WPPSS-FTS-133

WASHINGTON PUBLIC POWER

SUPPLY SYSTEM

INDIVIDUAL PLANT EXAMINATION
WASHINGTON NUCLEAR PLANT 2

MAIN REPORT

Revision 1: JULY 1994

Washington Public Power Supply System
3000 George Washington Way
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TABLE OF ACROYNMS

ADS	AUTOMATIC DEPRESSURIZATION SYSTEM
ANSI	AMERICAN NATIONAL STANDARDS INSTITUTE
ARI	ALTERNATE ROD INSERTION SYSTEM
ASEP	ACCIDENT SEQUENCE EVALUATION PROGRAM
ASME	AMERICAN SOCIETY OF MECHANICAL ENGINEERS
ATWS	ANTICIPATED TRANSIENT WITHOUT SCRAM
BHEP	BASIC HUMAN ERROR PROBABILITY
BPA	BONNEVILLE POWER ADMINISTRATION
BWR	BOILING WATER REACTOR
BWROG	BOILING WATER REACTOR OWNERS GROUP
CAS	CONTROL AIR SYSTEM
CCF	COMMON CAUSE FAILURE
CCFP	CONDITIONAL CONTAINMENT FAILURE PROBABILITY
CDF	CORE DAMAGE FREQUENCY
CEP	CONTAINMENT EXHAUST/PURGE SYSTEM
CET	CONTAINMENT EVENT TREE
CFAR	COMPONENT FAILURE ANALYSIS REPORT
CHR	CONTAINMENT HEAT REMOVAL
CIA	PRIMARY CONTAINMENT INSTRUMENT AIR
CN	CONTAINMENT NITROGEN SYSTEM
COND	MAIN CONDENSATE SYSTEM
CPI	CONTAINMENT PERFORMANCE IMPROVEMENT
CRD	CONTROL ROD DRIVE SYSTEM
CST	CONDENSATE STORAGE TANK
CSTS	CONDENSATE STORAGE AND TRANSFER SYSTEM
CW	CIRCULATING WATER SYSTEM
DCH	DIRECT CONTAINMENT HEATING
DEH	DIGITAL ELECTRO-HYDRAULIC CONTROL SYSTEM
DHR	DECAY HEAT REMOVAL
ECCS	EMERGENCY CORE COOLING SYSTEM
ECOM	ERROR OF COMMISSION
EDG	EMERGENCY DIESEL GENERATOR
EDR	EQUIPMENT DRAIN PROCESSING
EOM	ERROR OF OMISSION
EOP	EMERGENCY OPERATING PROCEDURE
EPG	EMERGENCY PROCEDURE GUIDELINES
EPRI	ELECTRIC POWER RESEARCH INSTITUTE
ESF	ENGINEERED SAFETY FEATURE
FCI	FUEL-COOLANT INTERACTIONS
FDR	FLOOR DRAIN PROCESSING
FP	FIRE PROTECTION SYSTEM
FSAR	FINAL SAFETY ANALYSIS REPORT
FVI	FUSSEL-VESSELLY IMPORTANCE
GSI	GENERIC SAFETY ISSUES

TABLE OF ACROYNMS (Cont'd)

HPCS	HIGH PRESSURE CORE SPRAY SYSTEM
HPME	HIGH PRESSURE MELT EJECTION
HRA	HUMAN RELIABILITY ANALYSIS
HVAC	HEATING, VENTILATION, AND AIR CONDITIONING
IDCOR	INDUSTRY DEGRADED CORE RULEMAKING GROUP
INPO	INSTITUTE OF NUCLEAR POWER OPERATIONS
IORV	INADVERTANTLY OPENED RELIEF VALVE
IPE	INDIVIDUAL PLANT EXAMINATION
IPEEE	INDIVIDUAL PLANT EXAMINATION EXTERNAL EVENTS
IPEM	INDIVIDUAL PLANT EXAMINATION METHODOLOGY
IPEP	INDIVIDUAL PLANT EXAMINATION PARTNERSHIP
IREP	INTERIM RELIABILITY EVALUATION PROGRAM
ISLOCA	INTERFACING SYSTEM LOSS OF COOLANT ACCIDENT
LER	LICENSEE EVENT REPORT
LFMG	LOW FREQUENCY MOTOR GENERATORS
LOCA	LOSS OF COOLANT ACCIDENT
LOOP	LOSS OF OFFSITE POWER
LPCI	LOW PRESSURE COOLANT INJECTION
LPCS	LOW PRESSURE CORE SPRAY SYSTEM
LPME	LOW PRESSURE MELT EJECTION
MCC	MOTOR CONTROL CENTER
MCCI	MOLTEN CORE-CONCRETE INTERACTIONS
MOV	MOTOR OPERATED VALVE
MSIV	MAIN STEAM ISOLATION VALVE
NCR	NONCONFORMANCE REPORT
NEI	NATIONAL ENERGY INSTITUTE
NPRDS	NUCLEAR PLANT RELIABILITY DATA SYSTEMS
NPSH	NET POSITIVE SUCTION HEAD
NRC	NUCLEAR REGULATORY COMMISSION
NREP	NATIONAL RELIABILITY EVALUATION PROGRAM
NSAC	NUCLEAR SAFETY ANALYSIS CENTER
NS4	NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM
NUMARC	NUCLEAR MANAGEMENT AND RESOURCES COUNCIL
PCS	POWER CONVERSION SYSTEM
PDS	PLANT DAMAGE STATE
PP	POWER PANEL
PPM	PLANT PROCEDURES MANUAL
PRA	PROBABALISTIC RISK ASSESSMENT
RCC	CLOSED COOLING WATER SYSTEM
RCIC	REACTOR CORE ISOLATION COOLING
RFW	REACTOR FEEDWATER SYSTEM
RHR	RESIDUAL HEAT REMOVAL SYSTEM
RHVAC	REACTOR BUILDING EMERGENCY COOLING SYSTEM
RPS	REACTOR PROTECTION SYSTEM

TABLE OF ACROYNMS (Cont'd)

RPV	REACTOR PRESSURE VESSEL
RRC	REACTOR RECIRCULATION COOLING SYSTEM
RRW	RISK REDUCTION WORTH
RWCU	REACTOR WATER CLEANUP SYSTEM
SBO	STATION BLACKOUT
SCRAM	SAFETY CONTROL ROD AXE MAN
SDV	SCRAM DISCHARGE VOLUME
SGT	STANDBY GAS TREATMENT
SJAE	STEAM JET AIR EJECTOR
SLC	STANDBY LIQUID CONTROL SYSTEM
SMS	SCHEDULED MAINTENANCE SYSTEM
SORV	STUCK OPEN RELIEF VALVE
SOV	SOLENOID OPERATED VALVE
SP	SUPPRESSION POOL
SPC	SUPPRESSION POOL COOLING
SRV	SAFETY RELIEF VALVE
STG	SOURCE TERM GROUP
SW	STANDBY SERVICE WATER SYSTEM
TAF	TOP OF ACTIVE FUEL
TMI-2	THREE MILE ISLAND UNIT 2
TSW	TURBINE BUILDING SERVICE WATER
USI	UNRESOLVED SAFETY ISSUES
WNP-2	WPPSS NUCLEAR PROJECT 2
WPPSS	WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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1.0 EXECUTIVE SUMMARY

1.1 Background and Objectives

This report presents the Individual Plant Examination (IPE) information for Washington Nuclear Plant-2 (WNP-2) requested by the NRC in Generic Letter 88-20 and Generic Letter 88-20, Supplement 1 (NUREG-1335). Its content and format are prepared in conformance to NUREG-1335. WNP-2 is a General Electric BWR 5 boiling water reactor with a Mark II containment operated by the Washington Public Power Supply System (The Supply System). It is located on the DOE Hanford Site near Richland, Washington.

The WNP-2 IPE represents a systematic evaluation using conventional probabilistic risk assessment (PRA) methodology to identify potential vulnerabilities to severe accidents. In performing this plant specific examination the objectives stated in the generic letter were achieved, namely:

- to develop an appreciation of severe accident behavior,
- to understand the most likely severe accident sequences that could occur at the plant,
- to gain a more quantitative understanding of the overall probabilities of core damage and fission product releases and,
- if necessary, to reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

In addition, evaluations per Generic Letter 88-20, Supplement 3 which requested incorporation of the Containment Performance Issues, have been completed as part of the Level 2 PRA and are reported herein as part of the IPE. In particular, the need for a containment vent or filtered vent in addition to the existing hardware is evaluated. Containment performance and phenomenological issues which are not well understood were generally resolved from information derived from referenced research activities and experiments performed under NRC or industry sponsorship.

During the original performance of the IPE for WNP-2, the Supply System maximized its use of in-house personnel. Experts from outside the corporation were used to train Supply System staff in the general application of PRA techniques and in the specific application of particular topics, such as human reliability analysis, and for several levels of technical review of the IPE/PRA output products. Supply System personnel performed the following tasks:

- data collection, in which system and design engineers who were familiar with the design and system functions were used as the principal investigators,
- system reliability modelling and quantification of plant models,

- thermal-hydraulic system analysis to confirm system success criteria, system and containment response during important accident sequences and to characterize radiological source terms,

- technical review of the analysis, and

- formal in-house independent review to validate the IPE process and results.

The most recently performed risk assessment, Revision 1 to the IPE, incorporates changes in plant design and procedures that have occurred since the design was originally defined for the IPE on the basis of a 1989 "freeze date." The current revision uses plant data and system design as it was at the end of 1993 except as specifically otherwise indicated. In addition, Revision 1 to the IPE also reflects enhancements which better describe the plant's response to issues of SRV reliability, Loss of Offsite Power and plant specific initiators. The process used for preparation and review of Revision 1 of the IPE was similar to that used during performance of the original IPE and, once again, the Supply System maximized its use of in-house personnel to perform and review the analyses. This was particularly true in the collection of up-to-date plant operating data and in the identification of design and procedure changes which have occurred since 1989.

Supply System staff performed the thermal-hydraulic analyses needed to refine system success criteria, to better understand plant response during severe accident sequences and to gain an enhanced understanding of containment response and any associated radiological source terms. Supply System staff reviewed the analysis and played a dominant role in applying the latest revision of the IPE to support resolution of USI/GSIs, and to recommend cost effective measures for changes to the plant where they can be demonstrated as being justified in the prevention or mitigation of severe accidents. Consultants (NUS Corp, Tenera LP) were used during the performance of Revision 1 to the IPE to provide a new perspective, remove unnecessary conservatism wherever possible, to increase the fidelity of the plant models and to add depth to the peer review process.

The effects of these analytical refinements can be seen in the results from the latest revision to the WNP-2 IPE in which the core damage frequency (CDF) is predicted to be 1.75E-05 per year as compared to 5.42E-05 per year predicted in the original study. This result reflects the effects of more realistic modelling of the constraints imposed upon maintenance of important equipment by the Plant Technical Specifications, sometimes referred to as "disallowed maintenance." Additional modelling changes which were incorporated to make the IPE a more faithful representation of WNP-2, include:

- a refined offsite power system model,

- a more rational method and data for treatment of the common cause failure of 18 SRVs to "open on demand."

In the Level 2 analysis, best estimate assumptions were used wherever possible, although whenever there appeared to be two or more possible interpretations of the modelling needs, the most conservative option was routinely selected. The current analysis predicts an annual radionuclide release frequency of 1.07E-5 per year, a value which corresponds to an average conditional failure probability of 0.61 for the WNP-2 containment.

1.2 Plant Familiarization

The Washington Nuclear Plant 2 (WNP-2) is a boiling water reactor of General Electric BWR 5 design. The reactor rated thermal power is 3323 MWt with a design electrical output of 1154 MWe. The Construction Permit was granted in March of 1973 and the plant began commercial operation in December 1984.

The reactor is housed in a free standing steel primary containment of a Mark II design. The free standing steel design is unique for domestic Mark II containments. The steel construction results in a decreased propensity for catastrophic failure modes, but has a lower ultimate pressure than a containment with a steel lined concrete design. With this design, the drywell which houses the reactor vessel is placed over the wetwell volume. The drywell to wetwell connection is through 99 downcomers and the drywell concrete floor is sealed to containment at its periphery with a metal Omega seal. The reactor sits on a pedestal that is recessed below the drywell floor area. The majority of the safety related equipment is housed in the secondary containment, also referred to as the reactor building. Subatmospheric pressure is maintained within this building to prevent the leakage of radioisotopes.

The plant design includes redundant and diverse systems, most of which have automatic initiation, to maintain all critical safety functions when the WNP-2 is being brought to a safe shutdown condition. These systems, which are very important to the IPE, include:

- reactivity control by CRD/RPS, backed up by ARI and SLC,
- reactor pressure control by 18 SRVs, which have a combined energy relieving capacity of greater than 100% normal reactor power,
- three high pressure coolant injection systems:

  RFW, HPCS, and RCIC, (CRD flow is available but has insufficient flow capacity to justify its being credited for core cooling by this analysis),
- automatic depressurization, with redundant initiation logic and seven ADS valves,
- low pressure coolant injection systems: COND, LPCS, LPCI, SW-Crosstie, and Fire Protection (FP) Water,

- three systems for containment heat removal: RHR, Power Conversion System (PCS), and CEP (Venting).

  NOTE: The term PCS is intended to represent the secondary system heat removal function provided by the feedwater, condensate and condenser systems, and their associated support systems.

WNP-2 has two independent offsite power sources and two independent divisions of emergency AC power which are capable of providing all of the electrical power needed to bring the plant to safe shutdown condition. A third independent source of emergency AC power is available to power one division of high pressure injection and a third offsite power source can be made available within several hours if manual hookup and initiation is successful.

Revision 4 of the BWR Owners Group EPG was used to develop WNP-2's symptom-based emergency operating procedures and WNP-2's operator training program is accredited by INPO.

The containment strength analysis performed for the IPE showed that containment failure is not expected to occur below 121 psig at 340°F, and that when failure does occur it is equally likely to result in membrane tears at any one of three more likely locations, two in the drywell, one in the wetwell. The strength analysis also showed that the probability of catastrophic containment failure is very small, so during the Level 2 analysis, the probability of gross failure was assumed to be no more than 1/100 except in the few accident sequences where pressurization was very rapid and consideration of the dynamic response of the containment became important.

Review of the reactor building layout resulted in the conclusion that consequential failures of the core cooling systems would only occur when containment fails in the wetwell region and releases steam and heat directly to the lower elevations of the reactor building. Failures above the 501 ft elevation (drywell floor level) were assumed to have no impact on operating injection systems.

1.3 Overall Methodology

The WNP-2 IPE was performed in accordance with Supply System procedures which ensure that the analysis follows the general precepts and methods outlined in the PRA procedures as described in NUREG/CR-2300, NUREG/CR-2815, NUREG/CR-4550, and the guidelines presented in Generic Letter 88-20, its supplements and appendices. The IPE is a full scope PRA which uses a Level 1 (Front-End) analysis to predict core-damage frequencies and a Level 2 (Back-End) analysis to predict containment performance characteristics during severe accidents. In the Level 1 analysis, a mission time of 24 hours was assumed for all equipment which plays a role in providing core protection functions after a transient and in the Level 2 analysis, a period of up to 40 hours was used to simulate and monitor predictions of containment performance following the occurrence of an accident sequence initiating event. Sensitivity analyses were performed to assess the relative and absolute importance of individual systems and components, to characterize the uncertainty of the frequency estimates for important sequences, and to identify and prioritize the beneficial, cost effective plant changes which may reduce core damage frequency or the frequency of release of fission products. The phenomenological uncertainties embodied in the Level 2 analysis were held to a minimum by using referenceable assumptions wherever possible. The final results of the containment performance assessment are presented as "best point estimates."

The WNP-2 IPE uses the "large fault tree/small event tree" approach to identify and quantify individual accident sequences. With this approach, functionally descriptive event trees are used to model the possible accident sequences which result in core damage, and initiator-specific containment event trees are used to represent possible containment behavior following a core melt accident. Detailed system fault trees are merged to quantify individual event tree sequences in both the Level 1 and Level 2 analyses, although the Level 2 system fault trees were much simpler. The NUPRA computer code was used to perform the quantification and both the RETRAN and MAAP computer codes were used to characterize plant, containment, and fission product behavior.

The effects of common cause failures are included as basic events in the WNP-2 system fault trees and are quantified by the "beta factor" method. This method provides a numerical estimate for the common cause failure probability which should be assigned to account for the likelihood that the plant would experience simultaneous failures of two or more identical components in a single train or multiple trains in a single system.

To ensure that the role of the operating staff was fully credited by the analysis, pre-accident human actions were explicitly modelled in the detailed system fault trees, and post-accident and recovery actions were linked to the functional headings in the event trees. A plant specific human reliability analysis (HRA) was used to predict pre-accident and post-accident human error probabilities for each of the modelled human actions. This analysis followed the guidelines provided by the "Accident Sequence Evaluation Program Human Reliability Analysis Procedure," NUREG/CR-4772. During performance of the Level 2 analysis, to compensate for the extremely severe stress levels which could be expected after a core melt accident, additional conservativism in the human error probabilities were used.

1.4 Summary of Major Findings

1.4.1 IPE Results

The results from the WNP-2 IPE, Revision 1, predict a mean core damage frequency of 1.75E-5 per year. This annual frequency estimate is comparable to those derived for other BWRs and shown in Table 1.4-1. The sequences which contribute to the WNP-2 core damage frequency are dominated by common cause and human error events, and the analysis failed to identify any vulnerabilities in which a single failure either initiates core damage directly or initiates consequential failures which, in turn, lead to core damage. The dominant accident sequences identified by the WNP-2 IPE are initiated by:

- Loss of offsite power (67%),
- Internal flooding (11%),
- Transients initiators (5%) and,
- Transient and Failure to SCRAM (ATWS) (3%).

Figure 1.4-1 provides a pie chart of the results and Table 1.4-2 provides a list of dominant sequences.

The Level 2 IPE results predict:

- a 39% chance that an accident sequence terminates with the containment intact,
- of the 61% of sequences which result in containment failure, 50% occur at or near the time of vessel failure (early), 50% occur much later in the sequence (late),
- of the late containment failures, 20% have insignificant releases because the release is scrubbed by the suppression pool, and,
- all early releases result in suppression pool bypass.

Figure 1.4-2 provides a pie chart frequency distribution for each of the radiological release categories and Table 1.4-3 provides a summary of the results.

1.4.2 Resolution of Generic Issues

USI A-45, Shutdown Decay Heat Removal (DHR), issue has been examined within the context of the results of the WNP-2 IPE. Loss of decay heat removal is a potentially important contributor to plant risk because its occurrence results in the "TW" accident sequences in which a fission product release occurs shortly after vessel failure. However, the dominant causes for loss of decay heat removal at WNP-2 are human errors associated with failure to initiate either RHR, suppression pool cooling or containment venting, and common cause failures of all available trains. The basic design of the DHR systems is not a contributor in either of these cases, in fact, the multiple decay heat removal systems and the diverse and independent support systems give the WNP-2 decay heat removal a high availability when the analysis only considers independent faults. Even with the inherent conservatisms introduced throughout the analysis, the total contribution from loss of decay heat removal sequences is sufficiently low to preclude the need for further evaluation, per NUREG/CR-1289 Guidelines.

USI A-17, Systems Interactions, issue has also been examined within the context and results of the IPE for internal events. The insights and results from this analysis lead to the conclusion that there are no vulnerabilities associated with system interactions. Similarly, because the predicted intersystem LOCA frequency for WNP-2 is negligible, it provides a basis for resolution of GSI 105, InterSystem LOCA.

1.4.3 Conclusions and Recommendations

The critical examination of WNP-2 systems and operations which was an integral part of the IPE identified the sequences which are documented in compliance with the guidelines of Generic Letter 88-20, and provided the basis for the search for potential plant vulnerabilities or outliers which could unduly influence core damage frequency or containment failure probability.

Several sensitivity analyses were performed for the Level 1 analysis to assess the potential reduction in core damage frequency which could be achieved from improvement in the reliability of the offsite power sources, from improvement of the onsite AC and DC distribution systems, and the extent to which common cause failures and human errors contribute to core damage frequency. Level 2 sensitivity analyses were performed to measure the effects that individual assumptions made during CET development and quantification may have on the overall Level 2 results. Specifically, the phenomenological effects of corium attack on the containment shell, assumptions about containment bypass, and the importance of individual human actions were studied.

The performance of the Level 1 and 2 IPEs and the sensitivity studies provided insights to assist in reaching conclusions regarding:

- no important vulnerabilities
- resolution of generic issues
- recommendations for procedural or hardware modifications
- resolving severe accident management issues

The results from the Level 1 and Level 2 analyses show that no further corrective actions are required, beyond the need to review the dominant accident sequences within the context of the sequence evaluation criteria provided by NEI (formerly NUMARC) in NUMARC 91-04. However, in an attempt to exploit the insights and benefits from performing the IPE, several specific recommendations have been made. Disposition of these recommendations will be achieved through the normal Supply System decision making and management processes. In brief, these recommendations are for plant engineering and operations staff to consider implementation of the following changes after they have been subject to comprehensive evaluation and have been shown to be cost-effective and important in reducing plant risk:

- Modify the isolated phase buses to allow expeditious alignment of the 500Kv highline to the plant AC distribution system via the main step-up transformer, following a loss of both the normal 230Kv and 115Kv offsite power sources.

- Increase the capacity of the 230Kv/115Kv plant bus transfer to maximize the effective availability of the 230Kv off-site power source.

- Install an additional battery charger which can both provide an additional source of charging to either DC bus and reduce the potential for common cause failure of the battery chargers.

- Enhance the existing piping inspection program to minimize the occurrence frequency for the important flooding events identified by the IPE, and implement any procedure changes which improve the operator's ability to detect and isolate these internal flooding events before they become serious enough to initiate an accident.

- Review the results from industry research on providing defense against common cause failures and confirm that existing WNP-2 operational and maintenance practices take full advantage of the insights which are currently available.

TABLE 1.4-1

Comparison of PRA Results (Internal Events)

Comparison of Internal Events PRA Results for BWRs	
Plant PRA	Mean Annual CDF (per year)
Shoreham	5.5E-5
Nine Mile Point, Unit 2	3.1E-5
Washington Nuclear Plant-2	**1.75E-5**
Peach Bottom, Unit 2	4.5E-6
Grand Gulf, Unit 1	4.4E-6
Mean Value, 20 BWRs	2.4E-5

TABLE 1.4-2

WNP-2 IPE Dominant Sequences

BRIEF SEQUENCE DESCRIPTION	FREQUENCY (per year)	% OF CDF
Station Blackout with HPCS failure and failure to recover offsite power in four hours	4.51E-06	25.8
Station Blackout with HPCS operating but failure to recover offsite power in ten hours	3.51E-06	20.1
Station Blackout with HPCS, RCIC failure and failure to recover offsite power in thirty minutes	2.71E-06	15.5
TSW flood that inops RHR and a failure of containment venting	9.99E-07	5.7
Loss of Offsite Power with DG 1 or 2 available, HPCS available, with loss of RHR cooling and failure to recover offsite power in ten hours	9.33E-07	5.3

TABLE 1.4-3

WNP-2 IPE Containment Performance Assessment
Summary of Results

Contributing Functional Scenarios	Frequency (per year)	CCFP*	Release Characteristics		Release Frequency (per year)
Long-term Station Blackout	8.02E-6	0.51	Late, small, scrubbed Late, small, unscrubbed Early, small, unscrubbed	(STG3) (STG5) (STG9)	4.76E-7 2.26E-6 1.33E-6
Short-term Station Blackout	2.71E-6	0.33	Late, small, scrubbed Late, small, unscrubbed Early, small, unscrubbed	(STG3) (STG5) (STG9)	3.75E-7 4.99E-7 2.05E-8
Station Blackout "Look Alike"	9.93E-7	0.53	Late, small, scrubbed Late, small, unscrubbed Early, small, unscrubbed	(STG3) (STG5) (STG9)	8.39E-8 2.68E-7 1.70E-7
Internal Flooding	3.16E-6	0.89	Late, small, unscrubbed Early, large, unscrubbed Very early, small, unscrubbed	(STG5) (STG10) (STG13P)	1.15E-6 9.53E-7 7.23E-7
Loss of DHR/CHR	1.37E-6	1.00	Very early, small, unscrubbed	(STG13P)	1.37E-6

* CCFP = Conditional Containment Failure Probability

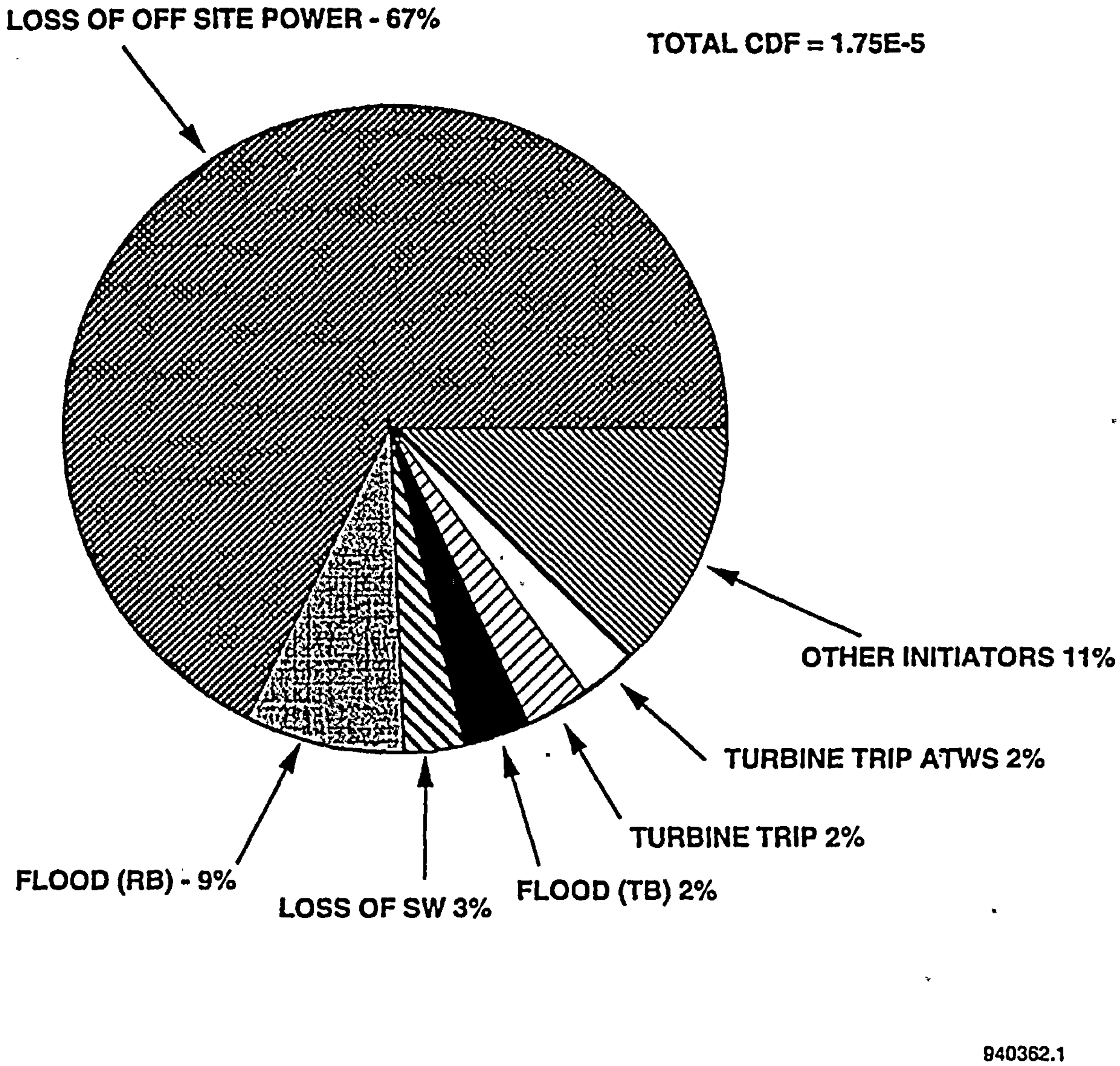

FIGURE 1.4-1 INITIATORS CONTRIBUTION TO CDF

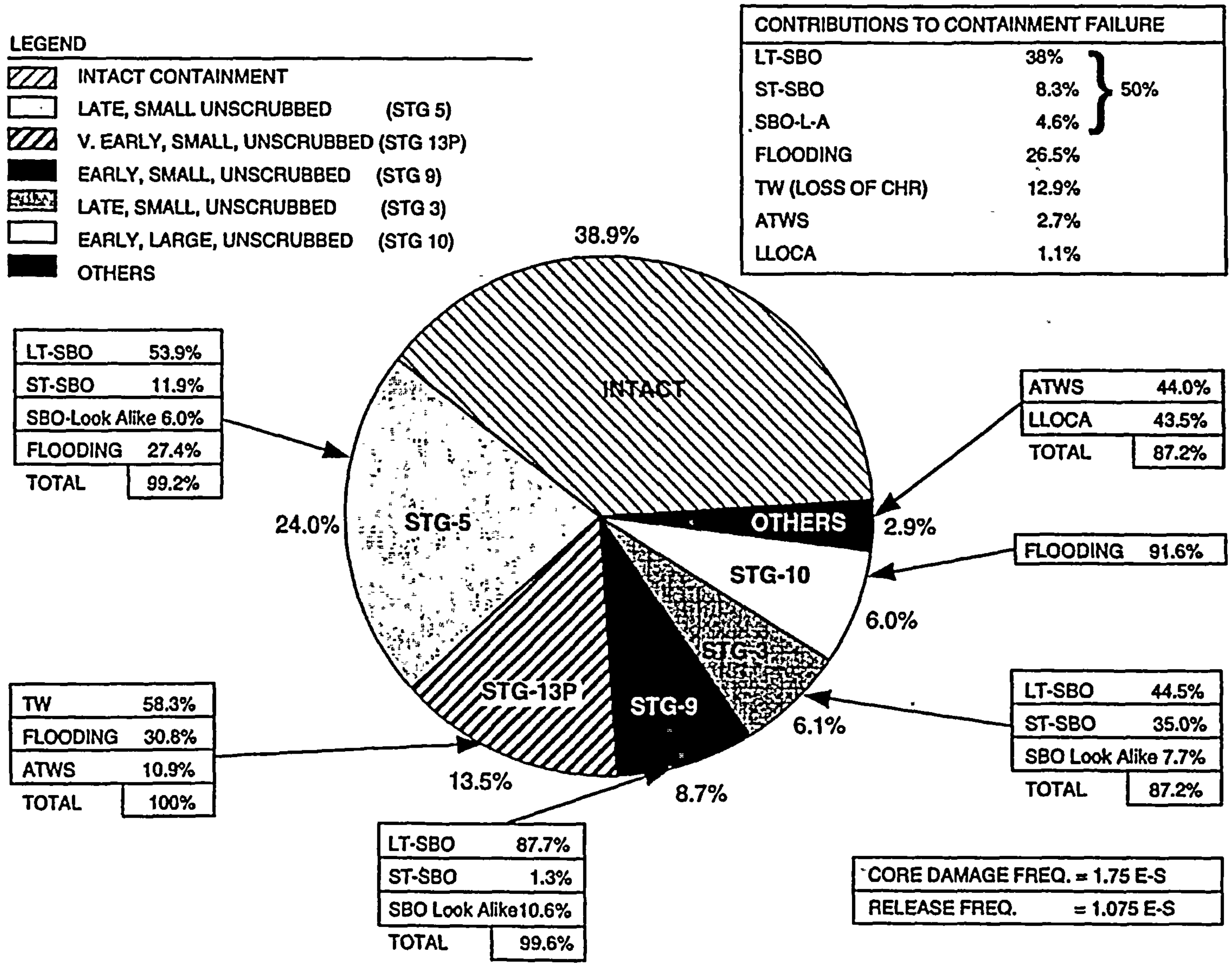

FIGURE 1.4-2 CONTRIBUTORS TO CONTAINMENT FAILURE AND RELEASE

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2.0 EXAMINATION DESCRIPTION

2.1 Introduction

In response to NRC Generic Letter No. 88-20, the Supply System has performed an Individual Plant Examination of WNP-2 for possible severe accident vulnerabilities. Consistent with the general purpose of the generic letter, in preparation of Revision 0 the Supply System has utilized exclusively in-house technical staff from various departments (Safety Analysis, Plant Technical, Engineering, Operations, and Licensing) to perform all aspects of the IPE work. System analyses in support of the IPE were treated as safety related analyses. As such they were performed, reviewed and approved in accordance with the approved in-house Engineering procedures for performing verified calculations. Consultants (Individual Plant Examination Partnership (IPEP), Principals TENERA and FAI) were utilized for peer review and comments on all aspects of the original IPE to ensure industry acceptable methods were used and a thorough examination performed.

Consultants (NUS Corp) were utilized in the current (Revision 1) revision to the IPE to provide a new perspective, reduce data conservatisms, provide more realistic models, and add depth to the peer review process. The modifications in the IPE from Revision 0 to Revision 1 can be categorized as:

- Reduction in Conservatisms - the application of common cause methodology in Revision 0 was overly conservative for multiple components such as SRVs, MSIVs, and circuit breakers. The reduction in CDF by improving this modeling was approximately 50%.

- Revision to Loss of Offsite Power - this revision includes a new two state model accounting for dependent and independent initiating line loss events and recalculating offsite power recovery factors. Although the sum of sequence frequencies for Loss of Offsite Power is similar between Revision 0 and Revision 1, the dominant sequences have changed and the model robustness improved.

- Model Enhancements - improved the HRA models, included CRD system models, and improved the MAAP analysis of containment response with respect to success criteria for SRVs and MSIVs. These enhancements had very mild impact on Revision 0 CDF but greatly improved model fidelity.

The in-house staff has identified the severe accident sequences, quantified the probabilities of core damage and fission product releases, addressed applicable USI and GSI, and recommended cost effective measures that would help prevent or mitigate severe accidents.

2.2 Conformance with Generic Letter and Supporting Material

This IPE report conforms to the NRC guidance given in Generic Letter 88-20 and its Supplement No. 1 (NUREG-1335). This report's section numbers and titles are the same as given in Table 2.1 of NUREG-1335. The examination method is a Level 1 and Level 2 PRA (consistent with the terminology contained in NUREG/CR-2300). Accident Management Strategies as described in Supplement 2 of the Generic Letter together with plant specific insights developed in this IPE report are being considered for the WNP-2 Severe Accident Management Program which is under development and being closely coordinated with the BWROG effort. Supplement 3 (Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the IPE for Severe Accident Vulnerabilities) has been considered herein. Criteria for selecting important severe accident sequences are in accordance with Appendix 2 of the Generic Letter. Documentation of examination details and results (referred to as Tier 2 documentation herein) are kept in a traceable manner under in-house document control as required by Section 10 of the Generic Letter. This IPE report contains all of the information required by Appendix 4 of the Generic Letter. Since the IPE represents recently completed analysis, this report reflects the current designs and practices at WNP-2 as of December 1993.

2.3 General Methodology

The general methodology used in this report is Level 1 and Level 2 PRA. This methodology and major tasks are as described in NUREG/CR-2300, "PRA PROCEDURES GUIDE" for Level 1 and Level 2 PRA. The major Level 1 tasks include information gathering of plant specific data, P&IDs, test, maintenance, operating procedures, and physical room, containment, and building information. The system analysis methodology utilized the small event tree-large fault tree approach. The event trees combined the initiating event with system functional successes or failures to delineate the accident sequences. The fault trees were developed to the component, relay, and sensor level of detail. The system modelling included human reliability and common cause failure events, utilizing plant specific failure data to the maximum extent possible. The human reliability analysis (HRA) utilized the ASEP HRA methodology. Common cause modeling was performed with the conservative beta factor method utilizing data primarily from NUREG/CR-4780. The event trees were quantified using the NUPRA computer code. The Level 2 tasks include grouping the Level 1 sequences into several bins or plant damage states, characterizing containment failure modes and locations, developing and quantifying logic trees and containment event trees, determining the magnitude of the radionuclide release, and performing sensitivity studies. The Modular Accident Analysis Program (MAAP), Revisions 7.02 and 7.03 were used to provide an integrated approach for modelling of plant thermal hydraulic response and fission product transport during severe accidents. In addition, research results in the open literature, IDCOR task reports, the Shoreham and Limerick PRAs, NUREGs, and engineering judgement were used in understanding accident progression and quantifying event trees. The

Supply System developed very detailed fault trees taking into account component failures, initiation and control failures (including logics and interlocks), support system failures, test and maintenance unavailabilities, operator errors, and common cause failures. Fault trees were developed down to the component, relay, and sensor levels beyond which component failure rate data do not exist.

The Level 1 analysis is coupled with the Level 2 analysis through the binning of the multitude of the Level 1 sequences into 19 groups of plant damage states with similar Level 2 characteristics. Results of a structural analysis performed for WNP-2 are used to assess containment strength, failure size and location. MAAP calculations, analytical models, and widely accepted research results are employed in the accident progression analysis. Based on this information, containment event trees and logic trees were developed to provide a description of the containment damage states. The containment event trees were quantified using the NUPRA code. The end states of the containment event trees were then grouped into a limited but complete set of unique release categories. For each category, a representative MAAP calculation was performed to provide estimate of the fission product release to the environment. Finally, a sensitivity analysis was performed to investigate important parameters that could have large impact on the likelihood or time of containment failure and the magnitude of the source term. The results were used to identify the areas for which potential improvements of the plant might be considered.

The Supply System developed several engineering standards used in developing the event and fault trees and quantifying the results. The convention for defining basic faults conforms to the WNP-2 Master Equipment List which has an unique ID for each component. The standards, together with configuration control of models and information, provided system boundaries, level of detail, compatibility between systems, and compatibility with event trees for developing fault trees. Because of the consistency throughout the process and the linking and merging capabilities of the NUPRA code, dependencies, common cause effects, and system interaction were fully accounted for by the IPE.

NUPRA, a PC-based program developed by NUS Corporation, was used to quantify the Level 1 and Level 2 accident sequences. It is an integrated program in that the databases, fault trees, event trees, quantification routines, sensitivity/uncertainty operations, and printing and plotting capabilities are all in one program. The fault tree linking approach is used to generate minimal cutset equations for various fault trees and accident sequences. NUPRA was verified and validated in accordance with the following NUS procedures:

CD-OP 5.1: "Computer Software Development Documentation, and Control"

CD-QAP 3.5: "Documentation, Verification, and Control of Software Programs"

It is the policy of the NUS Corporation to perform work related to nuclear power plant safety to the requirements of Title 10 of the Code of Federal Regulations, Part 50, Appendix B. After NUPRA was installed on in-house PCs, NUPRA fault tree and sequence results were checked against CAFTA and SETS results respectively. They were found to be identical. NUPRA has been verified and validated in accordance with the Supply System's computer code V&V procedure.

2.4 Information Assembly

Twenty-nine system notebooks were developed for the original WNP-2 IPE. An engineering instruction was developed and followed in the preparation, review, and approval of the system notebooks. Each notebook has the following sections:

1.0 Function
2.0 System Description
2.1 Support Systems
2.2 Instrument and Control
2.3 Test and Maintenance
2.4 Technical Specification Limitations
3.0 System Operation
4.0 Performance during Accident Conditions
5.0 Location within the Plant
6.0 Answers to IDCOR Questions
7.0 References

The system notebooks were primarily prepared by the system design engineers from Engineering, who were most familiar with the systems. The following reference materials were used in developing the system notebooks:

1. WNP-2 FSAR
2. WNP-2 System Operating Procedures
3. WNP-2 Surveillance Procedures
4. WNP-2 Maintenance Procedures
5. WNP-2 Technical Specifications
6. Piping and Instrument Diagrams
7. Electrical Wiring Diagrams
8. WNP-2 Systems Training Notebooks

The system notebooks were reviewed by both independent system design engineers from Engineering and the Shift Technical Advisors from the Plant Technical Department. There were a total of 42 people involved in the initial system notebook effort. The fault tree analysis for each system has been combined with the system notebooks and all of the system notebooks are under configuration control requiring updates with system modification. This ensures the IPE represents the As-Built, As Operated WNP-2 Plant.

Appendix D of IDCOR Technical Report 86.3B1 on "IPEM for BWRs" contains engineering insight questions developed for each functional heading of the Individual Plant Evaluation Methodology (IPEM) event trees. All of these insight questions were sorted and grouped under WNP-2 system names. These questions were answered in Section 6 of the system notebook for each system and were consulted by IPE engineers while developing system fault trees. Therefore, the WNP-2 IPE benefitted from the insights of the extensive IDCOR IPEM program to strengthen its PRA approach. The completed PRAs of Shoreham and Limerick were reviewed prior to initiating the WNP-2 IPE. During the IPE effort, NUREG-1150 analysis and, through our consultants, several ongoing PRA results, i.e., Monticello, Dresden, Pilgrim were used for consistency checks.

Twenty-four system fault trees were developed. After completion of a fault tree, the fault tree analyst together with 2 to 3 other people walked down the system. The walkdown team makeup was usually as follows:

- Fault Tree Analyst
- Walkdown Coordinator (former shift manager)
- System Engineer from Engineering
- Health Physics Technician

The purpose of the walkdown was to ensure the following:

1. The as-built system is consistent with the flow diagram used in the fault tree development.

2. The system lineup during normal operation is consistent with assumptions made in the fault tree development and described in the system notebook.

3. All local vulnerabilities (high room temperature, humidity, etc.) are accounted for in the fault tree development.

4. All support systems are accounted for in the fault tree development.

Walkdowns were performed in accordance with an Engineering standard, "IPE System Walkdown," developed specifically for the IPE effort. Walkdown Checklists that were filled out during the walkdown are part of the second level of documentation retained at the Supply System.

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3.0 FRONT END ANALYSIS

Section 3.0 contains the Level 1 PRA portion of the WNP-2 IPE. It is divided into four major sections as follows.

Section 3.1, Accident Sequence Delineation, includes the initiating event analysis, success criteria discussion, and the development of the event trees. Minor tabs are provided with event tree acroynms for ease of use.

Section 3.2, Systems Analysis, includes the WNP-2 physical system information and the system interdependencies. Each individual system description is minor tabbed for ease of use.

Section 3.3, Sequence Quantification, provides the quantification of the event trees, including fault tree linking. The basic event data utilized is characterized and provided in this section. The details of the Human Reliability Analysis and Common Cause Failure Analysis are described. The event tree functional equation solutions are presented and all sequences with a frequency greater than the quantification cutoff frequency of 1E-10 are presented.

Section 3.4, Results and Screening Process, summarizes the reportable (per Generic Letter 88-20) sequences and describes the major contributors. In this section, the rational for resolution of generic licensing issues (including Decay Heat Removal, A-45) are presented. Discussion of the results with respect to the screening criteria and with respect to the sensitivity analyses performed complete the Level 1 analysis.

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3.1 Accident Sequence Delineation

3.1.1 Initiating Events and System Success Criteria

3.1.1.1 Initiating Events

An initiating event is an event that disrupts normal plant operation and requires either automatic or manual reactor SCRAM. The initiating events considered in this report are plant-related internal events including internal flooding. External events, such as seismic events, tornadoes, and volcanoes, as well as internal fires will be examined in the IPEEE.

All of the initiating events considered in the Individual Plant Evaluation Methodology (IPEM) for Boiling Water Reactors, IDCOR Technical Report 86.3B1, were considered in this IPE. To identify the initiating events, a detailed review of the following information was used by IDCOR:

- Previous risk analyses (e.g., WASH-1400, Limerick PRA, Zion PRA, RSSMAP, and IREP were used to identify unusual initiators such as the loss of service water).
- Licensing basis accidents were reviewed.
- Potential common cause precursor events from the LER data base were examined to identify initiators such as internal flooding, loss of service water, etc.

The initiating events derived for the IPEM were as follows:

- Anticipated transients
- Manual shutdowns
- LOCAs
- Initiators with a failure to SCRAM
- Reactor pressure vessel rupture
- Other accident initiators
  - Loss of DC power
  - Internal flood
  - Reactor water level measurement anomalies
  - Loss of service water
  - Loss of AC bus

WNP-2 went into commercial operation in December 1984. A complete stress report on the reactor vessel has been prepared in accordance with ASME requirements. The stress analysis performed for the reactor vessel assembly (including faulted conditions) were completed using elastic methods. Therefore, catastrophic reactor vessel rupture was not considered in this IPE. If the reactor were to fail, it would most likely leak before failing. Such an accident initiator would be similar to that of a small LOCA. To complete the list of support system failures, loss of Containment Instrument Air (CIA) and loss of Control and Service Air system (CAS) were considered as initiating events in this report. The complete list of initiating events addressed in this analysis is shown in Table 3.1.1-3.

3.1.1.2 Initiating Event Frequencies

General Transients

EPRI NP-801 classifies BWR transients into 37 types such as electric load rejection, turbine trip, and recirculation pump trip. The transients belonging to General Transients, can be grouped on the basis of similarities into the following seven classes:

T_1 - MSIV closure
T_2 - Turbine trip
T_3 - Loss of offsite power
T_4 - Inadvertent opening of a relief valve
T_5 - Loss of feedwater
T_6 - Loss of condenser
T_7 - Control rod withdrawal

To this list of 7 classes, a manual shutdown class was added. Manual shutdown is defined as a planned or scheduled shutdown that is not required by Technical Specifications. An event tree was added to address these planned manual shutdowns and its plant specific initiating data was separated from the SCRAM data. Manual shutdowns required by Technical Specifications are included with the turbine trip events. Additionally, in this analysis, control rod withdrawal was considered to be part of the turbine trip events.

The INPO LER data base and SCRAM reports were reviewed to identify all reactor SCRAMs for WNP-2 in reactor modes 1, 2 and 3 beginning with the first year of commercial operation, i.e., beginning December 13, 1984, through December 31, 1993. A review was performed of the Nonconformance Reports (NCRs) and control room logs to further ensure that the data base for WNP-2 Group 1 accident initiators was complete. The data were divided into four categories: the first year of operation at less than or equal to 25% power; the first year of operation at greater than 25% power; after the first year of operation at less than or equal to 25% power; and after the first year of operation at greater than 25% power.

The WNP-2 turbine bypass system has 25% rated steam flow capacity. As discussed in Section 3.1.2.3, an ATWS event at 25% power or below does not contribute as much to the core damage frequency as an ATWS event occurring at greater than 25% power. The data are therefore separated along the 25% power line to facilitate analysis of both low power and high power ATWS events. Table 3.1.1-1 contains a summary of the number of transient events identified. For determining plant-specific initiating event frequencies, the data from the first year of operation were ignored since they are not considered to be representative.

Table 3.1.1-2 shows the total number of manual shutdowns and SCRAMs for the years 1985 through 1993. The numbers of SCRAMs are plotted in Figure 3.1.1-1 against year. As can be seen from the graph, the number of SCRAMs per year has been trending downward. In 1985 there were 21 SCRAMs, and in 1991 there were only four. Excluding the data from the first year of operation, the total number of SCRAMs (including manual shutdowns) occurring between 1986 and 1993 was 52. Therefore, the average number of events during this period is 52/8 = 6.5. It is assumed that WNP-2 will continue to experience on average 4 SCRAMs per year based on current trends. The assumption of 4 SCRAMs per year is considered conservative based on plant trends as depicted in Figure 3.1.1-1 which shows 4 SCRAMs is the maximum over the last four years of operations. By assuming that 4 SCRAMs per year will occur at WNP-2, the plant-specific initiator frequencies calculated from the data in Table 3.1.1-1 are multiplied by 4/6.5 = 0.6. Sensitivity studies regarding this assumption are included in Section 3.4.5.

Other Initiators

For relatively low frequency initiators which have not occurred at WNP-2, generic data, component failure rates or system fault trees were used. These initiators consist of: loss of offsite power, internal flooding, LOCAs, instrument line break, and loss of DC, TSW, CIA, and CAS. ATWS initiators have a plant-specific and generic component. Table 3.1.1-3 shows the initiating event frequencies that were used for the WNP-2 IPE. Descriptions of how the plant-specific initiating event frequencies were obtained are provided in the corresponding event tree discussions developed in Section 3.1.2.

3.1.1.3 Success Criteria

Coolant Injection

System success criteria are defined as the minimum set of components, trains and/or operator actions required to successfully fulfill a system's function in bringing the reactor to a safe shutdown state. Table 3.1.1-4, taken from GE NEDC-30936P, shows the system success criteria used for these functions. The criteria used for determining system flow rate success are based on the LOCA criteria of maintaining cladding temperature below 2200°F, and containment pressure and temperature below 45 psig and 340°F, respectively. Additional success criteria used are presented below.

SLC Initiation

Anticipated-transient-without-SCRAM (ATWS) events can be classified into those with the condenser available and those without the condenser available. For an ATWS with the condenser available, the operator has 40 minutes from ATWS initiation to initiate SLC injection (86 gpm). For an ATWS without the condenser being available, the operator has 20 minutes from ATWS initiation to initiate SLC injection. The time criteria are based on maintaining reactor power to an acceptable level to prevent core damage or containment failure.

Pressure Control

Safety Relief Valves must open to control reactor pressure to less than the ASME design limit of 1375 psig. For turbine trip and loss of feedwater events where MSIVs are open, it was assumed that 7 out of 18 SRVs must open. For MSIV closure, loss of Condenser, and loss of offsite power events where MSIVs are closed, it was assumed that 14 out of 18 SRVs must open. For any type of ATWS event, it was assumed that 17 out of 18 SRVs must open.

Containment Pressure Constraints on System Operation

To maintain adequate valve opening forces for the inboard MSIVs and SRVs, it is necessary to maintain defined minimum pressure differentials between the nitrogen providing the motive power to the actuators and the containment environment. This implies that containment pressure:

- must be less than 54 psig to ensure that the MSIVs remain open
- must be less than 62 psig to ensure that the SRVs remain open

The first criterion is important in determining whether the PCS can be made available as a heat sink for containment heat removal, whereas the second is important to the low pressure sequences in which RHR is lost. In these sequences, an increase in containment pressure beyond 62 psig will result in reclosure of the SRVs, RCS repressurization and a loss of low pressure injection.

Ultimate Containment Failure

Beyond the design basis values of 45 psig and 340°F, the containment will fail due to overpressurization and/or over temperature conditions as follows:

COMPONENT	INITIAL TEARING PRESSURE AT 340°F	PRESSURE FOR 28 IN^2 LEAK AT 340°F	INITIAL TEARING PRESSURE AT 600°F	PRESSURE FOR 28 IN^2 LEAK AT 600°F
Equipment Hatch	105 psig	121 psig	88 psig	103 psig
Wetwell above stiff'nrs	121 psig	121 psig	103 psig	103 psig
Drywell upper cone	122 psig	122 psig	104 psig	104 psig

TABLE 3.1.1-1

Summary of Group 1 Transient Events
Between December 13, 1984 and December 31, 1993

Major Category	MSIV Closure	Turbine	Total LOOP	IORV	LOF	LOC	CRW	Scheduled Manual Shutdown	Sum
1st Year, <= 25% Power	0	6	0	0	0	1	1	0	8
1st Year, > 25% Power	0	12	0	0	0	0	0	1	13
After 1st Year, <= 25% Power	1	7	0	0	1	0	0	0	9
After 1st Year, > 25% Power	2	32	0	0	2	0	0	7	43
TOTAL	3	57	0	0	3	1	1	8	73

MSIV - Main Steam Isolation Valve
LOOP - Loss of Offsite Power (all sources)
IORV - Inadvertent Opening of Relief Valve
LOF - Loss of Feedwater
LOC - Loss of Condenser
CRW - Control Rod Withdrawal

1st Year - 12-13-84 to 12-12-85

TABLE 3.1.1-2

History of Total Number of Shutdowns

Year	Total Number of Shutdowns
1985	21
1986	9
1987	9
1988	11
1989	5
1990	4
1991	4
1992	5
1993	5
	73

TABLE 3.1.1-3

Initiating Event Frequencies

	Initiation Event	Frequency Events/Year	Source
1	General Transients		
•	Turbine Trip	3.30	WNP-2 Specific
•	MSIV Closure	0.2	WNP-2 Specific
•	Loss of Condenser	0.05	WNP-2 Specific
•	Loss of Feedwater	0.1	WNP-2 Specific
•	Loss of Offsite Power	2.46×10^{-2}	WNP-2 Specific
•	IORV/SORV	0.2[a]	WNP-2 Specific
•	Manual Shutdown	0.5	WNP-2 Specific
2	LOCA		
•	Large LOCA (D > 6″)	3×10^{-4}	NREP EGG-EA-5887[b]
•	Medium LOCA (4″ < D < 6″)	3×10^{-3}	IPEM
•	Small LOCA (1″ < D < 4″)	8×10^{-3}	IPEM
•	Steam Line Break Outside Containment	2.17×10^{-5}	WNP-2 Specific[d]
•	ISLOCA	1.21×10^{-6}	WNP-2 Specific[d]
3	ATWS[c]		
•	Turbine Trip with Bypass (100% Power)	2.7	WNP-2 Specific
•	Turbine Trip with Bypass (25% Power)	.6	WNP-2 Specific
•	MSIV Closure	0.2	WNP-2 Specific
•	Loss of Condenser	0.05	WNP-2 Specific
•	Loss of Feedwater	0.1	WNP-2 Specific
•	SORV	4.0E-3	WNP-2 Specific
4	Special Initiators		
•	Loss of Division 2 DC	3×10^{-3}	IPEM
•	Loss of TSW	1.25×10^{-3}	IPEM
•	Loss of CIA	1.25×10^{-3}	IPEM
•	Loss of CN	1.25×10^{-3}	IPEM
•	Instrument Line Break	1×10^{-2}	IPEM
•	Internal Flooding (Category 6)	2.92×10^{-3}	WNP-2 Specific
•	Internal Flooding (Category 7)	1.60×10^{-5}	WNP-2 Specific
•	Internal Flooding (Category 14)	4.69×10^{-3}	WNP-2 Specific
•	Loss of SW	1.83×10^{-4}	WNP-2 Specific
•	Loss of CAS	1.25×10^{-3}	IPEM

NOTES

a	Frequency includes transfers from other event trees.
b	Per the Brunswick IPE, September 1992.
c	The ATWS event trees use these transient frequencies as initiating events and are followed in the trees by events for failures of the mechanical and electrical portions of RPS.
d	These values are calculated from generic pipe rupture data applied to WNP-2 plant specific piping lengths.

TABLE 3.1.1-4

SUCCESS CRITERIA FOR MITIGATING SYSTEMS

INITIATING EVENT	COOLANT INJECTION	CONTAINMENT HEAT REMOVAL
LARGE LOCA	HPCS (or) 1 LPCS Loop (or) 1 LPCI Pump*	1 RHR
INTERMEDIATE LOCA	1 LPCS Loop + 2 SRVs*** (or) 1 LPCI Pump + 2 SRVs*** (or) 1 Condensate Pump + 2 SRVs (or) HPCS (or) 1 FW Pump** FP Water or SW-Crosstie	1 RHR
SMALL LOCA	1 LPCS Loop + 3 SRVs (or) 1 LPCI Pump + 3 SRVs (or) 1 Condensate Pump + 3 SRVs (or) HPCS (or) RCIC (or) 1 FW Pump (or) FP Water or SW-Crosstie	1 RHR
TRANSIENT/IORV/SORV	Same as Small LOCA	1 RHR or PCS
ATWS	Dependent on Power Level	PCS

* In the long term (> 2 hours) a different combination may be required.

** MSIVs must be manually reopened, if previously closed, for this system to operate.

*** For liquid breaks less than 0.2 ft^2, 3 SRVs are needed.

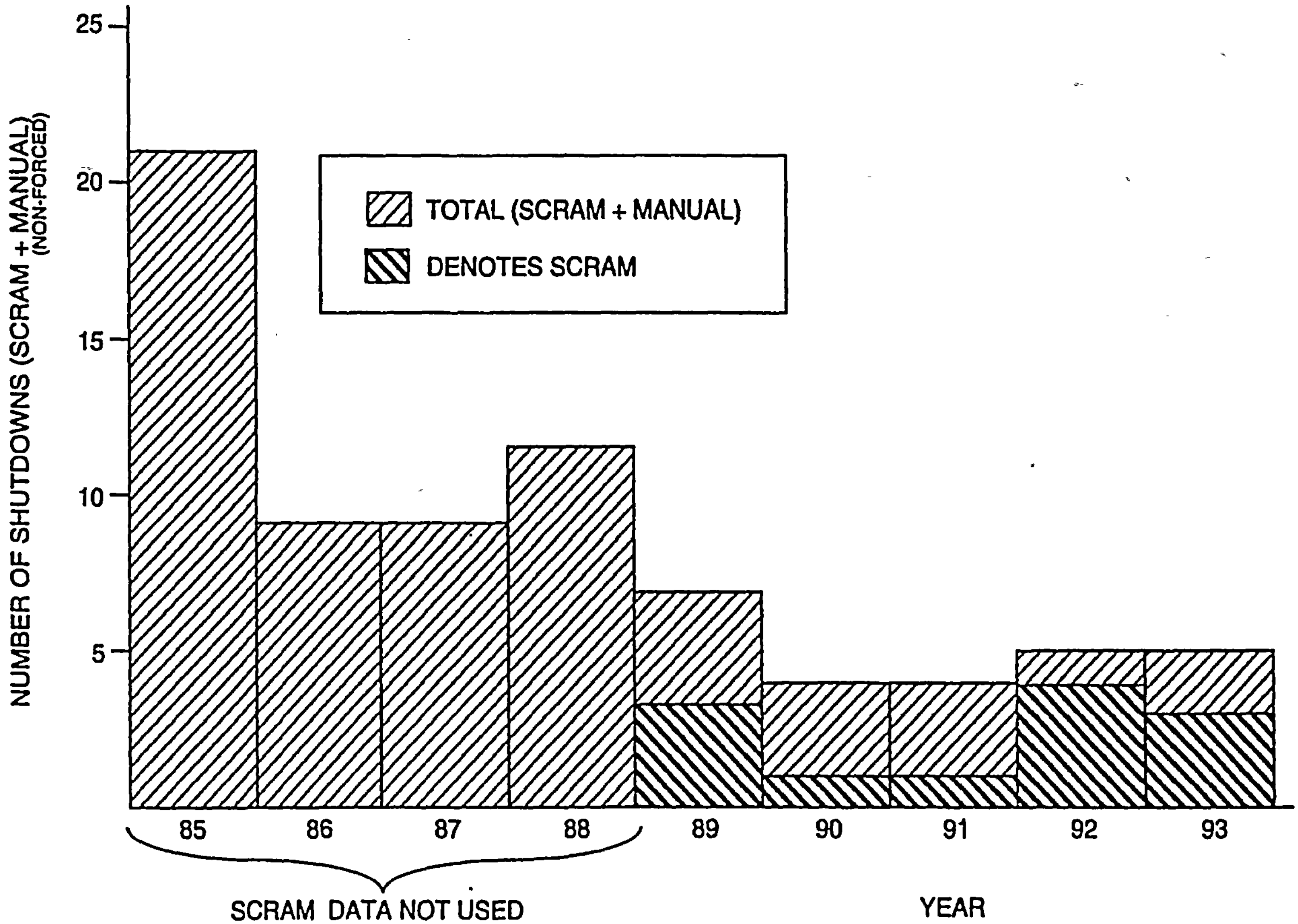

FIGURE 3.1.1-1 HISTORY OF TOTAL NUMBER OF SHUTDOWNS

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3.1.2 Front Line Event Trees

An event tree provides a means of describing all significant accident scenarios or sequences which may result from a specific initiating event. The accident sequences are defined in terms of the success or failure of the mitigating systems modeled in the event trees. The placement of these systems across the tree is based on the approximate timing in which they occur, proceeding from left to right. The probabilities of success or failure of the mitigating systems at the event tree branch points are determined by the use of system fault trees. The up-branches represent success of the respective systems, and the down-branches represent failure. Each fault tree model contains the credible modes of system unavailability identified due to hardware failure, human error, testing and maintenance, and common cause failure (if appropriate). Event trees are developed for the initiating events shown in Table 3.1.1-3, and are discussed in detail in the following subsections.

3.1.2.1 General Transient Event Trees

All plant transients demand control rod insertion, proper operation of the plant heat removal systems, etc. to ensure a safe shutdown. These transients can be grouped under turbine trip, MSIV closure, loss of condenser, etc., because of similar 1) systems available for accident mitigation, 2) initial conditions, 3) pressure, temperature and power. Moreover, Turbine Trip, MSIV closure, Loss of Condenser, Loss of Feedwater, and Manual Shutdown produce similar plant responses. Therefore, the accident sequences and plant end states are the same for the five accident initiators. The same general transient event tree will be used for the 5 accident initiators. However, the system success or failure probabilities at the event tree branch points may be different for the different initiators due to different thermal hydraulic conditions and different impact of each initiator on the availability of vessel makeup and heat removal using the Power Conversion System. The event tree branch point descriptions for the general transient event trees are described below. If they are modified for a specific initiator, that modifications is discussed in the section associated with the initiator event description.

C - Reactor Subcritical

Failure to bring the reactor subcritical is transferred to the appropriate transient ATWS event trees. The sequences assessed in this section are those in which control rods are successfully inserted. The Reactor Protection System is an extremely reliable system. Estimates of its unavailability range from 2×10^{-7}/demand [Reference : Utility Group on ATWS, "Comments of the Utility Group on ATWS," Docket No. PRM-50-29] to values in the range of 3×10^{-5} (NUREG-0460, WASH-1400). WNP-2 has added redundant vent and drain valves and 4 level transmitters to the SCRAM Discharge Volume (SDV) in accordance with BWROG recommendations. In early 1994, a single control rod failed to scram during surveillance testing at WNP-2. The cause of the failure was hardening of seals in the solenoid. Even if this failure mode was common to the other control rod mechanisms, failure to scram was highly unlikely because of diverse back-up scram valves. On-line preventive maintenance for this system is not currently allowed at WNP-2. The failure rate of 1.4×10^{-5}/demand to SCRAM is used for core damage quantification.

M - Safety Relief Valves Open

This functional event represents the opening of the safety relief valves to limit the reactor coolant pressure to within the primary system boundary design pressure (110% system pressure). Failure of a sufficient number of valves to open may lead to excessive pressure and a potential LOCA condition. For turbine trip from rated power without turbine bypass, 14 of the 18 valves are required to open initially to meet the success criteria. For the turbine trip from rated power with turbine bypass, 7 valves are assumed to have to open initially. These valves reclose and some experience subsequent cycling. It is assumed that if the valves open during the first cycle, they will open during the subsequent cycles if required.

The failure of the SRVs to open on high pressure is represented by a single common mode failure event with a value of 5.0E-8. This value was obtained derived from a common mode failure analysis of NPRDS data.

P - Safety Relief Valves Reclose

The safety relief valves that open as a result of the turbine trip must reclose to prevent discharge of an excessive quantity of reactor coolant and excessive heat to the suppression pool. Failure to reclose the safety relief valves is transferred to the IORV/SORV event tree. Based on the WNP-2 plant specific SRV failure rate 1.83×10^{-2}/demand, the probability of failure of seven SRVs to reclose in the first cycle is $7 \times 1.83 \times 10^{-2} = .128$. According to IPEM, IDCOR Technical Report 86.3B1, 85% of the SORVs will reclose when the reactor pressure drops below 200 psig. Therefore, the probability of SRVs failing to reclose is $.128 \times 0.15 = .019$.

Q - MSIVs Open, Condensate, Feedwater and PCS Systems Available

The condensate and feedwater system coupled with the PCS is the normal method of maintaining an adequate coolant inventory in the reactor vessel. The large condenser hotwell water inventory permits limited condensate/feedwater system operation without the immediate operation of the PCS. Since the PCS provides the return loop from the reactor to the condenser hotwell, its eventual operation is required for the hotwell inventory replacement necessary for extended condensate/feedwater system operation.

For the condensate/feedwater/PCS to successfully control inventory and transfer decay heat in the long term, the following systems must be available:

- One condensate, one booster and one feedwater pump must be operating while the reactor pressure is above 470 psig. For reactor pressure below 470 psig, one condensate and one booster pump must operate.
- Steam must be available from the reactor to the corresponding feedwater pump turbine. Two MSIVs (one inboard and one outboard) must be open.
- At least one of the main condenser circulating water pumps must be operable and delivering cooling water to the main condenser.
- At least one of the steam jet air ejectors or a mechanical vacuum pump must be operable and removing noncondensibles from the main condenser.
- Hotwell makeup from CST must be working if required.

Detailed fault trees for the condensate/feedwater/PCS systems are developed in system fault tree notebooks.

U - HPCS(U1) or RCIC(U2) Available

In addition to the feedwater system there are 2 other principal sources of water for maintaining the core inventory at high reactor pressure:

- HPCS has a operating range of 1130 psig to 0 psig. It has a motor driven pump and is designed to start automatically upon receipt of a L2 low water level or high containment pressure signal. It is assumed in this analysis that HPCS will not operate beyond its design temperature limit of 212°F.
- RCIC has a operating range of 1130 psig to 65 psig. It has a turbine driven pump and is designed to start automatically upon receipt of a L2 low water level signal. It is assumed that RCIC will not operate below a reactor to primary containment delta pressure of 65 psi.

Detailed fault trees for the HPCS and the RCIC systems are developed in system fault tree notebooks.

If reactor makeup from HPCS or RCIC is successful, removal of core decay heat can be performed with RHR via the suppression pool, cooldown via the Power Conversion System, or containment venting. If removal of decay heat cannot be performed with the use of these systems, containment failure pressure (121 psig) could be reached after a period of about 29 hours based on MAAP simulations. Breach of containment may cause failure of all injection which would lead to the core being uncovered and damaged. This scenario is discussed in this section in more detail under the W_2 - Containment Venting Available subject heading.

X - Timely Depressurization

In the unlikely event that there is an insufficient supply of coolant from high pressure sources, it would then be necessary for automatic or manual initiation of ADS to reduce reactor pressure below 470 psig to allow low pressure injection systems to maintain reactor inventory. WNP-2 has 7 ADS valves. Success criteria assumes that operation of 3 out of 7 ADS valves is required to depressurize the vessel so that the low pressure systems can be used. They will automatically actuate when the following signals are all present:

- Level 1 and Level 3
- 105 second time delay
- At least one low pressure ECCS pump is running.

The ADS may be actuated manually provided one low pressure ECCS pump is running. Operator failure to depressurize is analyzed in the human reliability analysis.

Since the reliability of depressurization is strongly sequence dependent, the reliabilities associated with automatic and manual depressurization may vary substantially with the sequence. Several items are of particular importance in the evaluated conditional probabilities:

- Battery power must be available to operate ADS solenoids.
- Automatic ADS would be inhibited in the case of a station blackout because of the low pressure pump operating requirement.
- Automatic ADS would occur if low reactor water level signals (L3 and L1) exist, one of the low pressure pumps is operating, Division 1 or 2, and the timer times out at 105 seconds.

- The WNP-2 emergency procedures direct the operator to depressurize if:

  a) Level cannot be maintained above
  -161″ (TAF)
  -192″ (2/3 Core Height)

  b) SP temperature cannot meet heat capacity temperature limit

  c) Drywell temperature cannot be maintained below 340°F

  d) Wetwell pressure and Suppression Pool (SP) level cannot be controlled below pressure suppression pressure limit.

- There is sufficient N_2 to operate the ADS valves. The CIA system supplies nitrogen to the 18 SRVs from the cryogenic nitrogen dewar. In the event cryogenic nitrogen is unavailable, two independent nitrogen bottle bank subsystems can deliver pressurized nitrogen to the 7 ADS valves and accumulators. A remote nitrogen cylinder connection is provided to each subsystem to permit supplementing the cylinder banks through manual connection of additional portable nitrogen cylinders, and thus maintaining pressure for an indefinite time.

- Without suppression pool cooling, the containment pressure may rise sufficiently to eliminate the required 88 psid differential pressure in the ADS pilot valves. The nitrogen supply pressure is required to be 150 psig.

A detailed fault tree for the ADS is developed in its system notebook. If ADS is unsuccessful in this sequence, no sources of reactor makeup are available, and core damage occurs.

V_1 - <u>LPCS or LPCI Available</u>

The LPCS is a single loop system. The LPCI is a 3 loop system. Either LPCS or one loop of LPCI is required to maintain reactor water inventory. These systems all have motor driven pumps taking suction from the suppression pool. They will initiate automatically upon receipt of a L1 low, low water level or high drywell pressure signal. In addition to the initiation signal, a further condition for the injection valves to open is the reduction in reactor pressure to less than 470 psig. Flow into the vessel commences as the reactor pressure drops below 400 psig. An LPCI initiation will automatically secure any other RHR lineup and reposition valves as necessary for the LPCI mode. Detailed fault trees for the LPCS and the LPCI are developed in system notebooks.

V_2 - Condensate System Available

Three condensate pumps take suction from the condenser hotwell via a single header. The condensate is directed to the suction of 3 condensate booster pumps. Water can be injected through the feedwater pumps or the bypass line into the reactor at low pressure. One condensate pump or one booster pump is required to maintain reactor water inventory. For maintaining reactor water inventory over extended periods, condenser hotwell makeup from CST is required for the condensate system. The operating range for the condensate pump is from 160 to 0 psig. The operating range for the booster pump is from 560 to 0 psig with a condensate pump operating. A detailed fault tree for the condensate system is developed in a system notebook.

V_3 - FP Water Available

An isolation valve and a fire hose connection are installed on the suction of the "A" condensate booster pump. FP water from two nearby fire hydrants can be injected into the reactor at low pressure. There are 4 fire pumps. Three of these (one diesel and two electric motor driven) take suction from the circulating water basin. The fourth pump (diesel) takes suction from the bladder tank which is filled from either the Well House Storage Tank or from the TMU system. One fire pump is sufficient to maintain reactor water inventory. A detailed fault tree for the FP water injection is developed in its system notebook and the process is detailed in the emergency procedures. However, the human reliability analysis performed for FP water implementation indicates a failure probability of 1.0 due to timing constraints. Therefore, FP water is excluded from the event tree.

V_4 - SW Crosstie to RHR-B Available

Service water from the SW B header can be lined up to the discharge of the RHR Heat Exchanger B via 2 keylocked valves. The keys are maintained in the control room under Administrative Control. When the valves are open, service water can be directed to any path associated with the RHR loop B for coolant injection into the core. The human reliability analysis for operator actions required to crosstie SW-B to RHR-B is performed in the human reliability analysis. The SW-B system pressure at the point of injection is 70 psig.

W_1 - RHR Available

The RHR system must provide a complete flow path from and to the containment (or reactor) through at least one RHR heat exchanger. In addition, the SW system must provide cooling water to the corresponding RHR heat exchanger from the spray pond. The RHR system has 2 loops for shutdown cooling mode, 2 loops for drywell spray mode, 2 loops for suppression pool spray/cooling mode, and 1 loop for vessel head spray mode. These modes are under manual control and are mutually exclusive. For Level 1 IPE purposes, only the suppression pool spray/cooling mode is modelled. That model includes the RHR pumps, heat exchangers, and all of the salient valves. Although there are other RHR modes for removing decay heat, those other modes do not significantly add to the RHR availability. A detailed fault tree for the RHR is developed in its system notebook.

Z - MSIVs Open and PCS Available

The use of the PCS as a method of containment heat removal is possible if at least one main steam line (2 MSIVs) and the corresponding turbine bypass line (turbine bypass valve) are maintained open. In addition, at least one of the main condenser circulating water pumps must be operable and delivering cooling water to the main condenser. If this method of containment heat removal is used, it must be done before containment pressure increases beyond 54 psig to ensure that MSIV could remain open. Based on MAAP simulations, this time period is estimated to be about 21 hours from accident initiation.

A detailed fault tree of the PCS method of containment heat removal is developed in its system notebook.

W_2 - Containment Venting Available

Without decay heat removal from containment, the suppression pool will eventually heat up and steam will be generated in the wetwell. Pressure in the containment will then continue to increase. The plant procedures direct drywell and wetwell venting through 30" and 24" exhaust butterfly valves. This could cause the "soft" HVAC ductwork to fail. However, ECCS equipment and motor control centers are not expected to be affected.

If decay heat removal is unavailable via RHR, the PCS or containment vent, containment pressure will rise due to suppression pool heat-up. The outcome of this scenario if the low pressure systems are operating is that ADS eventually will fail due to the loss of the necessary differential pressure between containment and the nitrogen supply to the valves. In this situation, reactor vessel pressure will rise and eventually exceed the pumping capability of the low pressure makeup systems. This results eventually in core uncovery and fuel damage.

If, instead, the high pressure makeup systems are operating (feedwater, RCIC or HPCS), containment failure will occur due to overpressure. Containment failure from severe overpressure can take two forms:

- catastrophic failure resulting in a large breach, rapid depressurization and possibly bulk boiling of the suppression pool. This causes loss of all injection and core melt because:

    - there is insufficient available NPSH for the pumps so they cavitate and fail,
    - release of steam to the reactor building which causes high temperatures in the ECCS pump rooms and resultant failure of pump motors or switchgear.

- a self limiting membrane tear which results in a controlled leak to the reactor building. This in turn may release enough steam to the reactor building to cause the temperature in the pump rooms to increase enough to initiate failure of the injection pump motors or critical switchgear. This eventually leads to core melt. A membrane tear is always expected to occur before catastrophic failure if the containment pressure increase is gradual.

Based on WNP-2 containment structural analyses, containment failure has been demonstrated to preferentially occur in any of three locations as discussed in Section 4.3. Two are located in the drywell region, and one is located above the horizontal stiffeners in the wetwell. The conditional occurrence probability for each failure location was found to be equal, so there is a 33% chance that the failure will occur in the wetwell and a 67% chance that it will occur in the drywell.

A failure in the wetwell region is assumed to fail the ECCS pumps because of their proximity to the leak location. Failures in the drywell, however, are expected to have no effect on the ECCS pump rooms because the pump rooms are closed, watertight and located at least three floors below the expected leak sites. The most likely outcome for leaks from the drywell to the reactor building wall will be that blow-out panels installed in the reactor building will fail and vent the steam to the environment.

On the basis of this qualitative argument and the estimate of equal likelihood for failure at each location, a 0.33 likelihood for failure of injection and subsequent core damage was assigned to each sequence in which containment overpressure was caused by loss of decay heat removal when high pressure makeup systems are operating. With the use of basic event CF-FAILS-INJECT, this likelihood was represented in the sequence quantification. The feedwater pumps are located outside of the Reactor Building and therefore would most likely be unaffected by the release of steam from containment. For simplification, the 0.33 value was in fact used for scenarios in which feedwater was postulated to be available.

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3.1.2.1.1 Turbine Trip

A variety of turbine or nuclear system malfunctions will initiate a turbine trip. Some examples are moisture separator and reheater drain tank high level, loss of DEH control fluid pressure, low condenser vacuum, and reactor high water level.

The sequence of events for a turbine trip at 105% power can be obtained from the WNP-2 FSAR. Turbine stop valve closure will initiate 1) turbine bypass operation, 2) reactor SCRAM via position signals to the RPS, and 3) recirculation pump trip to LFMG (Low Frequency Motor-Generator). For the turbine trip from rated power with the bypass system operating, neutron flux increases rapidly because of the void reduction caused by primary system pressure increase. However, the flux increase is limited by reactor SCRAM. Safety relief valves open when system pressure exceeds relief set points. As the pressure is relieved the reactor water level swells. The excess capacity of the feedwater system will compensate for the loss in water inventory through the SRVs. Decay heat can be dumped into the main condenser through the turbine bypass valves. The level swell may reach L8 thereby tripping the feedwater turbines. If feedwater is lost due to L8 trip or from equipment failure, vessel water level starts to decrease. At L2, MSIVs are closed, recirculation pumps are tripped and HPCS and RCIC are initiated. Relief valves cycle open and close on pressure. Reactor vessel pressure is maintained by steam generated due to decay heat of the fuel. HPCS or RCIC will provide sufficient water to restore the water level to the normal range. For the condition where both HPCS and RCIC are unavailable, water level will continue to drop. ADS will receive the required low water level signals (L3 and then L1) and the signal that one of the low pressure ECCS pumps is running. After the system timer times out at 105 seconds, the ADS valves will open and depressurize the reactor vessel. Depressurization can also be accomplished manually by the operator.

The RCIC system will isolate on low reactor pressure. When reactor vessel pressure drops below the shutoff head of the low pressure ECCS systems (LPCS and 3 LPCIs), these systems begin injecting coolant into the vessel and rapidly reflood it. For the unlikely condition when the LPCS and all three LPCIs are unavailable, the WNP-2 Emergency Operating Procedures direct that the condensate and the condensate booster systems be used to maintain the water level. For the very unlikely condition that LPCS, 3 LPCIs, condensate and condensate booster systems are all unavailable to restore level between L3 and L8, the Emergency Operating Procedures direct that the FP water be crosstied to the condensate system through COND-P-2A to restore the water level in the vessel. The same procedures also direct that the Standby Service Water loop B be crosstied to the RHR B to maintain level above top-of-active-fuel.

After reactor SCRAM, and water level is maintained, core decay heat must be removed to prevent containment failure. Heat can be removed from the suppression pool by the RHR Suppression Pool Cooling or Spray loop A or B. For the unlikely condition that Suppression Pool Cooling or Spray modes are unavailable for containment heat removal, decay heat can be removed by steaming through the MSIVs and the turbine bypass valves to the main condenser. Heat will be carried to the cooling towers by the circulating water system. Since decay heat is less than 5% of rated power 15 minutes after SCRAM, the mechanical vacuum pumps can be used when the Steam Jet Air Ejectors (SJAE) fail to produce vacuum in the condenser. The RPV low water level isolation interlocks on the MSIVs should be cleared if the low pressure coolant injection is successful and the MSIVs can be reopened per the EOPs. The operator will have sufficient time to do this. It takes approximately 21 hours for the decay heat to increase the temperature of the suppression pool to the point that the saturation pressure causes the MSIVs to automatically close.

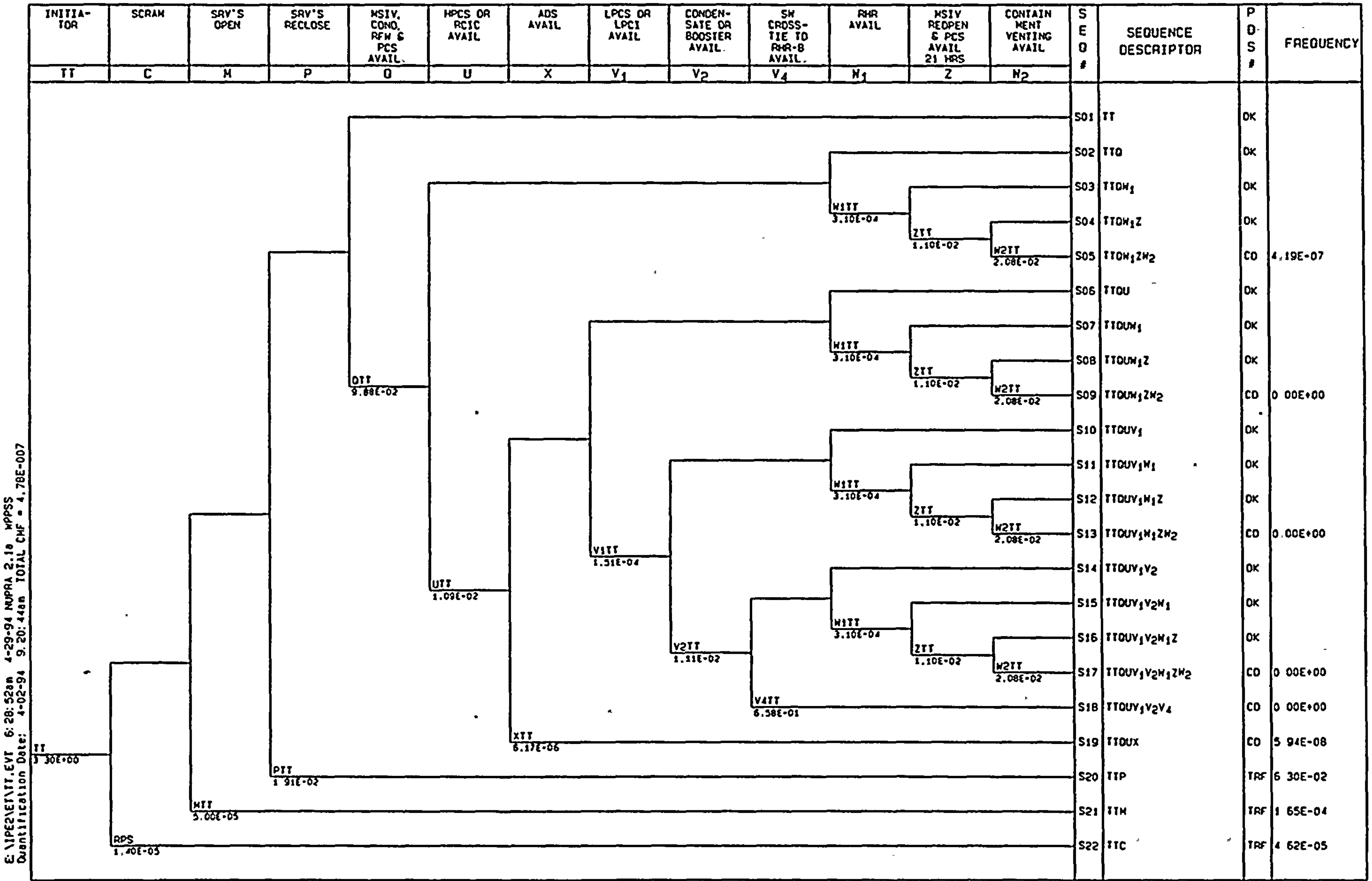
If drywell pressure reaches 39 psig, the Emergency Operating Procedures direct containment venting through 30″ and 24″ exhaust butterfly valves. The WNP-2 has 'hard' pipe from the containment to the SGT and 'hard' pipe from the SGT to the plant Stack. The SGT itself is 'soft' ductwork. Venting the containment at 39 psig will fail the SGT at 572′ elevation. Blow-down in the building at the 572′ elevation is not likely to affect the ECCS equipment at 420′ elevation or the critical motor control centers (MC-7B, 8B) at 522′ elevation due in part to the presence of blow-out panels in the Reactor Building exterior walls. Additionally, the ECCS equipment and motor-control centers are in sealed rooms having individual fan coolers.

Some of the containment heat can also be removed by venting the containment through the 2″ vent lines to the SGT to achieve a filtered, elevated release. Because of the small flow rate through the 2″ bypass valves, venting using these lines can only delay but not prevent containment failure.

Figure 3.1.2.1-1 is the general transient event tree applicable to turbine trip. Each of the functional events listed across the top of the event tree is discussed below. All T_T initiators are treated in the same manner (100% power) regardless of power level. This results in some degree of conservatism.

T_T - Turbine Trip Initiator

Section 3.1.1 discusses the initiating events. The average turbine trip frequency after the first year of commercial operation is 5.5/year. Using the discussion in Section 3.1.1, the present turbine trip frequency of 5.5 x 0.6 = 3.3/yr was used in the analysis.



TURBINE TRIP EVENT TREE

FIGURE 3.1.2.1-1

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3.1.2.1.2 Manual Shutdown

Manual shutdown is not a SCRAM transient but, rather requires a manual control rod insertion to bring the reactor subcritical in an orderly fashion. This characterizes events such as:

- Scheduled outages for maintenance
- Refueling outages

Since manual shutdowns are gradual, controlled events to bring the plant to a safe shutdown, they are characterized by the use of the normal makeup and heat removal systems (i.e., main condenser and feedwater/condensate systems). The sequence of events following a manual shutdown are shown below:

1. Reduce reactor power by reducing recirculation flow.
2. Feedwater injection used to control reactor water level within normal range.
3. Manually drive a selected set of control rods into the core.
4. Reduce power to 30%.
5. Transfer feedwater to manual control.
6. Reduce power to 10%.
7. Remove turbine from line and open turbine bypass valves to provide a heat sink.
8. Trip the reactor.
9. Control injection with feedwater-condensate and remove decay heat through the condenser.

Deviation from this controlled shutdown would generally be of a low likelihood or it could lead to a transient condition, which is included in the quantification of the other transient initiators. The event tree, Figure 3.1.2.1-2, is for the manual shutdown event. The discussion in Section 3.1.2.1-1 on turbine trip functional headings applies to the manual shutdown case with the following exceptions:

M_S - Manual Shutdown Initiator

The manual shutdown frequency after the first year of operation is 0.83/year. Using the discussion in Section 3.1.1, the present manual shutdown frequency is 0.83 x 0.6 = 0.5.

C - Reactor Critical

Failure to bring the reactor subcritical is negligible because of the controlled nature and the long time available during manual shutdowns.

M - Safety Relief Valves Open

The SRVs are not expected to open during manual shutdown. This function has a low likelihood of ever being challenged and, therefore, an extremely low likelihood of failure.

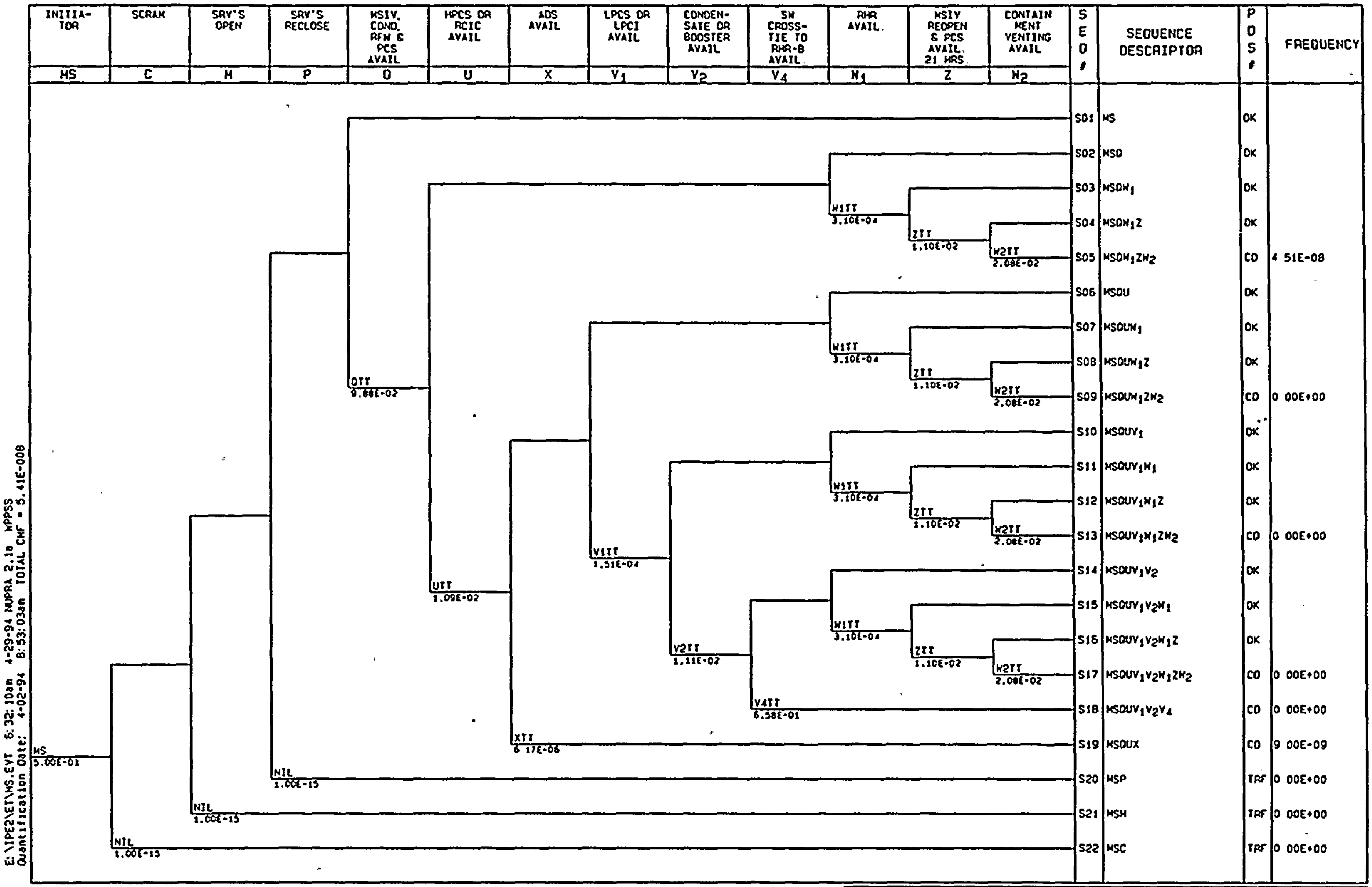
P - Safety Relief Valves Reclosed

Because of the low likelihood that the SRVs will be challenged during manual shutdown, the possibility of SRV's failure to reclose is also negligible.

Q - MSIVs Open, Condensate, Feedwater and PCS Systems Available

In a turbine trip, turbine stop valves close and the reactor SCRAMs. Safety relief valves open when the system pressure exceeds relief set points. The feedwater turbines may trip due to the water level swelling to L8. During a manual shutdown, however, the feedwater system is normally operating. The operators are familiar with the feedwater system because it is the primary method of maintaining reactor water inventory during normal operation. Therefore, the probability of feedwater failure during manual shutdown is less than that following a turbine trip; however, the feedwater fault tree developed in its system notebook as used in the other event trees is also used to bound this event.

Other event tree branch point headings are discussed in Section 3.1.2.1.



MANUAL SHUTDOWN EVENT TREE
FIGURE 3.1.2.1-2

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3.1.2.1.3 MSIV Closure

There are a number of incidents at nuclear power plants in which the main steam isolation valves may inadvertently or spuriously close causing a SCRAM challenge and a demand for the safe shutdown systems to operate. Two examples of MSIV closure incidents are:

1. A maintenance error during testing causes the MSIV closure logic to indicate a demand for MSIV closure.

2. A problem with a single MSIV during testing causes it to close suddenly and inadvertently, resulting in a high steam flow in the other steam lines. High steam flow results in a MSIV closure protective trip signal. Therefore, a closure of one MSIV may cause an immediate closure of all the other MSIVs depending on reactor conditions.

MSIVs can also close due to the following:

Main Steam Tunnel Area High Temperature
Main Steam Tunnel Area High delta Temperature
Main Steam Line High Flow
Main Steam Line Low Pressure
Reactor Low Water Level-2
Main Condenser Low Vacuum

The sequence of events for a MSIV closure at 105% rated power can be obtained from the WNP-2 FSAR. The event tree for the MSIV closure event is Figure 3.1.2.1-3.

MSIV closure initiates a reactor SCRAM via position signals to the RPS. Closure of these valves inhibits steam flow to the feedwater turbine terminating feedwater flow. Valve closure causes a trip of the main turbine and generator. Relief valves operate to limit primary system pressure. Because of the loss of feedwater flow, water level within the vessel decreases sufficiently to initiate trip of the recirculation pumps and initiate the HPCS and RCIC systems. The system pressure then begins dropping as the HPCS and RCIC systems subcool the core until flow is shutoff on high water level L8. Once the HPCS and RCIC shutoff, the water level starts dropping again and the pressure again rises to the SRV setpoint. With the loss of steam out of the relief valves the water level continues to drop, reinitiates the HPCS and RCIC, and the sequence of events described above is repeated.

For the condition that both HPCS and RCIC fail, water level continues to decrease towards the L1 setpoint. Automatic or manual depressurization can take place to reduce reactor pressure and allow low pressure coolant injection to restore reactor water level. These systems include LPCS, LPCI, Condensate, FP water and the SW crosstie to RHR-B. Buildup of decay heat requires the operation of the containment heat removal systems over the long term. Heat rejection can be achieved with successful operation of the RHR system, the PCS system (long term only, with MSIV open recovery), or venting. The functional headings in the MSIV closure event tree (Figure 3.1.2.1-3) are discussed below. All T_M initiators are treated in the same manner (100% rated power) regardless of power level. The functional hedings got discussed below are the same as those represented in Section 3.1.2.1.

T_M - MSIV Closure Initiator

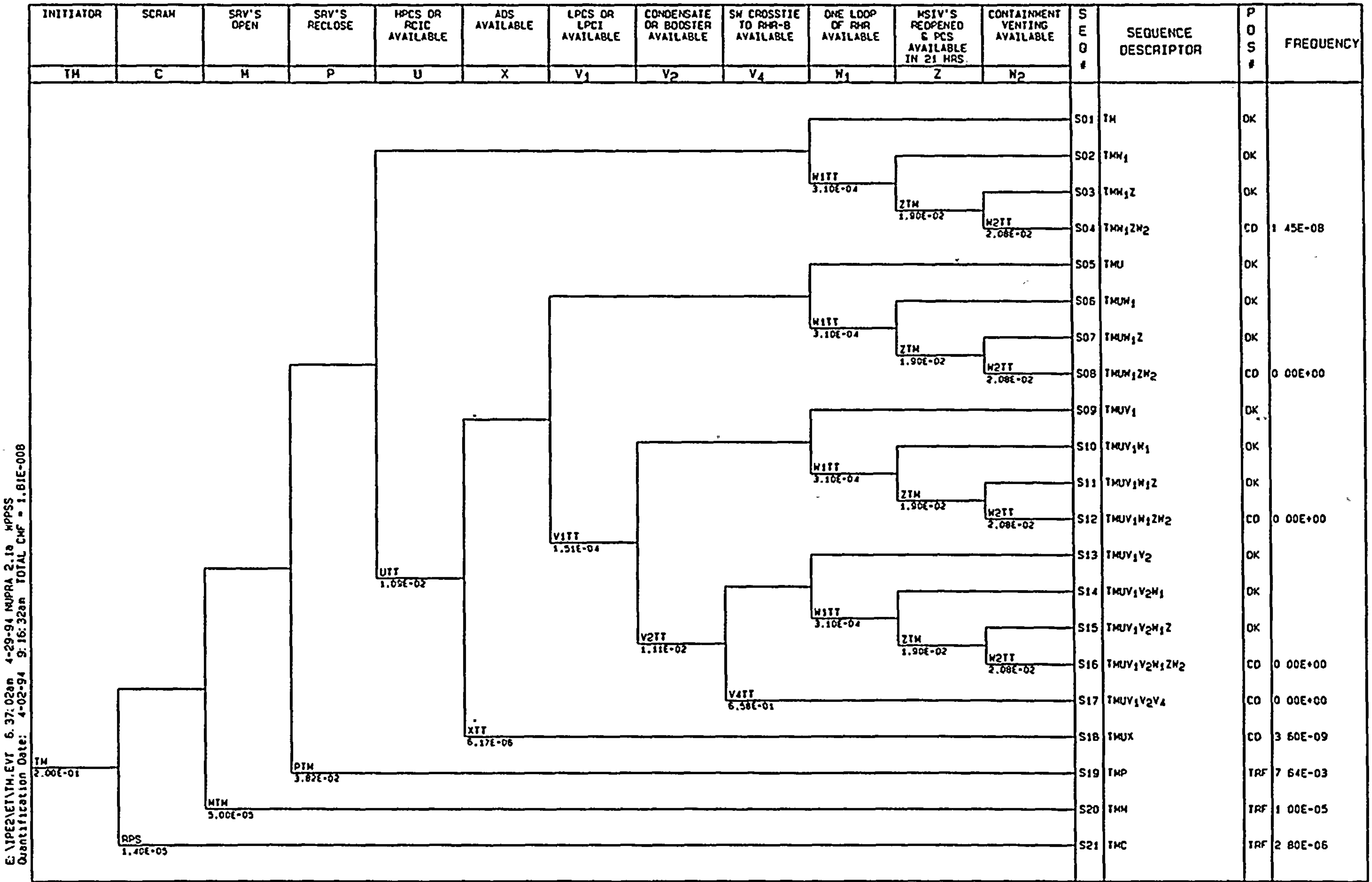
Section 3.1.1 discusses the initiating event frequencies. The MSIV closure frequency after the first year of operation is 0.333/year. Using the dicussion in Section 3.1.1, the present MSIV closure frequency is 0.6 x 0.333 = 0.2/yr.

Q - MSIVs Open, Condensate, Feedwater and PCS Systems Available

MSIV·closure inhibits steam flow to the feedwater pumps terminating feedwater flow. Although the operators are intimately familiar with the condensate/ feedwater system, the probability of reopening the MSIVs and recovering the feedwater system in a short time is conservatively assumed to be zero. Recovery in the longer term is discussed under heading Z below.

Z - MSIVs Open and PCS Available

Functional event Z incorporates a operator error probability for failure to open the MSIVs in addition to the other systems needed for the PCS. There are approximately 21 hours available to correct the cause of MSIV closure and allow the PCS to become the heat sink for containment heat removal. The operator actions to recover PCS by reopening MSIVs are analyzed in the human reliability analysis.



MSIV CLOSURE EVENT TREE
FIGURE 3.1.2.1-3

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3.1.2.1.4 Loss of Feedwater

A loss of feedwater flow could occur from pump failures, feedwater controller failures, operator errors, or reactor system variables such as high vessel water level (L8) trip signal.

The loss of feedwater event is the most challenging abnormal operational transient with respect to coolant inventory control since it results in the most rapid reactor coolant inventory loss. Feedwater serves two fundamental purposes:

1. Replenishes the reactor coolant inventory loss due to steam flow to the turbine and other paths;

2. Mixes with the relatively hot steam separator return flow to provide proper core inlet subcooling so that the void reactivity control is achieved.

Upon a loss of feedwater, reactor vessel water level starts to decrease rapidly due to the mismatch between coolant inventory loss (steam) and supply (feedwater). The rate of level decrease depends on the initial power level: higher initial power will cause faster level decrease. Because of diminishing injection of relatively cold feedwater, core inlet becomes warmer. This causes more void generation in the core, hence neutron flux decreases. As the power level is lowered, the turbine system flow starts to drop off because the pressure regulator is attempting to maintain pressure.

The level will continue to decrease and reach the low level SCRAM setpoint L3 where reactor SCRAM is initiated. SCRAM will cause a further rapid level reduction due to the redistribution of vessel downcomer water to fill the collapsed voids inside the core. Once the voids in the core have collapsed, level continues decreasing due to steaming to the main condenser through the turbine. Level eventually decreases to the L2 isolation setpoint.

The L2 trip will close the MSIVs, trip the recirculation pumps, and initiate HPCS and RCIC. Recirculation pump coastdown maintains higher than natural circulation core flow for a period of time. Reactor vessel pressure soon rises to the safety relief valve (SRV) setpoint. The pressure then remains at approximately the setpoint pressures as one or more SRVs cycle open and closed to maintain pressure control. Vessel pressure is then maintained by steam generated by the decay heat of the fuel. Vessel inventory continues to be lost as steam flows through the SRVs.

Under normal conditions, the high pressure makeup water systems will provide sufficient water to restore level to the normal range. Plant shutdown or restart can then be accomplished. For degraded conditions where all of the high pressure systems are unavailable, the water level will continue to drop. Under these conditions, depressurization of the reactor to the range where the low pressure systems can inject water into the reactor is necessary.

Due to the large capacity of the low pressure systems, they will rapidly reflood the reactor. Once the vessel is reflooded, the operator can then proceed to place the reactor in cold shutdown.

The functional headings in the loss of feedwater event tree, Figure 3.1.2.1-4, are discussed below. All loss of feedwater initiators are treated in the same manner (100% rated power) regardless of power level.

T_F - Loss of Feedwater Initiator

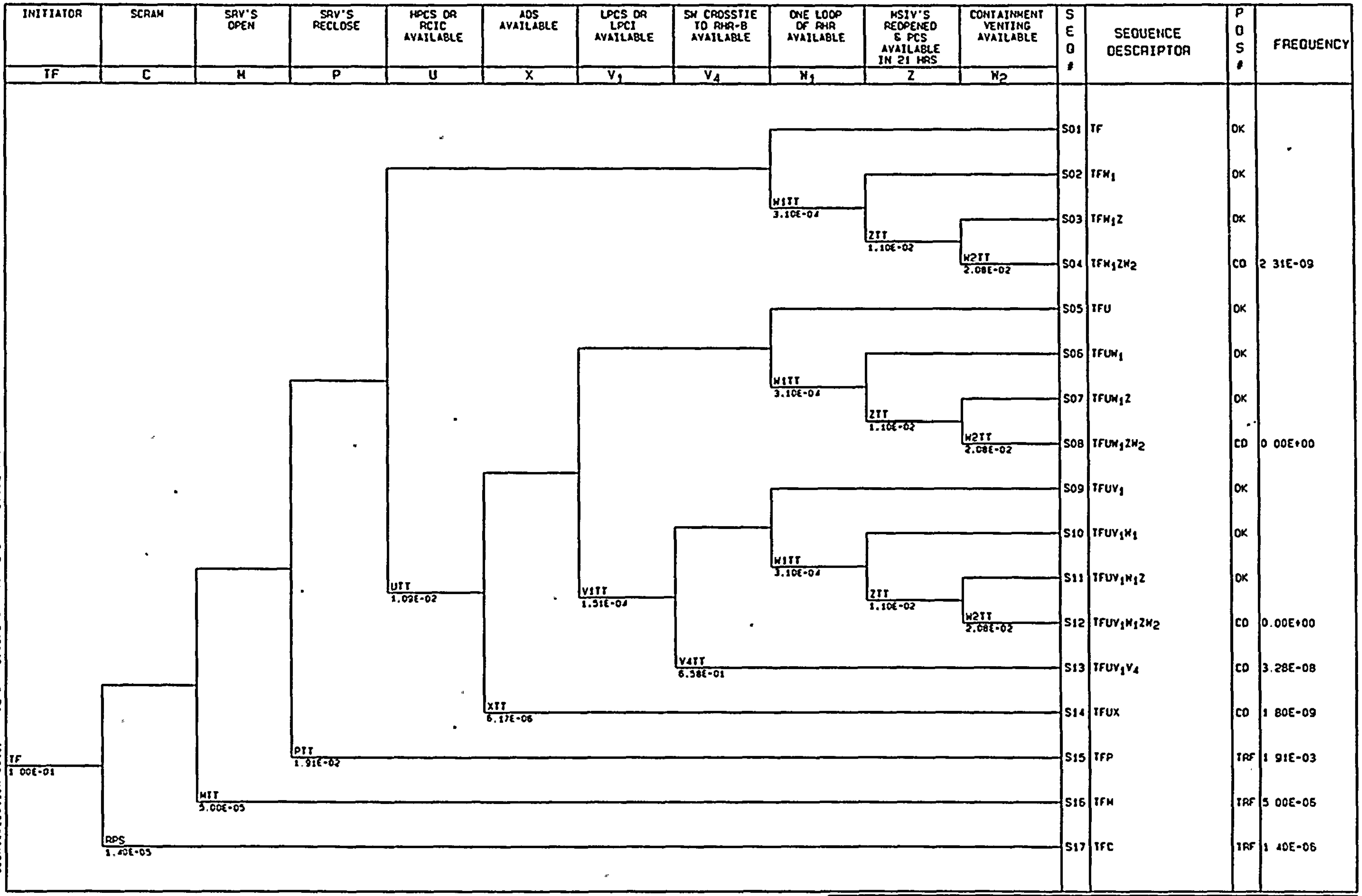
Section 3.1.1 discusses the initiating events. The loss of feedwater frequency after the first year of operation is 0.167/year. Using the discussion in Section 3.1.1, the present loss of feedwater frequency is 0.6 x .167 = 0.10/year.

Q - MSIVs Open, Condensate, Feedwater and PCS Systems Available

For the loss of feedwater event, the probability of recovering the feedwater system in a short time is conservatively assumed to be zero.

Other event tree branch point headings are discussed in Section 3.1.2.1.

E:\IPE2\ET\TF.EVT 6:39.58am 4-29-94 NUPRA 2.1a WPPSS
Quantification Date: 4-02-94 9:09:34am TOTAL CHF = 3.69E-008



LOSS OF FEEDWATER EVENT TREE
FIGURE 3.1.2.1-4

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3.1.2.1.5 Loss of Condenser

Following the loss of condenser vacuum at 2 inches Hg per second, the plant will first respond automatically by the closure of the turbine stop valves and the operation of the turbine bypass valves. Reactor SCRAM and recirculation pump trip to LFMG are initiated by the position switches on the turbine stop valves when the valves are less than 90% open. SRVs are actuated when the system pressure reaches relief setpoints. When the condenser vacuum is low enough, the MSIVs and the turbine bypass valves will close. This transient is similar to a normal turbine trip without bypass. It is an important transient to consider because the loss of condenser affects both the ability to provide coolant makeup (using the feedwater system) and the long-term containment heat removal (using the PCS). The functional headings in the loss of condenser event tree (Figure 3.1.2.1-5) that are different from those in section 3.1.2.1 are discussed below. All loss of condenser initiators are treated in the same manner (100% rated power) regardless of power level.

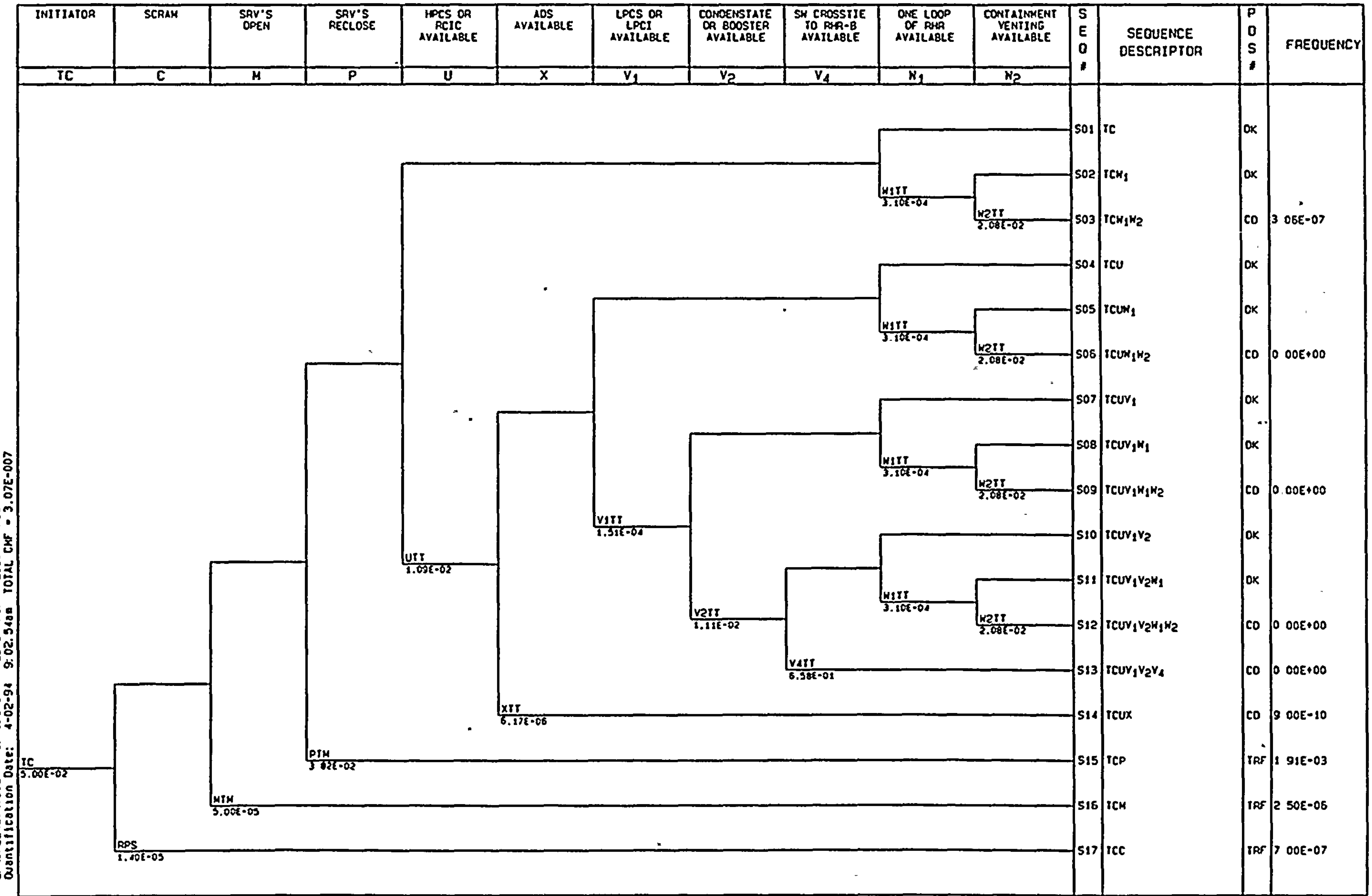
Tc - Loss of Condenser Initiator

The total number of loss of condenser events after the first year of operation is zero. For analysis purpose, the loss of condenser frequency is assumed to be 0.083/year. Using the discussion in Section 3.1.1, the present loss of condenser frequency is 0.083 x 0.6 = 0.05/year.

Q - MSIVs Open, Condensate/Feedwater and PCS Systems Available

Feedwater loss due to MSIV closure is conservatively assumed not to be recoverable in the loss of condenser event.

E:\IPE2\ET\TC.EVT 6:41:14am 4-29-94 NUPRA 2.1a WPPSS
Quantification Date: 4-02-94 9:02.54am TOTAL CMF = 3.07E-007



LOSS OF CONDENSER EVENT TREE
FIGURE 3.1.2.1-5

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3.1.2.1.6 Inadvertent/Stuck Open Relief Valve

Inadvertent opening and reclosing of a SRV has no significant effect on plant operation. However, a stuck open relief valve (SORV) can have significance. On noticing high SRV tail pipe temperature, high acoustic monitor indication, or suppression pool temperature increase, the operator at WNP-2 will reclose the valve as soon as possible and check that the reactor and turbine generator output return to normal. If the valve cannot be closed in 2 minutes, the operating staff will place the reactor mode switch in the shutdown position to SCRAM the reactor. It is assumed that there is a 2% probability that the operating staff is unaware of the SORV failure and fails to SCRAM. The discharge from the SRV to the suppression pool will cause a rise in suppression pool temperature and a rise in wetwell air space pressure. The 9 vacuum breakers between the wetwell and the drywell will cause subsequent drywell pressure rise to 1.68 psig. The reactor will then SCRAM automatically due to high drywell pressure.

The SORV will have approximately 800,000 lb/hr flow through the valve to the suppression pool. Before the SCRAM, the steam flow leaving the reactor causes a mild depressurization. The pressure regulator senses the nuclear system pressure decrease and within a few seconds closes the turbine control valve far enough to stabilize the reactor vessel pressure.

Potential undesirable states which could result include: 1) a SRV discharge line break in the wetwell air space, or 2) a long-term containment heat removal problem. It is assumed in this analysis that all SRVs discharge to the suppression pool. The case of a simultaneous SORV and SRV discharge line break is considered to be of very low probability and is therefore neglected.

Once the reactor is shut down, the SORV sequence of events is similar to the turbine trip sequence of events. However, since the reactor has been at full power, and has been releasing steam into the suppression pool for the time prior to SCRAM, the suppression pool temperature may have increased significantly. The MSIVs may close during this event due to low system pressure or low water level. The steam released decreases the time allowed for initiation of RHR to preclude suppression pool failure, loss of makeup, and eventual fuel damage or core melt.

The functional headings that are different from those in Section 3.1.2.1 in the IORV/SORV event tree (Figure 3.1.2.1-6) are discussed below. All T_I initiators are treated in the same manner (100% rated power) regardless of power level.

TI - <u>IORV/SORV Initiator</u>

The total number of IORV/SORV after the first year of commercial operation is zero (relief valve testing during startups are not included). To be conservative the IORV/SORV average frequency is assumed to be 0.083/year. Using the discussion in Section 3.1.1, the present SORV event frequency is 0.083 x 0.6 = 0.05/year. To this number, transfers from other event trees are added. For example, turbine trip with failure of one of the SRVs to reclose leads to a stuck open relief valve situation. The total frequency, with transfers included, is 2E-1/year.

C_1 - <u>Early Manual SCRAM</u>

The operator will be alerted to an IORV/ SORV condition by:

- high suppression pool temperature
- high drywell pressure
- increasing suppression pool level
- SRV acoustic monitor
- SRV discharge pipe temperature

Technical Specification 3/4.4.2 directs the operator to close the SORV within 2 minutes or if the suppression pool average water temperature is 110°F or greater, the operator is to place the reactor mode switch in the shutdown position. According to human reliability analyses, the probability of failing to perform a step-by-step task within 30 minutes is about 0.02. With all of the annunciators in a SORV situation, it is very unlikely for the operating staff not to close the SORV or place the mode switch in shutdown before 30 minutes. It is assumed that there is a 2% probability that the operating staff fails to observe the Technical Specifications and place the mode switch in shutdown.

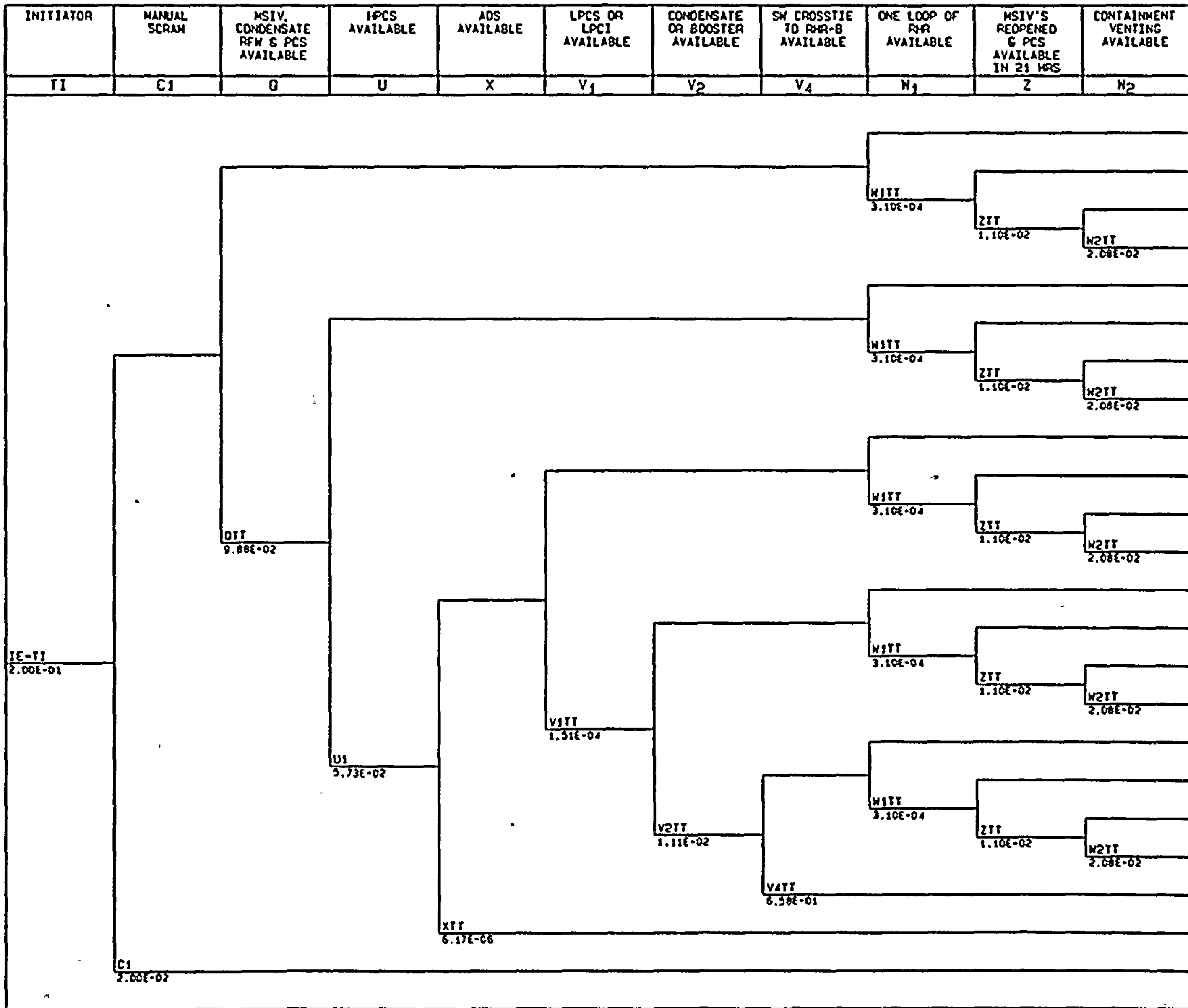
Q - <u>MSIVs Open, Condensate, Feedwater and PCS Systems Available</u>

This function is similar to the function in Section 3.1.2.1. The reactor is assumed to be initially at full power, and release steam to the suppression pool until time of SCRAM. If manual SCRAM is performed early, MSIVs will stay open and the RFW is a viable option for high pressure coolant injection. The feedwater availability should be approximately the same as that for the Turbine Trip event.

W_1 - RHR Available

This function is similar to the function in Section 3.1.2.1 with the exception that the heat removal requirements are greater for the IORV/SORV initiator because of the higher initial suppression pool temperature and pressure. It will take RHR a longer time period to remove heat from the containment for a SORV than for a turbine trip. Since this event is equivalent to a small break LOCA, 1 RHR loop is still sufficient for containment heat removal according to Table 3.1.1-4.

E:\IPE2\ET\TI.EVT 6:42:32am 4-29-94 NUPRA 2.1a WPPSS
Quantification Date: 4-02-94 9:12:53am TOTAL CMF = 7.68E-009



SEQ #	SEQUENCE DESCRIPTOR	POS #	FREQUENCY
S01	TI	OK	
S02	TIW_1	OK	
S03	TIW_1Z	OK	
S04	TIW_1ZW_2	CO	0 00E+00
S05	TIQ	OK	
S06	$TIQW_1$	OK	
S07	$TIQW_1Z$	OK	
S08	$TIQW_1ZW_2$	CO	3 82E-09
S09	TIQU	OK	
S10	$TIQUW_1$	OK	
S11	$TIQUW_1Z$	OK	
S12	$TIQUW_1ZW_2$	CO	0.00E+00
S13	$TIQUV_1$	OK	
S14	$TIQUV_1W_1$	OK	
S15	$TIQUV_1W_1Z$	OK	
S16	$TIQUV_1W_1ZW_2$	CO	0 00E+00
S17	$TIQUV_1V_2$	OK	
S18	$TIQUV_1V_2W_1$	OK	
S19	$TIQUV_1V_2W_1Z$	OK	
S20	$TIQUV_1V_2W_1ZW_2$	CO	0 00E+00
S21	$TIQUV_1V_2V_4$	CO	4 80E-10
S22	TIQUX	CO	3 38E-09
S23	TIC1	TRF	4 00E-03

IORV/SORV EVENT TREE
FIGURE 3.1.2.1-6

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REVIEWER__________ DATE______

3.1.2.1.7 Loss of Offsite Power

Electrical grid instabilities brought on by major shifts in electrical loads, lightning, storms or other disturbances, could cause damage to plant equipment. Protective relay schemes function at WNP-2 to mitigate damage in such instances by automatically disconnecting electrical sources and loads until electrical grid stability is regained.

Normal station power is supplied from the main generator via transformers TR-N1 and TR-N2. Startup power is supplied from the Ashe substation via a 230 Kv supply to transformer TR-S. Both the normal and startup transformers have the capacity to carry the full plant auxiliary load. A backup transformer, TR-B, is provided to supply critical buses SM-7 and SM-8 in the event that both the normal and startup transformers are lost. This transformer is supplied from the BPA Benton Substation via a 115 Kv line. During normal plant startup, plant electrical loads are supplied from the 230 Kv system and then are transferred to the main generator source when plant output has reached approximately 25% rated capacity. The startup transformer remains energized during plant operation, permitting the onsite AC electrical system to automatically transfer back in the event of a plant trip and loss of power from the generator. When the plant is in a shutdown condition, electrical power can be supplied via the 500 Kv line from the Ashe substation. This is accomplished by disconnecting the isolated phase bus duct links to isolate the main generator. Power is then established through the main step-up transformers to the AC distribution system. Experience has shown that it takes approximately 8 hours to make this transition.

The complete loss of offsite power (LOOP) at WNP-2 is a relatively low frequency initiator. Its effects, however, can be pervasive on plant systems which require AC power. If offsite power is lost, the balance-of-plant systems become unavailable, and the plant's safety systems ultimately require power from one of the emergency diesels or the recovery of offsite power for successful accident mitigation. For WNP-2, loss of offsite power means the power sources from the normal transformers, i.e., the startup transformer and the backup transformer are lost. Station blackout means emergency diesel generators 1 and 2 are also unavailable.

In general, the following sequence of events occur for the loss of offsite power initiator:

1. Recirculation pumps and condenser circulating water pumps trip off at zero seconds.

2. Due to loss of power to the SCRAM and MSIV relay solenoids, reactor SCRAM and MSIV closure is initiated at 2 seconds.

3. Feedwater turbines trip off at 4 seconds due to MSIV closure at 2 seconds.

4. Safety/relief valves open in the pressure relief mode of operation as the pressure increases beyond their setpoints.

5. Sensed reactor water level drops to the HPCS and RCIC initiation setpoint at approximately 36 seconds.

The event tree for the loss of offsite power initiator is provided in Figure 3.1.2.1-7.2. The tree models the significant sequences involved in providing onsite power, maintaining core cooling and containment heat removal, and restoring offsite power. The event tree and its functional events are discussed below.

T_E - <u>LOOP Initiator</u>

In the six years following WNP-2's first year of commercial operation, there were no events involving a complete loss of offsite power during normal operation. From a letter, dated November 1, 1988, sent by Carl M. Swanson, Bonneville Power Administration to G.C. Sorensen, Manager, Regulatory Programs, the following is quoted: "The WNP-2 - Ashe 500 Kv line was energized in April 1983, the WNP-2 - 230 Kv line in October 1978 and the Tap off the Benton - 451 115 Kv line in January 1973. For this limited time, however, the data shows that there has never been a time when all three sources were simultaneously out-of-service."

Based on data compiled by Bonneville Power Administration, the following power supply unavailabilities were derived:

500 Kv line = 0.1023/yr
230 Kv line = 0.048/yr
115 Kv line = 0.00123/yr

Taking into account common cause failures of lines and fast transfer failures, the probability of failure of all 3 offsite power sources was modelled in the fault tree provided in Figure 3.1.2.1-7.1. The loss of offsite power frequency derived from this tree is 2.46E-2/yr.

C - <u>Reactor Subcritical</u>

Failure to bring the reactor subcritical was not included in the development of LOOP tree since the initiating event frequency for such a scenario is less than 1E-7/yr. (The initiating event frequency for such a scenario is the combination of the likelihood for failure to SCRAM due to mechanical reasons, 4E-6, and the LOOP initiating event frequency of 2.46E-2/yr.)

DG - <u>Either Diesel Generator 1 or 2 Available</u>

In the event that offsite power becomes unavailable, AC emergency power can be supplied to the Division 1 and Division 2 4160 VAC buses via two independent diesel generators. The progression of events following a loss of offsite power with a failure to recover will be significantly different depending upon the availability of the emergency diesels. Successful diesel generator operation would make available several options for core cooling and containment heat removal, including LPCI, LPCS, RHR. Additionally, DC power would be supplied via the battery chargers allowing the continuous operation of RCIC and automatic ADS operation. Such functions are modelled in the sequences located on the upper event tree branch of function DG. With both diesels unavailable (represented by the lower event tree branch), a station blackout occurs. In this situation, HPCS can provide primary makeup if its dedicated diesel operates successfully. RCIC can also provide reactor coolant makeup for up to four hours prior to the depletion of station batteries.

U_1 - DG3, HPCS and its Cooling Available

Following a loss of offsite power, Division 3 4160V AC emergency power can be supplied to HPCS via diesel generator 3. HPCS can provide reactor coolant makeup which is required in a relatively short time following a SCRAM and throughout the reactor shutdown period. HPCS can supply adequate inventory makeup to the reactor following a loss of offsite power and successful SCRAM. The HPCS main pump seals and bearings are cooled by its own discharge. However, the pump room is cooled by the Reactor Building Emergency Cooling using the SW-C train. Therefore, the Reactor Building Emergency Cooling, the SW-C and the HPCS must all be available for successful inventory makeup.

Referring to the LOOP event tree diagram in Figure 3.1.2.1-7.2, two unique functional equations were used for the U_1 function - one representing HPCS operation in LOOP conditions (U1-LOOP), and a second representing HPCS operation in station blackout conditions (U1-SBO). Since HPCS has a dedicated diesel generator, it would be expected that the solutions for U1-LOOP and U1-SBO would be nearly identical, as in fact they are. Additionally, since the unavailability for U_1 is relatively high, the rare event approximation that allows for exclusion of success branch probabilities from event trees is not applicable. Therefore, functional equations were included on the success branches of the U_1 functional event.

U_2 - RCIC and its Cooling Available

RCIC is designed to start and run initially without any AC dependence. RCIC, although steam-turbine-driven, must rely on DC power (station batteries) for its operation. Following RCIC initiation, Loop B of the Standby Service Water system starts and the emergency cooling fan RRA-FN-6 of the RCIC pump room also starts if AC power is available. However, if AC power is unavailable, keeping the doors to the RCIC pump room open provides natural circulation sufficient to cool the room for continuous pump operation.

RCIC availability during a station blackout is strongly time dependent. This time dependence is principally due to the time varying auxiliary system requirements. The following considerations impact the availability of auxiliary systems required for RCIC coolant injection:

- Battery availability which is calculated to be 4 hours
- Room cooling requirements (RCIC system isolation on high room temperature by break detection logic)
- High suppression pool temperatures and containment pressure due to a lack of containment heat removal may have an adverse effect on the RCIC pump performance.

Two unique functional equations were generated for U_2: one representing RCIC operation in LOOP conditions (U2-LOOP), and a second representing RCIC operation in station blackout conditions (U2-SBO). The RCIC fault tree was solved to generate U2-LOOP by assuming the complete unavailability of offsite power, and U2-SBO by assuming the complete unavailability of AC power. Additionally, since the unavailability for U_2 is relatively high, the rare event approximation that allows for exclusion of success branch probabilities from event trees is not applicable. Therefore, functional equations were included on the success branches of the U_2 functional event. Discussion for the RCIC fault tree model development is provided in its system notebook.

X - <u>Timely Depressurization with ADS</u>

See subject discussion in Section 3.1.2.1. The ADS functional event is not used for station blackout sequences due to the unavailability of low pressure reactor makeup systems. Note that although the automatic ADS function is not available, the operating staff can manually depressurize the vessel. In the event that diesel generator 1 or 2 is available, the ADS functional event is placed in the event tree for sequences in which the high pressure makeup systems, HPCS and RCIC, are unavailable. The unavailability of ADS was prepared using the ADS fault tree and accounts for the complete unavailability of offsite power. A description of the fault tree's development of the tree is provided in its system notebook.

V - <u>Low Pressure Coolant Systems Available</u>

The available low pressure coolant systems during a loss of offsite power are LPCS, LPCI, and FP water. The cross-tie from service water train B to RHR train B is assumed to be unavailable for simplicity due to the possibility that diesel generator 2 is unavailable (which would cause the unavailability of SW train B). With the exception of the FP water pumps which are either motor-driven or diesel-driven, all low pressure pumps require 4160V AC power supplied by the diesel generators. Based on the fault trees developed for the low pressure systems, the unavailability for the low pressure systems is derived. The solution takes into account the complete unavailability of offsite power.

W_1 - <u>RHR Available</u>

See subject discussion in Section 3.1.2.1. The solution of functional event W_1 in the T_E event tree takes into account the complete unavailability of offsite power.

REC - <u>Recovery of Offsite Power Within X Hours</u>

Three time intervals for offsite power recovery are addressed in the LOOP event tree: 30 minutes, 4 hours, and 10 hours. The data values used for recovery of offsite power were developed from historical data contained in NSAC-194, "Losses of Off-Site Power at U.S. Nuclear Power Plants - Through 1992". Figure 3.1.2.1-7.3 presents the offsite power recovery curve developed in NSAC-194. From 1980 through 1992 there were 47 losses of offsite power at nuclear power plants ranging from 15 seconds to 5.5 days. The two longest outages of 5.5 days each were at the Turkey Point plants and were a result of Hurricane Andrew. These outages are considered not applicable to WNP-2 and were removed from the data in this analysis. The remaining 45 events ranged from 15 seconds to 19 hours. This data was used to revise the recovery of offsite power curve presented in NSAC-194.

In order to calculate the non-recovery of offsite power probability, the number of unrecovered events at specific time intervals were divided by the total number of events (45) yielding the non recovery probability. The probability of non-recovery at the critical time points of interest were based on this revised curve and are as follows:

NRAC30M (non-recovery in 30 minutes)	=	0.622
NRAC4 (non-recovery in four hours)	=	0.144
NRAC10 (non-recovery in ten hours)	=	0.0296

If power is recovered within the required time frame, the applicable sequences were assigned an end state of "OK." Although these sequences do not represent a safe stable state (that is, hot shutdown), such sequences are dominated by cutsets which could be recovered by the availability of offsite power. Failure to place the plant in a safe stable state would require several additional system failures due to the availability of offsite power, thus making the postulated core damage sequences for these situations relatively insignificant. Therefore, an end state of "OK" was assigned.

<u>Non-recovery of offsite power within 30 minutes (NRAC30M)</u>

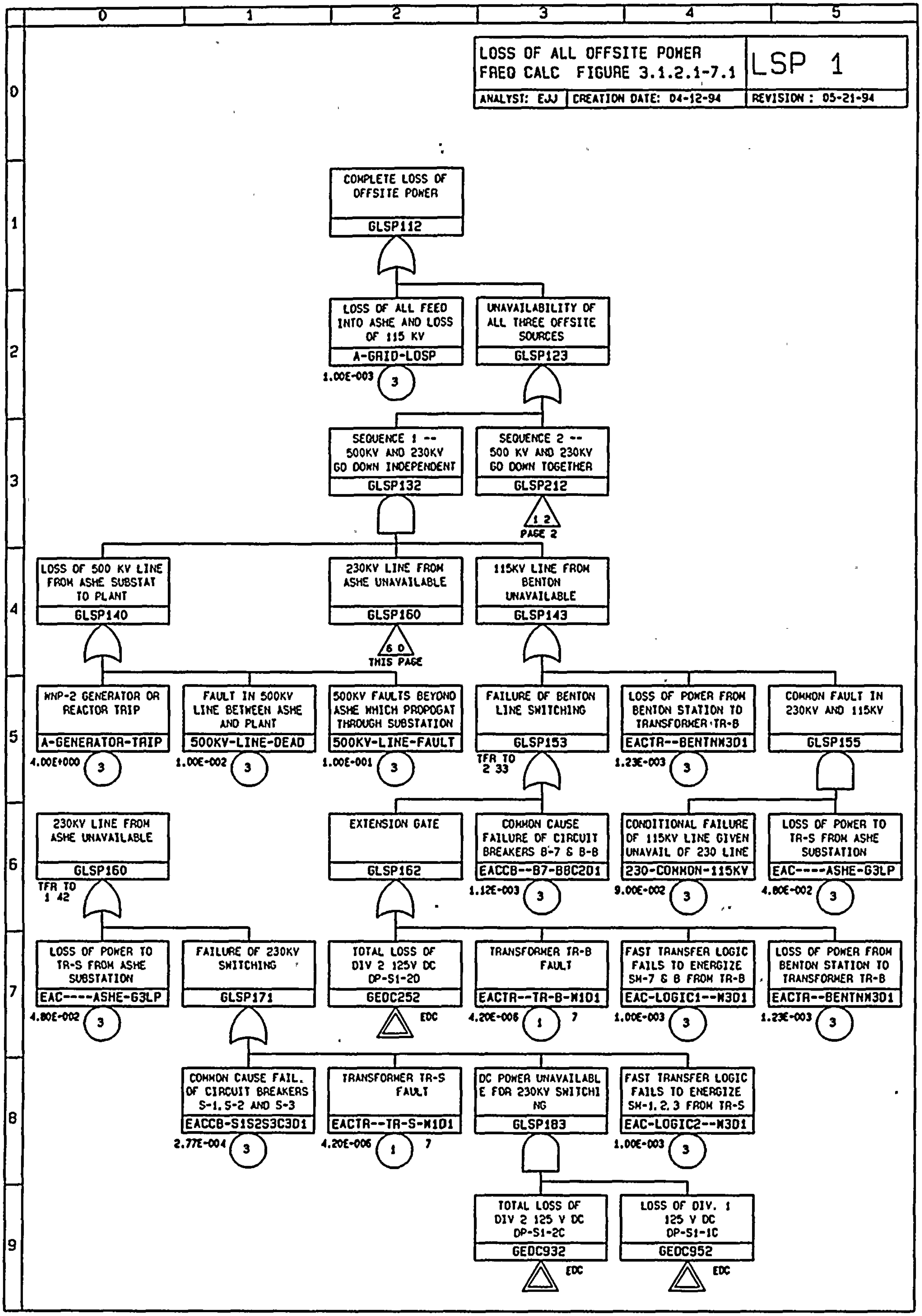
The first critical time period identified for offsite power recovery is 30 minutes due to the estimated time before core uncovery following a loss of offsite power and the subsequent failure of all injection sources. In the event that neither HPCS or RCIC is available and either the vessel cannot be depressurized automatically or the low pressure systems are not available, the recovery of offsite power within 30 minutes is questioned on the event tree. The 30 minute time interval is conservative since the core is actually predicted to uncover after 45 minutes based on MAAP simulations.

Non-recovery of offsite power within 4 Hours (NRAC4)

The second critical time identified by the LOOP event tree development is 4 hours following the loss of offsite power and the subsequent failure of Division 1 and 2 of AC Power (that is, failure of diesel generators 1 and 2). This time period is identified as critical based on the battery depletion time and its impact on RCIC operation. The 4 hour recovery time is questioned only following the failure of HPCS and successful operation of RCIC. In this state, RCIC is the only injection source available and will fail when the batteries deplete. Therefore power must be restored prior to battery depletion. This 4 hour time period is conservative since it is assumed to begin at the time of the initiator rather than from the time that the diesel generators 1 and 2 fail to operate.

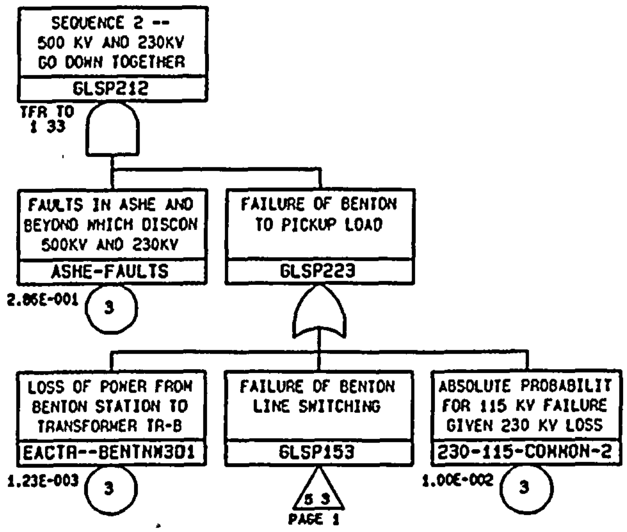
Non-recovery of offsite power within 10 Hours (NRAC10)

The third critical time period identified during event tree development is 10 hours following the loss of offsite power. The recovery of offsite power within 10 hours is questioned for the sequences where containment heat removal is unavailable. Without containment heat removal, suppression pool temperature is calculated based on MAAP simulations to rise to 212°F in approximately 10 hours. The HPCS switchover from CST to the suppression pool cannot be performed successfully at this suppression pool temperature, since the HPCS is assumed to fail at a temperature greater than 212°F. Injection may remain available for a short time through ADS actuation and low pressure injection success. In these sequences however, without containment heat removal, the containment will continue to heat up and containment pressure will increase to the point that the ADS valves can no longer be maintained open and the primary system will repressurize. Since no injection is available at this point, core uncovery and damage is assumed to occur. For a similar sequence under station blackout conditions, the non-recovery of offsite power within 10 hours leads to failure of all injection. HPCS fails due to high suppression pool temperature at 10 hours, RCIC already has failed at 4 hours due to battery depletion, and the low pressure systems remain unavailable due to the continued unavailability of AC power.

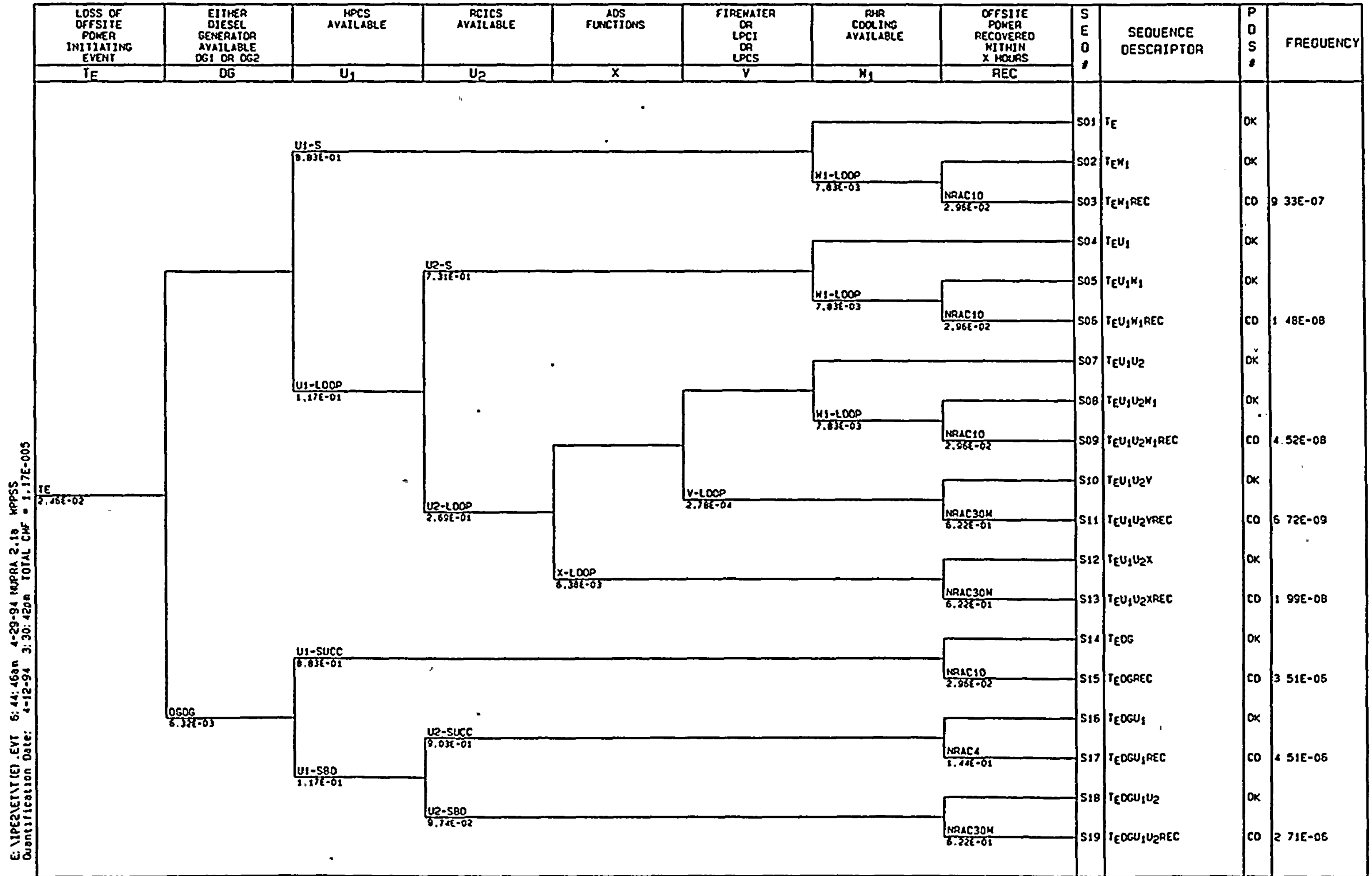


LOOP.LGC NUPRA 2.1a WPPSS

LOSS OF ALL OFFSITE POWER FREQ CALC FIGURE 3.1.2.1-7.1	LSP 2
ANALYST: EJJ · CREATION DATE: 04-12-94	REVISION : 05-21-94



LOOP.LGC NUPRA 2.1a WPPSS



LOSS OF OFFSITE POWER EVENT TREE
FIGURE 3.1.2.1-7.2

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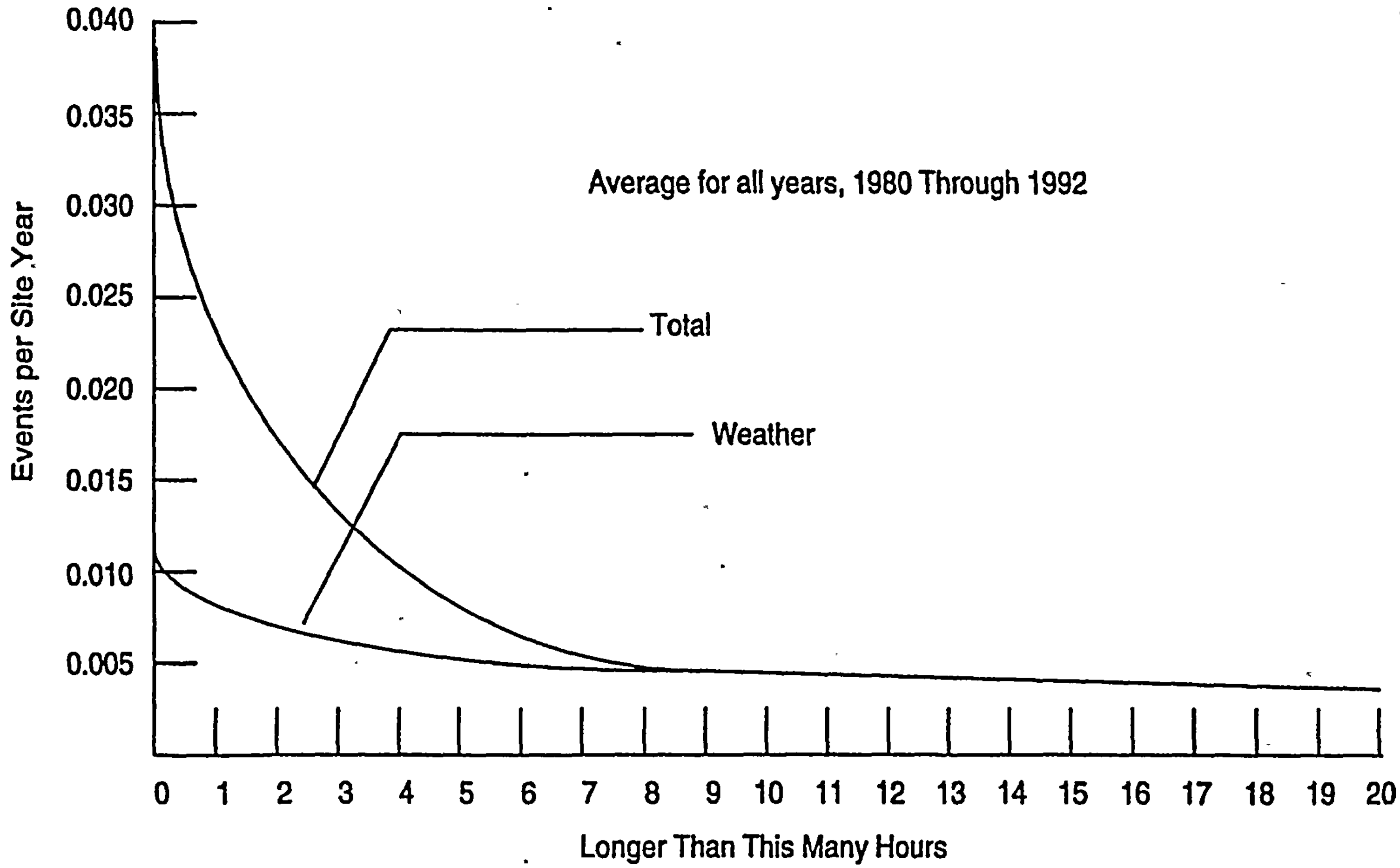

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Figure 3.1.2.1-7.3 Losses of all off site power per generating unit year longer than x hours

3.1.2.2 LOCA Event Trees

The LOCA evaluation for WNP-2 includes a number of discrete initiators to represent the spectrum of postulated LOCAs from the small to the large LOCA design basis accident. In considering pipe break location, pipe break size, and the level of operability required of the redundant ECCS systems, a very large number of combinations for analysis would result. Combinations have been reduced to make the analysis manageable. Five event trees depicting LOCA are used:

Small LOCA	(1″ < D < 4″)
Medium LOCA	(4″ < D < 6″)
Large LOCA	(D > 6″)

Steam line break outside the containment
Interfacing systems LOCA.

WNP-2 went into commercial operation in December 1984. The reactor pressure vessel is an ASME Class I component constructed to the requirements of the Summer 1971 Section III code. A complete stress report on the RPV has been prepared in accordance with ASME requirements. The stress analysis performed for the reactor vessel assembly (including the faulted condition) were completed using elastic methods. Therefore, catastrophic reactor vessel rupture was not considered in this IPE. If the reactor were to fail early in its lifetime, it would most likely leak before failing. Such an accident initiator would be similar to that of a small LOCA.

3.1.2.2.1 Small LOCA (1″ < D < 4″)

Flow through a small break is a constant enthalpy process. If the primary system break is below the reactor water level, the blowdown will consist of reactor water. Blowdown from reactor pressure to drywell pressure will flash approximately one-third of this water to steam and two-thirds will remain as liquid. Both phases will be at saturation conditions corresponding to drywell pressure.

If the primary system rupture is located so that the blowdown flow consists of reactor steam only, saturated steam will result in superheated conditions. A small reactor steam leak will impose the most severe temperature conditions on the drywell structures and the safety equipment in the drywell.

After a small break in a pipe connected to the reactor vessel inside the primary containment, the vessel pressure and water level tend to slowly decrease, with a corresponding increase in drywell pressure and temperature. When the drywell pressure reaches 1.68 psig, a signal will be generated to SCRAM the reactor, start the diesel generators, and initiate the ECCS systems. For small breaks, the excess capacity of the feedwater system will compensate for the loss in vessel inventory due to break flow. Furthermore, the HPCS system will begin injecting water into the vessel. The operating staff will later take over manual control of the water makeup system to maintain level and proceed to a cold shutdown state.

For the condition that both RFW and HPCS are unavailable, the reactor water level will continue to fall and finally reach the L2 trip setpoint. This trip will close the MSIVs, trip the recirculation pumps, and initiate RCIC. Once the MSIVs are closed, the reactor pressure soon rises to the SRV setpoint. The pressure then remains at basically the setpoint pressure as the SRV's cycle open and closed. The vessel pressure is maintained by steam generated by decay heat of the fuel.

The drywell pressure increase will lower the water level in the downcomer vents until the level reaches the bottom of the vents. At this time, noncondensibles and steam will start to enter the suppression pool. The steam will be condensed and the air will be carried over to the suppression chamber free space. The noncondensibles carryover will result in a gradual pressurization of the suppression chamber. Once all the drywell noncondensibles are carried over to the suppression chamber, pressurization of the suppression chamber will cease and the system will reach an equilibrium condition. The drywell will contain mostly superheated steam, and continued blowdown of reactor steam will condense in the suppression pool. The suppression pool temperature will continue to increase until the RHR heat exchanger heat removal rate is greater than the decay heat release rate.

The small LOCA event tree is shown in Figure 3.1.2.2-1. The evaluation of the event tree branch points is the same as that following the stuck open relief valve event with manual SCRAM. A generic value of 8E-3 is used for the small LOCA frequency.

3.1.2.2.2 Medium LOCA (4″ < D < 6″)

The sequence of events for the medium LOCA is similar to that for the small LOCA. The steam or liquid will blowdown at a higher rate. HPCS is still sufficient for reactor level makeup. However, RCIC alone is insufficient for level makeup in a medium LOCA situation. Due to the possible closure of the MSIVs when the water level drops to L2, the RFW for high pressure core injection and the PCS for the containment heat removal are assumed unavailable. The medium LOCA event tree is shown in Figure 3.1.2.2-2. A generic value of 3E-3 is used for the medium LOCA frequency. The description for the functional events in the tree is the same as the applicable ones described in Section 3.1.2.1.

3.1.2.2.3 Large LOCA (D > 6")

Figure 3.1.2.2-3A shows a schematic view of the flow paths to the recirculation line break. In the side adjacent to the recirculation loop suction nozzle, the flow will correspond to critical flow in the pipe cross-section. In the side adjacent to the recirculation loop injection nozzle, the flow will correspond to critical flow at the ten jet pump nozzles associated with the broken loop. In addition, the cleanup line crosstie will add to the critical flow area.

Rupture of a main steam line between the reactor vessel and the flow limiter (inside containment) results in the flow of primary system fluid and energy to the drywell. In the side adjacent to the reactor vessel, the flow will correspond to critical flow in the steamline break area. Blowdown through the other side of the break will occur because the steam lines are interconnected at a point upstream of the turbine by the bypass header. This interconnection allows primary system fluid to flow from the 3 unbroken steam lines, through the header and back into the drywell via the broken line. Flow will be limited by critical flow in the steam line flow restrictor.

For either the recirculation line break or steam line break, SCRAM will occur due to a high drywell pressure. For the recirculation line break, MSIVs will close due to low reactor water level, L2. For the steam line break, void formation in the reactor vessel water causes a rapid rise in the water level to the steam nozzles. L8 will trip RFW pumps. The closure of MSIVs will cut off motive power to the steam driven feedwater pumps. HPCS is initiated on high drywell pressure or when the reactor water level reaches L2. RCIC will be initiated at L2, but the lack of steam pressure for RCIC turbine will make it unavailable for coolant injection.

Primary system pressure will equalize with the drywell pressure fairly quickly. The drywell will contain primarily steam. The suppression chamber is pressurized by the carryover of noncondensibles from the drywell and by heatup of the suppression pool. As the vapor formed in the drywell is condensed in the suppression pool, the temperature of the suppression pool water peaks and the suppression chamber pressure stabilizes. The drywell pressure stabilizes at a slightly higher pressure, the difference being equal to the downcomer submergence.

The drywell and suppression pool will remain in this equilibrium condition until the reactor vessel refloods. During this period, the emergency core cooling pumps will be injecting cooling water from the suppression pool into the reactor. This injection of water will eventually flood the reactor vessel and spill into the drywell. The water spillage will condense the steam in the drywell and thus reduce the drywell pressure. As soon as the drywell pressure drops below the suppression chamber pressure, the drywell vacuum breakers will open and noncondensible gases from the suppression chamber will flow back into the drywell until the pressure in the two regions equalize.

The large LOCA event tree is shown in Figure 3.1.2.2-3B. It covers the worst of the recirculation line break and the steam line break in terms of demand on ECCS pumps for reflooding and containment heat removal. The functional headings in the large LOCA event tree are discussed below.

A - Large LOCA Initiator

Section 3.1.1 discusses the initiating events. Large LOCA is a design basis event. A generic frequency 3E-4/year was used in the event tree.

C - Reactor Subcritical

See subject discussion in Section 3.1.2.1. Although a large LOCA with failure to SCRAM is extremely unlikely, such an event is very difficult to mitigate and it is assumed to lead directly to core damage.

D - Vapor Suppression - Omega Seal and Vacuum Breakers OK

During a large LOCA challenge, high temperature/pressure primary fluid is released directly to the drywell. The suppression pool acts as a heat sink and vapor suppression mechanism for LOCAs. The drywell and suppression chamber are interconnected by 99 downcomer pipes. These downcomer vents allow the transfer of steam and air from the drywell to the suppression chamber after a postulated LOCA. The steam is condensed in the suppression pool which therefore provides a reduction in the pressure rise in the drywell. Nine vacuum breakers are provided to allow a return flow path for noncondensibles from the suppression chamber to the drywell.

The vapor suppression event represents the requirement for the downcomers to pass steam from the drywell to the suppression pool condensing a significant amount of steam produced in the postulated large LOCA. Contributors to the failure of this event following a large LOCA are (1) the drywell floor seal failure; (2) vacuum breaker failure; and (3) downcomer vent pipe failures. As has been shown previously in WASH-1400 and the Limmerick PRA the second and third items are low probability events and are therefore not generally considered contributors. The first item, failure of the drywell floor seal, is designed for design basis LOCA. It is judged to be extremely reliable with a conditional failure probability of 1E-4 per demand for the large LOCA.

Failure of the drywell floor seal (Omega seal) indicates that the released steam will no longer pass through the suppression pool. In this case heat and mass transfer takes place between the air space and the pool surface. This inefficient energy transfer mechanism cannot condense the large amount of steam released to the containment. Therefore, containment failure occurs immediately. Subsequent failure of all ECCS equipment may occur due to penetration failure, flooding, or an adverse environment in the reactor building.

U_1 - HPCS Available

HPCS injects water into the reactor vessel during the large LOCA event. It takes suction initially from the CST. When the CST reaches low level, the suction is automatically transferred to the suppression pool. RCIC will initiate at L2. However, due to the low reactor pressure to drive the RCIC turbine, RCIC is assumed unavailable during the large LOCA event.

V_1 - LPCS or LPCI Available

This function is similar to the function in Section 3.1.2.1.

W_1 - RHR Available

This function is similar to the function in Section 3.1.2.1 with the exception that the heat removal requirements are greater for the large LOCA event because of the high suppression pool temperature and pressure. Table 3.1.1-4 indicates that one loop of RHR is sufficient for containment heat removal.

W_2 - Containment Venting Available

See subject discussion in Section 3.1.2.1.

3.1.2.2.4 Large LOCA Outside Containment

WNP-2 is protected from LOCA outside containment by virtue of the isolation capability of the primary system lines which penetrate the containment. The steam lines, feed lines, high pressure injection lines (HPCS, RCIC, RWCU) all have a check valve, stop valve or both inside the primary containment. In the event of a LOCA outside the containment, the break can be detected by the operating staff due to high room temperature indication and high floor drain flow indication. The broken line will then be automatically or manually isolated. However, if automatic or manual isolation is unsuccessful, there may be a high environmental stress produced on equipment in the reactor building, i.e., ECCS operation may be compromised. The consequences of a core melt in this situation will be significant because of the direct pathway out of the primary system and containment.

The steamline break outside containment event tree is shown in Figure 3.1.2.2-4. Its effect on core damage frequency is typical of all large LOCAs outside containment. Its frequency is multiplied by a factor of three to include effects of RFW, and RWCU line breaks on the core damage frequency. The probabilities of HPCS and RCIC line breaks are very low because they are standby systems. The functional headings in the event tree which are different than Section 3.1.2.1 are discussed below:

A_{out} - Steamline Break Outside Containment

Section 3.1.1 discusses the initiating events. A steamline break outside containment is a design basis event and is not expected to occur during plant lifetime. There are four main steam lines. According to WASH-1400, pipe rupture rate is 8.59E-10/hr per section. WNP-2 availability is conservatively assumed to be 0.8. There are three sections between the in-board isolation valve and the high pressure turbine. The probability of a main steam line break is 8.59E-10 x 365 x 24 x 4 x 0.8 x 3 = 7.22E-5/year.

To include the effect of RFW and RWCU line breaks in the event tree, an initiating frequency of 3 x 7.22E-5 = 2.17E-4/year is used.

C - Reactor Subcritical

The reactor will get a SCRAM signal at low level L3. Although successful mitigation of a large LOCA outside containment with failure to SCRAM is very difficult, such an event is extremely unlikely. It is assumed to lead directly to core damage.

I - Containment Isolation in 15 Minutes

In case of a steamline break outside the containment, the break will be isolated immediately by MSIVs. A detailed fault tree for the NS^4 system is developed in the fault tree system notebook. Unavailability of NS^4 Group 1 isolation is used for functional event I.

Z - MSIVs Open and PCS Available

For one steamline break outside the containment, MSIVs will close due to high steam flow. If one unbroken line is reopened for containment heat removal using PCS, the steam line interconnection will allow primary system fluid to flow from the unbroken steam line through the header and back to the broken line. Containment heat removal using PCS is conservatively assumed unavailable for all LOCAs outside containment.

3.1.2.2.5 Interfacing systems LOCA

An interfacing systems LOCA involves the loss of isolation between the high pressure primary system and a low pressure system outside containment. For WNP-2, such a LOCA could possibly occur in the LPCS loop, LPCI loops, Shutdown Cooling Suction loop, or Shutdown Cooling Discharge loops. The isolation valves associated with the loops are indicated below:

System	Isolation
LPCS	LPCS-V-5 and LPCS-V-6
LPCI A	RHR-V-41A and RHR-V-42A
LPCI B	RHR-V-41B and RHR-V-42B
LPCI C	RHR-V-41C and RHR-V-42C
Shutdown Cooling Suction	RHR-V-8 and RHR-V-9 and (RHR-V-6A, -6B or -67)
Shutdown Cooling Discharge	RHR-V-50A and RHR-V-53A
Shutdown Cooling Discharge	RHR-V-50B and RHR-V-53B

Once isolation is breached, the subsequent integrity of the low pressure systems is dependent on a number of factors:

- interface leak rate,
- low pressure system relief valve capacity,
- instrumentation to detect,
- the ultimate strength of low pressure piping,
- the ultimate strength of seals, gaskets, flanges, and other components, in the low pressure system,
- operator surveillance, training and instruction,
- primary system pressure and temperature.

In the IPE, it is conservatively assumed that the rupture of low pressure system piping is inevitable once the high pressure/low pressure isolation fails.

Isolation valve failures can be due to rupture, human error, and spurious electrical signals, all of which are unlikely events. NUREG/CR-2815 has catastrophic leakage failure rates for motor-operated valves and check valves, but does not have spurious electrical signal rates. Human error of omission is negated by the self alignment of motor-operated valves to the closed position when remote manual switches are in AUTO. A fault tree was developed to calculate the interfacing systems LOCA initiating frequency (see Figure 3.1.2.2-5.1). The frequency is calculated to be 1.21E-6/yr based on the probabilities of simultaneous rupture of isolation valves for each low pressure system loops.

In an interfacing systems LOCA, the LOCA size could be large, medium or small. For a large interfacing systems LOCA, the primary system is automatically depressurized. For a medium or small interfacing systems LOCA, it is important that the primary system be depressurized. Since the isolation valves between the primary system and the low pressure system have already failed, the only way to keep the break flow to a minimum is by depressurizing the primary system. To achieve safe shutdown condition, the core must be covered with water and the break flow eventually stopped. It is assumed in the analysis that failure to depressurize with ADS does not allow the break flow to be minimized or stopped, leading directly to core damage.

When any one of the following emergency isolation signals is received, the reactor building Heating and Ventilation system shuts down and isolates the reactor building from the primary containment and the outside:

F - High Drywell Pressure

A - Reactor Vessel Low Water Level

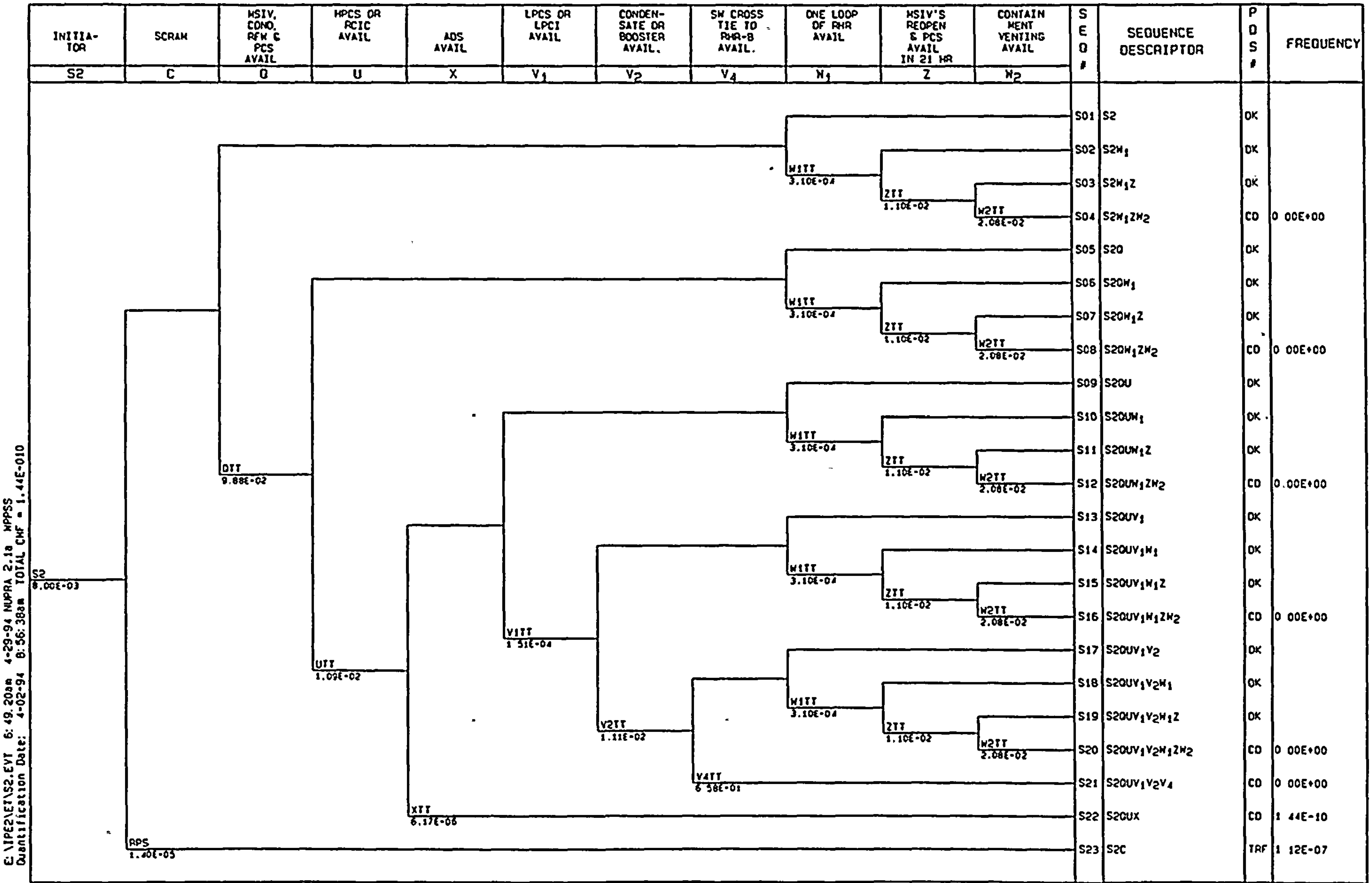
Z - High Radiation, Reactor Building Ventilation Exhaust Plenum

Air operated shutoff dampers in the HVAC supply ducts to the Div. 2 MCC Room, Div. 1 MCC Room, DC MCC Room, Div. 1 and 2 H_2 Recombiner Room, and Sampling and Analyzer Rooms, will automatically close on emergency isolation signals. Using the SW system as a heat sink, the reactor building Emergency Cooling system initiates to maintain ambient temperature in pump rooms, MCC rooms, and other rooms. There is no fire sprinkler system in the reactor building that will actuate due to high temperature.

It is assumed in this analysis that the break occurs in the LPCS line. The results would be similar for breaks in other lines. Water level will drop to L2 (closing MSIVs) and L3 (SCRAM) following the break. The interfacing systems LOCA event tree is shown in Figure 3.1.2.2-5. The functional headings in the event tree different from those in Section 3.1.2.1 are discussed below.

