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Technical Report

Title: ABWR Probabilistic Evaluations

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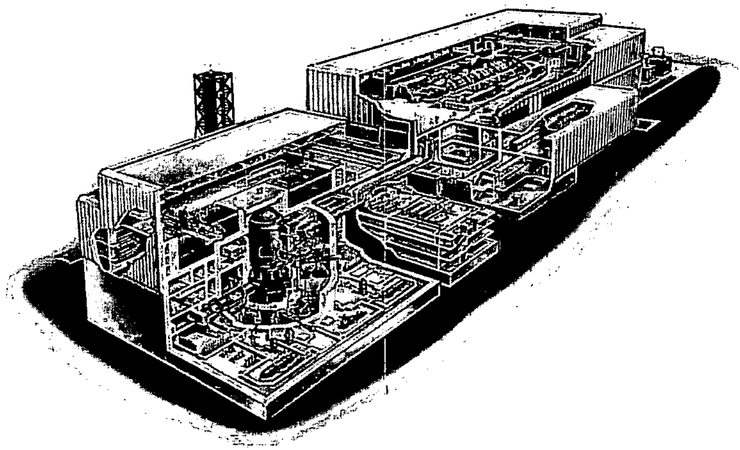
Technical Report

ABWR Probabilistic Evaluations

M.Ino
Approved by Nov 8, 2010
Safety & Dynamics Engineering Group

Toshiba Corporation
System Design & Engineering Department



Toshiba America Nuclear Energy ABWR Supplemental DCDRA Chapter 19D Documentation

Revision 0

August 2010

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Developed for

ETRANCO/MPR/Toshiba America Nuclear Energy

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Original Issue	AML/CLE	CLE/8-6-10	CLE/8-25-10	Note that all table of contents are aligned to the equivalent Chapter 19 documentation to allow for update to Chapter 19D.
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19D PROBABILISTIC EVALUATIONS

19D.1 INTRODUCTION

A focused-scope update of the original design certification PRA has been performed to address Toshiba ABWR design specific features and to gauge the impact of application ofcurrent PRA approaches on the existing design certification PRA metrics and overall conclusions. The focused-scope concentrated on the following areas of the PRA:

(1) Initiating event frequencies and component failure data

(2) Risk significant non-THERP-based human actions,

(3) Common cause failure treatment, and

(4) Propagation of revised Level 1 results through the developed Level 2 model.

(5) Shutdown accident sequence analysis

The scope for the PRA was specified by MPR Associates, Inc. based on direction from Toshiba. As documented in Reference 19D.2-3, changes from the original certified design were reviewed for their impact on the PRA. A two-step evaluation process was used by MPR to determine the impact of each design change. First, the design change was evaluated to determine if there was a change to a PRA-modeled System, Structure, and Component (SSC) that might affect the PRA results. If the level of detail is such that the change would not be "visible" in the PRA, then the SSC was not considered further. If the change would be "visible," then expert judgment was used to determine whether the change would have a risk-beneficial or a potentially risk-negative impact. Risk-beneficial design changes were not considered for further evaluation in the PRA because the design certification results would remain bounding. Retained risk-negative design changes are evaluated to quantify any change to the Core Damage Frequency and Large Release Frequency associated with the change. Of the changes from the original certified design, only the switch to a fan-cooled Ultimate Heat Sink was specified as needing to be included in the PRA model.

This Appendix provides a detailed compilation of the information gathered, as well as models and methods developed, during the course of performing the Toshiba Design Control Document Renewal and Amendment ABWR Probabilistic Risk Assessment. The overall methodology is summarized and analytic procedures are outlined. Data used in the analysis are presented and their sources identified.

Accident sequence event trees are presented and discussed and the basis for each branch point probability identified. Similarly, the ABWR containment event trees are provided, as is a detailed discussion of their bases. Fault trees for both front line emergency systems and vital support systems are presented and described, including the failure probability data used in their evaluation. Functional relationships are also illustrated by the inclusion of functional fault trees in Section 19D.6.

19D.2 MODELS AND METHODS DESCRIPTIONS

19D.2.1 SUMMARY OF METHODOLOGY

Methodology used in the ABWR PRA utilizes current methods for computing the frequency of core damage and radioactive release resulting from postulated accident sequences. Figure 19D.2-1 illustrates the basic procedure followed and defines the four major tasks of the analysis. Tasks I and II, frequency of core damage and radioactive release are discussed in this appendix and provide the input necessary to determine the magnitude and consequences of release, Tasks III and IV, which are discussed in Appendix 19E.

Fault trees were developed and evaluated for the major ABWR front line and support systems to determine the unavailability on demand of emergency core cooling and decay heat removal systems. Transient and loss-of-coolant accident (LOCA) events were consolidated into major accident event sequences which are described by the accident event trees of Subsection 19D.4. These event trees were used to calculate the frequency of core damage sequences using results from the fault trees of Subsection 19D.6, accident analysis success criteria, and detailed system analyses.

Outcomes of the event trees, other than successful termination of accident sequences, were transferred to containment event trees for further treatment to determine frequencies of radioactive releases to the environment. Containment event trees detailing the progression of potential accident sequences and fission product release paths are presented in Subsection 19D.5. These sequences were combined into appropriate accident classes by sorting transient, LOCA, loss of containment heat removal, and anticipated transient without scram (ATWS) events according to three criteria:

(1) the timing of the loss of containment integrity relative to core damage;

(2) the mass and energy discharge via safety/relief valves or drywell vents; and

(3) the timing of fission product releases.

Results of the containment event tree analyses provided the necessary input to model and assess fission product transport through the drywell and containment, calculate fission product release fractions associated with containment release paths, and determine potential consequences associated with each fission product release category. Details of these latter analyses are addressed in Appendix 19E.

19D.2.2 ANALYSIS OUTLINE

Figure 19D.2-2 diagrams the procedure for assessing frequency of core damage. Analysis begins with identification of events which could initiate an accident sequence and determination of their frequency. This process is discussed and the results are presented in Subsection 19D.3. Event trees were developed to trace the course of each accident sequence. These trees detail each step in each sequence, including the associated conditional probability, that could lead either to core damage or to successful termination of the event without core damage. A separate event tree was constructed for each initiating event analyzed. These trees are presented and discussed in Subsection 19D.4.

Fault trees were developed for each of the primary emergency core cooling and heat removal systems as well as critical support systems. These trees documented in Subsection 19D.6, provided detailed models to determine the ways in which each system could fail to provide its necessary function. The fault trees include the probability of component failure, operator action, testing and maintenance unavailabilities, interdependencies, and necessary support services such as electric power and service water.

To determine probabilities to apply to each branch of the various event trees, front line and support system trees were combined into functional fault trees, i.e., core cooling or heat removal, for evaluation. This was done to properly account for dependencies and commonalities between systems. Functional fault trees are included in Subsection 19D.6.

Accident sequences identified and evaluated in the event trees were examined and classified on the basis of similarity of timing, potential for fission product release, and containment response. Accident sequence classes used in the analysis are described in Subsection 19D.5. Total frequency of core damage and the major contributors are discussed in Subsection 19D.3.1. Assessed Frequency of core damage by accident class is presented in Table 19.3-4.

Figure 19D.2-3 illustrates the procedure for assessing frequency of fission product release from the core and from the containment. Analyses of core damage phenomenology provided the bases to identify and assess the probability and consequences of each of the different release mechanisms. These analyses are discussed in Appendix 19E. An analyses of the containment is also provided to assess the probability of loss of containment integrity. The containment analysis is discussed in Appendix 19F.

Based on the core damage and containment analyses, potential release mechanisms and pathways were identified. These are also discussed in Appendix 19F. Containment event trees were constructed to depict the sequence of events beginning with either core damage or loss of the heat removal function and leading to fission product release or safe termination of the event. Containment event trees are discussed in Subsection 19D.5. Potential containment release sequences were grouped into release categories which are discussed in Subsection 19D.5. Through evaluation of the containment event trees and the accident sequence classes, Table 19.3-4, a summary of release frequencies by accident class, was constructed.

Figure 19D.2-4 illustrates the procedure for assessing the quantity of fission products released. Source term magnitude is discussed in Subsection 19E.2.

The final PRA process, evaluating external consequences of the accident sequences, is depicted in Figure 19D.2-5. Considerations involved in the determination of external fission product transport and subsequent consequences of these releases are addressed in Subsection 19E.3. Application of the assessed frequency and magnitude of release to the external transport model for each release category yields the expected value of exposure and the consequent health effects.

19D.2.3 FAULT TREE-EVENT TREE ANALYSIS

Given an initiating event, probabilities associated with the accident sequences were evaluated in fault tree and event tree logic models. The CAFTA personal computer program, as described in Reference 19D.2-1, was used in all fault tree evaluations to probabilistically combine equipment and human failure events contributing to each accident sequence.

Mean values were used throughout the analysis. The fault and event tree models account for interdependencies including the following:

(1) component commonality at the system level, such as common initiating signals;

(2) common divisional services, such as common electrical buses or common service water;

(3) system dependencies, such as the ADS dependency on operability of at least one low pressure ECCS pump;

(4) the loss of either on-site or off-site power; and

(5) human errors.

Equipment failure rate inputs are generally at the component level, i.e., pump, valve, sensor, motor, relay, or circuit breaker. In a few cases, lower level data such as the failure of a lubrication system were used. In other cases, higher level reliability data were used for such items as diesel generators, MSIVs, turbines, safety relief valves, and heat exchangers.

Component reliabilities were assumed to be exponential, i.e., failure rates were treated as constants over periods of interest. This treatment assumes that components are properly tested and qualified when installed, are properly maintained, and that they are replaced before wearout effects begin to occur. These assumptions are validated by the design and maintenance requirements for ABWR equipment.

