-03

CASE NUMBER/SPLIT:    6      6.939E-03
     DEPENDENCIES:   60         60          3          7          56
    REQ. BRANCHES: (  5    +     6)   *     1    *     2    *     2
      DESCRIPTION:   16-20 %    > 20 %     SBO        CNTMT LIM SPRAY
CASE/BRANCH SPLIT:            1.724E-04 6.767E-03

CASE NUMBER/SPLIT:    7      1.697E-03
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```
     DEPENDENCIES:   60
    REQ. BRANCHES:    5
      DESCRIPTION:  16-20 %
CASE/BRANCH SPLIT:          2.687E-04 1.428E-03

CASE NUMBER/SPLIT:    8     4.650E-04
     DEPENDENCIES:   60
    REQ. BRANCHES:    6
      DESCRIPTION:   > 20 %
CASE/BRANCH SPLIT:          1.254E-04 3.396E-04

CASE NUMBER/SPLIT:    9     0.000E+00
      DESCRIPTION:      OTHERWISE                        $ SHOULD NEVER GO THIS PA
CASE/BRANCH SPLIT:          0.000E+00 0.000E+00


******** QUESTION:   64  HYDROGEN BURN LATE CONTAINMENT FAILURE
Q-TYPE/TIMES ASKED:      DEP. CALC. PROB.                                333908
         BRANCHES:          FAILURE   NO_FAILUR
                             1         2
   REALIZED SPLIT:          2.012E-02 9.799E-01

                  SUMMARY BY CASE

CASE NUMBER/SPLIT:    1     7.223E-04
     DEPENDENCIES:   63
    REQ. BRANCHES:    1
      DESCRIPTION:  DET_CF
CASE/BRANCH SPLIT:          7.223E-04 0.000E+00

CASE NUMBER/SPLIT:    2     9.993E-01
      DESCRIPTION:      OTHERWISE
CASE/BRANCH SPLIT:          1.940E-02 9.799E-01


******** QUESTION:   65  CONTAINMENT STATUS AT ACCIDENT PROGRESSION COMPLETION
Q-TYPE/TIMES ASKED:      DEP. INPUT PROB. INPUT PARM.                    338583
         BRANCHES:          EARLY_CF  LATE_CF   VENT      NO_LAT_CF
                             1         2         3         4
   REALIZED SPLIT:          2.415E-01 5.869E-02 2.952E-01 4.049E-01

                  SUMMARY BY CASE

CASE NUMBER/SPLIT:    1     2.281E-01
     DEPENDENCIES:    2
    REQ. BRANCHES:    2
      DESCRIPTION:  FAILED
CASE/BRANCH SPLIT:          2.281E-01 0.000E+00 0.000E+00 0.000E+00

CASE NUMBER/SPLIT:    2     2.851E-04
     DEPENDENCIES:    5
    REQ. BRANCHES:    2
      DESCRIPTION:  NOT ISOL
CASE/BRANCH SPLIT:          2.851E-04 0.000E+00 0.000E+00 0.000E+00
```

```
CASE NUMBER/SPLIT:     3     1.480E-03
     DEPENDENCIES:    24
    REQ. BRANCHES:     1
      DESCRIPTION:   FAILURE
CASE/BRANCH SPLIT:           1.480E-03 0.000E+00 0.000E+00 0.000E+00

CASE NUMBER/SPLIT:     4     1.164E-02
     DEPENDENCIES:    46
    REQ. BRANCHES:     1
      DESCRIPTION:   FAILURE
CASE/BRANCH SPLIT:           1.164E-02 0.000E+00 0.000E+00 0.000E+00

CASE NUMBER/SPLIT:     5     0.000E+00
     DEPENDENCIES:     2         3         9
    REQ. BRANCHES:     1    *    4    *    1
      DESCRIPTION:   INTACT    CRIT ATWS VENT
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00 0.000E+00 0.000E+00

CASE NUMBER/SPLIT:     6     2.304E-02
     DEPENDENCIES:     8         8        49        54
    REQ. BRANCHES:  (  1    +(   2    *    2))  *    1
      DESCRIPTION:   RHR_SPRY  RHR_POOL  NO_PL_BP  DRY-CCI
CASE/BRANCH SPLIT:           0.000E+00 2.303E-03 0.000E+00 2.074E-02

CASE NUMBER/SPLIT:     7     3.841E-01
     DEPENDENCIES:     8         8        49        54
    REQ. BRANCHES:  (  1    +(   2    *    2))  *   /1
      DESCRIPTION:   RHR_SPRY  RHR_POOL  NO_PL_BP /DRY-CCI
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00 0.000E+00 3.841E-01

CASE NUMBER/SPLIT:     8     0.000E+00
     DEPENDENCIES:     8        49         9
    REQ. BRANCHES:     2    *    1    *    2
      DESCRIPTION:   RHR_POOL  POOL BP   NO VENT
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00 0.000E+00 0.000E+00

CASE NUMBER/SPLIT:     9     2.952E-01
     DEPENDENCIES:     9
    REQ. BRANCHES:     1
      DESCRIPTION:   VENT
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00 2.952E-01 0.000E+00

CASE NUMBER/SPLIT:    10     1.025E-03
     DEPENDENCIES:     8                   6         6
    REQ. BRANCHES:     3    *    1    (    1    +    2)
      DESCRIPTION:   NO_RHR    SBO       NO INJECT RCIC
CASE/BRANCH SPLIT:           0.000E+00 1.025E-03 0.000E+00 0.000E+00

CASE NUMBER/SPLIT:    11     2.971E-03
     DEPENDENCIES:     8         3         6
    REQ. BRANCHES:     3    *         *    3
      DESCRIPTION:   NO_RHR    SB        HPCS
CASE/BRANCH SPLIT:           0.000E+00 2.971E-03 0.000E+00 0.000E+00
```

```
CASE NUMBER/SPLIT:   12      5.239E-02
     DEPENDENCIES:    8          6          6
    REQ. BRANCHES:    3    *(    1    +     2)
      DESCRIPTION:  NO_RHR    NO INJECT RCIC
CASE/BRANCH SPLIT:           0.000E+00 5.239E-02 0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   13      0.000E+00
     DEPENDENCIES:    8
    REQ. BRANCHES:    3
      DESCRIPTION:  NO_RHR
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00 0.000E+00 0.000E+00

CASE NUMBER/SPLIT:   14      0.000E+00
      DESCRIPTION:        OTHERWISE                        $ SHOULD NEVER GO THIS PA
CASE/BRANCH SPLIT:           0.000E+00 0.000E+00 0.000E+00 0.000E+00


********  QUESTION:   66  MODE OF LATE HYDROGEN AND OVERPRESSURE CONTAINMENT FAILURE
Q-TYPE/TIMES ASKED:       DEP. CALC. PROB.                         375044
          BRANCHES:          ANCHORAGE PN-D/NoCF
                                 1         2
    REALIZED SPLIT:          2.138E-02 9.787E-01

                    SUMMARY BY CASE

CASE NUMBER/SPLIT:    1      5.869E-02
     DEPENDENCIES:   65
    REQ. BRANCHES:    2
      DESCRIPTION:  LATE_CF
CASE/BRANCH SPLIT:           8.801E-03 4.989E-02

CASE NUMBER/SPLIT:    2      9.229E-01
     DEPENDENCIES:   64
    REQ. BRANCHES:    2
      DESCRIPTION:  NO_FAILUR
CASE/BRANCH SPLIT:           0.000E+00 9.229E-01

CASE NUMBER/SPLIT:    3      6.862E-04
     DEPENDENCIES:   63
    REQ. BRANCHES:    1
      DESCRIPTION:  DET_CF
CASE/BRANCH SPLIT:           0.000E+00 6.862E-04

CASE NUMBER/SPLIT:    4      1.781E-02
      DESCRIPTION:        OTHERWISE
CASE/BRANCH SPLIT:           1.258E-02 5.237E-03


********  QUESTION:   67  DRYWELL FAILURE DUE TO LATE HYDROGEN BURN IN CONTAINMENT
Q-TYPE/TIMES ASKED:       DEP. CALC. PROB.                         410122
          BRANCHES:           DW_FAIL   NO_DW_FAL
                                 1         2
    REALIZED SPLIT:          1.727E-02 9.829E-01
```

```
                    SUMMARY BY CASE

CASE NUMBER/SPLIT:    1      4.420E-02
     DEPENDENCIES:   62
    REQ. BRANCHES:    2
      DESCRIPTION:   LG_BURN
CASE/BRANCH SPLIT:           1.727E-02 2.693E-02

CASE NUMBER/SPLIT:    2      9.559E-01
      DESCRIPTION:       OTHERWISE                          $ NO LARGE BURNS IN CNTMT
CASE/BRANCH SPLIT:           0.000E+00 9.559E-01


********  QUESTION:   68  POOL BYPASS LATE
Q-TYPE/TIMES ASKED:        DEP. INPUT PROB.                           418904
         BRANCHES:              LAT_PL_BP NO_LAT_BP
                                   1         2
   REALIZED SPLIT:           2.877E-01 7.125E-01

                    SUMMARY BY CASE

CASE NUMBER/SPLIT:    1      1.727E-02
     DEPENDENCIES:   67
    REQ. BRANCHES:    1
      DESCRIPTION:   DW_FAIL
CASE/BRANCH SPLIT:           1.727E-02 0.000E+00

CASE NUMBER/SPLIT:    2      9.808E-03
     DEPENDENCIES:   66
    REQ. BRANCHES:    1
      DESCRIPTION:   ANCHORAGE
CASE/BRANCH SPLIT:           9.808E-03 0.000E+00

CASE NUMBER/SPLIT:    3      1.405E-01
     DEPENDENCIES:   55
    REQ. BRANCHES:    2
      DESCRIPTION:   AFTER_VB
CASE/BRANCH SPLIT:           1.405E-01 0.000E+00

CASE NUMBER/SPLIT:    4      1.194E-01
     DEPENDENCIES:   54
    REQ. BRANCHES:    1
      DESCRIPTION:   DRY-CCI
CASE/BRANCH SPLIT:           1.194E-01 0.000E+00

CASE NUMBER/SPLIT:    5      1.266E-02
     DEPENDENCIES:   62         3         3         3         3         3         7
    REQ. BRANCHES:    2    *((  3    +    4    +    6)   +(   1    +    5)   *(   1
      DESCRIPTION:   LG_BURN   OTHER TYP CRIT ATWS OTHERS     SBO        LOOP & SB PRI
CASE/BRANCH SPLIT:           5.271E-04 1.203E-02

CASE NUMBER/SPLIT:    6      7.006E-01
      DESCRIPTION:      OTHERWISE                           $ BYPASS PROB FOR SEQUENC
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CASE/BRANCH SPLIT: 6.793E-05 7.005E-01

APPENDIX H.5

PNPP IPE MAAP PARAMETER FILE

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***********************************************************************
**              MARK III/BWR 6 MAAP 3.0B PARAMETER FILE  07-4       **
**                  ---------------------------------------          **
**                          PERRY POWER PLANT                        **
**                                                                   **
**                             17Feb1992                             **
**                                                                   **
***********************************************************************
**                                                                   **
**              ARRANGED TO WORK WITH MAAP 3.0B REV 7.02PR           **
**                            MAAP72PY.EXE                           **
**                  BRITISH UNITS: FT-LB-HR-DEG F                    **
**                                                                   **
** ALL ELEVATIONS ARE REFERENCED TO SEA LEVEL UNLESS OTHERWISE NOTED. **
**                                                                   **
***********************************************************************
**
*BR
**
******************************************************************************
*ADDITIONAL ENGINEERED SAFEGUARDS
******************************************************************************
**
** ---------------
** INJECTION LOGIC
** ---------------
**
01   60.0    ZHHPCI        ELEVATION OF HIGH DOWNCOMER WATER LEVEL TO TRIP OFF
**                         HPCI
02  651.95   ZHRCIC        ELEVATION OF HIGH DOWNCOMER WATER LEVEL TO TRIP OFF
**                         RCIC @ 200 inches
03  651.95   ZHHPCS        ELEVATION OF HIGH DOWNCOMER WATER LEVEL TO TRIP OFF
**                         HPCS @ 200 inches
04  651.95   ZWSHCD        DESIRED WATER LEVEL FOR CONTROLING HPCI & RCIC FLOW
**                         TO BE USED WITH EVENT CODE( 295 ) @ 200 inches
**
05   0.34D6 PLIMSP(1)      TABLE FOR HEAT CAPACITY TEMPERATURE LIMIT CURVE( HCTL )
06   0.34D6 PLIMSP(2)
07   0.41D6 PLIMSP(3)
08   0.69D6 PLIMSP(4)          SET POINT OF THE FIRST GROUP SRVS
09   1.79D6 PLIMSP(5)
10   3.10D6 PLIMSP(6)                      VS.
11   4.83D6 PLIMSP(7)
12   7.42D6 PLIMSP(8)            SUPPRESSION POOL TEMPERATURE
13   3.78D2 TLIMSP(1)
14   3.67D2 TLIMSP(2)
15   3.64D2 TLIMSP(3)      NORMALLY HIGH SET POINT IS SPECIFIED FOR LOW SP
16   3.61D2 TLIMSP(4)      TEMPERATURE AND LOW SET POINT IS SPECIFIED FOR
17   3.55D2 TLIMSP(5)      HIGH SP TEMPERATURE.
18   3.50D2 TLIMSP(6)
19   3.44D2 TLIMSP(7)
20   3.37D2 TLIMSP(8)
**
21   62.0    ZHFW          HIGH WATER LEVEL TO TRIP FEEDWATER
```

