LOCA Aspects of the Combustion Engineering Advanced Light Water Reactor — System 80+

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Abstract

The C-E ALWR is an upgrade of the proven System 80 NSSS standard design and so is referred to as "System 80+." Both plants are rated at 3817 Mwt, but System 80+ incorporates a number of modifications to the design.

A best-estimate small break study addressed the economic concern of how large a break size can be tolerated without the two-phase fluid level falling below the top of the core. With a best-estimate analytical procedure and no single failure the core was shown to remain covered for breaks up to, and including, that of a 10 in. dia. break.

A large break licensing analysis confirmed that adequate reflood exists following the end of SIT discharge without a LPSI system.

Introduction

The paper provides analytical results of the response of the Combustion Engineering System 80+ NSSS to the Loss-of-Coolant Accident (LOCA). The purpose of these analyses is to show that: 1. for realistic pipe breaks (≤10 in. dia.) using best estimate methods, the reactor core remains covered throughout the transient; and 2. for large pipe breaks using licensing methods, low pressure safety injection pumps are not required for the new ECCS design with direct vessel injection.

Plant Changes Which Affect LOCA Response

The C-E System 80+ NSSS is an enhanced version of the NRC approved System 80 NSSS standard design. Both plants are rated at 3817 Mwt. Some of the System 80+ design changes that influence the response to a postulated LOCA include a larger heat transfer area in the steam generators and a larger pressurizer volume. The emergency core cooling system (ECCS) employs four (4) trains of high pressure safety injection (HPSI) pumps and safety injection tanks (SITs) which inject directly into the vessel annulus instead of the cold leg injection for System 80. Low pressure safety injection pumps are no longer needed in the ECCS. A comparison of the ECCS for System 80 and for System 80+ is given in Table 1.

LOCA Analyses

The LOCA analyses performed to date includes investigations of both small break and large break behavior of the reactor coolant system. These studies are discussed below.

Small Break Study

The purpose of the small break LOCA study was to determine if the core could remain covered with two-phase fluid for the duration of the transient following a realistic size break. In order to evaluate the expected plant performance in the unlikely event that a LOCA should occur, best estimate analytical procedures were employed without a worst single failure assumption.

Realistic breaks would, more likely, occur in the long lengths of tributary piping than in relatively short main cooling piping. However, as a conservative simplification the breaks in this study, were located at the bottom of a cold leg adjacent to the reactor vessel inlet nozzle. Break sizes analyzed are given in Table 2.

Table 1
Emergency Core Cooling System Features

Feature	System 80+	System 80
High pressure safety injection pumps (same characteristics)	4	2
Type of injection	Direct to vessel annulus	Redundant headers to each cold leg
Low pressure safety injection pumps	none	2
Safety injection tanks	4	4

Table 2
Small Break Sizes Investigated

0.05 ft² (worst small break for System 80 licensing analysis)
0.20 ft² (6 in. dia.)
0.55 ft² (10 in. dia.)

The analyses were performed with Combustion Engineering's improved small break LOCA computer program, CEFLASH-4AS (FII). This program incorporates a number of analytical models which are more realistic than those used in the current, approved licensing methodology. The single most important model improvement in the small break analysis is the ANS 5.1 decay heat function with a two standard deviation uncertainty in place of the Appendix K requirement of ANS 5.0 with a 20% uncertainty factor.

Small Break Results

The most important result for the spectrum of small break LOCAs with respect to the potential for fuel damage is the transient two-phase level inside the core support barrel. These results are shown in Figure 1 a, b and c for cold leg breaks of 0.05 ft², 0.2 ft² and 0.55 ft², respectively. These figures show that the core remains fully covered for the duration of the transient for all break sizes up to and including the largest one analyzed (0.55 ft²).

The core should also remain covered for breaks smaller than those analyzed (less than 0.05 ft²). This conclusion is based on results for the C-E System 80 standard plant. System 80 has a somewhat smaller RCS volume that does the System 80+ design of the ALWR, and also a lesser number of HPSI pumps of similar size (see Table 1).

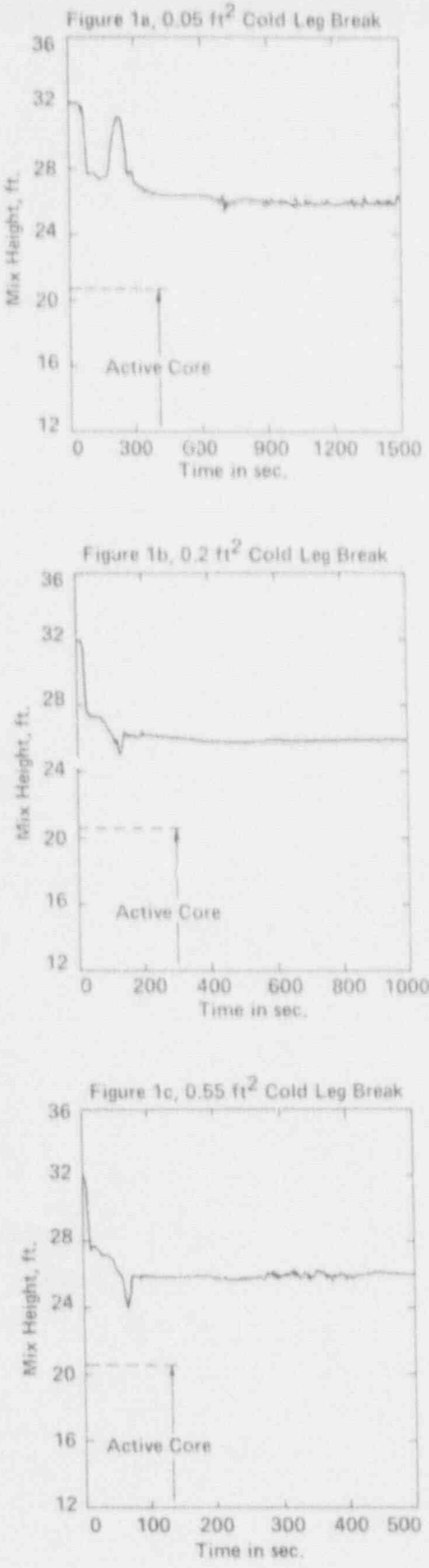


Fig. 1: Two-Phase Mixture Level Inside Core Support Barrel for Various Size Cold Leg Breaks

Large Break Study

The purpose of the large break licensing LOCA study was to determine whether modifications to the safety injection system could permit low pressure safety injection pumps to be eliminated from the ECCS (equivalent pumps would still be required for decay heat removal). These pumps were previously shown to be necessary for a brief period of time in the current generation NSSS LOCA licensing (Appendix K) analyses. That time period occurs immediately following the end of safety injection tank discharge during which the fluid loss rate from the core exceeds the net injection rate from a single operating HPSI pump. Such a situation is characterized by a decrease in the liquid level in the vessel annulus.

The improved ECCS design for the System 80+ plant provides a greater amount of HPSI flowrate than do the ECCS for the current generation plants. This is particularly important following the end of SIT discharge. Table 3 presents a comparison of the ECCS for System 80+ to that for a current generation C-E NSSS. The improved HPSI flowrate for System 80+, relative to current generation C-E plants, is a result of the direct injection into the vessel annulus and the use of four HPSI trains.

The large break LOCA analyses were run with C-E's currently approved licensing methodology (Ref. 1).

The conditions shown in Table 3 (loss of one diesel-generator) produce the lowest ECCS injection flowrates. However, these conditions do not necessarily result in the highest peak licensing break clad temperature. That results from full ECCS flow with maximum spillage which decreases containment pressure and worsens steam binding during core reflood.

Table 3 shows that the System 80+ plant with direct injection to the vessel has two (2) (net) HPSI pumps delivering water, whereas, System 80 has three-quarters (net) of one HPSI pump delivering water. This increment provides the necessary injection flowrate to match the fluid loss rate from the core.

Large Break Results

The large break study was focused on the double-ended break of a cold leg at the vessel inlet nozzle. This break size and location produces results (peak clad temperature) which are worst or very close to the worst. Peak clad temperature and local oxidation results are given in Table 4. The principal result from the large break study is that the LOCA licensing criteria can be met with the ECCS described in Table 3 (above). In particular, for the System 80+ ECCS, without LPSI pumps, the fluid level in the vessel annulus remains at its maximum value following the end of SIT discharge. This is shown in Figure 2. It is the liquid level in the vessel annulus which provides the static head to force reflood water up into the core. A maximum liquid level in the annulus assures maximum flow of cooling water to the core following the large break LOCA.

Table 3
ECCS Capability for Double-Ended Cold Leg Break (with loss of one diesel generator)

	System 80+	System 80
SITs delivering to vessel (net)	4	3*
HPSI pumps delivering to vessel (net) with loss of one diesel-generator	2	3/4*
LPSI pumps delivering to vessel (net) with loss of one diesel-generator	NA	1/2*
HPSI and LPSI delivery rate at SIT empty time with loss of one diesel-generator (approx.)	2260 GPM	3350 GPM
Flow required for boiloff (assuming core completely covered) (approx.)	1250 GPM	1250 GPM

*A portion of injection flow spills to containment through the cold leg break.

Table 4
ECCS Licensing Results for Double-Ended (Gulliotine) Break

	Peak Clad Temperature	Max Local Clad Oxidation (%)
System 80 (Palo Verde)	2169	13.1
System 80+	2168	13.3

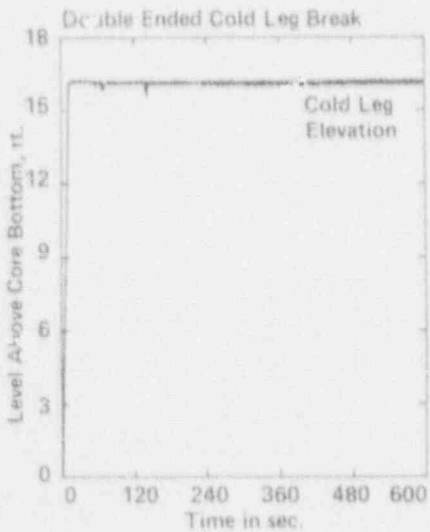


Fig. 2: Level in Downcomer

Conclusions

These LOCA analyses for the C-E System 80+ plant show that:

1. No core uncovery occurs for realistic pipe breaks and best estimate methods and
2. A LPSI pump is not required in the design to meet licensing criteria.

Reference

1. CENPD-132P, "Calculation Methods for the C-E Large Break LOCA Evaluation Model," August, 1974.

Question 720.6

In the logical expression of SGTR accident sequences provided by CE (floppy disk), five accident sequences (i.e., sequences 3, 4, 7, 9 and 10) were found to contain the complement of an event-tree top event, PMHG04BX. However, no such top event can be found in the SGTR event tree. Please explain this inconsistency.

Response 720.6

The fault tree, PMHG04BX, is defined as "Failure to Deliver Sufficient SI Flow to 4 of 4 Loops" (See page 6-277 in the System 80+ PRA Report). It is part of the "Failure of SIS for Medium LOCA" fault tree on page 6-274 of the System 80+ PRA Report. The complement of PMHG04BX was inadvertently used instead of the complement of the fault tree, PHOG01BX, which can be found on page 6-279 of the System 80+ PRA Report. PHOG01BX is defined as "Failure to Deliver SI Flow to 4 of 4 Loops - No Break". The primary difference between PMHG04BX and PHOG01BX is that PHMG04BX does not include the common cause failures of the mechanical components in the Injection System (They are included at a higher level in tree PHMG02BX on page 6-275 of the System 80+ PRA Report). This difference doe not impact the analysis because the downstream faulted events in SGTR sequences 3, 4, 7,9 and 10 do not include any of the mechanical equipment in the Injection System.

Question 720.9

In the event of a SGTR, many operator actions are needed to prevent core damage. They include manual control of EFW to maintain proper level in the intact SG; manual tripping of two out of 4 RCPs; and after identifying the ruptured SG, isolation of the ruptured SG by closing MSIVs, ADVs, and main feedwater isolation valves and by isolating SG blowdown, vents, drains, exhausts and bleedoffs. Are the required operator actions modeled in the fault trees? If yes, in which fault trees?

Response 720.9

See the response to Question 721.1.

Question 720.20

Please indicate where the accident sequence involving a SGTR with coincident loss of off-site power is modeled in the PRA? If it is not modeled, please model it in the PRA or provide an explanation for why it need not be modeled. Note that, for this sequence, turbine bypass system, MFW system and SG Blowdown system will not be available. Moreover, all RCPs will trip and pressurizer spray will not be available for lowering RCS pressure. Heat removal by SG will have to rely heavily on primary loop natural recirculation and it may take a longer time to bring the reactor to stable cold shutdown.

Response 720.20

All of the frontline system fault tree models in the SGTR event tree have all of the appropriate support system models linked into them. The electrical distribution system fault tree models include coincident loss of off-site power. Thus, all of the SGTR sequences include consideration of a coincident loss of off-site power. With a loss of offsite power, the RCPs trip and the main pressurizer spray is not available for RCS pressure control. Heat removal via the SGs relies on natural circulation in the primary loop. The reactor head vent subsystem portion of the safety depressurization system is available for RCS pressure control. Extra time is required for the cooldown on natural circulation. This extra time was assumed in the timing analysis. The models for the Startup Feedwater System and the Steam Generator Blowdown System address the availability of offsite power. The Turbine Bypass System was not credited in the analysis, with or without offsite power. In this analysis, a SGTR with a coincident full station blackout (loss of offsite power with failure of both diesel generators and the Alternate AC System) was assumed to lead to core melt because of the loss of RCS makeup capability.

Question 720.30

For an ATWS with a stuck open PSV, but with successful safety injection (see ATWS sequence #24) or for an ATWS with failure of boron delivery by the charging pumps, but with successful depressurization and safety injection (see ATWS sequence #7), the heat load of the IRWST can be expected to be significantly higher than for ordinary transients due to high reactor power. What is your basis for assuming that one containment spray pump is adequate to successfully cool the IRWST?

Response 720.30

The primary purpose for cooling the IRWST is to remove the residual heat from containment to prevent a containment overpressure failure. The containment spray system is designed such that one containment spray pump and its associated containment spray heat exchanger can remove sufficient energy from containment following a design basis large LOCA or Steam Line Break to prevent exceeding the containment design pressure. For an ATWS with an MTC that is not adverse, the reactor power initially increases. However, as the RCS temperature and pressure, the reactor power rapidly begins to decrease due to the moderator temperature reactivity feedback. The ATWS pressure peak is past with approximately 1 or 2 minutes with reactor power dropping to about 5% shortly thereafter. During this period, energy is being discharged from the RCS to the IRWST via the Primary Safety Valves (PSVs). Subsequent energy transfer would be via either the stuck open safety valve or the depressurization valve(s). Thus, the initial rate at which energy is transferred into containment from the RCS is less than for a design basis LOCA or Steam Line Break. With the boron addition associated with the successful safety injection, reactor power will continue to decrease to decay heat levels. Thus, while the total amount of energy added to containment (IRWST) following an ATWS with a stuck open PSV or successful depressurization may be slightly greater than for a design basis LOCA due to the higher initial power, this energy while be transferred to containment at a lower rate and over a greater period of time. Based on this, it was concluded that one containment spray pump and its associated heat exchanger could provide sufficient heat removal from the IRWST to prevent containment overpressurization.

Question 720.33

Describe in detail how reactor coolant pump (RCP) seal LOCAs are treated in the CESSAR 80+ PRA?

Response 720.33

RCP seal LOCAs are not treated explicitly in the System 80+ PRA. As an initiating event, RCP seal LOCAs are considered to be covered by the small LOCA event tree. The generic initiating event frequency presented in the EPRI ALWR PRA Key Assumptions and groundrules was based on all operating events which had a leak rate or leak size consistent with small LOCAs. Thus, RCP seal LOCAs can be considered to be covered by this initiating event frequency. The system responses needed to mitigate an RCP seal LOCA are the same as for any other small LOCA.

Based on operating experience and test data, as presented in CE NPSD-340, "A Combustion Engineering Review of NUREG-1032,: Evaluation of Station Blackout Events at Nuclear Power Plants", March, 1986, CE believes that the RCP seals used in CE plants will not develop excessive seal leakage under total Station Blackout conditions (SBO). Total SBO is defined here as a loss of offsite power combined with failure of both diesel generator and the alternate AC source. Therefore, consequential RCP seal LOCAs following a station blackout were not modeled in the System 80+ PRA.

Question 720.35

List the top 100 accident sequences where recovery was credited. Was recovery credited before or after sequence truncation? For each of the accident sequences that has been corrected for recovery actions, please indicate which safety functions were corrected and what recovery factors were used. Was recovery credited at the cutset level? If so, how many cutsets were involved?

Response 720.35

Table 720.35-1 (starting on next page) lists the accident sequences where the recovery was credited. The list comprises of 38 internal accident sequences and 11 seismically-induced accident sequences. Recovery was not credited in any tornado strike accident sequences because it was assumed that the offsite power could not be restored within the 24 hour mission time and that the alternate AC power source, namely combustion turbine, would not be available as a backup for the diesel generators. Table 720.35-1 indicates the safety functions that were corrected and/or recovery action(s) taken for a given accident sequence. The recovery factors that were used are to be found in terms of the non-recovery probabilities in Table 5-7 on page 5-50 of the System 80+ PRA Report (DCTR-RS-02, Rev. 0, Volume 1, January 1991). The recovery was credited at the cutset level. Recovery was applied to the point where the total contribution of the cutsets was at least 95% of the core damage frequency attributed to that particular sequence and the contribution of the individual cutset was less than 0.1%. Accordingly, the number of cutsets involved in the recovery process varied from one accident sequence to another sequence.

Question 720.36

Please provide dependency matrices showing major frontline system and their dependence on all the relevant support systems at the train level. Provide a similar matrix for support system dependency at the train level.

Response 720.36

C-E is currently updating the System 80+ PRA. The dependency matrices requested above will be included as a part of this update.

Table 720.35-1 : Accident Sequences were Recovery was Credited

Accident Sequence	Table no. in DCTR-RS-02 Vol.3 where Description is given	Safety Function(s) Corrected and/or Recovery Action(s) taken
2-Large LOCA	8.2.1-1	Containment Cooling; Open manual discharge valve(s)
2-Medium LOCA	8.2.2-1	Containment Cooling; Open manual discharge valve(s)
3-Medium LOCA	8.2.2-1	Start and Load Standby AC power or Restore Offsite power within 12 hours to provide motive power to Engineered Safety Feature (ESF) pumps
7-Small LOCA	8.2.3-1	Manually rackin equipment breakers to provide 125 VDC control power for the Engineered Safety Feature (ESF) pumps
10-Small LOCA	8.2.3-1	Start and Load Standby AC power or Restore Offsite power within 12 hours to provide motive power to Engineered Safety Feature (ESF) pumps
11-Small LOCA	8.2.3-1	Start and Load Standby AC power or Restore Offsite power within 12 hours to provide motive power to Engineered Safety Feature (ESF) pumps
10-SGTR	8.2.4-1	Manually rackin equipment breakers to provide 125 VDC control power for the Engineered Safety Feature (ESF) pumps
12-SGTR	8.2.4-1	Start and Load Standby AC power or Restore Offsite power within 12 hours to provide motive power to Engineered Safety Feature (ESF) pumps
13-SGTR	8.2.4-1	Manually rackin equipment breakers to provide 125 VDC control power for the Engineered Safety Feature (ESF) pumps; Start and Load Standby AC power to provide motive power to Engineered Safety Feature (ESF) pumps
7-LSSB (Large Secondary Side Break)	8.2.5-1	Start and Load Standby AC power or Restore Offsite power within 1 hour to provide motive power to Engineered Safety Feature (ESF) pumps

Table 720.35-1 : Accident Sequences were Recovery was Credited (cont'd)

Accident Sequence	Table no. in DCTR-RS-02 Vol.3 where Description is given	Safety Function(s) Corrected and/or Recovery Action(s) credited
17-LSSB	8.2.5-1	Start and Load Standby AC power to provide motive power to Engineered Safety Feature (ESF) pumps
4-LOFW (Loss of [main] Feedwater Flow)	8.2.6-1	Start and Load Standby AC power or Restore Offsite power within 9 hours or Restore Offsite power within 16 hours to provide motive power to Engineered Safety Feature (ESF) pumps
8-LOFW	8.2.6-1	Start and Load Standby AC power or Restore Offsite power within 1 hour or Restore Offsite power within 4 hours to provide motive power to Engineered Safety Feature (ESF) pumps
9-LOFW	8.2.6-1	Manually rackin equipment breakers to provide 125 VDC control power for the Engineered Safety Feature (ESF) pumps; Start and Load Standby AC power or Restore Offsite power within 4 hours to provide motive power to Engineered Safety Feature (ESF) pumps
8-TOTH (Other Transients)	8.2.7-1	Start and Load Standby AC power or Restore Offsite power within 4 hours to provide motive power to Engineered Safety Feature (ESF) pumps
9-TOTH	8.2.7-1	Manually rackin equipment breakers to provide 125 VDC control power for the Engineered Safety Feature (ESF) pumps; Start and Load Standby AC power or Restore Offsite power within 4 hours to provide motive power to Engineered Safety Feature (ESF) pumps

Table 720.35-1 : Accident Sequences were Recovery was Credited (cont.d)

Accident Sequence	Table no. in DCTR-RS-02 Vol.3 where Description is given	Safety Function(s) Corrected and/or Recovery Action(s) credited
3-LOOP (Loss Of Offsite Power)	8.2.8-1	Start and Load Standby AC power or Restore Offsite power within 1 hour or Restore Offsite power within 4 hours or Restore Offsite power within 16 hours to provide motive power to Engineered Safety Feature (ESF) pumps
7-LOOP	8.2.8-1	Start and Load Standby AC power or Restore Offsite power within 1 hour or Restore Offsite power within 4 hours to provide motive power to Engineered Safety Feature (ESF) pumps
9-LOOP	8.2.8-1	Start and Load Standby AC power or Restore Offsite power within 1 hour or Restore Offsite power within 4 hours to provide motive power to Engineered Safety Feature (ESF) pumps
11-LOOP	8.2.8-1	Start and Load Standby AC power or Restore Offsite power within 1 hour to provide motive power to Engineered Safety Feature (ESF) pumps
12-LOOP	8.2.8-1	Start and Load Standby AC power or Restore Offsite power within 12 hours to provide motive power to Engineered Safety Feature (ESF) pumps
3-CCWB (Loss of Component Cooling Water Div.)	8.2.9-1	Open manual discharge valve(s)
4-CCWB	8.2.9-1	Start and Load Standby AC power or Restore Offsite power within 16 hours to provide motive power to Engineered Safety Feature (ESF) pumps

Table 720.35-1 : Accident Sequences were Recovery was Credited (cont.d)

Accident Sequence	Table no. in DCTR-RS-02 Vol.3 where Description is given	Safety Function(s) Corrected and/or Recovery Action(s) credited
5-CCWB	8.2.9-1	Open manual discharge valve(s); Start and Load Standby AC power or Restore Offsite power within 16 hours to provide motive power to Engineered Safety Feature (ESF) pumps
8-CCWB	8.2.9-1	Start and Load Standby AC power or Restore Offsite power within 1 hour or Restore Offsite power within 4 hours to provide motive power to Engineered Safety Feature (ESF) pumps
8-125VB (Loss of 125VDC Bus B)	8.2.10-1	Start and Load Standby AC power or Restore Offsite power within 4 hours to provide motive power to Engineered Safety Feature (ESF) pumps
9-125VB	8.2.10-1	Manually rackin equipment breakers to provide 125 VDC control power for the Engineered Safety Feature (ESF) pumps
8-416KB (Loss of 4.16KV Bus B)	8.2.11-1	Start and Load Standby AC power or Restore Offsite power within 4 hours to provide motive power to Engineered Safety Feature (ESF) pumps
9-416KB	8.2.11-1	Manually rackin equipment breakers to provide 125 VDC control power for the Engineered Safety Feature (ESF) pumps; Start and Load Standby AC power or Restore Offsite power within 4 hours to provide motive power to Engineered Safety Feature (ESF) pumps
7-ATWS	8.2.12-1	Start and Load Standby AC power to provide motive power to Engineered Safety Feature (ESF) pumps
8-ATWS	8.2.12-1	Start and Load Standby AC power; Start and Load Standby AC power or Restore Offsite power within 9 hours to provide motive power to Engineered Safety Feature (ESF) pumps

Table 720.35-1 : Accident Sequences were Recovery was Credited (cont.d)

Accident Sequence	Table no. in DCTR-RS-02 Vol.3 where Description is given	Safety Function(s) Corrected and/or Recovery Action(s) credited
9-ATWS	8.2.12-1	Start and Load Standby AC power to provide motive power to Engineered Safety Feature (ESF) pumps
18-ATWS	8.2.12-1	Start and Load Standby AC power; Start and Load Standby AC power or Restore Offsite power within 12 hours to provide motive power to Engineered Safety Feature (ESF) pumps
25-ATWS	8.2.12-1	Start and Load Standby AC power or Restore Offsite power within 12 hours to provide motive power to Engineered Safety Feature (ESF) pumps
3-LHVAC (Loss of 1 HVAC division)	8.2.14-1	Open manual discharge valve(s)
4-LHVAC	8.2.14-1	Start and Load Standby AC power or Restore Offsite power within 12 hours to provide motive power to Engineered Safety Feature (ESF) pumps
5-LHVAC	8.2.14-1	Open manual discharge valve(s); Start and Load Standby AC power or Restore Offsite power within 16 hours to provide motive power to Engineered Safety Feature (ESF) pumps
8-LHVAC	8.2.14-1	Start and Load Standby AC power or Restore Offsite power within 1 hour or Restore Offsite power within 4 hours to provide motive power to Engineered Safety Feature (ESF) pumps
4-SEIS (Seismically-induce event)	8.3.2-1	Isolate failed motor jacket cooling heat exchanger for motor-operated pump

Table 720.35-1 : Accident Sequences were Recovery was Credited (cont.d)

Accident Sequence	Table no. in DCTR-RS-02 Vol.3 where Description is given	Safety Function(s) Corrected and/or Recovery Action(s) credited
5-SEIS	8.3.2-1	Isolate failed motor jacket cooling heat exchanger for motor-operated pump
8-SEIS	8.3.2-1	Isolate failed motor jacket cooling heat exchanger for motor-operated pump
9-SEIS	8.3.2-1	Isolate failed motor jacket cooling heat exchanger for motor-operated pump; Reclose breakers to provide power to the equipment
12-SEIS	8.3.2-1	Isolate failed motor jacket cooling heat exchanger for motor-operated pump; Reclose breakers to provide power to the equipment
15-SEIS	8.3.2-1	Isolate failed motor jacket cooling heat exchanger for motor-operated pump; Reclose breakers to provide power to the equipment
24-SEIS	8.3.2-1	Isolate failed motor jacket cooling heat exchanger for motor-operated pump
28-SEIS	8.3.2-1	Isolate failed motor jacket cooling heat exchanger for motor-operated pump
29-SEIS	8.3.2-1	Isolate failed motor jacket cooling heat exchanger for motor-operated pump; Reclose breakers to provide power to the equipment
32-SEIS	8.3.2-1	Isolate failed motor jacket cooling heat exchanger for motor-operated pump; Reclose breakers to provide power to the equipment
Seismically-induced SBO with Battery depletion	8.3.2-1	Isolate failed motor jacket cooling heat exchanger for motor-operated pump; Tie non-vital batteries to vital batteries within 8 hours

Question 720.38

By using the IRRAS input data provided by CE (on floppy disk), the staff requantified 65 out of the 67 zero-level fault trees. The two fault trees that could not be quantified (gave error messages) are POLX01EX and POLX01HX, both involving long-term decay heat removal. Please provide us a corrected floppy disk.

Response 720.38

The fault trees POLX01EX and POLX01HX, involving long-term decay heat removal with loss of 125 Vdc bus and 4.16 KV bus, respectively, were too large to run directly in IRRAS because of computer memory (RAM) limitation. Thus, following procedure, which is illustrated by an example, was employed to quantify these fault trees.

The fault tree POLX02EX was run as a sequence consisting of the fault trees PJOB01RE, failure to establish residual heat removal (RHR) or shutdown cooling flow for long-term heat removal with 125 VDC bus B unavailable, PAOG01ME, failure to deliver emergency feedwater (EFW) to either steam generator (SG) with 125 VDC bus B unavailable, and PMSA01ME, failure to deliver startup feedwater to either SG with 125 VDC bus B unavailable.

HR2_125B = PJOB01RE ↓ PAOG01ME ↓ PMSA01ME ↓

The cutsets for this sequence were obtained using IRRAS.

Similarly, the fault tree POLX03EX was run as a sequence consisting of the fault trees PJOB01RE, failure to maintain RHR or shutdown cooling flow for long-term heat removal with 125 VDC bus B unavailable, and PAOG01RE, failure to restart EFW system for long-term heat removal with 125 VDC bus B unavailable.

HR3_125B = PJOB01ME ↓ PAOG01RE ↓

The cutsets for this sequence were obtained using IRRAS.

The cutsets obtained for the sequences HR2_125B and HR3_125B were then processed through a proprietary utility code to eliminate the duplicate and nonsense cutsets, and then combined to obtain the cutsets for the fault tree POLX01EX. Finally, the fault tree POLX01EX was quantified by loading the cutsets for POLX01EX as a fault tree in IRRAS.

Similar procedure was followed to quantify the fault tree POLX01HX.

The final cutsets for POLX01EX and POLX01HX are provided on the floppy disk as POLX01EX.CUT and POLX01HX.CUT (in ASCII format as required by IRRAS).

Question 720.41

Please provide us a floppy disk containing the IRRAS input data for calculating the core damage frequencies attributable to tornado strike events.

Response 720.41

IRRAS was not used to calculate the core damage frequencies attributable to tornado strike event sequences. However, the cutsets for tornado strike event sequences from which the core damage frequencies were calculated are presented in Tables 8.2.1-2 through 8.2.1-8.

Question 720.45

Provide a comparison of the sequences in the CESSAR 80 PRA leading to (at least) 90 percent of the estimated core damage frequency with the corresponding sequences from the CESSAR 80+ PRA. Discuss the reasons for the improvement/worsening of the estimated core damage frequency in the corresponding 80+ PRA sequences.

Response 720.45

Table 8.1-4 in the System 80+ PRA Report, DCTR-RS-02, Rev. 0, January, 1991, provides a comparison between the estimated core damage frequency for System 80 and the estimated core damage frequency for System 80+ at the initiator level. The attached slides present the equivalent comparison at the initiator level with an assessment of the System 80+ design enhancements that resulted in the reduction in the estimated core damage frequency.

Question 720.46

Describe how loss off offsite power (LOOP) is modeled for seismic events (e.g., in the seismic event trees developed in figure 7.3-3).

Response 720.46

It was assumed that all seismic events resulted in a loss of offsite power lasting longer than 24 hours. The unavailability of the offsite power sources was reflected in the electrical distribution system models. Use of the Standby Alternate AC source was not credited because it is not seismically qualified.

RISK REDUCING FEATURES FOR DOMINANT SEQUENCE #1

SEQUENCE TYPE - Loss of Offsite Power (LOOP) including Station Blackout with Battery Depletion

REPRESENTATIVE DOMINANT SEQUENCE

(LOOP) (FAILURE of EFW)

FREQUENCY

OLD - 4E-5

NEW - 1E-7

FEATURES

- o Alternate AC Power Source (Gas Turbine)
- o Separate Offsite Power Source that Bypasses the Switchyard
- o Dedicated Battery for Each Diesel Generator
- o Four Train Emergency Feedwater (two with Turbine Driven Pumps
- o Turbine Generator Able to Run Back to Hotel Load.