Standby component failure rate data are applied on either a per demand or elapsed time since last test basis as appropriate. For some components, i.e., diesel generators, available data provide the basis for determining per demand unavailability directly.

Emergency procedures for operator action are based upon Generic Emergency Procedure Guidelines.

Throughout the analysis, operator error rates are included from RSC 08-06 (Reference 19D.2-1). Updated operator error rates were obtained for certain operator actions that were deemed important. This is documented in RSC CALKNX-2010-0506 (Reference 19D.2-2).

Calculated system and function demand unavailabilities are the composite of individual component failure probabilities, human action failure probabilities if applicable, unavailabilities of vital support systems, unavailability of electric power, and system unavailabilities due to on-line maintenance. Detailed fault trees for each front line and support system are presented in Subsection 19D.6.

19D.2.4 REFERENCES

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19D.2-2 Shehane, M., Update of Selected Human Action to support the Toshiba DCDRA, RSC Engineers, Inc., RSC CALKNX-2010-0506, July 2010.

19D.2-3 Kaufmann, S., Documentation of Toshiba ABWR Design Features to Include in the PRA Update, MPR 1230-1002-SK01, October 2010.

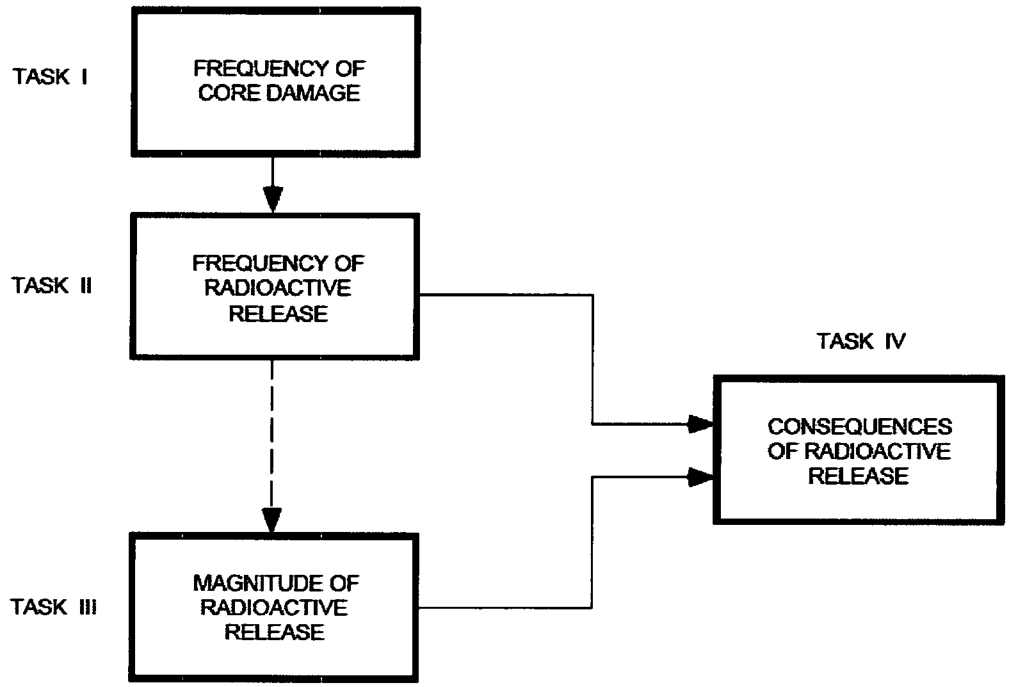


Figure 19D.2-1
Major Tasks of the PRA

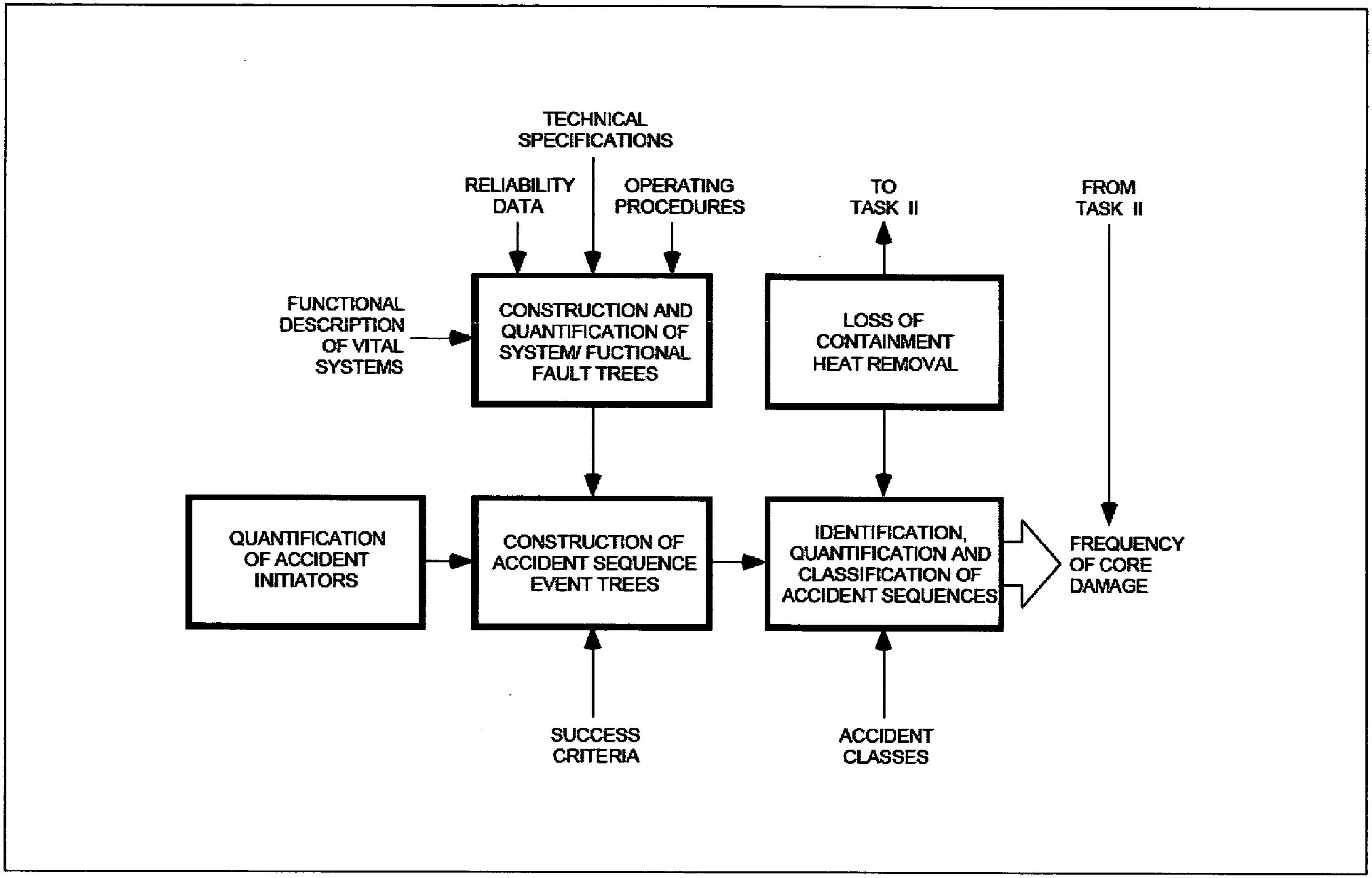


Figure 19D.2-2
Determination of Core Damage Frequency (Task I)