```
**                       USE OF THIS SET POINT IS QUESTIONABLE FOR CURRENT
**                       MAAP 3.0B REV 7 BECAUSE THE WAY FEEDWATER FLOW IS
**                       CALCULATED ENSURES THE WATER LEVEL TO STAY AT THE
**                       NORMAL OPERATING LEVEL, 'ZWSHO'
**
** ----
** RWCU
** ----
**
22   3.0     NRWCU       NUMBER OF REACTOR WATER CLEANUP PUMPS( 0.0 - 4.0, 3.0 )
23   0.43    WVRWCU      CONSTANT RWCU VOLUMETRIC FLOW RATE FOR EACH PUMP
**                       ( 0.41 - 0.44, 0.43)
24   0.0     TDRWCU      TIME DELAY FOR RWCU OPERATION ONCE AN INITIATION SIGNAL
**                       RECEIVED( 0.0 - 30.0, 0.0 )
25   1000.   NTRCHX      NUMBER OF TUBES IN ONE RWCU HEAT EXCHANGER
**                       ( 0.0 - 3000.0, 1000. )
26   10.     NBRCHX      NUMBER OF BAFFLES IN ONE RWCU HEAT EXCHANGER
**                       ( 0.0 - 15.0, 10. )
27   0.01656 XIDTRC      RWCU HEAT EXCHANGER TUBE INNER DIAMETER
**                       ( 0.0 - 0.0254, 0.01656 )
28   0.001245 XTTRC      RWCU HEAT EXCHANGER TUBE WALL THICKNESS
**                       ( 0.0 - 1.65D-3, 1.245D-3 )
29   0.0254 XTCRC        RWCU HEAT EXCHANGER TUBE CENTER TO CENTER SPACING
**                       ( 0.0 - 0.032, 0.0254 )
30   6.518   XSRC        RWCU HEAT EXCHANGER SHELL LENGTH( 0.0 - 16.2, 6.518 )
31   3.522D-4 RGFRC      RWCU HEAT EXCHANGER FOULING FACTOR
**                       ( 0.0 - 1.0D-3, 3.522D-4 )
32   17.3    KTRC        RWCU HEAT EXCHANGER TUBE WALL THERMAL CONDUCTIVITY
**                       ( 0.0 - [illegible]3.0, 17.3 )
33   0.31    XBCRC       RWCU HEAT EXCHANGER BAFFLE CUT LENGTH( 0.0 - 0.5, 0.31 )
34   1.5494  XIDSRC      RWCU HEAT EXCHANGER SHELL INNER DIAMETER
**                       ( 0.0 - 2.0, 1.5494 )
35   0.0     XSTRC       RWCU HEAT EXCHANGER BUNDLE TO SHELL GAP LENGTH
**                       ( 0.0 - 0.05, 0.02 )
36   0.0     NTURC       NTU FOR HEAT EXCHANGER( 0.0 - 1.5, 0.0 )                    **APL
37   2.0     FRCHX       TYPE OF RWCU HEAT EXCHANGER( 1.0 - 2.0, 2.0 )
**                          1 = STRAIGHT TUBE, 2 = U TUBE
38   0.0     ZLRWCU      RWCU TRIP OFF LEVEL
**
**
** --------------
** DRYWELL COOLER
** --------------
**
39   3.0     NFN         NUMBER OF DRYWELL COOLERS ( 3 )
40   20.0    WVFNO       VOLUMETRIC FLOW RATE OF EACH DRYWELL COOLER ( 20 )
41   5.0     TDFAN       TIME DELAY FOR DRYWELL COOLERS ( 5 )
42   1200.   NTFC        NUMBER OF TUBES IN EACH DRYWELL COOLER ( 1200 )
43   180.    ATFC        OUTSIDE AREA OF ALL TUBES IN EACH DRYWELL COOLER ( 180 )
44   1500.   AFINFC      AREA OF ALL FINS IN EACH DRYWELL COOLER ( 1500 )
45   0.5     FFINFC      DRYWELL COOLER FIN EFFICIENCY ( 0.5 )
46   0.001   RGFLHX      DRYWELL COOLER INSIDE FOULING FACTOR ( 0.001 )             ****
47   0.05    XDFNFC      DRYWELL COOLER FIN DIAMETER ( 0.05 )
48   0.001   XTTFC       DRYWELL COOLER TUBE THICKNESS ( 0.001 )
```

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49   240.    KTFC        DRYWELL COOLER THERMAL CONDUCTIVITY ( 240 )
50   10.     AFLMNF      MINIMUM FLOW AREA THROUGH DRYWELL COOLER ( 10 )
51   0.015   XIDTFC      DRYWELL COOLLER TUBE INSIDE DIAMETER ( 0.015 )
52   5.0     NREGFC      NUMBER OF NODES USED TO MODEL EACH DRYWELL COOLER ( 5 )
53   310.    TCWHX       INLET COOLING WATER. I.E. SERVICE WATER TEMPERATURE TO
**                       DRYWELL COOLER ( 310 )
54   110.    WCWFC       INLET COOLING WATER FLOW RATE TO EACH DRYWELL COOLER
**                       ( 110 )
55   8.D5    PHDWDC      HIGH DRYWELL PRESSURE TO TRIP DRYWELL COOLER( 8.D5 )
**
************************************************************************
*AUX BLDG/REACTOR BLDG INPUT
************************************************************************
**
**A MAXIMUM OF 9 NODES CAN BE REPRESENTED--THE NO. OF NODES INODRB IS GIVEN
**IN THE *CONTROL SECTION--NOTE THAT NODE INODRB+1 IS THE ENVIRONMENT BUT
**NO INFORMATION NEED BE ENTERED FOR IT
**THE FOLLOWING ARE MODELED AT PRESENT:
**
**   1.  WATER OVERFLOWS--THE SAME JUNCTIONS ARE USED AS IN THE GAS TRANSFERS
**       AND ARE SPECIFIED IN THE *TOPOLOGY SECTION BELOW; NOTE THAT ENTERING
**       A ONE IN THE APPROPRIATE SPOT MODELS FLOOR DRAINS WHICH INSTANTLY
**       DRAIN ALL ACCUMULATED WATER AWAY
**   2.  H2 BURNS
**   3.  CO2 FIRE SUPPRESSION SYSTEMS
**   4.  SPRAYS
**   5.  NATURAL CIRCULATION, BOTH UNIDIRECTIONAL AND COUNTER-CURRENT
**   6.  TWO HEAT SINKS/NODE-- ONE HEAT SINK REPRESENTS AN "OUTER" WALL WHICH
**       HAS THE NODE IN QUESTION ON ONE SIDE AND A USER-SPECIFIED NODE ON
**       THE OTHER SIDE; THE OTHER HEAT SINK REPRESENTS AN "INNER" WALL WHICH
**       HAS THE NODAL GAS TEMPERATURE ON BOTH SIDES.
**
01   63359.      VOLRB(I) VOLUME OF NODE 1
02   68440.      VOLRB(I) VOLUME OF NODE 2
03   59780.      VOLRB(I) VOLUME OF NODE 3
04   735800.     VOLRB(I) VOLUME OF NODE 4
05   629200.     VOLRB(I) VOLUME OF NODE 5
06   300800.     VOLRB(I) VOLUME OF NODE 6
07   281500.     VOLRB(I) VOLUME OF NODE 7
08   153400.     VOLRB(I) VOLUME OF NODE 8
09   5096200.     VOLRB(I) VOLUME OF NODE 9
**
**   FLOOR AREA ARE USED BOTH FOR WATER DEPTH AND TO REPRESENT CHARACTERISTIC
**   DIMENSION OF COMPT FOR BURN TIMES
**
11     1600.     FLOOR AREA FOR NODE 1
12     1600.     FLOOR AREA FOR NODE 2
13     1920.     FLOOR AREA FOR NODE 3
14    17100.     FLOOR AREA FOR NODE 4
15     9800.     FLOOR AREA FOR NODE 5
16     9200.     FLOOR AREA FOR NODE 6
17     6000.     FLOOR AREA FOR NODE 7
18     6850.     FLOOR AREA FOR NODE 8
19    32600.     FLOOR AREA FOR NODE 9
```

```
**
**   AT PRESENT, ONLY ONE EXTERIOR WALL PER NODE IS MODELED IN THE AUX CODE;
**   IE THE WALL HAS AUX CONDITIONS ON ONE SIDE AND THE ENVIRONMENT ON THE
**   OTHER--EITHER A STEEL OR CONCRETE WALL CAN BE MODELED BY INPUTTING THE
**   APPROPRIATE MATERIAL PROPERTIES
**
21     3510.     OUTER WALL AREA FOR NODE 1
22     3150.     OUTER WALL AREA FOR NODE 2
23     1000.     OUTER WALL AREA FOR NODE 3
24    19780.     OUTER WALL AREA FOR NODE 4
25     7360.     OUTER WALL AREA FOR NODE 5
26    12800.     OUTER WALL AREA FOR NODE 6
27    14060.     OUTER WALL AREA FOR NODE 7
28    57040.     OUTER WALL AREA FOR NODE 8
29    59700.     OUTER WALL AREA FOR NODE 9
**
31    3.2        THICKNESS OF OUTER WALL IN NODE 1
32    3.2        THICKNESS OF OUTER WALL IN NODE 2
33    0.021      THICKNESS OF OUTER WALL IN NODE 3
34    2.0        THICKNESS OF OUTER WALL IN NODE 4
35    3.0        THICKNESS OF OUTER WALL IN NODE 5
36    2.43       THICKNESS OF OUTER WALL IN NODE 6
37    3.0        THICKNESS OF OUTER WALL IN NODE 7
38    0.021      THICKNESS OF OUTER WALL IN NODE 8
39    1.0        THICKNESS OF OUTER WALL IN NODE 9
**
41    0.92       THERMAL CONDUCTIVITY OF OUTER WALL IN NODE 1
42    0.92       THERMAL CONDUCTIVITY OF OUTER WALL IN NODE 2
43    27.0       THERMAL CONDUCTIVITY OF OUTER WALL IN NODE 3
44    0.92       THERMAL CONDUCTIVITY OF OUTER WALL IN NODE 4
45    0.92       THERMAL CONDUCTIVITY OF OUTER WALL IN NODE 5
46    0.92       THERMAL CONDUCTIVITY OF OUTER WALL IN NODE 6
47    0.92       THERMAL CONDUCTIVITY OF OUTER WALL IN NODE 7
48    27.0       THERMAL CONDUCTIVITY OF OUTER WALL IN NODE 8
49    0.032      THERMAL CONDUCTIVITY OF OUTER WALL IN NODE 9
**
51    0.156      SPECIFIC HEAT OF OUTER WALL IN NODE 1
52    0.156      SPECIFIC HEAT OF OUTER WALL IN NODE 2
53    0.12       SPECIFIC HEAT OF OUTER WALL IN NODE 3
54    0.156      SPECIFIC HEAT OF OUTER WALL IN NODE 4
55    0.156      SPECIFIC HEAT OF OUTER WALL IN NODE 5
56    0.156      SPECIFIC HEAT OF OUTER WALL IN NODE 6
57    0.156      SPECIFIC HEAT OF OUTER WALL IN NODE 7
58    0.12       SPECIFIC HEAT OF OUTER WALL IN NODE 8
59    0.123      SPECIFIC HEAT OF OUTER WALL IN NODE 9
**
61    25.1       OUTER WALL HEIGHT IN NODE 1
62    25.7       OUTER WALL HEIGHT IN NODE 2
63    30.0       OUTER WALL HEIGHT IN NODE 3
64    43.0       OUTER WALL HEIGHT IN NODE 4
65    16.0       OUTER WALL HEIGHT IN NODE 5
66    16.7       OUTER WALL HEIGHT IN NODE 6
67    25.3       OUTER WALL HEIGHT IN NODE 7
68    110.0      OUTER WALL HEIGHT IN NODE 8
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69    67.1       OUTER WALL HEIGHT IN NODE 9
**
71    145.0      OUTER WALL DENSITY IN NODE 1
72    145.0      OUTER WALL DENSITY IN NODE 2
73    490.0      OUTER WALL DENSITY IN NODE 3
74    145.0      OUTER WALL DENSITY IN NODE 4
75    145.0      OUTER WALL DENSITY IN NODE 5
76    145.0      OUTER WALL DENSITY IN NODE 6
77    145.0      OUTER WALL DENSITY IN NODE 7
78    490.0      OUTER WALL DENSITY IN NODE 8
79    22.1       OUTER WALL DENSITY IN NODE 9
**
**   THE VENTILATION (OR "SGTS") SYSTEM IS MODELED BY SUPPLYING
**   A FORCED OUT FLOW AND/OR A FORCED IN FLOW--IF AC POWER IS AVAILABLE,
**   THIS FLOW IS ON UNTIL THE FIRE DAMPER SETPOINT(SEE BELOW) IS
**   REACHED IN A COMPARTMENT--THIS SHUTS FLOW DOWN IN THAT COMPT
**   FOR RECIRCULATING FLOWS, ENTER THE APPROPRIATE NODE ON THE SUCTION
**   SIDE IN FIELDS 161-169
**
81    0.0        VENTILATION FLOW OUT OF NODE 1
82    0.0        VENTILATION FLOW OUT OF NODE 2
83    0.0        VENTILATION FLOW OUT OF NODE 3
84    33.33      VENTILATION FLOW OUT OF NODE 4
85    33.33      VENTILATION FLOW OUT OF NODE 5
86    0.0        VENTILATION FLOW OUT OF NODE 6
87    0.0        VENTILATION FLOW OUT OF NODE 7
88    33.33      VENTILATION FLOW OUT OF NODE 8
89    0.0        VENTILATION FLOW OUT OF NODE 9
**
91    0.0        VENTILATION FLOW INTO NODE 1
92    0.0        VENTILATION FLOW INTO NODE 2
93    0.0        VENTILATION FLOW INTO NODE 3
94    0.0        VENTILATION FLOW INTO NODE 4
95    0.0        VENTILATION FLOW INTO NODE 5
96    0.0        VENTILATION FLOW INTO NODE 6
97    0.0        VENTILATION FLOW INTO NODE 7
98    0.0        VENTILATION FLOW INTO NODE 8
99    0.0        VENTILATION FLOW INTO NODE 9
**
101   3200.0     AEROSOL SETTLING AREA IN NODE 1
102   3200.0     AEROSOL SETTLING AREA IN NODE 2
103   3840.0     AEROSOL SETTLING AREA IN NODE 3
104   18810.0    AEROSOL SETTLING AREA IN NODE 4
105   19600.0    AEROSOL SETTLING AREA IN NODE 5
106   18400.0    AEROSOL SETTLING AREA IN NODE 6
107   12000.0    AEROSOL SETTLING AREA IN NODE 7
108   13700.0    AEROSOL SETTLING AREA IN NODE 8
109   65200.0    AEROSOL SETTLING AREA IN NODE 9
**
**   AEROSOL IMPACTION DATA;  IF IMPACTION IS MODELED IN A NODE, THE
**   IMPACTION AREA, DIAMETER (EG., GRATE THICKNESS), AND FLOW AREA MUST
**   ALL BE GIVEN
**
111   135.5      AEROSOL IMPACTION AREA IN NODE 1
```

```
112   135.5      AEROSOL IMPACTION AREA IN NODE 2
113   0.0        AEROSOL IMPACTION AREA IN NODE 3
114   0.0        AEROSOL IMPACTION AREA IN NODE 4
115   110.0      AEROSOL IMPACTION AREA IN NODE 5
116   0.0        AEROSOL IMPACTION AREA IN NODE 6
117   0.0        AEROSOL IMPACTION AREA IN NODE 7
118   0.0        AEROSOL IMPACTION AREA IN NODE 8
119   0.0        AEROSOL IMPACTION AREA IN NODE 9
**
**CHECK--THESE DIAMETERS ARE TOO LARGE FOR GRATING AND WON'T REMOVE VERY
**EFFICIENTLY
121   .0324      AEROSOL IMPACTION DIAMETER IN NODE 1
122   .0324      AEROSOL IMPACTION DIAMETER IN NODE 2
123   0.0        AEROSOL IMPACTION DIAMETER IN NODE 3
124   0.0        AEROSOL IMPACTION DIAMETER IN NODE 4
125   .0324      AEROSOL IMPACTION DIAMETER IN NODE 5
126   0.0        AEROSOL IMPACTION DIAMETER IN NODE 6
127   0.0        AEROSOL IMPACTION DIAMETER IN NODE 7
128   0.0        AEROSOL IMPACTION DIAMETER IN NODE 8
129   0.0        AEROSOL IMPACTION DIAMETER IN NODE 9
**
131   429.5      FLOW AREA AT GRATING ELEV. IN NODE 1
132   429.5      FLOW AREA AT GRATING ELEV. IN NODE 2
133   0.0        FLOW AREA AT GRATING ELEV. IN NODE 3
134   0.0        FLOW AREA AT GRATING ELEV. IN NODE 4
135   1320.0     FLOW AREA AT GRATING ELEV. IN NODE 5
136   0.0        FLOW AREA AT GRATING ELEV. IN NODE 6
137   0.0        FLOW AREA AT GRATING ELEV. IN NODE 7
138   0.0        FLOW AREA AT GRATING ELEV. IN NODE 8
139   0.0        FLOW AREA AT GRATING ELEV. IN NODE 9
**
**  SPRAYS (EG FIRE SPRAYS)--THESE ARE TURNED ON AND OFF MANUALLY USING
**  EVENT CODE 240;  THEY WILL ALSO COME ON IF THE NODAL TEMPERATURE
**  EXCEEDS THE SETPOINT VALUE INPUT BELOW
**
141   0.0        SPRAY MASS FLOW RATE IN NODE 1
142   0.0        SPRAY MASS FLOW RATE IN NODE 2
143   0.0        SPRAY MASS FLOW RATE IN NODE 3
144   0.0        SPRAY MASS FLOW RATE IN NODE 4
145   4812.4        SPRAY MASS FLOW RATE IN NODE 5
**NON-SPRAY FLOW POURED ON CABLE TRAYS NEGLECTED FOR CONSERVATISM
146  0.         SPRAY MASS FLOW RATE IN NODE 6
147  4812.4     SPRAY MASS FLOW RATE IN NODE 7
148  0.0        SPRAY MASS FLOW RATE IN NODE 8
149  24062.     SPRAY MASS FLOW RATE IN NODE 9
**
151  0.0        SPRAY FALL HEIGHT IN NODE 1
152  0.0        SPRAY FALL HEIGHT IN NODE 2
153  0.0        SPRAY FALL HEIGHT IN NODE 3
154  0.0        SPRAY FALL HEIGHT IN NODE 4
155  16.0       SPRAY FALL HEIGHT IN NODE 5
156  0.0        SPRAY FALL HEIGHT IN NODE 6
157  22.6       SPRAY FALL HEIGHT IN NODE 7  [SMALLEST USED]
158  0.0        SPRAY FALL HEIGHT IN NODE 8
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159  18.0      SPRAY FALL HEIGHT IN NODE 9
**
**  SPECIFY THE NODE NO. THAT THE VOL IN NODE 1, 2, ETC., RECEIVES ITS
**  INLET VENT FLOW FROM; USE INODRB+1 FOR ENVIRONMENT WHERE INODRB IS
**  NO. OF NODES IN THE MODEL; USE SMALLER NOS. IF A RECIRCULATING SYSTEM
**  EXISTS (IE TAKES FROM NODE 1 AND PUTS INTO NODE 4)
**
161   10.       INLET VENT FLOW SOURCE NODE FOR NODE 1
162   10.       INLET VENT FLOW SOURCE NODE FOR NODE 2
163   10.       INLET VENT FLOW SOURCE NODE FOR NODE 3
164   10.       INLET VENT FLOW SOURCE NODE FOR NODE 4
165   10.       INLET VENT FLOW SOURCE NODE FOR NODE 5
166   10.       INLET VENT FLOW SOURCE NODE FOR NODE 6
167   10.       INLET VENT FLOW SOURCE NODE FOR NODE 7
168   10.       INLET VENT FLOW SOURCE NODE FOR NODE 8
169   10.       INLET VENT FLOW SOURCE NODE FOR NODE 9
**
**  ENTER A ONE TO INSTANTLY DRAIN ALL WATER FROM NODE; THIS IS CONVENIENT
**  IF THE BLDG HAS AN EFFICIENT DRAIN SYSTEM THAT PUTS ALL THE WATER INTO
**  A LARGE SUMP (EG SEQUOYAH)
**
171  0.0       INSTANT DRAIN PARAMETER FOR NODE 1
172  0.0       INSTANT DRAIN PARAMETER FOR NODE 2
173  0.0       INSTANT DRAIN PARAMETER FOR NODE 3
174  0.0       INSTANT DRAIN PARAMETER FOR NODE 4
175  0.0       INSTANT DRAIN PARAMETER FOR NODE 5
176  0.0       INSTANT DRAIN PARAMETER FOR NODE 6
177  0.0       INSTANT DRAIN PARAMETER FOR NODE 7
178  0.0       INSTANT DRAIN PARAMETER FOR NODE 8
179  0.0       INSTANT DRAIN PARAMETER FOR NODE 9
**
181  -29.5     ELEV. OF FLOOR OF NODE 1 WITH RESPECT TO GROUND LEVEL
182  -29.5     ELEV. OF FLOOR OF NODE 2 WITH RESPECT TO GROUND LEVEL
183  18.0      ELEV. OF FLOOR OF NODE 3 WITH RESPECT TO GROUND LEVEL
184  18.0      ELEV. OF FLOOR OF NODE 4 WITH RESPECT TO GROUND LEVEL
185  -25.0     ELEV. OF FLOOR OF NODE 5 WITH RESPECT TO GROUND LEVEL
186  25.0      ELEV. OF FLOOR OF NODE 6 WITH RESPECT TO GROUND LEVEL
187  -29.5     ELEV. OF FLOOR OF NODE 7 WITH RESPECT TO GROUND LEVEL
188  -29.5     ELEV. OF FLOOR OF NODE 8 WITH RESPECT TO GROUND LEVEL
189  -25.0     ELEV. OF FLOOR OF NODE 9 WITH RESPECT TO GROUND LEVEL
**
**  CO2 MASS FLOWRATE FROM FIRE SUPPRESSION SYSTEM, IF ANY; THIS SYSTEM IS
**  ACTIVATED IF THE NODAL GAS TEMP EXCEEDS THE SETPOINT SPECIFIED BELOW
**
**
**
**
191  0.0       CO2 MASS FLOW RATE FOR NODE 1
192  0.0       CO2 MASS FLOW RATE FOR NODE 2
193  0.0       CO2 MASS FLOW RATE FOR NODE 3
194  0.0       CO2 MASS FLOW RATE FOR NODE 4
195  0.0       CO2 MASS FLOW RATE FOR NODE 5
196  0.0       CO2 MASS FLOW RATE FOR NODE 6
197  0.0       CO2 MASS FLOW RATE FOR NODE 7
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198  0.0        CO2 MASS FLOW RATE FOR NODE 8
199  0.0        CO2 MASS FLOW RATE FOR NODE 9
**
201  9135.0     AREA OF INTERNAL WALLS FOR NODE 1
202  9135.0     AREA OF INTERNAL WALLS FOR NODE 2
203  5550.0     AREA OF INTERNAL WALLS FOR NODE 3
204  10500.0    AREA OF INTERNAL WALLS FOR NODE 4
205  58660.0    AREA OF INTERNAL WALLS FOR NODE 5
206  5700.0     AREA OF INTERNAL WALLS FOR NODE 6
207  14600.0    AREA OF INTERNAL WALLS FOR NODE 7
208  40600.0    AREA OF INTERNAL WALLS FOR NODE 8
209  229750.0   AREA OF INTERNAL WALLS FOR NODE 9
**
211  1.64       THICKNESS OF INTERNAL WALLS FOR NODE 1
212  1.64       THICKNESS OF INTERNAL WALLS FOR NODE 2
213  2.12       THICKNESS OF INTERNAL WALLS FOR NODE 3
214  1.50       THICKNESS OF INTERNAL WALLS FOR NODE 4
215  2.01       THICKNESS OF INTERNAL WALLS FOR NODE 5
216  1.72       THICKNESS OF INTERNAL WALLS FOR NODE 6
217  1.62       THICKNESS OF INTERNAL WALLS FOR NODE 7
218  1.76       THICKNESS OF INTERNAL WALLS FOR NODE 8
219  2.30       THICKNESS OF INTERNAL WALLS FOR NODE 9
**
221  0.92       THERMAL CONDUCTIVITY OF INTERNAL WALLS IN NODE 1
222  0.92       THERMAL CONDUCTIVITY OF INTERNAL WALLS IN NODE 2
223  0.92       THERMAL CONDUCTIVITY OF INTERNAL WALLS IN NODE 3
224  0.92       THERMAL CONDUCTIVITY OF INTERNAL WALLS IN NODE 4
225  0.92       THERMAL CONDUCTIVITY OF INTERNAL WALLS IN NODE 5
226  0.92       THERMAL CONDUCTIVITY OF INTERNAL WALLS IN NODE 6
227  0.92       THERMAL CONDUCTIVITY OF INTERNAL WALLS IN NODE 7
228  0.92       THERMAL CONDUCTIVITY OF INTERNAL WALLS IN NODE 8
229  0.92       THERMAL CONDUCTIVITY OF INTERNAL WALLS IN NODE 9
**
231  0.156      SPECIFIC HEAT OF INTERNAL WALLS IN NODE 1
232  0.156      SPECIFIC HEAT OF INTERNAL WALLS IN NODE 2
233  0.156      SPECIFIC HEAT OF INTERNAL WALLS IN NODE 3
234  0.156      SPECIFIC HEAT OF INTERNAL WALLS IN NODE 4
235  0.156      SPECIFIC HEAT OF INTERNAL WALLS IN NODE 5
236  0.156      SPECIFIC HEAT OF INTERNAL WALLS IN NODE 6
237  0.156      SPECIFIC HEAT OF INTERNAL WALLS IN NODE 7
238  0.156      SPECIFIC HEAT OF INTERNAL WALLS IN NODE 8
239  0.156      SPECIFIC HEAT OF INTERNAL WALLS IN NODE 9
**
241  22.5       HEIGHT OF INTERNAL WALLS IN NODE 1
242  22.5       HEIGHT OF INTERNAL WALLS IN NODE 2
243  41.9       HEIGHT OF INTERNAL WALLS IN NODE 3
244  43.0       HEIGHT OF INTERNAL WALLS IN NODE 4
245  24.9       HEIGHT OF INTERNAL WALLS IN NODE 5
246  17.6       HEIGHT OF INTERNAL WALLS IN NODE 6
247  25.3       HEIGHT OF INTERNAL WALLS IN NODE 7
248  24.0       HEIGHT OF INTERNAL WALLS IN NODE 8
249  18.5       HEIGHT OF INTERNAL WALLS IN NODE 9
**
251  145.0      DENSITY OF INTERNAL WALLS IN NODE 1
```

```
252  145.0     DENSITY OF INTERNAL WALLS IN NODE 2
253  145.0     DENSITY OF INTERNAL WALLS IN NODE 3
254  145.0     DENSITY OF INTERNAL WALLS IN NODE 4
255  145.0     DENSITY OF INTERNAL WALLS IN NODE 5
256  145.0     DENSITY OF INTERNAL WALLS IN NODE 6
257  145.0     DENSITY OF INTERNAL WALLS IN NODE 7
258  145.0     DENSITY OF INTERNAL WALLS IN NODE 8
259  145.0     DENSITY OF INTERNAL WALLS IN NODE 9
**
**  USE INODRB+1 FOR ENVIRONMENT WHERE INODRB IS NO. OF NODES IN THE MODEL;
**  USE SMALLER NOS. IF AN OUTER WALL IN NODE 1 HAS A DIFFERENT NODE ON THE
**  OTHER SIDE; NOTE FOR WALLS THICKER THAN ROUGHLY 1-2 FEET, THE THERMAL
**  BOUNDARY LAYER DOESN'T PENETRATE THE WALL OVER THE TIME OF TYPICAL
**  TRANSIENTS AND ONE CAN FREELY LUMP WALLS WITH DIFFERENT NODES ON THEIR
**  OTHER SIDES
**
261  8.0       NODE # ON OTHER SIDE OF OUTER WALLS OF NODE 1
262  8.0       NODE # ON OTHER SIDE OF OUTER WALLS OF NODE 2
263  9.0       NODE # ON OTHER SIDE OF OUTER WALLS OF NODE 3
264  10.0      NODE # ON OTHER SIDE OF OUTER WALLS OF NODE 4
265  10.0      NODE # ON OTHER SIDE OF OUTER WALLS OF NODE 5
266  9.0       NODE # ON OTHER SIDE OF OUTER WALLS OF NODE 6
267  9.0       NODE # ON OTHER SIDE OF OUTER WALLS OF NODE 7
268  10.0      NODE # ON OTHER SIDE OF OUTER WALLS OF NODE 8
269  10.0      NODE # ON OTHER SIDE OF OUTER WALLS OF NODE 9
**
271  1.D10     TOTAL INIT. MASS OF H2O AVAILABLE FOR FIRE SPRAYS [LAKE]
272  0.0       TOTAL INIT. MASS OF CO2 IN FIRE SUPP. SYSTEM
273  90.0      INITIAL TEMPERATURE OF AUX. BLDG.
274  90.0      AUX BLDG. SPRAY WATER TEMPERATURE [ASSUMED=LAKE]
275  .003      AUX BLDG. SPRAY DROPLET DIAMETER
276  .50       INITIAL REL. HUMIDITY IN AUX. BLDG. COMPARTMENTS
277  90.0      ENVIRONMENT TEMPERATURE [ASSUMED]
278  14.7      ENVIRONMENT/AUX BLDG. PRESSURE
279  1.D10     FIRE DAMPER ACTIVATION TEMPERATURE [NONE]
280  212.      FIRE SPRAY ACTIVATION TEMPERATURE [HIGHEST USED]
281  1.D10     CO2 INJECTION ACTIVATION TEMPERATURE
282  1.D10     TOTAL AEROSOL MASS TO TEAR OUT SGTS FILTERS
283  100.0     SGTS FILTER DF
**@@@ REV 6 ADDITION FOR HYDROGEN COMBUSTION (TEMPORARY INPUT)
284  6.2       XRBRRB(1) CHARACTERISTIC RADIUS OF NODE 1 FOR H2 BURNS
285  7.8       NODE 2
286  4.2       NODE 3
287  20.8      NODE 4
288  10.1      NODE 5
289  14.7      NODE 6
290  14.5      NODE 7
291  7.5       NODE 8
292  24.[illegible]     NODE 9