RISK REDUCING FEATURES FOR DOMINANT SEQUENCE #2

SEQUENCE TYPE - TRANSIENTS

REPRESENTATIVE DOMINANT SEQUENCE
(LOFW) (FAILURE TO DELIVER EMERGENCY FW)

FREQUENCY
OLD - 1E-5
NEW - 3E-8

FEATURES

- o FOUR TRAIN EMERGENCY FEEDWATER SYSTEM
- o REDUNDANT SOURCES OF EMERGENCY FEEDWATER
  - 2 EFW TANKS
  - CONDENSATE STORAGE TANKS
- o HIGH RELIABILITY COMPONENT COOLING SYSTEM
  - TWO PUMPS PER TRAIN
  - NORMALLY RUNNING
- o START-UP FEEDWATER SYSTEM
  - FROM CONDENSATE STORAGE TANK
  - ACTUATED BEFORE EFW
- o FULL RUN-BACK CAPABILITY
- o TWO EFW ACTUATION SYSTEMS
  - REDUNDANT
  - DIVERSE

RISK REDUCING FEATURES FOR DOMINANT SEQUENCE #3

SEQUENCE TYPE - STEAM GENERATOR TUBE RUPTURE

REPRESENTATIVE DOMINANT SEQUENCE

(SGTR) (FAILURE TO DELIVER EFW)
(SGTR) (FAILURE OF SAFETY INJECTION)

FREQUENCY

OLD - 1E-5
NEW - 9E-8

FEATURES

- o FOUR TRAIN EMERGENCY FEEDWATER SYSTEM
- o FOUR TRAIN SAFETY INJECTION SYSTEM
- o SAFETY DEPRESSURIZATION SYSTEM

RISK REDUCING FEATURES FOR DOMINANT SEQUENCE TYPE #4

SEQUENCE TYPE - SMALL LOCA

REPRESENTATIVE DOMINANT SEQUENCE

(SMALL LOCA) (FAILURE OF SI RECIRCULATION)
(SMALL LOCA) (FAILURE OF SI INJECTION)

FREQUENCY

OLD - 9E-6
NEW - 4E-8

FEATURES

- o IN-CONTAINMENT REFUELING WATER STORAGE TANK
- o FOUR TRAIN SAFETY INJECTION SYSTEM
- o ELIMINATION OF RAS
- o SAFETY DEPRESSURIZATION SYSTEM

RISK REDUCING FEATURES FOR DOMINANT SEQUENCE #5

SEQUENCE TYPE - ATWS

REPRESENTATIVE DOMINANT SEQUENCES

(ATWS) (ADVERSE MTC)

FREQUENCY

OLD - 5E-6
NEW - 2E-7

FEATURES

- o LARGER PRESSURIZER
- o LARGER STEAM GENERATOR
- o SAFETY DEPRESSURIZATION SYSTEM
- o DIVERSE PROTECTION SYSTEM

Question 720.47

Please provide us a floppy disk containing the IRRAS input data for calculating all the seismic core damage sequences delineated by the seismic event tree presented by Figure 7.3-3.

Response 720.47

The core damage frequencies attributable to seismic event sequences were calculated by using the Seismic Integration Program (SIP) code. However, the cutsets for seismic event sequences from which the core damage frequencies were calculated are presented in Tables 8.3.1-2 through 8.3.1-15.

Question 720.48

Please provide the list of random failure probabilities as well as the fragility data of all the seismically-induced basic events used in quantifying the seisric fault trees and event trees.

Response 720.48

The random failure probabilities for basic events used throughout this PRA are presented in tables 5-2 through 5-7 of the System 80+ PRA Report (DCTR-RS-02 Rev. 0, January, 1991). The fragilities for seismically induced basic events used in this PRA are presented in table 7.3-2 of the PRA Report.

Question 720.49

Although seismic failures of the containment spray system (CSS/RHR) heat exchangers for long-term decay heat removal are considered in the seismic fault tree analysis, such failures, however, do not include a heat exchanger pipe break that could drain the contents of the IRWST and lead directly to core damage. Justify omitting such an accident scenario when estimating the seismic core damage frequency.

Response 720.49

This accident sequence was not included in the System 80+ seismic PRA. C-E is currently updating the System 80+ PRA and this seismic accident sequence will be evaluated, but it is not expected that the results will be significantly affected.

Question 720.51

A close examination of the seismic event analysis performed in the System 80+ PRA revealed that virtually no consideration was given to possible failures of important plant structures that may ensue from a seismic event. Please discuss the possible impact of such omissions on the core damage frequency estimates. In this connection, also provide the System 80+-specific fragility parameter calculations for the following structures and components: containment, reactor vessel, reactor internals, reactor-coolant piping, pressurizer, turbine building, main control room (including control room suspended ceiling, if any), condenser hotwell, emergency feedwater tanks, feedwater heaters, CRD guide tubes, CRD housings and fuel assemblies.

Response 720.51

An Advanced Reactor Severe Accident Program (ARSAP) contractor performed the initial portions of the System 80+ Seismic PRA. Part of these analyses, as described in section 7.1.8 of the System 80+ PRA Report, DCTR-RS--02, Rev. 0, January, 1991, and in Reference 66 of the PRA report, was a qualitative assessment of the design features of System 80+ as compared to the design features of plants for which a detailed seismic PRA had been performed. Based on this review, it was determined that the seismic capacity of the plant structures comprising the nuclear annex would be in excess of 3 g. Therefore, failure of these structures was not addressed in the seismic PRA. Equipment in structures outside of the nuclear annex which might have lower seismic capacities were not credited in the seismic analysis. System 80+ specific fragility parameters were not calculated for the seismic PRA. These calculations will be based on as procured information as part of the detailed plant design phase. The component fragilities

Response 720.51 (Cont.)

Assumptions and Groundrules. These fragilities are considered to be achievable. It is believed that any uncertainty in the assumed fragilities is overshadowed by the uncertainty in the seismic hazard curve.

Question 720.52

The staff believes that fires and internal floods can be significant contributors to estimated core damage frequency. Please provide a fire PRA and an internal flooding PRA.

Response 720.52

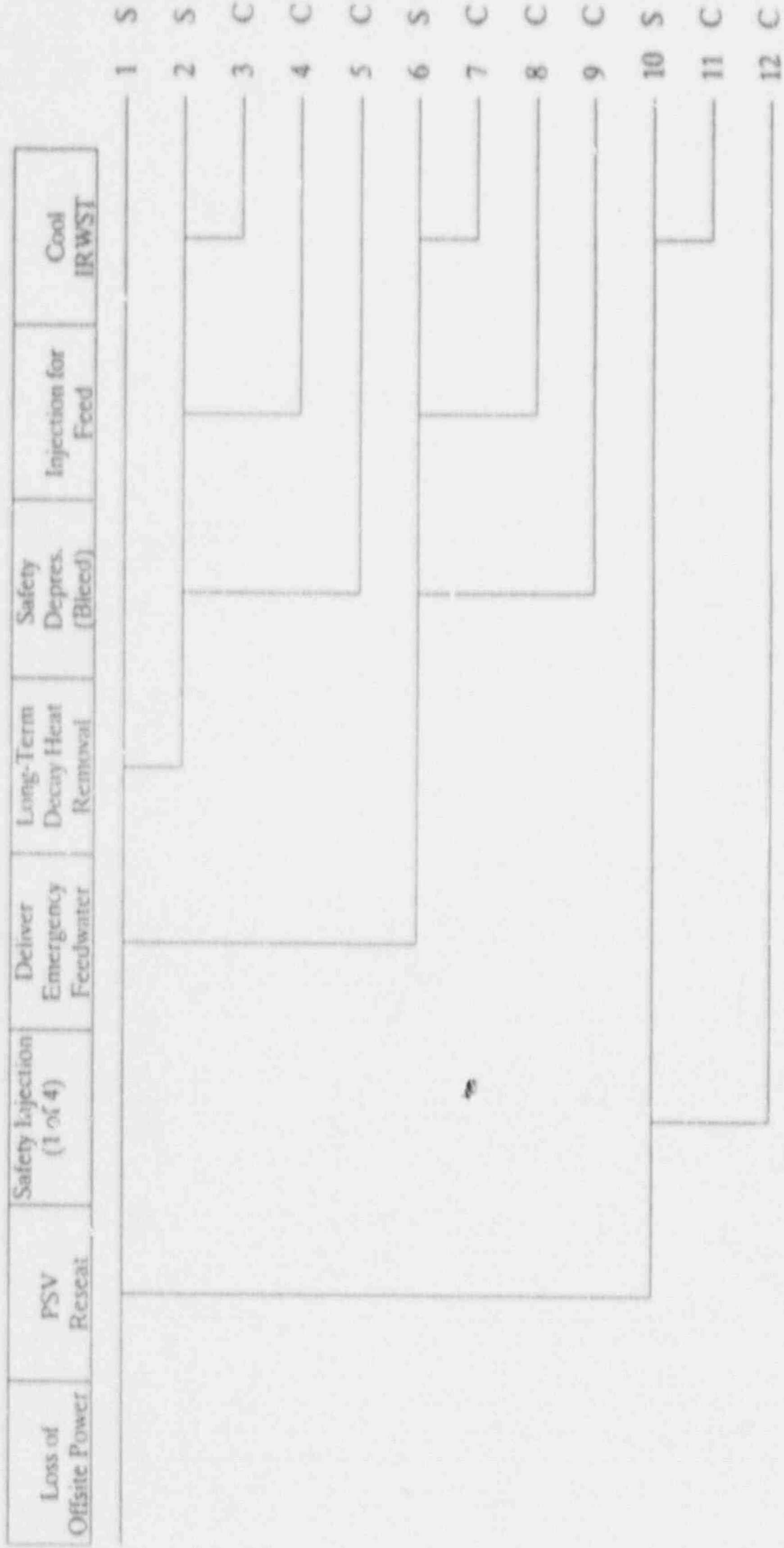
As discussed in sections 7.1.6 and 7.1.8 of the System 80+ PRA Report, DCTR-RS-02, Rev. 0, January, 1991, internal fires and internal floods were not considered to be significant risk contributors for System 80+ because of the high degree of separation and compartmentalization used in the System 80+ design. C-E is in the process of updating the System 80+ PRA. C-E will perform a more detailed evaluation of the risk potential for internal fires and floods as part of this PRA update. In addition, if the results of the qualitative analyses indicate that quantitative analyses are warranted, the quantitative analyses will also be included in the updated PRA.

Question 720.53

On page B-67 of the CESSAR, Appendix B, it is stated that "failure of a PSV to reseat after the primary side pressure decreases will result in a small Loss-of-Coolant Accident (LOCA) with offsite power unavailable. This is considered to be a small LOCA initiator for quantification of small LOCA frequencies." Essentially identical statements are also made on page B-144 for Tornado strike sequence analysis. Were these sequences actually transferred to the small-LOCA event tree? If so, what are their frequencies?

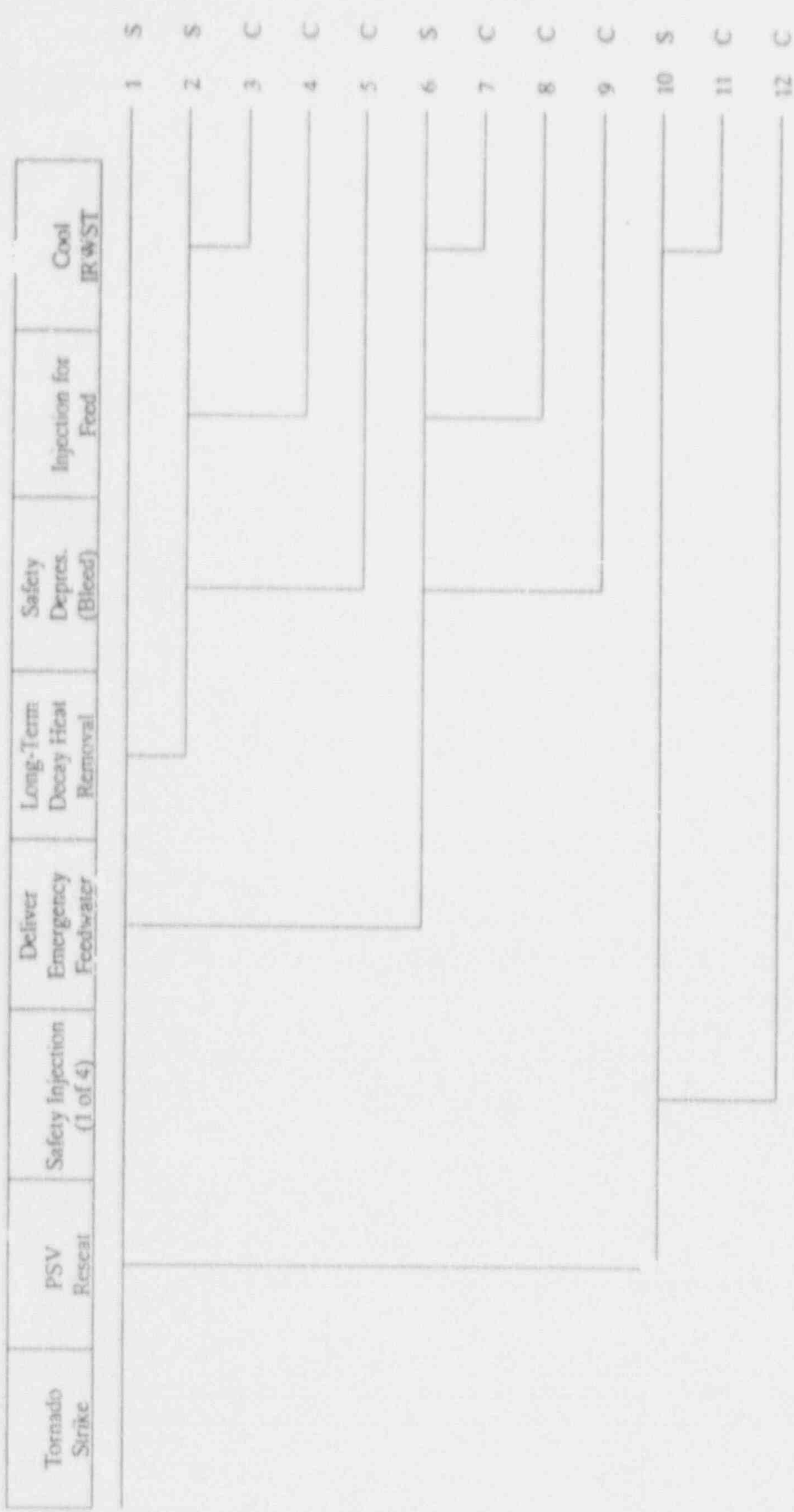
Response 720.53

The statements regarding PSV LOCAs following a LOOP or tornado strike initiator are incorrect. These "consequential" small LOCAs were treated within the appropriate event trees as shown in figures B3.1.8-1 and B4.2.3-1 (presented here for convenience). The statements on pages B-67 and B-144 were inadvertently left in from a previous iteration of the PRA. C-E is in the process of updating the System 80+ PRA and these statements will be deleted.



Amendment F
December 15, 1989

SYSTEM 80+	Loss of Offsite Power Event Tree	Figure B3.1.8-1



Amendment H
August 31, 1990

SYSTEM 80+™	Tornado Strike Event Tree	Figure B4.2.3-1

Question 720.54

On page 6-522 of the System 80+ PRA, it is stated that the electrical power for the CVCS (chemical and volume control system) is supplied by the 4.16 kV non-class 1E, 125 VDC non-class 1E, and 480 VAC non-class 1E power systems. More specifically, it is stated that the 4.16 kV non-class 1E power system provides the motive power to operate the charging pumps, the 480V non-class 1E Load Center provides the motive power to operate the boric acid makeup pumps, and the 480 VAC MCC system provides the motive power to operate the AC motor-operated valves. Focussing attention on the ATWS sequences delineated on the seismic event tree (Figure 7.3-3, sheet 2), which presumably assumes a concurrent loss-of-offsite power (LOOP), it is not clear whether the emergency diesel can supply power to the CVCS. If not, how can the event-tree top event, "Deliver Boron" be achieved under LOOP conditions?

Response 720.54

C-E agrees that the emergency diesels cannot provide power to the 4.16 kV permanent non-safety buses for a seismic event because the permanent non-safety buses are not seismically qualified and are assumed to be unavailable. Thus, the CVCS can not be used to "Deliver Boron" for ATWSs with a concurrent loss-of-offsite power (LOOP). The seismic ATWS sequences are modeled incorrectly. HPSI pumps would be required for boron delivery to provide long term reactivity control. The primary system would have to be depressurized using the depressurization valves before the HPSI pumps could be used for boron delivery. C-E is currently updating the System 80+ PRA to reflect some system design changes. The seismic ATWS sequences will be corrected in the revised PRA.

Question 720.56

Please concisely describe how the core damage frequency associated with seismically-induced station blackout sequences was calculated.

Response 720.56

See the responses to Questions 720.57 and 720.62.

Question 720.57

Please elaborate on the computational procedures for combining the fault trees shown in Figure 7.3-4 with the seismic hazard curve and the relevant fragility data shown in Table 7.3-2 to yield the core damage frequency.

Response 720.57

The fault tree presented in figure 7.3-4 represents the core damage sequence attributable to a seismically-induced station blackout with subsequent battery depletion. This model was solved using the methods discussed in the response to Question 720.62.

Question 720.58

What is the fragility for the seismically-induced basic event, SEISLOP (seismically induced failure of the switchyard) appearing in Figure 7.3-4? What recovery factor was used for the basic event HXZRCVR (operator fails to recover CCWS in seismic event)?

Response 720.58

The seismically-induced basic event, SEISLOP (seismically induced failure of the switchyard) appearing in Figure 7.3-4 is the same as the event, IE-SEISLOP (seismically induced loss of offsite power) presented in table 7.3-2. The fragility for this basic event was 0.3g. It was essentially assumed in the seismic analysis that all seismic events would result in a loss of offsite power lasting more than 24 hours.

The recovery factor used for the basic event, HXZRCVR (operator fails to recover CCWS in seismic event), was 1.0E-1 with an assumed error factor of 1.00. This value is presented in table 5-7 of the System 80+ PRA Report (DCTR-RS-02 Rev. 0, January, 1991). A copy of this table was provided as part of the response to question 720.48.

Question 720.59

Although seismically-induced large-break LOCAs are modeled in the seismic event trees, there is no modeling of possible seismically-induced small-break LOCAs, such as a break of a small RCS pipe or a break of the letdown line. Also, unlike the case with internal events analysis, no consideration is given to consequential steam generator tube rupture (SGTR) following an ATWS in the seismic events analysis. Justify the omission of these events from your seismic core damage frequency evaluation.

Response 720.59

The contractor that prepared the System 80+ seismic event trees determined that seismically-induced large and small LOCAs would occur at about the same seismic peak ground acceleration level and thus only modeled large LOCAs as being the more limiting of the two break sizes. Consequential SGTRs were not modeled because they are low probability sequences. C-E is currently updating the System 80+ PRA. Seismically-induced small LOCAs and consequential SGTRs will be evaluated for the seismic event trees in this update.

Question 720.60

Was any distinction made between ATWS events and non-ATWS events in assigning a probability for the seismic event-tree (Figure 7.3-3) top event, X (PSV Reseat)? What is the probability assigned to the top event, X, in quantifying the seismic accident sequences #11, #12, #31, and #32. Note that for ATWS events, the initial RCS pressure can be expected to be much higher, requiring more PSVs to open in order to relieve RCS pressure.

Response 720.60

There was no distinction made between ATWS and non-ATWS events in assigning a probability to the seismic event tree top event X (PSV Reseat). The probability assigned to top event X was 2.8E-2 per demand. As described in section 5.6.1 of the System 80+ PRA Report, DCTR-RS-02, Rev. 0, January, 1991, this probability is based on any one of four PSVs failing to reseat.

Question 720.62

Please provide a concise but systematic description of how the seismic fault trees are used in conjunction with the seismic hazard curve, component, and structure fragilities and other random failure probability data to obtain, through discretization and convolution, the frequency of individual accident sequence based on the Boolean expression derived from the seismic event trees.

Response 720.62

The seismic fault tree models presented in Appendix 7A of the System 80+ PRA Report, DCTR-RS-02, Rev. 0, January, 1991, were based upon the internal event fault tree models and include basic events representing both random failures of components and seismically-induced failure of components. These fault trees were initially solved to generate system cutsets using IRRAS 2 Beta Draft and SETS. For this preliminary solution of the seismic fault trees, the failure rates for the basic elements representing seismically-induced component failures were set to 1.0 to ensure that the cutsets containing these basic events were generated. Next, the IRRAS 2, Beta Draft, event sequence solution module was used to solve the "Boolean Equation" for each seismic core damage sequence using the seismic fault tree cutsets generated in the previous step. This produced a set of core damage cutsets for each seismic core damage sequence. These cutsets included both random failures and seismically-induced failures. These sequence cutsets were manually edited to ensure that the .NOT. logic had been properly applied and to combine cutsets containing seismic failures of like components based on the assumption that "one fail, all fail" for equivalent components.

Next, the Seismic Integration Program (SIP) computer code was used for the final solution of the cutsets for each seismic core damage sequence. SIP is a modified version of th SAMPLE code. The cutsets for a given seismic core damage sequence are converted to a Boolean equation in a FORTRAN subroutine which is then linked with the main body of the SIP code. The data file used by SIP contains three types of data; 1) a discretized version of the seismic hazard curve, 2) the median fragility and combined uncertainty for each seismically-induced failure, and 3) the mean failure probability for each random failure. SIP randomly selects a seismic acceleration from the input seismic hazard curve. Next, SIP calculates a probability of failure for each basic event representing a seismically-induced failure, given the selected seismic acceleration. The equation set for the seismic core damage sequence is then solved to generate a conditional core damage frequency estimate for the selected seismic event. Finally, an unconditional core damage frequency estimate is calculated as the product of the probability of having a seismic event with

Response 720.62 (Cont.)

the selected acceleration and the conditional core damage frequency estimate. This sampling process is repeated several thousand times for a given seismic core damage sequence. The core damage frequency estimates are used to calculate a mean core damage frequency and error factor for the specific seismic core damage sequence. SAMPLE was then used to calculated an overall mean seismic core damage frequency and error factor based on the mean core damage frequencies and error factors for all seismic core damage sequences.

Question 720.64

Please provide the detailed information regarding the seismic capacity of the fire protection system, including pumps, valves, and relevant equipment. Please indicate the median capacity as well as the parameters representing randomness and uncertainty.

Response 720.64

C-E did not credit the fire protection system in the seismic analysis. Therefore, the seismic capacities for the fire protection system equipment were not evaluated.

Question 720.65

Should the seismic fault tree top event, PAOG5MDX (motor-driven EFWP-102 unavailable, Figure 7A-1) contain, under GATE 2127, an additional event CVNOV138-143? Similarly, should the top event, PAOG9MDX contain, under GATE 2140, an additional event CVNOV238-243?

Response 720.65

C-E is currently updating the System 80+ PRA. These fault trees will be reviewed and revised appropriately as a part of this update.

Question 720.66

It appears that the following basic events are missing in some of the fault trees developed for high pressure injection system (Figure 7A-2) and "feed and bleed" (Figure 7A-3).

Basic Event	Fault tree Top Event where the Basic Event is Missing
CVNOV134-135	PHSG08DX, PHHB07DX
CVNOV136-137	PHSG09DX, PHBR08DX
CVNOV234-235	PHSG10DX, PHHB09DX
CVNOV136-137	PHSG11DX, PHBB10DX

Please comment on this.

Response 720.66

C-E is currently updating the System 80+ PRA. These fault trees will be reviewed and revised appropriately as a part of this update.

Question 720.67

In the seismic fault trees developed for containment spray system (Figure 7A-5), should the fault tree top event, PGOB13BX, contain an additional event, GVNOCS542 (flow diverted via the mispositioned valve CS-542)?

Response 720.67

C-E is currently updating the System 80+ PRA. These fault trees will be reviewed and revised appropriately as a part of this update.

Question 720.68

Comparison of the fault trees developed for long-term residual heat removal shown respectively in Figure 6.4.1-2 and Figure 7A-7 revealed that there are noticeable differences in specifying the input events to OR-gates PJOB07RX, PJOB17RX and PJOB22RX. Please explain these inconsistencies.

Response 720.68

The inconsistencies essentially consist of omission of basic events JVNOSD757 in the fault tree PJOB07RX and JVNOSD756 in the fault tree PJOB17RX, and incorrectly calling the same set of valves differently in the Figure 7A-7. C-E is currently updating the System 80+ PRA. These fault trees will be reviewed and revised appropriately as a part of this update.

Question 720.69

Is there a missing event, PMSA01MX (failure to deliver sufficient startup FW to either SG) under the the fault tree top event, POLX02BX (failure to establish RHR and failure to maintain secondary heat removal) developed for long-term decay heat removal (Figure 7A-6)?

Response 720.69

The Startup Feedwater System is a non-seismic system and, therefore, can not be credited to provide cooling water to the SGs during a seismic event. Hence, the event, PMSA01MX (failure to deliver sufficient startup FW to either SG), is not included under the the fault tree top event, POLX02BX (failure to establish RHR and failure to maintain secondary heat removal) developed for long-term decay heat removal during a seismic event (Figure 7A-6).

Question 720.70

It appears that certain basic events are omitted from some of the fault trees developed for long-term secondary side heat removal (Figure 7A-8). They are summarized as follows:

Basic Event	Fault tree where Omission is made
AVMX-EF2B2	PAOG4MBX
CVNOV138-143	PAOG5MBX (GATE 2510)
AVSX-EF2A2	PAOG6MBX
AVMX-EF2B2	PAOG8MBX
CVNOV238-243	PAOG9MBX (GATE 2532)

Please comment on this.

Response 720.70

The long-term decay heat removal is initiated later in the transient following the plant cooldown to residual heat removal entry conditions using secondary side heat removal. The Residual Heat Removal (RHR) System is then used for long-term heat removal. If the RHR System not available, secondary side heat removal must be maintained for long-term decay heat removal using the Emergency Feedwater (EFW) System. Therefore, successful operation of the EFW System is assumed when the long-term decay heat removal is implemented. The basic events CVNOV138-143 and CVNOV238-243 (Component Cooling Water valves for the EFW pumps not available due to maintenance error) need not be included in the fault trees since the emergency feedwater was successfully delivered during initial plant cooldown. Furthermore, since it was felt that the basic events AVMX-EF2B2 and AVSX-EF2A2 (common cause failures of SG isolation valves) are demand-type (fail to open) events and since the EFW System operated successfully prior to initiation of long-term decay heat removal, these events need not be included in the fault trees.

Question 720.71

Should the fault tree top event PC3N02BX (failure to deliver flow from CCW loop 1A) developed in Figure 6.3.3-2 contain an additional event PEEN01A6 (loss of 125 VDC control power for train A)? Similarly, should the fault tree top event PC4N02BX (failure to deliver flow from CCW loop 2A) developed in Figure 6.3.3-3 contain an additional event PEEN01B6 (loss of 125 VDC control power for train B)?

Response 720.71

The Electrical Distribution System (EDS) support systems for the Component Cooling Water (CCW) pumps in a particular division (or train) are similar to those for the front line systems in the same division. These EDS support systems are addressed under the frontline systems. Therefore, the duplication of these EDS support systems in the CCW model is not warranted. Hence, the additional events mentioned above need not be included in the corresponding fault trees.

Question 720.72

In Figure 7A-8, the basic event, AVSX-EF2A2, is defined to be "CCF of SG isolation valve set 2A(2), EF-100,EF-102." The same set of isolation valves, however, are defined to be EF-100/EF-101 in Figure 6.3.7-2. A similar discrepancy can also be found for the basic event AVSX-EF2B2 appearing in the same figures. Please clarify those inconsistencies.

Response 720.72

The definitions of SG isolation valve sets given in Figure 6.3.7-2 reflect the latest design of the Emergency Feedwater System. In Figure 6.3.7-2, the basic event AVSX-EF2A2 represents the CCF of SG isolation valve set EF-100/EF-101. However, in the same figure, the CCF of SG isolation valve set EF-102/EF-103 is represented by the basic event AVMX-EF2B2. On the other hand, in Figure 7A-8, the basic event AVSX-EF2A2 represents the CCF of SG isolation valve set EF-100,EF-102 and the basic event AVSX-EF2B2 represents the CCF of SG isolation valve set EF-101,EF-103. The basic events were called differently in Figure 7A-8 when the fault trees used in the seismic analysis were modified from the fault trees for the internal events. However, this inadvertent error does not impact the results since the appropriate numerical values of the probability and fragility been used in the respective analysis. C-E is currently updating the System 80+ PRA. These fault trees will be reviewed and revised appropriately as a part of this update.

Question 720.73

Comparison of the 125 VDC bus fault trees developed respectively for internal events (Figure 6.3.1-7) and seismic events (Figure 7A-16) revealed that many of the fault tree top events (inquiring about the availability of battery power) constructed for the latter contain an extra basic event, EBTA8HR (battery depleted-no recovery in 8 hours). (a) Should this basic event also be included in the corresponding top events for the former? (b) What probability was assigned to this basic event in the seismic quantification?

Response 720.73

A prolonged loss of offsite power is anticipated during a seismic occurrence, thus placing demands on the onsite power sources, such as the 125 VDC buses. In addition, a seismic occurrence may render a 125 VDC bus unavailable and the 125 VDC battery is more likely to be depleted since the offsite power cannot be restored for a prolonged time period. For this reason, the basic event EBTA8HR was included in the fault tree top events constructed for the seismic events. In the case of the internal events, Station Blackout is the *only* event where the 125 VDC bus may be depleted if the alternate standby AC power cannot be established or the offsite power cannot be restored within a certain time. A fault tree was developed to account for this scenario when determining the frequency of occurrence for the Station Blackout event (see Figure 4.8-2 on page 4-63 in the report

Response 720.73 (cont.d)

DCTR-RS-02, Rev. 0, Volume 1). Therefore, the basic event EBTA8HR should not be included in the corresponding top events constructed for the internal events.

Question 720.74

Are the fault trees, PEENO2A5 and PEENO2B5 (shown in Figure 7A-16) specifically developed for seismic events? Why there are no corresponding fault trees for internal events?

Response 720.74

The fault trees, PEENO2A5 and PEENO2B5, were initially developed for the internal events, but subsequently deleted because they were not used in the analysis. Although these fault trees were not used in the seismic analysis, they were inadvertently included in the report.

Question 720.75

It is not clear why the basic event, FSERAPS (no actuation signal from alternate protection system) is included in the actuation signal fault trees developed for seismic events, but is not included in those developed for internal events. What is the fragility assigned to this basic event in the seismic quantification?

Response 720.75

In the analysis for the internal events, it was assumed that, except for an Anticipated Transient Without Scram (ATWS) event, either the reactor trip is not required (Large and Medium LOCAs) or the reactor would automatically be tripped by the Reactor Protection System (RPS) on a safety parameter. The Alternate Protection System (APS) is provided to address an event in which the RPS fails to generate the trip signal. ATWS is such an event. The analysis for the internal events defined an ATWS as an occurrence of a transient requiring a reactor trip for reactivity control coupled with the failure of a reactor trip to occur. The failure of a trip could be due to either mechanical failure of the Control Element Assemblies (CEAs) or the failure of both the RPS and APS to generate a trip signal. The basic event FSERAPS does show up in many of the ATWS sequence cutsets. In addition, the APS generates an Alternate Feedwater Actuation Signal. Figure 7A-17 shows how the basic event FSERAPS is modeled in the fault tree model for the actuation signal. The seismically-induced failure of the actuation system is differentiated from the non-seismic failure at the top. No fragility value needs to be assigned to the basic event FSERAPS. However, the (mean) probability for this basic event is 2.6E-02 with an error factor of 3.0 (Table 5-2 in Volume 1 of DCTR-RS-02, Rev. 0).

Question 720.76

What are the probabilities and/or fragilities assigned to the following basic events: PC3N01MX (Figure 7.2-3), PC4N01MX (Figure 7.2-3), PCSEIS (Figure 7A-11), ECHATTER (Figure 7A-13), and ECHGSEIS (Figure 7A-16)?

Response 720.76

The basic events PC3N01MX and PC4N01MX shown in Figure 7.2-3 (Tornado-induced Station Blackout with Battery Depletion) were derived from the fault trees PC3N01MX and PC4N01MX for the internal events by adding an additional basic event CSWINTAKE (service water intake blockage due to tornado generated debris). This basic event, CSWINTAKE, was assigned a probability of 0.01 per demand. The fault trees PC3N01MX and PC4N01MX for the internal events were statistically combined with the basic event CSWINTAKE when analyzing the tornado strike sequences. Therefore, no direct probabilities were calculated for the basic events PC3N01MX and PC4N01MX shown in Figure 7.2-3. Similarly, the basic event PCSEIS (seismic failure of [Essential Service Water System] module) was derived based on the fault tree developed in Figure 7A-10 (on page 7A-108 in Volume 3 of DCTR-RS-02, Rev. 0). The basic event ECHATTER (failure to recover seismically-induced bus chatter) was assigned a probability of 0.05 per demand (Table 5-7 in Volume 1 of DCTR-RS-02, rev. 0). The fragility for the basic event ECHGSEIS (seismic failure of battery charger) was inadvertently left out of the Table 7.3-2 of the above mentioned report and has a value of 1.6g as specified in Table 7.3-1.