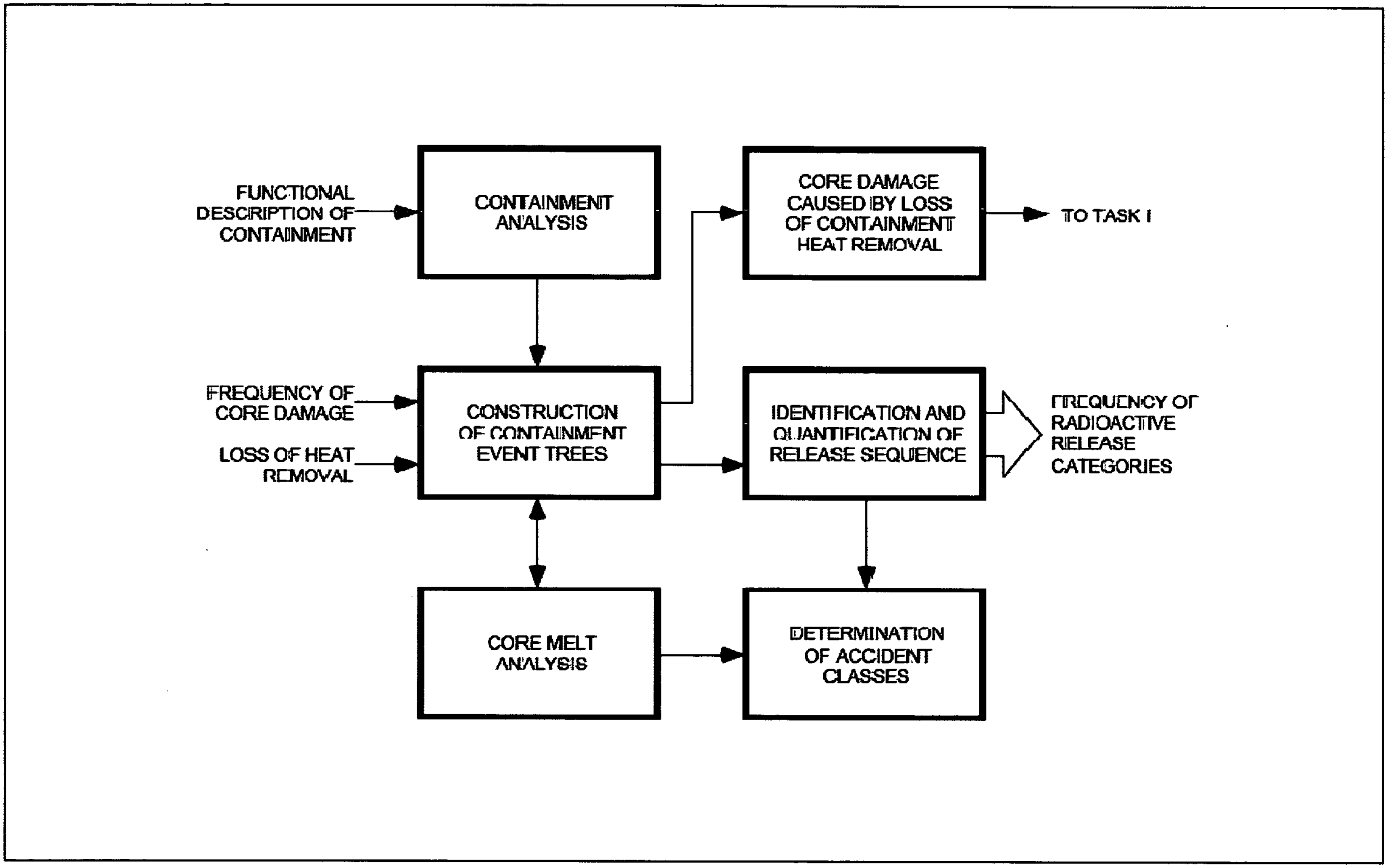


Figure 19D.2 3
Determination of Radioactive Release Frequency (Task II)

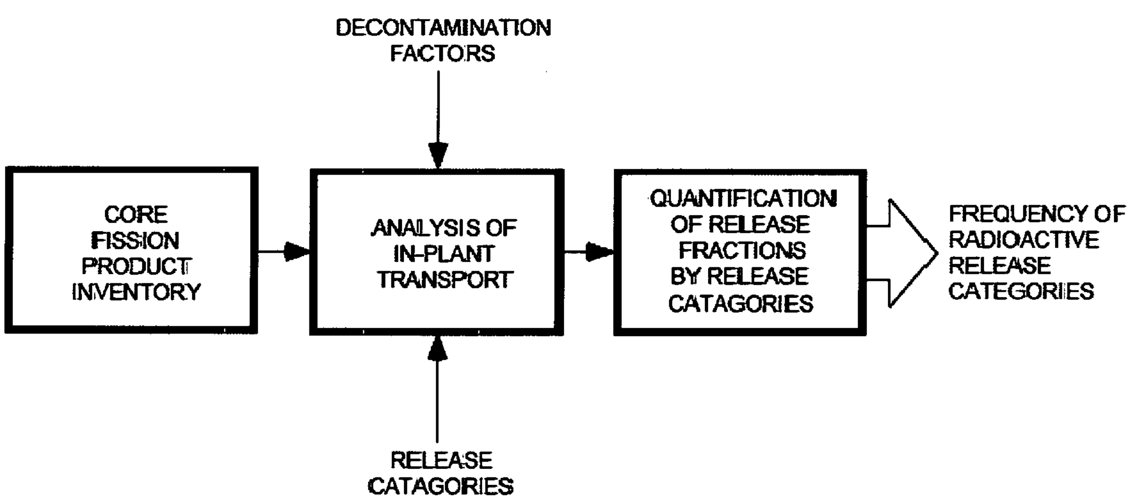


Figure 19D.2-4
Determination of the Magnitudes of Radioactive Release (Task III)

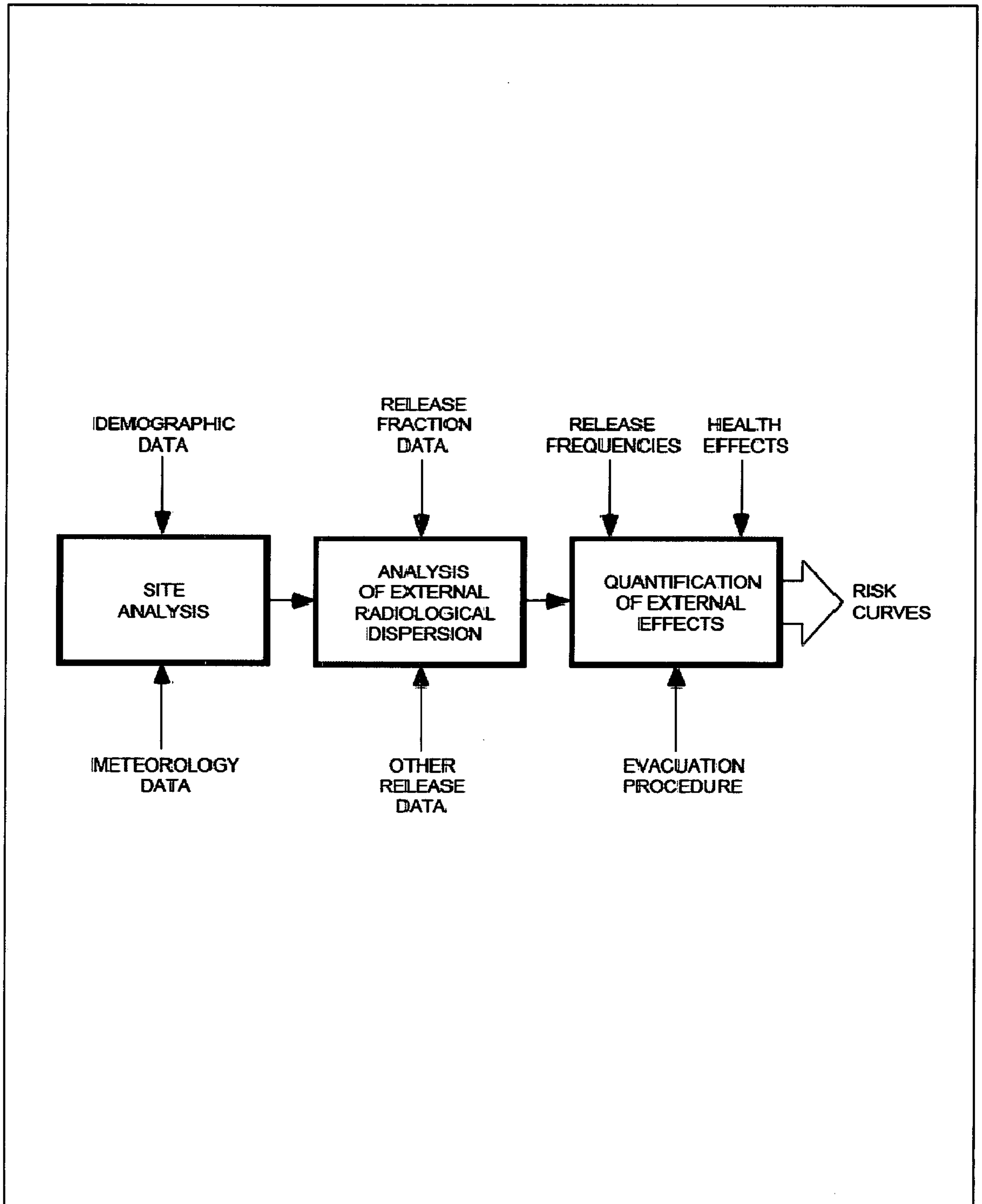


Figure 19D.2-5
Determination of Consequences of Radioactive Release (Task IV)

19D.3 INPUT DATA

19D.3.1 INITIATING EVENT FREQUENCIES

Initiating events and expected frequencies used in the ABWR PRA are presented in Table 19D.3-1. These values incorporate the latest information available in NUREG/CR-6928 (Reference 19D.3-1). From the documentation in RSC 08-06 (Reference 19D.3-2), four types of initiating events of dominant significance have consistently been identified:

(1) Shutdowns;

(2) Transients (scrams);

(3) Losses of offsite power; and

(4) Loss of coolant accidents (LOCAs).

These are the primary initiating events investigated in this analysis.

19D.3.1.1 Manual Shutdowns

Planned and unplanned manual shutdowns are controlled activities which present very mild challenges to the plant and seldom place demands on any standby safety equipment. These events are included in the analysis because of their frequency and since they do represent changes in operating states which could result in the initiation of accident sequences. On the basis of information provided in RSC 08-06 (Reference 19D.3-2), a value of one event per year was judged appropriate to present an average of all annual scheduled and unscheduled controlled shutdowns in a mature ABWR plant.

19D.3.1.2 Transients

The transient initiating events included in this analysis encompass all types of unplanned scrams that have been encountered at operating BWRs. The factors in the initiating event that would significantly affect the ensuing sequence of events are the following:

(1) Whether or not the reactor has been isolated due to MSIV closures or failures of turbine bypass valves;

(2) Whether or not feedwater has been lost;

(3) Whether or not offsite power has been lost; and

(4) Whether or not an automatic scram signal is generated.

These four conditions are encompassed by the four transient initiating events used in this analysis:

(1) Non-isolation event;

(2) Isolation/loss of feedwater event;

(3) Inadvertent (stuck) open relief valve; and

(4) Loss of offsite power.

19D.3.1.2.1 Non-Isolation Events

All events that do not result in an isolation of the reactor due to closure of MSIVs or failure of all turbine bypass valves are included in this group. A turbine trip with bypass is selected to represent the plant response for this group due to its relative severity and frequency.

19D.3.1.2.2 Isolation/Loss of Feedwater Event

All events that result in closure of the MSIVs except LOCAs and losses of offsite power are included in this group, as are turbine trip and load rejection transients with assumed failure of all turbine bypass valves. In addition, since the challenge to ECCS for loss of feedwater events is very similar to that for MSIV closure events, these events are also included in this group.