****
294  6.1       XHBRRB(1) CHARACTERISTIC HEIGHT OF NODE 1 FOR H2 BURNS
****             THIS IS FLOOR TO CEILING (NOT IGNITER TO CEILING)
295  202.      NODE 2
296  25.7      NODE 3
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297  17.3       NODE 4
298  8.5        NODE 5
299  6.1        NODE 6
300  6.1        NODE 7
301  4.0        NODE 8
302  14.2       NODE 9
****
304  0.D0       XIGRB  AVERAGE DISTANCE FROM IGNITERS TO CEILING
305  0.D0       NODE 2
306  0.D0       NODE 3
307  0.D0       NODE 4
308  0.D0       NODE 5
309  0.D0       NODE 6
310  0.D0       NODE 7
311  0.D0       NODE 8
312  0.D0       NODE 9
**
*************************************************************************
*COMPTA (MARKIII-MIDDLE WETWELL COMPARTMENT)
*************************************************************************
01   620.5      ZCAF        ELEVATION OF HCU DECK
02    170900.   VOLCA       VOLUME OF COMPARTMENT A
03       .50    RELHCA      RELATIVE HUMIDITY IN COMPT. A
04      4000.   ACAF        AREA OF COMPT. A FLOOR
05      2534.   ACACB       FLOW AREA BETWEEN COMPT. A AND COMPT. B
06      1900.   AWWCA       FLOW AREA BETWEEN WETWELL AND COMPT. A
07   620.5      ZWCAWW      CURB HEIGHT ON MIDDLE DECK
08     3.060    PPUR(1)     HYDROGEN MIXING COMPRESSOR FLOW CURVE:
09     4.071    PPUR(2)       - PRESSURE
10     4.941    PPUR(3)
11     6.149    PPUR(4)
12     6.668    PPUR(5)
13     7.159    PPUR(6)
14     7.300    PPUR(7)
15     7.412    PPUR(8)
16     47400.   WVPUR(1)      - FLOW
17     45300.   WVPUR(2)
18     43500.   WVPUR(3)
19     39600.   WVPUR(4)
20     34800.   WVPUR(5)
21     29700.   WVPUR(6)
22     20400.   WVPUR(7)
23     17200.   WVPUR(8)
24      2       NPURP       NUMBER OF HYDROGEN MIXING COMPRESSORS
25     -1.E10   ZLPUR       LOW WATER (LOCA) SIGNAL FOR DRYWELL PURGE
26      1.E10   PDWPUR      HIGH DRYWELL PRESSURE (LOCA) FOR DRYWELL PURGE
27      1.E10   PDPUR       PRESSURE DIFFERENTIAL SET POINT FOR DRYWELL PURGE
28      .00833  TDPUR       TIME DELAY FOR DRYWELL PURGE
29      13      NIGCA       NUMBER OF IGNITERS IN THE COMPT A
30      22.5    XIGCA       AVERAGE DISTANCE FROM FLOOR TO IGNITER
31      10      NIGBCA      NUMBER OF IGNITERS IN WETWELL SEEN BY COMPT. A
33      6828.   ASEDCA      AEROSOL SEDIMENTATION AREA
34      1200.   AIMPCA      COMPT A TOTAL IMPACTION AREA
35       .0208  XDIMCA      COMPT A MINIMUM GRATE DIAMETER (OR THICKNESS)
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36     2260.    AGRACA     COMPT A FLOW AREA THRU GRATE
37       .0     NHOOPA     NUMBER OF TENDONS IN HOOP DIRECTION
38       .0     XTREHA     VOLUME OF REBAR PER UNIT AREA OF OUTER WALL
39       .0     XTREZA     VOLUME OF REBAR PER UNIT AREA OF OUTER WALL
40       .0     XDHOPA     DIAMETER OF HOOP TENDONS
41       .0     ZACYL      HEIGHT OF THE CYLINDRICAL PART OF COMPT A WALL
42       .0     XDZFA      DISPLACEMENT IN AXIAL DIRECTION
43       .0     XDRFA      SAME AS 42 FOR THE RADIAL DIRECTION
44     29.4     XRBRCA     CHARACTERISTIC RADIUS OF COMP A FOR H2 BURNS
45     69.      XHBRCA     CHARACTERISTIC HEIGHT OF COMP A FOR H2 BURNS
**                         THIS IS FLOOR TO CEILING
**
**  *************************************************************************
*COMPTB (MARKIII-UPPER WETWELL COMPARTMENT)
*****************************************************************************
01   689.5      ZCBF       ELEVATION OF OPERATING DECK
02    823640.   VOLCB      VOLUME OF COMPT. B
03      .50     RELHCB     RELATIVE HUMIDITY IN COMPT.B
04   689.5      ZWCBWW     ELEVATION AT THE TOP OF THE UPPER DECK CURB
05      3427.   AWCB       UPPER POOL WATER SURFACE AREA
06       .0     PCPUR(1)   CONTAINMENT PURGE FAN CURVE:
07       .0     PCPUR(2)
08       .0     PCPUR(3)
09       .0     PCPUR(4)
10       .0     PCPUR(5)
11       .0     PCPUR(6)
12       .0     PCPUR(7)
13       .0     PCPUR(8)
14       .0     WVCPUR(1)
15       .0     WVCPUR(2)
16       .0     WVCPUR(3)
17       .0     WVCPUR(4)
18       .0     WVCPUR(5)
19       .0     WVCPUR(6)
20       .0     WVCPUR(7)
21       .0     WVCPUR(8)
22     32830.   VOLUPD     VOLUME OF WATER IN UPPER POOL DUMP
23   636.483    ZLUPD      LOW WATER L1 (LOCA) SIGNAL FOR UPPER POOL DUMP
24    16.58     PDWUPD     HIGH DRYWELL PRESSURE (LOCA) FOR UPPER POOL DUMP
25      .1445   TDDUMP     TOTAL TIME FOR UPPER POOL DUMP
26      .5088   TDUPD      TIME DELAY FOR UPPER POOL DUMP
27   667.343    ZCBF1      EFFECTIVE ELEVATION OF UPPER POOL FLOOR
28     39960.   VLAUPD     VOLUME OF WATER IN UPPER POOL AFTER UPPER POOL DUMP
29    47        NIGCB      NUMBER OF IGNITERS IN THE COMPT B
30    13.8      XIGCB      AVERAGE DISTANCE FROM FLOOR TO IGNITER
31    13        NIGBCB     NUMBER OF IGNITERS IN COMPT A SEEN IN COMPT B
33     10000.   ASEDCB     AEROSOL SEDIMENTATION AREA
         950.   AIMPCB     COMPT B TOTAL IMPACTION AREA
        .0208   XDIMCB     COMPT B MINIMUM GRATE DIAMETER (OR THICKNESS)
        1695.   AGRACB     COMPT B FLOW AREA THRU GRATE
3.       .0     NHOOPB     NUMBER OF TENDONS IN HOOP DIRECTION
38       .0     XTREHB     VOLUME OF REBAR PER UNIT AREA OF OUTER WALL
39       .0     XTREZB     VOLUME OF REBAR PER UNIT AREA OF OUTER WALL
40       .0     XDHOPB     DIAMETER OF HOOP TENDONS
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41      .0     ZBCYL      HEIGHT OF THE CYLINDRICAL PART
42      .C     XDZFB      DISPLACEMENT IN AXIAL DIRECTION
43      .0     XDRFB      SAME AS 42 FOR THE RADIAL DIRECTION
44    61.6     XRBRCB     CHARACTERISTIC RADIUS OF COMP B FOR H2 BURNS
45    67.5     XHBRCB     CHARACTERISTIC HEIGHT OF COMP B FOR H2 BURNS
**                        THIS IS FLOOR TO CEILING
**
*****************************************************************************
*CONCRETE PROPERTIES
*****************************************************************************
01     2420.   TCNMP     CONCRETE AVERAGE MELTING TEMP (SOLIDUS - LIQUIDUS)
02      501.   LHDEC     REACTION ENERGY FOR CONCRETE DECOMPOSITION
03      215.   LHCN      LATENT HEAT FOR CONCRETE MELTING
04   .35800    MFCN(1)   SIO2
05   .31300    MFCN(2)   CAO
06   .03600    MFCN(3)   AL2O3
07   .01220    MFCN(4)   K2O
08   .00082    MFCN(5)   NA2O
09   .00690    MFCN(6)   MGO+MNO+TIO2
10   .01440    MFCN(7)   FE2O3 -> FEO+O2
11   .0        MFCN(8)   FE
12   .00014    MFCN(9)   CR2O3
13   .04700    MFCN(10)  H2O
14   .21154    MFCN(11)  CO2
15   .0        MFCN(12)  O2
16    18.      DCSRCN    DENSITY OF REBAR IN CONCRETE
17      .20    CPCNO     SPECIFIC HEAT OF CONCRETE
**REMAINDER OF THE QUANTITIES ARE USED IN THE MARK III CONTAINMENT FAILURE
**MODEL AND NEED NOT BE SUPPLIED IF THE "SIMPLE" MODEL IS USED
**(SEE WETWELL SECTION)
18      .0     PTEN      ELASTIC YOUNGS MODULUS FOR TENDONS
19      .0     PEREB     ELASTIC YOUNGS MODULUS FOR REBAR
20      .0     PEPTEN    PLASTIC YOUNGS MODULUS FOR TENDONS
21      .0     PEPREB    PLASTIC YOUNGS MODULUS FOR REBAR
22      .0     PSSPH     PRESTRESS ON HOOP TENDONS
23      .0     PSSPZ     PRESTRESS ON AXIAL TENDONS
24      .0     PSSYHT    TENDON YIELD STRESS
25      .0     PSSYHR    REBAR YIELD STRESS
26      .0     PSSFHT    TENDON ULTIMATE STRESS
27      .0     PSSFHR    REBAR ULTIMATE STRESS
28      .0     PEL       ELASTIC YOUNGS MODULUS FOR LINER
29      .0     PEPL      PLASTIC YOUNGS MODULUS FOR LINER
30      .0     PSSYHL    LINER YIELD STRESS
31      .0     PSSFHL    LINER FAILURE STRESS
**
*****************************************************************************
*CONTROL CARDS
*****************************************************************************
01       3     IBWR       CONTAINMENT TYPE (MARK 1,2, OR 3)
02      49     IRSTW      UNIT NUMBER TO WRITE RESTART FILE (MAIN)
03      50     IHUW       UNIT NUMBER TO WRITE RESTART FILE (HEATUP)
04      40     IPOUT      UNIT NUMBER TO WRITE PROGRAM OUTPUT FILE
05       1     IPLT1      OPTION FOR PLTMAP VARIABLE LABEL LENGTH IN PLOT FILES
**                        =1, USE A8 FORMAT (AS HAS BEEN IN THE PAST)
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**                              =2, USE A15 FORMAT - MAX LENGTH OF ANY MAAP
**                              COMMON BLOCK NAME.
06   1000      IPTSMX     MAXIMUM NUMBER OF PLOTTED POINTS
07     10      IPTSPK     MAXIMUM NUMBER OF PLOT POINTS TRACED FOR FULL
**                              SCALE SPIKE
08    150      IPTSAV     NUMBER OF POINTS SAVED FOR NON-CHANGING PLOT [DUMMY]
09      0      ISUMM      SUMMARY DATA(0=ALL EVENTS,1=SHORTER LIST)
10     14      ISUM       SUMMARY FILE NUMBER
11      1      IRUNG      1=1ST ORDER R-K,2=2ND ORDER R-K
12   1         IFREEZ     1=DO FREEZE FRONT CALC. (0=NO CALC.)
13   12        INPGRP     NUMBER OF TRACE GAS TYPES (FISSION PRODUCTS)
14   0         IRET   WRITE RETAIN PLOT FILE (NOT USED)
15   10        IFPPLT RETAIN PLOT FILE UNIT NUMBER (NOT USED)
25   5         IH         NUMBER OF RADIAL NODES
26   10        JH         NUMBER OF AXIAL NODES
153  0         IINERT     0=CONTAINMENT NOT INERTED,1=CONTAINMENT INERTED
154  1         IILPCI     LPCI INJECTION: 1=LOWER PLENUM, 0=DOWNCOMER
155  0         INODRB     NUMBER OF REACTOR BLDG NODES
156  13        IAUXW      FILE TO WRITE AUX CODE INFO
157  0         IAUXR      FILE TO READ AUX CODE INFO
327  1         JNTGRT     = 1 : UTILIZE CONSISTENT TIMESTEPS BETWEEN
**                              DIFFUN (ICALL=3) AND INTGRT &
**                              DIFFP (ICALL=3) AND INTEGFP
**                              = 0 : UTILIZE THE SMALLER TIMESTEP OF
**                              DIFFUN (ICALL=3) AND LIMITING VALUES IN INTGRT &
**                              DIFFP (ICALL=3) AND LIMITING VALUES IN INTGFP
328  0      ITDLIM        = 1 : UTILIZE USER-INPUT CRITICAL PARAMETERS
**                              IN DETERMINING THE LIMITING TIMESTEP
**                              = 0 : UTILIZE ORIGINAL HARD-WIRED CRITICAL PARAMETERS
329  0      ISORT         = 1 : SORT OUT INTEGRATION DIAGNOSTIC FIGURES OF MERIT
**                              FOR ALL THE THERMAL HYDRAULIC VARIABLES IN INTGRT
**                              IN AVERAGE FRACTIONAL CHANGES OF STATE VARIABLES &
**                              FREQUENCY OF SIGN CHANGES OF THEIR RATES
**                              = 0 : NO SORTING
336  0      IEMBAL        = 1 : DO & PRINT OUT PRIMARY SYSTEM / CONTAINMENT
**                              MASS / ENERGY BALANCES
**                              = 0 : NO BALANCE
337  0       ICRBAL       = 1 : DO & PRINT OUT CORE NODE ENERGY BALANCE
**                              = 0 : NO BALANCE
**
*************************************************************************
*DRYWELL
*************************************************************************
01      .40     RELHDW     RELATIVE HUMIDITY IN DRYWELL
02    269900.   VOLDW      VOLUME OF DRYWELL
03   583.5      ZDWF       ELEVATION AT DRYWELL FLOOR
04      2449.   ADWF       AREA OF DRYWELL FLOOR
05   599.00     ZEDWWW     ELEVATION OF WEIR WALL BETWEEN DRYWELL AND WETWELL
06    8         NIGDW      NUMBER OF INGITERS IN THE DRYWELL
07   62.4       XIGDW      AVERAGE DISTANCE FROM FLOOR TO IGNITER
09     14851.   ASEDDW     AEROSOL SEDIMENTATION AREA
10     .017     ADWLEK     DRYWELL LEAK AREA - FOR MI, MII & MIII
11      9618.   AIMPDW     DRYWELL TOTAL IMPACTION AREA
12     .0208    XDIMDW     DRYWELL MINIMUM GRATE DIAMETER (OR THICKNESS)
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13     17175.   AGRADW      DRYWELL FLOW AREA THRU GRATE
14    3.08D-5   XDDROP      SPRAY DROPLET DIAMETER FOR CONTAINMENT SPRAYS
15     .0       XHSPDW      SPRAY FALL HEIGHT IN DRYWELL
16     .0       PCFAIL      CONTAINMENT FAILURE PRESSURE ONLY FOR MI & MIT
17   33.9       XRBRDW      CHARACTERISTIC RADIUS OF DRYWELL FOR H2 BURNS
18   77.1       XHBRDW      CHARACTERISTIC HEIGHT OF DRYWELL FOR H2 BURNS
**                          THIS IS FLOOR TO CEILING
**
************************************************************************
*ENGINEERED SAFEGUARDS
************************************************************************
01      1       NLPCI1      NUMBER OF RHR PUMPS IN LPCI LOOP 1
02      1       NLPCI2      NUMBER OF RHR PUMPS IN LPCI LOOP 2
03      1       NLPCI3      NUMBER OF RHR PUMPS IN LPCI LOOP 3
04      1       NLPCSP      NUMBER OF LPCS PUMPS
05       .0     NOT USED
06   7980       VMNCST      MIN. WATER VOLUME IN CST FOR HPCS/RCIC XFER
07    .01627    VWCST       SPECIFIC VOLUME OF CST WATER
08    .0        PHPCI(1)    HPCI PUMP FLOW CURVE            - N/A AT PY
09    .0        PHPCI(2)
10    .0        PHPCI(3)
11    .0        PHPCI(4)
12    .0        PHPCI(5)
13    .0        PHPCI(6)
14    .0        PHPCI(7)
15    .0        PHPCI(8)
16    .0        WVHPCI(1)
17    .0        WVHPCI(2)
18    .0        WVHPCI(3)
19    .0        WVHPCI(4)
20    .0        WVHPCI(5)
21    .0        WVHPCI(6)
22    .0        WVHPCI(7)
23    .0        WVHPCI(8)
24      291.    PLPCI(1)     RHR PUMP FLOW CURVE:
25      290.    PLPCI(2)       - PRESSURE
26      277.    PLPCI(3)
27      260.    PLPCI(4)
28      238.    PLPCI(5)
29      217.    PLPCI(6)
30      191.    PLPCI(7)
31      126.    PLPCI(8)
32          .0  WVLPCI(1)      - VOLUMETRIC FLOW
33     9023.    WVLPCI(2)
34    16040.    WVLPCI(3)
35    24060.    WVLPCI(4)
36    32080.    WVLPCI(5)
37    40100.    WVLPCI(6)
38    48120.    WVLPCI(7)
39    63360.    WVLPCI(8)
40      499.    PLPCS(1)     LPCS PUMP CURVE:
41      498.    PLPCS(2)       - PRESSURE
42      450.    PLPCS(3)
43      429.    PLPCS(4)
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44	390.	PLPCS(5)	
45	346.	PLPCS(6)	
46	381.	PLPCS(7)	
47	249.	PLPCS(8)	
48	.0	WVLPCS(1)	- VOLUMETRIC FLOW
49	9624.	WVLPCS(2)	
50	24060.	WVLPCS(3)	
51	32080.	WVLPCS(4)	
52	40100.	WVLPCS(5)	
53	48120.	WVLPCS(6)	
54	56140.	WVLPCS(7)	
55	59750.	WVLPCS(8)	
56	1387.	PHPCS(1)	HPCS PUMP CURVE:
57	1386.	PHPCS(2)	- PRESSURE
58	1212.	PHPCS(3)	
59	1104.	PHPCS(4)	
60	996.	PHPCS(5)	
61	779.	PHPCS(6)	
62	541.	PHPCS(7)	
63	271.	PHPCS(8)	
64	.0	WVHPCS(1)	- VOLUMETRIC FLOW
65	4812.	WVHPCS(2)	
66	16040.	WVHPCS(3)	
67	24060.	WVHPCS(4)	
68	32080.	WVHPCS(5)	
69	40100.	WVHPCS(6)	
70	48120.	WVHPCS(7)	
71	56140.	WVHPCS(8)	
72	1514.7	PRCIC(1)	RCIC PUMP 700 GPM FLOW CURVE:
73	1192.	PRCIC(2)	- PRESSURE
74	165.	PRCIC(3)	
75	14.7	PRCIC(4)	
76	14.7	PRCIC(5)	
77	14.7	PRCIC(6)	
78	14.7	PRCIC(7)	
79	14.7	PRCIC(8)	
80	5614.	WVRCIC(1)	- VOLUMETRIC FLOW
81	5614.	WVRCIC(2)	
82	5614.	WVRCIC(3)	
83	0.	WVRCIC(4)	
84	0.	WVRCIC(5)	
85	0.	WVRCIC(6)	
86	0.	WVRCIC(7)	
87	0.	WVRCIC(8)	
88	-1.E10	ZLHPCI	LOW WATER INITIATION FOR HPCI - N/A FOR PERRY
89	1.E10	PSHPCI	HIGH DRYWELL PRESSURE SET POINT FOR HPCI
90	1.E10	TDHPCI	TIME DELAY FOR HPCI
91	1.E10	PHHPCI	MINIMUM PRESSURE FOR HPCI TURBINE
92	645.925	LHPCS	LOW WATER LEVEL L2 INITIATION FOR HPCS
93	16.58	PSHPCS	HIGH DRYWELL PRESSURE SET POINT FOR HPCS
94	.0075	TDHPCS	TIME DELAY FOR HPCS
95	636.483	ZLLPCI	LOW WATER LEVEL L1 INITIATION FOR LPCI
96	16.58	PSLPCI	HIGH DRYWELL PRESSURE SET POINT FOR LPCI
97	.0103	TDLPCI	TIME DELAY FOR LPCI

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98    -1.E10    PLLPCI     RPV-WETWELL PRESS DIFFERENCE TO KEEP ADS OPEN
99   636.483    ZLLPCS     LOW WATER INITIATION FOR LPCS
100   16.58     PSLPCS     HIGH DRYWELL PRESSURE SET POINT FOR LPCS
101     .0103   TDLPCS     TIME DELAY FOR LPCS
102   -1.E10    PLLPCS     RPV-WETWELL PRESS DIFFERENCE TO CLOSE ADS VALVES
103  645.925    ZLRCIC     LOW WATER LEVEL 2 INITIATION FOR RCIC
104    1.E10    PSRCIC     HIGH DRYWELL PRESSURE SET POINT FOR RCIC
105     .0083   TDRCIC     TIME DELAY FOR RCIC
106   44.7      PHRCIC     MINIMUM VESSEL PRESSURE FOR RCIC TURBINE
107   68.       HCST       ENTHALPY OF CST
108   3.64E6    WSWHX      EMERG SERVICE WATER FLOW RATE THRU RHR HTXs
** THE SAFETY RELIEF VALVES MUST BE ENTERED IN ORDER OF INCREASING
** PRESSURE ACTUATION SET POINTS
109      .1151  ASRV1      FLOW AREA OF RELIEF VALVE TYPE #1
110      .1151  ASRV2      FLOW AREA OF RELIEF VALVE TYPE #2
111      .1151  ASRV3      FLOW AREA OF RELIEF VALVE TYPE #3
112      .1151  ASRV4      FLOW AREA OF RELIEF VALVE TYPE #4
** IF THE AREA OF GROUP #5 IS INPUT AS A NEGATIVE NUMBER THEN THE VALVE
** WILL DISCHARGE DIRECTLY INTO THE DRYWELL, IF POSITIVE IT WILL
** DISCHARGE INTO THE SUPPRESSION POOL
113      .1151       ASRV5      FLOW AREA OF RELIEF VALVE TYPE #5
114    1        NSRV1      NUMBER OF TYPE #1 RELIEF VALVES
115    4        NSRV2      NUMBER OF TYPE #2 RELIEF VALVES
116    4        NSRV3      NUMBER OF TYPE #3 RELIEF VALVES
117    9        NSRV4      NUMBER OF TYPE #4 RELIEF VALVES
118    1        NSRV5      NUMBER OF TYPE #5 RELIEF VALVES
119    1        NADS1      NUMBER OF ADS VALVES IN GROUP 1
120    1        NADS2      NUMBER OF ADS VALVES IN GROUP 2
121    2        NADS3      NUMBER OF ADS VALVES IN GROUP 3
122    4        NADS4      NUMBER OF ADS VALVES IN GROUP 4
**        RELIEF VALVES ARE GROUPED WITH NSRV VALVES IN EACH GROUP
**        PARAMETERS 123-127 ARE THE HIGH END TRIP PRESSURES
**        PARAMETERS 270-274 ARE THE LOW END TRIP PRESSURES
**        PARAMETERS 164-168 ARE THE DEAD BANDS
123  1129.7     PSRV1      PRESSURE SETPOINT FOR #1 RELIEF VALVE
124  1129.7     PSRV2      PRESSURE SETPOINT FOR #2 RELIEF VALVE
125  1129.7     PSRV3      PRESSURE SETPOINT FOR #3 RELIEF VALVE
126  1139.7     PSRV4      PRESSURE SETPOINT FOR #4 RELIEF VALVE
127  1119.7     PSRV5      PRESSURE SETPOINT FOR #5 RELIEF VALVE
128  636.483    ZLADS      LOW WATER LEVEL L1 FOR INITIATION OF ADS
129    1.E10    PSADS      HIGH DRYWELL PRESSURE SET POINT FOR ADS
130     .0325   TDADS      TIME DELAY FOR ADS ACTUATION
131    1.E10    TCHPCI     INLET TEMP LIMIT FOR HPCI           - N/A FOR PERRY
132  571.56     ZCLHPS     PUMP CENTER LINE ELEVATION FOR HPCS
133  571.56     ZCLLPI     PUMP CENTER LINE ELEVATION FOR LPCI
134  571.56     ZCLLPS     PUMP CENTER LINE ELEVATION FOR LPCS
135  185        TCRCIC     INLET TEMP LIMIT FOR RCIC
136   85        TWSW       EMERG SERVICE WATER TEMP (RHR HEAT EXCHANGERS,TCOLD)
137     .00278  TDDG1      HPCS LOAD DELAY TIME FOR DIESEL
138     .00139  TDDG2      LPCI LOAD DELAY TIME FOR DIESEL
139     .00417  TDDG3      LPCS LOAD DELAY TIME FOR DIESEL
**
**        THE HPSW SYSTEM CAN BE USED TO MODEL ANY INJECTION MODE SUCH AS
**        SERVICE WATER OR FIRE WATER, THE SYSTEM IS TOTALLY DEFINED BELOW
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**          FOR THE FIRE WATER ALTERNATE INJECTION
**
143   48.        HWHPSW      ENTHALPY OF FIRE WATER
144   .016072    VWHPSW      SPEC VOL OF FIRE WATER
145  135.7       PHPSW(1)    DIESEL DRIVEN FIRE WATER PUMP CURVE:
146  134.7       PHPSW(2)      - PRESSURE
147   14.7       PHPSW(3)
148   14.7       PHPSW(4)
149   14.7       PHPSW(5)
150   14.7       PHPSW(6)
151   14.7       PHPSW(7)
152   14.7       PHPSW(8)
153      0.      WVHPSW(1)     - VOLUMETRIC FLOW
154   1604.      WVHPSW(2)
155   7074.      WVHPSW(3)
156   7074.      WVHPSW(4)
157   7074.      WVHPSW(5)
158   7074.      WVHPSW(6)
159   7074.      WVHPSW(7)
160   7074.      WVHPSW(8)
161   16.58      PDWSPR      DRYWELL PRES SET PT FOR MARK III CONTAINMNT SPRAYS
162   23.55      PWWSPR      WETWELL PRES SET PT FOR MARK III CONTAINMNT SPRAYS
163     .1908    TDSPR       TIME DELAY FOR MARK III CONTAINMENT SPRAYS
164  137         PDSRV1      DEAD BAND FOR CLOSURE OF SRV#1
165  167         PDSRV2      DEAD BAND FOR CLOSURE OF SRV#2
166  100         PDSRV3      DEAD BAND FOR CLOSURE OF SRV#3
167  100         PDSRV4      DEAD BAND FOR CLOSURE OF SRV#4
168  107         PDSRV5      DEAD BAND FOR CLOSURE OF SRV#5
185  1514.7      PTURRI(1)   PPS-PWW VS. STEAM FLOW TO RCIC TURBINE
186  1192        PTURRI(2)
187   165        PTURRI(3)
188    74.7      PTURRI(4)
189    74.7      PTURRI(5)
190    74.7      PTURRI(6)
191    74.7      PTURRI(7)
192    74.7      PTURRI(8)
193  41490       WSTRCI(1)
194  34200       WSTRCI(2)
195  11000       WSTRCI(3)
196   6957       WSTRCI(4)
197   6957       WSTRCI(5)
198   6957       WSTRCI(6)
199   6957       WSTRCI(7)
200   6957       WSTRCI(8)
201    1.E10     PHTURH      HIGH TURBINE EXHAUST PRESSURE FOR HPCI
202   34.7       PHTURR      HIGH TURBINE EXHAUST PRESSURE FOR RCIC
203     .0                   NOT USED
204  481.3       PHLPCI      LOW RPV PRESSURE PERMISSIVE FOR LPCI
205  497.4       PHLPCS      LOW RPV PRESSURE PERMISSIVE FOR LPCS
206  593.242     ZHISP       HIGH SUPP. POOL LEVEL TRIP FOR HPCS/RCIC SUCT XFER
207   -1.E10     ZLSPR       LOW VESSEL WATER LEVEL FOR CONTAINMENT SPRAYS
208  750         NTHX        NUMBER OF TUBES IN RHR HTX
209    8         NBHX        NUMBER OF BAFFLES IN RHR HTX
210     .0725    XIDTHX      TUBE ID FOR RHR HTX
```