IS - <u>ISLOCA</u>

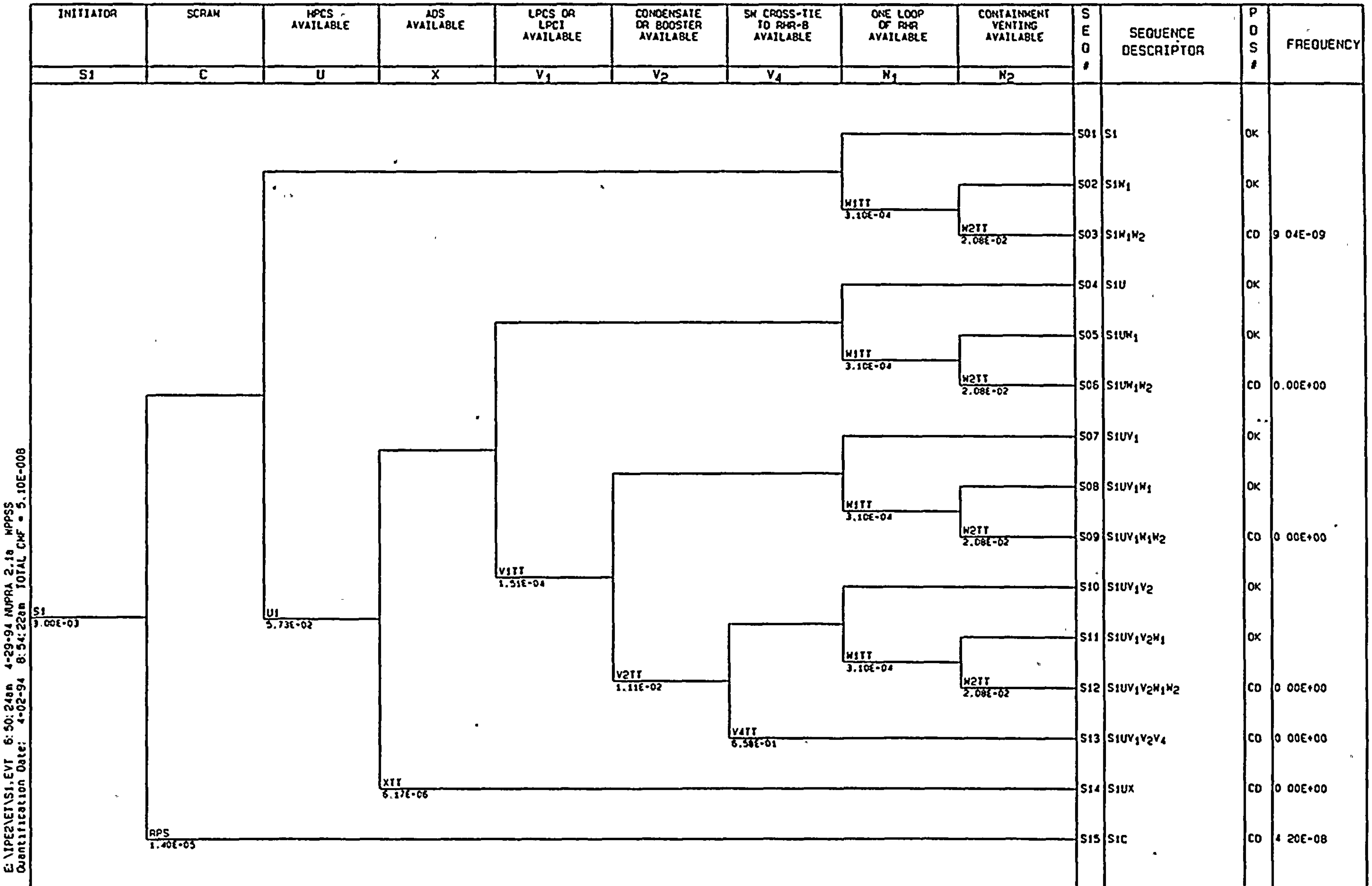
See the previous discussion for the interfacing systems LOCA initiating frequency.



SMALL LOCA EVENT TREE
FIGURE 3.1.2.2-1

AUTHOR________________DATE_______

REVIEWER________________DATE_______



MEDIUM LOCA EVENT TREE
FIGURE 3.1.2.2-2

AUTHOR__________ DATE______

REVIEWER__________ DATE______

POINT OF CRITICAL FLOW

A. RECIRCULATION LINE
B. CLEANUP LINE
C. COMBINED AREA OF ALL JET PUMP NOZZLES ASSOCIATED WITH THE BROKEN LOOP

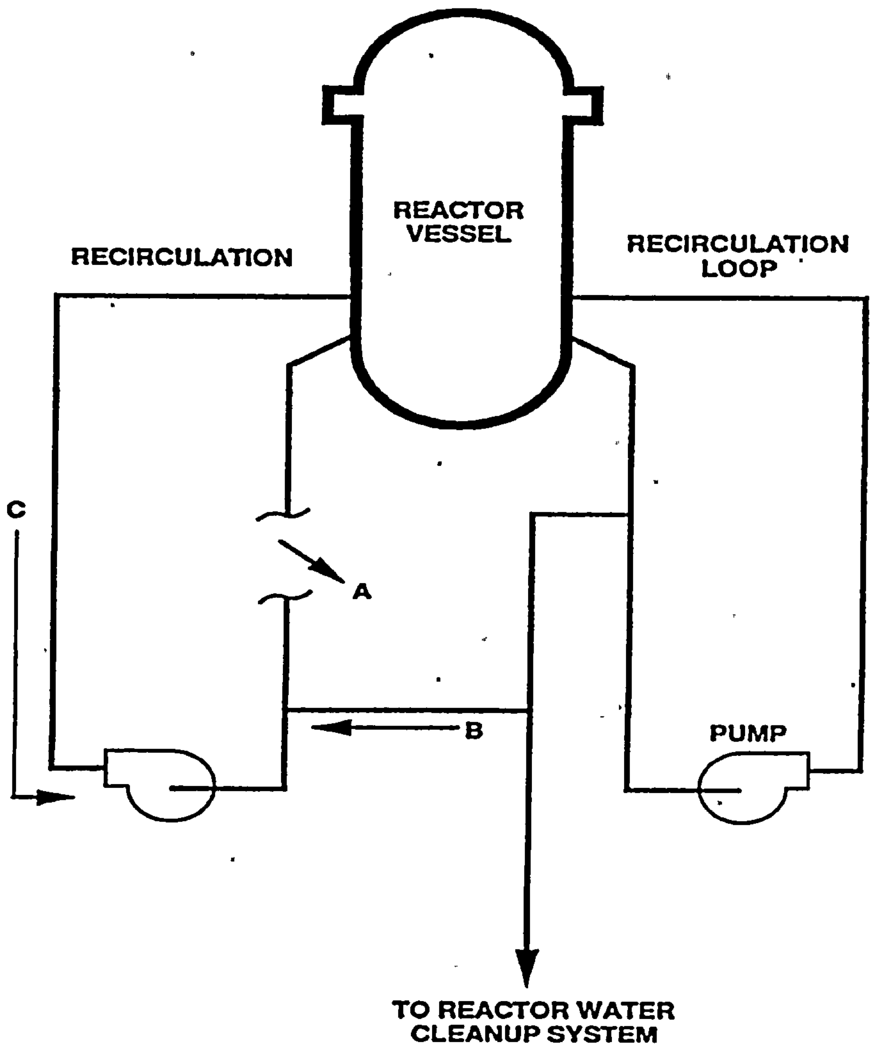
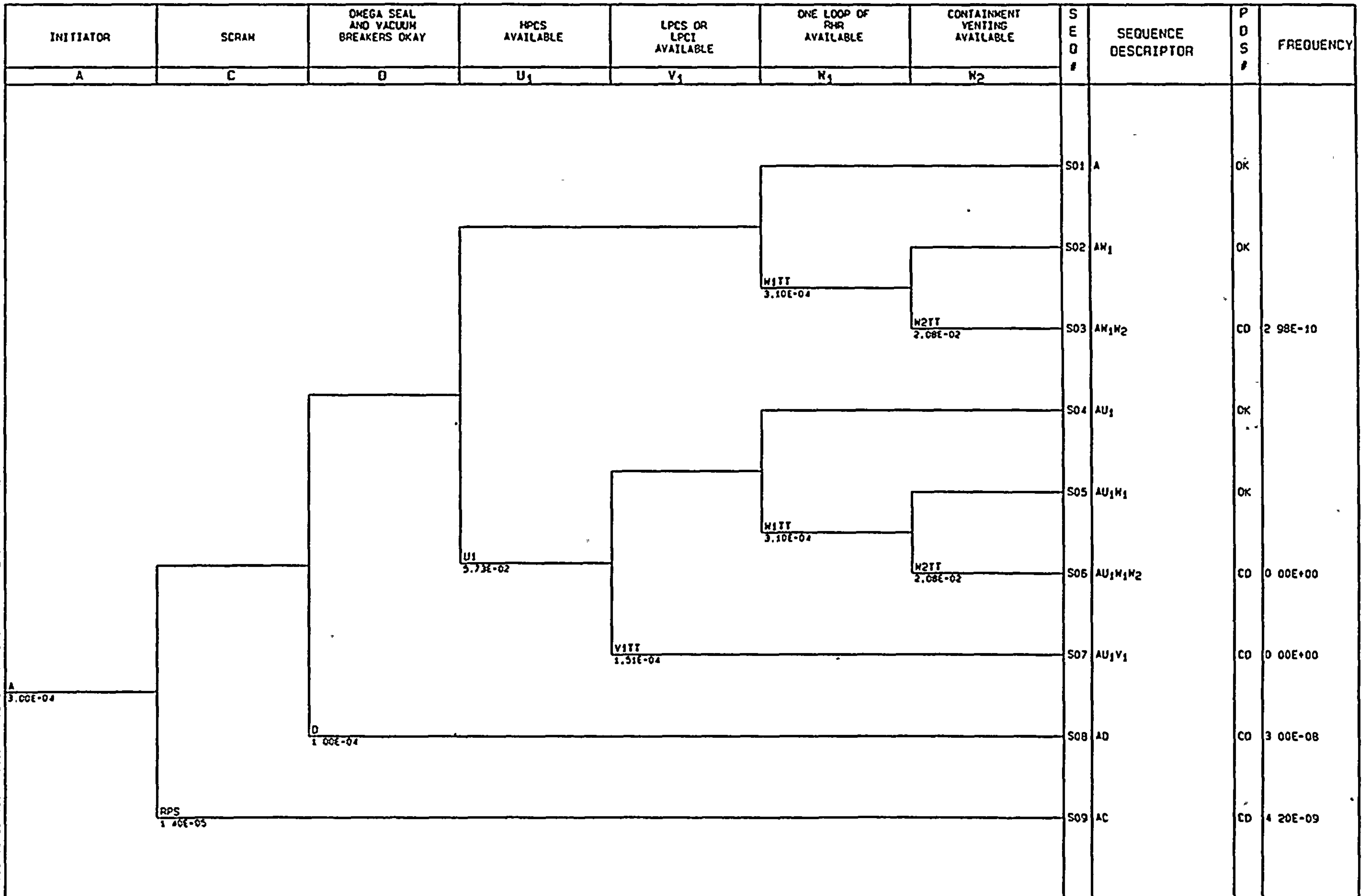


FIGURE 3.1.2.2-3 A SCHEMATIC SHOWING RECIRCULATION LINE BREAK



E:\IPE2\ET\A.EVT 8:11:14am 5-24-94 NUPRA 2.1a WPPSS
Quantification Date: 4-02-94 8:43:05am TOTAL CMF = 3.45E-008

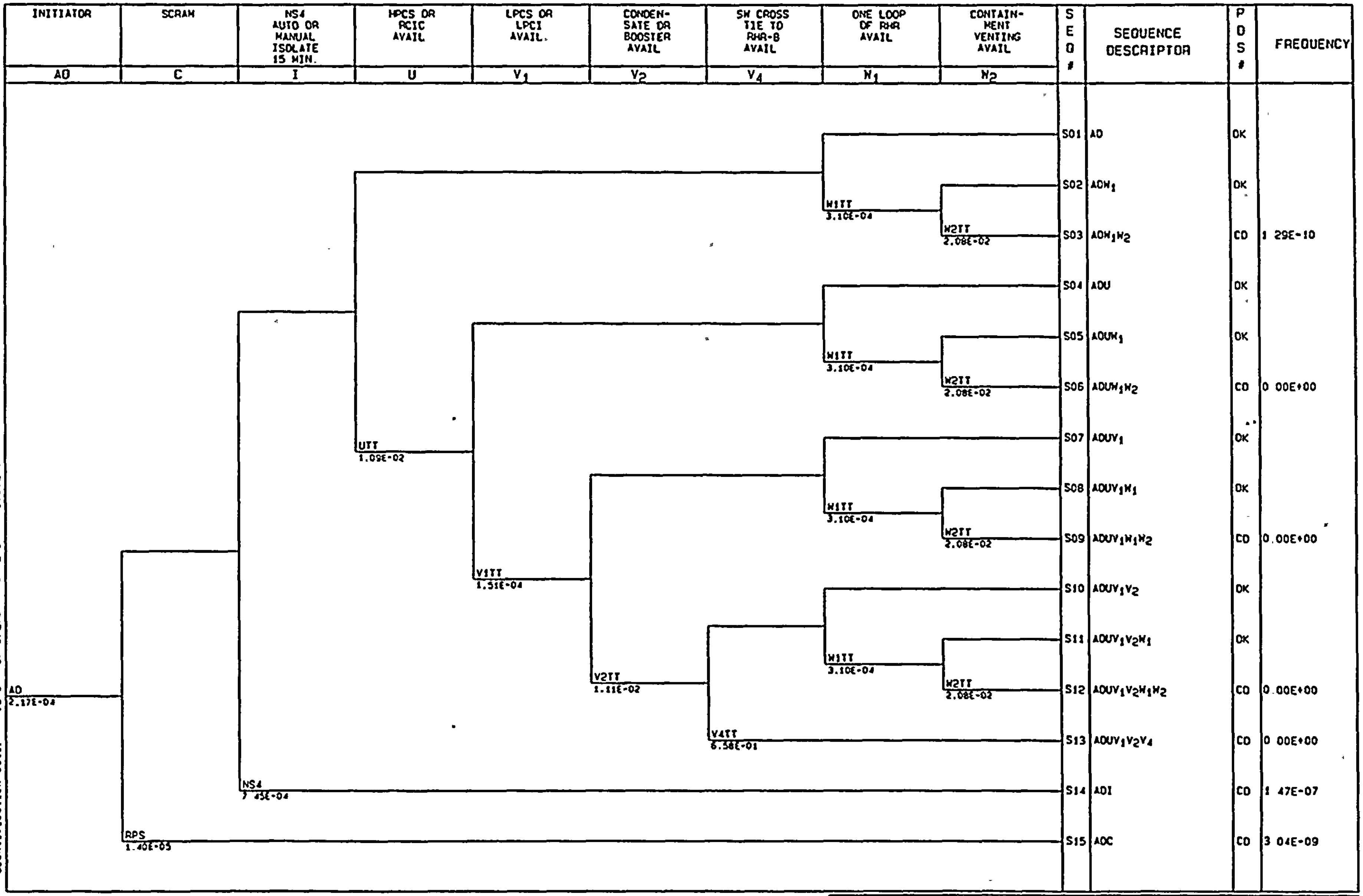
LARGE LOCA EVENT TREE

FIGURE 3.1.2.2-3B

AUTHOR__________ DATE______

REVIEWER__________ DATE______

E:\IPE2\ET\AO.EVT 6:52:24am 4-29-94 NUPRA 2.1a WPPSS
Quantification Date: 4-02-94 8:45:27am TOTAL CMF = 1.50E-007

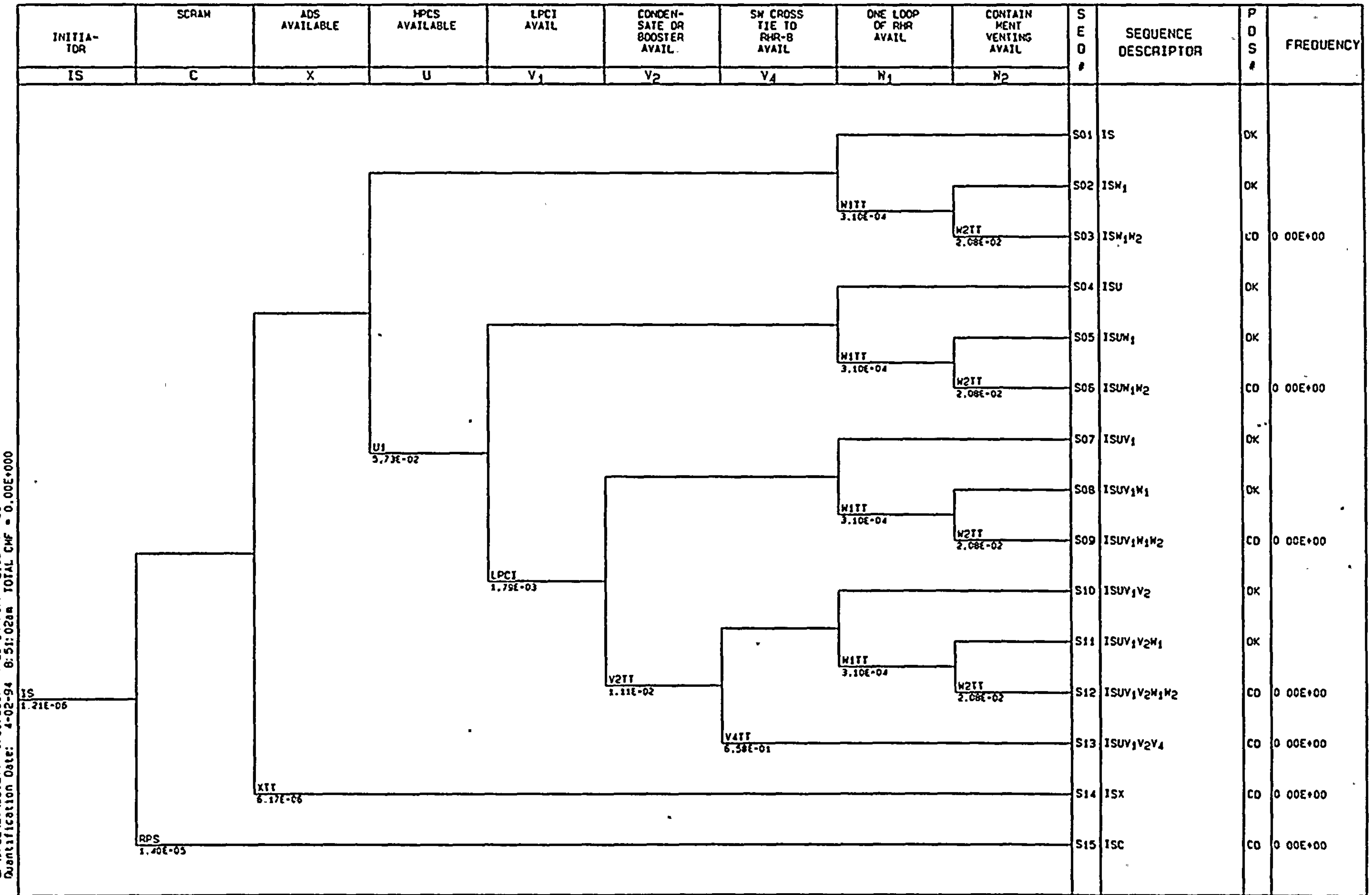


LARGE LOCA OUTSIDE CNTMT. EVENT TREE
FIGURE 3.1.2.2-4

AUTHOR__________ DATE______

REVIEWER__________ DATE______

E:\IPE2\ET\IS.EVT 6:33:58am 4-29-94 NUPRA 2.1a WPPSS
Quantification Date: 4-02-94 8:51:02am TOTAL CMF = 0.00E+000



ISLOCA EVENT TREE
FIGURE 3.1.2 2-5

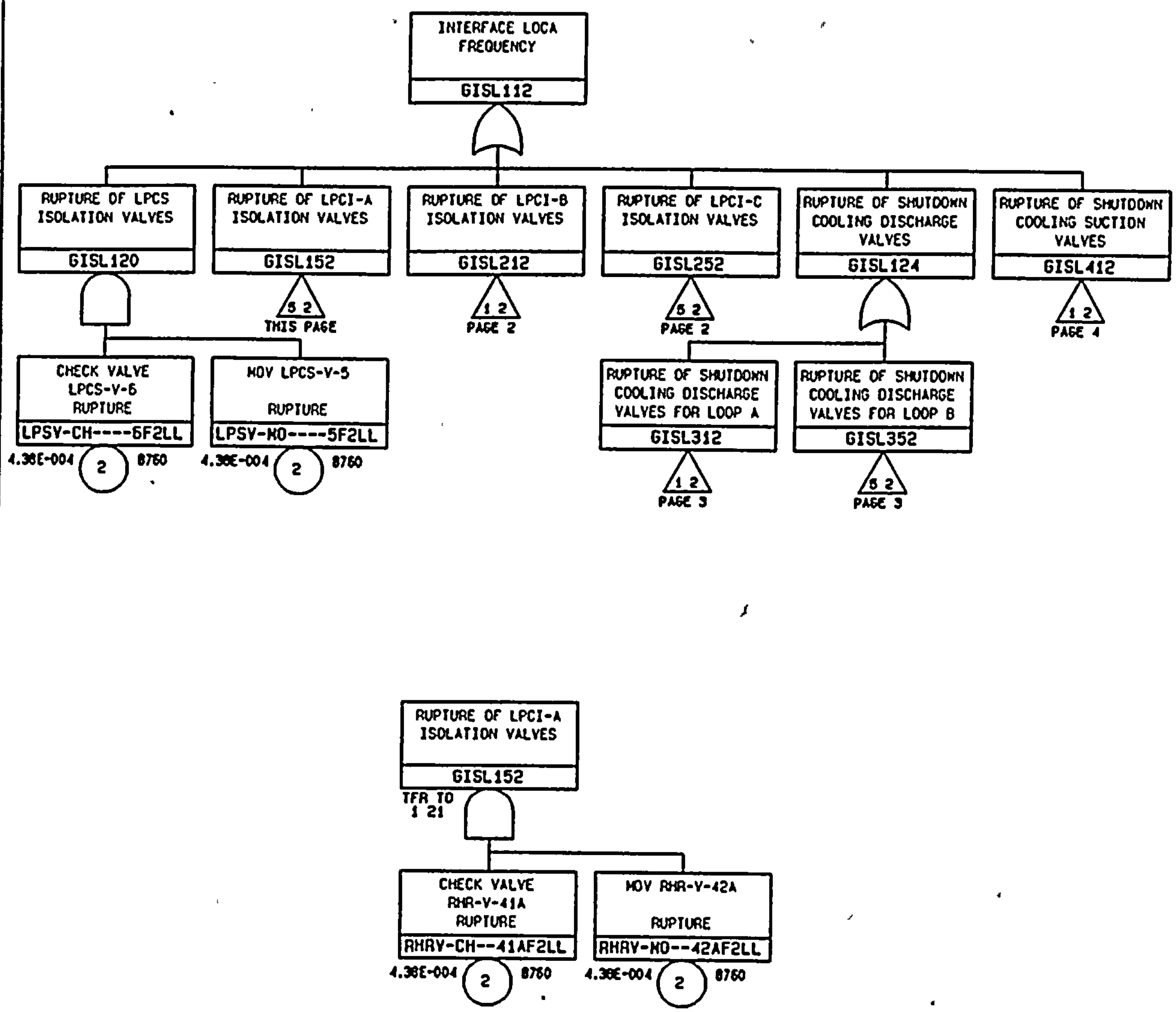
AUTHOR ____________ DATE ______

REVIEWER ____________ DATE ______

INTERFACE LOCA FREQUENCY
FIGURE 3.1.2.2-5.1

ISL 1

ANALYST: DPC	CREATION DATE: 02-15-1992	REVISION : 05-24-94

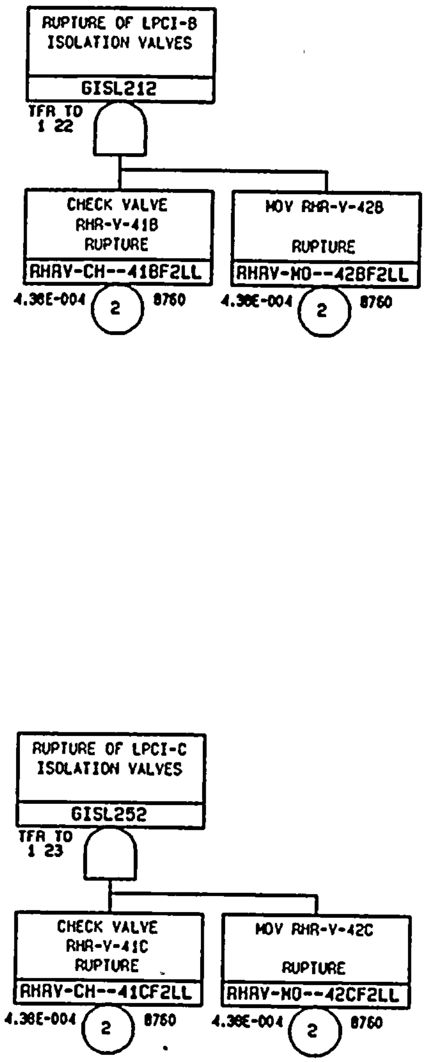


ISLOCA.LGC NUPRA 2.1a WPPSS

INTERFACE LOCA FREQUENCY
FIGURE 3.1.2.2-5.1

ISL 2

ANALYST: DPC | CREATION DATE: 02-15-1992 | REVISION : 05-24-94

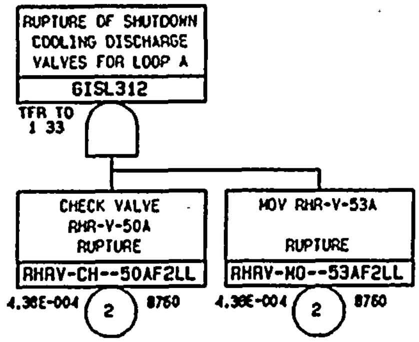


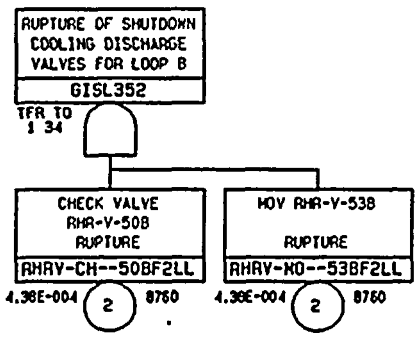
ISLOCA.LGC NUPRA 2.1a WPPSS

INTERFACE LOCA FREQUENCY
FIGURE 3.1.2.2-5.1

ISL 3

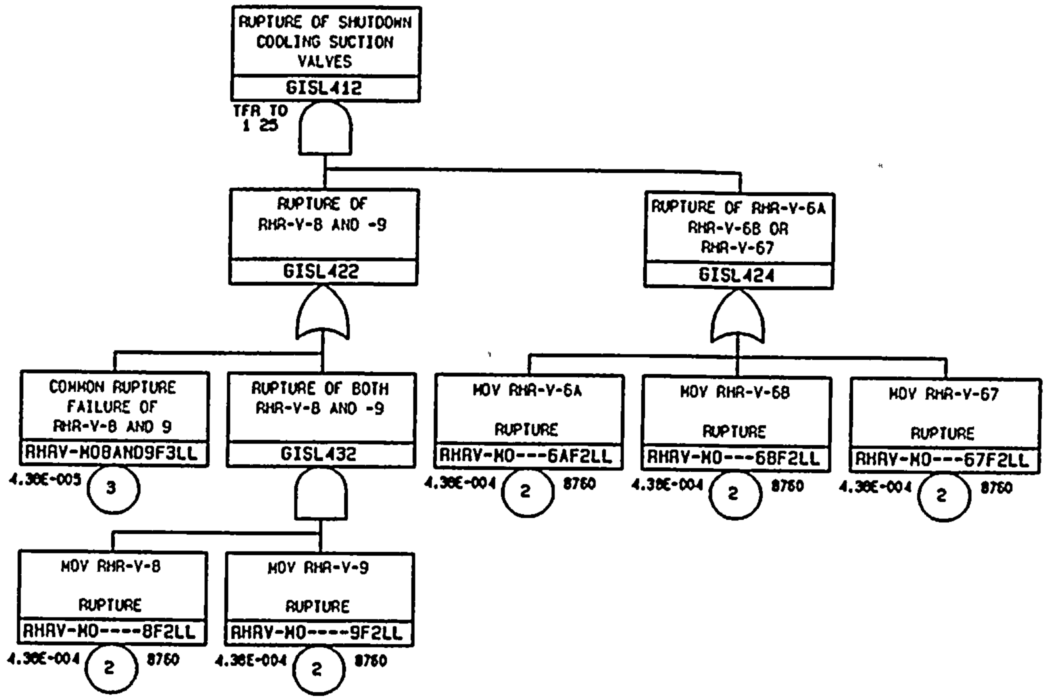
ANALYST: DPC	CREATION DATE: 02-15-1992	REVISION : 05-24-94





ISLOCA.LGC NUPRA 2.1a WPPSS

INTERFACE LOCA FREQUENCY FIGURE 3.1.2.2-5.1	ISL 4
ANALYST: DPC / CREATION DATE: 02-15-1992	REVISION : 05-24-94



ISLOCA.LGC NUPRA 2.1a WPPSS

3.1.2.3 ATWS Event Trees

This section discusses the event sequences of anticipated transients (turbine trip, MSIV closure, loss of feedwater, loss of condenser, and SORV) coupled with a failure to SCRAM. Since each of these transients has a different interaction with the mitigating systems available, a unique ATWS event tree is developed for each transient initiator as follows:

Transient	ATWS Event Tree Figure Number
Turbine Trip with Bypass, 100% Power	3.1.2.3-1A
Turbine Trip with Bypass, 25% Power	3.1.2.3-1B
MSIV Closure	3.1.2.3-2A
Loss of Condenser	3.1.2.3-3
Loss of Feedwater	3.1.2.3-4
SORV	3.1.2.3-5

Lower frequency initiating events (loss of offsite power, LOCA, loss of DC, loss of plant service water, loss of standby service water, loss of containment instrument air, loss of control and service air) coupled with a failure to SCRAM are assumed to lead directly to core damage. The discussions for these sequences are not provided in this section but are found in the event tree discussions corresponding to these initiators.

One principal distinction is made in the ATWS analysis between transients which proceed with the condenser available as a heat sink and those in which the condenser is unavailable. In a Turbine Trip, the most frequent transient, the condenser is available most of the time. Similarly, in a SORV transient, the pressure regulator senses the nuclear system pressure decrease and within a few seconds closes the turbine control valve far enough to stabilize the reactor vessel pressure. It is unlikely that the MSIVs will close due to low pressure in a SORV transient and the condenser will therefore remain available. On the other hand, MSIV Closure and Loss of Condenser are transients in which the condenser is not available. Also, in a loss of feedwater transient, the MSIVs are conservatively assumed to close due to low reactor water level (L2). The condenser is, therefore, assumed to be unavailable in a loss of feedwater ATWS transient.

For each transient initiator, a distinction can also be made regarding the power level prior to the transient. For transients initiated from low power (i.e., less than 25%), the Turbine Bypass system can handle all of the heat from the reactor core, allowing the operating staff to stabilize reactor power without SLC and to shutdown by individually inserting rods. For the MSIV Closure ATWS initiated from any power level, steam is dumped to the suppression pool via the SRVs. Since the RHR can only remove 0.74% of rated power with two trains available, the operating staff has to initiate SLC or manually insert rods to prevent core damage in essentially all instances. No credit for manually inserting rods was taken in the IPE analysis.

The WNP-2 plant response to an ATWS involves the reactor operators and the operation of a number of plant systems. The basic functions required to achieve a safe and stable condition are:

- reactivity control,
- primary system pressure control,
- system inventory control,
- containment heat removal.

In lieu of successful RPS SCRAM system operation, WNP-2 has alternate means of controlling reactivity, namely with Alternate Rod Insertion (ARI), ATWS Recirculation Pump Trip, EOC Recirculation Pump Trip (RPT), and Standby Liquid Control. These features are incorporated into the ATWS event trees as described below.

3.1.2.3.1A <u>Turbine Trip ATWS from 100% Power</u>

The applicable event tree for turbine trip ATWS from 100% power is provided in Figure 3.1.2.3-1A.

A best-estimate turbine trip ATWS simulation was performed using the RETRAN code. For an ATWS with bypass from 100% power, reactor level falls rapidly after the initial swell. L8 is approached but is never reached. The RFW system is on single element control, injecting water into the core. The turbine stop valve closes with a 0.1 second stroke time maximizing the pressure increase effects of the closing valve. When the turbine stop valve has closed 5%, a coincident trip initiates a SCRAM signal and a signal to trip recirculation pumps to LFMG. Insertion of all control rods was assumed to fail, and the reactor continues to operate at full power. The rapid increase in reactor pressure generates a rapid increase in reactor core power due to collapsing core voids. As reactor pressure increases to the safety/relief valve setpoints, the relief functions operate and four banks of relief valves open in order. Four banks of relief valves are open within 3 seconds. The steam dome pressure peaks at 1134 psia and the steam bypass valves open. The steam bypass capacity is sufficient (along with relief valve capacity), to limit the steam dome pressure peak to less than the ATWS high pressure trip setpoint. Therefore, the ATWS mitigation systems (ARI and RPT) are not predicted to be activated for this transient scenario. The relief valves begin to close and reactor power reaches a new semi-steady-state power level of approximately 40% of rated power. The RETRAN transient calculation was terminated at 30 seconds.

For the turbine trip ATWS, 4 groups of SRVs (total of 14 SRVs) must open during the initial reactor pressure rise. Three groups of SRVs (total of 12 SRVs) will reclose. During the second reactor pressure rise, group 2 (4 SRVs) will reopen and reclose. Group 1 (2 SRVs) will reclose when the reactor pressure drops to 1076 psig. If any of the required number of SRVs fail to open, the primary system boundary may breach with consequential core damage. If any of the required number of SRVs fail to reclose, the sequence is transferred to the SORV ATWS event tree (Section 3.1.2.3.5). The functional headings in the turbine trip ATWS event tree (Figure 3.1.2.3-1A) are discussed below.

TTC - Turbine Trip ATWS from Full Power With Bypass Initiator

Eighty-two percent of the turbine trip initiators occur at greater than 25% rated power and therefore the initiating event frequency is 2.7/yr (82% * 3.3 turbine trips per year).

C_M - Reactor Protection System (Mechanical)

The RPS is divided into mechanical and electrical functions for the purposes of this analysis. The mechanical function includes the operation of the CRD hydraulic system, the physical insertion of the control rods, and other mechanical parts as required. A value of 4E-6 per demand is used for the mechanical failure probability of the RPS.

C_E - Reactor Protection System (Electrical)

This portion of the RPS includes proper generation of a SCRAM signal from the sensors, logical processing of the signal, and the de-energizing of the SCRAM solenoids. A value of 1E-5 per demand is used for the electrical failure probability of the RPS.

R - Recirculation Pump Trip

There are 2 RPT systems: 1) ATWS-RPT, and 2) EOC-RPT which both function to trip the recirculation pumps. The ATWS-RPT shares the same trip inputs (low reactor water level or high reactor pressure) as the ARI system. The two events for which the EOC-RPT initiates are the closure of the turbine throttle valves and fast closure of the turbine governor valves. RETRAN analysis of the turbine trip ATWS indicates that the reactor power will drop to 40% following EOC-RPT initiation. Following successful RPT, alternate rod insertion will be initiated. If ARI is subsequently successful, the accident sequence transfers to the turbine trip event tree. If ARI subsequently fails, standby liquid control will be manually initiated to further reduce reactor power.

In the event that RPT fails, alternate rod insertion will also be initiated. If ARI fails, the operating staff can adjust feedwater flow to the vessel to control level, reduce power and maintain the core in which the fuel is cooled adequately and the energy is rejected to suppression pool. This will lead to a rapid increase in suppression pool temperature beyond the capability of the suppression pool cooling system, and eventually cause containment failure. Failure of containment may then cause loss of injection and core melt. Such a failure of all injection is assumed to occur with a 0.33 likelihood as discussed in Section 3.1.2.1 under the W_2 - Containment Venting Available subject heading. The availability of the feedwater system, however, makes the loss of injection less likely since the pumps are located outside of the Reactor Building. For simplicity, the 0.33 likelihood for failure of makeup was used despite its conservatism. The unavailability of the RPT is based on the RPT fault tree developed in its system notebook.

K - Alternate Rod Insertion

The function of the ARI is to vent the SCRAM air header for control rod insertion in the event of the rods fail to fully insert due to non-mechanical reasons such as improper generation of the SCRAM signal, RPS logic failures, or failure to de-energize the SCRAM solenoids. The ARI function is not questioned for accident sequences initiated by the mechanical failure of the RPS.

The unavailability of the ARI is based on the ARI fault tree developed in its system notebook. Identical basic event names are used in the ARI and RPT fault trees to account for the same trip inputs (low reactor water level or high reactor pressure).

M - Safety Relief Valves Open

Functional event M represents the opening of the safety relief valves to limit the reactor coolant pressure to within the primary boundary design pressure (110% system pressure). Failure of a sufficient number of valves to open may lead to excessive pressure and a potential LOCA condition. For the turbine trip ATWS, it is conservatively assumed that 17 of the 18 valves are required to open initially to be successful. The failure of the SRVs to open is represented by a single common mode failure event with a value of 5E-5. This value was derived from a common mode failure analysis of NPRDS data.

P - Safety Relief Valves Reclose

The safety relief valves that open as a result of the turbine trip ATWS must reclose to prevent 1) discharge of an excessive quantity of reactor coolant and 2) excessive heat to the suppression pool. Failure to reclose the safety relief valves is transferred to the SORV ATWS event tree. A study of ATWS events performed for WNP-2, estimates a total of 68 valve reclosures during pressure cycles for a MSIV closure ATWS before the reactor power stabilizes to 5%. It is assumed that if the valves reclose during the first cycle, they will reclose during the subsequent cycles if required. Therefore, the probability of a SRV not reclosing is $1.83 \times 10^{-2} \times 17 = 0.31$. Based on the IDCOR Technical Report 86.3B1 which found that 85% of SORVs will reclose when the reactor pressure drops below 200 psig, the probability of SRVs failing to reclose is $0.31 \times 0.15 = 4.67 \times 10^{-2}$.

C_3 - Standby Liquid Control Available

Two SLC pumps can provide an equivalent flow of 86 gpm of 13% sodium pentaborate solution by discharging into the HPCS header. The boron solution is thus into the reactor vessel upper plenum region, directly onto the top of the core. The SLC injection will bring the reactor power to 5% in a very short time once initiated and provides negative reactivity insertion in excess of the reactivity increase caused by the cooldown and decay of fission products. The WNP-2 Emergency Operating Procedures directs the operator to inject boron into the RPV with SLC if the reactor cannot be shutdown before suppression pool temperature reaches 110°F.

Two ATWS scenarios were simulated using MAAP. For conservatism, both simulations involved MSIV closure. The timing for events in the scenarios therefore represent a lower bound and the timing for other ATWS scenarios such as turbine trip would be longer. The first case is an MSIV closure ATWS with successful RPT but without SLC. The second case is a MSIV closure ATWS with RPT, without SLC and without coolant injection. For the first case, drywell pressure reaches 121 psig in approximately one hour. According to the structural analysis performed for the WNP-2 containment, the containment failure pressure is 121 psig as discussed in Section 4.3.

For the second MAAP simulation, the water level reaches TAF in 4.4 minutes and 2/3 active fuel in 5.1 minutes. The core melts, and the RPV breach occurs at 1.9 hours. There is a limiting time in the core melt progression beyond which coolant injection will not prevent core melt. This is taken to be the time when core plate failure occurs and it is estimated to be 1 hour. According to the SLC system notebook, it will take 54 to 61 minutes to inject the full tank of sodium pentaborate solution to the core at 43 gpm with one pump. At 86 gpm with two pumps it will take half of the time or 27 minutes. These times are sufficient to maintain the reactor subcritical during cooldown following the initial power reduction. Reactor power, and therefore steam discharge to the containment through the SRVs, will decrease as sodium pentaborate solution is being injected. To prevent containment overpressurization, SLC must be initiated within approximately 30 minutes for an MSIV closure analysis. In the analysis, it is conservatively assumed that SLC must be initiated within 20 minutes. For a turbine trip ATWS at 25% power with turbine bypass, conditions are less severe because steam is released to the condenser. The required SLC initiation time is conservatively assumed to be 40 minutes. SLC system unavailability is determined from the fault tree model provided in its system notebook.

If SLC is unsuccessful following a turbine trip ATWS, the core will continue to operate in a quasi-steady state condition at a nominal power level of 40% due to successful RPT operation. To maintain adequate core cooling at this power level it is necessary to maintain 40% reactor feed flow. Since none of the ECCS systems have a capacity which approaches this level, the core cooling can only be stabilized at this level if main feedwater is available. The SRVs have adequate capacity to handle all of this power and reject the necessary energy to the suppression pool, but in all likelihood the turbine bypass system will also be used to reject 25% power to the main condenser. This means that 15% will be rejected to the suppression pool, if uncorrected it will eventually increase in temperature to the point that the containment is threatened by overpressure. Overpressure failure of containment may then cause loss of injection and core damage. As discussed under the W2 - Containment Venting Available subject heading in Section 3.1.2.1, loss of injection from high pressure sources due to containment failure is modelled to occur with a 33% likelihood in the analysis. As discussed in Section 3.1.2.1, the 33% likelihood is considered conservative in this instance because feedwater is available. Unlike the ECCS pumps, the feedwater pumps are located outside of the Reactor Building and thus are not as likely to be affected by containment failure.

AI - ADS Inhibit and Maintain Low Level

Following SLC initiation, the Emergency Operating Procedures direct the operating staff to:

- inhibit ADS to prevent reactivity addition and boron dilution by the low pressure coolant injection systems,
- throttle injection into RPV except SLC and CRD until the top-of-active-fuel level is reached.

For automatic ADS actuation to occur, low reactor level signals L3 and L1 must exist, a 105 second time delay must be reached, and one low pressure ECCS pump must be running. The conditions for automatic ADS actuation may occur in such scenarios due to water levels controlled by the operators per WNP-2 emergency procedures. ADS inhibit is easily achieved by turning the control switches to INHIBIT. To throttle injection into the RPV, the operating staff would adjust feedwater flow if it is operating. If feedwater is unavailable, the operating staff must control HPCS or RCIC flow.

The above tasks can be performed in the control room. The operating staff is trained on these tasks but does not routinely perform them. Since the operator is already cued to initiate SLC by the same procedure, he will perform these tasks as follow-ons. MAAP results for a turbine trip without bypass ATWS indicate no big power surge or high vessel pressure beyond ASME Code limit even when all injection systems are operating. The failure probability to inhibit ADS and lower reactor water level during ATWS is calculated to be 5E-2 in the human reliability analysis.

Failure to inhibit ADS results in an accident sequence similar to that for failure of SLC as discussed above under the C_3 - Standby Liquid Control Available subject heading.

QU - High Pressure Coolant Injection

The Emergency Operating Procedures direct the operator to re-establish water level to between L3 and L8 after SLC operation is completed to allow sufficient mixing of the injected boron.

To re-establish water level, the operating staff can use RFW, HPCS or RCIC. Since there is ample time available for the operating staff to re-establish water level, the human error probability is negligible. Re-establishing water level is limited by the RFW, HPCS, and RCIC system unavailabilities obtained from the fault trees provided in their respective notebooks.

X - Depressurization

If the operating staff cannot re-establish water level using the RFW, HPCS or RCIC, the reactor must be depressurized to allow the use of low pressure systems. See subject discussion in Section 3.1.2.1.

V - Low Pressure Coolant Injection

After depressurization, the operating staff has the option of using Condensate, LPCS, LPCI, or the SW Crosstie. See subject discussions in Section 3.1.2.1.

W_1 - RHR Available

For a turbine trip ATWS with bypass from 100% power, successful recirculation pump trip to LFMG will lower the power level to 40%. Since 25% power goes to the condenser through the turbine bypass system, only 15% of reactor power is dumped to the suppression pool through safety relief valves. To bring the single-phased pool water temperature from 90°F to saturation temperature at 45 psig (containment design pressure) with 15% rated power, approximately 3 hours is required. Boron injection to the reactor vessel is accomplished in about one-half hour following SLC initiation and the reactor becomes subcritical. This allows the operator over 2 hours to initiate RHR before the suppression pool reaches the saturation temperature at 45 psig.

The human error in initiating RHR in this time period (> 2 hr) is negligible. RHR system unavailability is determined from the fault tree provided in its system notebook.

Z - MSIVs Open and PCS Available

See subject discussion in Section 3.1.2.1.

W_1 - Containment Venting Available

See subject discussion in Section 3.1.2.1.

3.1.2.3.1B Turbine Trip ATWS 25% Power

For turbine trip ATWS events at 25% power or less, the recirculation pumps are on the low frequency motor generator set. Most of the reactor power in this situation comes about from continued withdrawal of the control rods rather than from the recirculation flow. Therefore, RPT is relatively unimportant in a turbine trip ATWS with bypass from 25% rated power. For this case the turbine bypass system can handle all power generated from the reactor core. However, the operator does have to maintain the core inventory. If SLC, ADS-inhibit, and level control are successful, the operator can use RFW, HPCS, RCIC (or LPCS, LPCI, Condensate, SW Crosstie after depressurization). If SLC, ADS-inhibit or level control is not successful, reactor power level will remain at 25%. The operating staff can use the feedwater, HPCS, RCIC, and power conversion systems to stabilize the plant condition. The event tree for the turbine trip ATWS with bypass from 25% rated power is shown in Figure 3.1.2.3-1B. Functional headings for the event tree are the same as those for the 100% case with the exception that the RPT is deleted.

3.1.2.3.2 MSIV Closure ATWS

A best-estimate MSIV closure ATWS simulation from 100% power was performed using the RETRAN code. The MSIVs close with a three second stroke time (minimum valve stroke time), to maximize the pressure increase effects of the closing valves. When the MSIVs have closed 10%, a coincident trip initiates a SCRAM signal. The SCRAM was assumed to fail to insert any control blades and the reactor continues operating at power. Also at this time, loss of steam flow occurs because MSIV closure initiates a FW pump coastdown.

The rapid increase in reactor pressure generates a rapid increase in reactor core power due to collapsing core voids. As reactor pressure increases to the safety/relief valve setpoints, the relief valves operate and all five banks of relief valves open in order. Increasing reactor pressure trips the ATWS high pressure trip setpoint and results in a trip of the recirculation pumps. However, the ARI system has a 15 second delay time that must time out before the SCRAM header vents and the control blades can actually start to be inserted into the core. The void reactivity effect of the pressure relief provided by all the relief valves opening and the voids generated by rapidly rising core power limit the core power rise. The reactor vessel pressure follows the same response pattern. In accordance with the Emergency Operating Procedures, the operating staff will initiate suppression pool cooling using 2 loops of RHR. ARI is assumed failed and the falling reactor vessel level initiates both HPCS and RCIC on a level 2 trip signal. Before the suppression pool reaches 110°F, the operating staff must initiate SLC boron injection with two pump operation. As SLC boron injection flow is a relatively low flow rate, the operating staff is directed to throttle injection into RPV except SLC and CRD until top of active fuel (TAF) is reached. As vessel water level decreases towards TAF, the resulting increase in core voids will minimize fission power until SLC can inject enough boron into the reactor vessel to shut down the core.

The core decay heat is approximately 2.0% of full power, or 66.5 MWt. Two safety/relief valves cycle to remove decay heat to the suppression pool. The RHR heat exchangers are removing decay heat from the suppression pool. Peak pressure in the bottom of the reactor vessel is below the ASME code limit of 1375 psig for the reactor coolant pressure boundary.

Due to MSIV closure and the resulting loss of the main condenser as the sink for removing reactor heat, the suppression pool receives SRV blowdown for the entire transient. The suppression pool temperature rise and the wetwell maximum pressure do not challenge the containment wetwell design limits for an MSIV closure ATWS with successful SLC initiation.

If RPT succeeds following an MSIV closure ATWS, alternate rod insertion will be initiated. If ARI subsequently fails, standby liquid control is initiated. In the event that SLC is unable to provide boron solution to the reactor, reactor power will not be controlled at a level within the capacity of the available makeup systems (feedwater is not available). Therefore, reactor vessel level will drop and the core will eventually be uncovered and damaged. If RPT fails following a MSIV closure ATWS, the scenario is similar to the sequence described in the preceding paragraph. However, since the core energy level is initially much higher core melt will occur earlier.

For the MSIV Closure ATWS, 5 groups of SRVs (total of 18 SRVs) open during the initial reactor pressure rise. Four groups of SRVs (total of 16 SRVs) will reclose. During the 2nd to 7th reactor pressure rises, group 2 (4 SRVs) will reopen and reclose. During the 8th reactor pressure rise, group 1 (2 SRVs) will reopen and reclose. During the 9th and 10th reactor pressure rises, both group 1 and 2 (total of 6 SRVs) will reopen and reclose. After the 10th rise, group 1 (2 SRVs) will continue to cycle open and close relieving decay heat to the suppression pool. If any of the required number of SRVs fail to open (the probability of which is very low because of the large number of SRVs available for relief function), primary system boundary may breach with consequential core damage. If any of the required number of SRVs fail to reclose, the sequence is transferred to the SORV ATWS event tree (Section 3.1.2.3.5). The functional headings in the MSIV closure ATWS event tree (Figure 3.1.2.3-2A) are discussed below.

T_M - <u>MSIV Closure Initiator</u>

See subject discussion in Section 3.1.2.1.3.

C_M - <u>Reactor Protection System (Mechanical)</u>

See subject discussion in Section 3.1.2.3.1A.

C_E - <u>Reactor Protection System (Electrical)</u>

See subject discussion in Section 3.1.2.3.1A.

R - <u>Recirculation Pump Trip</u>

See subject discussion in Section 3.1.2.3.1A. RETRAN results of the MSIV Closure ATWS indicates that the reactor power will drop substantially following RPT, feedwater coastdown and SLC injection from a 100% power MSIV Closure ATWS. If RPT is unsuccessful, alternate rod insertion and SLC will be initiated.

K - <u>Alternate Rod Insertion</u>

See subject discussion in Section 3.1.2.3.1A.

M - Safety Relief Valves Open

See subject discussion in Section 3.1.2.3.1A.

P - Safety Relief Valves Reclose

See subject discussion in Section 3.1.2.3.1A.

C_3 - Standby Liquid Control Available

It is assumed that SLC must be initiated 20 minutes into the MSIV Closure ATWS event to prevent core damage. If SLC is unsuccessful, reactor power will exceed the capability of the available makeup systems to maintain level (feedwater is not available). The core will be uncovered and damaged. For the purposes of this study it was assumed that the core would melt before the energy rejected to containment threatened containment integrity.

AI - ADS Inhibit and Maintain Low Water Level

Failure of ADS-Inhibit results in a scenario similar to that discussed in the preceding paragraph for failure of SLC.

QU - High Pressure Coolant Injection

See subject discussion in Section 3.1.2.3.1A. Since FW is unavailable, HPCS and RCIC are the only available high pressure injection systems.

X - Depressurization

See subject discussion in Section 3.1.2.3.1A.

V - Low Pressure Coolant Injection

See subject discussion in Section 3.1.2.3.1A.

W_1 - RHR Available

See subject discussion in Section 3.1.2.3.1A.

Z - MSIVs Open and PCS Available

See subject discussion in Section 3.1.2.1.3.

W_1 - Containment Venting Available

See subject discussion in Section 3.1.2.1.

3.1.2.3.3 Loss of Condenser ATWS

Following the loss of condenser vacuum at 2 inches Hg per second, the plant will respond automatically by the closure of the turbine stop valves and the operation of the turbine bypass valves. When the condenser vacuum reaches 7″ Hg, the MSIVs and the turbine bypass valves will close. This transient is similar to the MSIV Closure ATWS. The event tree is exactly the same as that for the MSIV Closure ATWS with the exception that the initiator is replaced by the loss of condenser initiating frequency. The loss of condenser ATWS event tree is shown in Figure 3.1.2.3-3.

3.1.2.3.4 Loss of Feedwater ATWS

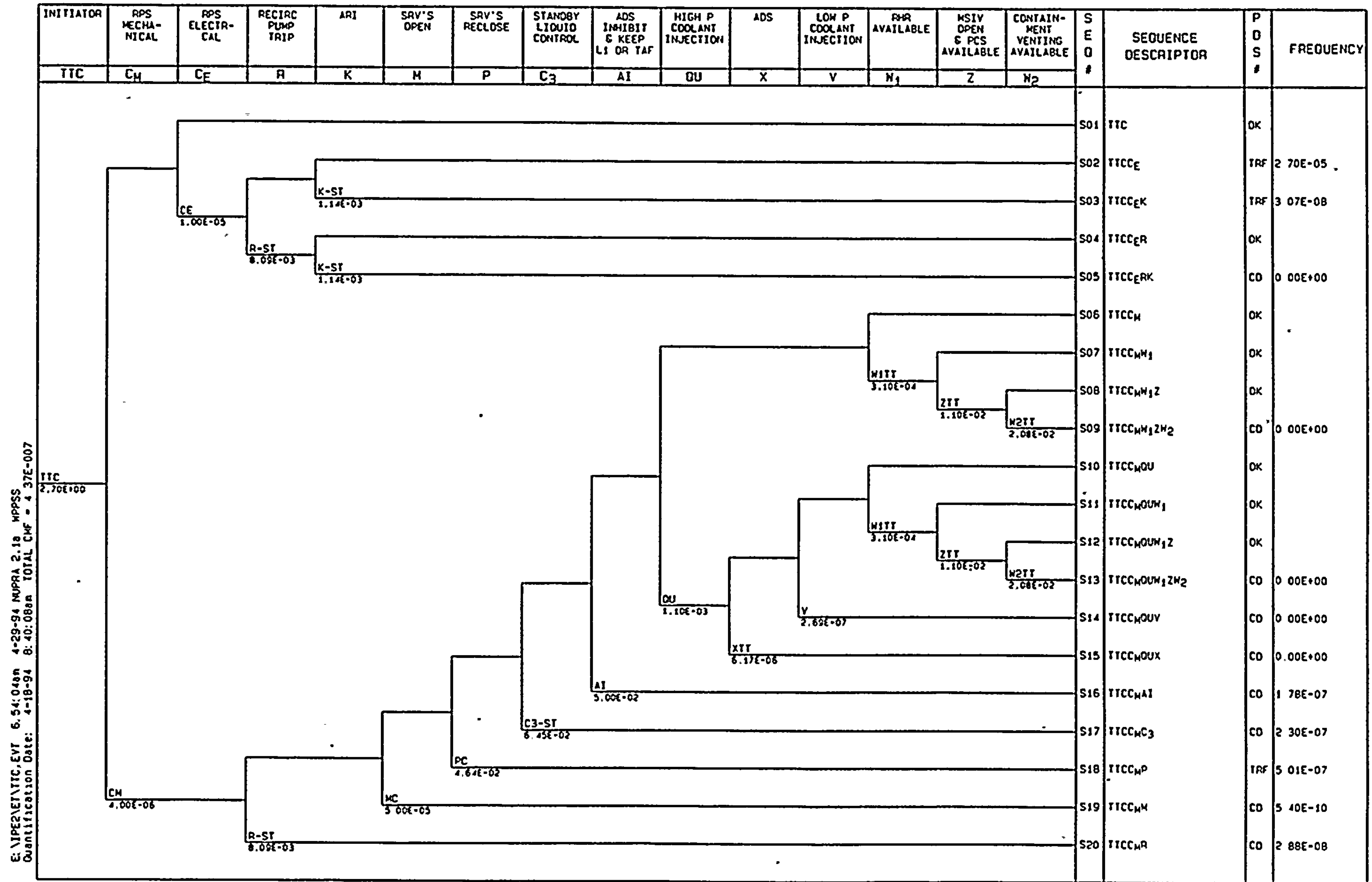
The loss of feedwater ATWS event tree is exactly the same as that for the MSIV Closure ATWS with the exception that the initiator is replaced by the loss of feedwater and that the condenser is available with a failure probability of 2.5%. The loss of feedwater ATWS event tree is shown in Figure 3.1.2.3-4.

3.1.2.3.5 Stuck Open Relief Valve ATWS

Indications of a stuck open relief valve are high SRV tailpipe temperature, high acoustic monitor indication, or high suppression pool temperature. Operators are directed to reclose the valve as soon as possible and check that the reactor and turbine generator output return to normal. If the valve cannot be closed in 2 minutes, the operator will place the reactor mode switch in the shutdown position to SCRAM the reactor. Because of the nature of the SORV challenge, the RPS and the EOC-RPT will not be actuated automatically as in the turbine trip and the MSIV Closure ATWS event sequences. If the reactor is not manually tripped, it will SCRAM eventually due to high drywell pressure.

If the reactor fails to SCRAM, but the recirculation pump trip is successful, the event can be mitigated by using SLC. It is assumed in the quantification that the necessary operator actions and timing are sufficiently characterized by those for the Turbine Trip ATWS. This is because the MSIVs remain open and the turbine bypass systems are available to transfer heat to the condenser.

The SORV ATWS event tree is shown in Figure 3.1.2.3-5. The initiating event frequency for TIC, 4E-3/yr, is determined from the combination of the stuck open relief valve initiating frequency, 2E-1/yr and the 2E-2 human error probability for failure to SCRAM. Refer to Section 3.1.2.3.1A for a discussion of the SORV ATWS event tree functional events.



TURB TRIP ATWS EVENT TREE (100% POWER)

FIGURE 3.1.2.3-1A

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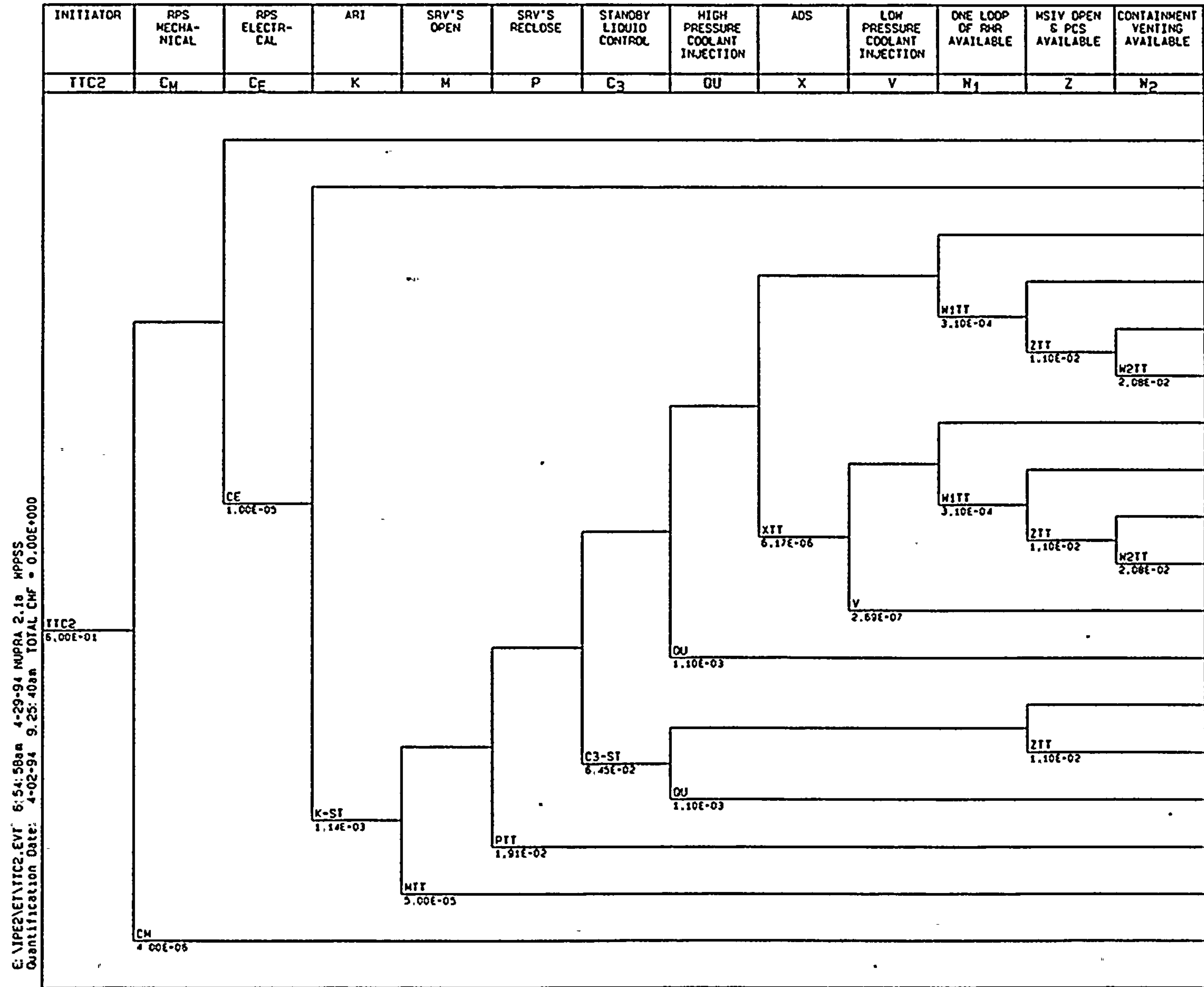
REVIEWER__________ DATE________



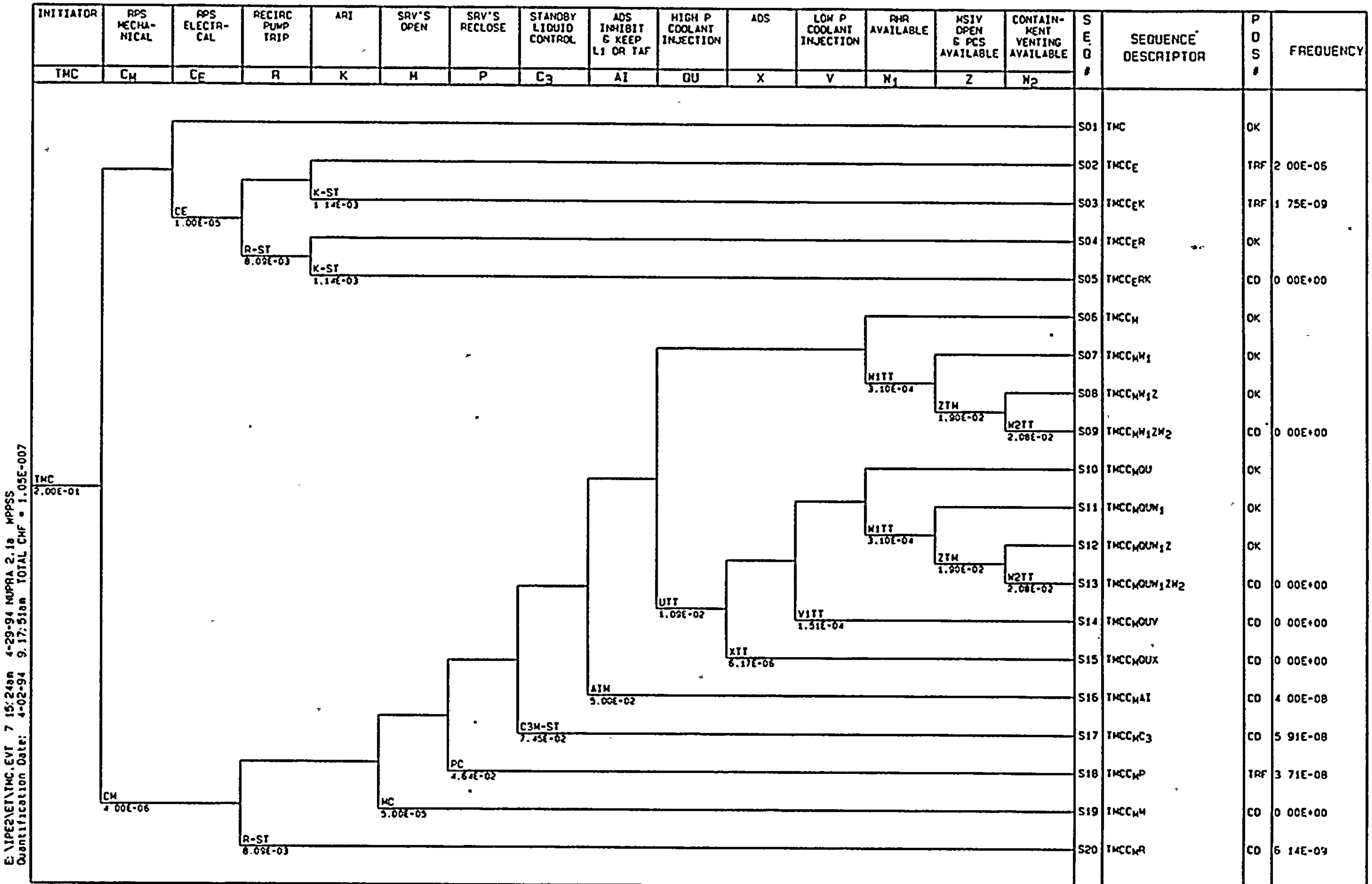
SEQ #	SEQUENCE DESCRIPTOR	POS #	FREQUENCY
S01	TTC2	OK	
S02	$TTC2C_E$	TRF	6.00E-06
S03	$TTC2C_EK$	OK	
S04	$TTC2C_EKW_1$	OK	
S05	$TTC2C_EKW_1Z$	OK	
S06	$TTC2C_EKW_1ZW_2$	CD	0.00E+00
S07	$TTC2C_EKX$	OK	
S08	$TTC2C_EKXW_1$	OK	
S09	$TTC2C_EKXW_1Z$	OK	
S10	$TTC2C_EKXW_1ZW_2$	CD	0.00E+00
S11	$TTC2C_EKXV$	CD	0.00E+00
S12	$TTC2C_EKQU$	CD	0.00E+00
S13	$TTC2C_EKC_3$	OK	
S14	$TTC2C_EKC_3Z$	CD	0.00E+00
S15	$TTC2C_EKC_3QU$	CD	0.00E+00
S16	$TTC2C_EKP$	TRF	0.00E+00
S17	$TTC2C_EKM$	CD	0.00E+00
S18	$TTC2C_M$	TRF	2.40E-06

TURB TRIP ATWS EVENT TREE (25% POWER)

FIGURE 3.1.2.3-18

AUTHOR__________ DATE______

REVIEWER__________ DATE______

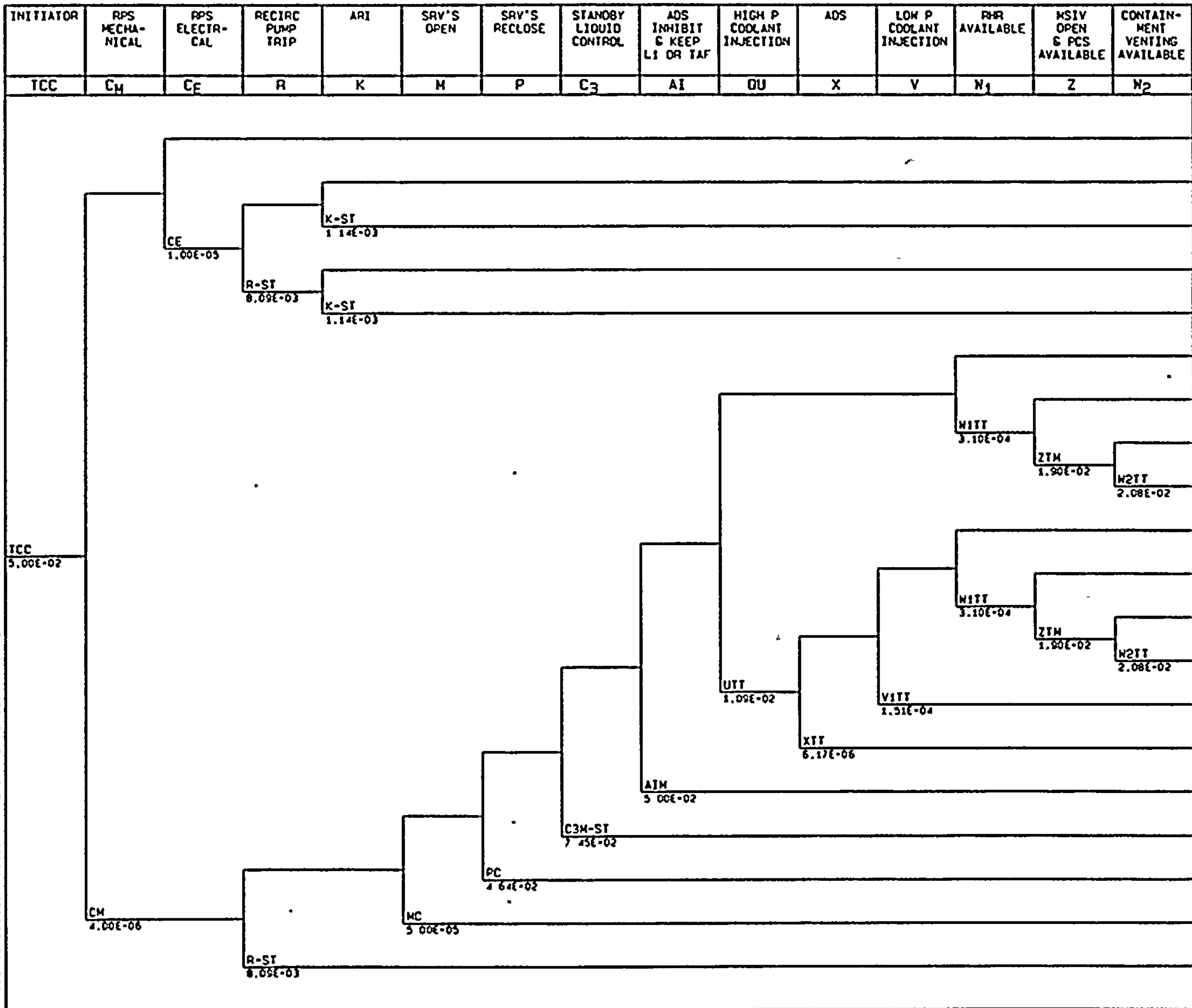


MSIV CLOSURE ATWS EVENT TREE
FIGURE 3.1.2.3-2A

AUTHOR________________ DATE________

REVIEWER________________ DATE________

E:\IPE2\ET\TCC.EVT 7:10:32am 4-29-94 NUPRA 2.1a WPPSS
Quantification Date: 4-02-94 9:05:18am TOTAL CMF = 2.50E-008



SEQ #	SEQUENCE DESCRIPTOR	POS #	FREQUENCY
S01	TCC	OK	
S02	$TCCC_E$	TRF	5 00E-07
S03	$TCCC_EK$	TRF	4 38E-10
S04	$TCCC_ER$	OK	
S05	$TCCC_ERK$	CD	0 00E+00
S06	$TCCC_M$	OK	
S07	$TCCC_MW_1$	OK	
S08	$TCCC_MW_1Z$	OK	
S09	$TCCC_MW_1ZW_2$	CD	0.00E+00
S10	$TCCC_MOU$	OK	
S11	$TCCC_MOUW_1$	OK	
S12	$TCCC_MOUW_1Z$	OK	
S13	$TCCC_MOUW_1ZW_2$	CD	0 00E+00
S14	$TCCC_MOUV$	CD	0.00E+00
S15	$TCCC_MOUX$	CD	0 00E+00
S16	$TCCC_MAI$	CD	1 00E-08
S17	$TCCC_MC_3$	CD	1 43E-08
S18	$TCCC_MP$	TRF	9 28E-09
S19	$TCCC_MM$	CD	0 00E+00
S20	$TCCC_MR$	CD	7 40E-10

LOSS OF CONDENSER ATWS EVENT TREE
FIGURE 3.1.2.3-3

AUTHOR________________DATE________

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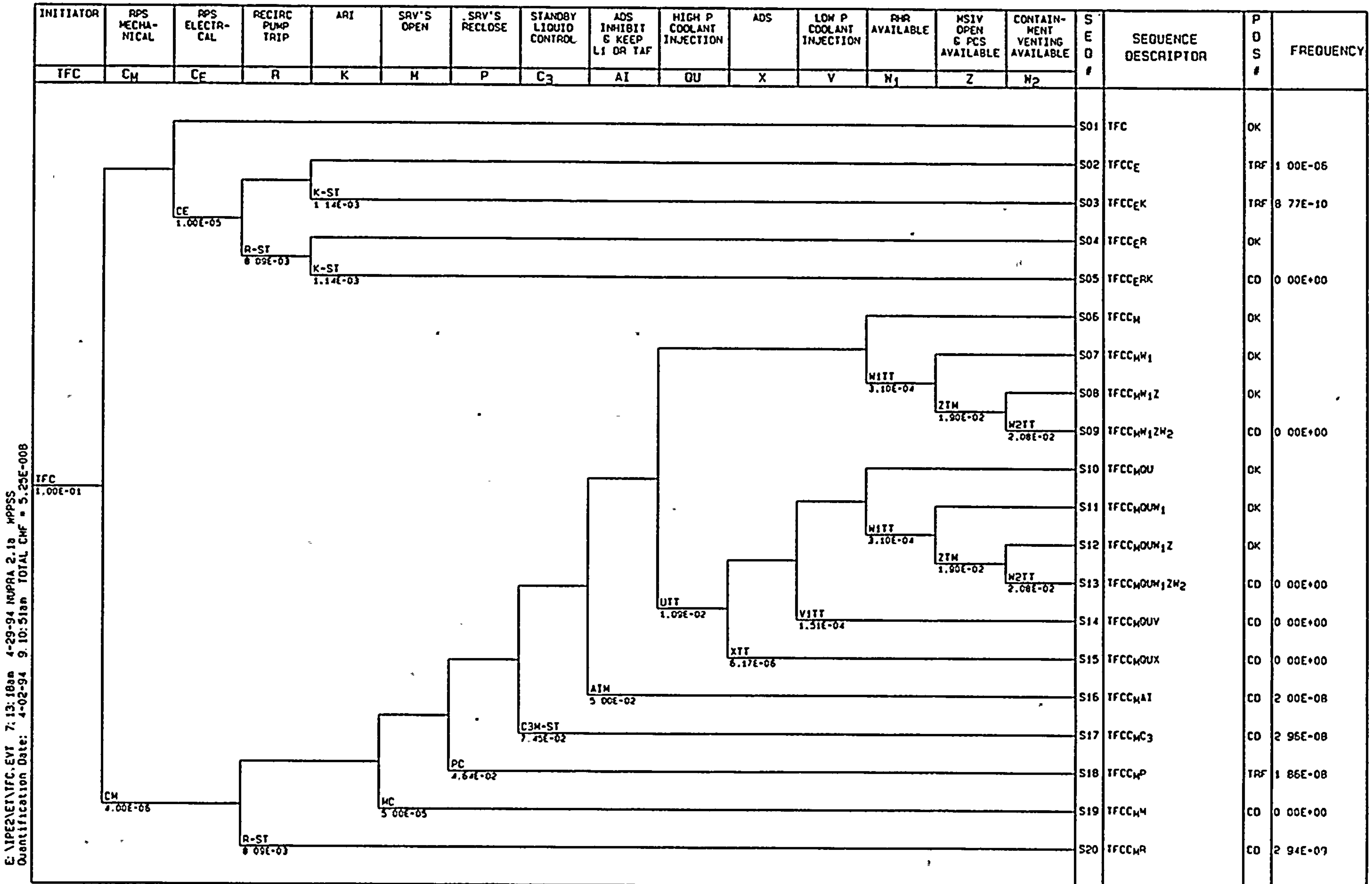
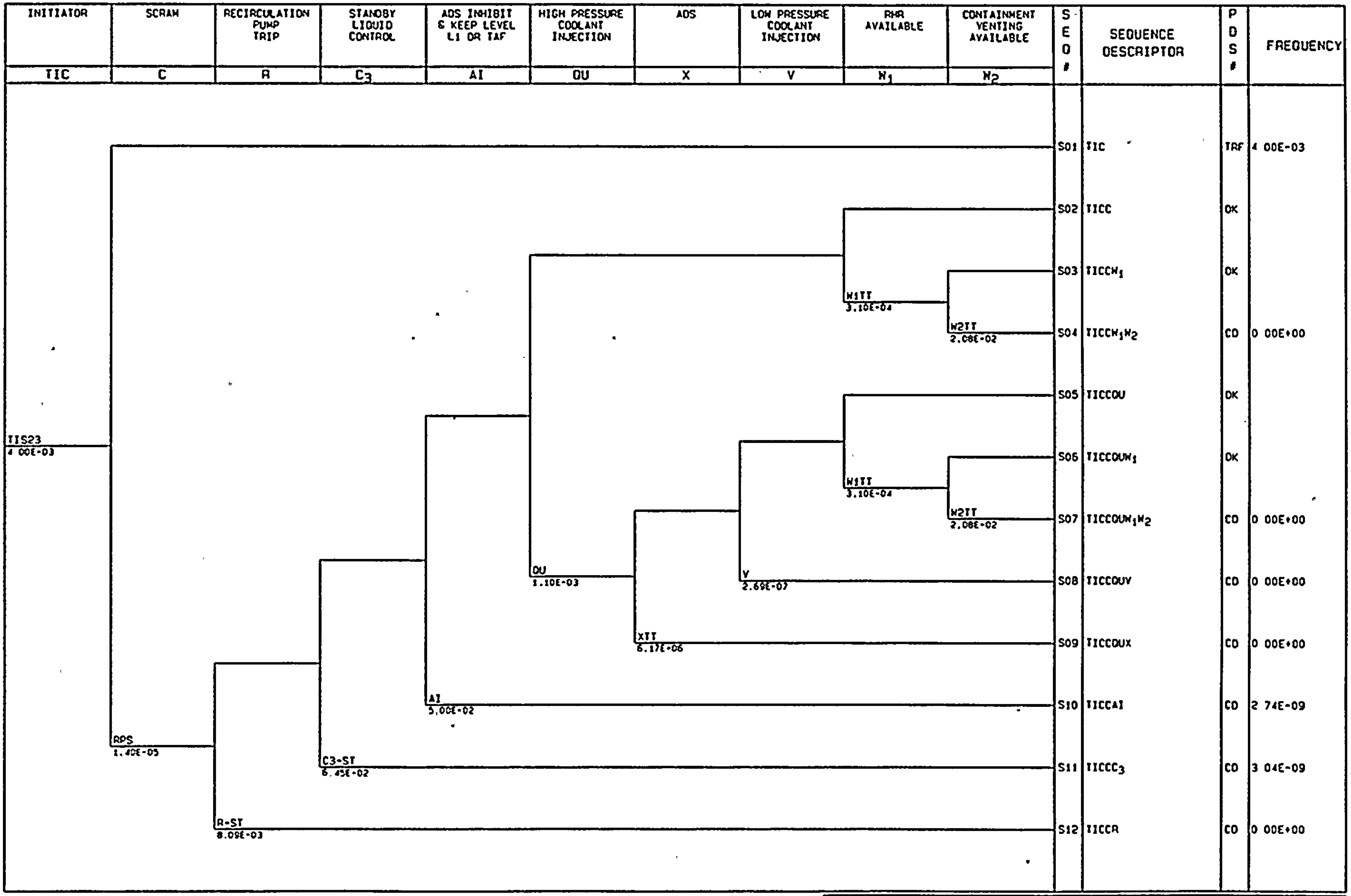


LOSS OF FEEDWATER ATWS EVENT TREE

FIGURE 3.1.2.3-4

AUTHOR____________________DATE________

REVIEWER____________________DATE________



E:\IPE2\ET\TIC.EVT 7:14:12am 4-29-94 NUPRA 2.1a WPPSS
Quantification Date: 4-02-94 9 14:37am TOTAL CMF = 5.78E-009

SORV ATWS EVENT TREE
FIGURE 3.1.2.3-5

AUTHOR________________DATE________

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3.1.2.4 Special Initiator Event Trees

Some support system failures, if they occur, can adversely impact multiple front-line systems. Internal flooding can also adversely impact multiple front-line systems. The event trees for the special initiators are developed and discussed in the following sections.

3.1.2.4.1 Loss of Division 2 DC

WNP-2's safety related DC loads are powered from 6 separate Class 1E DC power distribution systems. These systems consist of two 24 volt systems (Division 1 and 2), three 125 volt systems (Division 1, 2 and 3) and one 250 volt system (Division 1). The Division 1 and 2 24 volt DC systems provide redundant sources of DC power for the Nuclear Instrumentation, the Process Radiation Monitoring system, and selected Bypass Inoperative Status Indication systems. The Division 1 and 2 125 volt DC systems provide redundant sources of power for normal plant operation and operation of the emergency AC power systems and safety systems. The third 125 volt DC system (Division 3) is provided for the HPCS system. The 250 volt (Division 1) system provides DC power to the RCIC pump and valves, the emergency oil pumps associated with the Main Turbine and the RFW turbines, RHR-V-23 and RWCU-V-4 valves, and inverter E-IN-1.

According to the Technical Specifications, with either Division 1 or Division 2 of the DC system not energized during normal operation, the division must be reenergized within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. The impact of a loss of a single Division of DC power could be significant if a SCRAM occurs while the bus is out-of-service. At WNP-2, the loss of Division 2 DC eliminates control power for RHR-B and RHR-C, and one of two ADS and SRV channels. Plant Service Water (TSW) supplies cooling water to the RFW turbine oil coolers, condensate pumps, condensate booster pump lube oil coolers, CW pump and mechanical vacuum pump. TSW depends on Division 1 and 2 DC for control power. Therefore, the loss of Division 2 DC will have an indirect effect on reactor feedwater and the condensate systems. In the following it is assumed that Division 2 DC is out-of-service when the reactor is at 100% power and results in a reactor SCRAM. The event tree for the loss of Division 2 DC is shown in Figure 3.1.2.4-1. The functional events listed across the top of the event tree are discussed below:

T_{DC} - Loss of Division 2 DC Initiator

The initiating frequency of the loss of one Division 2 DC power coincident with or causing a reactor SCRAM, is obtained from the past performance history of DC buses in nuclear plants. The evaluated frequency is 6E-3 per reactor year. As pointed out by the NRC in NUREG-0666, operating experience has demonstrated that operator recovery is probable from DC bus faults. Based upon NUREG-0666, an operator success rate of 50% is conservatively given for conditions of a loss of DC bus coincident with a SCRAM. This rate applies to the high stress condition existing following such an incident. The calculated initiator frequency for loss of a DC bus coincident with SCRAM is then 3E-3 per reactor year.

C - Reactor Subcritical

The SCRAM system is independent of DC power. The SCRAM reliability subsequent to a loss of DC bus is the same as that for other accident sequences. See subject discussion in Section 3.1.2.1. Failure to bring the reactor subcritical is conservatively assumed to lead directly to core damage.

M - Safety Relief Valves Open

The safety mode of each SRV is actuated directly by the force exerted upon the main spring by reactor pressure. There is no dependency on the DC power for the safety mode operation. For SRV reliability, see Section 3.1.2.1.

P - Safety Relief Valves Reclose

See subject discussion in Section 3.1.2.1.

Q - MSIVs Open, Condensate, Feedwater and PCS Systems Available

See subject discussion in Section 3.1.2.1. MSIVs are not dependent on DC power. However, the loss of Division 2 DC has an impact on TSW availability which, in turn, has an impact on availabilities of condensate pumps, condensate booster pumps, reactor feedwater, mechanical vacuum pumps and circulating water pumps. System fault trees for TSW, COND, RFW and PCS systems are discussed in their respective system notebooks. Basic faults representing Division 2 DC unavailability are set to 1 for the three system fault trees.

U - HPCS or RCIC Available

RCIC is dependent on Division 1 DC, and HPCS on Division 3. A discussion on RCIC and HPCS availabilities is contained in their system notebooks.

X - Timely Depressurization

The loss of one logic channel due to the failure of Division 2 DC bus results in a higher conditional failure probability for timely depressurization. A fault tree for the ADS is developed in its system notebook. The unavailability of Division 2 DC power is accounted for in the solution of the system fault tree. In case of automatic initiation failure, it is important that the operating staff initiates ADS manually in approximately 30 minutes before the reactor core uncovers.

V_1 - LPCI-B or LPCI-C Available

The loss of Division 2 DC bus will disable the LPCI-B and LPCI-C low pressure injection loops. However the LPCS and LPCI-A will still be available. See subject discussion in Section 3.1.2.1.

V_2 - Condensate System Available

The loss of Division 2 DC has an impact on TSW availability which, in turn, has an impact on availabilities of condensate pumps and condensate booster pumps. System fault trees for TSW and COND systems are discussed in their respective notebooks. The fault tree solution generated for V_2 accounts for the unavailability of Division 2 DC power.

W_1 - RHR Available

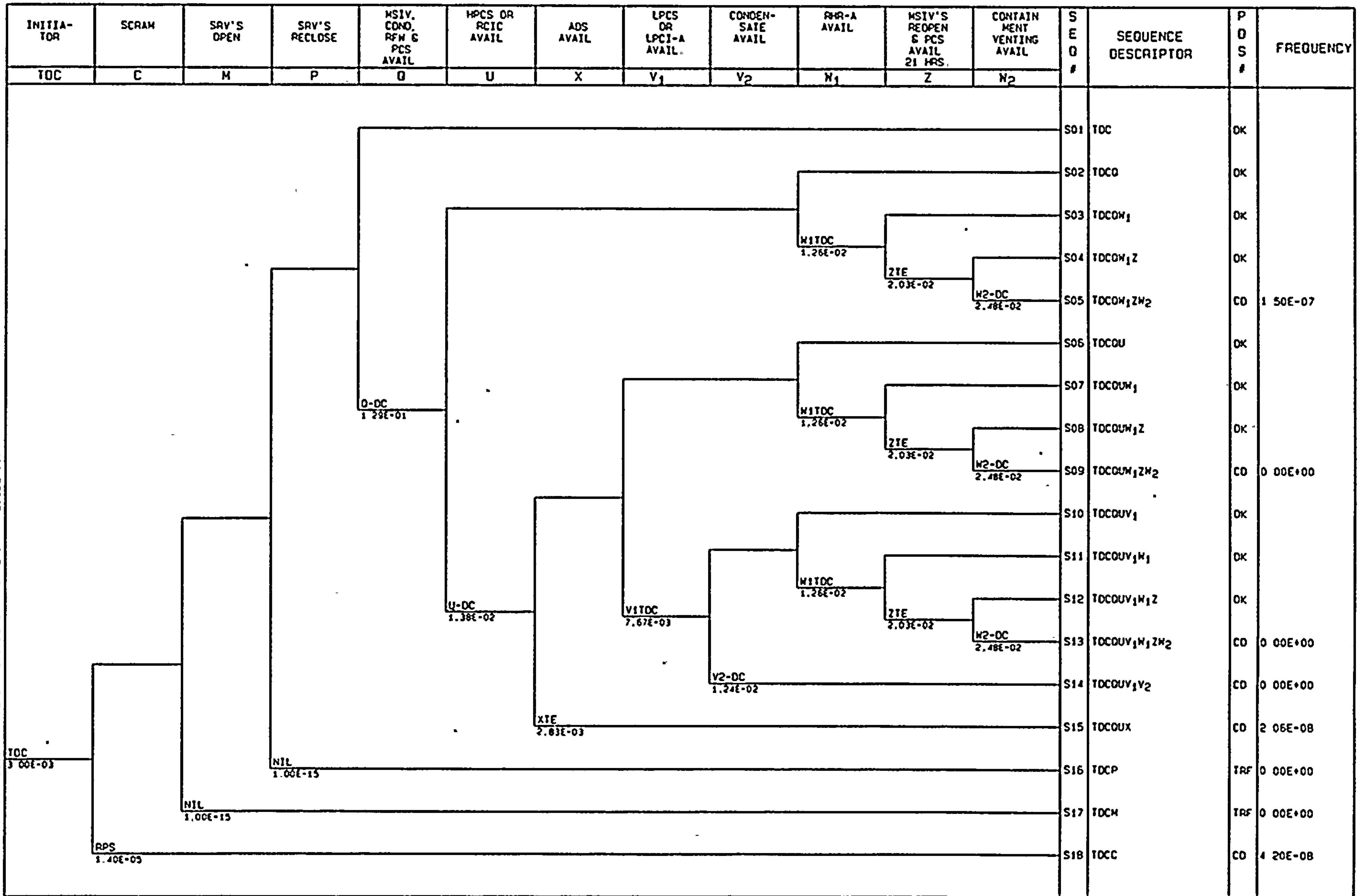
See subject discussion in Section 3.1.2.1. With the loss of Division 2 DC, only RHR Train A will be available for containment heat removal. A detailed fault tree for the RHR system is discussed in its system notebook.

Z - MSIVs Open and PCS Available

See subject discussion in Section 3.1.2.1. MSIVs are not dependent on DC power. However, the loss of Division 2 DC has an impact on TSW availability which, in turn, has an impact on the availabilities of the Circulating Water system and Condensate system. The Condensate system is required to cool the SJAE to prevent loss of condenser vacuum at greater than 5% reactor power. At less than 5% reactor power mechanical vacuum pumps can be used, which are dependent on TSW for cooling. The fault tree solution generated for Z accounts for the unavailability of Division 2 DC power.

W_2 - Containment Venting Available

See subject discussion in Section 3.1.2.1.



LOSS OF DIV. 2 DC BUS EVENT TREE
FIGURE 3.1.2.4-1

AUTHOR________________ DATE________

REVIEWER________________ DATE________

E:\IPE2\ET\DC.EVT 7:12:02am 4-29-94 NUPRA 2.1a WPPSS
Quantification Date: 4-02-94 9:07:57am TOTAL CWF = 2.13E-007

3.1.2.4.2 Loss of Plant Service Water

TSW takes suction from the Circulating Water basin. The system is operated with one TSW pump in service and the other one in standby. The motive power for the system is from the Division 1 or 2 AC. The control power for the system is from the Division 1 or 2 DC. Both divisions of AC or DC will have to be disabled to disable the TSW. The system supplies lube water to its own motors and seals. Loss of offsite power signal in combination with LOCA signal will trip the feeder breakers to SM-75 and 85 which supply power to the TSW. The TSW returns to the Circulating Water discharge tunnel where it combines with the circulating water prior to going to the cooling towers.

Equipment cooled by TSW are:

In Reactor Building:

1) Reactor Building Closed Cooling Heat Exchanger

In Turbine Building:

1) Mechanical Vacuum Pump Coolers
2) Condensate Pump Motor Coolers
3) Condensate Booster Pump Lube Oil Coolers
4) Reactor Feedwater Pump Turbine Oil Coolers
5) Air Compressor Intercoolers
6) Air Compressor Aftercoolers

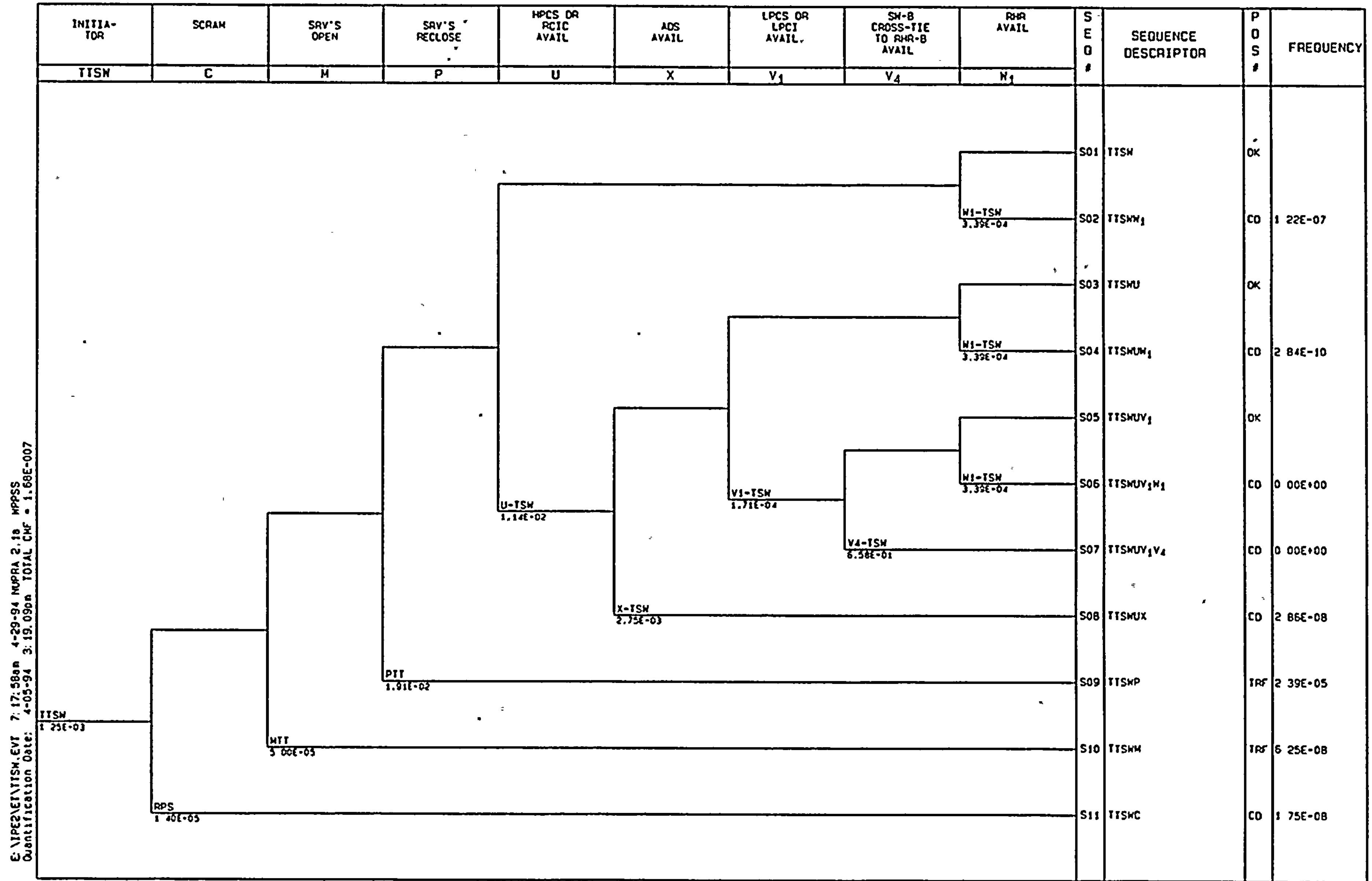
In Pumphouse:

1) Circulating Water Pump Bearing and Motor
2) Circulating Water Pump Priming Eductor

There are no Technical Specifications requirements on TSW. However, if the TSW pumps cannot be started, the plant must be shut down. This is because the components that are cooled by the TSW are essential for continued operation of the plant. Although there would most likely be adequate warning prior to a complete loss of TSW due to gradual reductions in performance, it is conservatively assumed that TSW suddenly becomes unavailable and the reactor is manually tripped. All of the equipment cooled by the TSW are assumed unavailable. The event tree for the loss of TSW is shown in Figure 3.1.2.4-2. The functional events that make up the event tree and are different than those given in Section 3.1.2.1 are discussed below.

T_{TSW} - <u>Loss of TSW Initiator</u>

There have been no sudden losses of TSW recorded based on a review of over 400 reactor years of reactor experience. It is conservatively assumed that 0.5 events have occurred. Assuming an incipient event in the operating history data base of 400 reactor years, we calculate a frequency of $0.5/400 = 1.25 \times 10^{-3}$ per reactor year for a complete and sudden loss of TSW.



LOSS OF PLANT SERVICE WATER EVENT TREE
FIGURE 3.1 2.4-2

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3.1.2.4.3 Loss of Standby Service Water

The Standby Service Water system (SW) consists of 2 motor driven pumps, 2 cooling water spray ponds and the necessary piping, valves, instrumentation and control. The system also has a third loop which cools the HPCS system equipment. During normal operation of the plant, the SW system is in standby mode. During an accident, the SW provides a heat sink for the RHR and the diesel generators. It also provides cooling water to the ECCS pump motors and the cooling coils of air handling units essential for the operation of critical components and control room ventilation. SW also cools critical motor control centers. Critical switchgear rooms are cooled by either SW or TSW. The bottom of the pump sumps are depressed below the spray pond bottom. This ensures that there is still sufficient submergence for the pumps at the lowest possible water level in the pond. A sand trap and a screen precede the pump sump to prevent heavy debris from entering the pump sump area. A skimmer wall-fixed screen prevents floating debris from entering the pumps.

During normal operation with both SW-A and -B inoperable, the Technical Specifications require restoration of at least one loop to operable status within 8 hours or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours. The event tree for the loss of SW is shown in Figure 3.1.2.4-3. All of the equipment cooled by the SW are assumed unavailable. The functional events used in the event tree which are different than those discussed in Section 3.1.2.1 are discussed below.

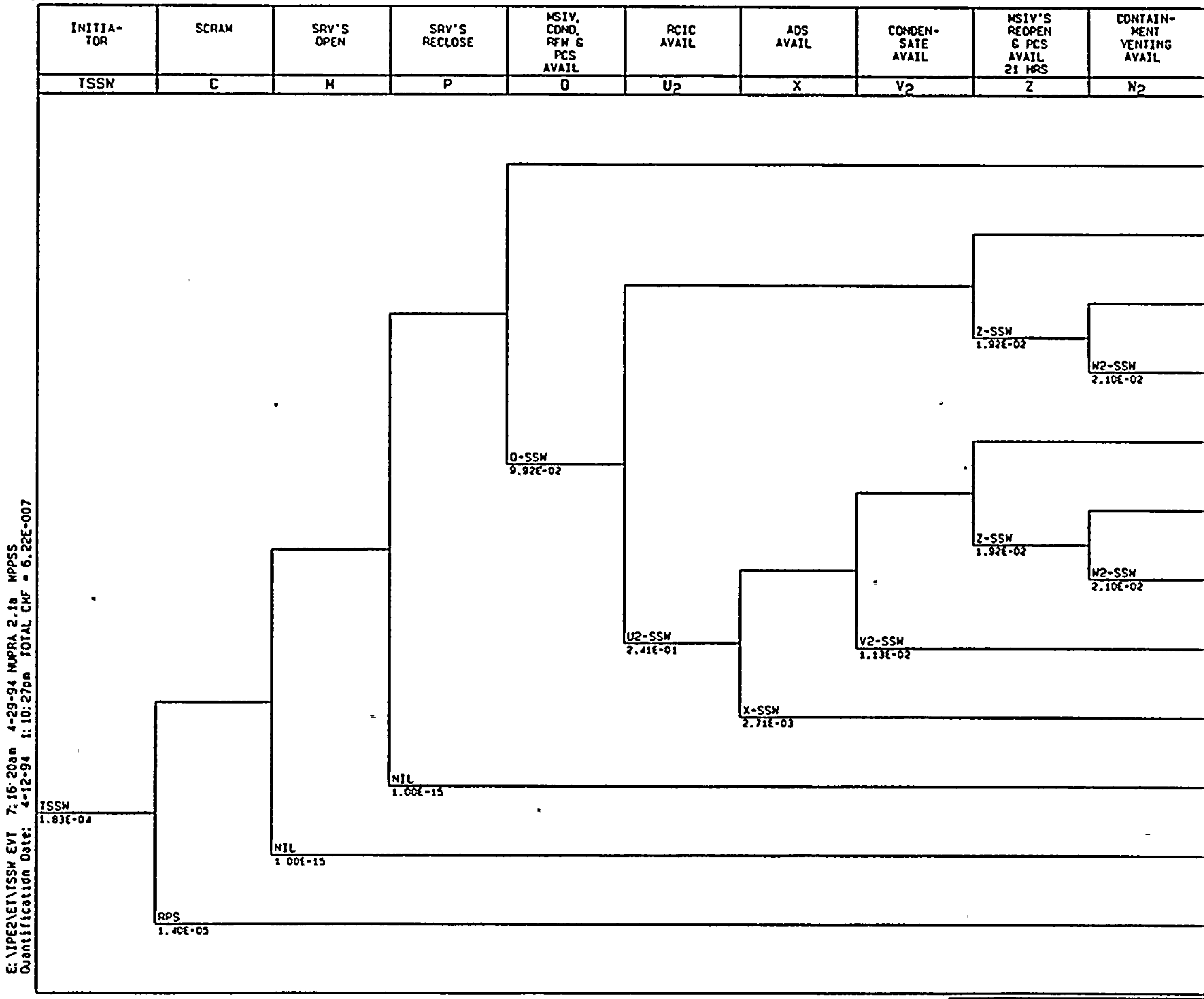
T_{SW} - Loss of SW Initiator

The frequency of the loss of Standby Service Water (T_{SSW}) initiating event was determined by fault tree analysis. SW is a standby system which if found to be unavailable and not able to be restored within the time specified by Technical Specifications, the plant must be shut down. A fault tree was built to reflect the logic that one train fails and the opposite train fails to run for 8 hours. The normal SW fault trees were then copied and modified to reflect the failure of each train in standby or failure to run during the monthly test. Train C of the Standby Service Water System was conservatively assumed to be unavailable since it has no impact upon the Technical Specification requirements to shutdown the plant.

The frequency of the loss of standby service water was calculated to be 1.83E-04/year and is dominated by common cause failures of the pumps, pump discharge valves, or pond return valves which are insensitive to the mission times.

U_2 - RCIC Available

See subject discussion in Section 3.1.2.1.7.2. With the RCIC pump room doors open, natural circulation is sufficient to cool the room for continuous pump operation. The operator action to open the doors when SW is unavailable is modeled in the RCIC fault tree.



SEQ #	SEQUENCE DESCRIPTOR	POS #	FREQUENCY
S01	TSSW	OK	
S02	TSSWQ	OK	
S03	TSSWQZ	OK	
S04	$TSSWQZW_2$	CO	2 00E-07
S05	$TSSWQU_2$	OK	
S06	$TSSWQU_2Z$	OK	
S07	$TSSWQU_2ZW_2$	CO	0 00E+00
S08	$TSSWQU_2V_2$	CO	4 09E-07
S09	$TSSWQU_2X$	CO	1 01E-08
S10	TSSWP	TRF	0 00E+00
S11	TSSWM	TRF	0 00E+00
S12	TSSWC	CO	2 56E-09

LOSS OF SERVICE WATER EVENT TREE
FIGURE 3.1.2.4-3

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3.1.2.4.4 Loss of Containment Nitrogen (CN)

The CN system supplies nitrogen to the 18 SRVs and the 4 in-board MSIVs using a cryogenic nitrogen supply. In the event that the cryogenic nitrogen is unavailable, two independent nitrogen bottle bank subsystems can deliver pressurized nitrogen to the 7 ADS valves and accumulators. A remote nitrogen cylinder connection is provided to each subsystem to permit supplementing the cylinder banks through manual connection of additional portable nitrogen cylinders, and thus maintaining pressure to the 7 ADS valves for an indefinite time.

On loss of CN, inboard MSIVs will close. The closure of the MSIVs will cause the reactor to SCRAM. It is assumed that the RFW is unavailable for coolant injection, and the condenser is unavailable for decay heat removal. Nitrogen bottle bank subsystems or remote nitrogen cylinder connector is assumed operational to keep ADS functional.

The event tree for the loss of CN is shown in Figure 3.1.2.4-4. The functional events listed across the top of the event tree and are different than discussed in Section 3.1.2.1 are discussed below.

T_{CN} - Loss of CN Initiator

A review of 400 reactor years of plant operational experience found no events in which the air system has been lost for greater than 4 hours. Only in cases where a LOOP or blackout exists for a long period does the reliability of the N_2 supply become a concern. Assuming 0.5 incipient event in the operating history data base of 400 reactor years, a frequency of $0.5/400 = 1.25 \times 10^{-3}$ per reactor year was calculated for a complete and sudden loss of the Containment Nitrogen system.

C - Reactor Subcritical

SCRAM valves and SCRAM Discharge volume vent and drain valves are not dependent on the CN system. See subject discussion in Section 3.1.2.1.

M - Safety Relief Valves Open

The safety mode of each SRV is actuated directly by the force exerted upon the main spring by the reactor pressure. There is no dependency on the CN for the safety mode operation. For SRV reliability, see Section 3.1.2.1.

X - Timely Depressurization

Each SRV valve has a 10 gallon accumulator to provide a source of N_2 in the event of a loss of the CIA. Moreover, each ADS valve has an additional 42 gallon accumulator to allow one actuation against maximum drywell pressure with the reactor pressure at 0 psig. A detailed fault tree for the ADS is developed in its system notebook. Basic fault representing the CN cryogenic unavailability is set to 1 for the ADS fault tree.

V_2 - Condensate System Available

There is no dependency of the COND system on the CIA. See subject discussion in Section 3.1.2.1.

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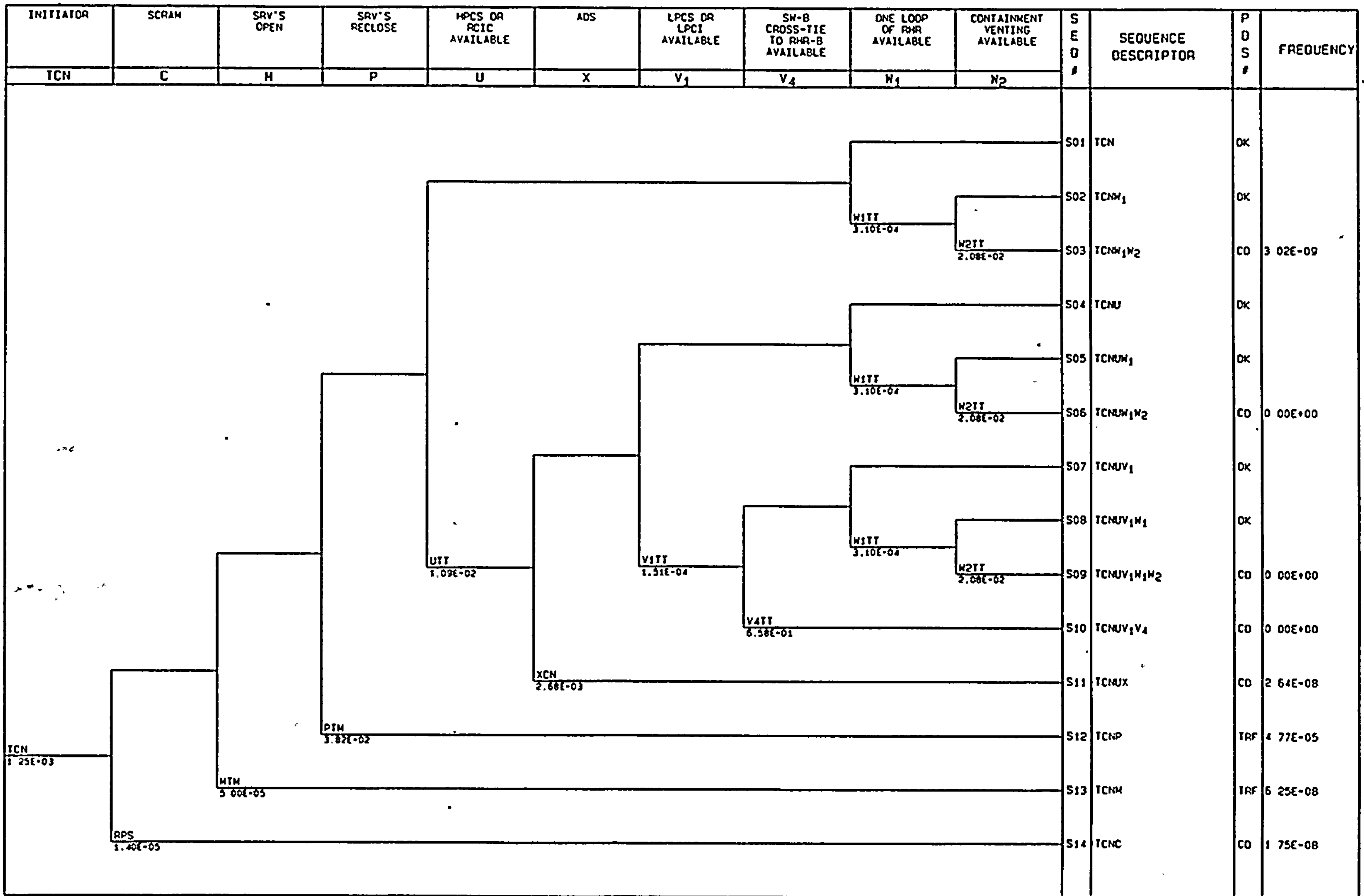
Quantification Date: 4-02-94 9:06:44am TOTAL CMF = 4.69E-008



LOSS OF CN EVENT TREE
FIGURE 3.1.2.4-4

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3.1.2.4.5 Loss of Control and Service Air

Four air compressors are arranged to operate in parallel or independently to supply a single, or any combination of 3 air receivers. One or two air compressors are normally running to carry the full plant load. The other air compressors will come on when the running compressors fail.

On loss of the CAS, the outboard MSIVs close. The SCRAM valves will fail open causing drive water to force the pistons upwards, thus inserting the control rods. However, the SCRAM vent and drain valves will fail closed, thus preventing loss of reactor water discharged from all CRDs during and after a SCRAM. The SCRAM discharge volume is sized such that the water accumulation in the volume with the vent and drain valves closed will not inhibit SCRAM. Air operated feedwater startup valves (RFW-FCV-10A and 10B) will fail as is on the loss of CAS. In the loss of CAS event tree development, RFW, COND, and FP water are assumed to be unavailable for coolant injection; and the condenser, unavailable for decay heat removal. The event tree for the loss of CAS is shown in Figure 3.1.2.4-5. The functional events listed across the top of the event tree are discussed below. Only the initiator functional heading is discussed, the others are discussed in Section 3.1.2.1.

T_{CAS} - Loss of CAS Initiator

See T_{CN} initiator discussion in Section 3.1.2.4.4. Initiator T_{CN} and T_{CAS} have the same frequency.

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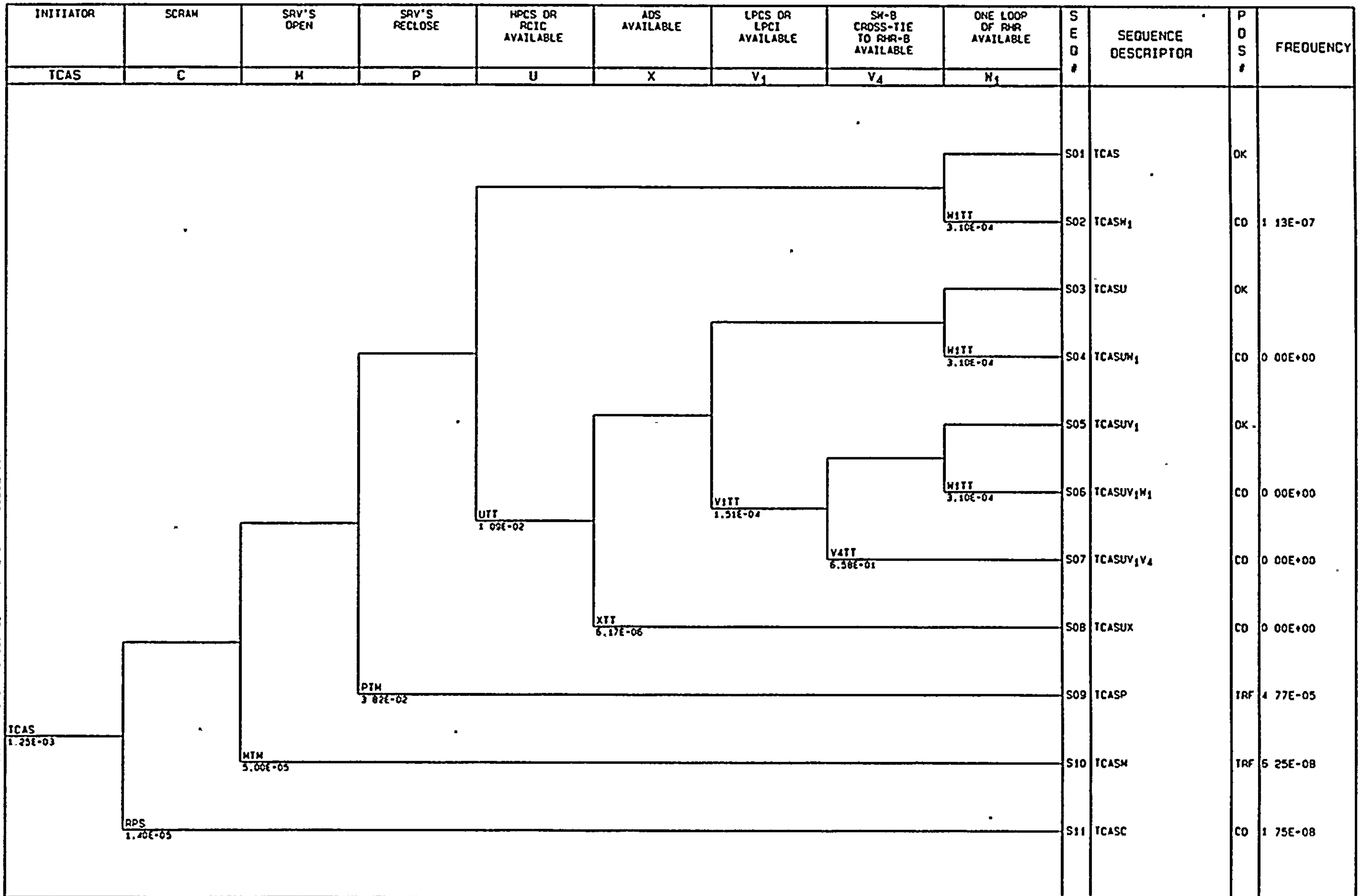
Quantification Date: 4-05-94 3:16:45pm TOTAL CMF = 1.31E-007



LOSS OF CAS EVENT TREE
FIGURE 3.1.2.4-5

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3.1.2.4.6 Reactor Water Level Instrumentation Line Break

Reactor water level measurement instrumentation affects both the operator's perception of the condition of the core and the automatic controls of normal and safety systems. As a result, postulated failure modes of this instrumentation, which can disable multiple systems, become important in the evaluation of plant safety.

WNP-2 has 4 condensing chambers MS-CU-4 A, B, C, and D (Figure 3.1.2.4-6.1) providing reference legs to the reactor water level instrumentation. Condensing chamber MS-CU-4B provides a reference leg to more instrumentation than the other condensing chambers do: 7 level sensors and 1 differential pressure transmitter.

The differential pressure transmitter is:

RFW-DPT-4B (L8 RFW Turbine trip)

The seven level sensors are:

MS-LIS-24D	(L8 trip of RCIC Steam Supply, L3 SCRAM, and L2 NS[4] groups 5 and 6 isolation)
MS-LIS-38B	(L3 SCRAM, and L3 ADS confirmation)
MS-LIS-37B,D	(L2 RCIC initiation)
MS-LIS-36C,D	(L2 recirculation pump trip, and L2 ATWS/ARI initiation)
MS-LS-61D	(L2 NS[4] group 1 and 2 isolation)

Except for channel functional tests which are performed monthly, all tests and preventive maintenance on level instrumentation are done when the reactor is shut down. Corrective maintenance is done in accordance with Technical Specifications.

In the following discussion, an instrument line of the condensing chamber MS-CU-4B is assumed ruptured outside the drywell but inside the secondary containment structure. The line is not isolated, and the rupture results in the release of primary system coolant to the secondary containment until the reactor is depressurized and the break flow terminated. Sensors connected to the line are conservatively assumed to indicate upscale regardless of the failure mode.

Symptoms may be one or any combination of the following:

a. Operator comparing readings with several instruments monitoring the same process variable such as reactor level, jet pump flow, steam flow, and steam pressure.

b. By annunciation of the control function, either high or low in the main control room.

c. By a half-channel SCRAM.

d. By a general increase in the area radiation monitor readings.

e. By an increase in the ventilation process radiation monitor readings.

f. By increase in area temperature monitor readings in the containment.

g. Leak detection system actuation.

The event tree of the reactor water level instrument line break is shown in Figure 3.1.2.4-6.2. The functional events listed across the top of the event tree that differ from the discussion in Section 3.1.2.1 are discussed below.

S_R - <u>Reactor Water Level Instrument Line Break Initiator</u>

The evaluated initiator frequency for instrument line breaks, and valve/human malfunctions is found as follows:

1) Observed leaks, breaks, valve induced leakages = 4E-2 per reactor year.

2) Combined frequency of (1) coupled with a coincident or dependent feedwater trip = 1E-2 per reactor year for a plant with four reference legs.

Q - <u>MSIVs Open, Condensate, Feedwater and PCS Systems Available</u>

During power operation, feedwater system vulnerability to a high level trip is increased due to the high level signal from the broken instrument line. The main turbine could also be tripped due to high reactor water level, L8. Feedwater is therefore unavailable for high pressure coolant injection.

U - <u>HPCS or RCIC Available</u>

In the event that level sensors MS-LIS-37B and D fail high, RCIC may fail to initiate. If level sensor MS-LIS-24D fails high, an L8 trip of RCIC could occur. It is conservatively assumed that RCIC is unavailable.

X - Timely Depressurization

The fault tree for the ADS system is discussed in Section 3.2.2.13. Operator response to manually initiate depressurization is included in the modelling.

V_1 - LPCS or LPCI Available

There are no low pressure coolant injection sensors connected with condensing chamber MS-CU-4B. See subject discussion in Section 3.1.2.1.

Z - MSIVs Open and PCS Available

Although MSIVs are assumed closed, they can be reopened given enough time. See subject discussion in Section 3.1.2.1.

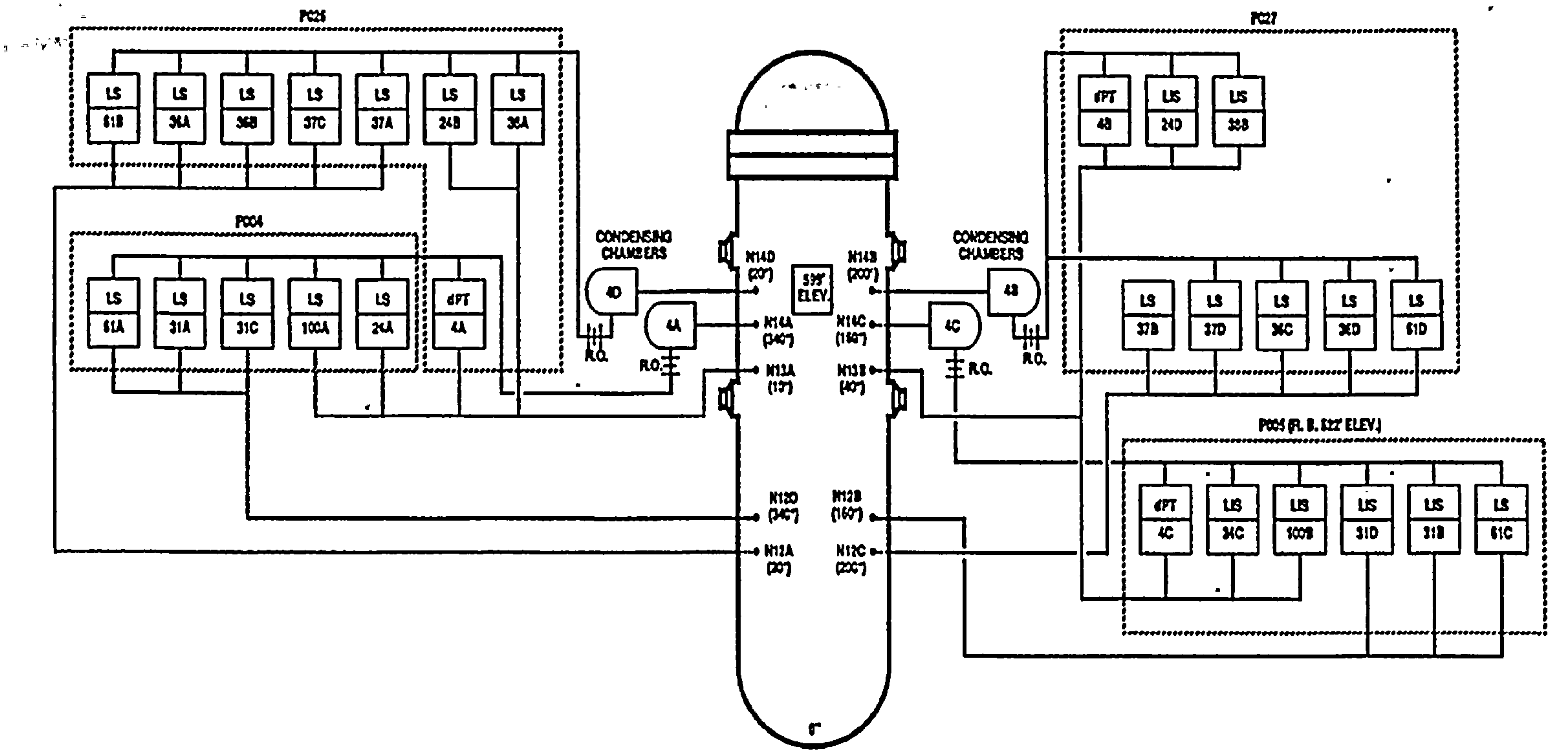

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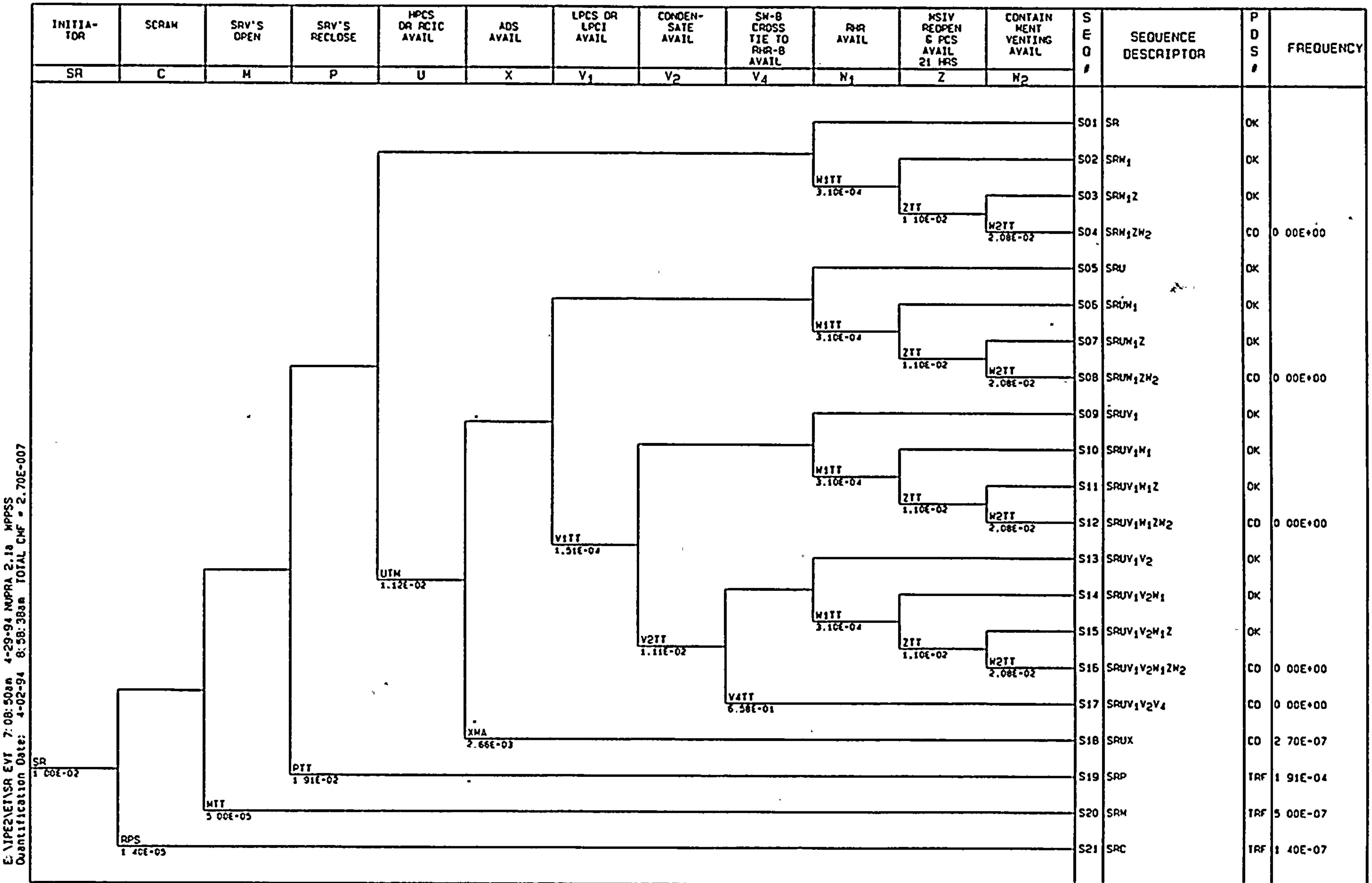
FIGURE 3.1.2.4-6.1 NUCLEAR BOILER LEVEL INSTRUMENTATION



LEVEL INSTRUMENT LINE BREAK EVENT TREE

FIGURE 3.1.2.4-6 2

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3.1.2.4.7 Internal Flooding

The contribution of internal flooding to core damage frequency can be determined by examining the potential water sources and the possible paths available for releasing water to IPE-related equipment. The following lists the water sources and their major delivery systems:

SOURCE	DELIVERY SYSTEM
Suppression Pool	LPCS, RHR, HPCS,
Condensate Storage Tanks	RCIC
Reactor Primary System	HPCS, RCIC
Condensate Hotwell	MS
Spray Ponds	COND, RFW
Circulating Water Basin	SW
Fire Protection Bladder Tank	CW, TSW, FP
Spent Fuel Pool	FP, FPC

Flooding can be due to a break in the water source container or in its delivery system. Breaks in the water source containers are discussed below.

Suppression Pool: - The suppression pool is part of the WNP-2 primary containment. Its design pressure and temperature are 45 psig and 275°F, respectively. Structural analysis indicates that the primary containment is not expected to fail below 121 psig, and that the failure location is above the suppression pool water levels. Therefore, under the normal operating condition of 75°F and 1 psig, there is very little probability that the suppression pool wall will rupture or leak. The possibility of a break in a piping section connected directly to the suppression pool is analyzed in the pipe break section of the flooding analysis.

Condensate Storage Tanks: - The condensate storage tanks are located outside the turbine generator building. If a break were to occur in one of these tanks, 400,000 gallons of water would drain to the ground outside without adversely affecting any other equipment assumed functional in the IPE.

Reactor Primary System: - Catastrophic reactor vessel rupture was not considered in this IPE. If the reactor were to fail early in its life time, it would most likely leak before failure. The accident sequence would be like that of a small LOCA. LOCAs of different sizes (small, medium, and large) are analyzed in Section 3.1.2.2, and their contributions to core damage frequency are determined in Section 3.3.7.

Condensate Hotwell: - The condensate hotwell is just below ground floor in the turbine generator building. As a passive system under very low pressure (external), the probability of a break is very small. If a break in the condenser hotwell is postulated, it would cause the loss of condenser vacuum. Loss of vacuum would cause the turbine stop valves to close

(18″ - 22″ Hg) and the MSIVs to close (7″ Hg). Closure of either set of valves would SCRAM the reactor. Event trees corresponding to Turbine Trip, MSIV Closure, and Loss of Condenser are analyzed in Section 3.1.2.1, and their contributions to core damage frequency are determined in Section 3.3.7. Water from the hotwell above the floor level will flow out into the turbine generator building ground floor and eventually out through the 15 foot wide equipment door on the west side of the building. The lowest equipment in the turbine generator building assumed functional in the IPE is installed 8″ above the floor. A condenser break would not submerge any of this equipment.

Spray Ponds: - The spray ponds are located northeast of the cooling towers at WNP-2. The water is below ground level and far from the power block. Flooding damage from the ponds themselves is not a concern.

Circulating Water Basin: - Like the spray ponds, the circulating water basin is located below ground level. Flooding damage from the basin itself is not a concern.

Fire Protection Bladder Tank: - The 400,000 gallon bladder tank for the fire protection system is located well away from the power block next to the potable water treatment building. The bladder tank is surrounded by an earthen berm which would contain the water in the event of a bladder tank rupture. Flooding damage from the fire protection bladder tank is not a concern.

Spent Fuel Pool: - The spent fuel pool is located on the 606′ elevation of the reactor building. It is a concrete pool with a steel liner, and is open to the atmosphere. Since it is open to the atmosphere, no over pressurization events are possible. The pool is also seismically qualified and an integral part of the structure of the reactor building. Rupture of the spent fuel pool is not believed to be possible without postulating an external event (such as a missile, or plane crash) which would define the initiator probability and is outside the scope of the flooding analysis.

Flooding scenarios due to breaks in delivery system piping are discussed below.

The PRA pipe break flooding analysis begins with a series of screening processes which are utilized to determine which areas of the plant are vulnerable to flooding damage of a nature serious enough to be risk significant. The Internal Flooding Evaluation Methodology (IFEM) Guidebook developed by the Individual Plant Evaluation Partnership (IPEP) is heavily relied upon to develop this screening process. A summary of the method used to develop this calculation is given below.

The first step of the screening process involves identifying all areas which contain equipment that may have an impact on core damage frequency. All components which were modeled in the system fault trees for the IPE were located. All areas of the plant which contain any of these pieces of equipment are considered within this calculation for flooding potential. Areas of the plant which do not contain equipment considered in the system fault trees are screened out.

The second step in the screening process is to take the set of areas which contain the equipment in the IPE system fault trees and define maximum flood levels for those areas. Those areas which cannot be substantially flooded or damaged by spray are eliminated from further consideration. This step involves defining the flood sources in each area of importance, including both the maximum expected rate of flow, the total quantity of water available to flood, the ability to detect the flood and the ability to isolate the flood source. The maximum flood level is calculated taking into account available drainage paths. If the maximum flood level in an area is low enough that the equipment considered in the IPE is not affected (and no IPE equipment is damaged by spray from the postulated pipe break), that area is eliminated from further consideration.

The data used for determining pipe break frequency is taken from the Failure Data Appendix of WASH-1400. This data was developed for "breaks of major, severance-type size." Therefore, flood levels were calculated assuming that the break in the piping was equivalent in area to the cross-sectional area of the pipe. The flow rate through the break was defined in a manner that is conservatively high. The flow rate across a circular orifice with an area equivalent to the cross-sectional area of the pipe was calculated using the normal operating pressure of the system. This flow rate was compared to the run out flow capacity of the supply pump. The lesser of these two values defines the highest flow rate possible out of the break.

High energy main steam and reactor feedwater lines are not considered flooding threats. One reason for screening them from the flooding analysis is that the majority of the water that is released goes out as steam and water vapor and is carried out of the building complex in gaseous form. The second reason is that the plant is designed to isolate high energy breaks very quickly. Thus, the total amount of water introduced to the building is small compared to other breaks. Thirdly, both main steam line break and loss of feedwater are handled as separate events in the WNP-2 IPE (Sections 3.1.2.2.4 and 3.1.2.1.4, respectively) and their contribution to the core damage frequency is determined in Section 3.3.7.

Once the flooding sources and flow rates are established, the method of detecting the flood and stopping the water source are identified. In most instances, the postulated floods will be detected when drain sump high level alarms go off. To determine the length of time that the flood source will release water into any given area, operator reaction times are established.

The operator response times used in the flooding calculation are consistent with the HRA analysis, and are discussed in more depth in the analysis of human reliability. The pertinent times for the flooding analysis are as follows:

a. If there is a requirement to use written procedures, assess a 5-minute delay, after correct diagnosis, before the first required post-diagnosis actions will be initiated.

b. Assess one minute as the required travel and manipulation time combined for each control room control action taken on the primary operating panels.

c. For required control actions on other than the primary control room operating panels, assess two minutes as the required travel and manipulation time for each such action.

These values are used to quantify the time it takes to terminate a flood source. Adding this to the time required to detect the flood source gives the total amount of time that the source is releasing water to an area.

Next, the possible drain paths out of the area are modeled and the maximum flood height is calculated. All submerged electrical or air operated equipment is assumed to fail. The flow out of all drain paths is also tracked to check for down stream effects.

The third step screens the remaining areas for failures which cause a shutdown transient. If an area does not contain any equipment which can cause the plant to shutdown, this area is removed from further analysis. Some judgement will be exercised in this step concerning the magnitude of the equipment loss. If the flooding causes an extensive amount of safety related equipment to be lost or safety related equipment in more than one division, it was retained for further evaluation even if no immediate shutdown is caused by the flood directly. In general, all floods in the reactor building or the turbine building are conservatively assumed to cause a plant shutdown, even if no specific cause of shutdown was identified.

The fourth step is to quantify the frequency of each of the flooding events in the areas which have passed all the screening criteria above, and determine the total contribution to the risk of core damage due to internal flooding. As stated earlier the rupture frequency data is taken from the Failure Data Appendix of the Reactor Safety Study, WASH-1400. The summary table of assessments for mechanical equipment lists the following computational median values:

Rupture/Plugging of pipe $\leq$ 3″ diameter	-	1E-9/hr per section
Rupture of pipe $>$ 3″ diameter	-	1E-10/hr per section
Rupture of all types of valves	-	1E-8/hr per valve

There are roughly one hundred flooding initiators identified which can affect equipment assumed functional in the IPE. Many of these initiators cover breaks in several different delivery systems within an area. For each initiator the number of piping sections and valves are counted, and the rupture frequency is calculated. The initiator frequency is determined on an annual basis by estimating the number of hours a given system is normally used and assuming a capacity factor of 0.8. Those initiators which have an initiation frequency less than 1E-6 are not considered further because even without assessing which systems are available to bring the plant to safe shutdown, they fall below the reporting criteria specified in NUREG-1335. The remaining initiators were grouped into fourteen categories based on the combination of systems that are damaged by the postulated flood.

For each of these categories the sum of all the initiating frequencies is determined and an event tree is developed which quantifies the risk of core melt. Of the 14 event trees, three result in core melt frequencies greater than 1E-6/yr. These three categories are described below.

Category FLD6: - Category FLD6 includes those flood initiators which cause the loss of the condensate (COND) and feedwater (RFW) systems as well as the loss of the control and service air system (CAS). (In a small fraction of these cases TSW is also lost because it is the system break initiating the flood. COND, RFW and CAS are the only IPE systems supported by TSW. So, the loss of TSW is inconsequential in this category of breaks.) This category of flood damage can be caused by submergence in the case of a 96" circulating water (CW) pipe break. It can also be caused by spray damage from breaks in the COND, RFW, TSW or fire protection piping in areas T104 or T105 in the turbine generator building. All together there are 58 sections of pipe and 41 valves whose rupture can cause a FLD6 category flood. This yields an initiator frequency of 2.92E-3/yr.

All floods in category FLD6 occur in the turbine generator building. Operator response times to terminate the flood were estimated to be about 9 minutes. However, if the operators fail to terminate the flood source in that time, no additional equipment would be damaged. The water from this flood would be exiting the turbine building on the west side through an equipment door which is always rolled open at least one foot. Figure 3.1.2.4-7.6 is the event tree for this category of flood initiators.

Category FLD14 - Category FLD14 consists of TSW line breaks at various locations in the plant, and COND, FP, and CW breaks in area T106 or the circulating water pump house. These breaks do not affect equipment other than TSW itself or systems supported by TSW. When TSW is lost due to pipe break or flooding effects, the supported systems, COND, RFW, and CAS are also lost. The event tree for this category is identical to the event tree in category FLD6 except for the initiator frequency. There are 104 sections of pipe and 66 valves whose rupture can cause a category FLD14 flood. This yields an initiator frequency of 4.69E-3/yr. Figure 3.1.2.4-7.14 is the event tree for this category of flood initiators.

Operator response times for termination of category FLD14 floods are 8 to 9 minutes. For the postulated floods in the circulating water pump house, the radwaste building and the turbine building, the time to termination is not critical. In these cases, equilibrium flood levels are reached and drainage paths are to outside the building. Additional time before termination does not result in the damage of any additional IPE equipment. However, for the sequence involving a TSW pipe break in area R504 of the reactor building, operator response to the break is important. The R504 TSW break with failure of the operators to terminate TSW in a timely manner is the category FLD7 flood.

Category FLD7 - Category FLD7 is a TSW pipe break in area R504 of the reactor building coupled with failure of the operator to terminate the flood source in a timely manner. In this flooding sequence the break flow is assumed to be the run out flow rate of the TSW pump, 23,000 gpm, into the general floor area of the 548′ elevation of the reactor building. This is the highest flow rate of any break in the reactor building. The drain paths from this area take water to the general floor areas on the 522′, 501′, and 471′ elevations as well as down the stairwells. The majority of the water goes down the northwest stairwell filling it until the doors on the 441 elevation break and water spills out into the hallway between the reactor building and the turbine building. However, a substantial quantity of water is carried away by the floor and equipment drains in the general floor areas. The floor drains in the reactor building go to four different sumps. These sumps are located in the RHR 'A' pump room, the RHR 'B' pump room, the RHR 'C' pump room and the HPCS pump room. If the operating staff fail to terminate the TSW flood source, these rooms eventually flood to a level where the ECCS pumps are affected. In addition, each of the ECCS pump rooms has an equipment hatch at the top. These hatches consist of concrete plugs in the floor of the 471′ elevation general floor area. The concrete plugs do not form a water tight seal, and some water from the 471′ elevation general floor area enters all of the ECCS pump rooms via this path.

The operating staff have about 20 minutes to terminate the TSW flood source. They will have a large number of indicators that there is a flood, and also indications that the problem is TSW. Within seconds of the TSW break, the main generator will trip on high temperature, causing a reactor SCRAM. High temperatures will also occur on the condensate pumps, the reactor feedwater pumps and the CAS compressors, indicating a problem with TSW. Early in the sequence, the high sump level alarms will go off on all four FDR sumps as well as the EDR sump in the CRD pump room. Between 4 and 6 minutes into the sequence the ECCS pump room flood level indicators in all three RHR pump rooms and in the HPCS pump room will alarm in the control room. The action required to terminate the flood is simply to switch the TSW pumps off from the control room.

If the operators fail to terminate the flood within 20 minutes, they begin to loose ECCS pumps. RHR-P-2A is lost first, followed by RHR-P-2C at about 21 minutes. At about 28 minutes, HPCS electrical connections are submerged, followed shortly by RCIC. The flow into the RCIC pump room is small, coming only from the leaks in the concrete plugs of the equipment hatch. But unlike the other pump rooms, the RCIC pump room has several electrical connections near the floor. At 35 minutes RHR-P-2B is lost. LPCS continues to be functional even if the operators fail to terminate the flood for another 2 hours. Figure 3.1.2.4-7.7 is the event tree for this category of flood initiators.

There are a total of 14 sections of TSW piping and 9 TSW valves whose rupture could cause this flooding sequence. This yields a frequency of 6.40E-4/yr. In this scenario the operating staff fail to terminate the flooding source within 20 minutes, but complete termination within 2 hours and 45 minutes. The operator failure probability is found using the data for a dynamic task type and a moderately high stress level. This yields a total human error probability of 2.5%. So, the initiation frequency for the category FLD7 flood is the product of the rupture frequency and the human error probability or 1.6E-5/yr.

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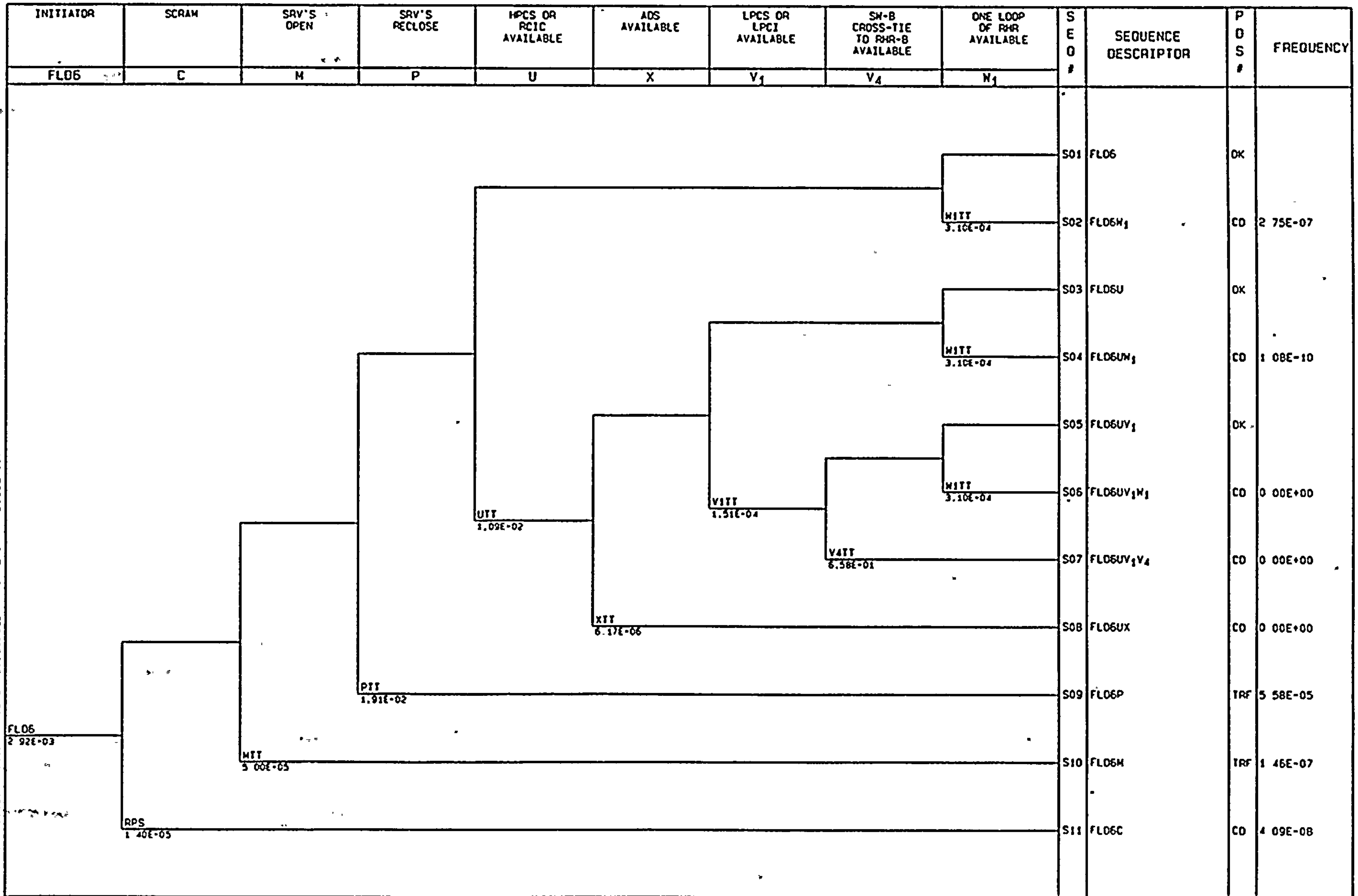

FLOODING EVENT TREE (CASE 6)
FIGURE 3.1.2.4-7.6

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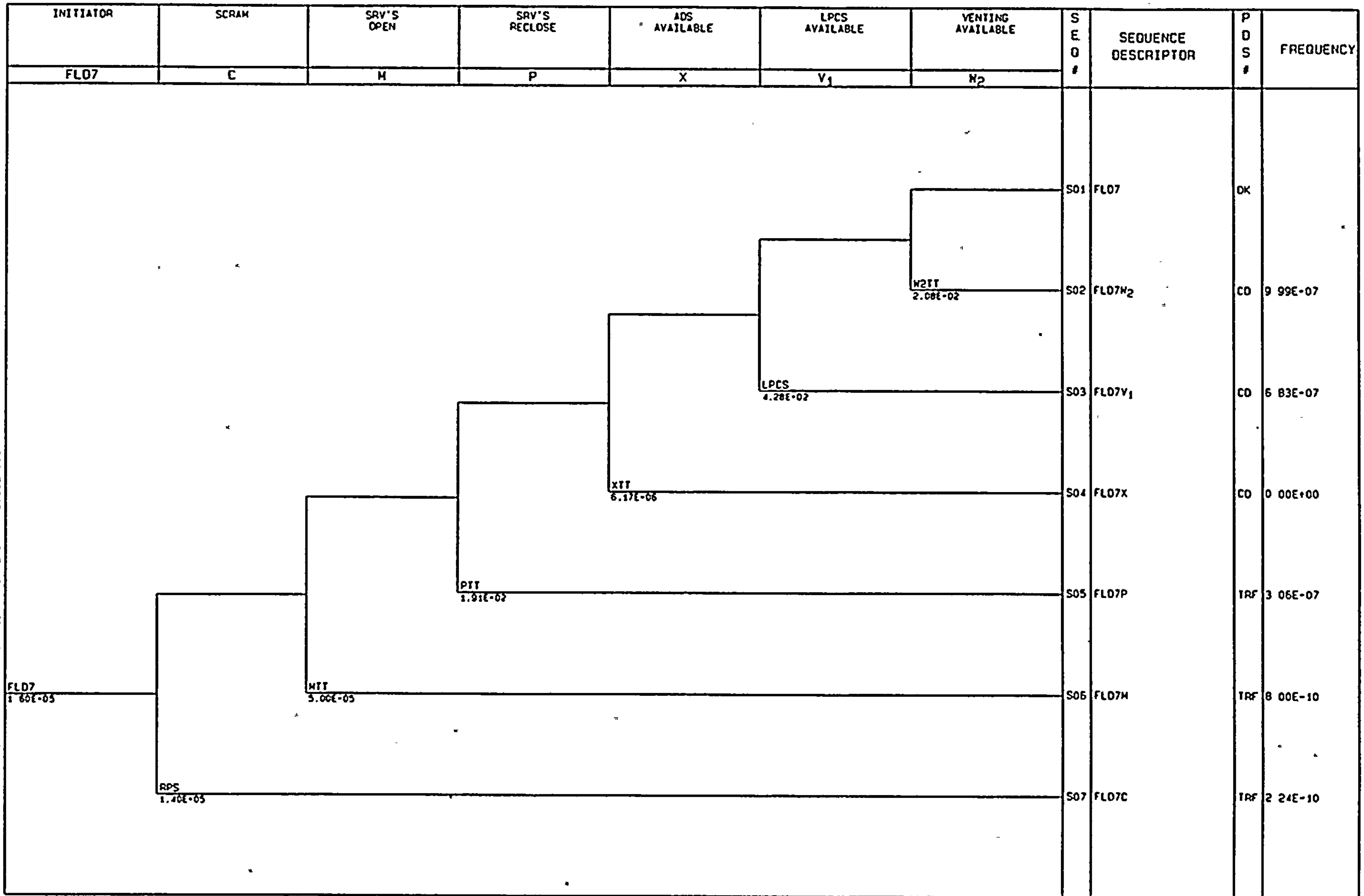

E:\IPE2\ET\FLD7.EVT 7:07:34am 4-29-94 NUPRA 2.18 WPPSS
Quantification Date: 4-12-94 3:26:05pm TOTAL CMF = 1.68E-006

FLOODING EVENT TREE (CASE 7)
FIGURE 3.1.2.4-7.7

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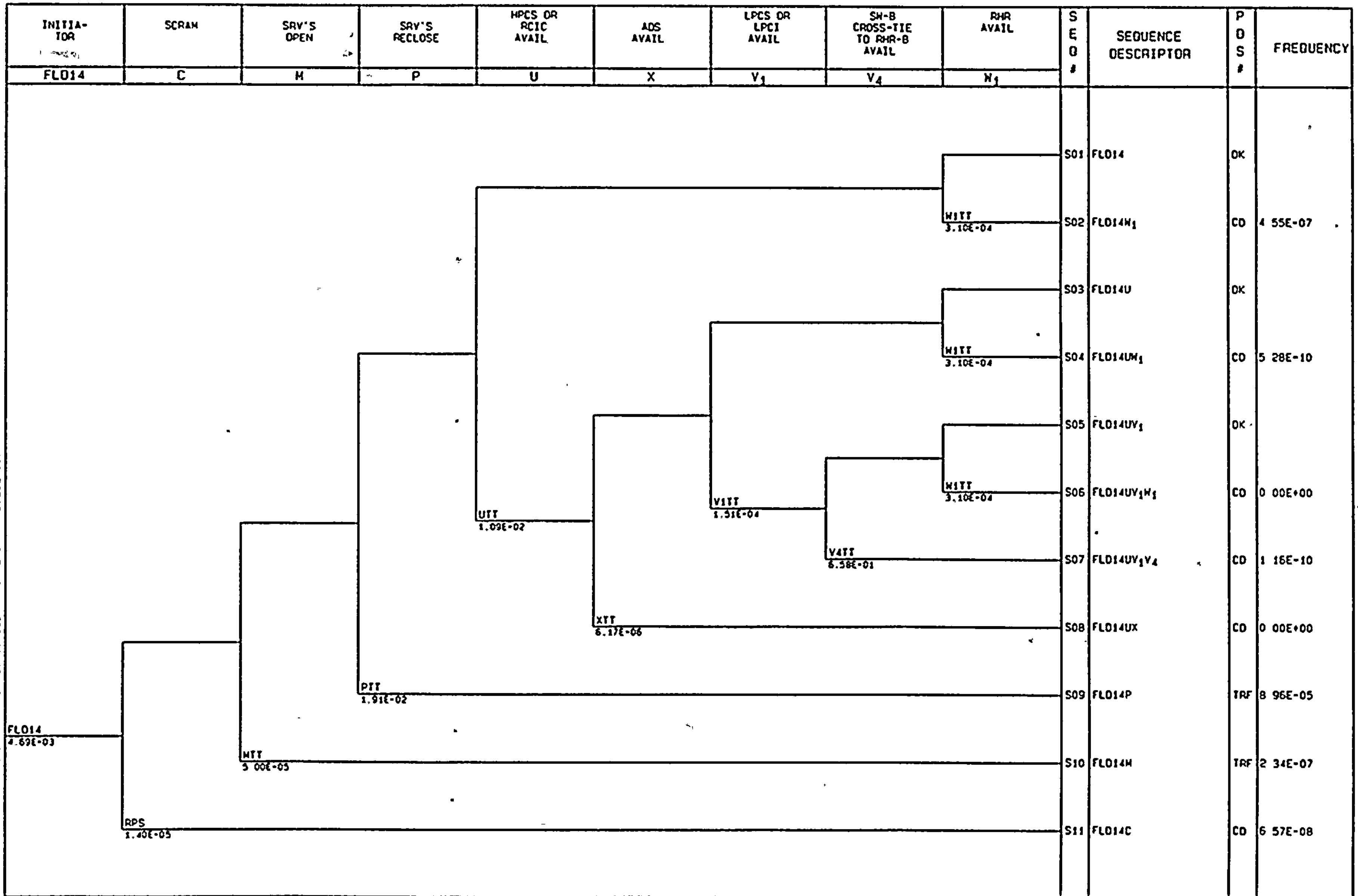

FLOODING EVENT TREE (CASE 14)
FIGURE 3.1.2.4-7.14

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3.1.3 Special Event Trees

Event trees for special initiators like loss of Plant Service Water, loss of Standby Service Water, etc., are discussed in Section 3.1.2.4.

3.1.4 Support System Event Tree

The NUPRA program used for quantification has "linking" and "merging" capabilities. The front-line system fault trees are logically linked to the support system fault trees. The effects that the support systems have on the front-line systems are properly accounted for. Therefore, support system event trees are not necessary.

3.1.5 Sequence Grouping and Bank End Interface

In order to conduct the level 2 analysis, dominant accident sequences obtained from the level 1 analysis are grouped based on the similarity in the effect that each has on containment performance. Each of the individual level 1 sequences which represent a contribution to overall core damage frequency of at least 1E-9/yr are placed into one of several groups which has the characteristics to represent its members throughout the level 2 analysis. In the level 2 portion of the analysis, consideration is given to how each sequence group is expected to affect the initial containment and RCS conditions or the availability of containment systems which are important to its response. The uniquely important characteristics of the core damage sequences within each sequence group are taken into account in the level 2 analysis with regard to system availability and unavailability. Discussion of the accident sequence groups used and their treatment is provided in Section 4.4.

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3.2 System Analyses

Each of the systems modeled in the WNP-2 IPE is briefly described in Section 3.2.1. The fault tree descriptions for each system are maintained in the second level of documentation and are retained at the Supply System. System interdependencies are discussed in Section 3.2.3 and presented in matrix format in four tables concluding that section.

3.2.1 System Descriptions

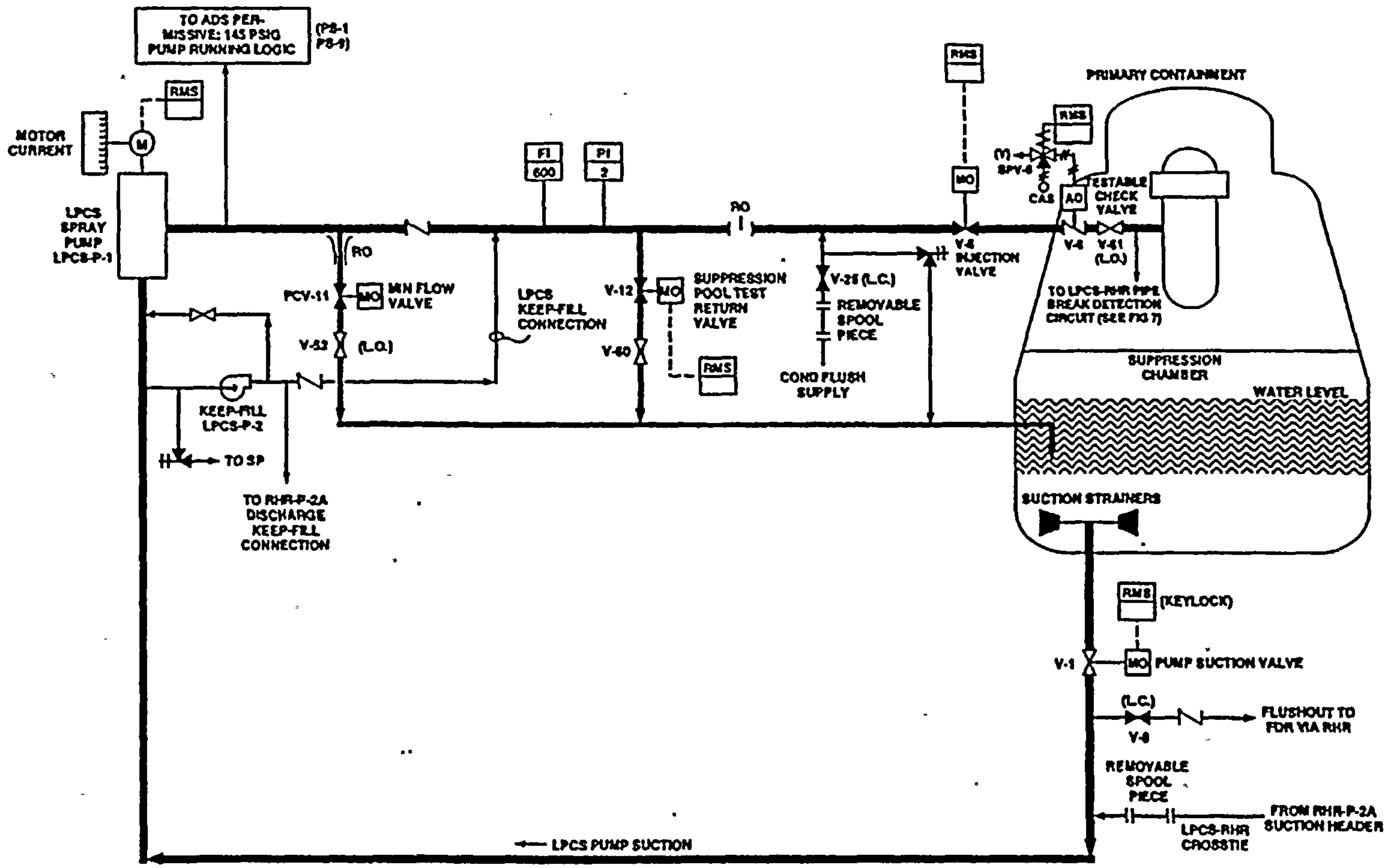
There are twenty-four system descriptions representing the systems modeled in the IPE.

3.2.1.1 LPCS System Description

The Low Pressure Core Spray (LPCS) system is an emergency core cooling system which pumps water from the suppression pool to the core through a sparger mounted above the reactor core. The LPCS is designed to provide cooling to the reactor core only when the reactor vessel pressure is low, for example, in the case of a large break LOCA. The LPCS system automatically starts when either a low reactor water level and/or high pressure in the drywell is sensed and sprays water into the core when the reactor vessel pressure is low enough.

The LPCS system consists of an electrically driven main pump and associated controls and piping capable of delivering water from the containment suppression pool through an in-vessel spray system to the top of the reactor core. During reactor operation, an auxiliary pump continuously pumps water from the suction side of the main pump to the piping down stream of the main pump discharge check valve, to assure that the elevated piping remains full of water. A test line connected to the main piping permits surveillance testing during reactor operation. A connection between the RHR and LPCS systems can be made by installation of a removable spool-piece during reactor shutdown, to permit surveillance testing of the entire system including spray nozzles without injecting suppression pool water into the reactor vessel. This system is depicted in the flow diagram shown on Figure 3.2.1-1.

To assure continuity of core cooling, primary containment isolation signals do not interfere with LPCS operation. The keep-fill system failure is not modeled as a failure mode of the LPCS (see RHR System Description, Section 3.2.1.10 for further discussion of this topic).



NOTES:
1. ALL VALVES SHOWN ARE LPCS SYSTEM
2. ALL INDICATIONS & SWITCHES ARE MOUNTED ON CONTROL ROOM PANEL P-602
3. SEE M-520 FOR ENGINEERING DIAGRAM
4. SP = SUPPRESSION POOL

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LPCS

FIGURE 3.2.1-1 SIMPLIFIED FLOW DIAGRAM: LPCS

3.2.1.2 CN System Description

The function of the Primary Containment Nitrogen Inerting (CN) system is to provide a supply of nitrogen for the following services:

- Primary containment inerting of both the drywell and the suppression chamber during normal reactor operation. Inerting is accomplished using the Containment Supply Purge and the Containment Exhaust/Purge systems.
- Actuation of the pneumatic components of the primary Containment Instrument Air (CIA) system during normal reactor operation. These pneumatic components consist of the control systems for four main steam isolation valves and their accumulators and for eighteen main steam safety/relief valves and their accumulators.
- Purging of the Traversing Incore Probe (TIP) system.
- Testing of the wetwell-to-drywell vacuum breaker valves.

The primary containment nitrogen inerting system is not a safety related system.

The system is designed to comply with the NRC staff position of April 2, 1981, requiring that "the General Electric pressure suppression containment systems identified by Mark I and Mark II, be inerted."

The system is designed to establish and maintain a nitrogen atmosphere in which the oxygen concentration can be controlled at less than 3.5% by volume in both the drywell and suppression pool during reactor operations.

The system is designed to supply sufficient nitrogen to provide one drywell air change in 45 minutes or one suppression chamber air change in 30 minutes at the design flow rate of 5000 cfm.

The system is designed to provide sufficient makeup to compensate for containment leakage during reactor operation.

The system is designed to Seismic Category II requirements as it has no post- accident function. However, the inert atmosphere precludes the possibility of a hydrogen burn in the event of an excessive metal/water reaction resulting from a LOCA.

The Primary Containment Nitrogen Inerting (CN) system is depicted on Figure 3.2.1-2. The system consists of a liquid nitrogen storage tank, a steam vaporizer to supply gaseous nitrogen for high flow requirements, two parallel flow ambient vaporizers and one electric vaporizer to supply gaseous nitrogen for low flow requirements and a pressure/temperature manifold skid containing pressure and temperature control valves.

For high flow requirements, liquid nitrogen at storage tank pressure and temperature (nominally 200 psig and -220°F) is supplied from the storage tank to a steam vaporizer. The vaporizer is a shell and tube heat exchanger with nitrogen flowing through the tubes and 50 psig saturated steam, (from the auxiliary steam system) flowing through the shell side. The steam vaporizer is rated to supply 5000 cfm of gaseous nitrogen between 60°F and 100°F to the pressure/temperature control manifold where the pressure of nitrogen is reduced to 30 psig by pilot operated pressure reducing valve CN-PCV-6. The high flow nitrogen stream of 300,000 scfh at 30 psig is utilized to inert the primary containment during reactor operation to reduce the residual oxygen concentration to below 3.5% by volume. The flow rate is indicated and controlled in the main control room.

To establish the inert atmosphere in the containment, nitrogen is introduced into the vessel through the 30" containment purge supply duct to either the drywell or the suppression pool (refer to the description for Primary Containment Purge system). The containment atmosphere is exhausted from the vessel through the drywell or wetwell purge exhaust penetrations. The purge stream is released through the plant vent and monitored. In the event airborne radiation levels in the containment purge are high, and for the first 24 hours of any purge, the containment purge exhaust will be sent through the Standby Gas Treatment system before discharge. During containment inerting, the drywell coolers and recirculation fans will be operated assuring a well mixed atmosphere.

Nitrogen Low Flow Requirements

Low flow nitrogen is required for the following services:

- 50-3000 scfh nitrogen at 186 psig is supplied (150 psig is required) to provide actuation of the pneumatic components of the primary Containment Instrument Air system.

  These Components include the actuators for four Main Steam Isolation Valves and their accumulators and for eighteen Main Steam Safety/Relief Valves and their accumulators.

- 300 scfh nitrogen at 110 psig is required for primary containment nitrogen inert atmosphere makeup, for testing of the primary containment wetwell-to-drywell vacuum breaker valves, and for purging of the TIP system components.

  For low flow requirements, vaporization of the nitrogen is performed by one of two redundant vaporizers followed by a 3 kw electric trim vaporizer that raises the exiting nitrogen gas temperature to 70°F. After leaving the electric vaporizer, the piping enters the pressure/temperature control manifold where it branches into a high pressure (186 psig) supply connection and a low pressure (110 psig) supply connection. Two pressure reducing valves are furnished to reduce the nitrogen pressure from tank pressure to 186 psig, for CIA loads, and 110 psig, for the other loads.

Flow integration is provided in each line but the flow integrator in the high pressure line is normally bypassed by the flow element. This is because the flow integrator is a low capacity device suitable only for nonload CIA system leakage determination; whereas, the flow element is sized to accommodate full-load requirements that could be imposed by maximum operation of the CIA system.