19D.3.1.2.3 Inadvertent (Stuck) Open Relief Valve (IORV)

This event begins with one or more relief valves opening and remaining open while the reactor is under otherwise normal operating conditions. It is the only initiating event considered where there is no immediate automatic scram signal.

19D.3.1.2.4 Loss of Offsite Power

The loss of offsite power event is defined as a complete loss of power from the grid. It requires the startup and use of emergency diesel generators or the combustion turbine generator to provide power to plant systems until external power to the plant is recovered. Offsite power loss could be due either to grid failure or failure of plant connections to the grid.

The expected loss of offsite power frequency and outage time distribution were derived from data presented in NUREG/CR-6890 (Reference 19D.3-4). The total frequency of this event is 3.59E-2/rcry. The data, documenting 24 loss of offsite power events, are based upon 724.3 site years of operation.

The loss of offsite power initiating event is broken into several intervals based upon length of time required for recovery. This is documented in RSC CALKNX-2010-0501 (Reference 19D.3-5).

19D.3.1.3 Loss of Coolant Accidents

Three types of coolant accidents (LOCAs) were considered for the ABWR: LOCAs within containment; LOCAs external to containment; and LOCAs in low pressure piping systems interfacing with high pressure piping systems connected to the reactor pressure vessel (RPV), also known as interfacing LOCAs.

19D.3.1.3.1 LOCAs Within Containment

As in a number of previous PRAs, the ABWR evaluation considers three sizes of inside containment LOCAs (small, medium, and large) which are established on the basis of reactor core cooling success criteria. The small break LOCA category represents break sizes characterized by very slow or no vessel depressurization and small gradual inventory loss similar to a transient. The medium break category differs from the small in that reactor coolant inventory is lost at a substantially greater rate such that RCIC has insufficient capacity to maintain water level. For the medium break category, the initial depressurization rate is sufficiently slow to require manual safety relief valve actuation (either ADS or non-ADS) for rapid depressurization. The large break LOCA is of sufficient size to provide rapid depressurization without the use of safety relief valves.

LOCA initiation frequencies corresponding to ABWR success criteria and presented in Table 19D.3-1 are based upon NUREG/CR-6928 data (Reference 19D.3-1). Expected frequencies for the three break size categories are a matter of considerable uncertainty.

For the ABWR analysis, breaks or leaks that are so small that they do not result in reactor trip on high drywell pressure are not considered as LOCAs.

19D.3.1.3.2 LOCAs External to Containment

The LOCAs external to containment were studied and the results are discussed in Subsection 19E.2.3.3. The risk associated with external LOCAs is calculated to be a very small fraction of the total risk of severe accidents evaluated in the PRA. Therefore, the external LOCAs are not analyzed separately as initiating events for the PRA.

19D.3.1.3.3 LOCAs in Interfacing Systems

Piping and components in the interfacing systems are designed for pressures which are much lower than the normal RPV pressure. However, certain equipment failures or operator errors can subject these pipes to the normal RPV pressure and cause a potential for LOCA in the interfacing system. To prevent such LOCAs, piping in interfacing systems for accepted practical regions has been redesigned on an ultimate rupture strength basis to withstand normal RPV pressure. Therefore, these systems are not expected to rupture and the interfacing system LOCAs, which are discussed in Subsection 19B.2.15, are not a concern for the ABWR plant. So interfacing LOCAs are not analyzed separately as initiating events in the PRA.

19D.3.1.4 Other Potential Initiators

Several other potential initiating events were evaluated for inclusion in the PRA. These events, and the reasons for not including them in the baseline PRA calculation, are discussed below.

19D.3.1.4.1 Reactor Pressure Vessel Failure

Disruptive failure of the RPV is an event that can be considered as being nearly incredible. There is no basis for estimating any specific probability or expected frequency of occurrence of such an event. Upper 99% confidence limits in order of 10^{-6} to 10^{-7} per year have been postulated but without firm statistical basis, since there has never been a disruptive failure of a reactor pressure vessel.

A survey of existing literature, data and reactor pressure vessel expert opinion yields the following pertinent points:

(1) RSC 08-06 (Reference 19D.3-2) states that the upper limit (99% confidence) probability of a disruptive RPV failure event in any one nuclear reactor during any service year falls within the range of 1E-7 to 1E-6 and the mean value of this probability would be expected to even smaller. This conclusion was based on 725,000 vessel-years of service in U.S. fossil-fueled power plants

without a disruptive failure. It quotes "...the estimated failure probability (for nuclear reactor vessels) may be reduced by an additional factor of 10 to 100 based on the detailed investigation of the influence and scheduling of the periodic inspections...".

(2) The "leak before break" phenomenon is a key consideration which justifies a failure rate less than 10^{-7} per vessel year (Reference 19D.3-2). RSC 08-06 further states that the "pressure vessel will have a considerable margin to failure by (a) brittle fracture, even with large postulated initial flaws and (b) that leak-before-break capability is maintained even after a LOCA." This means that long before a crack could propagate to the point that a disruptive failure could occur the crack would propagate through the vessel wall and be detected due to significant leakage. The leak detection system would detect the existence of leaks and allow shutdown of the reactor to avoid propagation of the crack and vessel failure.

(3) Reactor vessels are subjected to periodic inspections, in accordance with Section XI of the ASME Code. This inspection is generally more intensive than that for non-nuclear vessels, and consists of an ultrasonic inspection of weld joints before the vessel goes into service and every 10 years thereafter, supplemented by surface inspections (visual, liquid penetrant test and magnetic particle test).

(4) In recent years, a significant amount of research has been conducted in the area of pressure vessel integrity, and the factors relating to material specifications which play a key role in material embrittlement have been identified and well understood. The RPV material specifications and the RPV irradiation levels for the ABWR produce nil ductility temperature shifts that make the potential for nil ductility failures negligible.

(5) Recent work on the ABWR design evaluated large RPV bottom head breaks. Structural evaluations showed that loads on equipment and structures were insufficient to cause loss of structural integrity. These results show that the severe accident response for RPV failures would be no worse than for a large break LOCA severe accident.

(6) Reactor vessels are designed with a higher degree of protection from pressure transients and temperature events than are non-nuclear vessels. This higher degree of protection is assured by virtue of design measures, including overpressure relief devices and operational control procedures.

(7) Reactor vessels are designed and constructed in accordance with Section III of the ASME Code. These rules are more elaborate than the rules of Sections I and VIII, which are used for non-nuclear vessels.

(8) Reactor vessels are operated in accordance with the limitations specified in NRC license technical specifications and no such requirements are imposed on non-nuclear vessels.

Based on the above considerations, it is concluded that while it is not possible to quantify the probability of RPV failure with great precision, the failure probability of an RPV rupture for the ABWR plant is so low that its explicit inclusion in this analysis would not significantly impact the results. Furthermore, the RPV failure modes that are mechanistically plausible would produce consequences similar to the higher probability LOCA events because of the leak-before-break phenomenon.

19D.3.1.4.2 Loss of Main Control Area Envelope HVAC

The HVAC emergency cooling water (HECW) System delivers chilled water to the control building safety-related equipment area cooling coils, reactor building safety related electrical equipment area cooling coils, and the main control area envelope served by the control room habitability area cooling coils during shutdown of the reactor, normal operating modes, and abnormal reactor conditions. The HECW System consists of three mechanically separated divisions, A, B, and C. Each HECW division provides cooling to the control building safety-related equipment area and the reactor building safety-related electrical equipment area in its division. Also, either division B or C can independently cool the main control area envelope. Power is supplied to each division from independent Class 1E sources.

Each division of HECW consists of two parallel pumps, and refrigeration units, instrumentation, and distribution piping and valves to the cooling coils. System configurations for each division are illustrated in Figure 19D.3-1, 19D.3-2 and 19D.3-3. The HECW system is capable of removing all heat loads with four of the six units running and one pump and refrigerator unit from division "A" in standby mode and one of the four pump and refrigerator units from divisions "B" and "C" in standby. At any given time the division with two pumps and refrigeration units in operation provides cooling to the main control area envelope. The design philosophy is that if one of the refrigerators or pumps fails in this division, the standby refrigerator will automatically start and provide main control room area envelope cooling while the reactor building safety-related electrical equipment area and the Control Building Safety-related Equipment Area cooling requirements will continue to be met by the remaining refrigerator in the affected division.

Cooling water for the HECW refrigerators is provided by the corresponding division of the Reactor Building Cooling Water (RCW) System which in turn rejects heat through the Reactor Service Water (RSW) System to the ultimate heat sink. Each division of the RCW and RSW consists of two parallel trains interfacing through three heat exchangers.

RCW and RSW system design capacities are such that one RCW train, one RSW train, and two of the three RSW/RCW heat exchangers in operation are sufficient to successfully meet equipment and control area envelope heat removal requirements.

It is conservatively assumed that loss of one HVAC division results in the loss of cooling for one control building safety-related equipment area, and for one reactor building safety-related electrical equipment area, which leads to the failure of the ECCS equipment of that division. Accordingly, system fault trees modeling the loss of the HVAC function in each division have been incorporated directly into the ABWR functional fault tree and sequence evaluations.

The potential for failure of both control room HVAC divisions as an initiating event and the expected consequences have also been investigated. Frequency of control room HVAC failure was estimated by calculating the product of the operating HVAC division random failure frequency, assuming 80% plant capacity factor, and the conditional probability that the standby division will fail to start and run, given failure of the operating division. The operator is certain to notice the loss of HVAC serving the main control room and is expected to initiate a manual shutdown, which then becomes the initiating event. The scenario which then follows can be conservatively approximated to that for the control room fire risk screening analysis presented in the Appendix 19M Fire Protection Probabilistic Risk Assessment. Conservatism is introduced by the fact that, following HVAC failure, the operator has time to take many recovery actions, and control systems continue to operate for a significant length of time, whereas in case of a fire, control room functions can be lost in a relatively short time.