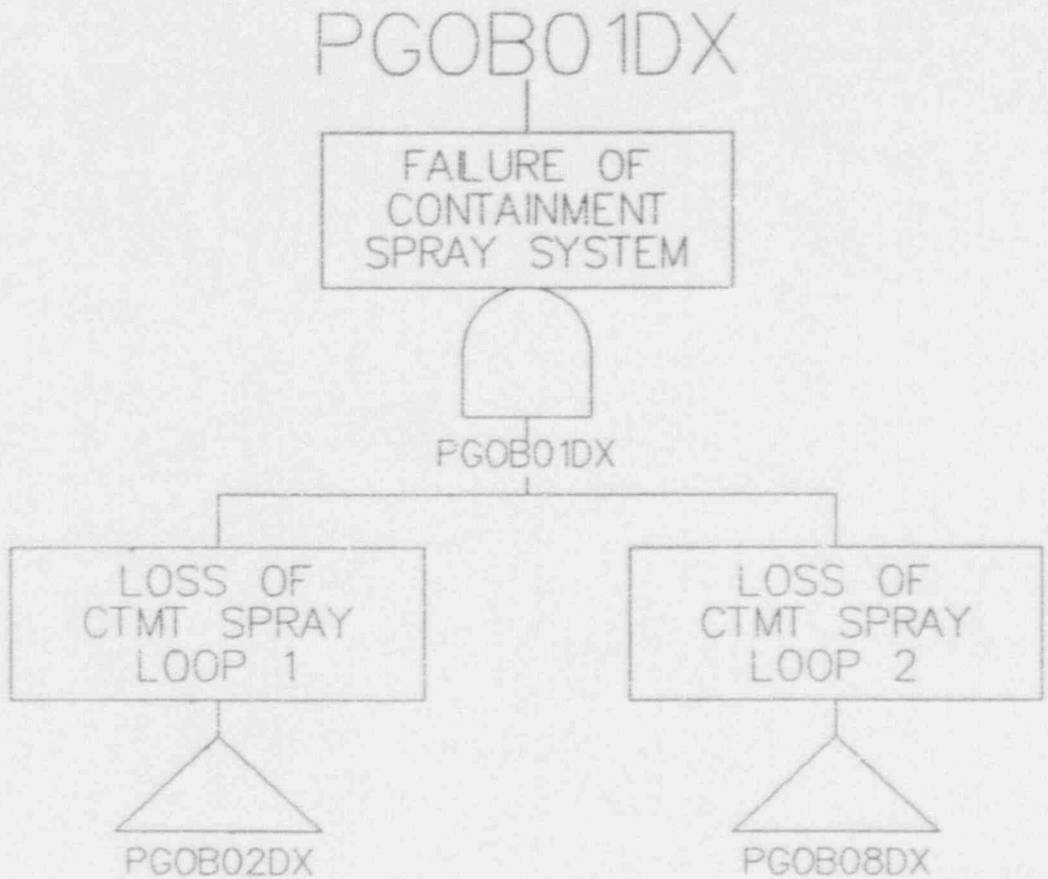
Question 720.77

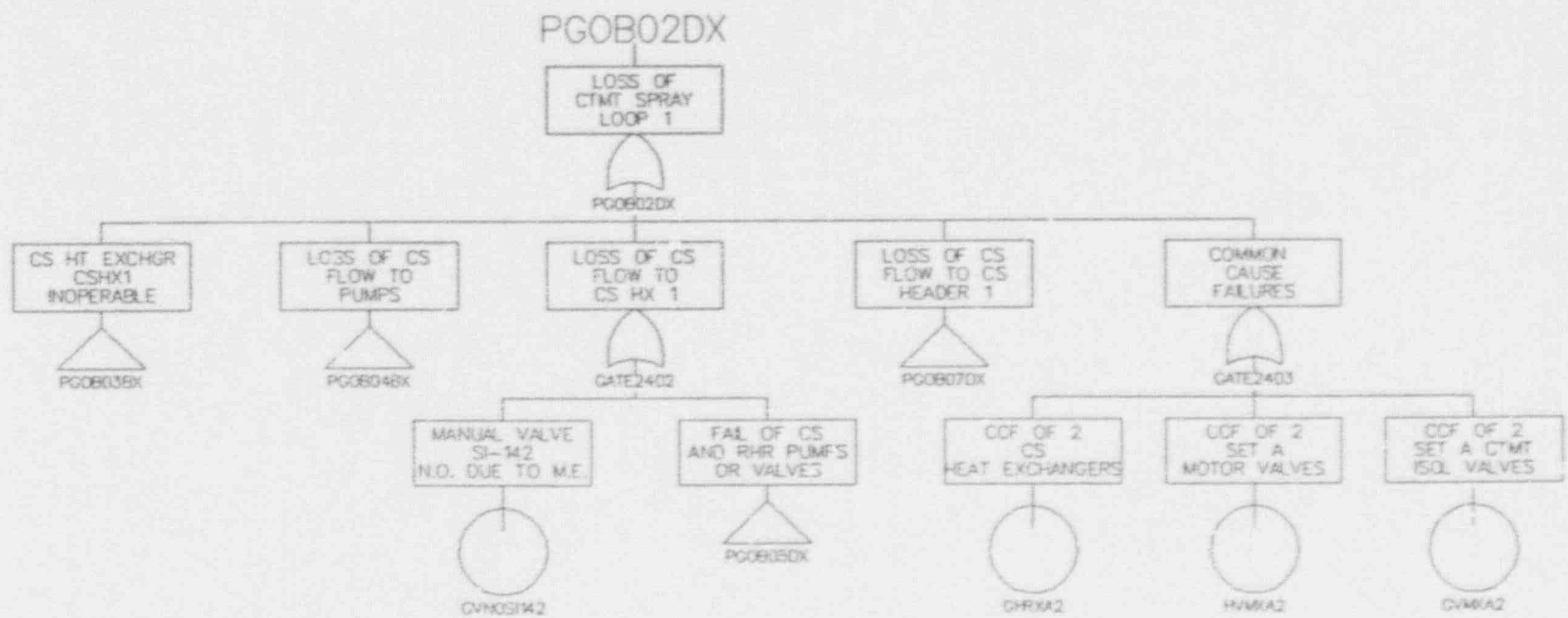
Please supplement the detailed fault trees (illustrating the fault tree logic) for the following zero-level fault tree top events which are missing in the System 80+ PRA:

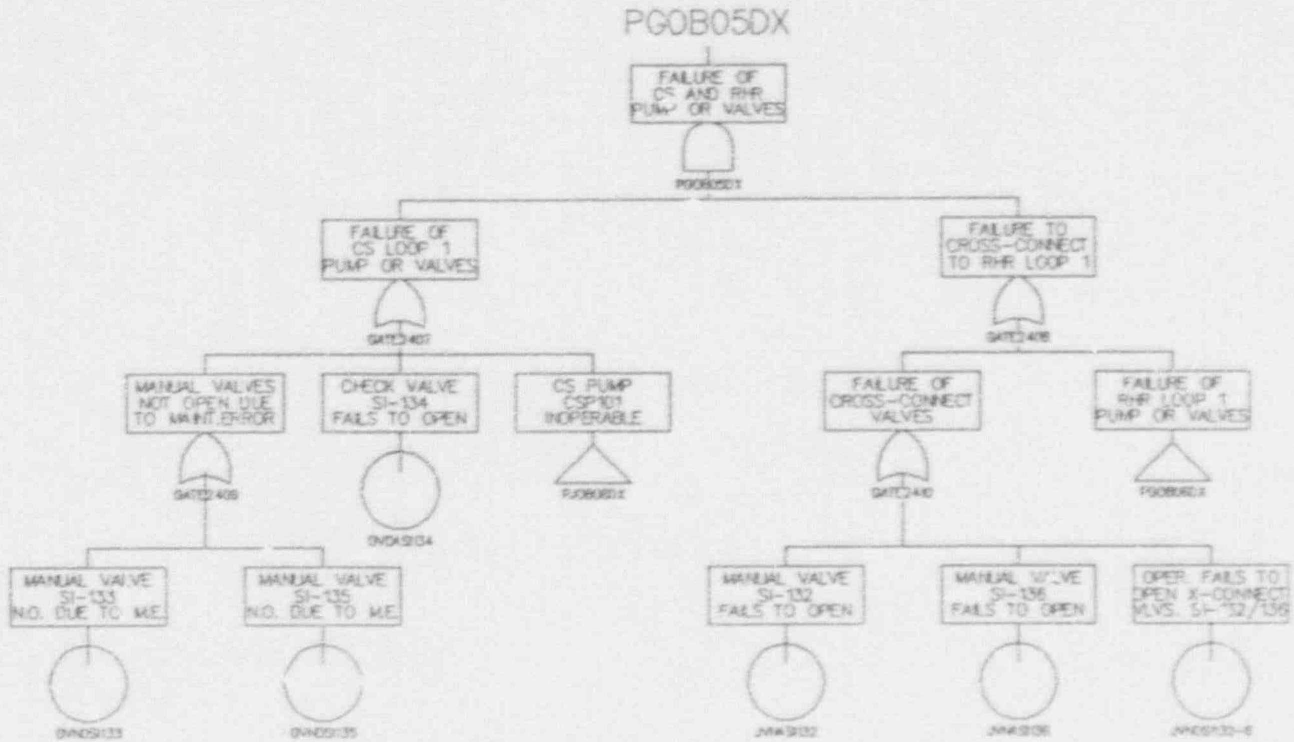
LCCSAPWR	PGSB01DX
LPWRCCSX	PGSB01EX
LPWRCCSY	PHBB01BX
LPWRPCCS	PHBB02EX
PAIB01MX	PJOB01BX
PAIB01RX	PLCH01BX
PGOB01DX	POLB01BX
PGOB01EX	PPAX1MBX
PGOB01SX	PVBB01EX
PGSB01BX	PVDB01BX
PGSB01CX	RCVR1

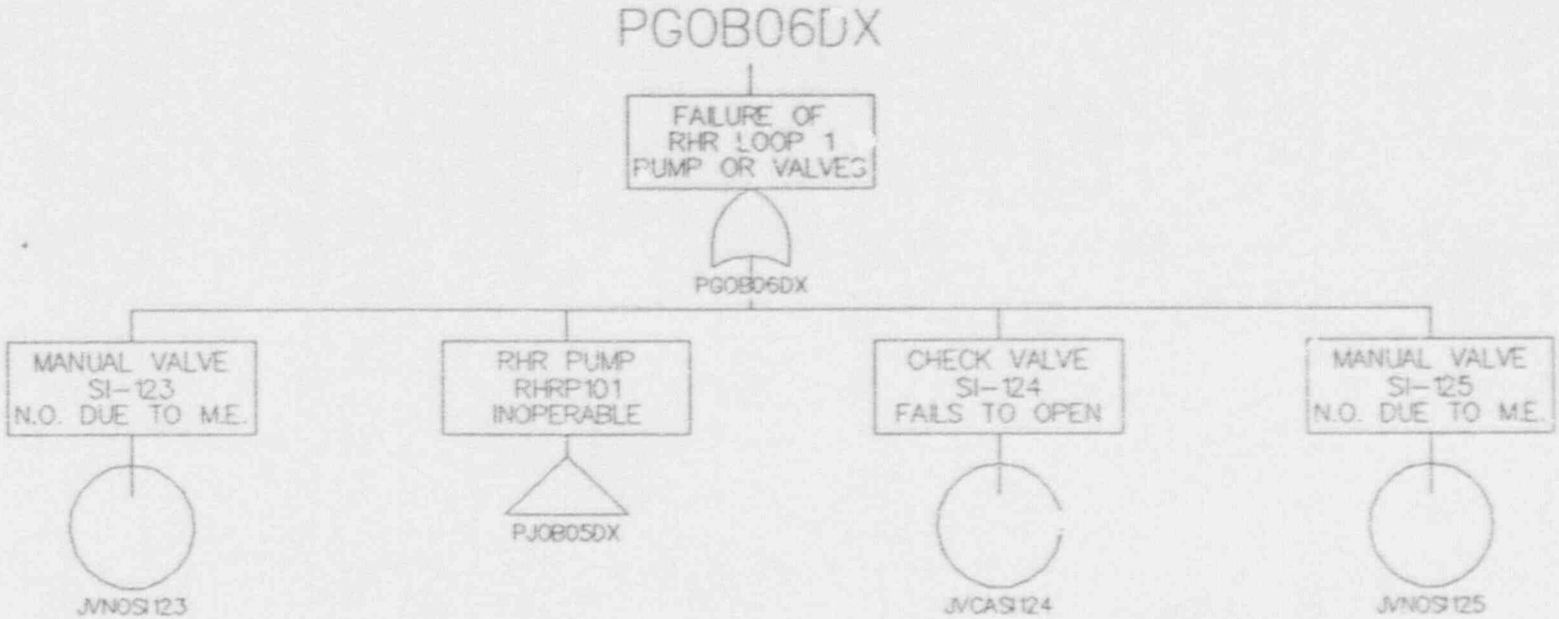
Response 720.77

The fault trees LCCSAPWR, LPWRCCSX, LPWRCCSY and LPWRPCCS were used only for design evaluation of the component control system but were not used in the System 80+ PRA analysis. Similarly, the fault trees PLCH01BX and PPAX1MBX were developed to model, respectively, the failure of safety injection tanks injection for medium LOCA and the RCS path unavailable for RCS pressure control logic but were not used in the PRA analysis as they were not needed. Therefore, these fault trees are not included here. The fault trees PAIB01MX, PAIB01RX are part of the zero-level fault tree top event POLB01BX, which is presented in Figure 6.4.1-3 starting on page 6-668 in volume 2 of the report DCTR-RS-02, rev. 0. The fault tree PAIB01MX (page 6-671) refers to two fault trees PAIB2MBX (page 6-382) and PAIB5MBX (page 6-385), which are part of the zero-level fault tree top event PAIB1MBX presented in Figure 6.3.7-5 starting on page 6-381 of the report. The fault tree PAIB01RX (page 6-672) eventually refers to two fault trees PAOG3MBX (page 6-360) and PAOG5MBX (page 6-362), which are part of the zero-level fault tree top event PAOG1MBX presented in Figure 6.3.7-2 starting on page 6-358 of the report. The fault trees PGOB01DX, PGOB01EX, PGOB01SX, PGSB01BX, PGSB01CX, PGSB01DX and PGSB01EX (and their subsequent level trees, if any) are included in their entireties as a part of this response. It is to be noted that the subsequent level(s) in any of the top level fault trees need not be in a sequential numeric order since the subsequent levels many times refer to other fault trees. The The fault tree PHHB01BX, is included as a part of the response to Question 720.11. The fault trees PHBB02EX, PVBB01EX and PVDB01BX are presented in Figure 6.3.6-13 starting on page 6-323, Figure 6.3.10-4 on page 6-460 and Figure 6.4.4-2 starting on page 6-874, respectively, in the report. Although only subsequent levels of the zero-level fault tree PJOB01BX are referred to by other fault trees, it is included here in its entirety. The fault tree RCVR1, which was used only to get the basic events representing the recovery actions in the IRRAS database, is also attached.

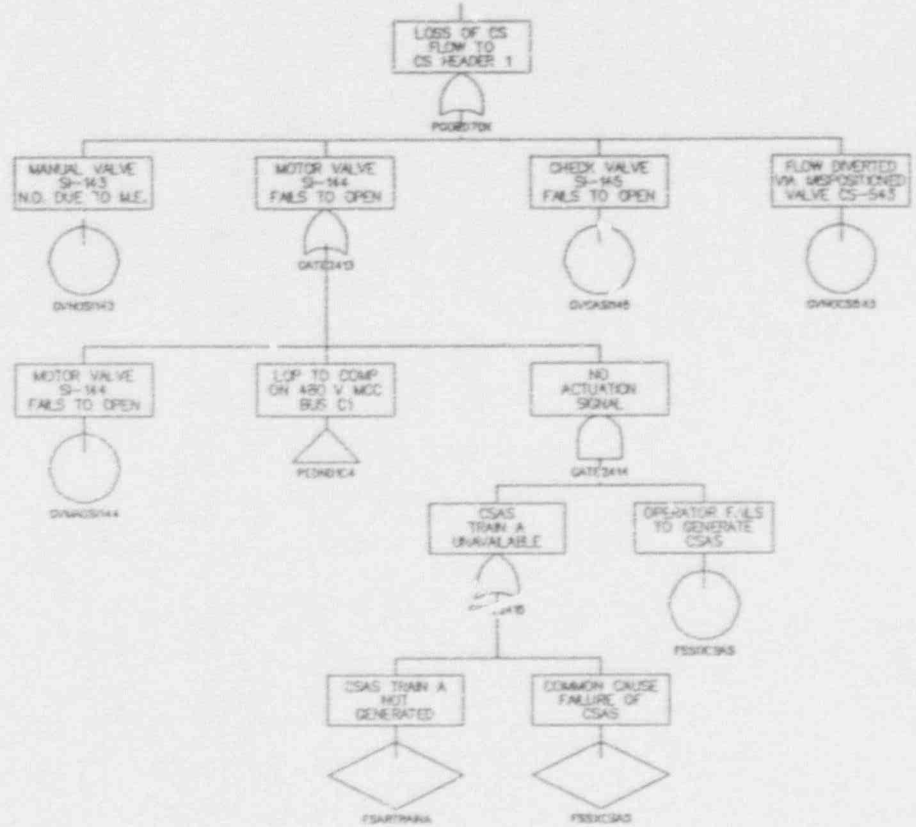

PGOB01DX
FAILURE OF CONTAINMENT SPRAY SYSTEM
PGOB01DX
LOSS OF CTMT SPRAY LOOP 1
LOSS OF CTMT SPRAY LOOP 2
PGOB02DX
PGOB08DX


PGOB02DX
LOSS OF CTMT SPRAY LOOP 1
PGOB02DX
CS HT EXCHGR CSHX1 INOPERABLE
PGOB03BX
LOSS OF CS FLOW TO PUMPS
PGOB04BX
LOSS OF CS FLOW TO CS HX 1
GATE2402
LOSS OF CS FLOW TO CS HEADER 1
PGOB07DX
COMMON CAUSE FAILURES
GATE2403
MANUAL VALVE SI-142 N.O. DUE TO M.E.
CVNOSI142
FAIL OF CS AND RHR PUMPS OR VALVES
PGOB05DX
CCF OF 2 CS HEAT EXCHANGERS
CHRXA2
CCF OF 2 SET A MOTOR VALVES
HVMXA2
CCF OF 2 SET A CTMT ISOL VALVES
CVMXA2


PG0B05DX
FAILURE OF CS AND RHR PUMP OR VALVES
FAILURE OF CS LOOP 1 PUMP OR VALVES
GATE2407
FAILURE TO CROSS-CONNECT TO RHR LOOP 1
GATE2408
MANUAL VALVES NOT OPEN DUE TO MAINT.ERROR
GATE2409
CHECK VALVE SI-134 FAILS TO OPEN
CS PUMP CSP101 INOPERABLE
FAILURE OF CROSS-CONNECT VALVES
GATE2410
FAILURE OF RHR LOOP 1 PUMP OR VALVES
MANUAL VALVE SI-133 N.O. DUE TO M.E.
MANUAL VALVE SI-135 N.O. DUE TO M.E.
MANUAL VALVE SI-132 FAILS TO OPEN
MANUAL VALVE SI-136 FAILS TO OPEN
OPER. FAILS TO OPEN X-CONNECT VLVS. SI-132/136

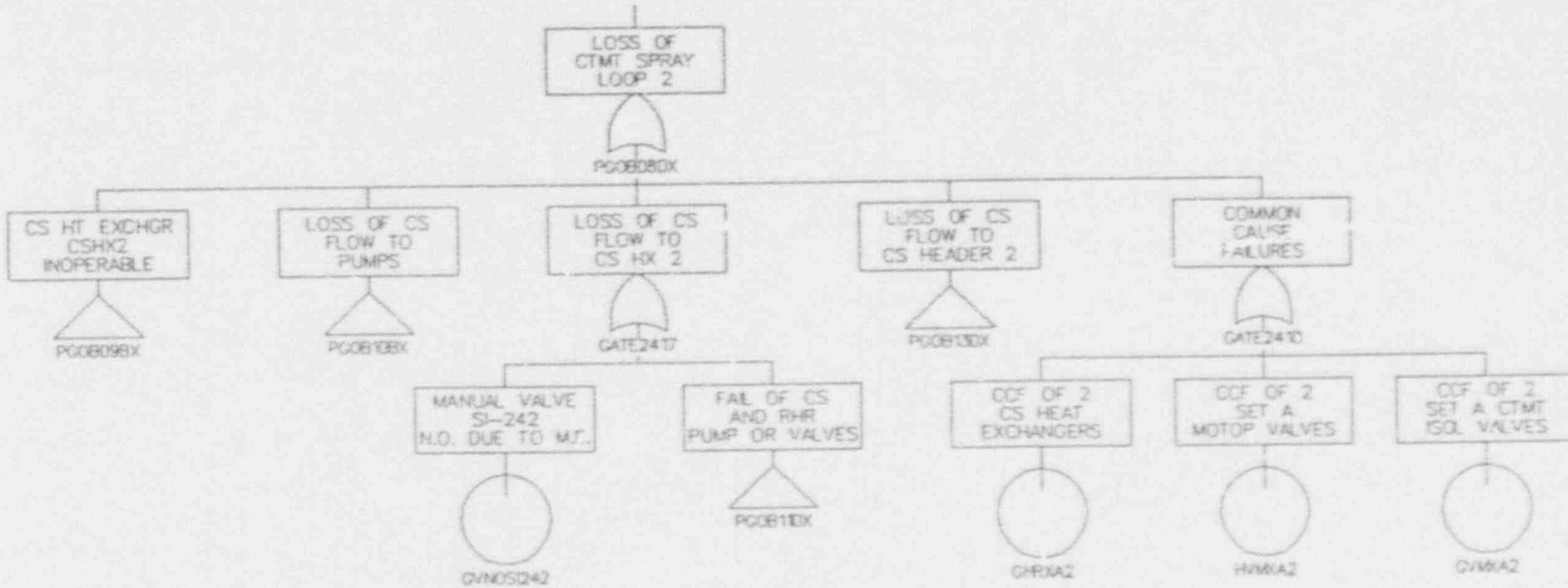

PGOB06DX
FAILURE OF RHR LOOP 1 PUMP OR VALVES
PGOB06DX
MANUAL VALVE SI-123 N.O. DUE TO M.E.
JVNOSI123
RHR PUMP RHRP101 INOPERABLE
PJOB05DX
CHECK VALVE SI-124 FAILS TO OPEN
JVCASI124
MANUAL VALVE SI-125 N.O. DUE TO M.E.
JVNOSI125

PG0807DX

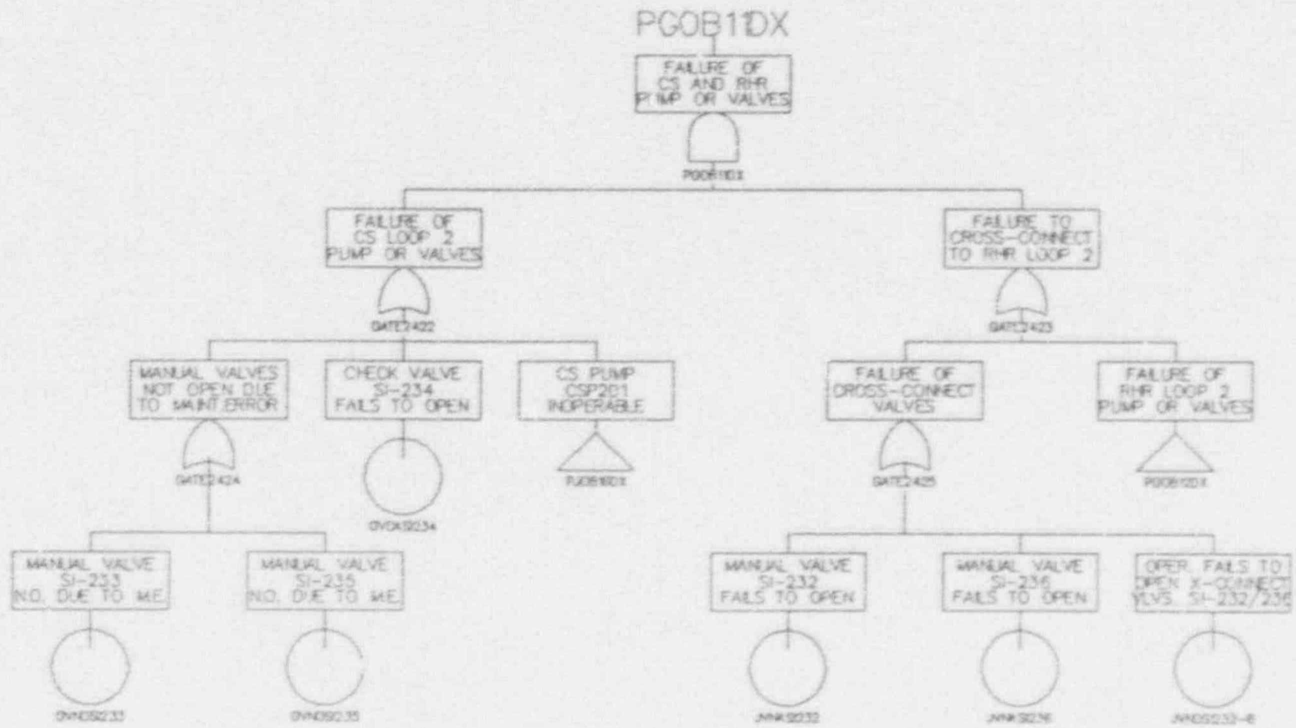


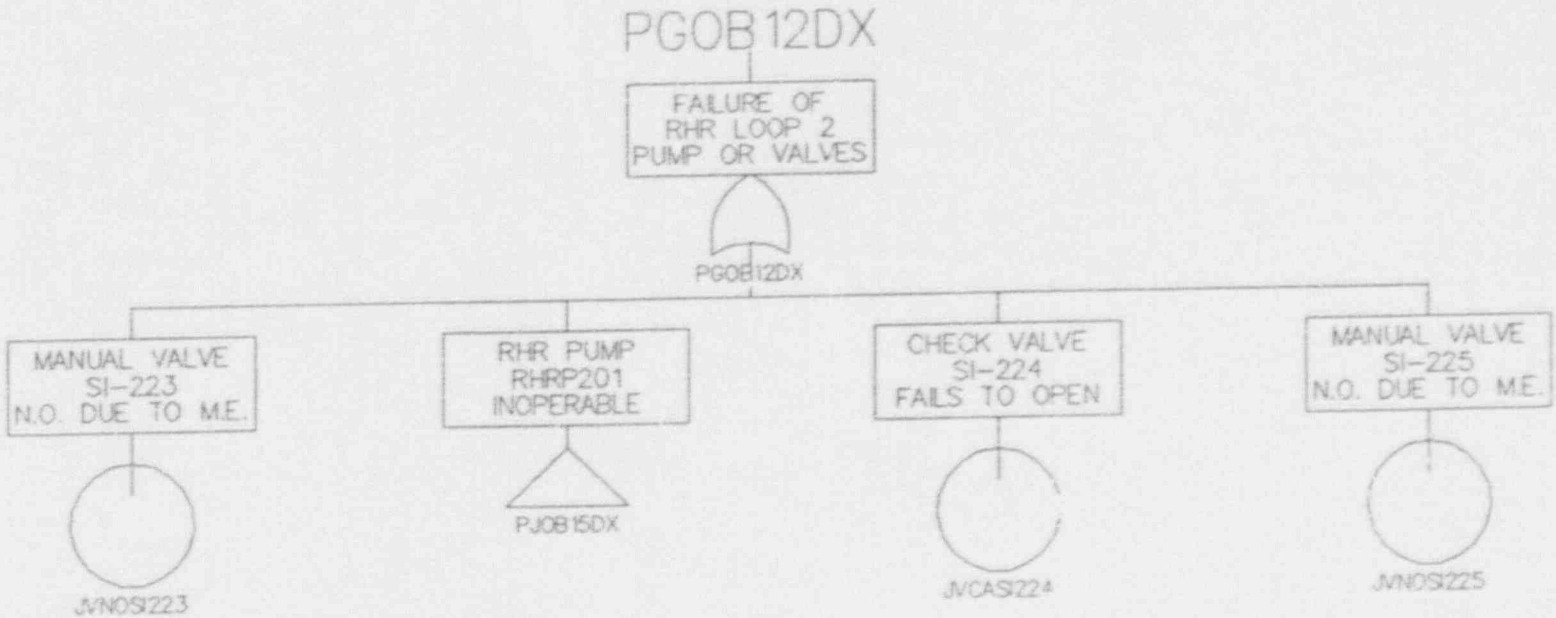
LOSS OF CS FLOW TO CS HEADER 1
MANUAL VALVE SI-143 N.O. DUE TO M.E.
MOTOR VALVE SI-144 FAILS TO OPEN
CHECK VALVE SI-145 FAILS TO OPEN
FLOW DIVERTED VIA MISPOSITIONED VALVE CS-543
MOTOR VALVE SI-144 FAILS TO OPEN
LOP TO COMP ON 480 V MCC BUS C1
NO ACTUATION SIGNAL
CSAS TRAIN A UNAVAILABLE
OPERATOR FAILS TO GENERATE CSAS
CSAS TRAIN A NOT GENERATED
COMMON CAUSE FAILURE OF CSAS

PGOB08DX



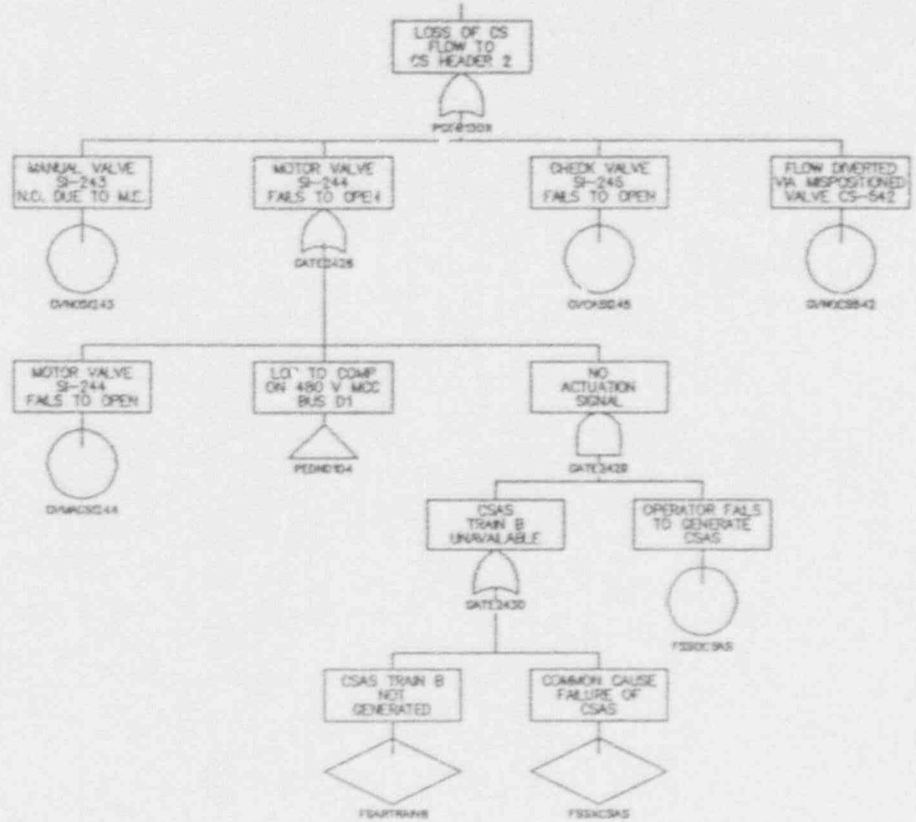
LOSS OF CTMT SPRAY LOOP 2
PGOB08DX
CS HT EXCHGR CSHX2 INOPERABLE
PGOB09BX
LOSS OF CS FLOW TO PUMPS
PGOB10BX
LOSS OF CS FLOW TO CS HX 2
GATE2417
MANUAL VALVE SI-242 N.O. DUE TO M.T.
GVNOSI242
FAIL OF CS AND RHR PUMP OR VALVES
PGOB11DX
LOSS OF CS FLOW TO CS HEADER 2
PGOB13DX
COMMON CAUSE FAILURES
GATE2410
CCF OF 2 CS HEAT EXCHANGERS
GHRXA2
CCF OF 2 SET A MOTOP VALVES
HVMXA2
CCF OF 2 SET A CTMT ISOL VALVES
GVMXA2


PGOB11DX
FAILURE OF CS AND RHR PUMP OR VALVES
FAILURE OF CS LOOP 2 PUMP OR VALVES
GATE2422
FAILURE TO CROSS-CONNECT TO RHR LOOP 2
GATE2423
MANUAL VALVES NOT OPEN DUE TO MAINT.ERROR
GATE2424
CHECK VALVE SI-234 FAILS TO OPEN
CS PUMP CSP201 INOPERABLE
FAILURE OF CROSS-CONNECT VALVES
GATE2425
FAILURE OF RHR LOOP 2 PUMP OR VALVES
MANUAL VALVE SI-233 N.O. DUE TO M.E.
MANUAL VALVE SI-235 N.O. DUE TO M.E.
MANUAL VALVE SI-232 FAILS TO OPEN
MANUAL VALVE SI-236 FAILS TO OPEN
OPER. FAILS TO OPEN X-CONNECT VLVS. SI-232/236