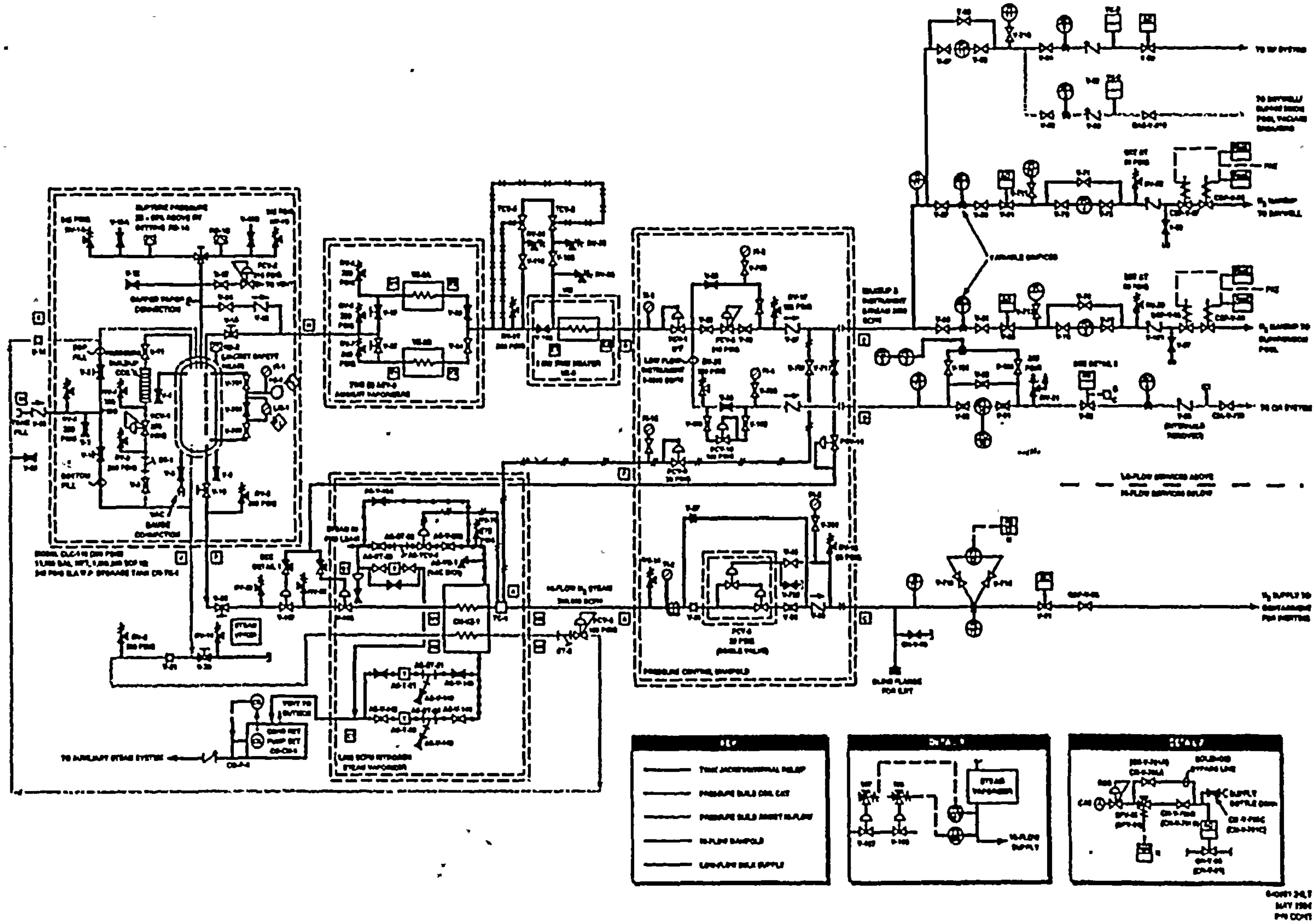

FIGURE 3.2.1-2 CONTAINMENT NITROGEN INERTING

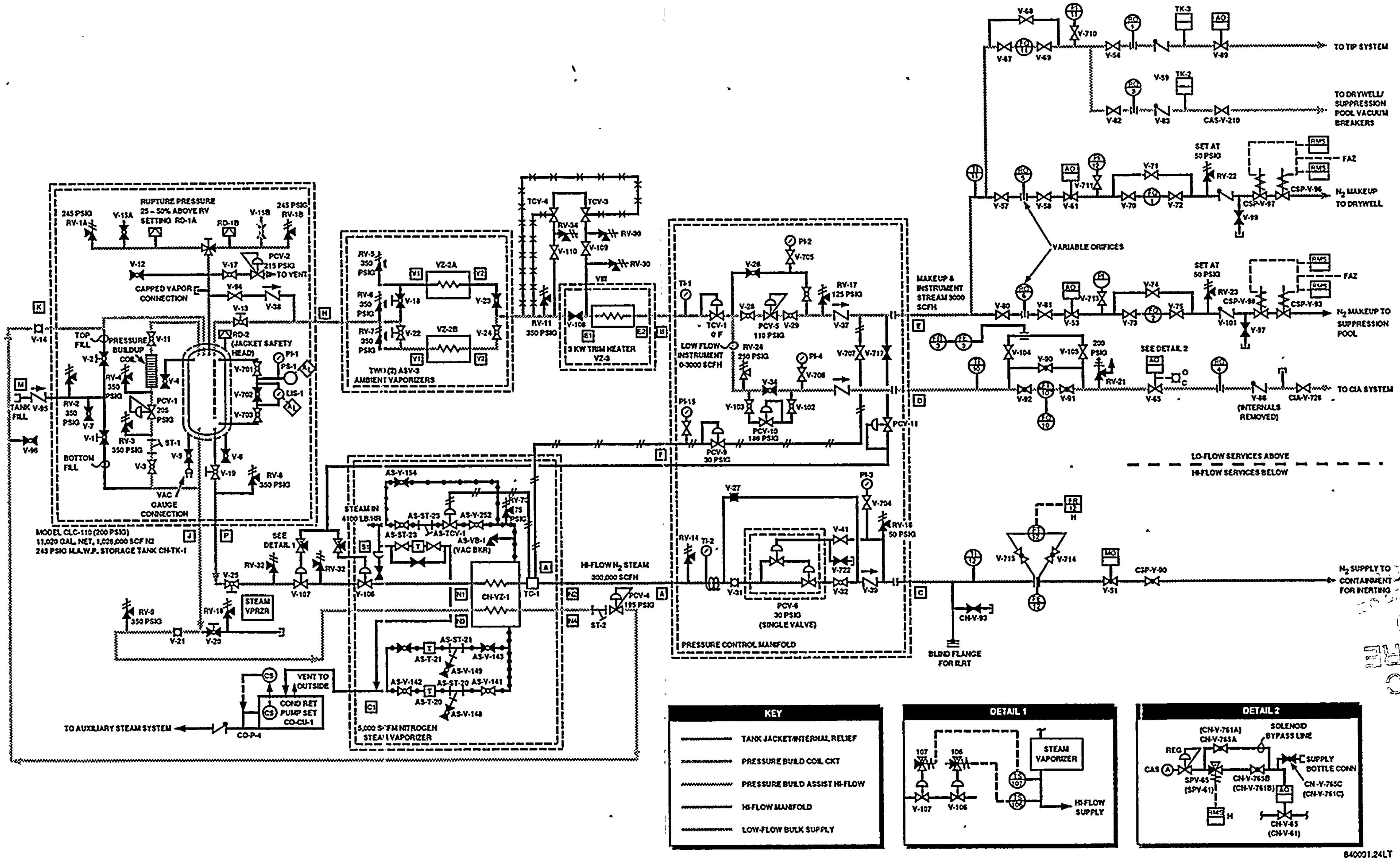

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FIGURE 3.2.1-2 CONTAINMENT NITROGEN INERTING

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3.2.1.3 CIA System Description

The CIA system provides nitrogen or air to all gas-operated components inside the Primary Containment Vessel (PCV).

The system is designed to supply clean, dry, compressed gas, nitrogen or air, to the following valve actuator accumulators and valve pilot controls inside primary containment: Seven individual accumulators, at the seven Main Steam Safety Relief Valves (SRV) dedicated to the Automatic Depressurization system (ADS) mode; four inboard Main Steam Isolation Valve (MSIV) actuator accumulators; eighteen SRV actuator accumulators for the power assisted pressure Relief Mode; and two Reactor Recirculation Cooling (RRC) pump seal staging drain valve pilot control valves. The loads are fed at a nominal pressure of 186 psig to the ADS, as regulated on the CN skid outside the Diesel Generator Building, and the balance of the loads (all nonsafety related) are fed at a reduced nominal pressure of 110 psig. To minimize the addition of air to an inerted containment due to leakage from the various components served by the system, nitrogen is available to meet both normal and abnormal system requirements. For normal plant operations, nitrogen is supplied from the CN system cryogenic storage vessel, which is also the source of nitrogen for inerting the primary containment atmosphere. Should the normal nitrogen supply become unavailable, the gas supply piping to the ADS function accumulators will automatically isolate from the cryogenic nitrogen supply and the ADS accumulator backup compressed gas manifold sub-systems will provide 180 psig nitrogen from banks of high pressure compressed nitrogen cylinders.

The 100 psig line does not have an automatic backup supply, but provision has been made for supplying air from the CAS system via manually actuated intertie valves in lieu of the normal CN supply.

The Containment Instrument Air system is primarily a pressurized nitrogen system, as shown in Figure 3.2.1-3. During normal reactor operation, the Containment Nitrogen system supplies pressurized nitrogen from an 11,000 gallon (1 million scf) cryogenic storage tank to meet the requirements of the following valves inside the Primary Containment Vessel:

Full supply pressure (186 psig) load:

- Seven dedicated accumulators to support the Automatic Depressurization System Mode of seven specific Main Steam Safety Relief Valves.

Reduced pressure (110 psig) loads:

- Four inboard Main Steam Isolation Valve accumulators.

- Eighteen SRV power assisted pressure Relief Mode individual accumulators.

- Two Reactor Recirculation Cooling (RRC) pump seal staging drain valve pilot control valves.

In case the cryogenic nitrogen source does not maintain supply pressure to the ADS accumulator safety related supply headers, two backup nitrogen cylinder bank subsystems are provided to automatically supply 180 psig nitrogen. A bank of 15 nitrogen cylinders supplies three of the ADS function accumulators, and a separate bank of 19 nitrogen cylinders supplies the other four ADS function accumulators (Figure 3.2.1-3). These backup subsystems provide a 30-day supply of nitrogen for the ADS function during a postulated LOCA condition. A remote nitrogen cylinder connection is provided to each sub-system to permit supplementing the cylinder banks through manual connection of portable nitrogen cylinders, and thus maintain the ADS function for an indefinite period following a postulated LOCA event. The remote cylinder connections are located in the Diesel Generator Building corridor, adjacent to the reactor building, outside the secondary containment boundary, permitting personnel access to the connections under post-accident conditions.

Remote Pressure Control Stations, each consisting of a bypassable Pressure Control Valve, are schematically located down stream of the junction of the backup nitrogen cylinders supply with the remote backup nitrogen cylinder supply, and are physically located in the Diesel Generator Building corridor. Thus, any problems with the pressure control valve could be accommodated by isolating it, to allow maintenance to be performed, while the ADS function header pressure was being maintained with the manual bypass valve. The Pressure Control Valve allows the cylinder regulators to be set at a higher, but broad range pressure.

The backup nitrogen cylinder banks are located in the reactor building railroad lock and are accessible during normal reactor operation. Cylinders are automatically valved to the supply piping in a sequential manner by a pressure controlled programmer for each bank, so that only the number of cylinders necessary to maintain pressure in the ADS accumulator supply lines are drawn upon. During normal reactor operation, if the pressure in a cylinder falls below 2200 psig, that cylinder is replaced with a cylinder charged to a higher pressure. The Department of Transportation (DOT) #3AA-3600+ gas cylinders used may only be charged to 3000 psig, a limitation imposed by the system piping. At the Technical Specification low allowable limit of 2200 psig (Technical Specification 4.5.1.e.1), 223 Standard Cubic Feet (SCF) of nitrogen are available, and at the lowest conservatively usable system pressure, 127 psig, 14.3 scf of nitrogen remains in each bottle.

It can be shown that 14 cycles of ADS bank 'A' and 15 cycles of ADS bank 'B' could be accomplished during a 30-day period assuming the most conservative set of pressure and leakage conditions. Thus, 14 ADS valve group actuations (with the word "group" denoting the simultaneous actuation of both banks) are available during the first 30 days after loss of the cryogenic nitrogen source.

Once opened, the ADS valves are not expected to be cycled during the post-accident period; nevertheless, the air supply was conservatively sized to allow for extra cycles, since they may be used for Alternate S/D Cooling.

In the event the cryogenic nitrogen source totally fails to supply the system requirements, the backup nitrogen cylinder banks will supply their respective ADS supply headers as described above and the headers will be automatically isolated from the common CIA supply line. The reduced pressure loads, enumerated above, will then be without a source of pressurized gas until the CAS intertie valves are opened. A normally open vent between the two in-line series block valves could then be closed (the vent is there to guarantee no backleakage from CIA to CAS, which could lower the oxygen content of CAS/SA and result in an adverse condition should a breathing-air tap be in use). The relatively low pressure CAS supply would still be inadequate for the ADS headers, though, and they would still automatically be supplied as discussed above.

Since each of the two backup nitrogen cylinder banks and the cryogenic nitrogen supply are independent of each other, a single component failure in one will not affect the operational function of the other. The two ADS header tie line isolation valves are each powered from a different division of the critical power supply.

During normal operation, the cryogenic nitrogen supply will maintain pressure in the inboard main steam isolation valve accumulators, and the main steam safety relief valve power assisted pressure relief accumulators, as well as in the ADS function accumulators. The cryogenic nitrogen supply piping and the CAS supply piping are not assumed to be serviceable under accident conditions. In such an event, the local accumulators at the MSIVs and SRVs provide a short term source of pressure for actuating these valves. The backup nitrogen cylinder bank subsystems will supply operating nitrogen pressure to the ADS accumulators at any time the normal supply does not function.

The solenoid valves on the ADS Mode SRVs could be pressurized to a pressure greater than their qualified pressure under certain accident conditions if the ADS header pressure were allowed to approach the piping design pressure; therefore, the ASME relief valves are set at a lower pressure to preclude this possibility.

3.2.1.4 RFW System Description

The reactor feedwater system primary function is to maintain the reactor vessel water level within predetermined limits during all reactor operating modes and provide the temperature, pressure and flow control of the feedwater to the reactor vessel.

The feedwater system provides a reliable source of high purity feedwater during both normal operation and anticipated transient conditions. The system is designed with sufficient capacity to provide for 115 percent of the feedwater flow at rated load. This provides sufficient margin to provide flow under anticipated transient conditions. The feedwater heaters are designed to provide the required temperature of feedwater to the reactor. The final feedwater temperature is 413.9°F at rated load.

The condensate and feedwater system is not required to effect or support the safe shutdown of the reactor or perform safety functions.

The feedwater system consists of reactor feedwater pumps, high pressure feedwater heaters, controls, instrumentation, piping, valves, and associated equipment to supply the reactor with heated, high quality feedwater. The system extends from the feedwater pump suction to the Reactor Pressure Vessel feedwater inlet penetrations and includes three subsystems which provide:

- A recirculating flow path through the feedwater pumps and HP pressure heaters and back to the condenser hotwell to provide system cleanup.

- A minimum flow path from each feedwater pump to the condenser to prevent pump over-heating and cavitation.

- A startup flow path to control reactor vessel level during low power operation.

Figure 3.2.1-4 is a simplified diagram of the feedwater system.

The feedwater system starts at the suction of the turbine driven reactor feedwater pumps. The feedwater pumps take suction from the discharge of the fifth stage low pressure condensate heaters via a common 30" header then through individual 24" suction headers each equipped with a manual isolation valve. The feedwater pumps consist of two fifty percent capacity turbine driven centrifugal pumps arranged in a parallel configuration. The pumps discharge through individual 24" discharge headers into a common 30" header. Each pump discharge header is equipped with a check valve, a motor operated isolation valve and a flow measuring device.

Each pump discharge header is also equipped with a 2" pressure equalizing header. The pressure equalizing header initiates down stream of the pump discharge isolation valve and routes feedwater through a restrictive orifice to a point just upstream of the pump discharge check valve. The pressure equalizing header is to allow pressurization of an isolated or off-line pump while at the same time preventing an excessive amount of equalizing header flow.

Each RFW pump has a minimum flow recirculation header. This header routes feedwater from a point upstream of the pump discharge check valve through an air operated flow control valve to the main condenser. The minimum flow recirculation systems allow a flow path for feedwater during system low flow conditions to prevent pump heat up or cavitation. The minimum flow control valves respond to their flow controller monitoring the individual pump discharge flow and are interlocked with the feedwater turbine stop valve positions; refer to Section 2.3.3 for detail.

There is also a 16" bypass header around the reactor feedwater pumps incorporating a motor operated control valve and a check valve. The bypass header routes condensate from the 24" RFW pump suction header to the common 30" RFW discharge header and is utilized to supply condensate to the reactor from the condensate system during startup preparation and low power operation when the feedwater pumps are off-line. The discharge flow from the reactor feedwater pumps or the feedwater pump bypass header is routed through the high pressure feedwater heaters. The high pressure feedwater heaters consist of two 50% capacity parallel heater exchangers which provide the last stage of feedwater heating prior to the reactor core. The heat exchangers combine high pressure turbine extraction steam and first and second stage moisture separator reheater drain condensate on the shell side to heat the feedwater routed through the tube side. Both high pressure feedwater heaters have motor operated inlet and outlet isolation valves.

The 24" outlet headers from the twin sixth stage HP feedwater heaters combine down stream of their respective discharge isolation valves in a common 30" header. This 30" header is the main feedwater supply header to the reactor.

The sixth stage HP feedwater heaters can be partially or completely bypassed. A 20" bypass header will route feedwater from the common 30" heater supply header, to a point down stream of the individual heater discharge isolation valves. This bypass header is equipped with a motor operated bypass valve.

A feedwater recirculation header and control system has also been provided. This system provides a flow path from the discharge of the sixth stage HP feedwater heater to the main condenser. During startup preparations, this flow path will allow the entire condensate and feedwater systems up to the discharge of the HP feedwater heater to be recirculated to the main condenser for water quality cleanup and feedwater flow control. A 16" header taps off between the outlet of each HP heater and its respective outlet isolation valve. Each header has a motor operated isolation valve.

These recirculation flow isolation valves have remote manual switches and position indicating lights located in the control room. The individual recirculation headers combine to form a common 16" header which returns to the main condenser via condenser connection 41. A flow control valve and flow measuring device are located in this common return header. This control system is utilized to measure and control the amount of recirculation flow to the main condenser.

The startup feedwater level control valves are installed in parallel in the 12" header located between the common feedwater recirculation header and the main feedwater supply header to the reactor. These startup low flow valves are utilized to provide make-up and level control to the reactor vessel during start-up when the feed pumps are off-line or operating at low speed. Both level control valves, controlled by the single element control system, are air operated control valves. An isolation valve is installed in the 12" line down stream of the level control valves.

The main 30" feedwater supply header branches into twin 24" headers (Line A and Line B) prior to leaving the turbine generator building. Each feedwater supply line has a flow measuring element which is utilized by the feedwater control system.

The 24" feedwater headers penetrate the primary containment and sacrificial shield structures of the reactor building to supply the reactor vessel. Each of these feedwater supply headers is equipped with three containment isolation valves in series: a motor operated gate valve and two check valves. The motor operated gate valve and a check valve equipped with a spring actuated operator are located outside containment between the reactor building wall and the primary containment structure. The second check valve in each line is located inside containment between the containment wall and the sacrificial shield. The Reactor Water Cleanup system 4 inch return header connects to the feedwater headers between the two isolation valves outside containment. Down stream of the second check valve each feedwater line branches into three supply lines to penetrate the sacrificial shield to connect to the reactor vessel feedwater supply nozzles.

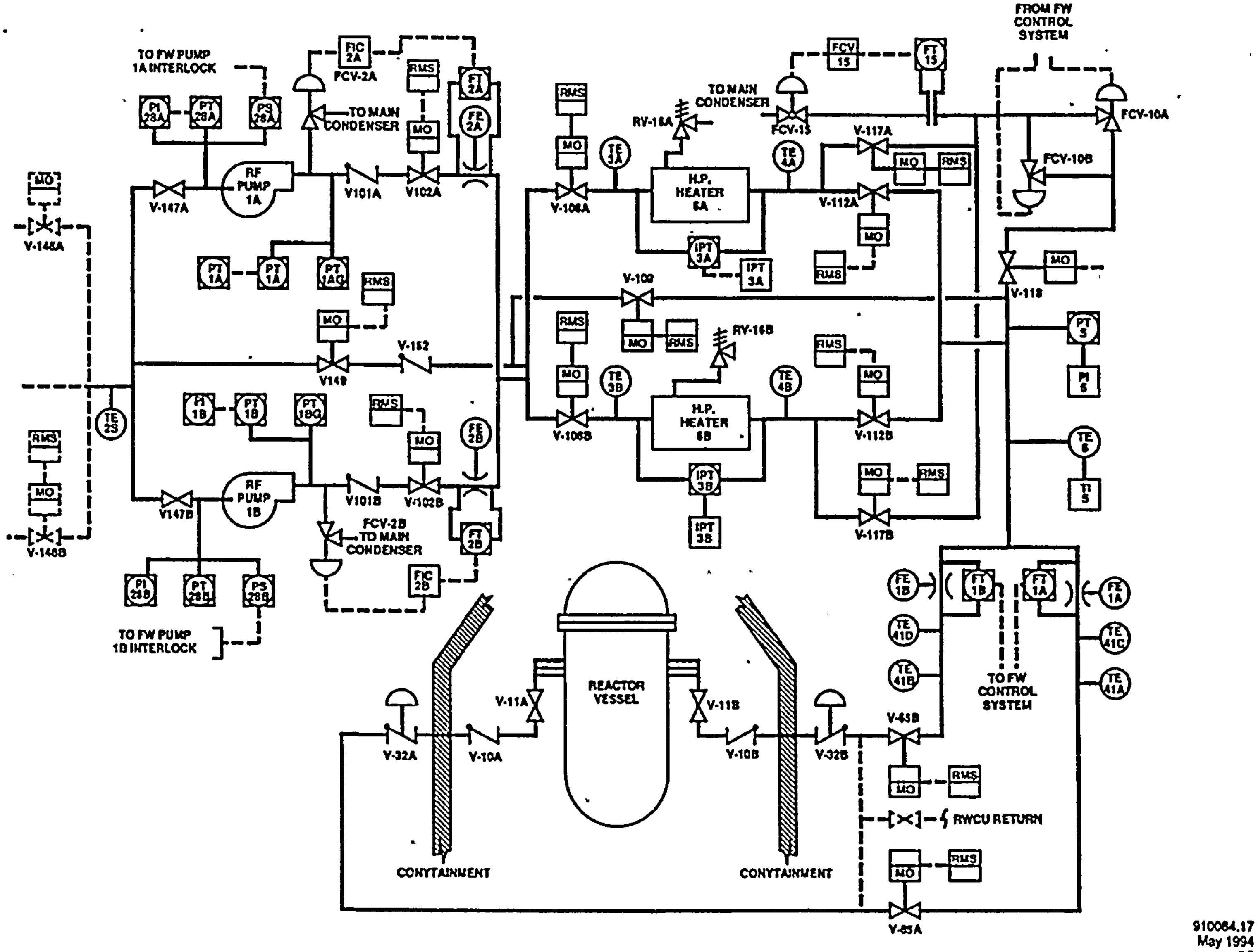

FIGURE 3.2.1-4 REACTOR FEEDWATER SYSTEM

3.2.1.5 SW System Description

The SW system is a Safety Class 1 system designed to provide cooling to the plant's safety related systems during reactor shutdown and accident conditions. The system is composed of two Standby Service Water loops and one HPCS service water loop, each containing a vertical pump and associated piping and valves. The system uses the ultimate heat sink water as coolant.

Primary functions of the system include the following:

- During normal shutdown operation to provide a heat sink for the Residual Heat Removal (RHR) system.
- During and after transient and accident conditions to provide a heat sink for the RHR system heat exchangers and the diesel generators, to serve some of the essential pump-motor units needing water cooling, and to maintain cooling to certain essential equipment, which are dependent on air cooling, by serving associated space coolers.
- After an accident to provide water for flooding of the containment if required.
- During loss of fuel pool cooling to provide make-up water in order to maintain the water level in the pool.

The SW consists of two (2) pumps driven by electric motors, two (2) cooling water spray ponds and the necessary piping, valves, instrumentation and controls as shown in Figure 3.2.1-5.

The system also consists of a third circuit which cools the HPCS system equipment. During normal operation of the plant, the SW system and the HPCS system are in the standby mode. The two loops of the SW system are maintained under pressure using the water leg pumps which prevent water hammer during pump start-up. The HPCS system has no water leg pump since all of the equipment that it serves is at grade.

During the normal and emergency shutdown mode of operation, the service water is taken from the spray ponds, routed to equipment requiring cooling during this mode of operation, and then returned to the spray ponds through circular headers connected to the spray trees for cooling of the standby service water. The HPCS loop operates in the same manner except the water is returned directly to the spray pond instead of through the spray trees. The two standby service water pumps and the HPCS service water pump each receive their power from independent electrical buses. Each bus may be powered from the two offsite power supplies or from its own diesel generator.

The standby service water pump of Loop A is located in pump house A and takes suction from Spray Pond A. Loop A serves the following equipment (partial list):

- RHR Heat Exchanger
- Diesel Engine 1A Heat Exchangers
- Cooling Coil Bank, Diesel Room A
- Cooling Coil, Diesel Room A
- Cooling Coil, LPCS Pump Room
- Cooling Coil, RHR Pump A Room
- RHR Pump Motor Cooling
- LPCS Pump Motor Cooling
- Cooling Coil, Switch Gear
- Cooling Coil, Cable Room
- Cooling Coil, Pump House
- Chiller Package Condenser
- Sampling & Analyzer Room 1A Cooling Coil
- Hydrogen Recombiner
- Hydrogen and Oxygen Analyzer
- Hydrogen Recombiner MCC Room Cooling Coils
- Fuel Pool Heat Exchanger
- Fuel Pool Pump Room Cooling Coil
- MCC Room Cooling Coils
- Battery and Battery Charger Room Cooling

The standby service water pump of Loop B is located in pump house B and takes suction from Spray Pond B. Loop B serves the following equipment (partial list):

- RHR Heat Exchanger
- Diesel Engine 1B Heat Exchangers
- Cooling Coil Bank, Diesel Room B
- Cooling Coil, Diesel Room B
- Cooling Coil, RHR Pump B Room
- RHR Pump B Motor Cooling
- Cooling Coil, RHR Pump C Room
- RHR Pump C Motor Cooling
- Cooling Coil, Switch Gear
- Cooling Coil, Cable Room
- Cooling Coil, Pump House
- Cooling Coil, RCIC Pump Room
- Diesel Generator Cable Cooling Coil
- MCC Room Cooling Coil
- Sampling & Analyzer Room 1B Cooling Coil
- Chiller Package Condenser
- Hydrogen Recombiner

- Hydrogen & Oxygen Analyzer
- Hydrogen Recombiner MCC Room Cooling Coil
- Fuel Pool Heat Exchanger
- Fuel Pool Pump Room Cooling Coil
- Battery and Battery Charger Room Cooling

Loops A and B are each provided with a branch-off upstream of the RHR heat exchangers facilitating emergency make-up to the fuel pool through a common 2" line. The branch lines each include a manual and motor operated globe valve.

Loop B is provided with a branch-off upstream of the RHR heat exchanger which facilitates transfer of water from the SW system to the containment via Loop B in the RHR system. The transfer line includes two motor operated shutoff valves plus a check valve, all located in the RHR side of the line.

Each loop is provided with a restricting orifice to maintain a higher system pressure on the RHR heat exchangers tube side than on the shell side, during the long-term reactor cooling mode to prevent any leakage of the radioactive reactor fluid into the SW system.

For optimum cooling during emergency mode, the heated service water from Loop A is returned to Spray Pond B. Similarly, the heated service water from Loop B is returned to Spray Pond A. To maintain equal water level in the spray ponds during operation, a siphon intertie line is provided.

The HPCS service water pump is located in pumphouse A, taking suction from Spray Pond A. The headers of this loop are 8" lines. The HPCS loop serves the following equipment:

- HPCS Diesel Generator Heat Exchanger
- Cooling Coil Bank, HPCS Diesel Room
- Cooling Coil, HPCS Diesel Room
- Cooling Coil, HPCS Pump Room

The heated HPCS service water is returned directly to Spray Pond A.

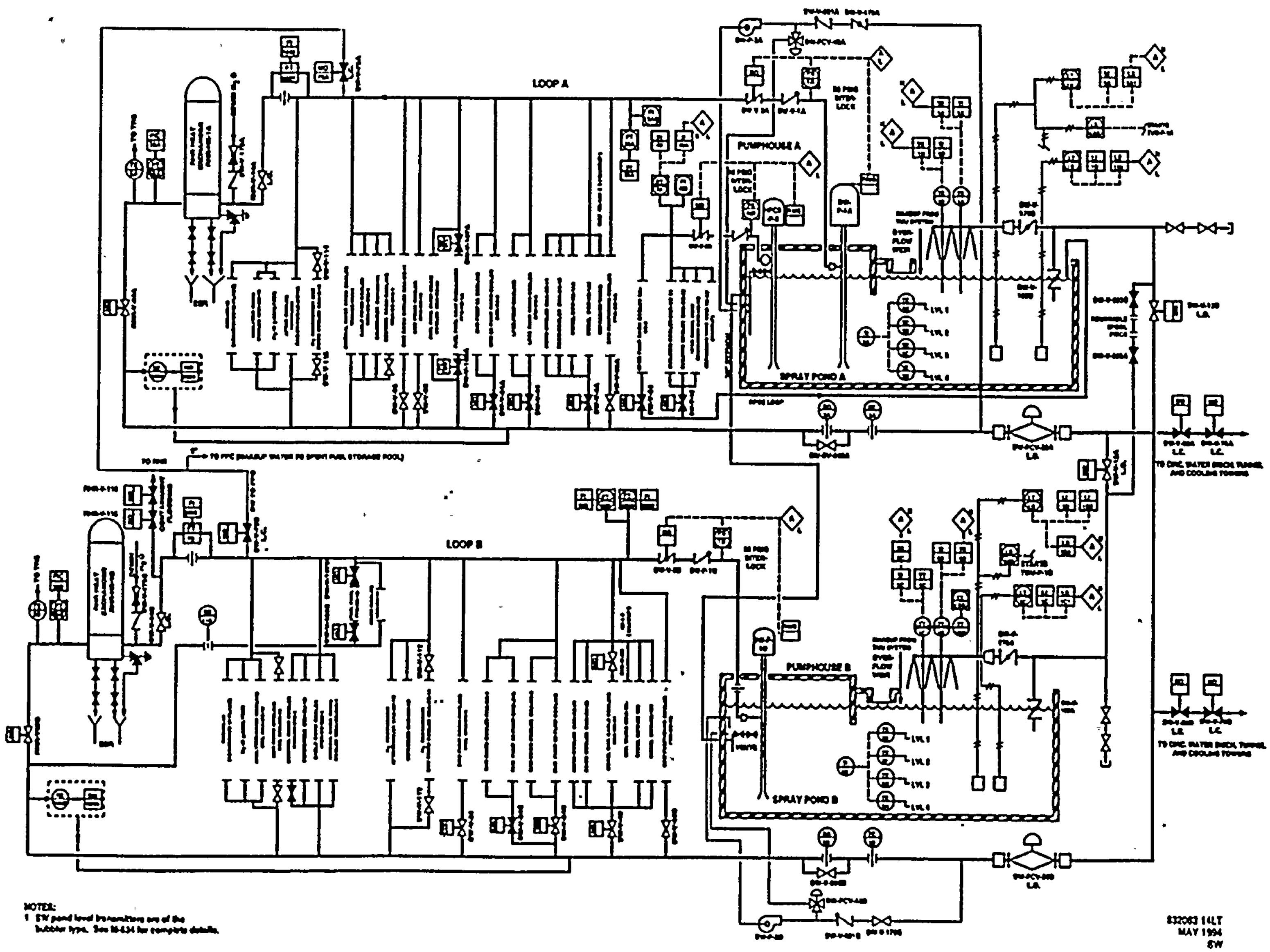

FIGURE 3.2.1-5 STANDBY SERVICE WATER SYSTEM

3.2.1.6 SLC System Description

The Standby Liquid Control system is a redundant and alternate method of manually shutting down the reactor, independent of the Control Rod Drive system. The system assures reactor shutdown by mixing a neutron absorber with the primary reactor coolant and is designed for use when control rod insertion capability is lost. The system is not a scram or a backup scram system for the reactor; it is an independent backup system for the Control Rod Drive system.

Overall Configuration - The SLC system (see Figure 3.2.1-6) consists of a heated storage tank containing a low temperature sodium pentaborate decahydrate solution, two positive - displacement pumps connected in parallel, two motor operated suction valves, two explosive actuated discharge valves, a test tank with its network of injection and recirculation pipes, and the necessary piping, valves, and instrumentation needed to inject neutron absorber solution into the reactor coolant.

The SLC system is manually initiated from the control room to pump the neutron absorber solution into the reactor via the HPCS injection line. Initiation of the system requires positive action from the main control room using keylocked switches located on P603.

The system has the capacity for controlling the reactivity difference between the steady state operating condition of the reactor with voids and the cold shutdown condition, including shutdown margin, to assure complete shutdown from the most reactive condition at anytime in core life. The time required for actuation and effectiveness of the SLC system is consistent with the nuclear reactivity rate of change predicted between rated operating and cold shutdown condition. Upon normal initiation, both suction valves will open, the pumps will start, and both explosive actuated discharge valves open. This establishes a flow path for the boron solution from the storage tank into the reactor vessel. The boron solution discharges inside the shroud through the HPCS spray header. The HPCS header is used to minimize the potential for power oscillations in the reactor (power chugging). Both trains are required to operate to meet the success criteria for SLC initiation.

The only routine operations associated with the SLC system involve monitoring the liquid level, concentration, and temperature, with periodic component testing to ensure the system will operate properly when required.

Sodium pentaborate decahydrate ($Na_2B_{10}0_{16}.10H_20$) is prepared by dissolving stoichiometric quantities of borax and boric acid in demineralized water. An air sparger is provided in the tank for mixing.

The saturation temperature of solution is 63°F at the low concentration (13.6%) and approximately 69°F at the high concentration (15%). The equipment containing the solution is installed in an environment in which the air temperature is maintained within the range of 78°F to 100°F. In addition, an electrical resistance heater system provides a backup heat source which maintains the solution temperature to prevent precipitation during storage. High or low liquid level causes an alarm in the control room.

The positive displacement pumps are sized to inject the solution into the reactor in 54 to 61 minutes, dependent on the amount of solution in the tank. The system design pressure between the pump inlet and the explosive valves is 1400 psig, and there are two relief valves to guard against overpressurization. To prevent bypass flow from one pump in case of relief valve failure in the line from the other pump, a check valve is installed down stream of each relief valve line in the pump discharge pipe.

The two explosive-actuated injection valves provide assurance of opening when needed and ensure that boron will not leak into the reactor even when the pumps are being tested. When the SLC system is actuated, storage tank liquid level, tank outlet valve position, pump discharge pressure, flow indication, and loss of continuity on the explosive valves indicate that the system is functioning. Cross piping and check valves assure a flow path through either pump and either explosive valve. Pump discharge pressure and system flow is indicated in the control room.

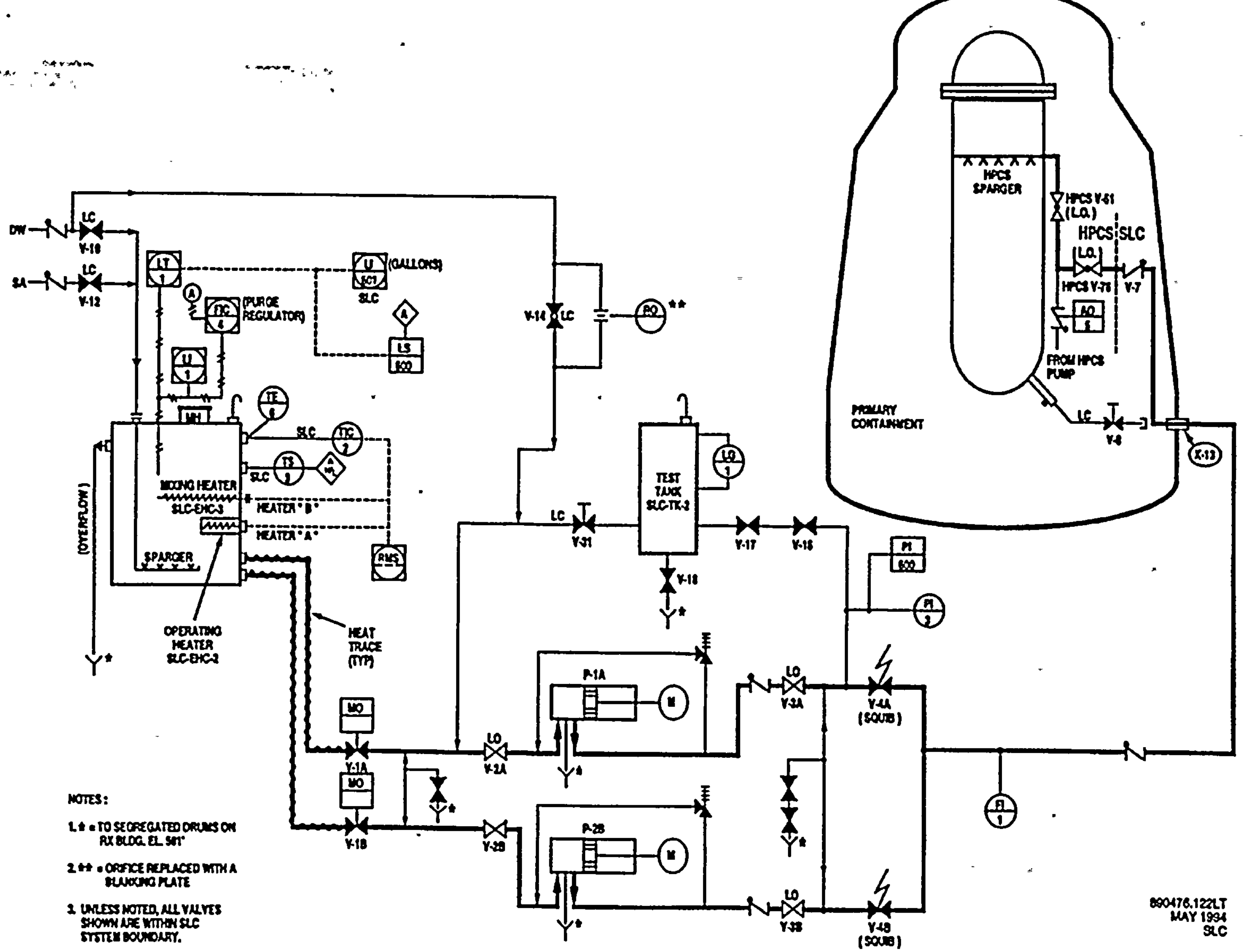

FIGURE 3.2.1-6 SLC FLOW DIAGRAM

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3.2.1.7 DC System Description

The Direct Current (DC) Electrical Power system is designed to provide a reliable source of DC power for the Plant's normal operation, Engineered Safety Features (ESF) Logic, Reactor Core Isolation Cooling (RCIC), High Pressure Core Spray (HPCS), Containment Atmospheric Control (CAC), Residual Heat Removal (RHR) valves, Diesel Generator (DG), Turbine Generator (TG) and Reactor Feedwater Turbine (RFT) oil pumps, Medium Voltage (4160v) and Low Voltage (480v) Switchgear as well as Security system loads.

The overall DC electrical power distribution system consists of ten separate DC systems at different voltage levels and importance to safety. Generally, each of these systems consists of a battery, a battery charger and a main distribution panel (bus) with further division down to motor-control centers and/or power panels and individual equipment loads. During normal operation the charger supplies the load and maintains a float-voltage of 2.25 volts per cell on the battery. The individual system battery is sized to supply all emergency DC loads for a minimum of two hours during loss of AC power, during accident conditions. For class 1E battery capacity, the B1-1 and B1-2 batteries have adequate capacity for a four hour station blackout coping period without load shedding (FSAR Section 1.5.2.2.5.2). Upon restoration of AC power, the battery charger is sized to supply the emergency loads, and to recharge the battery within twenty-four hours. The DC electrical power distribution system is provided as follows:

A: Generating Plant Safety Related DC Power Systems

The plant's safety related DC loads are powered from six separate, Class 1E, DC power distribution systems. These systems consist of two $\pm$ 24 volt systems (Division 1 and 2), three 125 volt systems (Division 1, 2 and 3) and one 250 volt system (Division 1). The safety related loads powered by these systems are divided to enable operation of redundant systems and equipment for plant shutdown. These distribution systems are designed and installed as Quality Class 1, Seismic Category I.

The Division 1 and 2, $\pm$ 24 volt DC systems provide redundant sources of DC power for the Startup Nuclear Instrumentation, the Process Radiation Monitoring system, and selected Bypass-Inoperative Status Indication (BISI) systems. Each of these DC systems consists of two, twelve-cell (EXIDE type 3CC-7), 150 amp-hour (nominal) Lead-Calcium batteries connected in series with the center leg grounded and two battery chargers each. The batteries are supported on Exide two-tier, seismic racks. The battery chargers are Power Conversion Products; single-phase, thyristor controlled, constant potential type chargers with single-phase 120 vac inputs and 24 vdc/25 amp outputs. See Figure 3.2.1-7.1.

The Division 1 and 2, 125 volt DC systems provide redundant sources of DC power for normal plant operation and operation of the emergency AC power systems and ESF systems. These DC systems each consist of a 58-cell (EXIDE type GN-13), 1140 amp-hour (nominal) Lead-Calcium battery and a battery charger. The batteries are supported on Exide two-tier, seismic racks. The battery chargers are Power Conversion Products; three-phase, thyristor controlled, constant potential type chargers with three-phase 480 vac inputs and 130 vdc/200 amp outputs. See Figure 3.2.1-7.2. A third 125 volt DC system (Division 3) is provided for the HPCS system. This DC system consists of a 58-cell (C & D type 3DCU-9), 100 amp-hour (nominal) Lead-Calcium battery and a battery charger. The batteries are supported on C & D two-step, earthquake protected racks. The battery charger is a C & D Battery, Inc.; three-phase, thyristor controlled, constant potential type charger with a three-phase 480 vac input and a 125 vdc/50 amp output. See Figure 3.2.1-7.3.

Since there are no redundant 250 volt loads for essential systems, only one 250 volt DC system is required. This system provides 250 volt power to RCIC pumps and valves and to the emergency oil pumps associated with the Main Turbine and the RFW Drive Turbines, valves RHR-V-23 and RWCU-V-4 as well as Inverter E-IN-1. This system consists of a 2520 amp-hour (nominal) Lead-Calcium battery, made up of two parallel strings of 116-cells (EXIDE type GN-15) each, and a battery charger. The batteries are supported on Exide two-tier, seismic racks. The battery charger is a Power Conversion Products; three-phase, thyristor controlled, constant potential type charger with a three-phase 480 vac input and a 260 vdc/400 amp output. See Figure 3.2.1-7.4.

B. Generating Plant Nonsafety related DC Power System

The plant's nonsafety related direct current control and instrumentation loads are powered from a separate, non-Class 1E, 125 volt power distribution system. This system consists of a 58-cell (EXIDE type GN-13), 1140 amp-hour (nominal) Lead-Calcium battery and a battery charger. The batteries are supported on Exide two-tier, seismic racks. The battery charger is a Power Conversion Products; three-phase, thyristor controlled, constant potential type charger with a three-phase 480 vac input and 130 vdc/200 amp output. This system is designed and installed as Quality Class 2, Seismic Category I. See Figure 3.2.1-7.5.

C. Makeup Water Pump House DC Power System

The switchgear control and ground-fault detection systems at the Makeup Water Pump House are powered by two non-Class 1E, 125 volt DC distribution systems (Division A and B). These redundant systems each consist of a 58-cell (EXIDE type 3CC-5), 100 amp-hour (nominal) Lead-Calcium battery and a battery charger. The batteries are supported on EXIDE two-tier, seismic racks. The battery chargers are Power Conversion Products; single-phase, thyristor controlled, constant potential type chargers with single-phase 120 vac inputs and 125 vdc/25 amp outputs. These nonsafety related systems were designed and installed as Quality Class 2, Seismic Category II. See Figure 3.2.1-7.6.

D. Security System DC Power System

This DC power distribution system (Division B) consists of a nonredundant, 480 volt AC, three-phase Uninterruptible Power system (UPS) with Static Bypass to provide emergency power to the plant's security loads for one hour. The battery portion of this UPS consists of a non-Class 1E, 180-cell (EXIDE type EX-31), 620 amp-hour (nominal), 405 volt Lead-Calcium battery. The batteries are supported on Exide two-tier, seismic racks. The rectifier-charger portion of the UPS is a twelve pülse, phase controlled solid state type unit with constant voltage and constant current circuitry. This UPS system was designed and installed as Quality Class 2, Seismic Category II. See Figure 3.2.1-7.7.

E. Other Batteries/Chargers

There are other batteries and battery chargers within the plant which are included in the systems which they support and are not considered part of the plant's DC electrical power system. They are:

- The three 12 volt automotive-type batteries associated with the starting air systems for the Division 1, 2 and 3 Emergency Diesel Generators.

- In addition, there are numerous battery-powered lighting units throughout the plant associated with the Emergency Lighting system.

- Four 24 volt batteries, two each associated with the starting systems for both Fire Protection diesels (fire-pumps FP-P-1 and -110) and,

- Two 24 volt Pyrotronics, Inc., battery/battery charger modules for the fire suppression control panels in the document vault and at the Makeup Water Pump House.

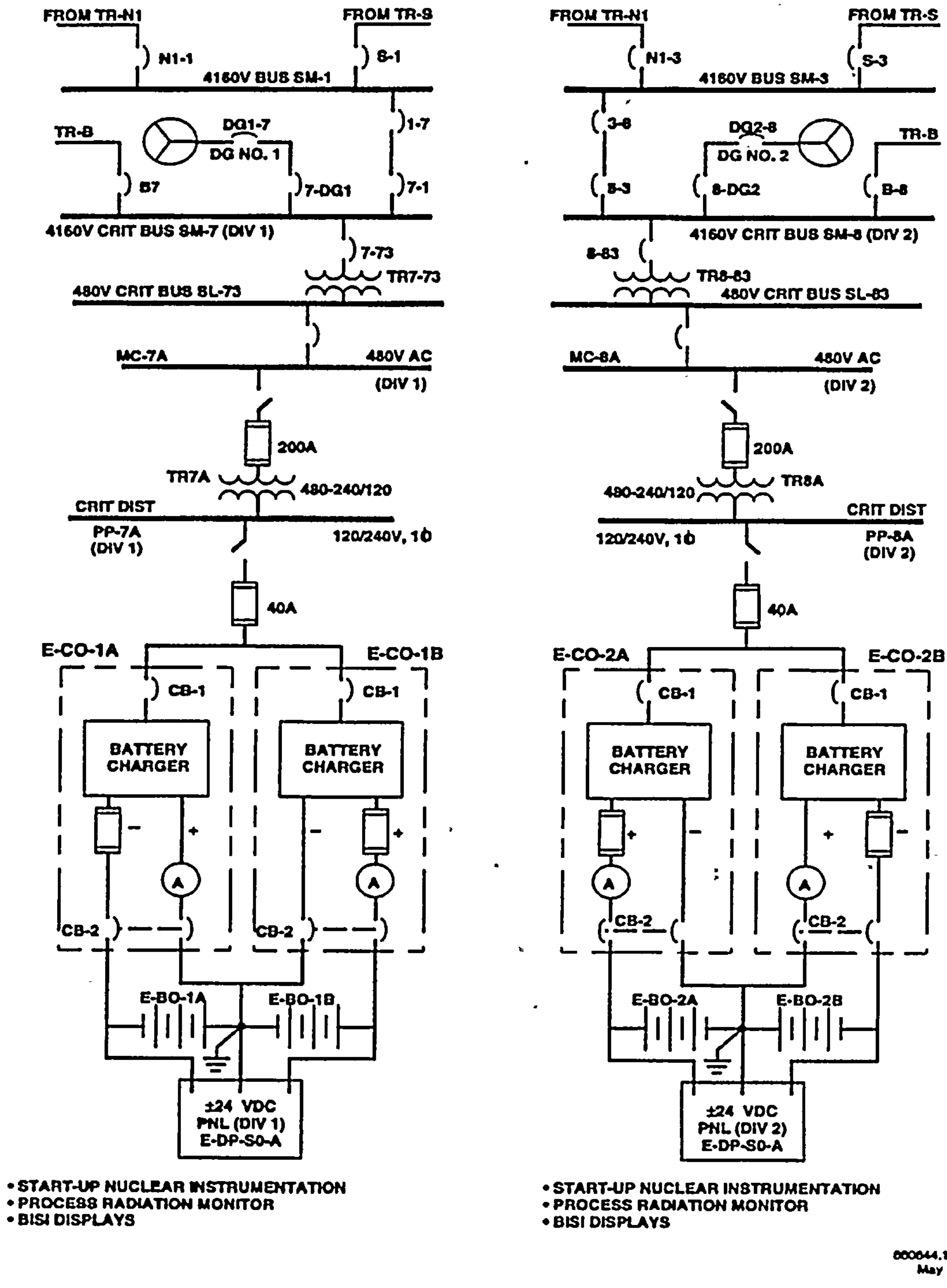

FIGURE 3.2.1-7.1 ±24 VDC SYSTEMS

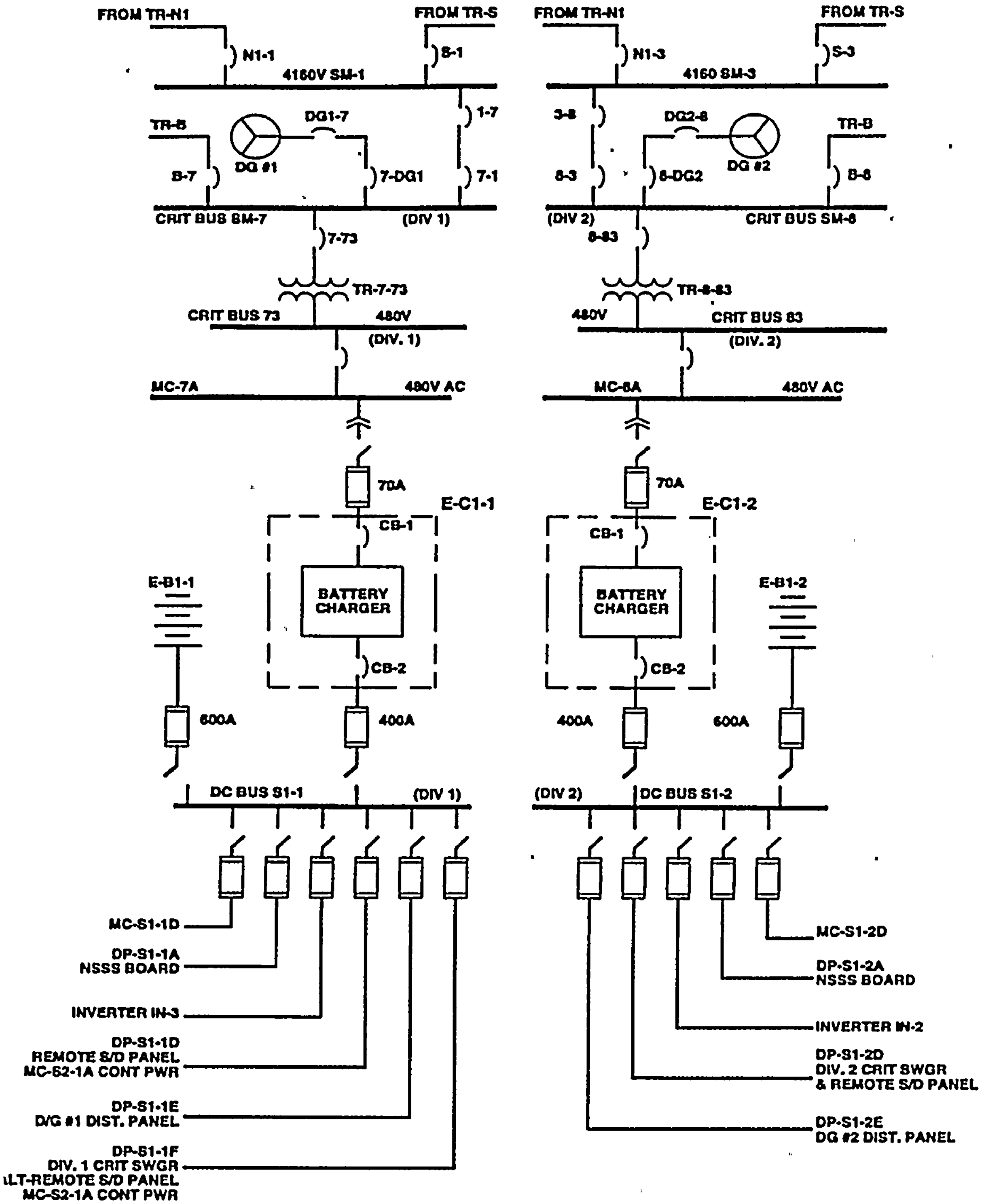

FIGURE 3.2.1-7.2 125V DC SYSTEMS

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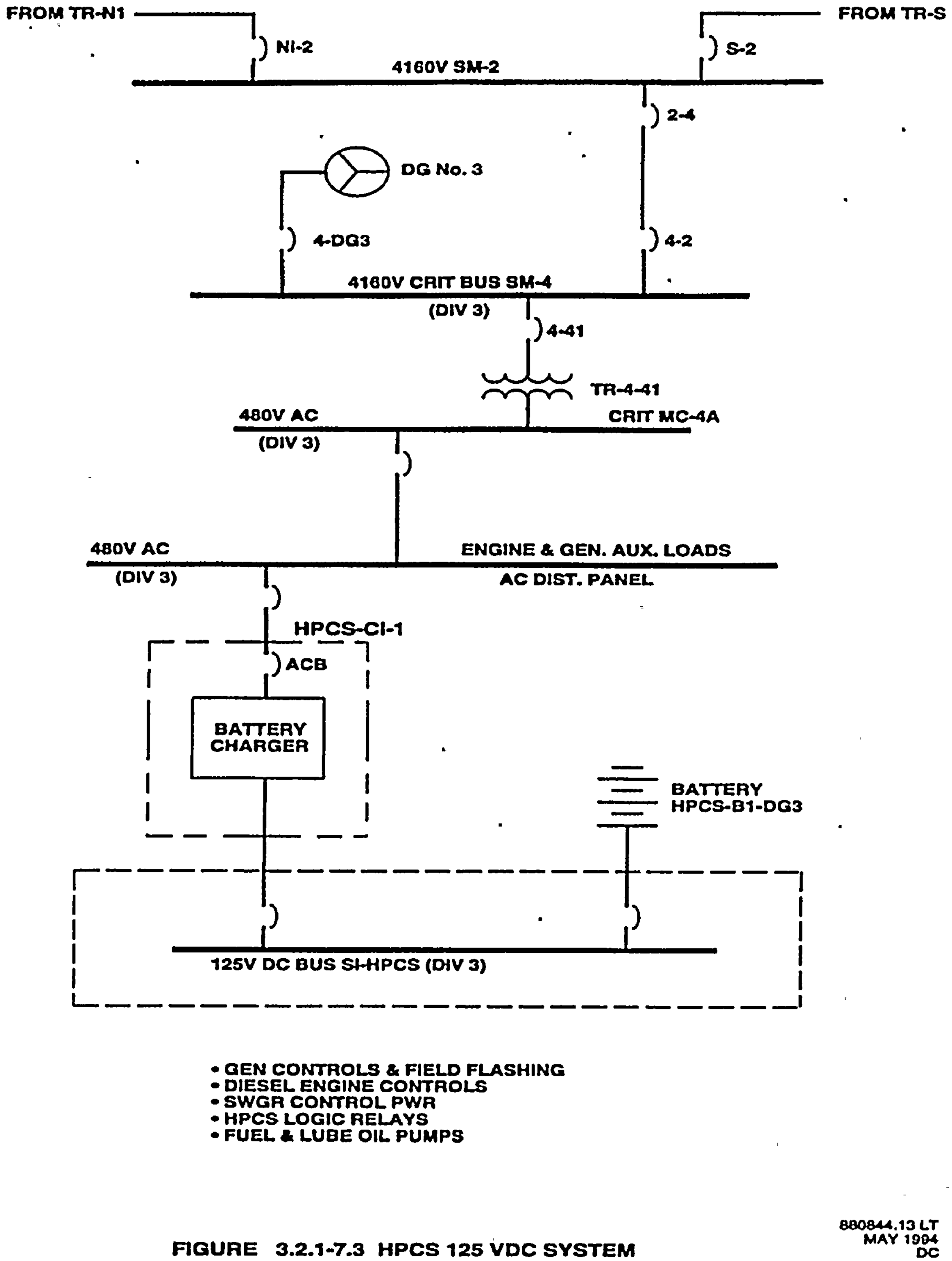

FIGURE 3.2.1-7.3 HPCS 125 VDC SYSTEM

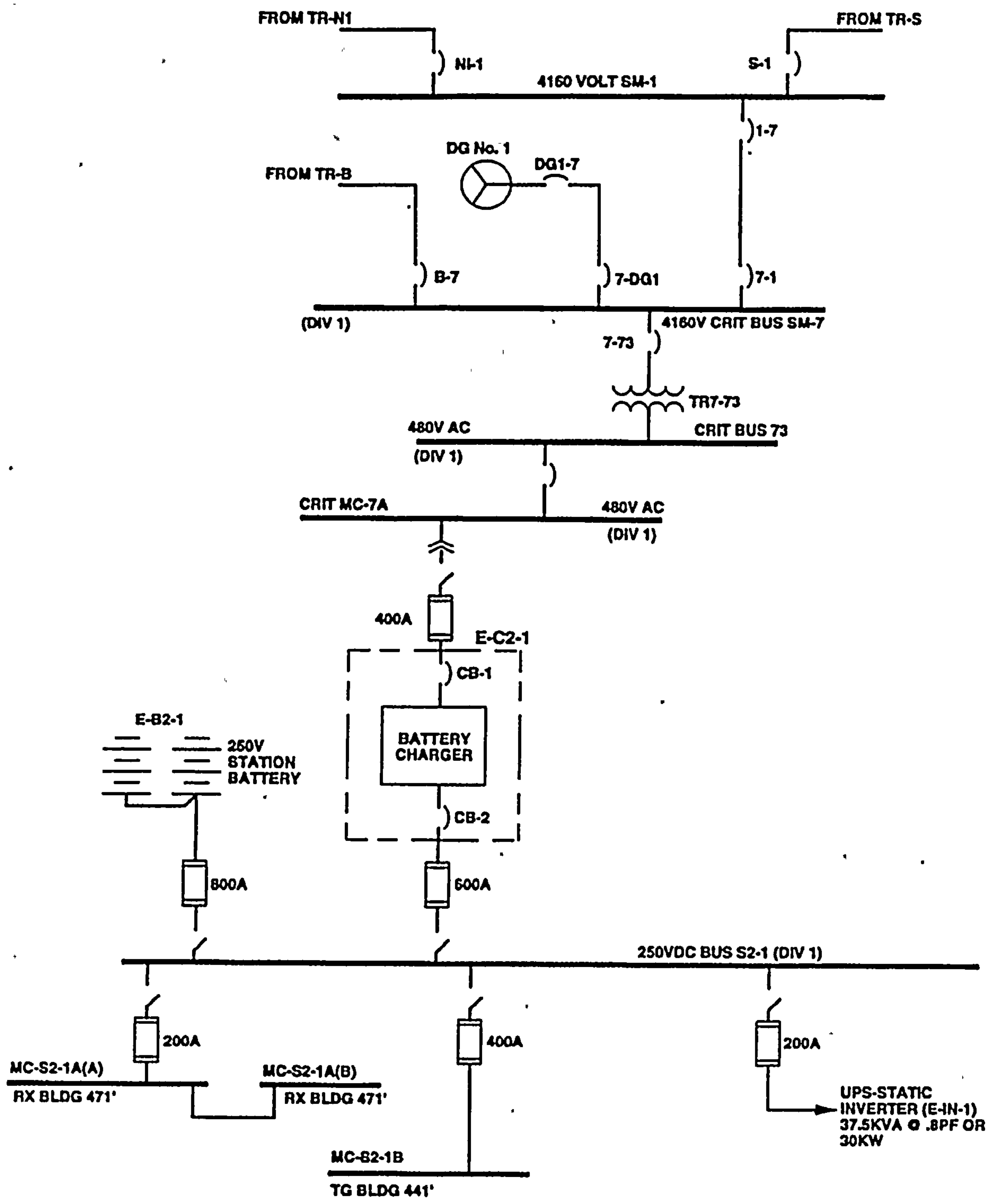

FIGURE 3.2.1-7.4 250V DC SYSTEM

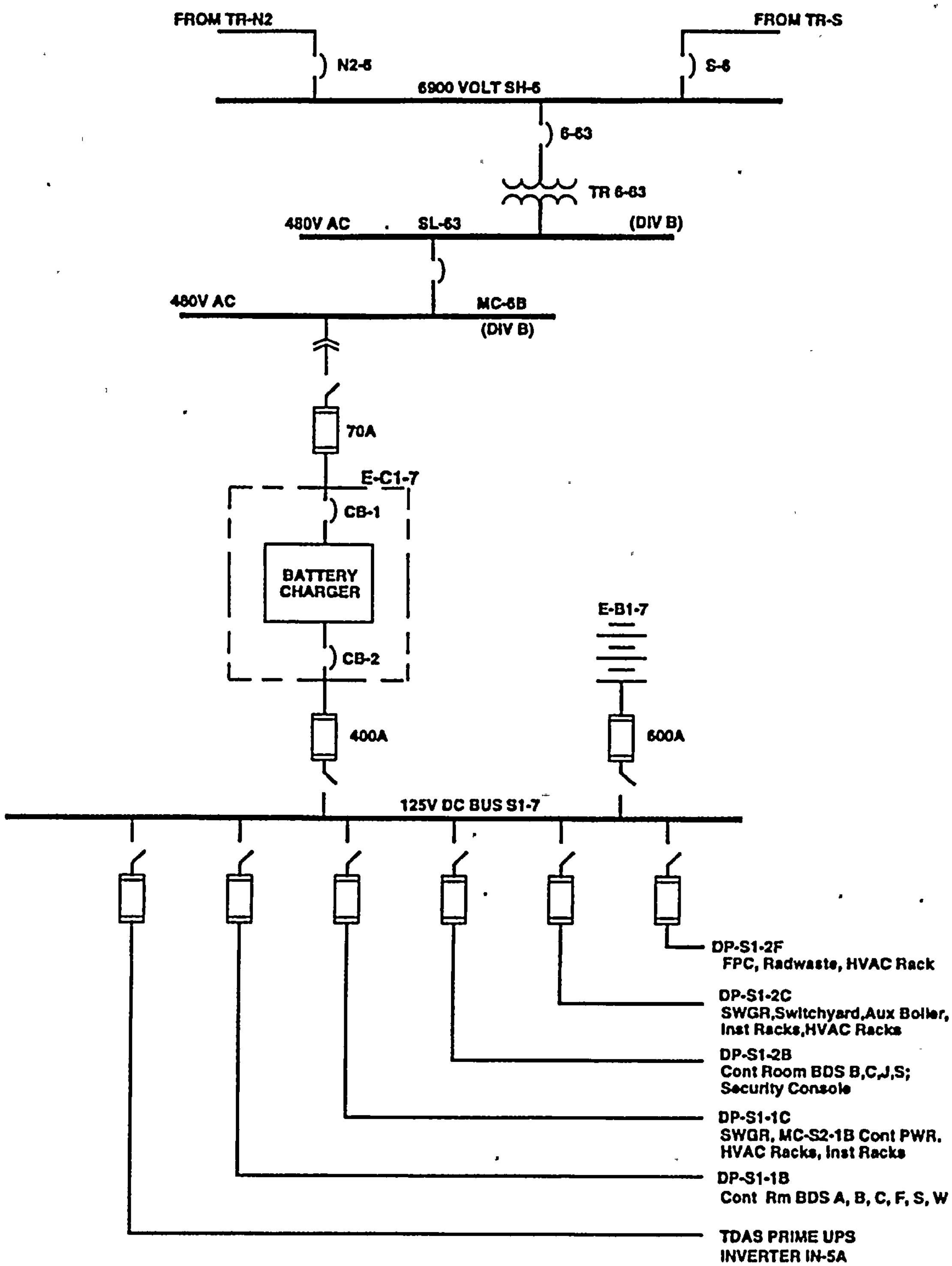

FIGURE 3.2.1-7.5 NON-CLASS 1E 125 VDC SYSTEM (SI-7)

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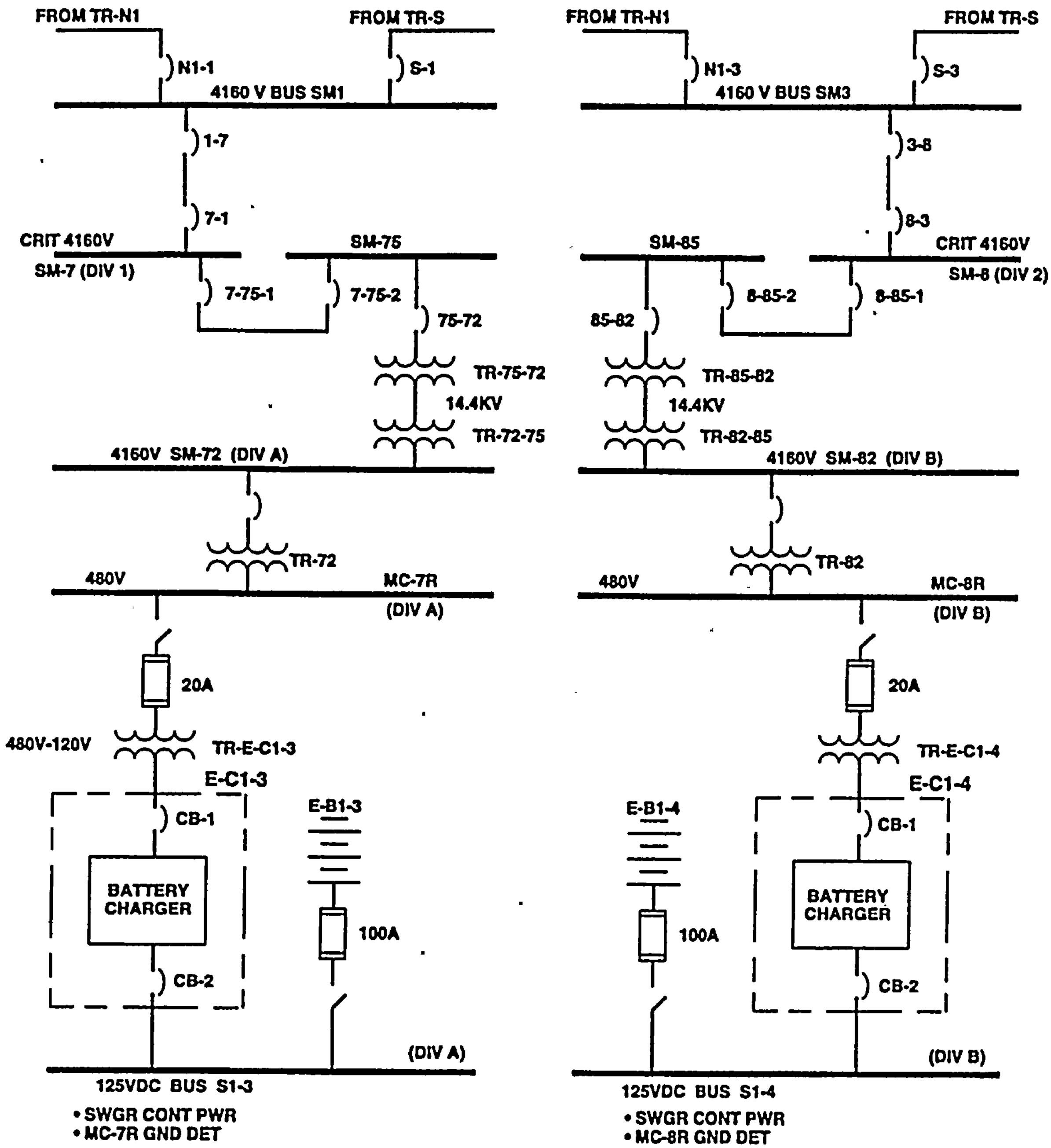

FIGURE 3.2.1-7.6 TMU PUMPHOUSE 125V DC SYSTEM

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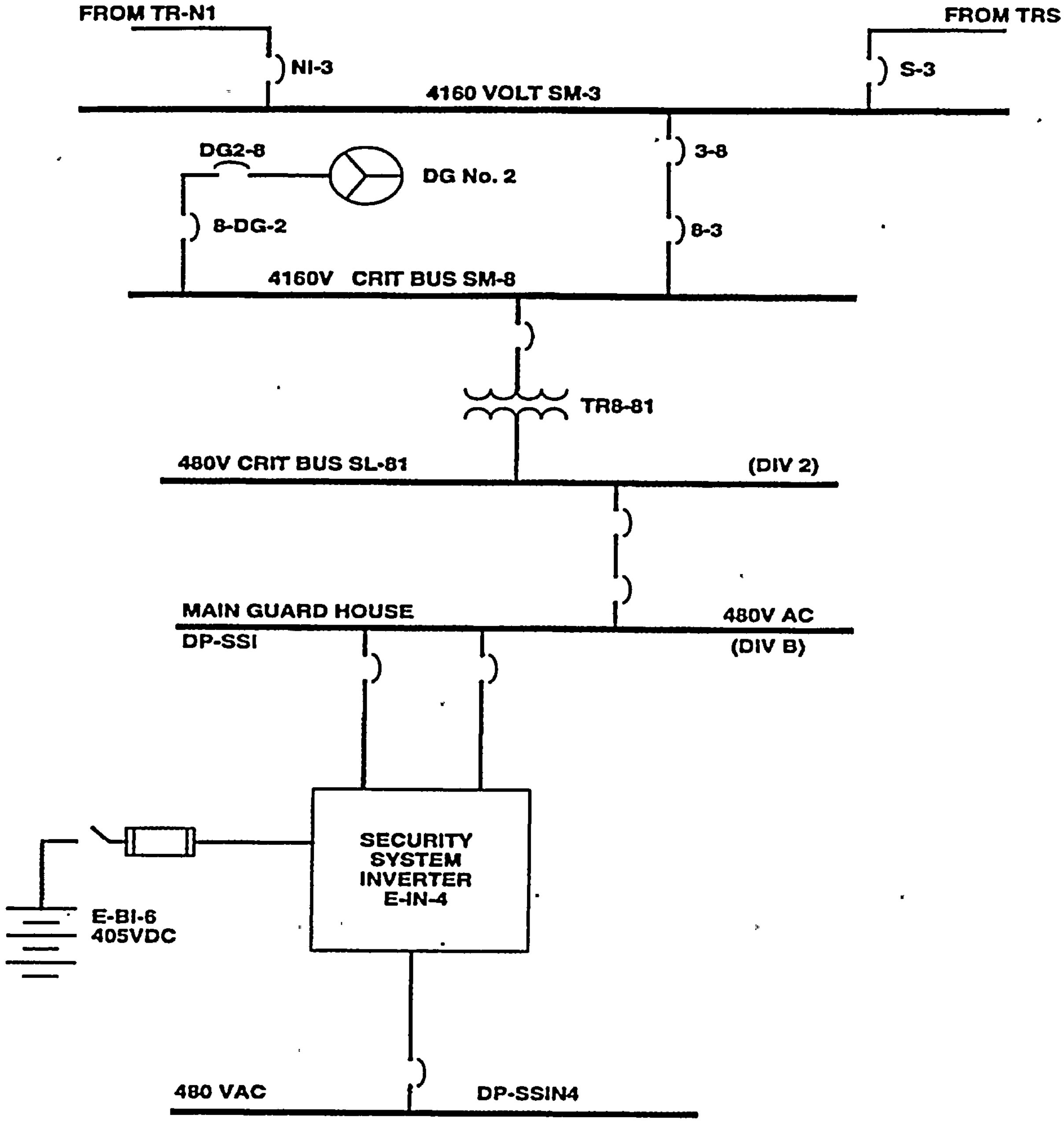

FIGURE 3.2.1-7.7 SECURITY 405 VDC SYSTEM

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3.2.1.8 HPCS System Description

The HPCS is an emergency core cooling system which pumps water through a peripheral ring spray sparger mounted above the reactor core. Coolant is supplied over the entire range of reactor system operating pressures. The primary purpose of HPCS is to maintain reactor vessel inventory after small breaks which do not depressurize the reactor vessel. HPCS also provides spray cooling heat transfer during large breaks which uncover the core.

The HPCS is designed to operate from normal power for surveillance testing and operation, from offsite auxiliary power or from a dedicated standby diesel generator power supply if offsite power is not available. System operation is initiated automatically upon reactor coolant system LOCA. During normal standby, valving is aligned to achieve injection of coolant with a minimum of valve position changes. Automatic system control features realign valve positions if the system is in a test mode upon receipt of an initiation signal generated by a LOCA.

The HPCS may be used to transfer water from condensate storage to fill the suppression pool during normal and emergency plant conditions, and may be used to backup the RCIC. Provisions for periodic surveillance testing, maintenance of system readiness, and continuous monitoring of system status are included in the design.

The HPCS consists of a single motor driven centrifugal pump, a spray sparger in the reactor vessel located above the core (separate from the LPCS sparger), and associated system piping, valves, controls, and instrumentation. A simplified system flow diagram is shown in Figure 3.2.1-8.

With the exception of the testable check valve on the discharge line, all active HPCS equipment is located outside the primary containment. Suction piping is provided from the condensate storage tanks and the suppression pool. Such an arrangement provides the capability to use reactor grade water from the condensate storage tanks when the HPCS system functions to backup the RCIC system. In the event that the condensate storage water supply becomes exhausted or is not available, automatic switchover to the suppression pool water source will assure a water supply for continuous operation of the system. HPCS pump suction is also automatically transferred to the suppression pool if the suppression pool water level exceeds a prescribed value. The condensate storage tanks reserve a minimum of 135,000 gallons of water for use by HPCS and RCIC.

After the HPCS injection piping enters the reactor vessel, it divides and enters the core shroud at two points near the top of the shroud. A semi-circular sparger is attached to each outlet. Nozzles are spaced around the spargers to spray the water radially over the core and into the fuel assemblies. The Standby Liquid Control (SLC) system also injects through the HPCS spargers, whether or not there is flow from HPCS.

The HPCS discharge line to the reactor is provided with two isolation valves. One of these valves is an air actuated testable check valve located inside the drywell as close as practical to the reactor vessel. HPCS injection flow causes the check valve to open during LOCA conditions (i.e., neither power nor air is required for valve actuation during injection). If the HPCS line should break outside the containment, the check valve in the line inside the drywell will prevent loss of reactor water outside the containment. The other isolation valve (which is also referred to as the HPCS injection valve) is a motor operated gate valve located outside the primary containment as close as practical to HPCS discharge line penetration into the containment. This valve is capable of opening with the maximum differential pressure across the valve expected for any system operating mode including HPCS pump shutoff head. Containment isolation valve 3/4" leak test connections are provided with two normally closed valves to assure containment integrity.

A low water level signal or a high drywell pressure signal initiates the HPCS and its support equipment. The system can also be placed in operation manually.

The system is designed to pump water into the reactor vessel over a wide range of pressures. For small breaks that do not result in rapid reactor depressurization, the system maintains reactor water level and depressurizes the vessel. For large breaks the HPCS system cools the core by spray. The HPCS system delivers rated flow into the reactor vessel within 27 seconds following receipt of the automatic initiation signal.

The HPCS automatically stops with a high water level in the reactor vessel by signaling the injection valve to close and it automatically starts again when a low water level is signaled. The HPCS system also serves as a backup to the RCIC system in the event feedwater flow is lost.

When the system is started, initial flow rate is established by primary system pressure. At a reactor pressure of 1130 psig, a design basis 1550 gpm core spray flow is maintained until reactor pressure decreases or the vessel is intentionally depressurized. As vessel pressure decreases, flow will increase. When vessel pressure decreases to a 200 psi differential between the reactor vessel and the suction source (either the condensate storage tank or suppression pool), the system reaches its rated core spray flow of 6350 gpm. The HPCS pump motor size is based on peak horsepower requirements.

The elevation of the HPCS pump is sufficiently below the water level of both the condensate storage tanks and the suppression pool to provide a flooded pump suction and to meet pump NPSH requirements with the containment at atmospheric pressure and the suction strainer 50% plugged.

A motor operated valve is provided in the suction line from the suppression pool. The valve is located as close to the suppression pool penetration as practical. This valve is used to isolate the suppression pool water source when HPCS system suction is from the condensate storage system.

A check valve, flow element, and restricting orifice are provided in the HPCS discharge line from the pump to the injection valve. The check valve is located below the minimum suppression pool water level and is provided so the piping down stream of the valve can be maintained full of water by the discharge line fill system. The flow element is provided to measure system flow rate during LOCA and test conditions and for automatic control of the minimum low flow bypass gate valve. The measured flow is indicated in the main control room. The restricting orifice was sized during the system preoperational test to limit system flow to the values given on the HPCS system process diagram to prevent pump runout.

A low flow bypass line with a motor operated gate valve connects to the HPCS discharge line upstream of the check valve on the pump discharge line. The line bypasses water to the suppression pool to prevent pump damage when other discharge line valves are closed. The valve automatically closes when flow in the main discharge line is sufficient to provide required pump cooling, and automatically reopens when flow in the main discharge line drops below minimum flow requirements. The bypass valve is normally closed, interlocked with the pump breaker to remain closed until a pump start signal is present.

To assure continuous core cooling, primary containment isolation signals do not interfere with HPCS operation.

The HPCS system incorporates relief valves to protect the components and piping from inadvertent overpressure conditions. One relief valve is located on the discharge side of the pump down stream of the check valve to relieve thermally expanded fluid. A second relief valve is located on the suction side of the pump.

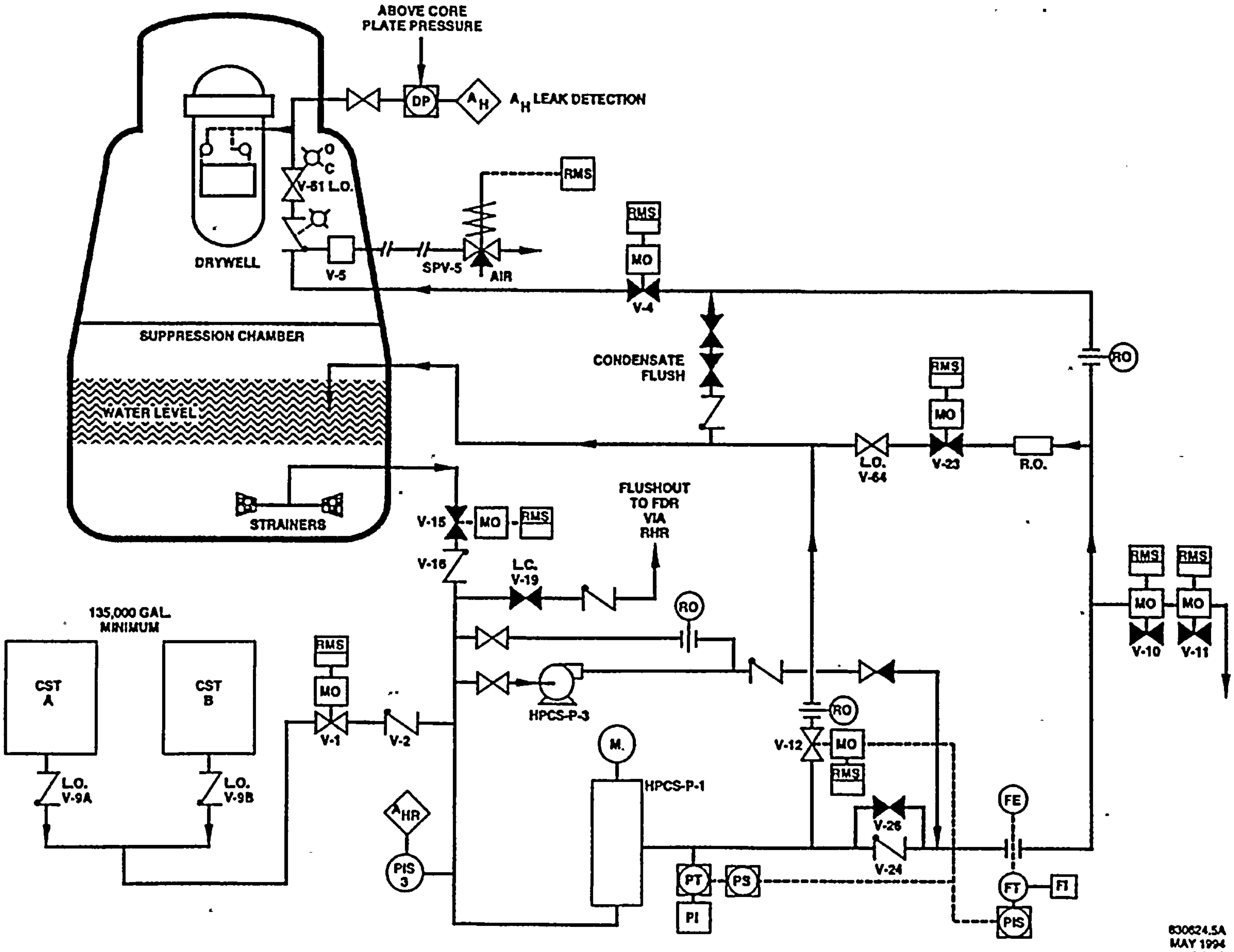

FIGURE 3.2.1-8 HIGH PRESSURE CORE SPRAY FUNCTIONAL DIAGRAM

3.2.1.9 RCIC System Description

The RCIC is designed to supply makeup water to the Reactor Vessel when the Reactor is isolated from the Main Condenser with the Reactor Feedwater system not in operation.

The system is to allow complete plant shutdown under conditions of loss of normal feedwater by maintaining sufficient water inventory until the Reactor is depressurized to a level where the Shutdown Cooling system is placed in operation. The system can be manually initiated or will automatically initiate on a Level 2 low reactor water level to maintain vessel inventory and prevent activation of the low pressure ECCS.

During an ATWS emergency, a flexible hose can be connected between the SLC system and the RCIC pump suction to allow the RCIC to borate the reactor vessel in the event of SLC malfunction. The RCIC is also designed to be operated from the remote shutdown panels.

Overall Configuration - The RCIC system consists of a steam-turbine driven pump and associated valves and piping capable of delivering water to the Reactor Vessel. Figure 3.2.1-9 is a simplified diagram of the system.

The RCIC Turbine is driven by the steam produced from decay heat. The steam is extracted from Main Steam Line B upstream of the main steam isolation valves. Water is normally taken from the Condensate Storage Tank (CST). In the event that the water supply from the condensate storage tank becomes exhausted, level instrumentation initiates an automatic switchover to the suppression pool as the water source for the RCIC pump. In addition, the operator does have the option to manually shift suction to the suppression pool. Water from either source is pumped into the Reactor Vessel via the head spray/RCIC injection line.

Turbine exhaust is directed to the suppression pool where it is condensed. The RCIC turbine casing and exhaust piping is protected against overpressure by a dual rupture disk arrangement.

To protect the RCIC pump from overheating if it is run against shutoff head or at low flow rates, a motor operated minimum flow bypass valve allows flow to the suppression pool. The bypass valve is capable of operating within five seconds against full differential pressure.

To facilitate operational testing of the RCIC system, a line branches off the discharge piping to allow recirculation to the CST. Two motor operated valves, a test flow control valve and a test shutoff valve allow system testing from the control room. Both valves will automatically close if there is a RCIC initiation signal or the suppression pool suction valve is fully open.

Some of the RCIC pump discharge flow is directed through a pressure control valve and a cooling water supply valve. The flow then passes through the tubes of the lube oil cooler and into the barometric condenser.

The lube oil cooler removes heat from the turbine lubricating oil system. Oil flow through the lube oil system is accomplished by a turbine attached lube oil pump. The lube oil pump supplies oil to the turbine bearings and to the governor valve control system.

The RCIC barometric condenser processes gland leak off from the governor valve, trip and throttle valve, and seals from the turbine shaft glands to contain radioactive steam. The steam is condensed by a water spray from the lube oil cooler water effluent supply line. Noncondensables are removed by a 250V DC powered vacuum pump and discharged to the suppression pool. Condensate and liquid from the spray is pumped back either to the suction side of the RCIC during RCIC operation or to the EDR when a high level is reached in the vacuum tank with RCIC-V-45 closed. Startup of the barometric condenser equipment is automatic upon RCIC system initiation. However, failure of the barometric condenser equipment does not prevent the RCIC system from fulfilling its design objectives.

The RCIC system controls automatically start the system and bring it to the design injection flow rate within 30 seconds after receipt of a Reactor Vessel Level 2 water level signal. The RCIC system automatically stops either when a high water Level 8 in the Reactor Vessel is signaled, low steam supply pressure is signaled, or when other system parameters generate a trip signal.

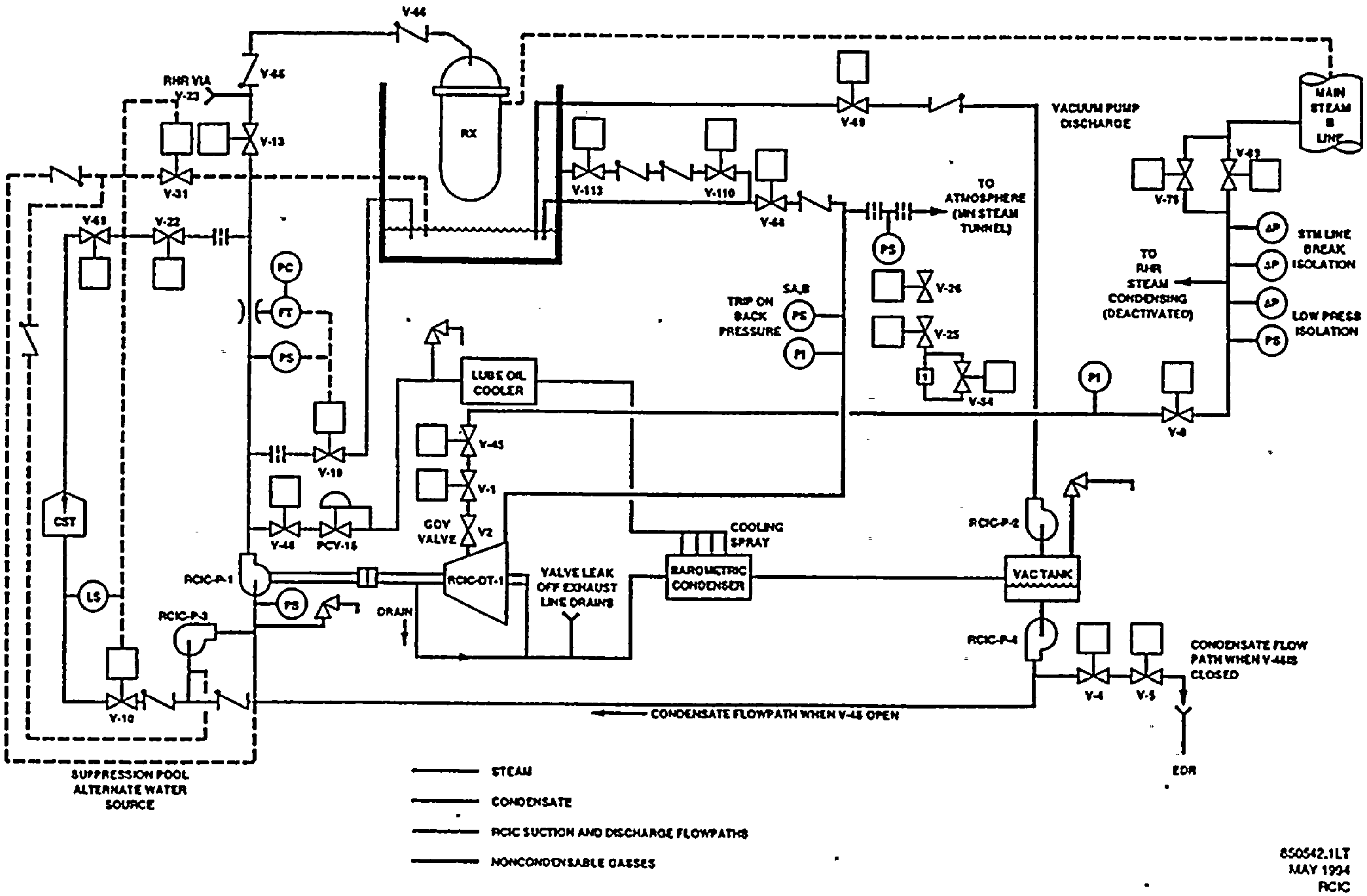

FIGURE 3.2.1-9 RCIC

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3.2.1.10 RHR System Description

The RHR system is one of several systems that protect the reactor core and fuel against overheating. This system performs four major functions which comprise the following operating modes:

- Low Pressure Coolant Injection (LPCI)
- Containment Spray Cooling
- Suppression Pool Cooling
- Shutdown Cooling

The low pressure coolant injection system is an operating mode of the RHR system. The LPCI system is automatically actuated by low water level in the reactor and/or high pressure in the drywell and, when reactor vessel pressure is low enough, uses the three RHR motor-driven pumps to draw suction from the suppression pool and inject cooling water flow into the reactor core to cool the core by flooding. Each loop has its own suction and discharge piping and separate vessel nozzle which connects with the core shroud to deliver flooding water on top of the core. The system is a high volume core flooding system.

Containment spray cooling mode is established by operator action to lower the ambient pressure in the containment and suppression pool. Suppression pool water is circulated through the RHR heat exchanger to spray spargers in the drywell and suppression pool. The functional design basis for the containment spray cooling mode is that there should be two redundant means to spray into the drywell and suppression pool vapor space to reduce internal pressure to below design limits.

Suppression pool cooling mode is established by operator action to lower the temperature of suppression pool water which is absorbing heat from the vessel introduced by exhaust steam from the RCIC turbine, from the main steam relief valves, or from the vessel and pipe in general. Suppression pool water is circulated through the RHR heat exchangers which cools this water. This mode is also used to cool the containment following a loss-of-coolant accident (LOCA).

Shutdown cooling mode is established by operator action to remove residual heat from the reactor vessel after pressure in the reactor has been reduced to less than 48 psig following blowdown to the main condenser. In this mode reactor water is taken from a recirculation suction line, pumped through the RHR heat exchangers and returned to the reactor vessel via the recirculation discharge lines. Part of the flow can be diverted by operator action to spray nozzles in the vessel head to help collapse the steam bubble in the dome.

Besides these four major functions the RHR system can perform the following secondary functions:

- Fuel Pool Cooling Assist Mode (Loop "B")
- Head Spray Mode - mentioned under Shutdown Cooling Mode (Loop "B")
- Containment Flooding Mode (Loops "A," "B," "C")
- Full Flow Test Mode (Loop "A," "B," "C")
- Pumping SW into the core through RHR system

In addition the RHR system, normally receiving electrical power from the AC distribution system, can operate from the DG system.

Overall Configuration - The RHR system consists of three pumped systems, two with heat exchangers and one without, and associated valves piping and controls capable of performing the four functions described in the previous section. Figure 3.2.1-10 is a simplified diagram of this system. The initial design of the RHR system had a fifth mode of operation, steam condensing. This mode was deactivated. Most of the piping, valves and instrumentation for this mode of operation were installed in the plant. This equipment has been de-energized, locked closed, or otherwise rendered inoperable and has been deleted from Figure 3.2.1-10 for clarity.

The low pressure RHR system (and LPCS) have deep fill systems to help prevent the potential for water hammer during rapid initiation (for example, automatic ECCS signal). The consequences of a water hammer that ruptures one of the low pressure system piping could be severe in that it could flood other redundant equipment or systems or result in loss of suppression pool inventory resulting in loss of suction supply to other ECCS injection systems. However, this potential failure mode of the low pressure ECCS systems was not modeled separately in the IPE. The unavailability of the keep-fill system is included in the unavailability due to maintenance of the ECCS system. The probability of a random failure of the keep-fill concurrent with a demand for ECCS injection has been shown to be negligible (IDCOR BWR Methodology Document).

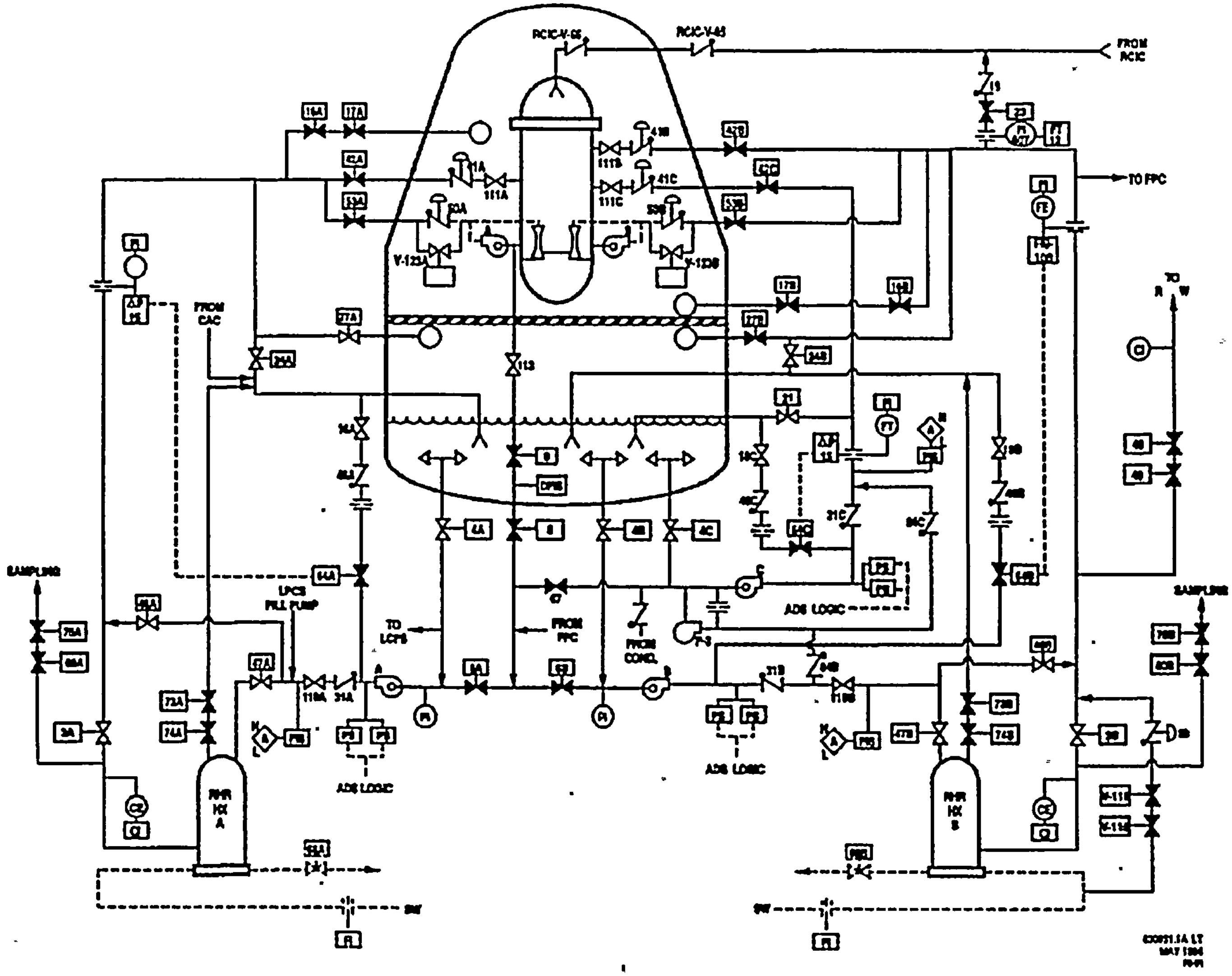

FIGURE 3.2.1-10
RHR System Normal Standby Configuration

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3.2.1.11 TSW System Description

The purpose of the TSW system is to supply cooling water to auxiliary equipment in the reactor building, turbine building, Service Building, Main Guardhouse, and Radwaste Building for removal of the operating heat load during normal plant operation.

The TSW system also supplies bearing lubricating water to the plant service water pumps and the circulating water pumps, cooling water to the circulating water pump motors, and water to the circulating water pump's priming eductors, which are located in the circulating water pumphouse.

The TSW system consists of two 100 percent capacity pumps taking suction from the circulating water basin and supplying water to the equipment as shown in Figures 3.2.1-11.1, 3.2.1-11.2, 3.2.1-11.3, and 3.2.1-11.4 representing a simplified schematic diagram of the TSW system.

The TSW system, after collecting heat from the various pieces of equipment, combines into a single header that dumps into the circulation water discharge tunnel for cooling in the cooling towers along with the circulating water.

The TSW pumps provide service water for initial filling of the CW system by the closure of the CW Bypass Line and pumping water through the TSW system to the CW discharge tunnel.

The TSW system is designed to function continuously during all modes of operation except during a simultaneous LOCA and Loss of Offsite Power conditions. In the event that plant service water becomes contaminated, a radiation monitor is provided in the circulating water system blowdown line to the river. The blowdown line is automatically isolated when radiation is detected. A second radiation element and switch is located in the TSW system piping. This switch provides a control room alarm when radiation is detected.

In addition to the chlorinated circulating water utilized by this system, the TSW system is equipped with a manual chlorination system. Sulfuric acid is also added to the circulating water for scale-corrosion control.

Required makeup to the TSW system is included as part of the overall circulating water system makeup requirements. .

A design engineering single-failure analysis has not been provided for the TSW system since this system serves only nonessential systems and is not required to perform a safety function. The system, however, incorporates features that assure continuity and reliability of operation. Plant service water pumps will be alternately operated to minimize wear, and the standby pump is available as a replacement during maintenance of the normally operating pump.

All piping, valves, and associated components of the TSW system are classified Seismic Category II. In the reactor building, system piping is supported to Seismic Category I.

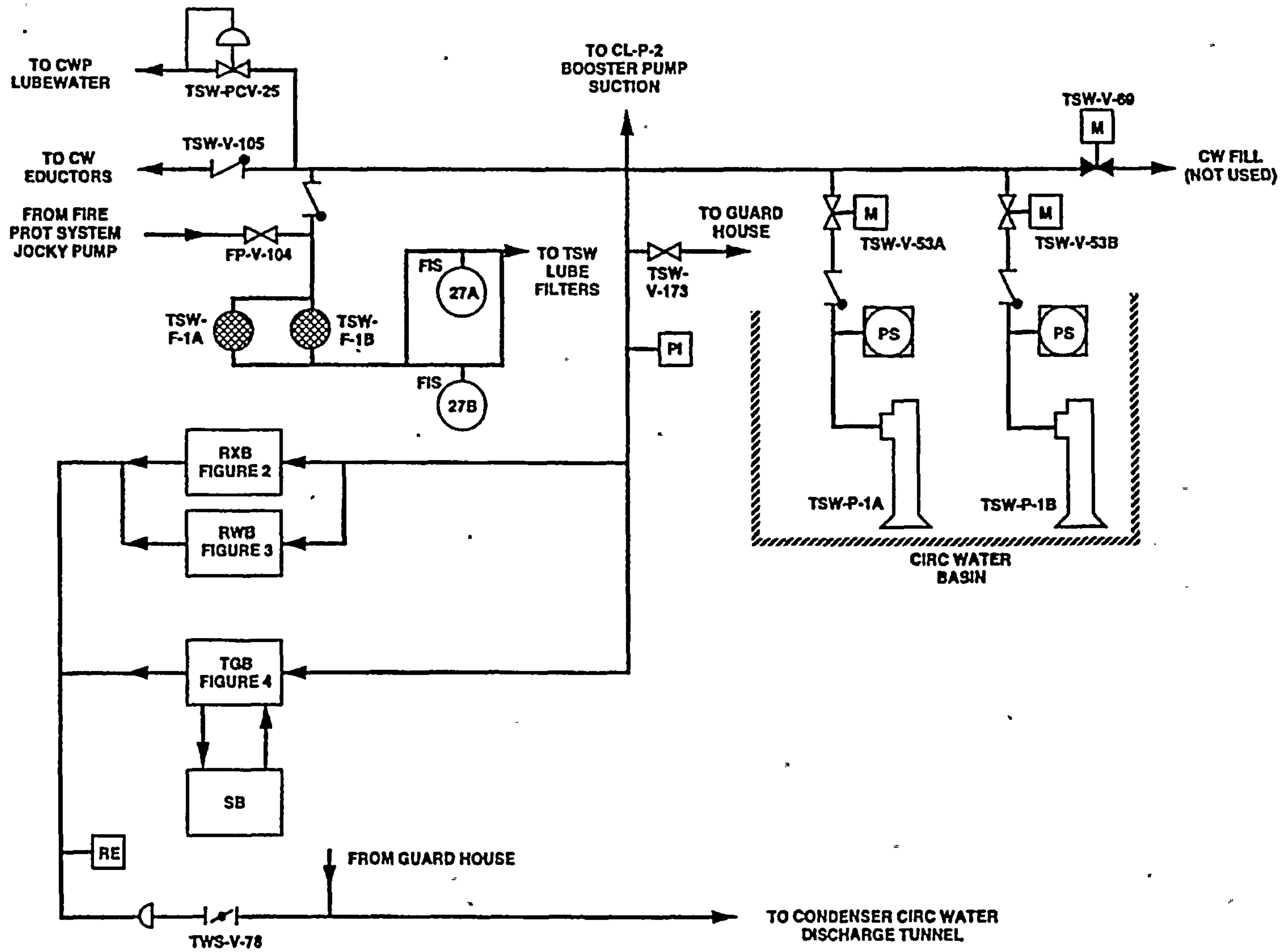

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FIGURE 3.2.1-11.1 PLANT SERVICE WATER SYSTEM

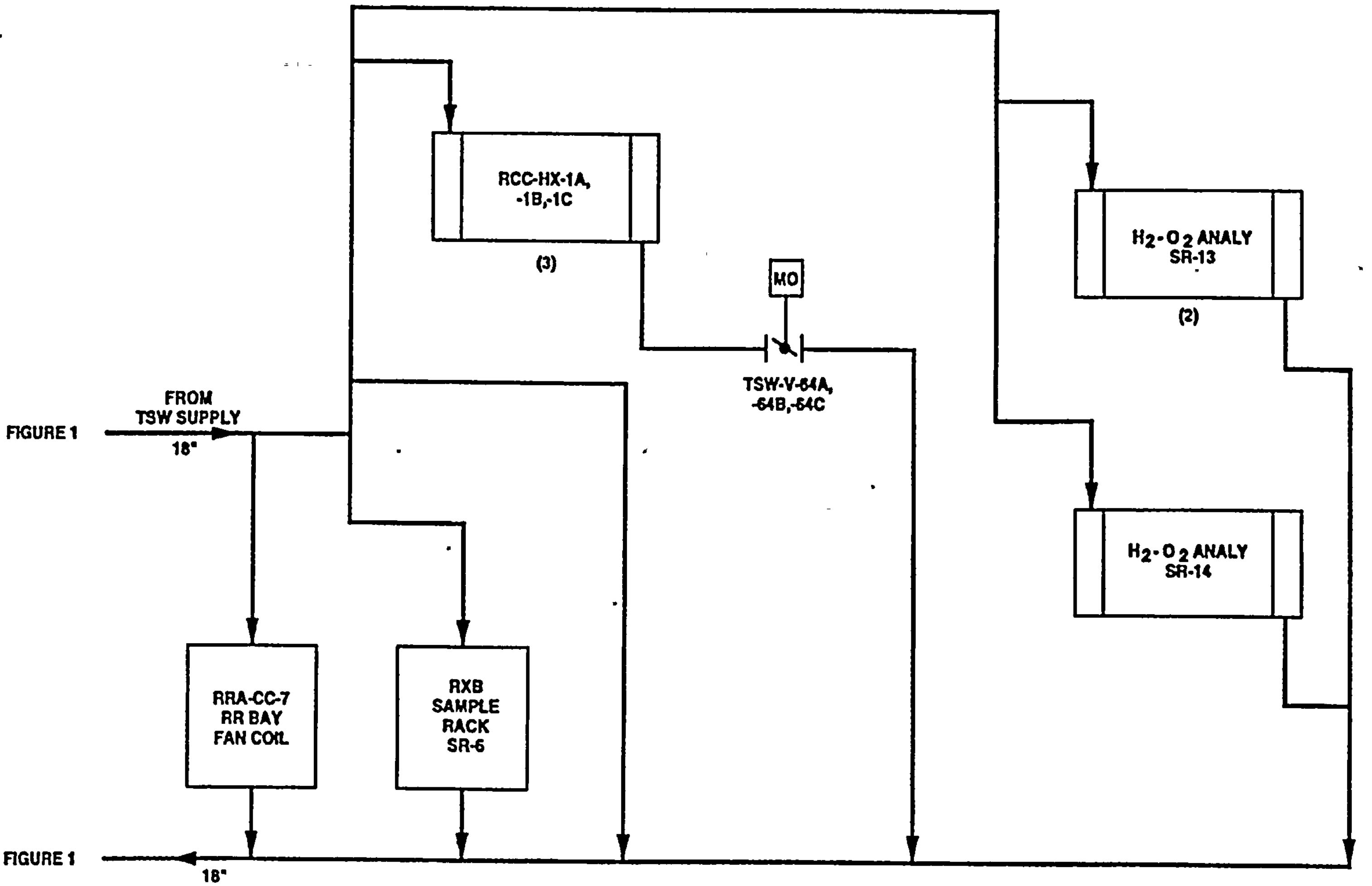

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FIGURE 3.2.1-11.2 TSW COOLED EQUIPMENT REACTOR BUILDING

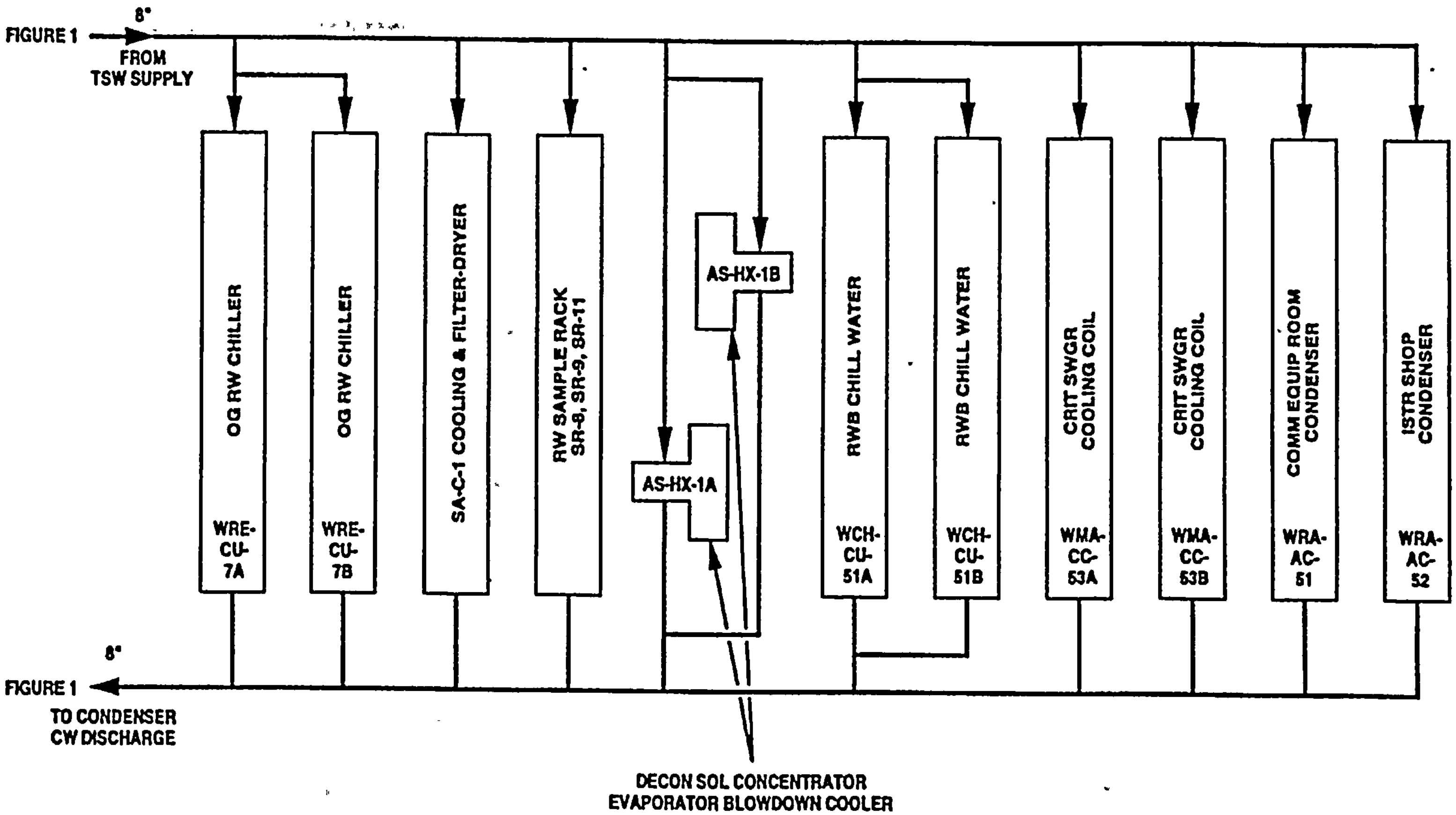

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FIGURE 3.2.1-11.3 TSW COOLED EQUIPMENT/RADWASTE BUILDING

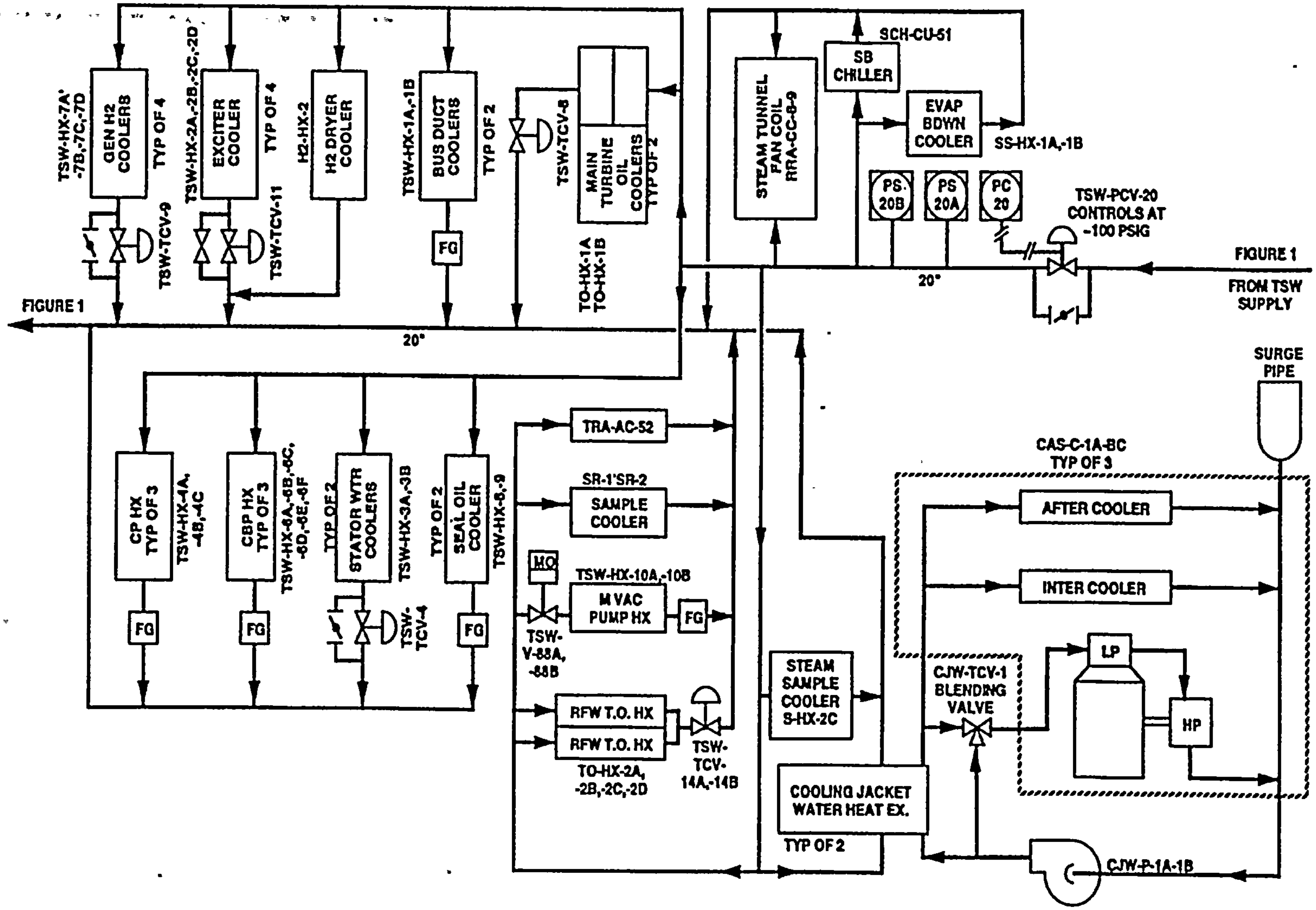

FIGURE 3.2.1-11.4 TSW COOLED EQUIPMENT SERVICE BUILDING & TURBINE GENERATOR BUILDING

3.2.1.12 Cond System Description

The main function of the condensate system (Figure 3.2.1-12) is to transport the condensed steam (condensate) in the condenser to the reactor feed pumps. The condensate is preheated by flowing through various stages of feedwater heaters. This preheating is required to increase the efficiency of the plant heat cycle.

The other functions of the condensate system are as follows:

- To provide for condensate makeup from and surplus condensate dumping to the condensate storage tank.
- To provide valving for isolation and bypassing of feedwater heaters, steam jet air ejectors and gland condenser.
- To conduct condensate to and from a full flow demineralizer. To provide for recirculation of 30% rated condensate flow through the demineralizer, pump and heater portions of the condensate system back to the condenser for a clean-up cycle during start up periods.
- Provide water for the exhaust hood spray system of the main turbine and desuperheating spray system.
- Provide coolant for the following:
    - Steam Jet Air Ejector Condensers
    - Gland Steam Condenser
    - Off-Gas Condenser
- Provide seal water for the following:
    - Condensate Pumps
    - Reactor Feedwater Pumps
    - Pumped Drain Tank Pumps
- Provides condensate makeup for the sealing steam evaporators.
- Provide a condensate supply to the control rod drive pumps. This supply also has an interface with the Condensate Storage and Transfer system.
- Provides an internal recirculation flow path for the condensate pumps.

- Provides condensate to pressurize the discharge subsystems of the following pumps when they are not in operation.
  - Reactor Building Condensate Supply Pump
  - Radwaste Building Condensate Supply Pump
  - Condensate Filter Demineralizer Backwash Pump

The main function of the condenser is to establish a heat sink for the main turbine exhaust which rejects its heat load to the circulating water system.

The other functions of the condenser are as follows:

- Deaerate the condensate in the plant cycle in the condenser deaeration section.
- Provide a heat sink for the turbine bypass subsystem during load rejection.
- Provide a heat sink for all drains and vents discharging into the condenser.

The condensate system consists of the pumps, feedwater heaters, valves, piping, instrumentation and controls necessary to provide the above listed functions. Each of the three nominal one-third capacity condensate pumps takes suction from the condenser hotwell supply header. Each suction line is equipped with manual isolation valves and relief valves. Each pump discharge line is equipped with check and manual isolation valves and then combined with the other discharge lines in a common discharge header.

Under low flow conditions all the condensate flows through the gland seal steam condenser. When flow through this condenser exceeds a preset value, a differential pressure indicator-controller opens the flow control valve in the bypass line. The bypass line is also used to pass the full condensate flow when the gland condenser is out-of-service prior to turbine shutdown. The condenser is equipped with manual isolation inlet and outlet valves.

From the gland steam condenser discharge, the condensate is passed through the condenser of one of two full capacity air ejectors. When flow through the operating air ejector condenser exceeds a preset value, a differential pressure indicator-controller opens this flow control valve in the bypass line. Each air ejector condenser inlet and outlet line is equipped with motor operated isolation valves. The outlets of both air ejector condensers and the bypass line join into a 36" header, which includes a removable spool piece.

Down stream of the air ejector condensers, the header leads to the off gas condenser and then to a flow test spool piece needed to permit installation of ASME flow nozzle assemblies if used for an ASME performance test.

Down stream of the spool piece, the header leads to the condensate filter demineralizers. From the demineralizers, the header runs to the three nominal one-third capacity condensate booster pumps and has a 20 inch branch header that feeds both the condenser high level dump line to the condensate storage tanks and the condensate system recirculation line back to the condenser. Each booster pump is equipped with manual suction and discharge isolation valves, a discharge check valve, and a suction line relief valve. In each pump discharge line a flow nozzle is used in a minimum flow control circuit. The pumps discharge to a common header.

An isolation valve and fire hose connection are installed on the suction of the "A" condensate booster pump. This provides the ability to use FP water as an emergency makeup water source for the reactor vessel.

Branching off from the header are three lines; each to a series of four nominal one-third capacity low pressure feedwater heaters and a bypass line. The single bypass line is shared by all three strings of heaters. The three heater strings are identical. Within each string the heaters are arranged for bypassing in the following groups (1&2), (3&4). At the outlet of both groups the three strings and the bypass line join into common headers. Motor operated isolation valves are included in each string at the inlet and outlet of each group. The bypass line section around each group is equipped with a motor operated globe valve for throttling control of bypass flow. The bypass line valves are normally closed, the line is used whenever a group of heaters within any one string is out-of-service, or for feedwater temperature reduction for nuclear fuel economy at the end of the fuel cycle.

Branching off from the header at the outlet of the heaters are two 24 inch lines each to a nominal half capacity low pressure feedwater heater and a 20 inch bypass line. The inlet and outlet of each heater is equipped with a motor operated isolation valve, the bypass line is equipped with a motor operated globe valve. The bypass line is used whenever either one of the heaters is out-of-service and for feedwater temperature reduction. The discharge of the heaters and the bypass line join into a 30" header which runs to the reactor feed pumps. The condensate system turns into the reactor feedwater system at the inlet to the feed pumps.

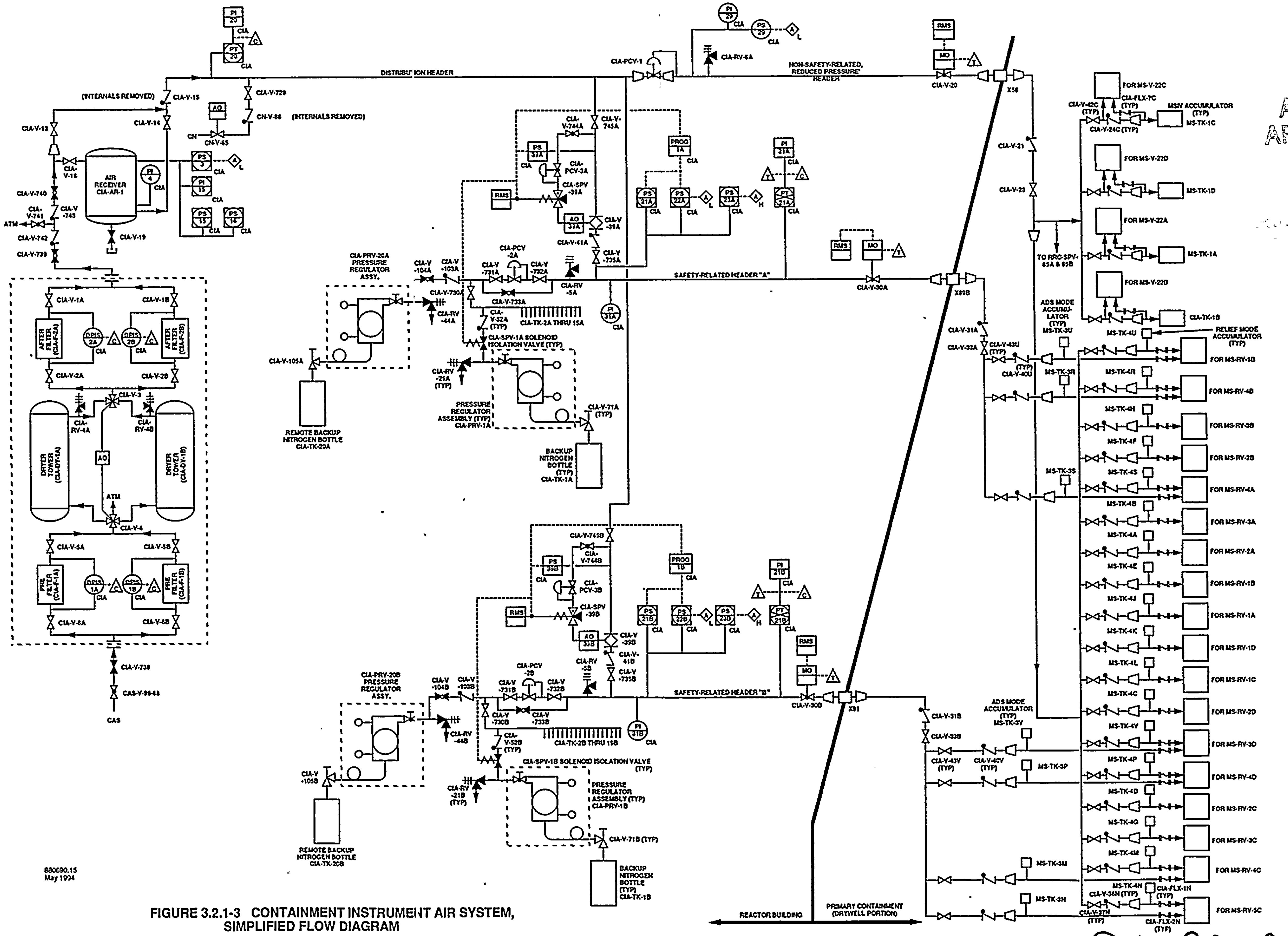

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FIGURE 3.2.1-3 CONTAINMENT INSTRUMENT AIR SYSTEM, SIMPLIFIED FLOW DIAGRAM

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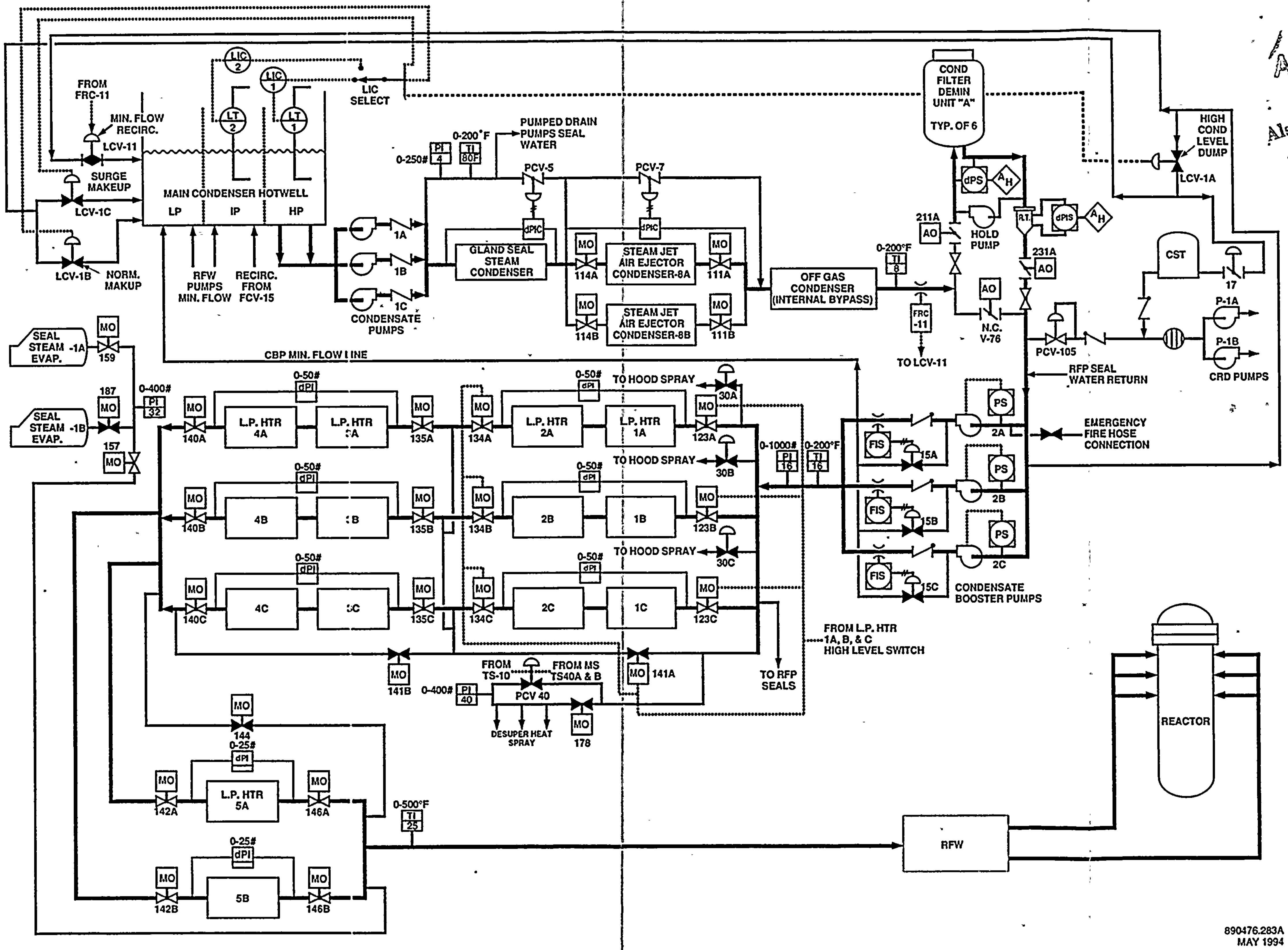

NOTE: ALL VALVES ARE SHOWN IN NORMAL OPERATING POSITION

FIGURE 3.2.1-12 CONDENSATE AND DEMINERALIZER

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3.2.1.13 ADS/SRV System Description

The Automatic Depressurization system (ADS) is designed to provide depressurization of the Reactor Coolant Pressure Boundary so that the RHR Low Pressure Coolant Injection and the Low Pressure Core Spray systems can operate to flood the reactor vessel to protect the fuel barrier from excessive temperature. Depressurization is accomplished by activating seven safety/relief valves, which vent steam to the suppression pool. ADS is activated if the High Pressure Core Spray system cannot maintain the Reactor water level following a small break in the Reactor Pressure Boundary. The safety relief valves also prevent overpressure and subsequent failure of the reactor pressure vessel.

There are eighteen Safety/Relief Valves (SRVs) located on the main steam lines inside the Primary Containment. Seven of those safety/relief valves are associated with the ADS system. The safety/relief valves are dual actuated types, mechanically self-actuating under conditions of high reactor pressure (safety mode) and electro-pneumatically actuated: 1) manually via control switches; 2) by logic circuity under high Reactor pressure: or 3) LOCA conditions for the seven ADS associated valves only (relief actuation). See Figure 3.2.1-13.1.

The ADS associated safety/relief valves receive their signal to open and depressurize the primary system from low reactor vessel water level provided that one of the low pressure emergency core coolant systems is operating. The ADS valves will automatically open after a 105 second time delay and remain open until manually reset after the initiating signal has cleared. See ADS Logic Diagram Figure 3.2.1-13.2.

Each of the seven ADS valves is provided with an additional pneumatic accumulator (two total each valve), connected in parallel, which are sized to provide sufficient capacity to ensure adequate supply pressure to the valve actuator. Each additional accumulator has a 42 gallon capacity and is sized to operate its ADS valve one time with maximum drywell pressure and a vessel pressure of 0 psig.

The eighteen SRVs exhaust via a tailpipe to the suppression chamber. Each safety/relief valve discharges to a point below minimum water level in the suppression pool. Water in the tailpipe more than a few feet above the suppression pool level would cause excessive pressure at the SRV discharge when the SRV opened. For this reason, redundant 10 inch vacuum relief valves are provided on each SRV discharge line to prevent drawing an excessive amount of water up into the line as a result of steam condensation following termination of relief operation. Each vacuum relief valve pair is situated with the valves in parallel. See Figures 3.2.1-13.3 and 3.2.1-13.4.

Each of the safety/relief valves has two methods of verifying the valve position and leakage, acoustic monitors and tailpipe thermocouples. Both the acoustic monitors and the thermocouple outputs are indicated in the main control room.

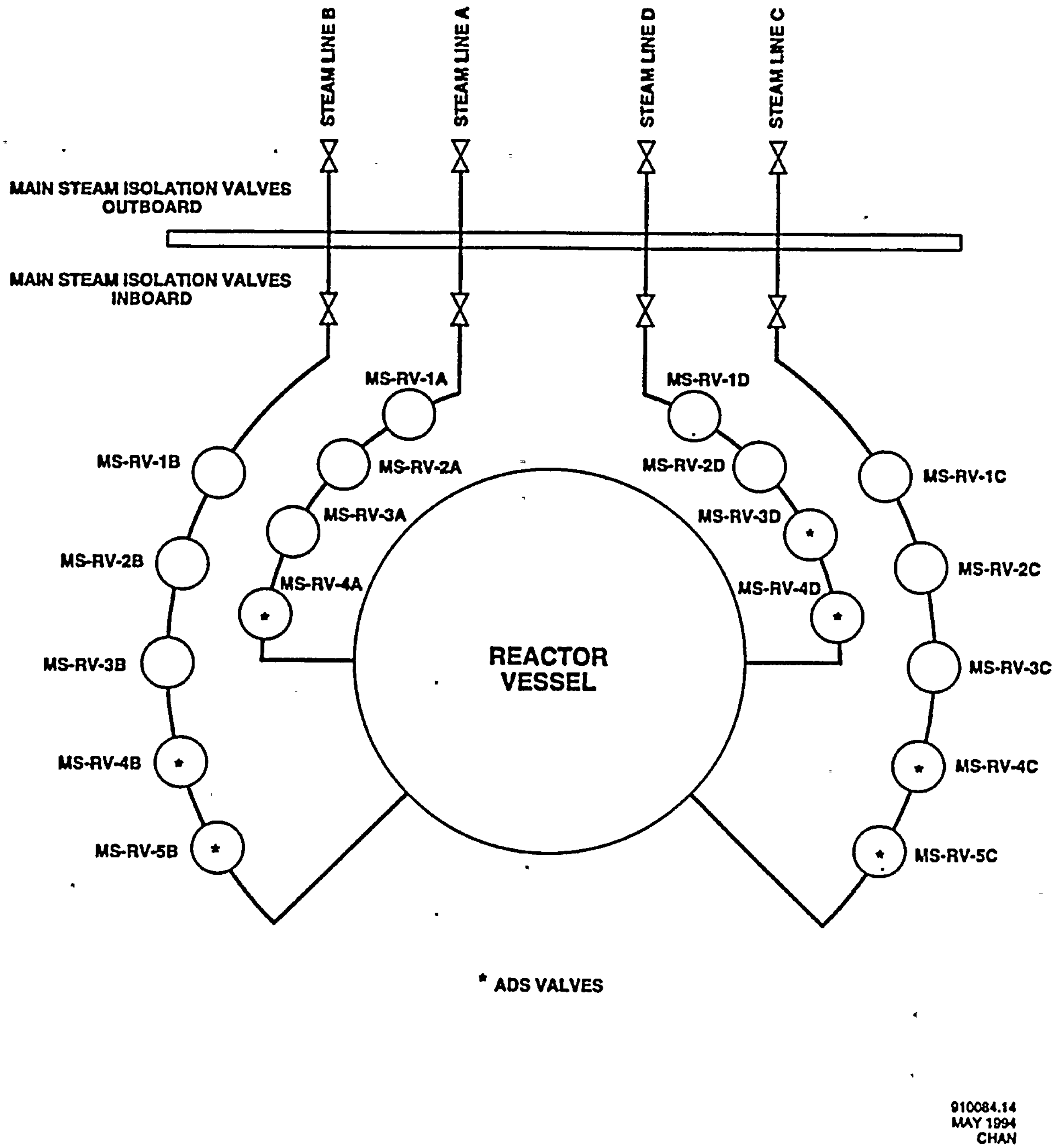

FIGURE 3.2.1-13.1 RELIEF VALVE LOCATIONS

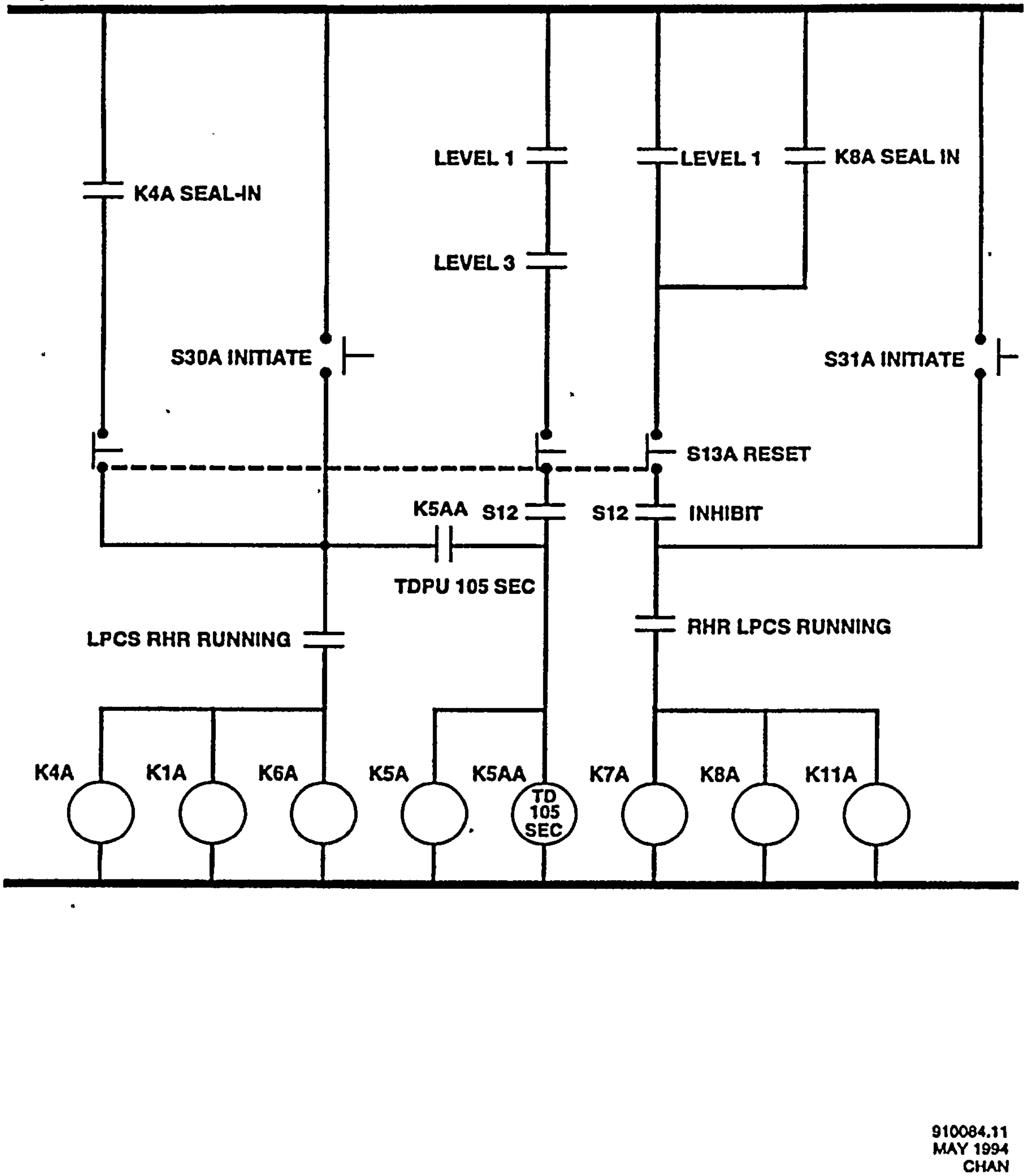

FIGURE 3.2.1-13.2 SIMPLIFIED ELEM. DIAGRAM OF ADS LOGIC

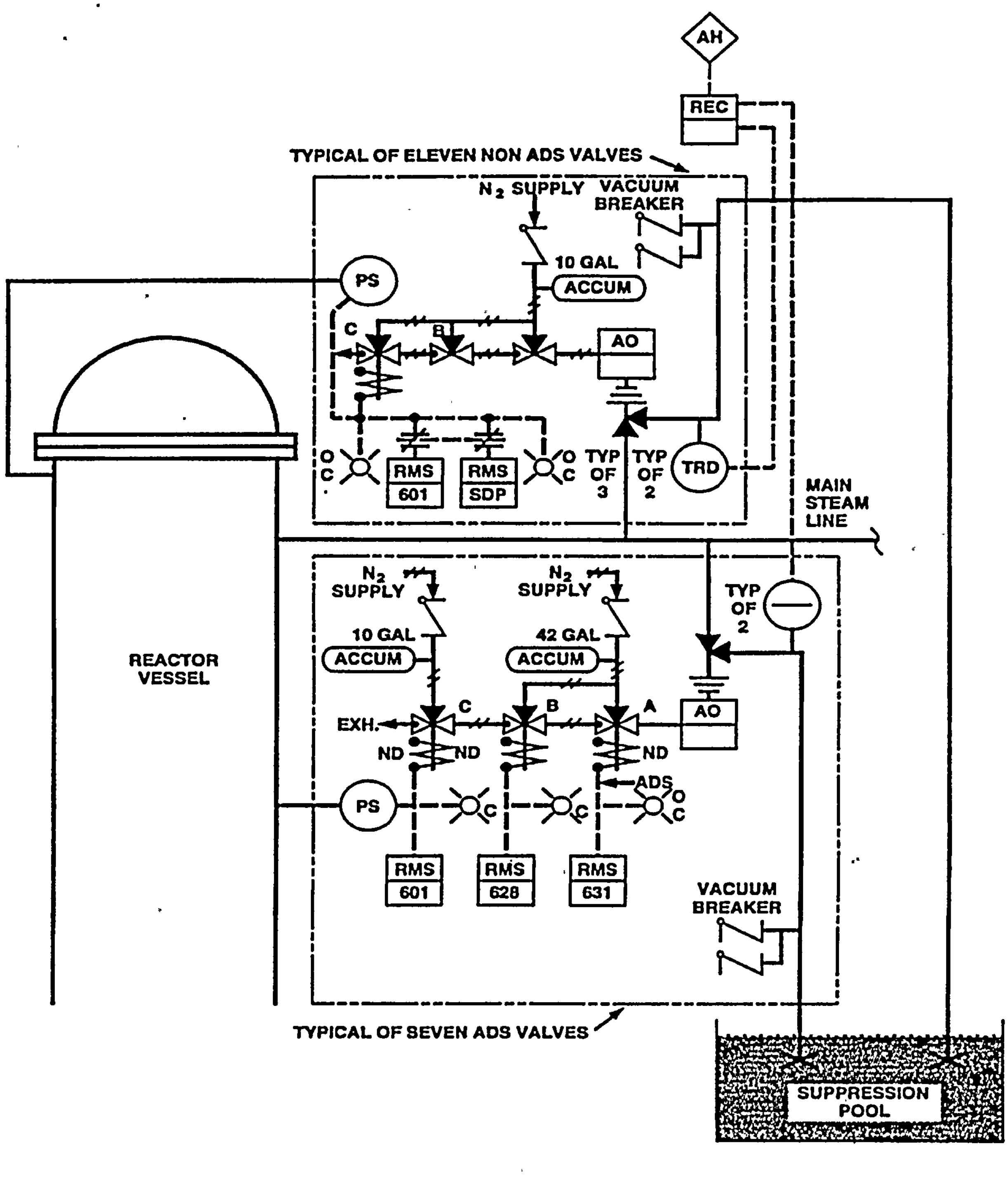

FIGURE 3.2.1-13.3 SAFETY/RELIEF VALVE CONTROLS

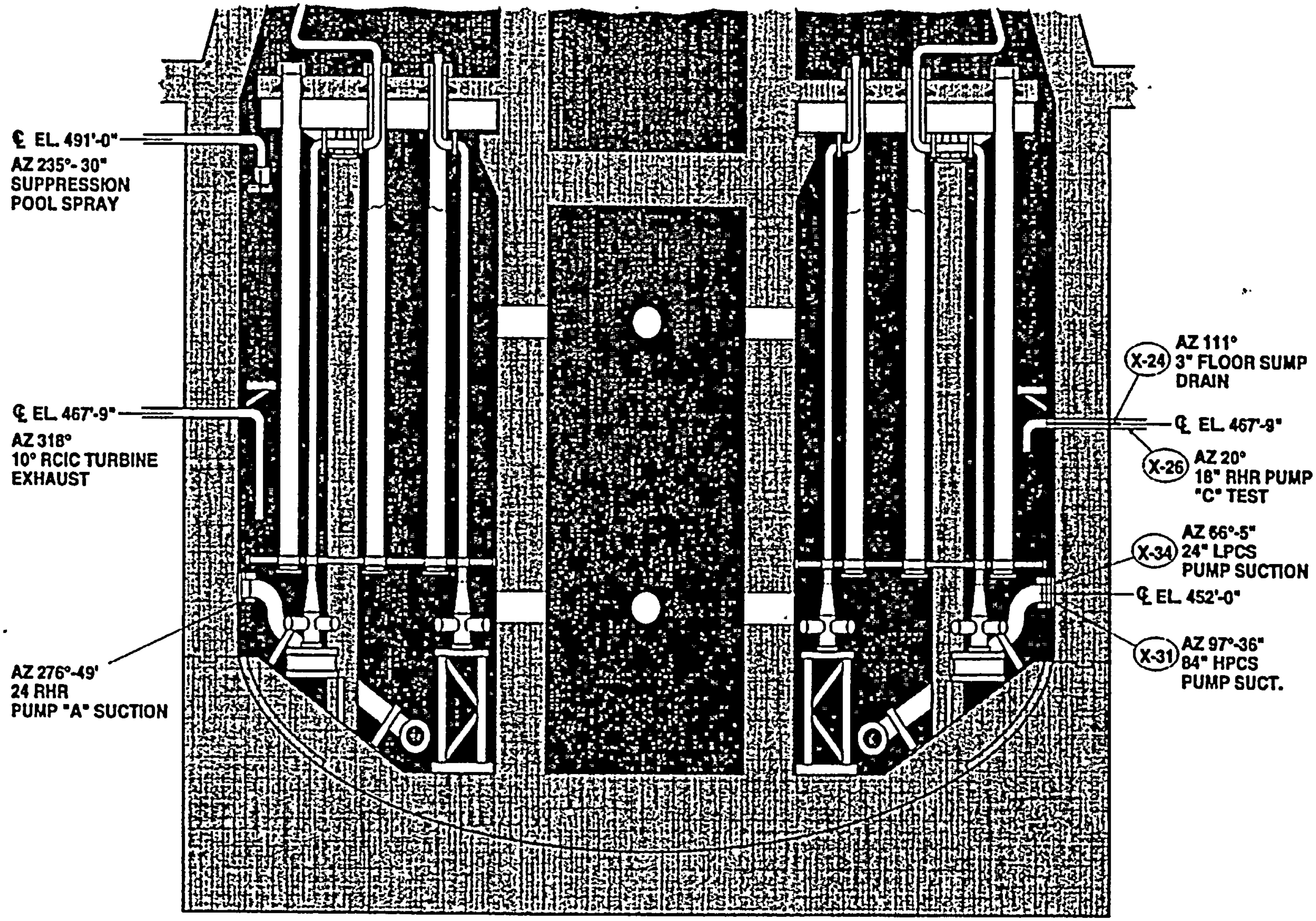

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FIGURE 3.2.1-13.4 SUPPRESSION POOL CROSS-SECTION

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3.2.1.14 AC System Description

The AC Electrical Distribution system provides power to the entire Plant during all operating and shutdown modes.

Figure 3.2.1-14 is a simplified one-line diagram of the system. The Plant is connected to the Bonneville Power Administration's (BPA) utility grid by the offsite power system and the Plant operates from the onsite system.

Offsite System

The offsite power system consists of the following elements and the connections between them.

- Main Generator
  1230 MVA, .975 Pf. 25Kv,
  3 phase, 60Hz, 1800 rpm

- Main Step-up Transformer Bank
  E-TR-M1, E-TR-M2, E-TR-M3, E-TR-M4 (1 spare)
  1140/1276MVA FOA, 55°C/65°C
  Each - 380 MVA, 1 phase, 60Hz 500/25Kv
  1300Kv BIL. Z=16% on 1140 MVA
  Connected delta-wye

- Isolated Phase Bus Duct
  Section connecting main generator to main step-up transformer 25Kv. Winding Side - 30,000A, 3 phase, 25Kv
  Forced air cooled
  Section feeding normal auxiliary transformers 1200A, 25Kv

- Generator Grounding Equipment
  Transformer Ratings:
  100KvA, 1 phase 60Hz
  22000 - 480 volts
  Insulation Class-25Kv
  BIL-150Kv
  Resistor-Insulation Class 25Kv
  Ampere Rating 600

- Normal Auxiliary Power Transformers
  E-TR-N1
  24/32/40/26.9/33.6/44.8 MVA OA/FA/FA,
  25-4.16/4.16 Kv 50°C/65°C, 3 phase,
  60 Hz 3 winding, taps ± 2.5%,
  14.4/19.2/24 MVA, X winding Z_{HX} = 10.02% on 24 MVA
  9.6/12.8/16 MVA, Y winding Z_{HY} = 7.22% on 16 MVA,
  Connected delta-wye, wye Z_{XY} = 9.0% on 9.6 MVA

  E-TR-N2
  16.3/21.6/27/18.1/22.7/30.2 MVA OA/FA/FA
  55°C/65°C - 3 phase, 60 HX, 2 winding
  25-6.9Kv, Z=7.5% on 27 MVA, Taps ± 2.5%
  Connected delta-wye

- Start-up Auxiliary Transformer E-TR-S
  42/56/70 MVA OA/FA/FA 65°C
  230-4.16/6.9 Kv, 3 winding,
  3 phase, 60 Hz,
  Connected wye-wye, wye
  Taps ± 5%
  18/24/30 MVA X winding, $Z_{H\text{-}X}$ = 4.9% on 18 MVA
  24/32/40 MVA Y winding, $Z_{H\text{-}Y}$ = 12.1% on 24 MVA
  $Z_{X\text{-}Y}$ = 20.9% on 24 MVA

- Backup Auxiliary Transformer E-TR-B
  10/11.2/14 MVA OA/OA/FA 55°C/65°C
  115-4.16 Kv, 2 winding
  3 phase 60 Hz
  Connected wye
  Z=6.96% on 10 MVA, Taps ± 2.5%

The offsite system has two separate connections to the utility grid, 230 Kv and 115 Kv. The 230 Kv connection provides the preferred power source to the Divisions 1, 2 and 3 critical buses that operate ESF systems, and the 115 Kv connection provides power to the Division 1 and 2 critical buses if the 230 Kv source is not available. During normal operations the Plant is started from the 230 Kv system and then transferred to the main generator source when the Plant output has reached 25% of rated capacity. The startup transformer remains energized to permit the onsite AC Electrical system to be automatically transferred back in the event of a Plant trip and loss of power from the generator.

When the Plant is in a shutdown condition, power can be obtained from the 500 Kv grid. This is accomplished by disconnecting the isolated phase bus links to isolate the main generator. Power can then be established through the main step-up transformers to the AC Distribution system. Experience has shown that it takes approximately eight (8) hours to make this transition.

Onsite System

The onsite power system provides power to the entire Plant including balance of Plant non-Class 1E loads and the engineered safety features (ESF) Class 1E loads. The onsite power system is connected to the offsite power system by circuit breakers housed in metal clad switchgear units E-SM-1, E-SM-2, E-SM-3, E-SH-5 and E-Sh-6. This arrangement provides two main distribution systems for the Plant:

- 6.9 Kv - E-SH-5 and E-SH-6
- 4.16 Kv - E-SM-1, E-SM-2 and E-SM-3

This document is limited to discussion of the onsite system that interfaces with the ESF systems. Since the 6.9 Kv system and associated sub-systems provide power to non-Class 1E loads only, no further discussion is provided.

The following systems provide power to ESF systems:

- 4.16 Kv system
- 480 V system
- 120/240 Class 1E instrumentation power system
- Standby diesel generator

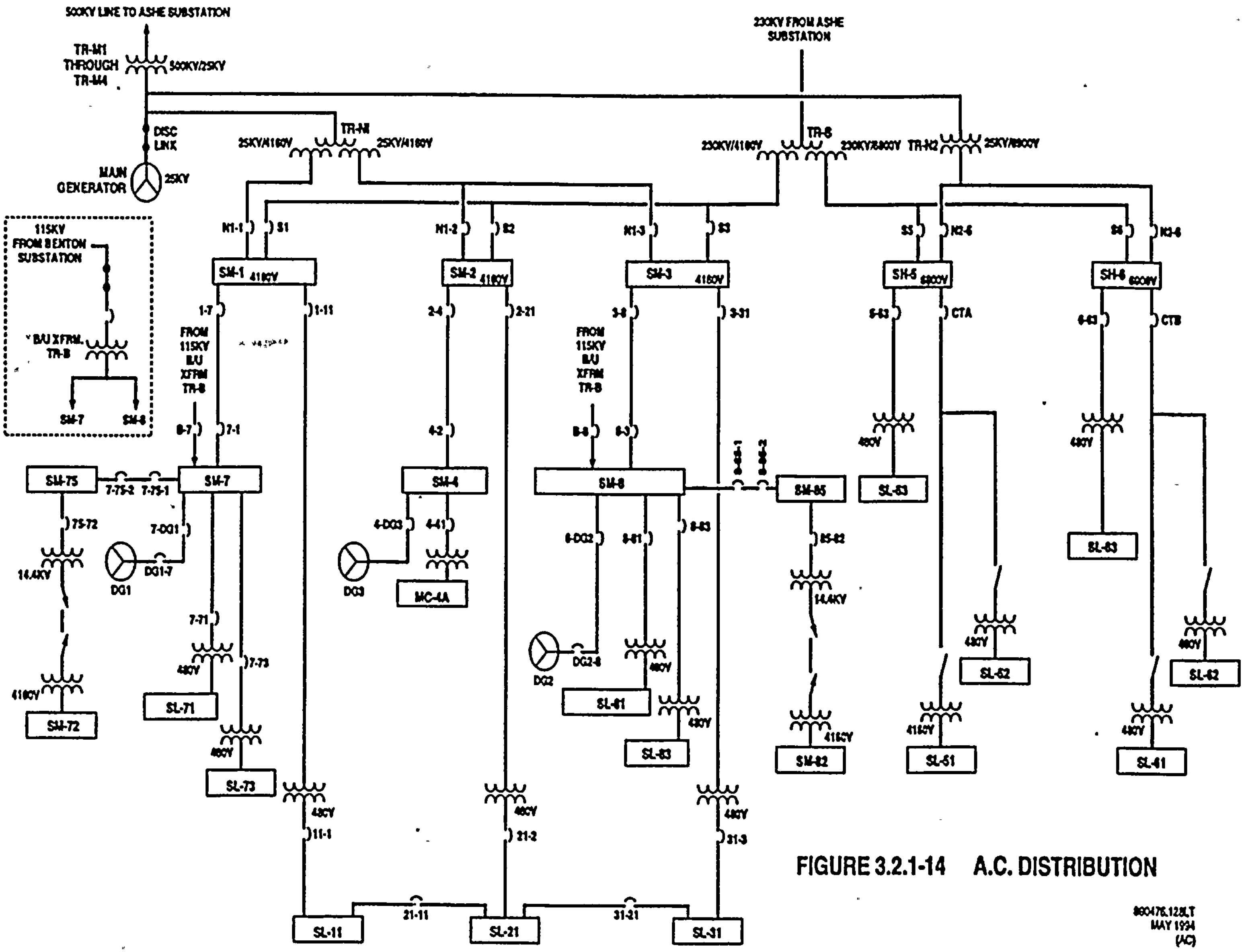

FIGURE 3.2.1-14 A.C. DISTRIBUTION

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3.2.1.15 <u>CAS System Description</u>

The Control and Service Air systems are two compressed air distribution systems that are arranged to complement and supplement each other in that each can supply the headers of the other. The systems can be isolated from each other through an isolation valve (SA-PCV-2). Figures 3.2.1-15A and 3.2.1-15B show the main system features. (Note that the "CAS" designation is used both for the entire Control and Service Air system as well as for the Control Air portion of the system. The meaning intended is apparent from the context.)

The Control Air portion of the system (CAS) supplies clean, dry, oil-free compressed air at 90 to 110 psig to outside-containment instrumentation, controls and local accumulators for valve actuators. The system is designed to provide uninterrupted service during plant operation. It is vital to plant operation but loss of CAS air pressure will not jeopardize the plant's safe shutdown capability. In the event CAS pressure downstream of the dryer falls to less than 75 psig, the valve bypassing the air dryer will automatically open to supply moist air to the distribution piping. The CAS supplies actuator air to the four outboard MSIVs through local check valves and accumulators at each valve actuator. Loss of air pressure at the actuator will allow the valves to be closed by the actuator springs and thus to fail-safe. CAS air also operates CRD system valves which, on loss of air pressure, will result in either slow insertion or scram of the control rods.

The Service Air portion of the system (SA) supplies clean, oil-free compressed air at 90 to 110 psig to the plant for general station services, such as operating air-powered tools and equipment, and as a source of air for backwashing filters and demineralizers, and for breathing air purifiers. The SA system is designed to be isolated from the CAS supply piping, at CAS supply air pressure less than 80 psig, to conserve air for CAS use. The SA system air is not vital to plant operation or to plant safety.

Four rotary screw-type, positive displacement, flood lubricated, single stage compressors supply air to the CAS system. Three of these are designated CAS compressors and the remaining one is designated an SA compressor.

The CAS compressors discharge to a single header. The air then flows through two desiccant dryers to three air receivers. From the receivers the flow is through one of two pre-filter banks, through one of two refrigeration type dryers and through one of two after-filter banks. The air is then distributed to the station loads through supply headers. Downstream of the after-filters the piping is arranged such that the refrigeration-type dryers and the associated pre- and after-filters can be bypassed. An isolable connection point to the SA system header is also provided. The CAS equipment is located in the turbine building, 441' elevation, adjacent to the auxiliary boiler room in the southeast corner of the building.

A specific sub-system designated the Cooling Jacket Water system (CJW), provides closed loop flow of Cooling Water to the CAS compressor lube oil heat exchanger, aftercooler and refrigeration dryer condenser. This system rejects heat to the Plant Service Water (TSW) system. The CJW system automatically commences operation whenever a CAS compressor starts. One of two CJW pumps is normally running with one of two CJW Heat Exchangers in service. Under duress the Fire Water system can be connected to replace the cooling provided by the TSW or the CJW system. The CJW does not serve the SA system.

The SA air compressor is located in the Radwaste Building, 467' elevation, near the southwest corner of the building. Compressed air leaving the SA-C-1 compressor is continuously monitored for carbon monoxide content and the compressor will shut down if concentration of CO exceeds limits for breathing air. Depending on pressures and demands in the CAS and SA systems, this compressor will supply air to the CAS system. A refrigerant dryer/filter arrangement dries the air which then flows through a charcoal filter and on to the system loads as well as to the CAS receivers and/or drying towers via the SA system isolation valves. The TSW system provides cooling water to the SA compressor coolers and SA dryer.

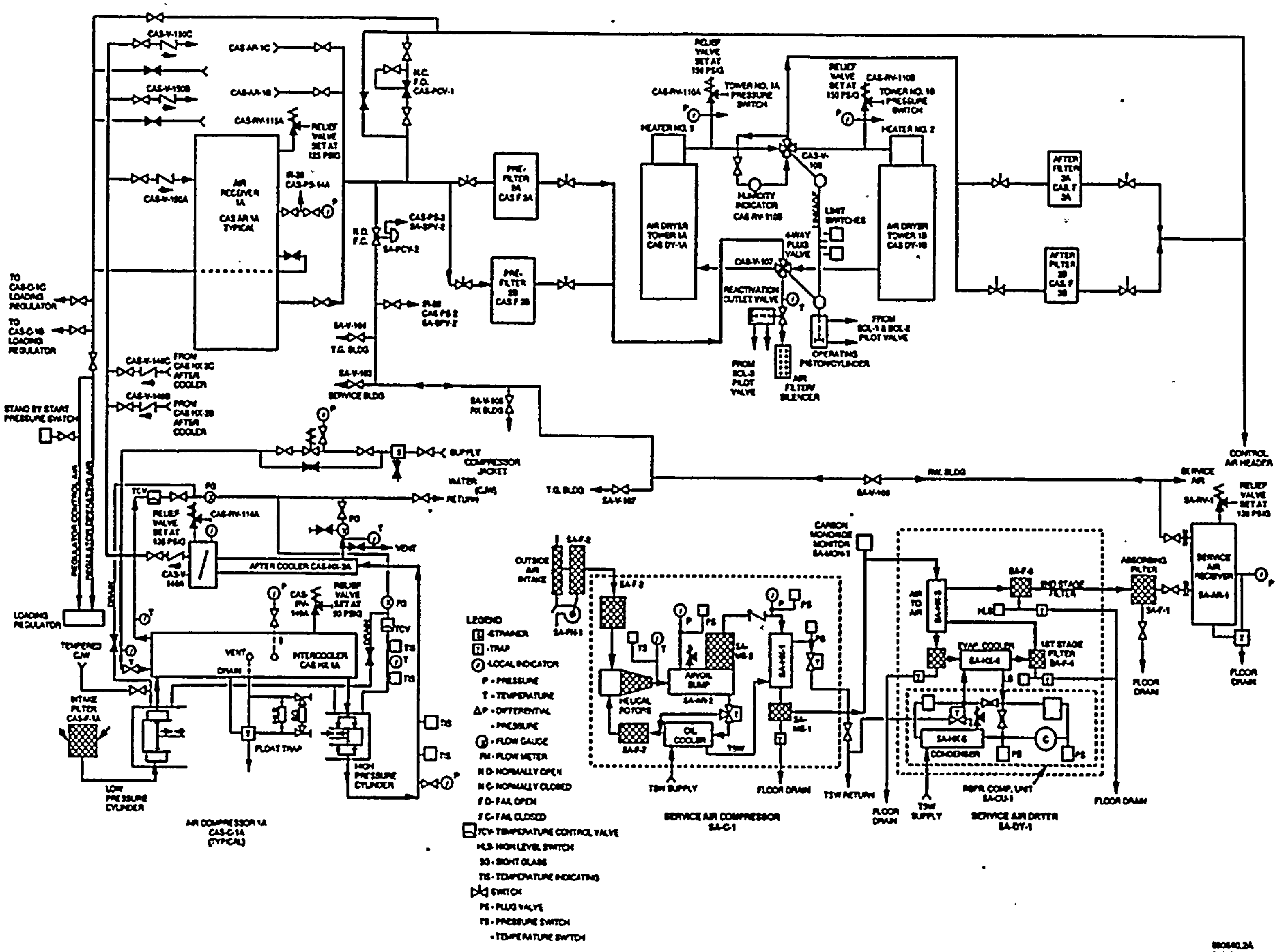

FIGURE 3.2.1-15A. CONTROL AND SERVICE AIR SYSTEM, SIMPLIFIED FLOW DIAGRAM

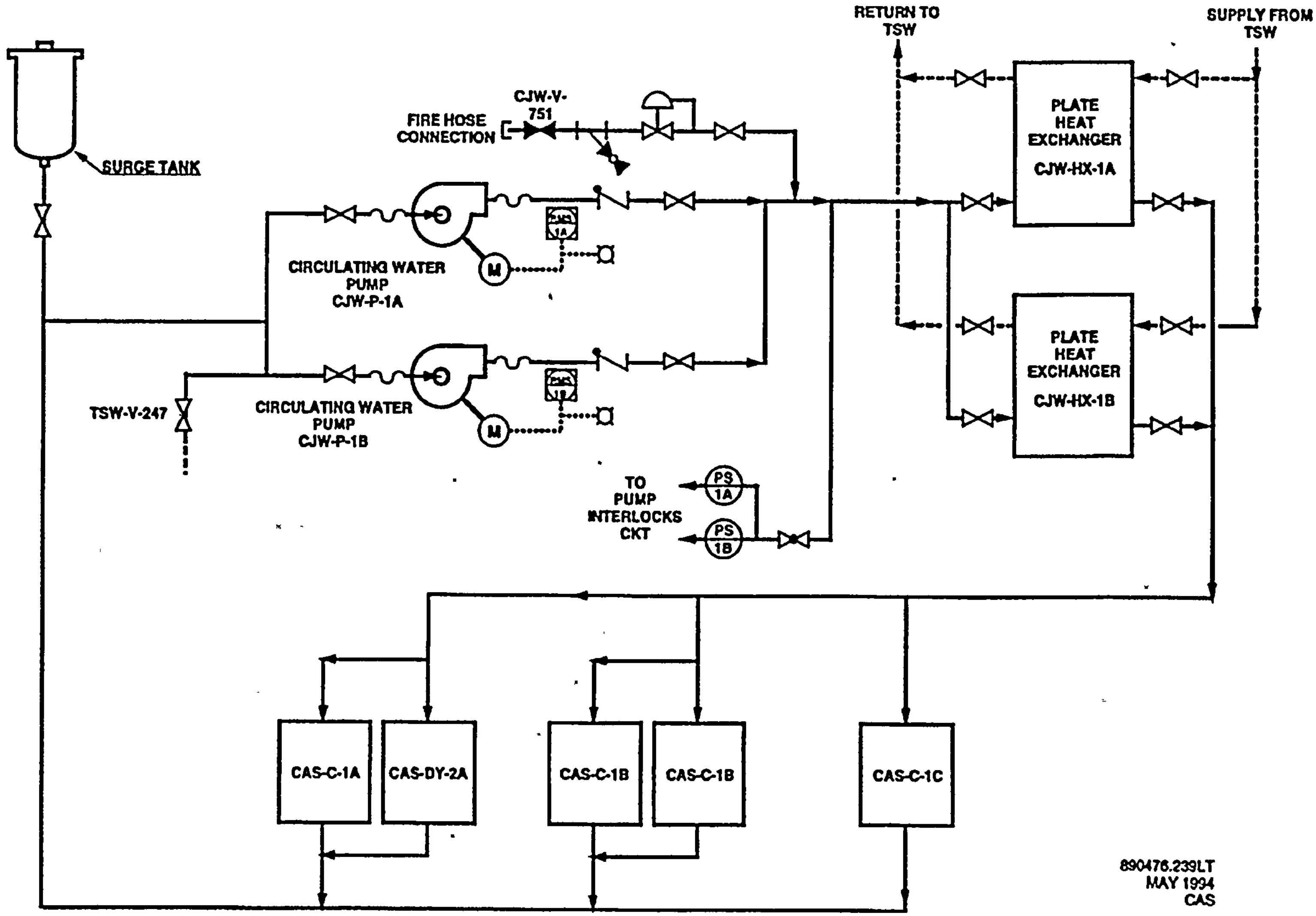

FIGURE 3.2.1-15B CAS COOLING JACKET WATER

3.2.1.16 NS4 System Description

The NS4 (Nuclear Steam Supply Shutoff System) includes the instrument channels, trip logics, and actuation circuits that automatically initiate valve closure providing isolation of the primary containment and/or reactor vessel to limit the release of radioactive materials. The NS4 also initiates the securing or startup of other equipment, but these initiations are not limited to NS4-generated isolation signals.

Sensor elements are located in the nuclear boiler system, reactor protection system, main steam system, standby liquid control system, reactor water cleanup system, residual heat removal system, recirculation system, and leak detection system. Their contact outputs connect directly to relay circuits or connect to relay circuits via sensor circuits located on local panel racks; or in the case of the main steam line high radiation inputs, from local sensors to radiation monitors, and then to relay circuits. The relay circuits in turn control annunciator circuits, indicator circuits, and isolation valve control circuits. The isolation valves are operated by AC or DC motor, direct solenoid, or pilot solenoid and service or instrument air pressure. For those valves which are motor operated, the relay control circuits cause the motor to close the valve for the trip condition. Air operated isolation valves are actuated through solenoid controlled pilot air valves. Switches located on control room panels permit manual operation of the isolation valves for testing and as a backup to automatic trip signals. System annunciators are located in the control room.

The NS4 system is divided up into seven groups. Each of these groups (referred to as "isolation groups") consists of a different group of valves and/or equipment that will automatically isolate, secure or start up as required. Isolation groups 1, 2, 5, 6 and 7 are referred to as NS4 isolation groups. Isolation Groups 3 and 4 are referred to as the Balance of Plant (BOP) isolation groups. The scope of this evaluation covers groups 1, 2, 5, 6 and 7 only. The NS4 isolation signals are high drywell pressure and reactor low water level.

