– FINAL REPORT –

PORV RELIABILITY STUDY AND SETPOINT ANALYSIS

FOR

TENNESSEE VALLEY AUTHORITY

AND

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

CONTRACT NO. 600-5253

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BABCOCK & WILCOX
Nuclear Power Group
Nuclear Power Generation Group
P. O. Box 1260
Lynchburg, Virginia 24505

Babcock & Wilcox

Babcock & Wilcox
Nuclear Power Group
Nuclear Power Generation Division
Lynchburg, Virginia

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PORV Reliability Study and Setpoint
Analysis for the 205-FA Owners Group

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EXECUTIVE SUMMARY/ABSTRACT

This report justifies the use of pre-TMI (as-designed) trip setpoints on the 205-fuel assembly pilot-operated relief valve (PORV) isolation system. The proposed system design using these setpoints comprises a single PORV and a single block valve with an automatic closure feature.

The supporting analysis verifies that the system design fulfills both operational and reliability requirements. The system ensures normal PORV operation and prevents high-pressure injection (HPI) actuation on low reactor coolant pressure if the PORV should fail open. Failure to isolate the PORV relief path is limits to 1.66×10^{-4} (TVA)/1.26×10^{-4} (WPPSS) failures per reactor year. Restoration of the designed PORV function will not lead to unacceptable challanges for the safety valves, which will have a failure rate of 9.73×10^{-6} failures per reactor year.

Consequently, B&W recommends that the automatic PORV block valve closure system be installed. In addition, the mandatory reactor trip on turbine trip should be eliminated since reliability requirements are easily achieved even at the elevated PORV challenge rate.

The advantages of this design include an enhanced ability to isolate the PORV relief path (compared to the 177-FA design), as well as fewer reactor protection system challenges and reactor trips. As a result, plant availability is increased. Plant safety will also be enhanced by permitting turbine and reactor runbacks. Should grid separation occur, these features will ensure that the integrity of the power supply defense systems is maintained.

The system does have two drawbacks: In some cases, reactor coolant system depressurization may actuate the engineered safety features actuation system if maximum instrument error is encountered. Also, the pressurizer code safety valves may be challenged if the PORV is inoperable and HPI has been actuated. However, the probability of either of these event sequences occurring is small.

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1. INTRODUCTION

Following the loss-of-coolant accident (LOCA) at the Three Mile Island Unit 2 (TMI-2) facility, the NRC re-evaluated the power-operated relief valve (PORV) system requirements. Plant configuration changes were recommended to reduce the probability of PORV failures. Operating plants were required to raise PORV setpoints, lower high-pressure reactor protection system (RPS) setpoints, and install anticipatory reactor trips upon main turbine trips. These modifications have reduced plant availability by increasing the number of reactor trips. The severity of these plant upsets can be reduced while meeting PORV reliability requirements. By returning the setpoints to their pre-TMI values and by installing an automatic PORV isolation system, both goals can be achieved.

The NRC has formalized guidance for the PORV system changes. The guidance is included in sections II.K.3.1 and II.K.3.2 of NUREG-0737. Section II.K.3.2 requires a report documenting the various actions that have been taken to decrease the probability of a small break LOCA caused by a stuck-open PORV or safety valve. If these actions reduce the probability of a small break LOCA caused by a stuck-open PORV so that it is not a significant contributor to the probability of a small break LOCA due to all causes, then no other actions are needed. If the contribution of the PORV to the total probability is more significant, then II.K.3.1 requires installation of an automatic PORV isolation system.

This report provides the rationale for maintaining the PORV and the high-pressure RPS trip setpoints at their as-designed values thus reducing unnecessary reactor trips by allowing the PORV to operate as intended. Since maintaining the PORV's intended function results in a moderate challenge rate to the valve, an automatic PORV block valve isolation system is necessary to achieve overall system reliability as required by II.K.3.2. An isolation system description and reliability analysis are included to verify that the system will not be a major contributor to the probability of a small break LOCA. In addition, it is shown that safety valve reliability is not significantly affected by the isolation system.

1.1. Background

Following the accident at TMI, the NRC required changes to the PORV opening and high-pressure reactor trip setpoints and the addition of an anticipatory reactor trip on turbine trip for all the operating plants. These changes have increased the number of reactor trips per month caused by minor overpressure events, turbine trips, and feedwater upsets. As intended, the modifications have reduced the number of challenges to the PORV, but they have concurrently increased the number of challenges to the reactor protection system (RPS) and other safety systems required to support a trip. Data collected has shown that of the 87 reactor trip events from September 1979 through December 1981, 40% were caused by high RCS pressure and 29% by the anticipatory reactor trip on main turbine trip.

In order to improve plant availability by reducing the number of reactor trips, the operating plant owners embarked on a program to return the PORV and high-pressure reactor trip setpoints to their pre-TMI values. These actions would increase the number of PORV challenges, necessitating the installation of an automatic PORV closure system. A preliminary conceptual system design was prepared for the Florida Powder Corporation in May 1980. In principle, the proposed design was identical to that proposed for backlog B&W 205-FA units. It consisted of a single PORV and a single block valve with an automatic closure feature. The system improved the probability of isolating a failed-open PORV by a factor of 25. However, its failure rate was still too high not to be considered a major contributor to the probability of a small break LOCA.

1.2. Scope

The results of the original automatic PORV isolation system proposed for Florida Power showed that the failure rate for isolating the PORV relief path prior to ESFAS actuation was 9.7×10^{-4} per reactor year. In order for the PORV not to be considered a significant contributor to the probability of a small break LOCA due to all causes, the calculated failure rate had to be reduced to approximately 3×10^{-4} per reactor year. To achieve this rate, a more detailed analysis was conducted for the 205-FA plants. It addressed four major areas:

PORV Relief System Setpoints — The automatic PORV isolation system was subjected to dynamic setpoint analysis using the POWER TRAIN V (PT-V) code. Setpoint selection was based on (1) the expected minimum closure pressure for the

PORV to preclude automatic block valve closure during normal PORV operation, (2) PORV block valve closure early enough to avoid ESFAS actuation due to low RCS pressure following a stuck open PORV (assuming no additional failures causing loss of RCS pressure control), (3) PORV block valve stroke time, and (4) nominal errors on applicable setpoints and instrument strings.

PORV/Safety Valve Demand Frequency – The demand frequencies of the PORV and safety valves were predicted for the backlog 205-FA plants. Various overheating events, such as turbine trips, reactor trips, and feedwater pump trips were considered, as well as overcooling events resulting in HPI repressurization. The PT-V code was used to model the overheating transients, while the KPRZ code was used for the overcooling transients.

PORV Relief Path Reliability – The probability of an open PORV flow path depends on the PORV demand frequency, the probability of a failed-open PORV (given that it has opened), and the probability of no block valve closure (given a stuck-open PORV). The probability calculations were based on valve hardware faults, valve operator faults, control faults, and human action probabilities.

Safety Valve Reliability – The probability of safety valve failure depends on the demand frequency, PORV position (open or closed), and the phase of the effluent (liquid or vapor). The probabilities for steam relief were estimated from applicable experience on steam safeties and B&W operating experience. Water relief probabilities were estimated using EPRI valve tests and applicable B&W experience.

1.3. Results

The results of these analyses indicate three significant points. First, by using an isolation valve closing setpoint of 2170 psig, ESFAS will not be actuated if nominal (as designed) trip setpoints are used. Premature isolation valve closure during normal PORV operation will also be prevented on more than 95% of the isolation valve challenges. Second, PORV and safety valve failure rates will be limited to 1.66×10^{-4} (TVA)/1.26×10^{-4} (WPPSS) and 9.73×10^{-6} failures per reactor year, respectively. At these levels, neither component can be considered a significant contributor to the probability of a small break LOCA. Finally, the demand frequency analysis indicates that a main turbine trip will generate about 1.12 PORV lifts per reactor year (about 26% of the total demand). However, the additional challenges do not significantly affect the reliability of the automatic PORV isolation system.

1.4. Organization

In order to logically evaluate the PORV isolation system, the body of this report is organized as follows. First, the basic conceptual design of the automatic PORV isolation system is described briefly to clarify system operation. Next, a block valve setpoint analysis is included to justify the closing setpoint choice. Given this setpoint, the demand frequency of the PORV and safety valves are predicted for various overheating/overcooling transients. With these predictions, the reliability of the PORV and safety valves is discussed. Finally, the post-TMI requirement of an anticipatory reactor trip on main turbine trip is evaluated objectively.

2. SYSTEM DESCRIPTION

The PORV has been deemed a probable source of failure that could lead to a small break LOCA. Should the PORV stick open or fail to reseat properly, coolant could be lost continuously from the RCS. A PORV relief path isolation system was designed to mitigate this event. The isolation system must function automatically to block the PORV whenever coincident "PORV flow" and low RC pressure signals are received. The system must also provide manual overrides for all automatic functions and allow the isolation valve to be opened by manual means alone. Within this framework, failure to close the PORV relief path must be less than 1×10^{-3} failures per reactor year to keep the system from being considered a significant contributor to the probability of a small break LOCA.

On 205-FA units, the PORV isolation system will consist of a single PORV mounted downstream from a block valve with an automatic closure feature. Original design setpoints will be used to ensure normal PORV operation. For a typical transient, an overheating event for example, the system response can be anticipated. Under design conditions, as the RC pressure rises above 2295 psig, the PORV opens to limit additional pressure increases. Following the transient the RC pressure will drop below 2270 psig, and the PORV will close to maintain RC pressure.

For off-design operation, the PORV may fail to open or may open but fail to close. If the PORV fails to open and the RC pressure reaches 2355 psig, the high-pressure RPS will trip the reactor. On the other hand, the PORV may open but fail to close when RC pressure drops below the 2270-psig closing setpoint. If the pressure continues to drop to 2170 psig and the PORV remains open, the block valve will close to maintain RC pressure. Should the block valve fail to close, the RPS will trip on low RC pressure at 1987 psig (TVA)/2000 psig (WPPSS).

3. PORV ISOLATION VALVE SETPOINT

Since the PORV failure at TMI-2, an automatic PORV isolation system has been proposed to increase system reliability. For proper operation, the PORV opening and high-pressure reactor trip setpoints must be maintained at their original design values. An isolation valve closing setpoint of 2170 psig (100 psi below the PORV closing setpoint) was originally recommended to prevent unnecessary cycling of the isolation valve. This setpoint should also prevent low RC pressure ESFAS actuation and prevent lifting of the code safeties for most transients. The following analysis is included to verify that the 2170-psig block valve closing setpoint satisfies all three design criteria.

Closure of the isolation valve during normal PORV operation defeats the original purpose of the PORV. The pressure sensors for the PORV and the isolation valve are located in the pressurizer and at the hot leg tap, respectively. Due to elevation differences and frictional losses during transients, a pressure difference exists between the two sensors which may cause premature isolation valve closure.

To evaluate the effects of this pressure difference, a Monte Carlo simulation was performed. POWER TRAIN V runs supplied representative pressure differentials between the PORV and isolation valve closing setpoints for various transients. The Monte Carlo simulation utilized a range of representative pressure differentials and accounted for instrument errors. This analysis predicted the probability of an isolation valve closure, prior to PORV closure, to be less than 5%. Consequently, the present 2170-psig setpoint should allow normal PORV operation, prevent unnecessary cycling of the isolation valve, and automatically mitigate a failed-open PORV small break LOCA.

The closing setpoint of 2170 psig prevents low RC pressure ESFAS actuation. Overheating and overheating/overcooling transients run on the hybrid computer code PT-V verify this value. Table 1 lists the nominal and error-adjusted setpoints used in the analysis. On the TVA model, an error-adjusted closing setpoint of 2120 psig (110 psi below the actual 2230-psig PT-V setpoint) prevents reactor trips on low RC pressure for most transients. The following will probably trip the reactor on low RC pressure:

Trip one RC pump at 100% EOL

Trip one RC pump at 80-100% BOL

However, with a reactor trip-induced pressure drop of approximately 200 psi, the lowest RC pressure achieved is approximately 1885 psig. This is 75 psi above the low RC pressure ESFAS setpoint, 1840 psig (error-adjusted). Hence, even with a lower setpoint (2120 versus 2230 psig) and a low RC pressure reactor trip, low RC pressure ESFAS actuation does not occur.

In addition, the WPPSS PT-V model verifies the setpoint of 2170 psig. Closing setpoints of 2180 and 2215 psig were used on the WPPSS model. When using the 2180-psig setpoint, the reactor trips on low RC pressure following a turbine trip with error-adjusted setpoints. The lowest pressure produced following the trip is 1865 psig. This pressure is 45 psi above the ESFAS actuation setpoint of 1820 psig (error-adjusted). When using 2180 or 2215 psig as the closing setpoint, tripping one of two feedwater pumps will trip the reactor on low pressure and possibly actuate the ESFAS. The lowest pressure produced is 1775 psig, 45 psi below the error-adjusted ESFAS setpoint (1820 psig), but 15 psi above the nominal ESFAS setpoint (1760 psig). A feedwater pump trip with a coincident failed PORV inherently seems to trip the reactor on low pressure and actuates ESFAS, regardless of the isolation valve closing setpoint. Therefore, with the possible exception of a feedwater pump trip, a closing setpoint of 2170 psig prevents low RC pressure ESFAS actuation.

In addition, the isolation valve closing setpoint is low enough to prevent lifting of the pressurizer safety valves. Repressurization of the RCS occurs after closing the isolation valve. With the PORV now blocked, only the pressurizer spray and the high-pressure reactor trip can decrease RC pressure. The highest repressurization occurs for an RC pump trip transient on the WPPSS model. In this case, pressurizer pressure may reach 2305 psig. A further increase in pressure will trip the reactor on high RCS pressure. Hence, the high-pressure reactor trip ensures that repressurization will never lift the code safety valves.

If the 5% probability that the isolation valve will interfere with normal PORV operation is unacceptable, the PT-V analysis can be used to verify another setpoint. Preliminary PT-V results indicate that the lowest nominal closing setpoint that can be justified is 2060 psig, while the lowest error-adjusted (low side) setpoint is 2110 psig. Thus, the present analysis can be used to select and justify a setpoint lower than 2170 psig.

In summary, the PORV isolation valve closing setpoint of 2170 psig satisfies all design criteria. This setpoint prevents low RC pressure ESFAS actuation and prevents lifting of the pressurizer code safety valves. In addition, normal PORV operation is preserved, while unnecessary cycling of the isolation valve is prevented.

Table 1. Setpoints for PORV Isolation Valve Closing Setpoint Analysis

	TVA setpoints, psig		WPPSS setpoints, psig	
	Nominal	With NAIEs[a]	Nominal	With NAIEs[a]
PORV block valve closing	2170 (2230)[b]	2120 (2180)	2170 (2230)	2120 (2180)
RPS low RC pressure	1987 (2047)	2012 (2072)	2000 (2060)	2025 (2085)
Low RC pressure ESFAS	1700 (1760)	1750 (1810)	1700 (1760)	1760 (1820)

[a] NAIEs: Non-accident instrument errors.