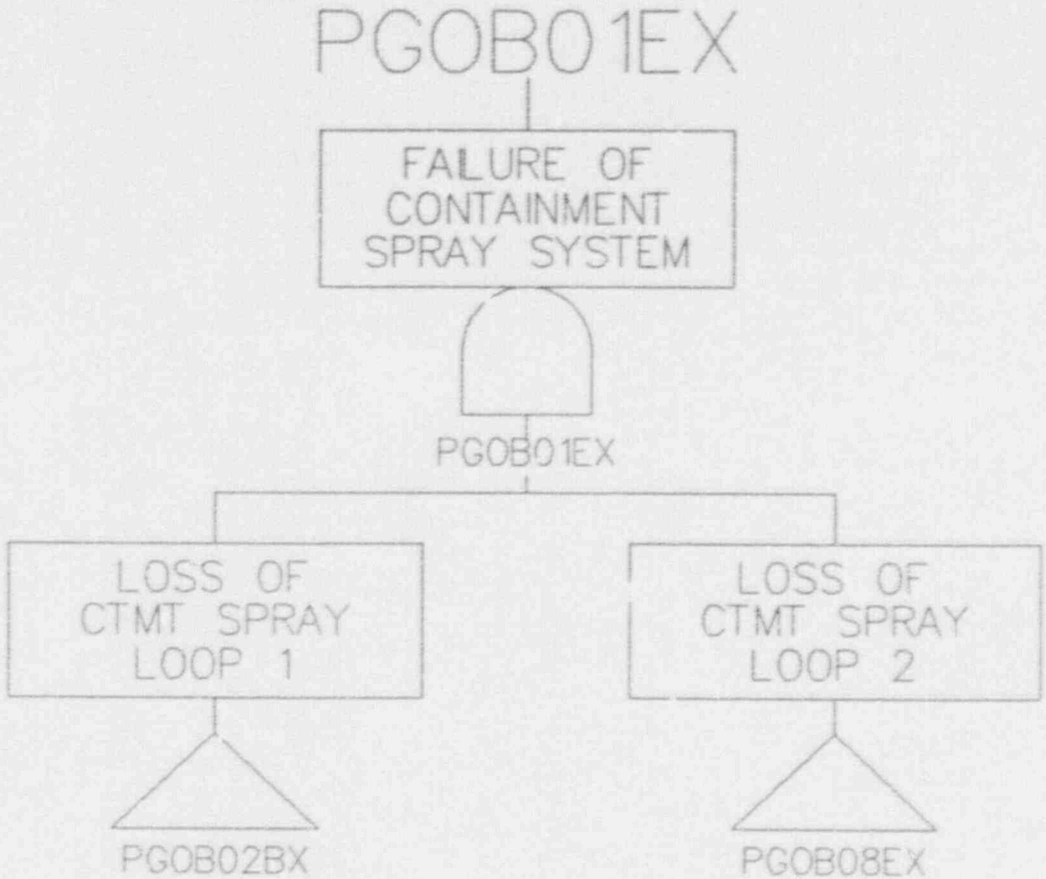


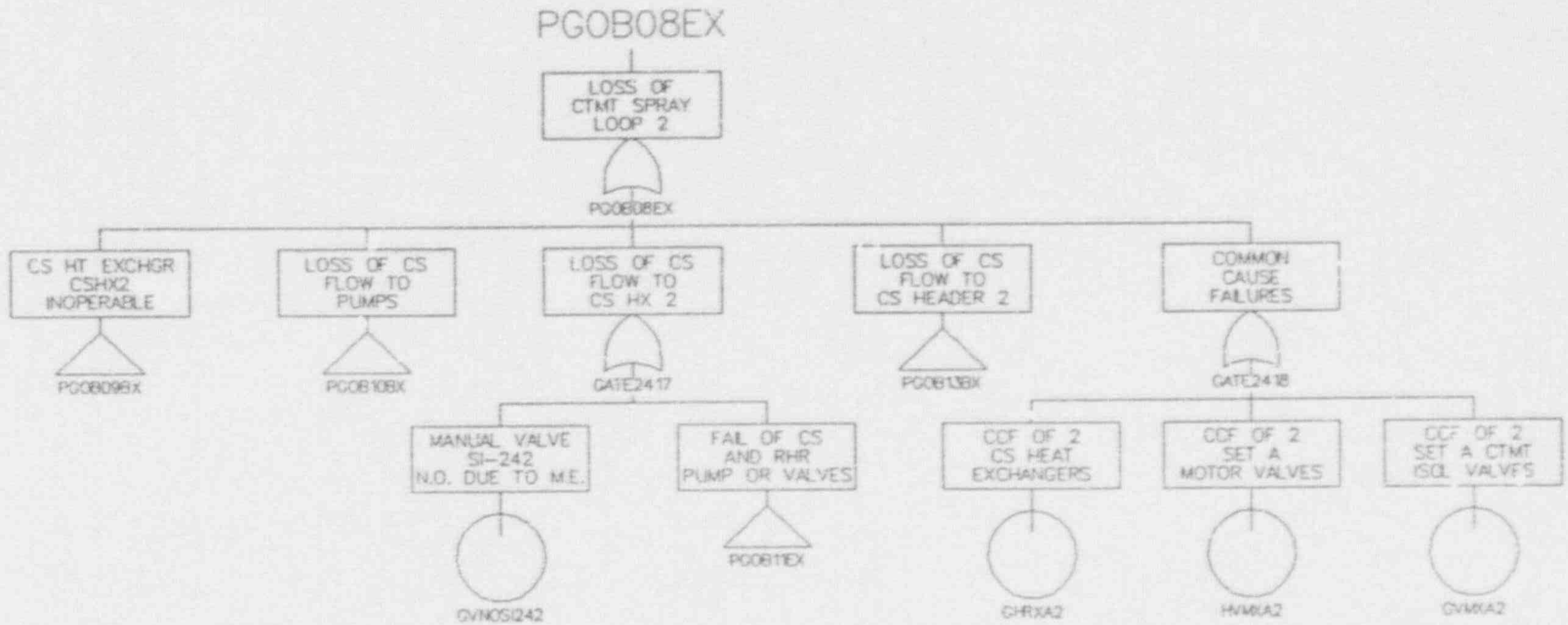
PGOB12DX
FAILURE OF RHR LOOP 2 PUMP OR VALVES
PGOB12DX
MANUAL VALVE SI-223 N.O. DUE TO M.E.
RHR PUMP RHRP201 INOPERABLE
CHECK VALVE SI-224 FAILS TO OPEN
MANUAL VALVE SI-225 N.O. DUE TO M.E.
JVNOSI223
PJOB15DX
JVCASI224
JVNOSI225

PGOB13DX

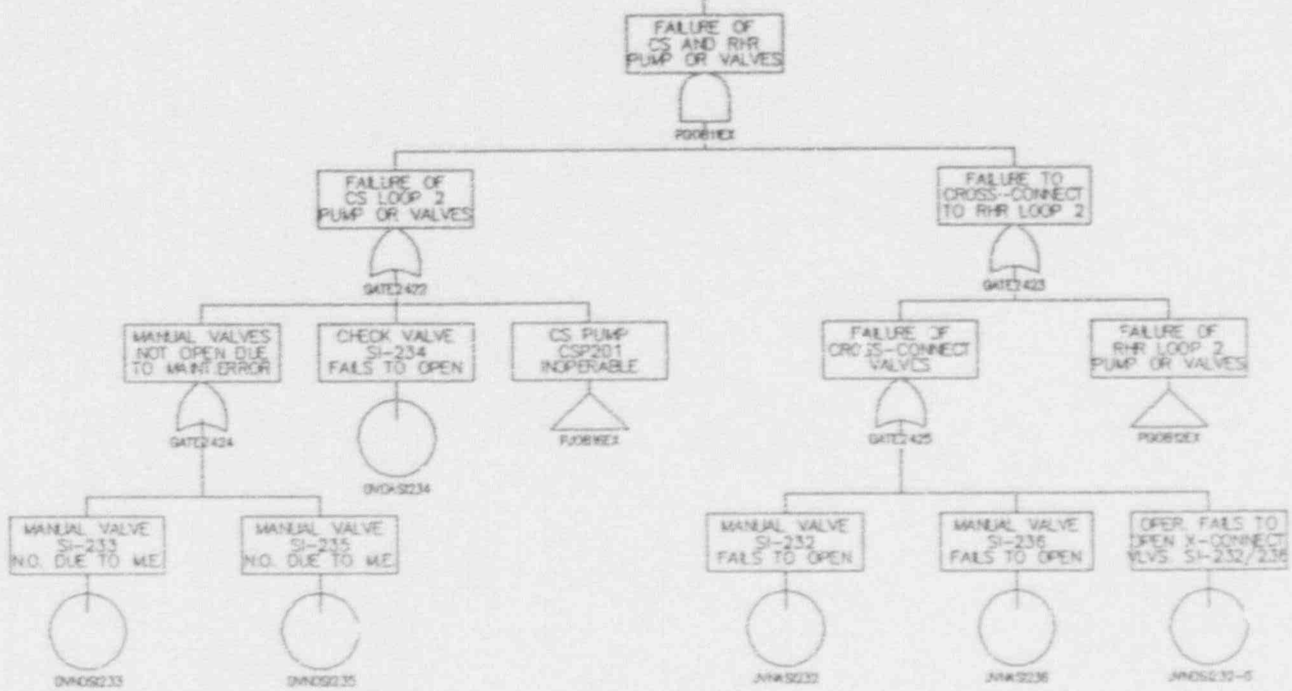


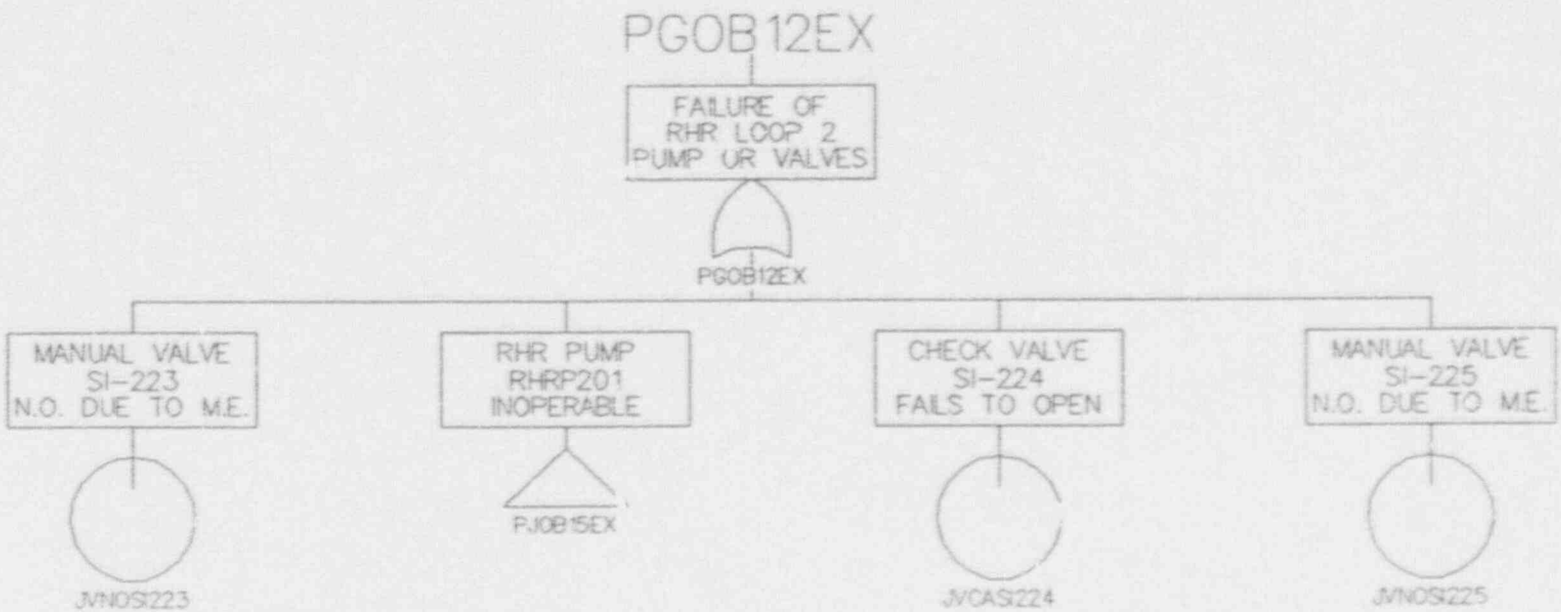
LOSS OF CS FLOW TO CS HEADER 2
MANUAL VALVE SI-243 N.C. DUE TO M.E.
MOTOR VALVE SI-244 FAILS TO OPEN
CHECK VALVE SI-245 FAILS TO OPEN
FLOW DIVERTED VIA MISPOSITIONED VALVE CS-542
MOTOR VALVE SI-244 FAILS TO OPEN
ON 480 V MCC BUS D1
NO ACTUATION SIGNAL
CSAS TRAIN B UNAVAILABLE
OPERATOR FAILS TO GENERATE CSAS
CSAS TRAIN B NOT GENERATED
COMMON CAUSE FAILURE OF CSAS


PGOB01EX
FAILURE OF CONTAINMENT SPRAY SYSTEM
PGOB01EX
LOSS OF CTMT SPRAY LOOP 1
LOSS OF CTMT SPRAY LOOP 2
PGOB02BX
PGOB08EX

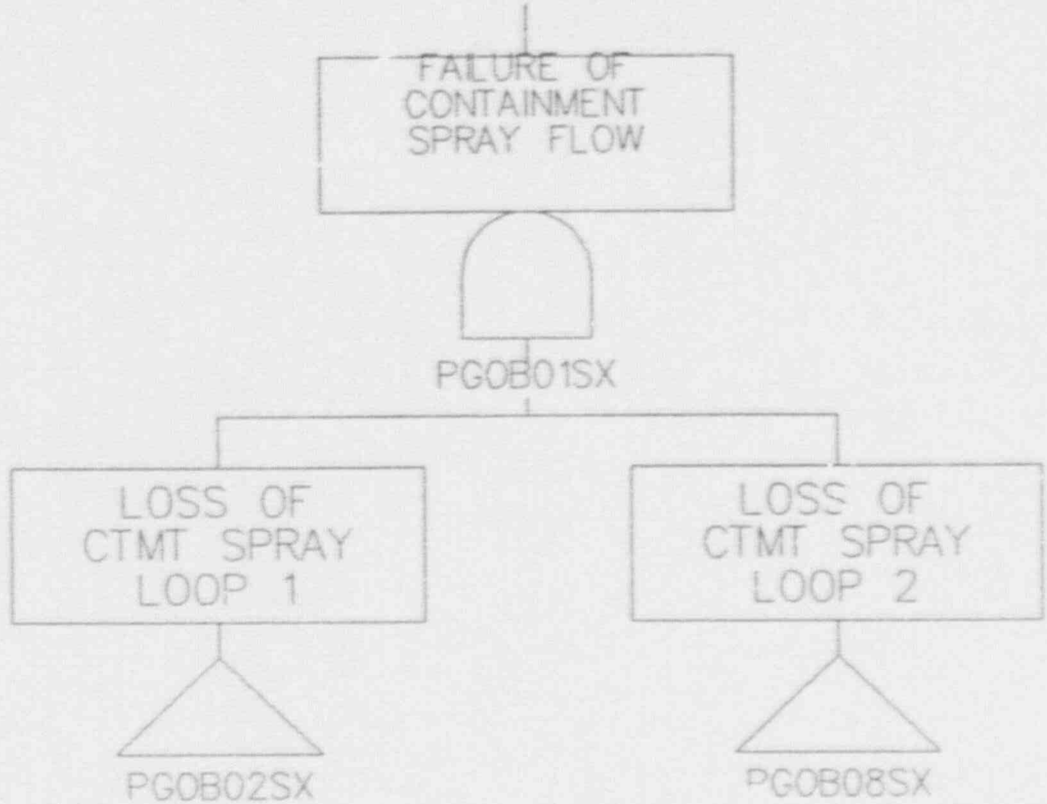

PGOB08EX
LOSS OF CTMT SPRAY LOOP 2
PGOB08EX
CS HT EXCHGR CSHX2 INOPERABLE
PGOB09BX
LOSS OF CS FLOW TO PUMPS
PGOB10BX
LOSS OF CS FLOW TO CS HX 2
GATE2417
MANUAL VALVE SI-242 N.O. DUE TO M.E.
GVNOSI242
FAIL OF CS AND RHR PUMP OR VALVES
PGOB11EX
LOSS OF CS FLOW TO CS HEADER 2
PGOB13BX
COMMON CAUSE FAILURES
GATE2418
CCF OF 2 CS HEAT EXCHANGERS
GHRXA2
CCF OF 2 SET A MOTOR VALVES
HVMXA2
CCF OF 2 SET A CTMT ISOL VALVES
GVMXA2

PGOB11EX

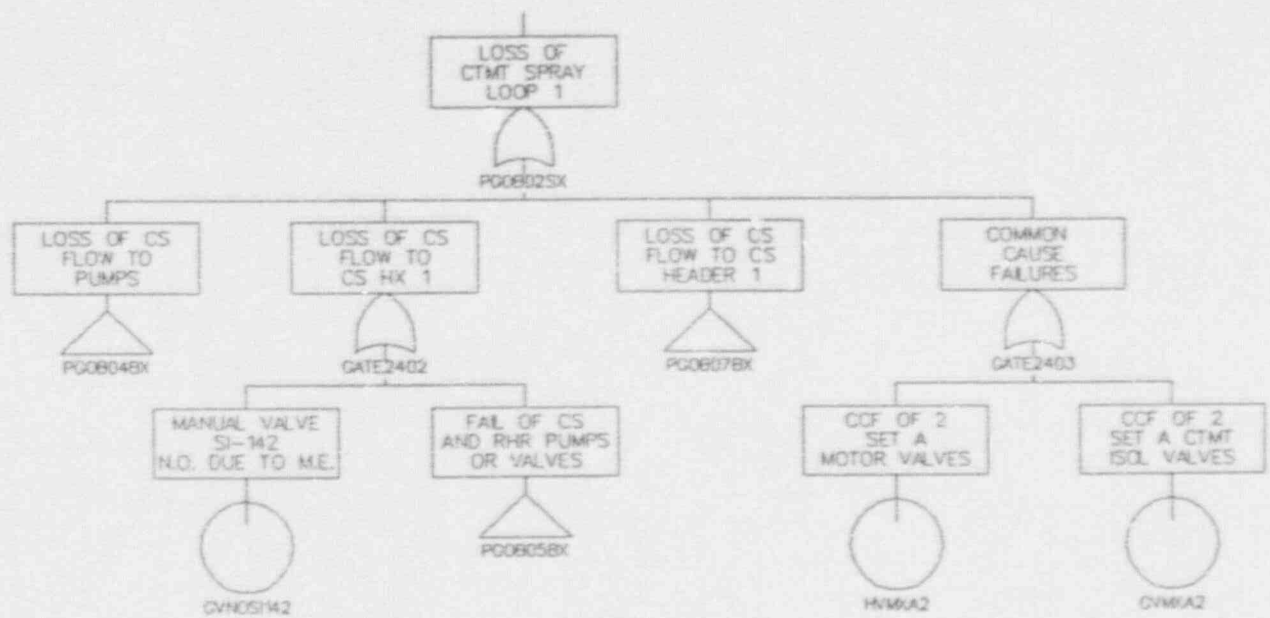

FAILURE OF CS AND RHR PUMP OR VALVES
FAILURE OF CS LOOP 2 PUMP OR VALVES
GATE2422
FAILURE TO CROSS-CONNECT TO RHR LOOP 2
GATE2423
MANUAL VALVES NOT OPEN DUE TO MAINT ERROR
GATE2424
CHECK VALVE SI-234 FAILS TO OPEN
CS PUMP CSP201 INOPERABLE
FAILURE OF CROSS-CONNECT VALVES
GATE2425
FAILURE OF RHR LOOP 2 PUMP OR VALVES
MANUAL VALVE SI-233 N.O. DUE TO M.E.
MANUAL VALVE SI-235 N.O. DUE TO M.E.
MANUAL VALVE SI-232 FAILS TO OPEN
MANUAL VALVE SI-236 FAILS TO OPEN
OPER. FAILS TO OPEN X-CONNECT VLVS. SI-232/236


PGOB12EX
FAILURE OF RHR LOOP 2 PUMP UR VALVES
PGOB12EX
MANUAL VALVE SI-223 N.O. DUE TO M.E.
RHR PUMP RHRP201 INOPERABLE
CHECK VALVE SI-224 FAILS TO OPEN
MANUAL VALVE SI-225 N.O. DUE TO M.E.
JVNOSI223
PJOB15EX
JVCASI224
JVNOSI225

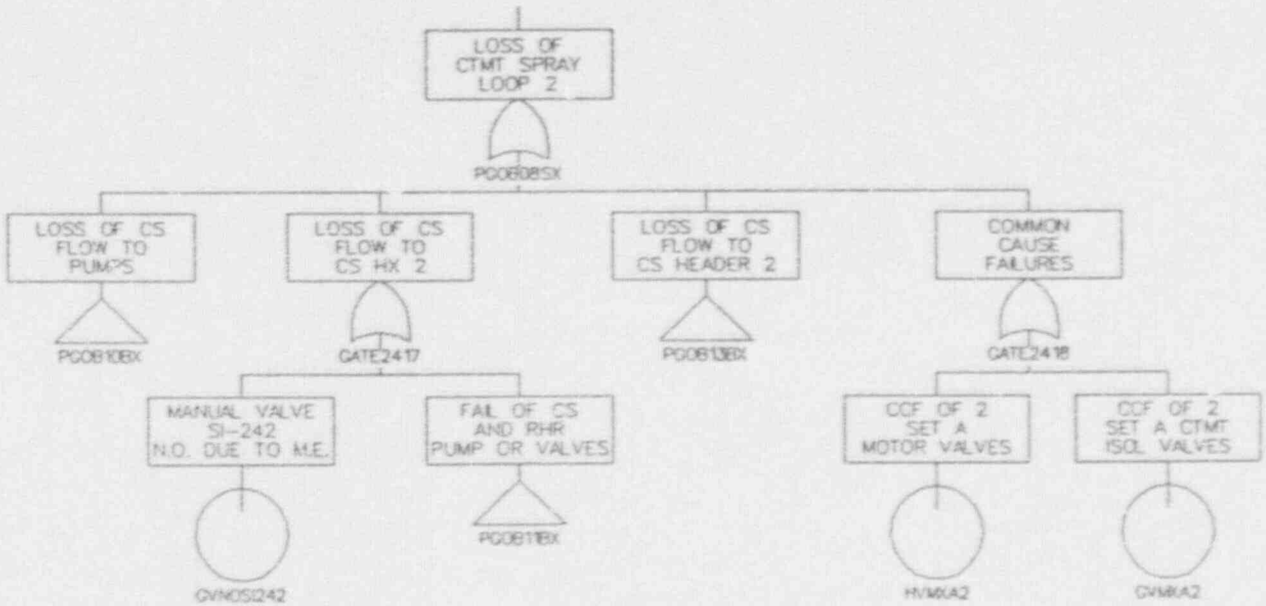
PGOB01SX


FAILURE OF
CONTAINMENT
SPRAY FLOW
PGOB01SX
LOSS OF
CTMT SPRAY
LOOP 1
LOSS OF
CTMT SPRAY
LOOP 2
PGOB02SX
PGOB08SX

PGOB02SX

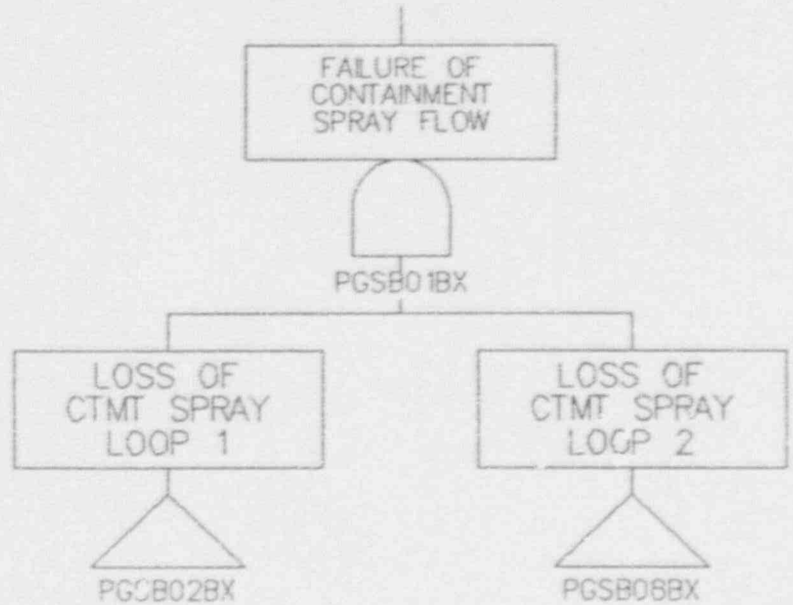

LOSS OF CTMT SPRAY LOOP 1
PGOB02SX
LOSS OF CS FLOW TO PUMPS
PGOB04BX
LOSS OF CS FLOW TO CS HX 1
GATE2402
LOSS OF CS FLOW TO CS HEADER 1
PGOB07BX
COMMON CAUSE FAILURES
GATE2403
MANUAL VALVE SI-142 N.O. DUE TO M.E.
GVNOSI142
FAIL OF CS AND RHR PUMPS OR VALVES
PGOB05BX
CCF OF 2 SET A MOTOR VALVES
HVMXA2
CCF OF 2 SET A CTMT ISOL VALVES
CVMXA2

PGOB08SX

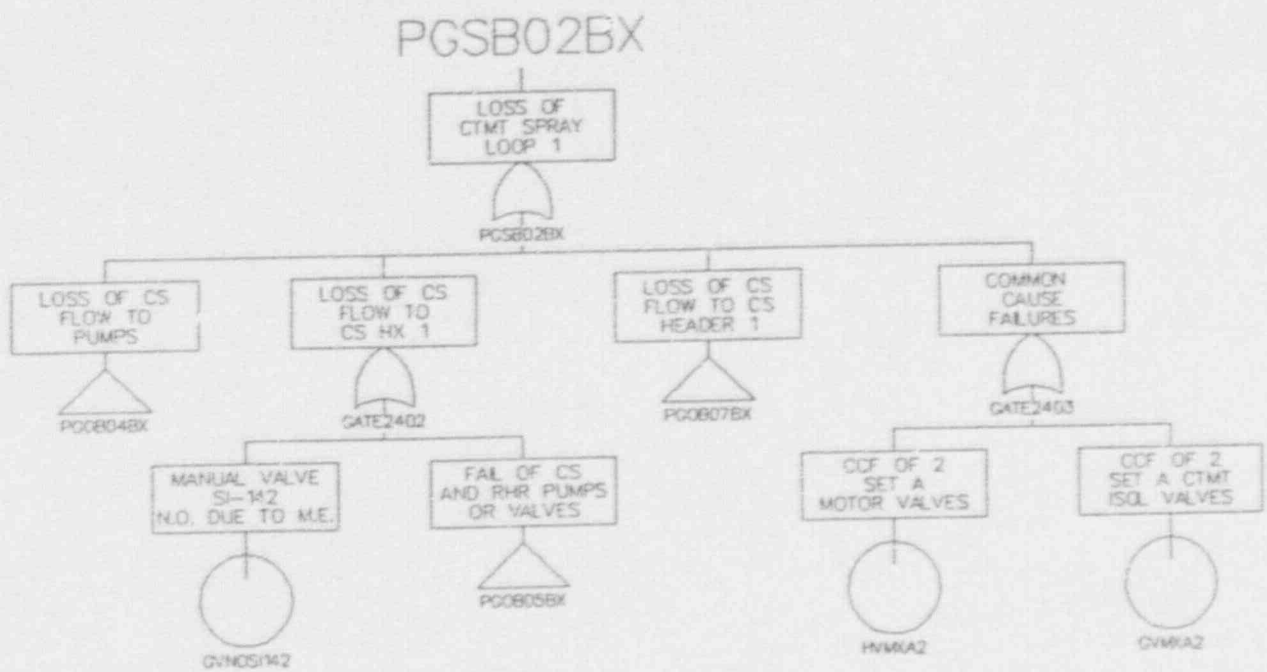


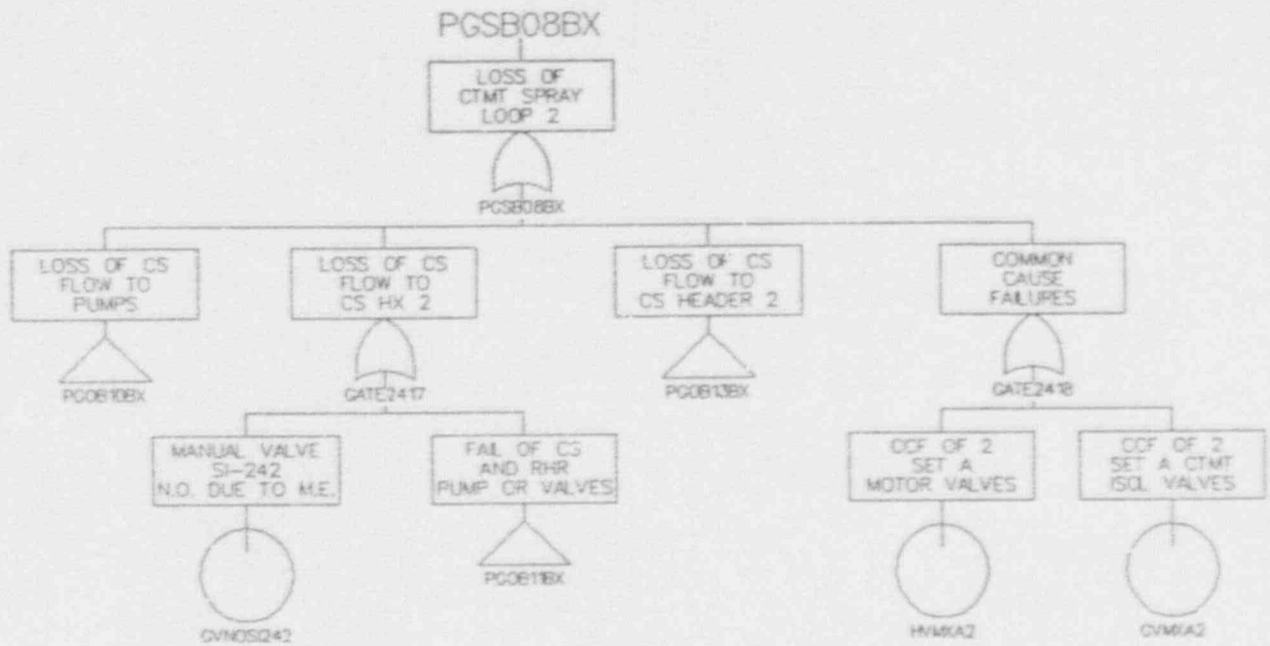
LOSS OF
CTMT SPRAY
LOOP 2
PGOB08SX
LOSS OF CS
FLOW TO
PUMPS
PGOB10BX
LOSS OF CS
FLOW TO
CS HX 2
GATE2417
LOSS OF CS
FLOW TO
CS HEADER 2
PGOB13BX
COMMON
CAUSE
FAILURES
GATE2418
MANUAL VALVE
SI-242
N.O. DUE TO M.E.
GVNOSI242
FAIL OF CS
AND RHR
PUMP OR VALVES
PGOB11BX
CCF OF 2
SET A
MOTOR VALVES
HVMXA2
CCF OF 2
SET A CTMT
ISOL VALVES
GVMXA2