The following is a brief description of the isolation groups:

Group 1. - Main Steam Isolation Valves (MSIVs) and
Main Steam Line Drain Valves

Group 2 - Reactor Water Sample Valves

Group 3 - Primary and Secondary Containment
Ventilation and Purge Systems

Group 4 - Miscellaneous Balance of Plant systems
(examples: Reactor Closed Cooling Water,
Fuel Pool Cooling, Circulating Water, etc.)
Traversing In-Core Probe (TIP) system

Group 5 Residual Heat Removal (RHR) system

<u>Group 6</u> - Residual Heat Removal system (Shutdown Cooling Mode)

<u>Group 7</u> - Reactor Water Cleanup system (RWCU)

3.2.1.17 CST System Description

The primary function of the Condensate Storage and Transfer system (CSTS) is to store and supply condensate for general plant use. See Figure 3.2.1-17.

To perform this function the CSTS consists of two condensate storage tanks, one reactor building, one radwaste condensate supply pump, and one condensate filter/demineralizer backwash pump.

The demineralized water system and the liquid radwaste system are the primary sources of makeup water to the condensate storage tanks. The tanks accommodate a surge volume for condensate returned to the tanks after treatment in the liquid radwaste system.

The CST system supplies makeup condensate for the condenser hotwell which is gravity fed from the storage tanks. Bleedoff water from the condensate system is returned to the storage tanks from the discharge of the condensate demineralizer.

A separate line from the condensate storage tanks supplies the control rod drive pumps with condensate by gravity flow. Condensate is supplied for various reactor building services, including fuel pool makeup, by the reactor building condensate supply pump. Condensate is supplied for various radwaste building services by the radwaste building condensate supply pump. The condensate filter demineralizer backwash pump supplies condensate for backwashing purposes to the condensate filter/demineralizers.

The Condensate Storage and Transfer system can provide condensate to the RCIC system, the HPCS system, and the RHR loops by the line-up of manual valves or installing removable spool pieces. A minimum inventory of 135,000 gallons in the condensate storage tanks is reserved for the RCIC and HPCS pumps. This assures the immediate availability of a sufficient quantity of condensate for emergency core cooling and reactor shutdown. Although this minimum is maintained in the condensate storage tanks for the RCIC and HPCS pumps, the water in the suppression pool is considered the emergency source of water for these pumps. The reserve of water is maintained by monitoring the level in the condensate storage tanks and by preventing condensate transfer when this reserve level is reached. The RCIC and HPCS pumps are gravity fed from the condensate storage tanks. A standpipe is located on the common condensate supply line leading to the HPCS and RCIC pumps inside the reactor building. The standpipe is used to indicate either a low water level condition in the condensate storage tanks or a loss of suction supply from the tanks. The pipe also contains a sufficient amount of water for HPCS and RCIC pump suction supply during switchover from the tanks to the suppression pool.

The Condensate Storage and Transfer system also facilitates testing and/or flushing of the High Pressure Core Spray, Low Pressure Core Spray, Residual Heat Removal, and the Reactor Core Isolation Cooling systems.

The condensate storage tanks are Seismic Category II; however, they are located inside a Seismic Category I concrete dike which is designed to retain the condensate from both tanks. Drainage from the dike is routed to the radwaste system for processing. During precipitation, drainage from the dike is sampled and analyzed for radioactivity before being discharged (and is monitored during discharge) to the storm drain header.

Condensate storage tank level is monitored in the main control room. High and low-level alarms are provided to prevent overflow and to prevent the water level from dropping below the required reserve level for RPV makeup. Level switches provide low-low annunciation and interlock with the HPCS and RCIC systems. Building condensate supply pumps are manually started with COND-P-4 or COND-P-5 normally running. The other pumps are brought on as needed.

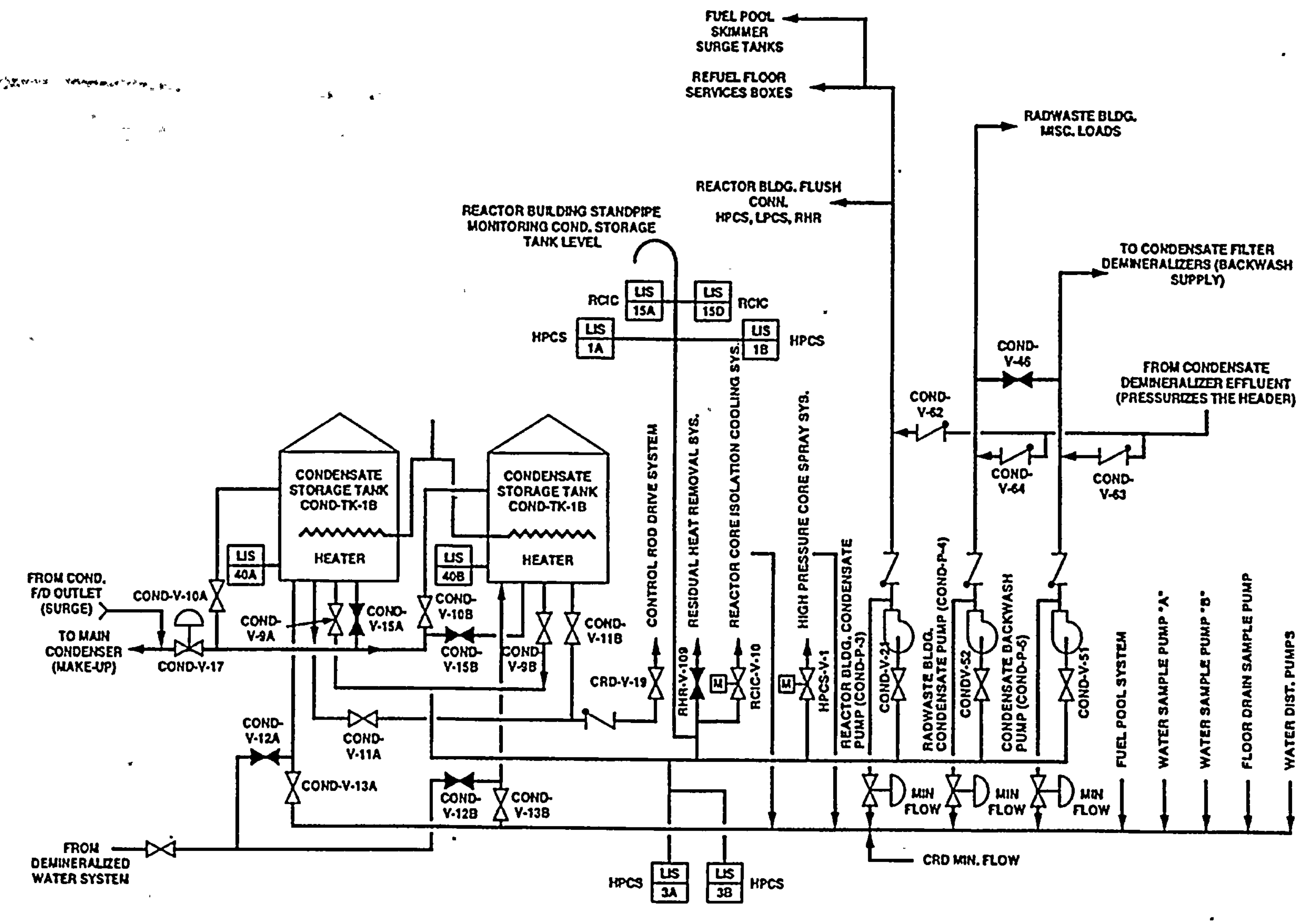

FIGURE 3.2.1-17 CONDENSATE STORAGE AND SUPPLY SYSTEM

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3.2.1.18 Power Conversion System Description

For IPE purposes, The Power Conversion system consists of the following sub-systems:

- Condenser
- Hotwell Level Control
- Circulating Water
- Steam Jet Air Ejectors
- Mechanical Vacuum Pumps
- MSIVs
- Turbine Bypass Valves

Condenser Description:

The condenser is a single pass, 3-bank, divided waterbox type surface condenser. The Circulating Water system provides the cooling medium for condensing the incoming steam. Circulating Water from a common inlet splits to enter the three separate heat exchanger banks. Each heat exchanger bank consists of an inlet, intermediate and an outlet waterbox connected by tubes. Steam entering the condenser from low pressure turbines, first contacts the Number 1 feedwater heaters, where it provides some heat input to the feedwater. As the steam enters the tube bundles within the condenser, it condenses and drains to the inner bottom plate. The inner bottom plate directs the condensate to the hotwell. The condensate flow in the hotwell is directed by two partitions which requires condensate entering the hotwell to flow the length of the condenser, before it can flow back toward the condensate pump suction header branch lines. This arrangement ensures the necessary holdup time to allow N-16 gamma decay.

The condenser is partitioned to provide three shell side pressure zones. The low pressure zone is at the circulating water inlet end of the condenser shell. The high pressure zone is at the circulating water outlet end of the condenser shell; the intermediate zone is in between the HP and LP zones.

Hotwell Control Description:

The hotwell level control valve stations consist of a normal make-up station, a surge make-up station, and a high-level dump station. The two make-up stations allow condensate to be gravity drained from the condensate storage tanks to the condenser hotwell. The high level dump station will route condensate from the condensate booster pump bypass header to the condensate storage tanks. All three level control valve stations are air-operated and operate sequentially from the output of the selected level indicator controller.

HOTWELL LEVEL CONTROL

	Full Open	Full Closed
LCV-1A (Dump Vlv)	3# (+6")	8.4# (+0.6")
LCV-1B (Norm M/U)	12# (-3")	9.6# (-0.6")
LCV-1C (Surge M/U)	15# (-6")	12# (-3")

The condensate storage tank isolation valve will auto close on a low level in either CST. This is to ensure that the CSTs are not drained to less than the Technical Specification requirement to meet ECCS requirements.

Each of the level taps from the three pressure zones in the condenser hotwell provide a sample point for the condenser water quality sample pumps. These pumps draw suction from the bottom of each level tap. The sample pump discharge is routed to the turbine building sample rack. A line from the sample pump discharge header is routed back to the condenser hotwell via the top of each level tap. This line provides a continuous flow path for sample water to prevent the pump from overheating.

Circulating Water Description:

The circulating water system (Figure 3.2.1-18) is designed to supply cooling water to the main condenser and auxiliary cooling systems, and release the heat collected to the atmosphere via mechanical draft cooling towers. The cooling towers are designed to remove 7.96×10^9 BTU/hr from the circulating water system. The circulating water system has a design flow rate of 550,000 gpm to the condenser and a design flow rate of 570,000 to the cooling towers.

The system also provides means for the blow-down, makeup and chemical treatment of the circulating water.

The circulating water system is a nonsafety related system designed to Seismic Category II requirements. It is a closed cycle cooling system that utilizes 3 pumps, six cooling towers, a chlorination system, a sulfuric acid system, and two bulk chemical feed systems. The three pumps, each making up one third capacity of the system, are located in the circulating water pumphouse. Each pump is set in its own bay with stop logs and a screen at the inlet. The pumps take suction from the circulating water pumphouse intake and discharge to a common header that runs to the circulating water intake tunnel. Water is supplied to the condenser by three lines from the intake tunnel and is discharged to the discharge tunnel through three lines. A single line from the discharge tunnel supplies all six cooling towers. After passing through the cooling towers, the water is collected in return headers to the circulating water pumphouse intake.

During plant startup and shutdown, when the heat content of the circulating water is relatively low, the system has the capability of bypassing the cooling towers and returning the water directly to the circulating water pumphouse intake. This operation is limited to having only one circulating water pump running when the cooling tower bypass valve is open and the cooling tower supply shut off valves must be closed. Flow through the bypass is equivalent to two cooling towers.

The system's blow-down flow is directed from the pump discharge header to the river to help maintain system chemistry. The blow-down line is equipped with automatic vent valves at most high points.

Circulating water pump bearing lubrication and motor cooling water is supplied from the plant service water system via a pressure reducing valve and the lube water filters. The plant service water also supplies water to provide the driving force for the primary eductor of each circulating water pump for startup.

Steam Jet Air Ejector Description:

The first stage ejectors take suction from the same suction header as the mechanical vacuum pumps. A branch line from the common header to each first stage SJAE contains an air operated suction valve. The first stage ejectors discharge to the shell side of the SJAE condenser.

The second stage ejector takes suction from the SJAE condenser and exhausts through an air operated discharge valve into the offgas system. An orificed bypass line has been added in parallel with the second stage SJAEs. This was done to ensure that the steam flow is high enough to get the required dilution of H_2 in the offgas system.

Cooling and condensing water is supplied to the SJAE condensers from the main condensate system. The condensed steam, from the first stage ejectors, is routed through a loop seal to the main condenser.

Mechanical Vacuum Pump Description:

The mechanical vacuum pumps take suction from three air outlet pipes located at the inlet end of each section (LP zone) of the main condenser. The three lines from the condenser form a common header which branches to the suction of both mechanical vacuum pumps. The common header also receives discharge from two sample lines associated with the offgas system.

The common header is equipped with an air operated shutoff valve located between the three condenser lines and the first branch lines. Each branch line to the mechanical vacuum pump contains an air operated suction valve.

Each vacuum pump discharges to an air separator. The air is discharged from the air separators through an air operated discharge valve to a common header which goes to the reactor building elevated release duct.

Seal water is provided to the air separators from the demineralized water system. Water is supplied through a float operated valve which maintains proper level in the separator. A manual bypass valve is supplied for initial filling and in case the float valve malfunctions.

MSIV Description:

Two isolation valves are installed in each steam line, for a total of 8 MSIVs. One set of valves is within the drywell and the others are located just outside the drywell, in the steam tunnel.

The MSIVs require air to open.

The MSIVs will close with spring and/or air pressure.

The MSIVs are opened pneumatically by way of a 20" piston and cylinder assembly.

The valves will fail closed upon a loss of pneumatic pressure.

Two AC powered, solenoid operated, pilot actuated valves route air to desired ports, thereby positioning the MSIV. The electrical supplies for the valves come from two separate sources, RPS buses A and B. Each MSIV has 2 pilot operating solenoids. One solenoid is powered from RPS Bus A and one solenoid is powered from RPS Bus B. A loss of both RPS buses is required to cause the valves to close.

An accumulator, mounted on the MSIV, provides backup pneumatic pressure to close the valve when both solenoids are de-energized or pneumatic supply pressure to the valve operator fails.

Opening and closure speeds of the valve are controlled by two adjustable pressure compensated flow control valves. Each valve is set up so that flow is permitted only in one direction, (open or close). The opening and closing speeds are set by adjusting one of two flow control valves, each one is adjusted separately.

The MSIVs will automatically close on any of the following signals:

- Reactor low water level
- Main steam line high radiation
- Main steam line high steam flow

- Main steam line low pressure
- Main steam line tunnel high temperature or high ventilation system differential temperature.
- Main condenser low vacuum

The MSIVs can also be manually closed by their associated control switches on P601 or by arming and depressing the four NS[4] pushbuttons on P601 (any combination of "A" or "C", and "B" or "D" pushbuttons).

Turbine Bypass Valve Description:

The turbine bypass consists of four hydraulically operated control valves which are mounted on a single valve manifold. They are connected to the main steam line header upstream of the turbine main stop valves by four 10-inch lines. Each valve outlet discharges into the manifold which is piped directly to pressure-reducing perforated pipes located in the condenser shell.

The turbine bypass system controls reactor steam pressure by sending excess steam flow directly to the main condenser. This permits independent control of reactor pressure and power during reactor vessel heatup to rated pressure prior to and while the turbine is brought up to speed and synchronized under turbine speed-load control and when cooling down the reactor. Following main turbine generator trips and during power operation when the reactor steam generation exceeds the transient turbine steam requirements, the turbine bypass valves control reactor over-pressure within its capacity and in accordance with the steam generation rate.

The turbine bypass system capacity is 25 percent of rated reactor steam flow.. The bypass system can accommodate a 25 percent turbine load rejection without causing a significant change in reactor steam flow.

The turbine bypass valves are capable of remote manual or automatic operation.

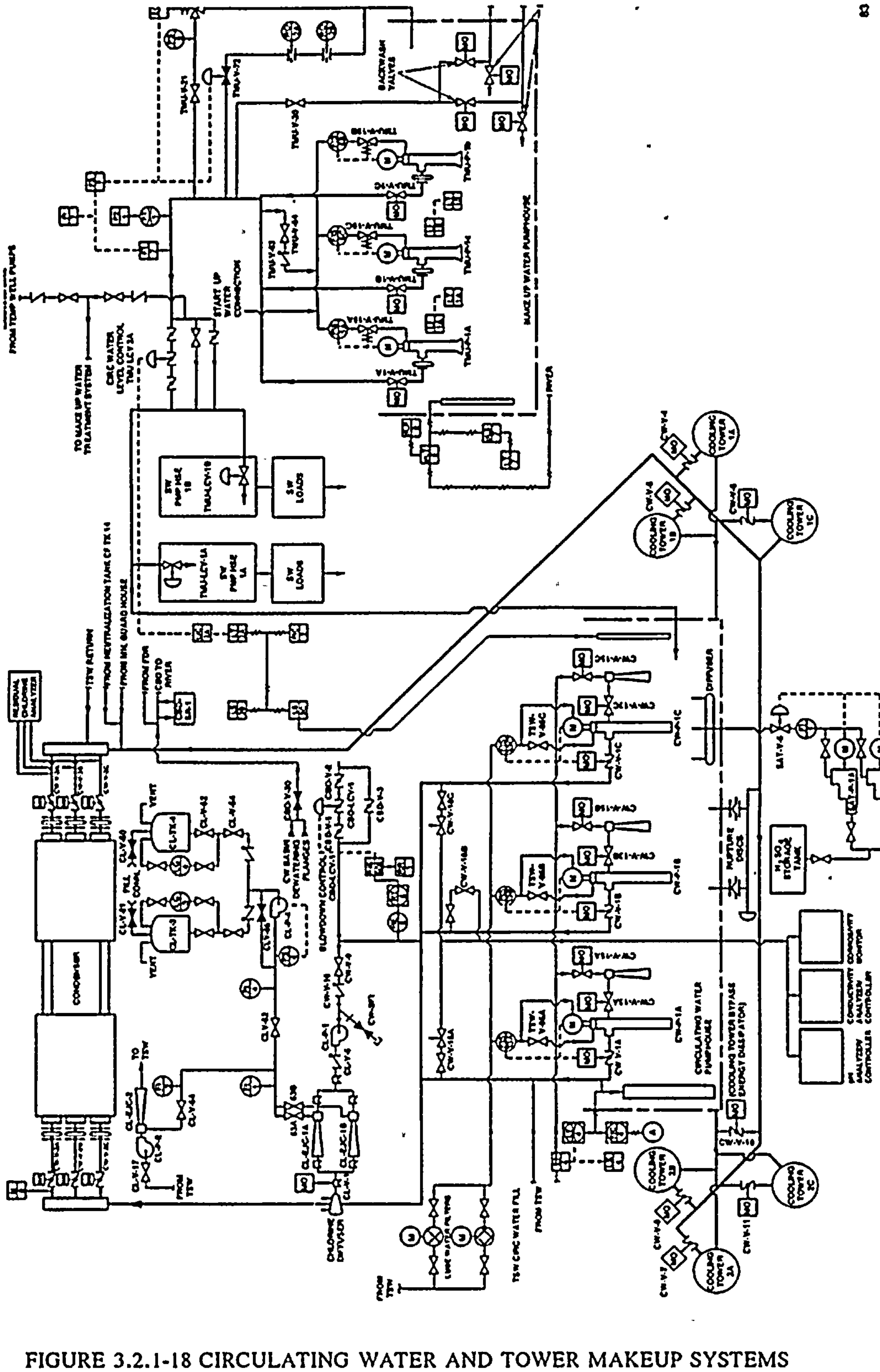

FIGURE 3.2.1-18 CIRCULATING WATER AND TOWER MAKEUP SYSTEMS

3.2.1.19 FP Water System Description

The WNP-2 Fire Protection systems consist of passive and active systems that will detect, extinguish, or contain fire in any fire area. Buildings are divided into fire areas or zones and separated by fire barriers or spatial separation. Fire detection is provided in most areas. These alarms annunciate in the main control room. Fire protection water (Figure 3.2.1-19) is provided by 4 fire pumps and a circular yard loop with sectional control valves. Automatic suppression is provided in areas required as noted in FSAR Appendix F, Fire Hazard Analysis.

The passive systems, i.e., fire walls, fire doors, fire dampers, penetration fire seals and electrical cable barrier protection are used in the safe shutdown analysis. Active systems, wet, pre-action and deluge sprinkler systems, gas (Halon 1301 or CO_2) systems, or manual systems, i.e., hoses stations, hydrants, and extinguishers are not included as part of the safe shutdown system. The fire detection and alarm system is used for the early notification of a fire, but they also are not used for the safe shutdown analysis.

In summary, the active or alarm systems are not used for safe shutdown and thus total loss will not affect the ability of the plant to safely shut down due to a fire. However, the passive parts of the fire protection system are needed to limit the spread of a fire and assure the plant can be safely shut down. The system function during a fire will be addressed in the WNP-2 IPEEE.

There are two water supplies for the fire protection system; 3 fire pumps taking suction from the Circulating Water Pump House (CWPH) intake basin and one fire pump taking suction from a 400,000 gallon bladder tank. The CWPH intake basin is considered to be an unlimited water source as the maximum refill is 25,000 gpm while maximum calculated evaporation rate of the cooling towers is about 18,500 gpm. The second source (the 400,000 gallon bladder tank) can be refilled in 8 hours or less.

The Fire Protection system can supply water to the RPV under emergency conditions. The WNP-2 emergency procedures provide a step-by-step procedure to connect the fire water system to the suction on condensate pump 2A. However, because of timing constraints, credit for this function was not taken in the IPE.

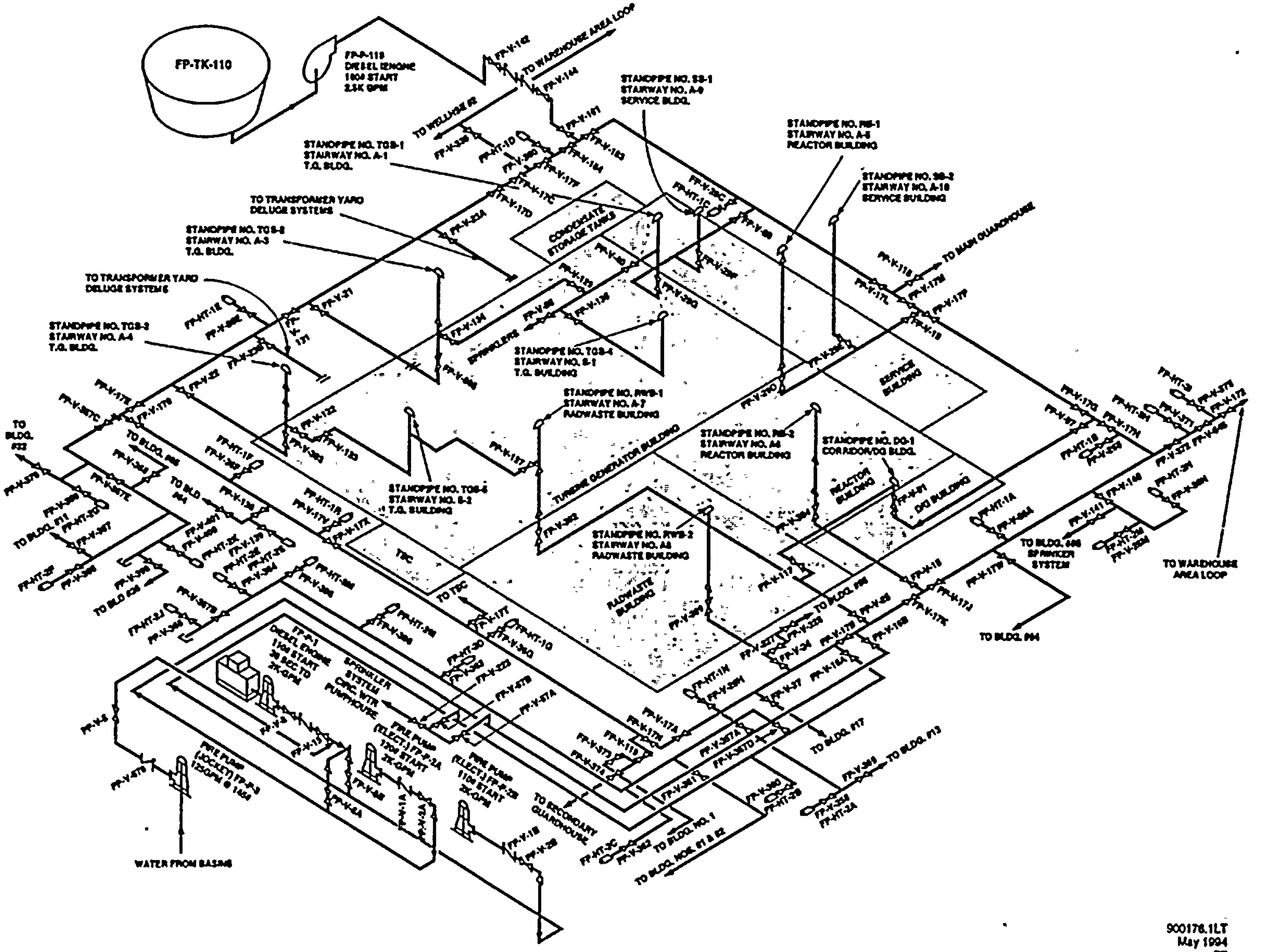

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FIGURE 3.2.1-19 FIRE PROTECTION WATER SUPPLY

3.2.1.20 Reactor Building Emergency Cooling System Description

The function of the reactor building Emergency Cooling system (RHVAC) is to maintain ambient temperature in the rooms housing critical equipment. The equipment, which is housed in individual rooms, requires a controlled environment to operate. The RHVAC system provides a controlled environment, in the event of a LOCA, by cooling recirculated room air.

Each of the fourteen rooms housing critical equipment is provided with an individual fan coil unit which is located within the room. The exception is the fuel pool heat exchanger and pump room which contains two fan coil units. Each fan coil unit is comprised of a direct drive centrifugal fan (except for the fuel pool heat exchanger and pump room fan coil units) and a water cooling coil in a sheet metal housing. The fuel pool heat exchanger and pump room fan coil units have vaneaxial fans. Water is supplied to the water coils by the standby service water system. During normal plant operation, all fifteen fan coil units are in standby. The units serving the pump rooms start upon actuation of their associated pumps. The units serving the MCC equipment rooms and the analyzer rooms start automatically upon any signal which isolates the reactor building. The units serving the fuel pool heat exchanger and pump room start on loss of offsite power or an isolation signal. Figures 3.2.1-20.1 and 3.2.1.20.2 are a simplified diagram of the system.

All units recirculate the air within the room they serve, removing the heat generated in the room via the water coil, to maintain temperatures below the design limits.

The RHVAC system is only operated during emergency, test, and reactor start-up and shutdown operations. During normal plant operation, the system does not operate. The reactor building HVAC system supplies the ventilating requirements to maintain the designed ambient conditions.

Supply air to the various MCC rooms and analyzer rooms enters the rooms through the respective intake damper, which are energized to open. The air is recirculated through the rooms and is exhausted through a backdraft damper to the main exhaust system. The backdraft dampers are set to maintain a +0.25 inch W.G. static pressure in the room.

Supply air to the pump rooms enters through ducting which directs the air to each room from the discharge of supply fan. The main exhaust fans take suction on the rooms and discharge to the elevated release point.

Supply air to the fuel pool heat exchanger and pump room enters through an intake damper which is energized to open. The air is circulated through the room and is exhausted through a damper which is also energized to open.

Various plant and system signals are used to automatically start the individual fan coil units which make up the RHVAC system, and the SW system which is used to remove heat from the units. The fan coil units then operate at full capacity and full cooling water flow to maintain the area which they serve at or below design maximum room temperature. In all areas but the pump rooms, normal air supply dampers close to isolate the rooms when the fan coil units start.

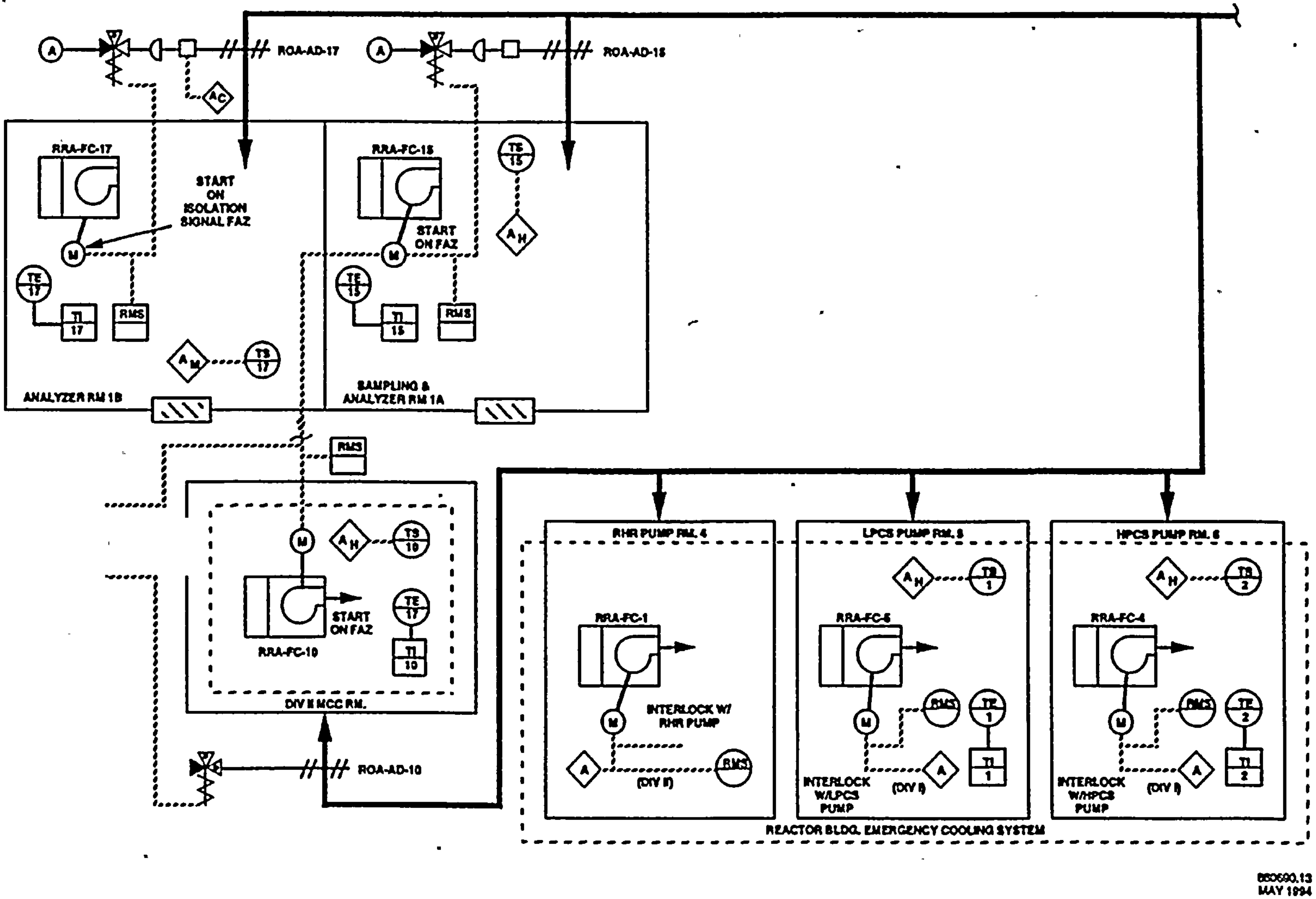

FIGURE 3.2.1-20.1 REACTOR BUILDING EMERGENCY COOLING

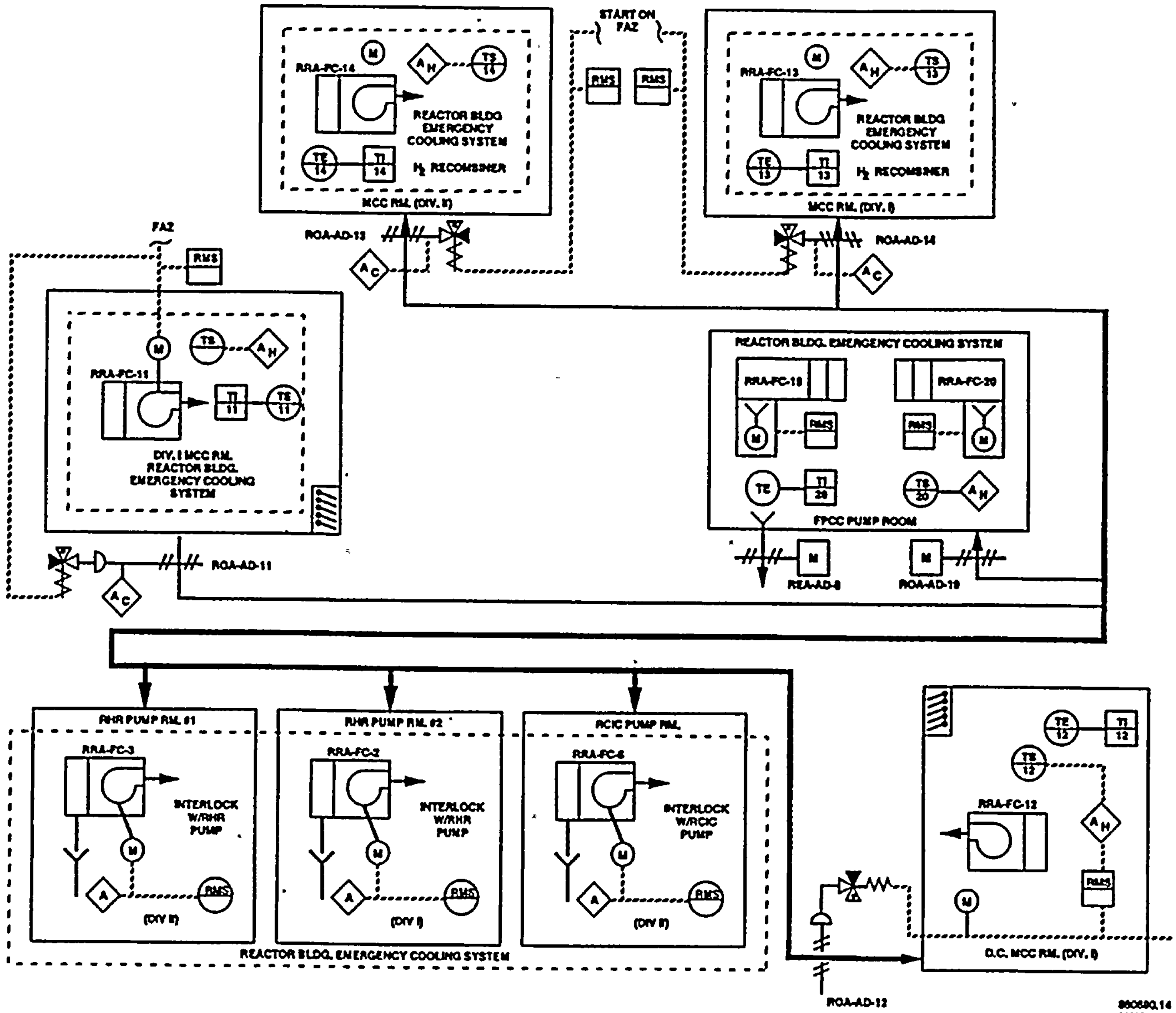

FIGURE 3.2.1-20.2 REACTOR BUILDING EMERGENCY COOLING

3.2.1.21 RPT System Description

There are two recirculation pump trip (RPT) systems: 1) ATWS-RPT, 2) EOC-RPT. The ATWS-RPT shares the same trip inputs as the ATWS-ARI system to trip the recirculation pumps.

The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine throttle valves and fast closure of the turbine governor valves.

The ATWS-RPT is designed to trip the recirculation pumps in the event of low reactor vessel water level or reactor vessel high pressure that presumes an ATWS event. This action reduces recirculating water flow within the vessel thus reducing power levels. ATWS-RPT, ATWS-ARI and SLCS combine to mitigate the consequences of an ATWS.

The EOC-RPT is designed to trip the recirculation pumps to reduce fuel thermal consequences during main turbine or generator trip transients. The trip inputs are provided by the turbine governor valves and control valves which are arranged in a two out of two and a one out of two twice logic, respectively. Both logic arrangements are bypassed below 30% of rated power to allow continued reactor operation.

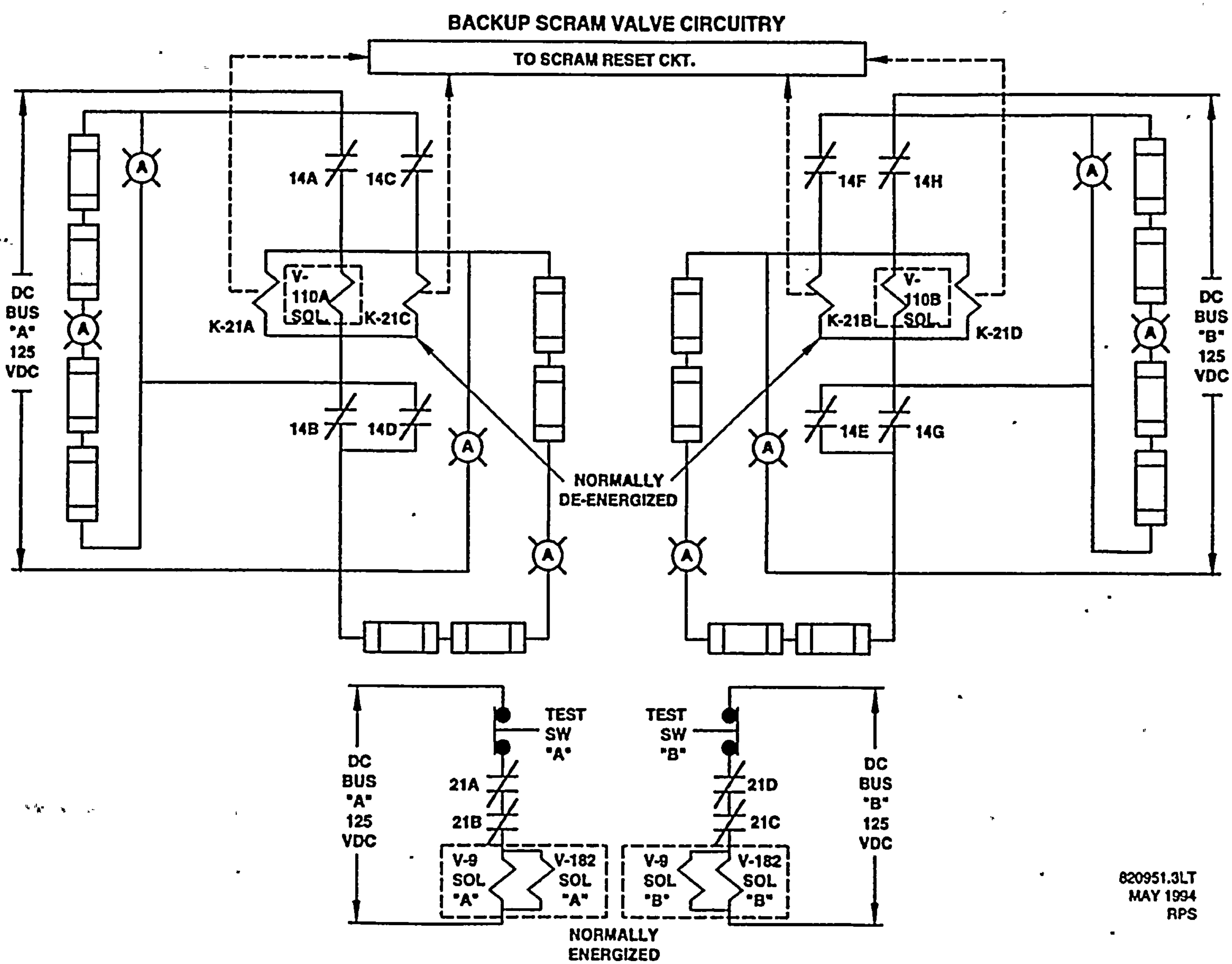

FIGURE 3.2.1-21 SCRAM LOGIC: BACKUP SCRAM VALVES SOLENOIDS (V-110A ANS B)
SDV VENT & DRN VALVES PILOT SOL. (V-9 AND 182)

3.2.1.22 ARI System Description

The function of the ATWS Alternate Rod Insertion system (ARI) is to vent the scram air header for control rod insertion in the event of an abnormal operating occurrence and lack of full RPS action to scram. See Figure 3.2.1-22.

The ATWS-ARI system is designed to provide a path to reactor shutdown which is diverse and independent from the Reactor Protection system. High reactor vessel pressure or low reactor vessel water level initiate ATWS-ARI to reduce scram air header pressure thus allowing the control rod drives to insert control blades for reactor shutdown.

There are four control rod drive header vent paths for the ATWS-ARI system. Each vent path has two valves connected in series. In addition, for testing purposes, a valve was added for makeup air from the CAS system to prevent system bleeddown.

The ATWS-ARI system is energized to operate, as opposed to the RPS which is deenergized to operate. ATWS-ARI consists of four sensor trip channels for each monitored variable with two sensors in each division. Therefore, two high reactor vessel pressure signals or two low reactor water level signals will actuate one division of ATWS-ARI. Both divisions of ATWS-ARI must be actuated to complete a full system actuation thereby venting the scram air headers.

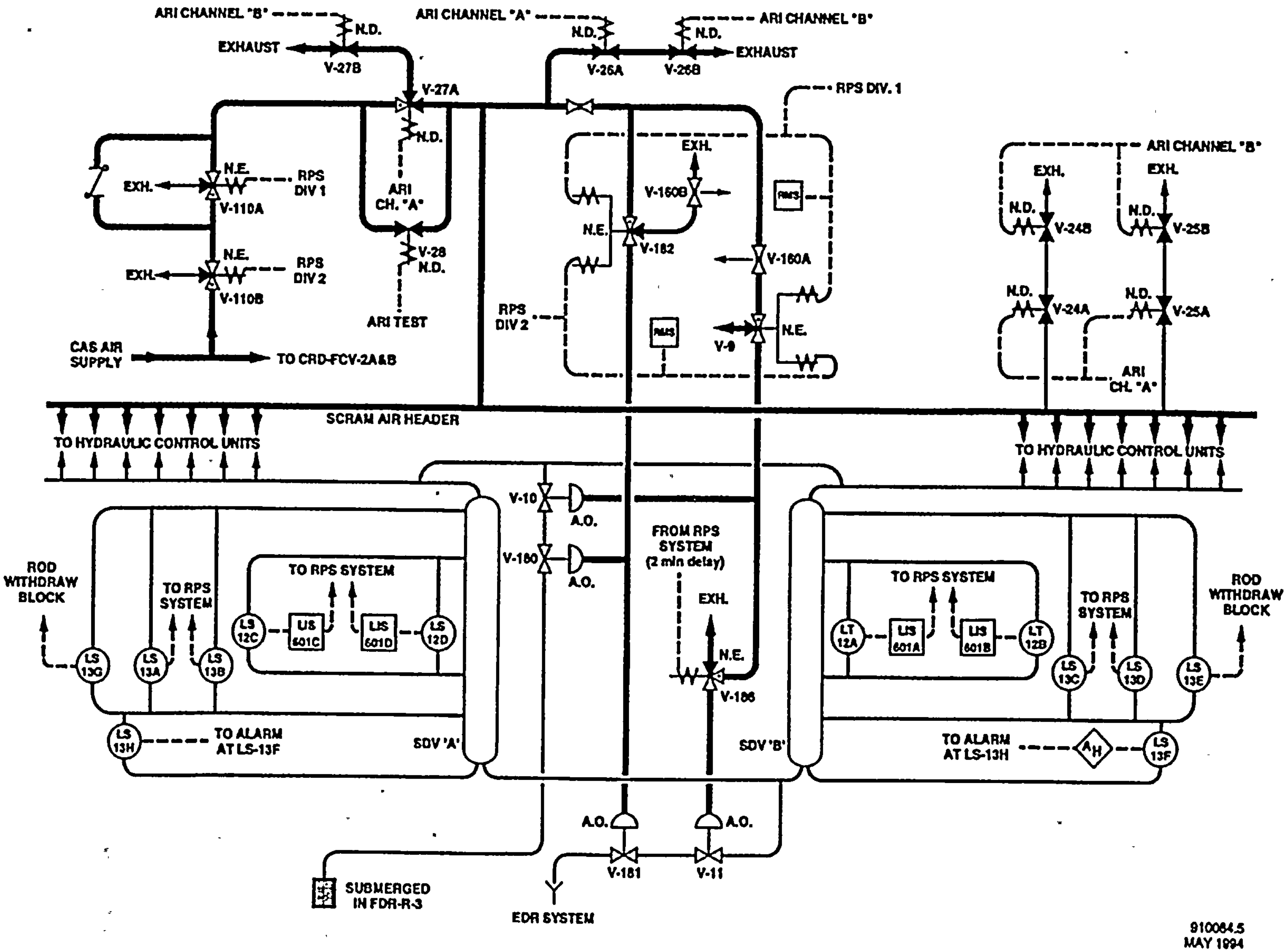

FIGURE 3.2.1-22 SCRAM DISCHARGE VOLUME & SCRAM AIR HEADER

3.2.1.23 Control Rod Drive System

The purpose of the Control Rod Drive system is three-fold. The first is to provide a means to control reactor power. The second is provide a means of shaping both the axial and radial flux profiles to achieve optimum core performance and fuel utilization. The third is to provide adequate negative reactivity to shut down the reactor from any normal or abnormal operation condition.

The CRD system consists of six basic units:

- the control rods or blades,
- the control rod drive mechanism,
- the hydraulic control units,
- the hydraulic supply system,
- the scram discharge volume, and
- the scram air header.

There are 185 control rods, each of which has its own control rod drive mechanism and hydraulic control unit. All 185 sets are supported by a single hydraulic supply system. The rods, drives, and hydraulic control units are safety-related, quality Class 1 components. The hydraulic supply portion of the CRD system is nonsafety-related, quality Class 2. There are two hydraulically connected scram discharge volumes which together support all 185 hydraulic control units. The scram discharge volumes are safety-related, quality Class 1 assemblies.

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3.2.1.24 Reactor Closed Cooling Water System

Certain plant systems and locations require cooling but are potentially contaminated such that direct cooling through the Plant Service Water can introduce a path for the release of contamination to the atmosphere. The Reactor Closed Cooling Water System (RCC) is a closed loop cooling system that removes heat from these potentially contaminated systems and locations and rejects heat to the Plant Service Water system (TSW). Its function then is to provide the required cooling while lowering the probability of releasing contamination to the atmosphere.

The RCC system provides cooling to equipment located in the primary containment, the reactor building and in the radwaste building. The System is not safety-related and has no safety classification, except for the following three portions: 1) Piping and valves forming part of the containment (ASME III-2), 2) Piping and valves associated with the fuel pool heat exchangers (ASME III-3) and 3) RCC system piping inside containment (ASME III-3). This system is not required for safe shutdown of the reactor plant after a Loss of Coolant Accident (LOCA).

The system consists of three centrifugal pumps arranged in parallel, three single-pass heat exchangers also arranged in parallel, a surge tank, and the requisite piping, valves, instrumentation, and controls. A chemical addition tank exists but is no longer used. Each of the three RCC pumps and three RCC heat exchangers provide half of the system capacity during normal condition operation. Therefore, two pumps and two heat exchangers are normally in operation while the third pump and heat exchanger are in standby. In the standby mode, the standby pump will start automatically if one of the running pumps trips. Heat is removed from the two normally operating heat exchangers by the plant service water system (TSW) from which the heat is ultimately rejected to the environment via the Circulating Water system and the Cooling Towers.

Figure 3.2.1-24 is a simplified flow diagram of the system.

In the Primary Containment, the RCC system provides cooling to the Drywell Coolers, the RRC Pump Motors, seals and motor bearings, and to the Drywell Sump EDR Heat Exchangers. In the Reactor Building, the primary loads are the RWCU Pump Motor Coolers and Nonregenerative Heat Exchangers, CRD Pump Seals and Bearings, and the Fuel Pool Heat Exchangers. The sole RCC loads in the Radwaste Building are the Offgas Glycol Refrigeration machines.

A surge tank is provided to accommodate cooling water volume changes from thermal expansion and contraction, to provide a means to add makeup water to the system, and to provide adequate NPSH for the pumps. The makeup water is supplied from the Demineralized water (DW) system.

The RCC Pumps and Heat Exchangers are located in the Reactor Building at elevation 548′.

The surge tank is also in the Reactor Building, at elevation 572′.

In the event of high Dry Well Pressure or low Reactor Water Level signal, all three RCC Pumps will trip, all RCC Containment isolation valves will close and the Reactor Building to Radwaste Building isolation valve automatically closes. When the isolation signals are reset (using the isolation logic reset pushbuttons in the Control Room) the RCC pumps will auto start but the isolation valves have to be opened using their control switches.

The system utilizes both air and motor operated control valves with position indication the Main Control Room. Relief valves are provided on each reactor building closed cooling water heat exchanger to protect the shell side (Reactor closed cooling water side) from overpressure as well as on the RWCU heat exchangers, fuel pool heat exchangers, reactor building equipment drain heat exchanger, and the equipment drain condenser. Each RCC Heat Exchanger is provided with a motor operated valve on the discharge side. These valves are opened and closed manually.

RCC cooling water is provided through two main branches from the exchanger discharge header. One branch provides cooling water to the equipment inside the primary containment whereas the other serves the equipment in the reactor building and radwaste building. Upon loss of offsite power, cooling water supply to the equipment in the reactor building and the rad waste building will be cut off by a motor operated valve and only one RCC pump will remain operating. Cooling to equipment inside primary containment is maintained. The RCC Pumps are supplied through 480V critical buses SL-71 and SL-81. The pumps do not trip upon loss of voltage and hence will automatically restart when power is available.

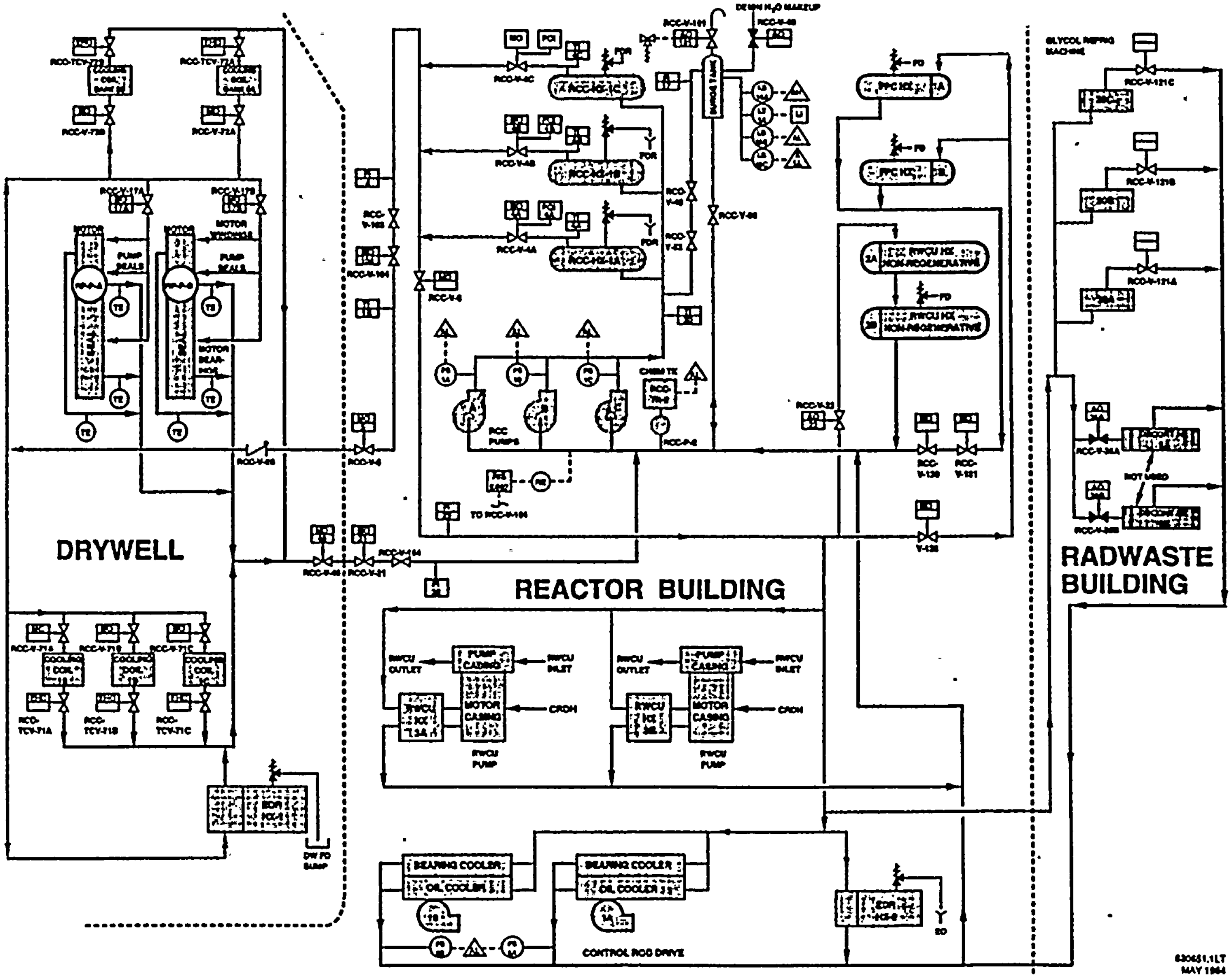

FIGURE 3.2.1-24 REACTOR CLOSED COOLING WATER SYSTEM

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3.2.2 Fault Trees

Fault trees are used to determine the probabilities of system failures. A fault tree represents a deductive logic model, reasoning from the general to the specific. At the top of a fault tree, an undesired event (system failure) is postulated. The undesired event can be due to a number of causes. Those causes can be due to a number of subcauses and so on. Graphically, the fault tree branches into chains of causes, subcauses and so on. At the bottom of the tree are the basic faults which contribute to the undesired event at the top of the tree. The Supply System has developed detailed fault trees taking into account component failures, initiation and control failures (including logics and interlocks), support system failures, test and maintenance unavailabilities, operator errors, and common cause failures. Fault trees are developed down to the relay and sensor levels beyond which component failure rates are not readily available. There are a total of 24 system fault trees developed for the IPE. They are described in calculation files which are retained at the Supply System.

All systems are dependent, in some way, on other systems for successful operation. System dependencies were accounted for by explicitly including the dependencies within the models. This was accomplished by modeling each dependency on other systems or portions of systems with an "external transfer" to the appropriate gate within the system model of the supporting system fault tree. The linking of the fault tree then incorporates the full model to fully account for all dependencies and assuring the dependencies and conditional probabilities are carried through the quantification of the event trees.

3.2.3 System Dependencies (Dependency Matrices)

Front line systems (RPS, ADS, RCIC, ECCS, etc.) are dependent on support systems (AC, DC, TSW, SW, CIA, etc.) and support systems are dependent on each other. Also front line systems are dependent on each other. Finally, accident initiators can effect support systems. Tables 3.2.3.1 through 3.2.3.4 show the system dependencies at WNP-2. Three types of dependencies are considered. Interdependence means neither system can operate without the other system also operating. For example, if offsite power is not available, the service water system requires that the diesel generators be in operation. The diesel generators, in turn, require that the service water system be available for cooling. Complete dependence means that if one system is unavailable, the other system is unavailable but not vice-versa. Partial or delayed dependence means that the failure of one system will not totally disable the other system. For example, the failure of the AC system will fail the motor driven pumps of the fire protection system. However, the fire protection system has 2 diesel-driven pumps which can still operate. An example of delayed dependence is that loss of room cooling will not immediately cause the failure of other systems. But, as the temperature in a pump room increases, the likelihood of pump failures also increases.

System fault trees have external transfers to other system fault trees at appropriate points. Linking of fault trees during fault tree reduction and cutset generation ensures system dependencies are accounted for in the IPE results. System dependency matrices and their justifications follow.

Justification for the Dependency Matrices in Tables 3.2.3-1 and 3.2.3-4

- <u>RODS</u>

During normal reactor operation each of the 2 logic channels (A and B) of the RPS energizes one of the two 3-way solenoid-operated scram pilot valves (117, 118). When normally energized, the pilot valves supply control air pressure to the diaphragm actuators of the inlet and outlet scram valves, maintaining the scram valves closed. Upon initiation of scram, both RPS logic channels are deenergized thereby venting control pressure from the scram valve actuators and permitting the scram valves to open. Scram accumulators consisting of high-pressure nitrogen provides the source of energy through water to scram the control rods. Upon loss of control air, the reactor will be in full scram. Upon loss of Division 1 (or Division 2) AC which supplies power to the RPS, the reactor will be in half scram. Upon loss of both divisions of AC, the reactor will again be in full scram. Therefore, scram with control rods is only partially dependent on the RPS instrumentation for the redundant signals (Matrix 1) and on nothing else (Matrix 4).

- <u>ARI</u>

The ARI is designed to provide reactor scram independent of the RPS. High reactor vessel pressure or low reactor vessel water level initiates the ARI to vent scram air header pressure thus scramming the control rods. The ARI is energized to operate as opposed to the RPS which is deenergized to operate. Division 1 and Division 2 125V DC are used for the ARI logic and valve actuation. Both divisions must be actuated to complete a full system actuation thereby venting the scram air headers. Therefore, the ARI actuation is dependent on both divisions of DC and is partially dependent on the instrumentation for the redundant signals (Matrix 1). Since the ARI and the RPT-ATWS are actuated by the same instrumentation, the ARI can be considered partially dependent on the RPT-ATWS (Matrix 4).

- <u>RPT-ATWS</u>

The RPT-ATWS is designed to trip both 60 Hz and 15 Hz breakers for the recirculation pumps in the event of an ATWS. Division 1 and Division 2 125V DC are used for the RPT-ATWS logic and actuation. Either division can complete a RPT-ATWS actuation tripping both recirculation pumps. Therefore, the RPT-ATWS actuation is partially dependent on each division of DC and on the instrumentation for the redundant signals (Matrix 1). Since the RPT-ATWS and the ARI are actuated by the same instrumentation the RPT-ATWS can be considered partially dependent on the ARI (Matrix 4).

- <u>RPT-EOC</u>

The RPT-EOC is designed to shift the recirculation pumps to the slow speed in the event of a turbine or generator trip. The trip inputs are provided by the turbine governor valves and control valves which are arranged in a two out of two and a one out of two twice logic, respectively. Division 1 and Division 2 125V DC are used for the RPT-EOC logic and actuation. Either division can complete a RPT-EOC actuation shifting both Recirculation Pumps to the LFMG. Therefore, the RPT-EOC actuation is partially dependent on each division of DC and on the instrumentation for the redundant signals (Matrix 1).

- <u>SLC</u>

Except for a common section of passive injection piping, the SLC system has 2 loops. Loop A (inlet valve, pump, injection valve, etc.) is dependent on Division 1 AC. Loop B is dependent on Division 2 AC. The positive displacement pump in each loop is rated at 43 gpm. System initiation is manual. When both loops are actuated, the total injection into the core is 86 gpm. Therefore, the SLC (A) actuation is dependent on Division 1 AC; and the SLC (B), on Division 2 AC (Matrix 1). Heat tracing on piping is not mandatory because it has been shown that the reactor building HVAC can maintain elevated temperatures for sodium pentaborate solution. If the installed heat tracing is not working, building HVAC may be needed to prevent solution crystallization.

- <u>ADS</u>

There are 18 SRVs, seven of which are ADS valves. Each ADS is a 2 loop emergency system. There are 3 solenoid valves for relief actuation. The system operates by energizing the solenoid, thereby allowing air to fill air cylinder. This action will mechanically open the valve. Power supply for 2 solenoids is from Division 1 125V DC. Power supply for the remaining solenoid is from Division 2 125V DC. Each ADS valve has 2 accumulators (10 gallon and 42 gallon) to allow for one actuation against the maximum drywell pressure with zero reactor pressure in the unlikely event that the Nitrogen Inerting system and the Containment Instrument Air system fail. The ADS will automatically actuate when the following signals are all present:

- Level 1 and Level 3
- 105 second time delay
- At least one low pressure ECCS pump is running

The ADS may be actuated manually provided one low pressure ECCS pump is running. Therefore, ADS I and II are shown as redundant systems with ADS(I) dependent on Division 1 DC and ADS(II) on Division 2 DC (Matrix 1). Both ADS I and II are partially dependent on instrumentation, and CIA (Matrix 1) and LPCS and LPCI A, B and C (Matrix 4).

- SRV

There are 18 SRVs seven of which are ADS valves. A pressure switch is supplied with each relief valve. The switch actuates the DC solenoid operated valve which permits nitrogen from CIA (or air from CAS) to enter the air operator cylinder which in turn mechanically opens the valve. Each valve may also be opened and closed manually. A ten gallon accumulator is supplied with each valve to provide a source of N_2 to the air cylinder in the event of a loss of CIA and CAS. The safety mode of the valve is actuated directly by the force exerted upon the main spring by reactor pressure. The power supply to SRVs is from DP-S1-1A (Division 1 DC). Therefore, the SRVs are totally independent for safety function (Matrix 4) but are partially dependent on CIA, CAS and the Division 1 DC for relief function (Matrix 1).

- MSIV - INBOARD

There are 4 inboard MSIVs one on each steam line. The MSIVs require N_2 from CIA to open. CAS can be fed to CIA. The MSIVs may be closed with spring or N_2/air pressure. Each MSIV has 2 solenoid operated pilot actuating valves in parallel routing N_2/air to open or close the valve. During normal operation one solenoid is energized from Division 1 AC and the other solenoid is energized from Division 2 AC. A loss of both AC divisions is required to cause the MSIVs to close. By energizing either or both of the solenoids, a MSIV can be opened. An accumulator, mounted on each MSIV, provides backup pneumatic pressure to close the valve when both solenoids are deenergized or pneumatic supply pressure (CIA) fails. The MSIVs will automatically close on any of the following signals:

- Reactor low water level (Level 2)
- Main steam line high radiation
- Main steam line high steam flow
- Main steam line low pressure (with Mode Switch in RUN)
- Main steam line tunnel high temperature or, high ventilation system differential temperature
- Main condenser low vacuum

Therefore, the opening of the inboard MSIVs is dependent on CIA, and is partially dependent on CAS, Division 1 AC, Division 2 AC, and instrumentation (Matrix 1). Main steam line tunnel cooling is provided by TSW. Steam tunnel cooling fans are powered off MC-7C and MC-8C. Tunnel isolates at 164°F high and 80°F delta-T. Loss of either fan at full power will cause tunnel isolation. Therefore, the opening of the inboard MSIVs is partially dependent on TSW (Matrix 1). Loss of the condensate system causes loss of the steam jet air ejector cooling which in turn can cause loss of condenser vacuum. Loss of condensate at power would cause loss of RFW which in turn would cause scram at L3 and MSIV closure at L2. Condenser vacuum can be established by running the mechanical vacuum pumps at < 5% power. However, SJAE (and COND) must be in-service above 5% power. Therefore, the opening of the inboard MSIVs is dependent on the condenser and the condensate system (Matrix 4). Although opening of MSIVs in mode 1 is dependent upon condenser vacuum, the low vacuum isolation is bypassed by the mode switch when the switch is not in run.

- MSIV - OUTBOARD

There are 4 outboard MSIVs, one on each steam line located in the steam tunnel. They work in the same way as the inboard MSIVs, with the exception that the CAS system is used instead of the CIA system. Steam tunnel cooling is provided by TSW. Steam tunnel cooling fans are powered off MC-7C and MC-8C. Tunnel isolates at 164°F hi and 80°F delta-T. Loss of either fan at full power will cause tunnel isolation. The CAS is cooled by the TSW. Therefore, the opening of the outboard MSIVs is dependent on CAS, and is partially dependent on Division 1 AC, Division 2 AC, instrumentation and TSW (Matrix 1). Loss of the condensate system causes loss of the steam jet air ejector cooling which in turn can cause loss of condenser vacuum. Loss of condensate at power would cause loss of RFW which in turn would cause scram at L3 and MSIV closure at L2. Condenser vacuum can be established by running the mechanical vacuum pumps at < 5% power. However, SJAE (and COND) must be in-service above 5% power. Therefore, the opening of the outboard MSIVs is dependent on the condenser and the condensate system (Matrix 4).

- CONDENSER

The condenser hotwell serves as a collection point for all three condenser pressure zones. The circulating water cools the condenser. The CW pump motors and bearings are cooled by TSW. The CW pumphouse is heated by electric heaters and cooled by drawing in outside air through louvers on the side of the building and exhausting it through roof vents. Six circulator mechanical draft cooling towers remove heat from the CW. During normal operation, the SJAEs take a suction on the main condenser and discharge through the ejector condensers to the off-gas system. The CW, cooling tower fans, and SJAE condenser cooling all use offsite power source. The motive power for the TSW is from SM-75 or 85. Loss of offsite power in combination with a LOCA signal will trip the feeder breakers to SM-75 and 85 from SM-7 (Division 1 AC) and SM-8 (Division 2 AC), respectively. The control power for the TSW is from Division 1 DC or Division 2 DC. Therefore, core cooling with the main condenser is dependent on the offsite power and the TSW, and partially dependent on Division 1 AC, Division 2 AC, Division 1 DC and Division 2 DC (Matrix 1). For core cooling with the main condenser, MSIVs must be open, and the condensate system must cool the SJAE to prevent loss of condenser vacuum at > 5% power. Since the inboard and outboard MSIVs are dependent on the CIA and CAS for opening, the condenser is shown dependent on CIA and CAS in Matrix 1. The main condenser is shown dependent on the MSIVs and the condensate system (Matrix 4).

- RFW

The feedwater system has 2 turbine driven feedwater pumps. Plant service water system supplies cooling water to the RFW turbine oil coolers. There are two methods of feedwater control. The first method employs air-operated (CAS) feedwater startup valves which fails as-is on a loss of air. During startup, this low flow valve maintains the vessel inventory level in response to operator manual control or automatically in response to the vessel level instrumentation. The second flow controlling device is the turbine driven variable speed feedwater pumps (usually in-service above 25% power) which automatically control the feedwater flow rate in response to its controller signal during power ascension and normal power levels. The sealing steam system is utilized to pressurize the shaft glands to prevent contaminated steam from escaping to the environment and condenser air in-leakage along the shaft. The condensate system, which is dependent on offsite power, has to operate before the RFW can be used as high pressure core injection. For the condensate system to operate, the water in the condenser hotwell must be available. Therefore, for high pressure core injection, the RFW is dependent on the offsite power, the TSW, and the CAS. Since TSW is dependent on Division 1 or 2 AC, and Division 1 or 2 DC, RFW is partially dependent on Division 1 or 2 AC and Division 1 or 2 DC (Matrix 1). The RFW is dependent on MSIVs, the condensate system and the condenser (Matrix 4).

- HPCS

The HPCS has a motor driven pump. The motive power for the HPCS is from the Division 3 AC. The control power for the HPCS is from the Division 3 DC. There is a water leg pump dependent on the Division 3 AC. The main pump seals and bearings are cooled by its own discharge. The pump room is cooled by the SW-C. Initiation of the system is either automatic or manual. Normal lineup is from condensate storage tanks. In the event that the condensate storage water supply becomes exhausted, suction is automatically switched over to the suppression pool. The NPSH requirement for the pump can be affected by the systems for containment pressure and temperature control (DW Coolers, Suppression Pool Cooling, DW Sprays, WW Sprays, and Venting). RCC to the drywell is isolated on an FA signal. The drywell fans become the air-H_2 mixing system. Therefore, the HPCS is dependent on the Division 3 AC, the Division 3 DC, the SW-C, and the reactor building emergency cooling, and is partially dependent on the instrumentation (Matrix 1). The HPCS is partially dependent on the Suppression Pool Cooling, DW Coolers, DW Sprays, WW Sprays, and the Venting (Matrix 4).

- RCIC

The RCIC has a turbine driven pump. The motive power for the valves is from the 250V Division 1 DC. The control power for the system is also from the 125V Division 1 DC. There is a water leg pump dependent on the Division 2 AC. Initiation of the system is either automatic or manual. Upon RCIC initiation, Loop B of the Standby Service Water starts and the emergency cooling fan of the RCIC pump room also starts. However, by keeping pump room doors open, natural circulation is sufficient to cool the room for continuous pump operation. Normal lineup is from the condensate storage tanks. In the event that the condensate storage water supply becomes exhausted, suction is automatically switched over to the suppression pool. The NPSH requirement for the pump and the exhaust pressure for the turbine can be affected by the systems for containment pressure and temperature control (DW Coolers, Suppression Pool Cooling, DW Sprays, WW Sprays, and Venting). Therefore, the RCIC is dependent on the Division 1 DC, and is partially dependent on the Division 2 AC, the SW-B, the instrumentation, and the reactor building Emergency Cooling (Matrix 1). If a SRV sticks open, the reactor can depressurize to a point such that the RCIC turbine does not have sufficient steam pressure for operation. The RCIC is partially dependent on the SRVs, DW Coolers, Suppression Pool Cooling, DW Sprays, WW Sprays, and Venting (Matrix 4).

- <u>LPCS</u>

The LPCS has a motor driven pump. The motive power for the LPCS is from the Division 1 AC. The control power for the LPCS is from the Division 1 DC. There is a water leg pump dependent on the Division 1 AC. The pump motor and the pump room are cooled by the SW-A. Initiation of the system is either automatic or manual. Injection starts after the reactor is sufficiently depressurized. The system takes a suction from the suppression pool and discharges the water to the core. The NPSH requirement for the pump can be affected by the systems for containment pressure and temperature control (DW Coolers, Suppression Pool Cooling, DW Sprays, WW Sprays, and Venting). Therefore, the LPCS is dependent on the Division 1 AC, the Division 1 DC, the SW-A, and the reactor building Emergency Cooling, and is partially dependent on the instrumentation (Matrix 1). The LPCS is partially dependent on the ADS I and II, the Suppression Pool Cooling, DW Coolers, DW Sprays, WW Sprays, and the Venting (Matrix 4).

- <u>LPCI</u>

The RHR has 3 LPCI loops pumping water from the suppression pool into the core region of the vessel. LPCI injection does not start until the pressure in the reactor vessel is low enough. The conditions that initiate LPCI are high drywell pressure or low reactor water level L1. There is a water leg pump keeping loops B and C filled. The loop A is kept filled using the water leg pump from the LPCS system. The motive power for the loop A is from the Division 1 AC whereas that for the loops B and C is from the Division 2 AC. The control power for the loop A is from the Division 1 DC whereas that for the loops B and C is from the Division 2 DC. Service Water A and B cool the loop A (B and C) pump seals and pump rooms, respectively. The NPSH requirements for the pumps can be affected by the systems for containment pressure and temperature control (DW Coolers, Suppression Pool Cooling, DW Sprays, WW Sprays and Venting). Therefore, the LPCI-A is dependent on the Division 1 AC, the Division 1 DC, the SW-A whereas the LPCI-B and C are dependent on the Division 2 AC, the Division 2 DC, the SW-B. (Matrix 1). The LPCI-A, B and C are dependent on reactor building Emergency Cooling and are partially dependent on the instrumentation (Matrix 1). The LPCI-A, B and C are partially dependent on the ADS I and II, the Suppression Pool Cooling, DW Coolers, DW Sprays, WW Sprays, and the Venting (Matrix 4).

- <u>CONDENSATE SYSTEM</u>

The condensate is a low pressure system. Three condensate pumps take suction from the condenser hotwell via a single header. They are motor driven pumps with the power supply from the offsite power. The motors are cooled by a combination of air and water. Water from the TSW is utilized to cool the upper thrust bearing of the motors. After passing through gland seal steam condenser, SJAE condensers, offgas condenser and the filter demineralizers, the condensate is directed to the suction of 3 condensate booster pumps. They are motor driven pumps with the power supply from the offsite power. The pumps are oil and water cooled. The water is from the TSW. Two lubricating oil systems are also cooled by the TSW. After passing through 5 series of heaters, the condensate is directed to the suction of the RFW system. TSW depends on Division 1 or 2 AC and Division 1 or 2 DC. RFW depends on CAS for startup valve control. Therefore, the condensate system is dependent on the offsite power, and the TSW, and is partially dependent on the Division 1 and 2 AC, and Division 1 and 2 DC (Matrix 1). The condensate system is partially dependent on the ADS I and II, the condenser and the RFW (Matrix 4).

- <u>FP WATER</u>

An isolation valve and fire hose connection are installed on the suction of the 'A' condensate booster pump. There are 4 fire pumps. Three of these (one diesel and two motor driven) take suction from the circulating water basin. The fourth pump (diesel) takes suction from the bladder tank which is filled from either the well house storage tank or from the TMU system. Since the FP water has to go through the COND and the RFW, it has the same dependencies as the COND and the RFW except for the offsite power and the TSW. Therefore, the FP water is partially dependent on the offsite power for its motor driven fire pumps (Matrix 1). Since the FP water is a low pressure system, it partially depends on RFW, and ADS I and II for depressurization before injection can occur (Matrix 4).

- <u>DW COOLERS</u>

Cooling of the drywell is provided by 5 fan coil units which recirculate containment air through cooling coils. Heat is transferred to the RCC system. The RCC system dumps its heat to the TSW system. RCC isolates on a LOCA signal. CRA-FNS-1A2, 1B2, 1C2, 2A1, 2B1, 4A and 4B all get a start signal to serve as drywell air mixers on F and A signals. There are 2 fans that draw air from the containment head area and return it to the general drywell area. In addition, there are 7 recirculation fans that provide recirculation of air in the drywell. Some fan coil units and fans depend on Division 1 AC; and others on Division 2 AC. During normal operation, 5 fan coils and some of the recirculation fans are running. All units are under manual control. Therefore, the DW Coolers are dependent on

Division 1 or 2 AC, the RCC and the TSW (Matrix 1). Since the upper DW Spray A could damage the fan coils and the recirculation fans; the DW Cooler operation is partially dependent on the DW Spray A (Matrix 4).

- <u>SHUTDOWN COOLING, SUPPRESSION POOL COOLING, DW SPRAYS, WW SPRAYS</u>

Shutdown Cooling, Suppression Pool Cooling, DW Sprays, and WW Sprays are different modes of the RHR system. These modes are under manual control and are mutually exclusive. The LPCI mode of the RHR takes precedence over all other modes. The motive power for the RHR loop A is from the Division 1 AC whereas that for the RHR loop B is from the Division 2 AC. The motive power for the shutdown cooling inboard isolation valve is from the Division 2 AC; and that for the shutdown cooling outboard isolation valve is from the Division 1 250V DC. Motive power for both shutdown cooling return valves RHR-V-53A and RHR-V-53B is from Division 1 480V AC. The control power for the RHR loop A is from the Division 1 DC whereas that for the RHR loop B is from the Division 2 DC. Service water A and B cool the loop A and B pump seals and pump rooms, respectively. Therefore, the Shutdown Cooling (A), the Suppression Pool Cooling (A), the DW Spray (A) and the WW Spray (A) are dependent on the Division 1 AC, Division 1 DC, SW-A, and the reactor building Emergency Cooling (Matrix 1). The Shutdown Cooling (B), the DW Spray (B) and the WW Spray (B) are dependent on the Divisions 1 and 2 AC, Division 2 DC, SW-B, and the reactor building Emergency Cooling (Matrix 1). The Shutdown Cooling (B) is also dependent on Div. 1 AC and Div. 1 DC because of RHR-V-53 and RHR-V-8. Because of the mutual exclusiveness, each RHR mode is partially dependent on the other modes not being operational (Matrix 4).

- <u>HEAD SPRAY, CONTAINMENT FLOODING</u>

Head spray takes suction from the recirculation loop A and discharges flow to the reactor vessel head via RHR-B to help depressurize the reactor and cool the upper vessel components. Reactor head spray valve RHR-V-23 is powered from Div. 1 DC. There are 2 motor operated valves along the suction line, RHR-V-8 and RHR-V-9. RHR-V-8 is powered from the Division 1 250V DC. RHR-V-9 is powered from the Division 2 AC. Service water from the service water B header can be lined up to the discharge of the RHR heat exchanger B via 2 key-locked valves. When the valves are open, service water can be directed to any path associated with the RHR loop B for containment flooding. Except for RHR-V-8 and RHR-V-23, the motive power for the RHR loop B is from the Division 2 AC. The control power for the RHR loop B is from the Division 2 DC. Service water B cools the loop B pump seal and pump room. Therefore, the head spray/flood is dependent on the Division 2 AC, Divisions 1 and 2 DC, SW-B, and the reactor building Emergency Cooling (Matrix 1). The containment flooding is dependent on the Division 2 AC, Divisions 1 and

2 DC, SW-B (Matrix 1). Because of the mutual exclusiveness, the head spray and the containment flooding are partially dependent on the other RHR modes not being in operation (Matrix 4).

- CONTAINMENT VENTING

The primary containment can be vented to the Standby Gas Treatment system or the reactor building Exhaust system through the containment exhaust isolation valves or their bypass valves. Along each of the exhaust paths from the wetwell or the drywell, there are 2 exhaust valves (or 2 bypass valves) in series. One valve is dependent on Division 1 AC; and the other, on Division 2 AC. The valves are normally closed, and fail closed upon loss of control power or air supply from the CAS. Each valve has a three position switch (Close, Norm, and Open) in the control room with spring return to "Norm" from the "Open" position. The solenoid admitting air to the valve actuator must be energized to open the valve. Opening is permissive on containment isolation signals F, A, and Z not being present. The valves only open manually, and once momentarily open contact is made at the switch, the open signal seals in. The seal-in feature drops and the valve closes if a containment isolation signal occurs when the valve is open. Also, placing the manual switch in the "Close" position breaks the seal-in open signal. The hand switch does not spring return to the "Norm" position from the "Close" position. Therefore, the containment venting is dependent on the Division 1 and 2 AC and CAS and is partially dependent on TSW because TSW cools CAS (Matrix 1).

Justification For The Dependency Matrix in Table 3.2.3.2

- OFFSITE POWER (SM-1, -2, and -3)

There are 3 offsite power sources: 230 kV from the Ashe Substation (startup transformer), 115 kV from the Benton Substation (backup transformer), and 500 kV from the Ashe Substation (normal transformer, main generator). The plant is started from the startup transformer and then transferred to the normal transformer when the plant output has reached 25% rated capacity. The startup transformer remains energized to permit the onsite AC system to be automatically transferred back in the event of a loss of power from the generator. If the startup transformer is unavailable, when the generator is out-of-service, noncritical loads are shed and the critical buses (Division 1 and 2) are automatically transferred to the backup transformer. Load transfer, load shedding, switchgear and bus operations are dependent on the DC power. If the main generator, the startup transformer, and the backup transformer are unavailable, power can be obtained from Diesel Generators or the 500 kV Ashe Substation. This is accomplished by disconnecting the isolated phase bus duct links to isolate the main generator. Power can then be established through the normal transformer to the plant. Experience has shown that it takes approximately 8 hours to make this transition. Therefore, the offsite power is partially dependent on the Division 1 and 2 DC.

- DIVISION 1, 2, and 3 AC (SM-7, -8, and -4)

Division 1 (SM-7) and 2 (SM-8) AC are supported by the normal transformer, the startup transformer, the backup transformer and the Diesel Generators 1 and 2, respectively. Division 3 (SM-4) AC is supported by the normal transformer, the startup transformer, and the Diesel Generator 3. Only loads supplied by the Division 1 or 2 power systems will enable operation of redundant systems and equipment for plant shutdown. Load transfer, load shedding, switch-gear and bus operations are dependent on the DC power.. In the event of a loss of offsite power the diesels supply power to the Standby Service Water system. The SW-A, B, and C in turn supply cooling to the Diesel Generators 1, 2, and 3, respectively. The individual Diesel Generator room cooling is performed by the SW-A, B, and C. Division 1 and Division 2 Diesel Generator area cable cooling is also performed by SW-B. Moreover, Reactor Building Emergency Cooling utilizes SW to cool the critical motor control centers. Critical switchgear rooms are cooled by either SW or TSW. Therefore, Division 1, 2, and 3 AC are interdependent on the SW-A, B and C respectively, and are partially dependent on the Division 1, 2, and 3 DC respectively, and on the reactor building Emergency Cooling, and the TSW.

- DIVISION 1, 2, and 3 DC

Each DC distribution system has a battery and a battery charger that are normally connected to the respective AC bus. During an emergency, the critical switchgear rooms utilize the TSW and/or the SW-A and B to cool the Division 1 and 2 battery and battery-charger rooms. Division 3 battery and battery-charger rooms are cooled by the HPCS diesel room cooling using the SW-C. Moreover, the SW-A and B cool the Division 1 DC MCC rooms, respectively. Therefore, the Division 1, 2, and 3 DC are partially dependent on the Division 1, 2 and 3 AC, the reactor building Emergency Cooling, and SW-A, -B, and -C, and the TSW.

- TSW

The TSW takes suction from the CW basin. The system is operated with one TSW pump in-service and the other one in standby. The motive power for the system is from the Division 1 or 2 AC. The control power for the system is from the Division 1 or 2 DC backed by AC. Both divisions of AC or DC will have to be disabled to disable the TSW. The system supplies lube water to its own motors and seals. Loss of offsite power signal in combination with LOCA signal will trip the feeder breakers to SM-75 and 85. This results in a trip of the TSW pumps. Therefore, the TSW is partially dependent on the Divisions 1 and 2 AC, Divisions 1 and 2 DC, and the instrumentation.

- SW

Each of the 3 Standby Service Water pumps receives its power from independent AC buses. In the event of a loss of offsite power, the Diesel Generators supply motive power to the SW. The control powers for SW-A and B are from Division 1 and 2 DC, respectively. The control power for SW-C is from Division 3 AC. The SW-A, B, and C in turn supply cooling to the Diesel Generators 1, 2, and 3, respectively. During normal shutdown or emergency condition, water is taken from the 2 spray ponds, routed to the heat exchangers and equipment that require cooling, and returned to the ponds where the water is cooled by the spray system prior to recycle. Standby Service Water pumphouses are heated by electric blast heaters. They are cooled by either a fan which draws in outside air, or when the associated Standby Service Water pump is running, by an air handling unit which is cooled by the SW. Each of the SW loops will auto initiate when the corresponding RHR pump, LPCS pump, RCIC turbine steam stop valve or Diesel Generator starts (or opens). Therefore, the SW-A, B, and C are interdependent with the Division 1, 2, and 3 AC, respectively. The SW-A and B are dependent on the Division 1 and 2 DC, respectively. The SW is partially dependent on the instrumentation.

- RCC

The Reactor Closed Cooling Water system removes heat from potentially contaminated systems in the reactor building, primary containment, and radwaste building. The RCC system consists of 3 parallel strings of pumps and heat exchangers, with two strings in-service at one time. The TSW cools the RCC heat exchangers. The motive power for the RCC is from the Division 1 or 2 AC. The control power for the RCC is from the Division 1 or 2 DC. There are 2 primary containment supply outboard isolation valves which are dependent on Division 1 AC. There is a primary containment return inboard isolation valve which is dependent on the Division 2 AC. There is also a primary containment return outboard isolation valve which is dependent on the Division 1 AC. Closing of any of the isolation valves will stop the RCC from cooling the drywell chillers. Therefore, the RCC is dependent on the Division 1 AC, the Division 2 AC, the Division 1 DC, the Division 2 DC, and the TSW.

- INSTRUMENTATION

Auto initiation of safety systems requires instrumentation. Instrumentation is powered by either AC or DC. In an emergency, room cooling for the motor control centers may be required utilizing the SW. Therefore, instrumentation is dependent on Division 1 AC, Division 2 AC, Division 1 DC, Division 2 DC, Division 3 DC, and is partially dependent on the reactor building Emergency Cooling and the SW.

- CAS (CONTROL AIR ONLY)

Four air compressors are arranged to operate in parallel or independently to supply a single, or any combination of 3 air receivers. Air from the receiver tanks flows through two of four prefilters, two of four air dryer towers and two of four after filters. Two air compressors are normally running to carry the full plant load. The other air compressors will come on when the running compressors fail. Motive and control power for the compressors are:

Compressor 1 -- Division 1 AC, MC-7A/5B
Compressor 2 -- Division 2 AC, MC-8A/4B
Compressor 3 -- Division B (offsite) AC, MC-2P/6C
Compressor 4 -- MCC-6C (offsite) AC, MC-6C/7A

Air dryers are powered by the Division 1 and 2 AC. There are no motor operated valves. Cooling for the compressors is by an intermediate Compressor Jacket Water system (CJW) which is cooled by the TSW. CJWP-1A is powered off PP-7E-A-10 (Division A), and CJWP-1B is powered off MC-8A/3D (Division 2). Cooling can also be performed with FP water by manual connection. Therefore, the control air is dependent on the TSW, and is partially dependent on the offsite power, the Division 1 AC, and the Division 2 AC.

- CIA

The CIA supplies nitrogen to the relief mode accumulators of the 18 SRVs and the 4 in-board MSIVs from the nonsafety related cryogenic nitrogen storage tank. In the event the cryogenic nitrogen is unavailable, two independent safety related nitrogen bottle bank subsystems can deliver pressurized nitrogen to the 7 ADS valves and accumulators. There are air-operated isolation valves which will shut when the cryogenic supply pressure falls below 140 psig. There is a programmer which will provide signals to sequentially open the nitrogen bottles. A remote nitrogen cylinder connection is provided to each subsystem to permit supplementing the cylinder banks through manual connection of portable nitrogen cylinders, and thus maintaining pressure for an indefinite time. For opening MSIVs and relief function of SRVs, CAS can be manually crosstied to feed CIA. Compressors, air dryers, programmer, and valves are powered by offsite power, or Division 1 and 2 AC. The TSW supplies cooling water to the intercoolers, aftercoolers and the CJWS which cools the compressors. Cooling can also be accomplished by FP water. Therefore, the CIA is partially dependent on the offsite power, the Division 1 AC, the Division 2 AC, the TSW, and the CAS.

- REACTOR BUILDING EMERGENCY COOLING

Whenever any of the F, A, Z signals are received, ventilation supply air to the ECCS (and RCIC) pump rooms and the critical MCC rooms is secured by the automatic closing of ROA and REA air dampers. The reactor building Emergency Cooling system automatically starts. Its fan coil units for the different divisions are cooled by the corresponding SW loops. Power supplies for the different divisions are from the corresponding AC divisions. Therefore, different divisions of reactor building Emergency Cooling are dependent on the corresponding AC division and the SW, and are partially dependent on the instrumentation.

Justification For The Dependency Matrix in Table 3.2.3.3

- LOSS OF OFFSITE POWER

Loss of offsite power will have a partial effect on all of the support systems indicated in the matrix. As long as the onsite power is available, all support systems are still available.

- LOCAs

LOCAs will cause low reactor water level, high drywell pressure, or both. The RCC will be unavailable because of FA load shed.

- LOSS OF DIVISION 2 DC

Since the offsite power and Division 2 AC, load transfer, load shedding, switchgear and bus operations are dependent on the Division 2 DC, the offsite power and Division 2 AC will be partially effected. The control power for the TSW and the reactor building Emergency Cooling is from the DC. Therefore, the TSW and the reactor building Emergency Cooling will be partially effected. The RCC has primary containment isolation valves which are dependent on the Division 2 DC for control. Loss of the Division 2 DC could isolate the RCC from cooling the drywell chillers. Motor operated valves fail 'As Is' on loss of AC or DC motive power.

- LOSS OF SW

The SW provides cooling to the Diesel Generators, the ECCS pump rooms, AC MCC, Division 2 DC MCC, Division 1, 2, and 3 DC battery and battery charger rooms.

- REACTOR WATER LEVEL MEASUREMENT LINE BREAK

Reactor water level measurement line break will cause high drywell pressure. The RCC will be unavailable because of FA load shed. Reactor water level indication will be erroneous.

→ DEPENDENT

↓ INDEPENDENT

FRONTLINE SYSTEMS / SUPPORT SYSTEMS		RODS	ARI	RPT AT-WS	RPT EOC	SLC	ADS I	ADS II	SRV	MSIV IN	MSIV OUT	CONDENSER	RFW	HPCS	RCIC	LPCS	LPCI A	LPCI B	LPCI C	CONDENSATE	FIREWATER	DW CHILLERS A	DW CHILLERS B	SHUT-DOWN COOLING A	SHUT-DOWN COOLING B	POOL COOLING A	POOL COOLING B	DW SPRAY A	DW SPRAY B	WW SPRAY A	WW SPRAY B	HD SPRAY FLOOD	CONTAINMENT VENT
AC	SM. 1,2,3											D	D							D	P												
AC	SM. 7					D				P	P	P	P			D	D			P		D		D	D	D		D		D			D
AC	SM. 8					D				P	P	P	P		P			D	D	P			D		D		D		D		D	D	D
AC	SM. 4													D																			
DC	DIV 1		D	P	P		D		P			P	P		D	D	D			P				D	D	D		D		D		D	
DC	DIV 2		D	P	P			D				P	P					D	D	P					D		D		D		D	D	
DC	DIV 3													D																			
TSW										P	P	D	D							D		D	D										P
SW	SW A															D	D							D		D		D		D			
SW	SW B														P			D	D						D		D		D		D	D	
SW	HPCS SW C													D																			
RCC																						D	D										
INSTRUMENT		P	P	P	P		P	P		P	P			P	P	P	P	P	P														
CAS									P	P	D	D	D							D													D
CIA							D	D	P	D		D																					
Rx BLDG EMERGENCY COOLING														D	P	D	D	D	D					D	D	D	D	D	D	D	D	D	

I = INTERDEPENDENCE
D = COMPLETE DEPENDENCE
P = PARTIAL OR DELAYED DEPENDENCE

900065.1
MAY 1994

Table 3.2.3-1 Matrix of Frontline System Dependencies on Support Systems

→ DEPENDENT

↓ INDEPENDENT

SUPPORT SYSTEMS (independent) \ SUPPORT SYSTEMS (dependent)		AC				DC			TSW	SW			RCC	INSTRUMENT	CAS	CIA	Rx BLDG EMERG. COOLING
		SM. 1, 2,3	SM. 7	SM. 8	SM. 4	DIV 1	DIV 2	DIV 3		SWA	SWB	SWC					
AC	SM. 1,2,3	I													P	P	
	SM. 7		I			P			P	I			D	D	P	P	D
	SM. 8			I			P		P		I		D	D	P	P	D
	SM. 4				I			P				I		D			D
DC	DIV 1	P	P			I			P	D			D	D			
	DIV 2	P		P			I		P		D		D	D			
	DIV 3	P			P			I						D			
TSW			P	P		P	P		I				D		P	P	
SW	SW A		I			P				I				P			D
	SW B			I			P				I			P			D
	HPCS SW C				I			P				I		P			D
RCC													I				
INSTRUMENT									P	P	P	P		I			P
CAS															I	P	
CIA																I	
Rx BLDG EMERGENCY COOLING			P	P	P	P	P	P						P			I

I = INTERDEPENDENCE
D = COMPLETE DEPENDENCE
P = PARTIAL OR DELAYED DEPENDENCE

910084.8
MAY 1994

Table 3.2.3-2 Matrix of Support System Dependencies on Other Support Systems

→ DEPENDENT

↓ INDEPENDENT

SUPPORT SYSTEMS / INITIATORS	AC				DC			TSW	SW			RCC	INSTRUMENT	CAS	CIA	Rx BLDG EMER. COOLING
	SM. 1, 2,3	SM. 7	SM. 8	SM. 4	DIV 1	DIV 2	DIV 3		SWA	SWB	SWC					
TURBINE TRIP																
MSIV CLOSURE																
LOSS OF CONDENSER																
LOSS OF FEEDWATER																
LOSS OF OFFSITE POWER	D	P	P	P	P	P	P	P	P	P	P	D	P	P	P	P
IORV AND SORV.																
MANUAL SHUTDOWN																
LARGE LOCA												D				
MEDIUM LOCA												D				
SMALL LOCA												D				
LARGE LOCA OUTSIDE CONTAINMENT	DEPENDS ON LOCA LOCATION & RESPONSE TIME															
ATWS																
LOSS OF DIV. 2 DC			P			D		P		D		D	P	P		P
LOSS OF SW		P	P	P	P	P	P		D	D	D		P			D
LOSS OF CN															P	
RX WATER LEVEL MEASUREMENT LINE BREAK												D	D			
RX BLDG FLOOD	DEPENDS ON FLOOD LOCATION & RESPONSE TIME															
Loss of TSW								D				D		P		

D = COMPLETE LOSS
P = PARTIAL OR DELAYED LOSS

910084.9
May 1994

Table 3.2.3-3 Matrix of Support System Losses Due to Initiators

→ DEPENDENT

↓ INDEPENDENT

		RODS	ARI	RPT		SLC		ADS		SRV	MSIV		CONDENSER	RFW	HPCS	RCIC	LPCS	LPCI			COND	FIREWATER	DW CHILLERS		SHUTDOWN COOLING		SUPP. POOL COOLING		DW SPRAY		WW SPRAY		FLOOD	VENT
				ATWS	EOC	A	B	I	II		IN	OUT						A	B	C			A	B	A	B	A	B	A	B	A	B		
RODS		I																																
ARI			I	P																														
RPT	ATWS		P	I																														
	EOC				I																													
SLC							I																											
ADS	I							I									P	P	P	P	P	P												
	II								I								P	P	P	P	P	P												
SRV										I						P																		
MSIV	IN										I		D	D																				
	OUT											I	D	D																				
CONDENSER											D	D	I	D							P													
RFW														I							P	P												
HPCS															I																			
RCIC																I																		
LPCS								P	P								I																	
LPCI	A							P	P									I					P		P		P		P		P			
	B							P	P										I					P		P		P		P		P	P	
	C							P	P											I														
CONDENSATE											D	D	D	D							I													
FIREWATER																						I												
DW CHILLERS	A														P	P	P	P	P	P			I											
	B														P	P	P	P	P	P				I										
SHUTDOWN COOLING	A																								I		P		P		P			
	B																									I		P		P		P	P	
SUPP. POOL COOLING	A														P	P	P	P	P	P					P		I		P		P			
	B														P	P	P	P	P	P						P		I		P		P	P	
DW SPRAYS	A														P	P	P	P	P	P			P	P	P		P		I		P			
	B														P	P	P	P	P	P						P		P		I		P	P	
WW SPRAYS	A														P	P	P	P	P	P					P		P		P		I			
	B														P	P	P	P	P	P						P		P		P		I	P	
FLOOD																										P		P		P		P	I	P
VENT															P	P	P	P	P	P													P	I

I = INTERDEPENDENCE
D = COMPLETE DEPENDENCE
P = PARTIAL OR DELAYED DEPENDENCE

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Table 3.2.3-4 Matrix of Frontline System Dependence on Other Frontline Systems

3.3 Sequence Quantification

The accident sequence quantification section provides the quantification of the event trees presented in Section 3.1. Sections 3.3.1 and 3.3.2 describe the data used for component unavailabilities and Sections 3.3.3 and 3.3.4 discuss the methodology used to quantify human reliability and common cause failure, respectively. Solutions for the functional equations of each event tree branch point or heading is given in Section 3.3.5 and the individual sequence solutions are presented in Section 3.3.7. The WNP-2 IPE does not utilize support system state methodology, therefore, Section 3.3.6 is not applicable to this report.

The WNP-2 IPE utilized plant specific data to the extent it was available on NPRDS. If plant specific data was insufficient, industry specific NPRDS data was used. If industry data was unavailable, then generic, published unavailabilities were used. The human reliability analysis utilized plant specific procedural steps in evaluating errors during recovery actions. Common cause failure analysis was performed by the beta factor, or multiple Greek letter if appropriate, methodology. The quantification was then performed by linking of the fault trees, solving each event tree heading for branch point values, and then numerically solving each event tree sequence for a minimum cutset solution. A truncation value of 1E-10 was used for all sequence evaluations.

3.3.1 List of Generic Data

In performing the Individual Plant Examination, the Supply System made use of WNP-2 specific data to the maximum extent possible. For situations where plant specific data were not available, generic data were used; for example, highly improbable initiator events which have not occurred at WNP-2.

As discussed in Section 3.1.1, the following generic initiator frequencies were used in this report:

Initiator	Frequency, per year
Large LOCA	3×10^{-4}
Medium LOCA	3×10^{-3}
Small LOCA	8×10^{-3}
Loss of Division 2 DC	3×10^{-3}
Loss of TSW	1.25×10^{-3}
Loss of CIA	1.25×10^{-3}
Instrument Line Break	1×10^{-2}
Loss of CAS	1.25×10^{-3}
Loss of CN	1.25×10^{-3}

For generic component failure rates, data from INPO's Nuclear Plant Reliability Data System (NPRDS) were used in this report. For the following component failure rates which were not found in the NPRDS, data from NUREG/CR-2815, NUREG-0460 or WASH-1400 were used:

COMPONENT FAILURE	FAILURE RATE	SOURCE	COMMENTS
Cooling Coil Leakage	3×10^{-9}/hr	NUREG/CR-2815	Heat Exchanger Tube Leakage Used
DG fails to start	6×10^{-5}/hr	NUREG/CR-2815	
DG fails to run	3×10^{-3}/hr	NUREG/CR-2815	
Electric Heater fails	1×10^{-5}/hr	NUREG/CR-2815	Wire Open Circuit Used

COMPONENT FAILURE	FAILURE RATE	SOURCE	COMMENTS
Strainer Plugged	3×10^{-5}/hr	NUREG/CR-2815	
Fuse Open Circuit	3×10^{-6}/hr	NUREG/CR-2815	
Limit Switch fails	6×10^{-6}/hr	NUREG/CR-2815	
MCC, PP or Switch Gear fails	3×10^{-8}/hr	NUREG/CR-2815	Bus Failure Used
Diesel-driven Pump fails to start	1×10^{-6}/hr	NUREG/CR-2815	
Diesel-drive Pump fails to run	1×10^{-6}/hr	NUREG/CR-2815	
Pipe ruptures	8.59×10^{-10}/hr	WASH-1400	Per Pipe Section
Rupture Disk fails	3×10^{-8}/hr	NUREG/CR-2815	Orifice Rupture Used
Transformer fails	6×10^{-7}/hr	NUREG/CR-2815	
Tube ruptures	8.59×10^{-8}/hr	NUREG/CR-2815	Assumed 100 Times Pipe Rupture Rate
Manual Valve fails to operate	2×10^{-7}/hr	NUREG/CR-2815	
Motor Operated Valve fails to remain open or closed	2×10^{-7}/hr	NUREG/CR-2815	
Wire short to power	3×10^{-8}/hr	NUREG/CR-2815	
Wire open circuit	1×10^{-5}/hr	NUREG/CR-2815	
Wire short to ground	1×10^{-6}/hr	NUREG/CR-2815	
Mechanical failure to scram	1×10^{-5}/demand	NUREG-0460	
Electrical failure to scram	2×10^{-5}/demand	NUREG-0460	

For components which have not failed at WNP-2, the following NPRDS industry failure rates were used in this report:

Component Failure	Failure Rate/Hr
Compressor fails	8.48×10^{-6}
Chiller (cooler + fan) fails	8.59×10^{-6}
Flow Sensor fails	9.58×10^{-7}
Fan fails	2.61×10^{-6}
Level Sensor fails	3.64×10^{-6}
Motor Generator Set fails	6.78×10^{-6}
Pressure Sensor fails	1.81×10^{-6}
Temperature Transmitter fails	1.78×10^{-6}

3.3.2 Plant Specific Data and Analysis

As discussed in Section 3.1.1, WNP-2 specific transient initiator frequencies were used in this report for components with sufficient NPRDS data. See Table 3.1.1-3. NPRDS is the INPO data base system that has the component engineering data, component failure/repair data, and plant operating data. Data are being reported regularly by utilities nationwide to INPO. Considerable improvements have been made in recent years to the data base as far as data accessibility and quality. Besides utilities, organizations like EPRI, INPO, NRC and DOE are also using the data base for various purposes. The Supply System has been reporting WNP-2 specific data to INPO for 30 safety and safety-related systems containing approximately 4,300 components for the last ten years.

NPRDS component failure/repair data are grouped under component names or application names. A component name is a term for identifying a certain kind of hardware, e.g., pump, valve, transformer, etc. An application name is a term for identifying a specific application of a hardware, e.g., condensate booster pump, feedwater containment isolation valve, startup transformer, etc.

NPRDS offers the Component Failure Analysis Report (CFAR) feature. The standard report of the CFAR feature has 3 comparison options. The first option compares the failure rates of WNP-2 components to the average failure rates of similar components industry-wide. The second option compares the failure rates of component applications. The third option combines options one and two. The time period used for failure rate comparisons is 18 months dating back from the time of the last comparison data update. The INPO staff updates the comparison data on a monthly basis. The 18-month period is used to provide relatively recent data trends and at the same time, provide enough data to have statistically valid comparisons.

Before the NPRDS data were used for the WNP-2 IPE, in-house checking of the NPRDS data was performed. Raw failure data at the Plant are contained in the Maintenance Work Requests. Information on the Maintenance Work Requests are reported to INPO in the form of Failure Master Reports. INPO takes the information from the Failure Master Report and calculates WNP-2 component failure rates. Those component failure rates are retrievable from the INPO computer by IPE analysts using the CFAR option. Checking was done to verify that data fidelity was maintained at every step. Because of the vast amount of components reported over many years, checking was done for five representative components (check valves, motor operated pumps, MOP motors, motor operated valves and MOV operators) and over two time periods (from November 1, 1987 to April 30, 1989 and from January 1, 1989 to June 30, 1990).

Results of the comparison check are shown in Table 3.3.2-1. It was concluded that data fidelity was maintained at every step for both time periods. WNP-2 component failure rates (shown in Table 3.3.2-1) calculated by INPO are verified to be the same as those by manual calculations done in-house. Although the industry-wide failure rates have a broader data base than the WNP-2 failure rates, Table 3.3.2-1 indicates that both failures rates are of the same order of magnitude for the same components. This suggests that the 18-month period used by INPO does provide enough data to have statistically valid failure rates. It also suggests that the component failure rates vary little with time.

For this report, the component failure rates in Table 3.3.2-2 from the WNP-2 CFAR standard report were used.

All generic and plant specific component failure rates are stored in the NUPRA program as a parameter file. Failure rates in the NUPRA parameter file are used by the NUPRA program to calculate the unavailabilities of the component basic faults in the fault trees. There are five types of unavailabilities which are specific for each of the component basic faults during fault tree construction. The five types of unavailabilities are discussed below:

Type 1 - Unavailability on demand of a monitored, on-line component whose failure is detected and repaired during operation.

Type 2 - Unavailability on demand of a component which is tested periodically.

Type 3 - Unavailability of a component on demand. Unavailabilities due to testing, preventive maintenance, repair, and human error are placed under this type.

Type 4 - Unavailability of a component during running.

Type 5 - Module: This is a pseudo basic event representing the top event unavailability of a subset of a fault tree. The basic events that occur in the module do not occur in any fault tree or module.

The Supply System has two computer data bases pertinent to component unavailabilities, SMS and TSS. The SMS data base contains the historical record of man-hours spent on every component in the plant during scheduled preventive maintenances. The TSS data base contains the historical record of man-hours spent on every component in the plant during Technical Specification surveillances. The man-hours spent during the last 2 or 3 preventive maintenances and surveillances were averaged and were used in calculations of maintenance and testing unavailabilities for system fault trees. Corrective maintenance is reported as part of the NPRDS data base.

The Component Failure Master feature of the NPRDS Retrieval Subsystem provides detailed information on component failures including out-of-service hours. For each system fault tree, the Failure Master Reports since December 1984 are obtained for all components associated with the system. The out-of-service times were summed and the repair unavailabilities calculated for all system components.

TABLE 3.3.2-1

Failure Rates of Supply System vs. Industry Average

Failure Rate (hr^{-1})	Supply System			Industry Average		
	Time Period*		Failure #	Time Period		Failure #
Check Valves	(I)	4.01 E-6	4	(I)	3.51 E-6	700
	(II)	5.02 E-6	5	(II)	3.14 E-6	657
Motor Operated Pumps (MOP)	(I)	1.48 E-5	7	(I)	1.42 E-5	721
	(II)	1.70 E-5	8	(II)	1.44 E-5	759
MOP Motors	(I)	3.51 E-6	4	(I)	2.57 E-6	88
	(II)	4.42 E-6	4	(II)	1.84 E-6	93
Motor Operated Valves	(I)	2.66 E-6	20	(I)	3.33 E-6	3044
	(II)	2.80 E-6	21	(II)	2.78 E-6	2650
MOV Operators	(I)	5.55 E-6	14	(I)	4.52 E-6	876
	(II)	4.75 E-6	12	(II)	4.02 E-6	816

* Time period (I) = Nov 1, 1987 to April 30, 1989 (18 month span)
Time period (II) = Jan 1, 1989 to June 30, 1990 (18 month span)

TABLE 3.3.2-2

Plant Specific Failure Rate Data for WNP-2

Component	Failure Rate/Hr
Damper	6.94×10^{-6}
Air/Fluid Operated Valve	4.49×10^{-6}
Battery	1.09×10^{-5}
Battery Charger	9.54×10^{-6}
Circuit Breaker	2.57×10^{-6}
Flow Transmitter	1.59×10^{-6}
Heat Exchanger	2.19×10^{-6}
Inverter	5.72×10^{-6}
Motor driven Pump	1.18×10^{-5}
Relief Valve	3.32×10^{-6}
Safety Relief Valve	4.24×10^{-6}
Temperature Sensor	8.85×10^{-7}
Check Valve	6.94×10^{-6}
Motor operated Valve	7.92×10^{-6}
Vacuum Breaker	3.32×10^{-6}
Level Transmitter	1.52×10^{-5}
Turbine Driven Pump	4.57×10^{-5}
Pressure Transmitter	5.96×10^{-6}
Relay	1.27×10^{-6}
Remote Manual Switch	4.50×10^{-7}
Solenoid-Operated Valve	7.31×10^{-6}

3.3.3 Human Error Analysis and Probabilities

This IPE considers human interactions in determining the plant responses to accident sequences. The methodology used in WNP-2 IPE integrates human reliability analysis with plant's hardware response analysis. Dependencies that exist between human interactions, and between human interactions and hardware unavailabilities in accident sequences are taken into account.

The methodology consists of four stages. In stage 1, plant logic models (event trees and fault trees) are developed. Basic events representing human interactions (calibration, preventive maintenance, procedure-driven human actions, recoveries, etc.) are defined in the logic models. In stage 2, modelling and quantification of the probabilities of human interactions are performed. In stage 3, recovery actions which are not procedure-driven are quantified. In stage 4, the products from the first three stages are reviewed.

3.3.3.1 Stage 1: Development of Plant Logic Models

Event trees and system fault trees are developed and documented in Section 3.1.2 and system notebooks, respectively. Basic events representing human interactions are defined in system fault trees, but not in event trees. Some important human interactions that are quantified in stage 2 are listed in the following:

- Use RFW to control reactor level following turbine trip
- Manual initiation of HPCS if AUTO fails
- Manual initiation of RCIC if AUTO fails
- Depressurize the reactor using ADS
- Manual initiation of LPCS if AUTO fails
- Manual initiation of LPCI if AUTO fails
- Crosstie SW to RHR-B
- Initiate Suppression Pool cooling/spray after accident
- Open MSIVs to recover Power Conversion System
- Perform containment venting
- Provide alternate room cooling to RCIC
- Initiate RPT manually
- Initiate ARI manually
- Inhibit ADS and lower reactor water level to control power during ATWS
- Initiate SLC during ATWS

3.3.3.2 Stage 2: Quantification of Human Error Probabilities

The ASEP HRA Procedure (Accident Sequence Evaluation Program Human Reliability Analysis Procedure) has been used to quantify human actions for WNP-2 IPE [1, 2]. The kinds of actions analyzed include human errors before the onset of an accident (pre-accident errors), and errors and recovery actions following the start of an accident (post-accident errors).

The pre-accident errors are caused by failure of test and maintenance personnel to restore components to operation after maintenance or miscalibration of multiple sensors. These could render system or component unavailabilities.

The post-accident errors are caused by actions performed by operations personnel after annunciation of some abnormal event has occurred. These actions include failures in manually initiating a system, aligning and actuating a system for injection, switching the system from injection to recirculation, and recovering a failed system.

3.3.3.2.1 Pre-Accident Human Reliability Development

A simple approach has been taken to quantify the pre-accident human errors. Similar to the method used in NUREG-1150 [3], a nominal failure probability of 0.03 was assigned as basic HEP (Human Error Probability or BHEP) for all pre-accident failures. This BHEP represents a combination of a generic HEP of 0.02 assessed for an error of omission (EOM) and a generic HEP of 0.01 assessed for an error of commission (ECOM). It is conservatively assumed that an ECOM is always possible if an EOM does not occur.

The BHEP of 0.03 need to be modified for the effects of dependence and recovery factors (RFs). Whether two human interactions are completely dependent, highly dependent, or independent is based on the following factors:

components upon which human actions take place are in series (not in parallel),

human actions on components in series are performed within two minutes of each other,

components upon which human actions take place are within four feet of each other,

written requirements for each human action performed.

For WNP-2 IPE, zero dependency is assumed for components upon which human actions performed are in series. Also, zero dependency is used because pre-accident tasks are generally more than two minutes from each other. However, to be conservative, if different tasks are performed by the same individual using the same plant procedure, complete human error dependency is modelled in the fault trees.

Recovery factor is defined as a factor that prevents or limits the undesirable consequences of a human error. The RFs taken into account include:

Error recovery by post-maintenance or post-calibration test,

Error recovery by a second person verifying the task or by the original person checking the task at a different time and place. A written verification (or checkoff list) must be used during the check otherwise no recovery credit will be given for either check.

Other factors such as compelling signals, and shiftly or daily check with checkoff list are not considered.

Each of the factors reduces the nominal value by a factor of 10 [3]. Therefore, the total failure probability (F_T) is

$$F_T = 0.03 \times 0.1 \times 0.1 = 3.0E\text{-}4$$

The pre-accident errors considered in the system fault trees are listed in Table 3.3.3-1.

3.3.3.2.2 <u>Post-Accident Human Reliability Development</u>

<u>Manual Back-up of Automatic Actions</u>

This category consists of manual initiation of a system (HPCS, RCIC, LPCS, LPCI, ARI, or fan coil unit) if auto initiation fails, and depressurization the reactor using ADS. Table 8-5 of Reference 2 lists the HEP and EF (Error Factor). HEP is the probability that an error will occur when a given task is performed. EF is defined as the square root of the ratio of the upper to the lower uncertainty bound. The listed HEP is 0.001 and EF is 10 for operators performing a post-diagnosis immediate emergency action for the reactor vessel/containment critical parameters. These critical parameters are reactor power level, water level in the core, reactor pressure, and containment temperature and pressure.

Assuming 0.001 is a medium probability and applying an error factor of 10, the mean HEP is 2.66E-3.

$$EF=10=\sqrt{\frac{X_{95}}{X_{05}}}=e^{1.645\sigma}$$

$$\sigma=1.3997$$

$$HEP_{mean}=HEP_{medium}\ e^{\frac{\sigma^2}{2}}=2.66E-3$$

In Reference 4, this type of actions is quantified as the probability of failure to execute the required response or slip probability. Based on simulator data collected by General Physics, the probability range is between 0.01 and 0.03. Due to the improved control boards, more training, and more simulator time in the nuclear power plants, a reduction in these value by a factor of five (i.e., 0.002 - 0.006) is recommended. Therefore, the estimated HEP of 2.66E-3 based on ASEP HRA is in good agreement with the industry data. Table 3.3.3-2 lists the errors considered in the system fault trees.

Human Errors in Using Symptom-oriented Emergency Operating Procedures (EOPs)

Two separate time-dependent probabilities are estimated: the probability of performing the correct diagnosis within its allowable time, and the probability of performing the correct post-diagnosis actions (steps) within their allowable time. The ground rules taken from the ASEP HRA procedure for quantifying post accident human interactions are listed below:

If there is a requirement to use written procedures, assess a 5-minute delay, after correct diagnosis, before the first of the required post-diagnosis action will be initiated.

Assess one minute as the required travel and manipulation time combined for each control room control action taken on the primary operating panels.

For required control actions on other than the primary control room operating panels, assess two minutes as the required travel and manipulation time for each control action.

If some safety related system fails after the operating crew is using the Emergency Operating Procedure, reclassify as dynamic any step-by-step tasks related to the use of the EOP.

If EOPs are not well designed, operators are not EOP-trained, or operators do not actually use EOPs, post-diagnosis actions related to EOPs are classified as dynamic.

If an individual operator must perform more than one task simultaneously with good cues for when he must shift from one task to another, each task is classified as dynamic.

At least a moderately high level of stress is assessed for a minimum of two hours after the initiation of an abnormal event.

The occasion of a large LOCA is assessed as resulting in extremely high stress until such time as recirculation is established.

Extremely high stress is assessed for occasions in which more than two primary safety system fails to function.

Regarding action error probability, it is conservatively assumed that there is a novice person backed by a more experienced person in performing the action to fulfil the safety function. Tables 3.3.3-3 and 3.3.3-4 [1] are used for assessing diagnosis error probability (HEP_D) and action error probability (HEP_A), respectively.

The steps involved in any task are taken from WNP-2 plant procedures. The time required for each step has been estimated by a shift manager. In most cases, a 5-minute delay time is assumed before initiation of the first step. The total time allowable for operator to take action to prevent core damage or containment failure is based on plant specific MAAP calculation results.

In Table 3.3.3-4, the human behavior can be classed as step-by-step if the operator has a clearly understood set of rules to follow in responding to a well-understood transient or situation. The behavior is classed as dynamic if it is applied to unfamiliar situations in which personnel have to interpret, diagnose, or use some level of decision making.

Quantification of this type of human errors has been documented in detail in Reference 5. The results are listed in Table 3.3.3-5.

3.3.3.3 Stage 3: Quantification of Non-Procedure-Driven Recovery Actions

WNP-2 has plant procedures for RPV flooding and containment flooding. During a severe accident, the RPV flooding procedure can be utilized to arrest core melt progression; and the containment flooding, to cool corium in the pedestal. Systems not available at the beginning of an accident can be made available later in the accident by recovery. System recovery rates credited in the Level 2 analyses are discussed in Section 4.0.2.

3.3.3.4 Stage 4: Review of the Human Error Analysis

Review of the products from the first three stages is discussed in Section 5.0.

3.3.3.5 References

1. Ericson, D.M. Jr., et. al., "Analysis of Core Damage Frequency: Internal Events Methodology. ," NUREG/CR-4550, Vol. 1, Rev. 1, January 1990.

2. Swain, A.D., "Accident Sequence Evaluation Program Human Reliability Analysis Procedure," NUREG/CR-4772, February 1987.

3. U.S. NRC, " Severe Accident Risks: An Assessment Five U.S. Nuclear Power Plants," NUREG-1150, Vol. 2, October 1990.

4. Parry, G.W., et. al., "An Approach to the Analysis of Operator Actions in Probabilistic Risk Assessment," EPRI TR-100259, June 1992.

5. Pong, L.T., "Human Reliability Analysis for WNP-2 IPE," Calculation No. NE-02-93-42, Washington Public Power Supply System, January 1994.

TABLE 3.3.3-1

Pre-Accident Human Errors

Basic Event ID	Point Est	Description
CIAHUMNBUTAKX3XX	3.E-04	EO FAILS TO REPLACE SPECIFIC BOTTLE WHEN REQUIRED
CIAHUMNBUTBKX3XX	3.E-04	EO FAILS TO REPLACE SPECIFIC BOTTLE WHEN REQUIRED
CIAHUMNTK20AX3XX	3.E-04	EO FAILS TO REPLACE BOTTLE (CIA-TK-20A) WHEN REQUIRED
CIAHUMNTK20BX3XX	3.E-04	EO FAILS TO REPLACE BOTTLE (CIA-TK-20B) WHEN REQUIRED
CIAHUMNV--20J3XX	3.E-04	TM ERROR ON CIA-V-20
CIAHUMNV-30AJ3XX	3.E-04	TM ERROR ON CIA-V-30A
CIAHUMNV-30BJ3XX	3.E-04	TM ERROR ON CIA-V-30B
CN-HUMNTK--1X3XX	3.E-04	PERSONNEL FAIL TO REORDER LIQUID NITROGEN WHEN REQUIRED
HPSHUMNLS1-3M3LL	3.E-04	HPCS-LS-1A/1B/3A/3B MISCALIBRATION OF CST LEVEL SENSOR
HPSHUMNLS2ABM3LL	3.E-04	HPCS-LS-2A AND -B MISCALIBRATION OF SP LEVEL SENSOR
HPSHUMNPM---X3LL	3.E-04	HUMAN ERROR DURING PREVENTIVE MAINTENANCE
HPSHUMNPVOPEX3LL	3.E-04	OPERATOR ERROR, PUMP & VALVE OPERABILITY TEST
HPSHUMNSYSTMJ3LL	3.E-04	OPERATOR ERROR REPAIR MAINTENANCE, TESTED COMPONENT
LD-HUMN-603AM3LL	3.E-04	LD-TS-603A HUMAN ERROR PPM 7.4.3.2.4
LD-HUMN-603BM3LL	3.E-04	LD-TS-603B HUMAN ERROR PPM 7.4.3.2.66
LPSHUMN---ICX3LL	3.E-04	TEST & MAINTENANCE ERROR, INSTRUMENT AND CONTROL
LPSHUMN--V-5X3LL	3.E-04	TEST & MAINTENANCE ERROR FOR LPCS-V-5
LPSHUMN--V-6X3LL	3.E-04	TEST & MAINTENANCE ERROR FOR LPCS-V-6
LPSHUMNFIS-4M3LL	3.E-04	LPCS-FIS-4 FLOW INDICATING SENSOR MISCALIBRATED
LPSHUMNPVOPEX3LL	3.E-04	TEST & MAINTENANCE ERROR FOR PUMP & VALVE OPERABILITY
LPSHUMNSYSTMJ3LL	3.E-04	SYSTEM REPAIR ERROR
MS-HUMN-100AM3LL	3.E-04	MS-LIS-100A MISCALIBRATION
MS-HUMN-100BM3LL	3.E-04	MS-LIS-100B MISCALIBRATION
MS-HUMN-BDPSM3LL	3.E-04	PRESSURE SENSOR MS-PS-48B & 48D TESTING
MS-HUMNBDLISM3LL	3.E-04	LEVEL SENSOR MS-LIS-37B & 37D TESTING
MS-HUMNLE24BM3LL	3.E-04	MS-LIS-24B & 24D HUMAN ERROR PPM 7.4.3.1.1.50
MS-HUMNLE31AM3LL	3.E-04	MS-LIS-31A & -31C MISCALIBRATION PP,7.4.3.3.1.51
MS-HUMNLE31BM3LL	3.E-04	MS-LIS-31B & -31D MISCALIBRATION PPM7.4.3.3.1.52
MS-HUMNLE36AM3LL	3.E-04	MS-LIS-36A & -36C CALIBRATION ERROR
MS-HUMNLE36BM3LL	3.E-04	MS-LIS-36B & -36D CALIBRATION ERROR
MS-HUMNLE37AM3LL	3.E-04	MS-LIS-37A & -37C MISCALIBRATION
MS-HUMNLE37BM3LL	3.E-04	MS-LIS-37B & 37D MISCALIBRATION
MS-HUMNLS37AM3LL	3.E-04	MS-LIS-37A & 37C HUMAN ERROR PPM 7.4.3.3.3.4

Basic Event ID	Point Est	Description
MS-HUMNLS37BM3LL	3.E-04	MS-LIS-37B & 37D HUMAN ERROR PPM 7.4.3.3.8
MS-HUMNP413CM3LL	3.E-04	MS-PS-413C MISCALIBRATION
MS-HUMNPE47AM3LL	3.E-04	MS-PS-47A & -47C MISCALIBRATION PPM7.4.3.3.1.53
MS-HUMNPE47BM3LL	3.E-04	MS-PS-47B & -47D MISCALIBRATION PPM7.4.3.3.1.54
MS-HUMNPE48AM3LL	3.E-04	MS-PS-48A & -48C MISCALIBRATION
MS-HUMNPE48BM3LL	3.E-04	PRESSURE SENSOR MS-PS-48B & 48D TESTING
MS-HUMNPS45AM3LL	3.E-04	MS-PS-45A & -45C CALIBRATION ERROR
MS-HUMNPS45BM3LL	3.E-04	MS-PS-45B & -45D CALIBRATION ERROR
RCIHUMN—P1X3LL	3.E-04	HUMAN ERROR DURING PUMP OIL CHANGE
RCIHUMNLS15AM3LL	3.E-04	RCIC-LS-15A HUMAN ERROR PPM 7.4.3.5.1.6
RCIHUMNLS15BM3LL	3.E-04	RCIC-LS-15B HUMAN ERROR PPM 7.4.3.5.1.6
RCIHUMNPS--6M3LL	3.E-04	RCIC-PS-6 MISCALIBRATION
RCIHUMNPS-9AM3LL	3.E-04	RCIC-PS-9A & 9B MISCALIBRATION
RCIHUMNPS12AM3LL	3.E-04	RCIC-PS-12A HUMAN ERROR PPM 7.4.3.2.1.50
RCIHUMNPS12BM3LL	3.E-04	RCIC-PS-12B HUMAN ERROR PPM 7.4.3.2.1.51
RCIHUMNPS12CM3LL	3.E-04	RCIC-PS-12C HUMAN ERROR PPM 7.4.3.2.1.50
RCIHUMNPS12DM3LL	3.E-04	RCIC-PS-12D HUMAN ERROR PPM 7.4.3.2.1.51
RCIHUMNPS13AM3LL	3.E-04	RCIC-DPIS-13A HUMAN ERROR PPM 7.4.3.2.1.20
RCIHUMNPS13BM3LL	3.E-04	RCIC-DPIS-13B & 7B HUMAN ERROR PPM 7.4.3.2.1.80
RCIHUMNPS22AM3LL	3.E-04	RCIC-PS-22A HUMAN ERROR PPM 7.4.3.2.1.50
RCIHUMNPS22BM3LL	3.E-04	RCIC-PS-22B HUMAN ERROR PPM 7.4.3.2.1.49
RCIHUMNPS22CM3LL	3.E-04	RCIC-PS-22C HUMAN ERROR PPM 7.4.3.2.1.50
RCIHUMNPS22DM3LL	3.E-04	RCIC-PS-22D HUMAN ERROR PPM 7.4.3.2.1.49
RCIHUMNPVTSTX3LL	3.E-04	HUMAN ERROR DURING RCIC OPERABILITY TEST
RHRHUMN--PMAX3LL	3.E-04	PREVENTIVE MAINTENANCE ERROR (RHR LOOP B)
RHRHUMN--PMBX3LL	3.E-04	PREVENTIVE MAINTENANCE ERROR (RHR LOOP B)
RHRHUMN--PMCX3LL	3.E-04	PREVENTIVE MAINTENANCE ERROR (RHR LOOP C)
RHRHUMN-FIXAJ3LL	3.E-04	REPAIR ERROR (RHR LOOP A)
RHRHUMN-FIXBJ3LL	3.E-04	REPAIR ERROR (RHR LOOP B)
RHRHUMN-FIXCJ3LL	3.E-04	REPAIR ERROR (RHR LOOP C)
RHRHUMNTESTAX3LL	3.E-04	TESTING ERROR (RHR LOOP A)
RHRHUMNTESTBX3LL	3.E-04	TESTING ERROR (RHR LOOP B)
RHRHUMNTESTCX3LL	3.E-04	TESTING ERROR (RHR LOOP C)
RRAHUMNFC-01J3D2	3.E-04	OPERATOR ERROR REPAIR MAINTENANCE, RRA-FC-01
RRAHUMNFC-02J3D1	3.E-04	OPERATOR ERROR REPAIR MAINTENANCE, RRA-FC-02
RRAHUMNFC-03J3D2	3.E-04	OPERATOR ERROR REPAIR MAINTENANCE, RRA-FC-03
RRAHUMNFC-04J3D3	3.E-04	OPERATOR ERROR REPAIR MAINTENANCE, RRA-FC-04
RRAHUMNFC-05J3D1	3.E-04	OPERATOR ERROR REPAIR MAINTENANCE, RRA-FC-05

Basic Event ID	Point Est	Description
RRAHUMNFC-06J3D2	3.E-04	OPERATOR ERROR REPAIR MAINTENANCE, RRA-FC-06
RRAHUMNFC-10J3D2	3.E-04	OPERATOR ERROR REPAIR MAINTENANCE, RRA-FC-10
RRAHUMNFC-11J3D1	3.E-04	OPERATOR ERROR REPAIR MAINTENANCE, RRA-FC-11
RRAHUMNFC-12J3D1	3.E-04	OPERATOR ERROR REPAIR MAINTENANCE, RRA-FC-12
SLCHUMN---TMX3XX	3.E-04	SLC UNAVAILABLE DUE TO T/M ERROR, LIKE SLC-V-31 LE
SLCHUMN--SLCJ3XX	3.E-04	OPERATOR ERROR REPAIR MAINTENANCE, TESTED COMPONENT
SLCHUMNBORONX3XX	3.E-04	INSUFFICIENT BORON IN INJECTION FLUID DUE TO PM ER
SW-HUMNSWP1AJ3LL	3.E-04	HUMAN ERROR DURING PREVENTIVE MAINTENANCE ON PUMP OR VALVE
SW-HUMNSWP1BJ3LL	3.E-04	HUMAN ERROR DURING PREVENTIVE MAINTENANCE ON PUMP OR VALVE
TSWHUMNSWP1BJ3PB	3.E-04	UNAVAILABILITY FROM HUMAN ERROR DURING PUMP OR VALVE
TSWHUMNSWP1BX3PB	3.E-04	HUMAN ERROR DURING PREVENTIVE MAINTENANCE ON PUMP OR VALVE

TABLE 3.3.3-2
Manual Back-up of Automatic Actions Considered in the System Fault Trees

Basic Event Id	Point Est	Description
ADSHUMNSTARTH3LL	2.66E-003	OPERATOR DOES NOT INITIATE ADS
HPSHUMNSTARTH3LL	2.66E-003	OPERATOR FAILS TO INITIATE.HPCS WHEN REQUIRED TO
LPSHUMNINITIH3LL	2.66E-003	OPERATOR NEGLECTS TO START LPCS WHEN IT IS NEEDED
RCIHUMNSTARTH3LL	2.66E-003	OPERATOR FAILS TO INITIATE RCIC WHEN REQUIRED TO
RHRHUMNLPCISTART	2.66E-003	OPERATOR FAILS TO INITIATE LPCI MANUALLY
ARIHUMN-----H3LL	2.66E-003	OPERATOR DOES NOT RESPOND WHEN HE IS SUPPOSED TO
RRAHUMNRFC10H3D2	2.66E-003	CONTROL ROOM OPERTR DOES NOT TURN ON RRA-FC-10 FAN COIL UNIT WHEN REQUIRED
RRAHUMNRFC11H3D1	2.66E-003	CONTROL ROOM OPERTR DOES NOT TURN ON RRA-FC-11 FAN COIL UNIT WHEN REQUIRED

TABLE 3.3.3-3

Nominal Model of Estimated HEPs and EFs for Diagnosis within Time T by Control Room Personnel of Abnormal Events Annunciated Closely in Time [1].

T, minutes	Median joint HEP_{d-em} for diagnosis of a single or the first event	EF
1	1.0	n/a
10	0.1	10
20	0.01	10
30	0.001	10
60	0.0001	30
1500	0.00001	30

TABLE 3.3.3-4

Operator Performance HEPs [1].

	Step-by-Step, Moderate Stress		Step-by-Step, Extreme Stress		Dynamic, Moderate Stress		Dynamic, Extreme Stress	
Operator	HEP	EF	HEP	EF	HEP	EF	HEP	EF
HEP_{op}: operator. action	0.02	5	0.05	5	0.05	5	0.25	5
HEP_r: recovery action	0.2	5	0.5	5	0.5	5	0.5	5

TABLE 3.3.3-5

Human Error Probabilities in using Symptom-oriented Emergency Operating Procedures

Basic Event Id	Point Est	Description
RHRHUMNSP-COOLL	1.E-5 (Transient) 3.E-5 (LOCA)	Human Errors in Following Procedure (PPM 2.4.2) to Bring the RHR System into Suppression Pool Cooling Mode
SLCHUMN20MINUTES	0.035	Human Errors in Following Procedure (PPM 5.1.4/PPM 2.4.1) to Initiate SLC during MSIV Closure ATWS in 20 minutes
SLCHUMN40MINUTES	0.025	Human Errors in Following Procedure (PPM 5.1.4/PPM 2.4.1) to Initiate SLC during MSIV Closure ATWS in 40 minutes
ZM	8.E-3	Human Error In following Procedure (PPM 5.5.7) to Open MSIVs to Recover PCS
VENTFAIL	0.06	Human Error in following Procedure (PPM 5.5.14) to Perform Containment Venting
AI	0.05	Human Error in Following Procedure (PPM 5.1.2) to Inhibit ADS and Lower Level to Control Power during MSIV Closure ATWS
AIM	0.05	Human Error in following Procedure (PPM 5.1.2) to Inhibit ADS and Lower Level to Control Power during Non-MSIV Closure ATWS
FP-HUMNSYS62H3LL	0.036	Human Error in following Procedure (PPM 5.5.3) to Connect Firewater Condensate System during Long Term SBO Sequence
RHRHUMNMKIIFLOOD	0.09	Human Error in Following Procedure (PPM 5.5.17) to Flood Containment
RHRHUMNSWCRTIELL	0.157 (SBO) 1.0 (Transients)	Human Error in following Procedure (PPM 5.5.17) to Cross-tie SW to RHR-B
RFWHUMN-1A--H3LL	0.034	Human Error in following Procedure (PPM3.3.1/PPM 5.1.1) to Use RFW to Control Level Following Turbine Trip
RPTHUMN-RPTH3LL	1.0	Human Error in following Procedure (PPM 5.1.2) to Initiate RPT Manually
RCIHUMNRCOLH3LL	0.002	Human Error in Following Procedure (PPM 5.6.1) to Provide Alternate Room Cooling to RCIC

3.3.4 Common Cause Failure Data

Common cause failures were modelled in the WNP-2 system fault trees in accordance with the Beta Factor methodology. For greater than two components, the (N+1)/2 beta factor method of NUREG/CR-4550 was used. Data for use in the analysis is from NUREG/CR-4780.

The common cause failure modelling was part of a wider evaluation aimed at analyzing and estimating the effects of dependencies in and among plant systems. The important dependencies are those that cause the probability of system failure (or part of a system) to be larger than the product of the system's (or the system part's) failure probabilities.

The common cause failure modelling treated those dependencies that were not explicitly evaluated in the IPE. The list below gives dependencies explicitly treated in the IPE and their method of treatment:

Support System Dependencies: Transfers to support system fault trees were included at appropriate points in system fault trees. Linking of fault trees during fault tree cutset generation ensured such dependencies were accounted for correctly in IPE results.

Shared Components Among Front Line Systems: Each plant component has a unique identification on the Master Equipment List. A basic fault associated with the component has the same unique identification plus failure mode. The same basic fault was used in more than one system fault tree if the component is shared between systems.

Human Errors: Common human errors were included in the IPE by having the same basic event designation. Human errors such as incorrect calibration of sensors or instruments were included as same events in system models if the same operator and procedure are used to calibrate different sensors or instruments.

External Events: Dependencies among component failures due to the effects of "external" events (earthquake, fire, external flood, tornado, and heavy wind) were excluded from the IPE at this time. The effects of these events will be evaluated in the IPEEE.

Some potential causes of dependent component failures other than those listed above include common design, manufacture, installation errors, and internal physical similarities. The beta method provides a conservative way to implicitly model all common cause failures not explicitly modelled in the IPE.

Generally, only common cause failures within a system were considered. However, the LPCS and LPCI components were treated for common cause between systems and the TSW and SW components were treated for common cause between those systems. For each system only components in redundant loops were considered for common cause failures. The following components in redundant loops were considered for common cause failures:

Diesel Generators (failure to start and run)

Pumps (failure to start and run)

Motor operated Valves (failure to open or close on demand)

Circuit Breakers (failure to open or close on demand)

Batteries

Battery Chargers

Air-Operated Valves (failure to open or close on demand)

Safety/Relief Valves (failure to open or reclose on demand)

Check Valves (failure to open on demand; failure to remain closed)

The following beta factors taken from NUREG/CR-4780 (except the SRV beta factor was calculated using formula from NUREG/CR-4550) were used:

COMPONENT	BETA FACTOR (β)	
Diesel Generators	$\beta_2 = 0.05$	$\beta_3 = 0.026$
Motor operated Valves	$\beta_2 = 0.08$	$\beta_3 = 0.043$
SRVs	$\beta_2 = 0.088$ (i>10)	
Pump		
• Safety Injection	0.17	
• RHR	$\beta_2 = 0.11$	$\beta_3 = 0.061$
• Containment Spray	0.05	
• Service Water	0.03	
Battery Chargers	$\beta_2 = 0.10$	$\beta_3 = 0.055$
All Others	0.10	

The beta factors were applied at component level with consistent basic event identifications.

3.3.5 Quantification of Unavailability of Systems and Functions

All systems are dependent, in some way, on other systems for successful operation. System dependencies were accounted for by explicitly including the dependencies within the models. This was accomplished by modeling each dependency on other systems or portions of systems with an 'external transfer' to the appropriate gate within the system model of the supporting system fault tree. The linking of the fault tree then incorporates the full model to fully account for dependencies.

The event trees in Section 3.1.2 identify functional equations associated with each of the branch points. A functional equation represents the unavailability of one or more front-line systems and their support systems to perform a specific function within the conditions identified in the event tree. The event tree conditions which impact the operation of systems are accounted for by the use of house events included in the fault trees. These house events are logical switches which can be set to logical true or logical false to turn on or off portions of the fault trees. For example, the loss of division 2 DC power (TDC) event tree represents an initial condition where division 2 DC power is failed. This condition is represented by inserting a house event (XHOSNDIV2DC) in the DC power fault tree which when set to 1.0 (logical true) fails division 2 DC power. Due to the linking of the models, this will propagate through the linked model and also represent failure of any components dependent upon division 2 DC power. The house events used in the WNP-2 IPE models are described in Table 3.3.5-1. The settings of the house events are controlled by updating the data associated with the house events prior to each solution of the fault tree with a house event data base. The WNP-2 IPE utilizes 22 different groups of settings referred to as "house event BED files" to update the various fault trees under different conditions. The house event BED files used in the WNP-2 IPE models and the specific logical settings (1 = true, 0 = false) within each file are summarized in Table 3.3.5-2.

Functional equations may be developed by one of three methods, 1) by solution of a single linked front-line system fault tree, 2) by a single basic event developed by the analyst, or 3) by a boolean combination of two or more single linked front-line system fault trees.

Prior to evaluation, each system fault tree is linked to its dependent system fault trees as previously discussed. Within this process, the logic loops which arise when the trees are combined must be identified and broken. Logic loops arise due to the system interdependencies which exist in any plant, e.g., AC power is dependent on Service Water via the diesel and room cooling requirements, and Service Water is in turn dependent on AC power for component operation. The logic loops are broken by duplicating and renaming support system fault trees and removing the dependency. For example, in the above situation, the service water tree which is linked in the AC power model is duplicated and renamed from the service water model which is used elsewhere and the AC power dependencies are removed. This special service water model is only linked in to support AC power, in all other cases the general service water model is used retaining all dependencies.

Fault trees are discussed in detail in the system notebooks (retained at the Supply System as part of the second tier of documentation). Basic event probabilities (component failures, test/maintenance unavailabilities, human errors and common cause failures) are discussed in Sections 3.3.1 through 3.3.4.

In order to solve the WNP-2 event trees, 142 cutset equations were developed. These cutset equations are broken down into the three types discussed above as follows:

1) 70 equations generated by solution of fault trees
2) 51 equations consisting of single basic events
3) 21 equations generated by combination of 2 or more of the type 1 solutions

The final functional equations were in general solved with a truncation value of 1E-8, however in several cases a value of 1E-7 was used to limit the number of cutsets generated to a manageable amount. This was primarily done to optimize the later sequence quantification. Initially all of the functional equations were solved using a truncation value of 1E-8. The cutsets were reviewed to ensure that using a higher truncation value resulted in acceptable coverage and that no potentially significant dependencies were lost.
Table 3.3.5-3 summarizes the solutions for the type 1 and 2 equations above. Table 3.3.5-4 summarizes the results of the type 3 functional equations generated.

TABLE 3.3.5-1 - House Event Descriptions and Locations in Fault Trees		
House Event	**Description**	**Fault Tree(s) Where Found**
XHOS-500KV-SS	500 kv Sensitivity Study	eac
XHOS-ATWS	ATWS Sequence	dam
XHOS-BLADDERTANK	Bladder Tank Empty	fpw
XHOS-CWPUMPHOUSE	Circ. Water Basin Empty	fpw, pcs
XHOS-LOOP	Loss of Offsite Power Occurs	crd, eac, tsw, xtsw, ytsw, ztsw
XHOS-NO-SBO	No SBO Conditions Exist	edc, swa, swb, xedc, xswa, xswb, yedc, yswa, yswb, zedc, zswa, zswb
XHOS-SBO	SBO Conditions Exist	edc, rec, xedc, xrec, yedc, zedc
XHOS-STEAM	Rx. Steam Pressure < 65 psig	pcs, rcic, rfw
XHOS-TSSW	Loss of Standby Service Water Initiating Event Occurs	swa, swb, swc, xswa, xswb, xswc, yswa, yswb, zswa, zswb
XHOS-TSW	Loss of Plant Service Water Initiating Event Occurs	tsw, xtsw, ytsw, ztsw
XHOSCSTLL	CST Low Level	crd, hpcs, pcs, rcic
XHOSFA	Drywell Pressure > 1.65 psig or L2 Exists	rhr, tsw, xtsw, ytsw, ztsw
XHOSFL1	Drywell Pressure > 1.68 psig or L1 Exists	rhr, swa, xswa, yswa, zswa
XHOSL2	L2 Exists	swb, xswb, yswb, zswb
XHOSL8	Rx Water Level > L8	hpcs, rcic
XHOSLDSHED	Essential Electrical Loads Shed	eac
XHOSNCN	CN System out-of-service	cn
XHOSNCSTLL	No CST Low Level	hpcs
XHOSNDIV1AC	Division 1 AC out-of-service	eac

TABLE 3.3.5-1 - House Event Descriptions and Locations in Fault Trees		
House Event	**Description**	**Fault Tree(s) Where Found**
XHOSNDIV1DC	Division 1 DC out-of-service	edc, xedc, yedc, zedc
XHOSNDIV2AC	Division 2 AC out-of-service	eac
XHOSNDIV2DC	Division 2 DC out-of-service	edc, tsw, xedc, xtsw, yedc, ytsw, zedc, ztsw
XHOSNF	Drywell Pressure < 1.68 psig	rhr
XHOSNFA	Drywell Pressure < 1.68 psig or L2 Does Not Exist	hpcs, rhr
XHOSNFL1	Neither Drywell Pressure < 1.68 psig or L1 Exists	ads, lpcs, rhr, swa, swb, xswa, xswb, yswa, yswb, zswa, zswb
XHOSNL2	L2 Does Not Exist	rcic
XHOSNNS4	Containment Not Isolated	slc
XHOSNS4	Containment Isolated	rfw, rhr, slc
XHOSNSPHL	No Supp. Pool High Level for Transfer From CST	hpcs
XHOSRXHP	RX Pressure > 470 psig	lpcs, rhr
XHOSSPHT	Suppression Pool Temperature Too High for Pump Seals	hpcs, lpcs, rcic, rhr
XHOSSPLL	Suppression Pool Empty	hpcs, lpcs, rcic, rhr
XHOSTORNADO	Tornado @ WNP-2	eac, swa, swb, swc, xswa, xswb, xswc, yswa, yswb, zswa, zswb
XHOSUVLOCASIGNAL	Undervoltage and LOCA Signals Exist	eac
XHOSVOLCANO	Excessive Volcanic Ash @ WNP-2	eac.lgc, swa.lgc, swb.lgc, swc.lgc, xswa.lgc, xswb.lgc, xswc.lgc, yswa.lgc, yswb.lgc, zswa.lgc, zswb.lgc

TABLE 3.3.5-2 - House Event Settings in House Event BED Files																						
HOUSE EVENT NAME	HS-SBO	HS-ATWS	HS-TE	HS1-DC	HS1-SSW	HS1-TSW	HS1	HS2-SSW	HS2-TSW	HS2	HS3	HS4-DC	HS4-SSW	HS4-TSW	HS4	HS5	HS6-SSW	HS6-TSW	HS6	HS7	HS8	HS9
XHOS-500KV-SS	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
XHOS-ATWS	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
XHOS-BLADDERTANK	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
XHOS-CWPUMPHOUSE	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
XHOS-LOOP	1	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
XHOS-NO-SBO	0	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
XHOS-SBO	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
XHOS-STEAM	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
XHOS-TSSW	0	0	0	0	1	0	0	1	0	0	0	0	1	0	0	0	1	0	0	0	0	0
XHOS-TSW	0	0	0	0	0	1	0	0	1	0	0	0	0	1	0	0	0	1	0	0	0	0
XHOSCSTLL	0	0	0	0	0	0	0	1	1	1	1	1	1	1	1	0	0	0	0	0	0	1
XHOSFA	1	1	1	1	1	1	1	1	0	1	0	0	0	0	0	1	0	0	0	0	1	1
XHOSFL1	1	1	1	0	0	0	0	1	1	1	0	1	1	1	1	1	0	0	0	0	0	1
XHOSL2	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	0	0	0	1	1
XHOSL8	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
XHOSLDSHED	1	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
XHOSNCN	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0
XHOSNCSTLL	1	1	1	1	1	1	1	0	0	0	0	0	0	0	0	1	1	1	1	1	1	0
XHOSNDIV1AC	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

TABLE 3.3.5-2 - House Event Settings in House Event BED Files																						
HOUSE EVENT NAME	HS-SBO	HS-ATWS	HS-TE	HS1-DC	HS1-SSW	HS1-TSW	HS1	HS2-SSW	HS2-TSW	HS2	HS3	HS4-DC	HS4-SSW	HS4-TSW	HS4	HS5	HS6-SSW	HS6-TSW	HS6	HS7	HS8	HS9
XHOSNDIV1DC	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
XHOSNDIV2AC	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
XHOSNDIV2DC	0	0	0	1	0	0	0	0	0	0	1	1	0	0	0	0	0	0	0	1	0	1
XHOSNF	1	1	1	1	1	1	1	0	1	0	1	1	1	1	1	1	0	0	0	0	0	0
XHOSNFA	0	0	0	0	0	0	0	0	1	0	1	1	1	1	1	0	1	1	1	1	0	0
XHOSNFL1	0	0	0	1	1	1	1	0	0	0	1	0	0	0	0	0	1	1	1	1	1	0
XHOSNL2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	1	1	0	0
XHOSNMAKEUP	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
XHOSNNS4	0	0	0	0	0	0	0	0	1	0	1	1	1	1	1	0	1	1	1	1	1	0
XHOSNS4	1	1	1	1	1	1	1	1	0	1	0	0	0	0	0	1	0	0	0	0	0	1
XHOSNSPHL	1	1	1	1	1	1	1	0	0	0	0	0	0	0	0	1	1	1	1	1	1	0
XHOSRXHP	0	1	0	1	1	1	1	0	0	0	0	0	0	0	0	1	1	1	1	1	1	0
XHOSSPHL	0	0	0	0	0	0	0	1	1	1	1	1	1	1	1	0	0	0	0	0	0	1
XHOSSPHT	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
XHOSSPLL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
XHOSTORNADO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
XHOSUVLOCASIGNAL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
XHOSVOLCANO	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

TABLE 3.3.5-3: Functional Equation Results						
Eqn. File	Logic File	House Event BED File	Solved Gate	Top Event Unavail.	No. Cutsets	Truncation Value
U2	RCIC.LGC	HS1	GRCI112	1.91E-1	363	1.00E-8
U2-DC	RCIC.LGC	HS1-DC	GRCI112	2.41E-1	376	1.00E-8
U2-SSW	RCIC.LGC	HS1-SSW	GRCI112	2.41E-1	406	1.00E-8
U2-TSW	RCIC.LGC	HS1-TSW	GRCI112	1.98E-1	479	1.00E-8
U2M	RCIC.LGC	HS6	GRCI112	1.96E-1	361	1.00E-8
U2-LOOP	RCIC.LGC	HS-TE	GRCI112	2.69E-1	290	1.00E-8
U2-SBO	RCIC.LGC	HS-SBO	GRCI112	9.74E-2	206	1.00E-8
U1	HPCS.LGC	HS1	GHPS112	5.72E-2	196	1.00E-8
U1-DC	HPCS.LGC	HS1-DC	GHPS112	5.72E-2	196	1.00E-8
U1-TSW	HPCS.LGC	HS1-TSW	GHPS112	5.77E-2	466	1.00E-8
U1-LOOP	HPCS.LGC	HS-TE	GHPS112	1.17E-1	90	1.00E-8
U1-SBO	HPCS.LGC	HS-SBO	GHPS112	1.17E-1	84	1.00E-8
XTT	ADS.LGC	HS5	GADS112	6.17E-6	77	1.00E-8
X-SSW	ADS.LGC	HS6-SSW	GADS112	2.71E-3	197	1.00E-8
X-TSW	ADS.LGC	HS6-TSW	GADS112	2.75E-3	739	1.00E-8
XTE	ADS.LGC	HS7	GADS112	2.83E-3	255	1.00E-8
XMA	ADS.LGC	HS6	GADS112	2.66E-3	17	1.00E-8
XCN	ADS.LGC	HS8	GADS112	2.68E-3	30	1.00E-8
X-LOOP	ADS.LGC	HS-TE	GADS112	6.39E-3	1755	1.00E-8
V2TT	COND.LGC	HS4	GCON162	1.11E-2	689	1.00E-8
V2-DC	COND.LGC	HS4-DC	GCON162	1.24E-2	796	1.00E-8
V2-SSW	COND.LGC	HS4-SSW	GCON162	1.13E-2	783	1.00E-8
LPCS	LPCS.LGC	HS2	GLPS112	4.27E-2	222	1.00E-8
LPCS-DC	LPCS.LGC	HS9	GLPS112	4.27E-2	227	1.00E-8
LPCS-TSW	LPCS.LGC	HS2-TSW	GLPS112	4.34E-2	334	1.00E-8
LPCS-TE	LPCS.LGC	HS-TE	GLPS112	1.11E-1	142	1.00E-8
LPCI	RHR.LGC	HS2	GRHR2912	1.79E-3	1749	1.00E-8
LPCI-TSW	RHR.LGC	HS2-TSW	GRHR2912	1.83E-3	2012	1.00E-8

TABLE 3.3.5-3: Functional Equation Results						
Eqn. File	Logic File	House Event BED File	Solved Gate	Top Event Unavail.	No. Cutsets	Truncation Value
W1TT	RHR.LGC	HS4	GRHR100	3.10E-4	1612	1.00E-8
W1-TSW	RHR.LGC	HS4-TSW	GRHR100	3.39E-4	1863	1.00E-8
W1TDC	RHR.LGC	HS3	GRHR100	1.26E-2	243	1.00E-8
LPCI1	RHR.LGC	HS9	GRHR100	1.26E-2	243	1.00E-8
LPCIA	RHR.LGC	HS2	GRHR2921	3.47E-2	221	1.00E-8
RHRA	RHR.LGC	HS4	GRHR452	1.26E-2	238	1.00E-8
RHRB	RHR.LGC	HS4	GRHR572	1.87E-2	237	1.00E-8
LPCIB	RHR.LGC	HS2	GRHR3012	4.00E-2	221	1.00E-8
LPCIC	RHR.LGC	HS2	GRHR3112	4.93E-2	219	1.00E-8
LPCI-TE	RHR.LGC	HS-TE	GRHR2912	1.01E-2	3610	1.00E-8
W1-LOOP	RHR.LGC	HS-TE	GRHR100	7.84E-3	3282	1.00E-8
ZTT	PCS.LGC	HS6	GPCS131	1.10E-2	674	1.00E-8
V4TT	V4TT.LGC	HS4	GV4T112	6.58E-1	161	1.00E-8
V4-TSW	V4TT.LGC	HS4-TSW	GV4T112	6.58E-1	161	1.00E-8
C3	SLC.LGC	HS6	GSLC112	7.95E-2	396	1.00E-8
C3M	SLC.LGC	HS1	GSLC112	8.95E-2	397	1.00E-8
K	ARI.LGC	HS1	GARI112	1.35E-3	207	1.00E-8
R	RPT.LGC	HS1	GRPT112	1.16E-2	264	1.00E-8
V3TT	FPW.LGC	HS2	GFPW122	4.04E-2	61	1.00E-8
V3-TE	FPW.LGC	HS-TE	GFPW122	4.18E-2	37	1.00E-8
NS4	NS4.LGC	HS1	GNS4212	7.45E-4	794	1.00E-8
ZTE	ZTM.LGC	HS7	GZTM112	2.04E-2	782	1.00E-8
Z-SSW	ZTM.LGC	HS6-SSW	GZTM112	1.92E-2	768	1.00E-8
FEED	FEED.LGC	HS6	GFEE112	3.97E-2	493	1.00E-8
FEEDDC	FEED.LGC	HS7	GFEE112	4.10E-2	507	1.00E-8
FEEDSSW	FEED.LGC	HS6-SSW	GFEE112	3.99E-2	522	1.00E-8
522	RFW.LGC	HS6	GRFW522	5.97E-2	415	1.00E-8
162	RFW.LGC	HS6	GRFW162	4.94E-2	304	1.00E-7

TABLE 3.3.5-3: Functional Equation Results						
Eqn. File	Logic File	House Event BED File	Solved Gate	Top Event Unavail.	No. Cutsets	Truncation Value
1622	RFW.LGC	HS6	GRFW1622	2.92E-2	297	1.00E-7
522DC	RFW.LGC	HS7	GRFW522	6.70E-2	322	1.00E-8
162DC	RFW.LGC	HS7	GRFW162	8.03E-2	290	1.00E-7
1622DC	RFW.LGC	HS7	GRFW1622	6.01E-2	276	1.00E-7
522SSW	RFW.LGC	HS6-SSW	GRFW522	6.01E-2	385	1.00E-8
162SSW	RFW.LGC	HS6-SSW	GRFW162	4.98E-2	310	1.00E-7
1622SSW	RFW.LGC	HS6-SSW	GRFW1622	2.96E-2	285	1.00E-7
W2TT	W2TT.LGC	HS6	GW2T112	2.08E-2	522	1.00E-8
W2-SSW	W2TT.LGC	HS6-SSW	GW2T112	2.10E-2	475	1.00E-8
W2-DC	W2TT.LGC	HS7	GW2T112	2.48E-2	324	1.00E-8
FAIL	FAIL.LGC	NA	GFAI112	1.00	1	1.00E-8
DAM	DAM.LGC	HS5	GDAM112	8.21E-3	64	1.00E-10
DAM-ATWS	DAM.LGC	HS-ATWS	GDAM112	2.96E-2	1581	1.00E-10
EDG-1	EAC.LGC	HS-TE	GEAC912	7.42E-2	77	1.00E-8
EDG-2	EAC.LGC	HS-TE	GEAC2012	7.46E-2	70	1.00E-8
A	IE.LGC	NA	GIE-120	3.00E-4	1	1.00E-8
AO	IE.LGC	NA	GIE-123	2.17E-4	1	1.00E-8
S1	IE.LGC	NA	GIE-121	3.00E-3	1	1.00E-8
S2	IE.LGC	NA	GIE-122	8.00E-3	1	1.00E-8
SR	IE.LGC	NA	GIE-124	1.00E-2	1	1.00E-8
TC	IE.LGC	NA	GIE-151	5.00E-2	1	1.00E-8
TCAS	IE.LGC	NA	GIE-180	1.25E-3	1	1.00E-8
TCN	IE.LGC	NA	GIE-181	1.25E-3	1	1.00E-8
TDC	IE.LGC	NA	GIE-182	3.00E-3	1	1.00E-8
TF	IE.LGC	NA	GIE-152	1.00E-1	1	1.00E-8
TI	IE.LGC	NA	GIE-155	1.25E-1	1	1.00E-8
TM	IE.LGC	NA	GIE-153	2.00E-1	1	1.00E-8
TSSW	IE.LGC	NA	GIE-183	1.00E-4	1	1.00E-8

TABLE 3.3.5-3: Functional Equation Results						
Eqn. File	Logic File	House Event BED File	Solved Gate	Top Event Unavail.	No. Cutsets	Truncation Value
TT	IE.LGC	NA	GIE-154	3.30	1	1.00E-8
TTSW	IE.LGC	NA	GIE-184	1.25E-3	1	1.00E-8
TE	IE.LGC	NA	GIE-185	2.46E-2	1	1.00E-8
FLD6	IE.LGC	NA	GIE-220	2.92E-3	1	1.00E-8
FLD7	IE.LGC	NA	GIE-221	1.60E-5	1	1.00E-8
FLD14	IE.LGC	NA	GIE-222	4.69E-3	1	1.00E-8
AI	ATWS-EV.LGC	NA	GATW120	5.00E-2	1	1.00E-8
AIM	ATWS-EV.LGC	NA	GATW121	5.00E-2	1	1.00E-8
C1	ATWS-EV.LGC	NA	GATW122	2.00E-2	1	1.00E-8
CE	ATWS-EV.LGC	NA	GATW123	1.00E-5	1	1.00E-8
CM	ATWS-EV.LGC	NA	GATW124	4.00E-6	1	1.00E-8
RPS	ATWS-EV.LGC	NA	GATW155	1.40E-5	1	1.00E-8
TCC	ATWS-EV.LGC	NA	GATW154	5.00E-2	1	1.00E-8
TFC	ATWS-EV.LGC	NA	GATW153	1.00E-1	1	1.00E-8
TIC	ATWS-EV.LGC	NA	GATW152	2.05E-2	1	1.00E-8
TMC	ATWS-EV.LGC	NA	GATW151	2.00E-1	1	1.00E-8
TTC	ATWS-EV.LGC	NA	GATW180	2.70	1	1.00E-8
TTC2	ATWS-EV.LGC	NA	GATW181	6.00E-1	1	1.00E-8
D	SINGLES.LGC	NA	GEV-120	1.00E-4	1	1.00E-8
MC	SINGLES.LGC	NA	GEV-121	5.00E-5	1	1.00E-8
MS	SINGLES.LGC	NA	GEV-122	5.00E-1	1	1.00E-8
MTM	SINGLES.LGC	NA	GEV-123	5.00E-5	1	1.00E-8
MTT	SINGLES.LGC	NA	GEV-124	5.00E-5	1	1.00E-8
NIL	SINGLES.LGC	NA	GEV-155	1.00E-15	1	1.00E-20
PC	SINGLES.LGC	NA	GEV-154	4.64E-2	1	1.00E-8
PTM	SINGLES.LGC	NA	GEV-153	3.82E-2	1	1.00E-8
PTT	SINGLES.LGC	NA	GEV-152	1.91E-2	1	1.00E-8
NRAC10	SINGLES.LGC	NA	GEV-151	2.96E-2	1	1.00E-8

TABLE 3.3.5-3: Functional Equation Results						
Eqn. File	Logic File	House Event BED File	Solved Gate	Top Event Unavail.	No. Cutsets	Truncation Value
NRAC4	SINGLES.LGC	NA	GEV-180	1.44E-1	1	1.00E-8
NRAC30M	SINGLES.LGC	NA	GEV-181	6.22E-1	1	1.00E-8
IS	SINGLES.LGC	NA	GEV-182	1.21E-6	1	1.00E-8
ZM	SINGLES.LGC	NA	GEV-183	8.00E-3	1	1.00E-8
U1-SUCC	SINGLES.LGC	NA	GEV-184	8.83E-1	1	1.00E-8
U2-SUCC	SINGLES.LGC	NA	GEV-185	9.03E-1	1	1.00E-8
U1-S	SINGLES.LGC	NA	GEV-220	8.83E-1	1	1.00E-8
U2-S	SINGLES.LGC	NA	GEV-221	7.31E-1	1	1.00E-8

TABLE 3.3.5-4: Merged Functional Equation Summary					
OCL File	Equation Generated	OCL Logic (1)	Frequency	No. of Cutsets	Truncation Value
ATWS-ST.OCL	R-ST	R \ DAM-ATWS	8.10E-3	61	1.00E-9
ATWS-ST.OCL	K-ST	K \ DAM-ATWS	1.14E-3	12	1.00E-9
ATWS-ST.OCL	C3M-ST	C3M \ DAM-ATWS	7.45E-2	279	1.00E-9
ATWS-ST.OCL	C3-ST	C3 \ DAM-ATWS	6.45E-2	279	1.00E-9
DGDG.OCL	DGDG	EDG-1 * EDG-2	6.32E-3	1377	1.00E-8
UTT.OCL	UTT	U1 * U2	1.09E-2	3694	1.00E-8
UTT.OCL	U-DC	U1-DC * U2-DC	1.38E-2	3514	1.00E-8
UTT.OCL	U-TSW	U1-TSW * U2-TSW	1.14E-2	5482	1.00E-8
V1TT.OCL	V1TT	LPCS * LPCI	1.51E-4	1131	1.00E-8
V1TT.OCL	V1-TSW	LPCS-TSW * LPCI-TSW	1.71E-4	1358	1.00E-8
V-LOOP.OCL	V-LOOP	LPCS-TE * LPCI-TE * V3-TE	2.78E-4	887	1.00E-8
V.OCL	V	V1TT * V2TT	2.69E-7	10	1.00E-8
V1TDC.OCL	V1TDC	LPCS-DC * LPCI1	7.67E-3	524	1.00E-8
UTM.OCL	UTM	U1 * U2M	1.12E-2	3840	1.00E-8
ZTM.OCL	ZTM	ZTT * ZM	1.90E-2	675	1.00E-8
QTT.OCL	QTT	(162 * 1622) + 522 + FEED	9.88E-2	752	1.00E-8
QTT.OCL	Q-DC	(162DC * 1622DC) + 522DC + FEEDDC	1.29E-1	653	1.00E-8
QTT.OCL	Q-SSW	(162SSW * 1622SSW) + 522SSW + FEEDSSW	9.92E-2	722	1.00E-8
QU.OCL	QU	QTT * UTT	1.10E-3	7678	1.00E-8
VFD.OCL	VFD	LPCS * LPCIA	8.45E-3	1101	1.00E-9
VFL.OCL	VFL	LPCIB * LPCIC	1.07E-2	561	1.00E-8

(1) See Table 3.3.5-3 for input equation information
Logic symbols - * = Boolean AND, + = Boolean OR, \ = Delete

3.3.6 Generation of Support System States and Quantification of Their Probabilities

This report does not use the support state methodology. System dependencies are accounted for by fault tree linking. This section is therefore not applicable for this report.

3.3.7 Quantification of Sequence Frequencies

Following completion of the development of the functional equations (Section 3.3.5) which define the cutsets for each event tree heading, the accident sequence quantification is performed to develop the accident sequence frequencies. The event trees in Section 3.1.2 define 177 sequences which result in a core damage (CD) end state and 53 sequences which result in transfers to other event trees. Each accident sequence is uniquely defined by the functional successes and functional failures along the accident sequence path through the event tree. The quantification of each accident sequence involves combining the functional equations for each functional failure with a Boolean "AND" and performing a boolean reduction of the resultant equation. After the system failures are ANDed together, the failure equations for the each functional failure with the Boolean "AND" and performed a Boolean reduction of the resultant equation for a minimal cutset solution. After the system failures are ANDed together, the failure equations for the functional successes are deleted from this equation. Accounting for the success paths is an essential step to avoid developing incorrect cutsets. One additional step is performed for each accident sequence, deleting the disallowed cutsets which arise due to modeling simplifications. This step is done by performing a delete operation on the sequence equation using an equation containing the disallowed cutsets, such as (HPCS maintenance * RCIC maintenance). A truncation value of 1E-10 was used for all accident sequence quantification steps.

Each of the transfer sequences was reviewed and the transfer sequence frequency was compared to the initiating event frequency of the event tree into which it transfers. In most cases the transfer frequency was 2 or more orders of magnitude below the initiating event frequency and is insignificant. The only transfers which were significant were the transient induced stuck open relief valve transfers to the IORV/SORV event tree. The cutsets associated with these transfers were combined with the IORV/SORV initiating event term for the event tree evaluation.

The quantification of the 177 accident sequences results in a core damage frequency of 1.75E-5/year from all internal initiating events and internal flooding. The dominant contributor to core damage is the station blackout scenarios which account for approximately 67 percent of the total core damage frequency. Table 3.3.7-1 lists the 72 sequences with frequencies above 1E-10 and identifies their contribution to the total core damage frequency. Table 3.3.7-2 lists the 72 sequences by initiating event and summarizes each initiating event contribution to the total core damage frequency. The event trees in Section 3.1.2 show the sequence names and the numerical values for the sequence core damage and transfer frequencies.