[b] Setpoints in parentheses are those used in POWER TRAIN V; 60 psi has been added to this setpoint to translate the setpoint from the hot leg top to the tap of the core.

4. PORV/SAFETY VALVE DEMAND FREQUENCY

In contrast to the operating 177-FA plants, the 205-FA design requires that the PORV setpoint be lower than the high-pressure reactor trip setpoint. This alignment increases the number of PORV challenges and raises questions about the reliability of the PORV and the safety valves. Operating experience from 177-FA plants (prior to the TMI-2 incident) indicates that a variety of transients may lift the PORV. Similar transients at the 205-FA plants should also generate PORV lifts. The following analysis predicts the number of PORV/safety valve lifts on the 205-FA units for transients in which either or both valves lift. With these demand requirements, the reliability of the PORV and the safety valves can be ascertained.

Challenges to the PORV and/or safety valves depend on the specific transient and plant being considered. Differences between the 205- and 177-FA plants eliminate the loss-of-main-feedwater transient. The anticipatory reactor trip on loss of both main feedwater pumps and on high flux/feedwater flow ratio should trip the 205-FA reactor before the PORV lifts. Also, differences between the TVA and WPPSS plants result in different transient lists for each plant. TVA's interlock to trip the reactor upon turbine trip — if reactor power is greater than 76% — eliminates a turbine trip from the transient list for TVA above 76% power. Based on 177-FA operating experience and plant differences, the resultant transient list includes the following:

Turbine trip with reactor trip (TVA > 76% reactor power)
Turbine trip without reactor trip
Trip one FW pump
Trip one RC pump
Trip two RC pumps (one per loop)
Load rejection
Ramp one FW valve 50% closed
Rod drop
Overcooling with HPI/MU repressurization

This list, consisting primarily of moderately frequent events, does not include random instrument failures that occur as a result of hardware failures or human error.

Two computer programs were used to determine the number of PORV and safety valve lifts. POWER TRAIN V (PT-V), a hybrid code, determines the number of PORV and/or safety valve lifts for overheating transients. The TVA PT-V model was used for both the TVA and WPPSS plants. This is justified since the differences in heat generation and removal between the two plants tend to offset each other. Comparison of a few WPPSS runs and the TVA runs verifies this point. Since PT-V cannot model high-pressure injection, KRPZ, a non-equilibrium pressurizer code, was used. KPRZ ascertains the number of PORV and/or safety valve lifts for overcooling events with HPI/MU repressurization.

The overheating transients run on PT-V (TVA model) gave the number of PORV lifts. Table 2 shows the number of PORV lifts for beginning-of-life (BOL) and end-of-life (EOL) conditions. The results indicate the maximum number of definite lifts plus or minus the number of possible lifts. The number of possible lifts represents variations in the PORV setpoint and in plant conditions at the beginning of the transient. These variations can cause peak pressures that previously missed the PORV setpoint, but later actuate the PORV in the same transient. In determining the PORV lifts, PT-V limits were observed and proper AFW actuation and control were assumed. These lifts are valid over the reactors' 70-100% power range. Below 70% power, the PORV lifts approach zero since the plant, with the aid of the ICS, can handle RC pressure upsets without challenging the PORV. Consequently, the majority of the PORV lifts will occur at high power levels.

PT-V and KPRZ provide the number of lifts for the overcooling events with HPI/MU repressurization. PT-V models overcooling transients prior to ESFAS actuation. Pressurizer conditions (such as pressure, level, insurge, temperature, etc.) from PT-V enable KPRZ to model post-ESFAS events. Insurge flow was assumed to be primarily due to high-pressure injection. The modeling also assumed that the operator throttles HPI 10 minutes after ESFAS actuation in an effort to control pressurizer level and subcooled margin. Post-ESFAS events modeled on KPRZ predict that an HPI repressurization will generate 129 ± 13 PORV lifts. The normal repressurization due to makeup flow following a reactor trip is controlled by the pressurizer spray. In this case, the PORV is not challenged. Therefore, only the overcooling with HPI repressurization lifts the PORV and may lift the pressurizer safety valves.

The same transients were repeated with the PORV blocked. For the overheating transients, the pressurizer safety valves do not lift since the reactor trips on high RC pressure, and auxiliary feedwater controls steam generator level to remove decay heat. For overcooling with makeup repressurization, the pressurizer spray maintains pressure below the PORV setpoint. Therefore, the safety valves do not lift for this transient either. Overcooling by HPI repressurization was the only transient that lifted the safety valves. As with the operable PORV case, the operator throttles HPI to control level 10 minutes after HPI begins. Throttling HPI limits the safety valve lifts to 15 ± 2 lifts per valve. Therefore, only overcooling with HPI repressurization will lift a safety valve.

Since both the PORV and the safety valves may be challenged, the lifts may be coincident, or out of phase. Both operable and inoperable PORVs were considered. With an operable PORV, the time difference between the two lifts is not applicable since the PORV or the pressurizer spray (overcooling with makeup repressurization) maintains pressure below the safety valve setpoint. For an inoperable PORV with overcooling and makeup (MU) repressurization, the pressurizer spray again maintains pressure below the safety valve setpoint. As a result, the time difference between lifts is again not applicable. However, for an inoperable PORV with overcooling by HPI repressurization, both safety valves do lift. In this case, the valves lift approximately 145 seconds apart (about 2.5 minutes).

In conclusion, input to the PORV reliability analysis consists of transients that lift the PORV, the number of PORV/safety valve lifts, and the time differences between PORV and safety valve lifts. Operating experience on 177-FA plants has provided the basis for the transient list. KPRZ indicates that the only transient that lifts the safety valves occurs for an inoperable PORV with HPI/MU repressurization. None of the overheating transients lifts the safety valves. However, note that the number of valve lifts should be regarded as representative of the expected number of lifts since no operating data are available.

Table 2. PORV Lifts

Transient	Lifts/demand,[a] BOL	Lifts/demand,[a] EOL	No. of lifts/yr[b]
Turbine trip w/ reactor trip	0 ± 0 > 76% pwr 1 ± 1 < 76% pwr	0 ± 0 > 76% pwr 1^{+0}_{-1} < 76% pwr	0 Negligible
Turbine trip w/o reactor trip	1 ± 1	1^{+0}_{-1}	1.12
Trip one FW pump	4^{+1}_{-4}	1^{+1}_{-0}	0.92
Trip one RC pump	2^{+0}_{-1}	2^{+0}_{-1}	0.04
Trip two RC pumps	1 ± 0	1 ± 0	Negligible
Load rejection	1 ± 0	1 ± 0	0.10
Ramp one FW valve 50% closed	2^{+0}_{-1}	1^{+1}_{-0}	0.91
Overcooling			
HPI repress'n[c]	129 ± 13		0.51
MU repress'n	0 ± 0		0
Rod drop			
0.09% $\Delta k/k$	2^{+1}_{-0}		0.74
0.06% $\Delta k/k$	2^{+1}_{-0}		
0.03% $\Delta k/k$	2^{+1}_{-0}		

[a] These lifts are valid over the power range from 70 to 100%. Below 70% power, the lifts will go to zero.

[b] Predictions made with point estimates for BOL.

[c] Includes operator corrective action.

5. PORV RELIEF PATH RELIABILITY

Having specified a PORV demand history, the reliability of the 205-FA automatic PORV isolation system can be evaluated. To meet NRC requirements, failure to isolate the PORV relief path must not appreciably impact the value of 1.0×10^{-3} failures per reactor year. Isolation of the PORV does increase the demand on the pressurizer code safety valves, however. As a result, safety valve reliability must also be evaluated, as discussed in section 6.

The probability of PORV isolation system failure was determined using a fault tree analysis. Fault trees were constructed for two classes of initiating events: pressure transients and spurious system operation. A statistical analysis was also performed, which predicted the PORV's challenge frequency. Dominant cut sets for each fault tree were obtained using the fault tree analysis program FTAP. With PORV challenge frequency and FTAP results as input, the SAMPLE code was used to predict the distribution of system failures.

Failure data and initiating event frequencies are listed in Appendixes C and D.

To evaluate the reliability of the PORV isolation system, the analysis was organized as follows: statement of assumptions, fault tree analysis, human reliability analysis, PORV challenge frequency, failure data, uncertainty analysis, and definition of mission success.

In any complex problem, simplifying assumptions are a necessity. For the automatic PORV isolation system, the following assumptions were made:

1. Degraded failures were not considered. That is, components were assumed to operate properly or were treated as failed.
2. Failures of passive components, such as test points, were disregarded due to their infrequent occurrences.
3. A monthly equipment test interval was assumed. Therefore, interim failures would not be discovered until the succeeding test.

4. Operator errors of commission were not included in the fault tree.

5. The failure rate for the block valve was based on an average electric-motor-operated gate valve of that size and operator.

6. Target Rock valves have experienced 125,000 total cycles (100,000 bench test and 25,000 field experience) on the pressurizer spray with no failures. Since the spray valve is not subjected to the same environment as the PORV, the value of zero failures in 25,000 cycles was used in the Bayesian updating procedure. This procedure uses the prior experience of the Dresser PORV (4 failures in 400 demands) and the evidence of zero failures in 25,000 cycles to arrive at a modified value for the Target Rock valve in the PORV application.

A fault tree analysis, consistent with the methodology described in the Fault Tree Handbook (NUREG-0492), was used to evaluate the reliability of the PORV/PORV block valve system. The fault trees for this system are included in Appendix A. The GRAP software package (graphic reliability analysis package) was used to construct and evaluate the fault trees. Fault trees were constructed with enough detail to identify the components that are dominant contributiors to system failure. No attempt was made to account for failures due to external events, such as fires, floods, or earthquakes.

The FTAP code was used for identification of minimum cut sets, quantification of the fault trees, ranking of basic event importance, and identification of major contributors to system failure.

A human reliability analysis (HRA) was also performed, which was consistent with the methodology described in NUREG/CR-1278. The basic human error probabilities used in this analysis are found in Chapter 20 of the Handbook. Probability tree diagrams for the human tasks of interest are presented in Appendix B.

With the framework of the fault tree and human reliability analysis set, the PORV demand frequency was predicted. PORV lifts were initiated using seven transient sources. The number of lifts for each source, in a specified period of time, is described by a Poisson distribution. Each PORV lift may result in one or more cycles. The number of cycles for each source is described by a multinomial distribution. This distribution changes linearly from the beginning to the end of the year. The statistical treatment involved combining the Poisson and multinomial distributions to describe the random number of cycles. Thereafter, the frequency of one, two, etc. cycles could be obtained, regardless of the source, by means of simulation.

The complete list of generic data used in this analysis is given in Appendixes C and D. Failure data and initiating event frequencies were obtained from various sources. Repair times for components in the power distribution sytem were supplied by plant personnel.

An uncertainty analysis was also performed. The SAMPLE code was used to evaluate uncertainties in the system unavailability results. Range factors obtained from the Reactor Safety Study were used to construct lognormal distributions. These distributions were localized around the point-estimate failure probabilities of the dominant unavailability contributors. Three parameters influenced the form of the sample function used in this analysis. The form depended on the product of two terms, the simulated PORV demand frequency and the system response to the pressure transients, plus the contribution due to spurious system operation. The uncertainties surrounding system unavailability were evaluated in terms of the mean, the 5%, and the 95% levels of system probability distribution.

To finally judge the PORV isolation system, a formal definition of mission success is required. Mission success can be defined in terms of either system operation or reliability. In terms of system operation, mission success is defined as the ability to isolate the PORV relief path prior to low RC pressure ESFAS actuation (1700 psig). System failure, therefore, is defined as any failure within the system boundaries that results in depressurization to the ESFAS actuation setpoint. In terms of reliability, the NRC requires a ceiling failure rate of 1.0×10^{-3} failures per reactor year for small break LOCAs. To provide a margin of safety, B&W has used a figure of 3.0×10^{-4} failures per reactor year. Consequently, system failure in this case is defined as a system with a probability of failure greater than 3.0×10^{-4}. With these definitions, mission success can be evaluated for the systems considered.

The results of this study indicate that the 205-FA automatic PORV isolation system satisfies both definitions of mission success. Operationally, the isolation system (with original design trip setpoints) prevents low RC pressure ESFAS actuation, effectively modulates RC pressure, reduces unnecessary reactor trips, and increases plant availability. From a reliability standpoint, the results are given in Table 3 at the mean, 5%, and 95% confidence levels. At the 95% confidence level, for example, failure to isolate the PORV relief path is limited to 1.66×10^{-4} (TVA)/1.26×10^{-4} (WPPSS) failures per reactor year. Therefore, both backlog plants will easily achieve the NRC reliability requirements.

Aside from strict design criteria, two other aspects of the design are worth mentioning. The results indicate that the Target Rock valves are extremely reliable and that the presence of the ATOG displays and PORV position switch in the control room increase operator awareness. However, there is one distinct drawback to this design. Improved isolation of the PORV relief path inevitably leads to elevated safety valve demand as discussed in section 6.

Table 3. PORV Automatic Block Valve Isolation System Failure Probability and Confidence Limits

	Failure probability/year		
	Mean	5% confid. limit	95% confid. limit
TVA	6.00×10^{-5}	1.31×10^{-5}	1.66×10^{-4}
WPPSS	4.99×10^{-5}	1.35×10^{-5}	1.26×10^{-4}

6. SAFETY VALVE RELIABILITY

A reliable automatic PORV isolation system had been developed for the 205-FA plants. With this system, isolation of the PORV relief path is maximized. Isolation of the PORV, however, should increase demand on the pressurizer code safety valves. Consequently, a safety valve reliability analysis was conducted.

A small break LOCA due to a failed-open safety valve may occur along either of two pathways. The pathways identified include overcooling with subsequent repressurization and overheating transients.

To quantify the LOCA probabilities, event sequences were constructed for the overcooling scenario and for three overheating events. The event sequences and supporting failure data are listed in Appendix E. The overcooling transient was initiated by assuming that the ESFAS actuates on low RC pressure. No attempt was made to predict the frequency of occurrence of the three overheating events analyzed. This method was chosen because the existing auxiliary feedwater designs are very reliable and, in the event of a total loss of feedwater, HPI feed along with some form of pressurizer bleed would be used to cool the core.

The following assumptions were used in analyzing the overcooling scenario:

1. The PORV relief path is isolated.
2. After 10 minutes of inadvertent HPI operation, the probability that the operator will throttle HPI and realign normal makeup is 1.0.
3. There is some type of uncertainty as to the type of discharge passed through the safety valves. However, a conservative failure estimate can be made by assuming that the discharge is water or two-phase (worst case).