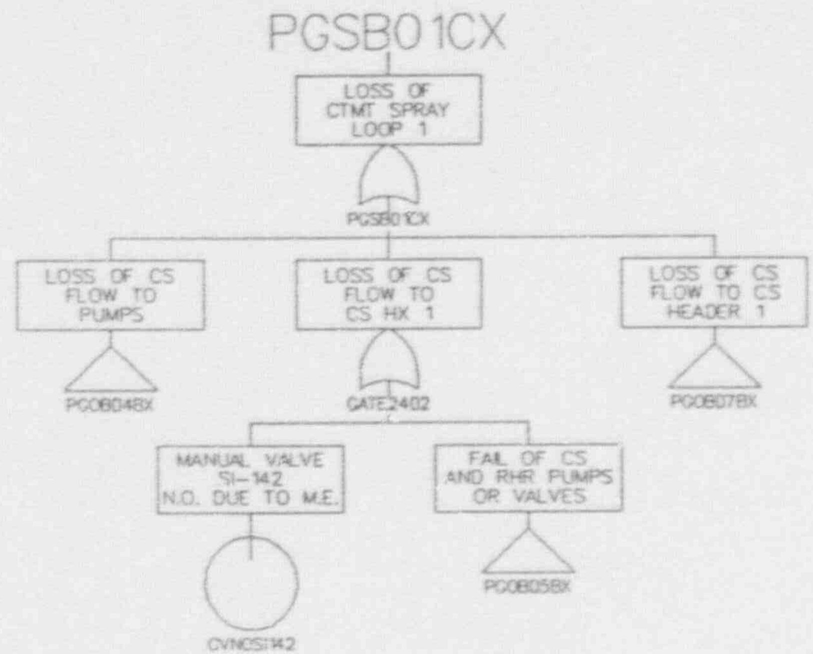
PGSB01BX

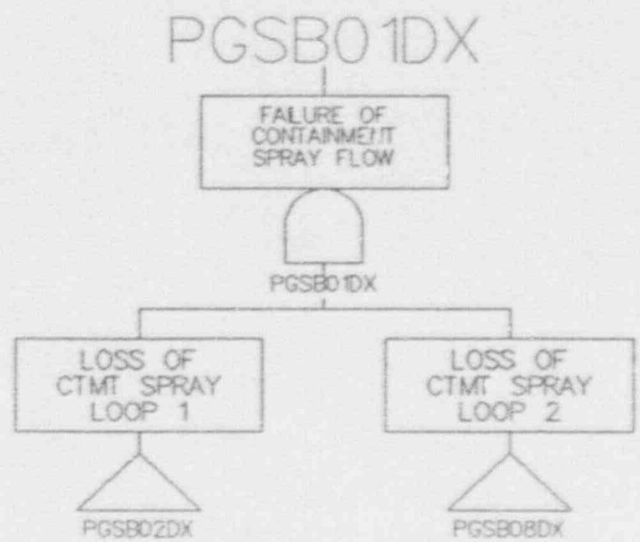


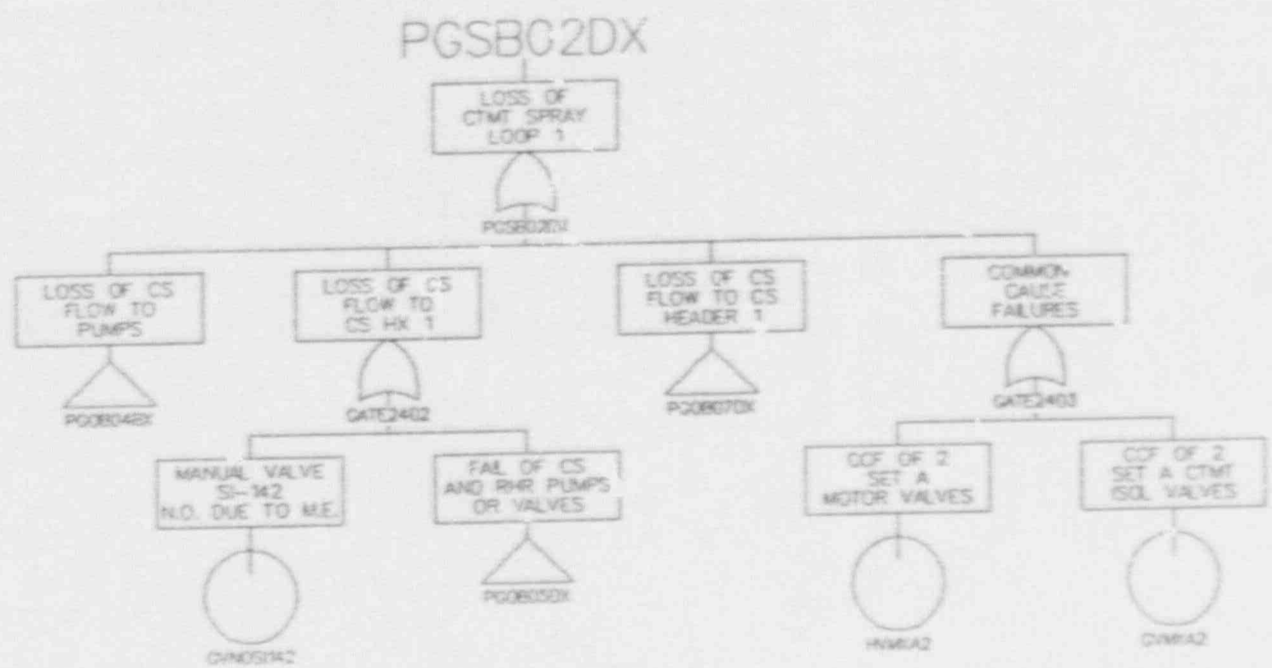
FAILURE OF CONTAINMENT SPRAY FLOW
PGSB01BX
LOSS OF CTMT SPRAY LOOP 1
LOSS OF CTMT SPRAY LOOP 2
PGSB02BX
PGSB06BX

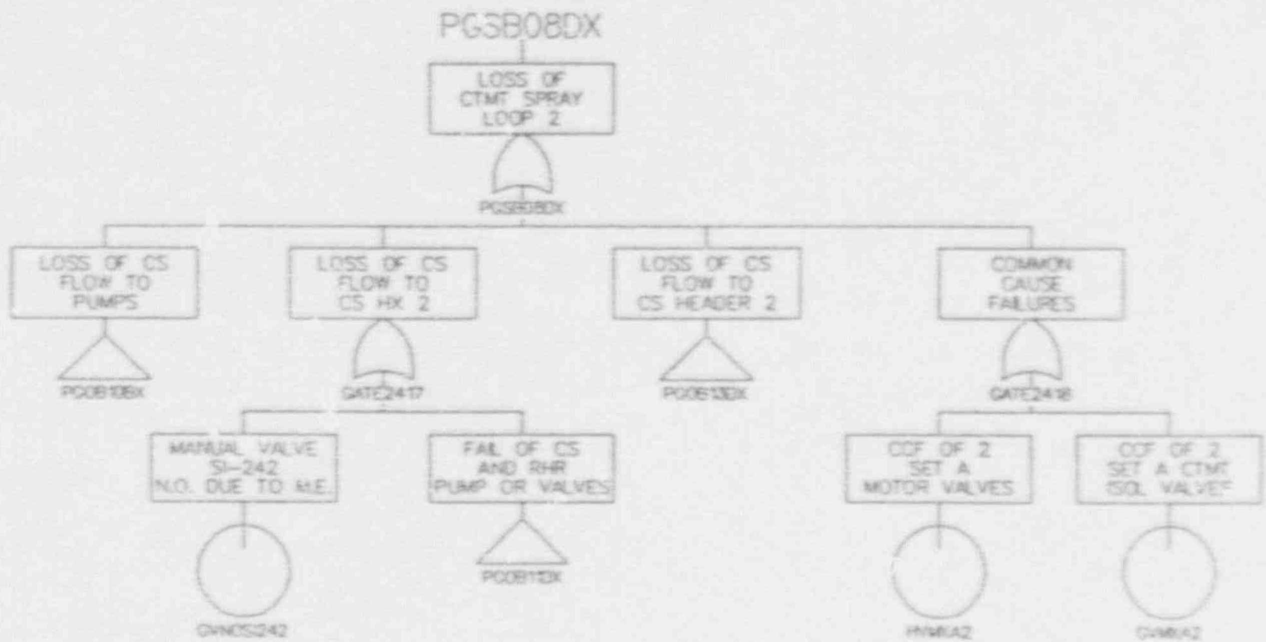

PGSB02BX
LOSS OF CTMT SPRAY LOOP 1
PGSB02BX
LOSS OF CS FLOW TO PUMPS
PGOB04BX
LOSS OF CS FLOW TO CS HX 1
GATE2402
MANUAL VALVE SI-142 N.O. DUE TO M.E.
GVNOSI142
FAIL OF CS AND RHR PUMPS OR VALVES
PGOB05BX
LOSS OF CS FLOW TO CS HEADER 1
PGOB07BX
COMMON CAUSE FAILURES
GATE2403
CCF OF 2 SET A MOTOR VALVES
HVMXA2
CCF OF 2 SET A CTMT ISOL VALVES
GVMXA2


PGSB08BX
LOSS OF CTMT SPRAY LOOP 2
PGSB08BX
LOSS OF CS FLOW TO PUMPS
PGOB10BX
LOSS OF CS FLOW TO CS HX 2
GATE2417
MANUAL VALVE SI-242 N.O. DUE TO M.E.
GVNOSI242
FAIL OF CS AND RHR PUMP OR VALVES
PGOB11BX
LOSS OF CS FLOW TO CS HEADER 2
PGOB13BX
COMMON CAUSE FAILURES
GATE2418
CCF OF 2 SET A MOTOR VALVES
HVMXA2
CCF OF 2 SET A CTMT ISOL VALVES
CVMXA2

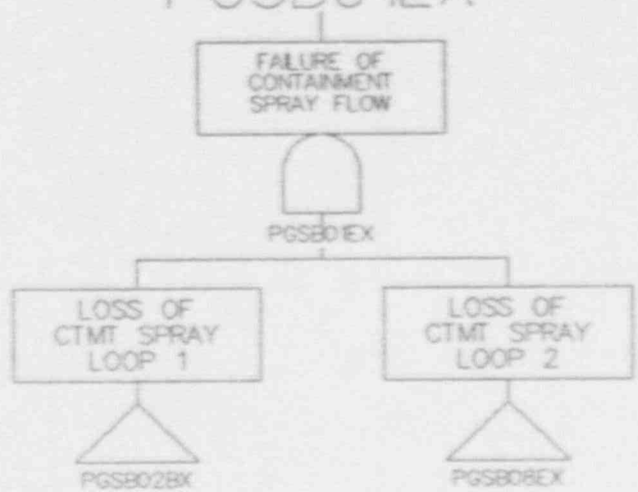

PGSB01CX
LOSS OF CTMT SPRAY LOOP 1
PGSB01CX
LOSS OF CS FLOW TO PUMPS
PGOB04BX
LOSS OF CS FLOW TO CS HX 1
GATE2402
LOSS OF CS FLOW TO CS HEADER 1
PGOB07BX
MANUAL VALVE SI-142 N.O. DUE TO M.E.
CVNOSI142
FAIL OF CS AND RHR PUMPS OR VALVES
PGOB05BX

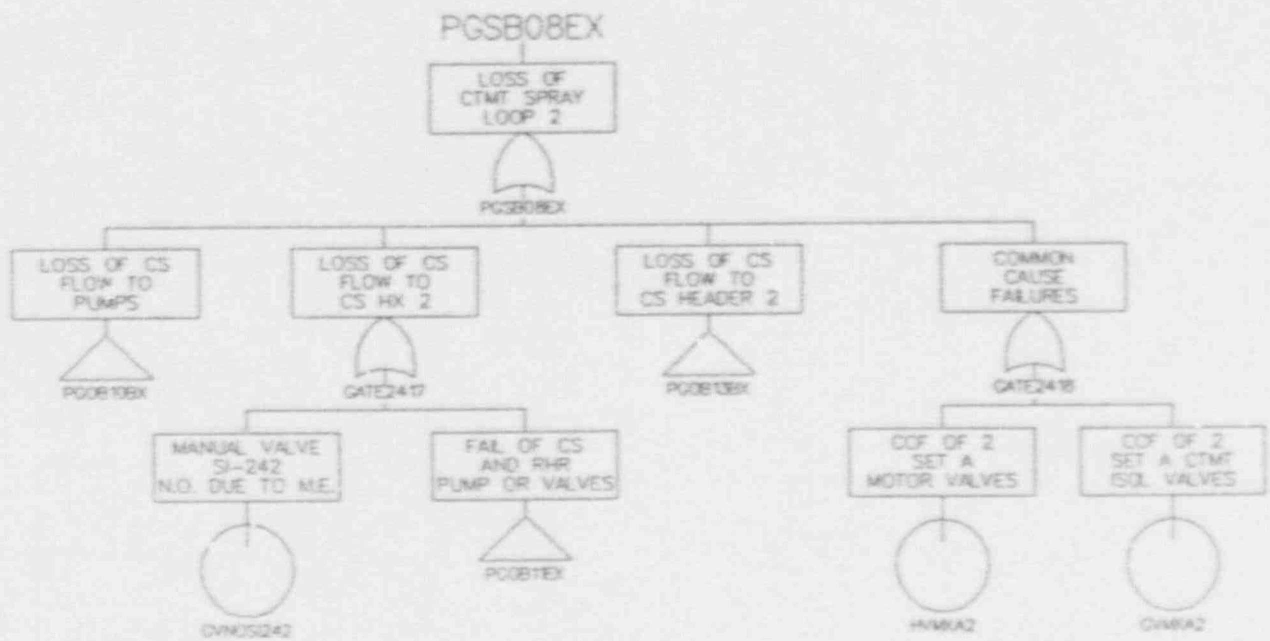

PGSB01DX
FAILURE OF CONTAINMENT SPRAY FLOW
PGSB01DX
LOSS OF CTMT SPRAY LOOP 1
LOSS OF CTMT SPRAY LOOP 2
PGSB02DX
PGSB08DX


PGSBC2DX
LOSS OF CTMT SPRAY LOOP 1
LOSS OF CS FLOW TO PUMPS
LOSS OF CS FLOW TO CS HX 1
LOSS OF CS FLOW TO CS HEADER 1
COMMON CAUSE FAILURES
GATE2402
GATE2403
MANUAL VALVE SI-142 N.O. DUE TO M.E.
FAIL OF CS AND RHR PUMPS OR VALVES
CCF OF 2 SET A MOTOR VALVES
CCF OF 2 SET A CTMT ISOL VALVES

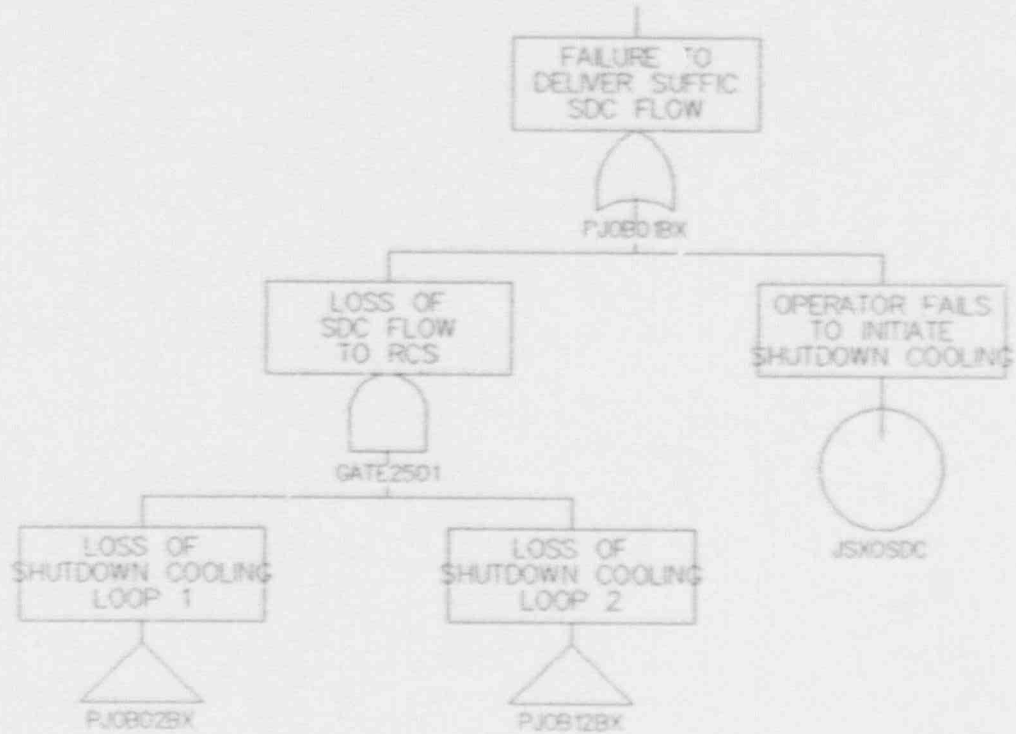

PG3B08DX
LOSS OF CTMT SPRAY LOOP 2
LOSS OF CS FLOW TO PUMPS
LOSS OF CS FLOW TO CS HX 2
GATE2417
LOSS OF CS FLOW TO CS HEADER 2
COMMON CAUSE FAILURES
GATE2418
MANUAL VALVE SI-242 N.O. DUE TO M.E.
GVNOS242
FAIL OF CS AND RHR PUMP OR VALVES
CCF OF 2 SET A MOTOR VALVES
CCF OF 2 SET A CTMT ISOL VALVES

PGSB01EX


FAILURE OF CONTAINMENT SPRAY FLOW
PGSB01EX
LOSS OF CTMT SPRAY LOOP 1
LOSS OF CTMT SPRAY LOOP 2
PGSB02BX
PGSB08EX

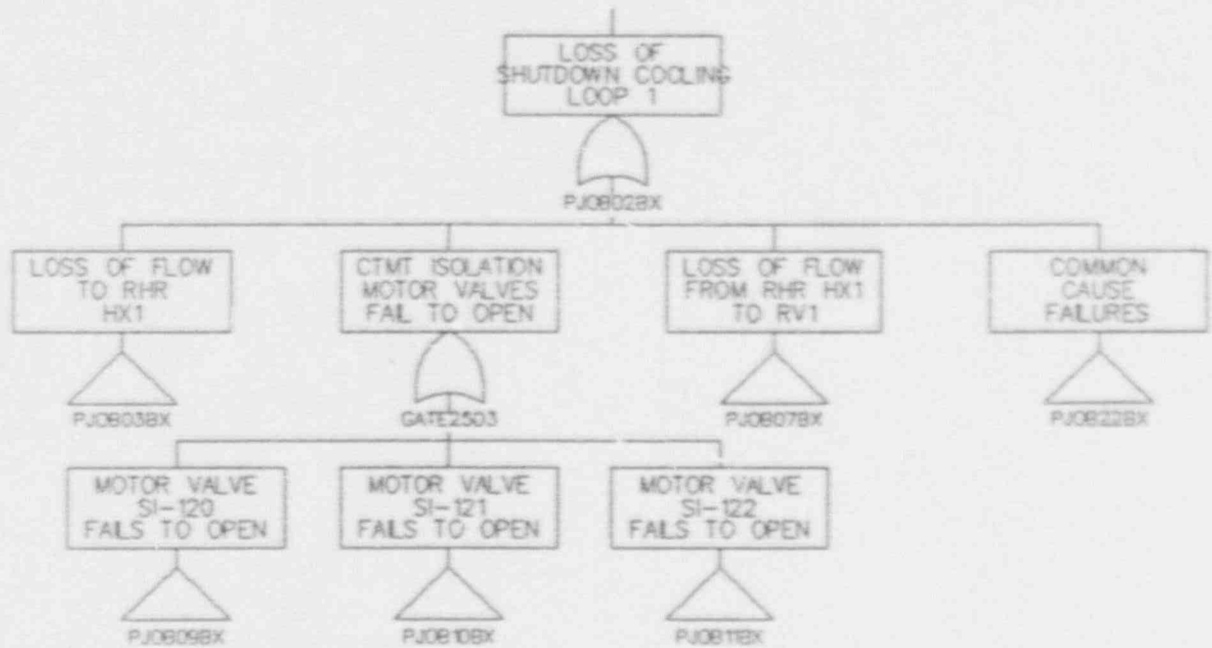

PGSB08EX
LOSS OF CTMT SPRAY LOOP 2
PGSB08EX
LOSS OF CS FLOW TO PUMPS
PGOB10BX
LOSS OF CS FLOW TO CS HX 2
GATE2417
MANUAL VALVE SI-242 N.O. DUE TO M.E.
CVNOSI242
FAIL OF CS AND RHR PUMP OR VALVES
PGOB11EX
LOSS OF CS FLOW TO CS HEADER 2
PGOB13BX
COMMON CAUSE FAILURES
GATE2418
CCF OF 2 SET A MOTOR VALVES
HVMKA2
CCF OF 2 SET A CTMT ISOL VALVES
CVMKA2

(PJOB01BX)

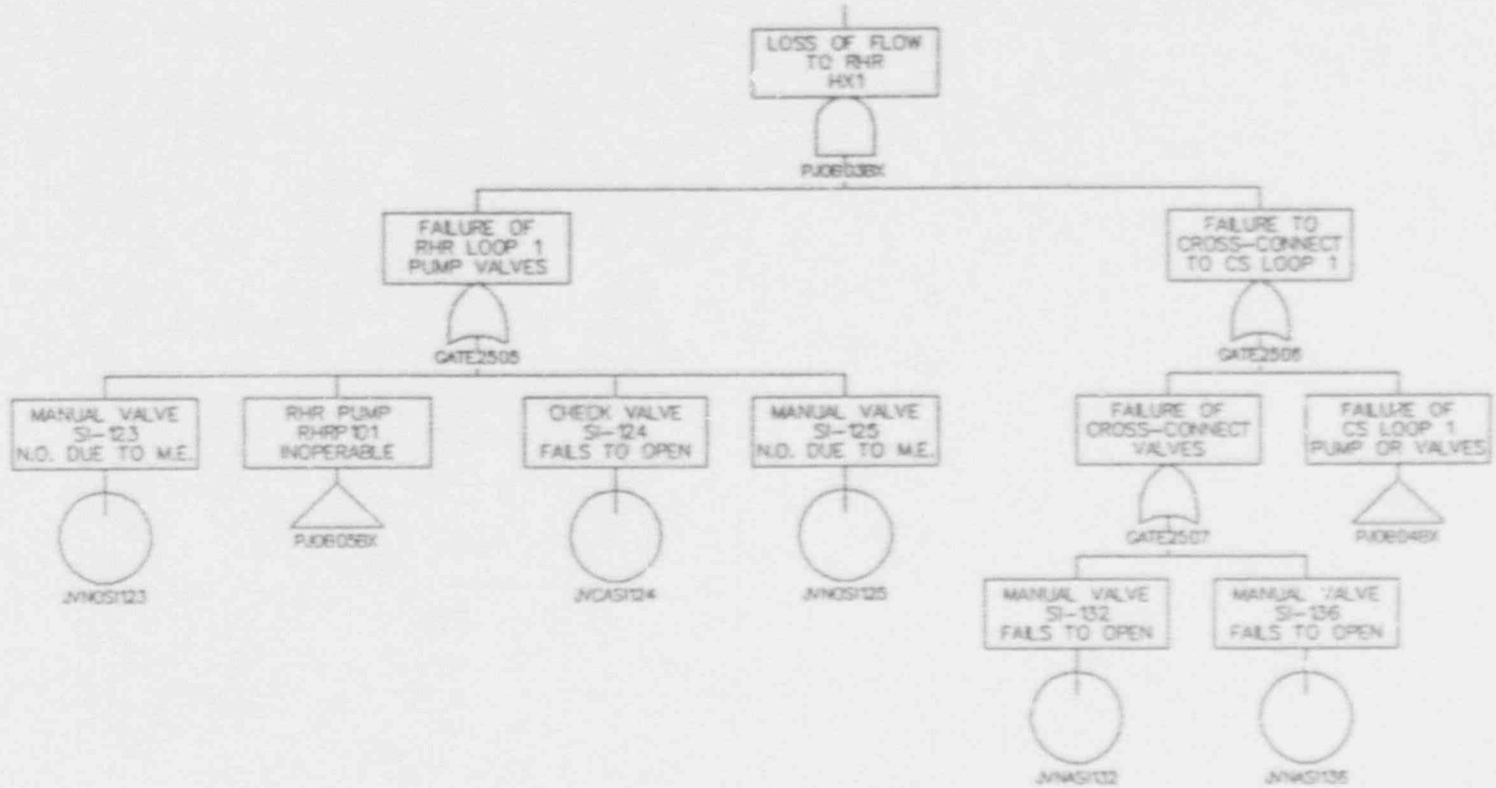


FAILURE TO
DELIVER SUFFIC
SDC FLOW
PJOB01BX
LOSS OF
SDC FLOW
TO RCS
GATE2501
OPERATOR FAILS
TO INITIATE
SHUTDOWN COOLING
JSXOSDC
LOSS OF
SHUTDOWN COOLING
LOOP 1
PJOB02BX
LOSS OF
SHUTDOWN COOLING
LOOP 2
PJOB12BX

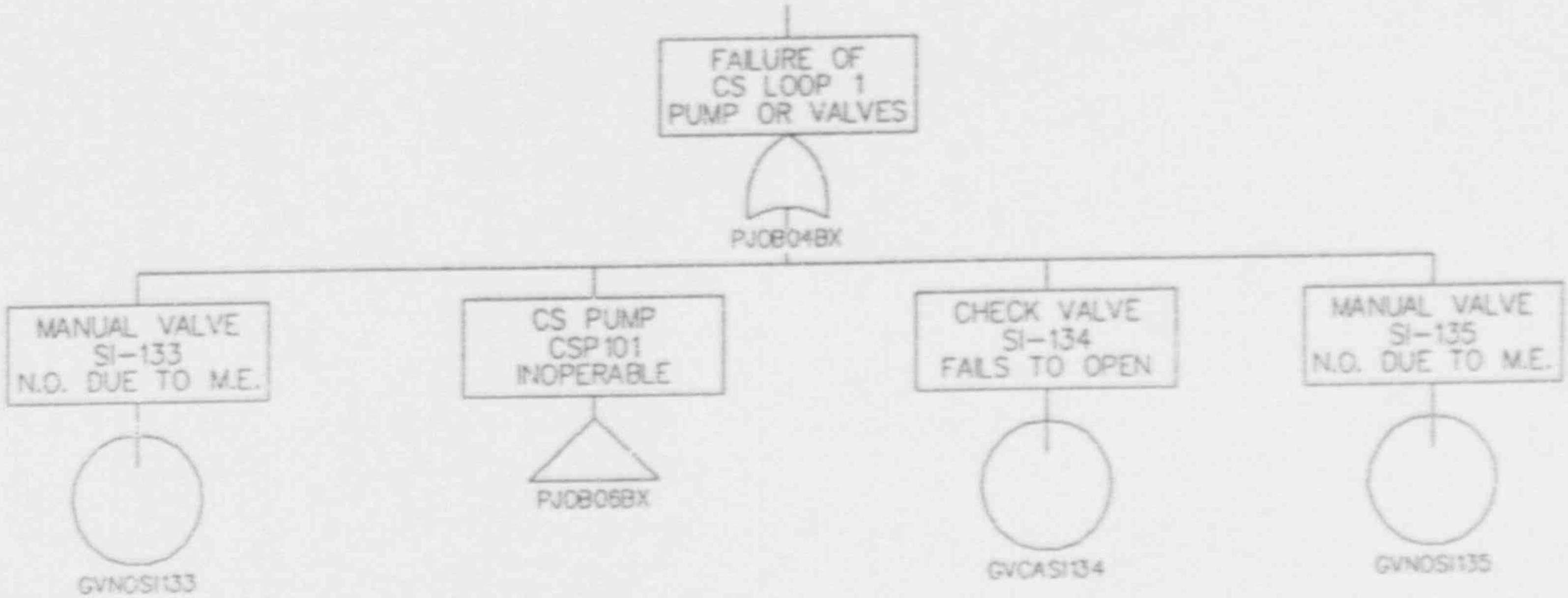
(PJOB02BX)


LOSS OF SHUTDOWN COOLING LOOP 1
PJOB02BX
LOSS OF FLOW TO RHR HX1
PJOB03BX
CTMT ISOLATION MOTOR VALVES FAIL TO OPEN
GATE2503
LOSS OF FLOW FROM RHR HX1 TO RV1
PJOB07BX
COMMON CAUSE FAILURES
PJOB22BX
MOTOR VALVE SI-120 FAILS TO OPEN
PJOB09BX
MOTOR VALVE SI-121 FAILS TO OPEN
PJOB10BX
MOTOR VALVE SI-122 FAILS TO OPEN
PJOB11BX

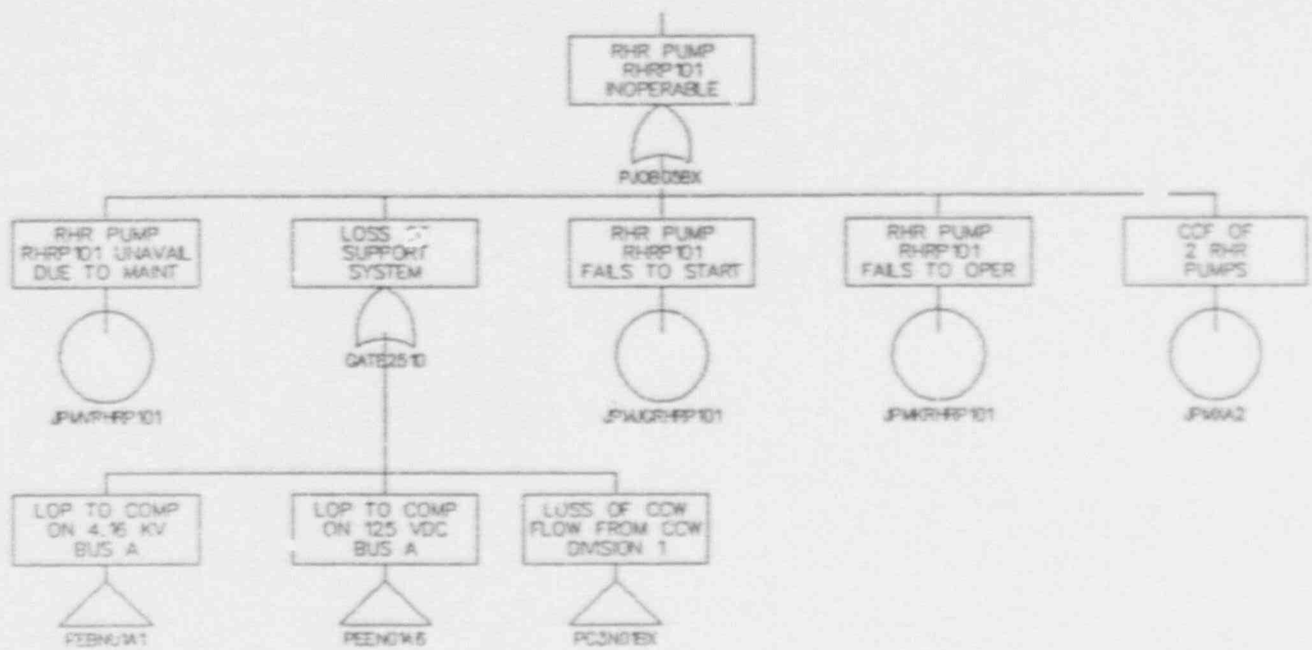
(PJOB03BX)


LOSS OF FLOW TO RHR HX1
PJOB03BX
FAILURE OF RHR LOOP 1 PUMP VALVES
GATE2505
FAILURE TO CROSS-CONNECT TO CS LOOP 1
GATE2506
MANUAL VALVE SI-123 N.O. DUE TO M.E.
JVNOSI123
RHR PUMP RHRP101 INOPERABLE
PJOB05BX
CHECK VALVE SI-124 FAILS TO OPEN
JVCASI124
MANUAL VALVE SI-125 N.O. DUE TO M.E.
JVNOSI125
FAILURE OF CROSS-CONNECT VALVES
GATE2507
FAILURE OF CS LOOP 1 PUMP OR VALVES
PJOB04BX
MANUAL VALVE SI-132 FAILS TO OPEN
JVNASI132
MANUAL VALVE SI-136 FAILS TO OPEN
JVNASI136