211	.00542	XTCHX	TUBE CENTER TO CENTER SPACING FOR RHR HTX
212	.1042	XTCHX	TUBE CENTER TO CENTER SPACING FOR RHR HTX
213	19.42	XSHX	SHELL LENGTH FOR RHR HTX
214	.0025	RGFOUL	FOULING FACTOR FOR RHR HTX
215	9.3	KTHX	THERMAL CONDUCTIVITY FOR TUBE WALL (RHR HTX)
216	1.75	XBCHX	BAFFLE CUT LENGTH FOR RHR HTX
217	4.667	XIDSHX	SHELL ID FOR RHR HTX
218	.0729	XSTHX	BUNDLE TO SHELL GAP LENGTH FOR RHR HTX
219	.0	NTUHX1	NTU FOR RHR HTX #1
220	.0	NTUHX2	NTU FOR RHR HTX #2
221	2	NHX1	NUMBER OF RHR LOOP #1 HTX
222	2	NHX2	NUMBER OF RHR LOOP #2 HTX
223	2	FHX	TYPE OF RHR HTX(1=STRAIGHT TUBE,2=U TUBE)
224	24.8	TDBATT	BATTERY OPERATION TIME FOR STATION BLACK-OUT
233	.0	ZHDLPI1	LPCI NPSH FOR GIVEN FLOWS
234	1.3	ZHDLPI2	
235	1.0	ZHDLPI3	
236	.7	ZHDLPI4	
237	.5	ZHDLPI5	
238	.5	ZHDLPI6	
239	.5	ZHDLPI7	
240	.5	ZHDLPI8	
241	.0	ZHDLPS1	LPCS NPSH FOR GIVEN FLOWS
242	5.5	ZHDLPS2	
243	3.2	ZHDLPS3	
244	2.5	ZHDLPS4	
245	2.0	ZHDLPS5	
246	1.7	ZHDLPS6	
247	1.5	ZHDLPS7	
248	1.5	ZHDLPS8	
249	574.833	ZCLRCI	PUMP CENTER LINE ELEVATION FOR RCIC
250	-1.E10	ZCLHPI	PUMP CENTER LINE ELEVATION FOR HPCI
251	.0		NOT USED
252	.0		NOT USED
253	.0		NOT USED
254	.0	TGDWHX(1)	COOLING CURVE FOR DRYWELL COOLERS
255	.0	TGDWHX(2)	TEMP IN DRYWELL VS. HEAT LOSS RATE
256	.0	TGDWHX(3)	
257	.0	TGDWHX(4)	
258	.0	TGDWHX(5)	
259	.0	TGDWHX(6)	
260	.0	TGDWHX(7)	
261	.0	TGDWHX(8)	
262	.0	QGDWHX(1)	HEAT LOSS RATE FOR DRYWELL COOLERS
263	.0	QGDWHX(2)	
264	.0	QGDWHX(3)	
265	.0	QGDWHX(4)	
266	.0	QGDWHX(5)	
267	.0	QGDWHX(6)	
268	.0	QGDWHX(7)	
269	.0	QGDWHX(8)	
270	1125.7	PSRVL1	LOW END PRESSURE SETPOINT FOR #1 RELIEF VALVE
271	1125.7	PSRVL2	LOW END PRESSURE SETPOINT FOR #2 RELIEF VALVE
272	1125.7	PSRVL3	LOW END PRESSURE SETPOINT FOR #3 RELIEF VALVE