Failure rates for the pressurizer safety valves (PSVs) can be ascertained by examining the failure rates of the main steam safety valves (MSSVs). This is possible because both operate on the same principle; i.e., they both work against the closing force of a spring, and they both require an additional sudden opening force when they reach their trip setpoints.

Differences between the PSV and MSSV must also be pointed out:

- The fluid passing through a PSV should contain fewer suspended particulates than that passing through an MSSV.
- The PSV is stainless steel whereas the MSSV is predominantly carbon steel. Rusting of the carbon steel will introduce additional foreign matter into the fluid.
- The PSV is an ASME Class I component, while the MSSV is an ASME Class II valve.
- The PSV must operate with a variable backpressure, while the MSSV operates with a fairly constant backpressure. As a result, the PSV design is more sophisticated and has more components that may fail.

The first three differences suggest that the PSV may have a lower failure rate than the MSSV, while the last point suggests the opposite.

Cumulative B&W operating experience indicates that there have been aproximately 2850 MSSV demands. In all these cases, there has not been a single failure due to a valve reseating problem (remain in full-open position). A failure rate based on zero failures in 2850 demands was computed using a χ^2 50% level test. The calculated failure rate for the steam relief was found to be 2.43×10^{-4} per demand. The failure rate for water relief was estimated to be 100 times larger than for steam relief, i.e., 2.43×10^{-2} per demand.

The safety valve failure rate was determined using a Bayesian updating procedure. The prior distribution was assumed to be lognormal with a mean of 2.43×10^{-2} per demand. This lognormal distribution was then combined with the evidence of five safety valve water demands with no failures to determine the probability of failure. Four EPRI safety valve test programs (September 1981) and a single demand at Crystal River 3 (February 26, 1980) accounted for valve performance history.

The results of this investigation indicate that an uncontrolled small break LOCA through the pressurizer code safety valves is not a probable event. During the course of this analysis, two paths were identified as dominant contributors to the probability of a safety valve failure. These are overcooling with subsequent repressurization and overheating transients. The probability of a LOCA due to overcooling events was found to be 9.73×10^{-6} per reactor year, while the cumulative frequency of occurrences for the overheating transients was calculated to be $6.2/ \times 10^{-5}$ per reactor year. In addition, the unavailability of the PORV relief path was estimated to be 7.23×10^{-3} per year.

The impact of the automatic PORV isolation system on safety valve reliability is insignificant because the unavailability of the PORV relief path is so low. The automatic isolation system achieves all operational requirements and NRC-mandated reliability requirements as originally designed.

7. ANTICIPATORY REACTOR TRIP ON TURBINE TRIP

Following the PORV failure at TMI-2, the NRC required PORV system modifications on all operating plants. Changes were made to the PORV opening and high pressure reactor trip setpoints. The addition of an anticipatory reactor trip on main turbine trip was also required. These modifications have decreased PORV challenges, but have concurrently increased the number of reactor trips (through RPS challenges). The intent of these modifications was to reduce PORV challenges and thus reduce the probability of a PORV failure. However, the probability of PORV failure can be reduced using alternative approaches that do not detract from plant performance.

On all 205-FA units, an automatic PORV isolation system using pre-TMI-2 (as-designed) trip setpoints has been proposed. This system consists of a single PORV and a single block valve with an automatic closure feature. The use of the original design trip setpoints will ensure normal PORV operation, reduce reactor trips, and increase plant availability. However, the question of the anticipatory reactor trip upon main turbine trip still remains.

The anticipatory reactor trip upon turbine trip was mandated to help reduce the number of PORV challenges. Operating experience verifies that it has achieved this objective, but at the expense of plant availability. However, with the improved 205-FA design, it is no longer necessary to limit PORV challenges.

The annual PORV challenge rate was predicted for both backlog plants at BOL conditions (worst case). The annual challenge rate depends on two factors; the number of challenges per transient and the number of transients per reactor year. The results of these calculations are given in Table 2.

Three operating regimes exist in the TVA plant since it was designed with an interlock to trip the reactor upon main turbine trip (provided reactor power is greater than 76%). Above 76% power, a turbine trip followed by a reactor trip will generate zero PORV lifts. From 70 to 76% power, a turbine trip will generate an insignificant number of lifts since the reactor rarely operates in this power range. Below 70% power, PORV lifts due to all causes approach zero.

On the WPPSS plant, only two operating regimes exist since the WPPSS plant was designed without a mandatory trip. Above 70% power, a turbine trip will generate 1.12 PORV lifts per reactor year, and below 70% power PORV lifts due to all causes again approach zero.

The number of PORV challenges due to a turbine trip has been predicted as 1.12 per reactor year. The addition of an anticipatory reactor trip on turbine trip can reduce this number to zero. Projected yearly PORV demand due to all causes should be in the 4-5 challenge range. With the addition of the automatic PORV isolated system, the NRC-mandated reliability requirements can be easily achieved, even with turbine trip-induced PORV challenges. Therefore, the need for the anticipatory reactor trip on main turbine trip seems marginal at best.

The post-TMI modifications to the PORV relief path system must be re-evaluated. They represent but one way to reduce the probability of a PORV failure (reduced PORV challenges). They also tend to increase the number of RPS challenges, increase the number of reactor trips, and reduce plant availability. B&W's automatic PORV isolation system will achieve the NRC's PORV reliability requirements without these modifications. As a result, the PORV will be able to control RC pressure for minor overpressure events and avoid the unnecessary reactor trips, which have been a consequence of the post-TMI modifications.

8. CONCLUSIONS

An automatic PORV isolation system has been designed for B&W's 205-FA units. The system will operate reliably to increase plant availability by reducing the number of reactor trips. This will be accomplished using pre-TMI-2 trip setpoints to ensure proper RC pressure control and reduced RPS challenges.

In addition, five significant conclusions can be drawn from the supporting analysis:

1. A block valve closing setpoint of 2170 psig will not actuate the ESFAS using nominal trip setpoints, but it will prevent premature isolation valve closure on 95% or more of the isolation valve challenges.
2. The PORV should be challenged annually on approximately 4.34 occasions on WPPSS and 3.22 occasions on TVA.
3. The number of PORV challenges due to a turbine trip represents about 26% of the total demand.
4. By using the automatic PORV isolation system, the probability of failing to isolate the PORV relief path will be limited to 1.66×10^{-4} (TVA) 1.26×10^{-4} (WPPSS) failures per reactor year. The NRC requires a failure rate of 1×10^{-3} failures per reactor year for isolation of the PORV relief path.
5. The reliability of the pressurizer code safety valves will not be significantly affected by the isolation system. With the automatic PORV isolation system installed, the probability of a safety valve failure will be 9.73×10^{-6} failures per reactor year.

9. RECOMMENDATIONS

Based on the supporting system justification, B&W recommends that the automatic PORV isolation system be installed on all 205-FA units as designed. In addition, the post-TMI requirement of an anticipatory reactor trip on main turbine trip should be abolished. Even though turbine trip-induced PORV challenges represent a significant portion of the total demand, the 205-FA design can suitably isolate the PORV relief path at the enhanced rate. In addition, unnecessary reactor trips will be avoided, and plant availability will be increased by the elimination of the mandatory reactor trip.

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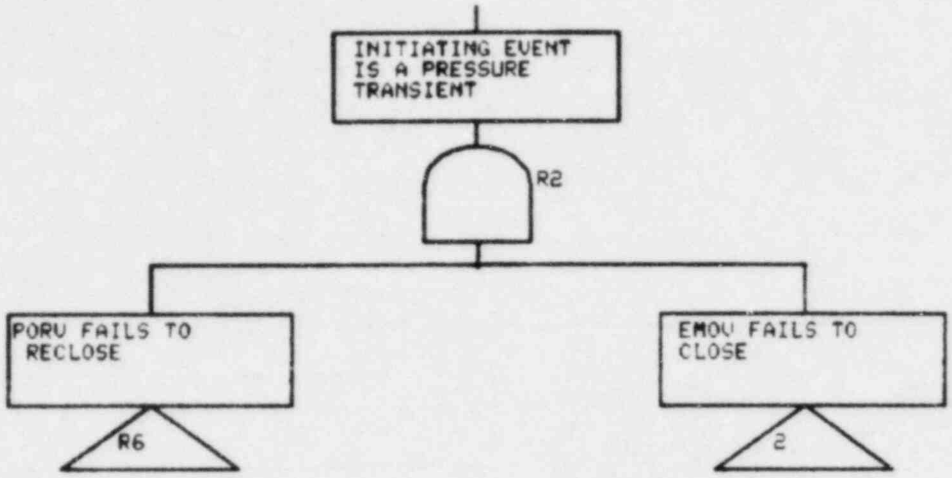
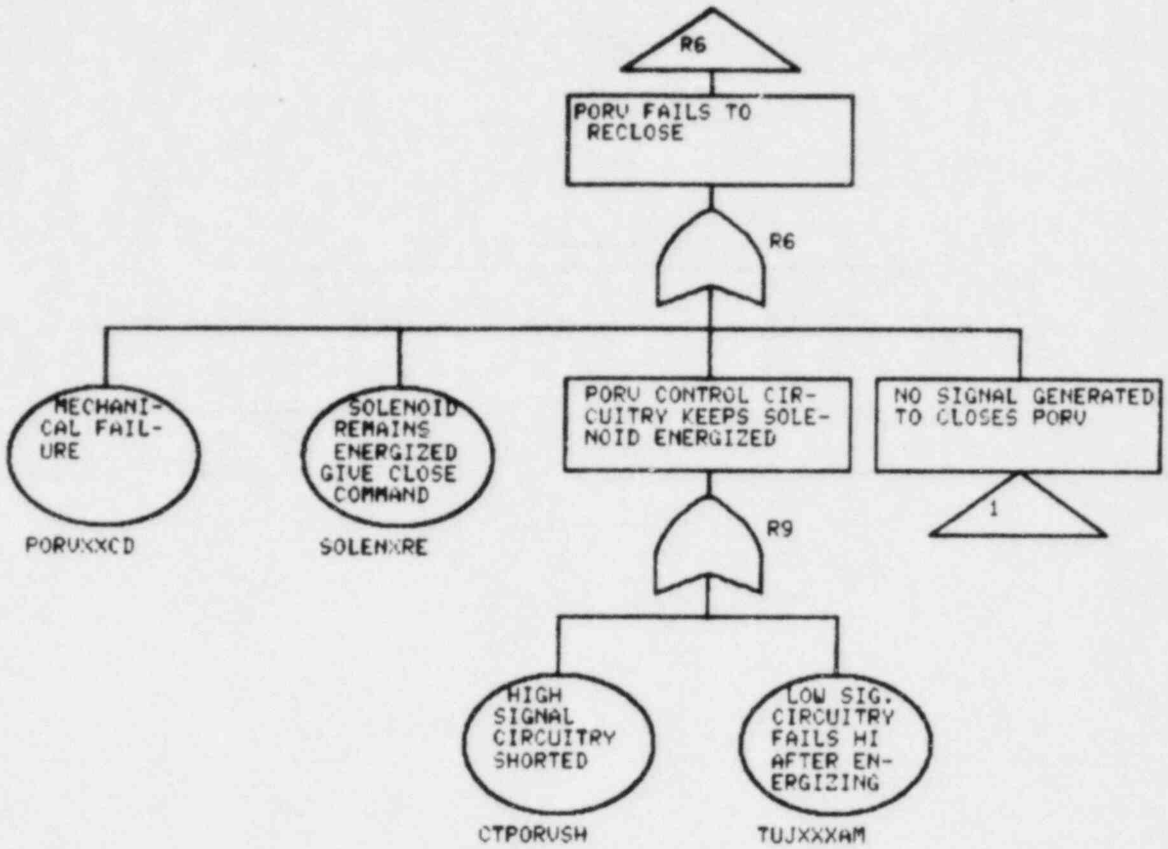
APPENDIX A