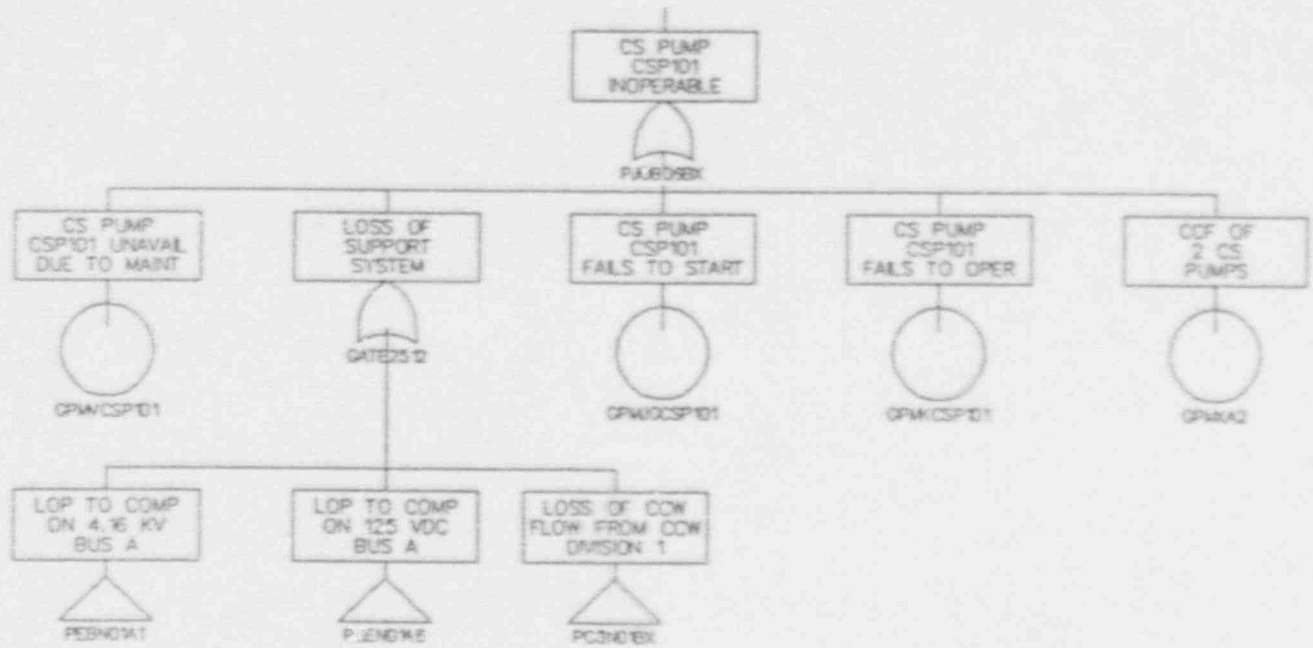
(PJOB04BX)


FAILURE OF CS LOOP 1 PUMP OR VALVES
PJOB04BX
MANUAL VALVE SI-133 N.O. DUE TO M.E.
GVNOSI133
CS PUMP CSP101 INOPERABLE
PJOB06BX
CHECK VALVE SI-134 FAILS TO OPEN
GVCASI134
MANUAL VALVE SI-135 N.O. DUE TO M.E.
GVNOSI135

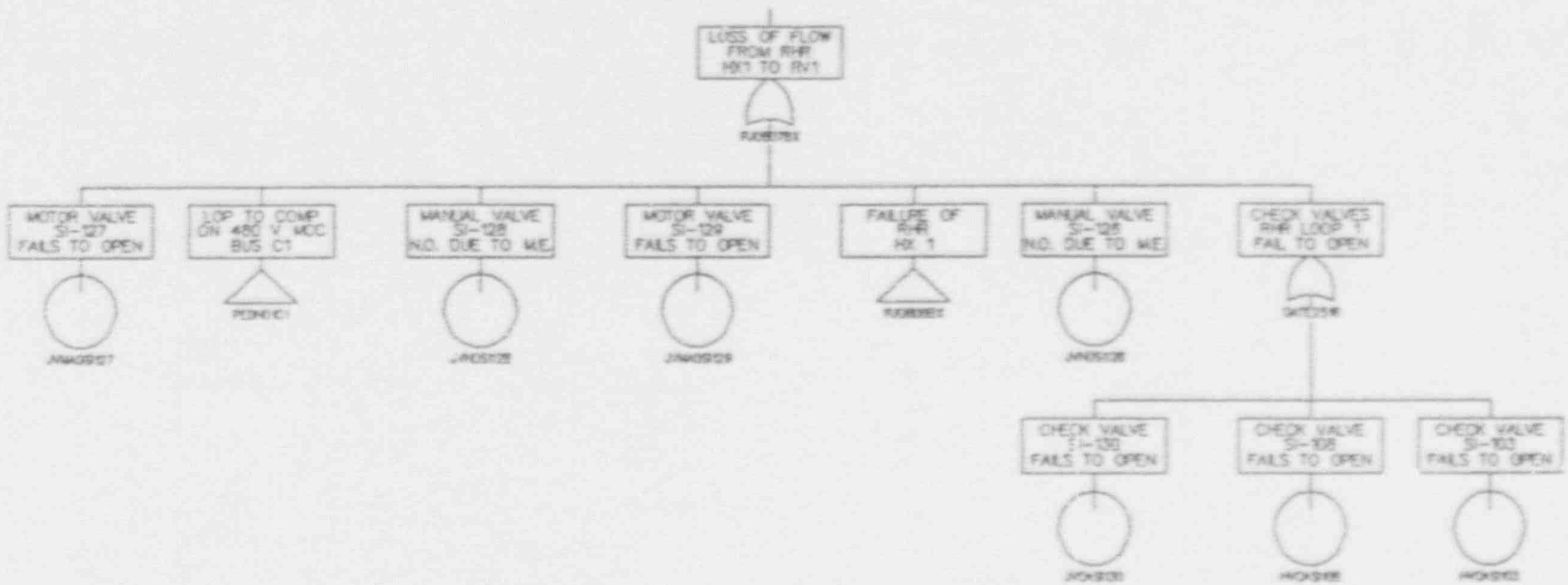
(PJOB05BX)


RHR PUMP
RHRP101
INOPERABLE
PJOB05BX
RHR PUMP
RHRP101 UNAVAIL
DUE TO MAINT
JPIWPHRP101
SUPPORT
SYSTEM
GATE2510
RHR PUMP
RHRP101
FAILS TO START
RHR PUMP
RHRP101
FAILS TO OPER
CCF OF
2 RHR
PUMPS
LOP TO COMP
ON 4.16 KV
BUS A
LOP TO COMP
ON 125 VDC
BUS A
FLOW FROM CCW
DIVISION 1

(PJOB06BX)

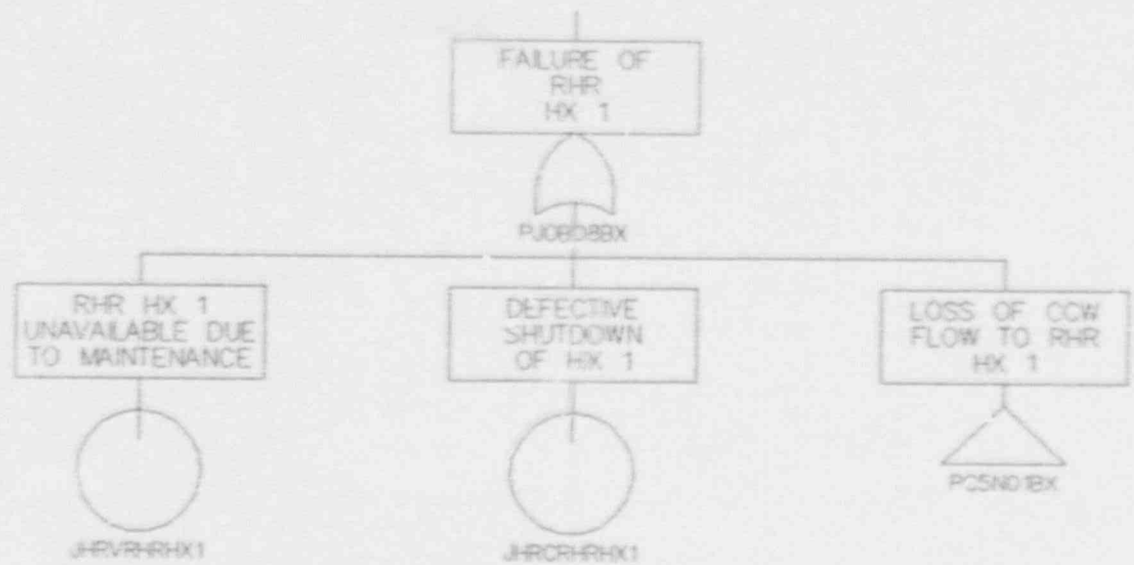

CS PUMP CSP101 INOPERABLE
CS PUMP CSP101 UNAVAIL DUE TO MAINT
LOSS OF SUPPORT SYSTEM
CS PUMP CSP101 FAILS TO START
CS PUMP CSP101 FAILS TO OPER
CCF OF 2 CS PUMPS
GATE2512
LOP TO COMP ON 4.16 KV BUS A
LOP TO COMP ON 125 VDC BUS A
LOSS OF CCW FLOW FROM CCW DIVISION 1
PC3N01BX

(PJOB07BX)

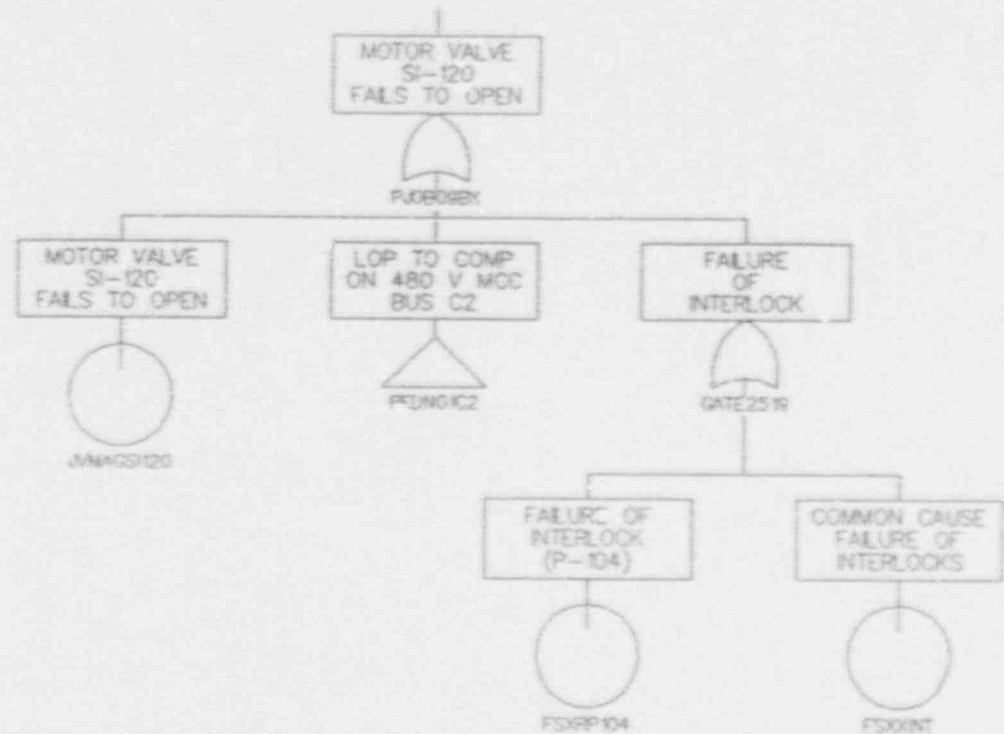


LOSS OF FLOW FROM RHR HX1 TO RV1
MOTOR VALVE SI-127 FAILS TO OPEN
LOP TO COMP ON 480 V MCC BUS C1
MANUAL VALVE SI-128 N.O. DUE TO M.E.
MOTOR VALVE SI-129 FAILS TO OPEN
FAILURE OF RHR HX 1
MANUAL VALVE SI-126 N.O. DUE TO M.E.
CHECK VALVES RHR LOOP 1 FAIL TO OPEN
CHECK VALVE SI-130 FAILS TO OPEN
CHECK VALVE SI-108 FAILS TO OPEN
CHECK VALVE SI-103 FAILS TO OPEN

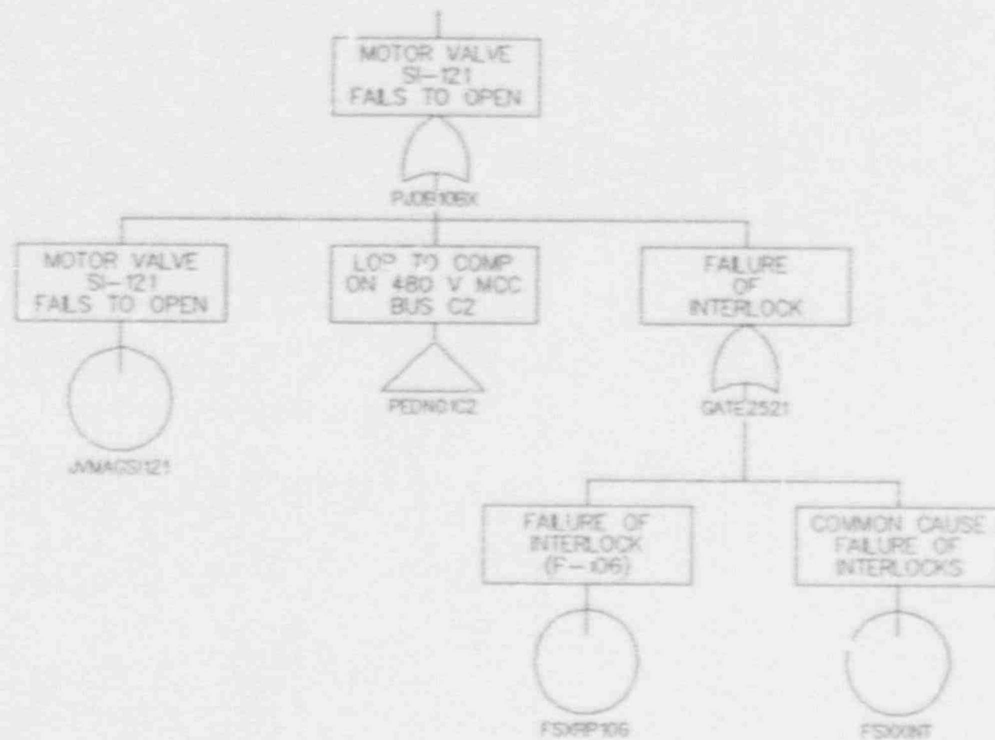
(PJOB08BX)


FAILURE OF RHR HX 1
PJOB08BX
RHR HX 1 UNAVAILABLE DUE TO MAINTENANCE
JHRVRHRHX1
DEFECTIVE SHUTDOWN OF HX 1
JHRCRHRHX1
LOSS OF CCW FLOW TO RHR HX 1
PCSN01BX

(PJOB09BX)

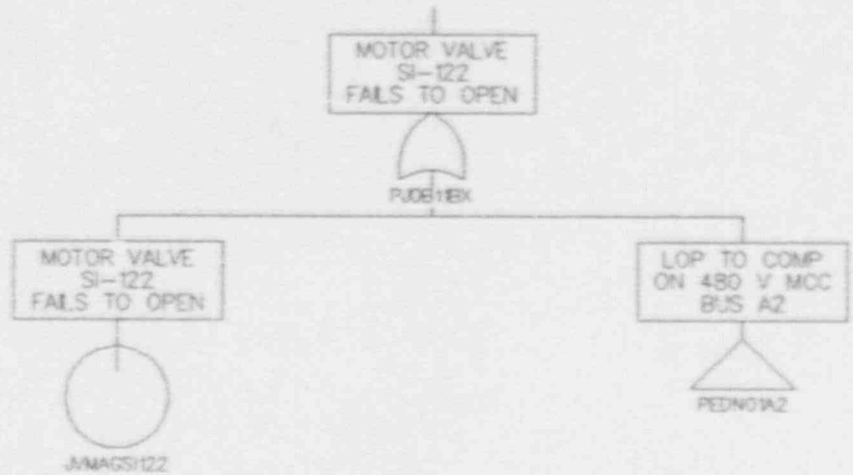

MOTOR VALVE SI-120 FAILS TO OPEN
PJOB09BX
MOTOR VALVE SI-120 FAILS TO OPEN
LOP TO COMP ON 480 V MCC BUS C2
FAILURE OF INTERLOCK
JVMAGSI120
GATE2519
FAILURC OF INTERLOCK (P-104)
COMMON CAUSE FAILURE OF INTERLOCKS
FSXRP104
FSXXINT

(PJOB10BX)



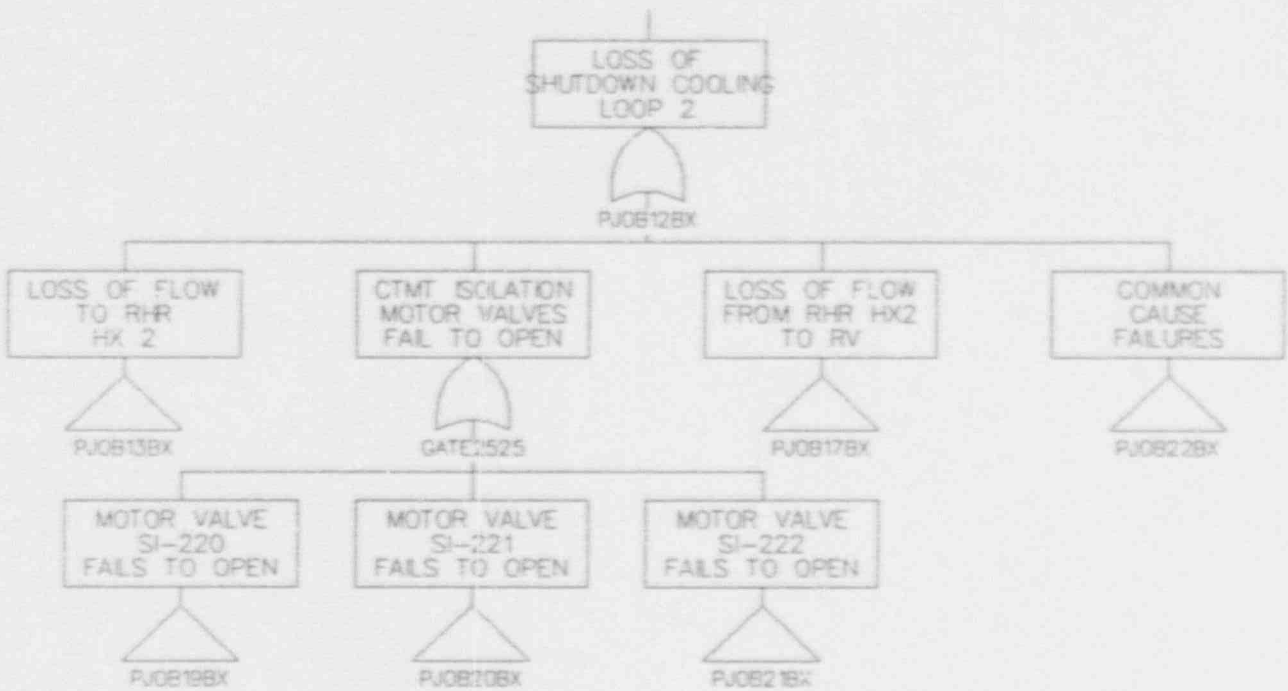
MOTOR VALVE SI-121 FAILS TO OPEN
PJOB10BX
MOTOR VALVE SI-121 FAILS TO OPEN
LOP TO COMP ON 480 V MCC BUS C2
FAILURE OF INTERLOCK
JVMAGSI121
PEDN01C2
GATE2521
FAILURE OF INTERLOCK (F-106)
COMMON CAUSE FAILURE OF INTERLOCKS
FSXRP106
FSXXINT

(PJOB11BX)



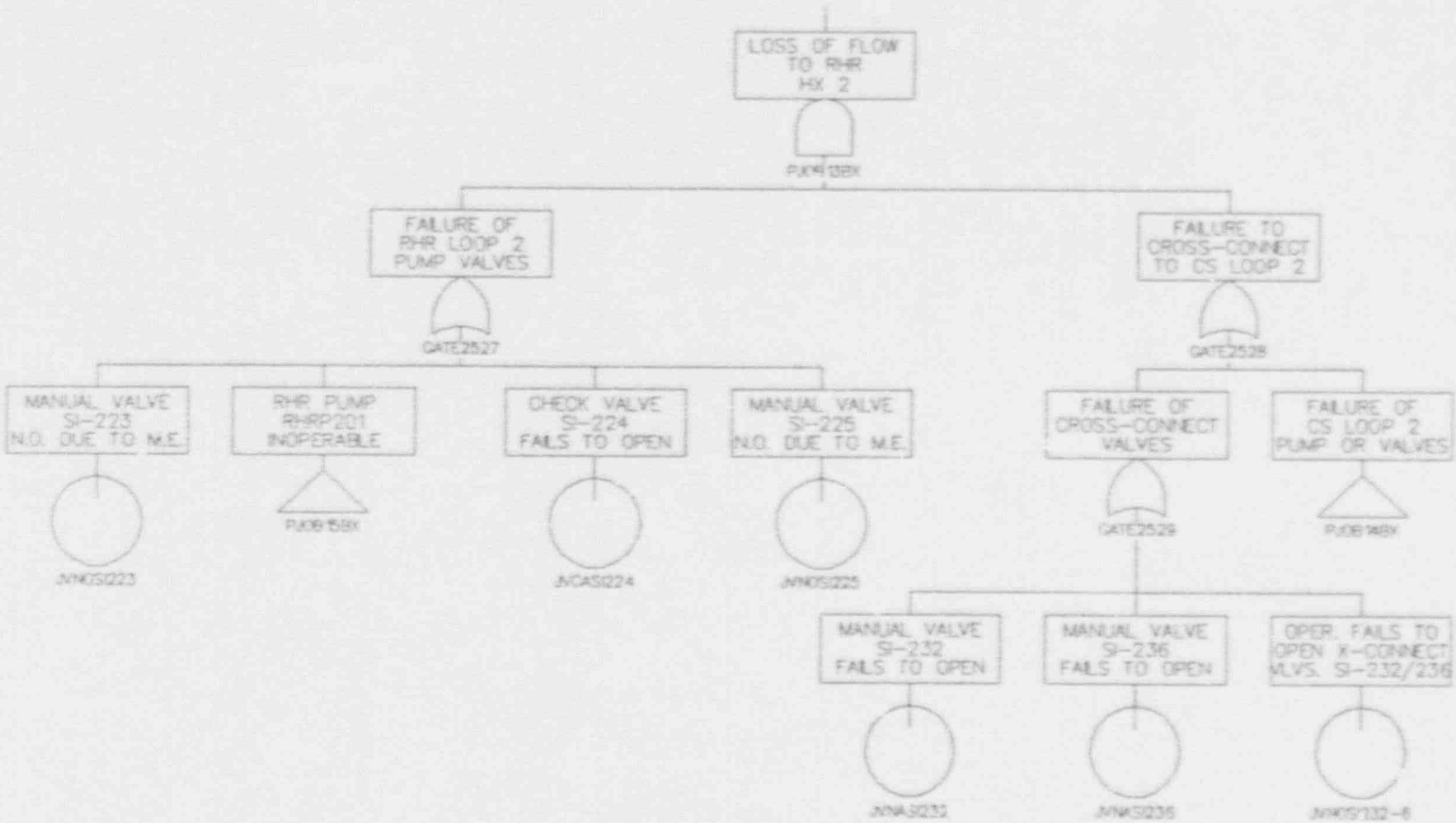
MOTOR VALVE
SI-122
FAILS TO OPEN
PJOB11BX
MOTOR VALVE
SI-122
FAILS TO OPEN
JVMAGSI122
LOP TO COMP
ON 480 V MCC
BUS A2
PEDNO1A2

(PJOB12BX)



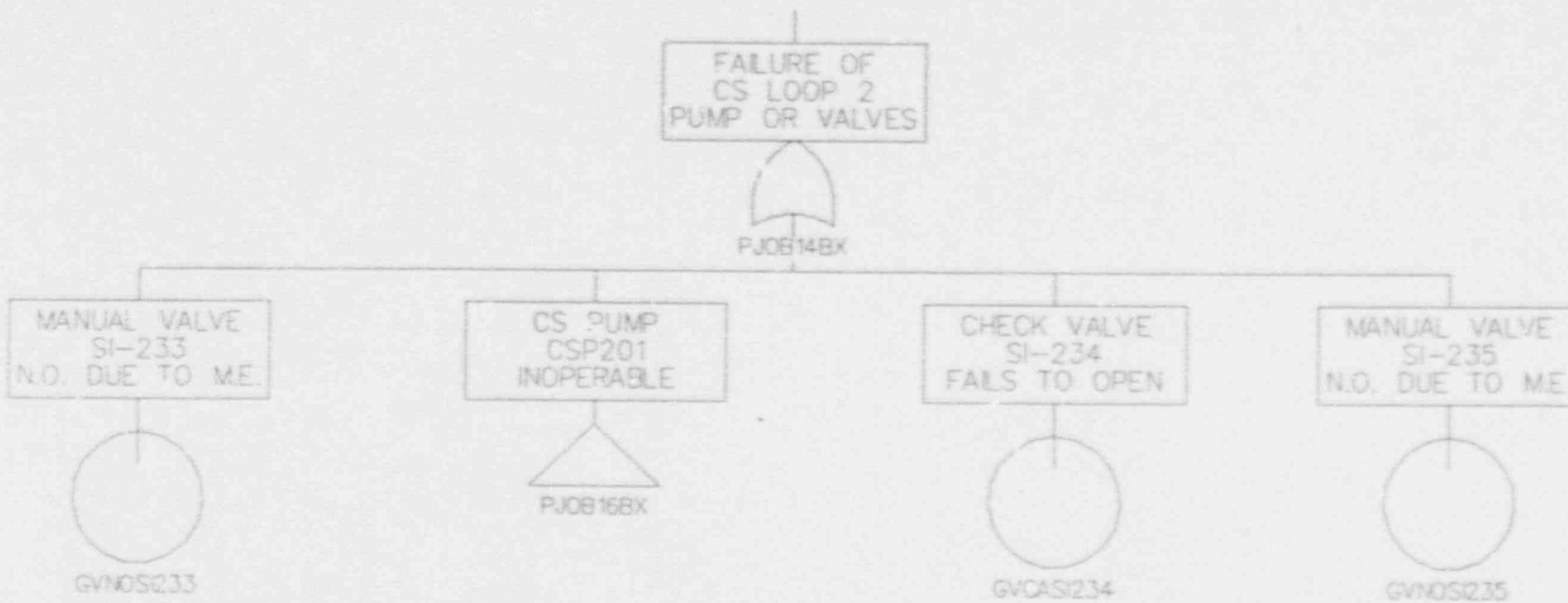
LOSS OF SHUTDOWN COOLING LOOP 2
PJOB12BX
LOSS OF FLOW TO RHR HX 2
PJOB13BX
CTMT ISOLATION MOTOR VALVES FAIL TO OPEN
GATE2525
LOSS OF FLOW FROM RHR HX2 TO RV
PJOB17BX
COMMON CAUSE FAILURES
PJOB22BX
MOTOR VALVE SI-220 FAILS TO OPEN
PJOB19BX
MOTOR VALVE SI-221 FAILS TO OPEN
PJOB20BX
MOTOR VALVE SI-222 FAILS TO OPEN
PJOB21BX

(PJOB13BX)

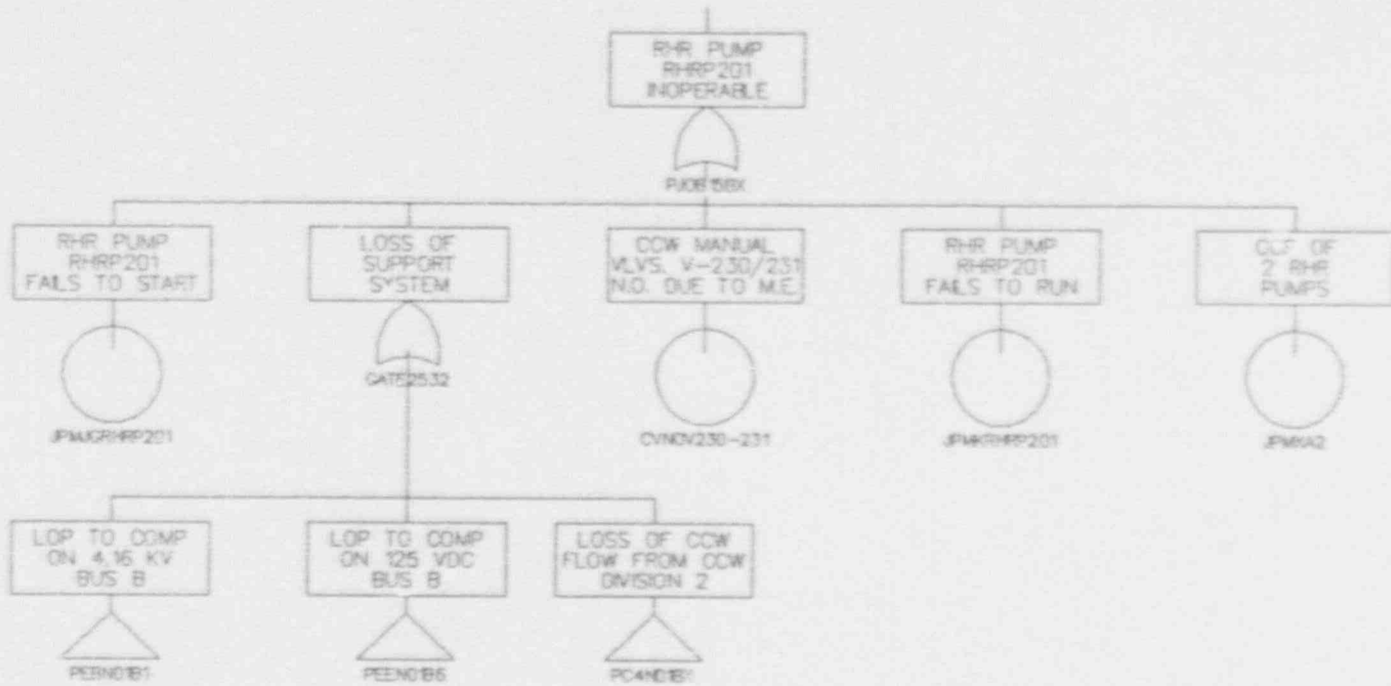


LOSS OF FLOW TO RHR HX 2
FAILURE OF RHR LOOP 2 PUMP VALVES
GATE2527
FAILURE TO CROSS-CONNECT TO CS LOOP 2
GATE2528
MANUAL VALVE SI-223 N.O. DUE TO M.E.
RHR PUMP RHRP201 INOPERABLE
CHECK VALVE SI-224 FAILS TO OPEN
JVCASI224
MANUAL VALVE SI-225 N.O. DUE TO M.E.
FAILURE OF CROSS-CONNECT VALVES
GATE2529
FAILURE OF CS LOOP 2 PUMP OR VALVES
MANUAL VALVE SI-232 FAILS TO OPEN
MANUAL VALVE SI-236 FAILS TO OPEN
OPER. FAILS TO OPEN X-CONNECT VLVS. SI-232/236

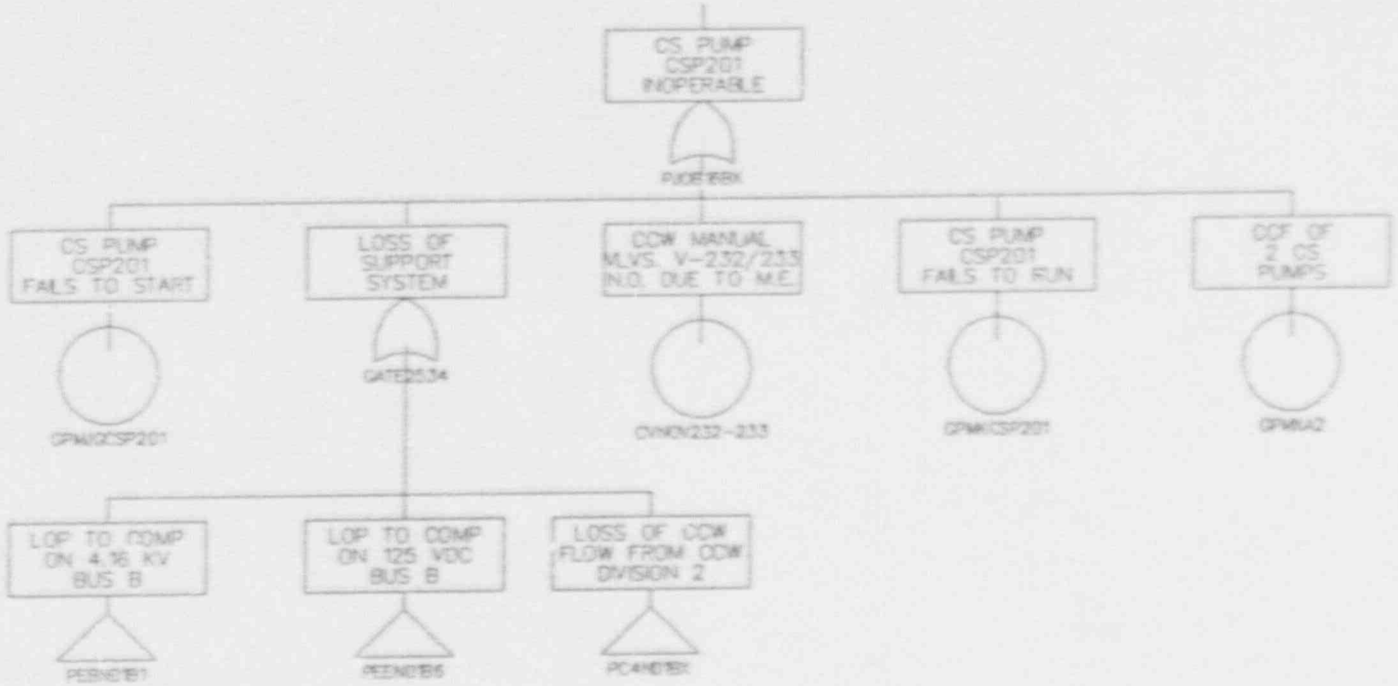
(PJOB14BX)