TABLE 3.3.7-1: Summary Accident Sequence Quantification Results - No Grouping			
Sequence	Frequency	% of TCDF	Sequence Name
T(E)S17	4.51E-006	25.8%	T(E)DGU(1)REC
T(E)S15	3.51E-006	20.1%	T(E)DGREC
T(E)S19	2.71E-006	15.5%	T(E)DGU(1)U(2)REC
FLD7S02	9.99E-007	5.7%	FLD7W(2)
T(E)S03	9.33E-007	5.3%	T(E)W(1)REC
FLD7S03	6.83E-007	3.9%	FLD7V(1)
FLD14S02	4.55E-007	2.6%	FLD14W(1)
TTS05	4.19E-007	2.4%	TTQW(1)ZW(2)
TSSWS08	4.09E-007	2.3%	TSSWQU(2)V(2)
TCS03	3.06E-007	1.7%	TCW(1)W(2)
FLD6S02	2.75E-007	1.6%	FLD6W(1)
SRS18	2.70E-007	1.5%	SRUX
TTCS17	2.30E-007	1.3%	TTCC(M)C(3)
TSSWS04	2.00E-007	1.1%	TSSWQZW(2)
TTCS16	1.78E-007	1.0%	TTCC(M)AI
TDCS05	1.50E-007	0.9%	TDCQW(1)ZW(2)
AOS14	1.47E-007	0.8%	AOI
TTSWS02	1.22E-007	0.7%	TTSWW(1)
TCASS02	1.13E-007	0.6%	TCASW(1)
FLD14S11	6.57E-008	0.4%	FLD14C
TTS19	5.94E-008	0.3%	TTQUX
TMCS17	5.91E-008	0.3%	TMCC(M)C(3)
T(E)S09	4.52E-008	0.3%	T(E)U(1)U(2)W(1)REC
MSS05	4.51E-008	0.3%	MSQW(1)ZW(2)
S1S15	4.20E-008	0.2%	S1C
TDCS18	4.20E-008	0.2%	TDCC

TABLE 3.3.7-1: Summary Accident Sequence Quantification Results - No Grouping			
Sequence	**Frequency**	**% of TCDF**	**Sequence Name**
FLD6S11	4.09E-008	0.2%	FLD6C
TMCS16	4.00E-008	0.2%	TMCC(M)AI
TFS13	3.28E-008	0.2%	TFUV(1)V(4)
AS08	3.00E-008	0.2%	AD
TFCS17	2.96E-008	0.2%	TFCC(M)C(3)
TTCS20	2.88E-008	0.2%	TTCC(M)R
TTSWS08	2.86E-008	0.2%	TTSWUX
TCNS11	2.64E-008	0.1%	TCNUX
TDCS15	2.06E-008	0.1%	TDCQUX
TFCS16	2.00E-008	0.1%	TFCC(M)AI
T(E)S13	1.99E-008	0.1%	T(E)U(1)U(2)XREC
TTSWS11	1.75E-008	0.1%	TTSWC
TCNS14	1.75E-008	0.1%	TCNC
TCASS11	1.75E-008	0.1%	TCASC
T(E)S06	1.48E-008	0.1%	T(E)U(1)W(1)REC
TMS04	1.45E-008	0.1%	TMW(1)ZW(2)
TCCS17	1.43E-008	0.1%	TCCC(M)C(3)
TSSWS09	1.01E-008	0.1%	TSSWQU(2)X
TCCS16	1.00E-008	0.1%	TCCC(M)AI
S1S03	9.04E-009	0.0%	S1W(1)W(2)
MSS19	9.00E-009	0.0%	MSQUX
T(E)S11	6.72E-009	0.0%	T(E)U(1)U(2)VREC
TMCS20	6.14E-009	0.0%	TMCC(M)R
AS09	4.20E-009	0.0%	AC
TIS08	3.82E-009	0.0%	TIQW(1)ZW(2)
TMS18	3.60E-009	0.0%	TMUX

TABLE 3.3.7-1: Summary Accident Sequence Quantification Results - No Grouping			
Sequence	Frequency	% of TCDF	Sequence Name
TIS22	3.38E-009	0.0%	TIQUX
TICS11	3.04E-009	0.0%	TICCC(3)
AOS15	3.04E-009	0.0%	AOC
TCNS03	3.02E-009	0.0%	TCNW(1)W(2)
TFCS20	2.94E-009	0.0%	TFCC(M)R
TICS10	2.74E-009	0.0%	TICCAI
TSSWS12	2.56E-009	0.0%	TSSWC
TFS04	2.31E-009	0.0%	TFW(1)ZW(2)
TFS14	1.80E-009	0.0%	TFUX
TCS14	9.00E-010	0.0%	TCUX
TCCS20	7.40E-010	0.0%	TCCC(M)R
TTCS19	5.40E-010	0.0%	TTCC(M)M
FLD14S04	5.28E-010	0.0%	FLD14UW(1)
TIS21	4.80E-010	0.0%	TIQUV(1)V(2)V(4)
AS03	2.98E-010	0.0%	AW(1)W(2)
TTSWS04	2.84E-010	0.0%	TTSWUW(1)
S2S22	1.44E-010	0.0%	S2QUX
AOS03	1.29E-010	0.0%	AOW(1)W(2)
FLD14S07	1.16E-010	0.0%	FLD14UV(1)V(4)
FLD6S04	1.08E-010	0.0%	FLD6UW(1)

TABLE 3.3.7-2: Summary Accident Sequence Quantification Results - Grouped by Initiating Event			
Sequence	Frequency	% of TCDF	Sequence Name
T(E) Sum = 1.18E-005		67.2%	
T(E)S17	4.51E-006	25.8%	T(E)DGU(1)REC
T(E)S15	3.51E-006	20.1%	T(E)DGREC
T(E)S19	2.71E-006	15.5%	T(E)DGU(1)U(2)REC
T(E)S03	9.33E-007	5.3%	T(E)W(1)REC
T(E)S09	4.52E-008	0.3%	T(E)U(1)U(2)W(1)REC
T(E)S13	1.99E-008	0.1%	T(E)U(1)U(2)XREC
T(E)S06	1.48E-008	0.1%	T(E)U(1)W(1)REC
T(E)S11	6.72E-009	0.0%	T(E)U(1)U(2)VREC
FLD7 Sum = 1.68E-006		9.6%	
FLD7S02	9.99E-007	5.7%	FLD7W(2)
FLD7S03	6.83E-007	3.9%	FLD7V(1)
TSSW Sum = 6.21E-007		3.6%	
TSSWS08	4.09E-007	2.3%	TSSWQU(2)V(2)
TSSWS04	2.00E-007	1.1%	TSSWQZW(2)
TSSWS09	1.01E-008	0.1%	TSSWQU(2)X
TSSWS12	2.56E-009	0.0%	TSSWC
FLD14 Sum = 5.22E-007		3.0%	
FLD14S02	4.55E-007	2.6%	FLD14W(1)
FLD14S11	6.57E-008	0.4%	FLD14C
FLD14S04	5.28E-010	0.0%	FLD14UW(1)
FLD14S07	1.16E-010	0.0%	FLD14UV(1)V(4)
TT Sum = 4.78E-007		2.7%	
TTS05	4.19E-007	2.4%	TTQW(1)ZW(2)
TTS19	5.94E-008	0.3%	TTQUX
TTC Sum = 4.37E-007		2.5%	

TABLE 3.3.7-2: Summary Accident Sequence Quantification Results - Grouped by Initiating Event

Sequence	Frequency	% of TCDF	Sequence Name
TTCS17	2.30E-007	1.3%	TTCC(M)C(3)
TTCS16	1.78E-007	1.0%	TTCC(M)AI
TTCS20	2.88E-008	0.2%	TTCC(M)R
TTCS19	5.40E-010	0.0%	TTCC(M)M
FLD6 Sum = 3.16E-007		1.8%	
FLD6S02	2.75E-007	1.6%	FLD6W(1)
FLD6S11	4.09E-008	0.2%	FLD6C
FLD6S04	1.08E-010	0.0%	FLD6UW(1)
TC Sum = 3.07E-007		1.8%	
TCS03	3.06E-007	1.7%	TCW(1)W(2)
TCS14	9.00E-010	0.0%	TCUX
SR Sum = 2.70E-007		1.5%	
SRS18	2.70E-007	1.5%	SRUX
TDC Sum = 2.13E-007		1.2%	
TDCS05	1.50E-007	0.9%	TDCQW(1)ZW(2)
TDCS18	4.20E-008	0.2%	TDCC
TDCS15	2.06E-008	0.1%	TDCQUX
TTSW Sum = 1.69E-007		1.0%	
TTSWS02	1.22E-007	0.7%	TTSWW(1)
TTSWS08	2.86E-008	0.2%	TTSWUX
TTSWS11	1.75E-008	0.1%	TTSWC
TTSWS04	2.84E-010	0.0%	TTSWUW(1)
AO Sum = 1.50E-007		0.9%	
AOS14	1.47E-007	0.8%	AOI
AOS15	3.04E-009	0.0%	AOC
AOS03	1.29E-010	0.0%	AOW(1)W(2)

TABLE 3.3.7-2: Summary Accident Sequence Quantification Results - Grouped by Initiating Event			
Sequence	**Frequency**	**% of TCDF**	**Sequence Name**
TCAS Sum = 1.30E-007		0.7%	
TCASS02	1.13E-007	0.6%	TCASW(1)
TCASS11	1.75E-008	0.1%	TCASC
TMC Sum = 1.05E-007		0.6%	
TMCS17	5.91E-008	0.3%	TMCC(M)C(3)
TMCS16	4.00E-008	0.2%	TMCC(M)AI
TMCS20	6.14E-009	0.0%	TMCC(M)R
MS Sum = 5.41E-008		0.3%	
MSS05	4.51E-008	0.3%	MSQW(1)ZW(2)
MSS19	9.00E-009	0.0%	MSQUX
TFC Sum = 5.25E-008		0.3%	
TFCS17	2.96E-008	0.2%	TFCC(M)C(3)
TFCS16	2.00E-008	0.1%	TFCC(M)AI
TFCS20	2.94E-009	0.0%	TFCC(M)R
S1 Sum = 5.10E-008		0.3%	
S1S15	4.20E-008	0.2%	S1C
S1S03	9.04E-009	0.1%	S1W(1)W(2)
TCN Sum = 4.69E-008		0.3%	
TCNS11	2.64E-008	0.2%	TCNUX
TCNS14	1.75E-008	0.1%	TCNC
TCNS03	3.02E-009	0.0%	TCNW(1)W(2)
TF Sum = 3.69E-008		0.2%	
TFS13	3.28E-008	0.2%	TFUV(1)V(4)
TFS04	2.31E-009	0.0%	TFW(1)ZW(2)
TFS14	1.80E-009	0.0%	TFUX
A Sum = 3.45E-008		0.2%	

TABLE 3.3.7-2: Summary Accident Sequence Quantification Results - Grouped by Initiating Event			
Sequence	**Frequency**	**% of TCDF**	**Sequence Name**
AS08	3.00E-008	0.2%	AD
AS09	4.20E-009	0.0%	AC
AS03	2.98E-010	0.0%	AW(1)W(2)
TCC Sum = 2.51E-008		0.1%	
TCCS17	1.43E-008	0.1%	TCCC(M)C(3)
TCCS16	1.00E-008	0.1%	TCCC(M)AI
TCCS20	7.40E-010	0.0%	TCCC(M)R
TM Sum = 1.81E-008		0.1%	
TMS04	1.45E-008	0.1%	TMW(1)ZW(2)
TMS18	3.60E-009	0.0%	TMUX
TI Sum = 7.69E-009		0.0%	
TIS08	3.82E-009	0.0%	TIQW(1)ZW(2)
TIS22	3.38E-009	0.0%	TIQUX
TIS21	4.80E-010	0.0%	TIQUV(1)V(2)V(4)
TIC Sum = 5.78E-009		0.0%	
TICS11	3.04E-009	0.0%	TICCC(3)
TICS10	2.74E-009	0.0%	TICCAI
S2 Sum = 1.44E-010		0.0%	
S2S22	1.44E-010	0.0%	S2QUX
TTC2 Sum = 0.00E+000		0.0%	
IS Sum = 0.00E+000		0.0%	

3.3.8 Internal Flooding Analysis

See Section 3.1.2.4.7 for internal flooding analysis discussion.

3.4 Results and Screening Process

Following the identification of the dominant events and sequences, the next step is to determine which of them are important, if any, and against what criteria to compare them to make that determination. Three criteria are used in assessing the importance or significance of the results.

- Generic Letter 88-20, Supplement 1 and the associated NUREG-1335, require that sequences with a frequency of > 1E-7 per year or that contribute greater than 5% to the CDF be reported.

- NRC's safety goal policy states that plants must have less than 1E-4 per year frequency for a core damaging event.

- NEI (formally NUMARC) has published a set of evaluation criteria in NUMARC 91-04 to assist in determining the level of action that should be taken for a given sequence group frequency.

The outcomes of this assessment are:

- The reportability criteria (sequences with a frequency of > 1E-7 per year or that contribute greater than 5% to the CDF) is met. These sequences are listed in Section 3.4.1 below.

- The WNP-2 IPE has demonstrated a CDF of 1.75E-5 per year for internal events which provides sufficient margin to 1E-4 per year that inclusion of external events will still meet the safety goal.

- Regarding the NUMARC criteria, the Loss of Offsite Power sequence group is the only group that does not meet the "Less than 1E-6: No specific action required" criteria. The flood in the Turbine Building is borderline and is included with the Loss of Offsite Power in the criteria "1E-5 to 1E-6: Ensure Severe Accident Management Guideline is in place." Due to the low frequency of CDF for WNP-2, the percentage criteria are not applied to the results.

The impact of these criteria on the WNP-2 results and sensitivity studies are evident in the list of recommendations presented in Section 6.0.

3.4.1 Application of Generic Letter Screening Criteria

The calculated core damage frequency is 1.75 x 10^{-5}/year. Systemic sequences with a frequency greater than 1 x 10^{-7}/year are shown in Table 3.4.1-1 and of these, the five sequences which contribute more than 5% to total CDF are discussed in detail below. A list of initiating events and their contribution to core damage frequency is given in Table 3.4.1-2 and displayed in Figure 3.4.1-1. The only two initiating events whose sequences sum to greater than 5% of the total core damage frequency are:

- Loss of Offsite Power
- TSW Flood in the Reactor Building (FLD7).

Of the events listed in Table 3.4.1-1, five (5) contribute greater than 5% to the CDF. These five are listed below and discussed in Sections 3.4.1.1 - 3.4.1.5. The remaining events with a frequency greater than 1E-7 per year are briefly discussed in Section 3.4.1.6.

WNP-2 IPE DOMINANT SEQUENCES (>5% contribution to CDF)

BRIEF SEQUENCE DESCRIPTION	FREQUENCY	% OF CDF
Station Blackout with HPCS failure and failure to recover offsite power in four hours	4.51E-06	25.8
Station Blackout with HPCS operating but failure to recover offsite power in ten hours	3.51E-06	20.1
Station Blackout with HPCS, RCIC failure and failure to recover offsite power in thirty minutes	2.71E-06	15.5
TSW flood that inops RHR and a failure of containment venting	9.99E-07	5.7
Loss of Offsite Power with DG 1 or 2 available, HPCS available, with loss of RHR cooling and failure to recover offsite power in ten hours	9.33E-07	5.3

3.4.1.1 Station Blackout Lasting Greater Than Four Hours (Sequence TE-S17)

For the WNP-2 IPE, Station Blackout (SBO) scenarios are defined as having a loss of offsite power with coincident failures of Emergency Diesel Generators EDG-1 and EDG-2 which results in loss of power to 4Kv buses in Division 1 and Division 2.

Sequence TE-S17 results from loss of offsite power, station blackout and an additional failure of either HPCS or EDG-3. RCIC system injects successfully until the batteries become depleted or containment pressure reaches approximately 20 psig. All balance of plant and containment heat removal systems remain unavailable until offsite power is restored. If offsite power is not restored before RCIC is rendered unavailable by battery depletion or high containment pressure, the resulting loss of injection will initiate core uncovery and consequential core damage.

3.4.1.2 Station Blackout Lasting Greater Than Ten Hours (Sequence TE-S15)

Sequence TE-S15 represents a long term station blackout sequence, i.e., loss of offsite power followed by coincident failures of DG 1 and DG 2. High pressure core spray (HPCS) is able to provide injection since its operation is not limited by four hour battery lifetime and it is able to draw water from the condensate storage tank. After approximately ten hours, the CST will be depleted and continued HPCS operation is contingent upon its being successfully realigned to the suppression pool. However, because successful injection for ten hours without decay heat removal means that suppression pool water temperature will exceed the design temperature of the HPCS pump, it is assumed to fail and initiate core uncovery and core damage. To arrest the sequence before core damage, offsite power must be recovered before the switchover from CST to suppression pool, about ten hours after the initiating event.

3.4.1.3 Station Blackout Without Injection (Sequence TE-S19)

Sequence TE-S19 represents a station blackout sequence which differs from TE-S17 only in the fact that RCIC is unsuccessful, either for mechanical reasons or from lack of room cooling because the normal source of room cooling, standby service water, is unavailable following loss of 4Kv. Inadequate RCIC room cooling can be averted if the operators are able to open the RCIC pump room door within 30 minutes. Unless offsite power is recovered within 30 minutes of the failure of RCIC, the total loss of injection will result in core damage.

3.4.1.4 Internal Flood With Failure of Containment Venting (Sequence FLD7-S02)

Sequence FLD7-S02 is initiated by a TSW piping break in the Reactor Building which disables all ECCS pumps except LPCS. Reactor SCRAM is successful and reactor pressure is initially controlled successfully with the SRVs. ADS functions successfully so that LPCS operation can be initiated to provide core cooling. However, failure to vent the containment by the operators allows the containment pressure to reach 62 psig, at which point the ADS valves reclose. The RCS repressurizes above the shut-off head for LPCS and initiates loss of RCS injection, core uncovery and subsequent core damage.

3.4.1.5 Loss of Offsite Power With Loss of Suppression Pool Cooling (Sequence TE-S03)

This is a loss of offsite power scenario where one diesel fails and one diesel operates. HPCS operates successfully, however, the RHR loop associated with the operable diesel is not available due to test and maintenance. Like the station blackout sequence described in section 3.4.1.2, with HPCS available and no suppression pool cooling, offsite power must be recovered in ten hours. If offsite power is not recovered in ten hours, core uncovery is initiated and core damage follows.

3.4.1.6 Remaining Sequences With Frequency of Occurrence $>10^{-07}$

The five sequences, described above, which contribute greater than 5% each to the CDF account for approximately 71% of the total core damage frequency. The sum of the nineteen sequences with a frequency of occurrence greater than 1E-07 (see Table 3.4.1-1) accounts for 95% of the total core damage frequency. Each of these sequences, with frequency greater than 1E-07 but do not contribute more than 5% to the total core damage frequency, are briefly described as follows:

- Reactor Building Flood With LPCS Failure (FLD7-S03) - This is a flood in the reactor building from TSW piping break that disables all ECCS pumps except LPCS. ADS is implemented successfully, but the LPCS is unavailable either because the pump/motor fails to start or because the isolation valve fails to open. Since the flood disables all other sources of injection, failure of LPCS results in a complete loss of RCS injection, core uncovery is initiated and core damage results.

- Turbine Building Flood With Loss of Containment Heat Removal (FLD14-S02) - This is a balance of plant piping break that floods turbine building causing loss of the power conversion system (PCS). The reactor successfully scrams, HPCS is providing inventory with decay heat being rejected to the suppression pool through the SRVs. The containment heat removal systems fail resulting in eventual containment failure which initiates loss of injection and core damage.

- Turbine Trip Transient With Loss of Containment Heat Removal (TT-S05) - This is a turbine trip plant transient in which the PCS is unavailable and cannot be recovered prior to containment failure. Containment failure occurs for this sequence since RHR is unavailable and containment venting is not initiated. Containment fails from overpressurization and leads to consequential failure of the injection system resulting in core uncovery and core damage.

- Loss of Standby Service Water With Loss of Condensate (TSSW-S08) - A loss of standby service water does not initiate SCRAM, however, it results in a manual shutdown due to Tech Specs. The unavailability of the condensate system results in loss of the PCS and without standby service water for RHR cooling, the containment will overpressurize resulting in consequential failure of the injection system. Core uncovery and core damage result.

- Loss of Condenser With Loss of RHR and Venting (TC-S03) - This sequence results in a SCRAM with injection available and pressure control via the SRVs. However, loss of the condenser means loss of the PCS. With the unavailability of RHR and venting, the containment will overpressurize, resulting in consequential failure of the injection systems. Core uncovery and core damage results.

- Turbine Building Flood With Loss of Containment Heat Removal (FLD6-S02) - This is a balance of plant piping break that floods turbine building causing loss of the power conversion system (PCS). The reactor successfully scrams, HPCS is providing inventory with decay heat being rejected to the suppression pool through the SRVs. The containment heat removal systems fail resulting in eventual containment failure which initiates loss of injection and core damage.

- Instrument Line Break With Loss of HPCS and ADS (SR-S18) - It is assumed that the instrument line break results in consequential failure of the RCIC and feedwater systems. Therefore, two of the high pressure injection sources are failed, as well as, the PCS for decay heat removal. With failure of the HPCS and the ADS function, the reactor inventory decreases and core damage results.

- ATWS With Failure of Standby Liquid Control (TTC-S17) - This sequence is an ATWS event in which a turbine trip from 100% reactor power is followed by mechanical failure of the reactor SCRAM system. Recirculation pump trip is successful and the SRVs open and reclose to limit RCS pressure. Attempts to control reactor power by actuation of the SLC system are unsuccessful, either because the operators are unable to respond successfully with the available time or one of the two SLC pump's fail to start. Both SLC pumps are required to inject boron to successfully bring the reactor subcritical. It is assumed that the HPCS and RCIC cannot provide adequate flow to maintain adequate core cooling, therefore, the core uncovers and consequential damage results.

- Loss of Standby Service Water With Loss of Containment Heat Removal (TSSW-S04) - This sequence is similar to the loss of SW with loss of feedwater sequence described above. In this sequence, the decay heat removal function of the PCS is unavailable because the MSIVs cannot be reopened and containment venting fails. This results in eventual containment failure which initiates loss of the injection systems and consequential core damage.

- ATWS With Failure to Inhibit ADS (TTC-S16) - This sequence is an ATWS event in which a turbine trip from 100% reactor power is followed by mechanical failure of the reactor SCRAM system. Recirculation pump trip is successful and the SRVs open and reclose to limit RCS pressure. Attempts to control reactor power by actuation of the SLC system are successful. However, ADS is not inhibited when the reactor water level reaches L1. The rapid depressurization and subsequent flooding of the core by low pressure injection systems results in the introduction cold water and rapid dilution of boron from the core region. This results in a transient increase in core reactivity and a return to power. The failure assumed in the analysis is failure to inhibit ADS and failure to maintain level above L1 with the high pressure injection systems. It is conservatively assumed that the containment fails resulting in injection system failure and consequential core damage.

- Loss of DC With Heat Removal Failure (TDC-S05) - This sequence is initiated with the loss of one divisional DC bus failure. High pressure injection is successful, but the MSIVs fail close rendering the PCS unavailable. The alternate train RHR fails, attempts to reopen the MSIVs fail, and venting is unsuccessful. The containment fails resulting in injection system failure and consequential core damage.

- Large LOCA Outside Containment With Failure To Isolate (AO-S14) - A large steam line outside primary containment ruptures releasing steam to the reactor building. The reactor SCRAMs but the ruptured steam line is not isolated. The continued steam release to the reactor building initiates failure of all injection systems. This results in core uncovery and core damage.

- Loss of Plant Service Water with Loss of RHR (TTSW-S02) - The loss of plant service water initiates an indirect SCRAM and eventually loss of the PCS. Injection sources are available and pressure control is available. With loss of RHR, the containment overpressurizes resulting in consequential failure of the injection systems. Core uncovery and core damage results.

- Loss of Control and Service Air With RHR Failure (TCAS-S02) - On loss of CAS, the outboard MSIVs close, initiating a SCRAM. Air operated feedwater startup valves fail as is on loss of CAS. The feedwater system is unavailable for injection and the condenser is unavailable for decay heat removal. This sequence assumes RHR is unavailable for decay heat removal and venting is not available. This leads to containment failure resulting in injection system failure and consequential core damage.

TABLE 3.4.1-1
Summary Accident Sequence Quantification Results > 1.0E-07/year

Sequence	Frequency	% of TCDF	Sequence Name
TE-S17	4.51E-006	25.8%	T(E)DGU(1)REC
TE-S15	3.51E-006	20.1%	T(E)DGREC
TE-S19	2.71E-006	15.5%	T(E)DGU(1)U(2)REC
FLD7-S02	9.99E-007	5.7%	FLD7W(2)
TE-S03	9.33E-007	5.3%	T(E)W(1)REC
FLD7-S03	6.83E-007	3.9%	FLD7V(1)
FLD14-S02	4.55E-007	2.6%	FLD14W(1)
TT-S05	4.19E-007	2.4%	TTQW(1)ZW(2)
TSSW-S08	4.09E-007	2.3%	TSSWQU(2)V(2)
TC-S03	3.06E-007	1.7%	TCW(1)W(2)
FLD6-S02	2.75E-007	1.6%	FLD6W(1)
SR-S18	2.70E-007	1.5%	SRUX
TTC-S17	2.30E-007	1.3%	TTCC(M)C(3)
TSSW-S04	2.00E-007	1.1%	TSSWQZW(2)
TTC-S16	1.78E-007	1.0%	TTCC(M)AI
TDC-S05	1.50E-007	0.9%	TDCQW(1)ZW(2)
AO-S14	1.47E-007	0.8%	AOI
TTSW-S02	1.22E-007	0.7%	TTSWW(1)
TCAS-S02	1.13E-007	0.6%	TCASW(1)

TABLE 3.4.1-2
Summary Accident Sequence Quantification Results - By Initiator

Sequence/Frequency	% of TCDF	Sequence Name
TE Sum = 1.18E-005	67.2%	Loss of Offsite Power
FLD7 Sum = 1.68E-006	9.6%	Flood in Reactor Building
TSSW Sum = 6.21E-007	3.6%	Loss of Standby Service Water
FLD14 Sum = 5.22E-007	3.0%	Flood in Turbine Bldg/CW House
TT Sum = 4.78E-007	2.7%	Turbine Trip Transient
TTC Sum = 4.37E-007	2.5%	Turbine Trip ATWS, 100% Power
FLD6 Sum = 3.16E-007	1.8%	Flooding in Turbine Building
TC Sum = 3.07E-007	1.8%	Loss of Condenser Transient
SR Sum = 2.70E-007	1.5%	Water Level Instr Line Break
TDC Sum = 2.13E-007	1.2%	Loss of Div 2 DC
TTSW Sum = 1.69E-007	1.0%	Loss of Plant Service Water
AO Sum = 1.50E-007	0.9%	LOCA Outside Primary Containment
TCAS Sum = 1.30E-007	0.7%	Loss of Control and Service Air
TMC Sum = 1.05E-007	0.6%	MSIV Closure ATWS
MS Sum = 5.41E-008	0.3%	Manual Shutdown Event
TFC Sum = 5.25E-008	0.3%	Loss of Feedwater ATWS
S1 Sum = 5.10E-008	0.3%	Medium Break LOCA
TCN Sum = 4.69E-008	0.3%	Loss of Containment Nitrogen
TF Sum = 3.69E-008	0.2%	Loss of Feedwater Transient
A Sum = 3.45E-008	0.2%	Large Break LOCA
TCC Sum = 2.51E-008	0.1%	Loss of Condenser ATWS
TM Sum = 1.81E-008	0.1%	MSIV Closure Transient
TI Sum = 7.69E-009	0.0%	IORV/SORV Transient
TIC Sum = 5.78E-009	0.0%	SORV ATWS
S2 Sum = 1.44E-010	0.0%	Small Break LOCA
TTC2 Sum = 0.00E+000	0.0%	Turbine Trip ATWS, 25% Power
IS Sum = 0.00E+000	0.0%	Interfacing System LOCA

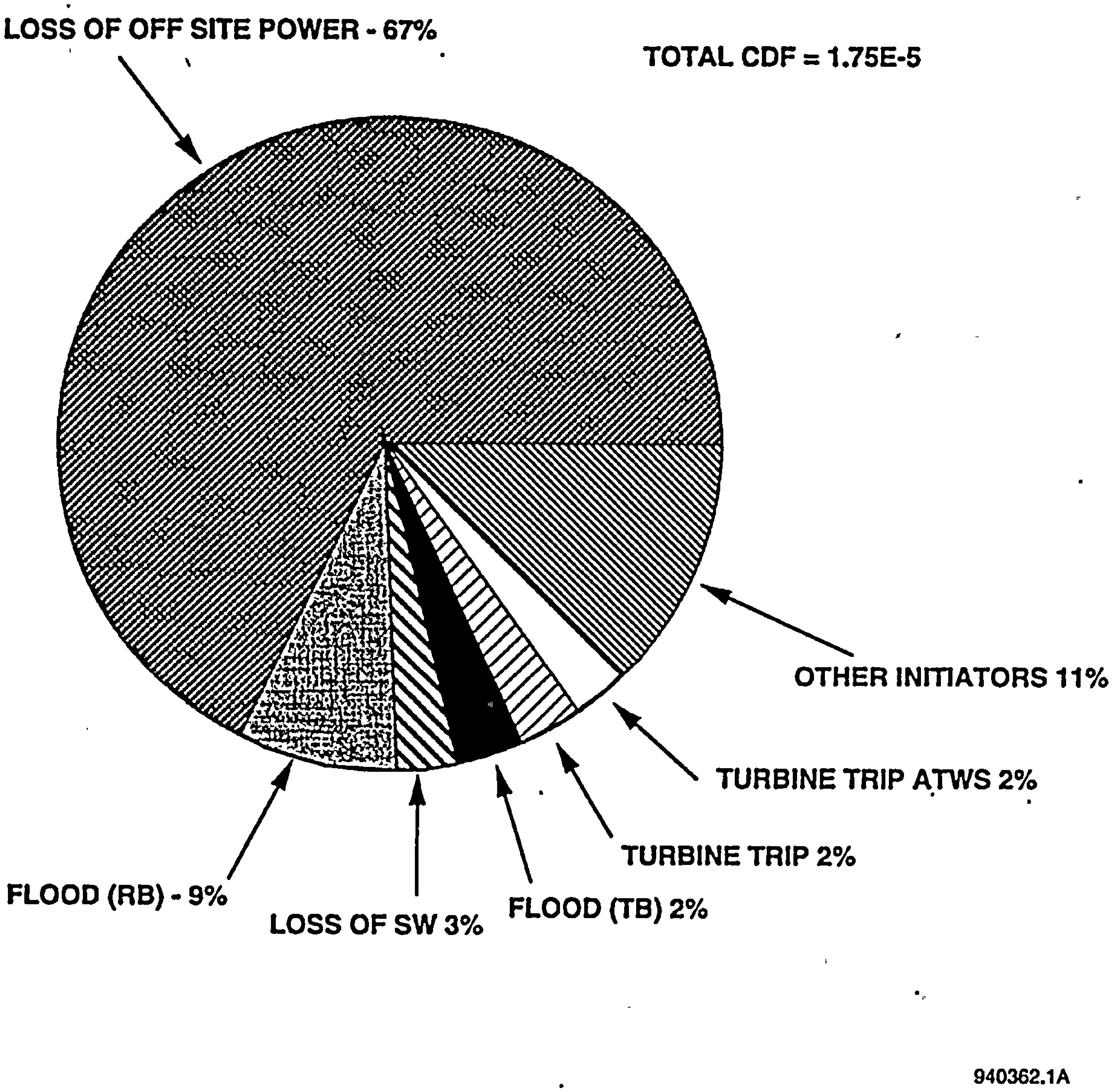

FIGURE 3.4.1-1 INITIATORS CONTRIBUTION TO CDF

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3.4.2 Vulnerability Screening

The WNP-2 Individual Plant Examination has identified no vulnerabilities in the WNP-2 design or operation. For WNP-2, vulnerability screening is based on:

- sequence groups >1E-6 that require modifications per NUMARC 91-04 guidelines,
- total CDF must be within the NRC's safety goal of 1E-4,
- sequences that indicate a plant specific feature that is an outlier to comparable BWR PRAs.

None of the sequence groups indicate a frequency that would require modification to WNP-2 hardware or procedures per the NUMARC guidelines. The Loss of Offsite Power is addressed in Station Blackout Procedures and the recommendations on insights from other sequences in the 1E-5 to 1E-6 range will contribute to the BWROG development of severe accident management guidelines. The core damage frequency of 1.75E-5 per year is well within the NRC's safety goal and provides ample margin to accommodate the external events contribution which is currently in progress. Several comparable BWR PRAs have been examined and WNP-2 does not exhibit any plant specific feature that could be considered an outlier. Therefore, it is concluded that WNP-2's IPE has not identified any vulnerabilities.

Table 3.3.7-1 shows the sequences with a frequency greater than 1E-08, i.e., contribute 0.1% or greater to the total core damage frequency. The functional failure headings for each sequence is also given. Each sequence is composed of basic events and the importance of a basic event is proportional to the number of sequences it impacts, as well as, its magnitude. By studying the characteristics of the basic event importance, it can be determined whether or not the basic event should be considered a vulnerability and it can indicate which basic events are most suitable for sensitivity analysis.

Tables 3.4.2-1 and 3.4.2-2 present the results of the importance measure of the basic events used in the Level 1 analysis. Table 3.4.2-1 is ordered by the Fussel-Vesely Importance (or, equivalently,Risk Reduction Worth), whereas, Table 3.4.2-2 orders the basic events by Risk Achievement Worth. The top fifty basic events are given for each table.

The Fussel-Vesely Importance/Risk Reduction Worth (FVI/RRW) is indicative of those basic events whose decrease in unavailability or probability of occurrence would most decrease the core damage frequency. The top twenty events in the list in Table 3.4.2-1 represent those events which could decrease the core damage frequency by 5% or more. Basic events below those top twenty would have less than 5% impact even if they never failed. Inspection of Table 3.4.2-1 shows that, excluding the initiating events, the top twenty basic events are primarily human recovery actions or human reliability events. The basic events in this list that are not related to human actions are those associated with diesel generator and AC circuit breaker failures.

The Risk Achievement Worth (RAW) is indicative of those basic events whose increase in unavailability or probability of occurrence would most increase the core damage frequency. Inspection of Table 3.4.2-2 shows that, excluding the initiating events, the top twenty basic events by RAW are common cause failures. Using the greek letter method for calculating common cause failure rates is conservative and there is little likelihood these values would increase. The one human failure basic event in the top events is also a common cause type event in that generally there are two RHR loops available for suppression pool cooling to remove decay heat.

The basic events that are in both FV/RRW and RAW list's top thirty are:

- the initiating events: loss of standby service water and flooding from a TSW piping break, and
- common cause failures: mechanical failure to scram rods and all three diesels fail.

On a systems basis, those that have components that appear on both lists are primarily: AC (including diesel generators), DC, and RHR/SW.

TABLE 3.4.2-1
Importance Analysis Sorted by Fussell-Vesely Ranking

Format: Rank	EVENT NAME / * EVENT DESCRIPTION *	POINT EST.	F-V IMPORT	RSK ACMT	RISK RED
1	TE	2.460E-002	6.721E-001	27.65	3.050
	* LOSS OF OFF-SITE POWER FREQUENCY IN EVENTS PER YEAR *				
2	U2-SUCC	9.030E-001	2.579E-001	1.03	1.347
	* 1 - (U2-SBO): SUCCESSFUL RCIC UNDER SBO *				
3	NRAC4	1.440E-001	2.579E-001	2.53	1.347
	* NON RECOVERY OF OFFSITE POWER WITHIN 4 HOURS *				
4	NRAC10	2.960E-002	2.577E-001	9.45	1.347
	* NON RECOVERY OF OFFSITE POWER WITHIN 10 HOURS *				
5	EACEDG123---C3LL	8.210E-004	2.533E-001	309.33	1.339
	* COMMON CAUSE FAIL. OF EDG-1,-2, AND -3 *				
6	U1-SUCC	8.830E-001	2.009E-001	1.03	1.251
	* 1 - (U1-SBO): SUCCESSFUL HPCS UNDER SBO *				
7	EACENG-EDG-1S4D1	2.955E-002	1.641E-001	6.39	1.196
	* EDG-1 FAILS TO RUN 10 HOURS *				
8	NRAC30M	6.220E-001	1.566E-001	1.10	1.186
	* NO RECOVERY OF OFFSITE POWER WITHIN 30 MINUTES *				
9	EACENG-EDG-2S4D2	2.955E-002	1.561E-001	6.13	1.185
	* EDG-2 FAILS TO RUN FOR 10 HOURS *				
10	CF-FAILS-INJECT	3.300E-001	1.461E-001	1.30	1.171
	* INJECTION FAILS DUE TO CONTAINMENT FAILURE *				
11	EACENG-EDG-1W2D1	2.129E-002	1.171E-001	6.38	1.133
	* EDG-1 FAILS TO START *				
12	EACENG-EDG-2W2D2	2.129E-002	1.114E-001	6.12	1.125
	* EDG-2 FAILS TO START *				
13	FLD7	1.600E-005	9.618E-002	6012.35	1.106
	* FREQUENCY OF TSW FLOOD IN RB WITHOUT OPERATOR SHUT *				
14	VENTFAIL	6.000E-002	7.674E-002	2.20	1.083
	* OPERATOR FAILS TO VENT CONTAINMENT *				
15	RCIHUMNRCOOLH3LL	5.000E-002	6.489E-002	2.23	1.069
	* OPERATOR FAILS TO OPEN PUMP ROOM DOOR & SU FANS PE *				

Format: Rank	EVENT NAME / * EVENT DESCRIPTION *	POINT EST.	F-V IMPORT	RSK ACMT	RISK RED
16	EACCB--CBB-7B2L1	1.117E-002	5.996E-002	6.31	1.064
	* CIRCUIT BREAKER B-7 FAILS TO OPEN PPM 7.4.8.1.1.1. *				
17	EACCB--CBB-8B2L2	1.117E-002	5.701E-002	6.05	1.060
	* CIRCUIT BREAKER B-8 FAILS TO OPEN PPM 7.4.8.1.1.1. *				
18	U1-S	8.830E-001	5.339E-002	1.01	1.056
	* 1 - (U1-LOOP): SUCCESS OF HPCS W/ LOOP *				
19	RHRBTM	6.310E-003	4.871E-002	8.67	1.051
	* TM UNAVAILABILITY OF RHR-B *				
20	EACENG-EDG-3S4D3	2.955E-002	4.674E-002	2.53	1.049
	* EDG-3 FAILS TO RUN FOR 10 HOURS *				
21	TSSW	1.830E-004	3.554E-002	95.17	1.037
	* LOSS OF STANDBY SERVICE WATER FREQUENCY,EVENTS/YR *				
22	CM	4.000E-006	3.547E-002	8867.86	1.037
	* MECHANICAL FAILURE OF SCRAM SYSTEM *				
23	RCIP-TD----1R2LL	2.428E-002	3.330E-002	2.34	1.034
	* RCIC-P-1 (TURB. PMP VALVE & GOVERNOR) FAILS TO START *				
24	EACENG-EDG-3W2D3	2.129E-002	3.305E-002	2.52	1.034
	* EDG-3 FAILS TO START *				
25	FLD14	4.690E-003	2.983E-002	7.33	1.031
	* TYPE 14 FLOOD FREQUENCY PER YEAR *				
26	TT	3.300E+000	2.756E-002	0.98	1.028
	* WNP-2 TURBINE TRIP FREQUENCY IN EVENTS PER YEAR *				
27	TTC	2.700E+000	2.501E-002	0.98	1.026
	* WNP-2 TURBINE TRIP ATWS FREQUENCY IN EVENTS PER YEAR *				
28	HPSP-MD----1R2LL	1.264E-002	2.437E-002	2.90	1.025
	* HPCS-P-1 MOTOR DRIVEN PUMP FAILS TO START *				
29	DEPLETE-B2-1	3.000E-001	2.305E-002	1.05	1.024
	* B2-1 BATTERY DEPLETES AFTER BATT. CHARGER FAILS *				
30	EDCC2-1REPAIR	3.250E-001	2.284E-002	1.05	1.023
	* REPAIR OF BATTERY CHARGER C2-1 *				
31	EAC----ASHE-G3D1	3.760E-003	2.225E-002	6.89	1.023
	* LOSS OF POWER TO TR-S FROM ASHE SUBSTATION *				
32	ADSHUMNSTARTH3LL	2.660E-003	1.922E-002	8.20	1.020
	* OPERATOR DOES NOT INITIATE ADS *				

Format: Rank	EVENT NAME / * EVENT DESCRIPTION *	POINT EST.	F-V IMPORT	RSK ACMT	RISK RED
33	FLD6	2.920E-003	1.806E-002	7.17	1.018
	* TYPE 6 FLOOD FREQUENCY PER YEAR *				
34	HPCSTM	9.810E-003	1.805E-002	2.82	1.018
	* HPCS IN TESTING, PREVENTIVE MAINTENANCE OR REPAIR *				
35	TC	5.000E-002	1.755E-002	1.33	1.018
	* LOSS OF CONDENSER FREQUENCY IN EVENTS PER YEAR *				
36	HPSV-MO----4P2LL	8.505E-003	1.570E-002	2.83	1.016
	* HPCS-V-4 MO GATE VALVE DOES NOT OPEN *				
37	SR	1.000E-002	1.544E-002	2.53	1.016
	* WNP-2 RX LEVEL INSTRUMENT LINE BRK FREQUENCY,EVENT *				
38	RHRHUMNSP-COOLLL	3.000E-005	1.542E-002	514.82	1.016
	* OPERATOR FAILS TO INITIATE SP COOLING OR SPRAY *				
39	RRAFC11	1.341E-003	1.486E-002	12.06	1.015
	* MODULE RRAFC11 RRA-FC-11 RELATED COMPONENTS FAIL *				
40	RPSRODS	1.400E-005	1.479E-002	1057.64	1.015
	* FAILURE TO SCRAM THE REACTOR *				
41	RRAFC02	1.346E-003	1.314E-002	10.75	1.013
	* MODULE RRAFC02 RRA-FC-02 RELATED COMPONENTS FAIL *				
42	TDC	3.000E-003	1.216E-002	5.04	1.012
	* WNP-2 LOSS OF DIV.2 DC FREQUENCY IN EVENTS PER YEAR *				
43	LPSP-MD----1R2LL	1.264E-002	1.193E-002	1.93	1.012
	* LPCS-P-1 MOTOR/PUMP/COUPLING FAILS TO START *				
44	SW-P-MDSWP1AR2LA	9.905E-004	1.183E-002	12.93	1.012
	* FAILURE OF SSW PUMP MOTOR TO START ON DEMAND, MECH *				
45	SW-P-CCPUMP1CCLL	2.010E-005	1.167E-002	581.45	1.012
	* COMMON CAUSE FAILURE FOR SW-P-1A AND 1B 0.03*(G420) *				
46	RCIV-MO---45P2LL	8.505E-003	1.079E-002	2.26	1.011
	* RCIC-V-45 MECHANICAL FAILURE TO OPEN *				
47	AI	5.000E-002	1.035E-002	1.20	1.010
	* FAILURE TO INHIBIT ADS OR KEEP L1 OR TAF FOR TT ATWS *				
48	TTSW	1.250E-003	9.658E-003	8.72	1.010
	* WNP-2 LOSS OF PLANT SERVICE WATER FREQUENCY,EVENTS *				
49	RHRP-MD---2AR2LL	9.905E-004	9.214E-003	10.29	1.009
	* RHR-P-2A MOTOR DRIVEN PUMP FAILS TO START *				

Format: Rank	EVENT NAME / * EVENT DESCRIPTION *	POINT EST.	F-V IMPORT	RSK ACMT	RISK RED
50	RRAFC10	1.341E-003	8.934E-003	7.65	1.009
	* MODULE RRAFC10 RRA-FC-10 RELATED COMPONENTS FAIL *				

TABLE 3.4.2-2
Importance Analysis Sorted by Risk-Achievement-Worth Ranking

Format: Rank	EVENT NAME / * EVENT DESCRIPTION *	POINT EST.	F-V IMPORT	RSK ACMT	RISK RED
1	EDCC1--ALLBCC2LL	6.000E-008	4.485E-003	74751.67	1.005
	* COMMON CAUSE FAIL. OF BATTERY CHARGERS C1-1,-2,-7 *				
2	CM	4.000E-006	3.547E-002	8867.86	1.037
	* MECHANICAL FAILURE OF SCRAM SYSTEM *				
3	FLD7	1.600E-005	9.618E-002	6012.35	1.106
	* FREQUENCY OF TSW FLOOD IN RB WITHOUT OPERATOR SHUT *				
4	EDCC1--C1127C2LL	3.600E-008	4.648E-005	1292.14	1.000
	* COMMON CAUSE FAIL. OF BATTERY CHARGERS C1-1, C1-2 *				
5	RPSRODS	1.400E-005	1.479E-002	1057.64	1.015
	* FAILURE TO SCRAM THE REACTOR *				
6	SW-P-CCPUMP1CCLL	2.010E-005	1.167E-002	581.45	1.012
	* COMON CAUSE FAILURE FOR SW-P-1A AND 1B 0.03*(G420) *				
7	RHRHUMNSP-COOLLL	3.000E-005	1.542E-002	514.82	1.016
	* OPERATOR FAILS TO INITIATE SP COOLING OR SPRAY *				
8	RHRP-MD-2ABCC3S4	1.120E-005	5.397E-003	482.85	1.005
	* COMMON CAUSE FAILURE TO RUN OF RHR-P-2A,B,C *				
9	RHRP-MD-2ABCC3R2	8.820E-006	4.182E-003	475.09	1.004
	* COMMON CAUSE FAILURE TO START OF RHR-P-2A,B,C *				
10	RHRHXSH--1ABC2LL	5.260E-006	2.356E-003	448.85	1.002
	* COMMON CAUSE FAILURE OF RHR-HX-1A,B *				
11	RHRV-MO-48ABC2LL	2.980E-006	1.161E-003	390.63	1.001
	* COMMON CAUSE FAILURE TO CLOSE RHR-V-48A,B *				
12	SW-V-CH--1ABC2LL	1.260E-006	4.044E-004	321.97	1.000
	* COMMON CAUSE FAIL. OF SW-V-1A AND SW-V-1B *				
13	RHRV-MO-47ABC2LL	1.350E-006	4.200E-004	312.14	1.000
	* COMMON CAUSE FAILURE OF RHR-V-47A,B *				
14	RHRV-MO-4ABCC3LL	1.350E-006	4.200E-004	312.14	1.000
	* COMMON CAUSE FAIL. OF RHR-V-4A,B,C TO REMAIN OPEN *				
15	RHRV-MO--3ABC2LL	1.350E-006	4.200E-004	312.14	1.000
	* COMMON CAUSE FAILURE OF RHR-V-3A,B *				
16	EACEDG123---C3LL	8.210E-004	2.533E-001	309.33	1.339
	* COMMON CAUSE FAIL. OF EDG-1,-2, AND -3 *				

Format: Rank	EVENT NAME	POINT EST.	F-V IMPORT	RSK ACMT	RISK RED
	* EVENT DESCRIPTION *				
17	RHRV-CH31ABCC3LL	1.260E-006	3.759E-004	299.30	1.000
	* COMMON CAUSE FAILURE OF RHR-V-31A,B,C *				
18	SW-VLCCVLV12CCLL	1.490E-007	3.687E-005	248.47	1.000
	* COMON CAUSE FAILURE FOR POND RETRN VLVE SW-12 *				
19	SW-VLCCVALV2CCLL	1.490E-007	3.687E-005	248.47	1.000
	* COMON CAUSE FAILURE FOR DISCHARGE VALVE SW-V2 *				
20	TSSW	1.830E-004	3.554E-002	195.17	1.037
	* LOSS OF STANDBY SERVICE WATER FREQUENCY,EVENTS/YR *				
21	EACCB--7-73-G1D1	1.799E-005	1.903E-003	106.79	1.002
	* CIRCUIT BREAKER 7-73 FTRC *				
22	WMADIS3MC-7FW1D1	1.799E-005	1.903E-003	106.79	1.002
	* FUSED DISCONNECT FROM MC-7F TO WMA-FN-53A FTRC *				
23	EACCB--SL73FG1D1	1.799E-005	1.903E-003	106.79	1.002
	* CIRCUIT BREAKER FROM SL-73 TO MC-7F FTRC *				
24	EACTR--TR773W1D1	4.200E-006	2.875E-004	69.45	1.000
	* TRANSFORMER TR-7-73 FAULT *				
25	EACCB-EDG1-2C2D1	9.250E-005	6.137E-003	67.34	1.006
	* COMMON CAUSE FAIL. OF EDG-1 AND EDG-2 OUTPUT BREAK *				
26	EACSM--SM-7-W1D1	2.100E-007	1.189E-005	57.63	1.000
	* SWITCHGEAR SM-7 FAULT *				
27	EACSL--SL-73W1D1	2.100E-007	1.189E-005	57.63	1.000
	* LOAD CENTER SL-73 FAULT *				
28	EACMC--MC-7FW1D1	2.100E-007	1.189E-005	57.63	1.000
	* MC-7F FAULT *				
29	SL73	2.240E-005	1.077E-003	49.08	1.001
	* SL-73, ITS BREAKER OR ITS TRANSFORMER FAULT *				
30	EACCB--SL73AG1D1	1.799E-005	7.802E-004	44.37	1.001
	* CIRCUIT BREAKER FROM SL-73 TO MC-7A FTRC *				
31	AO	2.170E-004	8.606E-003	40.65	1.009
	* LARGE LOCA OUTSIDE CONTAINMENT FREQUENCY,EVENT *				
32	TE	2.460E-002	6.721E-001	27.65	3.050
	* LOSS OF OFF-SITE POWER FREQUENCY IN EVENTS PER YEAR *				
33	EDCC1--C1-12C2LL	9.600E-008	2.550E-006	27.56	1.000
	* COMMON CAUSE FAIL. OF BATTERY CHARGERS C1-1 AND C1 *				

Format: Rank	EVENT NAME / * EVENT DESCRIPTION *	POINT EST.	F-V IMPORT	RSK ACMT	RISK RED
34	SW-FL--SST3AE4LA	3.600E-005	8.976E-004	25.93	1.001
	* SW PUMP 1A SUCTION STRAINER SW-ST-3A PLUGGED *				
35	CASCRM-TIMRAW4LL	2.061E-004	3.908E-003	19.95	1.004
	* CAS DRYER SKID 'A' TIMER, FAILS *				
36	CASV-CH-238AP1LL	1.666E-004	2.892E-003	18.36	1.003
	* CHECK VALVE CAS-V-238A DOES NOT OPEN *				
37	CASV-CH-240AP1LL	1.666E-004	2.892E-003	18.36	1.003
	* CHECK VALVE CAS-V-240A DOES NOT OPEN *				
38	D	1.000E-004	1.716E-003	18.16	1.002
	* FAILURE OF DRYWELL FLOOR SEAL OR VACUUM BREAKERS *				
39	EDCDISCS1-1AW1LL	1.799E-005	2.991E-004	17.62	1.000
	* FAILURE OF 200 AMP. FUSED DISCONNECT DP-S1-1A *				
40	EDCDISC-S11AW1LL	1.799E-005	2.991E-004	17.62	1.000
	* FAILURE OF TERMINAL FUSED DISCONNECT DP-S1-1A *				
41	CJWV-AOTCV1-W4LL	1.078E-004	1.711E-003	16.87	1.002
	* CJW-TCV-1, WATER MIXING VALVE, FAILS *				
42	CASV-AO235A-W4LL	1.078E-004	1.711E-003	16.87	1.002
	* CAS-V-235A, DRYER INLET,STICKS CLOSED WHEN TIMER S *				
43	CASV-AO233A-W4LL	1.078E-004	1.711E-003	16.87	1.002
	* CAS-V-233A, DRYER INLET,STICKS CLOSED WHEN TIMER S *				
44	CJWV-AOLCV1-W4LL	1.078E-004	1.711E-003	16.87	1.002
	* CJW-LCV-1 FAILS TO PROVIDE MAKEUP WATER *				
45	SL83	2.240E-005	3.422E-004	16.28	1.000
	* SL-83, ITS BREAKER OR ITS TRANSFORMER FAULT *				
46	CASRV--115B-W4LL	7.968E-005	1.189E-003	15.93	1.001
	* CAS-RV-115A FAILS OPEN *				
47	CJWRV-----1BW4LL	7.968E-005	1.189E-003	15.93	1.001
	* AUTO VENT VALVE CJW-AV-1B LOSS OF FUNCTION *				
48	CJWRV--753--W4LL	7.968E-005	1.189E-003	15.93	1.001
	* CJW-RV-753 FAILS OPEN *				
49	CJWRV-----1AW4LL	7.968E-005	1.189E-003	15.93	1.001
	* AUTO VENT VALVE CJW-AV-1A LOSS OF FUNCTION *				
50	CJWRV-----1CW4LL	7.968E-005	1.189E-003	15.93	1.001
	* AUTO VENT VALVE CJW-AV-1C LOSS OF FUNCTION *				

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3.4.3 Decay Heat Removal Evaluation (USI A-45)

NUREG-1289 considers six specific alternative courses of action in reaching resolution of USI A-45:

1. No corrective action,
2. Perform detailed risk assessment,
3. Install various modifications,
4. Install hardpipe containment vent,
5. Install a dedicated hot shutdown DHR system,
6. Install a dedicated cold shutdown DHR system.

In this section, IPE results related to decay heat removal are discussed in detail. Course 2 is taken to reach resolution of USI A-45.

3.4.3.1 Systems Available for DHR

This IPE takes credit for three methods by which decay heat can be removed from the reactor vessel and containment: main condenser, RHR, and containment venting. There are other methods of removing decay heat (e.g., RWCU, containment coolers, free wheeling of RCIC turbine, etc.). Their capabilities to remove all of the decay heat were not evaluated in detail since the three systems analyzed meet the guidelines for resolution of USI A-45.

Main Condenser:

The main condenser is the preferred decay heat removal system during a normal shutdown until reactor pressure drops to 48 psig when RHR shutdown cooling is normally placed in-service. Important support system requirements for the main condenser include the circulating water, steam jet air ejector or mechanical vacuum pumps, offsite power, containment instrument air, control and service air.

RHR:

If the main condenser is unavailable, RHR in any of its four modes can be used to remove decay heat.

RHR suppression pool cooling can be used to remove decay heat from the reactor vessel via the SRVs and the suppression pool. The Emergency Operating Procedures direct operators to use suppression pool cooling to control suppression pool temperature below 90°F. Either offsite or on-site emergency power may be used to operate RHR pumps and valves. The standby service water is the ultimate heat sink for RHR operation.

RHR shutdown cooling can remove decay heat provided that the reactor pressure is below 98 psig. The motive power for the shutdown cooling inboard isolation valve, RHR-V-9, is from Division 2 AC; and that for the shutdown cooling outboard isolation valve, RHR-V-8, is from the Division 1 250V DC. Motive power for both shutdown cooling return valves RHR-V-53A and RHR-V-53B is from Division 1 480V AC. Division 1 AC, Division 2 AC, and Division 1 250V DC must all be available in order to initiate shutdown cooling. Because of commonalities with suppression pool cooling, shutdown cooling may only provide additional redundancy for sequences which result from shutdown cooling valve failures. If an extended period without DHR is postulated, increased containment temperature and pressure conditions may result. In this situation, a group 6 isolation signal resulting from 1.68 psig containment pressure will preclude shutdown cooling from being placed in-service. Shutdown cooling is not credited in this IPE.

On increasing containment pressure, the Emergency Operating Procedures direct operators to use suppression pool sprays before wetwell pressure reaches 8 psig provided suppression pool level is below 51'. When suppression pool pressure exceeds 8 psig or drywell temperature exceeds 340°F, the same procedure directs operators to use drywell sprays provided that the drywell spray initiation limit is not exceeded.

Venting:

Containment heat removal can also be accomplished by venting through Containment Exhaust Purge system using 30" and 24" exhaust butterfly valves. Piping from the containment to the SGT is 'hard'. So is the piping from the SGT to the stack. However, the SGT is 'soft' ductwork. When the containment is vented at 39 psig as directed by the Emergency Operating Procedures, it is very likely that there will be a break in the SGT. Steam, radionuclides, if present, and noncondensible gases will be released to the reactor building. Pressurization of the reactor building will cause the blow-out panels around and over the refueling floor to blow out. Release to the environment will be through the blow-out panels. Required support systems include Divisions 1 and 2 AC, and Control and Service Air (CAS).

3.4.3.2 Thermal-Hydraulic Analysis & ECCS Response

The decay heat fraction as a function of length of irradiation and time from shutdown can be obtained from GE NEDO-24810A. For infinite irradiation time, the decay heat fraction decreases from 5.5% at 10 seconds to 2.6% in 10 minutes, to 0.9% in 10 hours, and to 0.35% in 1 week. The RHR system has two heat exchangers. The design duty of each heat exchanger is 41.6E6 Btu/hr which converts to 12.19 Mw or 0.37% thermal power. These numbers indicate that if the main condenser is not available, the containment temperature and pressure cannot be expected to decrease in less than 10 hours with two RHR heat exchangers operating, or in less than one week with only one RHR heat exchanger operating. The condenser and the turbine bypass valve system, which are designed to take 25% rated steam flow, are able to remove the decay heat without any trouble.

The MAAP code was used to investigate the effect of RHR on containment temperature and pressure rises for turbine trip without bypass but with core inventory makeup. For the case of both RHR heat exchangers operating, the results indicate that wetwell gas temperature and pressure start to level off after approximately 10 hours into the turbine trip accident. Maximum wetwell gas temperature and pressure reached are 125°F and 3.3 psig, respectively. Maximum drywell gas temperature and pressure reached are 290°F and 4.5 psig, respectively.

For the case of one RHR heat exchanger operating, wetwell gas temperature and pressure start to level off after approximately 16 hours in the turbine trip accident. Maximum wetwell gas temperature and pressure reached are 145°F and 5.5 psig, respectively.

For the case of no RHR heat exchanger operating, containment temperature and pressure continue to increase with time. The drywell pressure reaches 121 psig at about 29 hours into the accident. The decay heat fraction (for infinite irradiation time) after 15 hours is less than 0.8%, which is very close to the decay heat removal capability of the two RHR heat exchangers. Therefore, the accident will be arrested if the RHR loops A and B can be initiated before containment failure occurs.

Insufficient decay heat removal has an effect on core inventory makeup. WNP-2 has RFW, HPCS, and RCIC for high pressure coolant injection and LPCS, LPCI, COND, FP water and SW cross-tie for low pressure coolant injection. RFW, COND, FP water and SW cross-tie take suction from water sources outside the containment. HPCS initiates on L2 or high drywell pressure ($P > 1.68$ psig). RCIC initiates on L2. At the beginning of an accident, HPCS and RCIC take suction from the Condensate Storage Tanks (CST). Upon receipt of a high suppression pool or low CST level signal, HPCS suction is automatically transferred to the suppression pool. RCIC suction transfers on low CST level only. LPCS and LPCI initiate on L1 or high drywell pressure. Their injection valves open only when the reactor is depressurized to 470 psig. LPCS and LPCI take suction from the suppression pool. The following are the effects of decay heat removal on core inventory makeup:

- If core cooling is adequate, pump flows are throttled to preserve NPSH requirements for HPCS, RCIC, LPCS, and LPCI.

- High suppression pool pressure (25 psig) will trip the RCIC turbine.

- The reactor must be depressurized to 470 psig to allow low pressure system injection into the core. CIA nitrogen supply pressure to the ADS valves is 150 psig. The differential pressure between the nitrogen supply and the containment atmosphere must be at least 88 psid to open the ADS valves. Therefore, the containment pressure cannot be more than 62 psig in order for ADS to operate.

- For reactor feedwater to be operable, the MSIVs must remain open. CN nitrogen supply pressure to the MSIVs is 100 psig. The differential pressure between the nitrogen supply and the containment atmosphere must be at least 46 psid in order to keep the MSIVs open. Therefore, the containment pressure cannot be more than 54 psig for the MSIVs to remain open.

3.4.3.3 Probabilistic Analysis

The loss of decay heat removal sequences (categorized as TW in Level 2 analysis) are characterized by failure of all available decay heat removal systems (RHR, Main Condenser (PCS), and containment Vent). These sequences proceed to containment failure prior to core damage.

The initiating events of most importance to this accident class are:

1. Plant Transients Initiated By:

   -Loss of condenser vacuum
   -Loss of containment air
   -Loss of DC
   -Stuck open SRV
   -Turbine trip
   -Loss of service water
   -Loss of containment nitrogen
   -Loss of feedwater
   -MSIV closure
   -Loss of plant service water

   The sequences of importance to the loss of decay heat removal from these initiators are those in which high pressure injection is successful but PCS and RHR are unavailable either due to the initiator or due to subsequent failures. Containment venting is not implemented and the containment fails from overpressurization leading to consequential failure of the injection systems.

2. Flooding in the Turbine Building

   The flooding results in consequential failure of the PCS, however, high pressure injection is successful. RHR and containment venting are subsequential failures leading to containment failure from overpressurization with consequential failure of the injection systems.

3. Medium LOCA

   The medium LOCA sequences important to this category result in consequential failure of the PCS (due to containment pressure isolation signal) with high pressure injection available. Upon subsequent failure of RHR and containment venting, the containment fails on overpressurization with consequential failure of the injection systems.

The current analysis of the loss of decay heat removal sequences considers the following factors:

- The transient initiator sequences, such as Loss of the Main Condenser, do not credit recovery of the PCS prior to containment overpressure failure. It would be appropriate to credit PCS recovery as there is approximately 21 hours available for action. This could effectively eliminate these sequences contribution to the TW category.

- The flooding sequences (FLD6 and FLD14), do not credit the use of the containment vent to prevent overpressure failure. For these floods, loss of TSW would disable the CAS system due to loss of cooling and venting cannot be performed without CAS. However, the WNP-2 abnormal procedures instruct the operator to align the FP water system for CAS cooling in case TSW is lost. The operator has approximately 20 hours to take this action prior to containment pressure reaches 49 psig. This action can be easily accomplished in this amount of time, and would allow the venting system to be credited for this sequence. Credit for venting would effectively eliminate this sequence from the TW category.

- The dominant cause of failure to vent the containment is the operator action "Failure of Operator to Vent Containment." The probability of failure to vent is a function of the emergency procedure instructions requiring the operator to vent, the time available to vent, the effect of venting on systems located in the reactor building, and the operator stress level at the time when venting is initiated. The procedural instructions for containment venting at WNP-2 are explicit in the EOPs. If additional data, e.g., simulator studies, were used in the human reliability analysis, the venting human error probability could be lowered.

If a hard vent line is installed, e.g., at the SGT "soft" area, the effect of venting on injection system availability will remain the same in the IPE analysis. This is because primary containment venting will not adversely affect the operability of the RPV injection systems which are located in the lower levels of the reactor building. The hardened vent would increase the time available to commence venting and lower the diagnosis error probability. However, the similar level of operator stress is unlikely to affect the action error probability. Therefore, the human error probability for failure to vent the primary containment at WNP-2, which is dominated by action error, will not be lower. As a result, the WNP-2 decay heat removal risk will not be reduced due to installation of a hardened vent. In addition, the Level 2 sensitivity analyses (Section 4.9) concluded that the hardened vent would have a very limited benefit in reducing offsite consequences.

Based on the above considerations, it is felt that the current analysis is bounding. The sequences that contribute to the loss of decay heat removal function and resulting in core damage sum up to 1.4E-6 per year. According NUREG-1289, the NRC staff has selected the goal that the quantifiable contribution to core damage frequency related to DHR failure should not be greater than 1.0E-5 per reactor-year to provide a margin for the unquantificable contribution. Therefore, the contribution of DHR failure at WNP-2 is well within the NUREG criteria.

3.4.3.4 <u>Conclusions</u>

The conclusions of the WNP-2 decay heat removal evaluation to resolve USI A-45 are as follows:

1. The current DHR reliability for WNP-2 is high as a result of multiple means of decay heat removal, including the main condenser, RHR (suppression pool cooling, shutdown cooling, wetwell and drywell sprays), and the containment vent (wetwell and drywell vents). Based on the IPE analysis, the reliability of the mechanical and electrical equipment for all three of these systems is good as evident from an unavailability of approximately 1.4E-6/yr for loss of decay heat sequences. There are no single failure events in any WNP-2 system (front line or support system) that can prevent decay heat removal.

   The WNP-2 design has several diverse and independent support systems for DHR (two divisions of standby service water, plant service water, control air, containment instrument air, AC power, and DC power). The most critical supporting systems, standby service water (required for RHR heat removal) and plant service water (required for main condenser and vent heat removal), are completely independent and even have independent locations and water suction sources. If the plant service water system is lost, WNP-2 has a procedure in place to manually align FP water for air compressor cooling to allow vent valve operation. Therefore, loss of any supporting system will not prevent decay heat removal.

   If all of the WNP-2 decay heat removal systems should fail, MAAP analyses show that there is approximately 29 hours to recover suppression pool cooling before the containment will reach its failure pressure.

2. The core damage frequency for loss of decay heat removal sequences is 1.4E-6/year. This provides sufficient margin to the NRC goal which is 1.E-5/yr.

3. Due to the redundancy and diversity of the WNP-2 decay heat removal systems, and the low calculated core damage frequency due to loss of DHR, a hardened containment vent path will not have a significant impact on reduction of CDF and offsite releases.

Therefore, USI A-45 is considered resolved for WNP-2 and the need for a hardened vent has been demonstrated to be nonbeneficial.

3.4.4 USI and GSI Screening

The Supply System Licensing Department reviewed NUREG-0933 (through Supplement 10) and NRC generic letters and bulletins (through November 1990) to identify USIs and GSIs that can be resolved for WNP-2 by the IPE effort. Two issues were identified: A-45 (Shutdown Decay Heat Removal) and A-17 (System Interactions in Nuclear Power Plants). Resolution of A-45 is discussed in Section 3.4.3. Resolution of A-17 is discussed in Section 3.4.4.1 below. Since this initial review, another generic issue given high priority by the NRC is GSI-105, Intersystem LOCA. Resolution of GSI-105 for WNP-2 is discussed in Section 3.4.4.2 below.

3.4.4.1 A-17 System Interactions in Nuclear Power Plants

According to NUREG-1229, a system interaction is an action or inaction of various systems (subsystems, divisions, trains), components, or structures resulting from a single credible failure within one system, component, or structure and propagation to other systems, components, or structures by inconspicuous or unanticipated interdependencies. There are three classes of system interactions: functionally coupled, spatially coupled, and humanly coupled. The IPE as required by Generic Letter 88-20 is limited to internal events, power operating conditions (Modes 1, 2 and 3), and human errors of omission. As such this IPE does not address system interactions due to seismic, fire, external flood or human errors during outage. Other than those limitations, system interaction has been included in the IPE throughout the IPE analyses.

Front line systems (RPS, ADS, RCIC, ECCS, etc.) are dependent on support systems (AC, DC, TSW, SW, CIA, etc.). Support systems are dependent on each other. Front line systems are dependent on each other. Finally, accident initiators can affect support systems. As shown in Section 3.2.3, four system dependency tables (Tables 3.2.3-1 and 3.2.3-4) are developed using information in system notebooks to show system interdependencies at WNP-2. Those tables are used to check fault tree 'external transfers' and event tree functional dependencies.

System fault trees were developed for this IPE. The following dependencies are explicitly treated in the fault trees:

Support System Dependencies: Transfers to support system fault trees were included at appropriate points in system fault trees. Linking of fault trees during fault tree cutset generation ensured all system interdependencies were accounted for correctly in the IPE results.

Shared Components Among Front Line Systems: Each plant component has a unique ID on the Master Equipment List. Basic fault associated with the component has essentially the same unique ID plus a failure mode. The same basic fault is used in more than one system fault tree if the component is shared between systems.

Human Errors: Common human errors are included in this IPE by having the same basic event designation. Human errors such as incorrect calibration of sensors, incorrect action in a series of steps, are modelled as same basic events in system fault trees if the same operator and procedure are involved.

Spatial Dependencies: The last two digits in a eighteen digit basic event designation are reserved for location identification of the component. If two or more components in the same location are disabled by internal flooding, those component basic events will be automatically set to 1 (unavailable) by the NUPRA program through its component failure parameter file.

Some potential causes of dependent component failures other than those listed above include common design, manufacture, and installation errors. The beta method (NUREG/CR-4780) is used to quantify all common cause failures which are implicitly modelled.

The IPE results are discussed in Section 3.4. The importance studies listed in Tables 3.4.2-1 and 3.4.2-2 indicate that common cause failures and human errors are the most important contributors to the dominant accident sequences. The sequences shown in Table 3.3.7-1 lead to the conclusion that there are no identified vulnerabilities because no single failure either by itself or by consequential failure that could cause core damage. It is confirmed that USI A-17 for WNP-2 is resolved for internal events.

3.4.4.2 GSI 105 Intersystem LOCAs

The intersystem LOCA presents a potential for creating a bypass of the containment with high radiological consequences. The intersystem LOCA also presents a potential for loss of decay heat removal by consequential failure of RHR. The intersystem LOCA is presented in Section 3.1.2.2.5 and addresses pressure isolation valve initiating events. An integral part of the resolution of this issue is also contained in the flooding analysis, Section 3.1.2.4.7, which addresses pipe and tank failures.

The conclusions that can be drawn from the IPE analyses support the conclusions reached in NUREG/CR-5928, i.e., "ISLOCA is not a risk concern for the BWR plant examined here [Note: BWR-4]." Typically, low pressure systems are isolated by at least two isolation valves plus a check valve. The intersystem LOCA frequency for WNP-2 is 1.21E-6 per year, and therefore, it does not contribute to the WNP-2 CDF. In addition, the pressure margin is high due to a relatively low reactor system pressure. Based upon the analysis presented in this IPE, GSI 105 is considered resolved for WNP-2.

3.4.5 Sensitivity of Level 1 Results

The core damage sequences are dominated by the loss of offsite and onsite power events. Therefore, the sensitivity analyses are focused on those events. As noted in Section 3.4.1 and 3.4.2, several of the high importance failures are common cause events. They will also be addressed. The important human actions are subjected to sensitivity studies to indicate whether their importance warrants further recommendations or procedural corrections. The results of the sensitivity analyses in this section have been used to help formulate recommendations and insights reported in Section 6.0.

Common Cause Failure Rates - Beta factors from NUREG/CR-4780 and NUREG/CR-4550 are used in the IPE. The most important of the common cause failures occur in the shutdown cooling systems (RHR, SW) and in the electrical support systems (primarily diesel generators and DC battery chargers). As the dominant common cause failures noted in Section 3.4.2 have not occurred at WNP-2, the beta method utilizing the generic data is felt to be bounding. EPRI has investigated common cause failures (NP-5777P) and developed strategies for defense against common cause failures. This investigation concluded that the beta method for analytically predicting the probability of common cause failures could be overestimated by a factor of ten. In addition to the DC power and diesel generator common causes evaluated later in this section, the other systems important from common cause failures are the RHR and SW systems in decay heat removal (SPC) mode. The CCF of RHR and SW components are high in risk achievement worth, therefore, a reduction in their values does not significantly lower core damage frequency (<1%).

Component Failure Rates - Where available, the WNP-2 component failure rates were taken from the WNP-2 specific NPRDS. In cases where NPRDS data for the WNP-2 components were zero, the NPRDS data for the generic industry components was used. This means the component had not failed at WNP-2 in the last 18 months and a realistic value would probably be lower than the generic value. This added a degree of conservatism to the analysis.

Human Reliability Analysis - The methodology used in the human reliability analysis does not credit the level of experience of the WNP-2 operating staff, nor the level of detail and training of the Emergency Operating Procedures. The top nontest-and-maintenance errors are discussed below and a sensitivity analysis presented to help illustrate the importance of these actions.

> Operator Fails to Vent When Required: The failure rate calculated for this action is 0.06/demand. An increase by a factor of 10 increases CDF by 69%, whereas, a decrease by a factor of 10 (or 100) decreases the CDF by 7%. The base value could overestimate the failure rate considering the level of training and detailed procedures

available to perform this action. Venting the containment requires consideration of evacuating all personnel from the reactor building. The tasks are assumed to be "dynamic" instead of "step-by-step" in the HRA analysis. This contributes to the relatively high value of the failure probability.

Operator Fails to Open RCIC Doors For Emergency Cooling: The failure rate calculated for this action is 0.05/demand. An increase by a factor of 10 increases CDF by 58%, whereas, a decrease by a factor of 10 (or 100) decreases the CDF by 6%. The base value could overestimate the failure rate considering the simplicity of the action and considering it is an explicit emergency procedure. Complicating factors are potential emergency conditions such a lighting and environmental conditions.

Operator Does Not Initiate ADS: The base value of this action is 0.00266/demand. An increase by a factor of 10 increases the CDF by 17%, whereas, a decrease by a factor of 10 (or 100) decreases the CDF by 2%. The base value could overestimate the failure rate considering the degree of training in sequences with failure of high pressure injection systems and that the WNP-2 has 18 SRVs available for manual depressurization. This is a fertile area for further analysis and investigation given its impact on core damage frequency and the importance of low pressure melt versus high pressure melt as determined in the Level 2 studies.

Operator Does Not Initiate Suppression Pool Cooling or Containment Sprays: The base value for this action is 3E-5 which is realistic given the high degree of training and familiarization for performing this action. An increase by a factor of 10 increases the CDF by 14 %, whereas, a decrease by a factor of 10 (or 100) decreases the CDF by 1%. This operator action is the only HRA term that appears on the risk achievement worth (RAW) list of the top 50 important RAW events (Table 3.4.2-2).

Test and Maintenance: Sensitivity analyses were not done on the important test and maintenance events as this is an area for future study on Tech Spec impact (AOT/STI) and for the Maintenance Rule.

AC Power - The loss of offsite power dominance of the core damage frequency implies that the onsite and offsite power sources should be examined closely for sensitivity of the results to the initial assumptions and as areas of potential vulnerabilities. Several of the basic events that dominant the sequences for loss of offsite power are due to common cause failures of redundant systems. These types of failures and the conservatisms inherent in the beta factor method are investigated also.

Initiating LOOP Frequency - It was demonstrated in Section 3.1.2, based on BPA data, that WNP-2 loss of offsite power frequency is 0.0246 events per year. Generic data, for example, NUREG-1032, shows a frequency of approximately 0.08 events per year. If the generic data was applicable to WNP-2, it would increase the core damage frequency by 151% to 4.39E-05 per year.

Recovery Probabilities - The data used in the analysis for recovery of an offsite power source in a given time interval, i.e., NRAC(30M), NRAC4, and NRAC10, were based on the generic data presented in NSAC 194. This data, particularly the longer outage times, are dominated by weather related outages, most of which are not applicable to the BPA-Northwest grid. Using BPA data on 230 Kv and 115 Kv line outage times, the core damage frequency can be reduced by 50%. The ability to backfeed WNP-2 from the 500 Kv line was not credited in the analysis due to the length of time (8 hours) that it takes to establish the link. If that procedure and hardware supported a connect time of less than 4 hours, it would reduce the core damage frequency by up to 58%. If the generic LOOP frequency of 0.08 events per year is used with the ability to backfeed from the 500 Kv source, the core damage frequency is still reduced by 37%.

HPCS Diesel Generator - The current analysis does not credit the ability to power the safety buses SM-7 and SM-8 from the HPCS diesel if operable. This alternate AC source of power is physically able to provide power since the diesel generator output is at the same voltage rating. If it was possible to make this connection in less than 10 hours, the core damage frequency could be reduced up to 25%.

DG Common Cause - The common cause failure of all three diesel generators is important as the only high pressure injection source remaining on loss of offsite power is RCIC. As a sensitivity study, if the CCF of the three DGs were reduced by a factor of 10, then the CDF is reduced by 23%. The factor of 10 is on the order of the EPRI results in NP-5777P.

DC Power - In contrast to the AC Power where significant risk reduction can be shown, the DC system is high in the Risk Achievement Worth (RAW). In particular, the common cause failure of the battery chargers are high on the RAW list (Table 3.4.2-2). Therefore, the sensitivity analysis focuses on what results occur if the reliability of the system or components is not maintained or improved. If the battery charger beta factor or common cause failure rate increases by a factor of 10, the core damage frequency increases by 4%. Reducing the unavailability of the chargers themselves has negligible impact on core damage frequency.

Initiating Event Frequencies - The initiating event frequencies for those transients having occurred at WNP-2 were based on an assumption that the total SCRAMs are 4 per year. That assumption is justified based on the latest four years of operation. A statistical mean of the WNP-2 scram data, excluding the first year of operation, shows the mean value is 6.5 SCRAMs. Using 6.5 instead of 4 SCRAMs, and renormalizing the general transient initiating frequencies results in an increase in CDF of +2%. If the same initiating frequencies are used for the ATWS initiating cases as well, the CDF increase is +5%.

4.0 BACK END ANALYSIS

4.0.1 Overview of the WNP-2 Level 2 Methodology

The general approach used in the quantification of the containment performance for WNP-2 utilized the following analytical steps:

First, the selection of the actual sequences or sequence cutsets were placed into each individual group based on their functional characteristics and the status of systems which are important to the containment performance assessment. This process was achieved using sequence descriptions and correlated tabulations of the status of all relevant systems to provide the basis for comparison (included in section 4.4). When each Level 1 sequence which had a frequency greater than 1E-9 had been assigned to a group, the group was processed with its associated initiator specific containment event tree (CET). The information developed during the grouping process was then used to establish the unique set of conditions which were to be superimposed on the CET node models during quantification of the CET.

The grouping process ensured that each sequence from the Level 1 analysis which had a frequency greater than 1E-9 was explicitly analyzed as part of the Level 2 analysis. In the few cases where approximations were needed to simplify the grouping it was done in a way which ensures that the results are conservative. This is achieved by using a technique in which individual sequences and their associated frequencies are subsumed within other, more severe sequences. This technique is sometimes called "conservative condensation".

Second, a set of containment event trees was developed to model accident progression and provide a description of the possible outcomes or containment damage states, which can result from each of the specific plant damage state identified by the Level 1 analysis. The time frame for the Level 2 analysis is assumed to extended for 40 hours after the initiating event. Though these trees are developed for each plant damage state, they also tend to correspond to initiator type. For WNP-2, containment event trees were developed for:

- Short-term station blackout (power restored within four hours, batteries remain available to provide DC power throughout an accident))
- Long-term station blackout (power not restored within four hours so that battery depletion results in loss of DC)
- Transient
- Anticipated transient without successful reactor SCRAM (ATWS)
- Small LOCA (break size less than 4", Primary System remains at pressure unless automatic or manual depressurization is initiated)
- Large LOCA (break size more than 4", Primary System will depressurize)

Third, quantification of the CETs to provide the estimated frequency for each individual sequence was accomplished by the insertion of the appropriate conditional probabilities at each of the CET branch nodes. Final quantification was the result of propagation of each initiating plant damage state and its associated occurrence frequency through its respective CET and accumulating these frequencies for each defined source term group or release category.

The CET branch node probabilities are calculated in one of two ways:

- from fault trees developed to identify each of the individual functional failures which are important to resolution of the node,
- split fractions which could be assigned to each CET branch node.

The primary difficulty encountered in the quantification of the containment event trees is one of ensuring that the dependencies between events are treated correctly so that simple Boolean algebra can be used to calculate sequence frequencies. This was accomplished by pruning the fault trees to represent sequence specific structures which reflect sequence dependencies correctly and return CET node probabilities which are independent.

The conditional probabilities used to quantify each CET are adjusted to match the specific conditions represented by the plant damage states. For example, if the Level 1 sequence cutsets show that the unavailability of high pressure injection was caused by hardware failure, the failure probability was assigned to be 1 in Level 2.

In the case of "suppression pool bypass", understanding and correctly treating the intersequence logic was facilitated with the use of a sub-event tree. This event tree was developed to identify all relationships between each of the sequence specific variables which influenced suppression pool bypass. Individual containment sequences were mapped onto this sub-event tree to identify the possible outcomes and the calculated endpoint probabilities were combined for like outcomes and returned to the main CET as probabilities for bypass/no bypass.

The final functional task performed during the construction of the overall Level 2 model for WNP-2 involved the definition of a set of criteria which could be used as the basis for grouping containment event tree end states into a limited, but complete, set of unique release categories. These categories were equally applicable to each CET, i.e. damage state descriptors which are initiator independent. The sequence characteristics ultimately adopted to characterize these release categories were:

- containment is/is not bypassed
- containment is/is not isolated
- fission products are/are not scrubbed
- time of containment failure (early/late)
- containment failure mode (large/small)

A set of simple logical rules was developed to use these characteristics to consistently sort and accumulate the frequency contribution from each sequence into one of twenty six defined source term bins.

To determine a representative source term for each bin which had an occurrence frequency greater than the assigned cut-off value, a representative sequence was selected from each bin and used to define a MAAP simulation which would provide an estimate of the fission product release.

4.0.2 Recovery Actions Credited in the Level 2 Analysis

Generally, failures which occurred within the original core melt sequences may be considered recoverable in the containment performance assessment, because the Level 2 analysis reflects plant behavior during a different time period. This treatment of recoverability is exemplified by the treatment of the potential for recovery of offsite power, in which the possibility of power recovery between core melt and vessel failure and between vessel failure and containment failure is credited in both short-term and long-term station blackout sequences. There are however, some notable exceptions, the most important of which is the assumption for recovery of the PCS as a means for removing energy from containment.

Following loss of offsite power, when power is restored recovery of PCS is expected to take about 8 hours. Therefore, the probabilities of recovery are determined by looking at the available window of opportunity in which a primary constraint is opening the MSIVs before containment pressure reaches 54 psig. Following a routine turbine trip, the PCS is usually recoverable without a great deal of difficulty so credit is very much determined by the expected non-response probability of the operating staff.

If the plant SCRAM was initiated by a transient involving failure of the MSIVs (unplanned closure), loss of containment air or loss of a DC bus, then the recoverability of PCS was not credited because reopening the MSIVs requires restoration of failed hardware - a situation which is not assumed to be very creditable following a core melt accident. The last case in which PCS is not considered to be recoverable is with the sequences in which turbine trip is initiated by a loss of condenser vacuum. The transient sequences were grouped using these criteria and each group evaluated separately so that the conditional probabilities associated with recoverability were treated correctly.

4.0.3 Assumptions

To ensure that the uncertainties in phenomenological behavior did not result in the underprediction of containment failure rates, in each of the cases described below in which there was a very limited set of information which could be used to guide the assessment, conservative split fractions are used to describe whether or not:

- sufficient melt could escape the pedestal cavity and damage the containment shell following a high pressure melt ejection (HPME),
- the forces exerted on the pedestal were large enough to cause structural failure of the pedestal during specific core melt and vessel failure sequences,
- corium released to the pedestal cavity during high and low pressure melt ejections remains in a coolable configuration.

In addition, during assignment of source term characteristics, a release to the reactor building was assumed to be released directly to the environment, i.e. no credit was taken for fission product retention, filtering, or deposition within the reactor building.

In some cases, the analytical uncertainties did not originate with phenomenological issues, but with the assignment of conditional failure probabilities for human actions and hardware failure. In these cases, conservatisms were introduced to compensate for:

- uncertainty in the magnitude of the effects from the very severe stressors acting upon the plant operating staff during a post core melt environment, and whether their performance would move beyond the normal bounds for which the human error probability data are generally applicable.

- uncertainty in the performance of critical hardware following a core melt accident, in which hardware reliability may be adversely affected by severe environmental stress, operation at the limit of its design envelope and difficulty in reestablishing normal hardware alignment and operation following a complete loss of all AC, DC, or critical cooling systems and degraded operation of control, monitoring and actuation systems.

4.0.4 Sensitivity Studies

A sensitivity analysis was performed for those sequence variables which appeared particularly uncertain. This was done by varying the split fractions associated with individual variables and determining whether or not the assumptions were important to the overall results. This provided an estimate of the overall uncertainty in the results. The final calculated results, however, are based on "best point value estimates" of the split fractions which were derived from information conveyed by the quantified Level 1 sequences or from published sources exemplified by NUREG-1150, NUREG/CR-4551 or NUREG/CR-5528.

4.1 Plant Data and Plant Description

System, component, and structure data that may be of significance in assessing severe accident progressions and containment challenges are summarized in the following sections.

4.1.1 Reactor Vessel and Primary System

Washington Nuclear Plant 2 (WNP-2) is a boiling water reactor of General Electric BWR 5 design. The rated reactor thermal power is 3,323 MWt which is produced by 764 fuel bundles. The reactor pressure vessel (RPV) is a low-alloy steel vessel with a stainless steel interior clad. The RPV is shown in Figure 4.1-1 including the layout of major internals and vessel penetrations. The RPV is 6.69" thick including the cladding with reinforcement in the area of the vessel where the jet pumps reside to a total thickness of 10.13". The bottom head is also reinforced to a thickness of 8" to accommodate the 185 control rod drive penetrations. The RPV has an internal diameter of 20'11" and an internal height of 72'11".

The weight of the RPV is 1,723,500 lbs. Of this weight, the bottom head is 210,500 lbs, the top head is 186,700 lbs, the head flange is 89,100 lbs, internal supports are 79,100 lbs, and the CRD housings and stub tubes are 84,300 lbs.

Within the vessel, the steam driers weigh 80,000 lbs, the separators and shroud head weigh 146,500 lbs, the shroud itself weighs 116,900 lbs, the top guide weighs 19,500 lbs, the core support weighs 20,500 lbs and the guide tubes weigh another 46,300 lbs.

The total mass of UO_2 in the core is 349,900 lbs with a normal enrichment of 2.6%. The UO_2 fuel pellets are clad in Zircaloy tubing with a thickness of 0.035". In addition, the fuel assemblies contain another 61,900 lbs of Zircaloy. The RPV contains 185 control rods with 14.86 lbs of B_4C and 224 lbs of stainless steel in each.

During normal operation the RPV contains 12,080 ft^3 of liquid and 8,926 ft^3 of steam. Reactor feedwater enters the vessel at a temperature of 420°F and an enthalpy of 527.6 Btu/lb. The steam flow rate is 14.3 million lbs per hour and the coolant flow rate through the core is about 108.5 million lbs per hour. The RPV operates at a saturation temperature and pressure of 549°F, 1020 psia. The design pressure of the RPV is 1250 psig. The vessel is protected from over pressure by 18 relief valves. Each safety relief valve has two setpoints - a relief setpoint and a safety setpoint. In the relief mode the SRVs lift in banks at pressures between 1076 psig and 1116 psig. In the safety mode the SRVs lift at pressures between 1150 psig and 1205 psig. The SRV flow ratings are given at 103% of their safety settings and vary from 865,366 lb_m/hr for the SRVs set at 1150 psig to 906,250 lb_m/hr for those set at 1205 psig. Each main steam SRV outlet is piped to a submerged quencher in the wetwell. The main steam SRV tail pipe vacuum breakers are located in the drywell.

RPV venting can also be accomplished via the power conversion system (PCS) if a main steam line can be opened (or remains open through the transient). Using this vent path routs steam from the RPV to the main turbine condenser. The main steam isolation valves (MSIV) are capable of opening at containment pressures less than or equal to 54 psig. The MSIVs are air to open spring to close valves and at containment pressures exceeding 54 psig the control air has insufficient pressure differential to overcome the MSIV closure springs.

The reactor vessel is supported by an annular pedestal that extends from the containment basemat through the drywell floor to the vessel.

Table 4.1-1 shows the design parameters and characteristics for the reactor vessel and primary system. Details and references can be found in the IPE system notebook.

4.1.2 Containment System

WNP-2 employs a Mark II pressure suppression containment design. The primary containment is a free standing steel vessel which is divided into two major regions, the drywell and the wetwell. Figure 4.1-2 illustrates the general layout of the WNP-2 containment. The drywell has the shape of a truncated cone with a cap and it houses the reactor and its associated primary system. The wetwell has the shape of a cylinder with a round bottom. The drywell floor serves as a pressure barrier between the drywell and the wetwell. There are ninety nine downcomer pipes which penetrate the drywell floor and provide a flow path for steam and gas from the drywell to a pool of water (suppression pool) in the wetwell in an accident situation. There are nine wetwell-to-drywell vacuum breakers which provide a flow path for noncondensible gases from the wetwell air space back to the drywell. Three vacuum breakers between the reactor building and the wetwell are provided to limit the external force resulting from the condensing steam within the containment.

The pedestal region is recessed relative to the drywell floor. There are two 8 foot by 6 foot sumps cast into the pedestal floor. The sumps each have ⅜" thick stainless steel covers and normally contain water to a depth of about 17" (490 gallons each). If the drywell sprays are used, the water collects in trenches in the drywell floor. The water in the trenches drains via two 4" drain lines to the FDR sump in the pedestal. The sumps have drain lines which are routed beneath the surface of the suppression pool before exiting the containment. The drain lines are closed as part of containment isolation. There are no downcomers in the pedestal region. Figure 4.1-5 shows a plan view of the pedestal region including the sumps and Figure 4.1-6 shows a cross-sectional view of the pedestal and sumps.

The physical dimensions of the steel containment are:

- Diameter of the cylindrical portion at the base of the cone is 86 feet.
- The diameter at the top of the cone is 39.5 feet and then narrows to 32 feet to carry the head.
- The bottom head has an inside height of 21.5 feet with ellipsoidal head of 2:1 ratio, and varies in plate thickness from 7/8" to 1-1/2".

- The top head has an inside height of 15.5 feet, 15/16" thickness and is bolted with a flanged joint.
- The drywell shell is 99 feet high.
- The suppression chamber is 72 feet high.
- The overall steel height is 171 feet.
- The vessel shell plate thickness varies from 1-1/2" at the lower head to 3/4" at the drywell conical section.

A reinforced concrete Biological Shield wall surrounds the primary containment. There is a 2-1/4" gap filled with insulation between the containment and the Biological Shield which serves to allow interference between the two structures. The concrete used in the WNP-2 biological shield wall and basemat is a basaltic variety. Laboratory analyses of concrete samples show a mass fraction of 0.0038 CO_2 and a fraction of 0.1292 CaO.

The primary containment vessel is reinforced with internal vertical and horizontal stiffeners, Figure 4.1-3. The vessel sits on the concrete mat foundation. The bottom of the wetwell is lined on the inside with reinforced concrete.

The drywell floor to the containment vessel gap is closed off by means of the Peripheral Seal, also referred to as the Omega Seal, Figure 4.1-4. The Seal is made of steel and is welded to the containment vessel and to the underside of the circular closure girder embedded in the drywell floor.

Normal suppression pool water level elevation is 466'2¾" and is controlled within 2" of this height during normal operation. This gives a suppression pool depth of 31 feet and downcomer submergence of 11'10".

The primary containment can be vented either from the wetwell or the drywell. Containment venting would be accomplished through the 24" containment purge lines. The emergency operating procedures direct the operators to vent containment if internal pressure rises above 39 psig. The operator is instructed to try to vent the wetwell region first and to vent the drywell only if the wetwell vent path fails to open. The butterfly isolation valves in the 24" purge lines are designed to be able to open against containment pressure as high as 49 psig.

Table 4.1-2 shows the containment design parameters and characteristics. Details and references about the containment can be found in the primary containment (IPE) system notebook.

4.1.3 ECCS and Other Water Injection/Recirculation System

The emergency core cooling systems (ECCS) at WNP-2 consist of the High Pressure Core Spray (HPCS) System, the Low Pressure Core Spray (LPCS) System, the Low Pressure Core Injection (LPCI) System, and the Automatic Depressurization System (ADS). In addition, WNP-2 can use the Reactor Feedwater (RFW) System or the Reactor Core Isolation

Cooling (RCIC) System for high pressure injection, the Condensate (COND) System for low pressure injection, the Residual Heat Removal (RHR) System for containment spray and/or suppression pool cooling, and a Standby Liquid Control (SLC) System for backup reactivity control. The paragraphs to follow give a brief description of these systems. For a more complete description, see Section 3.2.1.

4.1.3.1 High Pressure Injection

The HPCS system is a single pump system which injects water through a spray ring inside the shroud in the RPV. The pump is driven by an electric motor which is supplied by division 3 power. Division 3 has a dedicated diesel for backup power as well as a battery bank capable of sustaining HPCS operation for 4 hours following SBO. The HPCS system draws water from either the condensate storage tank or the suppression pool. The nominal flow rate is 6350 gpm and is achieved at a differential pressure of 200 psi. At an RPV pressure of 1130 psig, the HPCS system will deliver 1550 gpm to the core. The HPCS pump is located in the basement of the reactor building in a separate compartment.

The RCIC system is a single pump system which injects water into the RPV via a spray nozzle in the RPV head. The pump is driven by a turbine which is supplied from the RPV. The RCIC system is designed to run even if the dedicated batteries are the only power source. The RCIC system normally draws water from the condensate storage tank but can also draw water from the suppression pool. The RCIC system is designed to function at suppression pool temperatures of 240°F or less and suppression chamber pressures of 25 psig or less. The system is designed to provide 600 gpm of water to the vessel at any RPV pressure between 150 psig and 1158 psig. The RCIC pump and turbine are located in a dedicated compartment in the basement floor of the reactor building.

The RFW system is used to provide coolant during normal operation. The system contains two turbine driven pumps which are capable of providing 115% of full power coolant requirements at rated reactor pressure. The RFW system injects water through a ring header outside the shroud in the RPV. The system requires AC power to operate. The RFW pumps are located in the turbine building. RFW is generally unavailable for all sequences except ATWS in the Level 2 analysis.

The Control Rod Drive (CRD) system also provides a small source of high pressure injection. The CRD system has two 100% capacity pumps which are located on the basement floor of the reactor building. At a reactor pressure of 1000 psig a CRD pump provides 63 gpm of cooling water to the core through the cooling water orifices in each of the control rod drives at the bottom of the vessel. At a RPV pressure of 125 psig one CRD pump will provide 157 gpm of condensate. The CRD system requires AC power to operate.

The SLC system has the primary purpose of injecting sodium pentaborate into the RPV as a backup means to shut down the reactor if the reactor failed to SCRAM. However, the SLC system also provides a means of injecting 86 gpm of water into the RPV at any RPV pressure up to 1400 psig. Each of the two pumps in the system is a motor driven positive displacement motor driven pump and provides 43 gpm of water. The SLC system injects water through the HPCS spray ring inside the shroud of the RPV. One loop of SLC is powered by division 1 power; the other by division 2. The SLC pumps are located on the 548' elevation of the reactor building.

4.1.3.2 Low Pressure Injection Systems

The LPCS system is a single pump system which injects water through a spray ring inside the shroud in the RPV. The pump is driven by an electric motor which is supplied by division 1 power. The LPCS system draws water from the suppression pool. The system can take suction from the RPV or the condensate system only if a spool piece between the LPCS and RHR 'A' systems is manually installed. The nominal flow rate is 6350 gpm and is achieved at a differential pressure of 128 psi. At a pump differential pressure of 415 psi, the LPCS system will deliver 2000 gpm to the core. The LPCS pump is located in a dedicated compartment on the basement floor of the reactor building.

The LPCI system is actually an operating mode of the RHR system. All three loops of the RHR system can operate in LPCI mode. All three pumps are motor driven. RHR 'A' loop is dependent on division 1 power. RHR 'B' & 'C' loops are dependent on division 2 power. Flow from the LPCI mode of RHR begins to enter the vessel when RPV pressure falls below 225 psig. Each loop can provide the nominal flow rate of 7450 gpm when the pressure difference between the RPV and the suppression pool falls to 26 psi. The LPCI mode of RHR floods the core just inside the shroud in the RPV. The RHR system is normally lined up to draw water from the suppression pool. It can, however, be lined up to take suction from the RPV or the condensate system. Each RHR pump is located in a dedicated compartment on the basement floor of the reactor building.

Two of the RHR loops, 'A' & 'B', may be operated in containment spray cooling mode. In this mode suppression pool water is circulated through the RHR heat exchangers and routed to one of the containment spray. There are two spray headers in the drywell, and one in the wetwell. Either RHR loop may be aligned to spray drywell or the wetwell.

RHR loops 'A' and 'B' may also be operated in suppression pool cooling mode. In this mode an RHR loop is aligned to take suction from the suppression pool, cool the water in the RHR heat exchangers and return the water to the pool. As with the containment spray mode, this operating configuration is used to remove energy from the containment.

RHR loops 'A' and 'B' may also be used in the shutdown cooling mode. In this mode suction is taken from the RPV, run through the heat exchangers and returned to the RPV. The RPV pressure must be below 48 psig to enter this mode.

The. COND system normally provides water to the RFW pumps during plant operation. However, in emergency situation the condensate and condensate booster pumps can provide water to the vessel if the vessel has been depressurized. The condensate pumps draw water from the condenser hotwell. In addition, a fire hose connection and isolation valve are provided at the suction of the 'A' condensate booster pump. This allows the fire protection water tanks to be used as an emergency water source. There are three one-third capacity condensate pumps and three one-third capacity condensate booster pumps. All are located in the turbine generator building. All require AC power to operate. The COND system injects water through the RFW ring header outside the shroud in the RPV.

The Service Water (SW) system provides cooling water to the ECCS pumps, emergency room coolers, and the RHR heat exchangers. By manually installing a spool piece between the SW system and the RHR system, the SW system can be used to flood the RPV (and in the case of a vessel rupture, to flood containment). The SW system can not inject to the RPV until the vessel pressure falls below 70 psig.

4.1.3.3 Water Sources

The two primary sources of water for cooling the core in a transient are the condensate storage tanks (CST) and the suppression pool. The minimum volume of water allowed in the CST is 135,000 gallons. This water is maintained between the temperatures of 40°F and 140°F. The minimum volume of water in the wetwell is 1.03 x 10^6 gallons (this volume is sufficient for removing core decay heat for approximately 11 hours). The wetwell is maintained at a temperature below 90°F.

TABLE 4.1-1
Reactor Pressure Vessel
And Coolant Injection Systems
Principal Design Parameters And Characteristics

BWR 5 Vessel I.D.	251 inches
Number of Fuel Bundles	764
Rated Reactor Thermal Power	3323 MWt
Turbine Bypass Capacity	25%
Steam Flow Rate during Normal Operation	14.3×10^6 lb/hr
Design Pressure of the RPV	1250 psig
Number of Safety Relief Valves	18
HPCS System	
Number of Pumps	1
Nominal Flow Rate	6350 gpm @ 200 psid 1550 gpm @ 1130 psid
Driver Type	AC Motor
Injection Point	Inside Shroud
LPCS System	
Number of Pumps	1
Nominal Flow Rate	6350 gpm @ 128 psid
Driver Type	AC Motor
Injection Point	Inside Shroud
LPCI (RHR) System	
Number of Pumps	3
Nominal Flow Rate	7450 gpm @ 26 psid
Driver Type	AC Motor
Injection Point	Inside Shroud
RCIC System	
Number of Pumps	1
Nominal Flow Rate	600 gpm
Driver Type	Steam Turbine
Injection Point	RPV Head Spray

TABLE 4.1-2
Primary Containment
Drywell And Pressure Suppression Chamber
Principal Design Parameters And Characteristics

Pressure Suppression Chamber	
Internal Design Pressure	45 psid
External Design Pressure (Due to negative internal pressure)	2.0 psid
Drywell	
Internal Design Pressure	45 psid
External Design Pressure (Due to negative internal pressure)	2.0 psid
Drywell Floor Design ΔP	
Downward	25 psid
Upward	6.4 psid
Drywell Free Volume, Including Downcomer Vent Pipes	200,540 $ft.^3$ (max)
Pressure Suppression Chamber Free Volume	144,184 $ft.^3$ (max)
Pressure Suppression Pool Water Volume	137,262 $ft.^3$ (min)
Submergence of Downcomer Vent Pipe Below Pressure Suppression Pool Surface	11.67 ft. (min) 12.0 ft. (max)
Design Temperature of Drywell	340°F
Design Temperature of Pressure Suppression Chamber	275°F
Downcomer Vent Pipe Pressure Loss Factor	2.77
Total Downcomer Vent Pipe Area	309 $ft.^2$
Number of Downcomer Vent Pipes	99
Minimum Spacing of Downcomer Vent Pipes	4'-3"
Normal Operating Temperature - Suppression Chamber	135°F for air 75°F for H_2O 135°F (150°F Locally)
Normal Operating Pressure - Drywell and Suppression Chamber	-.5 psig to 1.0 psig

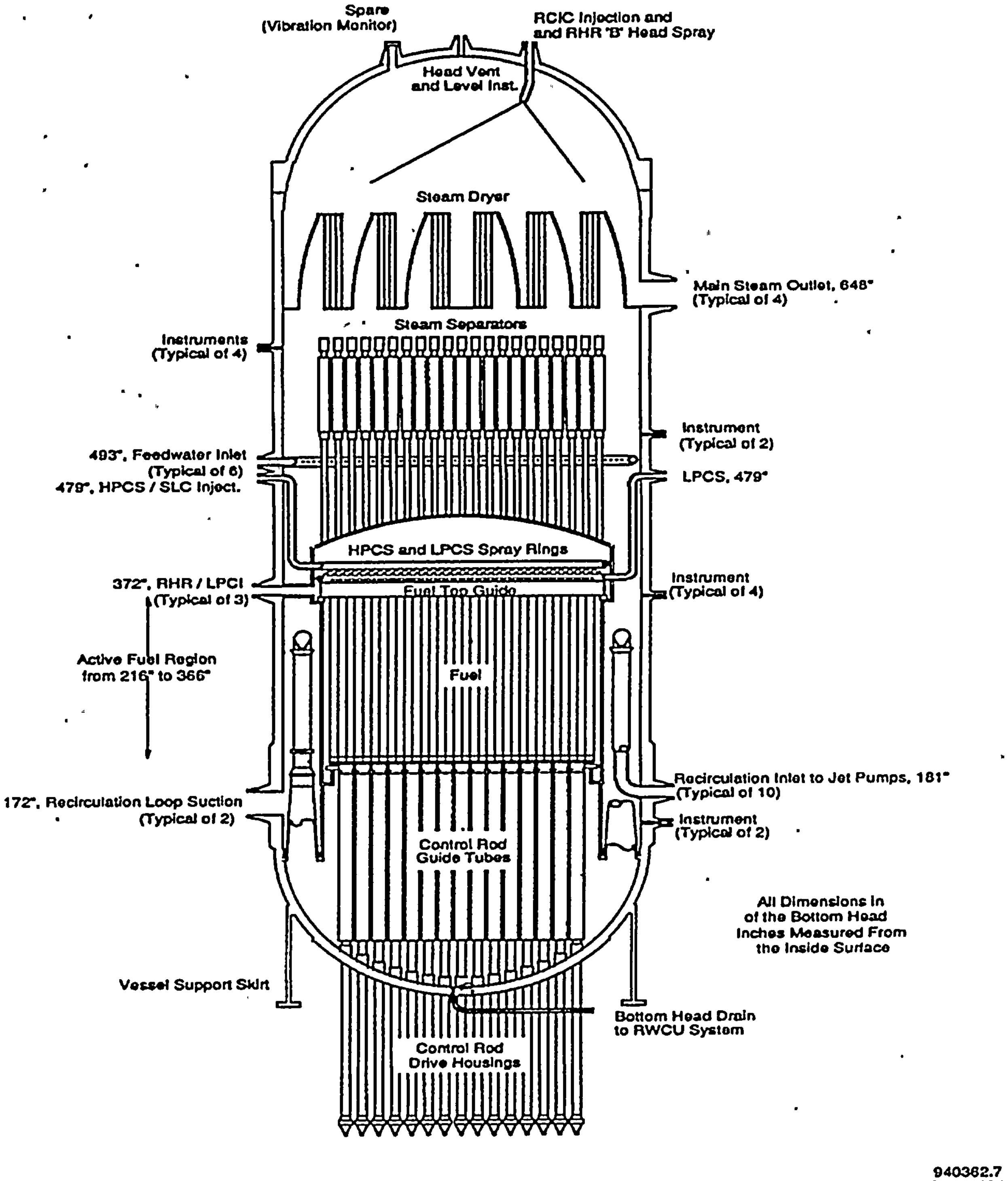

FIGURE 4.1-1 Reactor Pressure Vessel

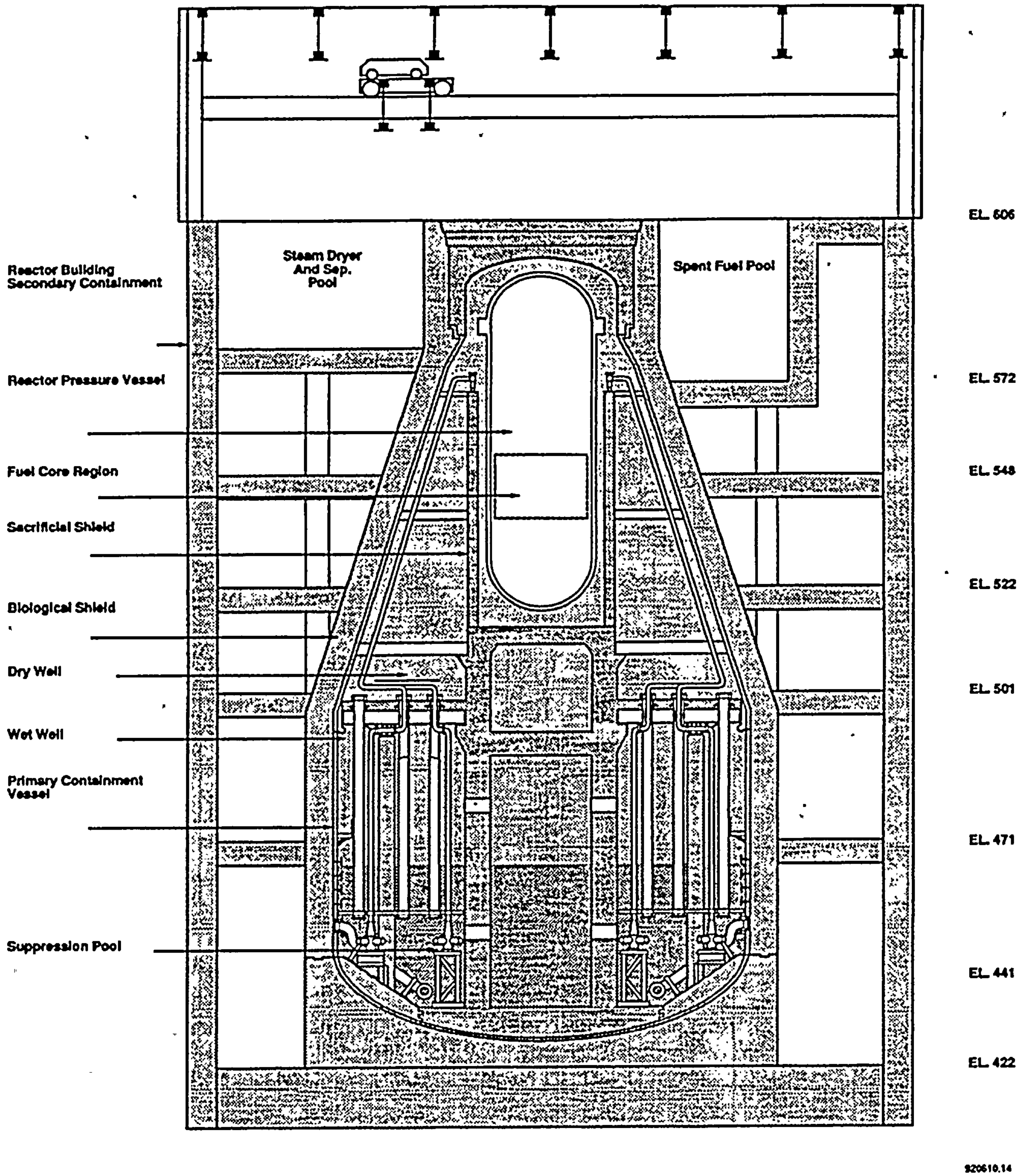

Figure 4.1-2 Primary Containment

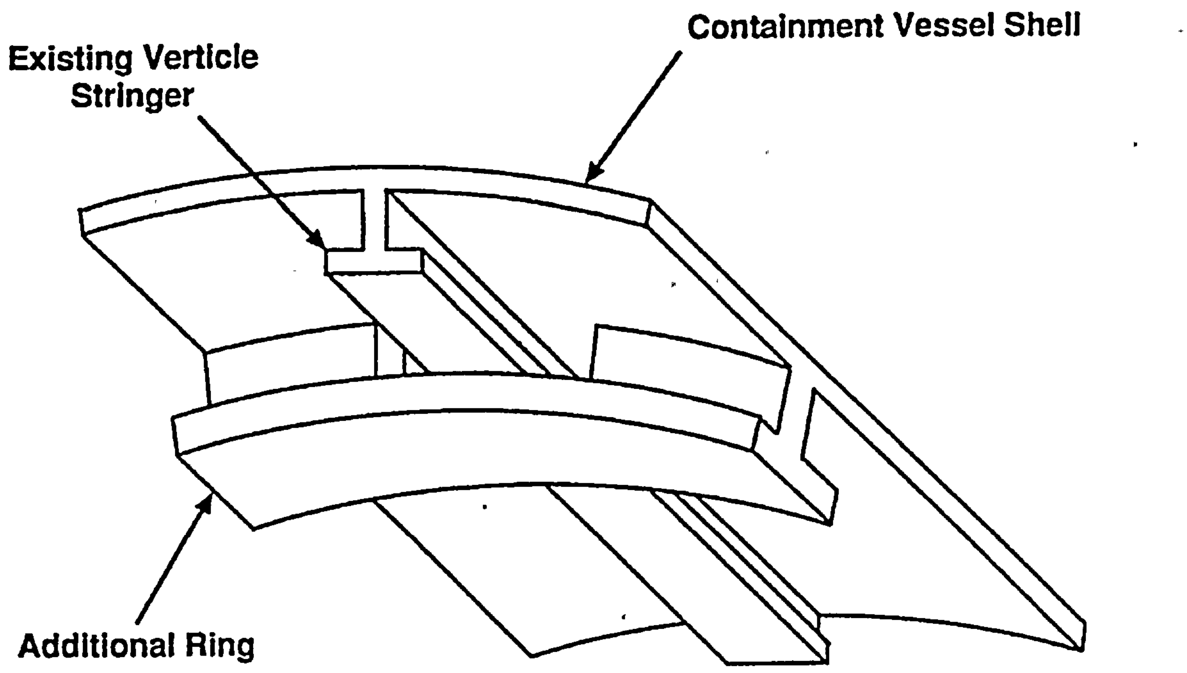

Figure 4.1-3 Stiffner Configuration

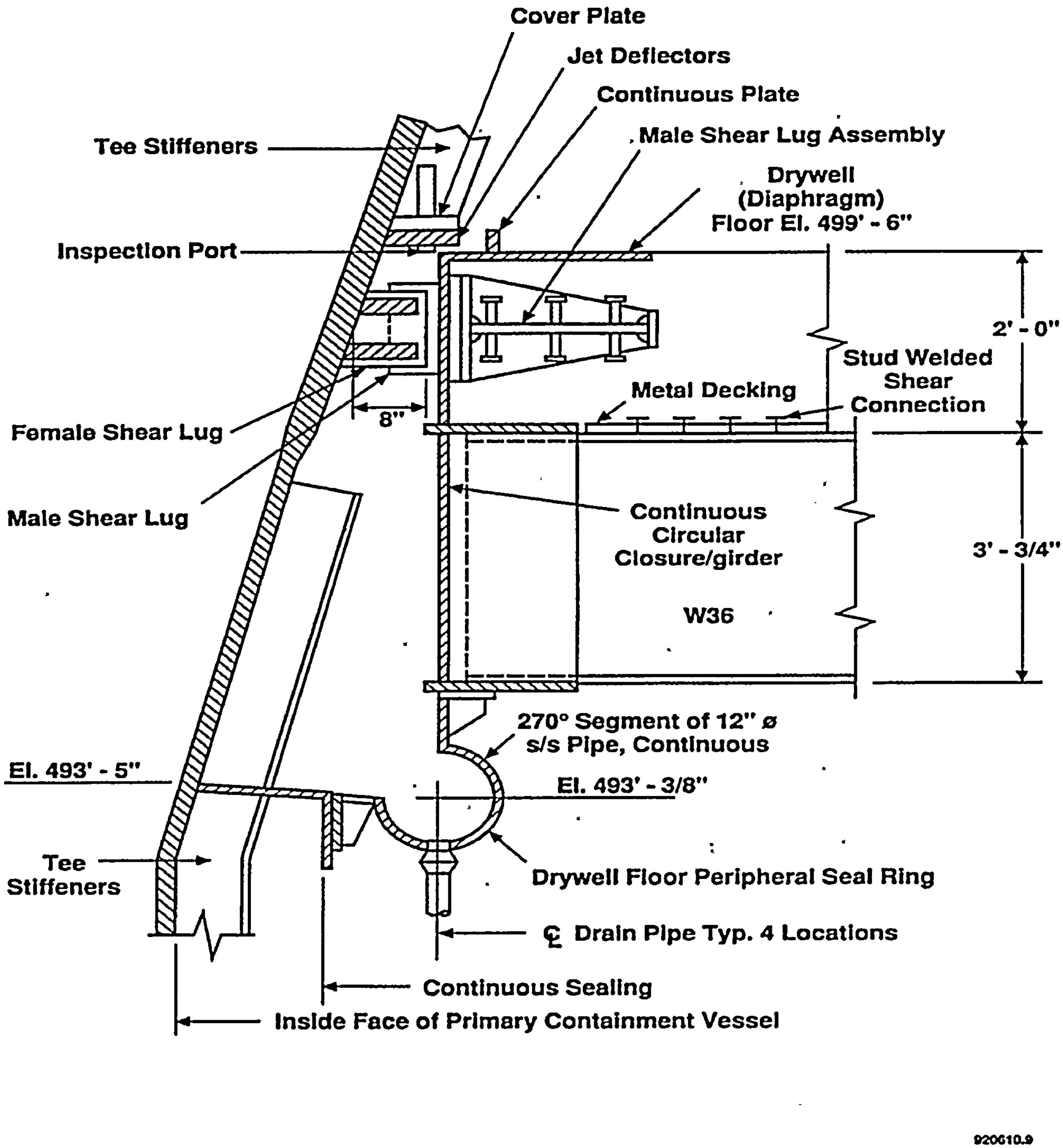

Figure 4.1-4 Drywell Floor Peripheral Seal, Circular Flanged Girder, Shear Lugs, Jet Deflectors

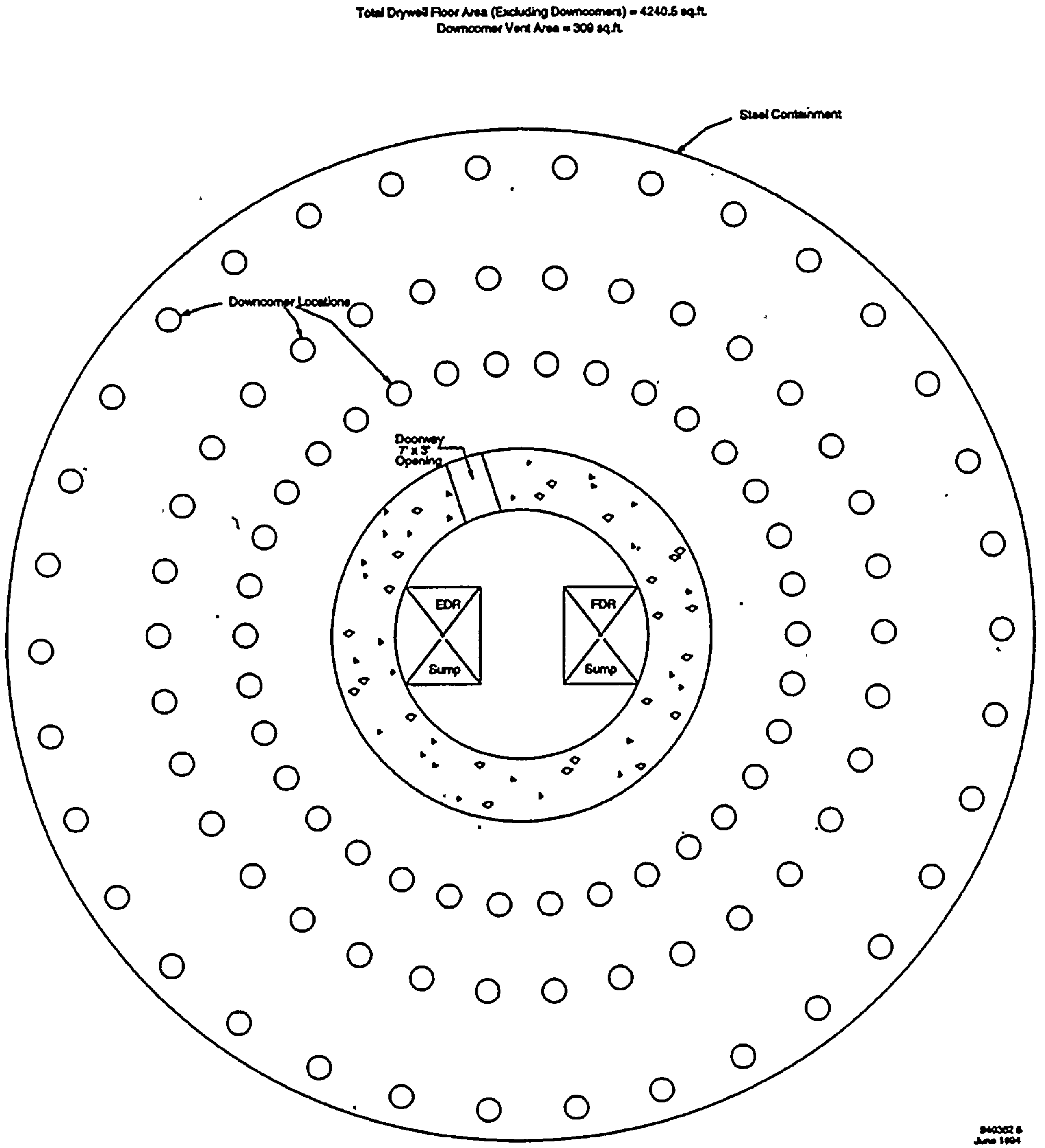

FIGURE 4.1-5 Pedestal Region, Plan View

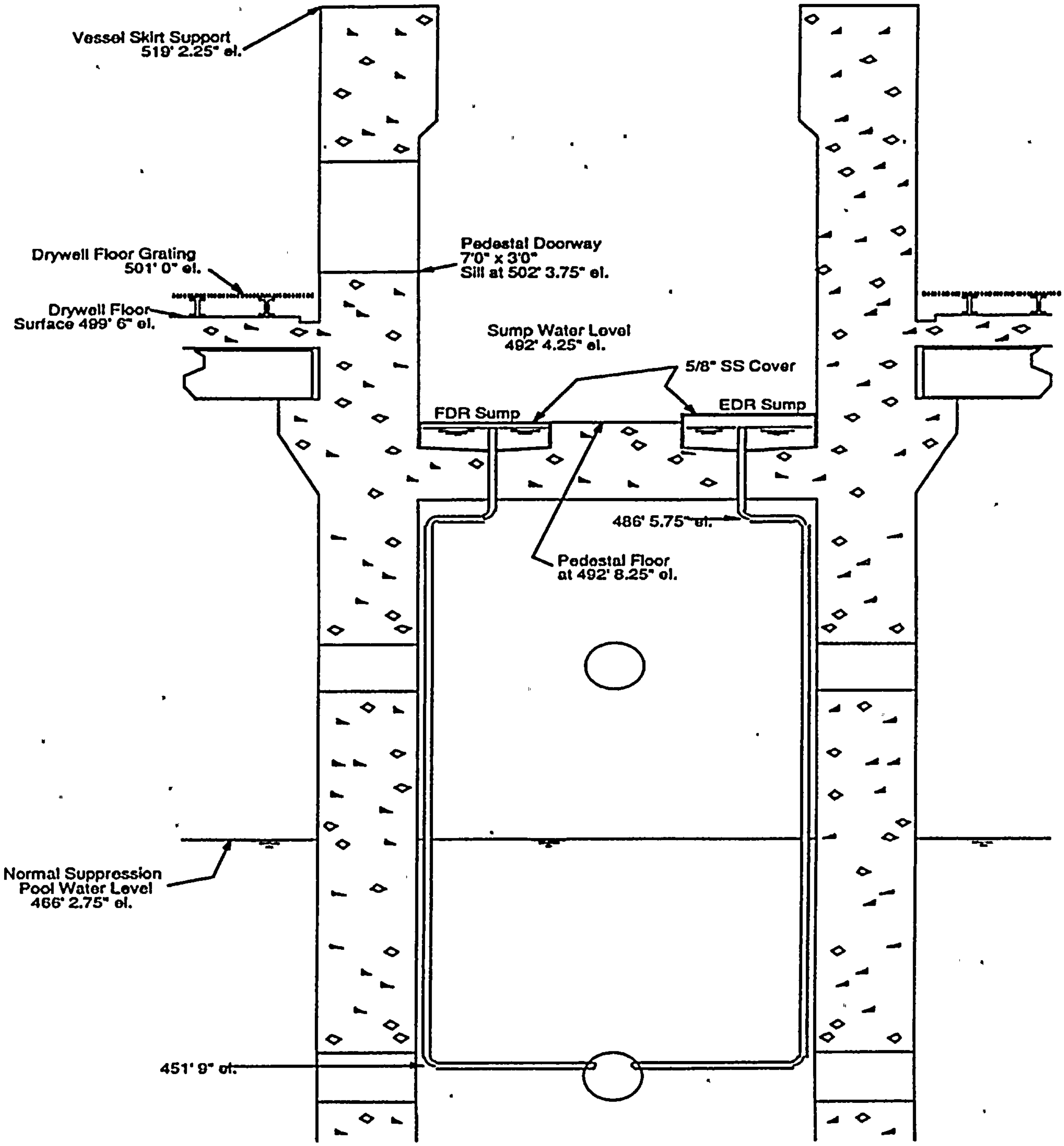

Figure 4.1-6 Pedestal Region, Cross Section

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4.2 Plant Models and Methods for Physical Processes

In the Level 2 IPE, the Modular Accident Analysis Program (MAAP), Revisions 7.02 and 7.03 are used to provide an integrated approach for modelling of plant thermal hydraulic response and fission product transport during severe accidents. Plant specific data derived from plant design documents and drawings are compiled into the MAAP parameter file. The WNP-2 specific MAAP parameter file provides a complete description of WNP-2 for MAAP simulation. Different severe accident events (station blackout, transients, ATWS, LOCA, etc.) are analyzed using the MAAP code. Important results are reported in appropriate sections.

In addition to MAAP results, research results in the open literature, IDCOR task reports, Shoreham and Limerick PRAs, NUREGs, and engineering judgement are used in understanding physical processes and developing event trees. In the following, models and methods for physical processes are discussed. References utilized in each section are contained at the end of the section. This format differs from the rest of the report because of the on-going research into the phenomena discussed make the results used dependent on the time references were generated.

4.2.1 Accident Progression

If adequate cooling water is not provided to the core, the fission product decay heat could result in the melting of the fuel and neighboring materials. During the period of core heatup and melting, steam can interact with the zircaloy cladding and produce hydrogen. This exothermic reaction further increases the heat up rate of the core. Volatile fission products (such as cesium and iodine) and the noble gases are released from the fuel. These fission products may deposit on the relatively cool surfaces in the reactor coolant system or may be swept into the suppression pool. Significant retention of the particulate fission product by the pool will occur, provided the integrity of the drywell wall is not compromised. The noble gases, which are neither deposited on surfaces nor trapped in the pool, will be released to the containment atmosphere.

If the core-melt accident is not arrested in the vessel, the fuel will melt through the lower head and fall into the pedestal region. Some of the molten material may be dispersed into the drywell. Following vessel breach, ex-vessel steam explosion, direct containment heating, hydrogen combustion, or molten core-concrete interaction could generate high pressure or high temperature conditions to challenge the containment integrity.

The MAAP code has been used to provide integral analyses of accident sequences from their inception until several hours after containment failure. The results of in-vessel and ex-vessel accident progression analysis are discussed in this section. Since MAAP 3.0B does not include models for direct containment heating, hydrogen detonation and steam explosion, these separate effects will be discussed in Sections 4.2.2 - 4.2.6.

4.2.1.1 In-vessel Accident Progression

The reactor vessel (Figure 4.2-1) is nodalized into core, shroud head, separator, upper head, upper downcomer, lower downcomer, recirculation loop, and lower head regions. Separate mass and energy conservation equations are solved for each of these regions. For regions containing both steam and water, conservation equations are solved for each constituent. If the calculated specific energy is higher than that for a saturated condition, a flashing rate and the average void fraction are calculated.

The reactor core is divided into eight radial regions and ten axial nodes per region. Each node is represented by a fuel pin segment with Zircaloy cladding, and adjacent coolant channel, and a section of Zircaloy fuel channel. Decay power is calculated on the basis of the ANSI/ANS Standard [1]. If the accident sequence is an ATWS, power is calculated based on the Chexal-Layman correlation. The constituents of the core (UO_2, Zr, and ZrO_2) are assumed to form an eutectic which melts at 2500°K with 275 kJ/kg latent heat. It is assumed that the molten mass will slump downwards to a lower node, mix with the lower node material, and thereby transfer mass and energy to lower and colder locations. When the lowest axial node in the radial channel is molten, all the molten material in that channel is then relocated to the lower core plate. There are several restrictions to the core slumping motion due to volumetric, hydrodynamic and steam generation considerations. Details are discussed in Reference 2.

Plant specific MAAP results indicate that for a short-term station blackout sequence without ADS actuation and without injection, the important event timings are:

- core uncovery - 47 minutes,
- core melting - 123 minutes,
- core plate failure - 215 minutes, and
- reactor vessel failure - 215 minutes.

Note that MAAP 3.0B does not distinguish the timing of core plate failure from that of vessel failure. The initial vessel failure size is 3.14 ft^2 and it increases to 3.66 ft^2. At vessel failure 73% of the molten core is discharged into the pedestal. It takes another 2 hours for the remaining molten core to be discharged. Table 4.2.1-1 compares the predicted accident progression results for WNP-2, La Salle [3] and a synthetic "Mark II CPI Plant" Design [4] using MAAP, BWRSAR and MELCOR. BWRSAR was developed by the Oak Ridge National Leboratory for modeling of the reactor vessel response, and MELCOR was developed by the Sandia National Laboratories for the reactor vessel and containment response. The CPI Plant integrates Susquehanna's BWR-4 reactor with a primary containment which incorporates elements of the Susquehanna, La Salle, and WNP-2 designs. The single deep-cavity design is representative of La Salle and WNP-2 that prevents spreading of the core debris to the ex-pedestal region of the drywell during the scenarios in which vessel fails under low pressure.

Modeling differences in RPV bottom head failure mode and flow type of melt out of vessel have significant effects on the predictions of event timings. The BWRSAR code assumes that after the core plate failure, the water in the reactor vessel bottom head must first be boiled away and the quenched debris must then reheat before the instrumentation tubes can fail. Therefore, a longer time is predicted for the failure of the bottom head penetration.

The predicted mass of hydrogen generation (1281 lb) for WNP-2 is less than those predicted for La Salle and CPI plants. The MAAP results were obtained using the local blockage model (FCRBLK=0). In this model, the movement of Zircaloy away from the melting region will terminate oxidation in that given node. The moving fuel and clad will have little effect on the gas flows and fission product release. To maximize the amount of hydrogen generation, the no blockage model was used by setting FCRBLK=-1 (in which fuel and clad material melting will have little impact on the hydrogen generation, gas flows, and fission product release rates). The mass of hydrogen generated is about 3050 lb, which is greater than the MELCOR and BWRSAR predictions.

4.2.1.2 <u>Ex-vessel Accident Progression</u>

Since WNP-2 has a smaller total containment volume, the resulting pressure increase due to vessel blowdown (~ 60 psia) is larger as seen in Table 4.2.1-1. Using the no core blockage model, the calculated pressure increase is about 87 psia. The pressure loading is still below the containment failure pressure (~121 psig). It is concluded that the pressure loading resulting from vessel blowdown will not fail the WNP-2 containment.

After vessel failure, BWRSAR assumes the metals with lower melting points would be released before the oxides. The release is characterized by an initial large pouring rate followed by a slower rate which is controlled by the decay heat level. The majority of the molten material is released in a period of about 160 min [4]. It is assumed that 100% of the melt remains in the pedestal region and subsequently causes severe molten core-concrete interaction (MCCI). Containment failure occurs at around 600 min due to noncondensable gas generation as seen in Table 4.2.1-1. In the MELCOR/LTAS analysis for La Salle, the melt could flow into the sump drain lines, and shortly thereafter (20 min) it could fail the lines and fall into the wetwell pedestal floor. When the calculation was terminated at 1170 min, the containment pressure was about 174 psia, which is below the LaSalle failure point 203 psia.

In the MAAP code, the core melt is assumed to produce substantial blockage formation above the core plate. This leads to a pool of core melt above the lower plenum. The core support structure is weakened by the load above it and collapses. The vessel is assumed to fail immediately after the core plate failure. Most molten material thus pours rapidly through the vessel penetrations for a period of about 1 min. The debris could be entrained from the pedestal to the drywell by the high velocity gas stream exiting the vessel. About 66% of the melt is predicted to be transported into the drywell. This leads to a less aggressive MCCI in the pedestal floor and a slower containment pressure increase. The hot debris in the drywell raises the gas temperature and eventually fails the drywell head flange seal. Containment failure is predicted to take place before drywell melt-through in both MAAP and BWRSAR/MELCOR.

4.2.1.3 References

1. "Decay Heat Power in Light Water Reactors," Revised American National Standard, ANSI/ANS-5.1-1979, 1979.

2. EPRI NP-7071-CCML

3. Dingman, S.E., Shaffer, C.J., Payne, A.C., and Carmel, M.K., "MELCOR Analyses for Accident Progression Issues," NUREG/CR-5331, SAND89-0072, January 1991.

4. Greene, S.R., Hodge, S.A., Hyman, C.R., and Tobias, M.L., "The Response of BWR Mark II Containments to Station Blackout Severe Accident Sequences," NUREG/CR-5565, ORNL/TM-11548, May 1991.

TABLE 4.2.1-1
Results for Short-term Station Blackout Sequence without ADS Actuation.

Plant	WNP-2	La Salle	Synthetic Mark II CPI Plant
Computer Code	MAAP 3.0B	MELCOR/LTAS	BWRSAR/MELCOR
Rated Power	3323 MWt	3293 MWt	3293 Mwt
Total Containment Volume	488,130 ft^3	526,880 ft^3	520,294 ft^3
Core Uncovery	47 min.	36 min.	38 min.
Core Melting	123 min.	71 min.	91 min.
Core Plate Failure	215 min.	281 min.	156 min.
Lower Head Penetration	215 min.	428 min.	246 min.
RPV Bottom Head Failure Mode / Flow Type of Melt out of Vessel	Gross Rupture / Slump-type Melt	Melt-through of Instrument Tubes / Flow-type melt	Melt-through of Instrument Tubes / Flow-type melt
In-Vessel Hydrogen Generation	582 kg (1281 lb)	1100 kg (2425 lb)	1014 kg (2235 lb)
Pressure Increase Due to Blowdown	0.41 MPa (60 psia)	0.19 MPa (28 psia)	0.37 MPa (54 psia)
Containment Failure due to overtemperature or overpressure	908 min ($T_{drywell}$ = 700°F)	No Containment Failure	600 min ($P_{wetwell}$ = 135 psig)
Percentage of debris staying in the pedestal	34%	100% (assumption)	100% (assumption)
Containment Pressure Increase Rate Due to MCCI	0.052 psia/min	0.16 psia/min	0.18 psia/min
Time for Pedestal Floor Melt-through / Floor Thickness	1564 min / 3.3 ft	20 min due to failure of the drain lines	730 min / 3 ft

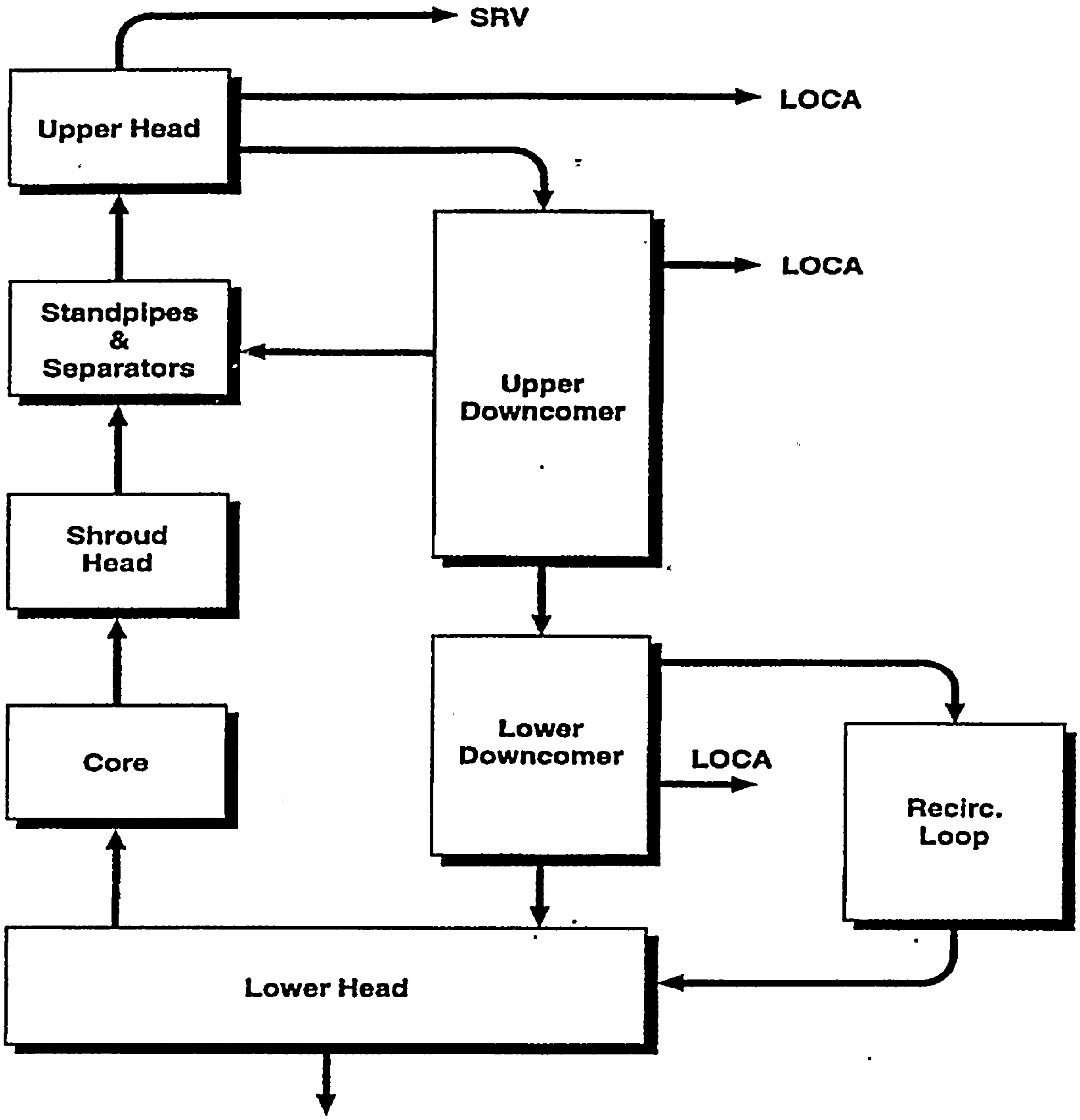

Figure 4.2-1 BWR Primary System Nodalization

4.2.2 Steam Explosions

In the event of a severe reactor accident leading to core melt, it is possible that molten fuel materials will come into contact with water, producing a steam explosion. Steam explosions can occur in the lower plenum inside the vessel, the pedestal area, or in the suppression pool.

4.2.2.1 Definition of the Steam Explosion Concern

Steam explosions are energetic interactions that sometimes occur when a molten material and liquid are brought together. The rapid transfer of thermal energy between the two materials occurs over a time scale of the order of milliseconds. If the energy is translated to mechanical work, it has the potential to cause significant damage.

The containment could be threatened by three possible damage mechanisms due to steam explosions:

(1) a solid missile such as the vessel upper head generated from the impact of a liquid slug (composed of molten fuel, water and structural material) accelerated by the vapor explosion,

(2) dynamic liquid phase pressure on structure,

(3) static overpressurization of the containment by steam production (steam spike).

4.2.2.2 Description of the Physical Process

Steam explosion processes involve rapid and coherent transfer of thermal energy between the hot molten fuel and the cold volatile liquid (water). Experimental observations suggest that a large-scale explosion progresses through four phases; (1) mixing, (2) triggering, (3) propagation, and (4) expansion.

When the molten fuel enters the coolant, several mechanisms (Rayleigh-Taylor instabilities, Kelvin-Helmholtz instabilities, boiling effect, etc.) act to break up the fuel stream into smaller particles (diameter ~ 10 mm). This is termed as the mixing phase. The coolant begins to vaporize at the fuel-coolant liquid interface as a vapor film separates the two liquids.

In order for an energetic explosion to occur, a trigger must be present in the system. Triggers are pressure or temperature disturbances that lead to the destabilization and collapse of the vapor film. The fuel and coolant are forced into intimate contact, accompanied by rapid heat transfer and formation of new interfacial area. Possible causes of these pressure disturbances are chemical interactions such as metal/water reactions, or solid/solid impact such as falling objects.

Once a trigger occurs in a region of pre-mixed fuel and coolant, the vapor film surrounding the fuel droplets collapses, and rapid heat transfer between the fuel and the coolant occurs. The collapse of the vapor layer surrounding the fuel droplets initiates further breakup of the droplets into fine fragments (diameter ~ 0.1 mm). This process propagates through the fuel-coolant mixture causing further fuel breakup and explosive vapor generation.

During the expansion phase, the high pressure coolant vapor expands against the inertial constraint of the surroundings and the mixture itself. Part of the fuel internal energy is transformed into the kinetic energy of the mixture and the surroundings. If the explosion process occurred inside the reactor vessel, the upward-directed kinetic energy could potentially damage the reactor vessel head and generate a missile to breach the primary containment. Traditionally this process is designated as alpha-mode failure. If the explosion occurred outside the vessel, the dynamic liquid phase shock waves, the high pressure vapor produced, and the slug kinetic energy can all do destructive work on containment structures.

4.2.2.3 Steam Explosions in WNP-2

In-vessel Steam Explosions

Comprehensive risk assessment efforts have been made to assess the likelihood of alpha-mode failure of large dry containments, in a hypothetical, unmitigated, low-pressure, core melt scenario [References 1-7]. The Reactor Safety Study [1] published the first attempt to quantify the likelihood of a steam-explosion-induced containment failure. More recently, Theofanous and coworkers applied a probabilistic framework for this purpose by taking account of recent steam explosion research findings [7]. Table 4.2.2-1 summarizes the results from different studies. In general, these results support the common expectation that in-vessel explosions do not pose a significant threat to containment.

Although the results in Table 4.2.2-1 are primarily for large dry containments, NUREG-1150 concludes that for the pressure-suppression containment types the likelihood of alpha-mode failure is low relative to other containment failure mechanisms [6]. For WNP-2, a probability of 10^{-4} is assigned.

If the reactor coolant system is at high pressure, the probability is set at 10^{-5}, one order of magnitude below the probability associated with the low-pressure case. This approach is similar to that used in NUREG-1150. The assigned probability reflects the experimental observation that high ambient pressure tends to suppress the occurrence of steam explosions.

TABLE 4.2.2-1
Quantification of Alpha-mode Failure.

Study or Authors	Estimate
Reactor Safety Study [1]	Probability = 10^{-2} (best estimate) 10^{-1} (upper bound)
Corradini & Swenson [2]	Probability = 10^{-4} (best estimate) 10^{-2} (upper bound)
Bohl & Butler [3]	Probability = 10^{-2} (upper bound)
Theofanous & Saito [4]	Probability $\sim 10^{-4}$
Swenson & Corradini [5]	Probability $< 10^{-4}$
NUREG-1150 [6]	medium relative frequency = 4×10^{-5}
Theofanous, Najafi and Rumble [7]	total probability with frequency $> 10^{-4}$ (per core melt) = 1.3×10^{-4}

Ex-vessel Steam Explosions

The potential for steam explosions to occur in a Mark-II suppression pool has been studied by Corradini and co-workers [8]. The containment analyzed is similar to that of WNP-2, in which the downcomers are located outside the pedestal region, and the containment wetwell wall is freestanding. The analysis showed that the dynamic pressure-time impulse from a steam explosion is highly unlikely to threaten the pedestal wall and the wetwell wall. Therefore, a containment failure probability of 0.001 is assigned.

When the RPV pressure is low at the time of vessel failure, the melt will be released to the containment as a low velocity jet. Ex-vessel steam explosions could occur in either the pedestal region or the drain line attached to the sump region. During normal plant operation, the average depth of water in the sump is about 17". Therefore, it is possible for steam explosions to take place in this area. Corradini [9] analyzed the pressure loads in LaSalle for explosions occurring under two conditions; (1) pedestal full of water and (2) pedestal empty with drain pipe water filled. The predictions suggest that the dynamic explosion pressure (~ 20 MPa or 2900 psia, pulse duration ~ 0.01 sec) maybe large enough to damage the drain line. Both the WNP-2 and LaSalle designs incorporate a deep in-pedestal drywell cavity, and have drain lines which penetrate the in-pedestal drywell floor. Therefore, for WNP-2, it is assumed that steam explosions in the pedestal region are likely to damage the drain line and cause wetwell bypass. Furthermore, the high temperature melt could also fail the drain line even if no steam explosion occurs. The probability of drain line failure is thus assigned to be 0.9.

When the RPV pressure is high at the time of vessel failure, the melt will be released to the containment as a high velocity jet. Significant fragmentation of the melt could happen. Four Sandia tests (SPIT-15, SPIT-17, HIPS-4W, and HIPS-6W) were conducted to investigate the effects of the discharge of pressurized material into water preexisting in a scaled test cavity that represents the Zion plant [10]. The results suggest that steam explosions occurred in the experiments and destroyed the model cavities. Therefore, a probability of 0.9 is conservatively assumed for structural failure of the containment due to steam explosion. Since it is questionable to extrapolate the small-scale results directly to the WNP-2 containment, sensitivity studies have been performed to investigate the impact on the source term and the results are presented in Section 4.9.

Steam Spikes

With its extensive pressure suppression capability, the Mark II containment is not susceptible to steam spikes. Moody et al. [11] analyzed ex-vessel steam pressure spikes in BWR Mark II containments. The predicted maximum pressure increase is less than 0.1 atm ($\sim$1.5 psid). This is substantially less than the containment design value. In conclusion, steam spikes would not challenge the containment integrity.

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4.2.3 Direct Containment Heating

4.2.3.1 Definition of the Direct Heating Concern

Direct containment heating (DCH) has been identified by several PRA (probabilistic risk assessment) studies as a major contributor to the probability of early containment failure during postulated severe accidents. DCH is relevant only to molten corium dispersal into the containment atmosphere following its release from a high pressure reactor vessel. In NUREG-1150, DCH is considered as possible if the reactor vessel pressure at the time of vessel breach is greater than about 200 psia [1]. This indicates DCH is possible for WNP-2 on the condition that leads to complete loss of RPV depressurization capability.

4.2.3.2 The Physical Process of Direct Containment Heating

DCH is characterized by forceful expulsion of the melt from the pressurized vessel, extensive fragmentation and dispersal of a large mass of molten corium into the containment atmosphere, rapid oxidation of the metallic components of the melt, and rapid transfer of this exothermic chemical heat and the corium sensible heat to the containment atmosphere [2].

To achieve the condition for direct containment heating, a high pressure expulsion of molten corium would have to occur in such a way that massive dispersal of finely fragmented metallic liquid into the containment atmosphere would take place. Although sufficient energy for this phenomenon would be available from the liquid corium in this scenario, even without oxidation, it has been one of the IDCOR conclusions on this issue (Issue #8) that fine particles of corium are not likely to be generated since a potential mechanism for this type of massive dispersal has not been found [3].

To date, experimental and analytical efforts have concentrated on pressurized water reactor systems. Experiments contributing to the current understanding of DCH have been performed at SNL, ANL, BNL, and Fauske and Associates, Inc. The experiments, which are sponsored by NRC, EPRI and IDCOR, include JETA-B Tests, the HIPS Program, Surtsey-DCH Tests, CWTI and Simulant Tests, Small-scale Simulant Tests for DCH Cutoff Pressure, Wood's Metal Simulant Tests. Reference 4 summarizes the findings from these experiments. Analytical tools such as the CONTAIN code with its DCH model, and the multi-dimensional KIVA code are still in the stage of verification against the experimental results [5,6]. At the present time, no verified models are available for estimating containment loads accompanying high pressure melt ejection and direct containment heating.

Applications of DCH models to boiling water reactor systems are extremely sparse. Reference 7 is one of the few examples that can be found in the open literature. In this work, Murata and Louie performed parametric CONTAIN calculations to estimate the containment response of the Grand Gulf plant [7]. The estimated pressure increase is about 4×10^5 Pa (58 psia) in the drywell.

4.2.3.3 Conclusions for WNP-2

A high pressure expulsion of molten corium during reactor vessel failure at high pressure is possible, although not very likely in WNP-2 (mainly because of the high redundancy for the depressurizing function, with eighteen valves, seven of which have double means of relief actuation plus instrument air backup supplies). In addition, no physical mechanism has been found for the hypothesized massive dispersal of finely divided metallic liquid into the containment atmosphere. Furthermore, past NUREG-1150 studies for Mark I plants showed that the melt progression in BWRs is likely to be a flow-type melt instead of a slump-type melt [8]. These factors tend to lower the likelihood of DCH.

In the rare event that DCH occurs, the drywell pressure increase can be approximated from that for Grand Gulf based on the following scaling factor [7,9]:

$$\text{Scaling Factor} = \frac{\left(\dfrac{\text{thermal power}}{\text{drywell volume}}\right)_{\text{WNP-2}}}{\left(\dfrac{\text{thermal power}}{\text{drywell volume}}\right)_{\text{Grand Gulf}}} = \frac{\left(\dfrac{3323\ \text{MWt}}{200{,}540\ \text{ft}^3}\right)}{\left(\dfrac{3883\ \text{MWt}}{262{,}000\ \text{ft}^3}\right)} = 1.118$$

$$(\Delta P_{\text{Drywell}})_{\text{WNP-2}} = \text{Scaling Factor} \times (\Delta P_{\text{Drywell}})_{\text{Grand Gulf}} = 65\ \text{psia}$$

The resultant pressure load is lower than the containment failure pressure for WNP-2 (which is 121 psig, see Section 4.3).

The issue of DCH can be summarized as follows [10]:

"The fundamental issue for direct containment heating is, whether a massive dispersion of particulated core debris into the drywell atmosphere would be possible. Five steps can be identified which would be necessary for such an outcome:

First, the reactor vessel must fail at high pressure and in such a way that a majority of the core debris is ejected from the reactor pressure vessel into the pedestal cavity.

Second, the containment geometry must be such that debris dispersal, as a result of high velocity steam from vessel blowdown following total ejection of core debris, could occur from the pedestal cavity.

Third, fine fragmentation of debris must occur and must be sustained.

Fourth, the debris particles must remain dispersed and be distributed throughout the containment atmosphere.

Fifth, the particles must remain airborne long enough to transfer energy and react chemically if in contact with steam.

Major impediments to these processes which render direct containment heating an unlikely phenomenon for BWRs like WNP-2 are the inherent inability to entrain and finely particulate debris due to the pedestal configuration that does not promote debris entrainment, and the presence of numerous structures in the drywell which highly promote debris deentrainment. Also, for the Mark II containment, the mitigating effects of the suppression pool make containment failure due to direct containment heating (if it could occur) very unlikely."

Accordingly, an appropriately low value of 1.0E-3 has been assigned to the conditional probability of DCH causing early containment failure.

4.2.3.4 <u>References</u>

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4.2.4 Hydrogen Generation and Combustion

4.2.4.1 Background of the Hydrogen Combustion Concern

The main concern with hydrogen combustion in the containment is that the high pressure generated might breach the containment and cause a release of radioactivity. During normal operating conditions, hydrogen combustion is precluded from occurring in a BWR Mark II drywell and wetwell because these atmospheres are inerted with nitrogen. Only if hydrogen is released to surrounding secondary building can combustion occur (secondary building is not modeled in the WNP-2 IPE). For short periods prior to shutdown and during startup, the plant Technical Specifications permit deinerting of the containment. Only then could hydrogen be a concern.

Attention to hydrogen combustion has been greatly increased following the TMI-2 accident on March 28, 1979. At the time of the accident there was great uncertainty about the behavior of the hydrogen-air mixture in the containment; it was feared that a detonation might take place and damage the containment building. About 40 percent of the total zirconium in the core of the TMI-2 reactor was estimated to be oxidized, producing about 460 kg of hydrogen [Reference 1, Section 4.3.1]. Estimates place the hydrogen concentration before combustion at 7.9 percent, and after combustion at 1.1 percent, so that the combustion utilized a concentration of 6.8 percent [Reference 1, Section 4.6.3]. The combustion occurred at about 09:50 hours into the accident, causing a very short, sharp peak in containment pressure [Reference 1, Figure 3-1] and a (more slowly decaying) peak in temperature. The temperature peaked at 650 C (1,200°F). The pressure went from +0.1 bar gage steeply to a peak of slightly below +2.0 bar gage (29 psig; 1 bar = 14.5 psi), then almost immediately back down to +0.2 bar gage. Duration of the pressure spike was no more than 200 seconds [Reference 2, Section 4.7.3.1 and Figure 4-33]. Based on the pressure and temperature responses, it is concluded that the combustion was a relatively slow deflagration [Reference 1, Section 4.6.3]. The deflagration posed no serious threat to the containment building [Reference 1, Page III]. Thus the TMI-2 accident has illustrated that hydrogen deflagration could produce a very high temperature in the containment atmosphere. Nevertheless, the resulting pressure increase (1.9 bar or 27.6 psi) is not necessarily extreme on a large scale. However, had the hydrogen concentration been more than twice as high as it was, a detonation would have been possible, although not very likely; a detonation would have resulted in a much higher pressure peak.

As a result of the TMI-2 accident, the decision has been made to have all BWR Mark I and Mark II containments inerted with nitrogen such that the oxygen mole fraction is less than 5%, which basically elimiates the hydrogen combustion threat. This is because hydrogen combustion can occur if hydrogen concentration is greater than about 6%, oxygen concentration greater than 5%, steam concentration less than 55%, and an ignition source exists [3]. As a result, the possibility of hydrogen combustion is avoided for inerted containments.

Consequently, the concern recently has focused on those occasions where the oxygen concentration in the containment atmosphere temporarily reaches 5 percent or more. This could occur during the process of inerting while in Mode 2 (power ascension) or during deinerting while decreasing power.

4.2.4.2 The Physical Process of Hydrogen Generation and Combustion

4.2.4.2.1 Hydrogen Generation

Hydrogen can be generated during postulated severe reactor core damage scenarios. According to Sherman [4], the sources of hydrogen generation can be grouped into two mechanisms: rapid production mechanisms and slow production mechanisms. In the rapid mechanisms, hundreds of kilograms of hydrogen could be generated in tens of minutes or less. While in the slow mechanisms, substantial amounts of hydrogen can be generated in tens of hour or longer. The rapid sources of hydrogen could be due to:

> **zirconium-steam reaction:** zirconium and water vapor react at high temperatures when there is insufficient cooling;
>
> **steel-steam reaction;**
>
> **molten core-water reactions;**
>
> **molten core-concrete interactions.**

The slow sources of hydrogen could be due to:

> **radiolysis of water by gamma radiation:** This source is comparatively small, partly because of recombination of hydrogen and oxygen, which occurs in parallel with the radiolytic decomposition of the H-O-H molecule. The net amount of hydrogen generated from this process is expected to be of potential significance only over extended periods of time during the post accident portion of a postulated severe accident.
>
> **corrosion of zinc-based paint and galvanized steel;**
>
> **corrosion of aluminum:** If the containment spray contains sodium hydroxide (for iodine removal), it will react with certain strongly reducing metals such as aluminum. However, this case does not apply because sodium hydroxide is not used as a spray additive at WNP-2.

The reaction between the Zr and steam is believed to be the main source of the hydrogen burned at Three Mile Island Unit 2. Therefore, only this mechanism is discussed here.

In the zirconium steam reaction, zirconium appears to be a superior reducing agent, compared to hydrogen. Given the required energy of activation, the H-O-H molecules dissociate (endothermic process in which the H-O bonds are broken), and then the oxygen molecules penetrate the normal oxide layer on the cladding surface (which cracks at elevated temperatures) and oxidize the zirconium metal. This dissociation and oxidation is described by the equation [Reference 5, Sections 7.121 through 7.130]:

$$Zr(s) + 2H_2O(g) \quad \text{--->} \quad ZrO_2(s) + 2H_2(g)$$

with the enthalpy of reaction being

$$DH_R = -594.6 \text{ KJ per mol of } ZrO_2(s) \text{ or per mol of } Zr(s), \text{ or}$$

$$DH_R = -297.3 \text{ KJ per mol of } H_2O(g) \text{ or per mol of } H_2(g)$$

Hydrogen keeps accumulating for as long as a hot zirconium-steam interface is available. Hydrogen blanketing, depletion of steam, fuel pin slumping and melt formation, or submergence in cold water will impede or terminate the hydrogen generation process.

The reaction rate of zirconium oxidation increases strongly with temperature; Glasstone [Reference 5, Page 464] gives an equation for the rate constant k:

$$k = 3{,}300 * \exp\{-22{,}900/T\} \qquad \text{with the temperature T in Kelvin}$$

For a few selected values of T we have:

Temperature	Rate Constant
T = 290°C = 563°K	k = 7.14E-15
T = 650°C = 923°K	k = 5.54E-08
T = 980°C = 1,253°K	k = 3.81E-05
T = 1,204°C = 1,477°K	k = 6.10E-04
T = 1,400°C = 1,673°K	k = 3.75E-03

Normal operating temperature for the outer surface of the zircaloy cladding is 285.9°C (559.1°K, saturation temperature at 70 bar). At this temperature the rate constant is very small; a thin, stable oxide layer forms, which prevents further oxidation. Very little hydrogen is generated below 650°C [Reference 1, Page 4-8]. Once the cladding temperature reaches about 980°C, the heat from the zirconium oxidation becomes appreciable compared to the nuclear decay heat. 1,204°C (2,200°F) is the acceptance limit for LOCA evaluation as specified in Appendix K to 10CFR Part 50. At about 1,400°C (Reference 6, Page 155, gives 1,800°K; Reference 7, Page 319, gives 1,400°K to 1,500°K for "beginning" of hydrogen generation), the reaction rate increases dramatically. The rate constant at 1,400°C is above the acceptance limit by a factor of 6: 3.75E-3/6.10E-4 = 6.1, and it is above the normal operating value by a factor of 3.75E-3/5.36E-15 = 7.03E+11, that is 703 billion. At about 1,870°C the zircaloy begins to melt. The process of zirconium oxidation is eventually stopped by hydrogen blanketing, when hydrogen gas accumulates in quantities sufficient to limit access to the cladding surface for water vapor. In addition to zirconium oxidation, steel oxidation occurs at appreciable rates when the temperature goes above 1,380°C. The melting temperature for carbon steel is 1,460°C.

Table 4.2.1-1 shows that the amount of hydrogen generated in-vessel is about 1280 lb for short-term station blackout sequence without ADS actuation. The calculated gas concentrations in the drywell after vessel breach (at approximately 3.76 hours) are 17% hydrogen, 13.6% nitrogen, and 69.4% steam. .

4.2.4.2.2 <u>Hydrogen Combustion</u>

The possible forms of combustion include ordinary deflagration, diffusion flames, accelerated flames, and detonations. A deflagration is a combustion wave that travels at subsonic speeds relative to the unburned gas. The pressure loads developed are quasi-static loads. A diffusion flame is one in which the burning rate is controlled by the rate of mixing of oxygen and fuel. Detonations are combustion fronts that travel at supersonic speeds relative to the unburned gas. Detonations can develop transient pressures that are much higher than those obtainable in an adiabatic isochoric complete combustion (AICC). Accelerated flames are a form of combustion intermediate between deflagrations and detonations.

The existence of diffusion flames is not considered to pose a direct threat to containment integrity in a severe accident since the resulting overpressure is quite low. Accelerated flames may give dynamic loads qualitatively similar to those caused by a detonation if the effective flame speed is high enough. Therefore, only the pressure loadings that may be expected from deflagrations and detonations will be considered.

Combustion could happen if the gas concentrations (hydrogen, oxygen, and steam) are within certain limits and an ignition source is present. It is commonly assumed that 6% hydrogen concentration is the lower limit for deflagration and 13% is the lean detonation limit.

Deflagrations

Hydrogen deflagration test data for reactor safety evaluations have been obtained in nine different test vessels varying in volume from 0.3 to 2100 m^3 [8,9]. The maximum pressure increase following deflagration measured from the experiments can be found in References 4 and 9. For hydrogen concentration between 6 and 12%, the normalized pressure rise is between ~0.14 and 4. Therefore, if the initial containment pressure is 15.4 psia, the containment pressure increase due to deflagration ranges from about 2 to 62 psia. This is lower than the containment failure pressure for WNP-2, which is 163.0, 136.0, and 118 psia at 70°F, 340°F, and 600°F.

In conclusion, deflagration alone will not challenge containment integrity. However, the pressure load generated from deflagration combined with that from vessel blowdown or core-concrete interaction may challenge containment integrity.

Detonations

Detonations can be developed by direct initiation or flame acceleration. Direct initiation requires a large energy input (~ 10 kJ) while ignition of deflagration requires only tenths of a millijoule. Therefore, it is likely that combustion in a detonable mixture will begin as a deflagration which may subsequently undergo a deflagration to detonation transition. Detonations have been achieved mostly under very special experimental conditions. Geometric arrangements which are suitable for generating exceptionally high turbulence during the chemical reaction seem to be favorable for achieving detonations. Experiments performed by the Sandia National Laboratories for the deflagration to detonation transition indicate that: with obstacles (high turbulence), a minimum of 15 volume percent of hydrogen in air is required; without obstacles (low turbulence), a minimum of 25 volume percent of hydrogen in air is required [Reference 10, Figure 2-13]. This does not mean that transition to detonation will occur above those concentrations; it only means that transition to detonation is possible, if geometric conditions are favorable.

Currently there is no suitable tool for predicting the pressure loads resulting from detonation in complicated geometries. The pressure loads predicted for simple geometries can be found in Reference 8. The generated pressure spike in the lean detonation limits (hydrogen concentration in the range of 13 to 15%) is about ten times the containment pressure prior to the occurrence of detonation. Therefore, it is highly likely that the developed transient pressure would challenge the integrity of the primary containment.

4.2.4.3 The Potential for Failure of the WNP-2 Containment due to Hydrogen Combustion

Frequency of Hydrogen Combustion

To estimate the frequency of hydrogen combustion in the WNP-2 containment, we consider the probability for two conditions to coincide:

1. The containment atmosphere has an oxygen content of 5 percent or more, and

2. The containment atmosphere has a hydrogen content of 6 percent or more.

The WNP-2 Technical Specifications in Section 3.6.6.2 require the drywell and suppression chamber atmosphere oxygen concentration to be less than 3.5 percent by volume within 24 hours after thermal power reaches 15% during startup and within 24 hours prior to going below 15% during shutdown.

Inerting involves changing the oxygen content of the containment from 21 percent to below 5 percent. This process takes 2 to 4 hours. For power ascensions, if we conservatively estimate that inerting is started about 15 hours after reaching 15 percent power, the maximum time for the containment atmosphere to be above 3.5 percent oxygen would be about 15 + 4 = 19 hours. If we assume the same time span for a power descent, and if we further assume 2 ascents and descents per year requiring inerting of the atmosphere, we have a total annual exposure time of

$$T = (4 * 19) / ([365 - 45] * 24) = 9.9E\text{-}3 \quad (<1\%) \sim 1E\text{-}2$$

for having an oxygen content of 5 percent or more in the containment.

Potential for Containment Failure due to Hydrogen Combustion

Combustion is assumed to occur 100% of the time that hydrogen reaches combustible range. This is very conservative because most scenarios which yield large quantities of hydrogen also produce an atmosphere with more than 55% steam. Once a combustion event starts, it is conservatively assumed that the likelihood of detonation is the same as that of deflagration. Also, it is assumed that detonations will fail the containment. Therefore, the potential for hydrogen combustion that damage the containment is

P = frequency of hydrogen combustion x probability of combustion that fails the containment
= 1E-2 x 0.5 = 5E-3

4.2.4.4 References

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4.2.5 Molten Core-Concrete Interaction

Based on the design configuration of WNP-2, if molten corium ejected from a failed reactor vessel (lower plenum) following an unlikely catastrophic event, it would immediately impinge on the pedestal floor and pedestal wall, and start attacking the concrete slab. One potential failure mechanism for the containment is the overpressure or overtemperature resulting from molten core-concrete interaction (MCCI). This failure mechanism is partly attributable to concrete ablation, and it can be analyzed by computer codes such as MAAP. MAAP results for a short-term station blackout sequence with ADS actuation indicate that the containment pressurization rate is about 0.07 psia/min. Therefore, it is likely that the overpressurization could induce late containment failure.

If no effective quenching mechanisms are available, MCCI may erode a large enough concrete volume to cause the pedestal wall to lose its load-carrying capability for the reactor vessel. Subsequently, the vessel might shift its position, and the containment penetrations might be torn out. MCCI may also erode the drywell floor and cause the molten corium to fall into the suppression pool. For the WNP-2 design, there is a huge pool of water directly below the pedestal floor. Although energetic fuel-coolant interactions (FCIs) could occur, Section 4.2.2 indicates that the dynamic pressure from a steam explosion is highly unlikely to threaten the pedestal wall and the wetwell wall. Nevertheless, FCIs are likely to disperse the corium and cause a rapid heat transfer from the melt to the water. This is expected to lead to a successful quenching of the debris; therefore, the containment basemat failure is not a concern [Reference 1].

The extent of concrete erosion due to MCCI is examined in the following sections.

4.2.5.1 Sequence of Thermal Attack

A number of experiments have been conducted by Sandia National Laboratory (SNL) and German research center KFK [Reference 1] in the last few years to enhance the understanding of phenomenological aspects of MCCI and to validate computer codes that are currently available, such as DECOMP (A subroutine of MAAP Code), CORCON and WECHSL. It is found that the sequence of thermal attack on the concrete can be broken into three phases:

PHASE (1)-Jet Attack

Initially, the molten corium is ejected from the vessel failure site as a of mixture of UO_2, Zr, ZrO_2, steel, and steel oxide. This jet stream, at a temperature of ~2000°C to 2800°C, leaves the pressure vessel swiftly and impinges on the pedestal surface locally. The overall duration of this severe attack, however, is short, lasting only from seconds to a few tens of seconds. During this time, the attack depth is small compared to the full dimension of concrete slabs. Analysis indicates that the typical penetration of a jet attack will be ~10 to 20 centimeters [References 1 and 2].

This short term, localized phenomenon is highly influenced by whether a water pool exists. The water content on the pedestal surface or in the sump will greatly reduce the seriousness of attack.

PHASE (2)-Early Aggressive Attack

After the phase of jet attack, the bottom of the pedestal cavity will then be covered at full length by the high temperature corium that initiates the second phase - "Early Aggressive Attack" to the concrete floor as well as the concrete wall.

In this phase, several chemical reactions with the concrete will occur at the following various temperatures as it is being heated up from between ~30°C to 40°C [Reference 3]:

(i) vaporization of free water completed at ~ 300°C

(ii) liberation of chemically bound water completed at ~ 600°C

(iii) melt and decomposition of the concrete occuring at ~ 1100°C - 1400°C

The first 2 reactions produce water which reacts with metallic constituents of corium to generate hydrogen. The third reaction can produce CO_2 which in turn reacts with high temperature metals in the debris to form carbon monoxide (CO). Both H_2 and CO are combustible.

One of the main characteristics of this attack mode is the vigorous stirring of molten corium by release of gases. This will induce corium circulation and promote convective heat transfer between corium and the concrete substrate. The pedestal material will be decomposed very fast. This attack, however, cannot be maintained for long due to the following reasons:

- the combination of sensible heat added to the concrete, the endothermic chemical reactions in releasing H_2O vapor and decomposing the concrete, and the latent heat of fusion for melting the substrate exhausts a considerable amount of energy from the incoming corium; and

- the concrete decomposition can produce some random slags into the pool, thus increasing its viscosity, and significantly reducing the heat transfer.

The aggressive attack, in general, requires more energy than can be generated by decay power from the corium. During the Early Aggressive Attack, such additional energy comes from exothermic chemical reactions and the initial stored energy carried by the corium. The exothermic chemical reactions result from oxidation of the metallic constituents by steam and CO_2 released from the pedestal concrete. It should be noted that the WNP-2 type concrete has very little $CaCO_3$ compared to the more commonly used Limestone type concrete

[Reference 4]. Likewise, the CO_2 generations from Basaltic type concrete during MCCI are relatively small and insignificant. The pedestal concrete of WNP-2 is of Basaltic type [Reference 5].

PHASE (3) - Long-Term Self-Limited Attack

At the end of Early Aggressive Attack, the heat transfer modes are limited to conduction and radiation. This is, however, not sufficient to transfer the sustained decay power from the debris and the exothermic heat from the gas-metal reactions, since concrete is a very poor heat conductor. Therefore, it is difficult to form large scale solidification of molten debris. Only a thin crust in the order of centimeters could be maintained. The concrete thermal attack still continues but in a slow and quasi-steady manner. The amount of concrete ablation can then be estimated from the internal energy generation minus the heat losses through the upper surface.

With the gaseous products (steam and CO_2) still forming during this slower thermal attack, the thin crust will intermittently be broken by the rising bubbles. The molten corium from the top may then slip through the crust and attack the surrounding concrete. In short, this type of attack is characterized by relatively low corium temperature (~2000°F), rise and fall of crust, and sustained concrete ablation at a slow and self-limited progression [References 3 and 6].

During this long-term attack, the debris remains at an essentially constant temperature. The concrete attack rate is much less than that in the earlier phases, and occurs over a much longer interval. This mode of attack is the part of MCCI that could eventually threaten the integrity of containment.

4.2.5.2 Controlling Physical Processes

The underlying physical processes that affect the extent of the thermal attack for each of the three phases described above are briefly described here. All of the major controlling processes are closely interrelated and difficult to separate. They are listed in the following:

- corium expulsion from the reactor vessel
- corium-water dynamic interaction and melt spreading
- debris mass configuration on the concrete
- melt bed deepening
- debris quenching due to water.