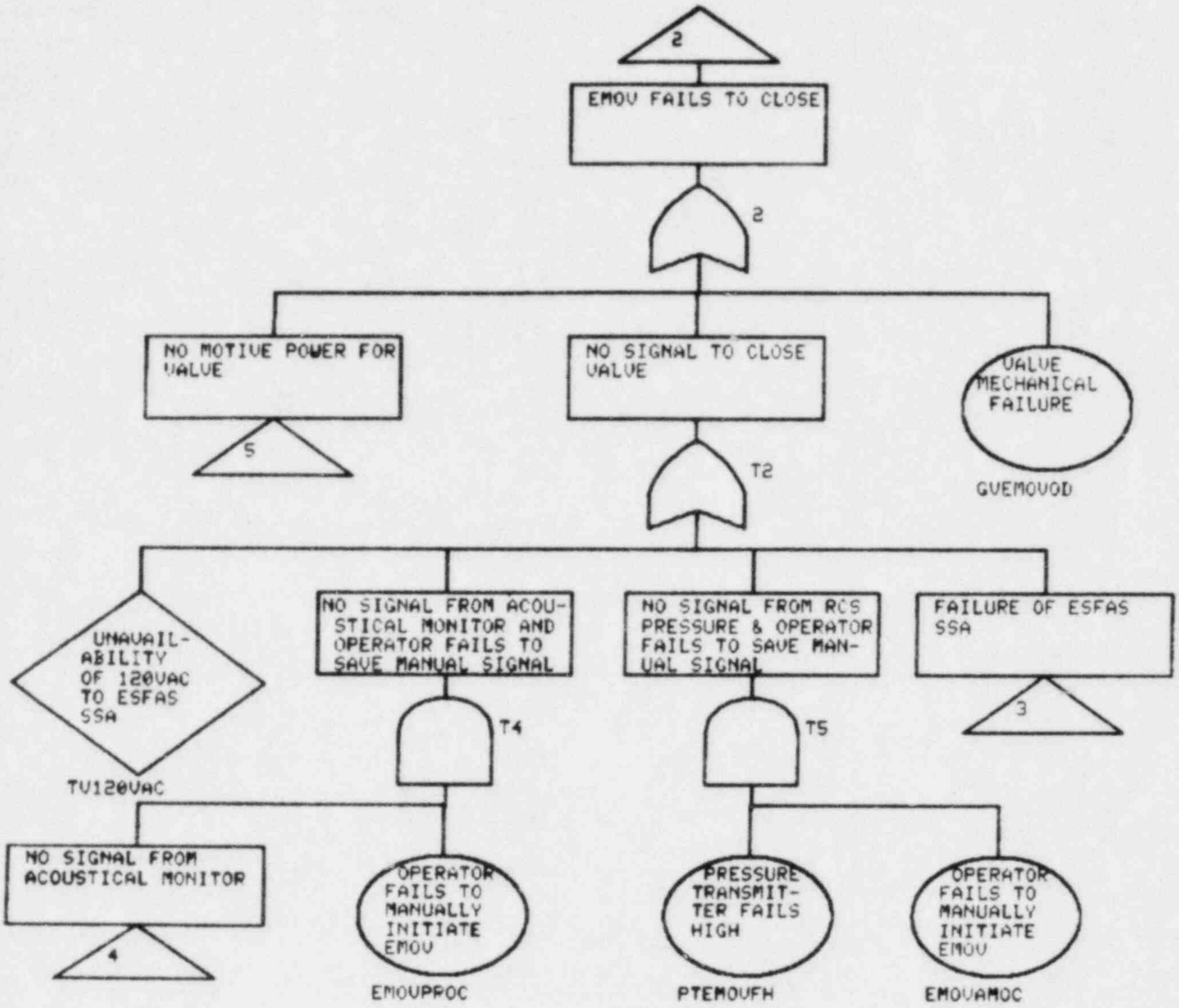
System Fault Trees

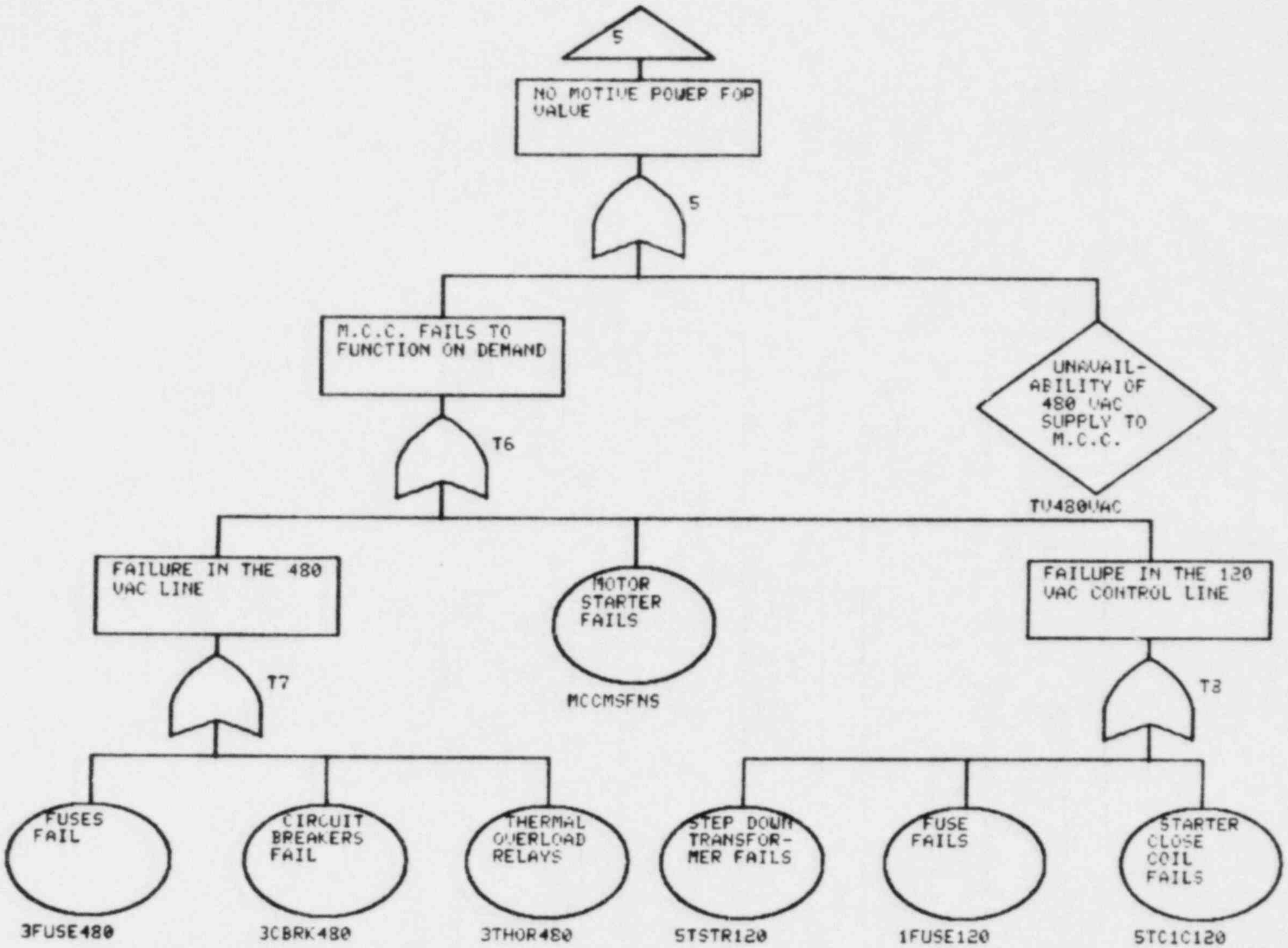
Top event	Sum of implicants
Initiating event is a pressure transient	1.29×10^{-5}
Initiating event is spurious PORV opening	2.78×10^{-5}
PORV relief path unavailable	7.23×10^{-3}

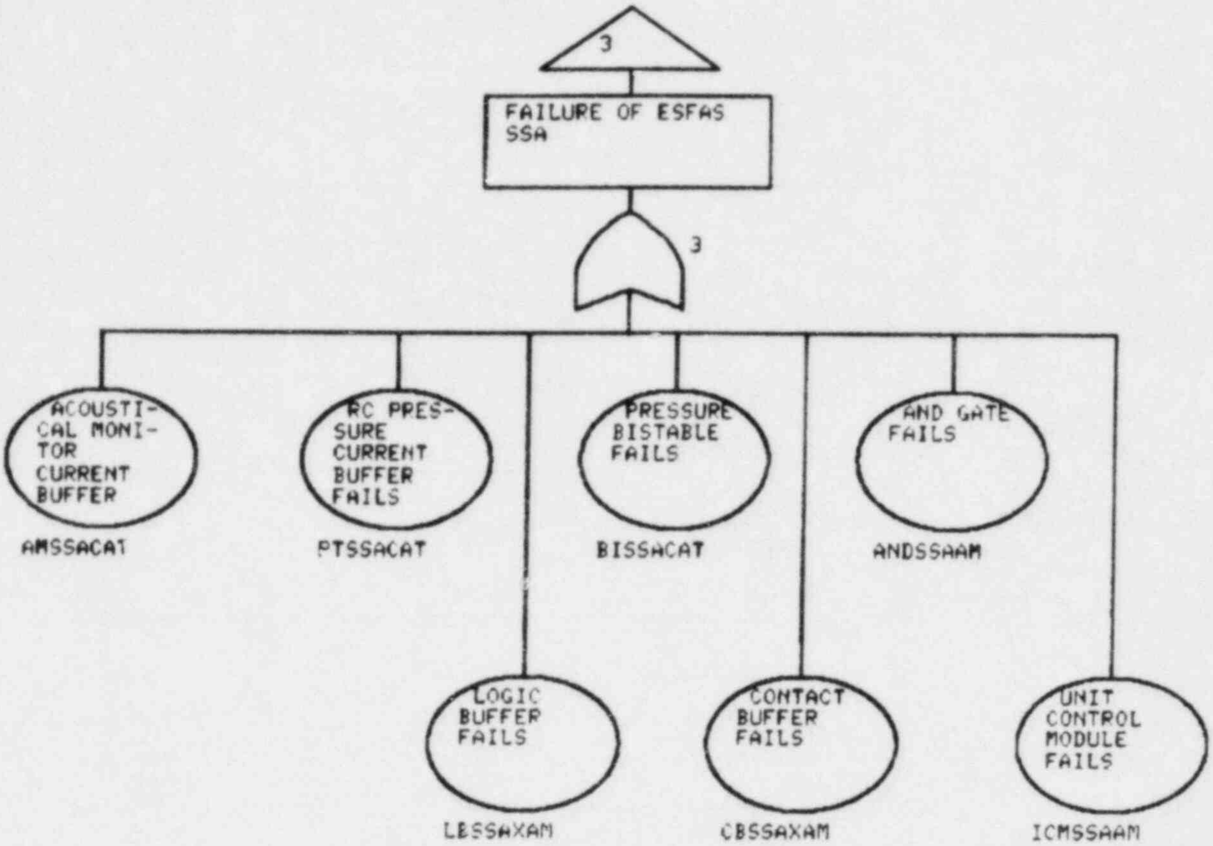
Note: These fault trees are representative of the TVA system design. The structure of the WPPSS trees is nearly identical except that the basic event "AMEMOVAM" (acoustic monitor fails) is replaced by "PSEMOVAM" (position switch fails).

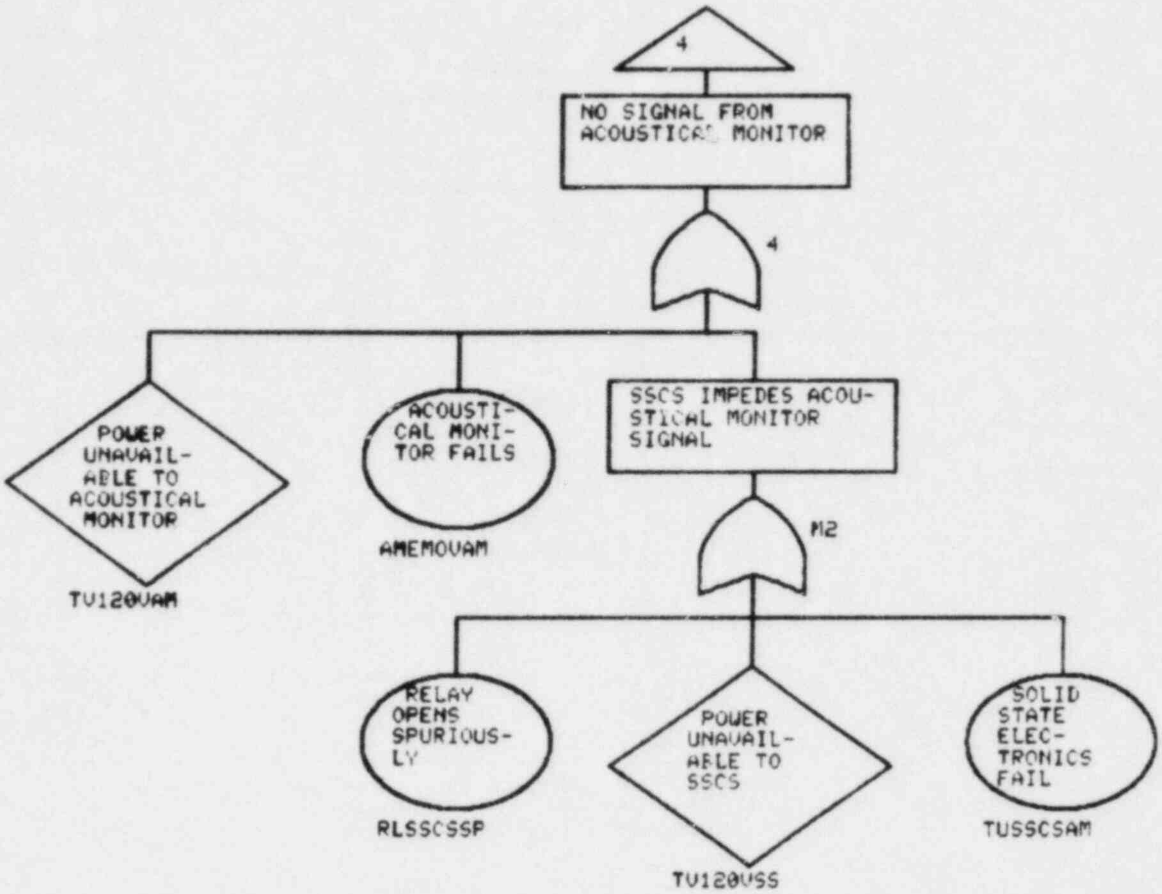

INITIATING EVENT
IS A PRESSURE
TRANSIENT
R2
PORV FAILS TO
RECLOSE
R6
EMOV FAILS TO
CLOSE
2

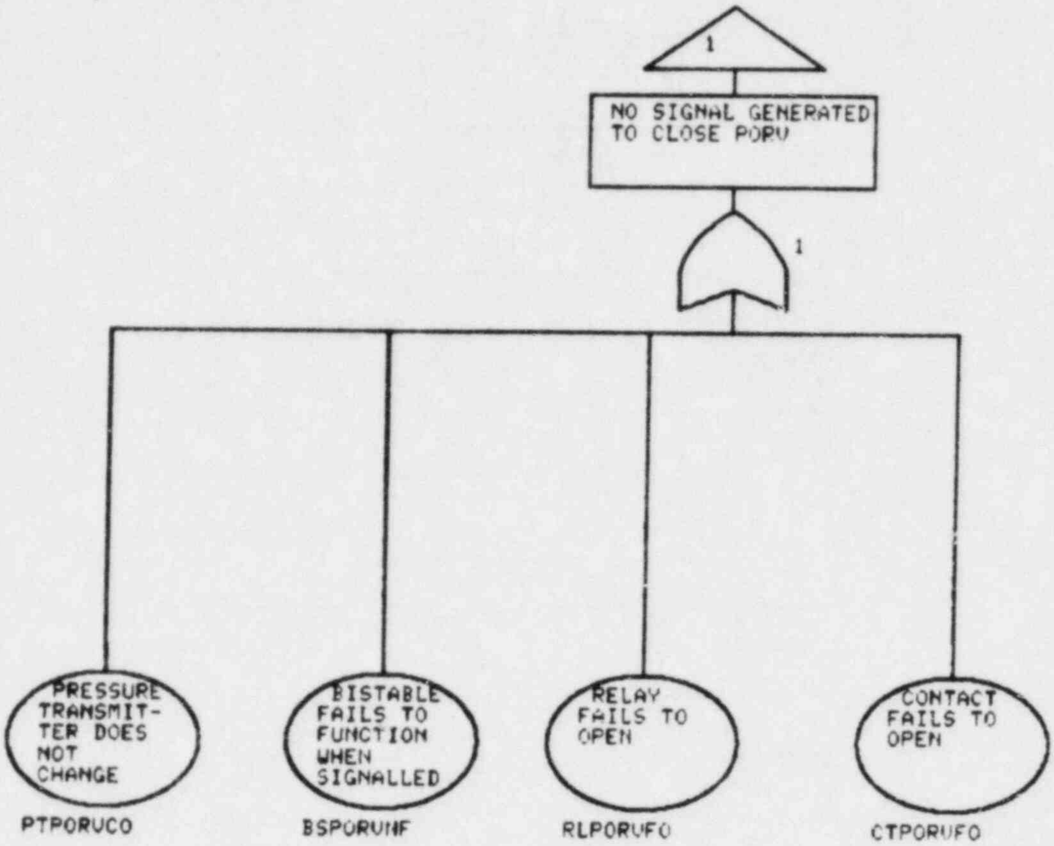

R6
PORV FAILS TO RECLOSE
R6
MECHANICAL FAILURE
PORVXXCD
SOLENOID REMAINS ENERGIZED GIVE CLOSE COMMAND
SOLENXRE
PORV CONTROL CIRCUITRY KEEPS SOLENOID ENERGIZED
NO SIGNAL GENERATED TO CLOSES PORV
1
R9
HIGH SIGNAL CIRCUITRY SHORTED
CTPORVSH
LOW SIG. CIRCUITRY FAILS HI AFTER ENERGIZING
TUJXXXAM