The sequence of events is illustrated in the event tree of Figure 19D.3-4. The assumption is made that the only ECCS equipment available to respond to this transient event is that which can be controlled from the remote shutdown panel. This capability includes operation of HPCFB, RHRA, RHRB, and four SRVs for reactor depressurization. In addition, RCIC and the Fire Protection System are available to provide core cooling, since each is capable of remote operation independent of the control room, and thus is unaffected by the HVAC failure.

No information is currently available regarding qualification temperature limits of essential equipment in the control room, the rate of temperature increase in the control room following HVAC loss, or the nature of and times available to initiate other possible recovery actions. Continued operation of the feedwater pumps (event tree node Q) will allow continued safe operation of the reactor while providing a reasonable time period to recover the HVAC. Consequently, the probability of failing to recover HVAC or to initiate alternate means of achieving adequate control room cooling was conservatively estimated to be 0.1. If feedwater is not available, no credit has been taken for recovery of HVAC.

On the above basis, core damage frequency resulting from the loss of control room HVAC initiating event was calculated to be less than 1.0E-8 per reactor-year. This result represents a conservative estimate within the context of the assumptions and constraints presented, and a more realistic analysis should produce an even lower CDF. Therefore, this potential initiator was dismissed from further consideration.

19D.3.1.4.3 Loss of a Single AC or DC Bus

Loss of a single AC or DC bus will not cause a reactor scram. Direct current power is used to supply channel sensors for each division of the reactor protection system. Scram signals are 2-out-of-4 channels of logic, with each of four DC divisional power supplies providing power to one of the four channels. Loss of one division of DC power will result in a trip in only 1-out-of-4 of the channels, therefore the 2-out-of-4 criterion for scram would not be met. Loss of a single DC power division will not result in an RPS scram signal. Therefore, loss of a single DC bus is not analyzed as a initiator in the PRA.

Alternating current power is supplied to plant equipment at and above 480 volts through three divisions. There are four divisions of 120 volt AC power. Division II of 480 Volt AC powers the battery charger for Division IV of 120 Volt AC power, which obtains an uninterruptible power supply from the batteries.

Scram solenoid groups A and B are powered by divisions II and III, respectively, of 120 volt vital AC power. Loss of division II or III of AC power will not result in scram, since both the A and the B scram solenoids must be de-energized to cause scram insertion of control rods. Loss of power to one set of solenoids would not result in a scram signal, so there would be no control rod insertion.

Power from AC buses is also supplied to reactor internal pumps (RIPs), to control rod drive (CRD) pumps, to feedwater, condensate and circulating water pumps, to service water and cooling water pumps, to HVAC system pumps, to cleanup water pumps, to drain pumps, to vacuum pumps, to battery chargers, to building fans, to gas compressors, and to many other electrical panels and components during plant operation. Loss of a single AC power bus could result in power loss to as many as three RIPs. This would result in reactor power reduction to approximately 90%, at which point power operation could continue without scram.

Loss of a single bus of AC power to any of the other equipment noted above will result in local impact on the system but little, if any, impact on plant operation. Systems impacted are designed with redundancy in that major components, such as feedwater pumps and CRD pumps, are each powered by different AC power divisions, load groups and/or buses. For example, each of three feedwater pumps is on a separate 6.9kV bus, each of three circulating water pumps is on a separate 6.9kV bus, two of four condensate pumps are on separate 6.9kV buses and the other two pumps are on a third 6.9kV bus, each of two CRD pumps is on a separate 6.9kV bus. The five normal cooling water refrigerators are distributed among these separate 6.9kV buses, each of three turbine building service water pumps is on a separate 6.9kV bus, and each of three turbine building cooling water pumps is on a separate 6.9kV bus. Similarly, reactor building cooling water pumps 1A and 1D (loop A) are on one 6.9kV bus, pumps 1B and 1E (loop B) are on a different 6.9kV bus, and pumps 1C and 1F (loop C) are on a third 6.9kV bus.

The four HVAC emergency cooling water (HECW) system refrigerators associated with control room cooling are on two different 480V buses so that loss of one bus will not disable all control room cooling.

Loss of a single component by loss of power is not expected to cause scram and usually will not result in power reduction. Loss of a single feedwater pump or condensate pump may result in a temporary power reduction, but full power can be achieved with the remaining two pumps. In some cases (such as loss of one circulating water pump on a hot day), a reduction in power might result. Loss of the two condensate pumps that are on one bus is expected to result in a loss of power without a scram.

Standby equipment that requires AC power, such as pumps for the residual heat removal (RHR) system and the high pressure core flooder (HPCF) system, is not normally operating and would not react directly to a loss of AC power. Thus, loss of a single bus of AC power to such equipment would not result in scram.

Other equipment supplied by AC or DC power either has adequate redundancy to avoid scram or is not related to scram signals. No single bus failure, AC or DC, will generate a scram signal or cause control rods to scram.

19D.3.1.4.4 Loss of One Division of the Reactor Service Water System

Loss of a single division of the reactor service water (RSW) system would not cause scram.

The RSW system is divided into three divisions which cool the three divisions of the reactor building cooling water (RCW) system. The RCW system cools a number of plant components. The RSW does not directly cool any components other than the RCW system.

Each of two RCW divisions cools five of the 10 reactor internal pumps (RIPs). Loss of either of these RCW divisions will cause its five RIPs to runback and trip from high RIP motor cooling water temperature. The ABWR is not to be licensed to operate with fewer than seven RIPs operating, so loss of five pumps would require shutdown. Thus, loss of one RSW division that provides cooling to five RIPs through the RCW division will lead to reactor shutdown, but not to a scram. Manual reactor shutdown has been analyzed (Subsection 19D.3.1.1).

The RCW system and the heating, venting and air conditioning (HVAC) normal chilled water system are used to cool the drywell. Drywell cooling is accomplished by three coolers which are in turn cooled by two RSW and two RCW divisions. In the most severe loss of a single RSW division two of the drywell coolers would be uncooled.

A study was made of single failure of equipment (loss of a division of both the RSW and RCW systems) during a loss of off-site power (LOOP). Drywell temperature increased from 330 to 348 K (57° to 75°C) eight hours after LOOP. For components in the drywell the upper limit for long term exposure is 353 K (80°C), so there would be no component damage requiring or causing scram for loss of a single RSW division. There would also be ample time for operator action to restore drywell cooling. The pressure increase resulting from the calculated drywell heating would not be high enough to cause scram. It is concluded that a single RSW division failure would not result in drywell conditions leading to scram.

Some of the equipment cooled by the RCW system does not require cooling during normal operation. This includes the residual heat removal (RHR) system, the reactor core isolation cooling (RCIC) system, the emergency diesel generators, the flammability control system (FCS), the high pressure core flooder (HPC F) system, and the standby gas treatment system (SGTS). For these items the loss of one division of RCW would not cause or require scram.

None of the other equipment cooled by the RCW system has functional requirements such that loss of one RCW division (or one RSW division) would result in reactor scram.

Such equipment includes the heating, ventilating and air conditioning (HVAC) system, the cleanup water (CUW) system, the fuel pool cooling (FPC) system, the containment air monitoring (CAM) system, the high conductivity waste (HCW) system, and the instrument air and service air systems. These systems operate during plant operation, but they have adequate redundancy to assure that plant operation can continue without scram in event of loss of cooling from one RCW division.

Therefore, loss of a single RSW system division is not analyzed as an initiator in the PRA.

19D.3.1.4.5 Reactor Vessel Water Level Instrumentation Failure

Failure of a single water level instrument or instrument channel would not cause scram.

The reactor pressure vessel (RPV) water level instruments are provided with four independent and separate divisions, electrically and mechanically. Water level signals are combined in 2-out-of-4 logic for reactor scram, so a failure in or of a single channel does not result in a scram, so there is no control rod insertion.

Partial loss of drywell cooling as a result of failure of one of the three plant service water systems could result in an average drywell temperature of 331 K (58° C) and local temperatures of 348 K (75° C) in the CRD area and 339 K (66° C) elsewhere in the drywell. This temperature increase would affect readings on all four water level instruments, because of piping inside the drywell, but not so much that a low water level trip would result.

Therefore, failure of a single water level instrument is not analyzed as an initiator in the PRA.

19D.3.1.4.6 Turbine Building Closed Cooling Water System

Total loss of the turbine building closed cooling water (TCW) system can result in scram. The TCW system provides cooling for the generator stator cooling water, and loss of this TCW cooling will lead to higher temperatures of the stator cooling water. As the stator temperature increases it will first give a high temperature alarm. The operator will act to reduce reactor power and will try to reestablish TCW system operation. If corrective action is not prompt and effective, the stator temperature will soon reach its high temperature trip and cause the generator to trip. This load rejection event will result in turbine trip and, at high power, the turbine trip will cause scram. Turbine trip is included in the basic initiating event frequency.

The TCW system has multiple pumps. If only one of these pumps is lost, the standby pump will start and there will not be transient temperature increases sufficient to result in equipment trips or scram. As noted in Subsection 19D.3.1.4.3, loss of a single AC bus will result in loss of no more than one TCW pump. Because of this redundancy of active TCW system equipment, the probability that the entire system could be disabled is low.

Total loss of TCW system cooling would impact cooling for other systems that are required for continued plant operation. This includes the feedwater pump adjustable speed drives, condensate pump drives, circulating water pump drives, transformers, and the isolated phase bus duct. Such cooling losses would require prompt operator action to either shut the plant down or restore TCW system operation. Otherwise, reactor scram could soon occur from high or low RPV water level, from turbine trip caused by low vacuum, from electrical system troubles, or from gradual loss of instrument air pressure.