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273  1135.7    PSRVL4     LOW END PRESSURE SETPOINT FOR #4 RELIEF VALVE
274  1115.7    PSRVL5     LOW END PRESSURE SETPOINT FOR #5 RELIEF VALVE
**
***********************************************************************
**
**@@@ REV 5 ADDITION, THIS EVTMES SECTION IS NEW
**
** THIS EVTMES SECTION DEFINES THE MAAP EVENT MESSAGES FOR EVENT CODES
**  1 THRU 250 IN THE PWR CODE, AND 1 THRU 300 IN THE BWR CODE.
**
** EXPECTED FORMAT OF THIS SECTION IS;
**
**  <NUMBER>  <FLAG>  <MESSAGE>
**
**  WHERE  <NUMBER> IS THE EVENT CODE NUMBER
**           <FLAG> IS THE EVENT FLAG
**        <MESSAGE> IS THE EVENT CODE MESSAGE
**
** EVENT FLAG TOKENS "1", "0", "T", "F", "TRUE", AND "FALSE" ARE
** ACCEPTABLE.  BE SURE TO END THIS SECTION WITH THE KEYWORD "END".
** ** COMMENTING ARE ALLOWED, AND THE FOLLOWING EVENT MESSAGES ARE THE
** DEFAULT MAAP MESSAGES, AS PRESENT IN MAAP BLOCK DATA.  HENCE, THIS
** SECTION IS NOT NECESSARY IF YOU LEAVE EVENT MESSAGES UNMODIFIED.
** NOTE THAT THE CHARACTER "!" IS TREATED AS A END OF LINE DELIMITOR,
** AND EVERYTHING AFTER "!" IS IGNORED.
**
***********************************************************************
*EVTMES
**
**   1 THRU 199 SET BY MAAP, NOT SETTABLE BY THE USER (INTERNAL CODES)
** 200 THRU 399 EITHER SET BY MAAP OR THE USER (EXTERNAL CODES)
** 400 THRU 699 USER DEFINED EVENT CODES
** THEREFORE, CHECK THE NEW LIST PROVIDED IN THIS FILE.
** YOU MAY NEED TO RENUMBER SOME OF YOUR PREVIOUS USER-DEFINED CODES
** TO AVOID COLLISIONS WITH NEW EXTERNAL CODES.
**
** EXPECTED FORMAT OF THIS SECTION IS;
**
**  <NUMBER>  <FLAG>  <MESSAGE>
**
**  WHERE  <NUMBER> IS THE EVENT CODE NUMBER
**           <FLAG> IS THE EVENT FLAG
**        <MESSAGE> IS THE EVENT CODE MESSAGE
**
** EVENT FLAG TOKENS "1", "0", "T", "F", "TRUE", AND "FALSE" ARE
** ACCEPTABLE.  BE SURE TO END THIS SECTION WITH THE KEYWORD "END".
** ** COMMENTING ARE ALLOWED, AND THE FOLLOWING EVENT MESSAGES ARE THE
** DEFAULT MAAP MESSAGES, AS PRESENT IN MAAP BLOCK DATA.  HENCE, THIS
** SECTION IS NOT NECCESSARY IF YOU LEAVE EVENT MESSAGES UNMODIFIED.
** NOTE THAT THE CHARACTER "!" IS TREATED AS A END OF LINE DELIMITOR,
** AND EVERYTHING AFTER "!" IS IGNORED.
**
  1  T  1 SRV (1113/1073 - 936) OPEN   (GRP 1)
  1  F  1 SRV (1113/1073 - 936) CLOSED (GRP 1)
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2	T	4 SRVs (1113 - 946) OPEN (GRP 2)
2	F	4 SRVs (1113 - 946) CLOSED (GRP 2)
3	T	4 SRVs (1113 - 1013) OPEN (GRP 3)
3	F	4 SRVs (1113 - 1013) CLOSED (GRP 3)
4	T	9 SRVs (1123 - 1023) OPEN (GRP 4)
4	F	9 SRVs (1123 - 1023) CLOSED (GRP 4)
5	T	ADS PERMISSIBLE-LP PUMP ON
6	T	ADS SIGNAL-LOW WATER, HIGH DW PRESSURE
7	T	HPCI ON
7	F	HPCI OFF
8	T	VESSEL FAILED
9	T	HIGH VESSEL PRESSURE SCRAM
10	T	SCRAM SIGNAL RECEIVED
11	T	SHROUD WATER SATURATED
11	F	SHROUD WATER SUBCOOLED
12	T	LOWER PLENUM WATER SATURATED
12	F	LOWER PLENUM WATER SUBCOOLED
13	T	LPCI LOOP 2 ON
13	F	LPCI LOOP 2 OFF
14	T	RHR HTX. #1 ON
14	F	RHR HTX. #1 OFF
15	T	RHR HTX. #2 ON
15	F	RHR HTX. #2 OFF
16	T	CORE PLATE FAILURE
17	T	CALL HEATUP
17	F	NO LONGER CALLING HEATUP
18	T	HIGH WATER LEVEL IN SUPP. POOL
18	F	NO LONGER HIGH LEVEL IN SP
19	T	FEEDWATER PUMP TRIPPED
19	F	FEEDWATER ON
20	T	ADS ON
20	F	ADS OFF
21	T	CORIUM CONTACTING PEDESTAL FLOOR
22	T	EX-VESSEL STEAM EXPLOSION IN PEDESTAL
23	T	INITIATION SIGNAL RECVD FOR LPCI #2
23	F	INITIATION SIGNAL LOST FOR LPCI #2
24	T	SUCTION PRESS LIMIT REACHED ON LPCI #2
25	T	DIESEL LOADING PERMISSIBLE FOR HPCS
26	T	HPCS ON
27	F	HPCS OFF
27	T	LPCI LOOP 1 ON
27	F	LPCI LOOP 1 OFF
28	T	LPCS ON
28	F	LPCS OFF
29	T	RCIC ON
29	F	RCIC OFF
30	T	CORE UNCOVERED
30	F	CORE COVERED
31	T	SHROUD WATER LEVEL < ELEVATION AT TOP OF JET PUMP
31	F	SHROUD WATER LEVEL > ELEVATION AT TOP OF JET PUMP
32	T	DIESEL LOADING PERMISSIBLE FOR LPCI
33	T	DIESEL LOADING PERMISSIBLE FOR LPCS
34	T	HP INJECTION SUCTION FROM SUPPRESSION POOL
34	F	HP INJECTION SUCTION FROM CST

35 T CORIUM IN LOWER PLENUM QUENCHED
35 F CORIUM IN LOWER PLENUM NOT QUENCHED
36 T CORIUM PRESENT IN LOWER PLENUM
36 F CORIUM NOT PRESENT IN LOWER PLENUM
37 T LOW WATER LEVEL IN CST
37 F NORMAL WATER LEVEL IN CST
38 T CORIUM AND WATER PRESENT IN LOWER PLENUM
38 F CORIUM AND WATER NOT PRESENT IN LOWER PLENUM
39 T LEVEL 8 HIGH WATER LEVEL
39 F RESET LEVEL 8 TRIP
40 T INITIATION SIGNAL RECVD FOR HPCI
40 F INITIATION SIGNAL LOST FOR HPCI
41 T INITIATION SIGNAL RECVD FOR HPCS
41 F INITIATION SIGNAL LOST FOR HPCS
42 T INITIATION SIGNAL RECVD FOR LPCI #1
42 F INITIATION SIGNAL LOST FOR LPCI #1
43 T INITIATION SIGNAL RECVD FOR LPCS
43 F INITIATION SIGNAL LOST FOR LPCS
44 T INITIATION SIGNAL RECVD FOR RCIC
44 F INITIATION SIGNAL LOST FOR RCIC
45 T HPCI TURBINE PUMP TRIPPED
45 F HPCI TURBINE PUMP NOT TRIPPED
46 T SUCTION PRESS LIMIT REACHED ON HPCS
47 T SUCTION PRESS LIMIT REACHED ON LPCI #1
48 T SUCTION PRESS LIMIT REACHED ON LPCS
49 T RCIC TURBINE PUMP TRIPPED
49 F RCIC TURBINE PUMP NOT TRIPPED
50 T LPCI #1 TO DRYWELL SPRAYS - OPEN
50 F LPCI #1 TO DRYWELL SPRAYS - CLOSED
51 T LPCI #2 TO DRYWELL SPRAYS - OPEN
51 F LPCI #2 TO DRYWELL SPRAYS - CLOSED
52 T LPCI #1 TO WETWELL SPRAYS - OPEN
52 F LPCI #1 TO WETWELL SPRAYS - CLOSED
53 T LPCI #2 TO WETWELL SPRAYS - OPEN
53 F LPCI #2 TO WETWELL SPRAYS - CLOSED
54 T LPCI #1 TO VESSEL - OPEN
54 F LPCI #1 TO VESSEL - CLOSED
55 T LPCI #2 TO VESSEL - OPEN
55 F LPCI #2 TO VESSEL - CLOSED
56 T LPCI #1 TO SUPPRESSION POOL - OPEN
56 F LPCI #1 TO SUPPRESSION POOL - CLOSED
57 T LPCI #2 TO SUPPRESSION POOL - OPEN
57 F LPCI #2 TO SUPPRESSION POOL - CLOSED
58 T LPCI LOOP 3 ON
58 F LPCI LOOP 3 OFF
59 T INITIATION SIGNAL RECVD FOR LPCI #3
59 F INITIATION SIGNAL LOST FOR LPCI #3
60 T SUCTION PRESS LIMIT REACHED ON LPCI #3
61 T 1 SRV (1103/1033 - 926) OPEN (GRP 5)
61 F 1 SRV (1103/1033 - 926) CLOSED (GRP 5)
62 T MSIV CLOSED
62 F MSIV OPEN
63 T LOSS OF AC POWER (LOCKED)
64 T REACTOR SCRAMMED

65	T	RECIRC PUMP TRIPPED
66	T	TURBINE STOP VALVES CLOSED
66	F	TURBINE STOP VALVES OPEN
67	T	LP PUMP PERMISSIBLE FOR WETWELL SPRAYS
68	T	WETWELL SPRAYS(MARKIII) ON
68	F	WETWELL SPRAYS(MARKIII) OFF
69	T	HPSW INJECTION ON
69	F	HPSW INJECTION OFF
70	T	PERMISSIBLE FOR RPT
71	T	CRD PUMP ON
71	F	CRD PUMP OFF
75	T	WATER IN CORE SATURATED
75	F	WATER IN CORE SUBCOOLED
76	T	CORIUM ENTRAINED IN PEDESTAL
76	F	CORIUM NO LONGER ENTRAINED IN PEDESTAL
77	T	WATER ENTRAINED IN PEDESTAL
77	F	WATER NO LONGER ENTRAINED IN PEDESTAL
78	T	LOW LEVEL TRIP FOR HPCI
78	F	RESET LOW LEVEL TRIP FOR HPCI
79	T	HIGH DRYWELL PRESSURE FOR HPCI
79	F	RESET HIGH DW PRESS. FOR HPCI
80	T	LOW LEVEL TRIP FOR HPCS
80	F	RESET LOW LEVEL TRIP FOR HPCS
81	T	HIGH DRYWELL PRESSURE FOR HPCS
81	F	RESET HIGH DW PRESS. FOR HPCS
82	T	LOW LEVEL TRIP FOR RCIC
82	F	RESET LOW LEVEL TRIP FOR RCIC
83	T	HIGH DRYWELL PRESSURE FOR RCIC
83	F	RESET HIGH DW PRESS. FOR RCIC
84	T	LOW LEVEL TRIP FOR LPCI
84	F	RESET LOW LEVEL TRIP FOR LPCI
85	T	HIGH DRYWELL PRESSURE FOR LPCI
85	F	RESET HIGH DW PRESS. FOR LPCI
86	T	LPCI FLOW > 0
86	F	LPCI FLOW = 0
87	T	LPCS FLOW > 0
87	F	LPCS FLOW = 0
88	T	DRYWELL VENT OPEN
88	F	DRYWELL VENT CLOSED
89	T	FIRST CALL TO ICRUST
90	T	LOW LEVEL TRIP FOR LPCS
90	F	RESET LOW LEVEL TRIP FOR LPCS
91	T	HIGH DRYWELL PRESSURE FOR LPCS
91	F	RESET HIGH DW PRESS. FOR LPCS
92	T	HIGH RCIC TURBINE EXHAUST
92	F	NO LONGER HIGH RCIC TURB. EXHAUST
93	T	LOW WATER TRIP FOR ADS
93	F	RESET LOW WATER TRIP FOR ADS
94	T	HIGH DRYWELL PRESSURE TRIP FOR ADS
94	F	RESET HIGH DRYWELL PRESSURE TRIP FOR ADS
95	T	PEDESTAL DOWNCOMER HAS FAILED
95	F	PEDESTAL DOWNCOMER NOT FAILED
96	T	AUX CONDENSER ON
96	F	AUX CONDENSER OFF

97	T	HIGH RPV PRESS INITIATION FOR ISO COND
97	F	NO HIGH RPV PRESS INITIATION FOR ISO COND
98	T	RX BLDG FIRE SPRAYS ON
98	F	RX BLDG FIRE SPRAYS OFF
99	T	RX BLDG CO2 FIRE SUPPRESSION ON
99	F	RX BLDG CO2 FIRE SUPPRESSION OFF
100	T	H2 BURNING IN RX BLDG
100	F	H2 NOT BURNING IN RX BLDG
101	T	BURNING IN PEDESTAL
101	F	BURNING OVER IN PEDESTAL
102	T	CORIUM TEMP. ABOVE CONCRETE MELTING IN PD
102	F	CORIUM TEMP. BELOW CONCRETE MELTING IN PD
103	T	WATER IN PEDESTAL
103	F	NO WATER IN PEDESTAL
104	T	WATER SATURATED IN PEDESTAL
104	T	WATER NO LONGER SATURATED IN PEDESTAL
105	T	CORIUM QUENCHED IN PEDESTAL
105	F	CORIUM NOT QUENCHED IN PEDESTAL
106	T	CORIUM AND WATER PRESENT IN PEDESTAL
106	F	NO CORIUM OR NO WATER PRESENT IN PD
107	T	CORIUM TEMP < CORIUM MELTING POINT IN PD
107	F	CORIUM TEMP > CORIUM MELTING POINT IN PD
108	T	FIRST CALL TO FREEZE
109	T	WETWELL VENT OPEN
109	F	WETWELL VENT CLOSED
110	T	RCIC SUCTION FROM SUPPRESSION POOL
110	F	RCIC SUCTION FROM CST
111	T	PRIMARY SYSTEM COUPLED
111	F	PRIMARY SYSTEM NOT COUPLED
112	T	MWLP G.T. MWMIN
112	F	MWLP L.E. MWMIN
113	T	ADS VALVES OPEN DUE TO LOW DRYWELL PRESSURE
113	F	ADS VALVES CLOSED DUE TO HI DRYWELL PRESSURE
114	T	ADS VALVES OPEN DUE TO HI RPV PRESSURE
114	F	ADS VALVES CLOSED DUE TO LOW RPV PRESSURE
115	T	START TO CALL FISSION PRODUCT MODELS
115	F	FISSION PRODUCT MODELS NOT CALLED
116	T	BURNING IN DRYWELL
116	F	BURNING OVER IN DRYWELL
117	T	CORIUM TEMP. ABOVE CONCRETE MELTING IN DW
117	F	CORIUM TEMP. BELOW CONCRETE MELTING IN DW
118	T	REVERSE FLOW THROUGH SUPPRESSION POOL VENTS
118	F	NORMAL FLOW RESTORED IN SUPPRESSION POOL VENTS
119	T	WATER IN DRYWELL
119	F	NO WATER IN DRYWELL
120	T	PDW > PPD
120	F	PDW < PPD
121	T	WATER TEMP. ABOVE SATURATION IN DRYWELL
121	F	WATER TEMP. BELOW SATURATION IN DRYWELL
122	T	CORIUM QUENCHED IN DRYWELL
122	F	CORIUM NOT QUENCHED IN DRYWELL
123	T	CORIUM AND WATER PRESENT IN DRYWELL
123	F	CORIUM AND WATER NOT IN DRYWELL
124	T	CORIUM TEMP < CORIUM MELTING POINT IN DW

124	F	CORIUM TEMP > CORIUM MELTING POINT IN DW
125	T	LOW DWNCMR - L PLEN PATH OPEN FOR CIRC
125	F	LOW DWNCMR - L PLEN PATH NOT OPEN FOR CIRC
126	T	UP DWNCMR - SEP PATH OPEN FOR CIRC
126	F	UP DWNCMR - SEP PATH NOT OPEN FOR CIRC
127	T	RECIRC LOOP OPEN FOR CIRC
127	F	RECIRC LOOP NOT OPEN FOR CIRC
128	T	TEMP LIMIT REACHED ON HPCI SUCT ON
128	F	TEMP LIMIT NOT REACHED ON HPCI SUCTION
129	T	HIGH TURB EXHAUST FOR HPCI
129	F	RESET HIGH TURB EXHAUST FOR HPCI
130	T	LOW RPV PRESSURE FOR HPCI
130	F	RESET LOW RPV PRESSURE FOR HPCI
131	T	BURNING IN WETWELL
131	F	BURNING OVER IN WETWELL
132	T	CORIUM TEMP > CONCRETE MELTING IN WETWELL
132	F	CORIUM TEMP < CONCRETE MELTING IN WETWELL
133	T	VACUUM BREAKERS OPEN
133	F	VACUUM BREAKERS CLOSED
134	T	WATER IN SUPP. POOL
135	T	SUPPRESSION POOL SATURATED
135	F	SUPPRESSION POOL NO LONGER SATURATED
136	T	CORIUM QUENCHED IN WETWELL
136	F	CORIUM TEMP. ABOVE WATER SATURATION IN WW
137	T	CORIUM AND WATER PRESENT IN WETWELL
138	T	CORIUM TEMP < CORIUM MELTING POINT IN WW
138	F	CORIUM TEMP > CORIUM MELTING POINT IN WW
139	T	RX BLDG FIRE WATER DEPLETED
139	F	RX BLDG FIRE WATER NOT DEPLETED
140	T	SUPP POOL LEVEL BELOW VENT PIPE
140	F	SUPP POOL LEVEL ABOVE VENT PIPE
141	T	TOP VENT OPEN (MIII)
142	T	TOP VENT COVERED (MIII)
143	T	TOP VENT OPEN & MID VENT COVERED (MIII)
144	T	MID VENT OPEN & BOTTOM VENT COVERED (MIII)
145	T	BOTTOM VENT OPEN (MIII)
146	T	MWCA > 0 KG.
146	F	MWCA = 0 KG.
147	T	BURNING IN MIDDLE CONTAINMENT (MIII)
147	F	BURNING OVER IN MIDDLE CONTAINMENT (MIII
148	T	MWCB > 0 KG.
148	F	MWCB = 0 KG.
149	T	BURNING IN UPPER CONTAINMENT (MIII)
149	F	BURNING OVER IN UPPER CONTAINMENT (MIII)
150	T	RX BLDG CO2 SUPPRESSION DEPLETED
150	F	RX BLDG CO2 SUPPRESSION NOT DELETED
151	T	RX BLDG DAMPERS CLOSED
151	F	RX BLDG DAMPERS OPEN
152	T	HYDROGEN MIXING SYSTEM ON
152	F	HYDROGEN MIXING SYSTEM ON
153	T	LOCA SIGNAL FOR UPPER POOL DUMP
154	T	UPPER POOL DUMP ACTIVATED
155	T	CONTAINMENT PURGE ON
155	F	CONTAINMENT PURGE OFF

156 T IGNITERS HAVE POWER
156 F IGNITERS DO NOT HAVE POWER
157 T HYDROGEN MIXING SYSTEM PERMISSIBLE
157 F HYDROGEN MIXING SYSTEM NO PERMISSIBLE
158 T END OF UPPER POOL DUMP
159 T BATTERY POWER UNAVAILABLE
159 F BATTERY POWER AVAILABLE
160 T LOW LEVEL FOR SCRAM
161 T HIGH DRYWELL PRESSURE FOR SCRAM
162 T HIGH DRYWELL PRESSURE FOR SPRAYS
163 T HIGH WETWELL PRESSURE FOR SPRAYS
164 T HIGH SUPP. POOL TEMP. FOR RCIC
165 T LOW RPV PRESSURE FOR RCIC
166 T CORIUM IN WETWELL
166 F CORIUM NOT IN WETWELL
168 T DRYWELL LEAK HAS PLUGGED
168 F DRYWELL LEAK NOT PLUGGED
169 T PLUGGED LEAK PATH BLOWN OPEN
169 F PLUGGED LEAK PATH NOT BLOWN OPEN
170 T CORE RADIAL REGION 1 HAS BLOCKED
170 F CORE RADIAL REGION 1 NOT BLOCKED
171 T CORE RADIAL REGION 2 HAS BLOCKED
171 F CORE RADIAL REGION 2 NOT BLOCKED
172 T CORE RADIAL REGION 3 HAS BLOCKED
172 F CORE RADIAL REGION 3 NOT BLOCKED
173 T CORE RADIAL REGION 4 HAS BLOCKED
173 F CORE RADIAL REGION 4 NOT BLOCKED
174 T CORE RADIAL REGION 5 HAS BLOCKED
174 F CORE RADIAL REGION 5 NOT BLOCKED
175 T INITIATION SIGNAL RECEIVED FOR DRYWELL COOLERS
175 F INITIATION SIGNAL LOST FOR DRYWELL COOLERS
176 1 REACTOR WATER CLEANUP SYSTEM(RWCU) ON
176 0 REACTOR WATER CLEANUP SYSTEM(RWCU) OFF
177 1 INITIATION SIGNAL RECEIVED FOR RWCU
177 0 INITIATION SIGNAL LOST FOR RWCU
178 1 TRIGGER SIGNAL RECEIVED FOR HPCI
178 0 TRIGGER SIGNAL LOST FOR HPCI
179 T SGTS FILTER AEROSOL LOADING EXCEEDED
179 F SGTS FILTER AEROSOL LOADING NOT EXCEEDED
180 1 TRIGGER SIGNAL RECEIVED FOR HPCS
180 0 TRIGGER SIGNAL LOST FOR HPCS
181 1 TRIGGER SIGNAL RECEIVED FOR RCIC
181 0 TRIGGER SIGNAL LOST FOR RCIC
182 1 HIGH LEVEL TRIP FOR FEED WATER
182 0 RESET HIGH LEVEL TRIP FOR FEED WATER
183 1 LPCI #1 TO RAD WASTE - OPEN
183 0 LPCI #1 TO RAD WASTE - CLOSE
184 1 LPCI #2 TO RAD WASTE - OPEN
184 0 LPCI #2 TO RAD WASTE - CLOSE
185 1 HIGH WATER LEVEL RESET SIGNAL RECEIVED FOR HPCI
185 0 HIGH WATER LEVEL RESET SIGNAL NOT RECEIVED FOR HPCI
186 1 HIGH WATER LEVEL RESET SIGNAL RECEIVED FOR RCIC
186 0 HIGH WATER LEVEL RESET SIGNAL NOT RECEIVED FOR RCIC
187 1 HIGH WATER LEVEL RESET SIGNAL RECEIVED FOR HPCS