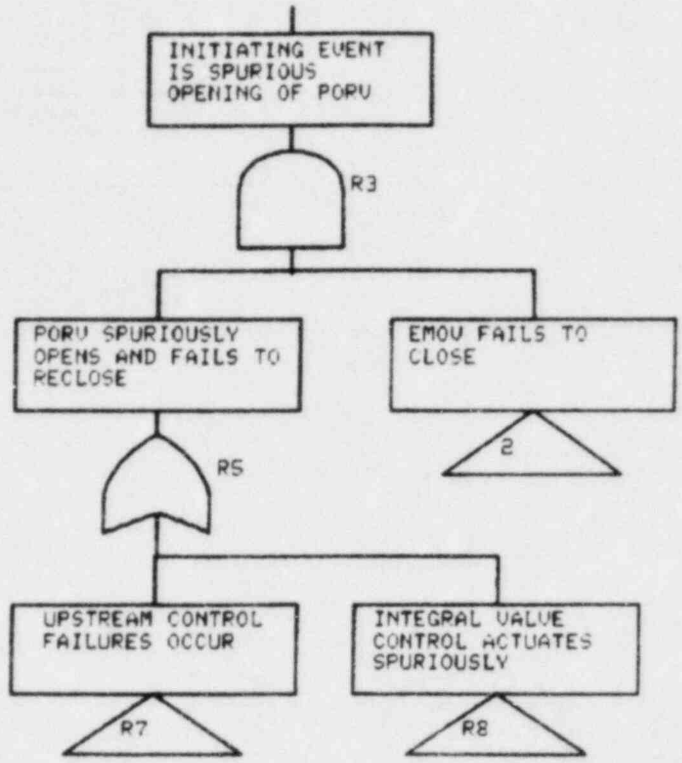

2
EMOV FAILS TO CLOSE
2
NO MOTIVE POWER FOR VALVE
5
NO SIGNAL TO CLOSE VALVE
T2
VALVE MECHANICAL FAILURE
GVEMOVOD
UNAVAIL-ABILITY OF 120VAC TO ESFAS SSA
TV120VAC
NO SIGNAL FROM ACOU-STICAL MONITOR AND OPERATOR FAILS TO SAVE MANUAL SIGNAL
T4
NO SIGNAL FROM RCS PRESSURE & OPERATOR FAILS TO SAVE MAN-UAL SIGNAL
T5
FAILURE OF ESFAS SSA
3
NO SIGNAL FROM ACOUSTICAL MONITOR
4
OPERATOR FAILS TO MANUALLY INITIATE EMOV
EMOVPROC
PRESSURE TRANSMIT-TER FAILS HIGH
PTEMOVFH
OPERATOR FAILS TO MANUALLY INITIATE EMOV
EMOVAMOC

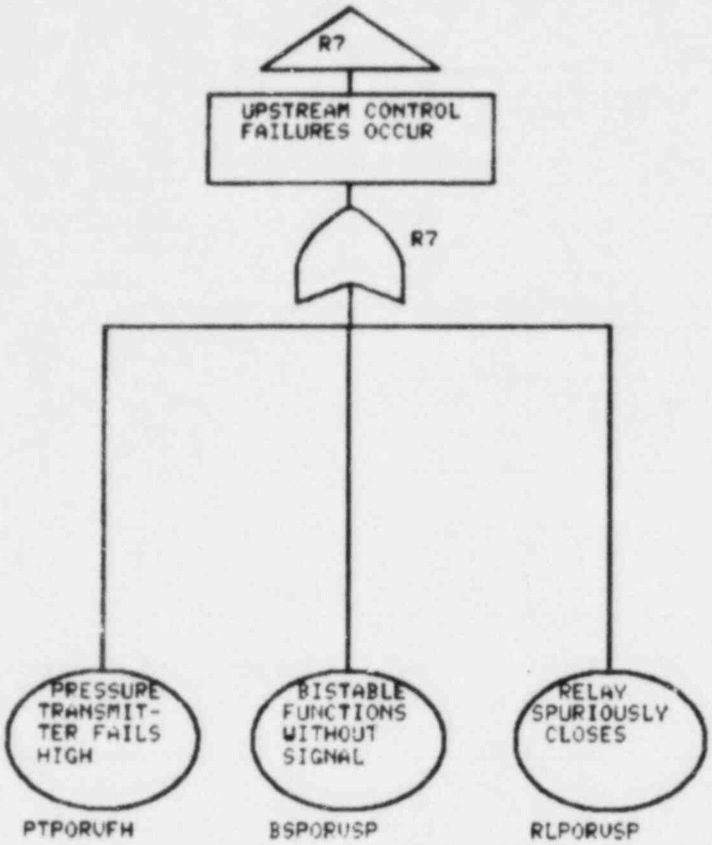

5
NO MOTIVE POWER FOR VALUE
5
M.C.C. FAILS TO FUNCTION ON DEMAND
UNAVAIL-ABILITY OF 480 VAC SUPPLY TO M.C.C.
TV480VAC
T6
FAILURE IN THE 480 VAC LINE
MOTOR STARTER FAILS
MCCMSFNS
FAILURE IN THE 120 VAC CONTROL LINE
T7
T8
FUSES FAIL
3FUSE480
CIRCUIT BREAKERS FAIL
3CBRK480
THERMAL OVERLOAD RELAYS
3THOR480
STEP DOWN TRANSFOR-MER FAILS
5TSTR120
FUSE FAILS
1FUSE120
STARTER CLOSE COIL FAILS
5TC1C120

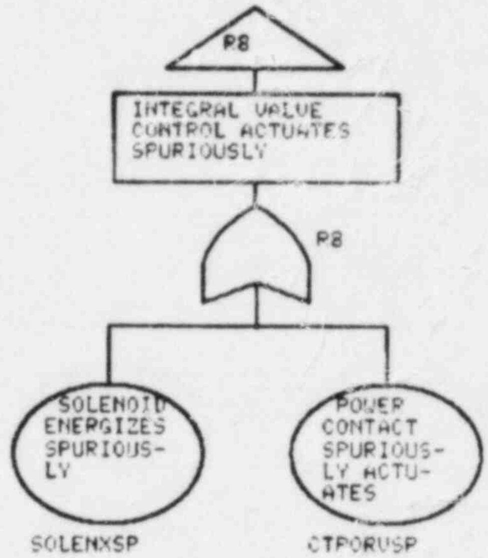

3
FAILURE OF ESFAS SSA
3
ACOUSTICAL MONITOR CURRENT BUFFER
AMSSACAT
RC PRESSURE CURRENT BUFFER FAILS
PTSSACAT
PRESSURE BISTABLE FAILS
BISSACAT
AND GATE FAILS
ANDSSAAM
LOGIC BUFFER FAILS
LBSSAXAM
CONTACT BUFFER FAILS
CBSSAXAM
UNIT CONTROL MODULE FAILS
ICMSSAAM

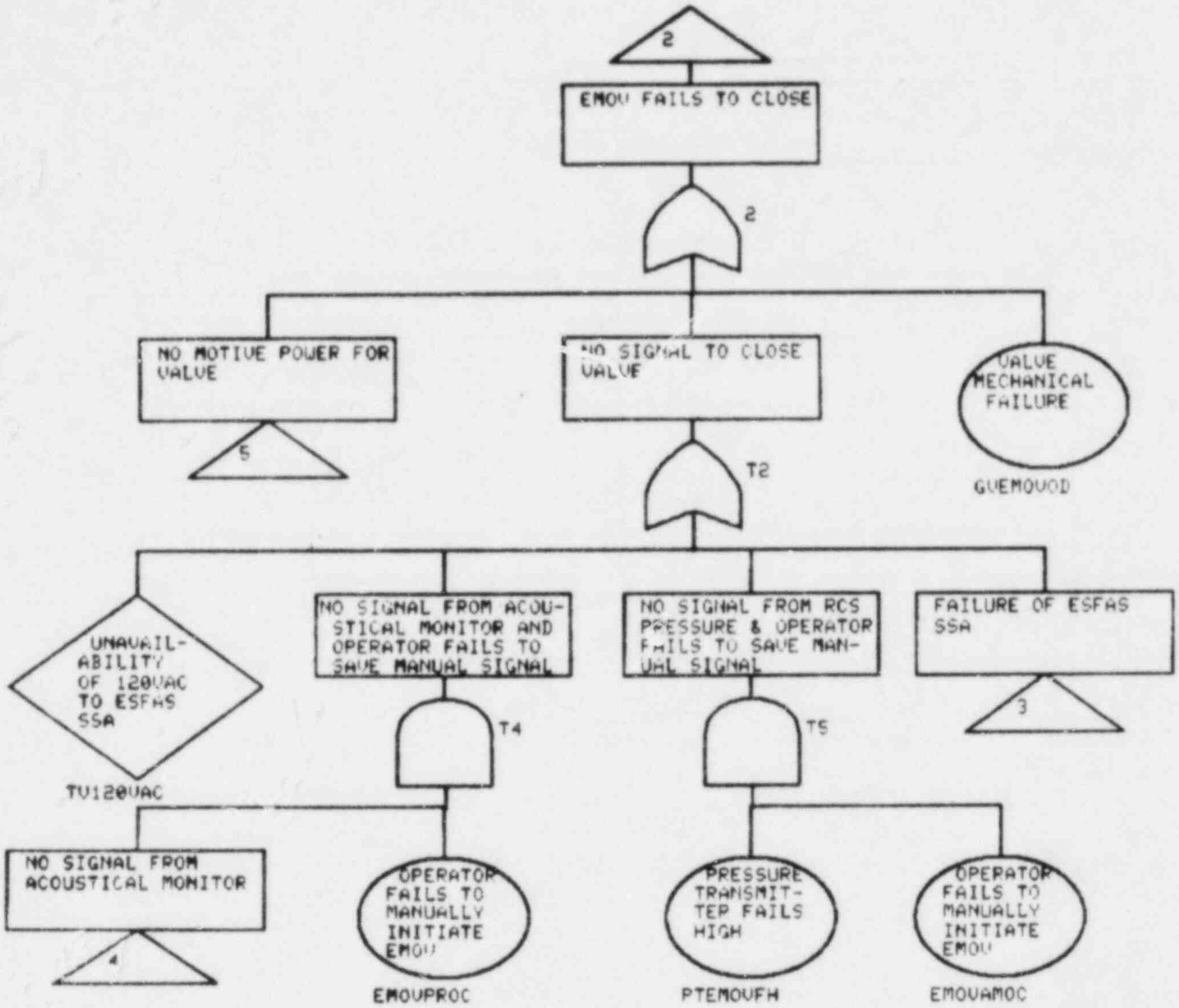

4
NO SIGNAL FROM ACOUSTICAL MONITOR
4
POWER UNAVAIL- ABLE TO ACOUSTICAL MONITOR
TV120VAM
ACOUSTI- CAL MONI- TOR FAILS
AMEMOVAM
SSCS IMPEDES ACOU- STICAL MONITOR SIGNAL
M2
RELAY OPENS SPURIOUS- LY
RLSSCSSP
POWER UNAVAIL- ABLE TO SSCS
TV120VSS
SOLID STATE ELEC- TRONICS FAIL
TUSSCSAM


1
NO SIGNAL GENERATED TO CLOSE PORV
1
PRESSURE TRANSMIT- TER DOES NOT CHANGE
BISTABLE FAILS TO FUNCTION WHEN SIGNALLED
RELAY FAILS TO OPEN
CONTACT FAILS TO OPEN
PTPORVCO
BSPORVNF
RLPORVFO
CTPORVFO

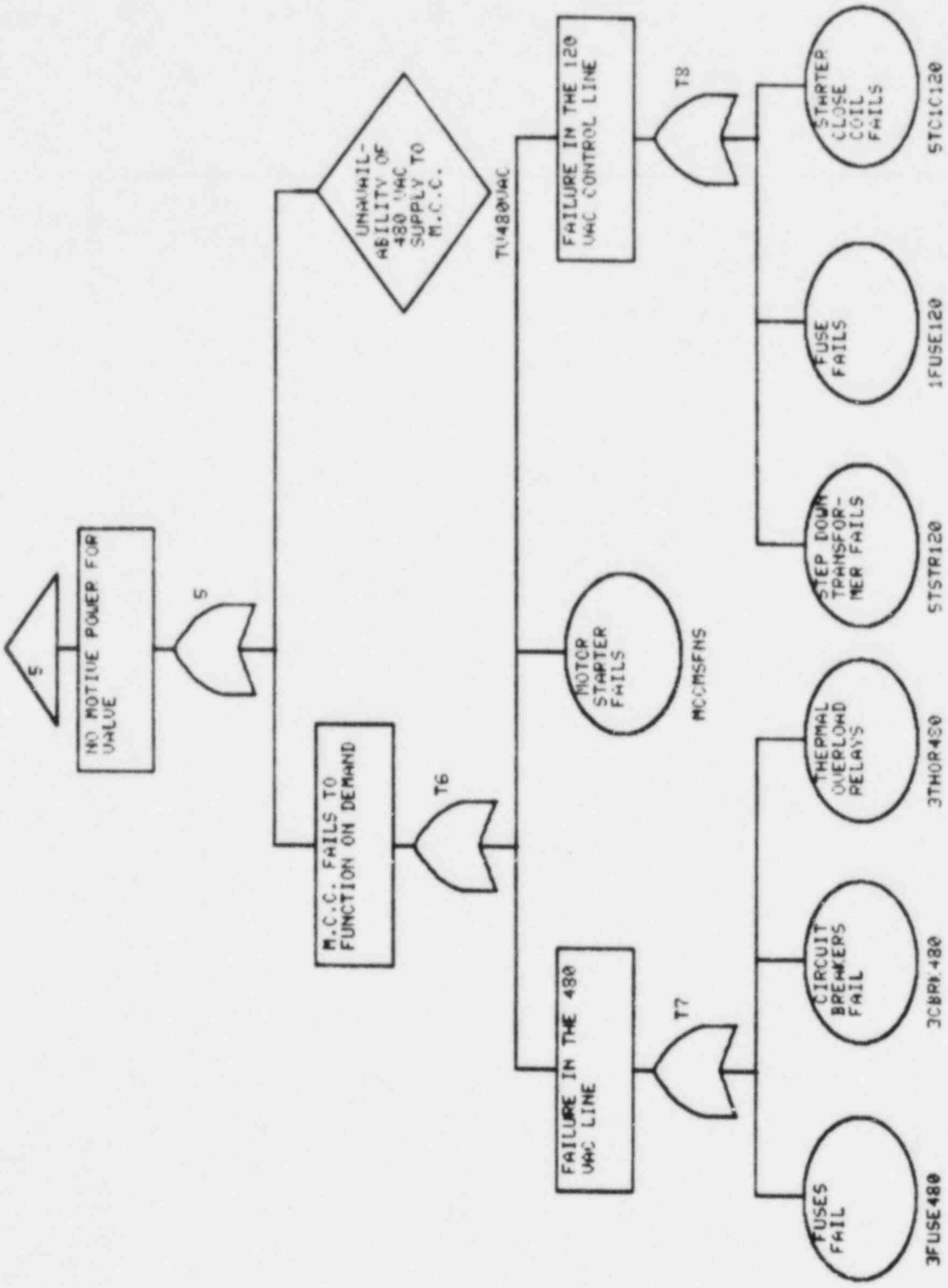

INITIATING EVENT IS SPURIOUS OPENING OF PORV
R3
PORV SPURIOUSLY OPENS AND FAILS TO RECLOSE
EMOV FAILS TO CLOSE
2
R5
UPSTREAM CONTROL FAILURES OCCUR
INTEGRAL VALVE CONTROL ACTUATES SPURIOUSLY
R7
R8