In WNP-2, there is one small pathway between pedestal cavity and drywell floor - a 7'x3' opening doorway. Because of this pathway, debris could be settling on the inside as well as the outside of the pedestal cavity depending on the vessel pressure during release. While the physical processes of the thermal attack on either the inside or the outside of the pedestal concrete should be similar, their ramifications may be different. The impact of the MCCI taking place outside of the pedestal will be discussed separately in Section 4.2.6.

The initial velocity of the melt spread depends on the vessel pressure and the failure size. As the initial spreading rate decreases, there is an increase in the time for heat transfer from the melt to concrete floor or overlaying water. The heat transfer rate is a strong function of water/debris dynamic interaction. If interactions between the melt and water cause the debris to disperse or violent oscillations at the melt-water interface, the heat transfer rates can be far greater than the rate associated with critical heat flux boiling. If not, the melt-water interface would undergo film boiling.

After the molten debris initially spreads out, two types of debris configurations are possible:

- a discontinuous debris bed with high porosity; or
- a continuous slab or partially molten pool.

Typically, a continuous slab configuration occurs because there is usually less water available for debris fragmentation. However, if there is sufficient water to quench, a discontinuous debris bed has been shown to occur. It is also possible that a discontinuous debris bed can evolve into a continuous debris slab. If insufficient porosity of the bed limits the dryout heat flux to less than decay power, the debris bed would heat up and eventually melt into a continuous debris configuration. The debris configuration strongly affects the quenching capability of overlaying water.

The depth of the corium pool also affects how effectively energy can be moved out. If the initial debris layer is so thick that the rate of heat removal from the debris is less than the internal heat generation rate, then the debris would reheat and ultimately remelt. Further spreading of molten or partially molten debris could then occur. The spreading can also be aided by gas agitation from concrete erosion. The debris would spread until the heat removal rate is greater than the internal energy generation, or it is constrained by adjacent walls. In Reference 7, the NRC has stated that a debris layer less than 25 cm in depth may be considered coolable.

The last process to discuss is quenching due to water. For a debris bed, the physical mechanism of cooling is water ingression into the bed coexisting with outflow of steam from the bed. The coolability limit for debris beds is a hydrodynamic limitation within the bed itself that strongly depends on the bed porosity, and less strongly depends on the corium depth. For a thin debris slab or shallow pool, conduction is an effective heat transfer mechanism and the slab can cool quickly with little or no cracking. However, for a thicker slab or deeper pool, coolability requires cracking of the slab or overlaying crust and ingression of water into the debris. Such cracking would be expected to occur as a result of the volume reduction associated with debris cooling and phase change, as well as due to bubbling of offgases produced by any thermal attack of concrete.

4.2.5.3 Containment Failure Modes due to MCCI

Erosion of concrete by molten corium to the pedestal could deteriorate the load-carrying capability of the pedestal walls sufficiently that the reactor vessel moves and could cause gross mechanical failures in the penetrations of piping connected to the reactor vessel. In general, MCCI could result in the following 4 scenarios:

(1) The corium is cooled in the pedestal based on the conditions of the debris quantity, its size, and configuration.

(2) The corium is not coolable. While attacking both pedestal floor and wall, the corium may erode through the pedestal floor and fall into the suppression pool. This may potentially threaten the integrity of the basemat. However, for WNP-2, this may not be considered as a failure scenario due to the cooling capability from the underlaying suppression pool. The debris will eventually settle in the wetwell and be quenched.

(3) Before the pedestal floor is melted through, the pedestal wall has been eroded sufficiently that containment integrity is lost.

(4) Some of the corium may be released through the 7'x3' opening doorway to the drywell floor, peripheral seal, steel liner, etc; and thermally attack them as soon as encountered.

For WNP-2, the primary concerns are either Scenario (2), (3) or (4). Scenario (3) is a late containment failure mechanism and would only be expected to occur many hours after reactor vessel failure. Scenario (4) will be discussed specifically in Section 4.2.6. Evaluation of the extent of concrete erosion is described in the following and documented in the documentation retained at the Supply System.

DETERMINATION OF THE CORIUM DEPTHS IN THE PEDESTAL

Based on NRC document [Reference 7], the existing experimental evidence suggests that core debris beds with depths greater than 25 centimeters may not be coolable. To determine whether a core debris bed is coolable, a calculation for corium layer thickness on the pedestal floor was performed. The debris bed depth can be represented by

$$X_c = \frac{V_c - V_s}{A} = \frac{f_1(1125.9 + 429.2f_2) - 175}{322} \text{ ft}$$

where

- V_c: volume of total core debris in the pedestal,
- V_s: volume of sumps (FDS & EDS) inside the pedestal,
- A: area of pedestal floor,
- f_1: fraction of molten corium released from core,
- f_2: fraction of M_{lh} (mass of lower plenum head and miscellaneous structures) that will be melted and dropped into the pedestal.

Depending on the degree of accuracy of f_1 and f_2, an estimate of corium depth X_c in the pedestal can be calculated. For an initial estimation for WNP-2, f_1 is ~ 40% (the majority of the balance will be in the drywell floor and suppression pool), and f_2 is ~ 50% (recommended by Reference 1). As a result, the corium depth in the floor, $X_{c,1}$ is estimated to be 1.12 ft (34.2 cm). In addition, the corium depth in the sump, $X_{c,2}$, is calculated to be 2.79 ft (85.0 cm).

The depth of the corium is 1.12 ft for 67.5% of the pedestal area and 2.79 ft for 32.5% of the area. Therefore, the average corium depth is

$$X_c = 0.675 \times 1.12 + 0.325 \times 2.79 \text{ ft} = 1.66 \text{ ft}$$

According to the NRC coolability criterion of 25 cm, sustained thermal attack is highly likely for WNP-2 in a severe accident.

DETERMINATION OF THE EXTENT OF MCCI

The results from the previous section indicate that the WNP-2 corium pool is not coolable. The erosion to the pedestal from debris thermal attack should thus be performed.

The following assumptions are made to form the bases:

- Sidewall gas bypasses the corium pool, hence it does not contribute chemical reactions, and heat.
- Zirconium is the only important source of chemical energy (comparing with Cr and Fe) in the core debris.
- Ratio of the erosion speed between sideward and downward, $r_s = U_s/U_d$, is a constant (=0.29).
- Erosion of the upper portion of the pedestal wall by radiative heat transfer is insignificant relative to the direct attack of the core debris from the corium pool in terms of containment failure.
- If MCCI causes the pedestal floor failure prior to pedestal wall failure, the corium will instantaneously be released into the suppression pool. Fuel-coolant interaction will proceed and eventually quench the debris, MCCI will then stop, and containment failure will not occur.
- Unless the core debris has eroded through the pedestal floor, MCCI will continue until containment failure occurs.

First, we need to determine whether the Zirconium in the corium has been depleted before the erosion process terminates. This is because the Zr makes a difference in the energy source term in the mathematical formulas to be used for time interval calculation. The presence of Zr will add energy due to the exothermic chemical reaction. The maximum amount of concrete erosion that can occur while Zr is present, $m_{cn,zr}$, is estimated to be 2.44×10^5 kg.

The maximum amount of concrete that can possibly be eroded in any situation based on our failure criterion, M_{max}, is the amount of erosion occurring when the pedestal floor is completely melted through. This is evaluated to be between 1.82×10^5 and 2.73×10^5 lb.

It is seen that $m_{cn,zr}$ is comparable to M_{max}. Therefore, Zr is likely not to be depleted when the corium melts through the pedestal floor. Thus, the Zr chemical reaction will continuously be a heat source during MCCI before failure of the pedestal floor.

Next step is to estimate the extent of pedestal erosion as a function of time. The derivation can be found from Eq. (3-11) of Reference 1. For any given sideward ablation depth $X_{c,s,c}$ in the concrete wall, the estimated time interval for erosion is shown in Table 4.2.5-1.

4.2.5.4 Conclusions for WNP-2

Based on the NRC criterion of coolability, the thermal attack of molten corium to the WNP-2 pedestal concrete may be sustained after vessel failure. The generated noncondensable gases and the presence of hot debris could cause late containment failure by increasing the containment pressure and temperature. However, the extent of concrete erosion would not jeopardize the containment integrity based on the following considerations:

From the load-carrying design of WNP-2 pedestal, the concrete thickness has been specified to provide approximately a safety margin of 2 [Reference 8]. In other words, the WNP-2 pedestal wall can sustain a MCCI up to a sideward erosion thickness of ~2.5 ft, which, according to Table 4.2.5-1, will occur within 43 hr. However, from the same table, as the time interval reaches 13.5 hour after MCCI starts, the pedestal floor has been melted through, and the corium would have fallen into the suppression pool. Loss of containment integrity due to concrete erosion, therefore is not a concern for WNP-2.

TABLE 4.2.5-1
Erosion Thickness vs. Time Interval (r_s = 0.29).

Sideward erosion thickness (ft), $X_{c,s,c}$	Downward erosion thickness (ft), $X_{c,d,c}$	Time interval after the start of MCCI, (hr)
0.5	1.72	6.27
0.75	2.59	10.01
0.97	3.33	13.47
1.0	3.45	14.04
1.25	4.31	18.34
1.45	5.00	21.96
1.75	6.03	27.64
2.0	6.90	32.00
2.5	8.62	43.06

4.2.5.5 References

1. "Molten Core-Concrete Interaction", FAI/91-113, August 1991 (Boro Malinovic to David Chan, 8/21/1991).

2. Mississippi Power and Light Co., "Grand Gulf Nuclear Station - Integrated Containment Analysis," IDCOR Technical Report 23.1GG, IDCOR, 1985.

3. FAI, "Preliminary Documentation of Phenomenological Models for Assessing Core-Concrete Interactions," IDCOR Task 15.3, April 1982

4. G. A. Greene, "CCI in the Mark I Containment Drywell and Local Liner Failure," NUREG-1079, Appendix B, Pg B-23.

5. Construction Technology Lab, Inc. to R. O. Vosburgh, "Chemical Analysis Report (Project 212654)," Dated 2/13/91

6. FAI, "Final Report on Core-Concrete Interactions," IDCOR Subtask 15.3, March 1983 .

7. <u>NRC Generic Letter 88-20, Individual Plant Examination for Severe Accident Vulnerabilities</u> - 10CFR50.54(f), Pg 1-8, Nov 23, 1988

8. Personal Communications with J. Twomey, March 1, 1992

4.2.6 Drywell Floor, Shell, and Peripheral Seal Failure

During a hypothetical, severe accident the reactor core is assumed to uncover, and later melt into a state of molten corium. Without proper cooling, this corium may eventually breach the pressure vessel and eject into the pedestal area.

When the RPV pressure is low at the time of vessel failure, the melt will drain to the pedestal by gravity. It will end up staying inside the pedestal cavity causing molten core concrete interactions (MCCI) as discussed in Section 4.2.5.

When the RPV pressure is high at the time of vessel failure, the melt will be released to the containment as a high velocity jet. Part of the corium may be dispersed to the drywell as fine particles through an opening doorway between pedestal cavity and drywell. That corium flowing to the drywell may simply stay locally in a certain domain of the drywell floor or possibly travel far enough to reach the containment shell and drywell floor peripheral seal. The MCCI on the drywell floor should basically follow the same sequences as the MCCI inside the pedestal cavity. It is, therefore, only the ramifications resulting from the current geometry, and the possibilities of corium attack of the containment shell and the drywell floor peripheral seal that will be discussed here.

4.2.6.1 Vessel Failure from A Depressurized State

If ADS is functional and activated before the vessel failure occurs, the vessel will then be depressurized. The debris would be ejected out of the vessel at a relatively low speed or drained out by gravity and the effects of jet attack would be minimized. Since the only pathway to the drywell floor is through a small 7'x3' doorway which is located at 9.375 ft (see Figure 4.1-6) above the top surface of the pedestal floor, the chances for debris to directly bounce upward and "flee" to the outside in a dynamic blowdown manner should be small and negligible. However, the maximum depth of the corium pool formed inside the pedestal cavity must be computed for comparison to the net distance between the doorway and the pedestal floor surface. If the former occupies an elevation greater than the latter, the molten corium will overflow to the pedestal floor. Whereas, in the opposite case, the MCCI will be contained inside the pedestal cavity, and thus the drywell floor and other peripheral structures will be kept relatively free of corium.

The corium height can readily be computed from (Eq. 2) of Section 4.2.5:

$$X_c = \frac{f_1 \cdot (1125.9 + 429.2\ f_2) - 175}{322} \quad \text{ft}$$

where
X_c = corium bed depth in the pedestal
f_1 = fraction of corium staying in the pedestal
f_2 = fraction of lower plenum steel melted into corium

To do a bounding (conservative) evaluation, it is assumed both f_1 and f_2 are 100%. Thus

$$X_c = [\ 1125.9 + 429.2 - 175\ ]\ /\ 322$$

$$= 4.29 \quad \text{ft}$$

This value of 4.29 ft is much less than the 9.375 ft depicted in Figure 4.1-6. Therefore, it is concluded that molten corium will be confined in the pedestal region. Melt-through of the pedestal floor due to MCCI is highly possible. However, MCCI should not be a risk to the floor and other structures in the drywell from a depressurized vessel.

4.2.6.2 Vessel Failure from A Pressurized State

If ADS has not been activated during an accident, the corium would be discharged into the pedestal at a higher speed from the pressure-elevated reactor vessel. Some corium might be momentarily entrained and swept out of the pedestal through the door opening and land on the drywell floor.

The amount of the corium that may end up reaching the drywell is a function of both vessel pressure and cavity geometry. In addition, modeling of the in-vessel melt progression could also impact the calculated results. In the BWRSAR analysis for a short-term station blackout without ADS actuation [1], melt release is characterized by an initial large pouring rate (lasting for about 50 minutes). This is followed by a slower rate (lasting for about 110 minutes) which is controlled by the decay heat level. Only part of the molten material from the initial pour could be swept out into the drywell and most material would be retained in the pedestal region. In the MAAP code, most molten material pours rapidly through the vessel penetrations for a period of about 1 min. The debris could be entrained from the pedestal to the drywell by the high velocity gas stream existing the vessel. About 66% of the melt is predicted to be transported into the drywell. Past NUREG-1150 studies for Mark I plants indicate that the melt progression in BWR is more likely to be a flow-type melt predicted by BWRSAR than the slump-type melt predicted by MAAP 3.0B [2]. Therefore, the amount of core debris splattered from the pedestal floor and escaping through the pedestal doorway could be small.

Regarding cavity geometry, the height of the door opening is 9.375 ft above the bottom floor. The possible pathways of corium jet stream include the melt stream impacting the pedestal floor and being deflected, or the melt being atomized by the gases into fine particles. Either way the corium will have to bounce backward 9.375 ft high from the pedestal floor and turn to the small 7'x3' opening door. This configuration is expected to limit the amount of debris that can be effectively transferred to the drywell.

COOLABILITY ASSESSMENT

If the dispersed corium falls on the drywell floor, the depth of accumulation can be calculated in a similar manner as in the derivation of Eq. 2, Section 4.2.5 except without the sump volumes. The corium depth in the drywell floor $X_{c,dw}$ is formulated in the following:

$$X_{c,dw} = \frac{f_{dw} f_s \; (1125.9 \; + \; 429.2 \; f_2)}{r_{sp} \; A_{dw}} \quad \text{ft} \qquad \text{(Eq1)}$$

where

f_{dw} = fraction of total corium being transferred to the drywell floor from reactor vessel

f_s = fraction of the corium staying on the drywell floor without draining to the suppression pool

f_2 = fraction of fuel plate being melted and dropped into the pedestal
= 0.5

A_{dw} = total drywell floor area (excluding downcomers)
= 4240.5 ft^2 (Pg 5.4.3, Ref. 3)

r_{sp} = ratio of corium spreading area to total drywell floor area

For a conservative estimation about corium coolability, one may use f_{dw} = 60%, and f_s = 80%; this leaves (Eq 1) into

$$X_{c,dw} = \frac{(0.6) \; (0.8) \; (1340.5)}{r_{sp} \; (4240.5)} \quad \text{ft} \qquad \text{(Eq2)}$$

In (Eq 2), r_{sp} is the ratio of the corium spreading area to total drywell floor area which corium occupies. The following table lists a few representative values of r_{sp} and its associated $X_{c,dw}$:

r_{sp}	$X_{c,dw}$ (cm)
0.2	23.1
0.5	9.25
0.75	6.17
1.00	4.62

Using the NRC 25 cm criterion for corium depth coolability [4], one can readily find that WNP-2 drywell floor to be coolable based on this conservative estimation. During an accident, if emergency cooling water such as containment spray is available, then MCCI on the drywell floor should not become a risk to its integrity.

In the case that no emergency cooling water is available, the hot debris cannot be cooled. The debris may impact the shell as a rapidly flowing jet and cause early containment failure. The drywell liner failure probability for Peach Bottom reported in NUREG-1150 is between 0.6 and 0.8 [5]. Since the probability of debris impingement on the shell is indeterminate, a value of 0.5 is used. Therefore, for WNP-2, the shell failure probability is: P = probability of debris impingement on the shell x probability of shell failure = 0.5 x 0.8 = 0.4. Sensitivity study performed to investigate its effect on the source term is presented in Section 4.9. If no early containment failure occurs, long-term MCCI could cause erosion of the downcomers or the omega seal. This will lead to suppression pool bypass.

4.2.6.3 References

1. Greene, S.R., Hodge, S.A., Hyman, C.R., and Tobias, M.L., "The Response of BWR Mark II Containments to Station Blackout Severe Accident Sequences," NUREG/CR-5565, ORNL/TM-11548, May 1991.

2. Kelly, D.L., Jones, K.R., Dallman, R.J., and Wagner, K.C., "An Assessment of BWR Mark II Containment Challenges, Failure Modes, and Potential Improvements in Performance," NUREG/CR-5528, EGG-2593, pg. 29, June 1990.

3. WNP-2 Calculation File, NE-02-90-25

4. <u>NRC Generic Letter 88-20, Individual Plant Examination for Severe Accident Vulnerabilities</u> - 10CFR50.54(f), Pg 1-8, Nov 23, 1988

5. NUREG-1150, pg. C-78, U.S. Nuclear Regulatory Commission, December 1990.

4.2.7 Containment Overtemperature and Overpressure

The conditions following a degraded core accident could challenge the structural integrity of the containment. The resulting pressures and temperatures may occur rapidly over a period ranging from a few seconds to a few minutes, or more slowly over several hours or days. Since the rate of increase may have significant influence on determining the ultimate mode of containment failure, it is important to distinguish between gradual and rapid pressure (or temperature) rises. Therefore, the loads are categorized into short term and long term loadings.

Short Term Loadings

Phenomena that could lead to pressure or temperature rises in a containment over a short period of time include high-pressure melt expulsion from the vessel, direct containment heating, deflagration or detonation of combustible gases, and rapid generation of steam through molten fuel-coolant interaction. Certain accident sequences, such as ATWS or large LOCA with failure of pressure suppression capability, could also generate rapid pressure rise to breach the containment. Short term loadings are likely to cause early containment failure (at or around the time of vessel breach). In some accident sequences, containment breach could release superheated steam to the ECCS pump rooms and lead to loss of injection. Core meltdown will follow. This type of failure is termed as "very early containment failure" in the current submittal.

A large or catastrophic failure is likely to be generated if the ultimate containment capacity is achieved rapidly during short term loadings. In this work, "containment failure mode is large" represents a catastrophic rupture, which is defined as the loss of a substantial portion of the containment boundary with possible disruption of the piping systems that penetrate or are attached to the containment wall. Since the flow rate of gas and aerosol out of the containment is high, large amounts of fission products could be released to the environment.

Long Term Loadings

Contributors to long term pressure and temperature buildup include steam created by heatup and boiling of the suppression pool, noncondensable gases from molten core-concrete interaction, and radiative and convective heating of the containment atmosphere by the hot debris. Containment failures induced by long term loadings tend to occur substantially after vessel failure. This allows time for radioactive material to deposit within the containment building and increase the opportunity for operator actions such as venting.

Gradual pressurization of the containment is likely to induce relatively small rupture areas. This type of small failure is considered more likely than the large failures generated by short term loadings. "Containment failure mode is small" represents a leak, which is defined as a containment breach that would arrest a gradual pressure buildup and would depressurize the containment within 2 hours. MAAP results show that either a leakage area equal to 28 in^2 (the failure area due to steam overpressurization reported in Section 4.3) or opening the 24" purge lines will effectively depressurize the containment in 1 hour.

4.2.7.1 Relationship to Source Term

Other than the noble gases, fission products are released in the form of aerosol with particle sizes ranging from less than 1 micrometer to as much as 10 micrometers. Substantial deposition or retention of these aerosols within the containment could occur depending on several factors such as the containment failure location, failure size, failure timing, and specific accident sequence characteristics.

For Mark II containments, failure locations that do not force flow through an effective pool or spray scrubbing mechanism before exiting the containment represent the largest concern. A plant specific calculation [1] identifies three failure locations of the containment shell due to overpressurization. Failure is believed to be equally likely to occur at these three locations. The drywell head seal could also fail due to containment overtemperature. Four failure locations are thus considered credible:

- Drywell near the upper cone/upper cylinder junction. This would result in suppression pool bypass although some credit could be taken for fission product retention by the reactor building.

- Drywell near the lower junction between the shell and the equipment hatch. This would also result in suppression pool bypass although some credit could be taken for fission product retention by the reactor building.

- Wetwell above the water line. In this case, no suppression pool bypass would occur.

- Drywell head. Suppression pool bypass would occur although some credit could be taken for fission product retention in the refueling bay.

Note that in this submittal, the reactor building and the refueling bay are not modeled. Therefore, fission product decontamination effects in these two areas are not considered.

Regarding failure sizes, the larger the hole in the containment, the more rapid the escape of fission products to the environment. The time available for fission products to deposit in the containment building will thus be reduced.

If release flow is not expected to pass through a water pool or spray mechanism, failure timing will be crucial from a source term perspective. Significant amount of fission products could be released to the environment in cases with early or very early containment failure. This is because during the entire accident the overall airborne fission product is the largest in the immediate time period of vessel failure. In cases with late containment failure, substantial fission product (including revaporization of the deposited materials) could be retained through naturally occurring deposition mechanism such as diffusion, impaction and gravitational settling.

The role played by specific accident sequences is best described by a few examples. ATWS sequences are often expected to cause containment overpressure failure hours ahead of vessel failure, while station blackout sequences are associated with containment overpressurization long after vessel failure. The availability of water in the containment prior to containment failure and the operability of engineered safety features, which are determined by accident sequences, could affect the pressurization process and retention of the fission products (as an example, the presence of water pool in the drywell or spray operation may significantly reduce the source term).

4.2.7.2 Reference

1. Shrivastara, H.P., WPPSS Calculation ME-02-91-77, Rev. 0, September 1991.

4.3 Containment Failure Characterization

4.3.1 WNP-2 Containment Failure Location and Modes

To discuss the failure location and modes, the containment loading and the containment strength analysis must be presented.

Containment Loading

The predominate loading progression for WNP-2 can be characterized by the following simplified sequence of conditions. A combined pressure-temperature rise along the water saturation curve will be followed by a period of rising temperature with little additional pressure increase, and then another period in which conditions follow the saturation curve in temperature and pressure. Depending on the sequence, containment failure can occur during any of these stages. The events causing these conditions are as follows;

Steam overpressure loading occurs while decay heat is transported to the suppression pool in the absence of suppression pool cooling. This occurs before vessel failure while the core is covered with water and the decay heat is being transferred by water/steam flows by either RHR, via the SRVs, or through a pipe break inside containment. Temperature and pressure conditions in both the drywell and the wetwell are approximately those of saturated water at the pool bulk temperature (for example, 103 psig at 340°F). The containment air tends to relocate to the drywell but enough remains in the wetwell so that the pool is slightly subcooled.

Overtemperature loading occurs after core cooling is lost, the core melts and the vessel fails and the debris is located in the pedestal without a continuous water supply to the pedestal. Decay heat is transferred to the pedestal concrete by conduction and to the drywell by radiation and convection. Pressure increases will be slight because the concrete is basaltic, so little non-condensible gas release occurs. There will be an increase in containment temperature at vessel failure but the jump will be limited. The upper regions of the drywell can reach temperatures of 400 to 900°F.

Steam overpressure loading can occur as the third phase if the pedestal floor collapses because of concrete ablation before the containment shell fails elsewhere. The core debris will relocate into the pool and the decay heat will be transferred directly to the water. The pressure and temperature conditions in the drywell and the wetwell will tend to return to equilibrium with the pool bulk temperature saturation values.

Fast pressure rise events such as vessel rupture, very large LOCAs, ex-vessel steam explosions and hydrogen combustion have been found to occur at low frequencies. In these cases it was assumed that gross failure would occur because of the uncertainty associated with the dynamic response of containment to acute temperature and pressure loading. Loss of pressure suppression because of vacuum breakers failing open can occur but the pressure loading tends to follow the steam overpressure case as there is usually flashing water involved in the primary system.

Containment Strength

Section 4.3.2 describes the calculation performed to obtain the containment shell pressure capabilities and leak area estimates. Conservative (low) strain limits were used to develop the pressure capability, and engineering judgement was used to develop the leakage area values. Estimates were made for containment temperatures of 70°F, 340°F and 600°F. Steam overpressurization failures will occur around the 340°F value. Overtemperature failures will occur at 600°F or higher. No significant containment loading occurs at 70°F so only the higher two temperatures are quoted in the following table. No estimate was made of the uncertainty levels for these values which are quoted in the following table.

COMPONENT	INITIAL TEARING PRESSURE (PSIG) AT 340°F	PRESSURE FOR 28 IN^2 LEAK (PSIG) AT 340°F	INITIAL TEARING PRESSURE (PSIG) AT 600°F	PRESSURE FOR 28 IN^2 LEAK (PSIG) AT 600°F
Equipment Hatch	105	121	88	103
Wetwell above stiff'nrs	121	121	103	103
Drywell upper cone	122	122	104	104

The leakage area for the vacuum breaker valves is estimated as 28 square inches assuming the complete failure of their non-metallic seals. This failure is assumed, as well as the butterfly valves in series with them. The pressure and temperature to cause these failures is not known and the report states that these valves may never attain the high containment temperatures and may never leak at all.

The three shell failures are all found to be of the leak-before-break type caused by material tearing. It is likely that a tear will occur of limited size sufficient to prevent further pressurization but without progressing to gross failure causing rapid depressurization. The calculation file [1] states the following:

> "..the failure first occurs in the wetwell region above the horizontal Tee stiffeners where the wall thickness is 1 5/16". Then a pressure increase of approximately 1 psi (insignificant increase) would cause a failure at the junction of the upper cone and the upper cylindrical vessel in the drywell.

Since the loading is gradually applied, a crack is expected to form and grow, at the weakest point at these locations, and then to stop when the leakage area is sufficient to relieve the containment pressure. Also the probability of failure at the second location is somewhat higher than the first location as this location is at the weld joint between a conical vessel and a cylindrical vessel. Also the change in wall thickness is significant in the second location.

Before the above (failure) pressures are reached, the leakage to the secondary containment may occur through the equipment hatch ...The equipment hatch leakage area .... has large uncertainty...

The high plastic strains (at the equipment hatch) are highly localized.."

The drywell head seal is expected to fail when the temperature reaches about 700°F, although the actual leakage area is somewhat uncertain. The estimate for Shoreham, provided in NUREG/CR-5528 indicates that:

Drywell temperature > 700°F, leakage area = 0.075 in^2
Drywell temperature > 800°F, leakage area = 7.2 in^2

This indicates an expected leakage area which approximates that of a leaking penetration, but, because of the uncertainty associated with the specific applicability of these estimates to WNP-2, in the Level 2 analysis it was assumed that the failure was equivalent to the "small" category shown in the CETs. This failure mode category represents a leak large enough to remove enough energy to limit further containment pressurization.

Containment Response to Chronic Overpressure

Long term loss of containment heat removal, hypothesized containment failure and consequential loss of injection represents an important class of sequences for the WNP-2 Level 1 analysis. The specific conditions associated with these sequences and their effects on possible containment response is described by the following.

In a long term loss of containment heat removal, energy is continuously transported from the core to containment by the water which is recirculated from the pool through the core and back to the pool. The pool temperature will rise monotonically with time over a period of several hours. Two outcomes are possible.

i) If the recirculating system does not fail at a temperature below 350°F then the containment will be subject to the steam overpressure loading. Based upon the insights from the containment strength analysis, it is safe to assume that a small membrane tear will be initiated with equal likelihood at any one of the three identified locations. It is possible, but unlikely, that failures could occur simultaneously at two or more locations. However, since the rate of increase of the effective net area of the breaks will stabilize at the point that there is balance between the energy removed by the break and the energy released to the containment, the effects on containment pressure will be similar whether there are single or multiple leaks. The only area of concern is one of whether there is a significant change in the probability that "small" containment failure will initiate a leak in the wetwell and cause suppression pool bypass if two or more simultaneous leaks are considered. This is a second order effect, in which the first failure (leak caused by membrane tear) should limit further containment pressure increases and prevent a second failure. This is unless the two failures occur at exactly the same time. General uncertainties and differences in local materials and their associated stresses under overpressure load make this case sufficiently unlikely that its possibility can be discounted and excluded from the WNP-2 analysis.

Failure in the wetwell at elevation 476′ will allow steam into the nominal 2 1/4″ gap between the concrete shell and the biological shield. Some of this presumably can flow into the reactor building at some elevation near this around penetration sleeves. Some may go higher but probably most will enter the reactor building at about this elevation.

Failure in the drywell at the equipment hatch will probably allow most of the steam to enter the reactor building at elevation 492′ (around the hatch) but some may go lower and some higher.

Failure in the drywell at the upper cylinder will probably allow steam to escape upwards to the refuelling floor but depending on the sealing arrangements around the drywell head some steam may flow in to the reactor building above elevation 492′.

This steam flow will equilibrate to a flow of some 1000 pounds per minute of steam. This steam will be escaping at > 300°F and > 100 psig. It will expand into the building at less than 20 psia and will become superheated at approximately 220°F and in a larger volume. The building blowout panels will soon fail and the steam will find paths to the outside environment. It is reasonable to assume that the steam will permeate at least all the available volumes above elevation 476′. This may lead to failure of systems providing cooling flow to the core and lead to situation ii) described below.

The pool will not "flash" as the ruptures are not expected to be catastrophic. The air in the containment will be swept out and the top of the pool will become saturated, but there will still be subcooling at the bottom of the pool due to the hydrostatic head above it. It is not known at this time whether the pumps can continue to operate under these conditions and they are assumed to fail.

If the pumps do not fail then the situation is stable and the core will not melt. The pool inventory is sufficient to maintain this state for many days.

ii) If the injection pumps fail before the containment fails, then heat addition to the pool will cease. If the operator has not depressurized the primary system before this point, and attempts to do so, the increase in the rate of energy addition to the pool may be enough to increase pool temperature to the point that containment fails. In any event, the core will become uncovered, melting and vessel failure will follow. This in turn may lead to conditions which result in overtemperature loading. Whether the containment will fail due to overtemperature or whether the pedestal floor will collapse first and result in containment failure due to overpressure will depend upon the characteristics of individual accident scenarios.

4.3.2 Containment Strength Assessment

The following summarizes the results of an analysis undertaken to assess containment strain under several regimes of pressure and temperature loading.

There is a 2¼ inch gap between the containment shell and the biological shield wall. The gap is filled with compressible insulation material consisting of polyurethane flexible foam sheets. Because of the relative compressibilities of polyurethane, containment shell and biological shield wall, polyurethane is neglected in the analysis. Figure 4.3-1 shows the containment thermal expansion as a function of temperature for the lower cylinder (wetwell) and the upper cylinder (below head flange). For the lower cylinder, the gap is 2¼ inches at 70°F and zero at 650°F. For the upper cylinder, the gap is 2¼ inches at 70°F, and 1.45 inches at 650°F. Figure 4.3-2 shows the containment pressures required to elastically strain the containment to close the gap at various temperatures. For the lower cylinder, the containment pressure required is 261 psig at 200°F. At 340°F, which is the containment design temperature, the containment pressure required is 180 psig. At 650°F, the containment pressure required is of course zero. For the upper cylinder, the gap cannot be closed without a containment pressure of over 1400 psig at 650°F.

In Section 4.3.2.1 that follows, the containment failure pressure and location are determined without considering the biological shield wall. The containment failure pressures are determined to be 148.0, 121.0, and 103.5 psig at 70°F, 340°F, and 600°F, respectively. Failure is equally likely to occur at the upper cone/upper cylinder junction or in the wetwell airspace. See Figure 4.3-3. However, in reality, the containment lower cylinder will be in contact with the biological shield wall at 103.5 psig and 600°F (representative of severe accident condition) while the containment upper cylinder will not. These results are based on the assumption that the containment is a shell of revolution without any penetration.

Penetrations are analyzed in Section 4.3.2.2 without the effect of the biological shield wall. Main steam and reactor feedwater piping penetrations, personnel airlock, equipment hatch, drywell head flange, and electrical penetrations are analyzed. Results indicate that the critical location is at the lower junction between the containment shell and the equipment hatch barrel. See Figure 4.3-4. Local failure at this location could occur when the containment pressure reaches 88 psig while the containment temperature is 600°F.

From the reactor building standpoint, the load on the biological shield wall is of displacement type and not of the force type before containment shell failure. The containment shell will first come into contact with the biological shield wall where the containment diameter is the largest (below EL. 501′). When the contact or interference grows, failure of the biological shield wall will be in circumferential direction based on simple shell analysis. In other words an axial or vertical gap will develop along the biological shield wall below the 501′ elevation to relieve the displacement load from the containment shell. Since the load is of displacement type the biological shield wall is not expected to fail in a catastrophic fashion.

Conservative calculations (Section 4.3.2.2) indicate that at 88 psig containment pressure and 600°F containment temperature a crack would develop at the lower junction between the containment shell and the equipment hatch barrel. The exact crack size is difficult to determine. However, based on numerous MAAP runs, a crack size of 2″ effective diameter will just begin to depressurize the containment. A crack size of 8″ effective diameter will rapidly depressurize the containment. The containment may be depressurized at a rate depending on the size of the failure location. Release of steam and radionuclides to the reactor building will be through the vertical crack below 501′ elevation developed by the interference between the containment shell and the biological shield wall.

Based on the analysis, the local failure is likely to occur at a higher temperature (600°F). In most cases, MAAP results show that the failure pressure of 121 psig at 340°F will be reached first. Therefore, in this study the containment failure pressure is assumed to be 121 psig.

4.3.2.1 Expected Containment Failure Pressures and Locations

A clean shell model of the WNP-2 primary containment was analyzed to obtain the failure pressure. This model was generated using 741 4-node shell elements STIFF-43 of the ANSYS computer program. The model included the lower cylindrical vessel, the lower and upper truncated conical vessels, the upper cylindrical vessel, and the torospherical head. In addition, the seven tee ring circumferential stiffeners and the longitudinal stiffeners in the wetwell, the two box type and the two tee ring circumferential stiffeners attached to the lower conical vessel, and the circumferential plate stiffener welded to the outside surface of the upper cylinder are also included. The model, however, did not include the penetrations which were analyzed in Section 4.3.2.2.

The material yield strength, ultimate strength, and the elongation at failure for SA-516, Gr. 70 which was primarily used for WNP-2 containment were obtained from the tensile tests performed by Koon-Hall Testing Corp., Portland, Oregon under a Supply System contract. These values were used in the containment analyses. Like certified material test reports (CMTR) for many of the plates used in WNP-2 primary containment construction, Koon-Hall test results indicate significantly higher yield strength than the minimum specified by the Code. Although the test values were higher than those provided by the Code, they were less than the CMTR values. As Koon-Hall tests provided only the yield and ultimate strengths and the corresponding elongation data, the material was assumed to behave linearly in both elastic and plastic strain ranges. The yield point and the point corresponding to the maximum load on the true stress-true strain diagram were connected by a straight line to describe the material strain-hardening. The extremely low tangent moduli obtained in this manner had little or no effect on the failure pressures calculated for three different containment temperatures. Upon completion of the analyses at these temperature, it was noticed that the failure pressures were directly proportional to the yield strength indicating that the strain-hardening model considered in these analyses was useful only for obtaining rapid convergence of the solutions.

Elastic/plastic analyses, in several small steps of pressure increments, were performed (see Reference 1) to determine the failure pressure at which either of the following two limits on the equivalent plastic strain would reach first:

- 1% at the middle surface of the shell, and
- 2% at any other location

The three temperatures for which the failure pressures were calculated are 70°F, 340°F, and 600°F to be 148.0, 121.0, and 103.5 psig, respectively. At these pressures and temperatures, the equivalent plastic membrane strain for elements located in the wetwell region above the horizontal tee stiffeners where the wall thickness is 1 5/16″ was approximately 1% indicating failure. A pressure increase of 1 psi caused the equivalent plastic membrane plus bending strain at the junction of upper cone and upper cylinder to increase to nearly 2%. As the difference in the failure pressures at these two locations is only 1 psi, it is equally probable to fail at either location.

It is interesting to note that the failure pressure for the WNP-2 containment obtained in Reference 2 is 133 psig and its location matches with the first failure location indicated above. The failure pressure seems to be low as a hoop strain equal to twice the strain at yield was conservatively used as the failure criterion in that study. Also, Reference 2 utilized two dimensional axisymmetric elements, therefore, the effects of the longitudinal stiffeners were included by increasing the shell thickness to provide an average equivalent shell stiffness. Reference 1 utilizes more appropriate shell elements for modelling the containment. Furthermore, a direct comparison of Reference 1 results with those of Reference 2 cannot be made as the failure pressure would greatly depend upon the value of the yield strength utilized. This is not given in Reference 2.

The WNP-2 containment failure pressures provided above are conservative. The failure criterion used allows only 1-2% plastic strain before the failure is assumed to occur. However, the test performed in Reference 3 and the finite element analyses performed in Reference 4 after completion of the Reference 3 test indicates that the maximum principal strain at rupture for SA-516, Gr. 70 test vessel was nearly 16%.

4.3.2.2 Containment Leakage

The purpose of this section is to identify the potential leakage paths that may occur before the above containment failure pressure is reached. The WNP-2 primary containment design documents indicate that all openings, except a few small penetrations where the shell thickness is adequate to provide the required reinforcement area, are reinforced per the ASME Code requirements. However, as large discontinuity stresses at the junction of the two intersecting shells may cause plastic strains, the following large penetrations were analyzed, in Reference 1, using finite element technique:

- Main Steam and Reactor Feedwater Piping Penetrations
- Personnel Airlock
- Equipment Hatch
- Drywell head flange
- Vacuum breaker valves
- Electrical penetrations

The analyses for the first two indicate that the shell failure at the shell intersections would not occur at a pressure below the containment failure pressure. For the piping penetrations, it was assumed, based upon the study performed in Reference 5, that the differential pressure and the differential thermal movements between the penetration and the pipe supports in the reactor building would have insignificant effect on the containment shell and any failure or plastic hinge formation due to these displacements would occur at the pipe fittings.

The two doors of the personnel airlock are pressure seating type and have double O-ring seals. The design of these doors is such that both doors cannot be opened simultaneously. Assuming the inner door is open, the outer door, which was fabricated from thinner plate, was analyzed in Reference 1. The resilience of the O-ring seals were neglected in this analysis. The resulting upper bound leakage area at the containment failure pressure is 0.56 in^2 or 0.84 effective diameter.

The equipment hatch is a large opening with a finished opening diameter of 12′-6″. The Control Rod Drive (CRD) hatch is an integral part of the equipment hatch and has a small opening of approximately 23″. The equipment hatch cover is spherical in shape and is mounted on the barrel with 64 hex head bolts. The O-rings utilized in the bolted flange joint are of silicone rubber and are certified for one time use at 340°F. The analysis of this joint, ignoring the seal resilience, indicates the flange separation would not occur until the containment is pressurized to 253 psig. The design of the CRD hatch cover is also such that no significant leakage would occur through this opening. The containment shell and the equipment hatch barrel were also analyzed to evaluate the stresses and the plastic strains at the shell intersection which is obviously the most critical location from a failure viewpoint. The failure at this location occurred when the containment pressure reached 88.0 psig while the containment temperature considered was 600°F.

The drywell head bolted flange connection utilizes 124 bolts of 2½″ nominal diameter and the seal is provided by the silicone rubber O-rings. The hand calculation performed in Reference 1 indicates that the flange separation would not take place below 97 psig.

The containment vacuum breaker valves, connected in series with the containment isolation butterfly valves, open to the secondary containment. The vacuum breaker valves as well as the butterfly valves utilize elastomer seals. However, the butterfly valves are connected to the wetwell air space where the temperature is relatively low because of the suppression pool water. The vacuum breaker valves are down stream of the butterfly valves. If there is no leak in the butterfly valves, the vacuum breaker valves will not experience the pressure and temperature of the wetwell air space.

The WNP-2 electrical penetrations were supplied by the Westinghouse Electrical Corporation and the Conax Buffalo Corporation. The Westinghouse penetration assemblies are of two types. The first, a modular type, uses potting compound to seal the cables in the modules. The modules are clamped to a header plate and utilize ethylene propylene and silicone O-rings for sealing. Reference 6 presents the results of thermal endurance tests performed by Westinghouse over the temperature range of 70°F to 392°F. The tests indicated that no significant leakage would occur at the highest test temperature and a mean life of 129 hours could be expected. This is also confirmed by the tests performed at the Sandia National Laboratories on similar Westinghouse penetration assemblies. The results of these tests are documented in Reference 7.

Westinghouse has also performed several qualification tests on the canister type penetrations which are also installed at WNP-2. In these penetrations there are no exposed organic material. The seal is provided by ceramic bushings which can withstand temperatures up to 1100°F (see Reference 8). It is thus concluded that the canister type penetrations also do not have significant leakage potential.

Ref. 7 also provides results of the tests performed on the Conax penetrations similar to those used at WNP-2. These tests indicate that the leak integrity would be maintained during a severe accident condition. It may be noted that Conax penetrations at WNP-2 would perform better as unlike the test penetrations, where silicone O-rings were used for seal between the header plate and the nozzle flange, the header plate is welded to the nozzle.

4.3.2 REFERENCES

1. Supply System Calculation ME-02-91-77, Rev. 0

2. Greimann, L., et al., "Final Report - Containment Analysis Techniques - A state-of-the-Art Summary", NUREG/CR-3653, March 1984.

3. Koenig, L., "Experimental Results for a 1:8-Scale Steel Model Nuclear Power Plant Containment Pressurized to Failure", NUREG/CR-4216, Dec. 1986

4. Clauss, D., "Comparison of Analytical Predictions and Experimental Results for a 1:8-Scale Steel Containment Model Pressurized to Failure", NUREG/CR-4209, September 1985

5. U.S. NRC, "Containment Performance Working Group Report - Draft Report forComment", NUREG-1037, May 1985

6. Westinghouse Document No. 75-7B5-BIGAL-R2, "Predicting the Thermal Life of Modular Penetrations", May 1975

7. Clauss, D., "Severe Accident Testing of Electrical Penetration Assemblies", NUREG/CR-5334, Nov. 1989

8. Westinghouse Document No. PEN-TR-80-100, "Qualification Tests for 5 kv Medium Voltage Penetrations for Susquehanna #1 and 2", Sept. 1980

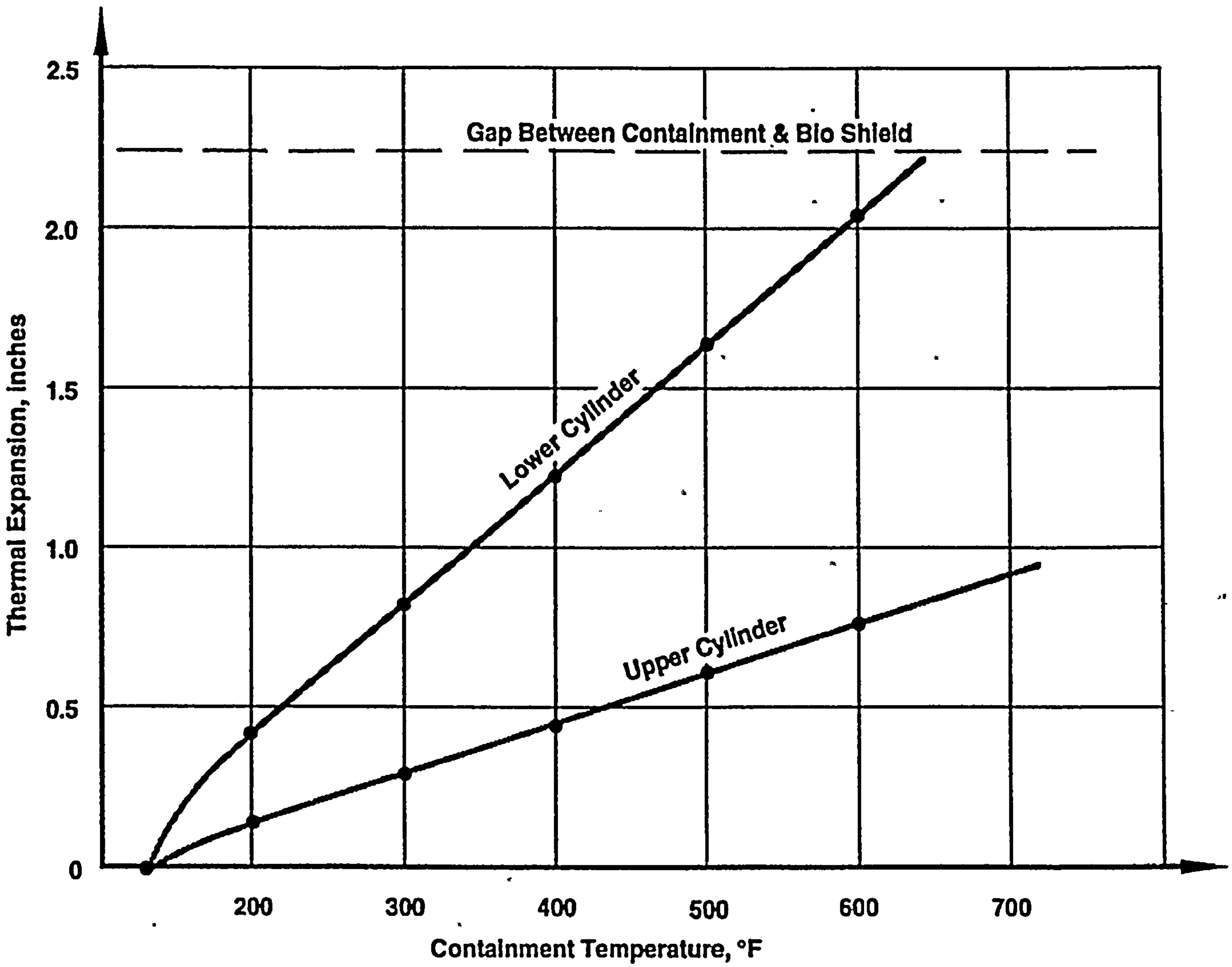

FIGURE 4.3-1 CONTAINMENT THERMAL EXPANSION

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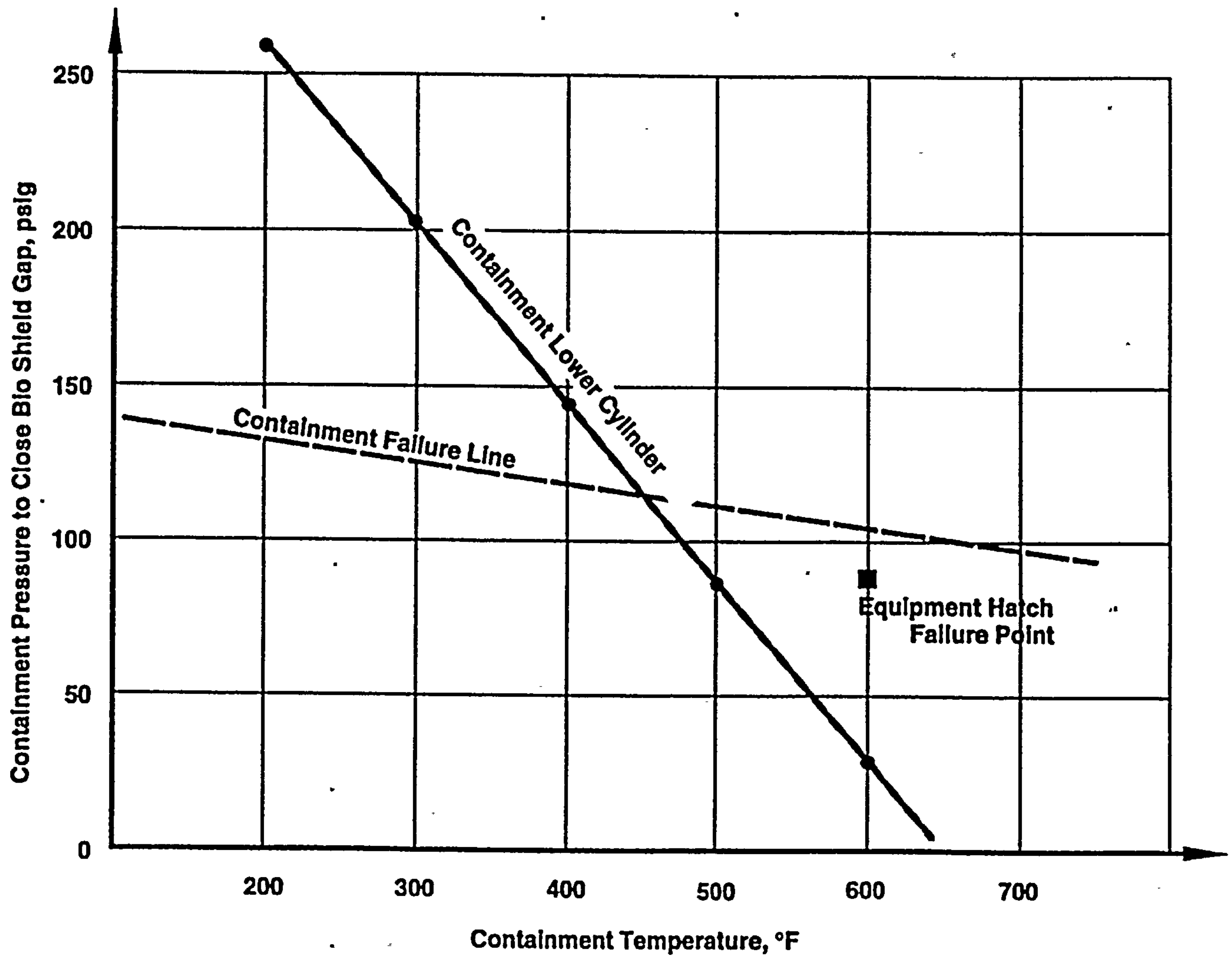

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Figure 4.3-2 Containment Pressure to Close Bio Shield Gap

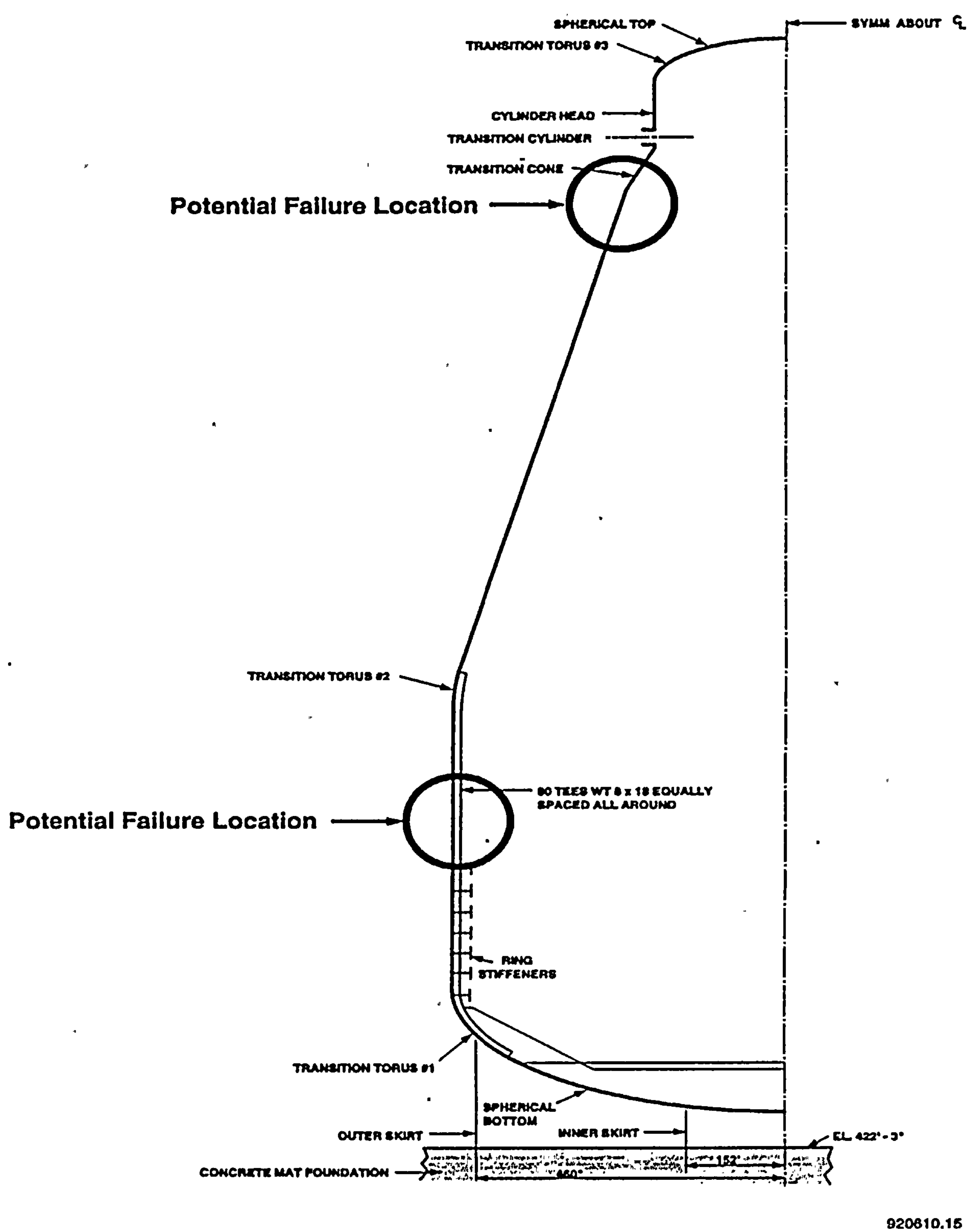

Figure 4.3-3 Other Potential Containment Failure Locations

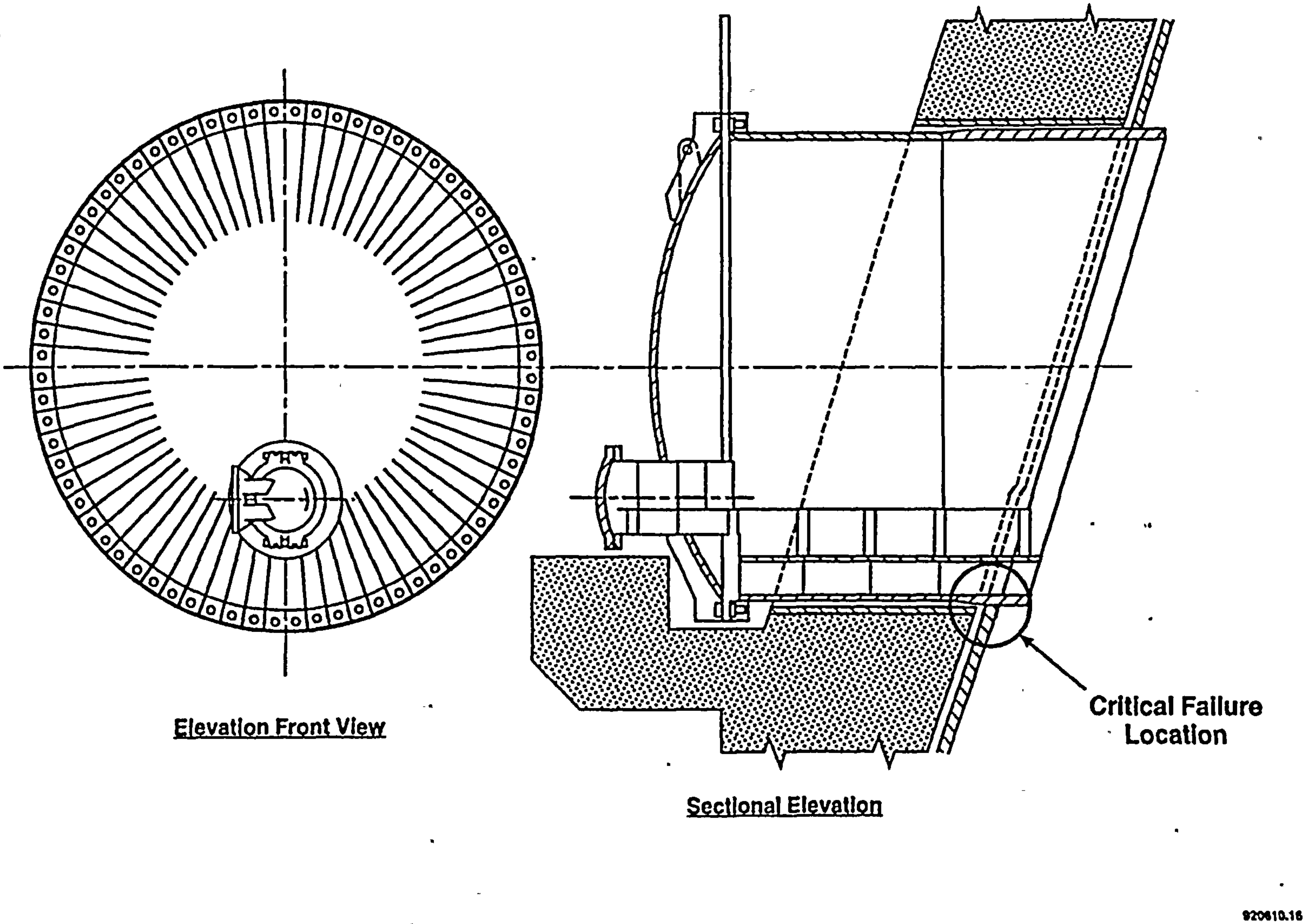

Figure 4.3-4 Critical Containment Failure Location

4.4 Bins and Plant Damage States

Each of the cutsets, representing specific accident scenarios, is placed into one of several plant damage state bins. Each bin has the overall characteristic that its effect on containment performance is unique. The basic categorization used to bin the WNP-2 Level 1 sequences is by initiator. This is because the decision was made to develop initiator specific CETs to assist in the management of Level 1/2 dependencies in the CETs and their associated fault trees.

The general (default) criteria for binning was as follows:

- Short-term station blackout (power restored within four hours, batteries remain available to provide DC power throughout an accident)
- Long-term station blackout (power not restored within four hours so that battery depletion results in loss of DC)
- Transient
- Anticipated transient without successful reactor SCRAM (ATWS)
- Small LOCA (break less than 4", primary system remains at pressure unless automatic or manual depressurization is initiated)
- Large LOCA (break more than 4", primary system will depressurize)

The second level of sequence binning took place within each initiator class. Those sequences which were expected to have a similar impact on containment response were combined into subgroups. The subgroups were then processed independently by the CET. The plant damage state frequency used to enter the CET was represented by the fraction of core damage frequency that each subgroup represented. The grouping of individual cutsets within an initiating event group was based upon their functional similarity, particularly in regard as to whether each represented unique possibilities for recovery or non-recovery of necessary plant equipment and systems during the Level 2 assessment. Typical of the grouping criteria were:

- recoverability of injection before vessel and/or containment failure,
- recoverability of containment heat removal before containment failure,
- time differences between the initiating event and containment failure (if it preceded core melt) or core uncovery,
- whether the core damage event was caused by failed support systems, e.g. AC and or DC power, standby service water or whether the core damage scenario resulted from failed hardware,
- sequence specific durations for loss of offsite power which appear in the Level 1 station blackout sequences,
- whether PCS or the containment system vent irretrievably failed during the core damage scenario so that it could not be used as a potential containment energy removal system in the Level 2 analysis.

Within each bin it was important to discriminate between those sequences in which a Level 1 critical function was lost because of hardware failures, and those in which the function was lost as a result of dependent system failures. Dependent system (e.g. "loss of AC, DC, Cooling") failures are considered to be potentially recoverable during the containment performance assessment, whereas those which result from hardware failures (e.g "injection pump fails to start") are considered non-recoverable. This general rule was modified in those cases in which there was an important common cause hardware failure. In this case some credit was given for the probability that one train could be recovered.

The recoverability for dependent systems was explicitly determined from the plant damage state definition and any limitations on individual success paths which were imposed by the effects of increasing containment pressure. The actual probability of non-recovery of the sequence was assumed to depend on either the hardware unreliability, "failure to start and run" or upon human unreliability, the non-response probability of the operating staff. After reviewing the accident specific situation in each of the important sequences it was determined that as soon as the needed support systems became available:

- the operating staff would be guided by the EOPs and would be focussed on the general functional tasks of initiating injection into the core or initiating containment heat removal with SPC or containment venting with the purge valves,
- there is a long time available for the operator to complete the necessary tasks before it had a significant impact on the sequence, and in general the tasks could be performed from the control room in a relatively short period of time.

This led to fairly low estimates for individual Human Error Probabilities (HEPs). However, there was a great deal of uncertainty about the magnitude of the effects that the imposed stress would have on operator performance during and immediately after core damage, core melt and vessel failure. To be sure that credit for successful human intervention was not overestimated, a standard HEP of 0.1 was assumed for all sequences. This corresponds to the HEP for a condition in which the time available to perform the action is approximately twice the median time taken to perform the action.

When needed support systems become available, the possibility of collateral plant damage or other unidentifiable effects on hardware reliability could make them less reliable than expected. To be sure that the "post core melt" unreliability of hardware systems was not underestimated from Level 1, some general conservative assumptions were made for Level 2:

- for a 2-train system, probability of failure to function on demand "after all support systems become available" was assumed to be 0.05,

- for a 1-train system, probability of failure to function on demand "after all support systems become available" was assumed to be 0.1,

- failure of actuation systems to function on demand "after the recovery of needed support systems in a post core-melt environment" was assumed to be 0.1.

Recovery of PCS to serve as a means for long term containment heat removal requires special attention because the nature of failure which initiated the transient and led to the plant SCRAM has an effect on its future availability.

Following loss of offsite power, recovery of PCS is expected to take about 8 hours, so the chances of recovery are determined by looking at the available window of opportunity in which the primary constraint is opening the MSIVs before containment pressure reaches 54 psig. Following a turbine trip, the PCS can usually be recovered without a great deal of difficulty so credit is primarily determined by the expected non-response probability of the operating staff.

If the plant SCRAM was initiated by a transient involving failure of the MSIVs (unplanned closure), loss of containment air or loss of a DC bus, then the recovery of PCS was not allowed. This is because reopening the MSIVs requires restoration of failed hardware. The last case in which PCS is not considered to be recoverable is with the sequences in which turbine trip is initiated by a loss of condenser vacuum. The transient sequences were grouped using these criteria and each group evaluated separately so that the conditional probabilities associated with recoverability were treated correctly.

Table 4.4-1 summarizes the resulting plant damage state bins and the system status. The total frequency considered in Table 4.4-1 represents about 99.9% of the Level 1 CDF. Section 4.4.1 describes the grouping process in greater detail.

TABLE 4.4-1
Individual Plant Damage States and their Contributing Sequences

Level 1 Systems Status						Level 2 Systems Status						
Sequence	LOOP	HP Inj	LP Inj	Cont. Intact	RCS Press	Press at VF	HP Inj	ADS	LP Inj	RHR	Vent	PCS
TES19 (ST-SBO) F = 2.71E-6	Yes	RCIC (F) HPCS (E)	U-LOOP	Yes	Hi or Lo	HPME or LPME	HPCS-ROSP	A	A-ROSP	A-ROSP	A-ROSP	A if P_{cont} < 54psig
TES15 (LT-SBO) F = 3.51E-6	Yes	HP inj fails	U-LOOP	Yes	Hi	HPME	Failed	A-ROSP	A-ROSP	A-ROSP	A - ROSP * P_{cont} < 49psig	A if P_{cont} < 54psig
TES17A (LT-SBO) F = 4.206E-6	Yes	RCIC (O) HPCS (E)	U-LOOP	Yes	Hi	HPME	HPCS-ROSP	A-ROSP	A-ROSP	A-ROSP	A - ROSP * P_{cont} < 49psig	A if P_{cont} < 54psig
TES17B (LT-SBO) F = 3.02E-7	Yes	RCIC (O) HPCS (F)	U-LOOP	Yes	Hi	HPME	failed	A-ROSP	A-ROSP	A-ROSP	A - ROSP * P_{cont} < 49psig	A if P_{cont} < 54psig
TES03 TES06 F = 9.48E-7	Yes	HPCS (O) RCIC (O)	1-train A Deadhead	Yes	Hi	HPME	Recovera ble	A	A	A-ROSP	A - ROSP * P_{cont} < 49psig	A if P_{cont} < 54psig
TES09 F = 4.52E-8	Yes	HP Inj failed	1 train · injecting	Yes	Hi - Repr.	HPME	Failed	U - P_{cont} > 62psig	A	A-ROSP	U - P_{cont} > 49psig	U - P_{cont} > 54psig

Level 1 Systems Status						Level 2 Systems Status						
Sequence	LOOP	HP Inj	LP Inj	Cont. Intact	RCS Press	Press at VF	HP Inj	ADS	LP Inj	RHR	Vent	PCS
TW: TCS03 TCASS02 TCNS03 TDCS05 TIS08 TFS04 TMS04 TSSWS04 TTS05 TTSWS02 MSS05 F = 1.38E-6	No	HPCS / RCIC (O)	failed	No	Hi	HPME	failed	F-P_{cont} > 62 psig	F (CF)	F (CF)	F	U - P_{cont} > 54psig
FLDTW: FLD14S02 FLD6S02 F = 7.3E-7	No	HPCS / RCIC (O)	failed	No	Hi	HPME	failed	failed/U (P_{cont} > 62 psig)	F (CF)	F (CF)	F	U - P_{cont} > 54psig
S1S03 (MLOCA) F = 9.04E-9	No	HPCS (O)	failed	No	Low	LPME	failed	failed	failed	failed	U - P_{cont} > 49psig	U
TTS19 F = 5.94E-8	No	All failed	LP deadhead	Yes	Hi	HPME	failed	failed	avail.	avail.	A if P_{cont} < 49psig	A if P_{cont} < 54psig
TES11 F = 6.72E-9	Yes	RCIC (F) HPCS (E)	LP failed	Yes	Lo	LPME	HPCS - ROSP	Avail	1 train - ROSP	1-train ROSP	A-ROSP	A - ROSP*P_{cont} < 54psig
TES13 F = 1.99E-8	Yes	RCIC (F) HPCS (E)	LP Deadhead	yes	Hi	HPME	HPCS-ROSP	Failed	1 train avail, 2 train ROSP	1 train avail, 2 train ROSP	A -ROSP * P_{cont} < 49psig	U

Level 1 Systems Status						Level 2 Systems Status						
Sequence	LOOP	HP Inj	LP Inj	Cont. Intact	RCS Press	Press at VF	HP Inj	ADS	LP Inj	RHR	Vent	PCS
TCNS11 TDCS15 TFS14 TMS18 TSSWS09 F = 6.25E-8	No	HPCS (F) RCIC (F)	LP Deadhead	Yes	Hi	HPME	Failed	Available	2 trains available	U	A if P_{cont} < 49psig	U
FLD7S02 TTSWS08 SRS18 F = 1.298E-6	No	HPCS (F) RCIC (F)	LP Deadhead	Yes	Hi	HPME	failed	failed	2 trains available	2 trains available	A if P_{cont} < 49psig	U
FLD7S03 TIS22 TSSWS08 MSS19 TFS13 F = 1.14E-6	No	HPCS (F) RCIC (F)	LP Failed	Yes	Low	LPME	failed	Available	failed	failed	A if P_{cont} < 49psig	U
AOS14 (LLOCA) F = 1.47E-7	No	HPCS (F) RCIC (F)	LP Failed	No - Bypass	Low	LPME	failed	n/a	failed	n/a	n/a	n/a
AS08 (LLOCA) F = 3.00E-8	No	F (CF)	F (CF)	No	Low	LPME	failed	U if P_{cont} >62psig	F (CF)	F (CF)	U - P_{cont} > 49psig	U - P_{cont} > 54psig
TMCS16 (ATWS) TCCS16 (ATWS) TFCS16 (ATWS) TICS10 AS09 S1S15 F = 1.19E-7	No	A	Operating	Yes	Low	LPME	A	ADS inhibit not initiated - depress. valves are open	O	A	A if P_{cont} < 49psig	U

Level 1 Systems Status						Level 2 Systems Status						
Sequence	LOOP	HP Inj	LP Inj	Cont. Intact	RCS Press	Press at VF	HP Inj	ADS	LP Inj	RHR	Vent	PCS
TMCS17 TCCS17 TFCS17 TICS11 TFCS20-100% TMCS20-100% TCNS14 TDCS18 TSSWS12 TTSWS11 FLD6S11 FLD14S11 TCASS11 F = 3.19E-7	No	A	A	Yes	Hi	HPME	A	A if P_{cont} < 62psig	A	A	A if P_{cont} <49psig	U
TTCS17 TTCS20	No	A	A	No	Hi	HPME	F	No	F	F	n/a	n/a
TTCS16	No	A	A	No	Lo	LPME	F	No	F	F	n/a	n/a
AOS15 F = 4.40E-7												

F	failed	LOOP	loss of offsite power
O	operating	E	electrical fault
ROSP	recovery of offsite power	VF	vessel failure
A	available	CF	containment failure
Repr.	repressurization	U	unavailable

4.4.1 Plant Damage State Grouping

Each of the individual Level 1 sequences, which represent a contribution to overall CDF of at least 1E-9, is placed into one of several plant damage state bins. Each bin has the characteristics to represent its members throughout the Level 2 analysis. The grouping is based on the similarity in the effect that each has on severe accident progression and containment performance rather than the similarity between their Level 1 characteristics. Each bin, therefore, has unique characteristics regarding containment condition before/during core degradation, reactor coolant system condition during core degradation, and containment safeguards system performance.

The following section describes the process and information used to group the Level 1 sequences in preparation for further assessment in the Level 2 analysis.. For each accident sequence within each PDS group, the following information is provided:

- the uniquely important characteristics of each of the core damage scenarios which has been identified by the Level 1 analysis
- how each sequence group is expected to affect the initial containment and primary system conditions or the availability of containment systems which are important to the containment response.

I. ·Short-term Station Blackout

ST-SBO/TE-S19 - Frequency = 2.71E-6/year

Level 1 Sequence Description

- Loss of Offsite Power.
- EDGs 1, 2 and 3 fail to start.
- RCIC is not available or fails to start so all injection is lost and the core melts within about 2 hours of the LOOP.

System Status for Level 2 (after core melt)

- HPME (high pressure melt ejection) or LPME (low pressure melt ejection) depends on operator actions.
- When offsite power is restored, HP injection will be available (assume EDG-3 failed). DC power is available throughout the core melt sequence.
- LP injection can be established when primary system is depressurized with manual ADS or vessel failure.
- CHR with suppression pool cooling is viable and the containment pressure is low enough to allow the use of the vent if power is restored within 8 - 10 hours.

- PCS requires 8 hours to become operable, and therefore is available for CHR if offsite power is restored before vessel failure.
- If power is not restored prior to vessel failure, it is likely that PCS will not be available. In the case that depressurization of primary system is unsuccessful, vessel blowdown would raise the containment pressure to beyond 54 psig and the MSIVs cannot be opened. In the case that depressurization of primary system is successful, the containment pressure will be above 54 psig approximately 6.5 hours after the initiation of the accident. The time window is less than that required for restoration of PCS. Therefore, PCS is assumed to be unavailable in this sequence.

II. Long-term SBO

LT-SBO/TE-S15 - Frequency = 3.51E-6/year

Level 1 Sequence Description

- Loss of offsite power.
- EDGs 1 and 2 fail to start or run, EDG-3 operates.
- HPCS is successful for a period of time but fails during its mission when the CST cannot be refilled and becomes depleted. HPCS cannot be realigned to the suppression pool because DC required to operate the necessary valves is not available. In addition, the temperature in the suppression pool will exceed the HPCS pump design limit of 212°F.
- Depletion of the batteries means that depressurization cannot be accomplished until power is recovered.
- Offsite power is not recovered within 10 hours so total loss of injection results in core melt about 15 hours after LOOP.

System Status for Level 2 (after core melt)

- HP injection is unavailable (even if power is recovered, suppression pool temperature will preclude its use).
- When offsite power is recovered, LP injection can be reestablished if system is manually depressurized or vessel failure occurs
- Recovery of offsite power should allow containment heat removal via SPC but containment pressure is probably too high for the vent (49 psig) or PCS (54 psig) to be used as a containment heat sink.

LT-SBO/TE-S17 - Frequency = 4.51E-6/year

TE-S17A	HPCS recoverable (EDG-3 failure)	- Frequency = 4.206E-6
TE-S17B	HPCS not recoverable (HPCS failure)	- Frequency = 3.02E-7

Note that the information used to distinguish between TE-S17A and TE-S17B which result in failure of HPCS because of EDG-3 failure rather than HPCS itself (other than EDG-3) fails was derived from a review of the individual sequence cutsets and the calculation of a split fraction based on the frequencies for each type. The top 30 cutsets were used to calculate the split fraction. The division of the Level 1 sequence into S-17A and B was necessary for Level 2 in the case of EDG-3 failure, HPCS can be recovered when offsite power is restored, whereas if the loss of HPCS is caused by local faults, such as failure of the pump, it is not recoverable.

Level 1 Sequence Description

- Loss of Offsite Power.
- EDGs 1, 2, and HPCS (or EDG-3) fail so RCIC is the only source of injection.
- There are no chargers for the batteries so DC fails within 4-6 hours.
- Loss of DC causes failure of RCIC so all injection is lost and the core melts.

System Status for Level 2 (after core melt)

- When offsite power is restored, HP injection will be recovered in TE-S17A but not in TE-S17B. LP injection can be established when primary system is depressurized after vessel failure.
- CHR with suppression pool cooling should be viable. Vent is available if the containment pressure is lower than 49 psig.
- Because PCS will require 8 hours to become operable, by the time it is available (14 hours after LOOP) the containment pressure will be above 54 psig and the MSIVs will close and eliminate PCS as a viable success path.

III. Loss of Offsite Power Transients (SBO look-alike)

Transient/TE-S03 *Frequency = 9.33E-7/year*
Transient/TE-S06 *Frequency = 1.48E-8 /year*

Level 1 Sequence Description for TE-S03

- Loss of offsite power.
- EDG 1 or 2 is available.
- HPCS is successful.
- The RHR loop which has power available to its bus is unavailable so there is no means for suppression pool cooling.
- Injection will fail if offsite power is not recovered within 12 hours and the potentially available train of RHR is not restored. This is because pump failure occurs due to high water temperature upon switchover from CST to the suppression pool.
- Loss of all HP injection results in core melt about 15 hours after LOOP.

Level 1 Sequence Description for TE-S06

- Loss of offsite power.
- EDG 1 or 2 is available.
- HPCS fails but RCIC succeeds.
- The RHR loop which has power available to its bus is unavailable so there is no means for suppression pool cooling.
- If offsite power is not recovered within 12 hours and the potentially available train of RHR is not restored, containment pressure will increase to a point that RCIC is no longer operable (about 20 psig/260°F) and the resultant loss of injection will initiate core melt..

System Status for Level 2 (after core melt)

- When offsite power is restored, HP injection will be recovered. LP injection can be established when primary system is depressurized after vessel failure.
- CHR with suppression pool cooling should be viable. Vent is available if the containment pressure is lower than 49 psig.
- Because PCS will require 8 hours to become operable, by the time it is available (14 hours after LOOP) the containment pressure will be above 54 psig and the MSIVs will close and eliminate PCS as a viable success path.

Transient/TE-S09   *Frequency = 4.52E-8 /year*

Level 1 Sequence Description

- Loss of offsite power.
- EDG 1 or 2 is available.
- All HP injection (HPCS and RCIC) failed.
- Primary system is depressurized successfully and a source of LP injection is established.
- The RHR loop which has power available is unavailable so there is no means for suppression pool cooling.
- If offsite power is not recovered within about 14 hours and the potentially available train of RHR is not restored, containment pressure will increase to more than 62 psig. At this point, the SRVs will reclose, the primary system will repressurize and the LP injection system will become incapable of injecting. This will result in loss of injection and core melt.

Systems Status for Level 2 (after core melt)

- HPME.
- HP injection is failed or lost inventory.
- Depressurization will occur upon vessel failure but containment pressure will likely be too high to initiate ADS before vessel failure.
- At least one train of LP injection and RHR should be available after recovery of offsite power.
- The PCS and vent will not be available because containment pressure will prevent the MSIVs and purge valves from being opened.

IV. Transients in which containment fails prior to core damage

Transient/TCS03	*Frequency*	=	*3.06E-7/year*
Transient/TCASS02	*Frequency*	=	*1.13E-7/year*
Transient/TCNS03	*Frequency*	=	*3.02E-9/year*
Transient/TDCS05	*Frequency*	=	*1.50E-7/year*
Transient/TIS08	*Frequency*	=	*3.82E-9/year*
Transient/TFS04	*Frequency*	=	*2.31E-9/year*
Transient/TMS04	*Frequency*	=	*1.45E-8/year*
Transient/TSSWS04	*Frequency*	=	*2.00E-7/year*
Transient/TTS05	*Frequency*	=	*4.19E-7/year*
Transient/TTSWS02	*Frequency*	=	*1.22E-7/year*
Manual-SD/MSS05	*Frequency*	=	*4.51E-8/year*

Level 1 Sequence Description (Transients)

- Plant transient initiated by:
  - Loss of condenser vacuum
  - Loss of containment air
  - Loss of DC
  - Stuck open SRV
  - Turbine trip
  - Loss of service water
  - Loss of containment nitrogen
  - Loss of feedwater
  - MSIV closure
  - Loss of plant service water
- Successful Reactor SCRAM.
- SRVs operate properly to maintain primary system pressure. HPCS or RCIC inject successfully.
- RHR is unavailable and the vent is not implemented.
- PCS is unavailable as a heat sink because the MSIVs are closed (high vacuum/ no motive air/no DC/not recovered following LOFW / failed due to flooding).
- Containment fails from overpressure and leads to consequential failure of injection.
- Core melt follows.

Flooding/FLD14S02 *Frequency = 4.55E-7/year*
Flooding/FLD6S02 *Frequency = 2.75E-7/year*

Level 1 Sequence Description (Flooding cases 6 and 14):

- Piping failure in BOP causes flooding which results in the non-recoverable loss of PCS (condensate, feedwater and circulating water).
- SCRAM is successful and the SRVs control primary system pressure.
- HPCS is successful and maintains primary system inventory with decay heat being rejected to the suppression pool through the SRVs.
- Containment heat removal systems are either unavailable or failed.
- Eventually containment fails from overpressure and initiates loss of injection and core melt.

MLOCA/S1S03 - *Frequency = 9.04E-9/year*

Level 1 Sequence Description (Medium LOCA)

- Medium LOCA initiates reactor SCRAM.
- SCRAM is successful.
- HPCS is available so HP injection maintains primary system inventory.
- RHR and vent failed. PCS is unavailable because of containment isolation closing the MSIVs.
- Containment pressure increases to the point of containment failure.
- The resulting release of energy to the reactor building fails ECCS pumps and causes loss of injection.

System Status for Level 2 (after core melt for non-transients and transients included above)

In the above transient and non-transient (LOCA and flooding) events in this category, containment failure occurs before core melt. This type of accident is termed "TW" sequence. Containment failure occurs because either all means of containment heat removal are lost or the suppression pool cannot absorb the release of energy from the primary system at a high enough rate. After containment failure there is a consequential loss of all injection which in turn initiates core melt.

- All sequences result in an HPME.
- LP injection and RHR will remain unavailable and unrecoverable because it is assumed that the release of energy to the lower reactor building from containment will fail the pump motors.
- HP injection will likely be lost following containment failure because of inadequate NPSH or high suppression pool temperature unless it can continue to inject water from a source external to the containment (CST).

- Total loss of injection means that there will be no ex-vessel cooling of the debris.
- The debris will be uncooled and the drywell/wetwell boundary will fail with the drain lines in the pedestal cavity to be the most likely point. Because the drain lines pass through the suppression chamber when they fail they will initiate suppression pool bypass resulting in fission products being unscrubbed.
- Containment pressure will be controlled by the amount of energy released from the breach in containment.

There are some important assumptions which are made for those sequences in which containment failure precedes core melt. The analysis assumes that for sequences which do not involve a loss of offsite power, an inexhaustible supply of water will be available via make-up to the CST. Because this make-up water will be cool, no thermodynamic considerations are expected to negatively impact the reliability of the HPCS pump. The HPCS pump will continue to operate reliably because it will never enter the mode of operation in which it is realigned to take suction from the suppression pool and be exposed to potentially damaging fluid temperatures. In the situation where the core melt sequence is initiated by a loss of offsite power, it is assumed that the systems required to transfer water to the CST will not be available. As a result, injection will fail when the CST is empty (about 10 hours) and the total loss of injection will lead to core melt.

V. Other Transient Groups

Transient/TTS19 Frequency = 5.94E-8/year

Level 1 Description

- Turbine trip.
- Successful SCRAM.
- SRVs operate accordingly to control primary system pressure.
- HPCS and RCIC unavailable.
- ADS is unsuccessful so there is no LP injection.
- Core melts with primary system at high pressure.

System Status for Level 2 (after core melt)

- HPME.
- After vessel failure, primary system depressurizes and the LP injection systems should be available (unchallenged or deadheaded during Level 1 sequence).
- Because both RHR and vent are not necessarily failed during the core damage scenario, they are potentially available for CHR after vessel failure. This means that containment heat removal with SPC should definitely be possible.

However, the operability of the support systems, which are needed to actuate the vent or initiate PCS as a heat sink, depends upon containment pressure.

The vent will normally be initiated at 39 psig and will be operable if containment pressure is less than 49 psig. However, a MAAP simulation shows that the increase in containment pressure following an HPME will tend to move the pressure from below to above the operability range. This means that the vent is likely to be non-functional in most HPME sequences. Since the MSIVs close when containment pressure reaches 54 psig, there is also a high likelihood that PCS will not be available for post-vessel failure containment heat removal in sequences involving HPME.

Transient/TE-S11 Frequency = 6.72E-9/year

Level 1 Sequence Description

- Loss of offsite power.
- EDGs 1 or 2 fail to start or run.
- HPCS fails because EDG-3 fails to start, and RCIC fails to start.
- Primary system is depressurized successfully but a source of LP injection cannot be established.
- If offsite power cannot be recovered within an hour the core melts - LPME at vessel failure.

System Status for Level 2 (after core melt)

- LPME.
- Total loss of injection means that there will be no ex-vessel cooling of the debris until recovery of power. When power becomes available, HPCS or one train of LP injection should be available and operable since primary system is depressurized.
- One train of RHR should be available after recovery of offsite power.
- Vent should be viable after recovery of offsite power because vessel fails under low pressure and the resultant containment pressure is less than 49 psig.

Transient/TE-S13 *Frequency = 1.99E-8/year*

Level 1 Sequence Description

- Loss of offsite power.
- EDGs 1 or 2 fail to start or run.
- HPCS unavailable because EDG-3 fails to start, and RCIC fails to start.
- Primary system is not depressurized successfully so the LP injection systems will be deadheaded.
- If offsite power cannot be recovered within an hour HP injection cannot be established and the core melts - HPME at vessel failure.

System Status for Level 2 (after core melt)

- After power is recovered, HPCS injection should be available. LP injection will not be available to arrest the sequence in-vessel unless primary system is depressurized.
- Total loss of injection means that there will be no ex-vessel cooling of the debris until recovery of power. Injection should be available after vessel failure because it was deadheaded but not failed.
- At least one train of RHR should be available after recovery of offsite power.
- Vent should be viable after recovery of offsite power because the short duration of the sequence should mean that containment pressure is less than 49 psig.
- PCS not available in time (before containment pressure gets to 54 psig).

Transient/TCNS11	*Frequency = 2.64E-8/year*
Transient/TDCS15	*Frequency = 2.06E-8/year*
Transient/TFS14	*Frequency = 1.80E-9/year*
Transient/TMS18	*Frequency = 3.60E-9/year*
Transient/TSSWS09	*Frequency = 1.01E-8/year*

Level 1 Sequence Description (Transient)

- Transient induced by loss of containment air, containment nitrogen, DC Div. 2 or feedwater, MSIV closure, loss of service water.
- HPCS fails (nonrecoverable).
- RCIC fails to start.
- Primary system is not depressurized successfully, so a source of LP injection cannot be established (in the case of TCN or TDC, the depressurization valves cannot be actuated).
- Core melts.

System Status for Level 2 (after core melt)

- Arrest of the sequence in vessel is not possible because all HP injection is lost.
- HPME at vessel failure.
- LP injection should be available after vessel failure because it was deadheaded not failed.
- RHR is not available and cannot be recovered.
- Vent should be viable if containment pressure is less than 49 psig.
- PCS is not available.

Flooding/FLD7S02	*Frequency*	=	*9.99E-7/year*
Transient/TTSWS08	*Frequency*	=	*2.86E-8/year*
Level Instr/SRS18	*Frequency*	=	*2.70E-7/year*

Level 1 Description

- Transient initiated by loss of plant service water, level instrument line break, and piping failure in reactor building causes flooding (FLD7 causes the nonrecoverable failure of all ECCS pumps except LPCS).
- Successful SCRAM.
- SRVs operate correctly to control primary system pressure.
- HPCS and RCIC unavailable.
- ADS is unsuccessful so there is no LP injection.
- Core melts with primary system at high pressure.

System Status for Level 2 (after core melt)

- HPME.
- After vessel failure, primary system depressurizes and the LP injection systems should be available (unchallenged or deadheaded during Level 1 sequence).
- Containment heat removal with SPC available and vent should be possible if containment pressure is less than 49 psig (both CHR systems are not necessarily failed in Level 1).
- PCS not recoverable unless plant service water is restored which is not credited in this analysis. PCS should be available for level instrument line break sequence.

Flooding/FLD7S03	*Frequency*	= *6.83E-7/year*
Transient/TIS22	*Frequency*	= *3.38E-9/year*
Transient/TFS13	*Frequency*	= *3.28E-8/year*
Transient/TSSWS08	*Frequency*	= *4.09E-7/year*
Manual SD/MSS19	*Frequency*	= *9.00E-9/year*

Level 1 Sequence Description

Either,

- Transient initiated by loss of Div 2 DC, loss of service water, flooding in the reactor building,
- SRVs operate properly to control primary system pressure,

or,

- transient initiates a stuck open SRV,

and,

- HP injection fails.
- ADS is implemented or the primary system depressurizes through the stuck-open SRV.
- There are no available sources of LP injection.
- Core melts.

System Status for Level 2 (after core melt)

- Total loss of injection means that there will be no in- or ex-vessel cooling of the debris.
- RHR is not available for CHR.
- Control power for the vent is dependent upon 120 VAC so vent should be viable as a means of CHR and the relatively short duration of the sequence should mean that containment pressure is less than 49 psig.
- PCS is not available for CHR because the Div 2 DC is needed to open the MSIVs (TDC) and a flood caused by failure of TSW will make PCS unavailable.

Large LOCA Containment Bypass/AOS14 - Frequency = 1.47E7/year

Level 1 Description

- Large LOCA outside containment (steam line break).
- Reactor SCRAM is successful.
- Main steam line fails to isolate.
- Steam released to reactor building initiates complete loss of injection.
- Core melts.

System Status for Level 2

- Containment is bypassed before injection failure.
- Core melts with injection unrecoverable so the debris will be uncooled.

Large LOCA/AS08 - Frequency = 3.00E-8/year

Level 1 Description

- Large LOCA.
- Successful SCRAM.
- Omega Seal failure causes suppression pool bypass. Containment overpressurizes because of insufficient energy transfer to the suppression pool.
- Containment failure initiates loss of all ECCS injection.
- Core melts.

Note that this scenario is different from the other TW sequences because though the mechanism by which the loss of injection is initiated by containment failure is similar, the timing of this event is quite different. In this scenario, containment fails a relatively short time after the initiator has occurred so there will be very little time to mobilize the emergency response program. In the conventional TW sequences, there may be 15 to 24 hours between the time of the initiator and containment failure.

The expected mode of containment failure in this scenario is also different. Because the pressure increase will be dramatic and severe, the likelihood that membrane tearing will stabilize pressure is much less certain. As a result, it was assumed that this sequence led to catastrophic failure of containment.

VI. ATWS sequences

Because of the uncertainty in the expected core behavior during ATWS events, some assumptions were implicit to the selection of sequence grouping. This is discussed below before the actual grouping is presented. The issue is determination of whether containment fails and initiates a consequential failure of all injection or whether the core melts, dislocates and becomes subcritical before containment failure. This latter case is relatively more benign because all normal core injection and containment heat removal systems are available so there is a high likelihood of arresting the sequence in-vessel.

The ATWS scenarios which are important to the WNP-2 study have the following characteristics:

ATWS Scenario Type 1

- ATWS is initiated by mechanical failure of the rods following a SCRAM demand.
- Recirculation pump trip and the actuation of the standby liquid control system is successful, but ADS is not inhibited.

This condition initiates a series of power oscillations as the core blows down vigorously to the suppression pool, some boron is flushed out of the primary system, and the core goes through a series of heat-up and cooldown events as cold water is added and then steamed off. In this scenario, it was assumed that power oscillations induces core melt before the energy added to the suppression pool is sufficient to increase containment pressure to a potentially damaging level.

If feedwater remains available, the scenario takes on characteristics which are similar to those exhibited by the type 2 scenario, described below.

ATWS Scenario Type 2

- ATWS is initiated by mechanical failure of the rods following a SCRAM demand.
- Recirculation pump trip is successful but actuation of the standby liquid control system does not occur.
- feedwater remains available.

In this scenario, the core will continue to operate in a quasi-steady state condition initiated at a nominal power level of 40%. To maintain adequate core cooling at this power level it is necessary to maintain 40% reactor feed flow. Since none of the ECCS systems have a capacity approaches this level, core cooling can only be stabilized at this level if main feedwater is available. The SRVs have adequate capacity to handle this power and reject the necessary energy to the suppression pool, but in all likelihood the turbine bypass system will also be used to reject 25% power to the main condenser. Therefore, 15% power will be rejected to the suppression pool. If uncorrected, suppression pool temperature will increase to the point that the containment is threatened by overpressure. Overpressure failure of containment then becomes an initiator for total loss of injection.

ATWS Scenario Type 3

- ATWS is initiated by mechanical failure of the rods following a SCRAM demand.
- Recirculation pump trip is successful but actuation of the standby liquid control system does not occur.
- Feedwater is not available.

This scenario starts off very much like scenario type 2 except that the main feedwater system is unavailable. The only sources of core injection are from the HP injection systems which have a combined total of about 5% of normal core feed so core water level will drop. As it does, power will decrease until presumably there will be a point at which the situation stabilizes. The unanswered question is whether the vessel level at which stabilization occurs will provide adequate cooling. For the purposes of this study, it was assumed that the core would melt before the rejected energy threatened containment integrity.

ATWS Scenario Type 4

- ATWS is initiated by mechanical failure of the rods following a SCRAM demand.
- Recirculation pump trip is unsuccessful.

In this scenario power continues to be generated at is nominal 100% rate, affected only by the change in the void fraction and other moderator or fuel temperature effects which are present.

After trip, two possibilities exists:

- If feedwater is available, the operators can control the vessel level, reduce power and maintain the core in a condition in which the fuel is cooled adequately and the energy is rejected to suppression pool. This will lead to a rapid increase in suppression pool temperature, beyond the capability of the SPC system, and eventually containment failure. Containment failure will then cause loss of injection and core melt.

- If feedwater is not available, the scenario becomes similar to ATWS scenario type 3. Because the core energy level is initially much higher, core melt will occur earlier. However, the broad time increments used in source term categorization, Section 4.6, will be unable to discriminate between this scenario and scenario type 3.

Grouping of the ATWS scenarios is described in the following.

ATWS/TMCS16	*Frequency = 4.00E-8/year*
ATWS/TCCS16	*Frequency = 1.00E-8/year*
ATWS/TFCS16	*Frequency = 2.00E-8/year*
ATWS/TICS10	*Frequency = 2.74E-9/year*

In addition, the following sequences are incorporated into this group since the Level 1 results show that these will lead to core damage.

AS09	*Frequency = 4.2E-9/year*
S1S15	*Frequency = 4.2E-8/year*

Level 1 Sequence Description

- Turbine trip causes loss of feedwater because the initiating event is one of the following:
  - loss of condenser vacuum
  - loss of feedwater
  - stuck open SRV
  - MSIV closure
  - turbine fault.
- Reactor fails to SCRAM (mechanical).
- Recirculation pump trip successful - power decreased to 40%.
- SRVs operate as designed to control primary system pressure.
- SLC is successful and brings down reactor power.
- ADS inhibit is not exercised so core starts to go through power oscillations as the LP injection systems flood the core, flush out boron and influence moderator temperature.
- Core melts.

Systems Status for Level 2

- Arresting this sequence in vessel should be possible since LP injection is already working and once the core melting/dislocation starts it should lose its critical geometry and become subcritical.
- LP system is injecting.
- RHR and vent available for CHR depend on timing and containment pressure.
- PCS not available (condenser/LOFW/MSIV closure/timing in SORV).

Note that if the power oscillations continue for a period of time before the core loses critical configuration, it could be possible to overpressurize containment and initiate failure of injection. Though considered during the current analysis, it was assumed that melting would occur before containment failure.

ATWS/TMCS17 (40% power)	*Frequency = 5.91E-8/year*
ATWS/TCCS17 (40% power)	*Frequency = 1.43E-8/year*
ATWS/TFCS17 (40% power)	*Frequency = 2.96E-8/year*
ATWS/TICS11 (40% power)	*Frequency = 3.04E-9/year*
ATWS/TFCS20 (100% power)	*Frequency = 2.94E-9/year*
ATWS/TMCS20 (100% power)	*Frequency = 6.14E-9/year*

In addition, the following sequences are incorporated into this group since the Level 1 results show that these will lead to core damage.

TCNS14	*Frequency = 1.75E-8/year*
TDCS18	*Frequency = 4.2E-8/year*
TSSWS12	*Frequency = 2.56E-9/year*
TTSWS11	*Frequency = 1.75E-8/year*
FLD6S11	*Frequency = 4.09E-8/year*
FLD14S11	*Frequency = 6.57E-8/year*
TCASS11	*frequency = 1.75E-8/year*

Level 1 Sequence Description (40% power - RPT successful)

- Turbine trip which is followed by a loss of feedwater because the trip was initiated by one of the following:
  - loss of condenser vacuum
  - loss of feedwater
  - Stuck open SRV
  - MSIV closure.
- Reactor fails to SCRAM (mechanical).
- Recirculation pump trip successful - power decreased to 40%.
- SRVs operate as designed to control primary system pressure.
- SLC is unsuccessful, power remains high.
- Unavailability of feedwater means that HPCS/RCIC remains the only means for adding water to the core. Power/flow mismatch results in very low vessel level and core melting.

Level 1 Sequence Description (100% power - RPT unsuccessful)

- Turbine trip initiated by a loss of feedwater because the initiating event is:
  - loss of feedwater
  - MSIV closure.
- Reactor fails to SCRAM (Mechanical).
- Recirculation pump trip not implemented.
- Power remains high so the energy released to the suppression pool through the SRVs will initiate containment failure.
- HPCS and RCIC are the only means for adding water to the core. Mismatch between power and flow results in very low vessel level and core melting.
- After the onset of fuel dislocation, the core will become subcritical.

System Status for Level 2

- Arrest in vessel should be possible since HP injection is available.
- RHR and vent are available for CHR since containment pressure is less than 49 psig.
- PCS is not available (condenser/LOFW/MSIV closure/timing in SORV)

ATWS/TTCS17 (100%) *Frequency = 2.30E-7/year*
ATWS/TTCS20 (100%) *Frequency = 2.88E-8/year*
ATWS/TTCS16 *Frequency = 1.78E-7/year*

In addition, the following sequence is incorporated into this group since the Level 1 results show that it will lead to core damage.

AOS15 *Frequency = 3.04E-9/year*

Level 1 Sequence Description

- Turbine trip occurs.
- SCRAM fails to occur (mechanical failure of CRDs).
- Feedwater starts to run back but remains available.
- The vessel is cooled with feedwater and maintained at a nominal power level of 40% or higher. The energy generated in excess of the capability of the turbine bypass system is transferred to the suppression pool.
- Suppression pool heats up quickly and eventually reaches a temperature at which the corresponding pressure threatens containment structural integrity.
- Containment fails and all injection is lost - the core melts.

System Status for Level 2 (40% and 100% power)

System status is the same as that for TW sequences - loss of containment integrity results in loss of all injection systems. After vessel failure the debris will be uncooled and the fission products unscrubbed. The timing will, however, be different because the increased energy addition rate to the suppression pool will increase its temperature much more quickly.

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4.5 Accident Progression, Containment Event Trees and Fault Trees

The Plant Damage States obtained from the IPE Level 1 analysis are further analyzed here to determine the containment failure mode and source term release probabilities.

4.5.1 Containment Event Tree Development

Containment event trees have been developed for each of the identified initiating event classes:

- Short-term station blackout
- Long-term station blackout
- Transient
- Anticipated transient without successful reactor SCRAM (ATWS)
- Small LOCA
- Large LOCA

Failure of containment isolation is not credited as a successful method for containment heat removal. In some cases this may be a conservative assumption because if a large penetration fails to isolate it may indeed provide an adequate means for containment energy removal and may prevent containment failure from overpressure. However, by making the assumption that failure of a penetration to isolate has no effect on containment sequences, simplification of the analysis is possible without seriously jeopardizing the validity of the results.

Each event tree (with the exception of large LOCA, which explicitly includes isolation) was quantified twice - once with the initiating frequency fraction for "isolation successful," once with the initiating split fraction for "failure to isolate." The results were then combined to find the cumulative frequencies for each specific containment damage state and associated source term.

During construction of the CETs, two general rules were used as guidance because they assist in the processing and understanding of the information they contain:

- events which exhibit the greatest amount of dependence with other events are shown as close to the beginning of the tree as possible
- events are shown in the order in which they are generally expected to occur

Simple fault trees are used to resolve each node for each accident sequence. The solution of the fault tree provides the relative likelihood with which each CET node branch represents the expected accident progression path. The fault tree solution returns values of:

- 1.0 or 0.0 (true or false) if the outcome can be definitively derived from known information which is explicitly defined in the (Level 1) plant damage state description, or by the outcome which has been defined for events which have occurred previously during the containment accident sequence.

- a probability, or split fraction, if the node cannot be resolved definitively because there is uncertainty in either the behavior of plant equipment which must respond during the accident, or because there is uncertainty associated with the phenomena which influence the outcome from the event represented by the CET node.

The events which are included in the containment event tree are described in the detailed description provided for each individual CET. However, there are some basic issues addressed during the development of the CETs. The following discussion provides some insight as to why certain events are, or are not, included in individual CETs. The findings from PRAs performed elsewhere gave the impetus to define a disposition for each item or whether or not they were relevant to the WNP-2 assessment.

- Can containment fail prior to vessel failure?

  Disposition:

  - The Level 1 results show that following Transients, ATWS and LOCA initiating events, loss of long-term heat removal resulted in containment failure, and resulted in the consequential failure of injection which initiated core melt. Some PRAs have provided additional information that containment failure does not necessarily mean the injection systems fail. However, these arguments were not developed for WNP-2 and the possibility of containment failure prior to vessel failure was provided for during CET development.

- Will containment fail at, or near, the time of vessel failure from hydrogen or steam explosions.

  Disposition:

  - The probability of in-vessel steam explosion is assessed to be 1.E-4 for the primary system at low pressure and 1.E-5 for primary system at high pressure (see the discussion in Section 4.2.2.3). As a result, in-vessel steam explosions are shown to be nearly impossible within the frequency cut-off used for the Level 2 analysis.

- There is a possibility that during some accident sequences in which injection is successful for a long enough period of time that containment pressure is increased to the point at which the sudden energy addition associated with vessel breach is sufficient to drive it quickly to the failure point. The fault tree (ECF) has been constructed to accommodate this possibility.

- The concern with hydrogen combustion was determined to be moot because the containment is inerted with nitrogen. Because the containment never becomes subatmospheric during sequences in which hydrogen may be important, even if there is a failed penetration, the amount of oxygen present will not be adequate to take part in the combustion process. This is consistent with the approach taken in NUREG-1335, which indicates that overpressurization due to combustion processes is not a potential containment failure mode for Mark II plants.

- Hydrogen deflagration in the reactor building may be of concern if venting is successful or if there is a failed penetration which could allow hydrogen to leave containment. This issue is a concern for a Level 3 analysis but beyond the scope of the current Level 2 analysis.

• Which sequences or events can be considered recoverable during the Level 2 analysis and what systems other than the mainline containment protection systems can be credited:

  - when high pressure injection systems are unchallenged during the core damage sequence and should be available when power is restored - if they are recovered prior to vessel failure, the sequence can be terminated if the in-vessel debris is in a coolable configuration.

  - when low pressure injection systems remain unchallenged during the core damage sequence and should be available when power is restored and the primary system has been depressurized (by either depressurization or vessel failure).

  - suppression pool cooling (Decay/Containment Heat Removal) should be available upon restoration of offsite power.

  - the fire protection system can provide additional cooling capability if containment pressure is not too high (less than 70 psig).

  - when the LP injection systems are available, the drywell sprays can provide containment heat removal and debris cooling functions and should be available following restoration of offsite power.

- service water is available for addition to the containment when offsite power is available and containment pressure is below 60 psig. Initiating this source of water addition to the containment may delay failure of an intact containment but unless long-term containment heat removal system is actuated the containment will eventually fail. The time for this sequence may, however be beyond the time frame of the current analysis which is assumed to be 40 hours.

- successful containment venting with the 24" wetwell purge to the SGT (vent) can prevent late containment failure if power has been restored (needed to vent).

- consideration is given to the possibility that impulse loading (ex-vessel steam explosion) at the time of vessel breach could cause pedestal failure, vessel dislocation and subsequent drywell failure, if there is water in the pedestal cavity and the corium is released in a form which is favorable for extensive fragmentation and extremely rapid heat transfer. There will be water in the cavity if the drywell sprays are actuated prior to vessel failure because the drywell drains to the cavity sumps. However, for sequences leading to vessel failure at low pressure, high or low pressure injection is not available. Drywell sprays cannot be actuated and no water will be present in the pedestal. In this case, ex-vessel steam explosions are not considered.

- Successful post vessel failure injection will control drywell temperatures to the extent needed to prevent failure of the drywell head seal (700°F) and prevent molten core-concrete interaction and pedestal floor failure if the debris ejected from the vessel is in a coolable geometry.

- If containment has failed prior to vessel failure (isolation failure), the energetic release of fission products to the containment at the time of vessel breach can result in a short-term increase in fission product release rate (puff release).

- If containment cannot be depressurized (vented or cooled), gross containment failure could initiate failure modes which lead to suppression pool bypass so that fission product scrubbing capability is lost. The most likely failure mode is however, a self limiting membrane tear at one of three locations, two in the drywell, one in the wetwell.

- The question as to whether the reactor building should be credited with fission product retention capabilities in the case of containment failure to the reactor building was excluded. It is left as an issue of concern for a Level 3 analysis. Any breach of containment was conservatively assumed to result in an environmental release.

4.5.1.1 Containment Event Tree for Short-term Station Blackout (ST-SBO)

The definition of a short-term SBO is one in which loss of offsite power initiates reactor SCRAM, onsite power fails to operate on demand, injection fails, and core damage becomes unavoidable. DC power from the batteries is available throughout the sequence so that upon recovery of offsite power, restoration of plant systems can be achieved without unusual difficulty. Depressurization of the primary system by the operating staff is also possible since DC remains available. The event tree is shown in Figure 4.5.1.1-1. The top events in the short-term station blackout event tree are discussed in the following.

Depressurization (DPR)

The first event is determined by whether or not the operating staff successfully implements manual depressurization. If it is successful, the sequence proceeds towards vessel failure at low pressure; if the staff is unsuccessful it becomes a potential high pressure sequence. For the SBO sequence, because total loss of onsite power results in loss of the 120vac instrument power, even though procedurally directed to initiate depressurization, because the operator is blind to conditions within the primary system a non-response probability of 0.1 was assumed. This compares somewhat unfavorably with the 2E-3 non-response probability expected under conditions in which onsite power is available and all process monitoring instruments remain functional.

Power Recovered Prior To Vessel Failure (PWRVF)

If an offsite power source can be recovered prior to vessel failure (assumed to occur when the core support plate fails), injection can be recovered and the core cooled sufficiently to prevent vessel failure. Failure of the core support plate is assumed to be appropriate as a surrogate event for vessel failure. This assumption is consistent with the overall WNP-2 philosophy, namely to minimize complexity in areas of the analysis in which there is relatively little effect on the overall results and insights from the analysis. In this particular case, the acceptability of the assumption was predicated on consideration of the following facts:

- the time between core plate failure and vessel failure predicted by MAAP 3.0B is relatively short.
- the current estimates for the time to vessel failure has been "discretized" into intervals for sequence groups. Five intervals are considered: 0 - 2, 2 - 4, 4 - 10, 10 - 24, > 24 hrs.
- there is a great deal of uncertainty in knowing whether the core debris is in an uncoolable configuration.

Therefore, the precision in the timing of events in the existing analysis does not support further resolution since the contribution to an increased probability of successful restoration of injection during the additional time which may be available is unlikely to have much effect. Even if injection is restored there is still a significant probability that the core cannot be cooled so injection has no effect on vessel protection. The node of the event tree is resolved simply with a split fraction which segregates the sequences by the relatively likelihood that power is or is not restored in time to prevent vessel failure. The time available for recovery is sequence specific but for short-term station blackout, core damage is expected to occur about two hours after the occurrence of the initial loss of offsite power and vessel failure approximately 1.5 hours later.

The split fraction for this node variable is taken from Table 4.5-1 which represents the time dependent power recoverability probabilities for WNP-2.

TABLE 4.5-1
Conditional Probability for Failure to Recover Offsite Power [NSAC-194].

TIME PHASE	DURATION OF TIME PHASE (HOURS)	CONDITIONAL PROBABILITY*
I	0 - 2	0.69
II	2 - 4	0.45
III	4 - 10	0.60
IV	10 - 24	0.53
V	> 24	0.75

* Data base same as Level 1, but intervals selected are different.

High Pressure Injection (HPI)

If injection can be recovered before vessel failure, cooling the debris will likely arrest the sequence. If depressurization is unsuccessful and the primary system is at high pressure then HP injection is the only possible source of core cooling and the split fraction for this node represents the probability that HPCS cannot be initiated following restoration of offsite power. The actual probability is derived from the fault tree for "Failure of HP Injection." (Note: Only random equipment failures which contribute to failure of high pressure injection systems are of concern because dependent faults have been addressed in the CET. Quantification of the probability of random equipment failures is modified from those assumed in the Level 1 analysis to reflect the increase in failure rate which could be expected after a core melt event in which operating conditions may be more severe than those seen during normal operation.)

Low Pressure Injection (LPI)

If the vessel is at low pressure and power is restored, then the introduction of LP injection can arrest the core melt sequence in-vessel. This event is quantified by examination of the fault tree for "Failure of LP injection" and the assignment of appropriate conditional probabilities for each random equipment failure which contributes to the top event. Dependent failures are addressed in the CET as they were for HP injection.

Power Recovered After Vessel Failure But Before Containment Failure (PWRCF)

If power can be recovered between the time of vessel failure and containment failure then:

- injection to the vessel, spilling through the damaged lower head, may possibly cool the debris and prevent the spread of damage via:
  - core-concrete interaction on the drywell floor or pedestal walls.
  - interaction between the melt and drywell downcomers, SRV standpipe collars, the drywell omega seal and the pedestal drain lines.
  - overtemperature within the drywell which results in premature drywell head failure.

- residual heat removal systems (suppression pool cooling, vent via purge valves and SGT, PCS) can be implemented in time to prevent containment failure.

The node is resolved with the application of the recovery/non-recovery split fraction derived from the conditional non-recovery probabilities defined in Table 4.5-1. In the case of short-term SBO, vessel failure is expected approximately 4 hours after the initiating event and containment failure is about 15 hours after the initiating event. The non-recovery split fraction is calculated to be 0.318 (=0.6 x 0.53).

Injection Recovered before Containment Failure (ICF)

The reasons for recovering offsite power before containment failure are identified above. This node merely asks for the split fraction for the success/failure probabilities for injection. Since the vessel has failed, the primary system is fully depressurized so any system which can transport adequate amounts of water to the containment could potentially provide success. Failure however, means that LP injection must have failed, a factor which is important to the later assessment of whether or not RHR can be recovered.

The appropriate branch point probabilities are derived from the fault tree for "Failure to Inject" and only reflect contributions from random equipment failures. Dependencies are already embodied in the event tree sequence logic.

Debris is Cooled (DC)

Even though power is restored and injection is recovered, it is still possible that the debris is in a configuration which is uncoolable. This is either because:

- the debris is released to the pedestal at low pressure where it forms a layer; about 4′ thick (see Section 4.2.6) which can only be cooled at the surface. In this case there may be insufficient area to provide adequate energy removal through the crust. Under the crust the material will remain molten and allow core-concrete interaction to continue.

- in an HPME the melt is released very energetically and splatters all over the walls of the pedestal, in which case the water running from the vessel will be unable to cool it and possibly result in overtemperature failure of the drywell head seal (> 700 °F).

- in an HPME it is expected that some of the melt will be ejected from the pedestal through the 3′ x 7′ pedestal opening, either as a result of its kinetic energy or because it is entrained in the very high velocity steam which is expected to accompany the release from the vessel. Because the pedestal is about 10′ deep, a significant portion of the melt debris is expected to remain in the cavity. But, the fraction which is released to the drywell will require operable drywell sprays if it is to be cooled successfully and prevent further interaction with the downcomer metal, the omega seal, or the containment shell.

The node split fraction is derived from the fault tree which has been developed for "fail to cool debris ex-vessel." The known conditions for system availability and their associated dependencies upon other systems are included in the CET. Therefore, the only concern to the fault tree quantification are the estimates of the relative likelihoods that each configuration the debris may take and random equipment failures or the non-response probabilities associated with failure to initiate the operation of required systems, for the conditions within each sequence. The combined probabilities from the fault tree solution are returned to the CET as a split fraction which gives the expected probabilities of success and failure for coolability of the debris.

Shell Failure (SF)

When an HPME initiates an energetic release of melt debris, some of the melt is expected to be ejected into the drywell. If this melt has a sufficient horizontal component to its velocity as it leaves the pedestal opening it may be deposited directly onto the containment shell where melt/metal interaction could initiate direct failure of the shell. This node in the CET is used to provide the possibility for assessing the importance of this containment failure mode and applies to all scenarios in which there is an HPME and the debris is not cooled.

The split fraction used to resolve this node, which was derived based on the information in NUREG-1150 for Peach Bottom, was discussed in Section 4.2.6.

RHR Recovered (RHR)

Recovery of RHR (suppression pool cooling) is sufficient for the containment to remain intact throughout each accident sequence. RHR success requires the operation of RHR pumps and their associated heat exchangers. Special considerations are:

- There is the possibility that if the RHR pumps are unavailable (failed) then the standby service water supply to the "B" heat exchanger can be realigned to the "B" flow path and used to inject standby service water into the core. However, at the point where the SW flow is introduced, its system developed head is only about 70 psig, so this technique cannot be used if containment pressure is above 50-60 psig. A similar situation exists with the fire protection water system.

- Since injection of service water or firewater does not actually remove energy from containment, but merely decreases the rate of containment heat up it cannot be credited as a success path for containment heat removal. In reality this could increase the time available to recover RHR, but it is not credited in this analysis.

The fault tree for RHR/Containment heat removal is used to estimate the branch fractions for each of the containment event sequences, with the conditional probabilities being implicitly or explicitly carried by the event tree logic. For example, the question of RHR is not asked if LP injection has failed.

If LP injection is successful the random equipment failures in the standby service water system become the predominant causes to be considered in the assessment of the failure probability for RHR.

Vent Recovered (VNT)

If containment heat removal with the RHR system is unsuccessful, it may be possible to remove sufficient energy from the containment via the purge system. The preferred path would be from the wetwell so that the suppression pool remains unbypassed. The vent should be available following the restoration of AC power when control power to the valves and instrument air to move the operators becomes available. There is a limitation on the vent however, namely that if the differential pressure across the valve disk exceeds 49 psig, the valve operators cannot generate enough force to open them. Since Emergency Operating Procedures indicate that the vent should be opened at 39 psig, there is a relatively small window of opportunity for implementation. MAAP simulations for important representative sequences were performed to confirm the feasibility of vent operation.

The fault tree for venting is used to estimate the split fraction for venting. The estimate is dominated by the human non-response probability and the extent to which containment pressure allows it.

Recovery of PCS as a Containment Heat Sink (PCS)

If the balance of plant systems can be restored to operability after offsite power is restored to the site, the main condenser can be used as a heat sink. This will require restoration of the circulating water system so that condenser vacuum can be reestablished, condensate and the MSIVs must be reopened. This is expected to take about 8 hours so the split fraction is based on a comparison between the time required to accomplish the PCS restoration tasks and the time window which is available to complete them successfully before containment fails.

It should also be noted that to operate the MSIVs the nitrogen used for motive power must have a pressure differential of 46 psig. Because of the nitrogen pressure feeding the MSIV actuators, the valves will likely not be operable when containment pressure exceeds 54 psig. The actual containment pressure was inferred from MAAP simulations which were performed for specific accident sequences.

Containment Failure Mode (CFM)

The containment strength characteristics are more fully described in Section 4.3 of this report, but in brief, when destructive containment overpressure occurs several possibilities exist. There may be:

- a catastrophic failure leading to a large breach of the containment pressure boundary which results in rapid containment depressurization and suppression pool boiling. It is assumed in the analysis that a large break will always result in a bypass of the suppression pool.

- Membrane tears which are self limiting, i.e. stop growing when the containment pressure stops increasing. This failure mode will result in stabilization of the event where the break is removing sufficient energy to prevent further increases in containment pressure. Eventually the containment pressure will start to decrease as decay heat levels drop.

  There are three potential sites for this type of failure, each of which is equally likely, see Section 4.3. These locations are:

  - junction of the upper cylinder and the cone
  - lower part of the equipment hatch
  - in the wetwell region near the horizontal stiffeners

Of these break locations, the first two are of particular importance because they result in suppression pool bypass.

The split fractions for this event are based on judgement and the results from the strength analysis in which small tears are expected to be the preferred failure mode. Therefore, given a severe overpressure or overtemperature, the probabilities for each containment failure mode are assumed to be:

- large: 0.01
- small: 0.99.

Note that overpressure rate would indicate tear also, except possibly ATWS where more rapid rate occurs.

If the failure is small, the probability of failure occurs in:

- drywell: 0.67
- wetwell: 0.33.

This last split fraction is the one used in resolution of the succeeding node, "Suppression Pool Bypass Occurs" (SPB).

Suppression Pool Bypass Occurs (SPB)

The resolution of this node for each containment sequence requires an understanding of the specific events which have already occurred and the specific conditions which resulted. To provide the understanding needed during exploration of this event it was found convenient to develop a sub-event tree instead of a fault tree. When the tree was completed it was only necessary to map the containment sequences onto it to determine which branches were conditionally applicable (1.0 or 0.0) and which split fractions were appropriate. The results were accumulated and returned to the main CET where they were used to define the conditional probability for the defined state (yes or no). The sub-event tree is shown in Figure 4.5.1.1-2.

4.5.1.1.1 Suppression Pool Bypass Occurs - Sub-event tree

The top events of this sub-event tree are discussed as follows.

Overpressure Causes Containment Structural Failure (CFM)

This node serves to separate the issues associated with structural failure of containment and the internal failures which result in suppression bypass when containment is vented to the reactor building (purge) or the condenser (PCS) or has a failed penetration.

Type of Containment Failure (SIZE)

This node allows discrimination between the possible outcomes from structural failure, i.e. whether it is a small or large failure.

Containment Failure Location is in Drywell (DIBYP)

If containment failure mode is large, then bypass is assumed. If small, then bypass will occur 67% of the time but for the remaining 33% of the time it may not.

Pressure At The Time Of Vessel Failure (PRS)

The information for this node is taken from the sequence in the CET and is represented as either High or Low with conditional probabilities of 1.0/0.0. The information is used later in the tree to determine the location and coolability of the debris which is ejected from the vessel at the time of failure.

LP Injection Available Before VF (LP)

This node is again represented by a true/false, 1.0/0.0, split fraction which defines the actual conditions represented by the sequence in the CET. The reason that this is important is that if LP injection is available but not used in a high pressure sequence then there is the possibility that the operator will initiate drywell sprays. This is only of interest during LOCAs. In the SBO sequence, recovery of injection is dominated by recovery of power - if power is not restored then there will be no way of having water in the cavity prior to vessel failure. If power is restored, injection will be possible and will prevent vessel failure.

Water in Cavity at time of Vessel Failure (WATER)

Because the drywell floor drains directly to the cavity, operation of the drywell sprays will lead to water accumulation in the pedestal cavity prior to vessel failure. This is an important factor in the assessment of the likelihood that a high pressure melt ejection will result in a steam explosion in the pedestal cavity. The node is resolved on the basis of the probability that in this sequence the operating staff will or will not actuate the drywell sprays.

Steam explosion in Cavity Fails the Drywell (EXP)

Even though there may be a high pressure melt ejection and water in the cavity, there is still the possibility that the steam explosion will not occur with sufficient power to fail the drywell. This node is quantified by judgement and is included to allow the possibility for sensitivity analysis to determine how important steam explosions may be.

Injection Recovered After Vessel Failure but Before Containment Failure (INJ)

The yes/no, 1.0/0.0, split fraction is provided by the specific sequence within the CET. It is important to answer whether the debris can possibly be cooled outside the vessel or whether it will be impossible.

Debris is Cooled (CL)

If there is a high pressure melt ejection the core may remain inside the pedestal or be transported into the drywell where it can contact the metal in the downcomers, the omega seal or the SRV standpipes. If this is the case, bypass failures are expected. The question to be resolved is the relative likelihood that the core can or cannot be cooled if injection (RHR) pumps are available.

In sequences which involve a low pressure melt, it is assumed that all the debris will remain inside the cavity and that if this is the case it will be of sufficient depth to preclude its being coolable. All LP melt ejection scenarios are assumed to result in failure of the concrete floor in the pedestal where the concrete is thinnest, namely in the sump region. There is also a high likelihood the sump drains which pass through the suppression chamber will fail as a result of interaction with the melt, a condition which leads to bypass. Because the melt is expected to travel from the cavity to the suppression pool relatively slowly (catastrophic failure of the floor is not anticipated), the possibility of a steam explosion in the suppression pool is not considered.

Overtemperature in the Drywell Causes Drywell Failure (DWOT)

If the debris is dispersed throughout the pedestal cavity or the drywell and it is uncooled, the drywell temperature will become high enough to initiate premature head seal failure or cause overtemperature failures where the melt is in direct contact with the metal in the drywell downcomers, omega seal, SRV standpipes or drains. This is expected whenever there is a high pressure melt ejection without drywell sprays (and vessel injection) or a low pressure melt with no injection to provide ex-vessel cooling.

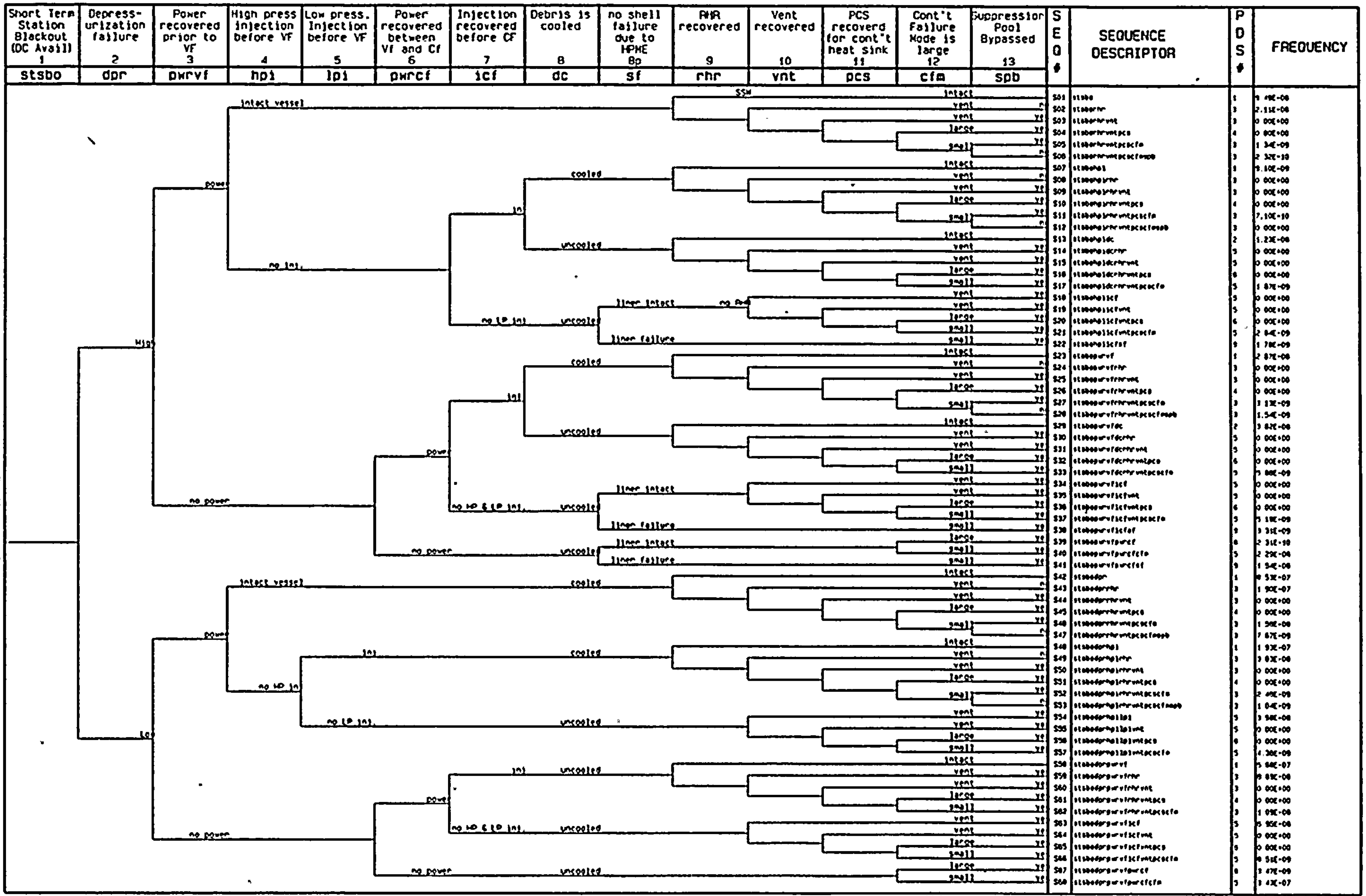
This node is only used to provide additional insight into the likelihood of containment failure modes.

Drywell Floor Failures Caused by Molten Core-Concrete Interaction (CCI)

If there is a low pressure melt, even if core injection is functional, the debris will be uncoolable and will result in MCCI and eventual failure of the pedestal floor. This node, in combination with the previous node (DWOT) provides grater insight into the possible causes of drywell bypass and the factors that may influence them.

Suppression Pool Bypass Occurs (BYP)

This node provides the final definitive state for the sequences as they are mapped onto the sub-event tree. The probabilistic and conditional events which are part of the accident sequence taken from the CET have been mapped onto the sub-event tree for drywell bypass and the resultant conditional probabilities for bypass/no bypass are returned to the main CET.



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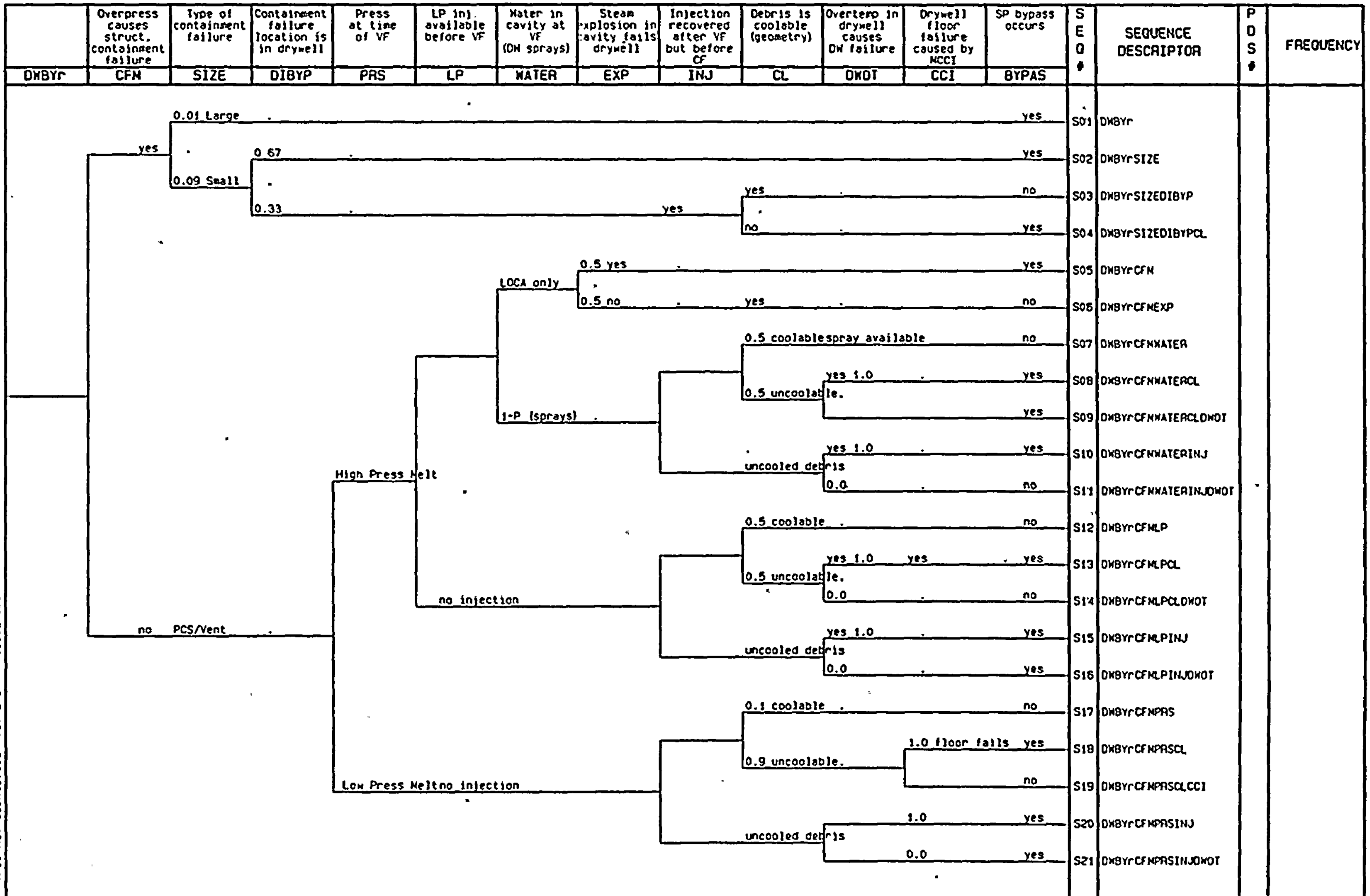
Quantification Date: 4-09-94 8:44:03am TOTAL CMF = 2.69E-006

Short term Station Blackout - WNP-2
Containment Event Tree - Isolated
Revision 1, April 8, 1994
(TE-S19)
Figure 4.5.1.1-1

O:\EI\INUSE\FIG\DWBYR.EVT 10:43:08am 6-10-94 NUPRA 2.1a WPPSS
Tree NOT Quantified TOTAL CMF = 0.00E+000



Suppression Pool Bypass - WNP-2
Containment Sub-event Tree
Revision 0, February 19th, 1994

Figure 4.5.1.1-2

4.5.1.2 Long-term Station Blackout

The difference between the long-term station blackout and the short-term station blackout CETs lies primarily with the initiating conditions or the plant damage state which results from the Level 1 sequences. In a short-term SBO, injection is lost early so that before the vessel fails the batteries have not yet been depleted so that depressurization is feasible. In the long-term SBO, initially injection continues for some time and injection stops when the batteries deplete. This means that there is no way to depressurize so all of the sequences result in vessel failure at high pressure.

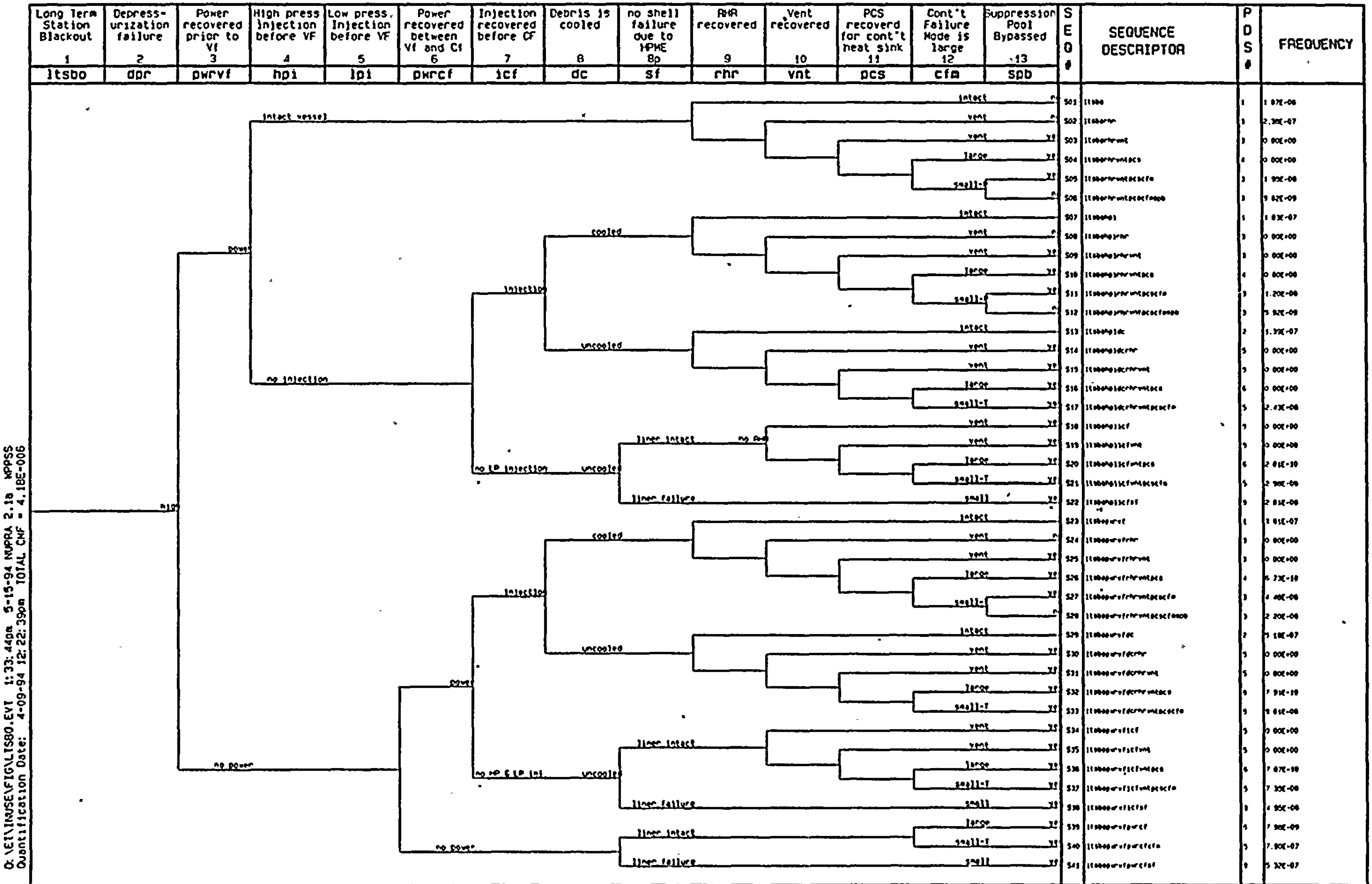
There are two classes of event resulting from the Level 1 analysis:

- those in which injection fails and power is not recovered within 4 to 10 hours.
- those sequences in which high pressure injection continues for a longer period of time and power is not recovered within 10 to 24 hours.

The treatment of all of the events is similar to that used in the case of short-term SBO except that the conditional probabilities for recovery of off-site power before vessel failure and before containment failure reflect the new time regimes.

Containment pressure is one of the other factors which are influenced by the extended injection times because of the increased energy which has been transferred to the suppression pool. This results in containment pressures being somewhat higher throughout the long-term containment sequence and has an effect on the probability that the vent can be implemented successfully after recovery of power. If containment pressure is above 49 psig, the vent is no longer feasible. The actual containment pressure was determined from MAAP simulations for individual sequences which represent the damage class.

The long-term SBO CET is shown in Figure 4.5.1.2-1.



Long term Station Blackout - WNP-2
Containment Event Tree - Isolated
Revision 1, April 8, 1994
(TE-S17A)
Figure 4.5.1.2-1

4.5.1.3 Transient Event Trees

The events in the transient event tree (see Figure 4.5.1.3-1) are similar to those already discussed for the short-term SBO CET, with the few exceptions outlined below.

Recovery of Offsite Power

The first major difference between the transient and the SBO CETs results from the fact that on-site power remains available on the safety related buses throughout the accident, so that recovery of offsite power is not relevant to the success of injection or the normal containment/DHR systems. The only point at which recovery of a source of offsite power is important to the transient containment event sequences is seen in the transient sequences in which the initiating event also includes failure of the electrical feed from the Ashe sub-station to the start-up transformer. In this case recovery of power to the BOP via the 500 kv backup or the restoration of the original Ashe feed must be successful prior to venting or the implementation of PCS as a heat sink. Venting requires the availability of containment air (AC dependent) and PCS requires full energization of the BOP. The non-restoration probabilities for these events are included in the fault trees from which the split fractions are determined.

Containment is intact at the onset of core damage (VECF)

The second major difference between the transient and short-term SBO CET relates to the first event in the tree, "Containment is intact at the onset of core damage," (VECF). This node allows the separation of TW sequences from other transient sequences. In TW, a transient is followed by successful SCRAM and injection but long-term containment heat removal remains unavailable. The reactor continues to reject energy into the suppression pool via the depressurization or safety relief valves until the saturation pressure associated with suppression pool temperature reaches the containment failure point. Containment is assumed to fail and the resultant energetic release of steam to the reactor building results in high temperatures/humidities in the RHR and HPCS pump rooms and subsequent failure of the drive motors. After failure of the injection systems, water over the core quickly boils off and core melt begins.

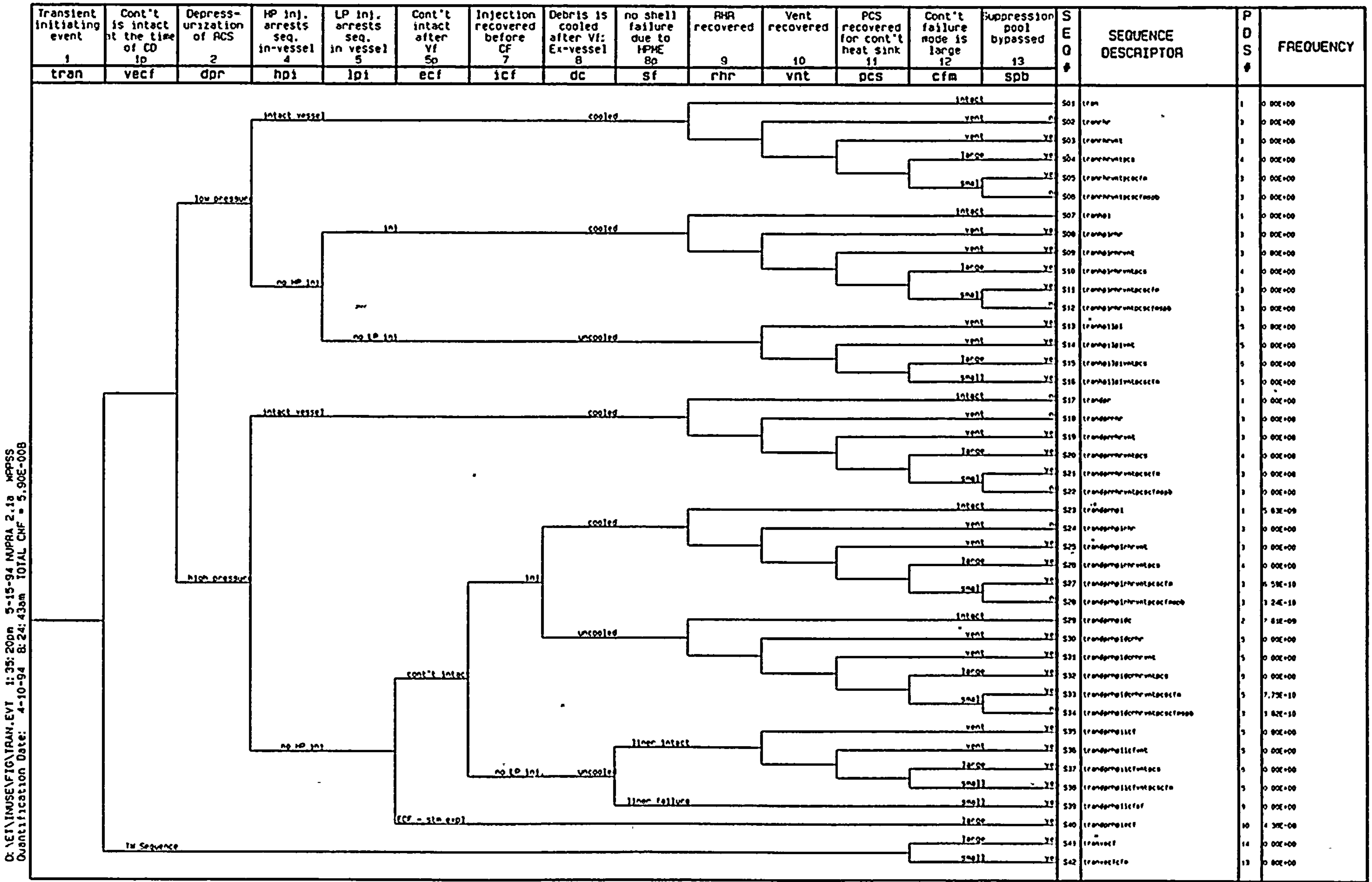
In the case of TW sequence, the only Level 2 issues are whether the containment breach is large or small. Bypass of the suppression pool will always be true based on the Level 1 outcomes.

The "failure to vent sequences" which represent long-term loss of containment heat removal, either because the sequences involve loss of RHR pumps, the unrecoverable failure of standby service water, or loss of the support systems required to vent will also be treated within this CET.

Containment is Intact After Vessel Failure (ECF)

The third difference between the short-term SBO CET and the transient CET is that because it is possible the drywell sprays may be operating to cool the drywell during a high pressure sequence (power is always available), even though there is no injection (HP injection failed). If this is the case it may be possible to have both a high pressure melt ejection and standing water in the pedestal cavity, conditions which may be favorable for a destructive steam explosion in the cavity.

The CET allows for consideration of the possibility that a steam explosion at the time of vessel failure will cause pedestal and subsequential containment failure and for assessment of the sensitivity of the final results to these assumptions.



Transient - WNP-2
Containment Event Tree - Isolated
Revision 1, April 8, 1994
(TT-S19)
Figure 4.5.1.3-1

4.5.1.4 Anticipated Transient without SCRAM (ATWS) Event Trees

The ATWS sequences which have been found important to the Level 1 results are:

- Failure to SCRAM following a transient which is the result of mechanical rod failure. This means that the failure to SCRAM cannot be recovered without the addition of an absorber (boron) to the primary system. The failure to SCRAM is followed by:

  - Successful implementation of the recirculation pump trip to reduce power to about 40%.
  - Successful operation of the SRVs to maintain system pressure at the SRV setpoint (open and reclose).
  - Successful actuation of the Standby Liquid Control (SLC) system.
  - Failure to inhibit ADS, which initiates rapid primary system blowdown. This in turn causes low vessel level and the actuation of injection which flushes boron from the system. Large amounts of energy are continuously rejected to the suppression pool through the depressurization valves during the succeeding period of alternating injection (controlled by vessel level) and vessel steaming when power oscillations are expected in the primary system.

  The containment is expected to fail from overpressure within a relatively short period of time. It is postulated that the release of high temperature steam into the reactor building which will follow containment failure will result in high ECCS pump room temperatures and fail the low pressure injection and RHR systems. This means that core melt and vessel failure become unavoidable, a situation similar to that seen in the TW sequences, except that the timing is much shorter.

  It is also possible the severe power oscillations which are expected to occur following depressurization of the primary system and the initiation of LP injection could of itself result in core melt. In which case core melt will precede containment failure by a short period of time, but, the net effect on source term appears similar so no attempt was made to discriminate between the two possibilities.

- Failure to SCRAM (mechanical failure of the rods) following a transient, which in turn is followed by:
  - successful implementation of the recirculation pump trip to reduce power to about 40%.
  - successful operation of the SRVs to maintain system pressure at the SRV setpoint
  - failure of the Standby Liquid Control (SLC) system.

The result of this event will be either:

- continued power generation within the core, severe increase on primary system pressure because of the inability to remove sufficient energy through the SRVs and eventual failure of the primary system pressure boundary (ATWS induced LOCA).

- continued energy generation in the primary system, rejection of large amounts of energy to the suppression pool which in turn results in overpressure failure of containment.

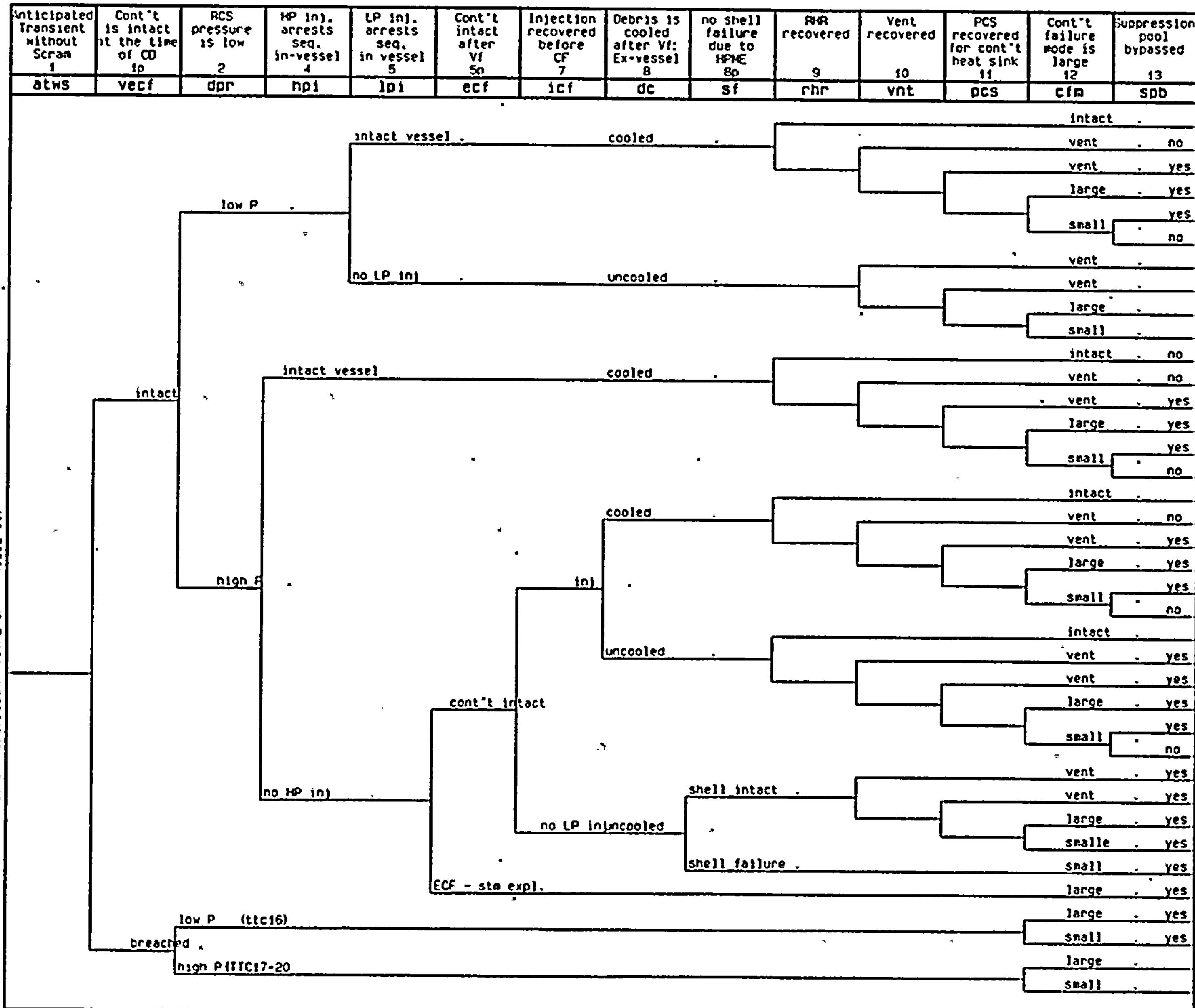
Of these two possibilities, the second seems most likely because the SRVs have the capability to remove 100% reactor power and the reactor will only be at 40% (nominal). The scenario then becomes similar to the TW sequence except for the relative timing of events. This difference is accommodated by selection of appropriate conditional probabilities for sequence quantification.

The events in the ATWS event tree (see Figure 4.5.1.4-1) are similar to those already discussed for the transient CET, with the exception outlined below.

Primary System Pressure is Low (DPR)

If containment failure occurs prior to core damage, the condition of primary system pressure is asked in the ATWS CET. This is to address the low pressure and high pressure sequences such as TTCS17 and TTCS20 which will results in different release categories.

If containment is intact prior to core damage, Level 1 conditions indicate that the low pressure injection system is already functioning (in sequence TMCS16). Therefore, the question of high pressure injection is not asked.



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
S01	atws	1	0.00E+00
S02	atwsrhr	3	0.00E+00
S03	atwsrhrvnt	3	0.00E+00
S04	atwsrhrvntpcs	4	0.00E+00
S05	atwsrhrvntpcscfm	3	0.00E+00
S06	atwsrhrvntpcscfmspb	3	0.00E+00
S07	atwslpi	5	0.00E+00
S08	atwslpivnt	5	0.00E+00
S09	atwslpivntpcs	6	0.00E+00
S10	atwslpivntpcscfm	5	0.00E+00
S11	atwsdpr	1	0.00E+00
S12	atwsdprrhr	3	0.00E+00
S13	atwsdprrhrvnt	3	0.00E+00
S14	atwsdprrhrvntpcs	4	0.00E+00
S15	atwsdprrhrvntpcscfm	3	0.00E+00
S16	atwsdprrhrvntpcscfmsp	3	0.00E+00
S17	atwsdprhpi	1	0.00E+00
S18	atwsdprhpirhr	3	0.00E+00
S19	atwsdprhpirhrvnt	3	0.00E+00
S20	atwsdprhpirhrvntpcs	4	0.00E+00
S21	atwsdprhpirhrvntpcscf	3	0.00E+00
S22	atwsdprhpirhrvntpcscf	3spb	0.00E+00
S23	atwsdprhpidc	2	0.00E+00
S24	atwsdprhpidcrhr	5	0.00E+00
S25	atwsdprhpidcrhrvnt	5	0.00E+00
S26	atwsdprhpidcrhrvntpc	6	0.00E+00
S27	atwsdprhpidcrhrvntpc	5fm	0.00E+00
S28	atwsdprhpidcrhrvntpc	3fm	0.00E+00
S29	atwsdprhpiicf	5	0.00E+00
S30	atwsdprhpiicfvnt	5	0.00E+00
S31	atwsdprhpiicfvntpcs	6	0.00E+00
S32	atwsdprhpiicfvntpcsc	5	0.00E+00
S33	atwsdprhpiicfsf	9	0.00E+00
S34	atwsdprhpiecf	10	0.00E+00
S35	atwsvecf	14	1.81E-09
S36	atwsvecfcfm	13	1.79E-07
S37	atwsvecfdpr	14p	2.58E-09
S38	atwsvecfdprcfm	13p	2.56E-07

D:\EI\INUSE\FIG\ATWS.EVT 9:58:14am 5-24-94 NUPRA 2.1a WPPSS
Quantification Date: 4-23-94 9:57:59am TOTAL CHF = 4.39E-007

Anticipated Transient Without Scram - WNP-2
Containment Event Tree
Revision 1, April 7, 1994
(TTC-17, TTC-16, TTC-20, AO-15)
Figure 4.5.1.4-1

4.5.1.5 <u>Small LOCA Containment Event Tree (Break Size < 4")</u>

The small LOCA containment event tree is similar to the transient event tree, even to the first event in which the question of containment integrity prior to core damage is resolved.

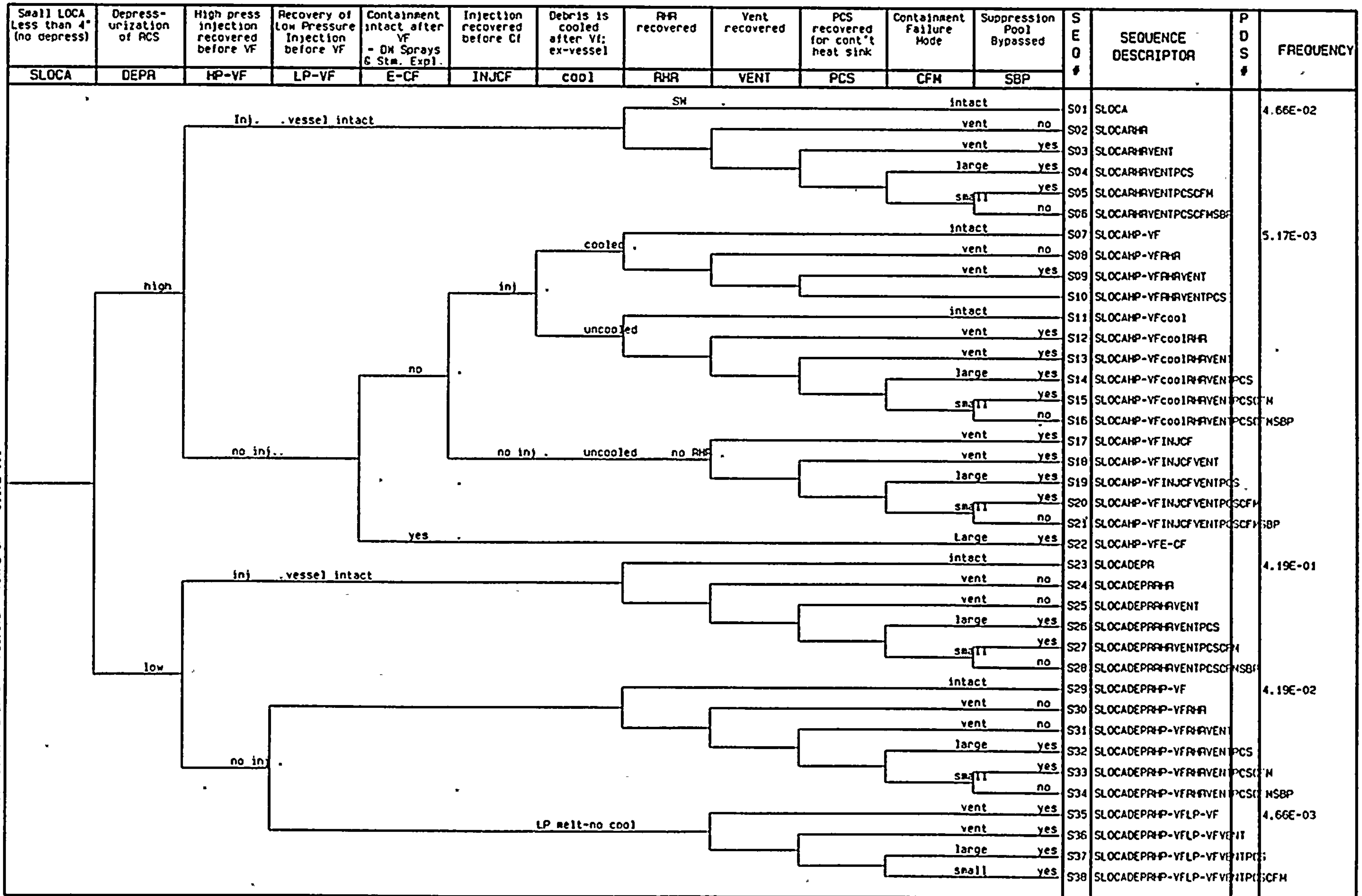
The reason for the similarity between the CETs is that because the LOCA break size is too small to remove enough energy from the primary system to depressurize it naturally, the LP injection systems do not become available without depressurization. This is identical to the case for transient induced accident sequences.

4.5.1.6 <u>Large LOCA Containment Event Tree (Break Size > 4")</u>

Because in the case of the large LOCA the break allows the release of sufficient energy to depressurize the primary system, all the sequences occur at low pressure. The CET is the same as the low pressure section of the small LOCA CET.

Because this tree is simpler than the others, the initial event "Containment Isolation is Successful" (ISOL) has been explicitly shown, rather than implicitly within the initiating damage state frequency.

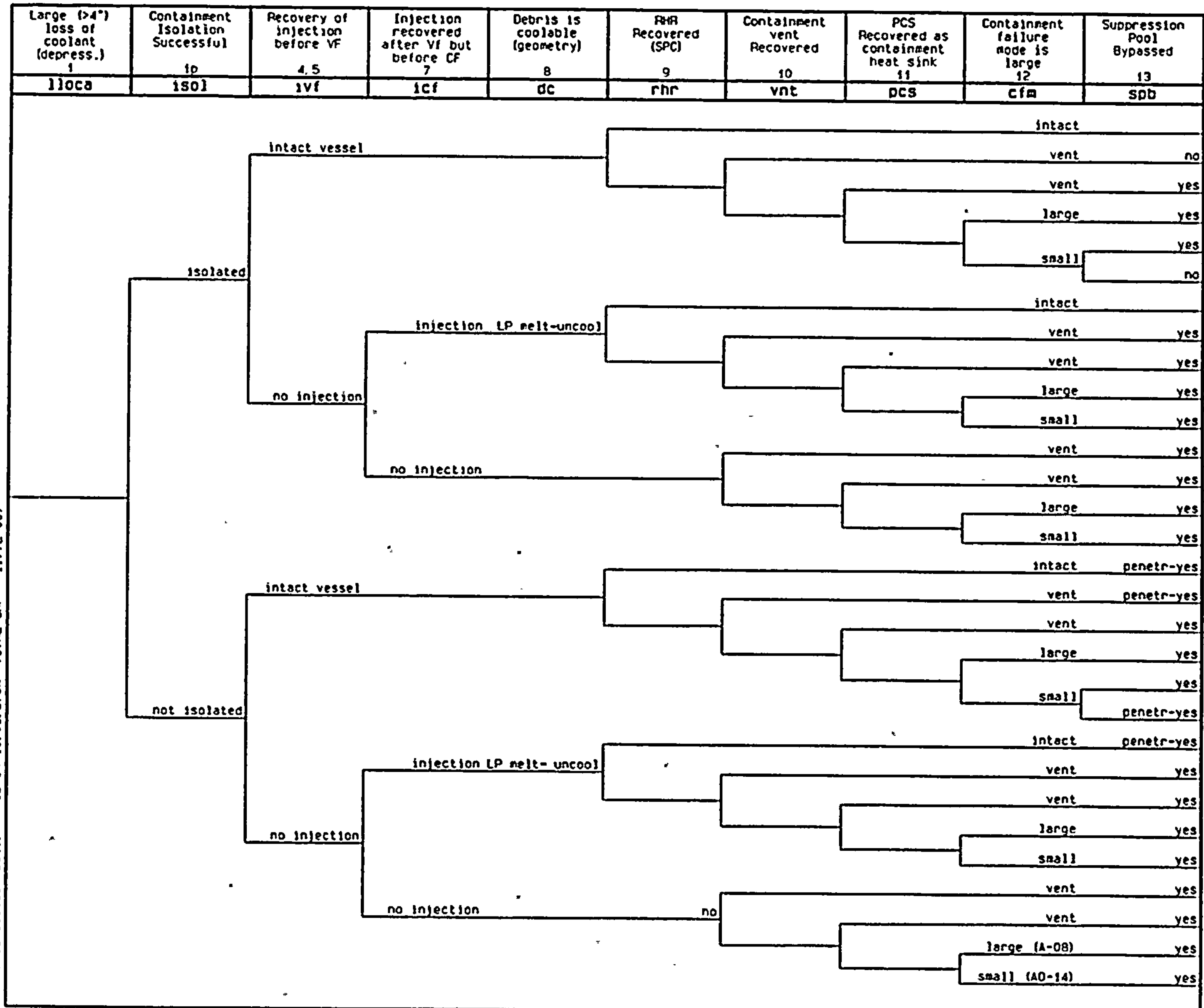
D:\EI\INUSE\FIG\SLOCA.EVT 10:08:08am 5-24-94 NUPRA 2.1a WPPSS
Quantification Date: 2-17-94 4:33:57pm TOTAL CMF = 5.17E-001



Small LOCA - WNP-2
Containment Event Tree

Revision 1, February 18th, 1994

Figure 4.5.1.5-1



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
S01	lloca	1	0.00E+00
S02	llocarhr	3	0.00E+00
S03	llocarhrvnt	3	0.00E+00
S04	llocarhrvntpcs	4	0.00E+00
S05	llocarhrvntpcscfm	3	0.00E+00
S06	llocarhrvntpcscfmspb	3	0.00E+00
S07	llocaivf	1	0.00E+00
S08	llocaivfrhr	3	0.00E+00
S09	llocaivfrhrvnt	3	0.00E+00
S10	llocaivfrhrvntpcs	4	0.00E+00
S11	llocaivfrhrvntpcscfm	3	0.00E+00
S12	llocaivficf	5	0.00E+00
S13	llocaivficfvnt	5	0.00E+00
S14	llocaivficfvntpcs	6	0.00E+00
S15	llocaivficfvntpcscfm	5	0.00E+00
S16	llocaisol	11u	0.00E+00
S17	llocaisolrhr	11u	0.00E+00
S18	llocaisolrhrvnt	11u	0.00E+00
S19	llocaisolrhrvntpcs	4u	0.00E+00
S20	llocaisolrhrvntpcscfr	11u	0.00E+00
S21	llocaisolrhrvntpcscfr	[illegible]	0 00E+00
S22	llocaisolivf	11u	0.00E+00
S23	llocaisolivfrhr	11u	0.00E+00
S24	llocaisolivfrhrvnt	11u	0.00E+00
S25	llocaisolivfrhrvntpc	4u	0.00E+00
S26	llocaisolivfrhrvntpc	[illegible]	0.00E+00
S27	llocaisolivficf	13u	0.00E+00
S28	llocaisolivficfvnt	13u	0.00E+00
S29	llocaisolivficfvntpc	17	2.99E-08
S30	llocaisolivficfvntpc	[illegible]	1.47E-07

O:\EI\IHUSE\FIG\LLOCA.EVT 1:37:30pm 5-15-94 NUPRA 2.1a WPPSS
Quantification Date: 4-13-94 10:03:37am TOTAL CHF = 1.77E-007

Loss Of Coolant Accident (LOCA) - Large (>4") - WNP-2
Containment Event Tree
Revision 1, April 8, 1994
(AO-S14, A-S08)
Figure 4.5.1.6-1

4.5.2 Fault Tree Development

A fault tree was developed for each of the events which occur within the containment event trees so that not only was it possible to confirm that all of the important influences on sequence behavior were explicitly considered, but also to calculate the branch fraction at each CET node. Because each fault tree had to be given a sequence-specific configuration each time it was used to calculate a branch fraction, it was adjusted to reflect all Level 1 and 2 conditional probabilities, the conditional probability for the top event calculated and returned to the appropriate node in the CET. In this way the CET event independence was maintained and the sequence frequencies could be calculated arithmetically without recourse to fault tree merging.

The logic, assumptions and the rationale for simplification which may be implicitly part of each fault tree is briefly described below. It must be remembered, however, that only appropriate pieces of each fault tree may have been used in quantifying individual sequence frequencies. In some cases, particularly in the quantification of drywell failure modes, the fault tree was used only to support structuring and quantification of the sub-event tree for suppression pool bypass.

4.5.2.1 Quantification of Basic Events in the Fault Trees

Because the operation of hardware may be affected by the severe environmental conditions which occur during a core melt accident. The following data were used for Level 2:

- 0.05 - Failure to start one train of a two train system, given the availability of all required support systems
- 0.1 - Failure to start a single train system, given the availability of all required support systems
- 0.1 - Operator fails to initiate operation of a system following recovery of required support systems

Exceptions to these assumptions are explicitly identified in the discussion of the fault tree structures below.

4.5.2.2 Fault Tree Top Events and Structures

The fault trees developed are shown in Figures 4.5.2.2-1 to 4.5.2.2-10.

Failure of Containment Isolation (ISL)

Faults which contribute to failure of containment isolation were subdivided by initiator so that if there were a difference in conditional failure probabilities they could be correctly accommodated. In each case the structure was similar and considered the following general issues:

- failure of a penetration to isolate on demand, either because the isolation system fails to actuate or one of the active penetrations fails because of local valve faults. the failure data was taken from results of the Level 1 systems reliability analysis:

  Group 1 - 9.3E-4 (MSIVs)
  Group 2 - Valves less than .75" so will have no effect
  Group 3 - Normally closed valves, > 2", will have no effect
  Group 4 - 3.5E-6 (Reactor Building. Closed Cooling and Fuel Pool Cooling)
  Group 5 - 1.6E-4 Sample lines (of no concern)
  Group 6 - 1.0E-3 (RHR)
  Group 7 - 3.9E-3 (RWCU)

  There is some probability that the operating staff can manually isolate a failed penetration, either by closing the errant valve or by closing other non-isolation valves downstream. The likelihood of successful isolation given failure of the automated system was assumed to be 0.5 except in station blackout sequences.

- failure of the wetwell vacuum breakers to remain closed during the sequence. The flow paths from the suppression chamber to the reactor building comprise two normally closed check valves in series. The check valves have magnets in the disk to hold them in the closed position. If they are open during normal operation the leakage should be detectable with the radiation monitoring system so there is the presumption that at the onset of the event they will be in the closed position. This means that since a pressure increase increases the seating force, the failure mode of concern will be rupture. Failure of a vacuum breaker flow path is computed as 1 out of 3 common cause rupture failure of two check valves in series and it is found to be negligible.

Failure to Start HP Injection Prior to Vessel Failure (HPI)

HP injection dependent system failures are embodied in the event trees, either explicitly as in the case of SBO or implicitly within the Level 1 plant damage states. This means that the random equipment failures which fail the hardware are of interest.

The tree must accommodate earlier failures which are not recoverable (failed in the Level 1 sequence) or occur during the time frame represented by the containment event trees. The fault tree includes three events:

- operator fails to align and initiate HP injection following recovery of any needed support systems
- pumps in a damaged condition, perhaps because at the time of loss of injection which cause core melt, they failed because of inadequate NPSH. In this case impeller damage is a possibility
- the pumps are in (apparent) good running condition but they fail to start on demand

Failure to Start LP Injection Prior to or After Vessel Failure (LPI)

LP injection dependent system failures are embodied in the event trees, either explicitly as in the case of SBO or implicitly within the Level 1 plant damage states. This means that the random equipment failures which fail the hardware are of interest.