187 0 HIGH WATER LEVEL RESET SIGNAL NOT RECEIVED FOR HPCS
188 1 DRYWELL COOLERS ON
188 0 DRYWELL COOLERS OFF
190 T CONT FAILED IN WW DUE TO STRAIN
191 T CONT FAILED IN WW DUE TO OVERPRESSURE
192 T CONT FAILED IN CA DUE TO STRAIN
193 T CONT FAILED IN CA DUE TO OVERPRESSURE
194 T CONT FAILED IN CB DUE TO STRAIN
195 T CONT FAILED IN CB DUE TO OVERPRESSURE
196 T SHUTDOWN COOLING ON
196 F SHUTDOWN COOLING OFF
** NOTE EVENT CODE 197 IS SET ONLY FOR ONE TIMESTEP IN WHICH
** EITHER THE REACTOR VESSEL OR CONTAINMENT FAILED.
197 T EITHER REACTOR VESSEL OR CONTAINMENT JUST FAILED
198 T AUTOMATIC PLOT SCALING IS ON
198 F EQUALLY SPACED PLOT SCALING IS ON
199 T CONTAINMENT FAILURE OR VENT OPEN
200 T HPCI MAN ON
200 F HPCI NOT MAN ON
201 T HPCI LOCKED OFF
201 F HPCI NOT LOCKED OFF
202 T LPCI LOOP 1 MAN ON
202 F LPCI LOOP 1 NOT MAN ON
203 T LPCI LOOP 1 LOCKED OFF
203 F LPCI LOOP 1 NOT LOCKED OFF
204 T HPCS MAN ON
204 F HPCS NOT MAN ON
205 T HPCS LOCKED OFF
205 F HPCS NOT LOCKED OFF
206 T LPCS MAN ON
206 F LPCS NOT MAN ON
207 T LPCS LOCKED OFF
207 F LPCS NOT LOCKED OFF
208 T FEEDWATER MAN ON
208 F FEEDWATER NOT MAN ON
209 T FEEDWATER MAN OFF
209 F FEEDWATER NOT LOCKED OFF
210 T RCIC MAN ON
210 F RCIC NOT MAN ON
211 T RCIC LOCKED OFF
211 F RCIC NOT LOCKED OFF
212 T TURBINE STOP VALVE CLOSED
212 F TURBINE STOP VALVE OPEN
213 T TURBINE BYPASS CLOSED
213 F TURBINE BYPASS OPEN
214 T MSIVS MAN OPEN
214 F MSIVS NOT MAN OPEN
215 T MSIVS LOCKED CLOSED
215 F MSIVS NOT LOCKED CLOSED
216 T PEDESTAL DOWNCOMER FAILED
216 F PEDESTAL DOWNCOMER NOT FAILED
217 T 1 SRV (1113/1073 - 936) MAN OPEN (GRP 1)
217 F 1 SRV (1113/1073 - 936) NOT MAN OPEN (GRP 1)
218 T 1 SRV (1113/1073 - 936) LOCKED CLOSED (GRP 1)

218	F	1 SRV (1113/1073 - 936) NOT LOCKED CLOSED	(GRP 1)
219	T	4 SRVs (1113 - 946) MAN OPEN	(GRP 2)
219	F	4 SRVs (1113 - 946) NOT MAN OPEN	(GRP 2)
220	T	4 SRVs (1113 - 946) LOCKED CLOSED	(GRP 2)
220	F	4 SRVs (1113 - 946) NOT LOCKED CLOSED	(GRP 2)
221	T	4 SRVs (1113 - 1013) MAN OPEN	(GRP 3)
221	F	4 SRVs (1113 - 1013) NOT MAN OPEN	(GRP 3)
222	T	4 SRVs (1113 - 1013) LOCKED CLOSED	(GRP 3)
222	F	4 SRVs (1113 - 1013) NOT LOCKED CLOSED	(GRP 3)
223	T	9 SRVs (1123 - 1013) MAN OPEN	(GRP 4)
223	F	9 SRVs (1123 - 1013) NOT MAN OPEN	(GRP 4)
224	T	9 SRVs (1123 - 1013) LOCKED CLOSED	(GRP 4)
224	F	9 SRVs (1123 - 1013) NOT LOCKED CLOSED	(GRP 4)
225	T	ADS MAN OPEN	
225	F	ADS NOT MAN OPEN	
226	T	ADS LOCKED CLOSED	
226	F	ADS NOT LOCKED CLOSED	
227	T	RHR HTX #1 MAN ON	
227	F	RHR HTX #1 NOT MAN ON	
228	T	RHR HTX #1 LOCKED OFF	
228	F	RHR HTX #1 NOT LOCKED OFF	
229	T	RHR HTX #2 MAN ON	
229	F	RHR HTX #2 NOT MAN ON	
230	T	RHR HTX #2 LOCKED OFF	
230	F	RHR HTX #2 NOT LOCKED OFF	
231	T	LPCI LOOP 2 MAN ON	
231	F	LPCI LOOP 2 NOT MAN ON	
232	T	LPCI LOOP 2 LOCKED OFF	
232	F	LPCI LOOP 2 NOT LOCKED OFF	
233	T	LPCI LOOP 1 TO DRYWELL SPRAYS-MAN ON	
233	F	LPCI LOOP 1 TO DRYWELL SPRAYS-NOT MAN ON	
234	T	LPCI LOOP 1 TO DRYWELL SPRAYS-LOCKED OFF	
234	F	LPCI LOOP 1 TO DRYWELL SPRAYS-NOT LOCKED OFF	
235	T	LPCI LOOP 2 TO DRYWELL SPRAYS-MAN ON	
235	F	LPCI LOOP 2 TO DRYWELL SPRAYS-NOT MAN ON	
236	T	LPCI LOOP 2 TO DRYWELL SPRAYS-LOCKED OFF	
236	F	LPCI LOOP 2 TO DRYWELL SPRAYS-NOT LOCKED OFF	
237	T	LPCI LOOP 1 TO WETWELL SPRAYS-MAN ON	
237	F	LPCI LOOP 1 TO WETWELL SPRAYS-NOT MAN ON	
238	T	LPCI LOOP 1 TO WETWELL SPRAYS-LOCKED OFF	
238	F	LPCI LOOP 1 TO WETWELL SPRAYS-NOT LOCKED OFF	
239	T	LPCI LOOP 2 TO WETWELL SPRAYS-MAN ON	
239	F	LPCI LOOP 2 TO WETWELL SPRAYS-NOT MAN ON	
240	T	LPCI LOOP 2 TO WETWELL SPRAYS-LOCKED OFF	
240	F	LPCI LOOP 2 TO WETWELL SPRAYS-NOT LOCKED OFF	
241	T	LPCI LOOP 1 ALIGNED TO VESSEL	
241	F	LPCI LOOP 1 NOT ALIGNED TO VESSEL	
242	T	LPCI LOOP 1 TO VESSEL-LOCKED OFF	
242	F	LPCI LOOP 1 TO VESSEL-NOT LOCKED OFF	
243	T	LPCI LOOP 2 ALIGNED TO VESSEL	
243	F	LPCI LOOP 2 NOT ALIGNED TO VESSEL	
244	T	LPCI LOOP 2 TO VESSEL-LOCKED OFF	
244	F	LPCI LOOP 2 TO VESSEL-NOT LOCKED OFF	
245	T	LPCI LOOP 1 TO SUPPRESSION POOL-MAN ON	

245	F	LPCI LOOP 1 TO SUPPRESSION POOL-NOT MAN ON
246	T	LPCI LOOP 1 TO SUPPRESSION POOL-LOCKED OFF
246	F	LPCI LOOP 1 TO SUPPRESSION POOL-NOT LOCKED OFF
247	T	LPCI LOOP 2 TO SUPPRESSION POOL-MAN ON
247	F	LPCI LOOP 2 TO SUPPRESSION POOL-NOT MAN ON
248	T	LPCI LOOP 2 TO SUPPRESSION POOL-LOCKED OFF
248	F	LPCI LOOP 2 TO SUPPRESSION POOL-NOT LOCKED OFF
249	T	SUCTION FOR HPCS MAN LINED UP TO SUPP POOL
249	F	SUCTION FOR HPCS NOT MAN LINED TO SUPP POOL
250	T	LOSS OF AC POWER
250	F	AC POWER RESTORED
251	T	LOSS OF DIESEL POWER
251	F	DIESEL POWER RESTORED
252	T	AUX BLDG DAMPERS SHUT
252	F	AUX BLDG DAMPERS NOT SHUT
253	T	NO H2 OR CO BURNING ALLOWED
253	F	H2 AND CO BURNING ALLOWED
254	T	SUCTION FOR RCIC MAN LINED UP TO SUPP POOL
254	F	SUCTION FOR RCIC NOT MAN LINED TO SUPP POOL
255	T	REACTOR MAN SCRAMMED
256	T	BREAK IN PRIMARY SYSTEM (LOCA)
256	F	NO BREAK IN PRIMARY SYSTEM
257	T	ATWS RUN
258	T	SLC INJECTION BEGUN
259	T	LPCI LOOP 3 MAN ON
259	F	LPCI LOOP 3 NOT MAN ON
260	T	LPCI LOOP 3 LOCKED OFF
260	F	LPCI LOOP 3 NOT LOCKED OFF
261	T	1 SRV (1103/1033 - 926) MAN OPEN (GRP 5)
261	F	1 SRV (1103/1033 - 926) NOT MAN OPEN (GRP 5)
262	T	1 SRV (1103/1033 - 926) LOCKED CLOSED (GRP 5)
262	F	1 SRV (1103/1033 - 926) NOT LOCKED OFF (GRP 5)
263	T	VACUUM BREAKERS-MAN OPEN
263	F	VACUUM BREAKERS-NOT MAN OPEN
264	T	VACUUM BREAKERS-LOCKED CLOSED
264	F	VACUUM BREAKERS-NOT LOCKED CLOSE
265	T	HYDROGEN MIXING SYSTEM MAN ON
265	F	HYDROGEN MIXING SYSTEM NOT MAN ON
266	T	HYDROGEN MIXING SYSTEM LOCKED OFF
266	F	HYDROGEN MIXING SYSTEM NOT LOCKED OFF
267	T	UPPER POOL DUMP MAN OPEN
267	F	UPPER POOL DUMP NOT MAN OPEN
268	T	UPPER POOL DUMP LOCKED CLOSED
268	F	UPPER POOL DUMP NOT LOCKED CLOSED
269	T	CONTAINMENT PURGE MAN OPEN
269	F	CONTAINMENT PURGE NOT MAN OPEN
270	T	CONTAINMENT PURGE LOCKED CLOSED
270	F	CONTAINMENT PURGE NOT LOCKED CLOSED
271	T	0 LPCS PUMPS ON
271	F	DEFAULT LPCS PUMPS ON
272	T	2 LPCS PUMPS ON
272	F	DEFAULT LPCS PUMPS ON
273	T	4 LPCS PUMPS ON
273	F	DEFAULT LPCS PUMPS ON