R7
UPSTREAM CONTROL FAILURES OCCUR
R7
PRESSURE TRANSMITTER FAILS HIGH
PTPORVFH
BISTABLE FUNCTIONS WITHOUT SIGNAL
BSPORVSP
RELAY SPURIOUSLY CLOSES
RLPORVSP


R8
INTEGRAL VALVE CONTROL ACTUATES SPURIOUSLY
R8
SOLENOID ENERGIZES SPURIOUSLY
SOLENXSP
POWER CONTACT SPURIOUSLY ACTUATES
CTPORVSP



2
EMOV FAILS TO CLOSE
2
NO MOTIVE POWER FOR VALVE
5
NO SIGNAL TO CLOSE VALVE
VALVE MECHANICAL FAILURE
GVEMOVOD
T2
UNAVAIL-ABILITY OF 120VAC TO ESFAS SSA
TV120VAC
NO SIGNAL FROM ACOU-STICAL MONITOR AND OPERATOR FAILS TO SAVE MANUAL SIGNAL
NO SIGNAL FROM RCS PRESSURE & OPERATOR FAILS TO SAVE MAN-UAL SIGNAL
FAILURE OF ESFAS SSA
3
T4
T5
NO SIGNAL FROM ACOUSTICAL MONITOR
4
OPERATOR FAILS TO MANUALLY INITIATE EMOV
EMOVPROC
PRESSURE TRANSMIT-TER FAILS HIGH
PTEMOVFH
OPERATOR FAILS TO MANUALLY INITIATE EMOV
EMOVAMOC


NO MOTIVE POWER FOR VALVE
5
5
M.C.C. FAILS TO FUNCTION ON DEMAND
T6
UNAVAILABILITY OF 480 VAC SUPPLY TO M.C.C.
TV480VAC
FAILURE IN THE 480 VAC LINE
T7
MOTOR STARTER FAILS
MCCMSFNS
FAILURE IN THE 120 VAC CONTROL LINE
T8
FUSES FAIL
3FUSE480
CIRCUIT BREAKERS FAIL
3CBRK480
THERMAL OVERLOAD RELAYS
3THOR480
STEP DOWN TRANSFORMER FAILS
STSTR120
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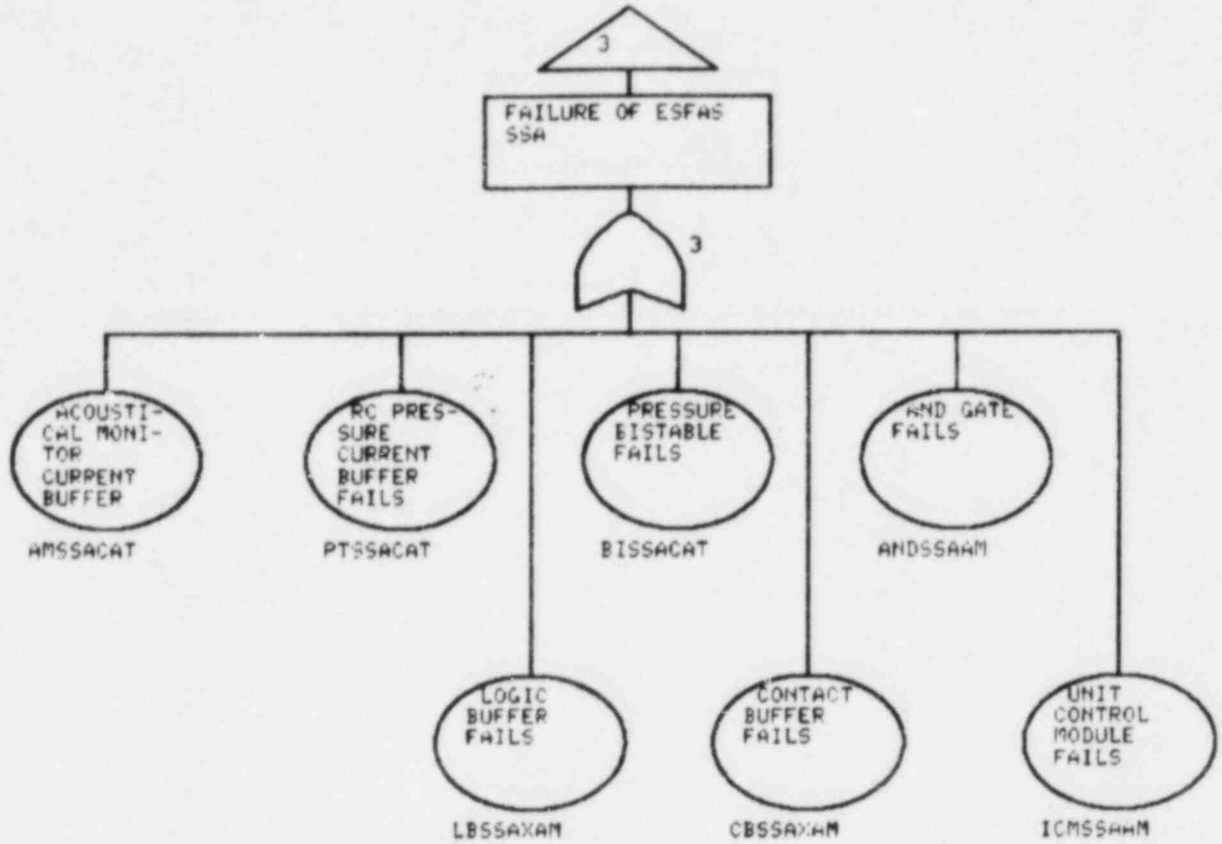
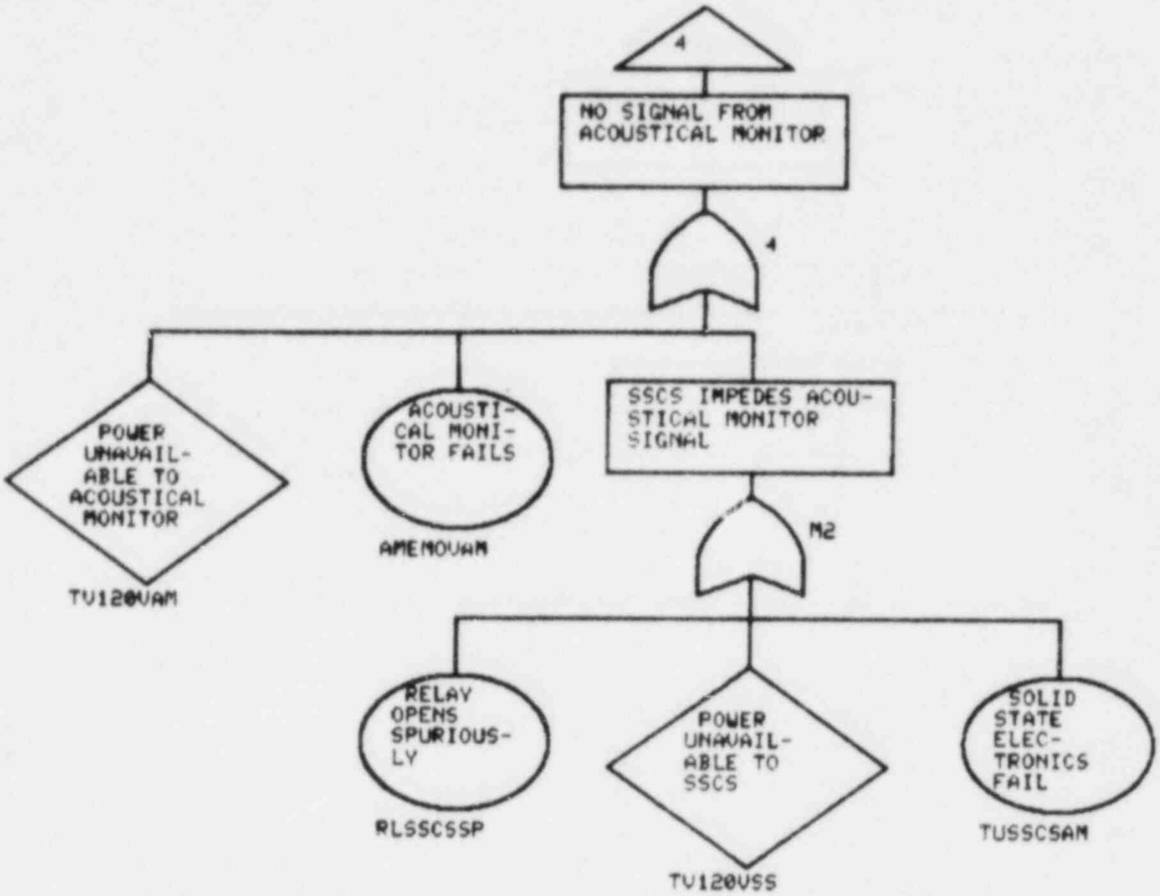
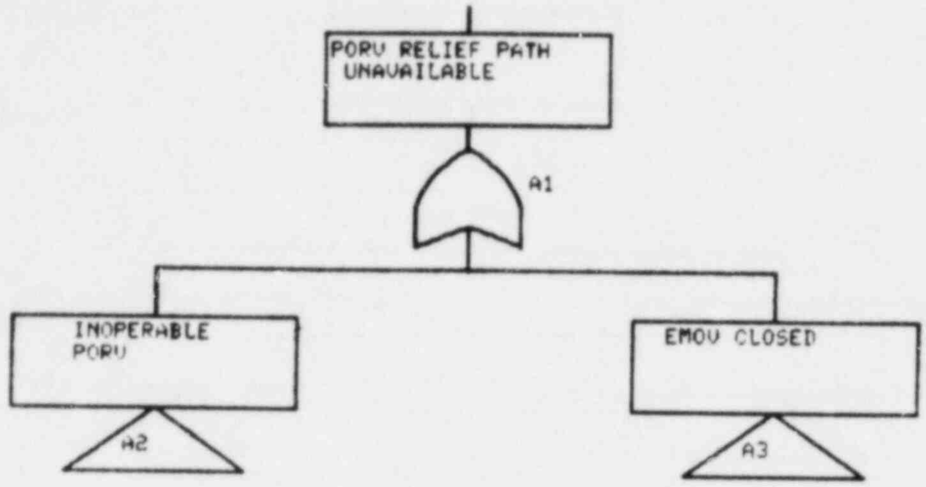
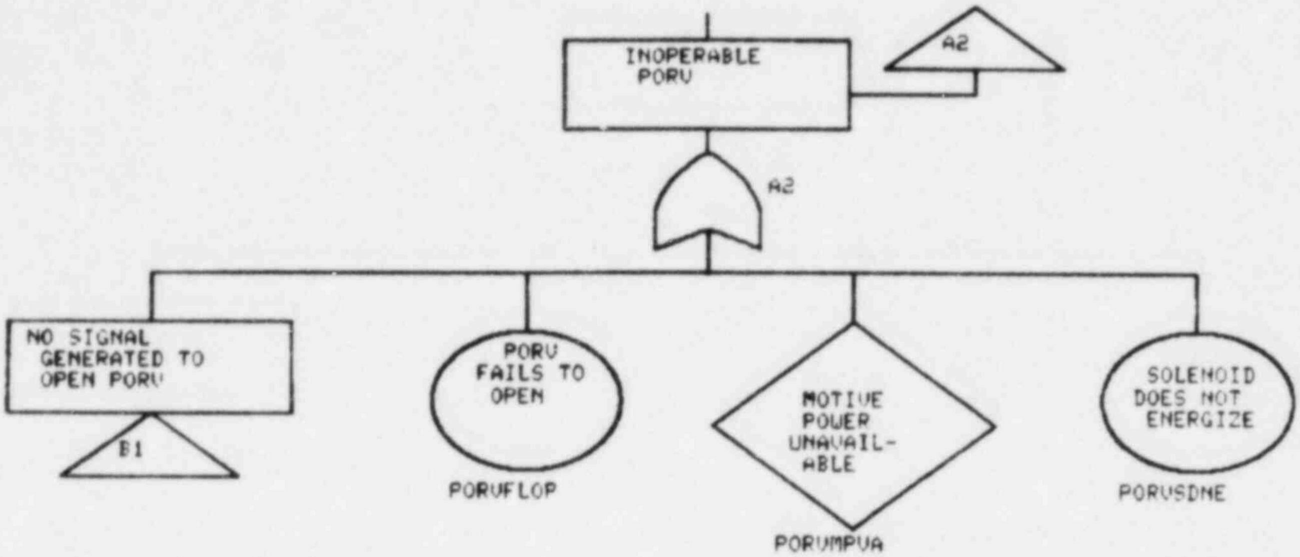
STARTER CLOSE COIL FAILS
STCIC120