LP injection can only be successful when the primary system pressure is below its shut-off head, but, this conditional probability is included in the event trees so it is not included as a conditional event within the fault tree.

The fault tree must accommodate earlier failures which are not recoverable (failed in the Level 1 sequence) or occur during the time frame represented by the containment event trees. The fault tree includes three events:

- operator fails to align and initiate LP injection following recovery of any needed support systems
- pumps in a damaged condition, perhaps because at the time of loss of injection which cause core melt, they failed because of inadequate NPSH. In this case impeller damage is a possibility
- the pumps are in (apparent) good running condition but they fail to start on demand

Failure of both LP and HP injection (HLP)

The structure of this fault tree is similar to HPI and LPI. Random equipment failures and operator errors are modeled for both high pressure and low pressure systems.

Early Containment Failure at Time of Vessel Failure (ECF)

The fault tree models challenges to containment integrity at or near the time of vessel failure. These challenges could result from a number of sources, including:

- in-vessel steam explosion,
- vessel blowdown,
- ex-vessel steam explosion,
- direct containment heating, and
- hydrogen combustion.

Debris Bed not Cooled (DC)

The importance of knowing whether or not the debris bed is cooled after the molten material is ejected from the vessel relates to the assessment of the likely failure modes for the drywell. This is used only in the HPME scenarios since after a low pressure vessel failure the total melt is expected to remain in the cavity. Therefore, it is expected that even if water is available the debris will be uncoolable.

If the debris is in the pedestal and:

- uncooled, overtemperature may cause failure of the drywell head; or

- accumulates on the pedestal floor in a configuration which is uncoolable even if though it covered with water, CCI can cause floor or pedestal wall failure.

If the debris is outside the pedestal and uncooled, it may cause overtemperature failure of the drywell or failure of the omega seal, downcomers or SRV standpipes.

In each case, a leakage path which bypasses the suppression pool results from drywell failure.

The fault tree structure attempts to assess the influences of each of these factors and provide the likelihood that water is available to cool the debris. The second question which needs an answer is, "if water is available, is the debris in a coolable geometry?" This is addressed outside the fault tree in the CET node split fraction.

The fault tree looks at debris inside and outside the pedestal and provides the opportunity to estimate the relative fraction for each in high pressure melt ejection sequences. It is assumed that in a low pressure melt ejection, all of the melt will remain in the pedestal cavity. This assumption is based upon the expectation that release from the vessel will not be highly energetic and because the opening in the pedestal is nine feet above the floor there will be little or no entrainment from the cavity.

It is assumed that for cooling of the debris ejected from the pedestal, drywell sprays must be available from the RHR pumps. This means that failure will result if either the operators do not restore them to operability following the failure of a support system, or fails to start them when they are needed. This can be because either they were damaged earlier in the sequence, or they are generally operable but fail to start on demand.

RHR not Restored Before Containment Failure (RHR)

To prevent containment failure it is essential to establish some form of containment heat removal. The preferred path is through the suppression pool cooling system. This requires the operation of RHR pumps and their associated RHR heat exchangers. The fault tree provides the means for assessing the likelihood of failure for this function by including:

- failure of the RHR pumps, either because they are in the failed state because they were damaged earlier in the sequence, or because they are nominally operable but fail to start on demand.

- Dependent failures are included implicitly within the CET, with the exception of standby service water, which must be available to provide the necessary heat sink.

  As a result, the faults in RHR-1 include failure to restore standby service water, failure to reestablish cooling following the restoration of standby service water

Containment not Vented Before Containment Failure (VNT)

The EOPs direct the operators to implement venting with the purge outlet valves when the containment pressure exceeds 39 psig. The fault tree has within its provisions to consider:

- operator failing to respond in time, because though directed to open the vent at 39 psig, he has a limited window of opportunity for completing the task. This is because if the containment pressure reaches 49 psig before the valves are opened the forces exerted by the differential pressure across the disk will exceed the capabilities of the valve operator, i.e. it won't open.

- recovery of needed support systems. The vent requires containment air, which in turn requires restoration of offsite power. Since this can be determined from the sequence conditions, this event is treated as a conditional event which is/is not true. There will probably be a need for manual alignment and starting of the compressors and cooling systems so the recovery of air also has a human component whose success probability will vary with sequence timing. The greater the time window within which the task can be successfully performed the greater the likelihood that the task will be successful.

Failure to Recover PCS (PCS)

Once there is full recovery of offsite power the only issue facing the operating staff is whether or not they have sufficient time available to recover PCS before containment failure occurs. The fault tree also has provision for an assessment of containment pressure, because the MSIVs likely cannot be opened if containment pressure exceeds 54 psig. After recovery of power after station blackout, this task is expected to take about 8 hours, but following a transient the time is expected to be much less, providing that the transient was not initiated by damage to the condenser or a major circulating water system failure.

Containment Failure Mode (CFM)

The factors considered within the CFM fault tree are:

- severe overpressure which results in a gross containment failure
- small containment failure which results in a breach of limited size

As indicated in Section 4.3, the most likely failure mode will involve membrane tears in one of three equally likely locations, referenced as locations 1, 2 and 3 in the fault tree. Two of these locations are in the drywell and result in suppression pool bypass, one at the base of the equipment hatch and the other at the junction of the cone and upper cylinder. The third location is in the wetwell above the horizontal stiffeners, so a release from this location would be scrubbed by the suppression pool.

When containment pressure stops increasing and the forces in the membrane stabilize, the nature of a membrane tear results in its arrest. This means that the failure increases in size until the breach removes enough energy to maintain a balance with decay heat, the breach will no longer grow, and containment pressure will no longer increase. This precludes the possibility that pressures will continue to climb until conditions favorable to some form of catastrophic failure result.

Despite the foregoing arguments against the likelihood that there will be a catastrophic failure of containment, the fault tree does include provisions for it. The large failure is expected to fail the suppression pool and result in an unscrubbed release - either because the suppression pool is damaged and loses its inventory, or because the event also results in catastrophic structural failure of the drywell.

The quantification of this fault tree was based on:

- 0.67 - probability that the failure is in the drywell, given a small containment failure
- 0.33 - probability that the failure is in the wetwell, given a small containment failure

Suppression Pool Bypassed at the Time of Containment Failure (SPB)

The Suppression Pool Bypass fault tree was used to quantify individual elements of the Bypass sub-event tree which then returned values to the main CET. This was done to overcome the difficulties in understanding and recognizing each of the sequence specific conditions which represent dependencies between events in the CET.. If these are treated incorrectly then the numerical solution of the CET, which assumes independence between events, would also be in error.

The first issue addressed by the fault tree logic is one of definition of the relationships between containment and drywell failure modes. This is addressed in the SPB fault tree in a manner which is essentially the same as that in the CFM fault tree, with the exception that consideration is give to the effects of venting. In essence, the effects of a small containment failure in the wetwell region are similar to those present when the vent is implemented with the purge system. Under these particular conditions, bypass will only occur if there is a failure of the pressure boundary between the drywell and the suppression pool. The remainder of the fault tree explores the possible reasons for failure of this boundary.

There are several possibilities considered:

- Failure of the pedestal floor or the pedestal walls from core-concrete interaction. The pedestal floor failure is the most likely mode because in the sump region it is only 3′ 3″ thick, whereas the walls are a nominal 5′ thick. There are drain lines between the drywell floor and the cavity which may represent potential vulnerabilities to MCCI, but to improve the manageability of the analysis only floor not wall failure was considered.

  The difference between the two modes of failure become even less important when the outcomes are considered - if the floor fails the melt will slump into the suppression pool, whereas if the pedestal wall fails, the vessel may fall, cause failure of the floor and again result in the core falling into the suppression pool. The only factor which is omitted from consideration by this simplification in approach is whether or not failure of the pedestal could also result in a breached containment.

- Failure of the drywell/suppression chamber pressure boundary. This can result from reactions between the melt debris and metal piping and accessories which fails:
  - the drywell downcomers.
  - the collars around the SRV standpipes where they penetrate the dry well floor.
  - the omega seal.

- Failure of the drywell head seal because of extreme temperatures and moderate pressure loading within the drywell

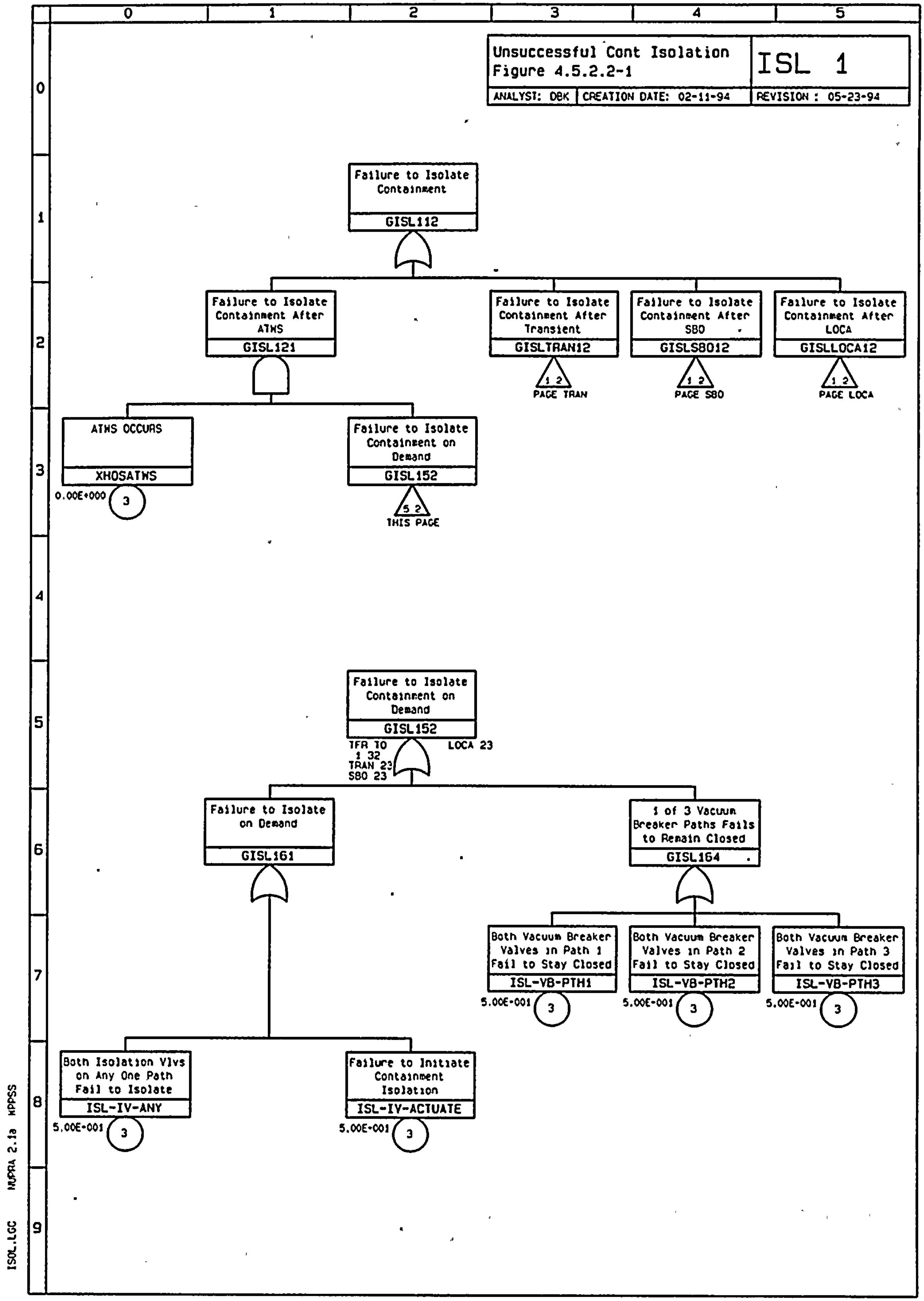
The resolution of these concerns within the fault tree is founded upon the following assumptions:

If the vessel is at high pressure when it fails, there is both the possibility that the melt will remain in the pedestal or that it will be ejected to the drywell through the 3′ x 7′ opening in the pedestal.

- If the melt is ejected from the cavity, drywell sprays are required to cool it.
- if the melt remains within the cavity it will be splattered over the walls and even if injection to the vessel is successful, water leaving the failed vessel will likely not cool all of the debris directly but the steam generated by the regions where water does come in contact with the melt will cool the drywell enough to keep it below the temperature (700 °F) where head seal failure become likely.

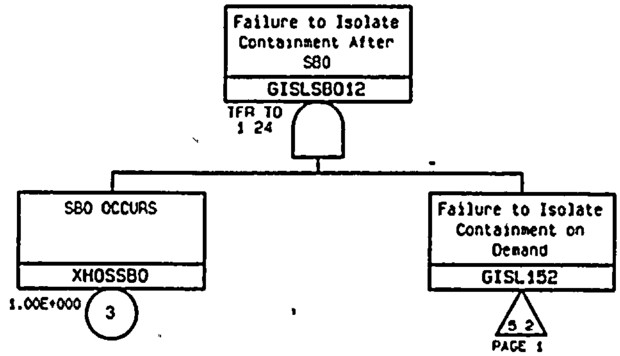
If the vessel fails at low pressure, the melt is expected to remain in the cavity and form a crusted pool in the bottom three feet of the pedestal. This depth of melt is expected to be enough to prevent its being cooled at the melt/concrete interface, even if it is flooded with cooling water. This means that MCCI will cause erosion/failure of the pedestal floor.

It is possible that the pedestal drain line which passes through the suppression chamber to the liquid waste system, will fail preferentially, in which case the melt may be released slowly to the suppression pool before it has sufficient contact time to cause a failure of the concrete floor. This event is also considered in the fault tree as a contributor to suppression pool bypass.



0
1
2
3
4
5
6
7
8
9
Unsuccessful Cont Isolation
Figure 4.5.2.2-1
ISL 1
ANALYST: DBK
CREATION DATE: 02-11-94
REVISION : 05-23-94
Failure to Isolate Containment
GISL112
Failure to Isolate Containment After ATWS
GISL121
Failure to Isolate Containment After Transient
GISLTRAN12
1 2
PAGE TRAN
Failure to Isolate Containment After SBO
GISLSBO12
1 2
PAGE SBO
Failure to Isolate Containment After LOCA
GISLLOCA12
1 2
PAGE LOCA
ATWS OCCURS
XHOSATWS
0.00E+000
3
Failure to Isolate Containment on Demand
GISL152
5 2
THIS PAGE
Failure to Isolate Containment on Demand
GISL152
TFR 10
1 32
TRAN 23
SBO 23
LOCA 23
Failure to Isolate on Demand
GISL161
1 of 3 Vacuum Breaker Paths Fails to Remain Closed
GISL164
Both Vacuum Breaker Valves in Path 1 Fail to Stay Closed
ISL-VB-PTH1
5.00E-001
3
Both Vacuum Breaker Valves in Path 2 Fail to Stay Closed
ISL-VB-PTH2
5.00E-001
3
Both Vacuum Breaker Valves in Path 3 Fail to Stay Closed
ISL-VB-PTH3
5.00E-001
3
Both Isolation Vlvs on Any One Path Fail to Isolate
ISL-IV-ANY
5.00E-001
3
Failure to Initiate Containment Isolation
ISL-IV-ACTUATE
5.00E-001
3
ISOL.LGC
NUPRA 2.1a
WPPSS

Unsuccessful Cont Isolation Figure 4.5.2.2-1		ISL SBO
ANALYST: DBK	CREATION DATE: 02-11-94	REVISION : 05-23-94



ISOL.LGC NUPRA 2.1a KPPSS

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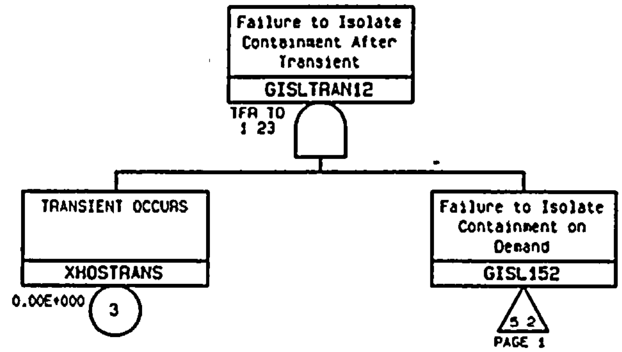
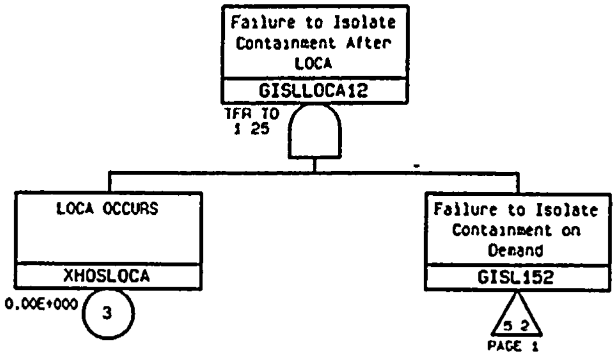
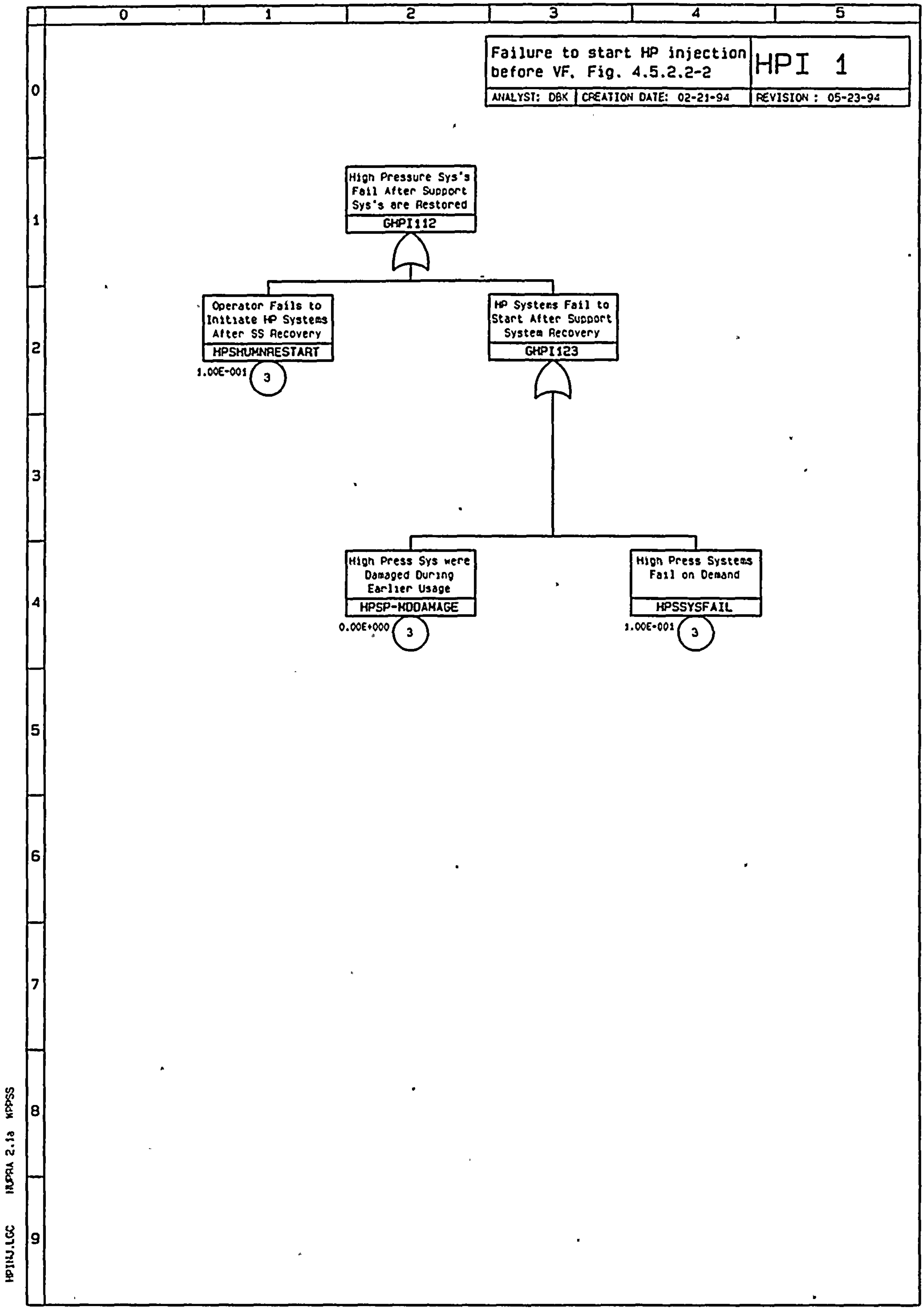


ISOL.LGC NUPRA 2.1a WPPSS

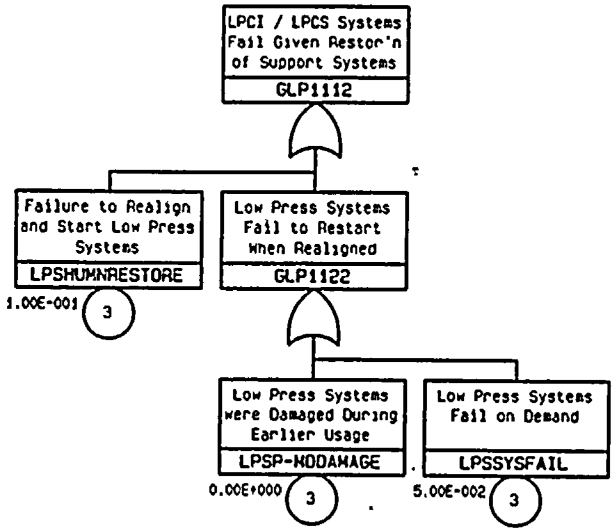
Unsuccessful Cont Isolation Figure 4.5.2.2-1		ISL LOCA
ANALYST: DBK	CREATION DATE: 02-11-94	REVISION : 05-23-94



ISOL.LGC NUPRA 2.1a WPPSS


Failure to start HP injection before VF. Fig. 4.5.2.2-2
HPI 1
ANALYST: DBK
CREATION DATE: 02-21-94
REVISION : 05-23-94
High Pressure Sys's Fail After Support Sys's are Restored
GHPI112
Operator Fails to Initiate HP Systems After SS Recovery
HPSHUMNRESTART
1.00E-001
3
HP Systems Fail to Start After Support System Recovery
GHPI123
High Press Sys were Damaged During Earlier Usage
HPSP-MODAMAGE
0.00E+000
3
High Press Systems Fail on Demand
HPSSYSFAIL
1.00E-001
3
HPINJ.LGC IRRAS 2.1a WPPSS

Failure to recover LP INJ before/after VF.Fig.4.5.2.2-3	LP1 1
ANALYST: DBK | CREATION DATE: 02-22-94	REVISION : 05-23-94

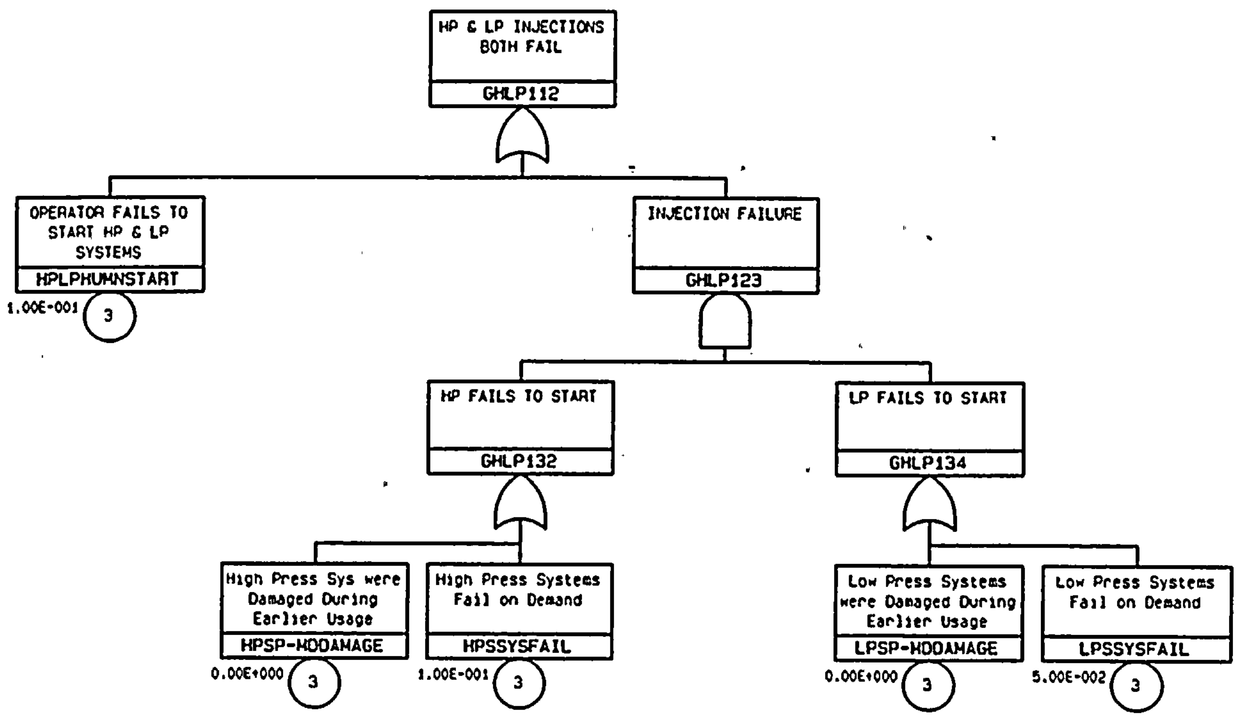


INJCF1.LGC NUPRA 2.1a WPPSS

HP and LP INJECTIONS FAILURE
Fig. 4.5.2.2-4

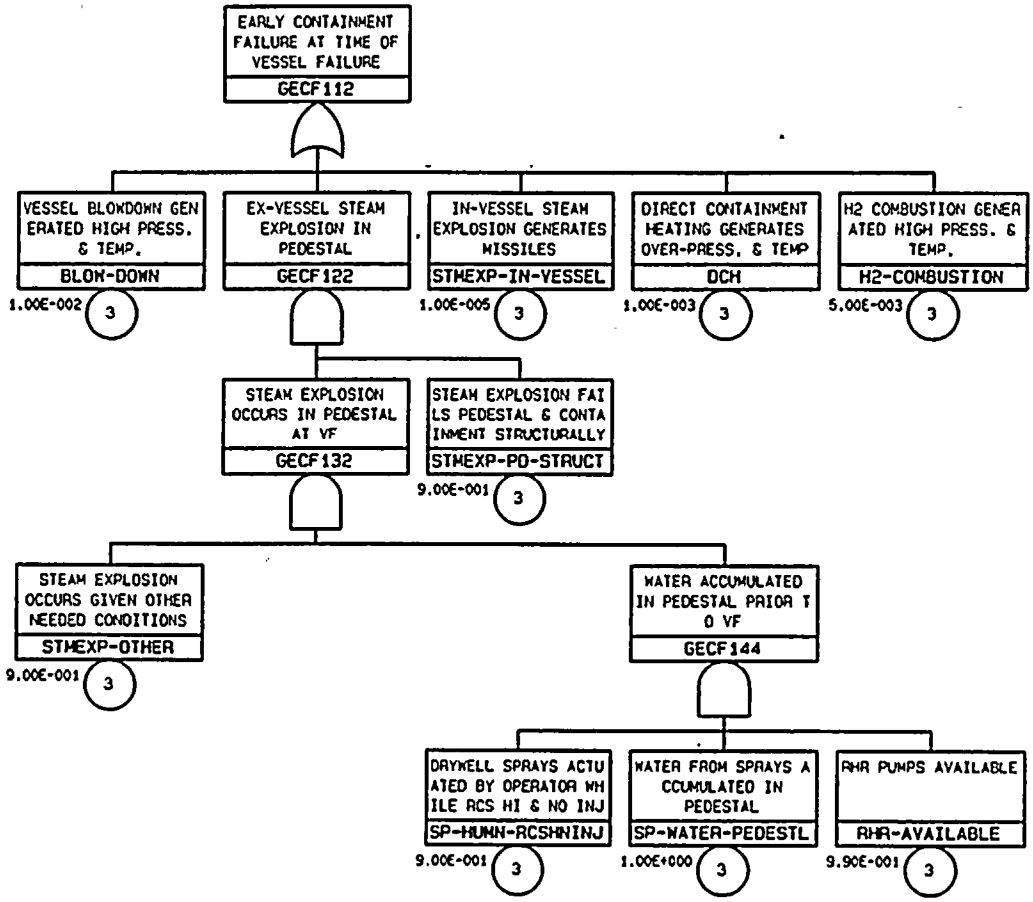
HLP 1

ANALYST: LTP | CREATION DATE: 02-23-94 | REVISION : 05-23-94



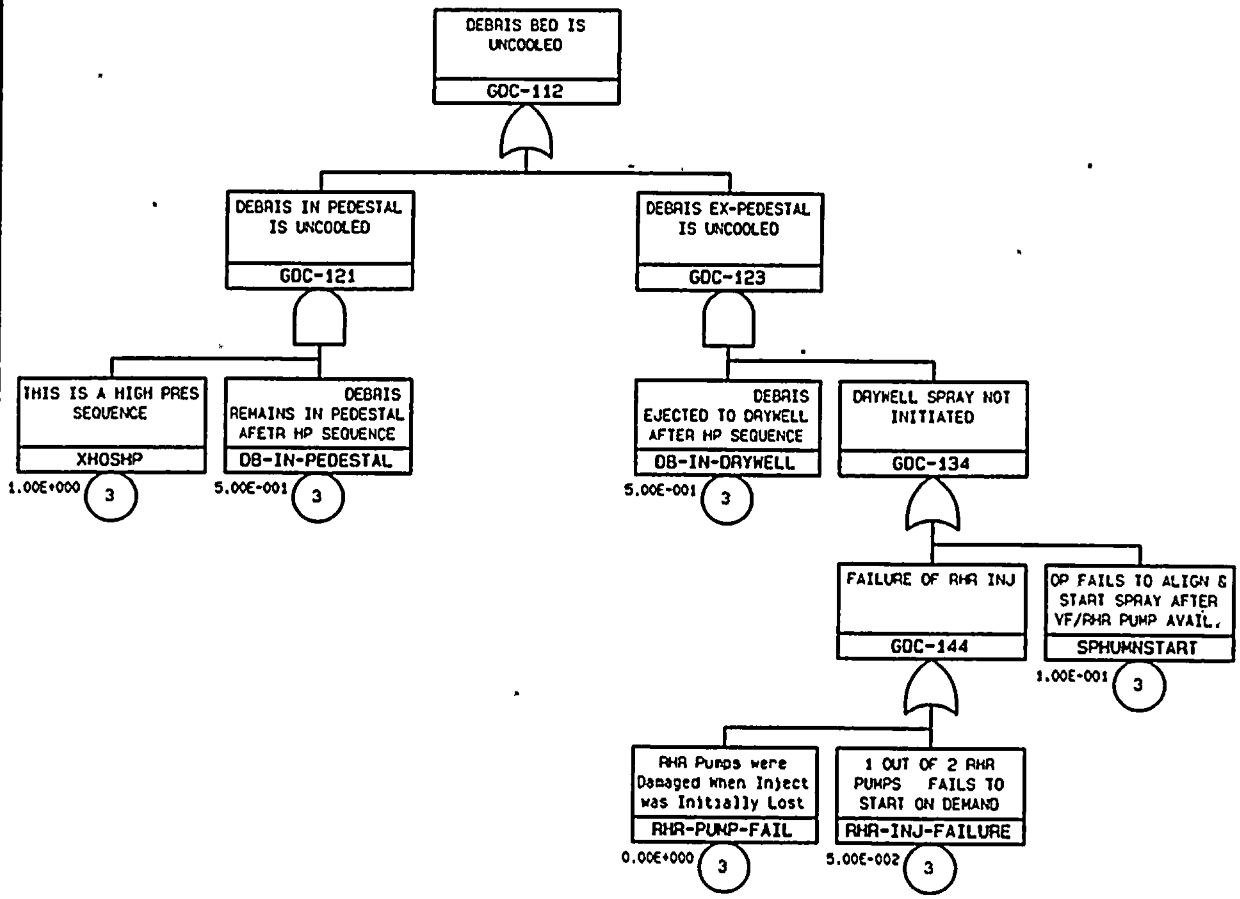
HP-LP.LGC NUPRA 2.13 KPPSS

EARLY CONTAINMENT FAILURE AT TIME OF VF. Fig.4.5.2.2-5a		ECF 1
ANALYST: LTP	CREATION DATE: 02-12-94	REVISION : 06-07-94



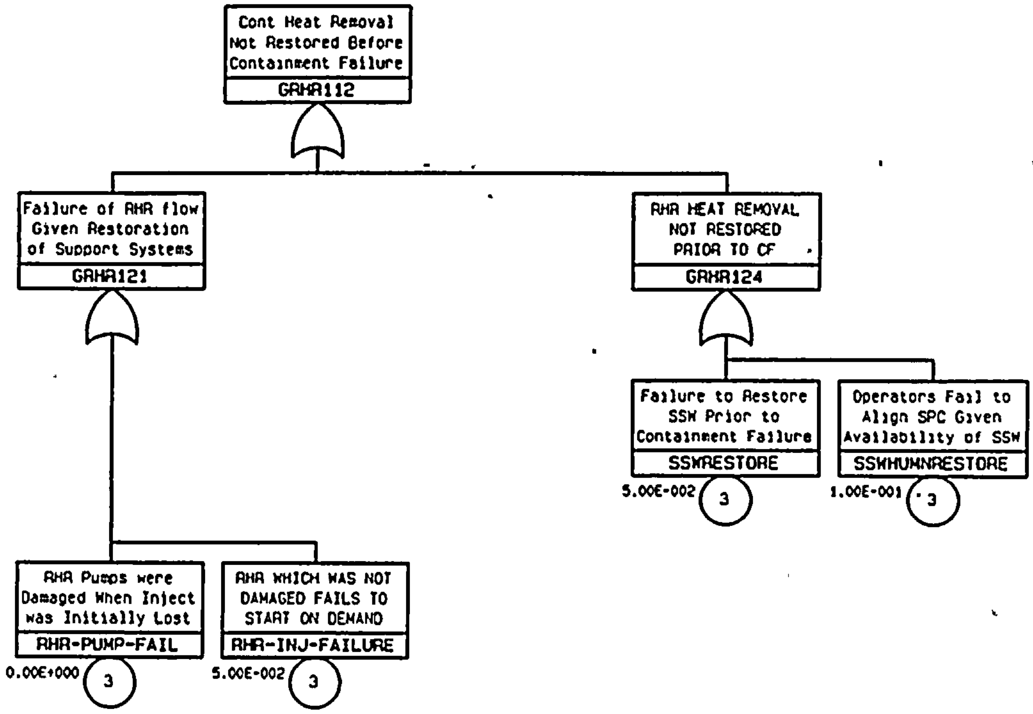
ECF LGC NUPRA 2 18 WPPSS

Debris bed not coolable Fig. 4.5.2.2-5b	DC- 1
ANALYST: ltp | CREATION DATE: 02-22-94	REVISION : 06-07-94



COOL.LGC NUPRA 2.1a WPPSS

RHR not restored before CF Fig. 4.5.2.2-6		RHR 1
ANALYST: DBK	CREATION DATE: 02-22-94	REVISION : 05-23-94

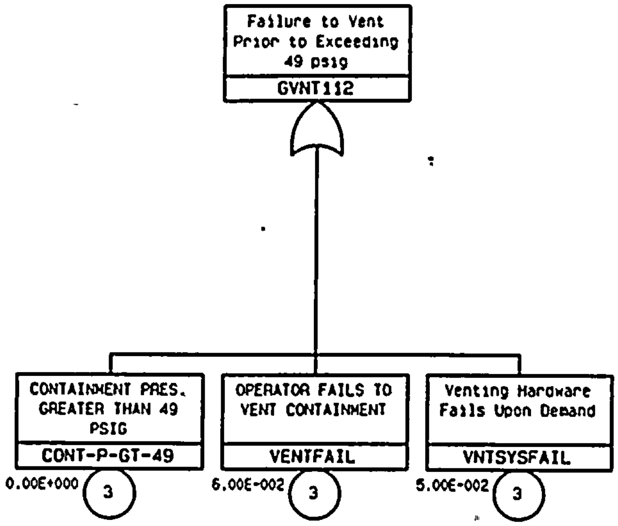


RHR.LGC NUPRA 2.1a WPPSS

No Venting before CF
Fig. 4.5.2.2-7

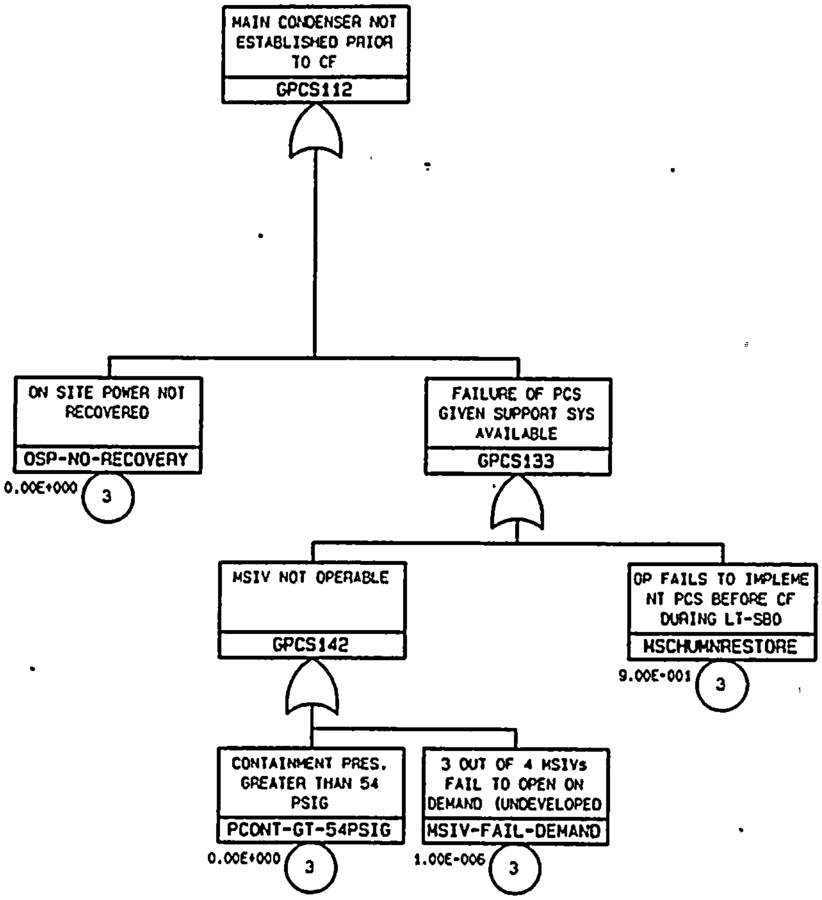
VNT 1

ANALYST: DBK | CREATION DATE: 02-14-94 | REVISION : 05-23-94

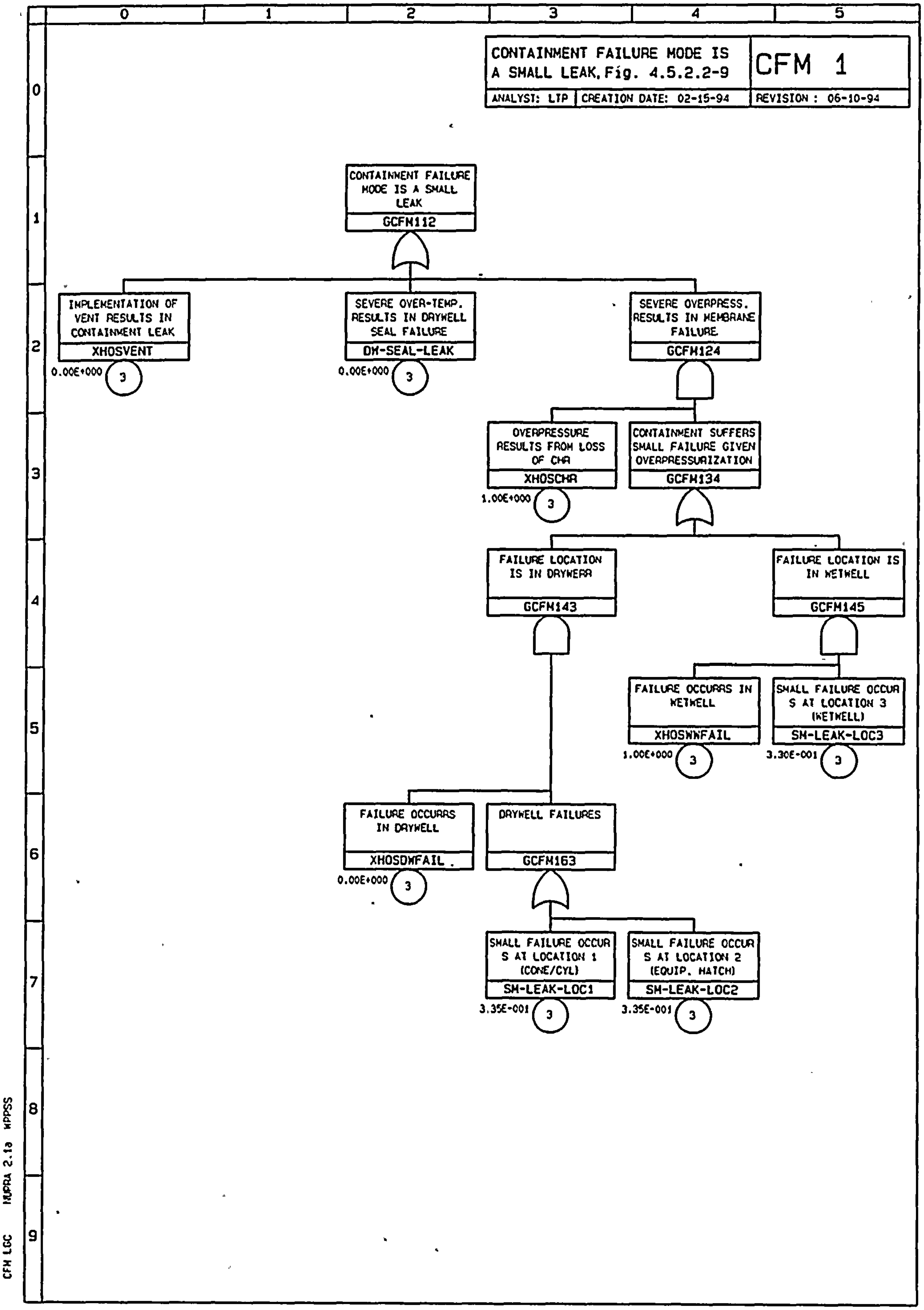


VENT.LGC NUPRA 2.1a NPPSS

Failure to recover PCS Fig. 4.5.2.2-8	PCS 1
ANALYST: ltd | CREATION DATE: 02-22-94	REVISION : 05-23-94

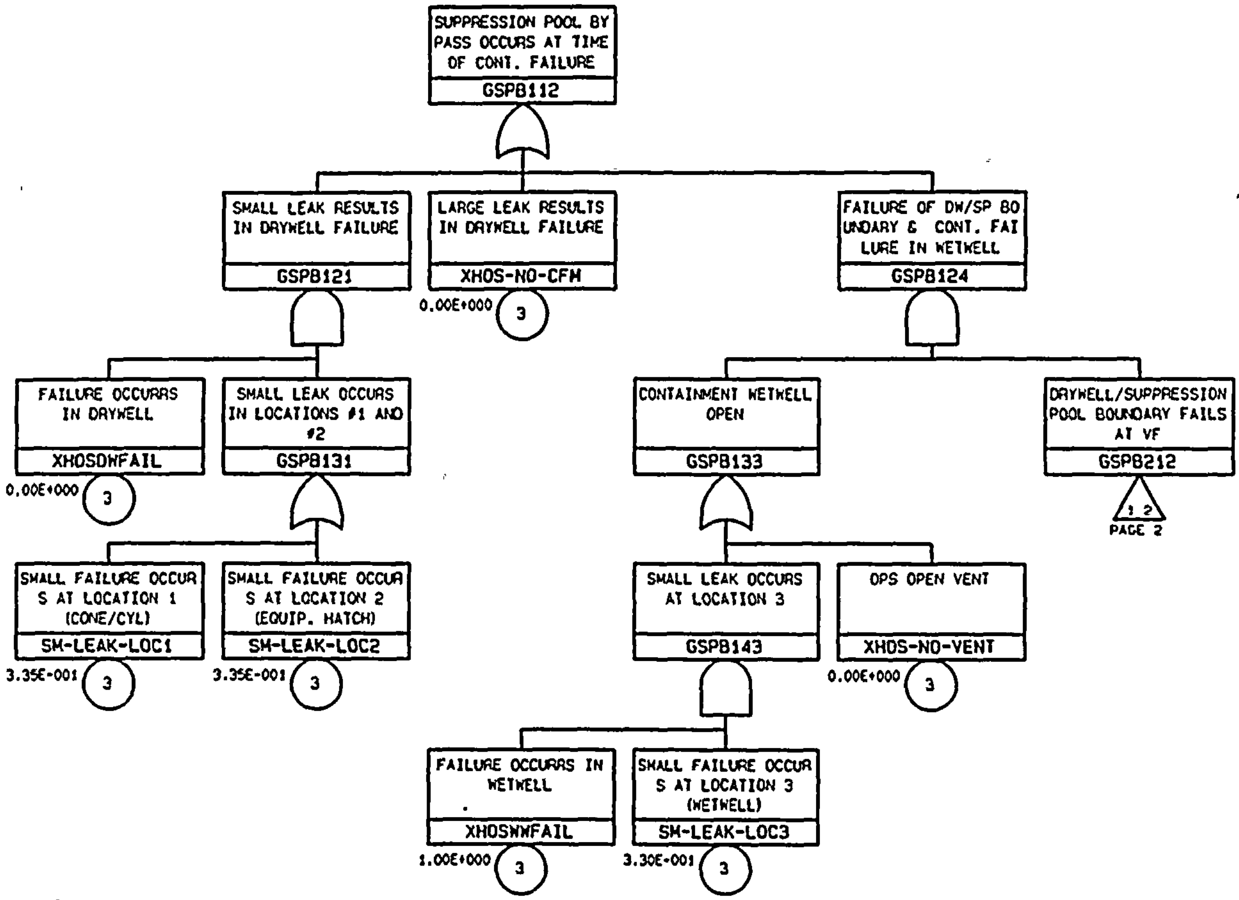


PCS LGC NUSPRA 2 18 WPPSS

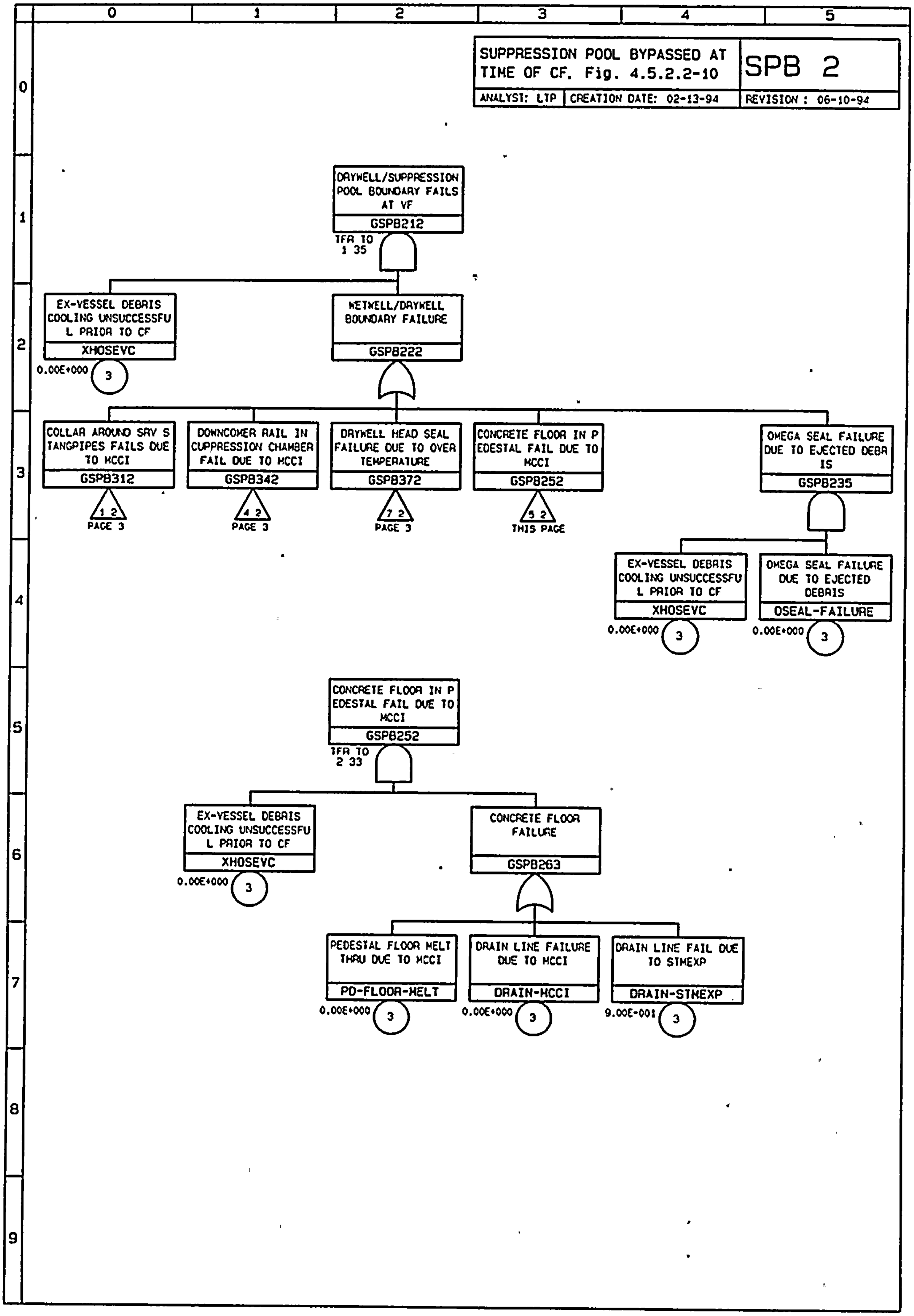


0
1
2
3
4
5
CONTAINMENT FAILURE MODE IS A SMALL LEAK, Fig. 4.5.2.2-9
CFM 1
ANALYST: LTP
CREATION DATE: 02-15-94
REVISION : 06-10-94
CONTAINMENT FAILURE MODE IS A SMALL LEAK
GCFM112
IMPLEMENTATION OF VENT RESULTS IN CONTAINMENT LEAK
XHOSVENT
0.00E+000
3
SEVERE OVER-TEMP. RESULTS IN DRYWELL SEAL FAILURE
DW-SEAL-LEAK
0.00E+000
3
SEVERE OVERPRESS. RESULTS IN MEMBRANE FAILURE
GCFM124
OVERPRESSURE RESULTS FROM LOSS OF CHR
XHOSCHR
1.00E+000
3
CONTAINMENT SUFFERS SMALL FAILURE GIVEN OVERPRESSURIZATION
GCFM134
FAILURE LOCATION IS IN DRYWERR
GCFM143
FAILURE LOCATION IS IN WETWELL
GCFM145
FAILURE OCCURRS IN WETWELL
XHOSWWFAIL
1.00E+000
3
SMALL FAILURE OCCURS AT LOCATION 3 (WETWELL)
SM-LEAK-LOC3
3.30E-001
3
FAILURE OCCURRS IN DRYWELL
XHOSDWFAIL
0.00E+000
3
DRYWELL FAILURES
GCFM163
SMALL FAILURE OCCURS AT LOCATION 1 (CONE/CYL)
SM-LEAK-LOC1
3.35E-001
3
SMALL FAILURE OCCURS AT LOCATION 2 (EQUIP. HATCH)
SM-LEAK-LOC2
3.35E-001
3
0
1
2
3
4
5
6
7
8
9
CFM LGC
NUPRA 2.1a
WPPSS

SUPPRESSION POOL BYPASSED AT TIME OF CF. Fig. 4.5.2.2-10	SPB 1
ANALYST: LTP / CREATION DATE: 02-13-94	REVISION : 06-10-94

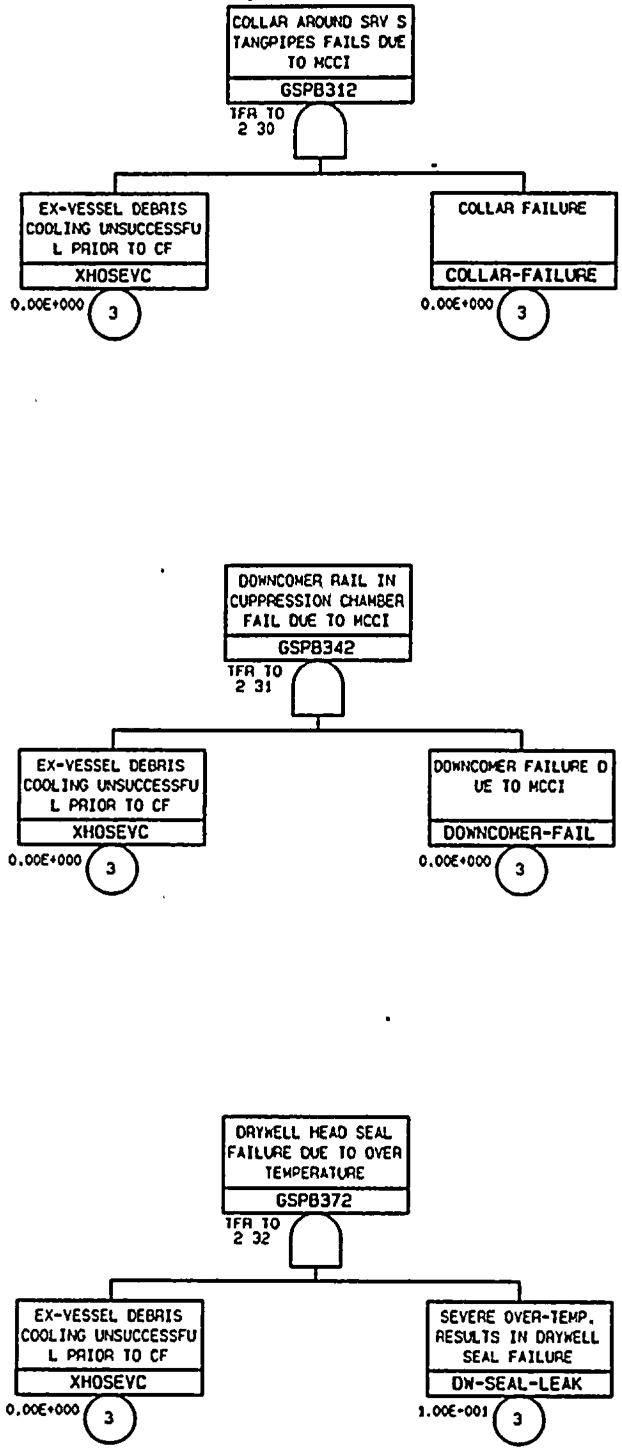


SPB.LGC NUPRA 2 1a WPPSS



SPB.LGC NUPRA 2 1a WPPSS

SUPPRESSION POOL BYPASSED AT TIME OF CF. Fig. 4.5.2.2-10		SPB 3
ANALYST: LTP	CREATION DATE: 02-13-94	REVISION : 06-10-94



SPB LGC NUPRA 2.1a KPPSS

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4.6 Radionuclide Release Characterization

4.6.1 Source Term Binning Logic

The grouped sequence cutsets for each initiator class represent the Level 1 plant damage states which are to be further analyzed with the containment event trees. The endpoints from the CETs represent the outcomes which result from complete severe accident sequences from the initiating event to the release of radionuclides to the environment. It is for each of these end points that analytical assessment of the fission product characteristics is needed.

To associate a unique atmospheric source term with each CET end state would result in a product in which it would be difficult to distinguish the effects from individual sequences. In fact, the large number of CET sequences and the similarity in their characteristics makes it unnecessary to develop a source term for each endpoint in the CET. What is required is a way of characterizing each outcome so that they can be grouped on the basis of similarity and their frequencies cumulated. This will result in a set of predefined release events, each of which has an associated occurrence frequency. To achieve this first requires the development of a source term logic grouping process, to group CET sequences into release categories with similar source term characteristics.

4.6.1.1 Release Category Grouping Parameters

The first step is one of definition of the set of important sequence parameters which can form the basis for CET sequence grouping into source term categories. The characteristics of each sequence are contained within the sequence description and include information which can be used to determine the state or conditions represented by the containment damage states and their associated source terms. Of particular interest are the following candidate sequence characteristics:

- Whether or not the sequence involves **containment bypass** - in this case a direct release to the environment will result, no scrubbing and no fission product retention in containment.

- Whether or not the **containment is isolated** - if isolation fails when conditions initially demand it, there will be a continuous release throughout the accident. In addition, there is likely to be a transient "puff" release at the time of vessel failure if the melt is released very energetically (HPME). An open penetration has the potential to increase the magnitude of the overall release because fission products which may have been naturally attenuated within containment are released before these mechanisms have an opportunity to act.

- Whether the sequence results in **termination with an intact vessel**. In this case the fission products will always be scrubbed before they are released to the containment or the environment i.e., "Debris Cooled In-Vessel."

- Whether or not the **release is unscrubbed** because there is a failure of the drywell pressure boundary which results in a direct release to the reactor building or to the suppression chamber and thence unscrubbed to the reactor building. Generally if the debris is cooled successfully, fission product scrubbing is assumed to exist because the drywell will remain intact. The single exception occurs in the case of a low pressure melt ejected to the pedestal. Flooding will be unsuccessful in cooling the debris so the pedestal floor will fail and initiate suppression pool bypass. However, since the molten debris is expected to fall into the suppression pool it will still be scrubbed.

- Knowledge of the **time of containment failure** is important because the longer the overall accident sequence, the more time there is available for natural in-containment processes such as deposition and plate-out to contribute to a reduction of the airborne fission products which are potentially available for release. The longer the time during which containment remains intact, the greater the time available to initiate mitigative actions, such as evacuation, sheltering and the use of blocking agents, to protect the general public.

- Knowing the expected **containment failure mode** is important for two reasons. The size of the failure (large, small) has an effect on the release-rate for radionuclides and its location (drywell, wetwell, vent) has an effect on the character of the release because some failure locations result in a direct, unscrubbed release of fission products to the reactor building or the environment.

A more detailed rationale for the selection of these parameters to represent the basis for source term grouping is provided below.

CONTAINMENT BYPASS

If the accident causes an opening of a path from the reactor coolant system outside of the containment boundary then the containment natural and engineered safeguards features are ineffective in reducing fission product releases. These scenario types, often referred to as interfacing system LOCAs, are characterized by a failure at the boundary between the high pressure primary system and a low pressure system such as the LPCS system.

The containment bypass core melt sequence is explicitly treated in the release category logic because it represents a direct path between the fission products released from the damaged core and the environment with little or no opportunity for retention or mitigation. Even though the Level 1 study contains no containment bypass sequences above the cutoff frequency, for the sake of completeness it is considered in the grouping process.

CONTAINMENT ISOLATION

Whether or not the containment is isolated at the time that core damage begins is important to the assessment of the release characteristics because if a penetration remains open there may be an environmental release of fission products from the core, drywell or containment atmosphere throughout the accident scenario. This release occurs early, before containment failure, and may result in an enhanced source term. This is because fission products which may otherwise have been retained within containment are released before natural deposition and decay processes have an opportunity to work.

During the WNP-2 analysis, however, failure of a penetration was assumed to be incapable of removing enough energy to prevent containment failure so all sequences involving "failure to isolate" are described with a combined source term, that from the penetration and that from the associated containment failure. The assumptions upon which this approach was based are conservative, because it is expected that the predicted source term for isolated and non-isolated scenarios will be somewhat overestimated. This assumption was made in order to simplify the overall computational process.

The other issue associated with isolation, is whether or not an unisolated containment will result in a "puff release" if there is a high pressure melt ejection. In this event, the energetic release from the vessel is hypothesized to result in a transient pressure and flow condition which sweeps fission products from the containment, through the failed penetration, into the reactor building and into the environment.

SEQUENCE ARRESTED IN-VESSEL

This characteristic is important for non-bypass cases because if injection can be restored before the core support plate fails, the core will be in a coolable geometry and the vessel will be saved. This means that the fission products will be largely contained within the primary system pressure boundary and those which are released to the containment will be scrubbed by the water in the primary system. Maintaining the debris in-vessel also negates concern for all of the other problems associated with ex-vessel release of the melt, i.e whether it is coolable and whether interaction between the melt and other containment components can result in failure of the drywell and whether the suppression pool is bypassed.

DRYWELL FAILURE

This attribute is important to the source term for sequences in which the vessel fails because failure of the drywell may result in a direct path to the reactor building or to the suppression chamber airspace which bypasses the scrubbing action normally provided by the suppression pool. The only possibility for fission product scrubbing is if the debris in the drywell is flooded.

Drywell failure is possible if:

- the drywell/reactor building pressure boundary fails from overpressure.

- drywell floor fails as a result of molten core-concrete interaction (MCCI).

- drywell integrity is lost as a result of failure of the penetrations or seals which serve as part of the pressure boundary between drywell and suppression chamber.

The accident scenarios which initiate each of these drywell failure modes fall into three basic categories:

1. Containment Failure Prior To Core Melt

   In this scenario class, core melt is initiated by a overpressure failure of containment which results in a large release of energy to the reactor building which in turn results in loss of all ECCS capabilities. Because there is no injection, after vessel failure the debris will be uncooled so the drywell will fail.

2. Low Pressure Melt Ejection (LPME)

   Following an LPME, because of the pedestal cavity geometry, the melt will accumulate to a depth which assures that it will be in an uncoolable geometry, even if injection flow to the vessel is successful. This means that MCCI will continue throughout the accident and will eventually cause failure of the drywell floor, thus providing a direct path to the suppression chamber. This means that for LPMEs:

   - if the debris is flooded, there will be fission product scrubbing throughout.

At first with the water covering the debris, and later by the suppression pool because after the floor fails the debris will fall into the pool. This is expected to happen slowly enough to obviate any concerns with steam explosions in the suppression pool.

- if the debris is not flooded, it is assumed that the fission products will be unscrubbed throughout the scenario, even though it is likely that eventually the floor will fail and the debris will fall into the suppression pool where scrubbing will occur.

3. High Pressure Melt Ejection (HPME)

In the case of an HPME, the issue is different because the debris will be splattered over the inside of the pedestal or ejected through the opening into the drywell. If the melt is uncooled, premature drywell failure from overtemperature may occur, or the interaction of the melt with metal in the omega seal, downcomers or SRV standpipes may cause local failures which result in loss of drywell integrity without structural failure of the floor. Actuation of the drywell sprays will result in debris cooling and it is assumed that drywell sprays will serve to scrub fission products from containment.

POOL BYPASS

If the drywell is intact (floor, shell and penetrations), the remaining issue of concern is whether all of the fission products released to the environment actually pass through the suppression pool so that scrubbing can take place. There are several ways in which suppression pool bypass can occur:

- Use of the PCS to remove containment energy - the main steam system bypasses the pool by providing an open path directly from the drywell, through the vessel, to the main condenser.

  During this analysis, a release to the main condenser was considered equivalent to a release to the environment.

- Loss of water in the suppression pool. This was accepted as a possibility during gross containment failures, although since these "large" failures were expected to also cause drywell failure, the information turned out to be redundant.

This issue is of unique concern for all sequences which do not explicitly involve containment bypass, drywell failure or core melt arrested in vessel.

TIME OF CONTAINMENT FAILURE (CF)

This descriptive source term attribute is important because it affects the time available for fission product release mitigation by natural removal processes, scrubbing, and retention of fission products. It is an issue for all scenarios which do not involve containment bypass, or core melt arrested in vessel.

The times selected to be representative of the overall accident time scale are very early, early and late. These are defined for this analysis as:

- Very early - prior to core damage,
- Early - at or near time of vessel failure,
- Late - significantly after vessel failure,
- Immediate - specifically for large LOCA sequence (A-S08) in which failure of the omega seal leads to loss of the vapor suppression function and subsequent containment failure.

The possibility of no containment failure exists and is assigned its own unique source term category.

The time of containment failure is directly related to the mechanical status of injection and availability of power.

MODE OF CONTAINMENT FAILURE (CFM)

This attribute is important because it governs the rate of fission product release to the atmosphere. It also affects the magnitude of release by controlling the time available for fission product attenuation in containment. "Containment failure mode is large" represents a catastrophic rupture, which is defined as the loss of a substantial portion of the containment boundary with possible disruption of the piping systems that penetrate or are attached to the containment wall. Since the flow rate of gas and aerosol out of the containment is high, large amounts of fission products could be released to the environment. "Containment failure mode is small" represents a leak, which is defined as a containment breach that would arrest a gradual pressure buildup and would depressurize the containment within 2 hours.

Containment failure mode is only considered to be an important discrimination for those sequences which have very early or early containment failure, because late failure of containment would have allowed time for effective fission product attenuation.

The attributes considered significant are vent, leak and rupture of containment. These are evaluated using the branch attributes shown in the CETs. Successful operation of the PCS to remove energy from containment is considered to have the same effect as the vent, except that use of the PCS will always result in suppression pool bypass since the path is directly from the drywell.

The information described above was used to coalesce the damage states represented by individual CET sequence end points into a relatively small group of source terms, which could then be evaluated for actual fission product releases with MAAP simulations. The source term groups used for the WNP-2 analysis are identified below.

4.6.1.2 Source Term Groups

Seventeen basic source term groups were identified into which CET damage states will fall, with a limited number of additional modified states, in which the expected release is augmented with a release from a failed penetration. These latter modified states were only explicitly developed for sequences which survived the truncation value of 1E-10. These source term groups used in the WNP-2 analysis are identified below:

For cases with isolated containment:

STG-1 - Fission products scrubbed * Containment Intact
STG-2 - Fission products not scrubbed * Containment Intact
STG-3 - Fission products scrubbed * small CFM * CF-late
STG-4 - Fission products scrubbed * large CFM * CF-Late
STG-5 - Fission products not scrubbed * small CFM * CF-Late
STG-6 - Fission products not scrubbed * large CFM * CF-late
STG-7 - Fission products scrubbed * small CFM * CF-Early
STG-8 - Fission products scrubbed * large CFM * CF-Early
STG-9 - Fission products not scrubbed * small CFM * CF-Early
STG-10 - Fission products not scrubbed * large CFM * CF-Early
STG-11 - Fission products scrubbed * small CFM * CF-Very Early
STG-12 - Fission products scrubbed * large CFM * CF-Very Early
STG-13 - Fission products not scrubbed * small CFM * CF-Very Early
STG-14 - Fission products not scrubbed * large CFM * CF-Very Early
STG-15 - Direct containment bypass * core in-vessel
STG-16 - Direct containment bypass * core ex-vessel
STG-17 - Fission products not scrubbed * large CFM * CF-Immediate

For cases with "CF prior to CM + HPME": a puff release is followed by the above generic release

STG-13P - Fission products not scrubbed * small CFM * CF-Very Early
STG-14P - Fission products not scrubbed * large CFM * CF-Very Early

For cases with unisolated containment:

STG-4U - Fission products scrubbed * large CFM (occurs late in time)
STG-6U - Fission products not scrubbed * large CFM (occurs late in time)
STG-10U - Fission products not scrubbed * large CFM (occurs around the time of VF)
STG-11U - Fission products scrubbed * small CFM * CF-Very Early
STG-12U - Fission products scrubbed * large CFM * CF-Very Early
STG-13U - Fission products not scrubbed * small CFM * CF-Very Early
STG-14U - Fission products not scrubbed * large CFM * CF-Very Early

4.6.1.3 Assigning Source Term Groups

To facilitate the use of the information carried within each of the sequences to categorize it into one of the source term groups identified above, a series of logical relationships were developed between information and the grouping criteria. The "rules" for categorization are provided below and the assigned source term groups are shown in the "PDS #" in Figures 4.7-1 through 4.7-32.

1. **Fission Product Scrubbing**

1A The debris is assumed to be flooded if the sequence involves:

LPME * Successful recovery of Injection (covers the debris in the pedestal) or,
HPME * active sprays (covers debris in- and ex-cavity with water) or,
Sequence is arrested in vessel (core injection floods debris)

1B The suppression pool is assumed to be bypassed if:

LPME * uncoolable (this is always the case - fails DW floor) or,
PCS is used for Containment heat removal or,
Large containment failure or,
Small containment failure in drywell or,
HPME and no sprays (overtemp/debris interaction causes failure of DW)

1C Fission Product scrubbing will occur if:

The debris is flooded or,
The suppression pool is NOT bypassed

2. **Containment Failure Mode**

"Intact" containment is assumed if sequence indicates:

Successful RHR * no Early or Very Early containment failure

Containment failure mode is assumed to be "small" if failure is initiated by chronic overpressure and results in:

Small structural tear of containment membrane or,
Vent is implemented or,
PCS is used to remove energy from containment

Containment failure mode is assumed to be "Large" if overpressure initiates "large" (catastrophic) structural failure which is defined by the sequence.

3. **Time of Containment Failure**

Containment is assumed to fail "Very Early" if containment failure initiates core melt.

Containment is assumed to fail "Early" if the sequence involves:

Steam explosion or,
ATWS

Containment is assumed to fail "Late" when failure is initiated by overpressure following loss of long term containment heat removal.

4. **Puff Release**

Puff Release is assumed to occur if:

Containment is NOT isolated * HPME (VF @ high primary system pressure).

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4.7 Containment Event Tree Quantification

4.7.1 Methodology

Each of the initiator specific containment event trees was constructed to allow the assignment of conditional branch point probabilities which could then be arithmetically combined with the initiating state frequency in the manner dictated by the logic of each individual sequence in the CET. The actual probabilities were synthesized by either node specific fault trees developed to identify each of the individual functional failures, or split fractions (1 or 0) assigned to each branch node based on the Level 1 system status. The CET end point frequencies were then computed with the NUPRA code in the conventional manner.

To simplify the management of the complex dependencies implicitly contained in the event trees, the effects of dependent systems were considered during CET development so that the fault trees only contain local component/human failures (fail to actuate, fail to start or fail to continue running). Generally the state of the support systems could be derived from the successes and failures of systems, earlier in the sequence. The support systems of particular concern to the analysis included:

- On- and off-site AC power (motive power for ECCS, containment air, containment venting to SGT via purge system)
- DC power (needed to open the MSIVs and to use the depressurization valves)
- Standby service water (needed for a heat sink during the operation of RHR)

The conditions associated with these systems were determined from the preceding sequence successes and failures and the corresponding system operability state determined. Hardware was also considered unavailable (and generally unrecoverable) if the sequence cutsets indicate that it was in a failed state prior to core damage. If the system was inferred to be unavailable it was not credited during fault tree solution; if it was inferred to be available, the probability of failure was based on the applicability of "failure to actuate on demand," "failure to start given actuation" or "failure to run for the required mission time given that it started successfully."

Because the severe environmental conditions to which the hardware may be exposed during a core melt event, the following values were used throughout the Level 2 analysis:

- single train system - probability of failure to start, 0.1
- two train system - probability of failure to start, 0.05
- two train system - probability of common cause failure to run is 100 times greater than common cause failure to run during normal operation

These values are intended to accommodate the operation of equipment at the limits of its design envelope:

- minimum NPSH for pumps
- higher than normal temperatures in the operating fluids
- higher than normal ambient operating temperatures
- the possibility of debris being ingested by running equipment

A similar situation exists for the operating staff. The stress and confusion levels will be extremely high during and after a core melt event, so even if there appears to be ample time to initiate a particular action, the assumed non-response probability are expected to be higher than if the actions were performed under normal operating conditions:

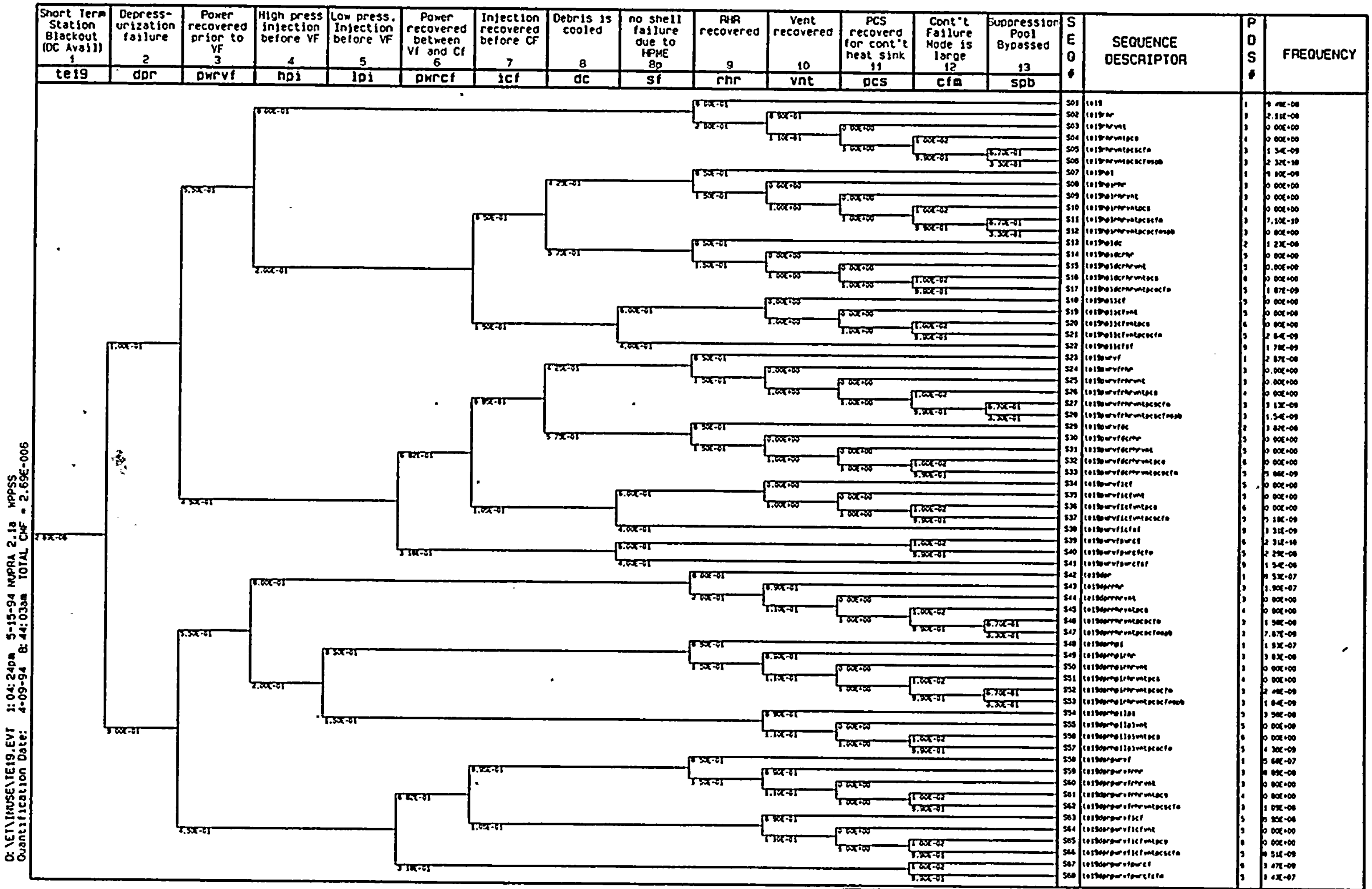
- non-response probability for recovery of needed systems, 0.1.

Detailed descriptions of the assignment of damage states and quantification are summarized in the second level of information retained at the Supply System.

4.7.2 Quantification Results

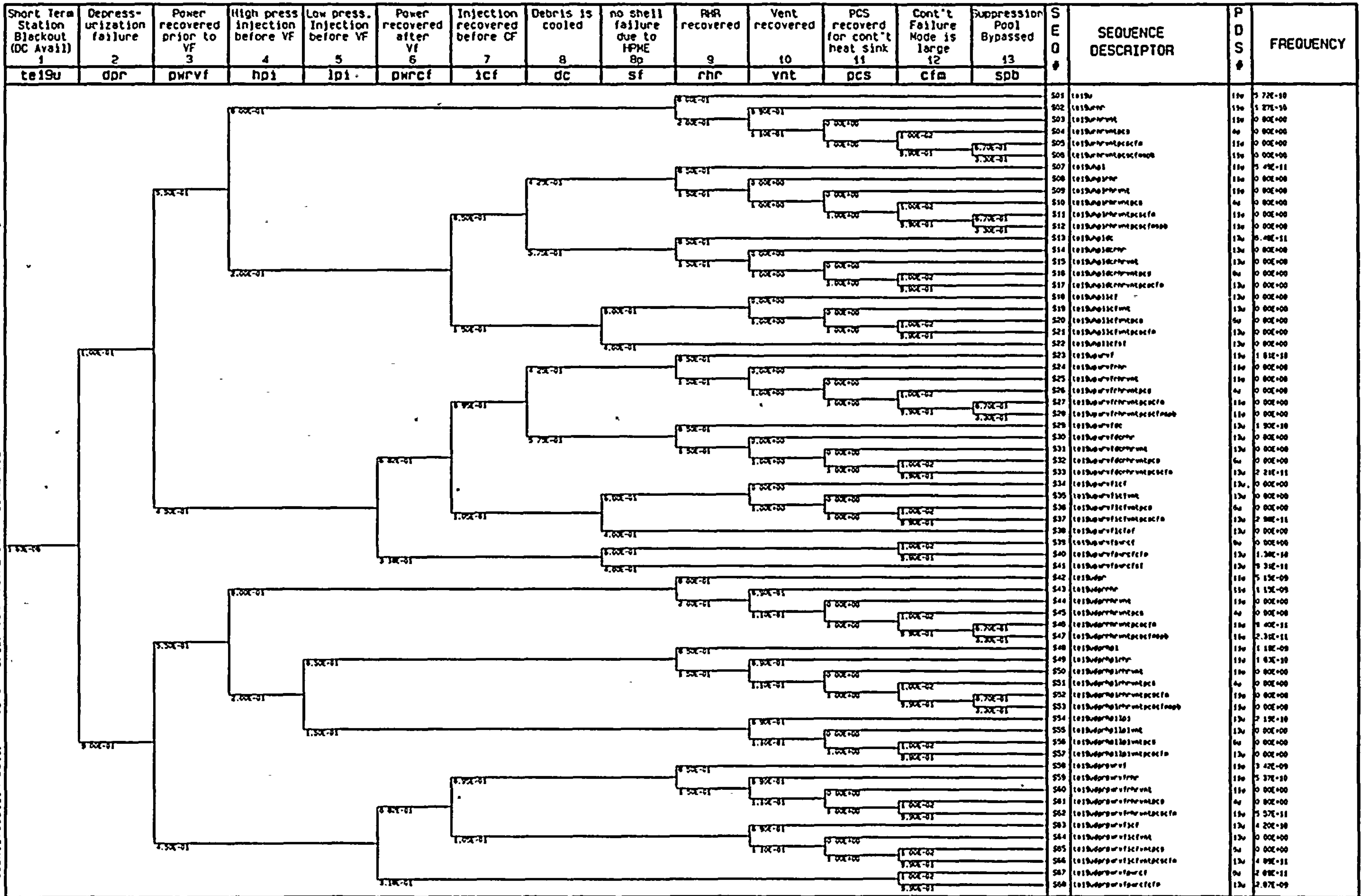
Nineteen plant damage states based on Level 1 results were analyzed further by means of the containment event trees to determine the probability of containment damage and radionuclide release. Except sequences in which containment failed prior to core damage, each containment event tree was quantified twice - once with the initiating frequency fraction for "isolation successful," once with "isolation failure."

The NUPRA code was used to quantify the CETs. The cutoff limit used for the final merge steps is 1.E-10 for "isolation successful" and 1.E-11 for "isolation failure." The frequencies for some initiators (the cases with failure to isolate) are lower than 1.E-10, therefore, were not analyzed. A total of 32 CETs were quantified and the results are presented in Figs. 4.7-1 through 32.



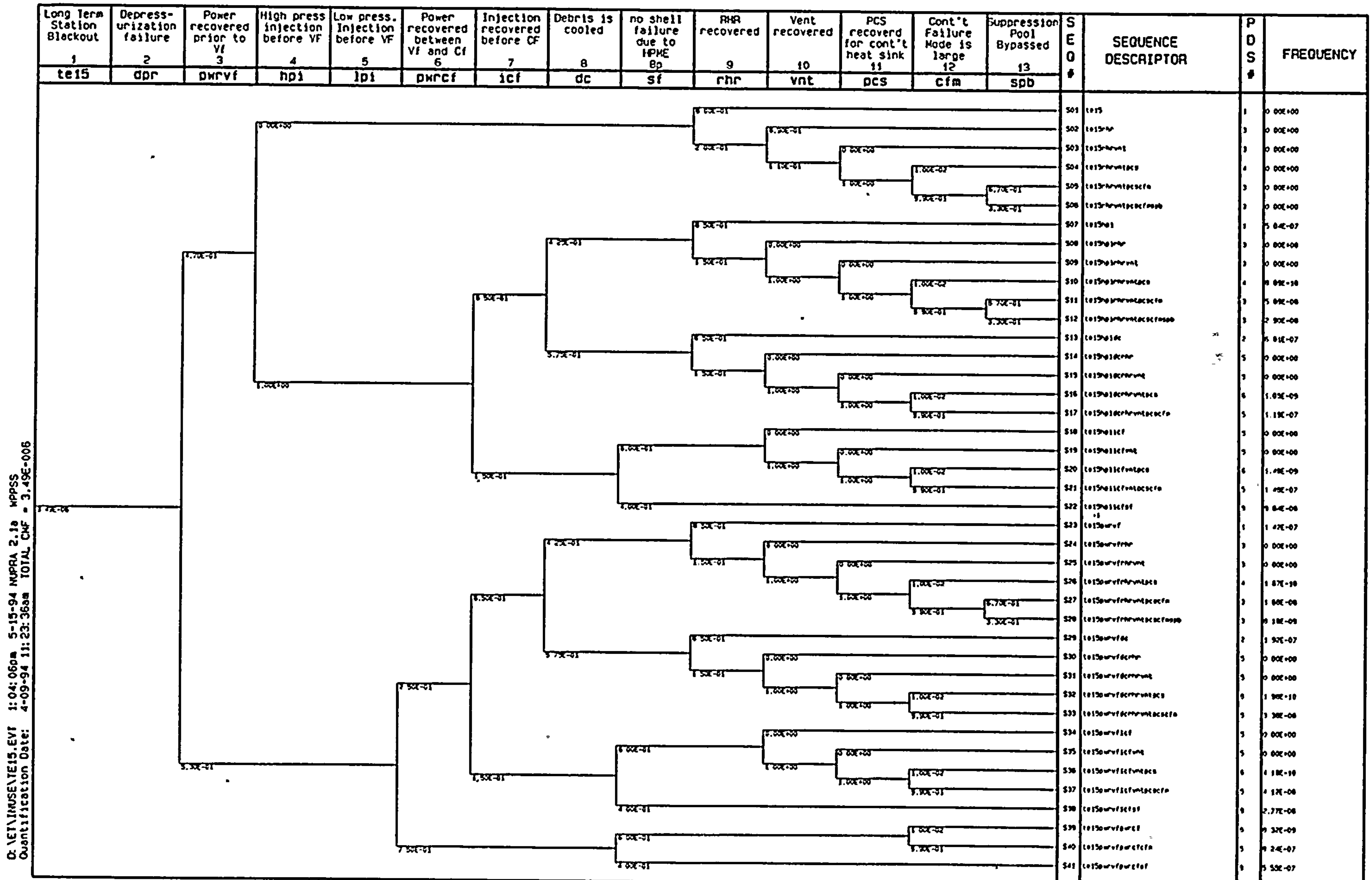
Short term Station Blackout - WNP-2
Containment Event Tree - Isolated
Revision 1, April 8, 1994
(TE-S19)
Figure 4.7-1

O:\ET\INUSE\TE19U.EVT 1:11:26pm 5-15-94 NUPRA 2.1a WPPSS
Quantification Date: 4-09-94 9:13:40am TOTAL CMF = 1.60E-008



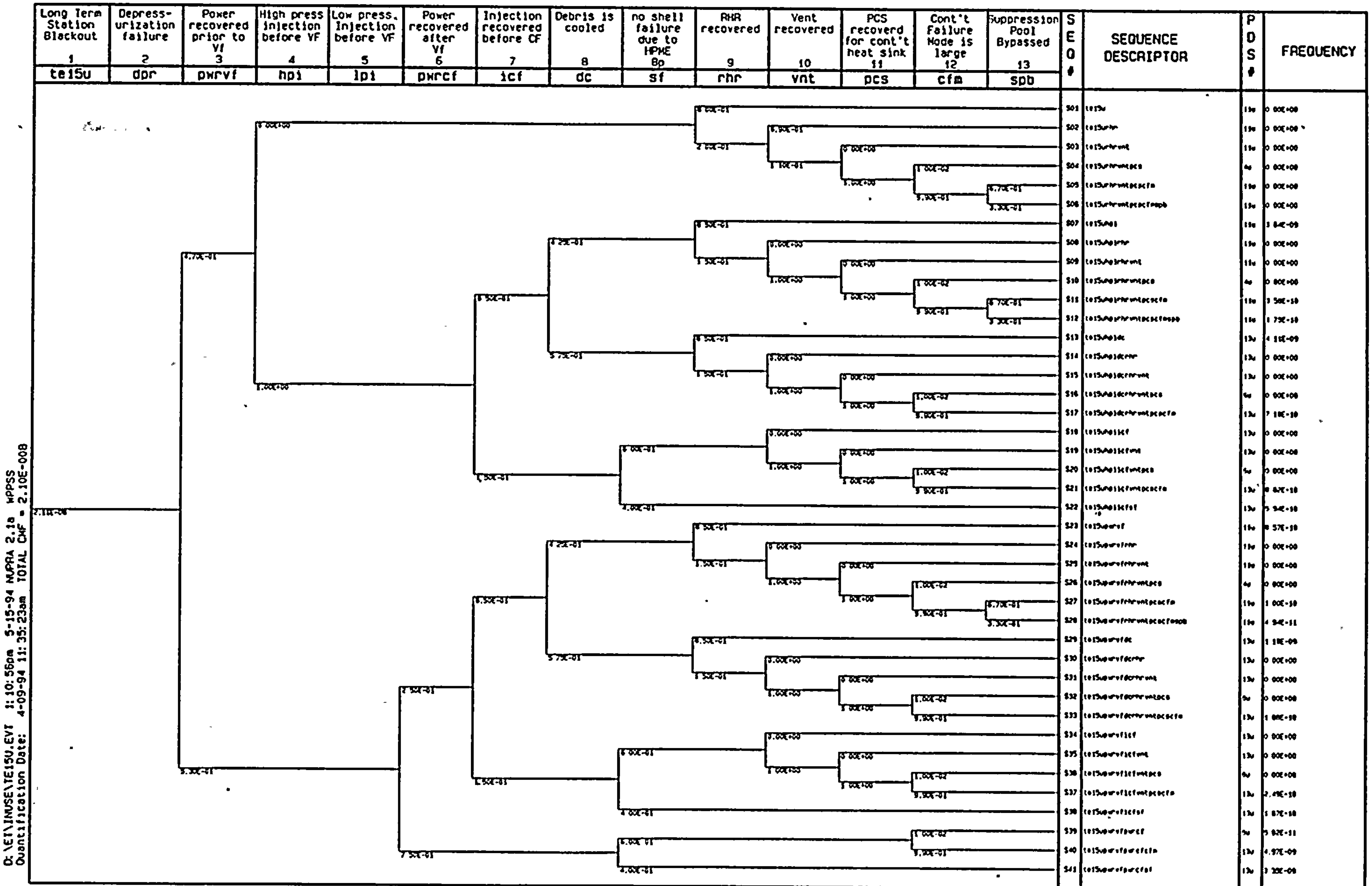
Short term Station Blackout - WNP-2
Containment Event Tree - Unisolated

Revision 1, April 8, 1994
(TE-S19)
Figure 4.7-2



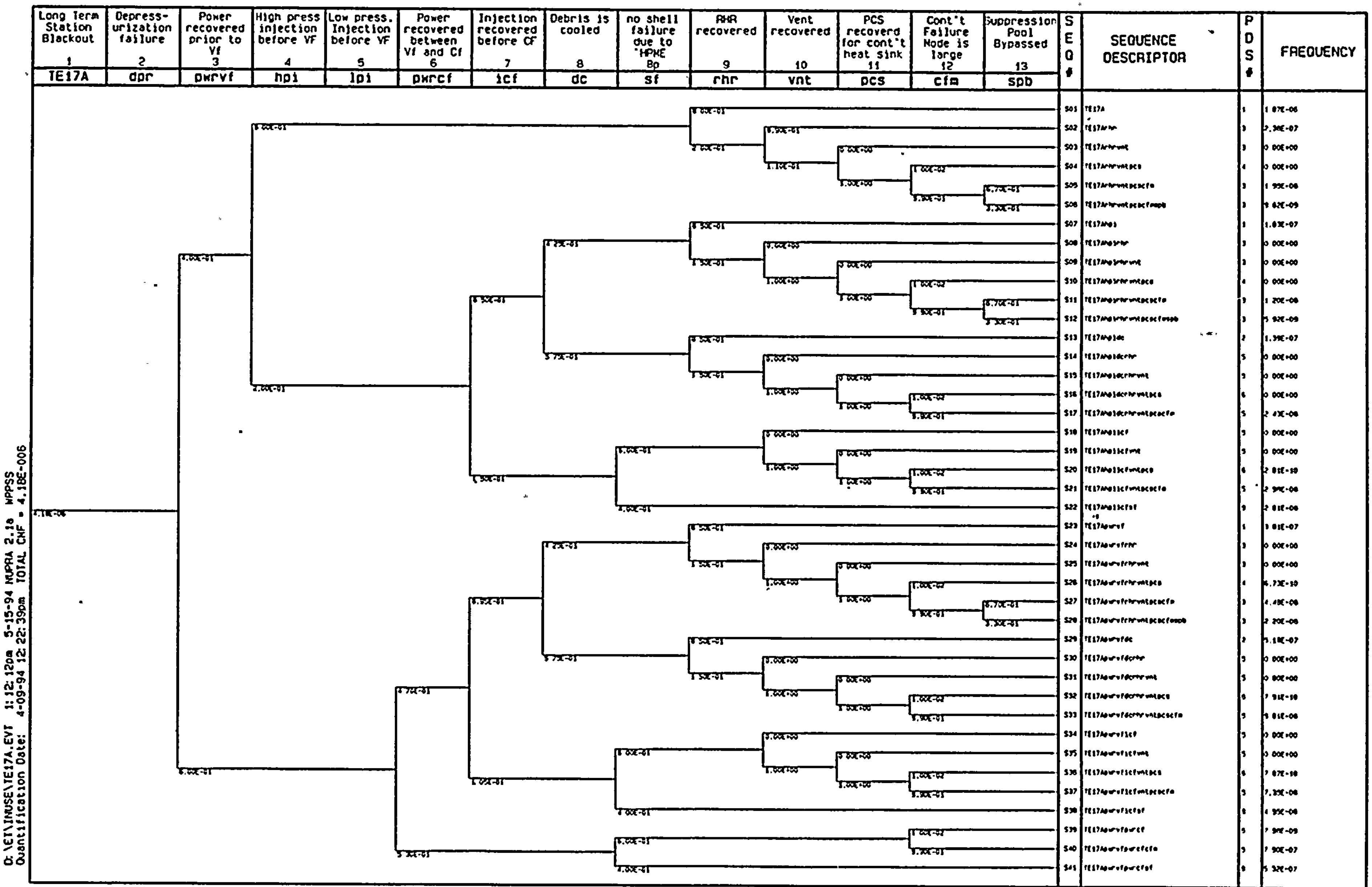
Long term Station Blackout - WNP-2
Containment Event Tree - Isolated

Revision 1, April 8, 1994
(TE-S15)
Figure 4.7-3

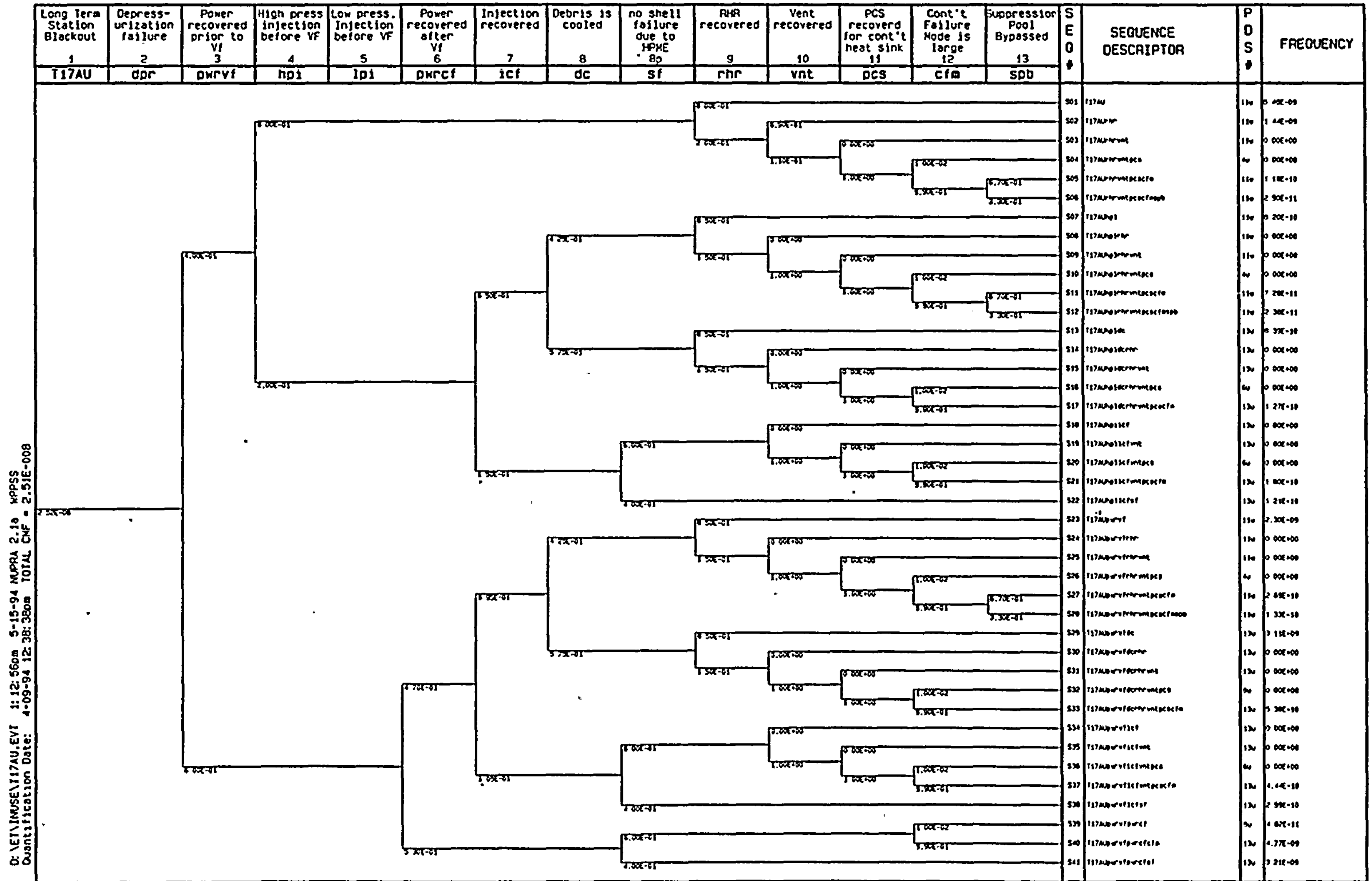


Long term Station Blackout - WNP-2
Containment Event Tree - Unisolated

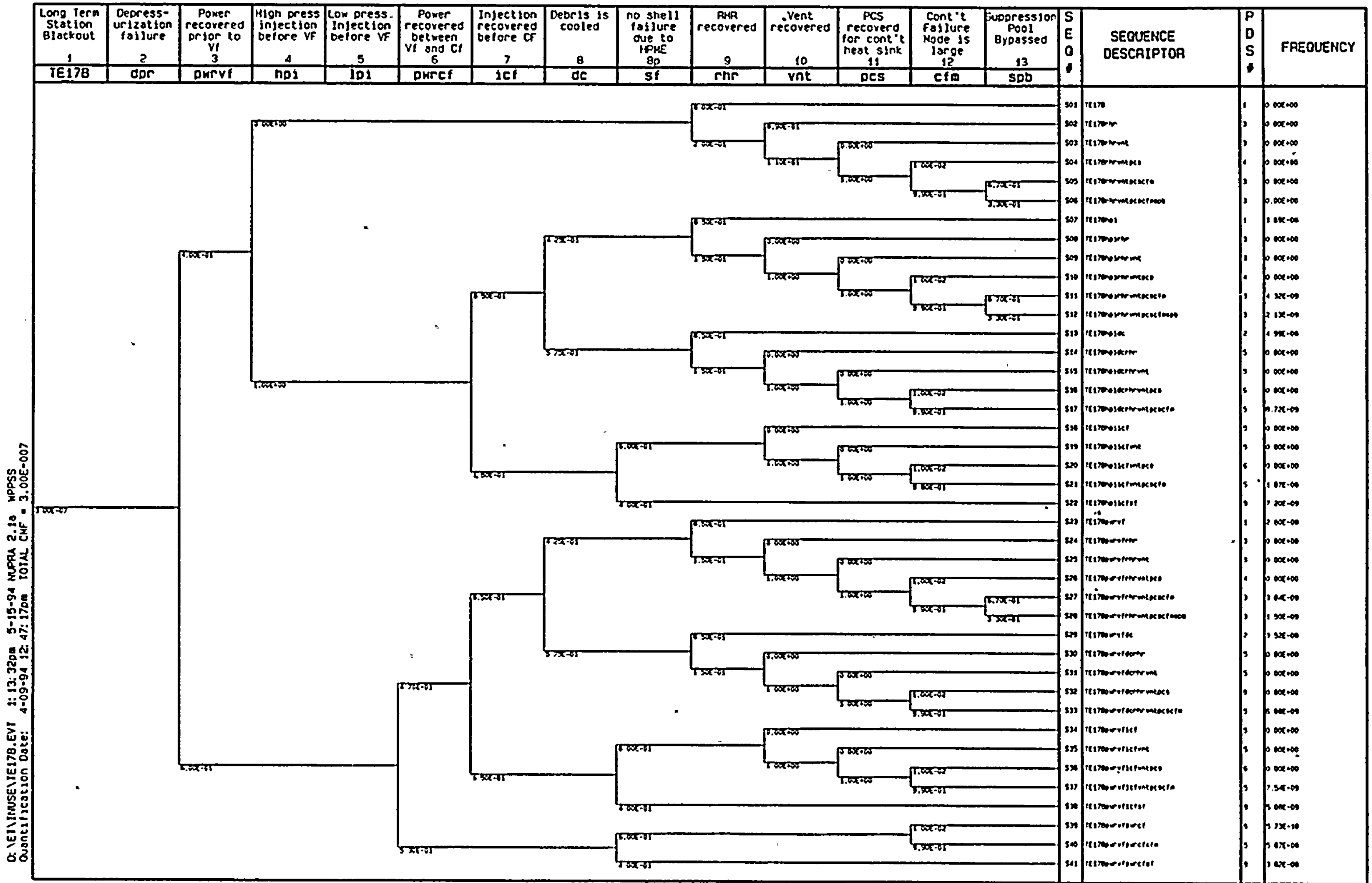
Revision 1, April 8, 1994
(TE-S15)
Figure 4.7-4



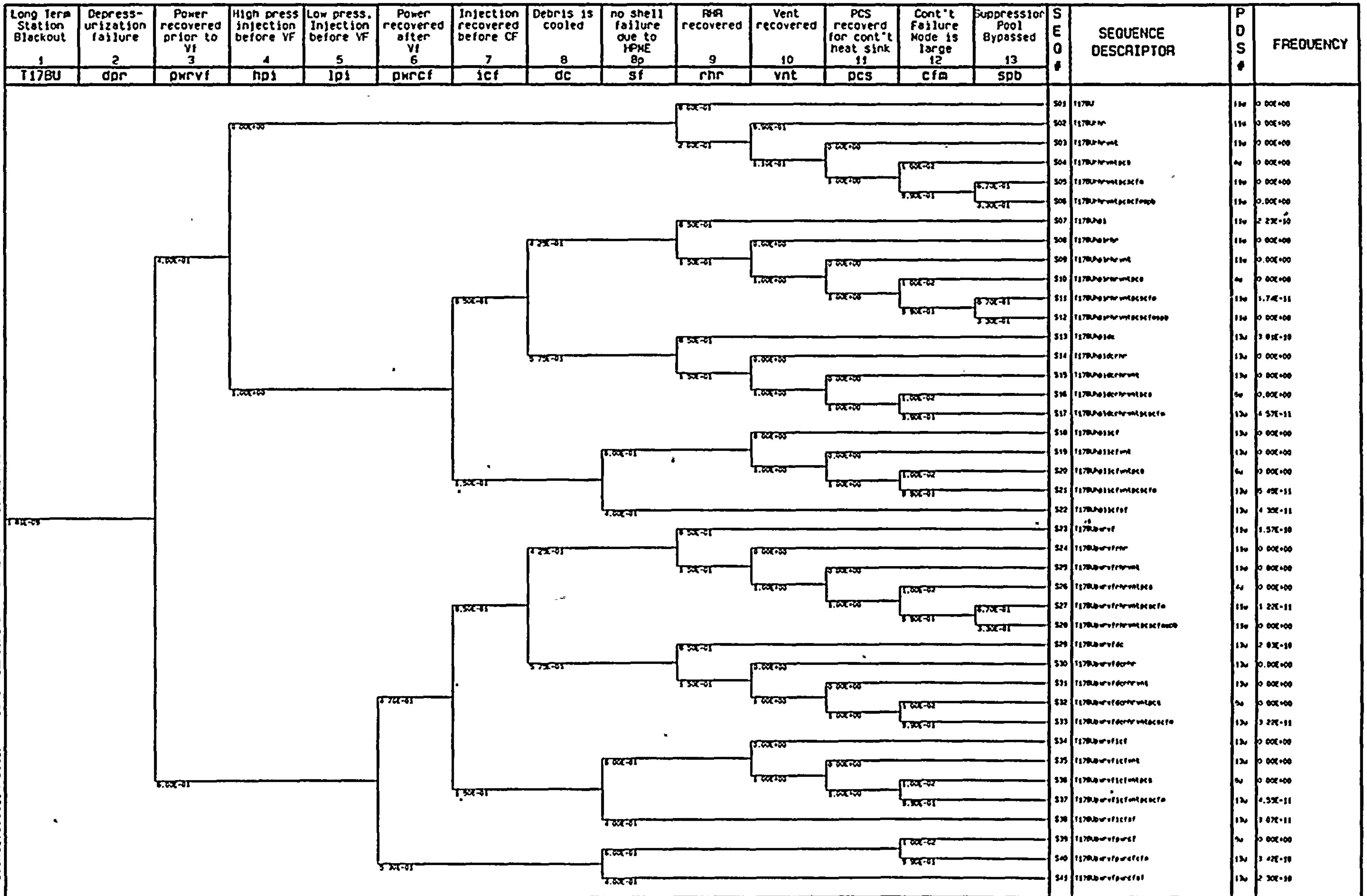
Long term Station Blackout - WNP-2
Containment Event Tree - Isolated
Revision 1, April 8, 1994
(TE-S17A)
Figure 4.7-5



Long term Station Blackout - WNP-2
Containment Event Tree - Unisolated
Revision 1, April 8, 1994
(IE-S17A)
Figure 4.7-6

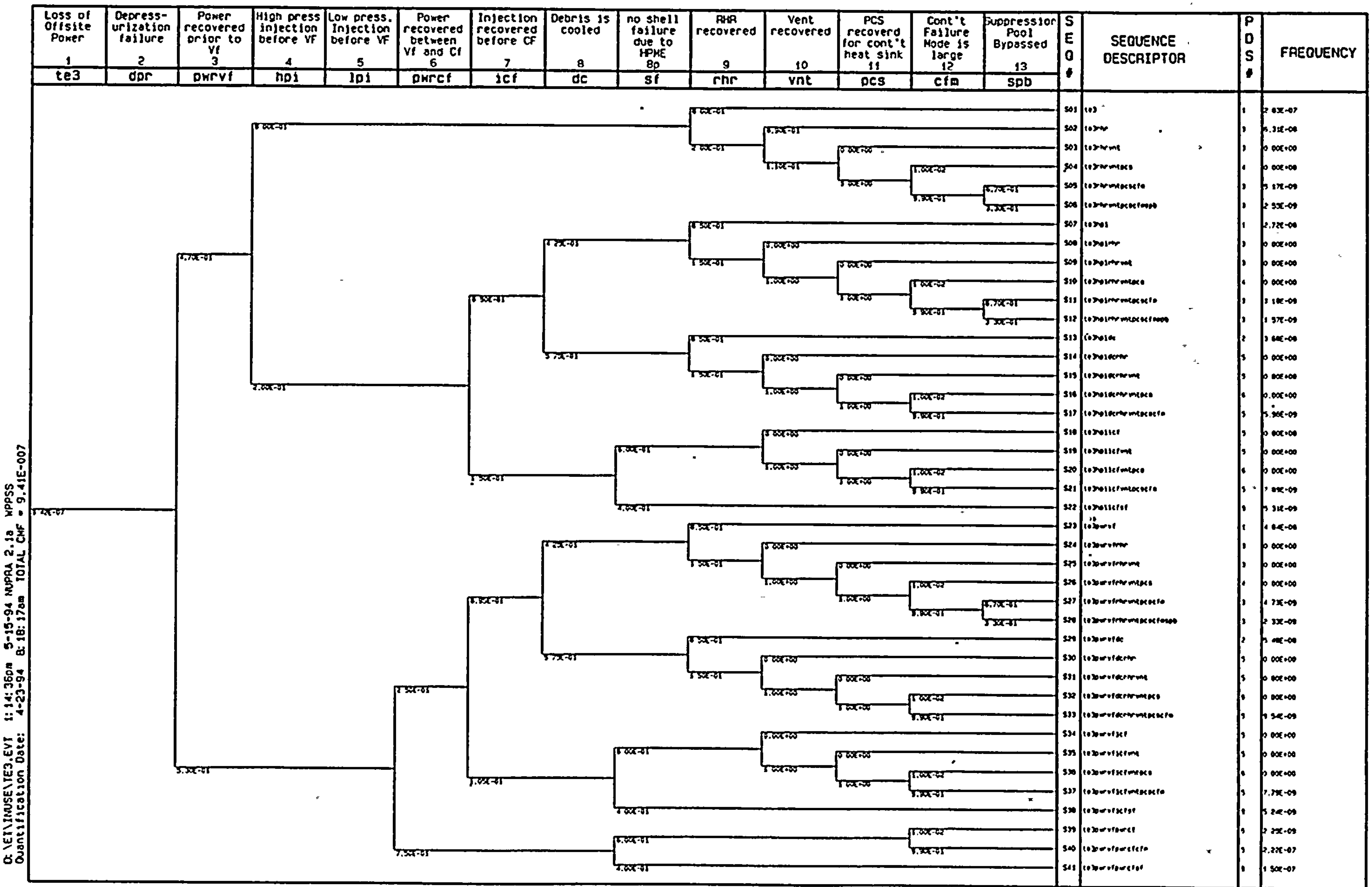


Long term Station Blackout - WNP-2
Containment Event Tree - Isolated
Revision 1, April 8, 1994
(TE-S178)
Figure 4.7-7



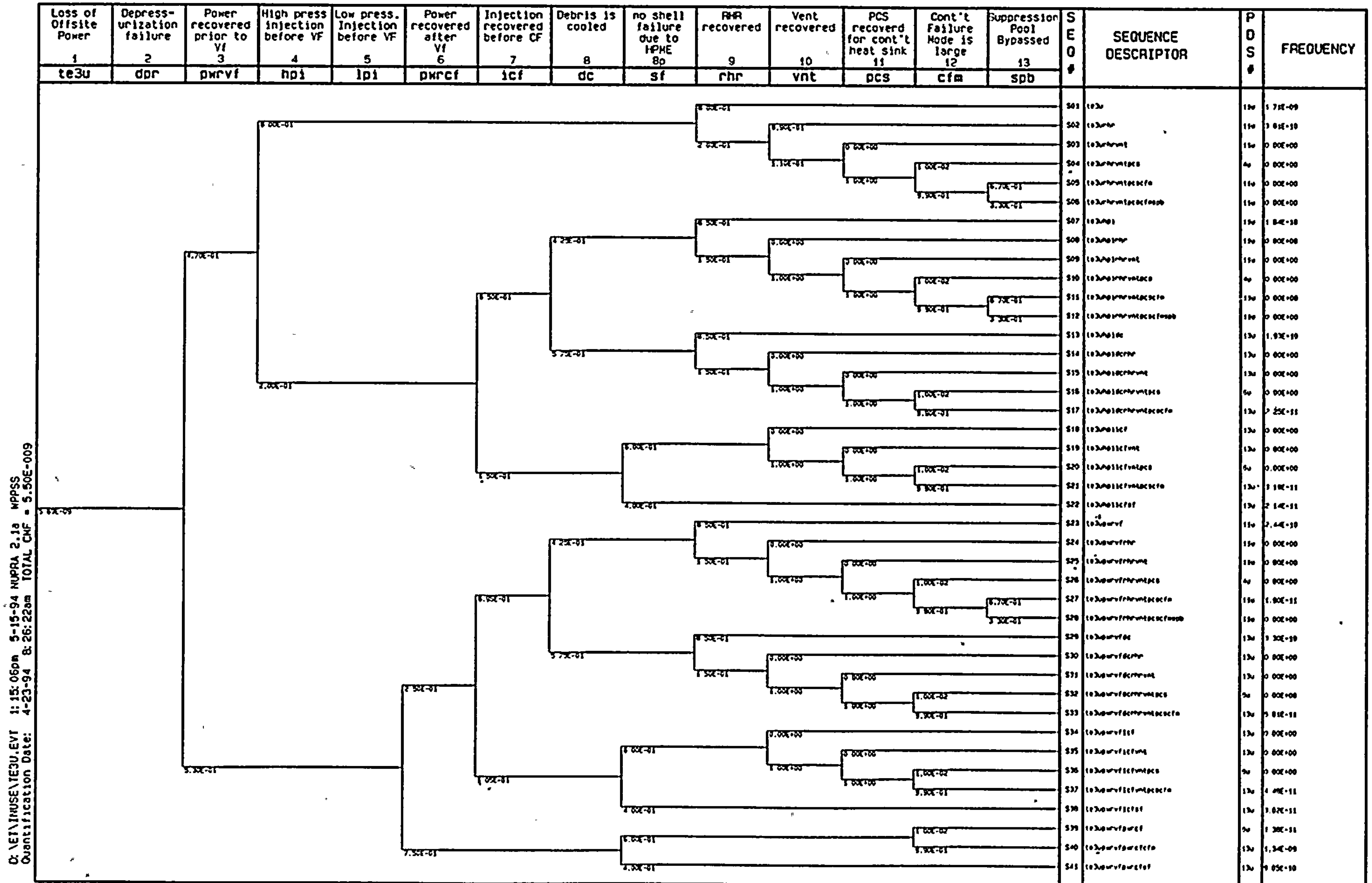
O:\ET\INUSE\T17BU.EVT 1:14:04pm 5-15-94 NUPRA 2.1a WPPSS
Quantification Date: 4-09-94 12:55:34pm TOTAL CMF = 1.75E-009

Long term Station Blackout - WNP-2
Containment Event Tree - Unisolated
Revision 1, April 8, 1994
(TE-S17B)
Figure 4.7-8



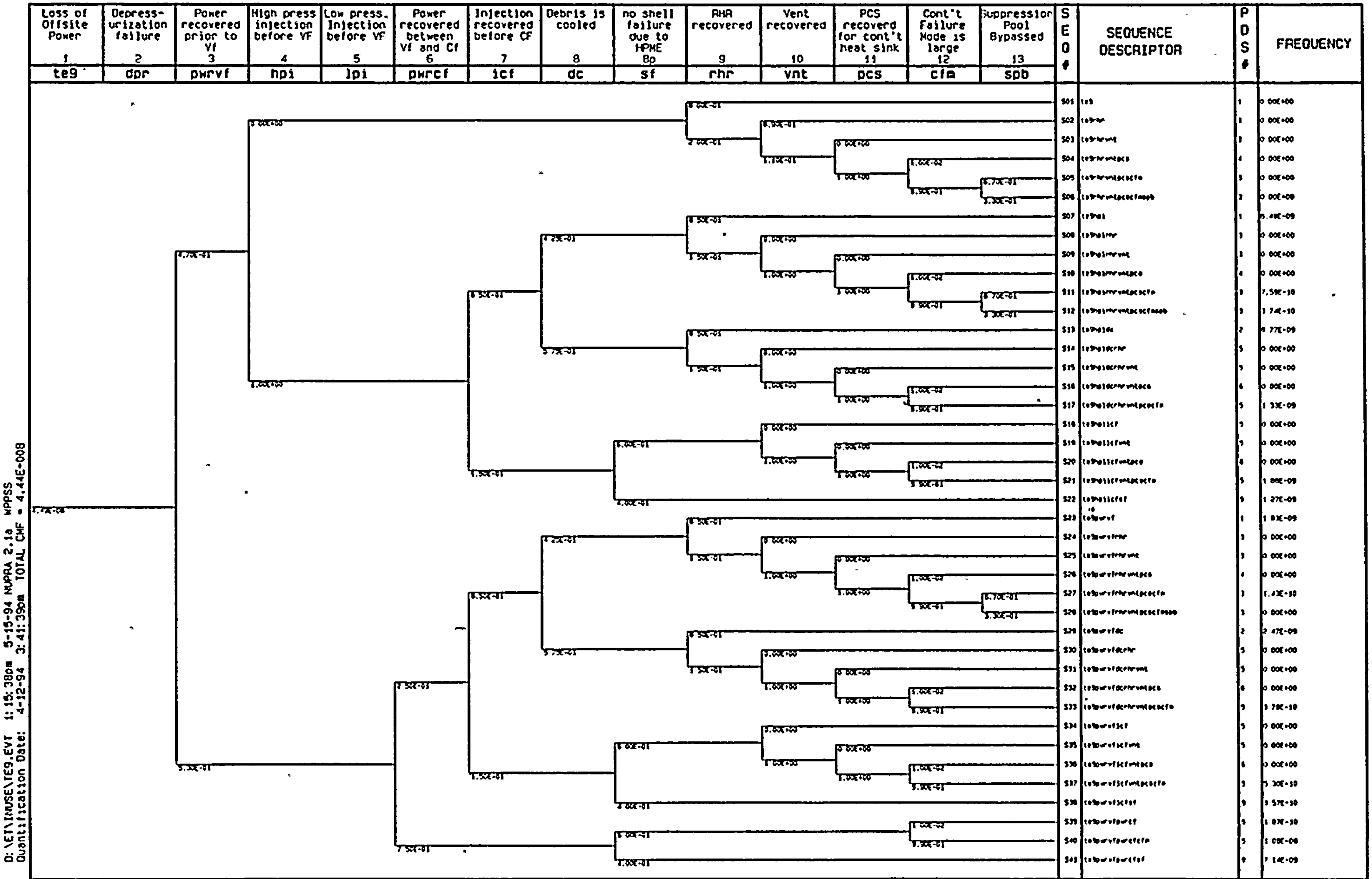
Long term Station Blackout - WNP-2
Containment Event Tree - Isolated

Revision 1, April 8, 1994
(TE-S03, TE-S06)
Figure 4.7-9



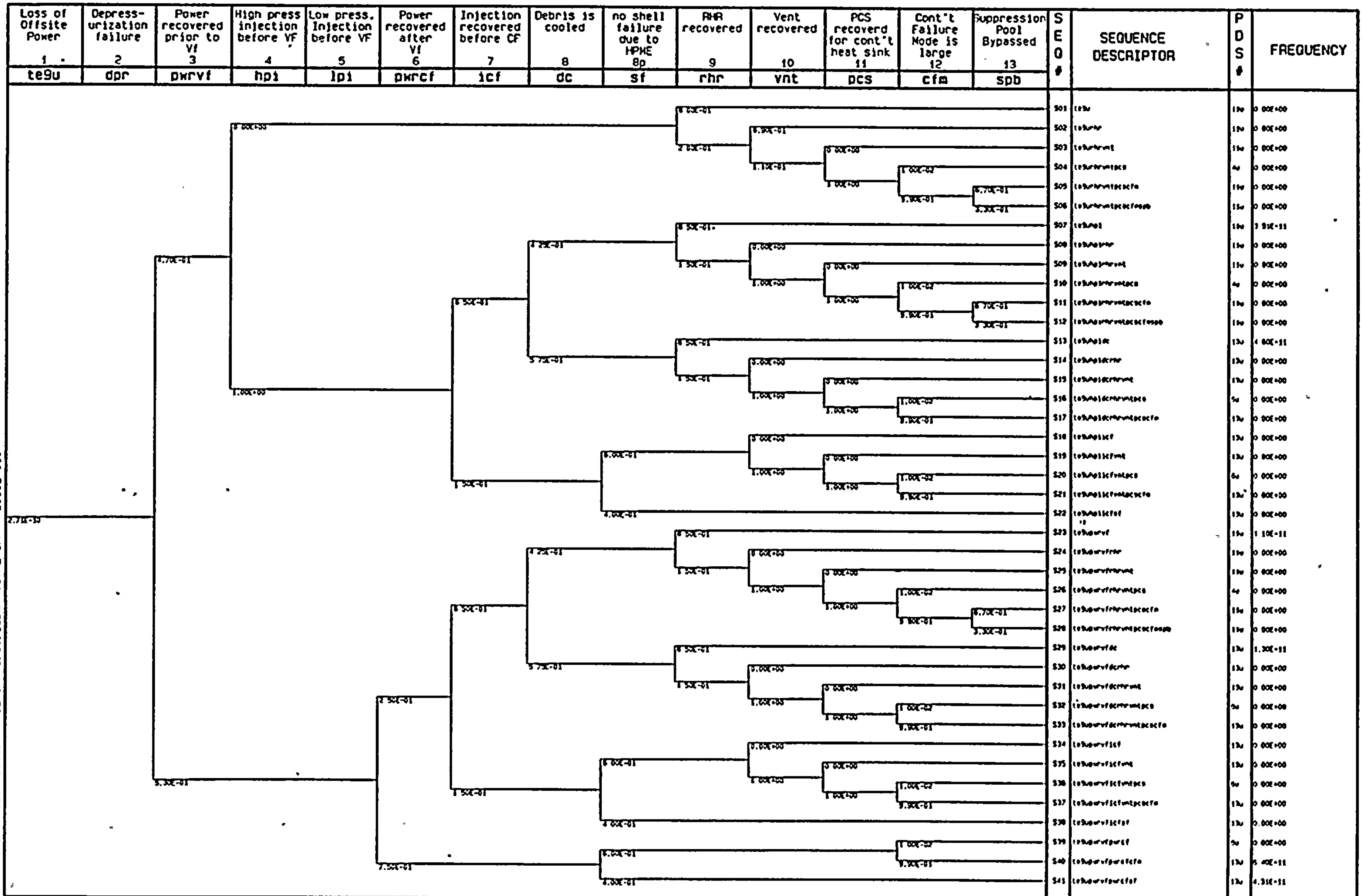
Long term Station Blackout - WNP-2
Containment Event Tree - Unisolated

Revision 1, April 8, 1994
(TE-S03, TE-S06)
Figure 4.7-10



Long term Station Blackout - WNP-2
Containment Event Tree - Isolated

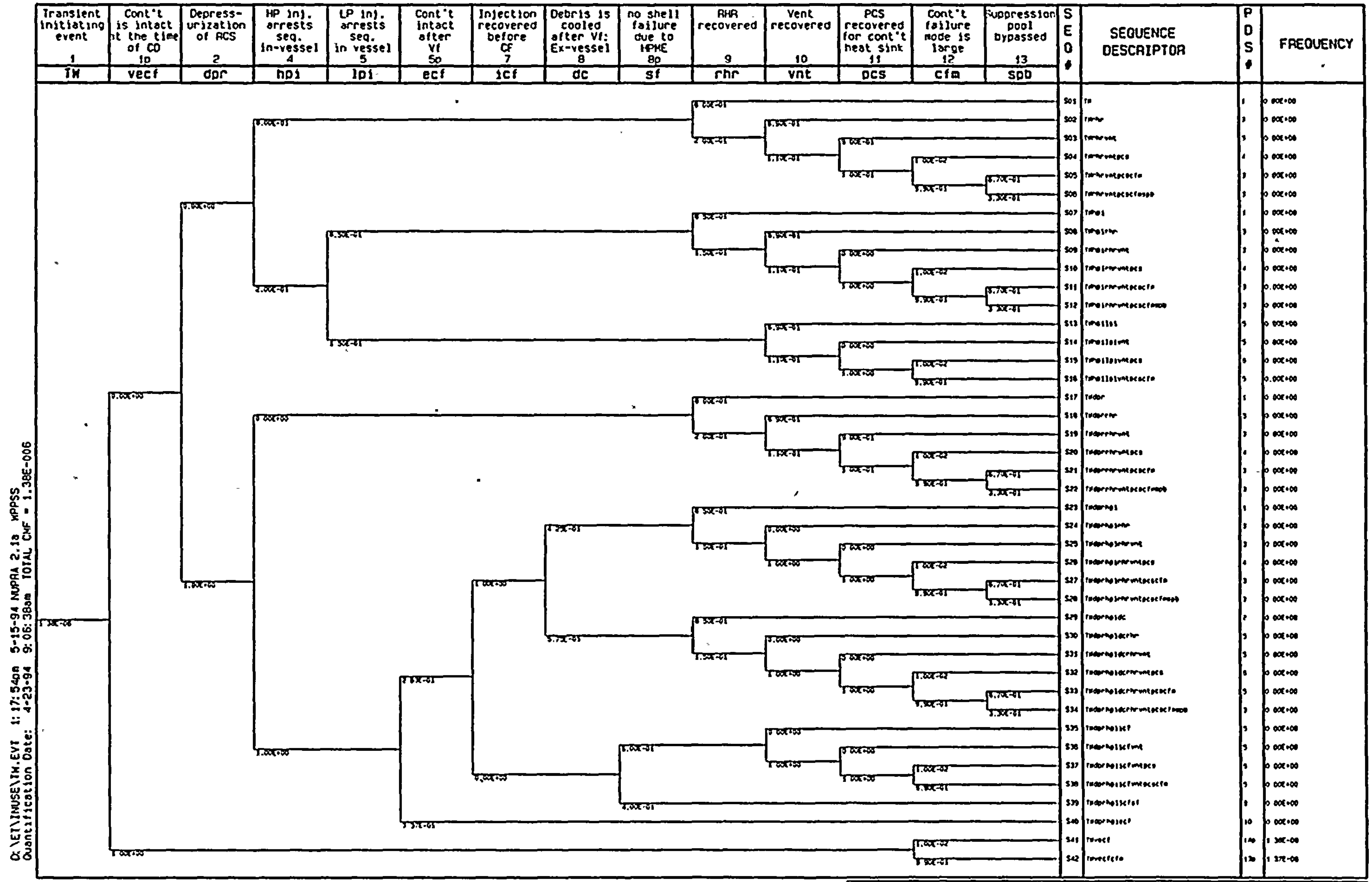
Revision 1, April 8, 1994
(TE-S09)
Figure 4.7-11



D:\ET\INUSE\TE9U.EVT 1:16:18pm 5-15-94 NUPRA 2.1b WPPSS
Quantification Date: 4-12-94 3:50:33pm TOTAL CMF = 2.16E-010

Long term Station Blackout - WNP-2
Containment Event Tree - Unisolated

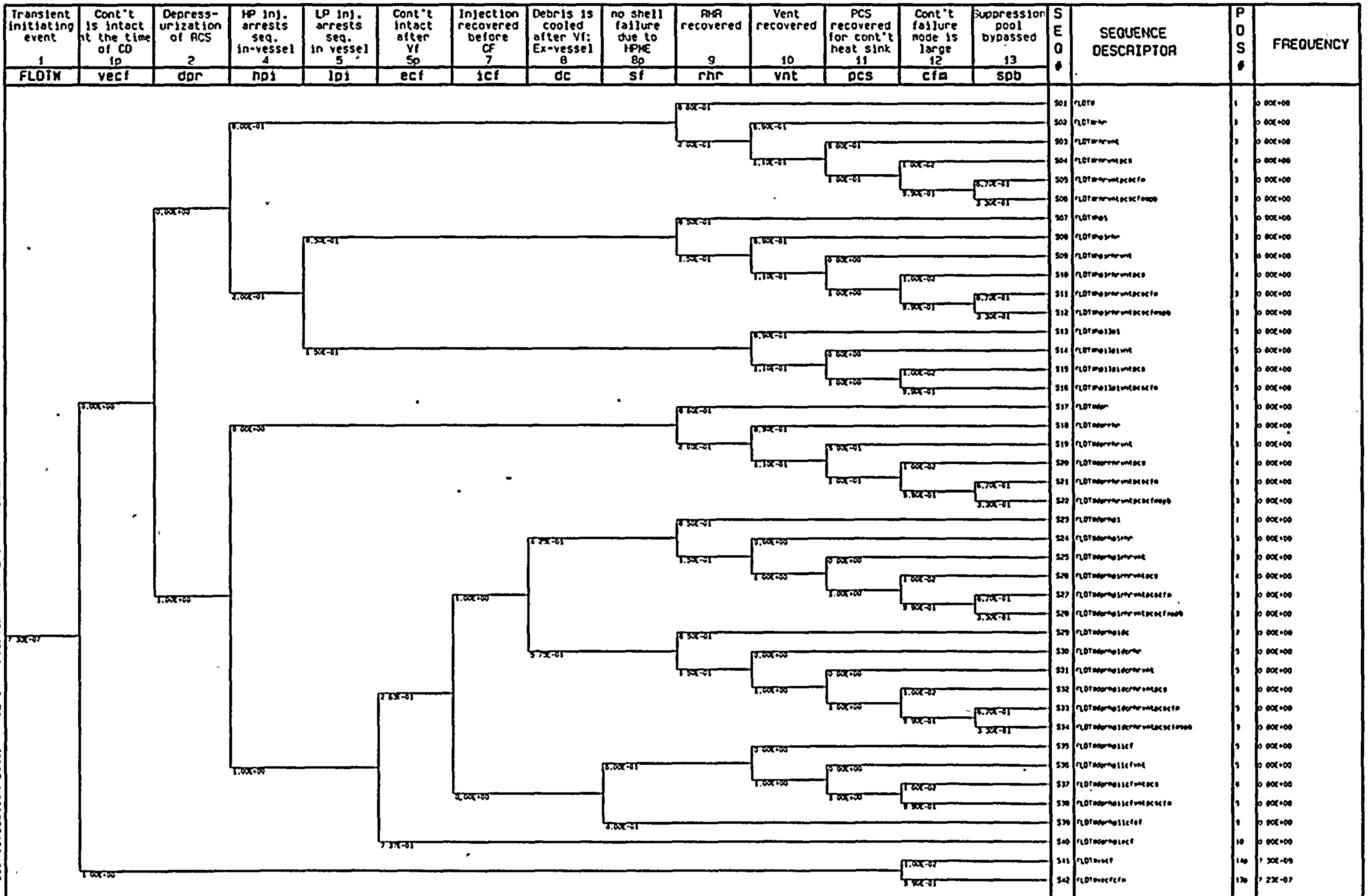
Revision 1, April 8, 1994
(TE-S09)
Figure 4.7-12



Transient - WNP-2
Containment Event Tree

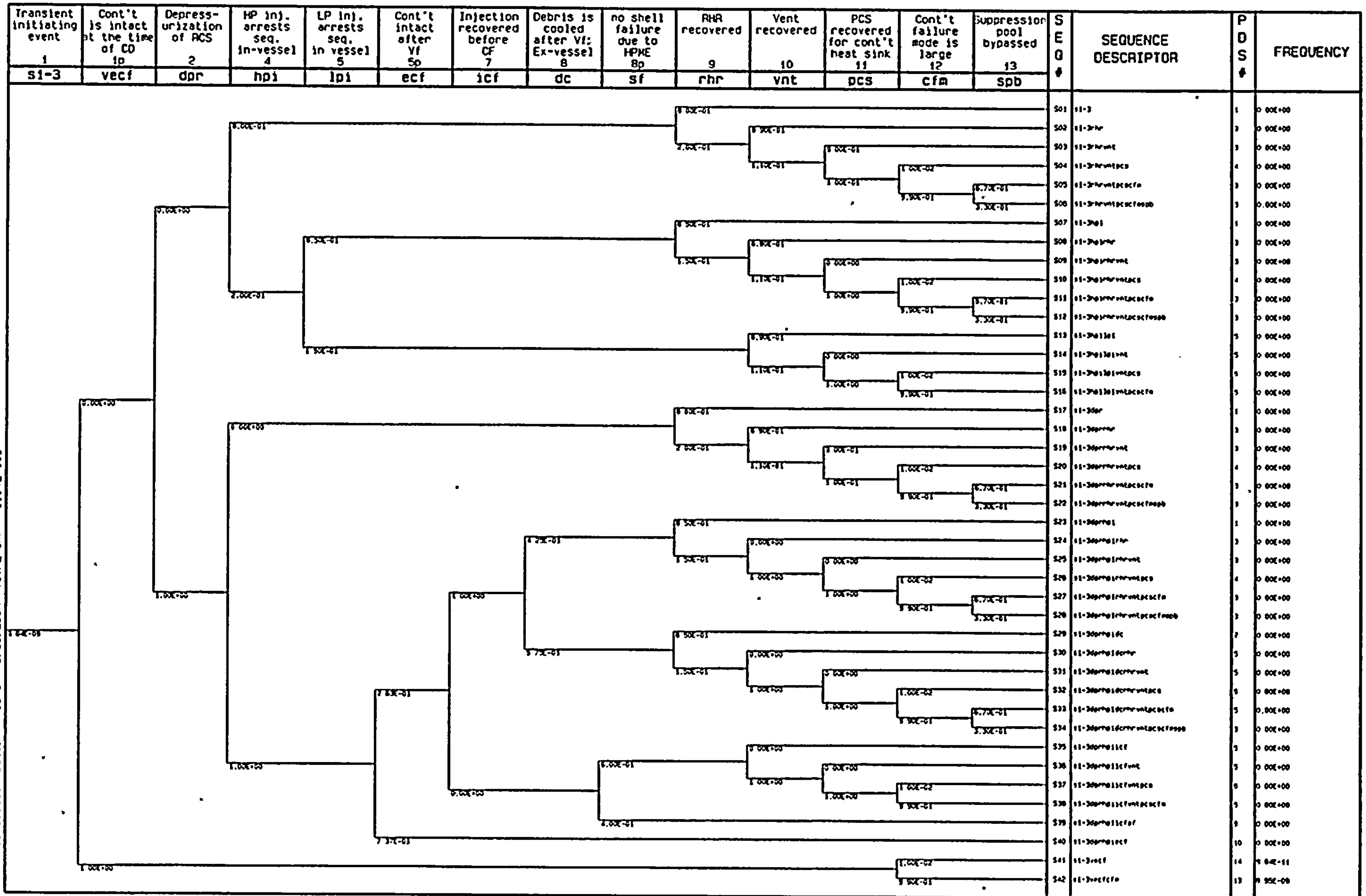
Revision 1, April 8, 1994
(TC-3, TCAS-3, TCN-3, TOC-5, TF-4, TW-4, TSSW-4, TT-5, TTSW-2
WS-5) Figure 4.7-13

O:\EET\INUSE\FLD14.EVT 1:18:26pm 5-15-94 NUPRA 2.1a WPPSS
Quantification Date: 4-12-94 4:05:08pm TOTAL CMF = 7.30E-007



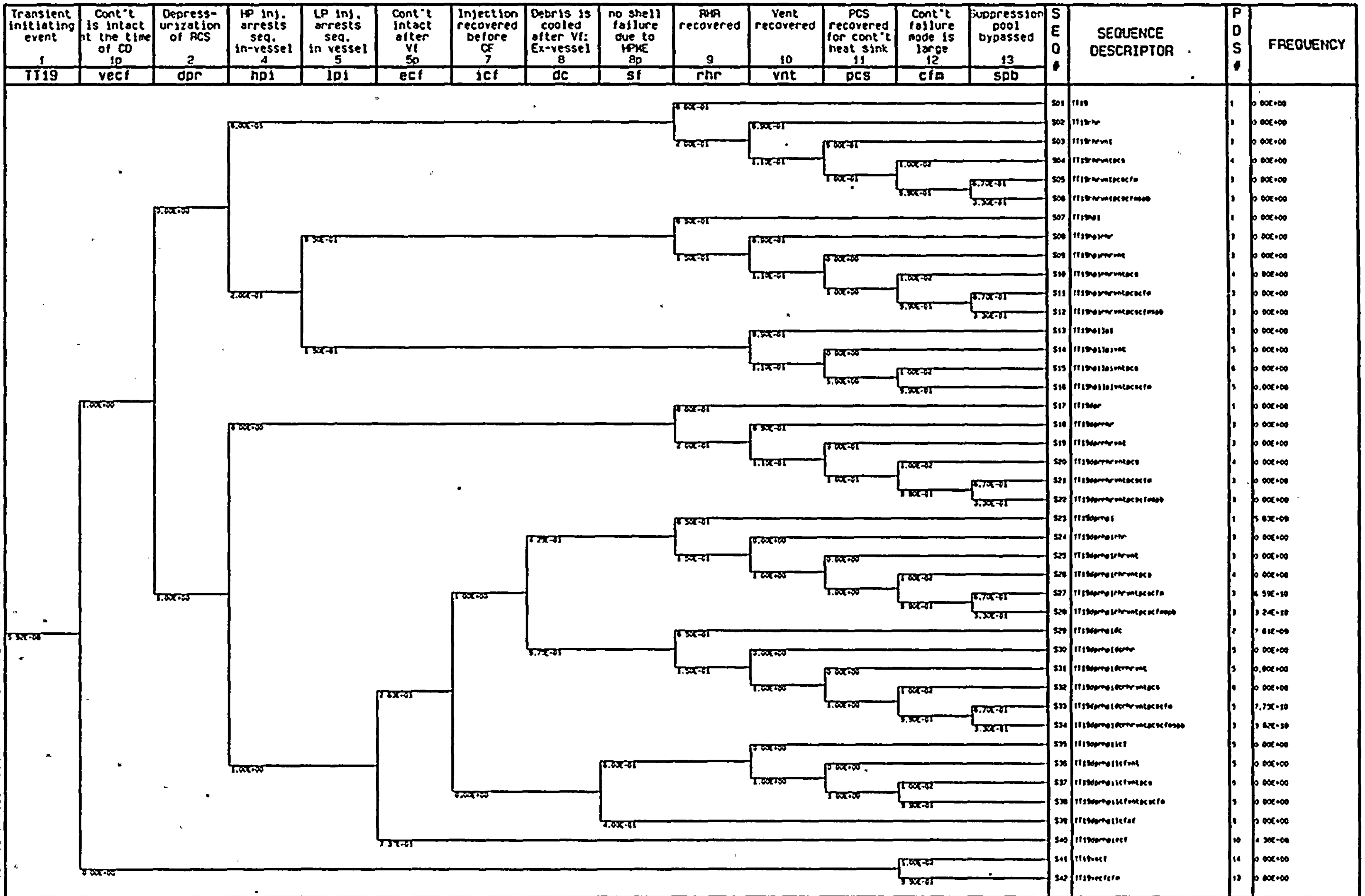
Transient - WNP-2
Containment Event Tree
Revision 1, April 8, 1994
(FLD14-S2, FLD6-S2)
Figure 4.7-14

D:\ET\INUSE\S1-3.EVT 1:19:00am 5-15-94 NUPRA 2.1a WPPSS
Quantification Date: 4-13-94 9:35:25am TOTAL CMF = 9.04E-009

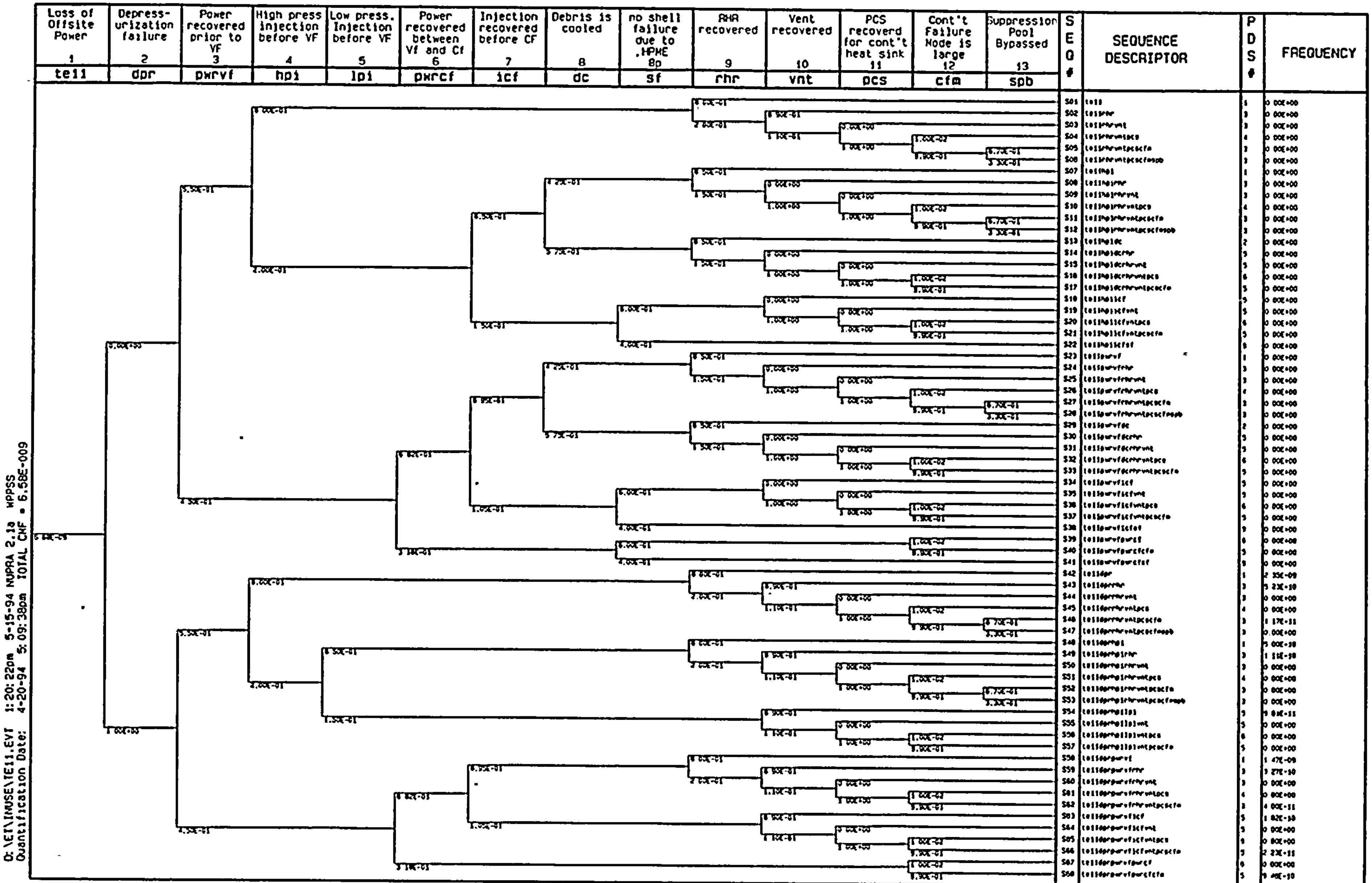


Transient - WNP-2
Containment Event Tree
Revision 1, April 8, 1994
(S1-S03)
Figure 4.7-15

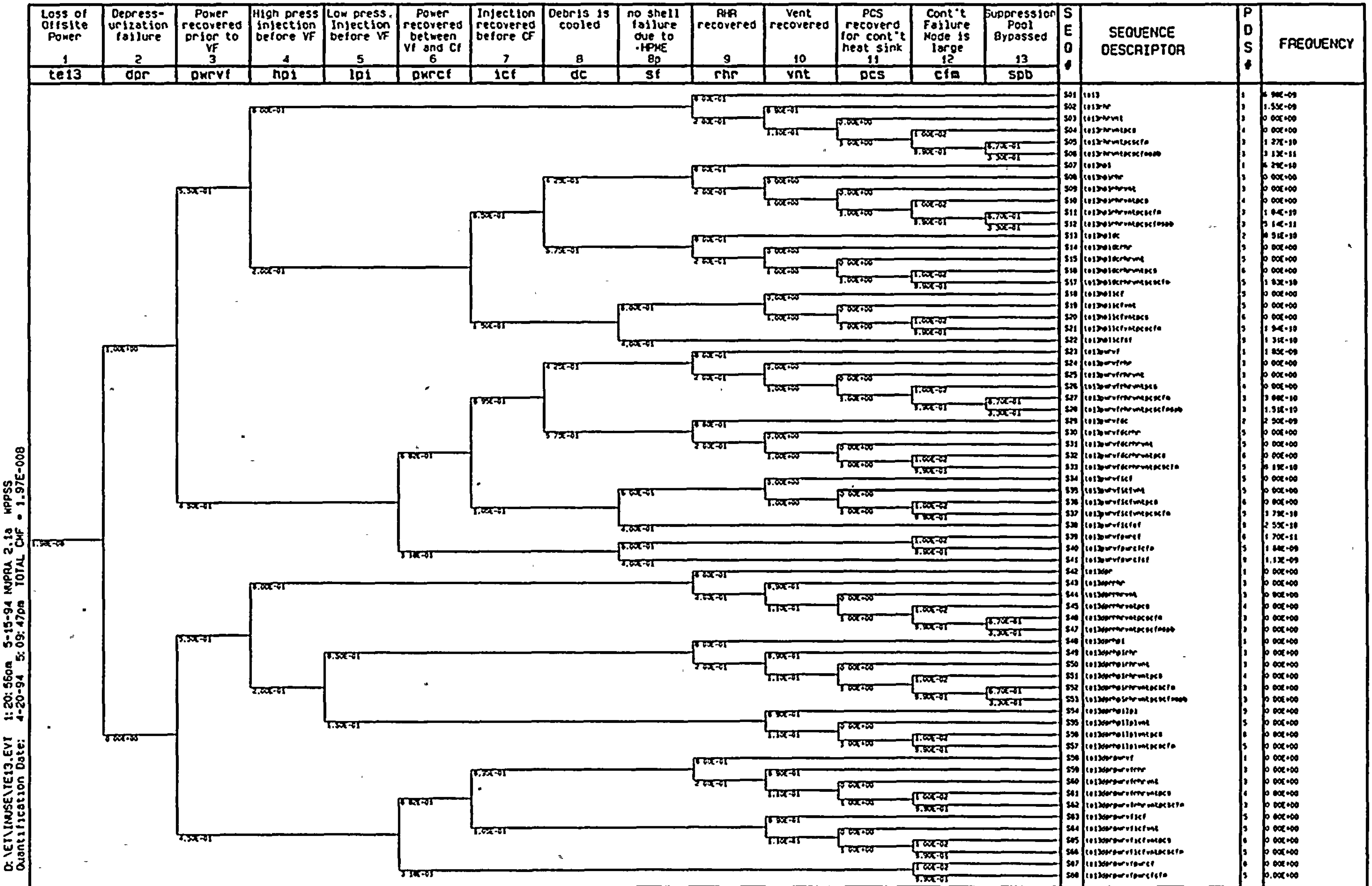
D:\ET\INUSE\TT19.EVT 1:19:30pm 5-15-94 NUPRA 2.1a WPPSS
Quantification Date: 4-10-94 8:24:43am TOTAL CHF = 5.90E-008



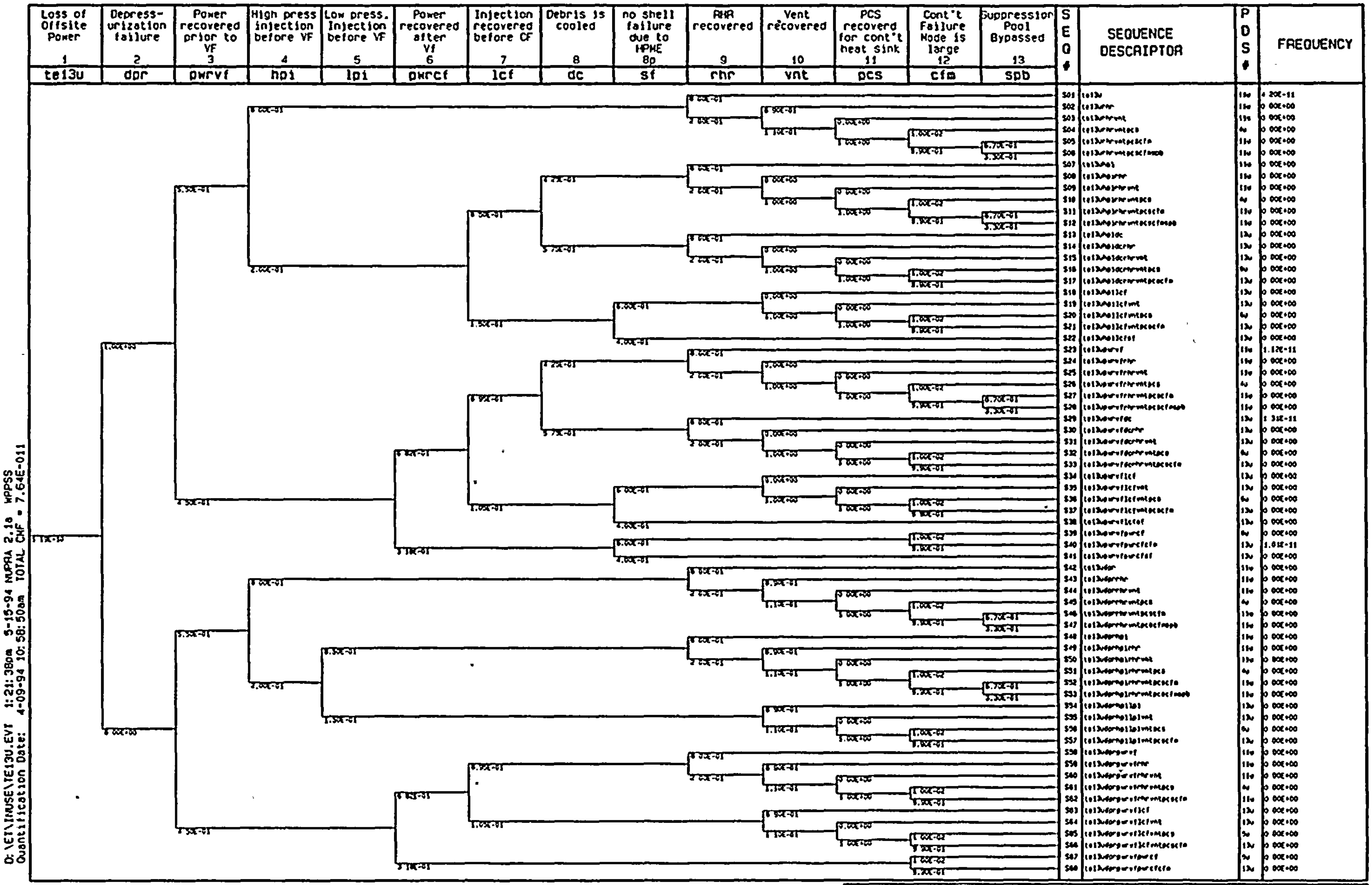
Transient - WNP-2
Containment Event Tree - Isolated
Revision 1, April 8, 1994
(TT-S19)
Figure 4.7-16



Short term Station Blackout - WNP-2
Containment Event Tree - Isolated
Revision 1, April 8, 1994
(TE-S11)
Figure 4.7-17

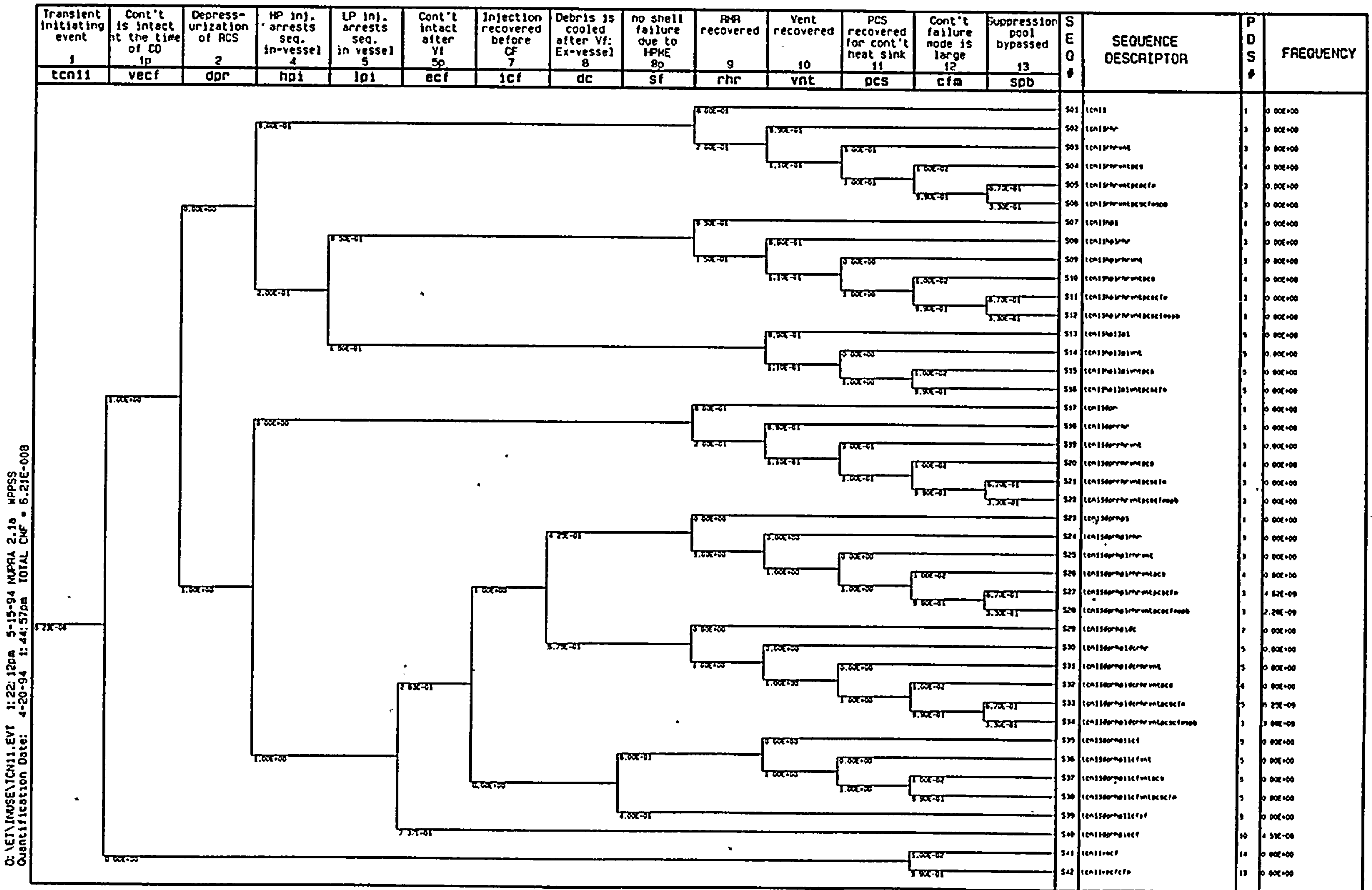


Short term Station Blackout - WNP-2
Containment Event Tree - Isolated
Revision 1, April 8, 1994
(TE-S13)
Figure 4.7-18



Short term Station Blackout - WNP-2
Containment Event Tree - Unisolated

Revision 1, April 8, 1994
(TE-S13)
Figure 4.7-19



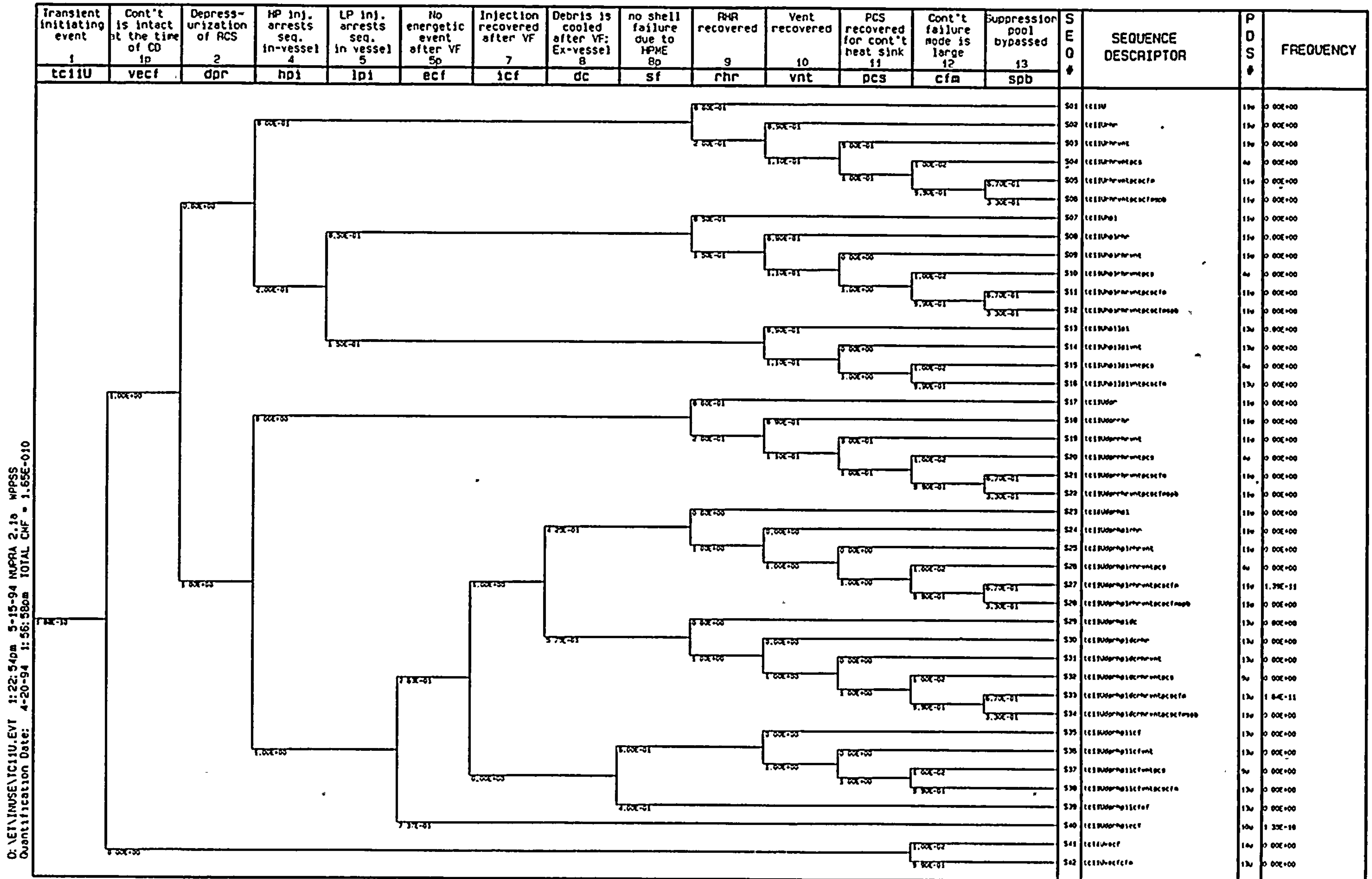
Transient - WNP-2

Containment Event Tree - Isolated

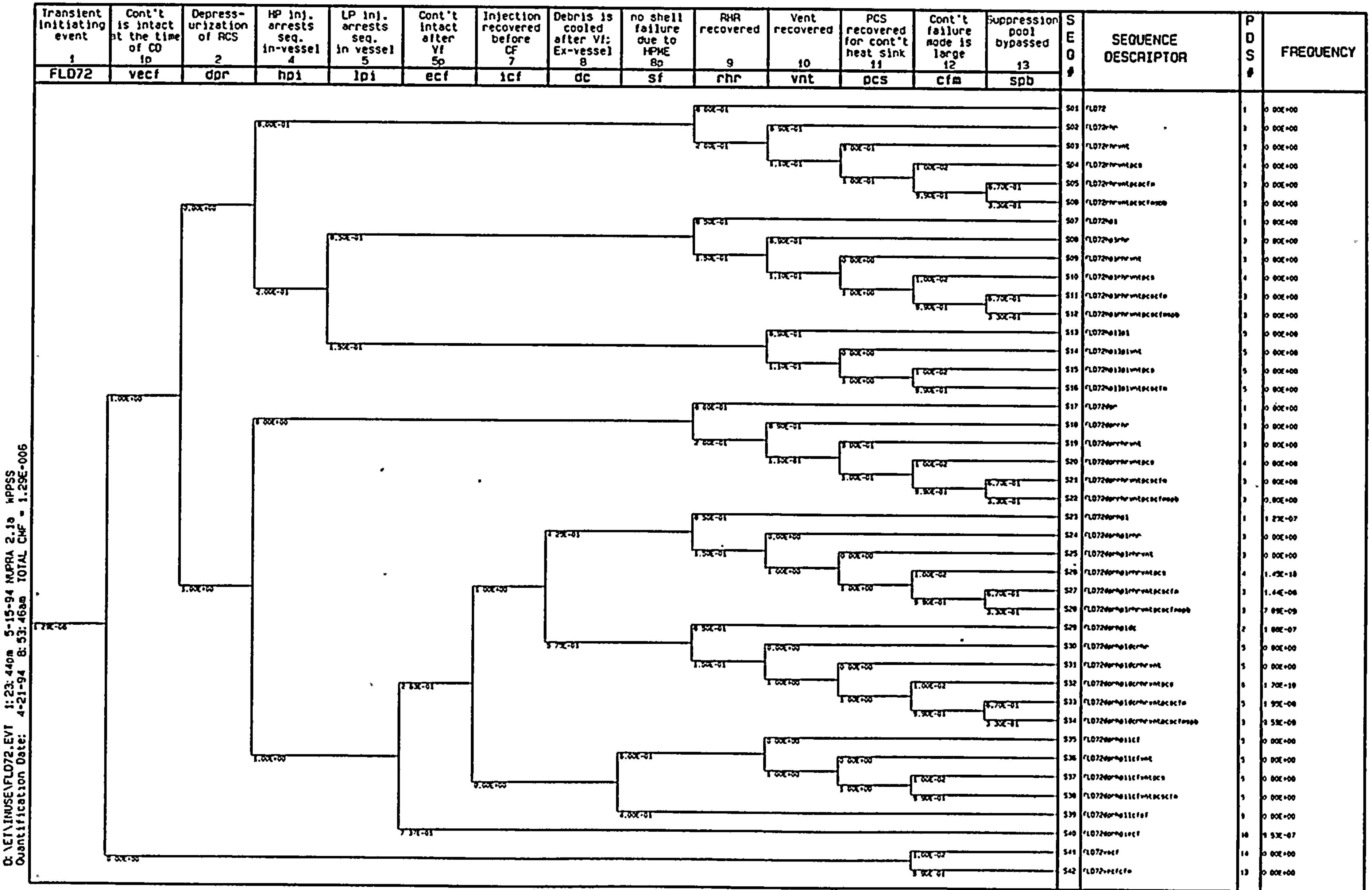
Revision 1, April 8, 1994

(TCN-S11, TDC-S15, TF-S14, TH-S18, TSSW-S09)

Figure 4.7-20

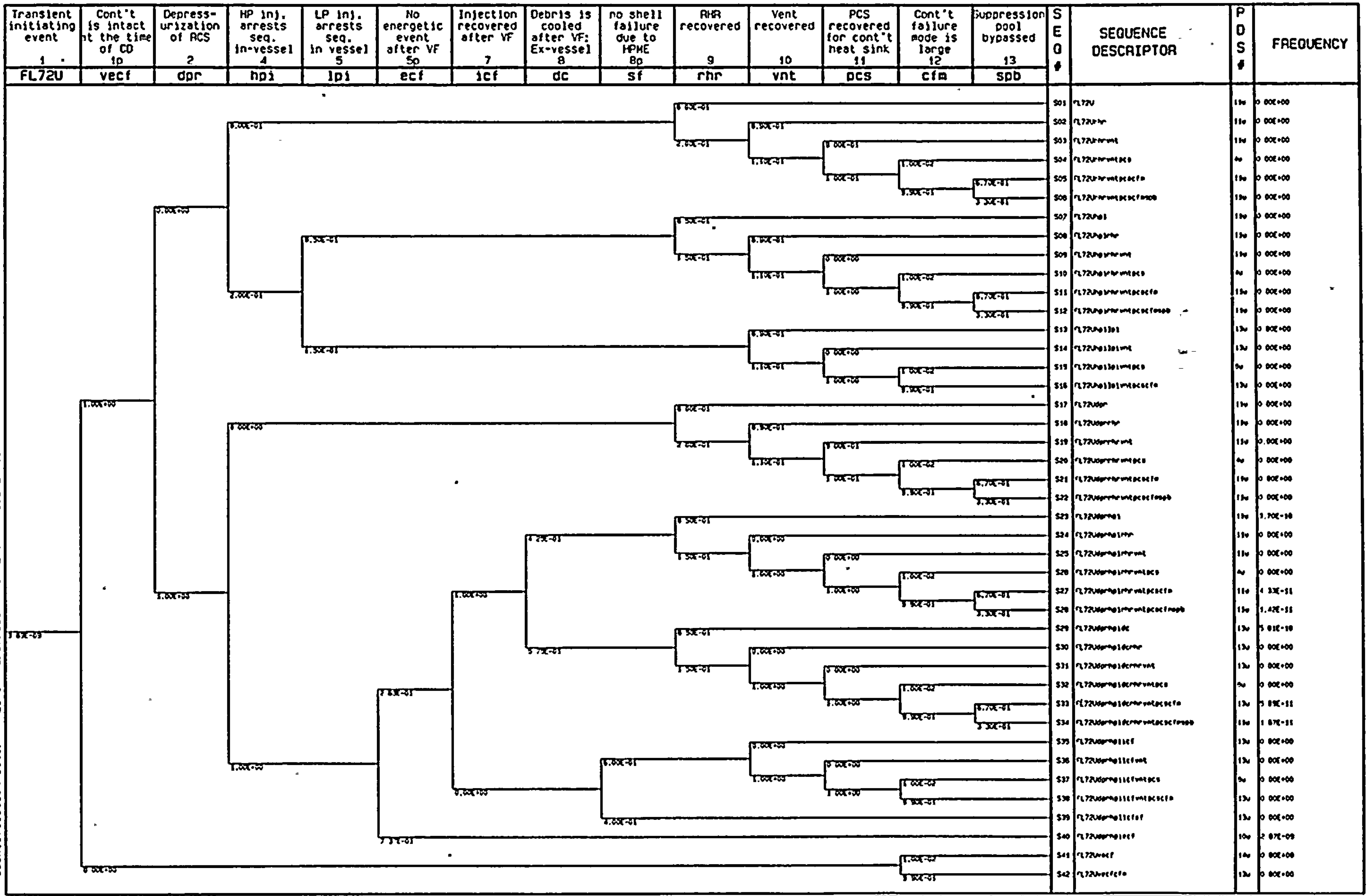


Transient - WNP-2
Containment Event Tree - Unisolated
Revision 1, April 8, 1994
(TCN-S11, TDC-S15, TF-S14, TM-S18, TSSW-S09)
Figure 4.7-21