FAILURE OF
CS LOOP 2
PUMP OR VALVES
PJOB14BX
MANUAL VALVE
SI-233
N.O. DUE TO M.E.
GVNOSI233
CS PUMP
CSP201
INOPERABLE
PJOB16BX
CHECK VALVE
SI-234
FAILS TO OPEN
GVCASI234
MANUAL VALVE
SI-235
N.O. DUE TO M.E.
GVNOSI235

(PJOB 15BX)

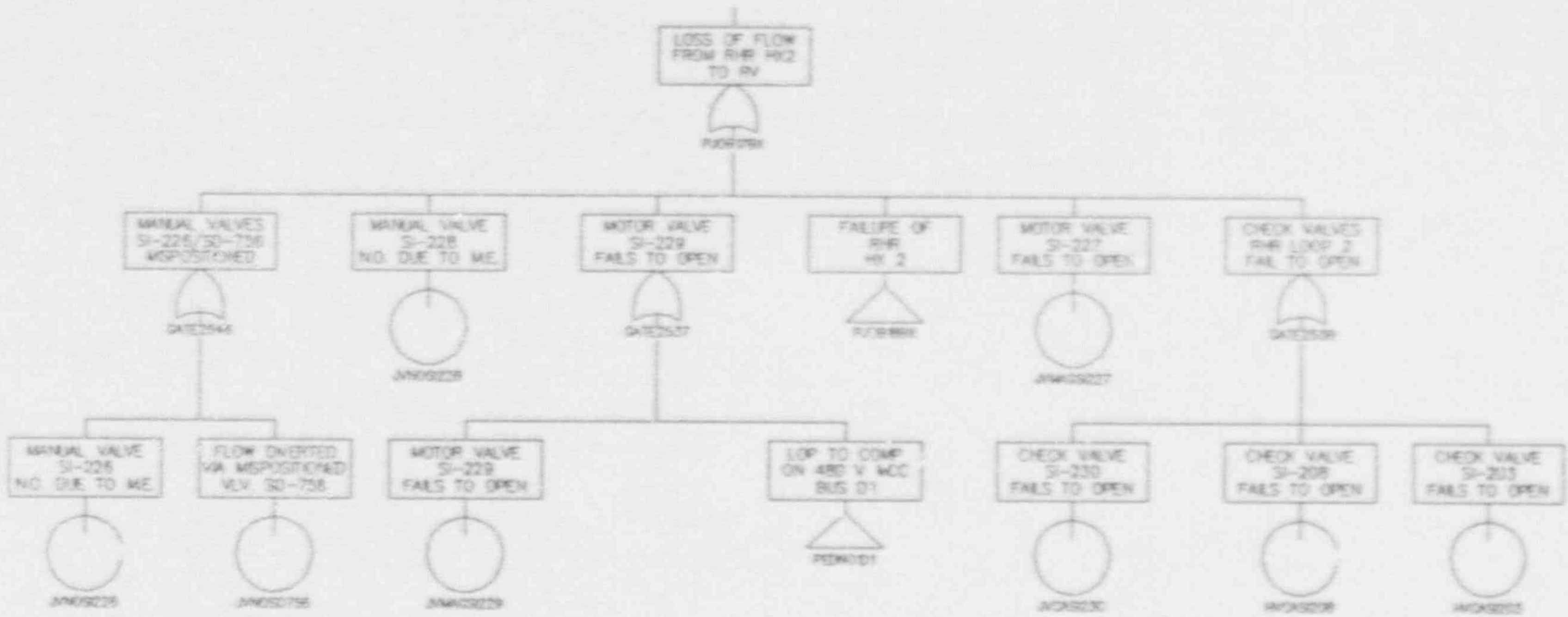

RHR PUMP RHRP201 INOPERABLE
PJOB15BX
RHR PUMP RHRP201 FAILS TO START
LOSS OF SUPPORT SYSTEM
GATE2532
CCW MANUAL VLVS. V-230/231 N.O. DUE TO M.E.
CVNOV230-231
RHR PUMP RHRP201 FAILS TO RUN
CCF OF 2 RHR PUMPS
LOP TO COMP ON 4.16 KV BUS B
LOP TO COMP ON 125 VDC BUS B
LOSS OF CCW FLOW FROM CCW DIVISION 2

(PJOB 16BX)



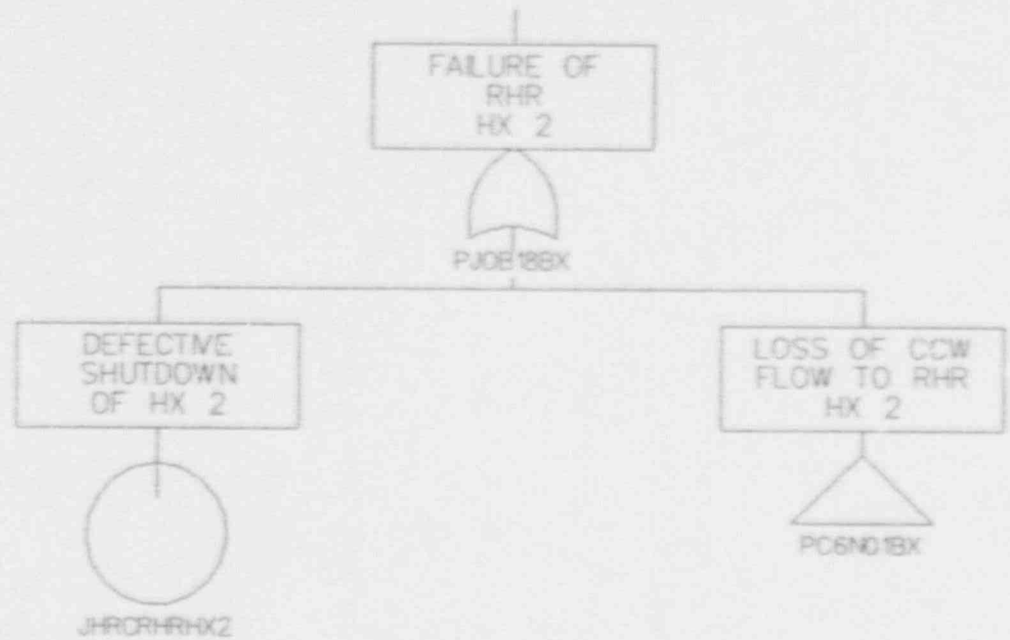
CS PUMP CSP201 INOPERABLE
CS PUMP CSP201 FAILS TO START
LOSS OF SUPPORT SYSTEM
GATE2534
CCW MANUAL M.V.S. V-232/233 N.O. DUE TO M.E.
CS PUMP CSP201 FAILS TO RUN
CCF OF 2 CS PUMPS
LOP TO COMP ON 4.16 KV BUS B
LOP TO COMP ON 125 VDC BUS B
LOSS OF CCW FLOW FROM CCW DIVISION 2

(PJOB17BX)

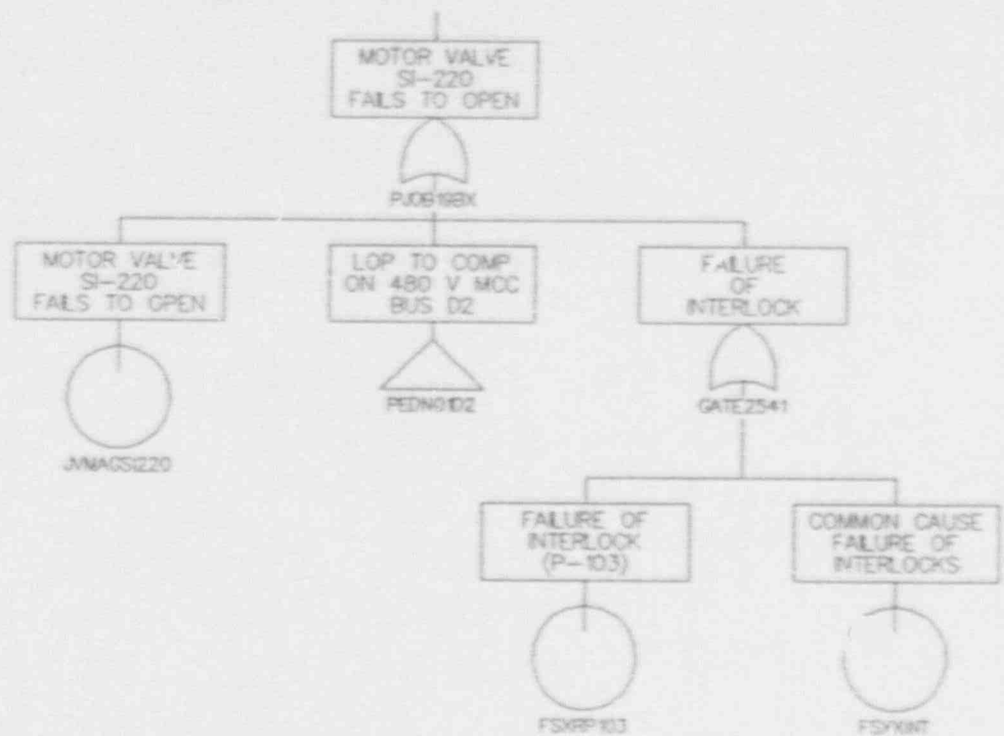


LOSS OF FLOW FROM RHR HX2 TO RV
MANUAL VALVES SI-225/SD-756 MISPOSITIONED
MANUAL VALVE SI-228 N.O. DUE TO M.E.
MOTOR VALVE SI-229 FAILS TO OPEN
FAILURE OF RHR HX 2
MOTOR VALVE SI-227 FAILS TO OPEN
CHECK VALVES RHR LOOP 2 FAIL TO OPEN
MANUAL VALVE SI-226 N.O. DUE TO M.E.
FLOW DIVERTED VIA MISPOSITIONED VLV. SD-756
MOTOR VALVE SI-229 FAILS TO OPEN
LOP TO COMP ON 480 V MCC BUS D1
CHECK VALVE SI-230 FAILS TO OPEN
CHECK VALVE SI-208 FAILS TO OPEN
CHECK VALVE SI-203 FAILS TO OPEN

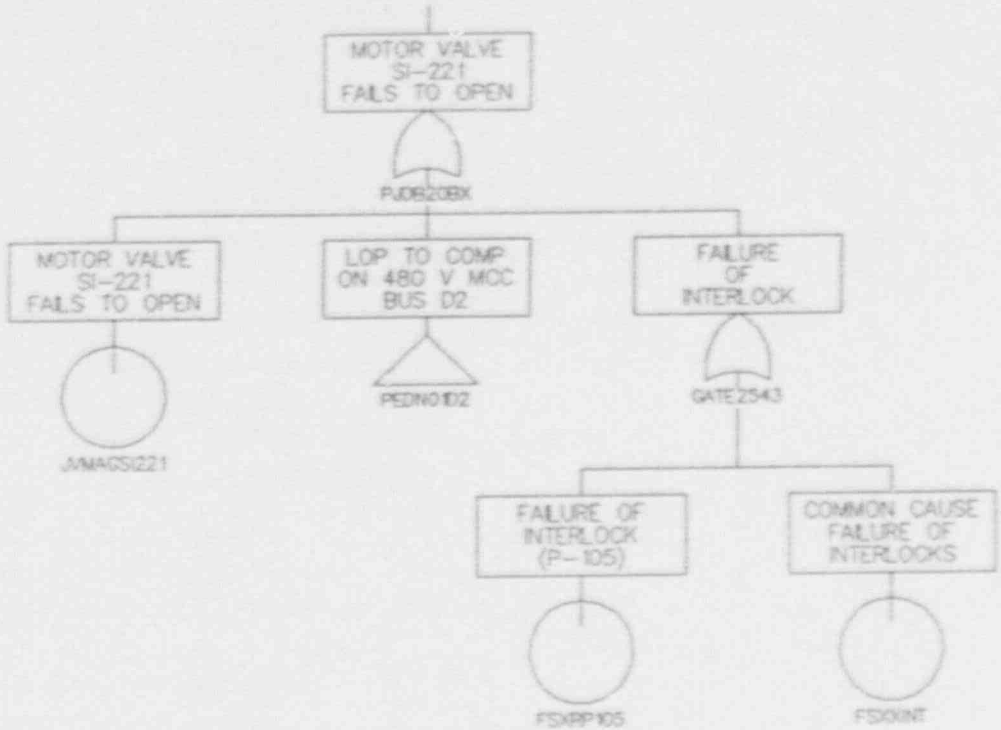
(PJOB18BX)


FAILURE OF RHR HX 2
PJOB18BX
DEFECTIVE SHUTDOWN OF HX 2
JHRCRHRHX2
LOSS OF CCW FLOW TO RHR HX 2
PC6N01BX

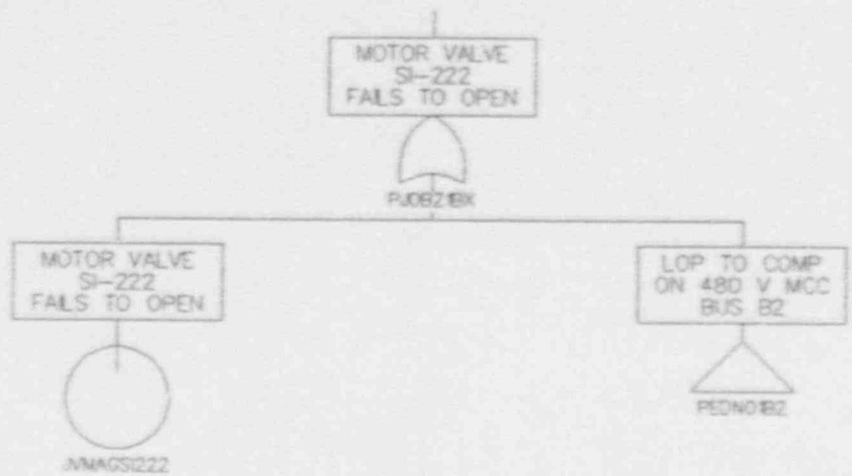
(PJOB19BX)


MOTOR VALVE SI-220 FAILS TO OPEN
PJOB19BX
MOTOR VALVE SI-220 FAILS TO OPEN
JVMAGSI220
LOP TO COMP ON 480 V MCC BUS D2
PEDN01D2
FAILURE OF INTERLOCK
GATE2541
FAILURE OF INTERLOCK (P-103)
FSXRP103
COMMON CAUSE FAILURE OF INTERLOCKS
FSYXINT

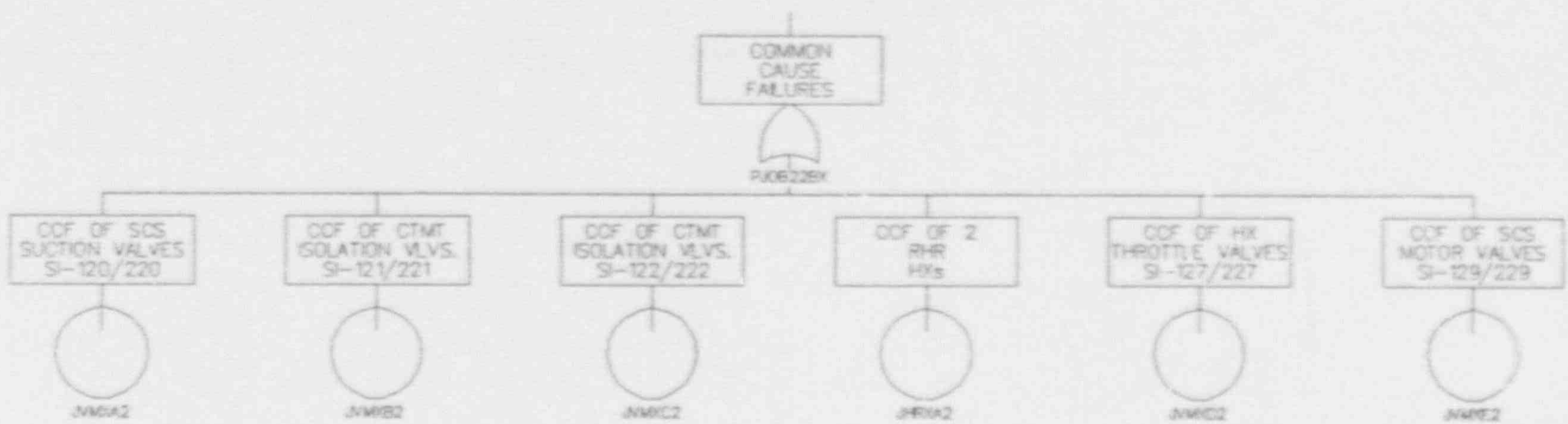
(PJOB20BX)


MOTOR VALVE SI-221 FAILS TO OPEN
PJOB20BX
MOTOR VALVE SI-221 FAILS TO OPEN
LOP TO COMP ON 480 V MCC BUS D2
FAILURE OF INTERLOCK
JVMAGSI221
PEDN01D2
GATE2543
FAILURE OF INTERLOCK (P-105)
COMMON CAUSE FAILURE OF INTERLOCKS
FSXRP105
FSXXINT

(PJOB21BX)

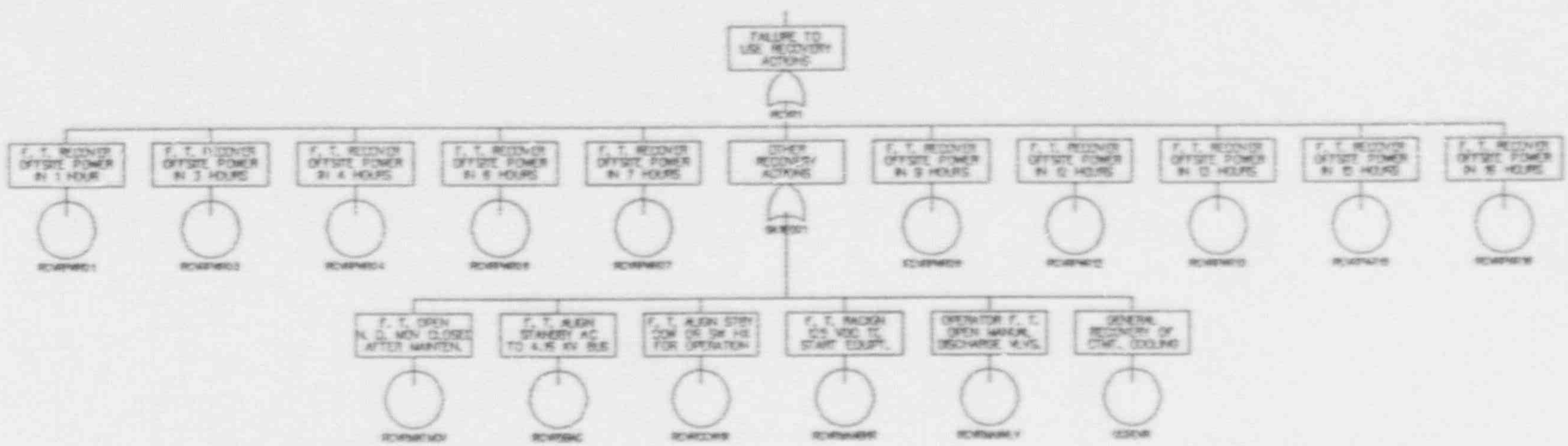

MOTOR VALVE
SI-222
FAILS TO OPEN
PJOB21BX
MOTOR VALVE
SI-222
FAILS TO OPEN
JVMAGSI222
LOP TO COMP
ON 480 V MCC
BUS B2
PEDN01B2

(PJOB22BX)



COMMON CAUSE FAILURES
PJOB22BX
CCF OF SCS SUCTION VALVES SI-120/220
CCF OF CTMT ISOLATION VLVS. SI-121/221
CCF OF CTMT ISOLATION VLVS. SI-122/222
CCF OF 2 RHR HXs
CCF OF HX THROTTLE VALVES SI-127/227
CCF OF SCS MOTOR VALVES SI-129/229
JVMXA2
JVMXB2
JVMXC2
JHRXA2
JVMXD2
JVMXE2

RECOVERY ACTIONS



Question 720.78

Succ. sful operation of high pressure safety injection pumps in loops B and D requires control power from 125 Vdc bus B. In the event of a failure of 125 Vdc bus B, only HPSI pumps in loops A and C are available for mitigating accidents. In the fault tree (Figure 6.3.6-13) developed for failure to deliver safety injection with 125 Vdc (bus B unavailable), therefore, should GATE 1836 be defined to be "failure to deliver sufficient SI flow to 2 of 2 loops" instead of 4 of 4 loops? Similarly, should GATE 1838 be defined to be "CCF of 2 of 2 loops" rather than "CCF of 4 of 4 loops"?

Response 720.78

As described in subsection 6.3.1.1.3.11 and depicted in Figure 6.3.1-2, the 125 VDC class 1E power system consists of six independent and physically separate load groups. Each load group includes a battery, a battery charger and DC distribution center. The battery chargers of load group channels A, C and division I are powered from division I of the 480 VAC power system. Similarly, the battery chargers of load group channels B, D and division II are powered from division II of the 480 VAC power system. Furthermore, the ESF equipment which is loaded on the 4.16 KV or 480 V bus is provided with redundant trip coils. Control power for the trip coil circuitries is assumed to be obtained from the 125 VDC buses A and I for division I equipment and from 125 VDC buses B and II for division II equipment. The HPSI pumps in loops B and D are powered from division II power system, thus the control power for these pumps can be from 125 VDC bus B or II. Therefore, in the event of a failure of 125 VDC bus B, the HPSI pumps in safety injection loops B and D do not necessarily become unavailable. For the HPSI pumps in loops B and D to completely fail, failure of both the 125 VDC buses B and II must occur. The fault tree (Figure 6.3.6-13) developed for failure to deliver safety injection flow with 125 VDC bus B unavailable correctly models such logic.

Question 720.79

Based on the fault trees shown in Figure 6.3.7-3 (8 of 9 and 9 of 9), loss of 125 VDC control power (bus B) will lead to loss of flow from emergency feedwater (EFW) sub-train B2 because of the failure of motor-driven EFW pump-104. In the fault tree (Figure 6.3.7-7) constructed for failure to deliver sufficient EFW to either steam generator with loss of 125 VDC vital bus B, therefore, should GATE 2110 contain just one event "loss of flow from EFW train B1"? Was the failure probability of the event, PAOG08EX, taken to be 1.0 in the fault tree quantification? Why are the availability tne emergency feedwater tanks not modeled in the fault tree?

Response 720.79

As described in subsection 6.3.1.1.3.11 and depicted in Figure 6.3.1-2, the 125 VDC class 1E power system consists of six independent and physically separate load groups. Each load group includes a battery, a battery charger and DC distribution center. The battery chargers of load group channels A, C and division I are powered from division I of the 480 VAC power system. Similarly, the battery chargers of load group channels B, D and division II are powered from division II of the 480 VAC power system. Furthermore, the ESF equipment which is loaded on the 4.16 KV or 480 V bus is provided with redundant trip coils. Control power for the trip coil circuitries is assumed to be obtained from the 125 VDC buses A and I for division I equipment and from 125 VDC buses B and II for division II equipment. The motor-driven EFW pump 104 in sub-train B2 is powered from division II power system, thus the control power for these pumps can be from 125 VDC bus B or II. Therefore, in the event of a failure of 125 VDC bus B, this pump does not necessarily become unavailable. For the EFW pump-104 to completely fail, failure of both the 125 VDC buses B and II must occur. The fault tree (Figure 6.3.7-7) developed for failure to deliver sufficient EFW to either steam generator with 125 VDC bus B unavailable correctly models such logic. The failure probability of the event PAOG08FX was accordingly derived, and *not* taken to be 1.0, in the fault tree quantification. The (un)availabilities of the emergency feedwater tanks was modeled in the developed events ALOSFPTINDO, loss of suction flow to turbine-driven EFW pump, and ALOSFPMINDO, loss of suction flow to motor-driven EFW pump.

Question 720.80

Loss of one component cooling water division, loss of 125 Vdc vital bus, or loss of a 4.16 KV vital bus all have some impact on the availability of the containment spray system. No clear treatment of the relevant impacts, however, can be found in the fault trees (Figure 6.3.13-2) developed for failure of the containment spray system. Please explain why they are not modeled.

Response 720.80

Treatment of relevant impacts of loss of one component cooling water division, loss of 125 Vdc vital bus, or loss of a 4.16 KV vital bus on the availability of the containment spray system was considered when developing fault tree for failure of the containment system. However, some of the fault trees detailing the logic were inadvertently left out of the report and the rest of the fault trees were included in other system or special functions. The impacts of loss of offsite power, loss of one division (B or II) of component cooling water and loss of a 125 VDC vital bus are modeled in the development of the fault trees for the special function "Failure to Successfully Cool the IRWST given a Loss of Offsite Power" (Figure 6.4.2-6), "Failure to Successfully Cool the IRWST given Loss of a Component Cooling Water division" (Figure 6.4.2-7) and "Failure to Successfully Cool the IRWST given Loss of a 125 VDC Vital Bus" (Figure 6.4.2-8) in volume 2 of DCTR-RS-02, Rev. 0, respectively. Only the fault tree for "Containment Spray (CS) pump CSP-101 inoperable (given loss of a component cooling water division), PGIB15CX, is included in Figure 6.4.2-7 (on page 6-846). Other relevant fault trees attached herewith are:

PJOB06DX : CS pump CSP-101 inoperable given a Loss of Offsite Power

PJOB16DX : CS pump CSP-201 inoperable given a Loss of Offsite Power

PJOB16EX : CS pump CSP-201 inoperable given Loss of a 125 VDC Vital Bus

The consequences of losing a 4.16 KV vital bus are identical to those for loss of a component cooling water division and, therefore, the case for losing a 4.16 KV vital bus was not explicitly modeled. Instead the fault trees developed for loss of a component cooling water division were used.

Similarly, the corresponding fault trees for the Residual Heat Removal (RHR) or Shutdown Cooling System (SDC) pumps needed in the special function "Failure to Successfully Cool the IRWST" were inadvertently left out of the report. These are also attched.

PJOB05DX : RHR pump RHRP-101 inoperable given a Loss of Offsite Power

PJOB15DX : RHR pump RHRP-201 inoperable given a Loss of Offsite Power

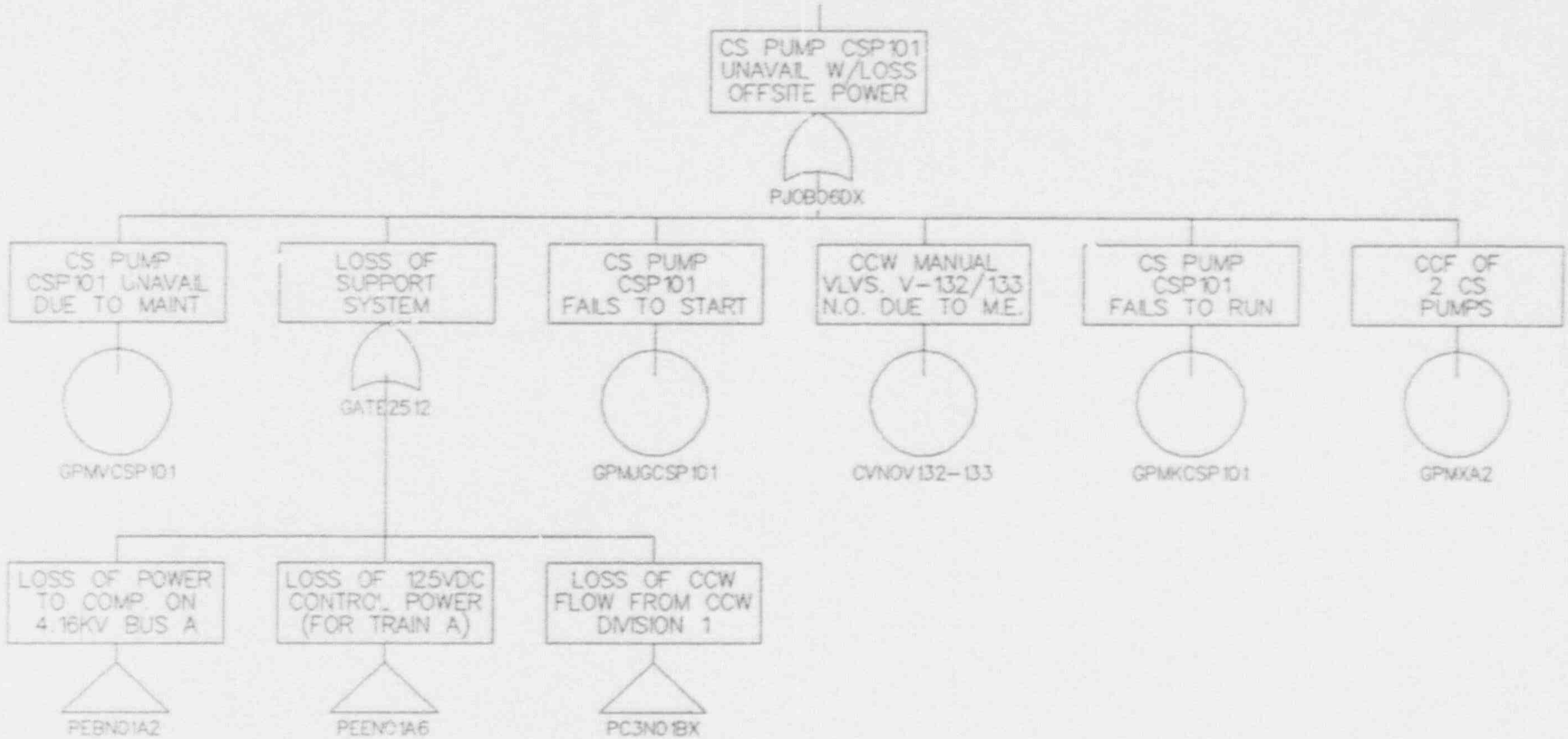
PJOB15EX : RHR pump RHRP-201 inoperable given Loss of a 125 VDC Vital Bus

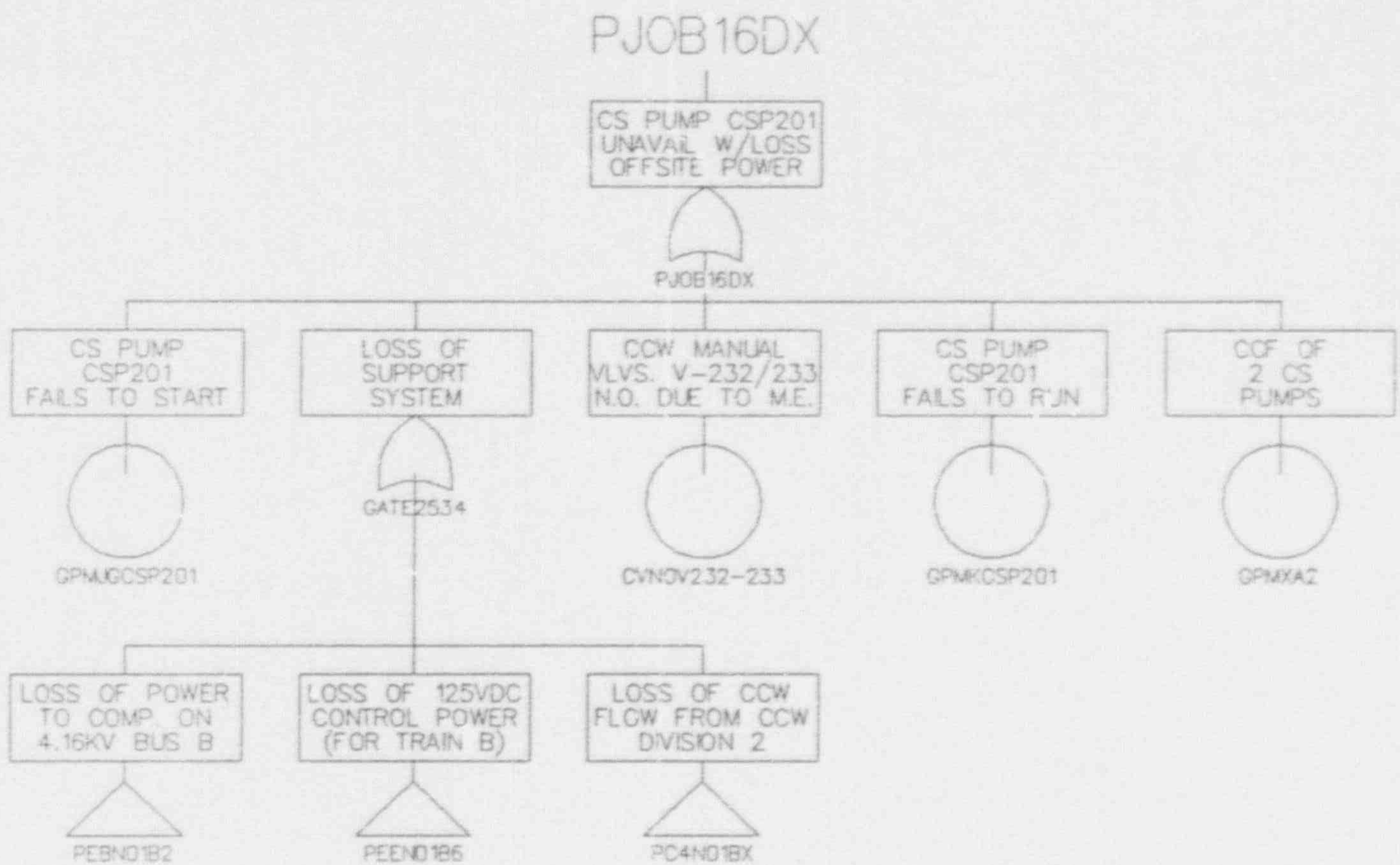
The fault tree developed for unavailability of RHR pump-101 given loss of a component cooling water division is presented as PGIB14CX in Figure 6.4.2-7 (on page 6-845).