Loss of TCW system cooling would have no impact on any plant safety equipment. Such equipment is cooled by the reactor building cooling water (RCW) system. Even if a scram were to occur, the net effect would be a negligible increase in the frequency of events already included in the PRA. Therefore, total loss of the TCW system is not analyzed as an initiator in the PRA.

19D.3.1.4.7 Trip of Circulating Water Pumps

As part of internal flood protection in the turbine building, instrumentation has been added to trip the circulating water pumps when a flood is detected in this building.

Reliable instrumentation built with redundancy is provided to assure a high probability of tripping the pumps on demand while assuring a low frequency of inadvertent trip of the pump. The probability that this instrumentation will fail to trip pumps is 1.7E-4 per year, including common mode failure.

An inadvertent trip of all circulating water pumps could cause scram because of loss of vacuum in the main condenser. It is estimated that the frequency of such trips is 1.8E-4 per year, which is smaller than similar initiating events already included in the internal event PRA. It is therefore concluded that no separate analysis is needed for including the reactor trip initiated by the inadvertent trip of a circulating water pump.

19D.3.1.4.8 Loss of Instrument Air

Total loss of instrument air for a prolonged duration will result in reactor scram because air pressure is required to keep scram valves closed on the control rod drives. Loss of pressure allows scram valves to open, and rods will be driven into the core by hydraulic pressure. Instrument air pressure is also required to keep MSIVs open, so loss of air pressure would cause MSIV closure which would also result in scram.

Instrument air is supplied by the lead air compressor which operates as needed to maintain pressure in a large accumulator. The lead compressor is normally off, and it starts and takes up load when accumulator pressure drops. If the lead compressor fails to start or to continue operation, the second, standby compressor will start automatically and operate to supply system needs. The operators are alerted by alarm if the lead compressor fails to start on demand so they can take corrective action.

If both compressors fail, pressure in the system will decay through air leakage and eventually cause a low pressure alarm to signal need for operator corrective action.

Operators can manually connect the service air system (which has two compressors) to replace any failed instrument air compressors. If leakage continues, eventually the scram valves will open and scram will occur. Also, MSIVs would close because of low air pressure and give a scram signal. If operators recognize that they will not be able to restore air pressure before scram, they may manually shut the reactor down without scram.

In event of a loss of offsite power, one instrument air compressor is automatically switched to the combustion turbine generator bus for power and the other can be switched to the CTG manually. They can both be manually transferred to the diesel generator bus, if necessary.

Nitrogen is used to open the safety/relief valves (SRVs) of the automatic depressurization system (ADS), so complete loss of air would not impact SRVs or the ADS.

Because of the backup compressor and the large accumulator, the loss of ABWR instrument air is expected to be a low probability event. Experience from operating BWRs shows that loss of instrument air caused, from 1983 to 1987, fewer than 0.05 scrams per plant year (A Risk-Based Review of Instrument Air Systems at Nuclear Power Plants, NUREG/CR-5472). Average U.S. BWR scram frequency during that period was greater than three scrams per year. Since 1987, efforts at scram frequency reduction have significantly reduced scram frequency from all causes. Scram frequency for U.S. BWRs in 1992 was below 1.3 scrams per year. For the ABWR the frequency of scrams resulting from the loss of instrument air is expected to be very small (<0.02 per year).

The loss of instrument air event is very similar to an isolation event (which is analyzed in the PRA with an event frequency of 0.293/rcry). Because of this, and since no other safety system needed for mitigation (such as ADS) is degraded significantly by loss of air, this event is judged to be already included in the isolation event analysis. It is therefore concluded that no separate analysis is needed for including in the PRA the reactor trip initiated by the loss of instrument air.

19D.3.1.5 Initiating Event Contribution to CDF

Table 19D.3-5 summarizes individual contributions to CDF from initiating events that were not screened out for the ABWR. The table shows that station blackout is the dominant contributor to CDF. Greater than 80% of CDF is related to the three station blackout initiating events. There are also slight contributions from the loss of offsite power events and the loss of feedwater event.

19D.3.2 GENERIC COMPONENT DATA

Applicable component failure rate data accompany each system fault tree presented in Subsection 19D.6. These data are primarily the values used in RSC 10-02 (Reference 19D.3-4). They have been collected from a number of primary sources and have been screened and modified appropriately for application to the ABWR analyses. All of the values represent mean values and have been used in the analyses as single-point best estimate values.

19D.3.3 HUMAN ERROR PROBABILITIES

Human error probabilities used in this analysis are presented in the applicable component failure rate data tables which accompany each system fault tree presented in Subsection 19D.6. They were taken predominately from the ABWR PRA (Reference 19D.3-2) for which they were collected from various other sources and modified, as appropriate, for the ABWR DCD application. More recent studies suggest that these values may be somewhat conservative. Updated values for selected operator actions of high importance were obtained and this process is documented in RSC CALKNX-2010-0506 (Reference 19D.3-6).

19D.3.4 MAINTENANCE AND TEST UNAVAILABILITIES

Equipment maintenance or test unavailabilities used in the initial ABWR PRA submittal were were based upon BWR experience. Consequently, T&M values for RCIC, HPCFB, HPCFC, RHRA, RHRB, and RHRC were each set to 2% (Subsection 19D.9.1) in the PRA model as shown in Table 19D.3-2. The final calculated CDF of 9.8E-8 reflects inclusion of these values. Sensitivity of CDF to ECCS T&M outage times is summarized in Subsection 19D.9.

19D.3.5 RECOVERY OF OFFSITE POWER AND DIESEL GENERATOR RESTORATION

Table 19D.3-3 represents the probability of losing offsite power as a function of time, given LOOP. Table 19D.3-4 presents DG recovery times for a station blackout given a failure of the three DG trains. The offsite power recovery probabilities and diesel generator values were developed from data presented in Reference 19D.3-4.

19D.3.6 REFERENCES

19D.3-1 Eide, S.A., Wierman, T.E., Gentillon, C.D., et al., Industry Average Performance for Components and Initiating Events at U.S. Commerical Nuclear Power Plants. USNRC, NUREG/CR-6928, February 2007.

19D.3-2 Eddy, C., Establishment of Model to Evaluate Plant Specific Changes, Reliability and Safety Consulting (RSC) Engineers, Inc. RSC 08-06, April 2010.

19D.3-3 Eide, S.A., Gentillon, C.D., Wierman, T.E., et al. Reevaluation of Station Blackout Risk at Nuclear Power Plants: Analysis of Loss of Offsite Power Events: 1986-2004. USNRC, NUREG/CR-6890, December 2005.

19D.3-4 Lee, A.M., Documentation of the RSC Generic Database for PSA Studies. RSC Engineers, Inc., RSC 10-02, April 2010.

19D.3-5 Lee, A.M., Update of the Toshiba DCD PRA MOR Component Failure and Initiating Event Data to Support the Toshiba DCDRA, RSC Engineers, Inc., RSC CALKNX-2010-0501, Revision 1, July 2010.

19D.3-6 Shehane, M., Update of Selected Human Action to support the Toshiba DCDRA, RSC Engineers, Inc., RSC CALKNX-2010-0506, July 2010.

Table 19D.3-1
Initiating Events Frequencies

Initiating Event		Frequency (/rcry)
Manual Shudown		1.06E+0
Isolation/Loss of Feedwater		2.93E-1
Loss of Heat Sink	1.97E-1	
Loss of Feedwater	9.59E-2	
Turbine Trip		8.30E-1
Inadvertent Open Relief Valve		2.23E-2
Loss of Offsite Power		3.59E-2
Less than 30 min.	9.68E-3	
30 min. to 2 hours	1.48E-2	
2 to 8 hours	9.01E-3	
Greater than 8 hours	2.41E-3	
Small LOCA (max diameter 1 in. liquid) (max diameter 4 in. steam)		5.00E-4
Medium LOCA (max diameter 5 in. liquid) (max diameter 5 in. steam)		1.04E-4
Large LOCA (break diameter >5 in.)		6.78E-6

Table 19D.3-2
ABWR System and Train Maintenance and Test Unavailability

System	T&M Unavailability
RCIC	0.02
HPCFB	0.02
HPCFC	0.02
RHR A	0.02
RHR B	0.02
RHR C	0.02

Table 19D.3-3
LOOP Recovery Calculations for Assessed Durations

LOOP Recovery Time	LOOP IE Frequency (/rcyr)	Offsite Power Non-Restoration Probability	LOOP Frequency by Duration (/rcyr)
0-0.5 hours	3.59E-2	2.69E-1	9.66E-3
0.5-2 hours	3.59E-2	4.13E-1	1.48E-2
2-8 hours	3.59E-2	2.51E-1	9.01E-3
>8 hours	3.59E-2	6.72E-2	2.41E-3

Table 19D.3 4
Diesel Generator Recovery Table

DG Recovery Time	Probability of Non-Recovery
0-0.5 hours	9.29E-1
0.5-2 hours	7.53E-1
2-8 hours	4.72E-1
>8 hours	2.96E-1

Table 19D.3-5
Initiating Event Contribution to CDF

Event	Description	CDF Contribution (/yr)	% of CDF
TM	Reactor Shutdown	2.90E-10	0.30%
TT	Turbine Trip	4.52E-10	0.46%
TIS	Isolation/Loss of Feedwater	5.73E-9	5.84%
TE2	Loss of Offsite Power for less than 2 hours	1.70E-9	1.73%
TE8	Loss of Offsite Power for 2 to 8 hours	4.27E-9	4.36%
TE0	Loss of Offsite Power for more than 8 hours	1.46E-9	1.49%
BE2	Station Blackout for less than 2 hours	3.47E-8	35.37%
BE8	Station Blackout for 2 to 8 hours	1.38E-8	14.07%
BE0	Station Blackout for more than 8 hours	3.32E-8	33.88%
TIO	Inadvertantly Open Relief Valve	3.38E-11	0.03%
S2	Small Break LOCA	2.26E-10	0.23%
S1	Medium Break LOCA	1.29E-9	1.31%
S0	Large Break LOCA	7.51E-11	0.08%
ATWS	Anticipated Transient Without SCRAM	8.24E-10	0.84%
Total		9.80E-8	100%