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274  T  HPSW INJECTION MAN ON
274  F  HPSW INJECTION NOT MAN ON
275  T  HPSW INJECTION LOCKED OFF
275  F  HPSW INJECTION NOT LOCKED OFF
276  T  AUX BLDG SPRAYS ON
276  F  AUX BLDG SPRAYS OFF
277  T  CRD PUMP MAN ON
277  F  CRD PUMP NOT MAN ON
278  T  CRD PUMP LOCKED OFF
278  F  CRD PUMP NOT LOCKED OFF
279  T  OPEN DRYWELL VENT
280  T  CLOSE DRYWELL VENT
281  T  LOCA OUTSIDE OF CONTAINMENT - OPEN
281  F  LOCA OUTSIDE OF CONTAINMENT - CLOSED
282  T  OPEN WETWELL VENT
283  T  CLOSE WETWELL VENT
284  T  DRYWELL COOLERS ON
284  F  DRYWELL COOLERS OFF
285  T  AUX CONDENSER MAN ON
285  F  AUX CONDENSER NOT MAN ON
286  T  AUX CONDENSER MAN OFF
286  F  AUX CONDENSER NOT MAN OFF
287  T  AUX BLDG CO2 ON
287  F  AUX BLDG CO2 OFF
288  T  SHUTDOWN COOLING MAN ON
288  F  SHUTDOWN COOLING NOT MAN ON
289  T  SHUTDOWN COOLING MAN OFF
289  F  SHUTDOWN COOLING NOT MAN OFF
290  T  IGNITORS FORCED ON
290  F  IGNITORS NOT FORCED ON
291  1  MECHANISTIC DRYWELL COOLER MAN ON
291  0  MECHANISTIC DRYWELL COOLER NOT MAN ON
292  1  MECHANISTIC DRYWELL COOLER LOCKED OFF
292  0  MECHANISTIC DRYWELL COOLER NOT LOCKED OFF
293  1  RWCU MAN ON
293  0  RWCU NOT MAN ON
294  1  FEEDWATER NOT TRIPPED BY MSIV CLOSURE
294  0  FEEDWATER TRIPPED BY MSIV CLOSURE
295  1  RCIC FLOW CONTROL ON
295  0  RCIC FLOW CONTROL OFF
296  T  BAR GRAPH DISPLAYS ON
296  F  BAR GRAPH DISPLAYS OFF
297  T  HEAT-UP DISPLAY STATUS ON
297  F  HEAT-UP DISPLAY STATUS OFF
298  T  VESSEL DISPLAY STATUS ON
298  F  VESSEL DISPLAY STATUS OFF
299  T  CONTAINMENT DISPLAY STATUS ON
299  F  CONTAINMENT DISPLAY STATUS OFF
** NOTE THIS FLAG IS SIMILIAR TO 197, IE., THAT IT IS AUTOMATICALLY SET
** WHEN VF OR CF IS TRUE ONLY FOR ONE TIMESTEP.  HOWEVER, THIS FLAG IS
** USED FOR RESETTING CUMULATIVE FIGURE OF MERITS MONITORED BY TOPSRT.
300  T  RESET CUMULATIVE FIGURE OF MERITS
301  1  LPCI LOOP 1 TO RADWASTE - MAN ON
301  0  LPCI LOOP 1 TO RADWASTE - NOT MAN ON
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302  1  LPCI LOOP 1 TO RADWASTE - LOCKED OFF
302  0  LPCI LOOP 1 TO RADWASTE - NOT LOCKED OFF
303  1  LPCI LOOP 2 TO RADWASTE - MAN ON
303  0  LPCI LOOP 2 TO RADWASTE - NOT MAN ON
304  1  LPCI LOOP 2 TO RADWASTE - LOCKED OFF
304  0  LPCI LOOP 2 TO RADWASTE - NOT LOCKED OFF
305  1  ISOLATION CONDENSER TUBE RUPTURED
305  0  ISOLATION CONDENSER TUBE NOT RUPTURED
309  1  HYDROGEN MIXING CLOSES UPON CONT. ISOLATION
309  0  HYDROGEN DOES NOT CLOSE UPON CONT. ISOLATION
310  1  SECOND DRYWELL PURGE LINE AVAILABLE
310  0  SECOND DRYWELL PURGE LINE NOT AVAILABLE
311  1  HEAT CAPACITY TEMPERATURE LIMIT (HCTL) - ON
311  0  HEAT CAPACITY TEMPERATURE LIMIT (HCTL) - OFF
312  1  ISO. COND. MAKEUP WATER - MAN ON
312  0  ISO. COND. MAKEUP WATER - MAN OFF
320  T  PRINT AND RESTART THIS TIMESTEP
400  1  TIME STEP BURNING IN CA - FLAG ON
400  0  TIME STEP BURNING IN CA - FLAG OFF
401  1  TIME STEP BURNING IN CB - FLAG ON
401  0  TIME STEP BURNING IN CB - FLAG OFF
402  1  TIME STEP BURNING IN DRYWELL - FLAG ON
402  0  TIME STEP BURNING IN DRYWELL - FLAG OFF
403  1  TIME STEP BURNING IN PEDESTAL - FLAG ON
403  0  TIME STEP BURNING IN PEDESTAL - FLAG OFF
404  1  TIME STEP BURNING IN WETWELL - FLAG ON
404  0  TIME STEP BURNING IN WETWELL - FLAG OFF
END
**
****************************************************************************
*FISSION PRODUCTS
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** INITIAL FISSION PRODUCT MASSES IN CORE REGION
01      935.82      Xe
02       62.15      Kr
03       40.12      I
04       56.44      Rb
05      500.44      Cs
06      151.68      Sr
07      253.97      Ba
08       87.52      Y
09      237.65      La
10      645.72      Zr
11         .24      Nb
12      573.19      Mo
13      142.20      Tc
14      416.00      Ru
15         .24      Sb
16       84.44      Te
17      503.09      Ce
18      194.44      Pr
19      655.20      Nd
20      130.07      Sm
21         .24      Np
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22     1796.74       Pu
26     2253.04       SN
27     1068.14       MN
28     2530.86       B4C
**   FOR DF VALUES ENTER A VALUE GE 1 OR SET EQUAL TO 0 TO HAVE CODE
**   CALCULATE THEM
** DF=1 ASSUMED FOR NOBLE GASES
** DF=1000 ASSUMED FOR ALL OTHER VAPORS
31          0.    FDFSP DRYWELL VENTS DECONTAMINATION FACTOR
32          0.    FDFRV SRV DECONTAMINTION FACTOR
33   0.028        FP GRP #1,  NOBLES
34   0.151        FP GRP #2,  CSI
35   0.0194       FP GRP #3,  TEO2
36   0.062        FP GRP #4,  SRO
37   0.05         FP GRP #5,  MOO2
38   0.1          FP GRP #6,  CSOH
39   0.           FP GRP #7,  BAO
40   0.           FP GRP #8,  LA2O3
41   0.           FP GRP #9,  CEO2
42   0.           FP GRP #10, SB
43   0.0194       FP GRP #11, TE2
44   0.           FP GRP #12, UO2
**
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*HEATUP
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01   12.57     XZFUEL     LENGTH OF ACTIVE FUEL
02     .0174   XRFUEL     RADIUS OF FUEL PELLET [TO THE CLAD WITH NO GAP]
03     .00267  XTCLAD     THICKNESS OF CLADDING
04   72987.2   MZRCAN     TOTAL MASS OF ZR IN ASSEMBLY CAN
05    0.0      MBCR       NOT USED
06     .0100   XZRCAN     CAN WALL THICKNESS
** NODE 1,1 IS BOTTOM-CENTER, 1,10 IS TOP-CENTER, 2,1 IS SECOND RADIAL
** RING OUT FROM CENTER AT THE BOTTOM OF THE CORE, ETC
07     .772    FPEAK(1, 1)   PEAKING FACTOR FOR NODE (1, 1)
08     .796    FPEAK(2, 1)   PEAKING FACTOR FOR NODE (2, 1)
09     .930    FPEAK(3, 1)   PEAKING FACTOR FOR NODE (3, 1)
10     .647    FPEAK(4, 1)   PEAKING FACTOR FOR NODE (4, 1)
11     .308    FPEAK(5, 1)   PEAKING FACTOR FOR NODE (5, 1)
15    1.384    FPEAK(1, 2)   PEAKING FACTOR FOR NODE (1, 2)
16    1.400    FPEAK(2, 2)   PEAKING FACTOR FOR NODE (2, 2)
17    1.649    FPEAK(3, 2)   PEAKING FACTOR FOR NODE (3, 2)
18    1.244    FPEAK(4, 2)   PEAKING FACTOR FOR NODE (4, 2)
19     .566    FPEAK(5, 2)   PEAKING FACTOR FOR NODE (5, 2)
23    1.459    FPEAK(1, 3)   PEAKING FACTOR FOR NODE (1, 3)
24    1.402    FPEAK(2, 3)   PEAKING FACTOR FOR NODE (2, 3)
25    1.573    FPEAK(3, 3)   PEAKING FACTOR FOR NODE (3, 3)
26    1.364    FPEAK(4, 3)   PEAKING FACTOR FOR NODE (4, 3)
27     .669    FPEAK(5, 3)   PEAKING FACTOR FOR NODE (5, 3)
31    1.378    FPEAK(1, 4)   PEAKING FACTOR FOR NODE (1, 4)
32    1.369    FPEAK(2, 4)   PEAKING FACTOR FOR NODE (2, 4)
33    1.442    FPEAK(3, 4)   PEAKING FACTOR FOR NODE (3, 4)
34    1.260    FPEAK(4, 4)   PEAKING FACTOR FOR NODE (4, 4)
35     .689    FPEAK(5, 4)   PEAKING FACTOR FOR NODE (5, 4)
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39   1.290      FPEAK(1, 5)   PEAKING FACTOR FOR NODE (1, 5)
40   1.270      FPEAK(2, 5)   PEAKING FACTOR FOR NODE (2, 5)
41   1.332      FPEAK(3, 5)   PEAKING FACTOR FOR NODE (3, 5)
42   1.156      FPEAK(4, 5)   PEAKING FACTOR FOR NODE (4, 5)
43    .668      FPEAK(5, 5)   PEAKING FACTOR FOR NODE (5, 5)
47   1.238      FPEAK(1, 6)   PEAKING FACTOR FOR NODE (1, 6)
48   1.207      FPEAK(2, 6)   PEAKING FACTOR FOR NODE (2, 6)
49   1.259      FPEAK(3, 6)   PEAKING FACTOR FOR NODE (3, 6)
50   1.082      FPEAK(4, 6)   PEAKING FACTOR FOR NODE (4, 6)
51    .645      FPEAK(5, 6)   PEAKING FACTOR FOR NODE (5, 6)
55   1.210      FPEAK(1, 7)   PEAKING FACTOR FOR NODE (1, 7)
56   1.178      FPEAK(2, 7)   PEAKING FACTOR FOR NODE (2, 7)
57   1.207      FPEAK(3, 7)   PEAKING FACTOR FOR NODE (3, 7)
58   1.020      FPEAK(4, 7)   PEAKING FACTOR FOR NODE (4, 7)
59    .621      FPEAK(5, 7)   PEAKING FACTOR FOR NODE (5, 7)
63   1.168      FPEAK(1, 8)   PEAKING FACTOR FOR NODE (1, 8)
64   1.220      FPEAK(2, 8)   PEAKING FACTOR FOR NODE (2, 8)
65   1.136      FPEAK(3, 8)   PEAKING FACTOR FOR NODE (3, 8)
66    .926      FPEAK(4, 8)   PEAKING FACTOR FOR NODE (4, 8)
67    .573      FPEAK(5, 8)   PEAKING FACTOR FOR NODE (5, 8)
71   1.049      FPEAK(1, 9)   PEAKING FACTOR FOR NODE (1, 9)
72   1.059      FPEAK(2, 9)   PEAKING FACTOR FOR NODE (2, 9)
73    .919      FPEAK(3, 9)   PEAKING FACTOR FOR NODE (3, 9)
74    .729      FPEAK(4, 9)   PEAKING FACTOR FOR NODE (4, 9)
75    .465      FPEAK(5, 9)   PEAKING FACTOR FOR NODE (5, 9)
79    .544      FPEAK(1,10)   PEAKING FACTOR FOR NODE (1,10)
80    .519      FPEAK(2,10)   PEAKING FACTOR FOR NODE (2,10)
81    .438      FPEAK(3,10)   PEAKING FACTOR FOR NODE (3,10)
82    .344      FPEAK(4,10)   PEAKING FACTOR FOR NODE (4,10)
83    .228      FPEAK(5,10)   PEAKING FACTOR FOR NODE (5,10)
87    [illegible]  XCHIM      UNHEATED FUEL LENGTH ABOVE TAF
88    .000125   XIZROX        INITIAL CLADDING OXIDE THICKNESS
89   2531.      MBCBLA        MASS OF B4C IN ALL CONTROL BLADES
90  36370.      MSSBLA        MASS OF STAINLESS STEEL IN ALL CONTROL BLADES
91    .74       MFFESS        FRACTION OF FE IN STAINLESS STEEL
92    .18       MFCRSS        FRACTION OF CR IN STAINLESS STEEL
93    .08       MFNISS        FRACTION OF NI IN STAINLESS STEEL
94   2600.      TCBMP         MELTING POINT OF CONTROL BLADE
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*HTSINKS
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01       1600.  AHS1     AREA OF WALL #1
02       9313.  AHS2     AREA OF WALL #2
03       6086.  AHS3     AREA OF WALL #3
04      10372.  AHS4     AREA OF WALL #4
05      15637.  AHS5     AREA OF WALL #5
06      40100.  AHS6     AREA OF WALL #6
07          3.8 AHS7     AREA OF WALL #7
08       9740.  AHS8     AREA OF WALL #8
09      1.427   KHS1     THERMAL CONDUCTIVITY OF WALL #1
10       .80    KHS2     THERMAL CONDUCTIVITY OF WALL #2
11      1.49    KHS3     THERMAL CONDUCTIVITY OF WALL #3
12     26.0     KHS4     THERMAL CONDUCTIVITY OF WALL #4
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13    26.0      KHS5     THERMAL CONDUCTIVITY OF WALL #5
14    26.0      KHS6     THERMAL CONDUCTIVITY OF WALL #6
15    26.0      KHS7     THERMAL CONDUCTIVITY OF WALL #7
16      .80     KHS8     THERMAL CONDUCTIVITY OF WALL #8
17     6.0      XHS1     THICKNESS OF WALL #1
18     5.0      XHS2     THICKNESS OF WALL #2
19     5.0      XHS3     THICKNESS OF WALL #3
20      .125    XHS4     THICKNESS OF WALL #4
21      .125    XHS5     THICKNESS OF WALL #5
22      .125    XHS6     THICKNESS OF WALL #6
23      .125    XHS7     THICKNESS OF WALL #7
24     3.6      XHS8     THICKNESS OF WALL #8
25      .0833   XLHSI1    INNER LINER THICKNESS FOR WALL #1
26      .0208   XLHSI2    INNER LINER THICKNESS FOR WALL #2
27      .041    XLHSI3    INNER LINER THICKNESS FOR WALL #3
28   .00067     XLHSI4    INNER LINER THICKNESS FOR WALL #4
29   .00067     XLHSI5    INNER LINER THICKNESS FOR WALL #5
30   .00067     XLHSI6    INNER LINER THICKNESS FOR WALL #6
31   .00067     XLHSI7    INNER LINER THICKNESS FOR WALL #7
32      .0      XLHSI8    INNER LINER THICKNESS FOR WALL #8
33      .0833   XLHSO1    OUTER LINER THICKNESS FOR WALL #1
34      .0      XLHSO2    OUTER LINER THICKNESS FOR WALL #2
35      .0237   XLHSO3    OUTER LINER THICKNESS FOR WALL #3
36      .0      XLHSO4    OUTER LINER THICKNESS FOR WALL #4
37      .0      XLHSO5    OUTER LINER THICKNESS FOR WALL #5
38      .0      XLHSO6    OUTER LINER THICKNESS FOR WALL #6
39      .0      XLHSO7    OUTER LINER THICKNESS FOR WALL #7
40      .0      XLHSO8    OUTER LINER THICKNESS FOR WALL #8
41   162.       DHS1      DENSITY OF WALL #1
42   150.       DHS2      DENSITY OF WALL #2
43   154.       DHS3      DENSITY OF WALL #3
44   490.       DHS4      DENSITY OF WALL #4
45   490.       DHS5      DENSITY OF WALL #5
46   490.       DHS6      DENSITY OF WALL #6
47   490.       DHS7      DENSITY OF WALL #7
48   150.       DHS8      DENSITY OF WALL #8
49      .190    CPHS1     SPECIFIC HEAT FOR WALL #1
50      .193    CPHS2     SPECIFIC HEAT FOR WALL #2
51      .191    CPHS3     SPECIFIC HEAT FOR WALL #3
52      .110    CPHS4     SPECIFIC HEAT FOR WALL #4
53      .110    CPHS5     SPECIFIC HEAT FOR WALL #5
54      .110    CPHS6     SPECIFIC HEAT FOR WALL #6
55      .110    CPHS7     SPECIFIC HEAT FOR WALL #7
56      .193    CPHS8     SPECIFIC HEAT FOR WALL #8
**ALL OF THESE EQUIPMENT HEAT SINKS ARE LOCATED IN GAS VOL. OF COMPARTMENT
57      .0      MEQPD     MASS OF EQUIPMENT IN PEDESTAL
58   3420000.   MEQDW     MASS OF EQUIPMENT IN DRYWELL
59    100000.   MEQWW     MASS OF EQUIPMENT IN WETWELL
60    439000.   MEQCA     MASS OF EQUIPMENT IN COMPT A
61   3900000.   MEQCB     MASS OF EQUIPMENT IN COMPT B
62      .0      AEQPD     AREA OF EQUIPMENT IN PEDESTAL
63     46000.   AEQDW     AREA OF EQUIPMENT IN DRYWELL
64     85782.   AEQWW     AREA OF EQUIPMENT IN WETWELL
65     60600.   AEQCA     AREA OF EQUIPMENT IN COMPT A
```

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66     11600.   AEQCB    AREA OF EQUIPMENT IN COMPT B
67      .0      HTOUTW   HEAT TRANSFER COEFF. AT OUTER WALL
68      .0      RGAPI1   INNER LINER TO WALL GAP RESISTANCE #1
69      .0      RGAPI2   INNER LINER TO WALL GAP RESISTANCE #2
70      .0      RGAPI3   INNER LINER TO WALL GAP RESISTANCE #3
71      .0      RGAPI4   INNER LINER TO WALL GAP RESISTANCE #4
72      .0      RGAPI5   INNER LINER TO WALL GAP RESISTANCE #5
73      .0      RGAPI6   INNER LINER TO WALL GAP RESISTANCE #6
74      .0      RGAPI7   INNER LINER TO WALL GAP RESISTANCE #7
75      .0      RGAPI8   INNER LINER TO WALL GAP RESISTANCE #8
76      .0      RGAPO1   OUTER LINER TO WALL GAP RESISTANCE #1
77      .0      RGAPO2   OUTER LINER TO WALL GAP RESISTANCE #2
78      .0      RGAPO3   OUTER LINER TO WALL GAP RESISTANCE #3
79      .0      RGAPO4   OUTER LINER TO WALL GAP RESISTANCE #4
80      .0      RGAPO5   OUTER LINER TO WALL GAP RESISTANCE #5
81      .0      RGAPO6   OUTER LINER TO WALL GAP RESISTANCE #6
82      .0      RGAPO7   OUTER LINER TO WALL GAP RESISTANCE #7
83      .0      RGAPO8   OUTER LINER TO WALL GAP RESISTANCE #8
84      .0      MEQWWS   MASS OF EQUIP. HEAT SINK WETWELL (SUBMERGED)
85      .0      AEQWWS   AREA OF EQUIP. HEAT SINK WETWELL (SUBMERGED)
86      .0      XTGAP1   GAP THICKNESS FROM LINER TO WALL FOR  #1
87      .0      XTGAP2   GAP THICKNESS FROM LINER TO WALL FOR  #2
88      .0      XTGAP3   GAP THICKNESS FROM LINER TO WALL FOR  #3
89      .0      XTGAP4   GAP THICKNESS FROM LINER TO WALL FOR  #4
90      .0      XTGAP5   GAP THICKNESS FROM LINER TO WALL FOR  #5
91      .0      XTGAP6   GAP THICKNESS FROM LINER TO WALL FOR  #6
92      .0      XTGAP7   GAP THICKNESS FROM LINER TO WALL FOR  #7
93      .0      XTGAP8   GAP THICKNESS FROM LINER TO WALL FOR  #8
94      .0      ZEQPD    AVERAGE HEIGHT OF PEDESTAL EQUIPMENT
95    49.       ZEQDW    AVERAGE HEIGHT OF DRYWELL EQUIPMENT
96    14.       ZEQWW    AVERAGE HEIGHT OF WETWELL EQUIPMENT - GAS SPACE
97    13.7      ZEQWWS   AVERAGE HEIGHT OF WETWELL EQUIPMENT - SUBMERGED
98    21.       ZEQCA    AVERAGE HEIGHT OF COMPT A EQUIPMENT
99    48.       ZEQCB    AVERAGE HEIGHT OF COMPT B EQUIPMENT
**
************************************************************************
*INITIAL CONDITIONS
************************************************************************
01   12.212E9   QPOWER     CORE POWER
02   1039.6     PPSO       INITIAL PRESSURE IN PRIMARY SYSTEM
03     14.7     PPDO       INITIAL PRESSURE IN PEDESTAL
04     14.7     PDWO       INITIAL PRESSURE IN DRYWELL
05     14.7     PWWO       INITIAL PRESSURE IN WETWELL
06   593.083    ZSPDWO     INIT.ELEV. OF WATER LEVEL IN DW SIDE OF SUPP.POOL
07   593.083    ZSPWWO     INIT.ELEV. OF WATER LEVEL IN PC SIDE OF SUPP.POOL
08   145.       TPDO       INITIAL TEMPERATURE IN PEDESTAL
09   144.       TDWO       INITIAL TEMPERATURE IN DRYWELL
10    89.       TWWO       INITIAL TEMPERATURE IN WETWELL
11    89.       TWSPO      INITIAL TEMPERATURE OF SUPPRESSION POOL WATER
12   651.615    ZWSHO      INITIAL ELEVATION OF WATER IN THE SHROUD
13   4524727.   MWCBO      MASS OF WATER IN UPPER POOL (MARKIII ONLY)
14     33343.   VCSTO      VOLUME OF WATER IN CONDENSATE STORAGE TANK
**
************************************************************************
```

```
**
**@@@ REV 5/6 ADDITION, NEW SECTION FOR USER-INPUT TIMESTEP CONTROL
**
** INTEGRATION CONTROL
**
** SI units only allowed
**
** ALLOWED SYNTAXES:
**
** 1. Fractional change limitation:
** INDEX  R  X-NAME F-NAME F-CHANGE  X-MIN  X-MAX  TRUE  #1  FALSE  #2
** where:
** INDEX = index of limiting variable
** R = a fractional change (ie, a rate) limitation
** X-NAME = state or aux variable name
** F-NAME = rate of change variable name
** F-CHANGE = fractional change
** X-MIN = minimum x value for limitation
** X-MAX = maximum x value for limitation
** The "TRUE #1" & "FALSE #2" are optional:
** TRUE #1 = used when event #1 true
** FALSE #2 = used when event #2 false
** when code #1 is true the control is on
** when code #2 is false the control is on
** either "TRUE" or "FALSE", or both "TRUE" & "FALSE" conditions
** can be used
** Example:
** 1   R  MWCOR    FMWCOR    0.04   1.E3   1.E10    FALSE  8
** timestep limiting variable 1 is MWCOR, rate of change FMWCOR
** its fractional change is 4% maximum during a timestep
** if MWCOR < 1.e3 kg it is not used to limit the timestep
** if MWCOR > 1.e10 kg it is not used to limit the timestep
** it is used when event code 8 is false, ie reactor vessel intact
**
** 2. Threshold specified explicitly:
** INDEX  T   X-NAME  F-NAME  THRESH
** INDEX  T+  X-NAME  F-NAME  THRESH
** INDEX  T-  X-NAME  F-NAME  THRESH
** where:
** T = a threshold limitation both ascending and descending
** T+ = an ascending threshold limitation
** T- = a descending threshold limitation
** THRESH = the threshold value
** Example:
** 2   T+  PPS  FPPS  7.75E6
** timestep limiting variable 2 is PPS, rate of change is FPPS
** the timestep will be limited if PPS attempts to cross 7.75 MPa
**  in an ascending manner (ie, as if a relief valve were to open)
**
** 3. Threshold specified by reference to parameter input:
** INDEX  T  X-NAME  F-NAME  T-NAME
** where:
** T-NAME = the variable name for the threshold
** Example:
```

```
** 3   T+  PPS  FPPS  PSRV1
** timestep limiting variable 3 is PPS, rate of change is FPPS
** the timestep will be limited if PPS attempts to cross PSRV1
**  in an ascending manner (ie, as if a safety valve were to open)
**  and PSRV1 is input in the parameter file already as the
**  safety valve setpoint
**
*INTEGRATION
**
** CRITICAL QUANTITIES FOR TIME LOST INFORMATION
**
** CATEGORY 1 -- GAS MASSES & TEMPERATURES
1   R  MGPS     FMGPS     0.04    1.E1    1.E10
2   R  TGPS     FTGPS     0.05    1.E2    1.E4
3   R  MGPD     FMGPD     0.04    1.E1    1.E10
4   R  TGPD     FTGPD     0.05    1.E2    1.E4
5   R  MGDW     FMGDW     0.04    1.E1    1.E10
6   R  TGDW     FTGDW     0.05    1.E2    1.E4
7   R  MGWW     FMGWW     0.04    1.E1    1.E10
8   R  TGWW     FTGWW     0.05    1.E2    1.E4
9   R  MGCA     FMGCA     0.04    1.E1    1.E10
10  R  TGCA     FTGCA     0.05    1.E2    1.E4
11  R  MGCB     FMGCB     0.04    1.E1    1.E10
12  R  TGCB     FTGCB     0.05    1.E2    1.E4
** CATEGORY 2 -- WATER MASSES, ETC.
13  R  MWPD     FMWPD     0.04    1.E4    1.E10
14  R  MWDW     FMWDW     0.04    1.E4    1.E10
**15  MSPDW = mass of water in wetwell downcomers (not used)
16  R  MSPWW    FMSPWW    0.04    1.E2    1.E10
17  R  MWCOR    FMWCOR    0.04    1.E3    1.E10    FALSE  8
18  R  MWOSH    FMWOSH    0.04    1.E3    1.E10
19  R  MWJET    FMWJET    0.04    1.E2    1.E10
20  R  XROF     FXROF     0.04    1.E-3   1.E3
21  R  MWAC     FMWAC     0.04    1.E10   1.E15
** CATEGORY 3 -- CRUST THICKNESSES
**22  XUCPD = upper debris crust thickness in pedestal (not used)
**23  XLCPD = lower debris crust thickness in pedestal (not used)
**24  XUCDW = upper debris crust thickness in drywell (not used)
**25  XLCDW = lower debris crust thickness in drywell (not used)
**26  XUCWW = upper debris crust thickness in wetwell (not used)
**27  XLCWW = lower debris crust thickness in wetwell (not used)
** CATEGORY 4 -- HARDWIRED AS IN ORIGINAL INTGRT
**28  DEBRIS INTERNAL ENERGY IN PEDESTAL
**29  DEBRIS INTERNAL ENERGY IN DRYWELL
**30  GAS MASS IN AUX BLDG
**31  GAS TEMP IN AUX BLDG
**32  SUBROUTINE HEATUP REQUIRED TIME STEP
**33  RPV CIRCULATION REQUIRED TIME STEP
**
END
**
******************************************************************************
*ISOLATION CONDENSER
******************************************************************************
```

```
01   -1.    VOLIC         VOLUME OF ISOLATION CONDENSER
**                        SPECIFY A NEGATIVE VOLUME TO INDICATE ABSENCE OF      88
**                        ISOLATION CONDENSER IN THE PLANT                      88
02    0.    MWICI         INITIAL MASS OF WATER IN ISOLATION CONDENSER
03    0.    TWICI         INITIAL WATER TEMPERATURE IN ISOLATION CONDENSER
04    0.    PPSIC(1)
05    0.    PPSIC(2)                      TABLE OF
06    0.    PPSIC(3)
07    0.    PPSIC(4)        ISOLATION CONDENSER HEAT REMOVAL RATE
08    0.    PPSIC(5)
09    0.    PPSIC(6)                         VS.
10    0.    PPSIC(7)
11    0.    PPSIC(8)      PARTIAL PRESSURE OF STEAM IN PRIMARY SYSTEM
12    0.    QIC1(1)
13    0.    QIC1(2)
14    0.    QIC1(3)       THE HEAT REMOVAL RATE BECOMES ZERO WHEN THE TUBES
15    0.    QIC1(4)       ARE UNCOVERED OR WHEN TUBES RUPTURE.
16    0.    QIC1(5)
17    0.    QIC1(6)
18    0.    QIC1(7)
19    0.    QIC1(8)
20    0.    ZWMAKE        WATER LEVEL TO WHICH THE MAKE-UP WATER FILLS THE IC
21    0.    WWMAKE        FLOW RATE OF MAKE-UP WATER
22    0.    HWMAKE        ENTHALPY OF MAKE-UP WATER
24    0.    PICI          INITIAL PRESSURE INSIDE THE ISOLATION CONDENSER
25    0.    ARUPIC        TUBE RUPTURE AREA
26    0.    XZRUP         HEIGHT OF THE TUBE RUPTURE ABOVE THE FLOOR OF IC
27    0.    AVEN          AREA OF THE ISOLATION CONDENSER VENT
**                        THIS VENT IS OPEN TO AMBIENT
28    0.    XWIC1(1)
29    0.    XWIC1(2)                   TABLE OF
30    0.    XWIC1(3)
31    0.    XWIC1(4)            WATER HEIGHT IN ISO. COND.
32    0.    XWIC1(5)
**                                        VS.
33    0.    FWIC1(1)
34    0.    FWIC1(2)        FRACTION OF TUBE H.T. AREA COVERED
35    0.    FWIC1(3)
36    0.    FWIC1(4)
37    0.    FWIC1(5)
**
38    0.      XWIC2(1)
39    0.      XWIC2(2)                  TABLE OF
40    0.      XWIC2(3)
41    0.      XWIC2(4)         WATER HEIGHT IN ISO. COND.
42    0.      XWIC2(5)
**                                         VS.
43    0.       VOLW2(1)
44    0.       VOLW2(2)        VOLUME OF WATER IN ISO CON.
45    0.       VOLW2(3)
46    0.       VOLW2(4)
47    0.       VOLW2(5) <------THIS MUST BE SAME AS 'VOLIC'
**
*************************************************************************
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```
*MODEL PARAMETERS FOR BWR
*************************************************************************
01    .005D0    FRCOEF   FRICTION COEFFICIENT FOR CORIUM DISCHARGED
**                       FROM VESSEL AS CALCULATED IN SUBROUTINE VFAIL
02    .10D0     FMAXCP   ONCE THE CORE FRACTION MELTS TO A VALUE BELOW
**                       FMAXCP,THE REMAINDER OF THE CORE SLUMPS TO THE
**                       LOWER PLENUM TO FAIL THE CORE PLATE
03    8.8D0     HTBLAD   FUEL CHANNEL TO CONTROL BLADE HEAT TRANS. COEFF
04    52.8D0    HTFB     NON-RADIATIVE FILM BOILING HEAT TRANS. COEFF.
05    10.D0     FDF1     DF FOR WATER POOLS OVER CORE DEBRIS (EXCLUDING
**                       SUPPRESSION POOL) OF 1 METER DEPTH
06    0.02D0    FEFFDR   DROP COLLECTION EFFICIENCY FOR SPRAY SWEEP-OUT
07    2240.D0   TCLMAX   CLAD FAILURE TEMP TO BEGIN FISSION PRODUCT REL
12       .1D0   FTENUR   UNOXIDIZED ZR MASS FRACTION LIMIT.
**                       THIS APPLIES TO NUREG-0772/KELLY FISSION PRODUCT
**                       RELEASES CALC'S ONLY.  (SEE MODEL PARAMETER
**                       FPRAT #41 BELOW)  IF CALC'D UNOXIDIZED ZR NODAL
**                       MASS IS GREATER THAN THIS LIMIT, THEN THE TE
**                       RELEASE RATE IS LIMITED, OTHERWISE THE TE RELEASE
**                       RATE IS NOT AFFECTED AS RECOMMENDED IN NUREG-0956.
**                       EXPERIMENTAL EVIDENCE HAS SHOWN THAT SIGNIFICANT
**                       AMOUNT RELEASED TE TENDS TO BIND WITH UNOXIDIZED ZR.
**                       THIS PARAMETER IS NOT TO BE CONFUSED WITH MODEL
**                       PARAMETER FTFREL (#43) BELOW.
13    1.0D      SCALFP   FISSION PRODUCT RELEASE RATES DIVIDED BY THIS VALUE
**                       IN FPRATB
14    176.D0    HTCMCR   CORIUM-CRUST HEAT TRANSF. COEFF. USED IN DECOMP
15    0.16D0    XCMX     MINIMUM CORIUM THICKNESS ON DRYWELL FLOOR AND PED
**                       FLOOR (MARK II ONLY) [DUMMY]
16    0.03D0    XDCMSP   PARTICLE SIZE (DIAMETER) FOR CORIUM AS IT FALLS
**                       INTO SUPPRESSION POOL (MARK II ONLY) [DUMMY]
17    2.78D-5   TDSTX    TIME DELAY AFTER CORIUM CONTACTS FLOOR TO TRIGGER
**                       STEAM EXPLOSION
18    1.53D0    FCHTUR   CHURN-TURBULENT CRITICAL FLOW PARAMETER
19    3.7D0     FDROP    DROPLET CRITICAL FLOW PARAMETER
20    3.D0      FFLOOD   FLOODING FLOW PARAMETER
21    1.35      FSPAR    PARAMETER FOR BOTTOM-SPARGED STEAM VOID FRACTION
22    3.D0      FVOL     PARAMETER FOR VOLUME SOURCE VOID FRACTION MODEL
23    1.4D-4    TIENTR   ENTRAINMENT EFFECTIVE EMPTYING TIME
**                       GIVEN THAT CORIUM AND WATER ARE ENTRAINED FROM PEDESTAL
**                       HOW LONG WILL IT TAKE TO EXPELL ENTIRE MASS
24    .90D0     EW       EMISSIVITY OF WATER
25    .80D0     EWL      EMISSIVITY OF WALL
26    .85D0     ECM      EMISSIVITY OF CORIUM
27    .6D0      EG       EMISSIVITY OF GAS
28    .85D0     EEQ      EMISSIVITY OF EQUIPMENT
29    0.5D0     FOVER    FRACTION OF CORE SPRAY FLOW ALLOWED TO BYPASS CORE
** THIS PARAMETER IS USED TO COMPUTE THE FRACTION OF CORE SPRAY THAT GOES
** INTO THE BYPASS REGION AND REFLOODS THE CORE FROM THE BOTTOM
30    1.D0      NPF      NUMBER OF PENETRATIONS FAILED IN LOWER HEAD AT TIME
**                       OF VESSEL FAILURE
31    2.D0      FCDCDW   DOWNCOMER PERIMETER PER METER FROM PEDESTAL DOOR
**                       (MARK II ONLY) [DUMMY]
32    0.10D0    FCHF     COEFFICIENT FOR CHF CORRELATION IN PLSTM
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33   .70DO      FCDBRK   DISCHARGE COEFFICIENT FOR PIPE BREAK
34   .13DO      FENTR    NUMBER TO MULTIPLY KUTATELADZE CRITERION BY TO
**                       REPRESENT DIFFICULTY (GT 1.DO) OR EASE (LT 1.DO)
**                       FOR MATERIAL TO BE BLOWN OUT OF CAVITY
35   1.00       SCALU    SCALING FACTOR FOR ALL BURNING VELOCITIES
36   1.00       SCALH    SCALING FACTOR FOR HT COEFFICIENTS TO PASSIVE
**                       HEAT SINKS
37   1.5DO      FUMIN    CLADDING SURFACE MULTIPLIER TO ACCOUNT FOR POTENTIAL
**                       CLAD RUPTURE
38   2.5        GSHAPE   PARTICLE COLLISION GAMMA SHAPE FACTOR
39   1.0        FSHAPE   CHI SETTLING SHAPE FACTOR
40   8.0        FAERDC   RATIO OF AIRBORNE AEROSOL MASS TO THE MASS WHICH
**                       WOULD LEAVE YOU IN STEADY-STATE WITH THE CURRENT
**                       SOURCE STRENGTH.THIS IS USED TO CONTROL THE SELECTION
**                       OF DECAY VS STEADY-STATE AEROSOL SETTLING CORRELATIONS
41  -1          FPRAT    FISSION PRODUCT RELEASE CORRELATION & CONTROL
**                       ENTER A VALUE TO SELECT CORRELATIONS:
**                         +1 OR -1  NUREG-0772 MODEL
**                         +2 OR -2  IDCOR/EPRI STEAM OXIDATION MODEL
**                       ENTER A SIGN TO SELECT RELEASE LIMITATIONS:
**                         + SIGN: RELEASE RATES DEFINED BY CORRELATIONS
**                         - SIGN: RELEASES FURTHER LIMITED BY SATURATION
**                            VAPOR PRESSURE FOR NONVOLATILES AND STRUCTURE
**                       THE + SIGN IS USEFUL WHEN THE IDCOR BLOCKAGE MODEL
**                       IS SELECTED, SINCE FLOW IN CORE NODES CAN GO TO
**                       ZERO DURING BLOCKAGE, THUS STOPPING RELEASES.
**                       THE - SIGN IS USEFUL WHEN NO BLOCKAGE IS
**                         ALLOWED AND THERE IS ALWAYS BULK FLOW.
**                       THE + SIGN ALSO IS USEFUL FOR SENSITIVITY.
**                       THE - SIGN ALLOWS THE PHYSICAL MECHANISM OF
**                         SATURATION TO BE CONSIDERED FOR RELEASE.
**                       HOWEVER, DIFFUSION COEFFICIENTS, VAPOR
**                         PRESSURES AND GEOMETRY ARE QUITE UNCERTAIN .
42   1.0        FCSIVP   GROUP 2 (CSI) & GROUP 6 (CSOH) VAPOR PRESSURE
**                       MULTIPLIER
**                       NEG NUMBER USES ANL CSOH VAP PRESS,
**                       POS NUMBER USES SANDIA CSOH VAP PRESSURE
43   0          FTEREL   0=TE BOUND UP IN ZIRCALOY, 1=NOT BOUND UP
**                       THIS APPLIES TO CUBICOITTI FISSION PRODUCT
**                       RELEASES CALC'S ONLY.  (SEE MODEL PARAMETER
**                       FPRAT #41 ABOVE)
44   7.25       PPLUG    PRESSURE DIFFERENCE TO BLOW OPEN PLUG IF LEAK PATH HAS
**                       PLUGGED AS A RESULT OF AEROSOLS
45   0.066      XHLEAK   WIDTH OF LEAK PATH
** MAAP ASSUMES THAT THE DRYWELL FAILURE OR LEAKAGE OPENING IS A LONG SLIT
** WITH AREA EQUAL TO ADWLEK(SEE DRYWELL INPUT) AND AN EFFECTIVE WIDTH EQUAL
** TO XHLEAK.BASED UPON THESE TWO VALUES THE LENGTH OF THE SLIT CAN BE
** CALCULATED
46   3121.      DKPLUG   MOREWITZ COEFF FOR PLUGGING
47   4040.      TEUTEC   CORE NODE EUTECTIC TEMPERATURE FOR MELTING NODE
48   107.       LHEU     LATENT HEAT OF FUSION OF EUTECTIC
49   1.D-6      XRSEED   SEED RADIUS FOR HYGROSCOPIC FORMATION
50   0.DO       TIDCF    IF EVENT CODE 216 IS SET TO 1 TO FAIL PEDESTAL
**                       DOWNCOMERSTHEN THE DOWNCOMERS WILL FAIL TIDCF
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**                              SECONDS AFTER VESSEL FAILURE
51      .0        ASTRN         STRAIN INDUCED CONTAINMENT FAILURE AREA
**                              FOR MARK III ONLY (WETWELL OR COMPT B) [DUMMY]
52     1.         AOVPR         GROSS OVER-PRESSURE CONTAINMENT FAILURE AREA
**                              FOR MARK III ONLY (WETWELL OR COMPT B)
53   1448.        TJBRN         JET BURN TEMP: IF GAS JET OUT OF PEDESTAL EXCEEDS
**                              THIS TEMP THEN H2 AND CO BURN AS THEY ENTER DRYWELL
54    .01D0       FASI          ACTIVITY COEFFICIENT FOR SIO2
55    .05D0       FASR          ACTIVITY COEFFICIENT FOR SRO
56    .05D0       FABA          ACTIVITY COEFFICIENT FOR BAO
57   1.D-8        FAKO          ACTIVITY COEFFICIENT FOR K2O
58   0.0          FCRBLK        =1  CORE BLOCKAGE/LOCAL NODE CUT-OFF
**                              =0  NO CORE BLOCKAGE/LOCAL NODE CUT-OFF
**                              -1  NO CORE BLOCKAGE/NO LOCAL CUT-OFF
59    .33D0       FEO           PRUPACHER-KLETT COLLISION EFFICIENCY
**                              (USE .33 FOR P-K MODEL;1 FOR FUCHS)
60   18.D0        FNUDRP        NUSSELT NO. WHICH GOVERNS HEAT CONDUCTION INTO
**                              A SMALL DROPLET (EG FROM CONTAINMENT SPRAYS);
**                              AVAILABLE DATA SUGGESTS THAT A CONSTANT VALUE IS
**                              FAIRLY ACCURATE
61   1310.        TAUTO         AUTOIGNITION TEMPERATURE FOR H2 BURNS
**                              A BURN WILL OCCUR IF THE GAS T EXCEEDS THIS VALUE
**                              NO MATTER WHAT THE H2 CONCENTRATION IS
62   0.75         XSTIA         STEAM MOLE FRACTION TO INERT A H2-AIR-H2O MIXTURE
**                              AT INCIPIENT AUTOIGNITION -- AT TEMPERATURES JUST
**                              BELOW AUTOIGNITION, THIS STEAM MOLE FRACTION WILL
**                              PREVENT A BURN.
63   1.00         DXHIG         OFFSET H2 MOLE FRACTION FOR DEFINITION OF IGNITION
**                              DURING A BLACKOUT SEQUENCE: THIS IS ADDED TO (OR TAKEN
**                              FROM, IF NEGATIVE) THE DOWNWARD FLAMMABILITY LIMIT
**                              IGNITION (GLOBAL BURNS) WILL OCCUR IF THE H2
**                              MOLE FRACTION EXCEEDS THE LIMIT PLUS THE OFFSET
64   2.0          FLPHI         FLAME FLUX MULTIPLIER (BETWEEN 1.0 AND 10.0)
**                              BEST-ESTIMATE USED FOR CONTAINMENTS WITH SPRAYS AND
**                              FANS OFF IS 2, WHEN FANS OR SPRAYS ARE ON, USE 1[illegible].
**                              IF DIFFUSION BURN ARE EXPECTED, WHEN H2 RATE > .15
**                              LB/SEC USE 0.25. [REF: EPRI RESEARCH PROJECT, Y101-1,
**                              THERMAL ANALYSIS OF HYDROGEN DIFFUSION FLAMES IN A 1/4
**                              SCALE BWR MARK III CONTAINMENT BUILDING, PG 13-3]
**
**
*******************************************************************************
*PEDESTAL
*******************************************************************************
01        293.4 APDF          AREA OF PEDESTAL FLOOR
02         28.  APDVT         AREA OF PEDESTAL-DRYWELL OPENING
03       8423.  VOLPD         FREE VOLUME OF PEDESTAL
04    585.542   ZWPDDW        ELEVATION OF WIER WALL
05    5[illegible].750   ZPDF          ELEVATION AT PEDESTAL FLOOR
06        .40   RELHPD        RELATIVE HUMIDITY IN PEDESTAL
07        .0    NIGPD         NUMBER OF IGNITERS IN THE PEDESTAL
08        .0    XIGPD         AVERAGE DISTANCE FROM FLOOR TO IGNITERS
09        .0                  NOT USED
10        .0    XWPDVT        WIDTH OF PEDESTAL DOOR (MARK II ONLY)
11        .0    ADCPD         AREA OF DRYWELL TO COMPT A LEAKAGE
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12      .0      NDCPD        NUMBER OF PEDESTAL DOWNCOMERS (MARK II)
13      .0                   NOT USED
14   301.       ASEDPD       AEROSOL SEDIMENTATION AREA
15      .0      AIMPPD       PEDESTAL TOTAL IMPACTION AREA
16      .0      XDIMPD       PEDESTAL MINIMUM GRATE DIAMETER (OR THICKNESS)
17      .0      AGRAPD       PEDESTAL FLOW AREA THRU GRATE
18     1.E10    ZWPDWW       PEDESTAL-WETWELL OVERFLOW ELEVATION
19     4.0      APSUMP       PEDESTAL SUMP TOTAL AREA
20   573.35     ZPSUMP       ELEVATION AT BOTTOM OF PEDESTAL SUMP
21     9.8      XRBRPD       CHARACTERISTIC RADIUS OF PEDESTAL CAVITY FOR H2 BURNS
22    28.3      XHBRPD       CHARACTERISTIC HEIGHT OF PEDESTAL CAVITY FOR H2 BURNS
**                           THIS IS FLOOR TO CEILING
**
*************************************************************************
**                    MAAP BWR PLOT FILES
*************************************************************************
**
**@@@ REV 5 ADDITION, THIS PLTMAP SECTION IS NEW
**
** YOU CAN HAVE UP TO 25 PLOT FILES AND UP TO 99 VARIABLES.
** BEGIN EACH PLOT FILE SECTION WITH THE WORK "PLOTFIL" FOLLOWED BY
** THE UNIT NUMBER YOU WANT THE FILE WRITTEN TO.  A NEGATIVE UNIT NO.
** WILL FORCE BINARY OUTPUT.
**
** NEXT, SELECT THE VARIABLES YOU WANT TO BE PLOTTED BY SIMPLY
** SPECIFYING THE VARIABLE NAMES.  PLOT FILES 41 THRU 48 DEFINED BELOW
** ARE IDENTICAL TO THE "OLD" HARDWIRED MAAP PLOT FILES.
**
** FOR THE CASE OF A VARIABLE NOT PRESENT IN THE MAAP COMMON BLOCK BUT
** YOU WANT TO PLOT IT OUT (OR USE IT IN USER DEFINED EVENTS CODES),
** COMMON/XPLTX/ PLT(500) WAS PROVIDED EXPRESSLY FOR THAT PURPOSE.
** INSERT THE LINE  "COMMON/XPLTX/ PLT(500)"  INTO THE ROUTINE THAT
** HAS THE LOCAL VARIABLE YOU WANT TO SAVE, AND ASSIGN THE VALUE OF
** THE VARIABLE TO ONE OF THE ARRAY PLT INDICES.  THEN SELECT THAT
** ARRAY INDICE TO BE PLOTTED IN THE PLOTFIL SECTION.
**
** BE SURE TO END THIS SECTION WITH THE KEYWORD "END", AND **
**  COMMENTING IS ALLOWED.
**
** A NEW ADDITION TO THE OLD PLTFIL SCHEME IS THE CONSTRAINTS TO
** THE MINIMUM/MAXIMUM PLOT DT AS DETERMINED BY THE AUTODT PLOT SCALER.
** PREVIOUSLY UNDER THE OLD AUTODT SCHEME, THE PLOT SPACING BETWEEN THE
** PLOTTED DATA POINTS CAN BE AS SMALL AS MAAP TIMESTEP OR AS LARGE AS
** THE MAAP RUN TIME.  THIS ADDITION WILL PROVIDE REASONABLE CONTROL OF
** THE WAY PLOTTING DATA POINTS MAY TURN OUT, EG., ELIMINATION OF
** VERY NOISY (IE., MANY DATA POINTS OVER SMALL TIME INTERVAL) AND
** VERY COARSE (IE., FEW DATA POINTS SPREAD OVER LARGER TIME INTERVAL)
** PLOTS.  PRESENTLY, THE PLOT FREQUENCY IS SET TO A MINIMUM OF 1 SEC,
** AND MAXIMUM OF 5 MINUTE AS SPECIFIED BELOW.  SOME MAY FIND THIS
** UNSUITABLE AND MAY WANT TO ALLOW LARGER OR SMALLER FREQUENCY.  THE
** FORMAT TO SPECIFY THE PLOT FREQUENCY CONSTRAINTS IS
**
**          FREQ  <MINIMUM PLOT DT>  <MAXIMUM PLOT DT>
**
```

```
** WHERE THE MAX/MIN PLOT DT IS SUPPLIED IN SECONDS.  NOTE THAT PLOT
** FREQUENCY CONSTRAINTS APPLIES TO ALL PLOT FILES.
**
** PLOT FILES CAN ALSO BE SETUP VIA INPUT DECK THROUGH LOCAL PARAMETER
** CHANGE.  SIMPLY SPECIFY 25,0,0 FOLLOWED BY THE SYNTAX EXPLAINED,
** PRACTICALLY IDENTICAL TO THE SETUP BELOW BUT WITHOUT THE *PLTMAP
** LINE) AND BE SURE TO END PLTMAP INPUT WITH THE KEYWORD END.
**
*PLTMAP
**
PLOTFIL 41  /  PERRY PLOT FILE .PL1
**
XCNDWP
XCNPDP
ZCMDW
ZCMPD
ZWPD
WVHPS
WVLPS
WVRCI
WVLP1
WVLP2
XHCMLP
XWCOR
XWSH
XWJET
XWDW
XSPDW
XSPWW
WCRD
PDW
PPD
PPS
PCB
PDWWW
QCORE
T15P
T110P
TCMDW
TGDW
TGCA
TCMPD
TGPD
TGPS
TWSP
TGCB
TGWW
VLCSTP
MGLOBE
MU2CT
MCOGLO
MH2GLO
MO2DWP
MSTDWP
```