Response 720.80 (cont.d)

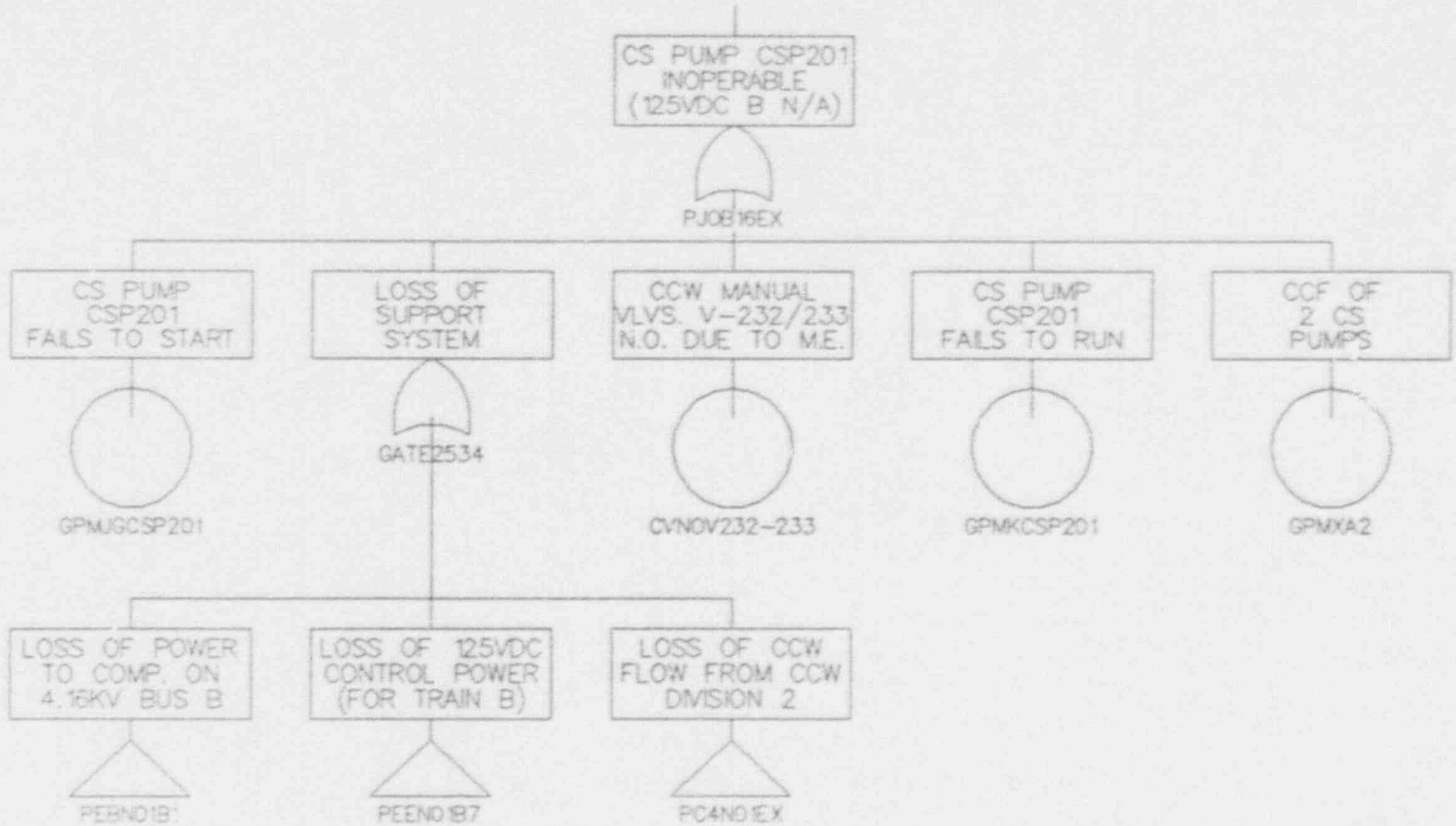
Also, it is to be noted that, as described in subsection 6.3.1.1.3.11 and depicted in Figure 6.3.1-2, the 125 VDC class 1E power system consists of six independent and physically separate load groups. Each load group includes a battery, a battery charger and DC distribution center. The battery chargers of load group channels A, C and division I are powered from division I of the 480 VAC power system. Similarly, the battery chargers of load group channels B, D and division II are powered from division II of the 480 VAC power system. Furthermore, the ESF equipment which is loaded on the 4.16 KV or 480 V bus is provided with redundant trip coils. Control power for the trip coil circuitries is assumed to be obtained from the 125 VDC buses A and I for division I equipment and from 125 VDC buses B and II for division II equipment. The CS and RHR pumps in train B are powered from division II power system, thus the control power for these pumps can be from 125 VDC bus B or II. Therefore, in the event of a failure of 125 VDC bus B, these pumps do not necessarily become unavailable. For the CS or RHR pump to completely fail, failure of both the 125 VDC buses B and II must occur. The fault trees, PJOB16EX and PJOB15EX, developed for unavailability of CS pump and RHR pump, given loss of a 125 VDC bus B, correctly model such logic.

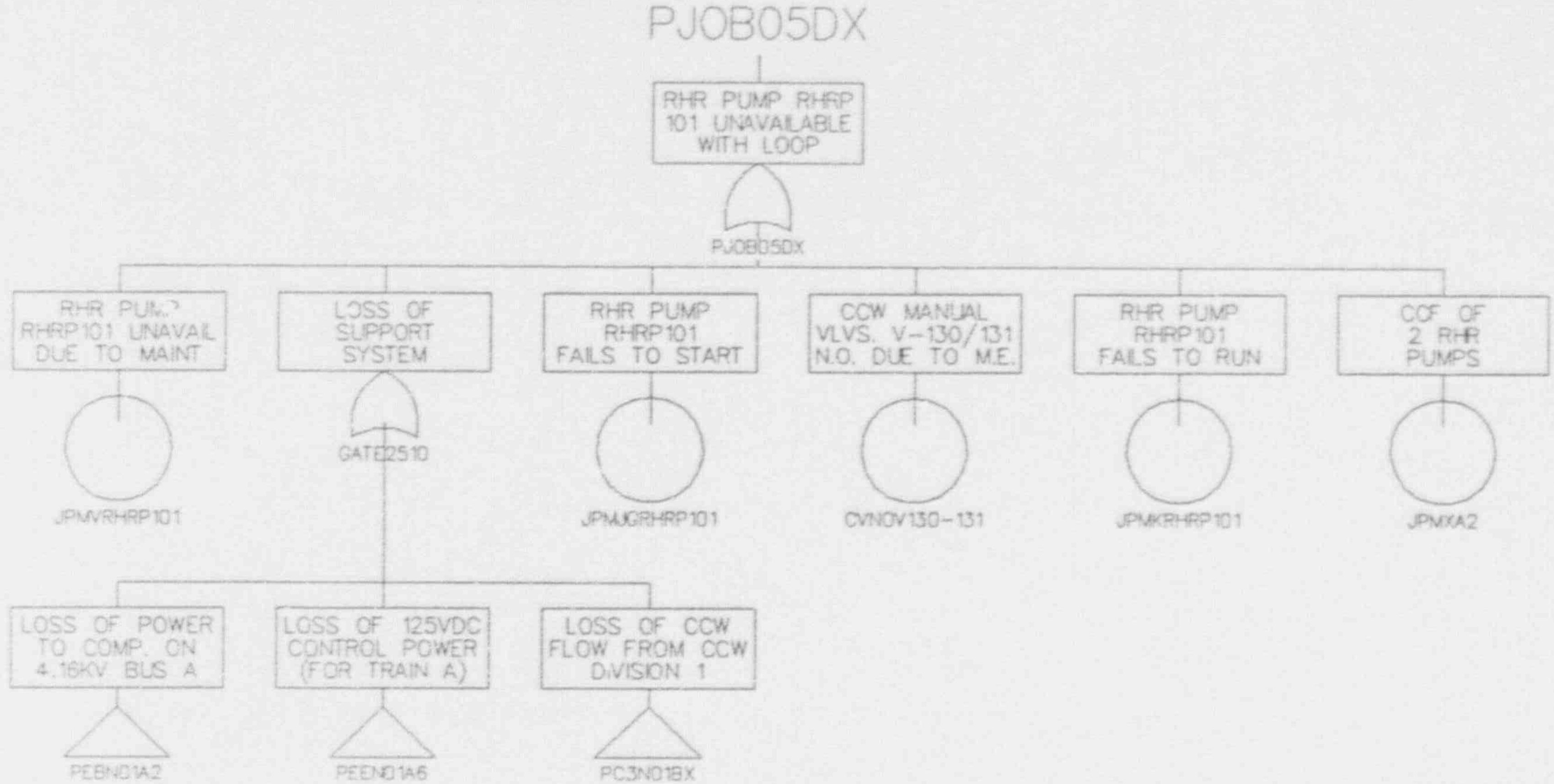
PJOB06DX

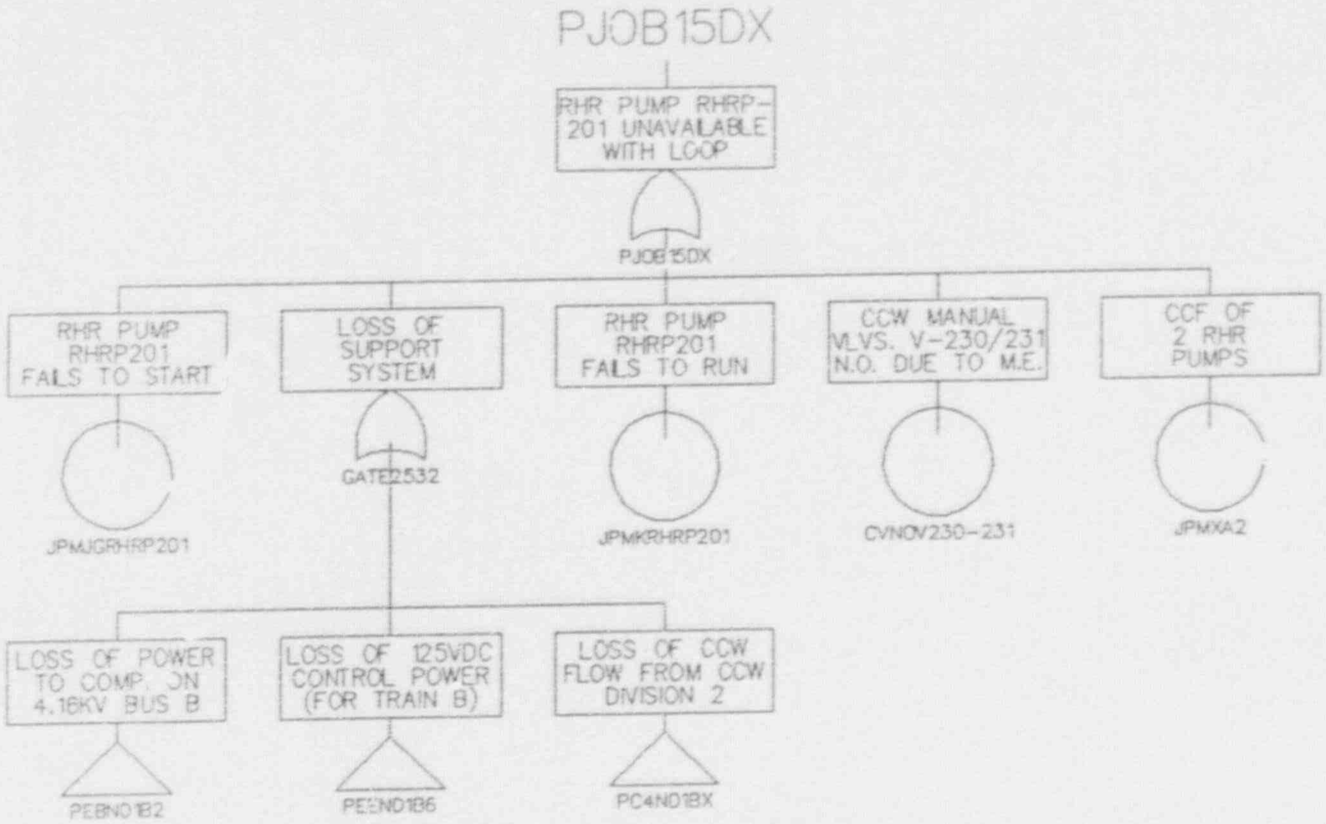

CS PUMP CSP101 UNAVAIL W/LOSS OFFSITE POWER
PJOB06DX
CS PUMP CSP101 UNAVAIL DUE TO MAINT
GPMVCSP101
LOSS OF SUPPORT SYSTEM
GATE2512
CS PUMP CSP101 FAILS TO START
GPMJGCSP101
CCW MANUAL VLVS. V-132/133 N.O. DUE TO M.E.
CVNOV132-133
CS PUMP CSP101 FAILS TO RUN
GPMKCSP101
CCF OF 2 CS PUMPS
GPMXA2
LOSS OF POWER TO COMP. ON 4.16KV BUS A
PEBN01A2
LOSS OF 125VDC CONTROL POWER (FOR TRAIN A)
PEEN01A6
LOSS OF CCW FLOW FROM CCW DIVISION 1
PC3N018X

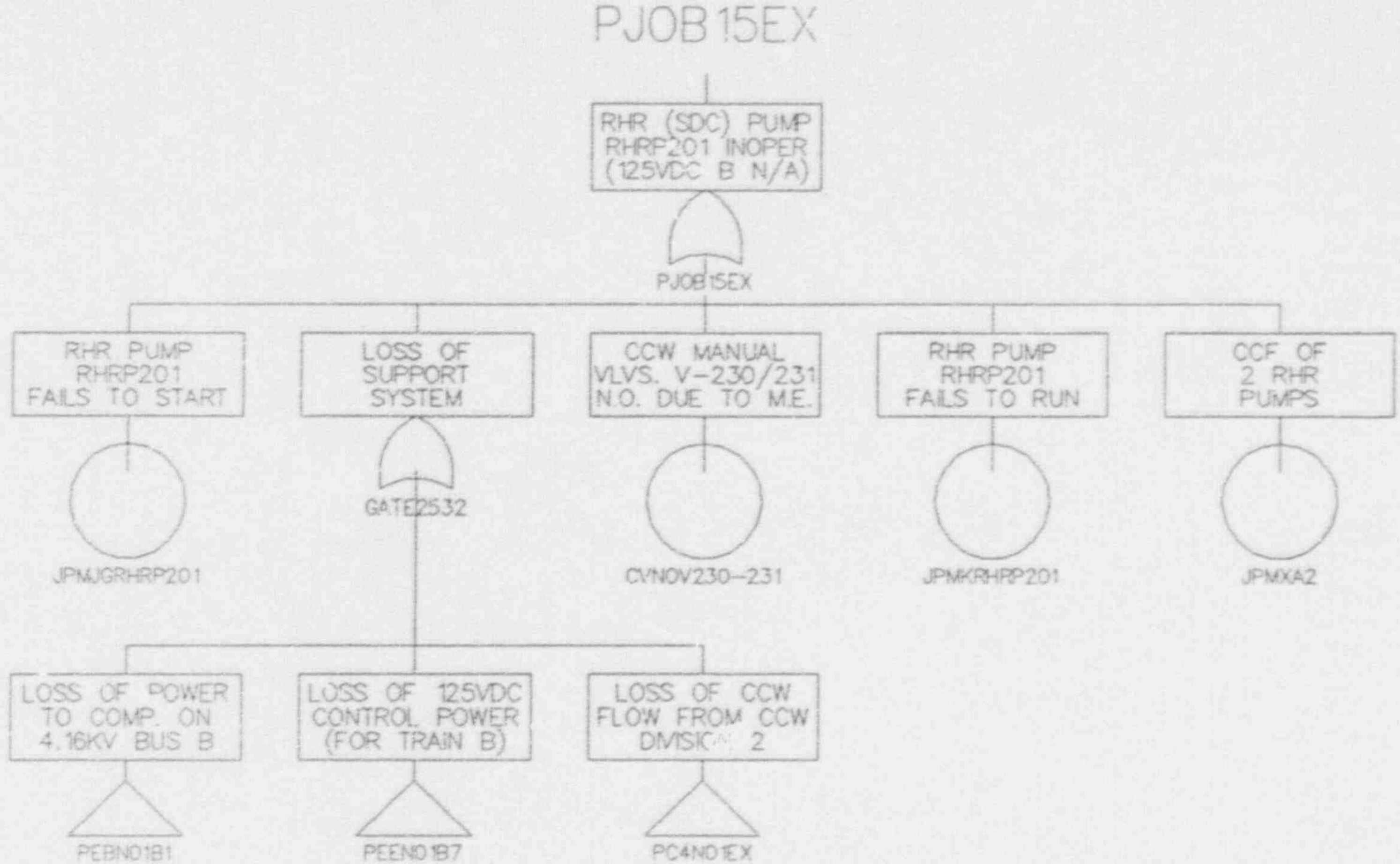

PJOB16DX
CS PUMP CSP201 UNAVAIL W/LOSS OFFSITE POWER
PJOB16DX
CS PUMP CSP201 FAILS TO START
LOSS OF SUPPORT SYSTEM
CCW MANUAL VLVS. V-232/233 N.O. DUE TO M.E.
CS PUMP CSP201 FAILS TO R'JN
CCF OF 2 CS PUMPS
GATE2534
GPMJGCSP201
CVNOV232-233
GPMKCSP201
GPMXA2
LOSS OF POWER TO COMP. ON 4.16KV BUS B
LOSS OF 125VDC CONTROL POWER (FOR TRAIN B)
LOSS OF CCW FLOW FROM CCW DIVISION 2
PEBN01B2
PEEN01B6
PC4N01BX

PJOB16EX


CS PUMP CSP201 INOPERABLE (125VDC B N/A)
PJOB16EX
CS PUMP CSP201 FAILS TO START
GPMJGCSP201
LOSS OF SUPPORT SYSTEM
GATE2534
CCW MANUAL VLVS. V-232/233 N.O. DUE TO M.E.
CVNOV232-233
CS PUMP CSP201 FAILS TO RUN
GPMKCSP201
CCF OF 2 CS PUMPS
GPMXA2
LOSS OF POWER TO COMP. ON 4.16KV BUS B
PEBN01B1
LOSS OF 125VDC CONTROL POWER (FOR TRAIN B)
PEEN01B7
LOSS OF CCW FLOW FROM CCW DIVISION 2
PC4N01EX


PJOB05DX
RHR PUMP RHRP 101 UNAVAILABLE WITH LOOP
PJOB05DX
RHR PUMP RHRP101 UNAVAIL DUE TO MAINT
JPMVRHRP101
LOSS OF SUPPORT SYSTEM
GATE2510
RHR PUMP RHRP101 FAILS TO START
JPMJGRHRP101
CCW MANUAL VLVS. V-130/131 N.O. DUE TO M.E.
CVNOV130-131
RHR PUMP RHRP101 FAILS TO RUN
JPMKRHRP101
CCF OF 2 RHR PUMPS
JPMXA2
LOSS OF POWER TO COMP. ON 4.16KV BUS A
PEBN01A2
LOSS OF 125VDC CONTROL POWER (FOR TRAIN A)
PEEN01A6
LOSS OF CCW FLOW FROM CCW DIVISION 1
PC3N01BX


PJOB15DX
RHR PUMP RHRP-201 UNAVAILABLE WITH LOOP
PJOB15DX
RHR PUMP RHRP201 FAILS TO START
JPMJGRHRP201
LOSS OF SUPPORT SYSTEM
GATE2532
RHR PUMP RHRP201 FAILS TO RUN
JPMKRHRP201
CCW MANUAL VLVS. V-230/231 N.O. DUE TO M.E.
CVNOV230-231
CCF OF 2 RHR PUMPS
JPMXA2
LOSS OF POWER TO COMP. ON 4.16KV BUS B
PEBN01B2
LOSS OF 125VDC CONTROL POWER (FOR TRAIN B)
PEEN01B6
LOSS OF CCW FLOW FROM CCW DIVISION 2
PC4N01BX


PJOB15EX
RHR (SDC) PUMP RHRP201 INOPER (125VDC B N/A)
PJOB15EX
RHR PUMP RHRP201 FAILS TO START
LOSS OF SUPPORT SYSTEM
CCW MANUAL VLVS. V-230/231 N.O. DUE TO M.E.
RHR PUMP RHRP201 FAILS TO RUN
CCF OF 2 RHR PUMPS
GATE2532
JPMJGRHRP201
CVNOV230-231
JPMKRHRP201
JPMXA2
LOSS OF POWER TO COMP. ON 4.16KV BUS B
LOSS OF 125VDC CONTROL POWER (FOR TRAIN B)
LOSS OF CCW FLOW FROM CCW DIVISION 2
PEBN01B1
PEEN01B7
PC4N01EX

Question 720.81

Sections 1.2, 1.3, and 4.3 - It is stated that "bounding site characteristics were used for the evaluation of external events such as seismic and tornado strike events....". However, it appears that the single seismic hazard curve (Figure B4.3.1-2) used in the CESSAR 80+ PRA has a much lower return period than those in the EPRI Requirements Document hazards curves at the higher acceleration levels. In addition, at sites where EPRI and Lawrence Livermore National Laboratory (LLNL) have both made hazard curve estimations, the LLNL curves have tended to be an order of magnitude greater in frequency for a given acceleration level. What is CE's basis then for stating that the single hazard curve used in the CESSAR 80+ PRA is bounding? Does CE claim that the annual seismic severe accident core damage frequency of 1.2E-6 presented in Table B1.3-1 is best estimate or conservative? Because of the wide range of uncertainty within the earth science community regarding earthquake potential and ground motion estimation in the central and eastern United States, the seismic hazard curves developed by both EPRI and (LLNL), both with four and five ground motion experts, should be used for the seismic hazard estimation.

Response 720.81

The seismic analysis information, including the seismic hazard curve, used in the System 80+ seismic PRA was taken from the early version of the EPRI ALWR PRA Key Assumptions and Groundrules Document and was selected considering the PRA goals (e.g., core damage frequency). It is C-E's understanding that analysis using the LLNL hazard curve may not be necessary, pending NRC's final position on demonstrating plant safety beyond the safe shutdown earthquake (reference: Meeting with NRC on November 26, 1991). Should analysis using the LLNL hazard curve be required, however, it is likely that a new core damage goal would be selected.

Question 720.82

Section 4.3 - How is the buckling failure mode of steel containment incorporated in the containment fragility descriptions for the System 80+ PRA?

Response 720.82

The buckling failure mode of steel containment was not incorporated in the containment fragility description for the System 80+ PRA. C-E is currently updating the System 80+ PRA. The potential impact of this failure mode will be evaluated as part of this update.

Question 720.83

Section 2.9 - What is the basis for the primary feed and bleed success criterion? In general, are the success criteria in the System 80+ PRA "best estimate" or design basis?

Response 720.83

The design basis for feed and bleed was that the reactor vessel level would remain at least 2 feet above the top of the core if feed and bleed operation was initiated within 30 minutes. Using C-E's design basis codes, the bleed valves were sized so that this could be accomplished using one bleed valve and two HPSI pumps. Additional MAAP analyses demonstrated that feed and bleed would be successful with one bleed valve and one HPSI pump. The upper portion of the core would uncover briefly, but the fuel temperature remained below 2200 Deg F. This was the basis for the feed and bleed success criterion in the System 80+ PRA. In general, the success criteria used in the System 80+ PRA are design basis criteria. However, in cases where the design basis success criteria were felt to be overly conservative, additional "best-estimate" thermal hydraulic analyses were performed using MAAP or CENTS to determine if less conservative success criteria would be viable. These "best-estimate" success criteria were used as appropriate.

Question 720.84

Section 3.1.8 - For loss-of-offsite power and station blackout sequences, the Standby AC Power System appears to make a significant contribution to accident prevention, yet it does not appear in Table B3.2-2, Component Importance for System 80+ PRA. Why not? Since the system is not safety grade, what is its assumed availability and reliability? What is this system's mathematical measure of importance?

Response 720.84

Use of the Standby AC Power System to provide AC power was treated as a recovery action. The basic event used to represent failure of this system is "RCVRSBAC". As presented in table 5-7 of the System 80+ PRA report, DCTR-RS-02, Rev. 0, January, 1991, the unavailability used for this event is 5.0E-2 with an error factor of 3.0. As presented in table 8.4-1 of the System 80+ PRA report and table B3.2-2, the Fussel-Vesely importance measure for this element is 1.84E-2.

Question 720.88

Section 4.2.3.4 - What is the "best estimate" recovery factor of offsite power after a major tornado?

Response 720.88

In the System 80+ tornado strike analysis, it was assumed that a tornado strike on site would result in a loss of offsite power lasting longer than the base 24 hour mission time. Therefore, recovery of offsite power was not credited in the System 80+ tornado strike analysis.

Question 720.89

Section 4.3.1 - The generic fragility values assigned to components in the System 80+ PRA differ from those recommended in Appendix A (Table A.3.4) of the EPRI Requirements Document. Clarify these differences. Note that the EPRI document has median capacity factors for different types of sites.

Response 720.89

The generic component fragilities used in the System 80+ PRA were based on an early version of Appendix A of the EPRI Requirements Document. C-E is currently updating the System 80+ PRA. The generic component fragilities will be updated as appropriate for this analysis. See the response to Question 720.81.

Question 720.92

Section 4.3.2 - Table B4.3.2-1 gives core damage frequency contributions for different sequences. Describe how these frequencies were derived. Provide the top fifty seismic cutsets. Does "ERF" mean "error Factor"? If so, how was it obtained.

Response 720.92

The response to Question 720.62 describes how the seismic core damage frequencies were calculated. The cutsets for the dominant seismic sequences are presented in Tables 8.3.2-2 through 8.3.2-15 in the System 80+ PRA Report, DCTR-RS-02, Rev. 0, January 1991. "ERF" does mean "error factor".

Question 720.93

Section 4.3.2 - Provide or reference the random failure rates and human error rates used in the seismic PRA analysis. Detail the differences in these rates between internal events and seismic events. Describe the human errors considered in seismic events.

Response 720.93

The random failure rates and human error rates used in the seismic PRA analyses are presented in tables 5-2 through 5-7 of the System 80+ PRA report, DCTR-RS-02, Rev. 0, January, 1991. The same random failure rates were used for both the internal events and seismic analyses. The same human error rates, to the extent that they overlapped, were also used for both the internal events and seismic analyses. The human actions unique to the seismic analyses tended to be special recovery actions such as reactivating equipment after relay chatter failures or isolating component cooling water to equipment whose jacket cooling heat exchagers had failed.

Question 720.94

Section 4.3.2.4 - Provide the basis that one to two hours will be available for operators to reclose switchgear breakers after seismically-induced relay chatter.

Response 720.94

The seismic event is the initiating event. Therefore, the relay chatter failures occur at time zero and the plant systems fail to respond to the transient because of the relay chatter failures. Relay chatter failures occur at relatively low seismic acceleration levels. Thus, in the sequences of concern, there is little if any additional seismically-induced damage. The transient can be terminated be resetting the relays and reactivating the safety systems, primarily the emergency feedwater system. This is equivalent to a standard transient in which secondary side heat removal must be established within approximately 2 hours of the initiating event in order to prevent core damage.

Question 720.96

Describe how the System 80+ PRA and its insights were used in identifying equipment to be tested/evaluated in the ITAAC program. Describe how test specifications were influenced by the PRA.

Response 720.96

The PRA was used as an integral part of the design process to gain insight into vulnerabilities and to evaluate design features proposed by the EPRI Utility Requirements Document. Items to be selected for ITAAC will be based on design evaluations which may include PRA insights but , in general, the PRA will not be used to define the ITAAC program. The PRA will however, provide significant input to the Reliability Assurance Program.

Question 720.97

Provide a comparison of the estimated core damage frequency, conditional containment failure probability, and offsite consequences for the System 80+ design to the Commission Safety Goals, if no credit is taken for operator actions other than control room-based alignment of alternative core cooling methods.

Response 720.97

C-E is currently updating the System 80+ PRA. The requested comparison will be provided in the updated PRA report.

Question 720.98

The bottom line core damage frequency estimate for the System 80+ design appears to be very low. Given these low estimates, address how the System 80+ PRA evaluated initiating events that have a lower frequency than those normally postulated, but may have more serious consequences. Examples include the questions, "What does a 1E-5/yr steam generator tube rupture initiator look like and how is it handled in the PRA?," "What do various 1E-5 common cause failures look like and how were they looked for and evaluated in the PRA?," and "What is the effect of multiple failures/equipment outages during modes other than full power?."

Response 720.98

The System 80+ PRA does not address initiating events with very low frequencies because there little information on what these initiating events might be. This approach is consistent with EPRI guidance and generally accepted methodology.

Question 720.99

Describe how the PRA has factored in the possibility of the need to deal with appreciable fuel damage in conjunction with RHR operation and waste processing?

Response 720.99

The C-E System 80+ PRA does not address the possibility of the need to deal with appreciable fuel damage in conjunction with RHR operation and waste processing. In the System 80+ PRA, sequences involving appreciable fuel damage are assumed to be core melt sequences and evaluated as severe accidents.

Question 720.100

Describe how the PRA modeled control systems and control system failure modes.

Response 720.100

The System 80+ PRA does not model control system failures. It was assumed, that with the improved component control system, control system failures would have minimal impact. C-E is currently updating the System 80+ PRA. The potential impact of control system failure will be reassessed during this update.

Question 720.101

Provide a discussion of how the System 80+ PRA was used to identify equipment/structures/components to be covered by the Reliability Assurance (RAP).

Response 720.101

A Reliability Assurance Program (RAP) plan is being developed in response to the "Request for Additional Information, Combustion Engineering System 80+, Performance and Quality Evaluation Branch, Generic Safety Issue II.C.4; Reliability Engineering". The System 80+ PRA is being used as the primary resource for development of this RAP plan. Structures, systems, and components and reliability criteria in the RAP will be consistent with those in the PRA.

Question 721.1

The credibility of an HRA analysis is highly dependent on the mix of expertise in the analysis team. In this regard, please provide information on the makeup of the team that performed and reviewed the HRA portion of the System 80+ PRA.

Response 721.1

For the System 80+ PRA, C-E performed a preliminary HRA analysis consistent with the EPRI HCR model and the methods described in the Handbook of Human Reliability Analysis. The analysis team consisted of the systems analysts with assistance from engineers holding an SRO. Most of the HRA was based on generic System 80 information and generally accepted operating procedures for C-E designed plants(e.g., Emergency Operating Guidelines in CEN-152). C-E recognizes the NRC's concern with the limitations of this analysis. C-E is currently updating the System 80+ PRA. This update will include improved and detailed HRA.

Questions 721.2 - 721.17

[These questions address the manner in which human reliability was included in the System 80+ PRA]

Response 721.2 - 721.17

Combustion Engineering agrees with NRC's comments and will resolve them in the revised PRA. It is expected that C-E/NRC meetings in the interim will ensure that resolutions in the revised PRA are adequate. This updated HRA will use the most up-to-date methods to the extent possible and will draw on more recent human reliability data such as that provided via NUCLARR.

Response 721.2 - 721.17 (cont.)

A new subsection will be added to each event analysis section to describe the human actions and performance shaping factors applicable to each initiating event. In addition, Chapter 5 will be expanded to describe the quantification of each operator action credited in the PRA.