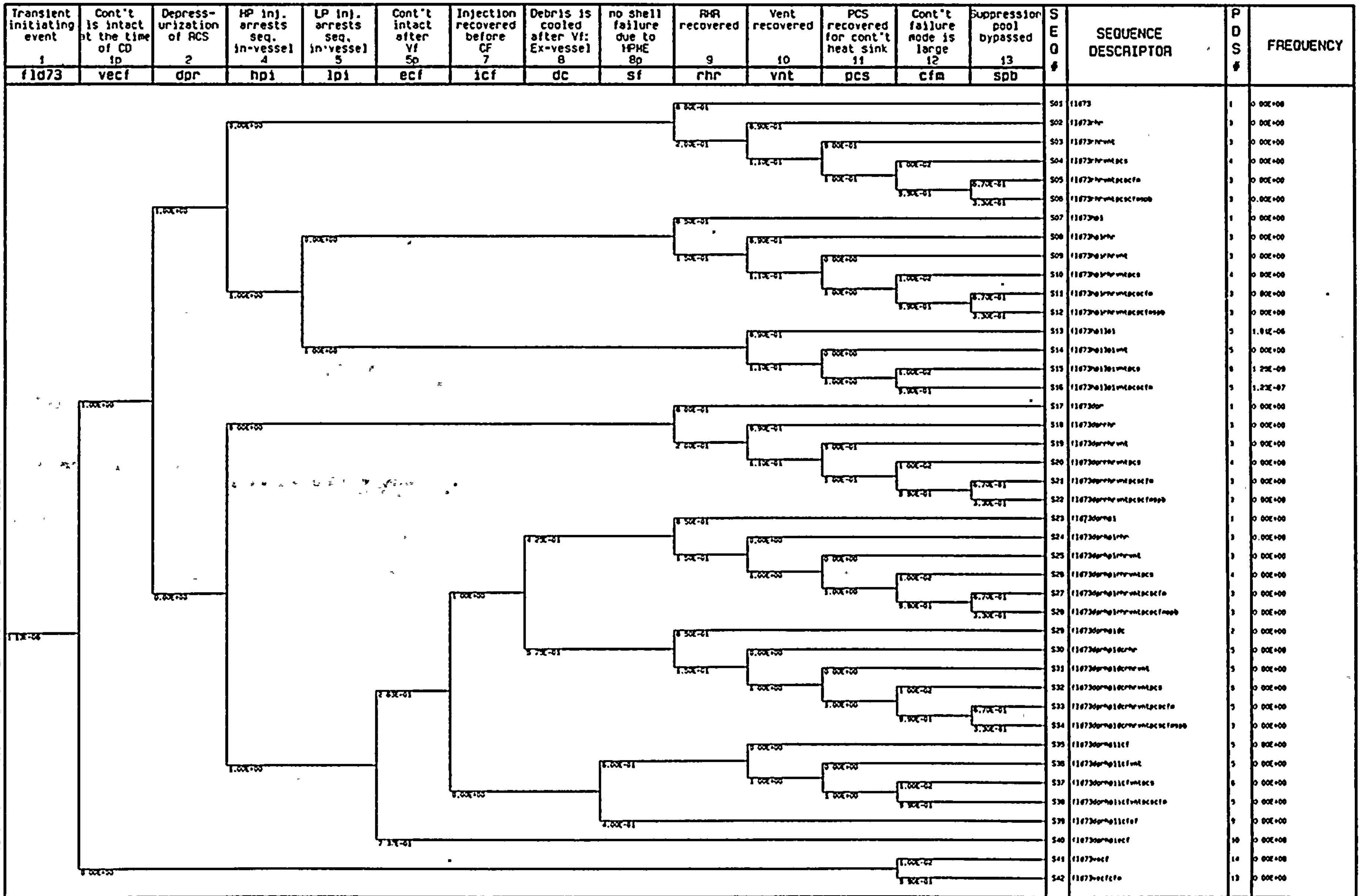
Transient - WNP-2
Containment Event Tree - Isolated
Revision 1, April 8, 1994
(FLD7-2, TTSW-08, SA-18)
Figure 4.7-22

O:\ET\INUSE\FL72U.EVT 1:24:16pm 5-15-94 NUPRA 2.1a WPPSS
Quantification Date: 4-21-94 8:57:39am TOTAL CMF = 3.87E-009



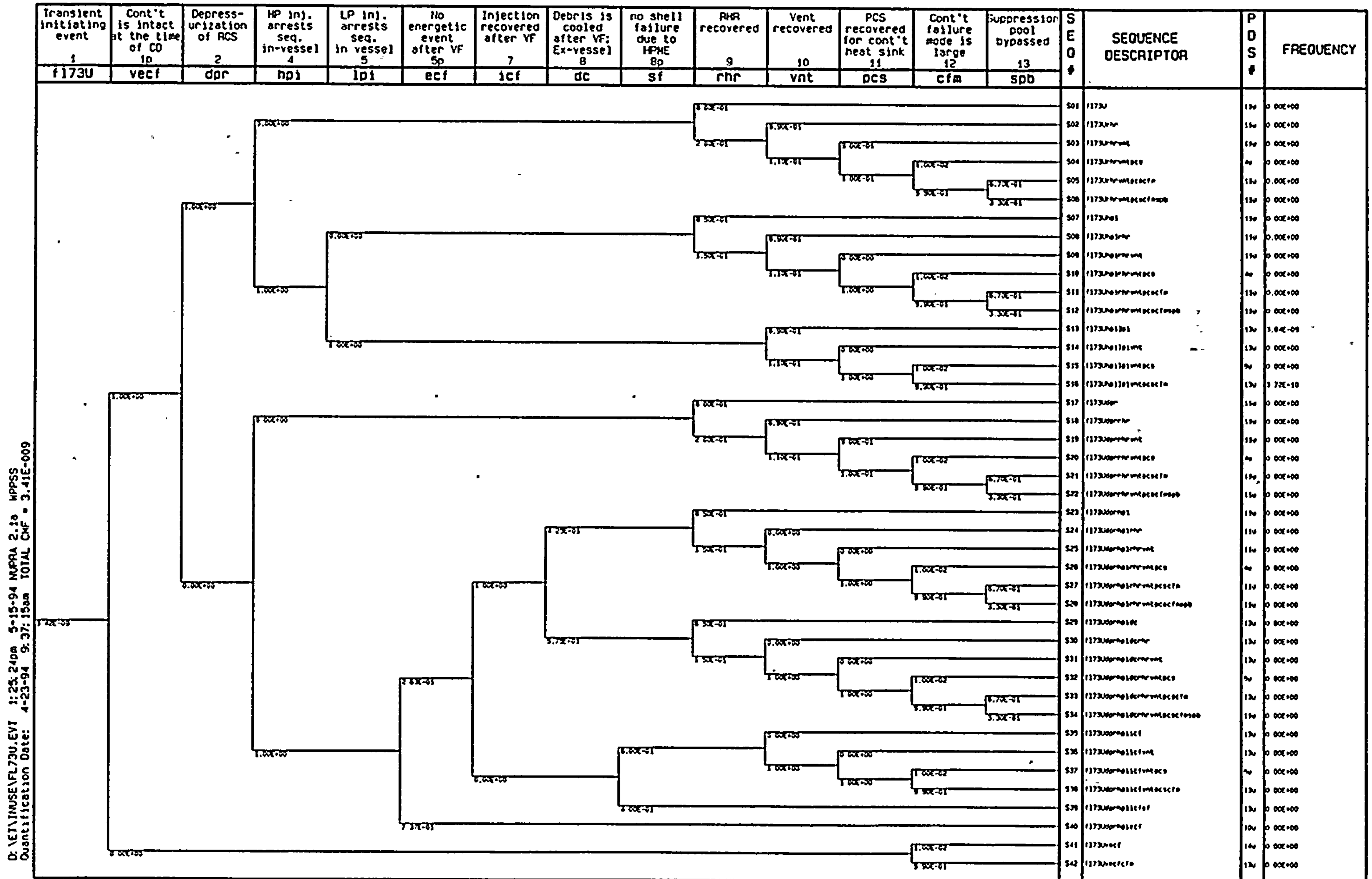
Transient - WNP-2
Containment Event Tree - Unisolated

Revision 1, April 8, 1994
(FLD7-2, TISN-08, SA-18)
Figure 4.7-23



D:\ET\INUSE\FLD73.EVT 1:24:56pm 5-15-94 NUPRA 2.1a WPPSS
Quantification Date: 4-20-94 2:44:30pm TOTAL CMF = 1.13E-006

Transient - WNP-2
Containment Event Tree - Isolated
Revision 1, April 8, 1994
(TDC-14, TI-22, TSSW-8, FLD7-3, MS-19, TF-13)
Figure 4.7-24



Transient - WNP-2

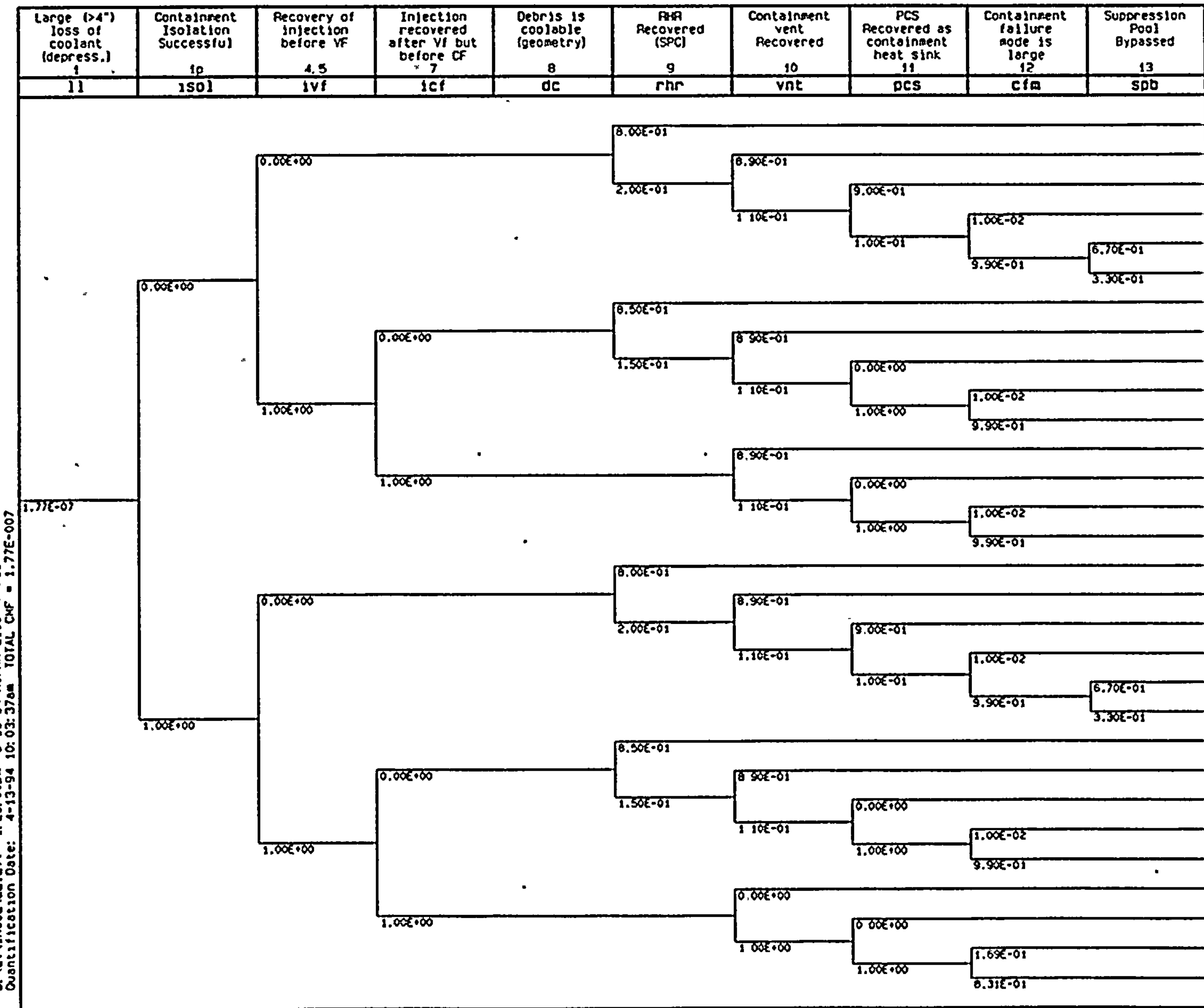
Containment Event Tree - Unisolated

Revision 1, April 8, 1994

(TDC-14, IETI-22, TSSW-8, FLD7-3, MS-19, TF-13)

Figure 4.7-25

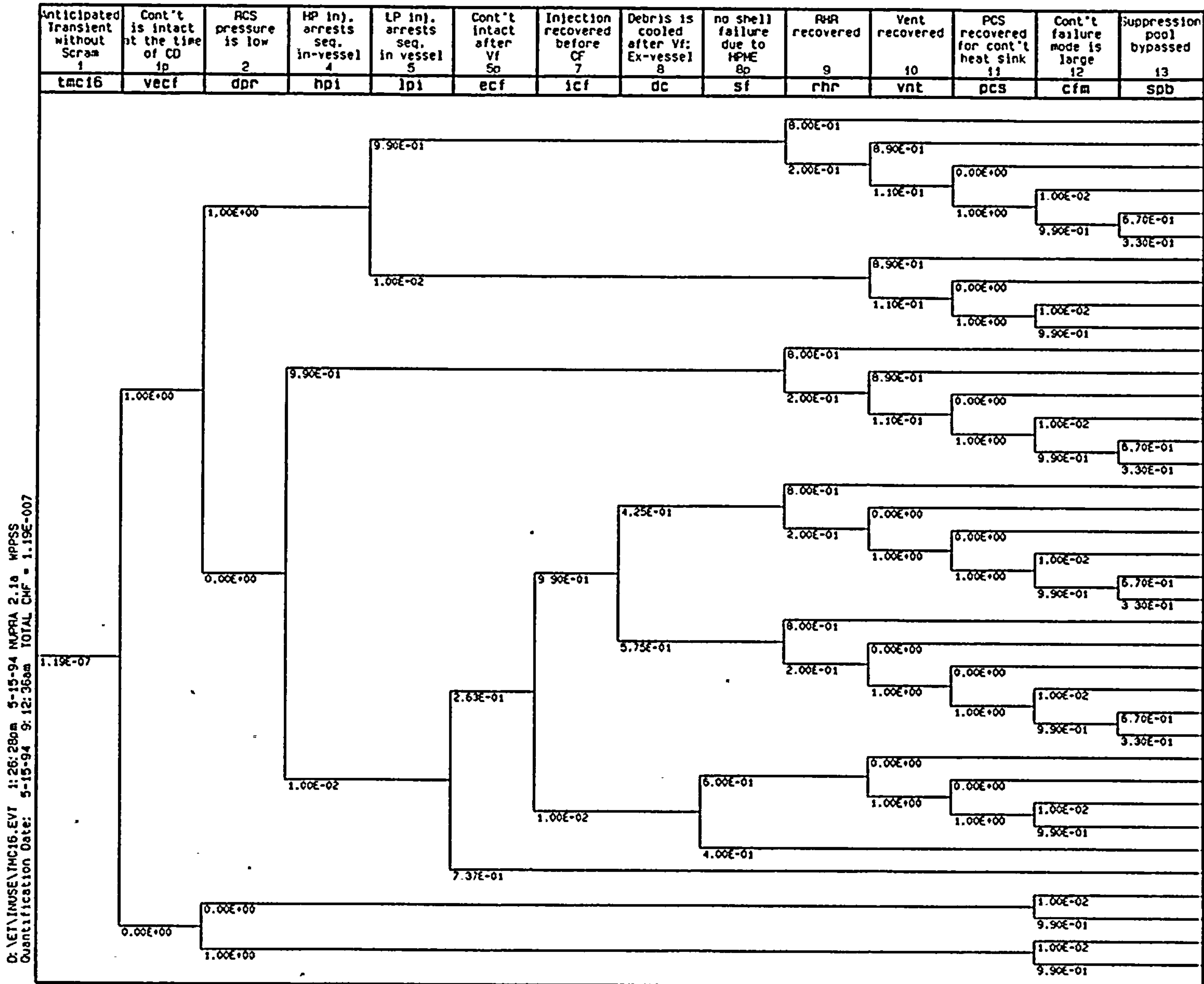
O:\ET\INUSE\LL.EVT 1:26:00pm 5-15-94 NUPRA 2.1a WPPSS
Quantification Date: 4-13-94 10:03:37am TOTAL CMF = 1.77E-007



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
S01	ll	1	0.00E+00
S02	llrhr	3	0.00E+00
S03	llrhrvnt	3	0.00E+00
S04	llrhrvntpcs	4	0.00E+00
S05	llrhrvntpcscfm	3	0.00E+00
S06	llrhrvntpcscfmspb	3	0.00E+00
S07	llivf	1	0.00E+00
S08	llivfrhr	3	0.00E+00
S09	llivfrhrvnt	3	0.00E+00
S10	llivfrhrvntpcs	4	0.00E+00
S11	llivfrhrvntpcscfm	3	0.00E+00
S12	llivficf	5	0.00E+00
S13	llivficfvnt	5	0.00E+00
S14	llivficfvntpcs	6	0.00E+00
S15	llivficfvntpcscfm	5	0.00E+00
S16	llisol	11u	0.00E+00
S17	llisolrhr	11u	0.00E+00
S18	llisolrhrvnt	11u	0.00E+00
S19	llisolrhrvntpcs	4u	0.00E+00
S20	llisolrhrvntpcscfm	11u	0.00E+00
S21	llisolrhrvntpcscfmspb	11u	0.00E+00
S22	llisolivf	11u	0.00E+00
S23	llisolivfrhr	11u	0.00E+00
S24	llisolivfrhrvnt	11u	0.00E+00
S25	llisolivfrhrvntpcs	4u	0.00E+00
S26	llisolivfrhrvntpcscfm	11u	0.00E+00
S27	llisolivficf	13u	0.00E+00
S28	llisolivficfvnt	13u	0.00E+00
S29	llisolivficfvntpcs	17	2.99E-08
S30	llisolivficfvntpcscfm	13u	1.47E-07

Loss Of Coolant Accident (LOCA) - Large (>4") - WNP-2
Containment Event Tree

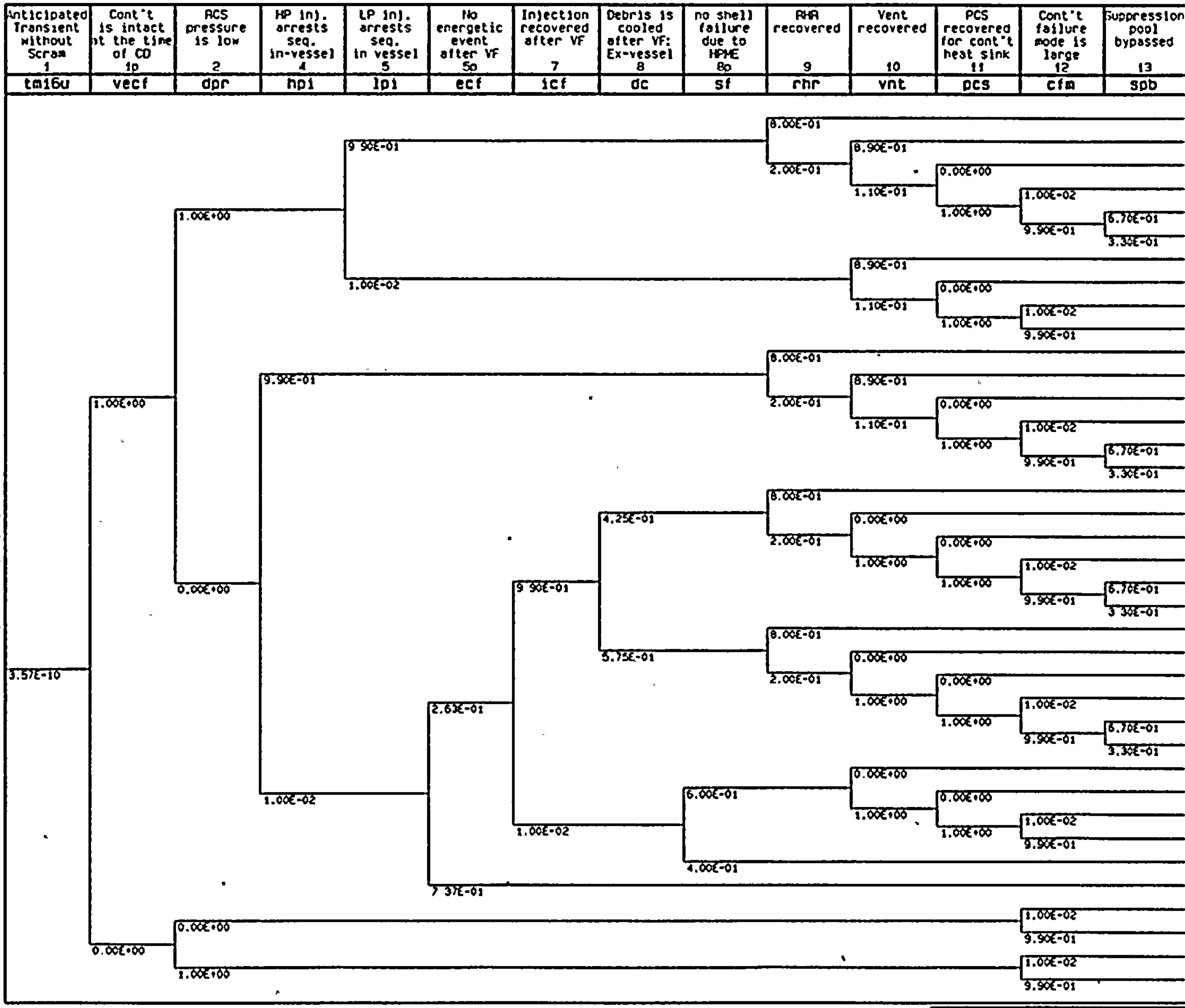
Revision 1, April 8, 1994
(AO-S14, A-S08)
Figure 4.7-26



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
S01	tmc16	1	9.42E-08
S02	tmc16rhr	3	2.10E-08
S03	tmc16rhrvnt	3	0.00E+00
S04	tmc16rhrvntpcs	4	0.00E+00
S05	tmc16rhrvntpcscfm	3	1.72E-09
S06	tmc16rhrvntpcscfmspb	3	8.47E-10
S07	tmc16lpi	5	1.06E-09
S08	tmc16lpivnt	5	0.00E+00
S09	tmc16lpivntpcs	6	0.00E+00
S10	tmc16lpivntpcscfm	5	0.00E+00
S11	tmc16dpr	1	0.00E+00
S12	tmc16dprrhr	3	0.00E+00
S13	tmc16dprrhrvnt	3	0.00E+00
S14	tmc16dprrhrvntpcs	4	0.00E+00
S15	tmc16dprrhrvntpcscfm	3	0.00E+00
S16	tmc16dprrhrvntpcscfm	3b	0.00E+00
S17	tmc16dprhpi	1	0.00E+00
S18	tmc16dprhpirhr	3	0.00E+00
S19	tmc16dprhpirhrvnt	3	0.00E+00
S20	tmc16dprhpirhrvntpcs	4	0.00E+00
S21	tmc16dprhpirhrvntpcsc	8m	0.00E+00
S22	tmc16dprhpirhrvntpcsc	8msp	0.00E+00
S23	tmc16dprhpidc	2	0.00E+00
S24	tmc16dprhpidcrhr	5	0.00E+00
S25	tmc16dprhpidcrhrvnt	5	0.00E+00
S26	tmc16dprhpidcrhrvntpc	6	0.00E+00
S27	tmc16dprhpidcrhrvntpcscfm	[illegible]	0.00E+00
S28	tmc16dprhpidcrhrvntpcscfm	[illegible]	0.00E+00
S29	tmc16dprhpiicf	5	0.00E+00
S30	tmc16dprhpiicfvnt	5	0.00E+00
S31	tmc16dprhpiicfvntpcs	6	0.00E+00
S32	tmc16dprhpiicfvntpcsc	8m	0.00E+00
S33	tmc16dprhpiicfsf	9	0.00E+00
S34	tmc16dprhpiecf	10	0.00E+00
S35	tmc16vecf	14	0.00E+00
S36	tmc16vecfcfm	13	0.00E+00
S37	tmc16vecfdpr	14p	0.00E+00
S38	tmc16vecfdprcfm	13p	0.00E+00

Anticipated Transient Without Scram - WNP-2
Containment Event Tree - Isolated

Revision 1, April 7, 1994
(TMC-16, TCC-16, TFC-16, TIC-10, A-09, S1-15)
Figure 4.7-27

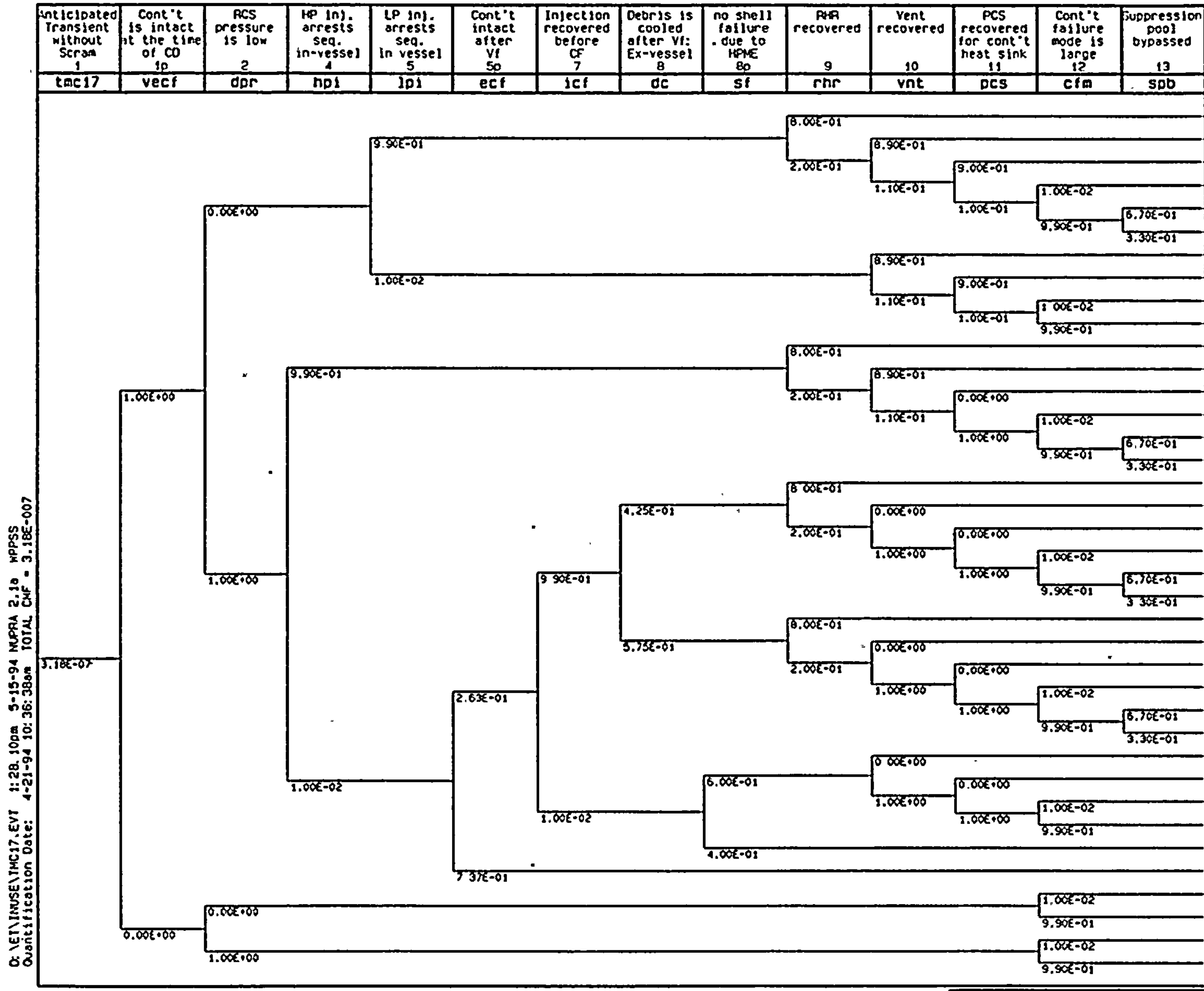


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
S01	tm16u	11u	2.83E-10
S02	tm16urhr	11u	6.29E-11
S03	tm16urhrvnt	11u	0.00E+00
S04	tm16urhrvntpcs	4u	0.00E+00
S05	tm16urhrvntpcscfm	11u	0.00E+00
S06	tm16urhrvntpcscfmspb	11u	0.00E+00
S07	tm16ulpi	13u	0.00E+00
S08	tm16ulpivnt	13u	0.00E+00
S09	tm16ulpivntpcs	6u	0.00E+00
S10	tm16ulpivntpcscfm	13u	0.00E+00
S11	tm16udpr	11u	0.00E+00
S12	tm16udprrhr	11u	0.00E+00
S13	tm16udprrhrvnt	11u	0.00E+00
S14	tm16udprrhrvntpcs	4u	0 00E+00
S15	tm16udprrhrvntpcscfm	11u	0.00E+00
S16	tm16udprrhrvntpcscfm	11u	0.00E+00
S17	tm16udprhpi	11u	0.00E+00
S18	tm16udprhpirhr	11u	0.00E+00
S19	tm16udprhpirhrvnt	11u	0.00E+00
S20	tm16udprhpirhrvntpcs	4u	0.00E+00
S21	tm16udprhpirhrvntpcs	11u	0.00E+00
S22	tm16udprhpirhrvntpcs	11u	0.00E+00
S23	tm16udprhpidc	13u	0.00E+00
S24	tm16udprhpidcrhr	13u	0.00E+00
S25	tm16udprhpidcrhrvnt	13u	0.00E+00
S26	tm16udprhpidcrhrvntp	6u	0.00E+00
S27	tm16udprhpidcrhrvntp	13u	0.00E+00
S28	tm16udprhpidcrhrvntp	13u	0.00E+00
S29	tm16udprhpiicf	13u	0.00E+00
S30	tm16udprhpiicfvnt	13u	0.00E+00
S31	tm16udprhpiicfvntpcs	6u	0.00E+00
S32	tm16udprhpiicfvntpcs	13u	0.00E+00
S33	tm16udprhpiicfsf	13u	0.00E+00
S34	tm16udprhpiecf	10u	0.00E+00
S35	tm16uvecf	14u	0.00E+00
S36	tm16uvecfcfm	13u	0.00E+00
S37	tm16uvecfdpr	14p	0.00E+00
S38	tm16uvecfdprcfm	13p	0.00E+00

C:\ET\IMUSE\TM16U.EVT 1:27:00pm 5-15-94 NUPRA 2.1a WPPSS
Quantification Date: 5-15-94 9:23:57am TOTAL CMF = 3.46E-010

Anticipated Transient Without Scram - WNP-2
Containment Event Tree - Unisolated

Revision 1, April 7, 1994
(TMC-16, TCC-16, TFC-16, TIS23-10)
Figure 4.7-28



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
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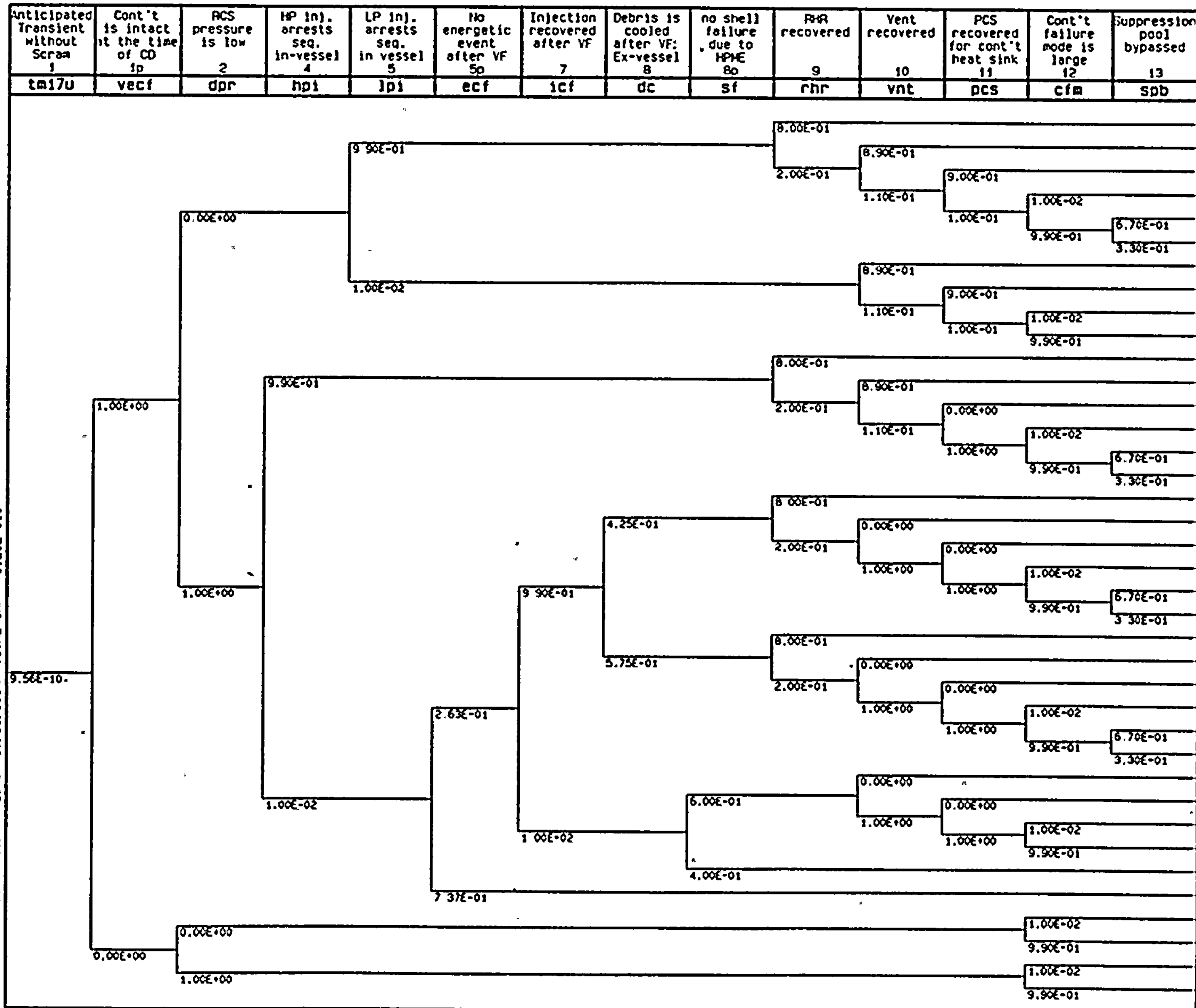
S03	tmc17rhrvnt	3	0.00E+00
S04	tmc17rhrvntpcs	4	0.00E+00
S05	tmc17rhrvntpcscfm	3	0.00E+00
S06	tmc17rhrvntpcscfmspb	3	0.00E+00
S07	tmc17lpi	5	0.00E+00
S08	tmc17lpivnt	5	0.00E+00
S09	tmc17lpivntpcs	6	0:00E+00
S10	tmc17lpivntpcscfm	5	0.00E+00
S11	tmc17dpr	1	2.52E-07
S12	tmc17dprrhr	3	5.60E-08
S13	tmc17dprrhrvnt	3	0.00E+00
S14	tmc17dprrhrvntpcs	4	0.00E+00
S15	tmc17dprrhrvntpcscfm	3	4.59E-09
S16	tmc17dprrhrvntpcscfmspb	3b	2.26E-09
S17	tmc17dprhpi	1	0.00E+00
S18	tmc17dprhpirhr	3	0.00E+00
S19	tmc17dprhpirhrvnt	3	0.00E+00
S20	tmc17dprhpirhrvntpcs	4	0.00E+00
S21	tmc17dprhpirhrvntpcs	3m	0.00E+00
S22	tmc17dprhpirhrvntpcs	3ms	0.00E+00
S23	tmc17dprhpidc	2	0.00E+00
S24	tmc17dprhpidcrhr	5	0.00E+00
S25	tmc17dprhpidcrhrvnt	5	0.00E+00
S26	tmc17dprhpidcrhrvntpc	6	0.00E+00
S27	tmc17dprhpidcrhrvntp	cfm	0.00E+00
S28	tmc17dprhpidcrhrvntp	cfm	0.00E+00
S29	tmc17dprhpiicf	5	0.00E+00
S30	tmc17dprhpiicfvnt	5	0.00E+00
S31	tmc17dprhpiicfvntpcs	6	0.00E+00
S32	tmc17dprhpiicfvntpcs	6m	4.96E-10
S33	tmc17dprhpiicfsf	9	3.34E-10
S34	tmc17dprhpiecf	10	2.29E-09
S35	tmc17vecf	14	0.00E+00
S36	tmc17vecfcfm	13	0.00E+00
S37	tmc17vecfdpr	14p	0.00E+00
S38	tmc17vecfdprcfm	13p	0.00E+00

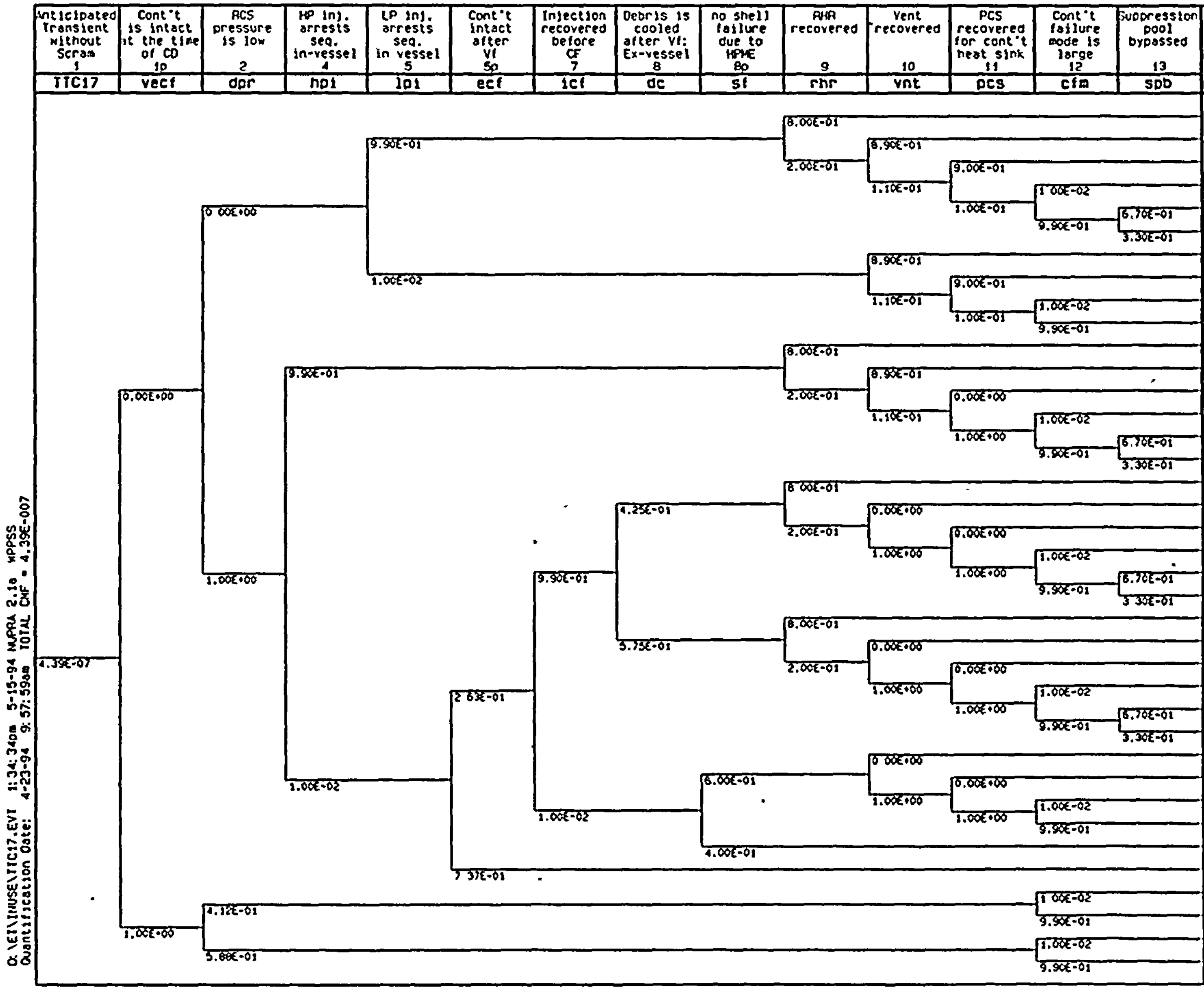
Anticipated Transient Without Scram - WNP-2
Containment Event Tree - Isolated

Revision 1, April 7, 1994, Figure 4.7-29
(TMC-17, TCC-17, TFC-17, TIC-11, TFC-20, THC-20, TCN-14, TOC-18, TSSW-12, TTSW-11, FLD6-11, FLD14-11, TCAS-11)



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
S01	tm17u	11u	0.00E+00
S02	tm17urhr	11u	0.00E+00
S03	tm17urhrvnt	11u	0.00E+00
S04	tm17urhrvntpcs	4u	0.00E+00
S05	tm17urhrvntpcscfm	11u	0.00E+00
S06	tm17urhrvntpcscfmspb	11u	0.00E+00
S07	tm17ulpi	13u	0.00E+00
S08	tm17ulpivnt	13u	0.00E+00
S09	tm17ulpivntpcs	6u	0.00E+00
S10	tm17ulpivntpcscfm	13u	0.00E+00
S11	tm17udpr	11u	7.57E-10
S12	tm17udprrhr	11u	1.69E-10
S13	tm17udprrhrvnt	11u	0.00E+00
S14	tm17udprrhrvntpcs	4u	0.00E+00
S15	tm17udprrhrvntpcscfm	11u	0.00E+00
S16	tm17udprrhrvntpcscfm	[illegible]	0.00E+00
S17	tm17udprhpi	11u	0.00E+00
S18	tm17udprhpirhr	11u	0.00E+00
S19	tm17udprhpirhrvnt	11u	0.00E+00
S20	tm17udprhpirhrvntpcs	4u	0.00E+00
S21	tm17udprhpirhrvntpcsc	[illegible]	0.00E+00
S22	tm17udprhpirhrvntpcsc	[illegible]	0.00E+00
S23	tm17udprhpidc	13u	0.00E+00
S24	tm17udprhpidcrhr	13u	0.00E+00
S25	tm17udprhpidcrhrvnt	13u	0.00E+00
S26	tm17udprhpidcrhrvntpc	6u	0.00E+00
S27	tm17udprhpidcrhrvntpc	[illegible]	0.00E+00
S28	tm17udprhpidcrhrvntpc	[illegible]	0.00E+00
S29	tm17udprhpiicf	13u	0.00E+00
S30	tm17udprhpiicfvnt	13u	0.00E+00
S31	tm17udprhpiicfvntpcs	6u	0.00E+00
S32	tm17udprhpiicfvntpcsc	[illegible]	0.00E+00
S33	tm17udprhpiicfsf	13u	0.00E+00
S34	tm17udprhpiecf	10u	0.00E+00
S35	tm17uvecf	14u	0.00E+00
S36	tm17uvecfcfm	13u	0.00E+00
S37	tm17uvecfdpr	14p	0.00E+00
S38	tm17uvecfdprcfm	13p	0.00E+00

Anticipated Transient Without Scram - WNP-2
Containment Event Tree - Unisolated
Revision 1, April 7, 1994
(TMC-17, TCC-17, TFC-17, TIS23-11, TFC-20, TMC-20)
Figure 4.7-30



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
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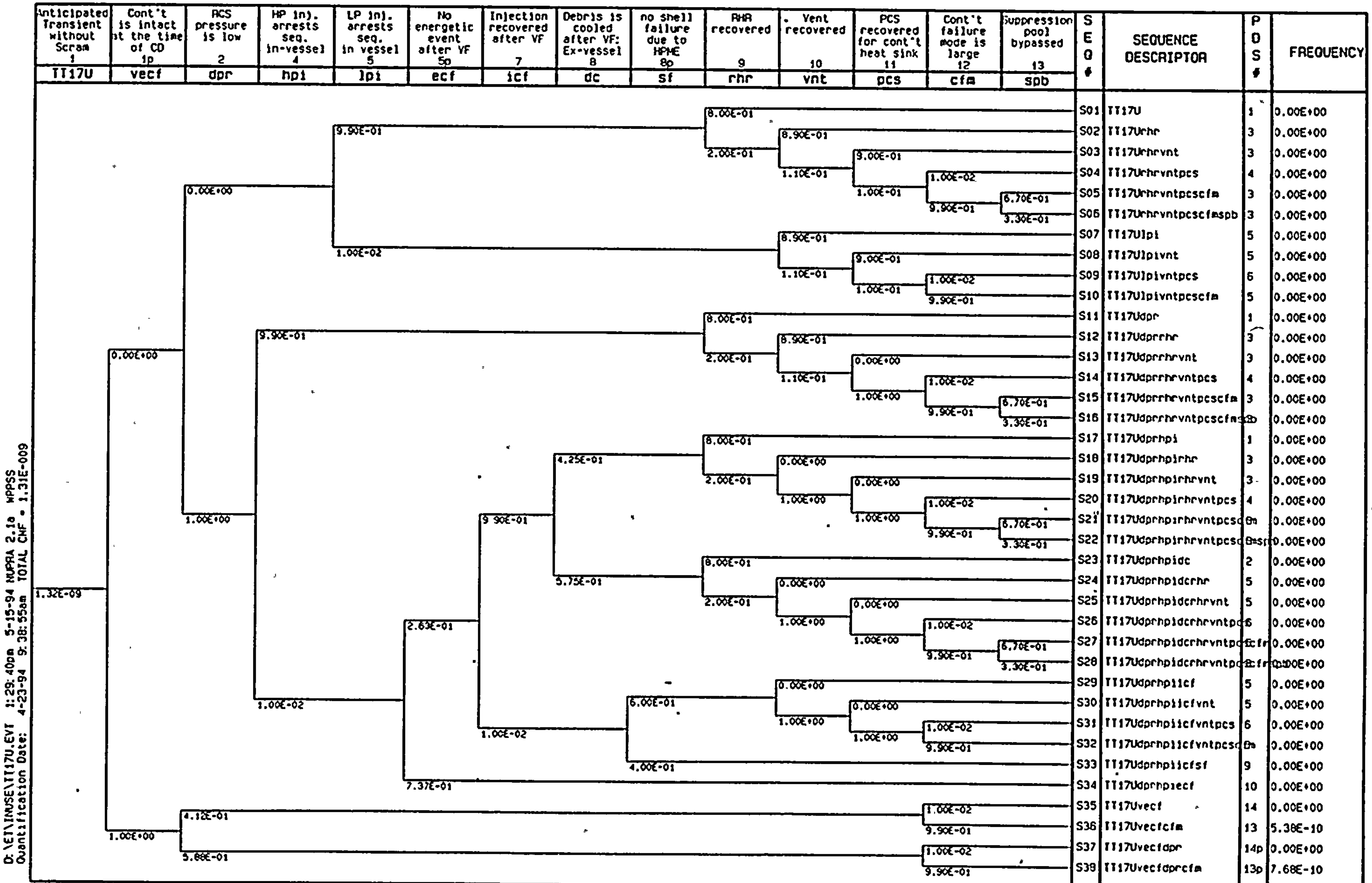
S02	TTC17rhr	3	0.00E+00
S03	TTC17rhrvnt	3	0.00E+00
S04	TTC17rhrvntpcs	4	0.00E+00
S05	TTC17rhrvntpcscfm	3	0.00E+00
S06	TTC17rhrvntpcscfmspb	3	0.00E+00
S07	TTC17lpi	5	0.00E+00
S08	TTC17lpivnt	5	0.00E+00
S09	TTC17lpivntpcs	6	0.00E+00
S10	TTC17lpivntpcscfm	5	0.00E+00
S11	TTC17dpr	1	0.00E+00
S12	TTC17dprrhr	3	0.00E+00
S13	TTC17dprrhrvnt	3	0.00E+00
S14	TTC17dprrhrvntpcs	4	0.00E+00
S15	TTC17dprrhrvntpcscfm	3	0.00E+00
S16	TTC17dprrhrvntpcscfmspb	[illegible]	0.00E+00
S17	TTC17dprhpi	1	0.00E+00
S18	TTC17dprhpirhr	3	0.00E+00
S19	TTC17dprhpirhrvnt	3	0.00E+00
S20	TTC17dprhpirhrvntpcs	4	0.00E+00
S21	TTC17dprhpirhrvntpcscfm	8	0.00E+00
S22	TTC17dprhpirhrvntpcscfmspb	8	0.00E+00
S23	TTC17dprhpidc	2	0.00E+00
S24	TTC17dprhpidcrhr	5	0.00E+00
S25	TTC17dprhpidcrhrvnt	5	0.00E+00
S26	TTC17dprhpidcrhrvntpcs	6	0.00E+00
S27	TTC17dprhpidcrhrvntpcscfm	[illegible]	0.00E+00
S28	TTC17dprhpidcrhrvntpcscfm	[illegible]	0.00E+00
S29	TTC17dprhpiicf	5	0.00E+00
S30	TTC17dprhpiicfvnt	5	0.00E+00
S31	TTC17dprhpiicfvntpcs	6	0.00E+00
S32	TTC17dprhpiicfvntpcscfm	8	0.00E+00
S33	TTC17dprhpiicfsf	9	0.00E+00
S34	TTC17dprhpiecf	10	0.00E+00
S35	TTC17vecf	14	1.81E-09
S36	TTC17vecfcfm	13	1.79E-07
S37	TTC17vecfdpr	14p	2.58E-09
S38	TTC17vecfdprcfm	13p	2.56E-07

Anticipated Transient Without Scram - WNP-2
Containment Event Tree
Revision 1, April 7, 1994
(TTC-17, TTC-16, TTC-20, AO-15)
Figure 4.7-31



Anticipated Transient Without Scram - WNP-2
Containment Event Tree
Revision 1, April 7, 1994
(TTC-17, TTC-16, TTC-20, AD-15)
Figure 4.7-32

4.8 Back End Results

4.8.1 Containment Damage States

The estimated frequencies for each of the damage states defined to characterize the performance of the WNP-2 containment are shown in the right hand columns of the initiator specific containment event tree trees presented in Figures 4.7-1 through 4.7-32. The accumulation of these results into a summary of the constituents of each individual containment damage state whose occurrence frequency is greater than 1.E-8/year is presented in Fig. 4.8-1 and Tables 4.8-1 and 2. Note, throughout this section of the report the terms "containment damage states (CDS)" and "source term groups (STG)" are considered synonymous and are used interchangeably.

These summary results show that the contribution from damage states CDS-1 and CDS-2, in which the containment remains intact, represents 38.9% to the total frequency. This therefore implies that the conditional frequency with which an environmental release of fission products can be expected for WNP-2 is estimated to be about 1.07E-5/year. Of the containment failures, the chances of late and early containment failure are about equal. In about 20% of the late containment failures, fission products will be scrubbed by the suppression pool and the release will be insignificant. In the early containment failures, the suppression pool will be bypassed. This is because drywell integrity could be lost as a result of failure of the penetrations or seals which serve as part of the drywell/wetwell boundary, or gross structural failure of the containment induced by severe overpressure from ex-vessel steam explosions.

Because the release frequency, 1.07E-5/year, represents the aggregate frequency for releases of different severity, the approximate individual constituents are outline below.

- 39% of the total release frequency (total = 1.07E - 5/year) is contributed by CDS-5.

  This damage state is characterized by an unscrubbed release of fission products through a relatively small breach in containment many hours after core melt, and generally follows a sequence in which the plant experiences a complete loss of all injection systems. This condition can result from:

  - station blackout
  - station blackout look-alike sequences in which loss of offsite power is accompanied by individual hardware failures in the HP injection systems
  - sequences in which LP injection is initially successful but increasing containment pressures result in reclosure of the depressurization valves, primary system repressurization and loss of LP injection.

The in-containment conditions which result in CDS-5 originate primarily with either a total loss of injection or debris which is in an uncoolable geometry. In both cases this leads to the presence of unquenched debris in the containment. When the effects from unavailability of the RHR, vent, and PCS systems to remove core decay heat, and noncondensable gases generated by molten core-concrete interactions are combined, the result is containment overpressures to a level which is sufficient to cause structural damage to the containment pressure boundary.

- 22% of the total release frequency (total = 1.07E - 5/year) is contributed by CDS-13P.

This damage state is characterized by an unscrubbed release of fission products through a small breach in containment, beginning at the time of core damage. Initially the release will be scrubbed, but after vessel failure the release will bypass the suppression pool and go directly to the environment. This damage state condition is typified by the TW sequence during which injection is successful but all viable means of containment heat removal are unavailable. Containment pressure continues to increase until it reaches the point at which it initiates a membrane tear in the vicinity of the wetwell horizontal stiffeners. The energetic release of steam from this location into the reactor building basement leads to consequential failure of all injection systems and core melt.

Individual TW sequences exhibit a great deal of similarity because sequence-to-sequence differences are small and tend to reflect the different ways in which the initiating events affect the availability of individual hardware systems whose operation is needed to maintain critical plant functions. The single sequence which exhibits marked difference is the one in which the core is initially cooled with main feedwater following a turbine trip ATWS. In this sequence it is assumed that feedwater will continue until containment pressure results in MSIV closure (54 psig) at which time, low reactor vessel level will initiate HPCS. Because there is an unusually high energy inventory in containment at the time of HPCS actuation, continued injection is expected to cause containment failure and consequential core melt.

When core cooling is assumed by HPCS, its limited ability to meet core cooling requirements following ATWS could introduce a high likelihood of inadequate core cooling and core melt. However, because of the uncertainty in the expected outcome and because core melt caused by containment failure is potentially much more severe than the case in which core melt renders the core subcritical and the probability of arresting the sequence in-vessel is very high, the analysis assumed the more conservative outcome.

Because the overall airborne fission product is the largest during the period immediately following vessel failure, transient sequences in which the containment fails prior to the blowdown which follows vessel breach at high primary system pressure have the potential for a significant release of fission products to the environment.

- 14% of the total release frequency (total = 1.07E - 5/year) contribution comes from CDS-9.

This damage state is characterized by a release through a small breach in containment beginning at or near the time of vessel failure. This damage state condition results from sequences which are initiated by a loss of offsite power and progress to long-term station blackout or look-alike sequences in which a source of HP injection substitutes for a third failed diesel generator. (Note: Long-term station blackout involves battery depletion and total loss of DC power, whereas during a short-term SBO, DC power remains available.)

The dominant containment failure mode for CDS-9 results from a situation in which a rapidly flowing jet of hot debris impacts the shell and causes early containment failure.

- 10% of the total release frequency (total = 1.07E - 5/year) contribution comes from CDS-3.

This damage state is characterized by a scrubbed release of fission products through a small breach in containment which occurs several hours after the time of vessel failure. This damage state condition results from sequences which are initiated by a loss of offsite power and progress to long-term or short-term station blackout sequences.

In this damage state, the debris is cooled following the recovery of injection or containment spray. The water pool above the debris and the suppression pool are available for fission product scrubbing. Containment fails as a result of overpressurization which follows a complete loss of long term containment heat removal capability.

- 9.7% of the total release contribution (total = 1.07E - 5/year) comes from CDS-10.

This damage state is characterized by a catastrophic failure of containment and a release of unscrubbed fission products at or about the time of vessel failure.

The dominant cause of this type of release is expected to be a high pressure transient sequence in which a failure of injection occurs before vessel failure. There is a pre-existing accumulation of water in the pedestal cavity as a result of operation of the drywell sprays so that when the melt is forcefully ejected at high pressure there is an ensuing steam explosion. This explosion is postulated to result in severe overpressure in the cavity, collapse of the pedestal and gross structural failure of containment.

The corresponding source terms for each of these dominant containment damage states are presented in Section 4.8.2.

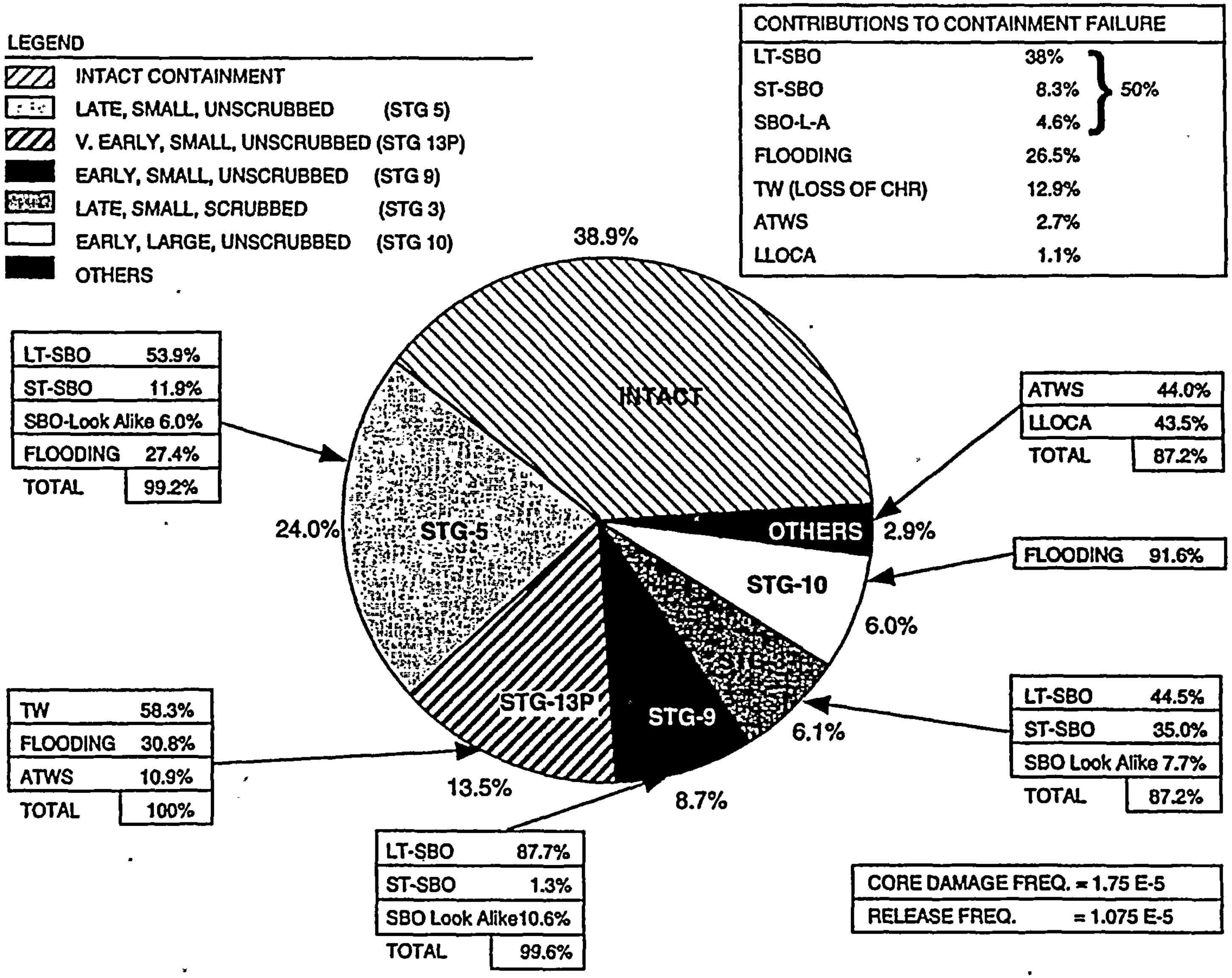


FIGURE 4.8-1 CONTRIBUTORS TO CONTAINMENT FAILURE AND RELEASE

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TABLE 4.8-1
Summary of Containment Damage State or Source Term Group Frequencies.

Containment Damage State or Source Term Group		Frequency (per year)	Fraction
1	Fission products scrubbed * Containment Intact	4.85E-006	27.8
2	Fission products not scrubbed * Containment Intact	1.94E-006	11.1
others	Containment Failed	1.07E-005	61.1

TABLE 4.8-2
Summary of Containment Damage State or Source Term Group Frequencies for Failed Containment

Containment Damage State or Source Term Group		Frequency (per year)	Fraction of the total release
5	Late CF * small CFM * fission products not scrubbed	4.19E-006	39.1%
13P	Very Early CF * small CFM * fission products not scrubbed	2.35E-006	21.9%
9	Early CF * small CFM * fission products not scrubbed	1.52E-006	14.2%
3	Late CF * small CFM * fission products scrubbed	1.07E-006	10.%
10	Early CF * large CFM * fission products not scrubbed	1.04E-006	9.7%
13U	Early CF * small CFM * Fission products not scrubbed	1.89E-007	1.76%
13	Early CF * small CFM * Fission products not scrubbed	1.88E-007	1.76%
11U	Early CF * small CFM * Fission products scrubbed	3.35E-008	<1%
17	Early CF * large CFM * Fission products not scrubbed	2.99E-008	<1%
6	Late CF * large CFM * Fission products not scrubbed	2.92E-008	<1%
14P	Early CF * large CFM * Fission products not scrubbed	2.37E-008	<1%

Note: The fractional contribution identified for each CDS represents the fractional contribution to total release frequency (i.e., 61% of the total CDF).

TABLE 4.8-3
Contribution of Frequency to the Source Term Group from Level 1 Dominant Sequences.

STG-5:

CET	Frequency (per year)
TE-15	1.16E-6
FLD7-3	1.13E-6
TE17A	1.01E-6
TE-19	4.99E-7
TE-3	2.53E-7
TE17B	8.97E-8
FLD7-2	1.95E-8
TE-9	1.47E-8
TCN-11	6.25E-9
TE-13	3.05E-9
TE-11	1.25E-9
TMC-16	1.06E-9
TT-19	7.75E-10
TMC-17	4.96E-10

Total: 4.19E-6

STG-13p:

CET	Frequency (per year)
TW	1.37E-6
FLDTW	7.23E-7
TTC17	2.56E-7

Total: 2.35E-6

TW consists of TC-3, TCAS-2, TCN-3, TDC-5, TI-8, TF-4, TM-4, TSSW-4, TT-5, TTSW-2, and MS-5.

FLDTW consists of FLD14-2 and FLD6-2.

TTC17 consists of TTC-17, TTC-20, TTC-16, AND AO-15.

STG-9:

CET	Frequency (per year)
TE-15	6.81E-7
TE-17A	6.02E-7
TE-3	1.61E-7
TE-17B	5.05E-8
TE-19	2.05E-8
TE-9	8.77E-9
TE-13	1.52E-9
TMC-17	3.34E-10

Total: 1.52E-6

STG-3:

CET	Frequency (per year)
TE-19	3.75E-7,
TE17-A	3.52E-7,
TE-15	1.13E-7,
TE17-B	1.10E-8,
TE-3	8.26E-8,
TMC-17	6.29E-8,
TMC-16	2.33E-8,
TCN-11	9.98E-9,
FLD7-2	3.11E-8,
TE-13	2.32E-9,
TT-19	1.37E-9,
TE-9	1.28E-9,
TE-11	1.01E-9,

Total: 1.07E-6

STG-10: CET Frequency (per year) FLD7-2 9.53E-7, TT-19 4.36E-8, TCN-11 4.59E-8, TMC-17 2.29E-9, **Total: 1.04E-6**	**STG-13u:** CET Frequency (per year) LL 1.47E-7 Others 4.20E-8 **Total: 1.89E-7**
STG-13: CET Frequency (per year) TTC-17 1.79E-7 S1-3 8.95E-9 **Total: 1.88E-7**	**STG-17:** CET Frequency (per year) LL 2.99E-8 **Total: 2.99E-8**

4.8.2 Source Term Analysis

Based on the contribution to the source term group, a representative accident sequence in Table 4.8-2 was selected to represent each group. MAAP analysis was performed for that particular accident sequence to determine the radionuclide source term.

The source terms are provided in terms of containment release fractions (FREL) of fission product species. This factor is defined as the fraction of radioactive material leaks to the environment (reactor building) to the initial inventory. In the MAAP code, the fission products are grouped into twelve species as seen in Table 4.8-4. Table 4.8-4 also lists the radionuclide inventory at the beginning of an accident.

During the heat-up and meltdown phase, the high volatile fission products such as noble gases, CsI and CsOH would be released from the core nearly completely. Tellurium, strontium, barium, lanthanum and cerium would be released later during core-concrete interactions. About 100% of the noble gases is expected to escape from the containment in all the containment damage states. CsI, CsOH, and tellurium are volatile and large fractions could be released to the environment in cases with suppression pool bypass. Barium, strontium, lanthanum, and cerium groups are substantially less volatile than tellurium. Therefore, the release fractions will be substantially smaller. The release fraction for specie 12 (UO_2 + NpO_2 + PuO_2) is not presented since it is negligible.

The calculated containment release fractions (FREL) versus time for each source term group are presented in Figs. 4.8-2 to 4.8-16 from time 0 (start of the accident) to 3.5 days. The following insights are gained based on the results:

> STG-10 represents the most severe release since a catastrophic containment failure occurred immediately after vessel breach. The large amount of airborne fission products would escape rapidly to the environment. The predicted release fraction of CsI (or CsOH) is about 0.44 at or around the time of containment failure.
>
> The fractional release of radioactive material in STG-3 is benign due to the suppression pool scrubbing effect. The calculated release fraction of CsI is less than 5E-4.
>
> In STG-5, vessel breach occurred at around 17 hrs. Due to the extended time required before containment failure, most of the aerosols are predicted to settle on the containment. Therefore, the release fractions following containment failure are small (~0.05 for CsI). However, since the drywell temperature stays high because of the presence of hot debris, the radioactive material deposited on surfaces in the vessel would be reevolved and result in a large late release at around 60 hrs.

Immediately following containment failure, the release characteristics of STG-13p and 9 are similar. The major difference between these two groups is in the release timing. The release of STG-13p started at 38 hours after the accident and that of STG-9 started at 18 hours. Also, in STG-13p, suppression pool flashes due to high pool temperature. Therefore, the containment pressure of STG-13p is higher than that of STG-9. This results in higher late releases.

TABLE 4.8-4
Fission Product Species and Initial Inventory.

Group	Fission Products	Inventory, lbm
1	Nobles and inert aerosols (Xe, Kr)	1,257E3
2	CsI + RbI	1.162E2
3	TeO_2	0.0*
4	Sr0	2.152E2
5	$Mo0_2$	9.636E2
6	CsOH + RbOH	7.475E2
7	Ba0	3.419E2
8	$La_2O_3 + Pr_2O_3 + Nd_2O_3 + Sm_2O_3 + Y_2O_3$	1.838E3
9	$Ce0_2$	7.486E2
10	Sb	7.918E0
11	Te_2	1.124E2
12	$UO_2 + NpO_2 + PuO_2$	3.531E5

* Since the initial mass of TeO_2 is zero, the release fraction of TeO_2, FREL(3), is calculated based on a total mass of 140.5 lb, which is converted from the initial mass of Te_2 (140.5 lb = 112.4 lb / 128 g/mole x 160 g/mole).

WNP-2: STG-5 (TE-15)

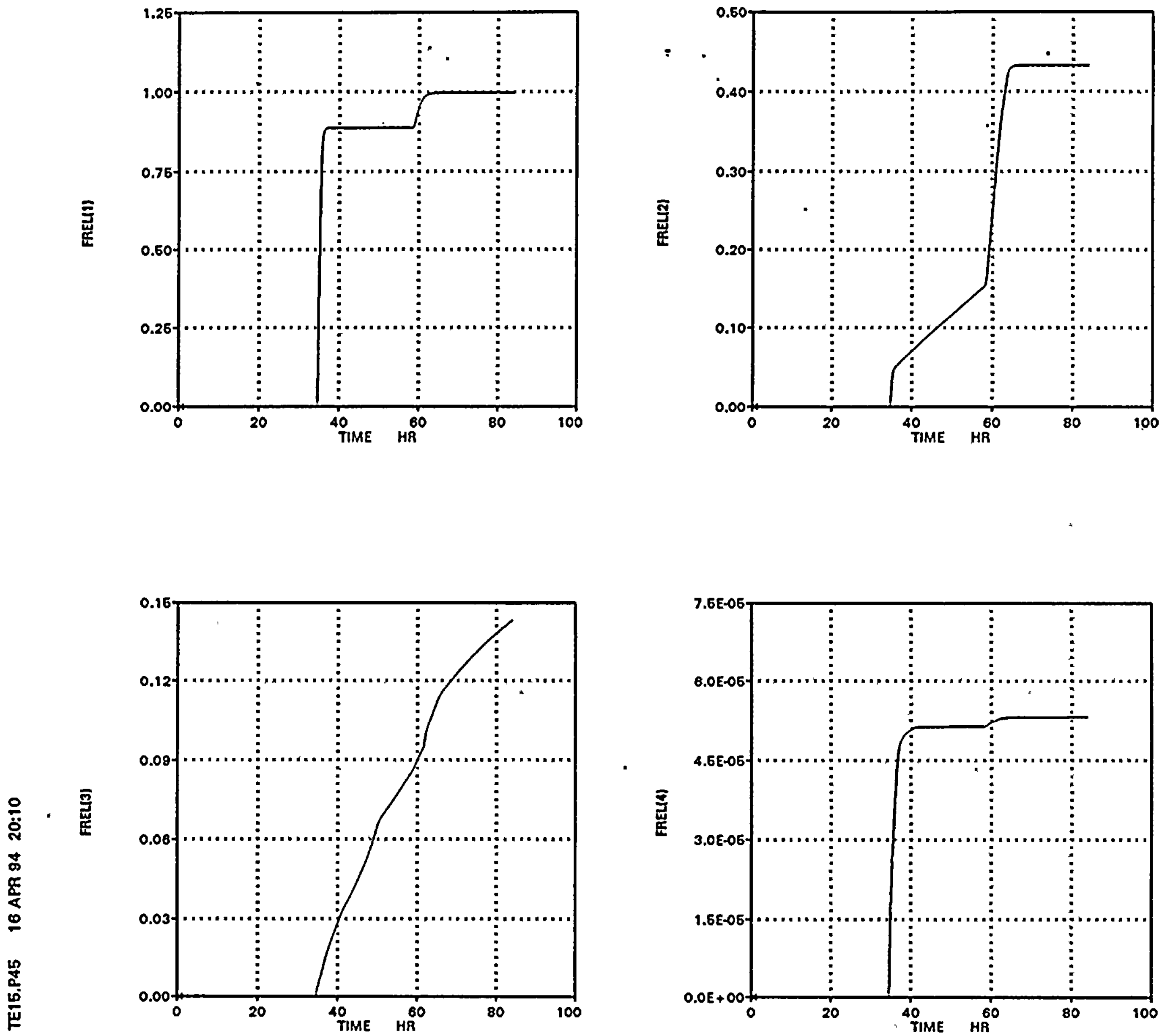


Figure 4.8-2 Release Fractions for Source Term Group 5 (1 - Nobles, 2 - CsI + RbI, 3 - TeO_2, 4 - SrO).

WNP-2: STG-5 (TE-15)

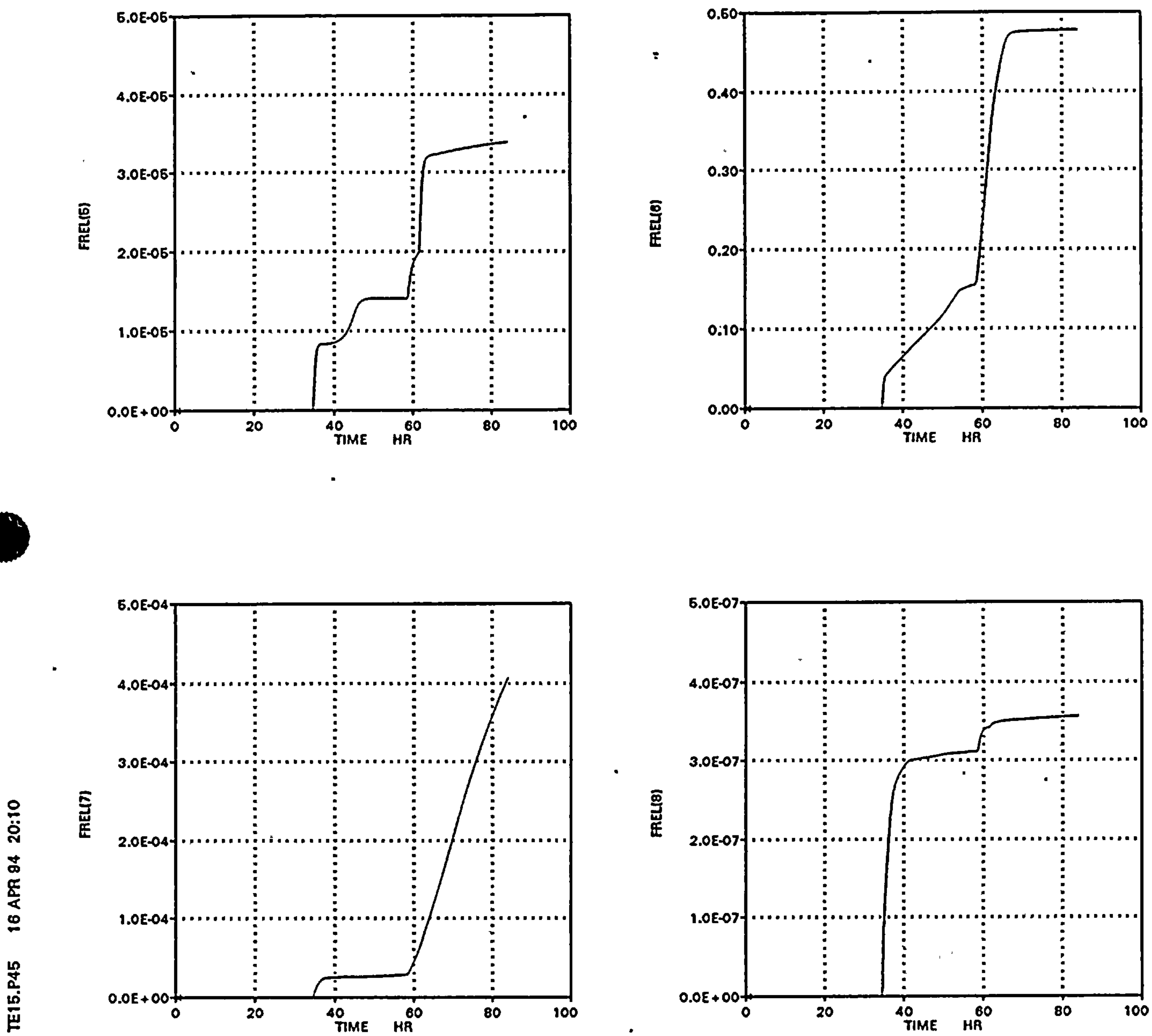


Figure 4.8-3 Release Fractions for Source Term Group 5 (5 - MoO_2, 6 - CsOH + RbOH, 7 - BaO, 8 - La_2O_3 + Pr_2O_3 + Nd_2O_3 + Sm_2O_3 + Y_2O_3).

WNP-2: STG-5 (TE-15)

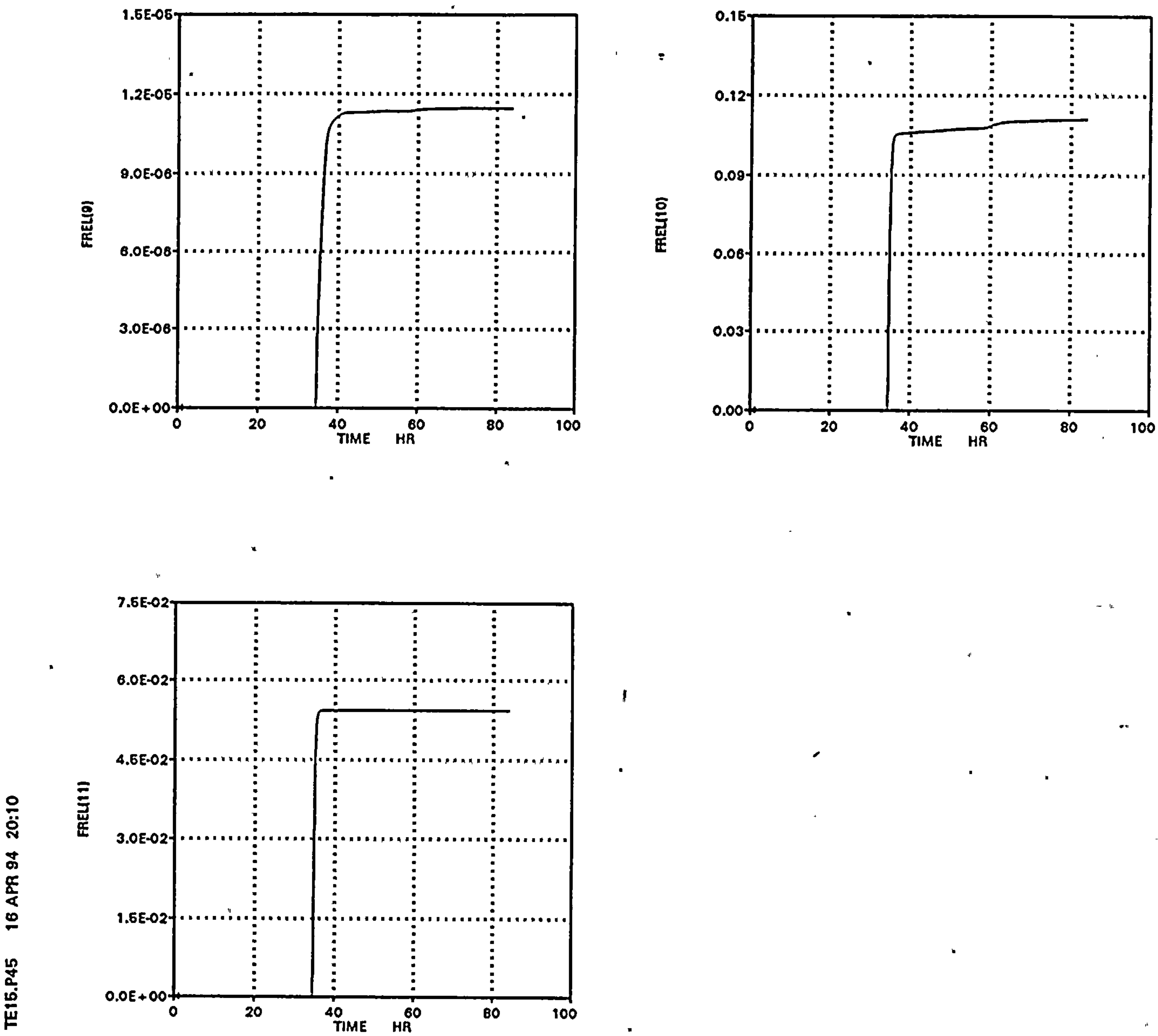


Figure 4.8-4 Release Fractions for Source Term Group 5 (9 - CeO_2, 10 - Sb, 11 - Te_2).

WNP-2: STG-13p (TW)

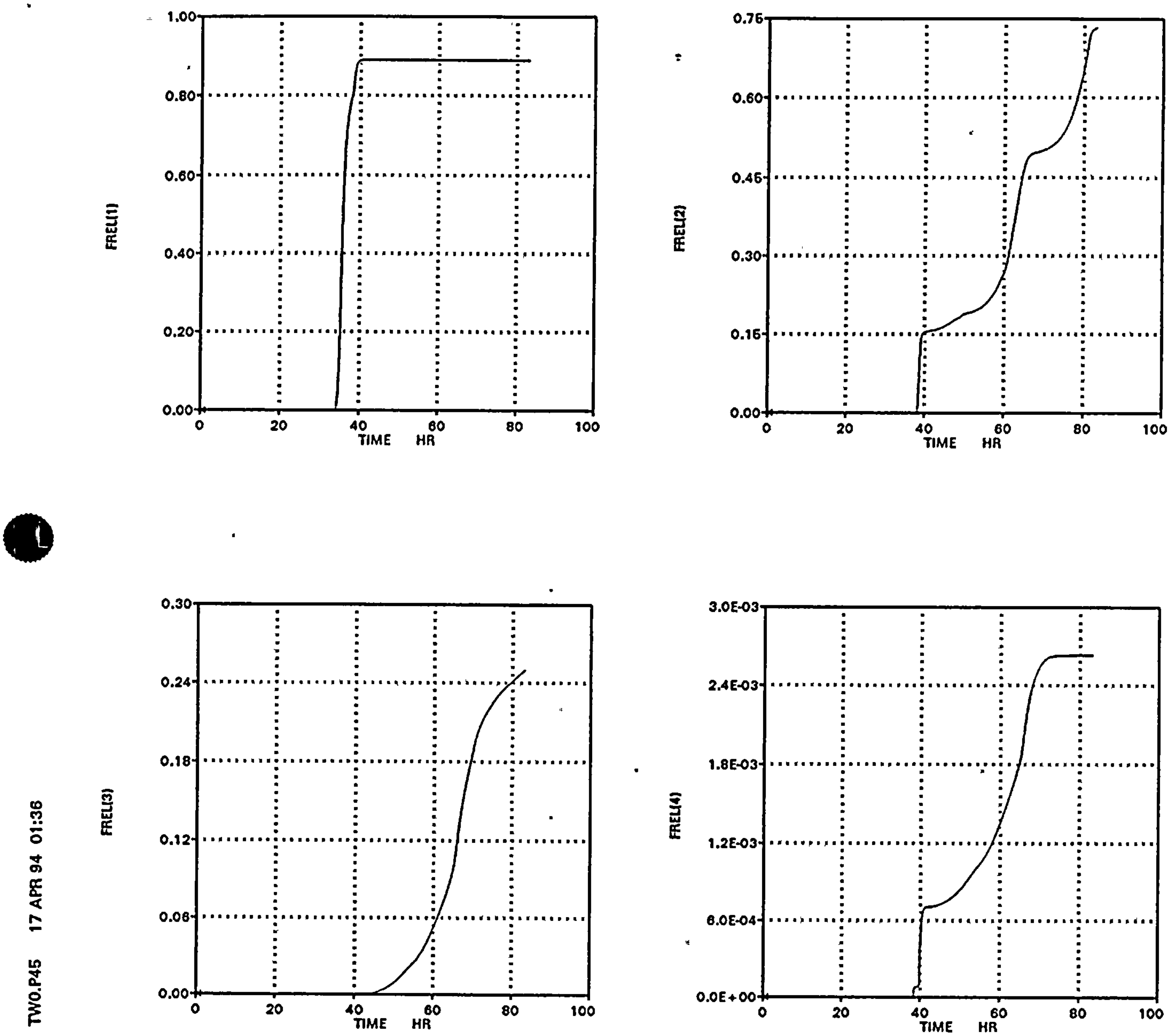
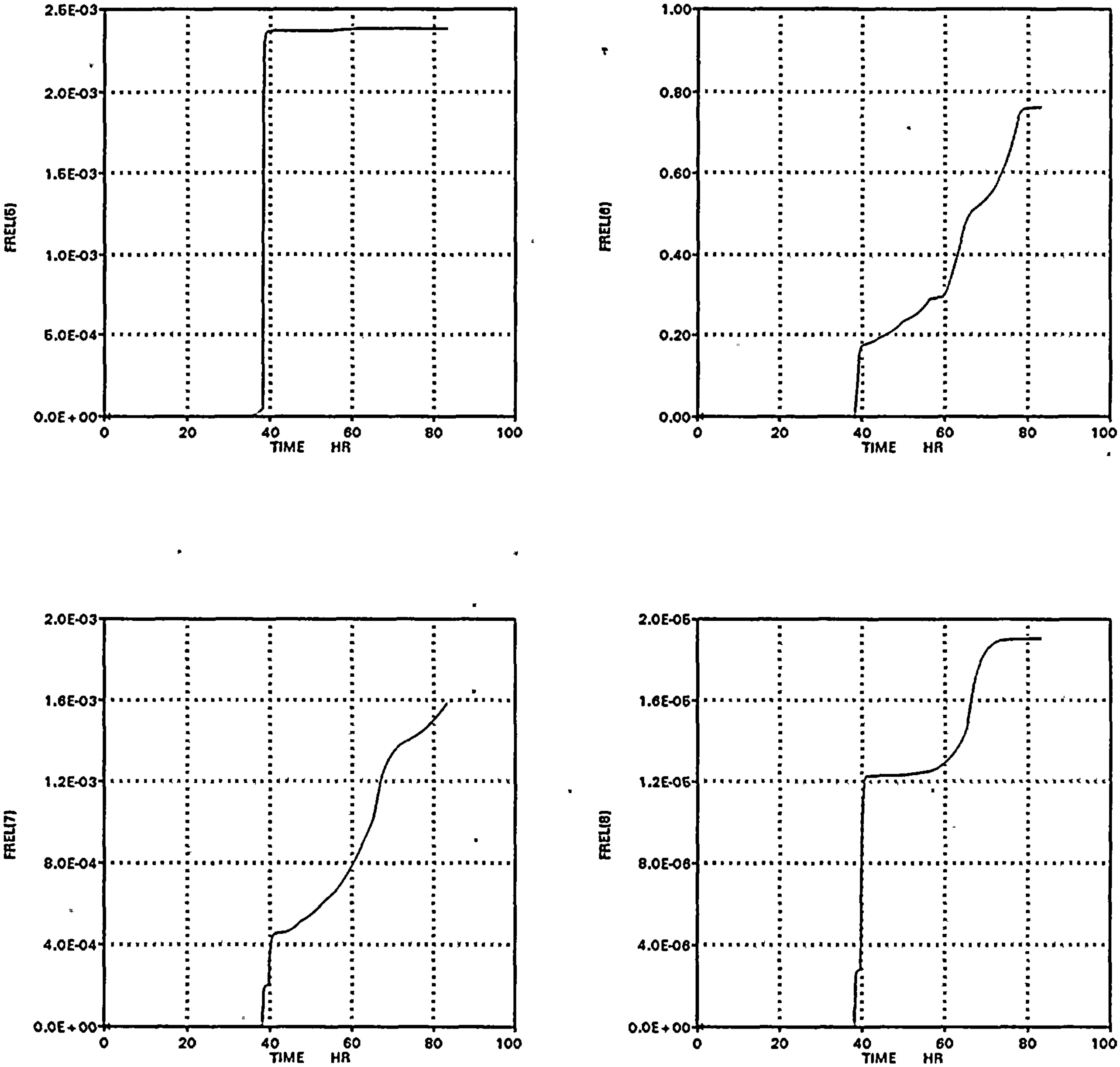


Figure 4.8-5 Release Fractions for Source Term Group 13p (1 - Nobles, 2 - CsI + RbI, 3 - TeO_2, 4 - SrO).

WNP-2: STG-13p (TW)



TW0.P45 17 APR 94 01:36

Figure 4.8-6 Release Fractions for Source Term Group 13p (5 - MoO_2, 6 - CsOH + RbOH, 7 - BaO, 8 - La_2O_3 + Pr_2O_3 + Nd_2O_3 + Sm_2O_3 + Y_2O_3).

WNP-2: STG-13p (TW)

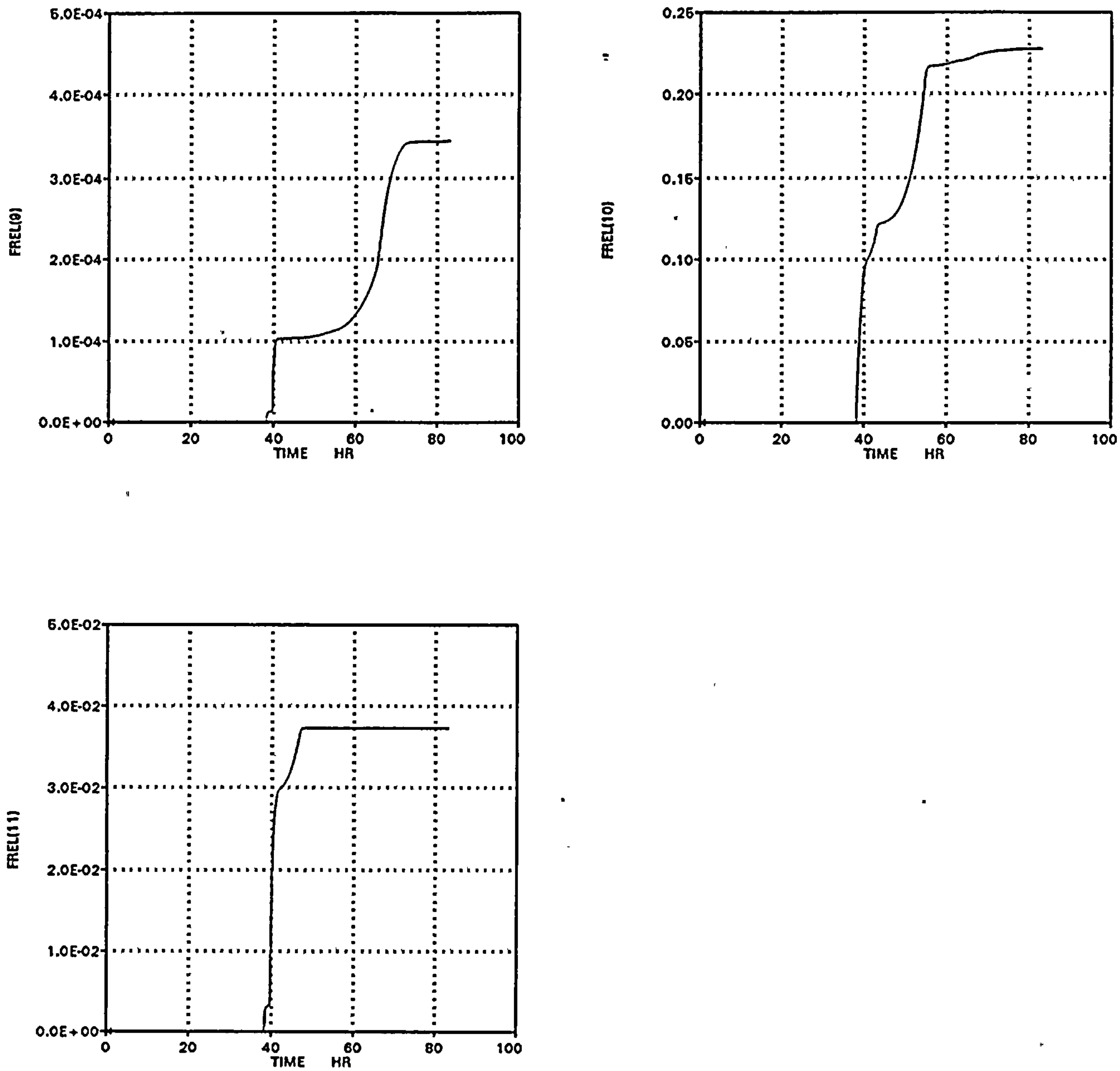


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Figure 4.8-7 Release Fractions for Source Term Group 13p (9 - CeO_2, 10 - Sb, 11 - Te_2).

WNP-2: STG-9 (TE-15)

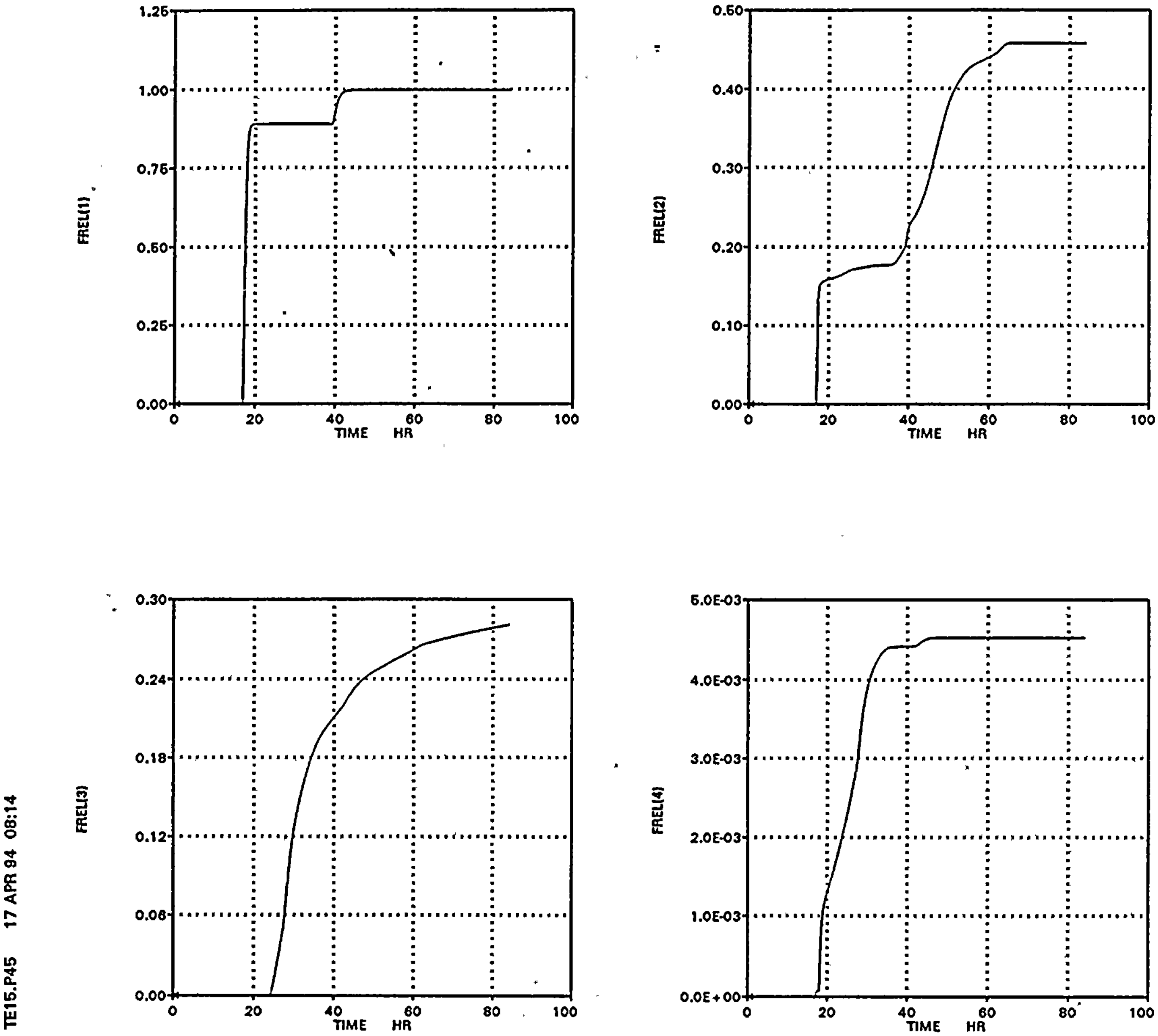

Figure 4.8-8 Release Fractions for Source Term Group 9 (1 - Nobles, 2 - CsI + RbI, 3 - TeO_2, 4 - SrO).

WNP-2: STG-9 (TE-15)

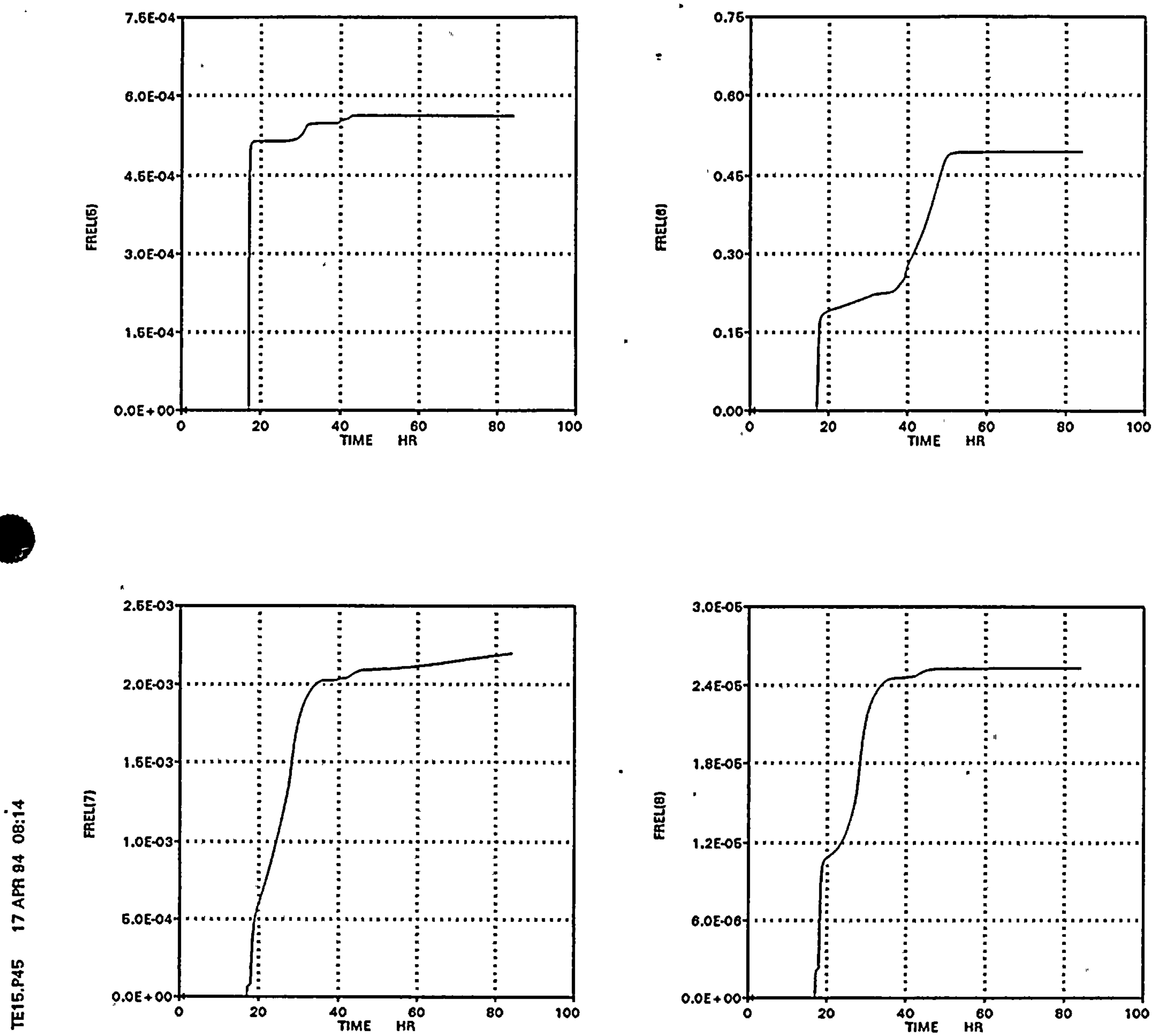

Figure 4.8-9 Release Fractions for Source Term Group 9 (5 - MoO_2, 6 - CsOH + RbOH, 7 - BaO, 8 - La_2O_3 + Pr_2O_3 + Nd_2O_3 + Sm_2O_3 + Y_2O_3).

WNP-2: STG-9 (TE-15)

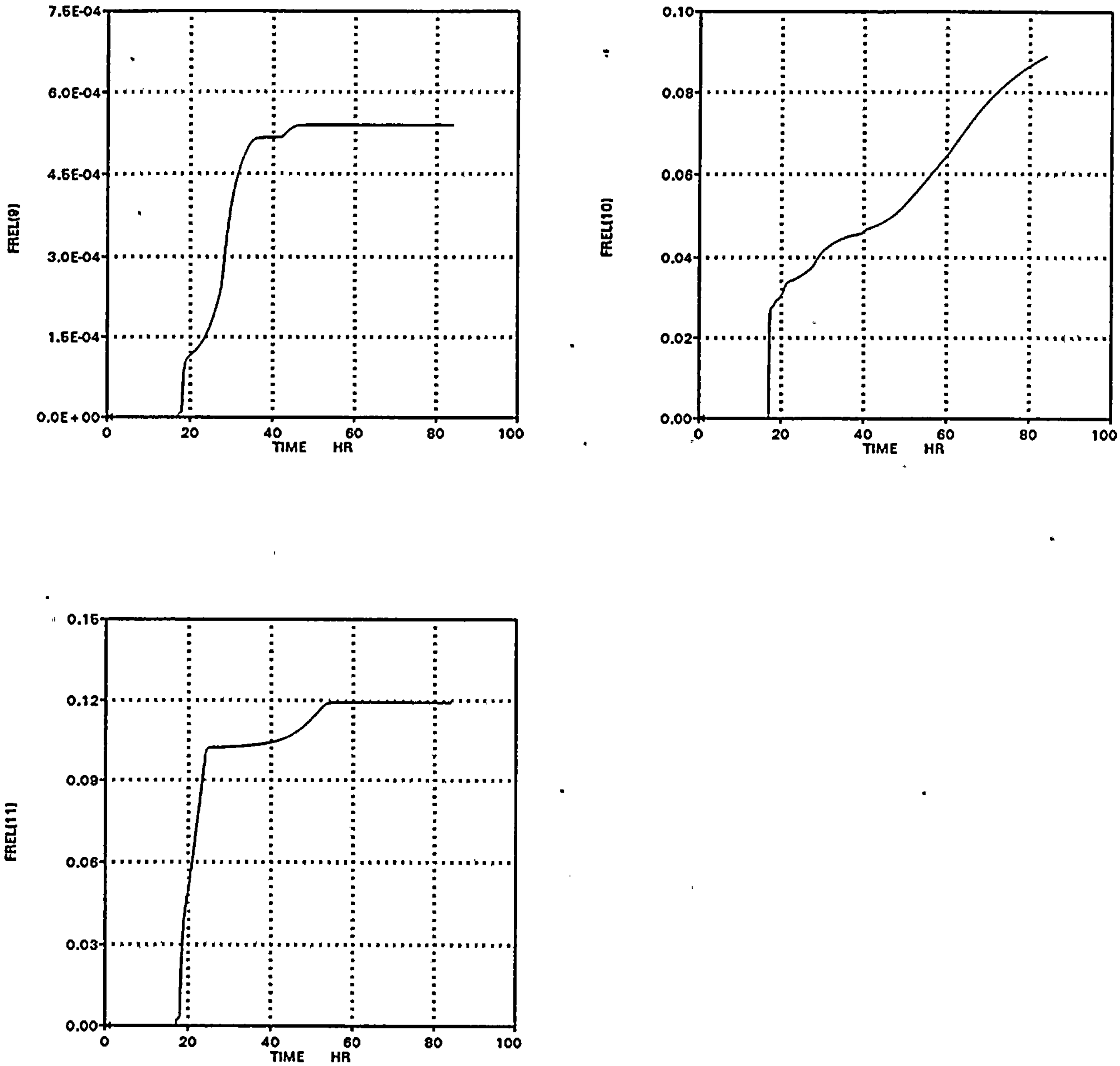

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Figure 4.8-10 Release Fractions for Source Term Group 9 (9 - CeO_2, 10 - Sb, 11 - Te_2).

WNP-2: STG-3 (TE-19)

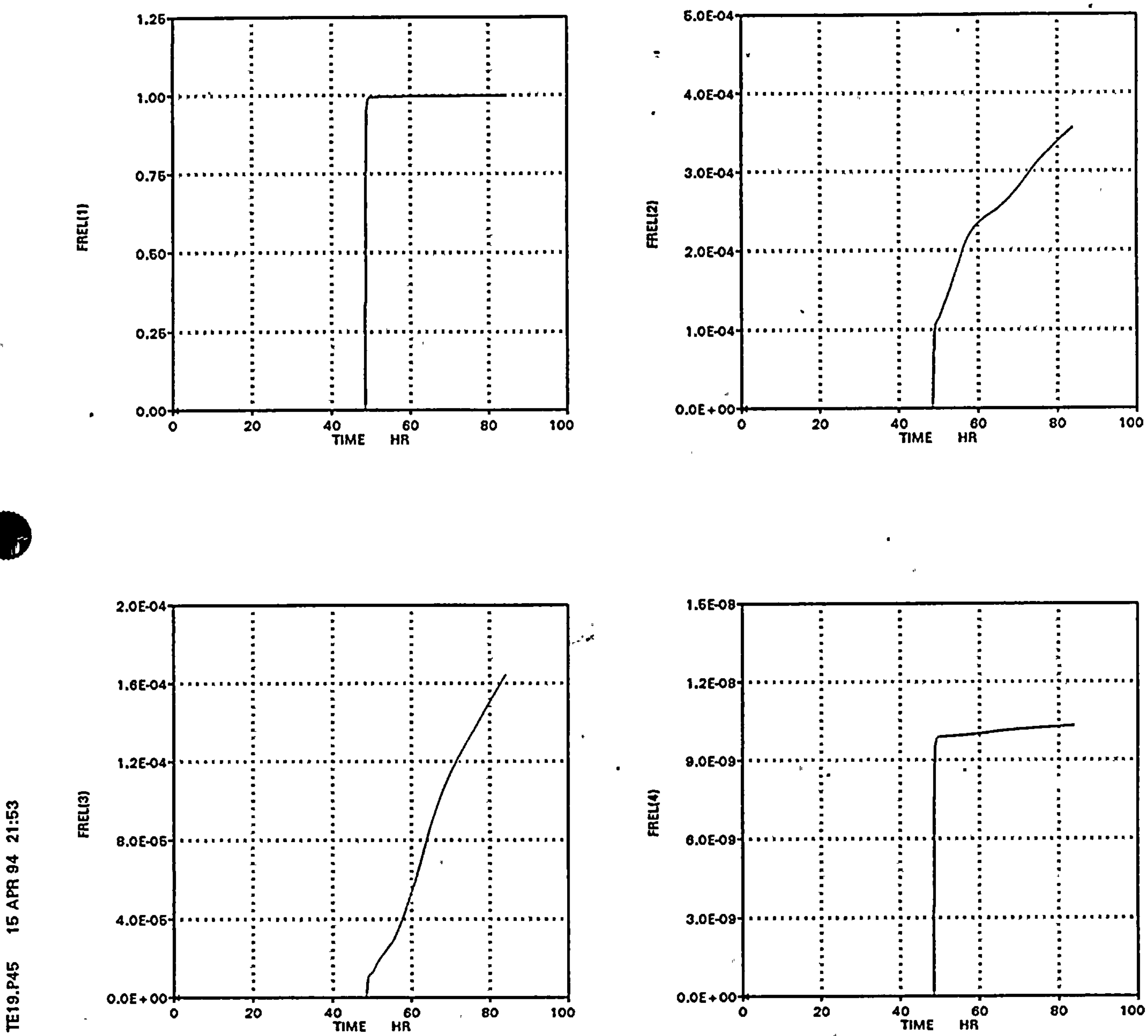


Figure 4.8-11 Release Fractions for Source Term Group 3 (1 - Nobles, 2 - CsI + RbI, 3 - TeO_2, 4 - SrO).

WNP-2: STG-3 (TE-19)

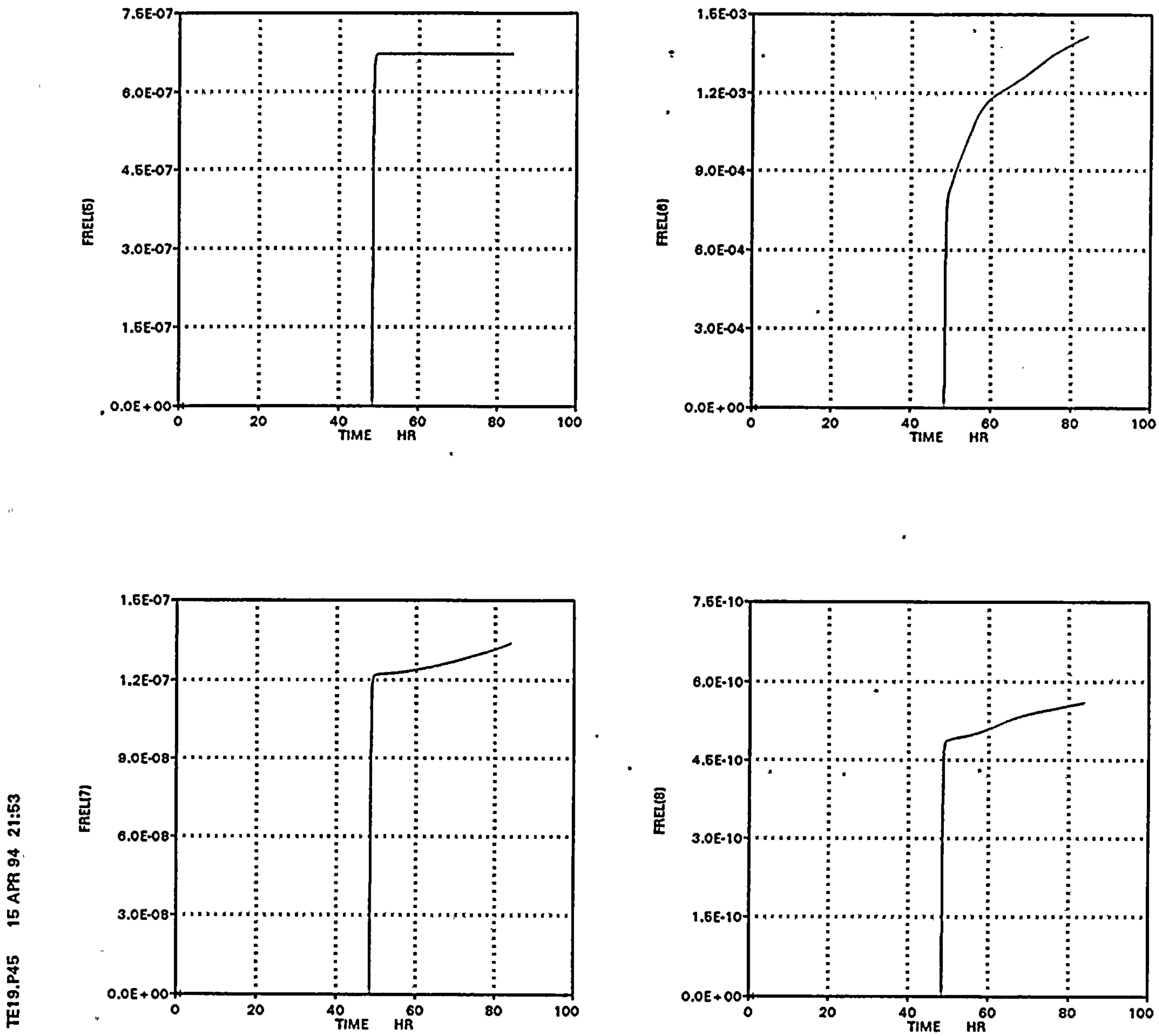


Figure 4.8-12 Release Fractions for Source Term Group 3 (5 - MoO_2, 6 - CsOH + RbOH, 7 - BaO, 8 - La_2O_3 + Pr_2O_3 + Nd_2O_3 + Sm_2O_3 + Y_2O_3).

WNP-2: STG-3 (TE-19)

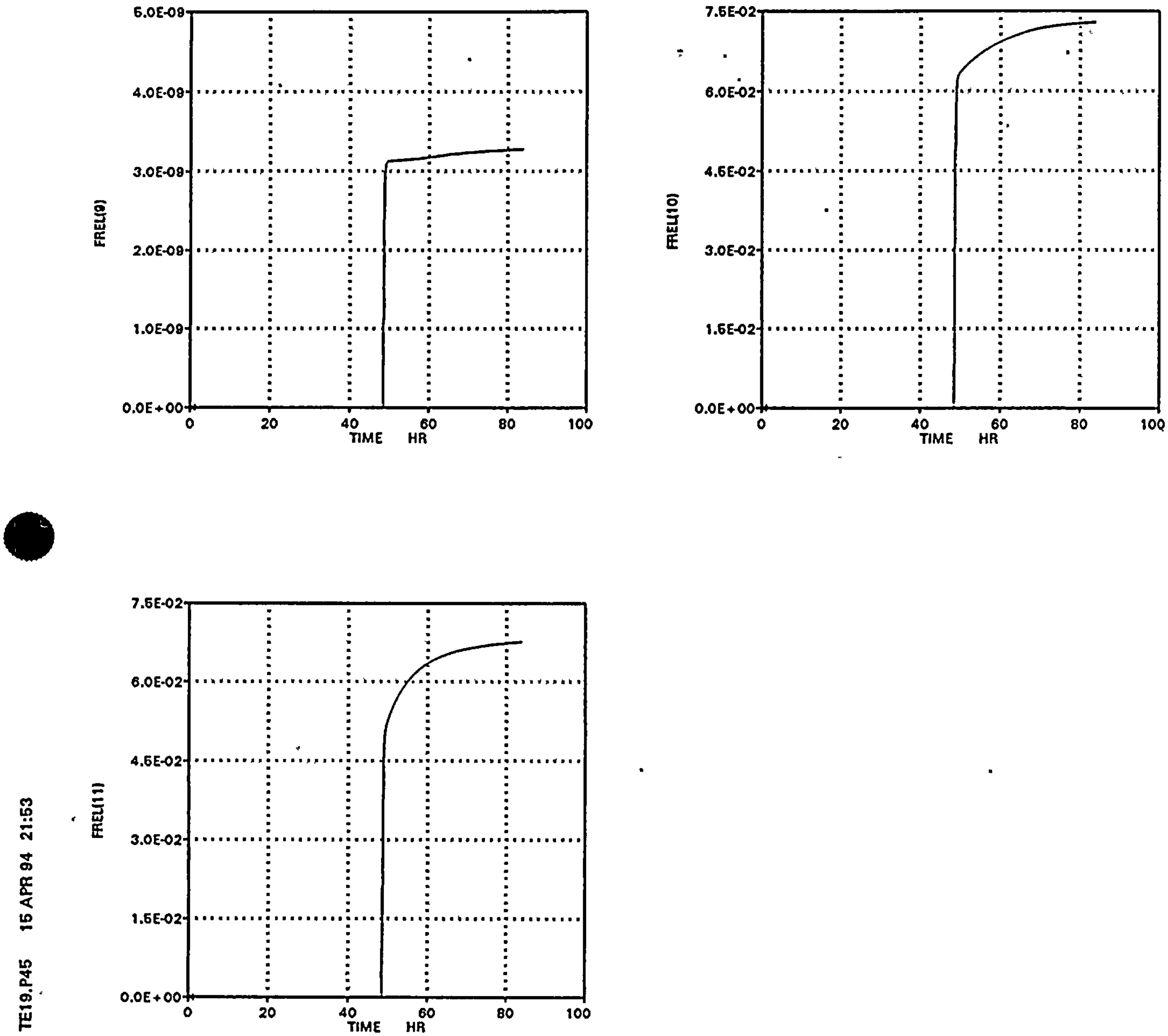


Figure 4.8-13 Release Fractions for Source Term Group 3 (9 - CeO_2, 10 - Sb, 11 - Te_2).

WNP-2: STG-10 (FLD7-2)

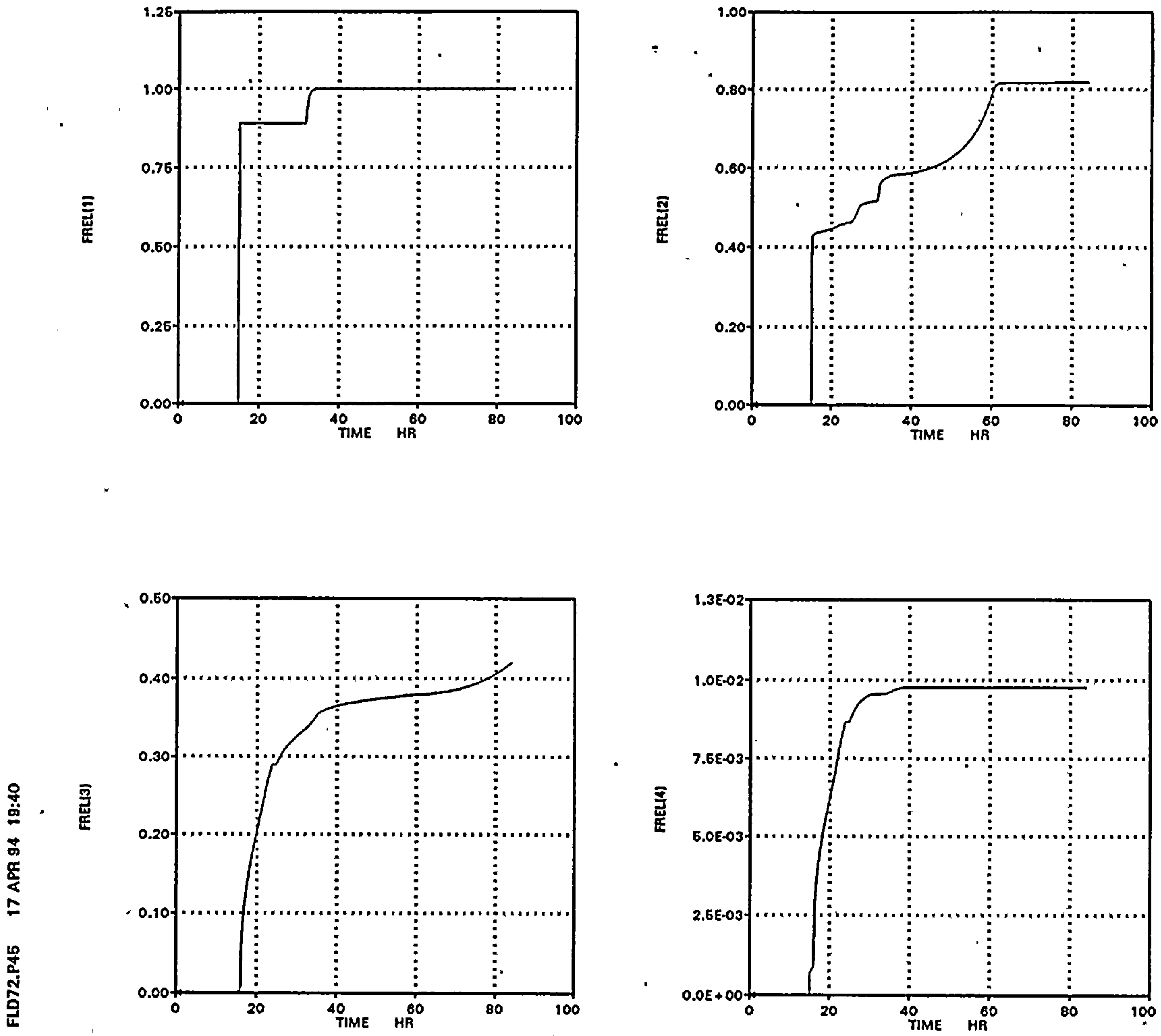


Figure 4.8-14 Release Fractions for Source Term Group 10 (1 - Nobles, 2 - CsI + RbI, 3 - TeO_2, 4 - SrO).

WNP-2: STG-10 (FLD7-2)

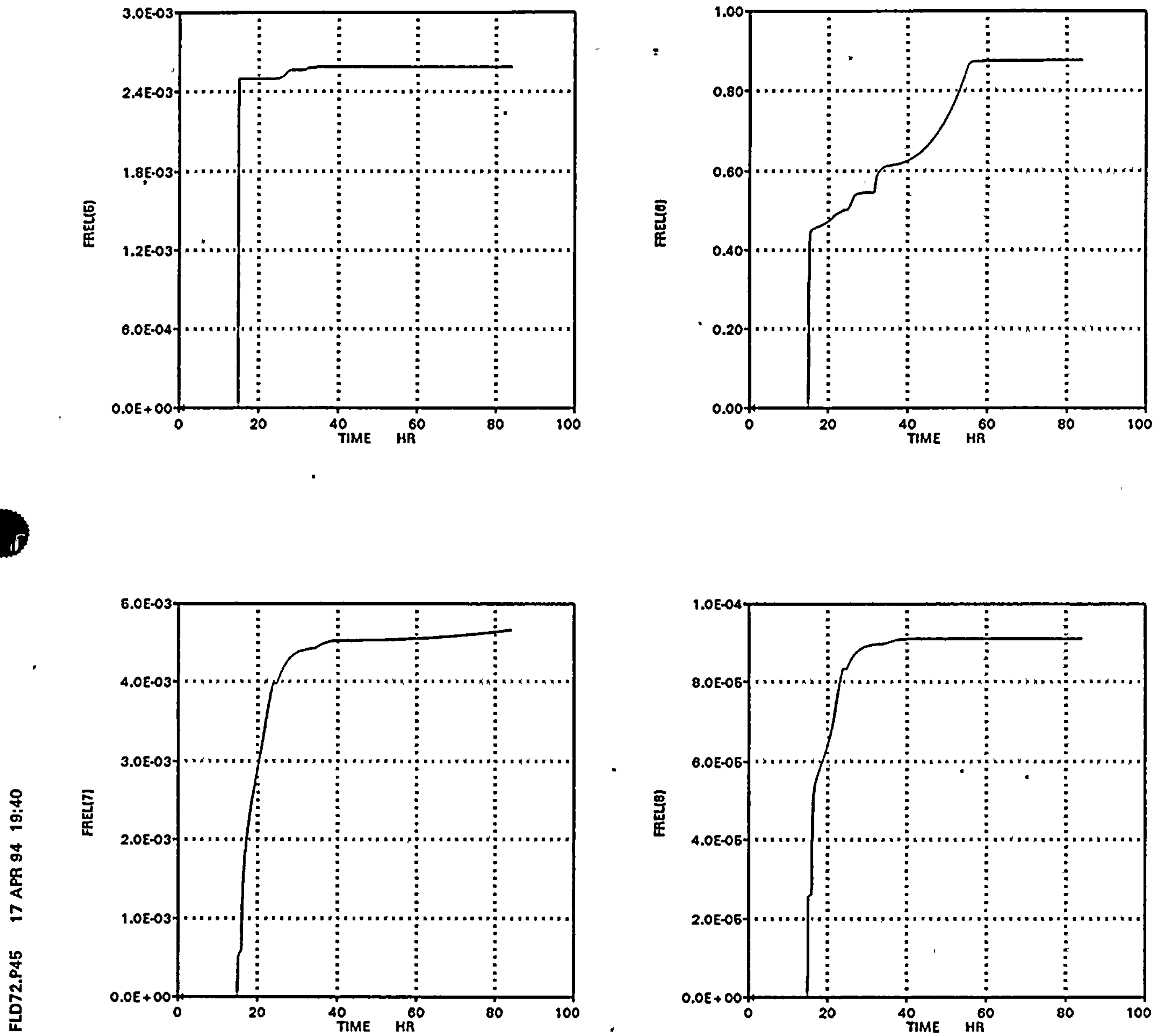
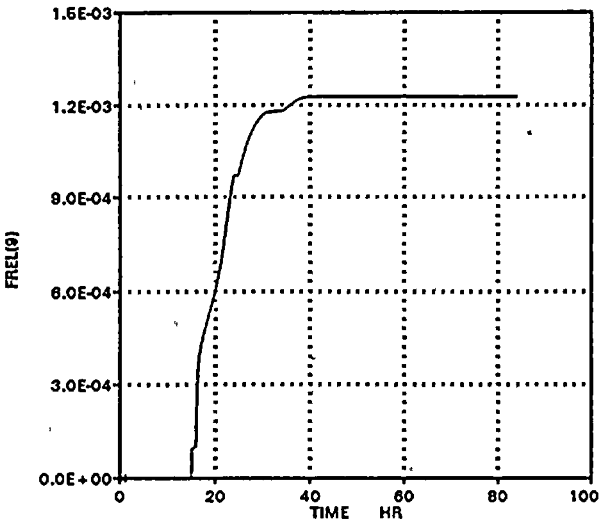
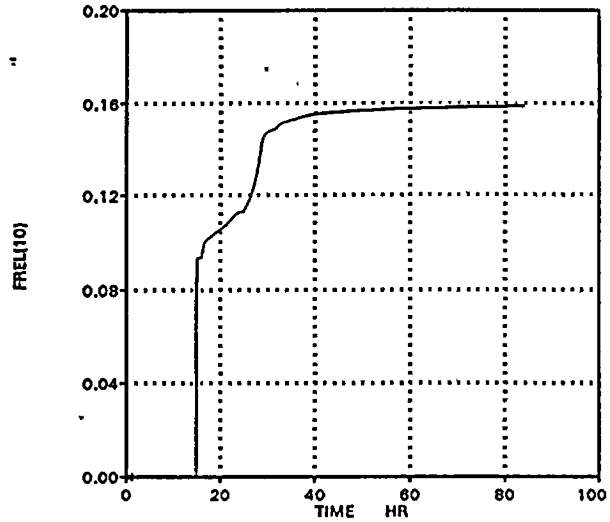


Figure 4.8-15 Release Fractions for Source Term Group 10 (5 - MoO_2, 6 - CsOH + RbOH, 7 - BaO, 8 - La_2O_3 + Pr_2O_3 + Nd_2O_3 + Sm_2O_3 + Y_2O_3).

WNP-2: STG-10 (FLD7-2)





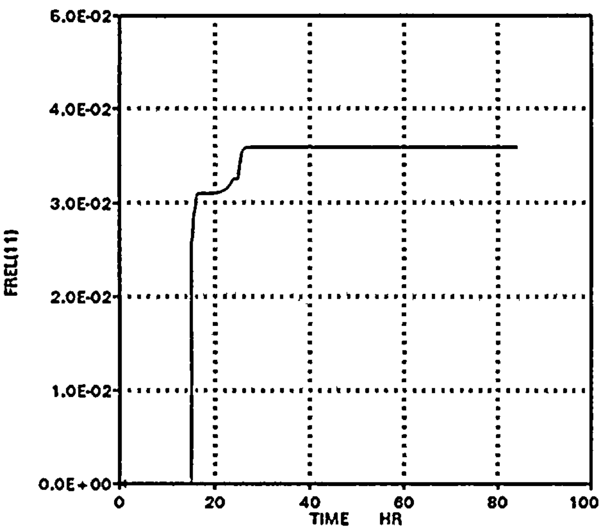


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Figure 4.8-16 Release Fractions for Source Term Group 10 (9 - CeO_2, 10 - Sb, 11 - Te_2).

4.9 Sensitivity of Back End Results and Insights

The CETs were requantified by varying important parameters that are likely to have the largest effect on the likelihood or time of containment failure and the magnitude of the source term. The results were then used to identify the areas for which potential improvements of the plant might be considered. The sensitivity studies performed for the Level 2 IPE are discussed in the following and the numerical results are listed in Tables 4.9-1 to 4.9-3.

Depressurization During A Short-term Station Blackout

During the natural course of a short-term station blackout, the primary system will remain at high pressure and at the time of vessel failure there will be an HPME. Because the batteries remain available, the operators have the opportunity to open the SRVs and depressurize the vessel. Therefore, at the time of vessel failure an LPME will result. The first sensitivity analysis was designed to provide an assessment of the relative benefits and shortcomings associated with each strategy. Table 4.9-1 lists the results.

If the primary system is depressurized during the station blackout, when there is no source of make-up, the inventory lost during blowdown will result in core melting about 1 hour sooner than if the system is left at high pressure. This provides an additional hour within which there is an opportunity to recover offsite power and prevent core melt. If the decision is made to depressurize the vessel so that it is at low pressure at the time of vessel failure, the net effect is that the incidence of STG-9 (early containment failure at or near the time of vessel failure) will be reduced at the expense of an increase in the frequency for STG-3 and 5 (late containment failures). However, if the decision is made not to depressurize the vessel, the frequency of STG-9 increases from 2.05E-8/yr to 2.07E-7/yr.

The initial conclusions from this brief analysis are that depressurization of the primary system should be delayed as long as possible, but, once core melting has been irreversibly initiated it would seem prudent to depressurize as quickly as possible. This has two benefits, namely if power is restored and all injection systems become available as potential sources of core cooling, and if depressurization is at least partially successful, the chances for delaying containment failure or for maintaining the integrity of containment are enhanced.

Decay Heat Removal Systems - RHR

Since an analysis of the importance of DHR to core damage frequency was included in the discussion of the Level 1 results, an analysis of the sensitivity of source term grouping results to the availability of DHR systems might provide additional insights. To gain an initial understanding of this issue and to do this in a relatively straightforward manner, the likelihood that RHR will be recovered or implemented prior to containment failure was assumed to be 1.0. The net shift in source term group frequencies becomes an indicator of

the maximum gain possible if RHR performance were improved. A large potential benefit measured by a dramatic shift towards more benign release groups would indicate that a search for improved performance may be warranted.

The results in the sensitivity analysis or "improvability assessment" for RHR showed that if the RHR systems were "perfectly available" from the Level 2 perspective, the fraction of "no containment failures" (STG-1 and 2) increased from 38.9% to 46.9% as seen in Table 4.9-2. As would be expected, this improvement was also manifest in a corresponding reduction in the frequency of "late failure" in STG-3 from 1.07E-6/yr to 9.98E-9/yr. These results do not necessarily indicate that loss of containment heat removal reliability is an important plant weakness. This does indicate that an examination of the options which may be available to enhance RHR reliability under severe accident conditions may be an area to evaluate the costs and benefits of possible improvements.

Decay Heat Removal Systems - Hardened Vent

This sensitivity study is to investigate the benefits of installing a hardened containment vent. In Level 2, assuming venting is always successful except in TW sequences, the effects on the source term group frequencies are listed in Table 4.9-2. The frequency of STG-5 (late containment failures) increased slightly from 4.19E-6 to 4.21E-6. In addition, the frequency of late catastrophic failures represented by STG-4 was eliminated and that of STG-6 (late CF * large CFM * unscrubbed) was reduced from 2.92E-8/yr to 2.29E-8/yr. It is concluded that once core damage occurs, hardened venting has very limited benefits in reducing offsite consequences.

Fragility of the Containment Shell

To better understand the issues and to estimate the changes in source term character as some of the base assumptions are changed, a limited series of sensitivity studies were undertaken. First, to explore the effects of assumptions about shell fragility when under assault from molten debris ejected from the pedestal after an HPME, the shell failure split fractions were varied to see what the resulting effects may be. This analysis had two aspects to it, because by varying the split fraction for shell failure/no failure from 60/40 to 10/90 it was possible to infer the effects of assumptions about:

- the amount and velocity of corium ejected from the deep pedestal cavity which could be available to come in contact with the containment shell
- the intrinsic fragility of the shell when exposed to corium

The results of the sensitivity study show that as the fragility of the shell increases or the assumption that less corium is ejected from the cavity, the fraction of very early containment failures represented by STG-9 decreased from 8.7% to 2.2%, and the fraction of late failures represented by STG-5 increased from 24.0% to 30.5%. This confirms the importance of the assumptions associated with shell melt through and with the mass and velocity of corium ejected from the pedestal and reconfirms the need to try to reduce the incidence of core damage events in which there is expected to be an HPME.

Drywell Sprays

In high pressure core melt sequences in which the LP systems are available, it may be useful to understand the value of initiating drywell sprays prior to vessel failure because there are competing risks. Because the drywell drains to the pedestal cavity, if the sprays are actuated prior to vessel failure, water will accumulate in the cavity to sufficient depth that a destructive steam explosion becomes a possibility following vessel failure. This is the negative side of spray operation. The positive contribution from drywell spray operation at the time of vessel failure comes from the fact if the sprays are operating at the time that there is an HPME, the cooling of the debris ex-vessel will be enhanced and the threat of corium ejection to the shell or corium attack on the downcomers may be reduced. This means that the likelihood of containment failure may be reduced.

Sensitivity studies were performed to gain insight into both of these issues by first changing the split fraction for the likelihood of steam explosions causing early containment failure (Node ECF), and then for the likelihood that the injections could be recovered following vessel breach (Node ICF) under HPME conditions. In one case the split fraction for early CF/no early CF was changed from 74/26 to 10/90 and in another case injection was assumed to be successful before containment failure.

The results of the analyses associated with the parameters which influence the likelihood of containment failure because of steam explosions in the pedestal and an inability to cool the debris ex-vessel after an HPME show competing characteristics. If the drywell sprays are not actuated prior to vessel breach so that at the time of vessel failure the pedestal will be dry and the likelihood of steam explosions becomes minimal, the overall performance of containment improves slightly. The increase in the conditional probability of an intact containment (STG-1 and 2) is 0.042 (38.9% to 43.1%).

If the drywell sprays are actuated prior to vessel breach, though providing a mechanism for water accumulation in the pedestal, the consequences are an enhanced cooling of the debris splattered within the cavity, and an increased probability that corium ejected from the cavity opening will not impact and damage the containment shell plate, downcomers or other penetrations in the drywell floor. This results in an improvement in containment performance. The probability of maintaining an intact containment under these conditions increases by 0.031 (38.9% to 42%).

Though there is uncertainty in both of the estimates of the improvement which could be expected if the drywell sprays are, or are not, actuated during the high pressure scenarios in which the low pressure systems are available but deadheaded, it appears that more benefit could be gained if the sprays are not actuated prior to vessel breach. The results of this sensitivity analysis indicate that further analysis of this issue may be warranted to assist in the optimization of the overall strategy which should be employed in managing severe accident and to confirm that the current emergency operating procedures reflect the benefit from any important insights which may follow.

Debris Coolability

This analysis is to investigate the effect of the core debris being coolable or not coolable since there is uncertainty regarding the geometry of the debris under HPME conditions. If the split fraction for debris cooled/not cooled was changed from 43/57 to 90/10, the conditional probability of an intact containment remains unchanged as seen in Table 4.9-3. In addition, about 1.6% of the contributions from STG-5 and 6 (fission products not scrubbed) is shifted towards more benign release groups (STG-3 and 4 - fission products scrubbed). This concludes that the current assumption of debris coolability is conservative.

TABLE 4.9-1
Effects of Vessel Depressurization on Source Term During A Short Term Station Blackout.

Base Case			No Depressurization			Depressurization		
STG-			STG-			STG-		
1	Freq= 1.74E-006	64.4%	1	Freq= 1.31E-006	48.2%	1	Freq= 1.79E-006	66.1%
5	Freq= 4.99E-007	18.5%	2	Freq= 4.85E-007	17.9%	5	Freq= 5.13E-007	18.9%
3	Freq= 3.75E-007	13.9%	5	Freq= 3.92E-007	14.5%	3	Freq= 3.85E-007	14.2%
2	Freq= 4.85E-008	1.8%	3	Freq= 2.99E-007	11.1%	11u	Freq= 1.31E-008	0.5%
9	Freq= 2.05E-008	0.8%	9	Freq= 2.07E-007	7.6%	6	Freq= 3.86E-009	0.1%
11u	Freq= 1.27E-008	0.5%	11u	Freq= 9.65E-009	0.4%	13u	Freq= 3.06E-009	0.1%
6	Freq= 3.70E-009	0.1%	13u	Freq= 6.50E-009	0.2%	6u	Freq= 2.33E-011	0.0%
13u	Freq= 3.29E-009	0.1%	6	Freq= 3.36E-009	0.1%			
6u	Freq= 2.09E-011	0.0%	4	Freq= 4.72E-010	0.0%			
			6u	Freq= 1.40E-011	0.0%			

TABLE 4.9-2
Results of Sensitivity Studies for Level 2 IPE

BASE CASE:

STG-1	Freq= 4.85E-006	27.8%
5	Freq= 4.19E-006	24.0%
13p	Freq= 2.35E-006	13.4%
2	Freq= 1.94E-006	11.1%
9	Freq= 1.52E-006	8.7%
3	Freq= 1.07E-006	6.1%
10	Freq= 1.04E-006	6.0%
13u	Freq= 1.89E-007	1.1%
13	Freq= 1.88E-007	1.1%
11u	Freq= 3.35E-008	0.2%
17	Freq= 2.99E-008	0.2%
6	Freq= 2.92E-008	0.2%
14p	Freq= 2.37E-008	0.1%
10u	Freq= 3.01E-009	0.0%
14	Freq= 1.90E-009	0.0%
4	Freq= 1.87E-009	0.0%
6u	Freq= 1.33E-010	0.0%

DECAY HEAT REMOVAL SYSTEM - RHR:

STG-1	Freq= 5.91E-006	33.8%
5	Freq= 3.87E-006	22.1%
13p	Freq= 2.35E-006	13.4%
2	Freq= 2.28E-006	13.1%
9	Freq= 1.52E-006	8.7%
10	Freq= 1.04E-006	6.0%
13u	Freq= 1.89E-007	1.1%
13	Freq= 1.88E-007	1.1%
11u	Freq= 3.38E-008	0.2%
17	Freq= 2.99E-008	0.2%
6	Freq= 2.70E-008	0.2%
14p	Freq= 2.37E-008	0.1%
3	Freq= 9.98E-009	0.1%
10u	Freq= 3.01E-009	0.0%
14	Freq= 1.81E-009	0.0%
6u	Freq= 1.33E-010	0.0%

DECAY HEAT REMOVAL SYSTEM - Hardened Vent:

STG-1	Freq= 4.85E-006	27.8%
5	Freq= 4.21E-006	24.1%
13p	Freq= 2.35E-006	13.4%
2	Freq= 1.94E-006	11.1%
9	Freq= 1.52E-006	8.7%
3	Freq= 1.06E-006	6.1%
10	Freq= 1.04E-006	6.0%
13u	Freq= 1.89E-007	1.1%
13	Freq= 1.88E-007	1.1%
11u	Freq= 3.37E-008	0.2%
17	Freq= 2.99E-008	0.2%
14p	Freq= 2.37E-008	0.1%
6	Freq= 2.29E-008	0.1%
10u	Freq= 3.01E-009	0.0%
14	Freq= 1.81E-009	0.0%
6u	Freq= 1.33E-010	0.0%
14u	Freq= 0.00E+000	0.0%
4u	Freq= 0.00E+000	0.0%
4	Freq= 0.00E+000	0.0%

Containment Shell FRAGILITY:

STG-5	Freq= 5.32E-006	30.5%
1	Freq= 4.85E-006	27.8%
13p	Freq= 2.35E-006	13.4%
2	Freq= 1.94E-006	11.1%
3	Freq= 1.07E-006	6.1%
10	Freq= 1.04E-006	6.0%
9	Freq= 3.81E-007	2.2%
13u	Freq= 1.89E-007	1.1%
13	Freq= 1.88E-007	1.1%
6	Freq= 4.06E-008	0.2%
11u	Freq= 3.35E-008	0.2%
17	Freq= 2.99E-008	0.2%
14p	Freq= 2.37E-008	0.1%
10u	Freq= 3.01E-009	0.0%
4	Freq= 1.87E-009	0.0%
14	Freq= 1.81E-009	0.0%
6u	Freq= 1.89E-010	0.0%

TABLE 4.9-2 (Cont'd)
Results of Sensitivity Studies for Level 2 IPE

DRYWELL SPRAYS - Recovered Before Vessel Breach:			DRYWELL SPRAYS - Recovered After Vessel Breach:		
STG-1	Freq= 5.18E-006	29.6%	STG-1	Freq= 5.12E-006	29.3%
5	Freq= 4.27E-006	24.4%	5	Freq= 3.83E-006	21.9%
2	Freq= 2.37E-006	13.5%	13p	Freq= 2.35E-006	13.4%
13p	Freq= 2.35E-006	13.4%	2	Freq= 2.22E-006	12.7%
9	Freq= 1.53E-006	8.7%	9	Freq= 1.30E-006	7.4%
3	Freq= 1.17E-006	6.7%	3	Freq= 1.11E-006	6.4%
13u	Freq= 1.90E-007	1.1%	10	Freq= 1.04E-006	6.0%
13	Freq= 1.88E-007	1.1%	13	Freq= 1.88E-007	1.1%
10	Freq= 1.41E-007	0.8%	13u	Freq= 1.87E-007	1.1%
11u	Freq= 3.48E-008	0.2%	11u	Freq= 3.54E-008	0.2%
6	Freq= 3.03E-008	0.2%	17	Freq= 2.99E-008	0.2%
17	Freq= 2.99E-008	0.2%	6	Freq= 2.68E-008	0.2%
14p	Freq= 2.37E-008	0.1%	14p	Freq= 2.37E-008	0.1%
4	Freq= 2.71E-009	0.0%	10u	Freq= 3.01E-009	0.0%
14	Freq= 1.81E-009	0.0%	4	Freq= 2.14E-009	0.0%
10u	Freq= 4.08E-010	0.0%	14	Freq= 1.90E-009	0.0%
6u	Freq= 1.33E-010	0.0%	6u	Freq= 1.33E-010	0.0%

TABLE 4.9-3
Sensitivity Study for Debris Coolability.

BASE CASE:			DEBRIS COOLABILITY:		
STG-1	Freq= 4.85E-006	27.8%	STG-1	Freq= 6.46E-006	37.0%
5	Freq= 4.19E-006	24.0%	5	Freq= 3.92E-006	22.4%
13p	Freq= 2.35E-006	13.4%	13p	Freq= 2.35E-006	13.4%
2	Freq= 1.94E-006	11.1%	9	Freq= 1.52E-006	8.7%
9	Freq= 1.52E-006	8.7%	3	Freq= 1.34E-006	7.7%
3	Freq= 1.07E-006	6.1%	10	Freq= 1.04E-006	6.0%
10	Freq= 1.04E-006	6.0%	2	Freq= 3.37E-007	1.9%
13u	Freq= 1.89E-007	1.1%	13	Freq= 1.88E-007	1.1%
13	Freq= 1.88E-007	1.1%	13u	Freq= 1.78E-007	1.0%
11u	Freq= 3.35E-008	0.2%	11u	Freq= 4.44E-008	0.3%
17	Freq= 2.99E-008	0.2%	17	Freq= 2.99E-008	0.2%
6	Freq= 2.92E-008	0.2%	6	Freq= 2.72E-008	0.2%
14p	Freq= 2.37E-008	0.1%	14p	Freq= 2.37E-008	0.1%
10u	Freq= 3.01E-009	0.0%	4	Freq= 4.79E-009	0.0%
14	Freq= 1.90E-009	0.0%	10u	Freq= 3.01E-009	0.0%
4	Freq= 1.87E-009	0.0%	14	Freq= 1.81E-009	0.0%
6u	Freq= 1.33E-010	0.0%	6u	Freq= 1.33E-010	0.0%

5.0 UTILITY PARTICIPATION AND INTERNAL REVIEW TEAM

5.1 IPE Program Organization

In the early 1980's, the Supply System established a Senior Management Review Group to maintain cognizance of the severe accident issues, work with the industry's IDCOR Program through the policy group and with technical reviews, and assess potential impacts to WNP-2 as the NRC formulated the Severe Accident Policy and drafted the regulations that became Generic Letter 88-20. The Supply System established the IPE team within the Engineering Directorate in anticipation of the release of Generic Letter 88-20. As the scope and cost of the individual plant examination using probabilistic risk assessment methodology became clear, the decision was made early to internalize the methodology to be able to benefit from the knowledge gained and apply it to other areas within the Supply System.

The IPE team members are listed on the cover sheet of this report. In addition, a former Shift Manager from Operations was made available full-time during the initial information gathering tasks, for the system walkdowns, and for input and review of the system analysis. This early Operations support was a significant contributor to the quality of the analyses and results. From the list of contributors shown on the Acknowledgement page, it is obvious that all technical organizations of the Supply System contributed to this effort. The cost of the internal events IPE is now approximately $3.2M including internal manpower, consulting services, and computer code acquisitions. The analysts assigned to this effort remained throughout the entire project providing the consistency and continuity necessary to bring the effort to fruition on schedule.

The Supply System staff performed all aspects of the IPE. This included the system information gathering, system modelling and analysis, initiator data gathering, event tree preparation and quantification, containment event tree preparation and quantification, phenomenological MAAP analyses, primary containment structural analysis, and sensitivity analysis. A consulting firm, Individual Plant Evaluation Partnership (IPEP, a partnership between Tenera, L.P., Fauske and Associates, Inc., and Westinghouse), was hired to assist in preparation the initial IPE. The function of IPEP was twofold, to provide training in PRA and special topics like Human Reliability Analysis, and secondly, to provide peer review of the IPE products as they were generated. IPEP had approximately 50 percent of the consulting contracts in the industry, making them eminently qualified to review the WNP-2 IPE as peer reviewers of both the Front-End and Back-End analyses. As applications demand increased for use of the IPE products, it became clear that more realistic models were necessary. This realism is crucial to ensure resources and priorities are focused on the items providing the most safety benefit. To ensure an independent assessment of the conservatisms that existed in the original IPE, a consultant, NUS Corp, was hired to recommend model improvements and eliminate data conservatisms. Revision 1 of the IPE is the result of implementing NUS's recommendations.

5.2 Composition of Independent Review Teams

The WNP-2 IPE has received a multi-tiered review in terms of technical review, peer review, independent in-house review, and management review. Technical reviews of the analysis were performed as the individual analyses were completed. The system notebooks were prepared by a system design engineer, reviewed by an independent system design engineer, and reviewed again by a Plant Technical Staff system engineer or Shift Technical Advisor (STA). The system analysis (fault tree) models were prepared in accordance with 10CFR50 Appendix B QA requirements and reviewed by IPEP for reasonableness compared to other plants models and results. The MAAP model input parameter file was also prepared to Appendix B requirements. Technical peer review of all quantification results and phenomenological issue positions and results were performed by IPEP for the original analysis and by NUS for the rebaselined Revision 1 analysis.

An independent In-house Review Team was established to review the WNP-2 IPE. This review team was composed of individuals not associated with the preparation of the IPE but who are experienced and knowledgeable of WNP-2 systems, safety evaluations, design basis safety analyses, reliability methodology, and/or training in these functional areas. The In-house IPE Review Team members and their organizational affiliation were:

Gordon Brastad	Systems Analysis (I&C)-Engineering Directorate
Doug Coleman	Assurance Eng-Licensing & Assurance Directorate
Alan Hosler	Licensing-Licensing & Assurance Directorate
Dave Injerd	Maintenance-Operations Directorate
Steve Kirkendall	Systems Analysis (Mech)-Engineering Directorate
Art Moore	Training-Operations Directorate (Supervisor-Safety Analysis for Revision 1 effort)
Bob Talbert	STA-Plant Tech Staff-Operations Directorate

These individuals also represent potential PRA applications users in their respective work functions. The In-house Review Team met for several half-day sessions. Each session was conducted to a) respond to concerns/questions raised at the previous session, b) have a member of the IPE team introduce a topic, e.g., system fault tree modelling, c) present the results in the appropriate section of the IPE Report, and d) respond to questions or record comments for resolution by the next meeting. This review process was used for all aspects of this report.

In addition to the In-house Review Team, IPEP reviewed a draft of the original report and provided comments. Their review was comprehensive in terms of ensuring the individual tasks and results were properly collated into this report and by performing a peer review of other plant's IPE Reports. NUS Corp has reviewed the Revision 1 report as well as the technical input to the report. A senior level peer reviewer, John Raulston of Tenera, L.P., was hired to provide an independent overview of the methodology used and the results obtained in Revision 1 of the IPE. Presentations have been made to the Corporate Nuclear Safety Review Board and to Supply System Senior Management to discuss the methodology and results of the IPE analysis. These presentations focused on what parameters most affect the CDF, what can be done about them, and the tasks remaining to complete the Severe Accident Program. The potential uses of the methodology and IPE results in other Supply System processes were also discussed.

5.3 Review Comments and Resolution

The WNP-2 IPE Report, Revision 0, was reviewed by the In-house Review Team and by IPEP (Tenera, L.P. and Fauske and Associates, Inc.). Revision 1 of the report was reviewed by NUS Corp and Tenera, L.P. These reviews were conducted at various stages of report completeness to ensure resolution of comments addressing technical correctness and issues to report content and clarity. The number and content of the comments are too numerous to include in this report. In all cases, however, satisfactory resolution was achieved between the PRA team and the reviewers. The following is a list of outstanding issues raised during the review that did not significantly impact the results but will be addressed in the long term as part of maintaining the IPE.

- The loss of offsite power scenarios from full power should include a SORV event. The SORV would have little impact on injection source success except it would contribute to RCIC unavailability as the reactor vessel depressurized. The magnitude of the impact was investigated and determined that the CDF increases by approximately 4.5% (.08E-05).

- The assumption that containment failure in the wetwell region (1/3 of the failure probabilities) would always lead to injection failure is too conservative. A more realistic evaluation would lower the "TW" sequences contribution to CDF. This evaluation will be done as part of the configuration control of the IPE.

- CST refill was not credited as a potential recovery action. During some scenarios, refilling the CST would prevent switch over to the suppression pool for HPCS or RCIC. One of the HPCS and RCIC failure modes is that on the switchover to suppression pool suction, the pool temperature exceeds their design temperature and the pumps are assumed to fail. By allowing continued suction on the CST, this failure mode could be avoided. This evaluation will be done as part of the configuration control of the IPE.

- A maximum human error limit should be applied to all sequences. If a sequence contains two or more human errors, it probably was truncated out during quantification. In reality, if an operating crew fails to perform the first action, the subsequent actions likely have a higher failure rate than if performed independently. This evaluation will be done as part of the configuration control of the IPE.

- The diesel generator failure data should be based on plant specific rather than generic data. The WNP-2 diesel generator data will be examined for inclusion as part of the configuration control of the IPE.

6.0 PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

The Level 1 and Level 2 results show that the loss of offsite power with concurrent loss of onsite power sources represent the largest contributor to plant risk. The flooding sequences that cause loss of decay heat removal from the power conversion system (condenser, feedwater) or loss of the RHR system also contribute significantly. One of the beneficial impacts of performing the IPE has been the gain in plant knowledge of major features and insights of the plant system response to severe accidents. Therefore, a general recommendation that results from the IPE is to disseminate this knowledge to the organizations of the Supply System that can utilize the knowledge in their day-to-day activity.

6.1 Unique Safety Features of WNP-2

BPA Power Grid-Although loss of offsite power is a major contributor, the BPA grid has been shown to be extremely stable. The recovery times and probabilities of recovery used in this IPE are based on generic data and believed to be bounding. It is worth noting that the region is not susceptible to adverse weather phenomena, such as hurricanes, that tend to lengthen the time to recover offsite power. During the latest major earthquake in California the northwest portion of the BPA grid was not lost and WNP-2 did not loose offsite power. The long down times from the generic data is bounding and the power grid can be considered a unique strength of WNP-2.

ATWS Mitigation Systems-Through the use of the alternate rod insertion (ARI), standby liquid control (SLC) injection system, and recirculation pump trip, the ATWS transients are not a significant contributor to WNP-2 core damage. The electrically caused ATWS sequences do not even appear in the final quantification due to the redundancy mentioned and the mechanically caused ATWS sequences require additional failures to lead to core damage.

Recessed Pedestal Cavity-The pedestal area under the reactor vessel is recessed below the drywell floor area. This allows most of the molten corium to accumulate in the pedestal area during low pressure melt scenarios. Therefore, the primary containment shell is protected from direct corium attack following vessel breach. Conservative estimates of melt carry out were used in the Level 2 analysis, and the sensitivity analysis on this feature indicates that for station blackout scenarios, operator action to depressurize the vessel once the core starts to melt would be beneficial. This will be a recommendation, but does not negate the benefit of the recessed pedestal for the majority of the scenarios.

DC System Redundancy-WNP-2 DC system design separates the safety and non-safety DC loads such that load shed on loss of AC is easily accomplished. This extends the battery lifetime for station blackout and loss of DC scenarios.

HPCS-The HPCS is an independent third divisional system with its own electrical support including diesel generator and battery. Its diesel generator cooling is powered from its own diesel, allowing the system to be operable during station blackout scenarios. The service water to the room coolers is dependent on other AC sources and is lost during station blackout. However, the room configuration and location is amenable to cooling by opening the door to the room.

6.2 IPE RECOMMENDED IMPROVEMENTS

Current WNP-2 design and operation has been demonstrated by the IPE to exhibit an acceptable level of risk to the general public health and safety. The following recommendations have been evaluated by a cost benefit screening process using IPE sensitivity studies. The hardware modification recommendations that pass this screening will be presented to the plant for evaluation using normal plant procedures. This will allow detailed cost benefit analysis to be done and implementation if they are indeed cost beneficial. Procedure recommendations will be made using normal plant procedures to ensure complete review and approval by affected organizations.

500 Kv Backfeed - When the plant is in a shutdown condition, power can be obtained from the 500 Kv grid. This is accomplished by disconnecting the isolated phase bus links to isolate the main generator. Power can then be established through the main step-up transformers to the AC Distribution system. Currently this procedure takes longer than eight hours to make this transition. Therefore, credit was not taken in the IPE for this alternate source of offsite power. It is recommended an evaluation be conducted of the costs and benefits of making a permanent hardware change so that the 500 Kv source is available within the four hour station blackout coping period. A core damage frequency reduction of up to 50% can be achieved, depending on the design and reliability of the modification. The procedural recommendation to depressurize during station blackout scenarios (see below) also provides meaningful core damage frequency reduction. Once the procedure revision is instituted, the addition of the 500 Kv backfeed breakers should be re-evaluated.

Startup Transformer Capability - When the 230 Kv voltage is below a preset value the plant bus transfer, load to the BOP buses is lost and the 115 Kv line is used to power the safety buses. This undervoltage (but not lost totally) unavailability of the 230 Kv line and startup transformers can be alleviated by increasing the capacity on the transfer. A core damage frequency reduction of 4% was determined by the sensitivity analysis. The evaluation of the costs and benefits of making this hardware modification was performed and the modification shown not to be cost effective.

125 VDC Battery Swing Charger - The common cause failure of the battery chargers is a significant contributor to the core damage frequency. A special reliability study was performed and potential modifications were assessed. The recommendation with the largest potential benefit is to provide a battery charger that is sufficiently isolated but can be made available to either division of the 125 VDC system. This swing charger should be of a

different model and type from the current designs. The evaluation of the costs and benefits of making this hardware modification was performed and was marginal for the cost involved. The common cause failure of the battery chargers appears high on the risk achievement worth list (see Table 3.4.2-2), therefore, a reduction in the common cause failure rate has little impact on reduction of the core damage frequency. However, improving the reliability, i.e., preventing degradation of the charger common cause failure rate prevents increasing the core damage frequency by 4% for each factor of ten degradation in charger common cause. The cost benefit screening did show improved maintenance practices would be cost effective.

ADS Inhibit Switch - Under ATWS scenarios, the water level is lowered to control power. If the ADS setpoint is reached, the operator must inhibit the ADS function within the 105 second time delay to prevent depressurization and possible low pressure injection that would sweep out boron from SLC. For non-ATWS scenarios, the use of the inhibit switch is not allowed, and in order to follow emergency procedure guidance, the operators must invoke the ADS inhibit function every 105 seconds or result in an unwanted depressurization. The failure to inhibit ADS contributes approximately 1.5% to the CDF. A licensing action to allow WNP-2 the same emergency procedure assumption as the other BWRs has been initiated.

Procedural Recommendations - Numerous procedure improvements have been made in the EOPs in response to the station blackout, ATWS, and industry initiatives. The current emergency procedures cover a large range of severe accident phenomena. Credit was taken in the IPE where the procedure already exists. The dominant sequences' operator actions were verified by a table top walk through of the emergency procedures. Recommendations will be made in the following areas:

> Reactor building and turbine building flooding: although adequate alarms and sumps exist, it is recommended an evaluation of procedures and training for the recognition and isolation of floods identified to cause multiple system failures be conducted to assess the costs and benefits of potential improvements. This evaluation will include an examination of surveillance of non-safety system piping and the costs and benefits of periodic inspection for those piping locations contributing to the flooding event consequences.
>
> Preventative maintenance for common cause: the majority of risk significant failures are due to common cause failures of similar components. It is recommended an evaluation of the costs and benefits of changes to maintenance practices to ameliorate common cause occurrences be conducted. The importance of common cause to overall plant risk will be emphasized to appropriate maintenance organizations. The importance of maintaining commitments to practices such as staggered maintenance of SOVs (NUREG-1275, Vol. 6) will be demonstrated.

Station Blackout Depressurization: It is recommended that the longer coping time if the vessel is maintained at pressure with the benefit of depressurizing once fuel melt starts, but before vessel breach, be evaluated for incorporation in the station blackout emergency procedures. Additional studies will be performed to evaluate the effects of depressurizing prior to battery depletion to lengthen the potential recovery time. The additional coping time due to depressurization has been evaluated to reduce CDF by up to 34%. The insights from these studies will be used for input to the appropriate BWROG severe accident groups and for consideration on WNP-2 plant specific procedures.

Drywell/Wetwell Bypass: the Omega seal design separating the drywell and wetwell air spaces is passive design with very low failure rate. However, its failure results in very large consequences. Therefore, it is recommended an evaluation of the costs and benefits of periodic inspection and maintenance of the Omega seal be made.

CN Supply to MSIVs and Vent Valves - A hardware modification of the air supply to the inboard MSIVs and the containment vent valves for backup from the containment nitrogen system was investigated. The modification would improve long term decay heat removal by providing redundant air supply to the valve's solenoids. As discussed in Section 3.4.3, the total contribution to CDF from loss of decay heat removal function is 1.4E-6 per year. The hardware modification costs are marginally cost effective. It is not recommended for further plant evaluation at this time but may increase in importance as other recommendations are instituted.

6.3 ADDITIONAL IPE INSIGHTS

A major benefit in performing the IPE is a greater awareness of the plant response to severe accidents, the systems and components most important to prevention and mitigation of the accident, and the importance of operator actions in the prevention and mitigation. Most of these insights have been recognized and are currently in the process of incorporation into a revision of the BWROG Emergency Procedures Guidelines. The Supply System is actively working with the appropriate BWROG committees in formulating the accident management procedures. Therefore, these recognized insights are not included in this report.

Other insights observed during sensitivity analyses for Level 1 and Level 2 include:

RHR - Improvement in the reliability of the RHR system for containment heat removal has a significant beneficial impact on both Level 1 and Level 2 results. This indicates that future work in this area could be cost beneficial.

Drywell Sprays - Sensitivity studies indicate competing effects of early initiation of drywell sprays. Since the drywell drains to the pedestal area, early initiation could flood the cavity and result in increased likelihood of steam explosion under high pressure melt scenarios. However, spray water could protect the shell and downcomers from melt ejected to the drywell. Under certain high pressure melt scenarios, when vessel breach occurs before the operator is expected to initiate sprays, the containment pressure will exceed the spray initiation limit immediately after vessel breach. Therefore, sprays would never be initiated during these scenarios to assist in debris cooling. These insights will be used to assist the BWROG efforts in developing an integrated containment response strategy for the emergency procedure guidelines.

System Importance - The systems most important for maintaining the current level of safety at WNP-2 are:

- Reactor Scram System (RPS/CRD)
- AC Power System (emphasis on onsite diesels)
- DC Power System (emphasis on battery chargers and maintenance)
- RHR in the suppression pool cooling mode
- Standby Service Water for decay heat removal function
- High Pressure Injection (RCIC, HPCS)

Input of the system importance information will be provided to the RCM effort as maintaining these system reliability has the most benefit in terms of core damage frequency.

Operator Action Importance - There are several operator actions that are important to the current level of safety at WNP-2:

Operator vents containment in time.
Operator opens RCIC pump room doors on loss of room cooling.
Amount of time RHR Train B is out of service for test and maintenance.
Operator initiates ADS.
Amount of time HPCS is out of service for test and maintenance.
Operator initiates suppression pool cooling or sprays in time.
Operator controls inventory or inhibits ADS during ATWS.

The importance and timing of these actions will be provided to the Training personnel for inclusion into Operator training processes.

Higher Pressure in Air Supply - It was noted in several sections, that when containment pressure exceeded a given level, the ability to open the MSIVs or the SRVs limited the operator ability to recover during the sequence. If the pressure of the air supply and nitrogen backup were raised, then additional margin, which translates to additional time would be available for corrective action. The analysis was based on 150 psig air supply pressure, which has recently been raised to 186 psig. The additional 36 psi results in a longer time period for the operator to recover by opening MSIVs or venting as necessary for mitigation of the sequence.

Adjust RCIC Initiation Setpoint - Currently the RCIC is initiated on the same water level as HPCS. On loss of feedwater type transients, earlier initiation of RCIC may prevent reaching HPCS initiation and more importantly, may prevent reaching MSIV closure with its subsequent loss of the condenser as a heat sink. This potential reliability improvement will be proposed for evaluation within normal Supply System processes.

Operator Action in ATWS - During ATWS sequences, due to the rapid energy addition to the containment, several sequences end with containment failure prior to core melt. Upon core melt, this provides a release path when the core becomes ex-vessel. One action the operator could do is to stop injection with all sources, except SLC, allowing the core to melt if SLC is unsuccessful prior to containment failure. Then the operators efforts can focus on preventing containment failure and containing the fission products inside. This strategy will be proposed for inclusion in the BWROG severe accident management efforts.

Applications Of The IPE - The insights, models, and results of the IPE can provide relevant information in several current efforts and processes at WNP-2. The more immediate and beneficial areas include:

- IPE models and techniques used on safety and BOP systems important for the Maintenance Rule.

- Providing models and techniques for risk profiles for planned and unplanned outages. This implementation of PRA methodology has been initiated at WNP-2.

- Revising Tech Spec AOT/STI values based on contribution to plant risk
- Modifying Tech Specs based on IPE insights, e.g., on loss of standby service water it is better to maintain the plant at hot standby than to require cold shutdown.
- Providing scenario descriptions for emergency drills and exercises.

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7.0 SUMMARY AND CONCLUSIONS

The Supply System has performed an Individual Plant Examination of the WNP-2 using latest plant specific system information, procedures, accident initiator data, and component failure data. The general methodology used in this report is a Level 1 and 2 PRA. This methodology and major tasks are as described in NUREG/CR-2300 "PRA PROCEDURES GUIDE" for Level 1 and Level 2 PRA. The system analysis methodology utilized the small event tree-large fault tree approach. A limited amount of generic data (e.g., NUREG/CR-2815, WASH-1400) were used where plant specific data was not available. System analyses were treated as safety related analyses and were performed to the in-house Engineering procedures. Core damage frequency and containment failure probability were quantified using the NUPRA code which has been verified and validated to the requirements of Title 10 of the Code of Federal Regulations, Part 50, Appendix B. Because of the consistency throughout the process and the linking and merging capabilities of the NUPRA code, dependencies, common cause effects, and system interactions were fully accounted for by the IPE.

The Supply System staff performed all aspects of the IPE. This includes the system information gathering, system modelling and analysis, initiator data gathering, event tree preparation and quantification, containment event tree preparation and quantification, phenomenological MAAP analyses, primary containment structural analysis, and sensitivity analysis. A consulting firm, Individual Plant Evaluation Partnership (IPEP-a partnership between Tenera, LP, Fauske and Associates, Inc, and Westinghouse), was hired to assist in the initial IPE. The function of IPEP was twofold, to provide training in PRA and special topics like Human Reliability Analysis, etc. and secondly, to provide peer review of the IPE products as they were generated. IPEP had approximately 50% of the consulting contracts in the industry, making them uniquely qualified to review the WNP-2 IPE as peer reviewers of both the Front-End and Back-End analysis. Consultants (NUS Corp. and Tenera LP) were used in revision 1 of the IPE to provide a new perspective, reduce data conservatisms, provide more realistic models and add depth to the peer review process. The WNP-2 IPE has received multi-tiered review in terms of technical review, peer review, independent in-house review, and management review.

The core damage frequency and conditional containment failure probability are calculated to be a mean value of 1.75E-5/year and 1.07E-5/year, respectively. There are no vulnerabilities identified in that a single failure either by itself or by consequential failure can cause core damage. The sequences are dominated by common cause and human error events.

Unresolved Safety Issues A-45, A-17 and Generic Safety Issue 105 have been examined and their resolutions verified by the IPE for internal events. Conclusions reached by the IPE analysis show that the most benefit in reduction of core damge frequency is realized by process improvements (e.g., maintenance and testing), and enhancements to operator training.

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