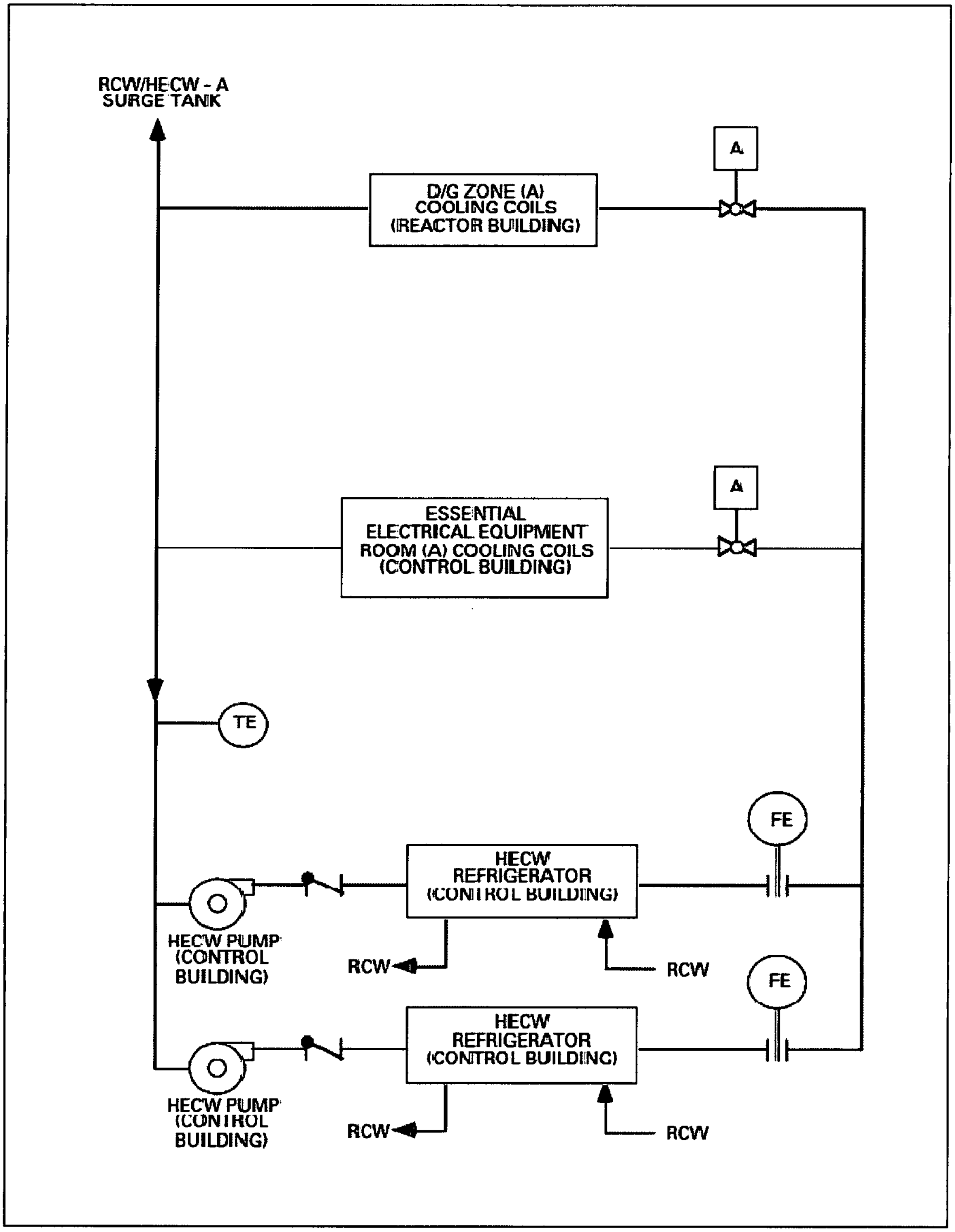


Figure 19D.3-1
HECW Division A

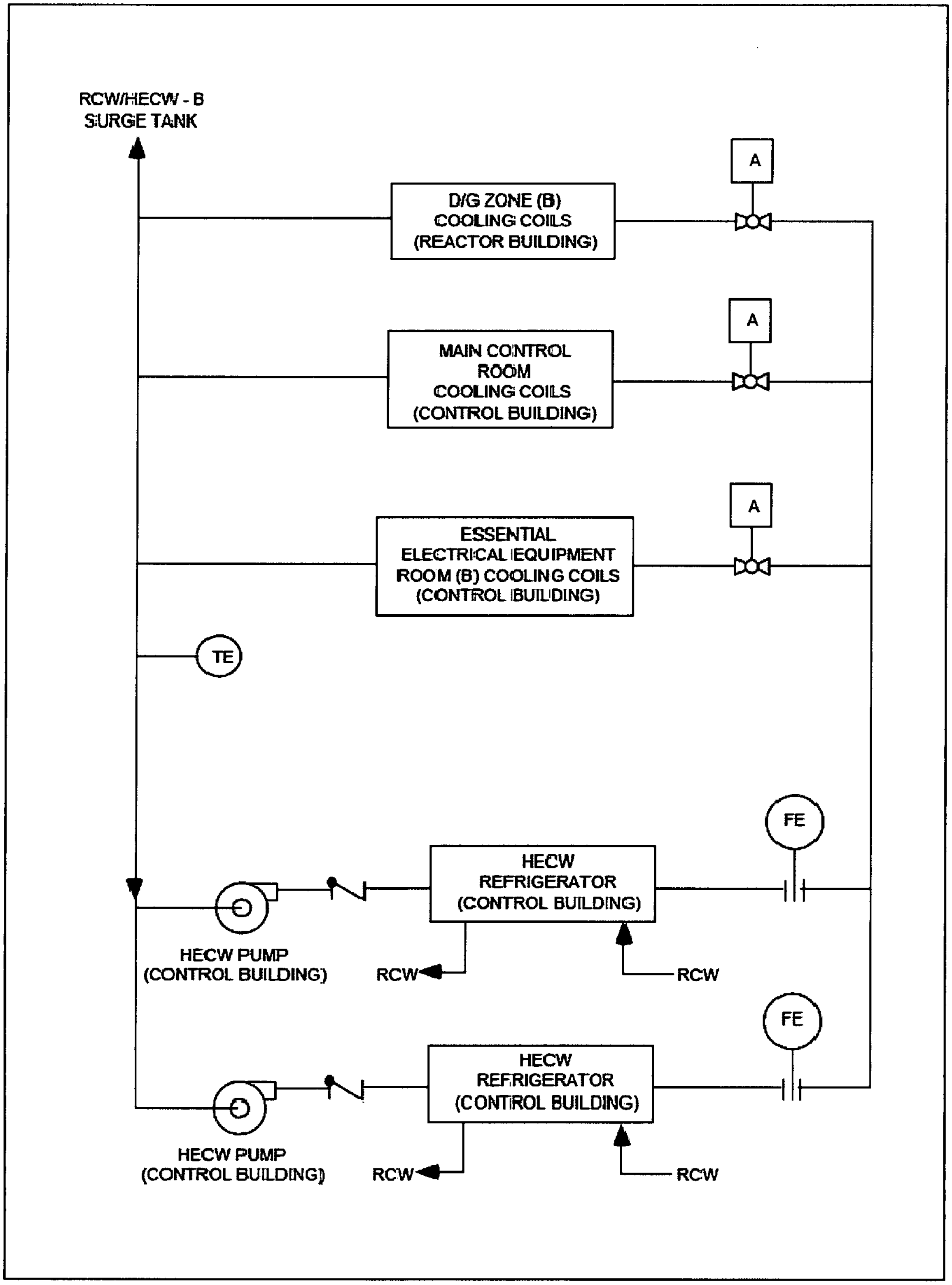


Figure 19D.3-2
HECW Division B

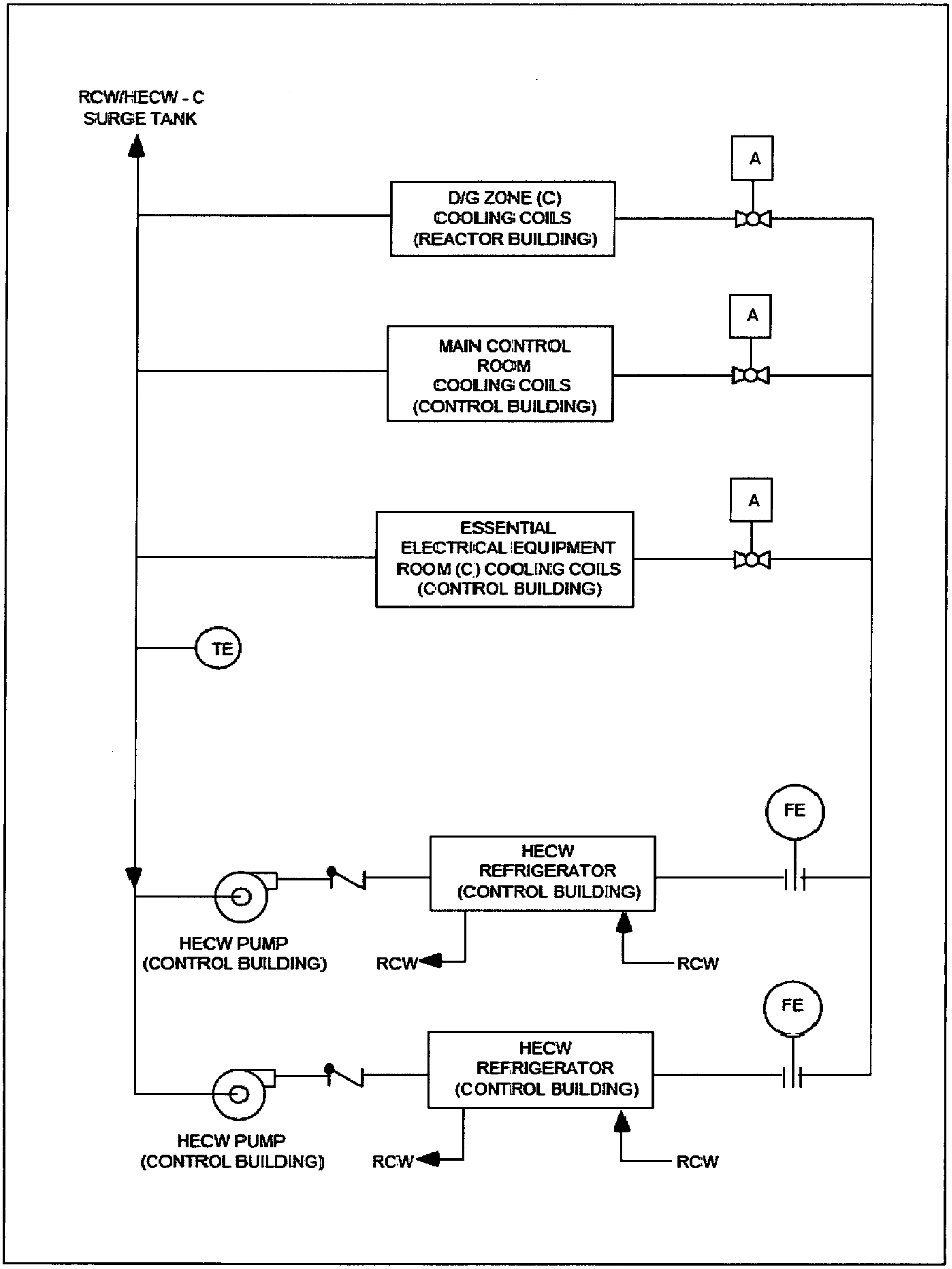


Figure 19D.3-3
HECW Division C

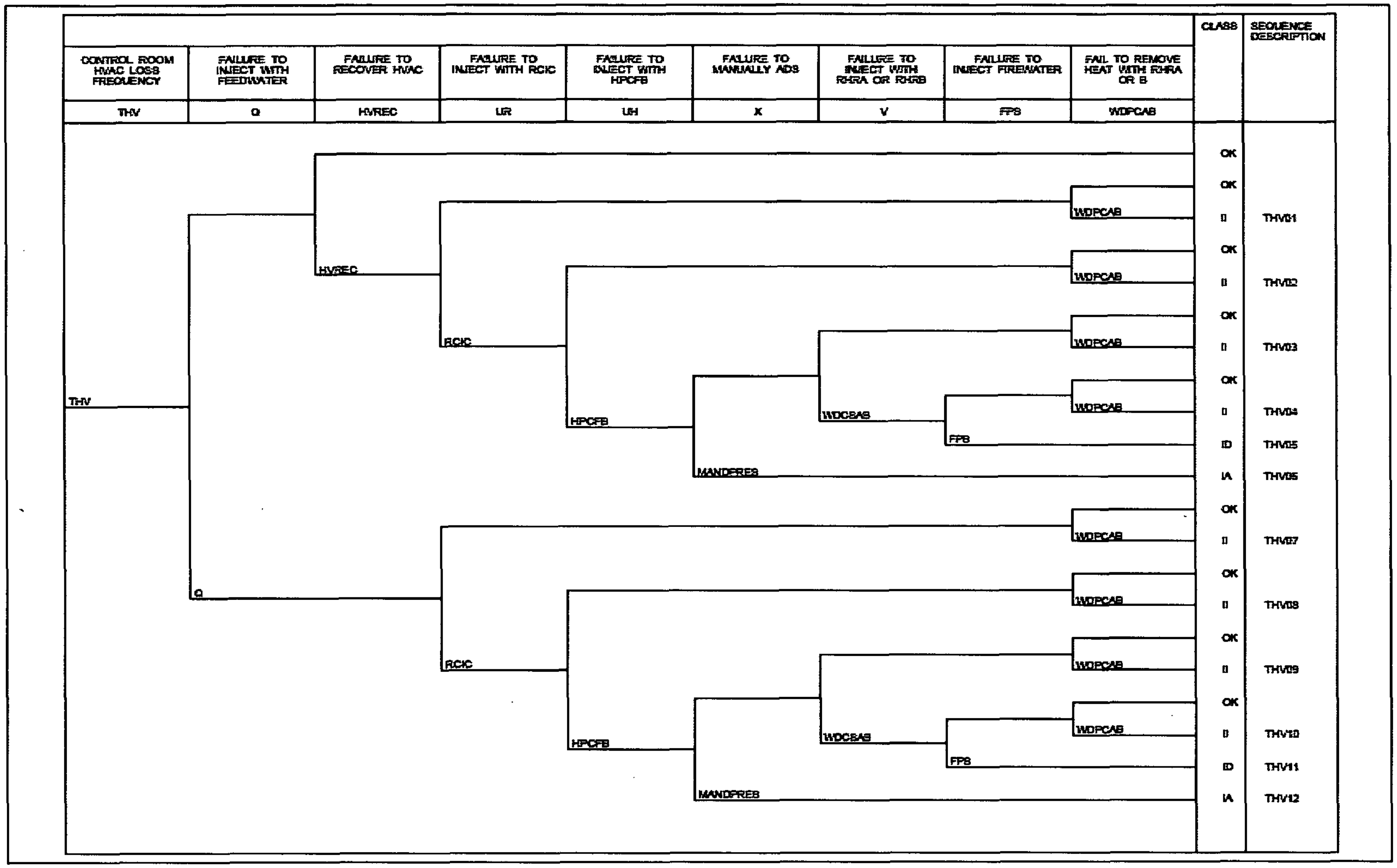


Figure 19D.3-4
Loss of Control Room HVAC

19D.4 ACCIDENT EVENT TREES

19D.4.1 ACCIDENT EVENT TREE ANALYSIS

19D.4.1.1 Introduction

This subsection describes the event trees used in the analysis to determine accident sequence frequencies. Each of these sequences leads to core damage, safe reactor shutdown, or to intermediate states which require additional treatment in the containment event trees of Subsection 19D.5 to establish final core states. Separate trees have been developed, as shown in Figures 19D.4-1 through 19D.4-15, for each of the initiating events considered. Accident event tree sequences which lead to core damage or loss of heat removal are further treated in the containment event trees of Subsection 19D.5 to determine frequencies of isotope releases to the environment.

19D.4.1.2 Accident Event Tree General Description

Figure 19D.4-1 is the event tree for the reactor shutdown initiating event. The initiating event name and symbol are provided at the top of the first column, and the first event has its frequency. The tree was developed by identifying the system functions required, in the approximate chronological order of occurrence, for successful reactor shutdown. Success and failure states of each system function are represented by branches in the tree, where the upper branch represents success and the lower branch failure. If a prior system function leads directly to success or failure in the accident sequence, analysis of the remaining system functions is unnecessary. Information given at the top of the column for each system function consists of a description of success and the symbol for conditional failure probability. The two columns labeled "CLASS" and "PROB" document the outcome of each accident sequence. The first column contains the classification of each sequence; either successful termination, core damage, or a sequence which is developed further in another accident tree or transferred to the appropriate containment event tree. The final column contains the yearly frequency of each sequence

19D.4.1.3 Safety Functions and Success Criteria