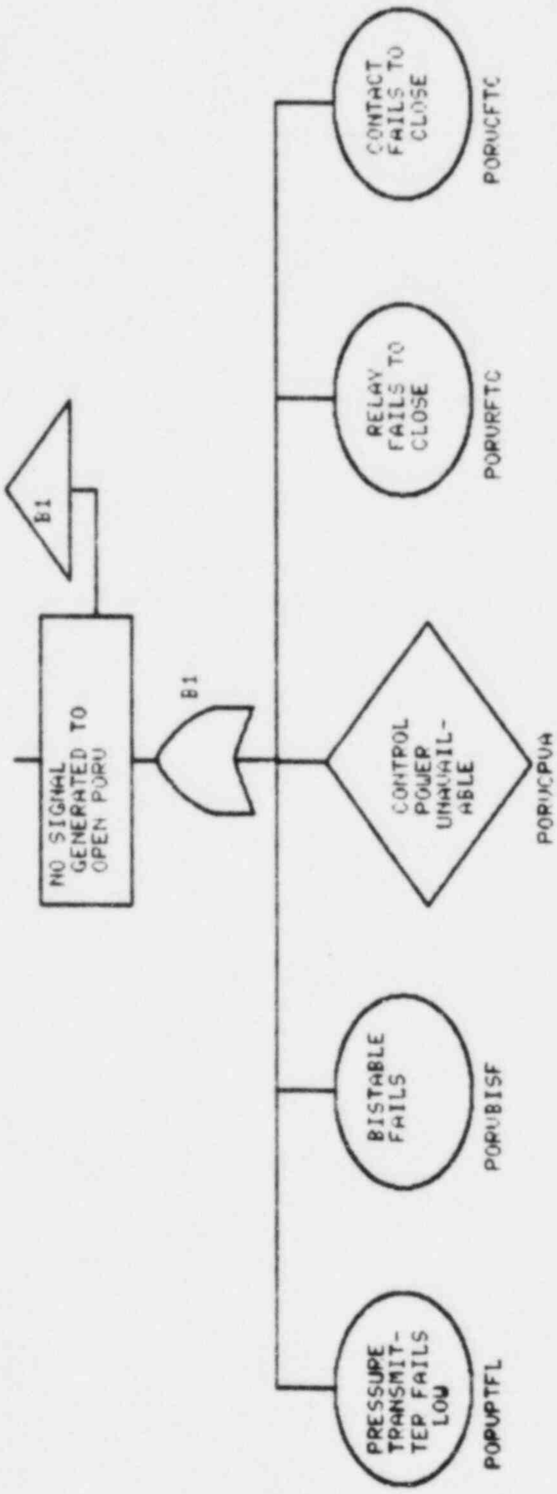

3
FAILURE OF ESFAS SSA
3
ACOUSTICAL MONITOR CURRENT BUFFER
AMSSACAT
RC PRESSURE CURRENT BUFFER FAILS
PTSSACAT
PRESSURE BISTABLE FAILS
BISSACAT
AND GATE FAILS
ANDSSAAM
LOGIC BUFFER FAILS
LBSSAXAM
CONTACT BUFFER FAILS
CBSSAXAM
UNIT CONTROL MODULE FAILS
ICMSSAAM

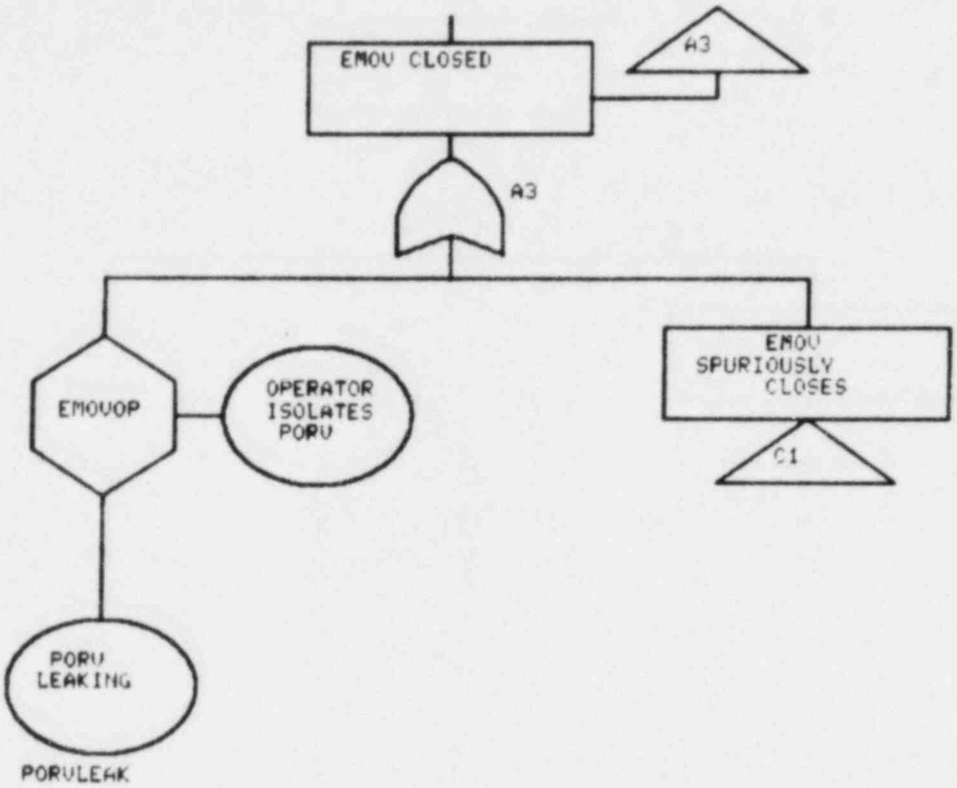

4
NO SIGNAL FROM ACOUSTICAL MONITOR
4
POWER UNAVAIL-ABLE TO ACOUSTICAL MONITOR
TV120VAM
ACOUSTI-CAL MONI-TOR FAILS
AMENOVAM
SSCS IMPEDES ACOU-STICAL MONITOR SIGNAL
M2
RELAY OPENS SPURIOUS-LY
RLSSCSSP
POWER UNAVAIL-ABLE TO SSCS
TV120VSS
SOLID STATE ELEC-TRONICS FAIL
TUSSCSAM


PORV RELIEF PATH
UNAVAILABLE
A1
INOPERABLE
PORV
A2
EMOV CLOSED
A3

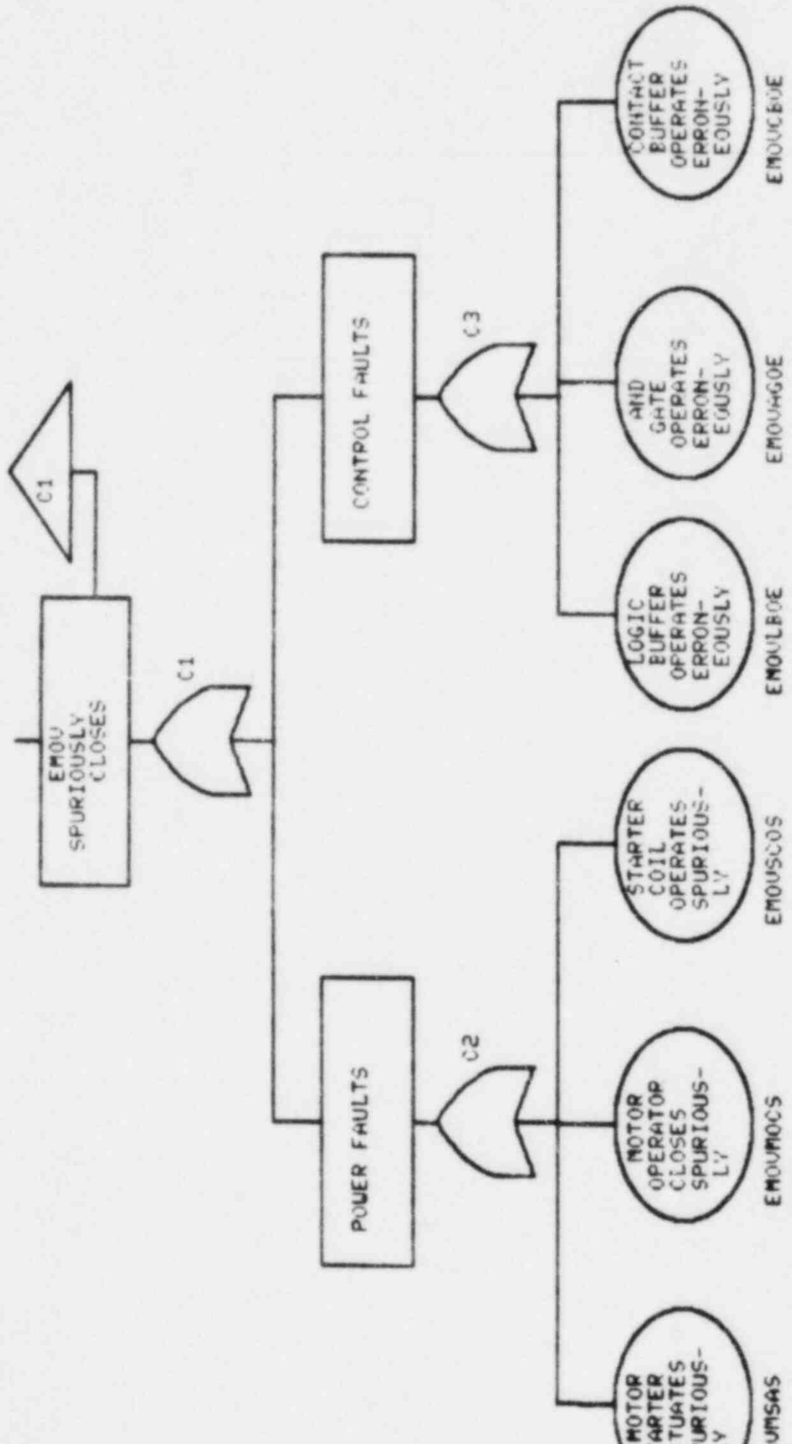


INOPERABLE PORV
A2
A2
NO SIGNAL GENERATED TO OPEN PORV
B1
PORV FAILS TO OPEN
PORVFLOP
MOTIVE POWER UNAVAIL-ABLE
PORVMPVA
SOLENOID DOES NOT ENERGIZE
PORVSDNE


NO SIGNAL GENERATED TO OPEN PORV
B1
B1
PRESSURE TRANSMIT-TER FAILS LOW
PORVPTFL
BISTABLE FAILS
PORVBISF
CONTROL POWER UNAVAIL-ABLE
PORVCPUA
RELAY FAILS TO CLOSE
PORVRFTC
CONTACT FAILS TO CLOSE
PORVCFTC



EMOV CLOSED
A3
A3
EMOVOP
OPERATOR ISOLATES PORV
EMOV SPURIOUSLY CLOSES
C1
PORV LEAKING
PORVLEAK

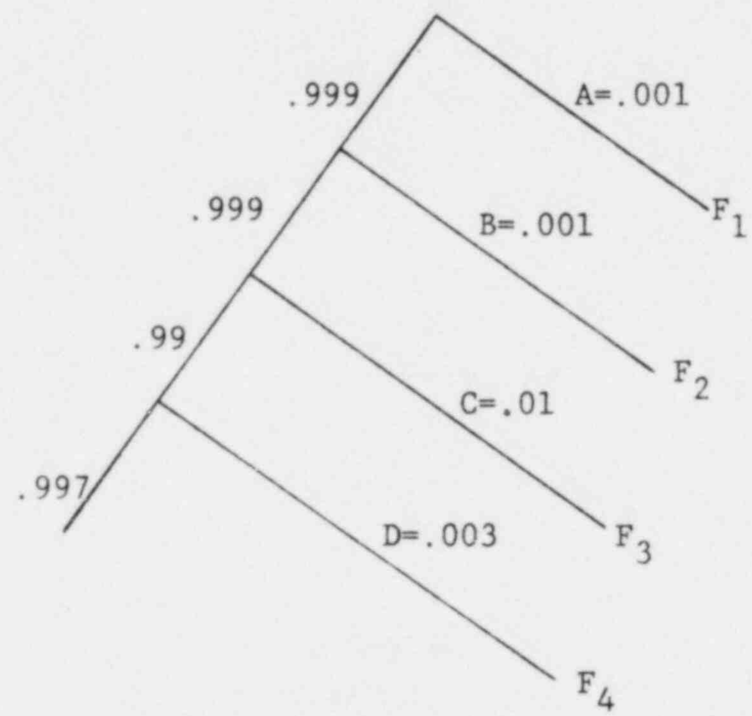


C1
EMOV SPURIOUSLY CLOSES
C1
POWER FAULTS
CONTROL FAULTS
C2
C3
MOTOR STARTER ACTUATES SPURIOUS-LY
EMOVMSAS
MOTOR OPERATOR CLOSES SPURIOUS-LY
EMOVMOCS
STARTER COIL OPERATES SPURIOUS-LY
EMOVSCOS
LOGIC BUFFER OPERATES ERRON-EOUSLY
EMOVLBOE
AND GATE OPERATES ERRON-EOUSLY
EMOVAGOE
CONTACT BUFFER OPERATES ERRON-EOUSLY
EMOVCBOE

APPENDIX B

Human Error Analysis

HPITHROC - Operator fails to throttle HPI



$$P(F) = F_1 + F_2 + F_3 + F_4$$

$$P(F) = 1.49 \times 10^{-2}$$

"A" = Operator fails to realize ESFAS initiates HPI pumps (Table 20-3).*

"B" = Fails to resume attention to legend light (Table 20-3).

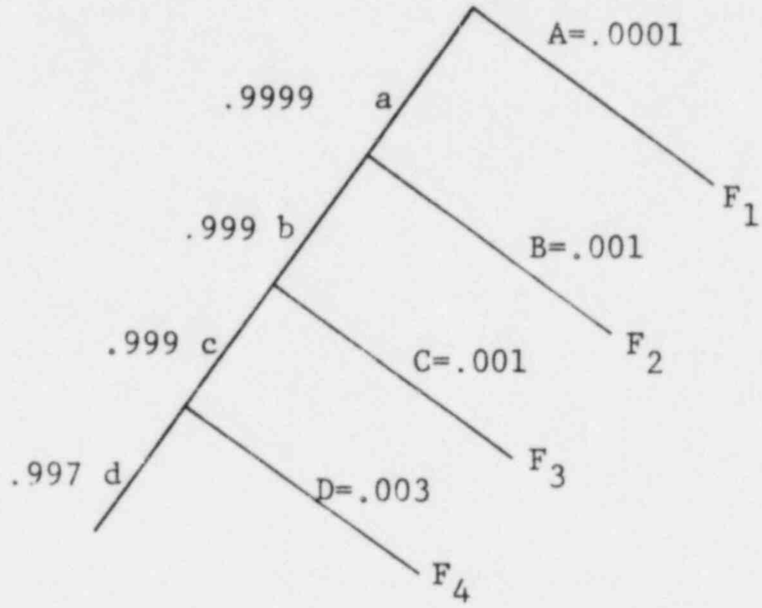
"C" = Fails to recognize the return of pressurizer level on ATOG scope (Table 20-5).

"D" = Fails to throttle HPI and realign normal make-up (Table 20-13).

*Note: Tables identified in this appendix are from NUREG/CR 1278.