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MU2DWP
MCRUST
MLCMLP
MU2PDP
WH2COR
WSTBRK
WWBRK
WSTCFW
NFCODW
NFC2DW
NFH2DW
NFO2DW
NFSTDW
NFCOCA
NFC2CA
NFH2CA
NFO2CA
NFSTCA
NFCOPD
NFC2PD
NFH2PD
NFO2PD
NFSTPD
NFCOCB
NFC2CB
NFH2CB
NFO2CB
NFSTCB
NFCOWW
NFC2WW
NFH2WW
NFO2WW
NFSTWW
**
PLOTFIL 42  /  PERRY PLOT FILE .PL2
**
FMCSIP
FMC[illegible]P
FMCSIW
FMCSIR
FREL(1)
FREL(2)
FREL(3)
FREL(4)
FREL(5)
FREL(6)
FREL(7)
FREL(8)
FREL(9)
FREL(10)
FREL(11)
FREL(12)
MFPTC
MFPTP
```

```
MFPREX(3)
MFPREX(4)
MFPREX(5)
MFPREX(6)
MFPREX(7)
MFPREX(8)
MFPREX(9)
MFPREX(10)
MFPREX(11)
MFPREX(12)
**
ENDPLOT
**
**************************************************************************
*PRIMARY SYSTEM
**************************************************************************
01    79.01     AFLCOR    FLOW AREA OF REACTOR CORE
02    67.17     ALSH      FLOW AREA IN LOWER SHROUD
03    34.77     AFLBYP    CORE BYPASS FLOW AREA
04   222.81     AUSH      FLOW AREA IN UPPER SHROUD
05    47.72     HCRD      SPECIFIC ENTHALPY OF FLOW IN CRD TUBES
06   395.96     HFW       SPECIFIC ENTHALPY OF FEEDWATER
07   343761.    MU2PS     TOTAL MASS OF UO2 IN CORE
08   748        NASS      NUMBER OF FUEL ASSEMBLIES IN REACTOR CORE
09    64        NPINS     NUMBER OF FUEL RODS IN A FUEL ASSEMBLY
10   177        NCRD      NUMBER OF CRD TUBES
11     5.63     NOFPS     SENSIBLE ENERGY STORED IN FUEL (FULL POWER SECONDS)
12     9.58E-4  TDMSIV    DELAY TIME FOR MSIV CLOSURE
13      .00097  TDSCRM    DELAY TIME FOR FULL SCRAM
14   11828.     TIRRAD    TOTAL EFFECTIVE IRRADIATION TIME FOR CORE
15    1460.     WVCRDI1   CRD PUMP CURVE:
16    2166.     WVCRDI2     - VOLUMETRIC FLOW
17    2166.     WVCRDI3
18    2166.     WVCRDI4
19    2166.     WVCRDI5
20    2166.     WVCRDI6
21    2166.     WVCRDI7
22    2166.     WVCRDI8
23    1047.7    PCRD1        - PRESSURE
24      14.7    PCRD2
25      14.7    PCRD3
26      14.7    PCRD4
27      14.7    PCRD5
28      14.7    PCRD6
29      14.7    PCRD7
30      14.7    PCRD8
31    1.778E7   WFWMAX    MAXIMUM FEEDWATER FLOW RATE (RUN OUT FLOW)
32    5.389E6   WBPMAX    MAXIMUM TURBINE BYPASS FLOW RATE
33     .1437    NXCORE    EXIT CORE QUALITY AT TIME ZERO
34    16.158    XDCORE    REACTOR CORE DIAMETER TO INNER SHROUD WALL
35    70.305    XHRV      INTERIOR HEIGHT OF REACTOR VESSEL
36     9.918    XRRV      INTERIOR RADIUS OF REACTOR VESSEL
37   616.919    ZBJET     ELEVATION AT BOTTOM OF JET PUMPS
38   609.016    ZBRDT     ELEVATION AT BOTTOM OF CRD TUBES
```

39	647.844	ZBSEP	ELEVATION AT BOTTOM OF STEAM SEPARATORS
40	605.194	ZBV	ELEVATION AT BOTTOM OF REACTOR VESSEL
41	622.070	ZCPL	ELEVATION AT CORE PLATE
42	631.171	ZTJET	ELEVATION AT TOP OF JET PUMPS
43	12.056	AJET	TOTAL AREA OF JET PUMPS
44	635.282	ZTOAF	ELEVATION AT TOP OF ACTIVE FUEL
45	657.785	ZTSEP	ELEVATION AT TOP OF STEAM SEPARATORS
46	651.615	ZWNORM	ELEV AT NORMAL SHROUD LVL (SAME AS INITIAL COND: ZWSHO)
47	619.206	ZLOCA	ELEVATION AT BREAK
48	2.702	ALOCA	AREA OF BREAK
49	653.574	ZWL8	ELEVATION AT LEVEL L8 TRIP
50	619.206	ZSRR	ELEVATION ON RPV WALL FOR RECIRC PUMP SUCTION PIPE
51	650.040	ZSCRAM	LOW WATER LEVEL L3 SCRAM
52	1094.4	PSCRAM	HIGH PRESSURE SCRAM SETPOINT
53	19500	MCSPT	MASS OF CORE SUPPORT PLATE
54	.734	TDSLC	TIME FOR SCRAM WITH SLC - 1 PUMP
55	.0	TIRR(1)	RECIRC PUMP COAST DOWN CURVE:
56	.0005556	TIRR(2)	- TIME AFTER TRIP
57	.001111	TIRR(3)	
58	.001667	TIRR(4)	
59	.002222	TIRR(5)	
60	.002778	TIRR(6)	
61	.004167	TIRR(7)	
62	.01111	TIRR(8)	
63	1.00	FWRR(1)	
64	.76	FWRR(2)	
65	.58	FWRR(3)	
66	.44	FWRR(4)	
67	.33	FWRR(5)	
68	.25	FWRR(6)	
69	.13	FWRR(7)	
70	.00	FWRR(8)	
71	113.	HSLC	INLET ENTHALPY OF SLC
72	14.7	PSLC(1)	SLC FLOW CURVE:
73	214.7	PSLC(2)	- PRESSURE
74	414.7	PSLC(3)	
75	614.7	PSLC(4)	
76	814.7	PSLC(5)	
77	1014.7	PSLC(6)	
78	1214.7	PSLC(7)	
79	1414.7	PSLC(8)	
80	352.9	WVSLC(1)	- FLOW
81	352.9	WVSLC(2)	
82	352.9	WVSLC(3)	
83	352.9	WVSLC(4)	
84	349.7	WVSLC(5)	
85	349.7	WVSLC(6)	
86	349.7	WVSLC(7)	
87	349.7	WVSLC(8)	
88	1.2E-5	TDRPT	DELAY TIME FOR RECIRC PUMP TRIP
89	636.474	ZLMSIV	LOW WATER LEVEL L1 FOR MSIV CLOSURE
90	645.915	ZLRPT	LOW WATER LEVEL L2 FOR RECIRC PUMP TRIP
91	1.E10	PHRPT	HIGH VESSEL PRESSURE FOR RECIRC PUMP TRIP
92	16.58	PDWSCM	HIGH DRYWELL PRESSURE FOR SCRAM

93	.031	FENRCH	NORMAL FUEL ENRICHMENT
94	23121.2	EXPO	AVERAGE BURNUP IN MWD/TONNE
95	.655	FCR	PRODUCTION OF U239 TO ABSORPTION IN FUEL
96	1.13	FFAF	RATIO OF FISSILE ABSORPTION TO TOTAL FISSION
97	.70	FQFR1	FISSION POWER FRACTION OF U235 AND PU241
98	.266	FQFR2	FISSION POWER FRACTION OF PU239
99	.08	FQFR3	FISSION POWER FRACTION OF U238
100	1.0000	XPCRDT	PITCH OF CRD TUBES
101	.8333	XDCRDT	OUTER DIAMETER OF CRD TUBE
102	54	NINST	NUMBER OF INSTRUMENT TUBES
103	.0120	XTHCRD	THICKNESS OF CRD TUBE WALL
104	.1692	XDINST	OUTER DIAMETER OF INSTRUMENT TUBE
105	.500	XDRIVE	LOWER CRD DRIVE OUTER DIAMETER
106	.01627	VWCRD	SPECIFIC VOLUME OF CRD WATER
107	.01632	VWCST	SPECIFIC VOLUME OF SLC WATER
108	64.7	PADSC	DRYWELL PRESS WHICH WILL CLOSE ADS VALVES
109	64.7	PADSO	DW PRESS WHICH WILL ALLOW ADS TO RE-OPEN IF CLOSED
110	.750	XTRV	THICKNESS OF LOWER VESSEL HEAD
111	.000000	TIFWCD1	FEEDWATER PUMP COAST DOWN:
112	.0000278	TIFWCD2	- TIME
113	.000500	TIFWCD3	
114	.001111	TIFWCD4	
115	.001444	TIFWCD5	
116	.001944	TIFWCD6	
117	.002333	TIFWCD7	
118	.00500	TIFWCD8	
119	1.536E7	WFWCD1	- MASS FLOW
120	1.64E7	WFWCD2	
121	1.23E7	WFWCD3	
122	1.54E7	WFWCD4	
123	1.15E7	WFWCD5	
124	3.07E6	WFWCD6	
125	.0	WFWCD7	
126	.0	WFWCD8	
127	809.7	PLMSIV	LOW RPV PRESSURE FOR MSIV CLOSURE
128	658.66	ZMSL	ELEVATION AT CENTER LINE OF MAIN STEAM LINE
129	28.81	XATWS(1)	NSAC 70 ATWS POWER VS RPV LEVEL REFERENED TO BAF
130	12.50	XATWS(2)	- REFERENCE TO THE BAF
131	10.00	XATWS(3)	
132	8.50	XATWS(4)	
133	.0	XATWS(5)	
134	.0	XATWS(6)	
135	.0	XATWS(7)	
136	.0	XATWS(8)	
137	.50	FQATWS(1)	FRACTION OF TOTAL POWER
138	.206	FQATWS(2)	
139	.13	FQATWS(3)	
140	.084	FQATWS(4)	
141	.0	FQATWS(5)	
142	.0	FQATWS(6)	
143	.0	FQATWS(7)	
144	.0	FQATWS(8)	
145	55700	MCS	MASS OF CORE SHROUD FROM TAF TO BOTTOM OF CORE
146	31800	MTG	MASS OF CORE TOP QUIDE

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147     47100   MSH        MASS OF SHROUD HEAD
148     70100   MSP        MASS OF STANDPIPES AND SEPARATORS
149     77200   MDR        MASS OF STEAM DRYERS
150   1586000   MUH        MASS OF RPV UPPER HEAD
151    449200   MUW        MASS OF RPV WALL FROM MSL ELEVATION TO TAF
152    352800   MLW        MASS OF RPV WALL FROM TAF TO BOTTOM OF JET PUMP
153     40000   MRR        MASS OF RECIRC DISCHARGE PIPING
154    162500   MLH        MASS OF RPV LOWER HEAD
155     35800   MSS        MASS OF SHROUD SUPPORT
156       116   XZRR       LENGTH OF RECIRC PIPE MODELED
157    1.72     XDRR       ID OF RECIRC PIPE
158      9205   ASEP       TOTAL SURFACE AREA OF ALL STANDPIPES + SEPARATORS
159   57.91     AGSEP      TOTAL GAS FLOW AREA OF ALL STANDPIPES
160   20100     ADR        SURFACE AREA OF STEAM DRYERS
161   10.00     AUDSS      FLOW AREA FROM UPPER DOWNCOMER TO SEPARATORS
162   39.       AUHUD      FLOW AREA FROM UPPER HEAD TO UPPER DOWNCOMER
163  640.564    ZBSTAN     ELEVATION AT BOTTOM OF STANDPIPES (AVERAGE VALUE)
164  1133.83    VSHED      TOTAL VOLUME INSIDE SHROUD HEAD
165  2076.27    VSEP       TOTAL VOLUME INSIDE STANDPIPES+SEPARATORS
166  675.505    ZTV        ELEVATION AT TOP OF RPV
167  445200.    QCO        RPV CONVECTION LOSSES AT TIME ZERO
168    10       FINPLT     REFLECTIVE INSULATION WITH FINPLT PLATES
169      .250   XTINS      THICKNESS OF INSULATION
170      .0     AIMPPS     PRIMARY SYSTEM TOTAL IMPACTION AREA
171      .0     XDIMPS     PRIMARY SYSTEM MINIMUM GRATE DIAMETER (OR THICKNESS)
172      .0     AGRAPS     PRIMARY SYSTEM FLOW AREA THRU GRATE
173  261        NSEP       NUMBER OF STEAM SEPARATORS
176    1.E10    PHY315     HIGH RPV PRESSURE FOR STARTUP
177      .0                NOT USED
178   -1.E10    ZLY315     LOW RPV WATER LEVEL ELEVATION TO TRIP OFF
196      .0     VRRMIN     (SWEDISH REACTORS ONLY)
197   20        NJET       NUMBER OF JET PUMPS
198      .003   XTJET      AVERAGE THICKNESS OF JET PUMP WALL
199  676.1      ZSPARG     ELEVATION OF RCIC SPARGER
200      .003   XSPARG     DROPLET DIAMETER FOR RCIC SPARGER
201    1.E10    TD315      TIME DELAY FOR ISO CONDENSER - N/A FOR PERRY
202  644.5      ZSPRCS     ELEVATION OF LPCS/HPCS SPARGER
203      .0003  XSPRCS     DROPLET DIAMETER FOR LPCS/HPCS SPARGER
**
***************************************************************************
*SUPPRESSION POOL(MARKIII ONLY)
***************************************************************************
01        482.  ASPDW      AREA OF DRYWELL SIDE OF SUPPRESSION POOL
02       5900.  ASPWW      AREA OF CONTAINMENT SIDE OF SUPPRESSION POOL
03          40  NVT1       NUMBER OF VENTS OF TYPE #1 -- TOP
04          40  NVT2       NUMBER OF VENTS OF TYPE #2 -- MID
05          40  NVT3       NUMBER OF VENTS OF TYPE #3 -- BOTTOM
06      2.29    XDIAVT     DIAMETER OF ONE SUPPRESSION POOL VENT
07    591.470   ZLLSP      ELEVATION OF SPMU POOL LOW LEVEL SETPOINT
08    587.000   ZVT1       ELEVATION OF TOP OF VENT TYPE #1
09    582.500   ZVT2       ELEVATION OF TOP OF VENT TYPE #2
10    578.000   ZVT3       ELEVATION OF TOP OF VENT TYPE #3
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*TIMING DATA
**************************************************************************
01   20.DO     TDMAX      MAXIMUM ALLOWED TIME STEP
02    1.D-3    TDMIN      MINIMUM ALLOWED TIME STEP
03    2.5D-2   FMCHMX     MAXIMUM MASS CHANGE (%) FOR INTEGRATION
04    2.5D-2   FUCHMX     MAXIMUM GAS TEMP CHANGE FRACTION FOR INTEGRATION
05    2.5D-2   MDFPMN     MIN FISS PROD MASS ALLOWED TO CONTROL TIME STEP
**
**************************************************************************
*TOPOLOGY
**************************************************************************
**
**THIS SECTION DEFINES THE WAYS THAT THE VARIOUS AUX NODES ARE CONNECTED
**TOGETHER--THERE ARE THREE FORMATS FOR ENTERING DATA THAT ARE DESCRIBED
**BELOW; THE LAST CARD IN THIS SECTION MUST BE "END"
**
**   1. "JUNCTION" CARDS--THIS IS DEFINED BY A CARD WITH A "J" IN COLUMN
**        1 FOLLOWED BY A CARD WITH THE FOLLOWING INFORMATION:
**        A. NODE NO. OF THE VOLUME ON THE UPSTREAM SIDE OF JUNCTION;
**        B. NODE NO. OF DOWNSTREAM VOLUME;
**        C. 1 IF JUNCTION IS IN A HORIZONTAL WALL (IE FLOW IS VERTICAL,
**           USE 0 IF JUNCTION IS IN A VERTICAL WALL);
**        D. ELEVATION OF THE BOTTOM OF THE JUNCTION ABOVE THE FLOOR
**           OF THE UPSTREAM NODE;
**        E. FACING THE HOLE, THE WIDTH OF JUNCTION;
**        F. FACING THE HOLE, THE HEIGHT OF JUNCTION;
**        G. LENGTH OF JUNCTION;
**        H. AREA OF JUNCTION
**
**        NOTE: IF WIDTH=HEIGHT, THE JUNCTION IS ASSUMED CIRCULAR, OTHERWISE
**        RECTANGULAR (USE WIDTH SLIGHTLY DIFFERENT THAN HEIGHT FOR SQUARE)
**        EVEN IF THE JUNCTION IS RECTANGULAR, THE AREA CAN BE DIFFERENT
**        THAN THE PRODUCT OF LENGTH AND WIDTH IF THE JUNCTION REPRESENTS THE
**        SUM OF SEVERAL HOLES WHICH HAVE THE SAME ELEVATION, ETC.
**
**   2. "FAILURE" CARDS--THIS IS DEFINED BY A CARD WITH AN "F" IN COLUMN
**        1 FOLLOWED BY A CARD WITH THE FOLLOWING INFORMATION:
**        A. NODE NO. OF NODE WHICH CAN FAIL (UPSTREAM NODE);
**        B. NODE NO. THAT THE FAILED VOLUME  BLOWS DOWN INTO;
**        C. 1 IF THE JUNCTION IS HORIZ (0 IF VERTICAL);
**        D. ELEVATION OF THE BOTTOM OF THE OPENING ABOVE THE FLOOR OF
**           THE FAILED NODE;
**        E. FACING THE HOLE, THE WIDTH OF JUNCTION;
**        F. FACING THE HOLE, THE HEIGHT OF JUNCTION;
**        G. LENGTH OF JUNCTION;
**        H. AREA OF JUNCTION;
**        I. DIFFERENTIAL PRESSURE REQUIRED TO FAIL THE NODE IF THE UPSTREAM
**           NODE HAS THE HIGHEST PRESSURE
**        J. DIFFERENTIAL PRESSURE REQUIRED TO FAIL THE NODE IF THE DOWNSTRM
**           NODE HAS THE HIGHEST PRESSURE
**
**        NOTE: IF WIDTH=HEIGHT, THE JUNCTION IS ASSUMED CIRCULAR, OTHERWISE
**        RECTANGULAR (USE WIDTH SLIGHTLY DIFFERENT THAN HEIGHT FOR SQUARE)
**        EVEN IF THE JUNCTION IS RECTANGULAR, THE AREA CAN BE DIFFERENT
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**       THAN THE PRODUCT OF LENGTH AND WIDTH IF THE JUNCTION REPRESENTS THE
**       SUM OF SEVERAL HOLES WHICH HAVE THE SAME ELEVATION, ETC.
**
**   3. "CONTAINMENT INTERFACE" CARD--ONE SUCH SET OF TWO CARDS SHOULD BE
**      PROVIDED
**      THE FIRST CARD SHOULD HAVE A "C" IN COLUMN ONE;
**       THE SECOND CARD GIVES:
**       A. THE NODE NO. WHICH RECEIVES FLUID FROM THE CONTAINMENT (OR
**          PRIMARY SYSTEM FOR V SEQUENCES) AND
**       B. ELEVATION ABOVE THE FLOOR OF THIS NODE OF THE TOP OF THE
**          JUNCTION THROUGH WHICH THE PRI SYS OR CONTMT EFFLUENT IS ISSUING
**
**IMPORTANT NOTE:
**THE MODEL WILL NOT RELIABLY FIND A
**SOLUTION FOR THE AUX BLDG FLOWS IN ONE SPECIFIC CIRCUMSTANCE:
**           1. TWO VOLUMES ONE ABOVE THE OTHER
**           2. PARALLEL FLOW PATHS CONNECTING THE TWO VOLUMES THROUGH
**              THE FLOOR OF THE UPPER VOLUME
**
**IT APPEARS THAT SUCH A SITUATION IS NUMERICALLY ILL-POSED
**TO AVOID PROBLEMS, IT IS RECOMMENDED IN SUCH A SITUATION TO
**LUMP THE TWO FLOWPATHS TOGETHER
**
JUNCTION 1-8
1 8 1 25.5 .17 13.12 4.0 8.07
JUNCTION 2-8
2 8 0 29.5 .17 34.35 3.5 7.86
JUNCTION 8-5
8 5 0 29.5 .17 44.94 3.0 7.86
JUNCTION 8-3
8 3 1 42.5 .17 33.41 5.0 11.36
JUNCTION 1-8
1 8 0 54.5 .17 34.64 3.5 5.89
JUNCTION 2-8
2 8 0 54.5 .17 34.64 3.5 5.89
JUNCTION 1-3
1 3 0 47.5 .17 39.65 5.0 6.74
JUNCTION 2-3
2 3 0 47.5 .17 39.65 5.0 6.74
JUNCTION 8-4
8 4 0 54.5 .17 69.18 3.0 11.76
JUNCTION 5-4
5 4 1 43.0 64.76 18.0 1.0 1428.94
JUNCTION 5-4
5 4 1 43.0 7.75 12.33 1.0 95.56
JUNCTION 9-10
9 10 0 25.0 14.0 20.0 2.0 280.0
FAILURE 21
1 7 1 29.5 8.1 7.9 2.67 64.0 3.1 3.1
FAILURE 22
1 7 0 30.17 3.0 7.17 10.5 21.5 3.9 3.9
FAILURE 23
1 8 0 30.17 3.0 7.17 3.0 21.5 3.9 2.2
FAILURE 24
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2 7 0 30.17 3.0 7.17 10.5 21.5 3.9 3.9
FAILURE 25
2 8 0 30.17 3.0 7.17 3.0 21.5 3.9 2.2
FAILURE 26
2 7 1 29.5 8.1 7.9 2.67 64.0 3.1 3.1
FAILURE 27
5 10 1 25.0 8.0 22.0 101.0 352.0 5.0 5.0
FAILURE 28
5 10 0 25.0 3.0 7.17 18.0 21.5 3.9 3.9
FAILURE 29
7 5 0 29.5 3.0 7.17 55.0 100.2 3.9 3.9
FAILURE 30
7 9 0 29.5 3.0 7.17 3.0 64.5 2.2 3.9
FAILURE 31
1 6 0 55.17 3.0 7.17 9.0 21.5 3.9 3.9
FAILURE 32
2 6 0 55.17 3.0 7.17 9.0 21.5 3.9 3.9
FAILURE 33
1 10 1 92.5 13.17 12.0 3.0 158.0 3.3 3.3
FAILURE 34
2 10 1 92.5 13.17 12.0 3.0 158.0 3.3 3.3
FAILURE 35
6 4 0 0.0 3.0 7.17 14.0 43.0 3.9 3.9
FAILURE 36
3 9 0 0.0 20.2 20.1 250.0 406.5 0.4 0.4
FAILURE 37
9 10 0 94.92 44.6 44.8 1.0 2000.0 0.7 0.7
FAILURE 38
6 9 0 0.0 8.0 7.83 4.0 125.3 2.2 3.9
FAILURE 39
9 10 0 88.0 3.0 7.17 4.0 43.0 2.2 3.9
FAILURE 40
6 9 0 19.0 6.0 7.83 3.0 94.0 2.2 3.9
FAILURE 20
7 6 1 29.5 3.0 7.17 25.0 43.0 3.9 2.2
FAILURE 41
8 10 1 92.5 .167 408.4 63.0 68.2 0.7 0.7
FAILURE 43
8 7 1 28.3 8.10 7.90 1.17 128. 1.31 1.31
CONTAINMENT INTERFACE
1   10.00
END
**
**********************************************************************
*USEREVT
**********************************************************************
**
** THIS DEFINES THE USER DEFINED EVENT CODES.  THE FOLLOWING SYNTAX
**  SHOWN BELOW ARE WHAT IS NORMALLY EXPECTED IN THE *USEREVT PARAMETER
**  SECTION.  ANYTHING ELSE IS IGNORED AND THE USER IS WARNED.
**
**  1) **        / COMMENTING
**  2) END       / END OF SECTION KEYWORD
**  3) <NUMBER> <EXPRESSION> / USER DEFINED EVENT CODE
```

```
**  4) <TRUE>    <MESSAGE>     / USER SUPPLIED TRUE MESSAGE
**     <FALSE>   <MESSAGE>     / USER SUPPLIED FALSE MESSAGE
**  5) SELECT <NUMBER1> <NUMBER2> ... ETC.
**     SELECT ALL
**
**
**  OPTIONAL USER DEFINED EVENT MESSAGE DISCRIPTION MESSAGE CAN FOLLOW
**   THE EVENT CODE EXPRESSION.  THE ONLY RESTRICTION IS THAT SUCH
**   MESSAGE NEED TO CONFORM TO THE FOLLOWING FORMAT AND MUST COME AFTER
**   THE EVENT CODES THE MESSAGES ARE DEFINED FOR.
**
**       <"TRUE"/"FALSE">  <USER DEFINED MESSAGE>
**
**  AND BOTH TRUE AND FALSE MESSAGE OR JUST ONE OR NONE COULD BE SUPPLIED.
**   TOKENS "TRUE", "T", "FALSE", AND "F" ARE ACCEPTABLE.  NOTE THAT THE
**   CODE WILL GENERATE THE EVENT MESSAGE FROM THE USER DEFINED EVENT CODE
**   EXPRESSION, AND SUPERSEDE IT WITH USER'S IF SUPPLIED.
**
**
**  THE SELECT KEYWORD IS USED TO "SELECT" THE USER DEFINED EVENT CODE
**   NUMBERS TO BE WRITTEN TO THE SUMMRY FILE AND LOG FILE IF NEGATIVE
**   WHEN THE CORRESPONDING EVENT CODE NUMBER STATUS HAS CHANGED.  IF
**   YOU WANT ALL, SIMPLY SAY SELECT ALL, AND THEY WILL ALL BE WRITTEN TO
**   THE SUMMRY FILE.  NOTE THAT NO MESSAGE IS WRITTEN TO THE LOG FILE
**   UNLESS YOU SELECTED A NUMBER WITH A NEGATIVE SIGN.  NOTE THAT WHEN
**   AN MAAP OPERATOR EVENT CODE STATUS CHANGED DUE TO USER DEFINED EVENT
**   CODE STATUS CHANGE, IT IS REPORTED TO THE LOG FILE ALWAYS.
**
**
**  AN VALID USER DEFINED EVENT CODE EXPRESSION FORMAT CONSISTS OF AN
**   EVENT CODE NUMBER AND CORRESPONDING EVENT CODE EXPRESSION, EG.,
**
**      <EVENT CODE NUMBER> <DEFINING EVENT CODE EXPRESSION>
**
**  EVENT CODE NUMBERS 400-699 ARE ALLOCATED FOR USER DEFINED EVENT CODE
**   DEFINITIONS.  ALSO ALLOWED ARE MAAP DEFINED OPERATOR EVENT FLAGS SO
**   THAT THE USER CAN HAVE THE FLEXIBILTY TO CONTROL HARDWIRED MAAP
**   FUNCTIONS, EG., MANUAL SCRAM, TURN CONTAINMENT SPRAYS ON, ETC.
**   THE MAAP OPERATOR EVENT FLAG CODES ARE 200-399.  CONSULT THE MAAP
**   USER MANUAL OR THE *EVTMES PARAMETER SECTION FOR DEFINITION OF
**   MAAP OPERATOR CODES.
**
**  AN VAILD EVENT EXPRESSION FORMAT IS EXPECTED TO CONFORM TO ONE OF THE
**   FOLLOWING FORMATS:
**
**  1) "EVENT"  <NUMBER>  <"TRUE"/"FALSE"> <"LOCKOUT">
**
**  2) <VARIABLE1/REAL1>  <REAL_OPERATOR>  <VARIABLE2/REAL2>  <"LOCKOUT">
**
**  3) <FORMAT1/FORMAT2> <LOGICAL_OPERATOR> <FORMAT1/FORMAT2> <"LOCKOUT">
**
**   WHERE    "EVENT"  = THE KEYWORD "EVENT"
**           <NUMBER>  = THE CORRESPONDING EVENT CODE NUMBER
**   <"TRUE"/"FALSE">  = THE KEYWORD "TRUE" OR "FALSE"
```

```
**
**        <VARIABLE1>  = THE MAAP COMMON BLOCK VARIABLE NAME
**            <REAL1>  = THE NUMBERIC REAL VALUE
**    <REAL_OPERATOR>  = SELF EXPLANATORY (SEE BELOW)
**        <VARIABLE2>  = THE MAAP COMMON BLOCK VARIABLE NAME
**            <REAL2>  = THE NUMBERIC REAL VALUE
**
**          <FORMAT1>  = IS THE FORMAT DEFINED ABOVE
**          <FORMAT2>  = IS THE FORMAT DEFINED ABOVE
** <LOGICAL_OPERATOR>  = SELF EXPLANATORY (SEE BELOW)
**
**        <"LOCKOUT">  = OPTIONAL "LOCKOUT" KEYWORD
**
**  THE FIRST FORMAT TYPE IS DEFINED AS A LOGICAL EXPRESSION, THE
**   SECOND FORMAT TYPE IS DEFINED AS A REAL EXPRESSION, AND THE THIRD
**   FORMAT TYPE IS DEFINED AS AN MULTIPLE EXPRESSION CONSISTING OF A
**   COMBINATIONS OF FORMAT1 AND/OR FORMAT2.
**
**  NOTE THAT "/" USED ABOVE IN DEFINING THE FORMATS IS EXPRESSED AS
**   "EITHER".  THUS THERE ARE TWO LOGICAL EXPRESSION, FOUR REAL
**   EXPRESSION, AND FOUR MULTIPLE EXPRESSION POSSIBLE COMBINATIONS.
**
**  OPTIONAL KEYWORD "LOCKOUT" TOKEN COULD BE ADDED AT END OF LINE.
**   THIS WILL PERMANETLY LOCK THE EVENT CODE TO TRUE ALWAYS ONCE THE
**   DEFINING EXPRESSION IS SATISFIED. TOKENS "L", "LKO", & "LO" ARE
**   ACCEPTABLE.
**
**  ALLOWABLE <REAL_OPERATOR> TOKENS ARE
**
**       >  or GT           (GREATER THAN)
**       <  or LT           (LESS THAN)
**       >= or GE or =>     (GREATER THAN OR EQUAL TO)
**       <= or LE or =<     (LESS THAN OR EQUAL TO)
**       =  or EQ           (EQUAL TO)
**       <> or NE           (NOT EQUAL TO)
**
**  AND ALLOWABLE <LOGICAL_OPERATOR> TOKENS ARE
**
**       AND or A
**       OR  or O
**
**
**  WE WILL NOW SHOW SOME EXAMPLES OF USER DEFINED EVENT CODES, AND
**   WE'LL START WITH A SIMPLE EXPRESSION AND END WITH A MULTIPLE
**   EXPRESSION EXAMPLE.
**
**  LET'S SAY WE HAVE THE FOLLOWING SIMPLE EXPRESSION;
**
**       401 PPS > 1.E6
**
**  EVENT CODE 401 IS TRUE WHEN PPS, THE REACTOR VESSEL PRESSURE, IS
**   GREATER THAN 1.E6 PASCALS, OTHERWISE IT IS FALSE.
**
**
```

```
**  THE SECOND EXAMPLE INVOLVING MULTIPLE EXPRESSION, AS SHOWN IS;
**
**     402 TGPS > 450 AND EVENT 301 TRUE LOCKOUT
**
**  EVENT CODE 402 IS SET PERMANETLY TRUE WHEN TGPS, THE REACTOR GAS
**   TEMPERATURE, IS GREATER THAN 450 KELVINS AND WHEN EVENT CODE 401
**   IS TRUE.
**
**
**  TWO IMPORTANT NOTES MUST BE MADE.  THE FIRST IS THAT ALL NUMBERS
**   ARE EXPECTED TO BE IN SI UNITS.  THIS IS DONE TO PREVENT CONFUSION
**   AS TO WHAT MAAP COMMON BLOCK VARIABLES HAVE DEFINING UNITS NUMBERS
**   ASSIGNED.  THOSE THAT DO, CAN BE EASILY CONVERETED TO/FROM SI AND
**   BRITISH UNITS.  THOSE THAT DO NOT ARE ALWAYS IN SI UNITS.  SINCE NOT
**   ALL MAAP COMMON BLOCK VARIABLES HAVE DEFINING UNIT NUMBERS, THE
**   POTENTIAL TO CONFUSE WHAT TYPE OF NUMBERS TO INPUT IS GREAT AND WE
**   WANT TO AVOID THIS SITUATION.  HOPEFULLY THIS WOULD BE RECIFIED IN
**   THE NEAR FUTURE.
**
**  THE SECOND NOTE IS THAT USER DEFINED EVENT CODES ARE EVALUATED
**   IMMEDIATELY AFTER MAAP EVENT CODES ARE EVALUATED.  USER DEFINED
**   EVENT CODES ARE EVALUATED SEQUENTIALLY FROM 400 TO 699 FIRST, AND
**   THEN USER DEFINED EVENT CODES 300-399 (WHICH HAVE A ONE TO ONE
**   CORRESPONDENCE WITH MAAP OPERATOR EVENT CODES) ARE EVALUATED NEXT.
**   THIS IS IMPORTANT TO NOTE SINCE IF THE EVALUATION OF AN EVENT CODE
**   DEPENDS ON THE STATUS OF ANOTHER EVENT CODE, YOU'LL WANT TO BE
**   SURE THAT THE "OTHER" EVENT CODE HAS ALREADY BEEN EVALUTED FIRST.
**
**  USER DEFINED EVENT CODES CAN ALSO BE DEFINED IN THE INPUT DECK VIA
**   LOCAL PARAMETER CHANGE.  SPECIFY 28,0,0 FOLLOWED BY THE SYNTAX
**   EXPLAINED ABOVE.  BE SURE TO END WITH THE KEYWORD END.
**
**  IT IS HOPED THAT THE ABOVE DESCRIPTIONS SHOULD BE MORE THAN ADQUATE
**   IN EXPLAINING THE PURPOSE OF USER DEFINED EVENTS.
**
END
**
************************************************************************
*WETWELL
************************************************************************
01   574.83    ZWWF        ELEVATION AT WETWELL FLOOR
02      .267   AVB         FLOW AREA THROUGH A DRYWELL VACUUM BREAKER
03     2       NVB         NUMBER OF DRYWELL VACUUM BREAKERS
04      .5     PSETVB      PRESSURE SETPOINT FOR VACUUM BREAKERS
05      .1     PDVB        DEAD BAND FOR VACUUM BREAKERS
06    265023.  VOLWW       TOTAL VOLUME OF WETWELL (PLUS SUPP POOL)
07      .5     RELHWW      RELATIVE HUMIDITY IN WETWELL
08    10       NIGWW       NUMBER OF IGNITERS IN THE WETWELL
09    [illegible]1.5     XIGWW       AVERAGE DISTANCE FROM SUPR PL WATER LEVEL TO IGNITERS
11      .0     AWWF        AREA OF WETWELL FLOOR (MARK II)
12      8273.  ASEDWW      AEROSOL SEDIMENTATION AREA
13      .49[illegible]   ACVENT      CONTAINMENT VENT AREA (FPCC VENT)
14   743.      AIMPWW      WETWELL TOTAL IMPACTION AREA
15      .0208  XDIMWW      WETWELL MINIMUM GRATE DIAMETER (OR THICKNESS)
```

```
16  2796.      AGRAWW     WETWELL FLOW AREA THRU GRATE
17    53.2     XHSPWW     SPRAY FALL HEIGHT IN WETWELL
18   633.      ZCFAIL     ELEVATION OF CONTAINMENT VENT IN WETWELL
19   579.42    ZSRVD      AVERAGE ELEVATION OF SRV DISCHARGE IN SUPP POOL
20   114.7     PCFM3      FAILURE PRESSURE OF CONTAINMENT
21      .0     XRCONT     CONTAINMENT RADIUS
22      .0     NHOOPW     NUMBER OF TENDONS IN HOOP DIRECTION
23      .0     XTREHW     VOLUME OF REBAR PER UNIT AREA OF OUTER WALL
24      .0     XTREZW     VOLUME OF REBAR PER UNIT AREA OF OUTER WALL
25      .0     XDHOPW     DIAMETER OF HOOP TENDONS
26      .0     ZWCYL      HEIGHT OF THE CYLINDRICAL PART OF THE WETWELL WALL
27      .0     XDZFW      DISPLACEMENT IN AXIAL DIRECTION
28      .0     XDRFW      SAME AS 29 FOR THE RADIAL DIRECTION
29      .0     NTENZ      NUMBER OF TENDONS IN AXIAL DIRECTION
30      .0     XDTENZ     DIAMETER OF TENDONS IN AXIAL DIRECTION
31     9.      XRBRWW     CHARACTERISTIC RADIUS OF WETWELL FOR H2 BURNS
32    27.4     XHBRWW     CHARACTERISTIC HEIGHT OF WETWELL FOR H2 BURNS
**                        THIS IS FLOOR TO CEILING
**
**          UNITS SYSTEM CLARIFICATION
** AS WAS PREVIOUSLY STATED, PLACING A *SI AT THE END OF THE
** PARAMETER FILE WOULD MAKE THE CODE EXPECT THE INPUT DECK TO BE IN SI
** UNITS FOR INPUT NUMBERS, AND ALL OUTPUT FILES WOULD BE IN SI UNITS.
** CONVERSLY, IF *BR WAS PLACED AT THE END OF THE PARAMETER FILE, INPUT
** UNITS WOULD BE EXPECTED TO BE IN BRITISH UNITS, AND ALL OUTPUT FILES
** WOULD BE IN BRITISH UNITS.
**
*BR
**
```