EMOVAMOC - Operator fails to close block valve

Based on [Acoustical Monitor Signal (TVA) or Position Switch (WPPSS)]



$P(F) = F_1 + F_2 + F_3 + F_4$

$P(F) = 5.09 \times 10^{-3}$

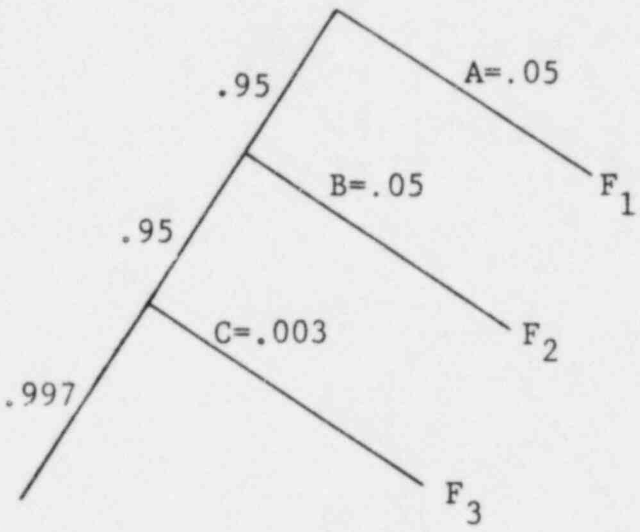
"A" = Fails to respond to alarm (Table 20-3) (.00005 to .001)

"B" = Incorrectly reads message (Table 20-3) (.0005 to .005)

"C" = Fails to resume attention (Table 20-3) (.0001 to .01)

"D" = Selects wrong MOV switch (Table 20-14) (.001 to .01)

EMOVPROC - Operator fails to close block valve based on RC pressure



$P(F) = F_1 + F_2 + F_3$

$P(F) = .1002 \approx .1$

"A" = Operator fails to detect low RC pressure display (Table 20-12).

"B" = Operator fails to properly diagnose that RC pressure drop is due to open PORV path (i.e.) fails to detect quench tank temperature/level rise. (Table 20-14)

"C" = Operator selects wrong MOV switch (Table 20-14).

APPENDIX C

Statistical Modeling of PORV Lifts

Assumptions

PORV lifts are initiated by seven transient sources with failure rates F_i, $i=1 \ldots 7$. Since the time to failure (initiation of transient) is assumed to be exponential, the number of times the PORV lifts (X_i) in time t for each transient is given by the Poisson distribution

$$\text{prob}(X_i) = \frac{(F_i t)^{X_i} \exp(-F_i t)}{X_i!} .$$

After a transient has been initiated, it may lead to a random number of PORV lifts. The probability distribution of a given number of PORV lists is different for each scurce, and it changes from the beginning to the end of each fuel cycle. For a given transient source, if the transient is initiated in the time interval $t+\Delta t$, the number of lifts (y_i) is given by the multinomial distribution

$$P(y_i/x_{i,t}) = \frac{X_i!}{y_{0i}!y_{1i}!y_{ki}!} P_{0i(t)}^{y_{0i}} P_{1i(t)}^{y_{1i}} P_{ki(t)}^{y_{ki}}$$

$$\sum y_{ji} = X_i$$

$$\sum P_{ji(t)} = 1$$

The marginal distribution of $P(y_{i,t})$ is obtained as

$$\sum_{x_i=0}^{\infty} P(y_i/x_i t)P(x_i)\Delta t = \sum_{x_i=0}^{\infty} \frac{(F\Delta t)^{X_i} \exp(-F\Delta t)}{X_i!} \times \frac{X_i!}{(x_i - y_{1i} - y_{2i} - y_{ki})!y_{1i}! + y_{ki}!}$$

$$\times (1 - P_{1it} - P_{2it} - \ldots - P_{kit}) \times (x_i - y_{1t} - y_{2t} - y_{kt}) \times P_{1t}^{y_{1i}} \ldots P_{kt}^{y_{kt}}$$

$$= \frac{(F\Delta tP_{it})^{y_{1t}} \exp(-F\Delta tP_{it})}{y_{1t}!} \times \frac{(F\Delta tP_{2t})^{y_{2t}} \exp(-F\Delta tP_{2t})}{y_{2t}!} \times \ldots$$

$$\times \frac{(F\Delta tP_{kt})^{y_{kt}} \exp(-F\Delta tP_{kt})}{y_{kt}!} .$$

Thus, each number (1, 2, etc.) of PORV lift cases for source i is distributed independently by Poisson distributions at any time interval $t+\Delta t$. The number of lifts over the entire time interval 0-t can be obtained by adding the Poisson distributions over the interval. If Δt is taken to be small, this amounts to integration. Thus, the number of single lifts is

$$\text{No. of lifts} = \frac{F\int_0^T (P_{1t}) \exp(-F \int_0^T P_{1t})}{y\,!},$$

$$k \text{ lifts} = \frac{F\int_0^T P_{kt} \exp(-F \int_0^T P_{kt})}{y_k!}, \text{ etc.}$$

Since the sum of independent Poisson distributions is again Poisson distributions, we can obtain the number of single lifts, double lifts, etc. for all transient sources. Thus, the number of single PORV lift cases for all transient sources will have a Poisson distribution with the following parameters:

$$G_1 = F_1 \int_0^T P_{11}(t)dt + F_2 \int_0^T P_{k2}(t)dt + \ldots + F_7 \int_0^T P_{17}(t)dt$$

and

$$G_k = F_1 \int_0^T P_{k1}(t)dt + F_2 \int_0^T P_{k2}(t)dt + \ldots + F_7 \int_0^T P_{k7}(t)dt.$$

If the Poisson distributions with parameters G_1, G_2, G_k are simulated in SAMPLE, yielding simulated variables z_1, z_2, z_k, then the total number of lifts for each simulation will be given as

$$\text{No. of lifts per simulation} = z_1 + 2z_2 + 3z_3 + \ldots + kz_k.$$

The probabilities $P_{0i}(t)$ ... $P_{ki}(t)$ were obtained from the histograms at the beginning and end of fuel life. Assuming that the change occurs linearly with time, the probabilities are given as

$$P_{0i}(t) = P_{0i}(0) + \frac{P_{0i}(t) - P_{01}(0)}{t} \times t$$

and

$$\int_0^T P_{0i}(t) = P_{0i}(0) + \frac{P_{0i}(T) - P_0(0)}{2} \times T$$

where $P_{0i}(0)$ and $P_{0i}(T)$ denote the probability of zero lifts at the beginning and end of the fuel cycle, respectively. The probabilities $P_{0i}(t)$ are seen to be appropriate multinomial probabilities since the sum over 0, 1, 2, etc. adds up to 1 for any value of t, given that this is true for the initial and final histograms. Similar modeling was used to derive the probabilities for the number of lifts equal to 1 ... k. This type of modeling was used for cases 1 and 2. In case 3, the number of transients in time t is assumed to be given by a Poisson distribution as before. However, in this case, the number of lifts for each transient will be defined by a normal distribution with specified mean and standard deviation (mean = nominal No. of lifts, std = $\Delta/2$, where $\pm\Delta$ denotes the maximum and minimum deviations from the mean).

The number of PORV lifts for case 3 is taken as normal with mean $x\mu$ and variance $x\sigma^2$, where x is the simulated Poisson value. Thus, a random value of x was obtained first, and then a random number of lifts could be determined:

$$\text{No. of lifts} = x\mu + z\sqrt{x\sigma^2}$$

where z is simulated normal with mean zero and a variance of 1.0.

Statistical Simulation Cases

Case 1	Case 2	Case 3
Turbine trip without reactor trip	Turbine trip with reactor trip	Overcooling: HPI repressurization
Trip one FW pump	Trip one FW pump	
Trip one RC pump	Trip one RC pump	
Load rejection	Load rejection	
Ramp one FW valve 50% closed	Ramp one FW valve 50% closed	
Rod drop	Rod drop	

Note: The expected contribution to total PORV demand from case 3 must be qualified by an operator error probability (operator fails to throttle HPI) before it can be added to cases 1 and 2.

Initiating Event Frequencies

Transient	Frequency, times/rx-yr*
Turbine trip	1.120
Trip one FW pump	0.229
Trip one RC pump	0.019
Load rejection	0.095
Ramp one FW valve 50% closed	0.457
Overcooling: HPI re-pressurization	0.263
Rod drop	0.372

*rx-yr: reactor year.

Notes

1. Rod drop frequency was determined over all power ranges. All other event frequencies were determined when the reactor was in operation above 70% power.
2. The fuel cycle was assumed to be 12 months.
3. Downtimes are inherent in the initating event frequency.

APPENDIX D

Failure Data

Code	Source	Unavailability
PORVXXCD	--	3.03×10^{-4}/d
SOLENXRE	IEEE, p. 387*	2.56×10^{-4}
PTPORVCO	IEEE, p. 428	1.10×10^{-3}
BSPORVNF	IEEE, p. 483	1.09×10^{-3}
RLPORVFO	IEEE, p. 155	3.54×10^{-5}
CTPORVFO	IEEE, p. 174	4.20×10^{-5}
PTPORVFH	IEEE, p. 428	2.19×10^{-3}/yr
BSPORVSP	IEEE, p. 483	1.80×10^{-3}/yr
RLPORVSP	IEEE, p. 155	3.6×10^{-4}/yr
GVEMOVOD	--	2.00×10^{-3}/d
CTPORVSH	IEEE, p. 174	6.02×10^{-6}
TUJXXXAM	---	5.48×10^{-5}
SOLENXSP	IEEE, p. 387	1.23×10^{-3}/yr
CTPORVSP	IEEE, p. 174	1.45×10^{-4}/yr
AMEMOVAM	--	1.52×10^{-2}
PSEMOVAM	IEEE, p. 452	4.89×10^{-4}
PTEMOVFH	IEEE, p. 428	9.13×10^{-5}
3 FUSE 480	IEEE, p. 193	2.30×10^{-5}
3 CBRK 480	IEEE, p. 148	4.71×10^{-5}
3 THOR 480	IEEE, p. 155	3.94×10^{-5}
MCCMSFNS	IEEE, p. 171	4.42×10^{-5}
1 FUSE 120	IEEE, p. 193	7.67×10^{-6}
STC1C120	IEEE, p. 162	2.45×10^{-5}
TUSSCS AM	--	5.48×10^{-5}
RLSSCS SP	--	1.69×10^{-5}
AMSSACAT	IEEE, p. 475	1.12×10^{-4}
PTSSACAT	IEEE, p. 475	1.12×10^{-4}
BISSACAT	IEEE, p. 483	3.75×10^{-4}
ANDSSAAM	--	5.48×10^{-5}
LBSSAXAM	MIL-HDBK 217-C	2.92×10^{-4}
CBSSAXAM	MIL-HDBK 217-C	2.92×10^{-4}
ICMSSAAM	IEEE, p. 177	3.10×10^{-5}

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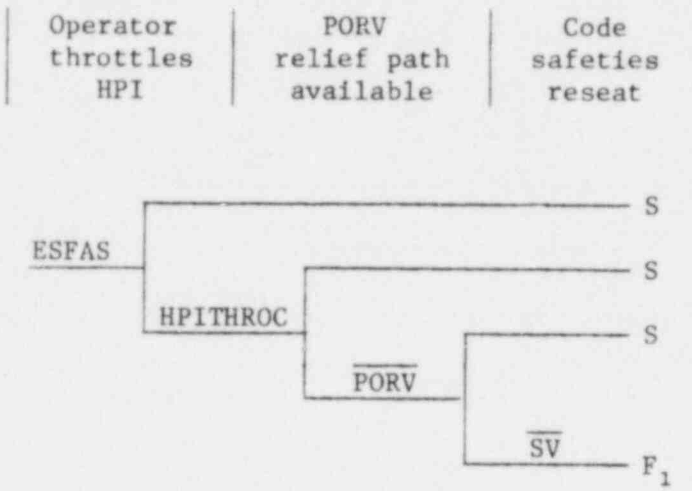
*IEEE: IEEE Std 500-1977.

Code	Source	Unavailability
TV120VAM	--	3.11×10^{-5}
TV120VAC	--	3.11×10^{-5}
TV120VSS	--	3.00×10^{-5}
TV480VAC	--	2.12×10^{-5}
5TSTR120	IEEE, p. 372	1.68×10^{-4}
PORVFLOP	--	1.00×10^{-3}/d
PORVLEAK	NPRDS, p. 573	1.70×10^{-3}
SVRESEAT	--	9.38×10^{-3}/d
PORVSDNE	IEEE, p. 387	2.56×10^{-4}
PORVMPVA	--	2.12×10^{-5}
PORVPTFL	IEEE, p. 428	1.10×10^{-3}
PORVBISF	IEEE, p. 483	1.09×10^{-3}
PORVRFTC	IEEE, p. 155	3.54×10^{-5}
PORVCFTC	IEEE, p. 174	4.20×10^{-5}
PORVCPVA	--	2.12×10^{-5}
EMOVMSAS	IEEE, p. 171	1.34×10^{-4}
EMOVMOCS	NPRDS, p. 617	1.75×10^{-3}
EMOVSCOS	IEEE, p. 162	2.02×10^{-6}
EMOVAGOE	--	7.20×10^{-6}
EMOVLBOE	MIL-HDBK 217C	3.84×10^{-5}
EMOVCBOE	MIL-HDBK 217C	3.84×10^{-5}

APPENDIX E

Event Sequences

1. Overcooling Scenario



$$F_1 = (0.263/\text{yr})(1.49 \times 10^{-2})(7.23 \times 10^{-3})(2.29 \times 10^{-2})(15)$$

$$= 9.73 \times 10^{-6}/\text{yr}^*$$

2. Overheating Events

F_2: loss of main feedwater and no auxiliary feedwater, given that normal electric power is available.

$$F_2 = (\text{LMFW})(\overline{\text{AFW}}/\text{AC})$$

$$= (1.78/\text{yr})(3 \times 10^{-5})$$

$$= 5.34 \times 10^{-5}/\text{yr}$$

F_3: loss of offsite power and no auxiliary feedwater, given that diesels are operative.

$$F_3 = (\text{LOOP})(\overline{\text{AFW}}/\text{diesels})$$

$$= (0.3/\text{yr})(3 \times 10^{-4})$$

$$= 9 \times 10^{-6}/\text{yr}$$

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*In this scenario the safety valves are challenged 15 times.

F_4: loss of offsite power and no auxiliary feedwater, given that diesels fail.

$$F_4 = (\text{LOOP})(\overline{\text{diesels}})(\overline{\text{AFW}}/\overline{\text{diesels}})$$

$$= (0.03/\text{yr})(3.2 \times 10^{-3})(3 \times 10^{-3})$$

$$= 2.88 \times 10^{-7}/\text{yr}$$

Event Sequence Failure Data

Event	Failure rate
LOOP	0.03/yr
$\overline{\text{diesels}}$	3.2×10^{-3}/day
$\overline{\text{AFW}}$/diesels	3×10^{-4}/day
$\overline{\text{AFW}}/\overline{\text{diesels}}$	3×10^{-3}/day
$\overline{\text{AFW}}$/AC	3×10^{-5}/day
LOFW	1.78/yr
ESFAS	0.263/yr
HPITHROC	1.49×10^{-2}/day
$\overline{\text{PORV}}$	7.23×10^{-3}/day
$\overline{\text{SV}}$	2.29×10^{-1}/day