Accident event trees developed in this analysis contain branches which address the primary safety functions of reactivity control, reactor pressure control, core cooling, and containment heat removal. These four functions are considered in all event trees except the reactor shutdown event in which reactivity control is, by definition, provided by event initiation.

Success criteria provide the bases for defining combinations of those functions required to bring the plant to a safe stable shutdown condition. The necessary combinations of minimum system requirements were established on the basis of best estimate predictions. Success criteria are provided in Subsection 19.3.1.3.1.

19D.4.1.4 Branch Point Probabilities

Branch point values were defined by the system fault tree documented in Subsection 19D.6. These trees and other estimates were linked with initiating event frequencies as defined by each sequence, and each sequence equation was solved directly to obtain contribution to core damage frequency. These sequence outcomes were then summed to obtain total core damage frequency.

19D.4.1.5 Accident Sequence Classification

As stated previously, the consequence of each accident sequence may be either successful termination (achievement of adequate core cooling and containment heat removal) or core damage. Each sequence that results in core damage or inadequate containment heat removal is assigned to an accident class for further evaluation in the containment event trees of Subsection 19D.5. The bases for sequence classification are discussed in detail in that subsection.

19D.4.1.6 ATWS and LOCA Sequence Treatment

Sequences leading to either ATWS or LOCA events are not processed to their final dispositions in the primary transient event trees, and consequently are transferred to other event trees for further development. For example, in the non-isolation event tree, Figure 19D.4-2, the bottom sequence representing a failure to scram is routed to the ATWS event tree, Figure 19D.4-15, for additional treatment. In these sequences, the event tree name (ATWS, in this example) is indicated in the accident class column.

19D.4.1.7 Accident Sequence Evaluation

The frequency of each accident sequence is developed by initiating event in the event trees of Figures 19D.4-1 through 19D.4-15. Sequence outcomes are summed by accident class and these totals are then routed to the corresponding CET for further analysis. Table 19D.4-17 provides a summary of accident event tree results by initiating event and accident class.

19D.4.2 EVENT TREE DESCRIPTIONS

This subsection provides a description of each of the event trees developed and illustrated in Figures 19D.4-1 through 19D.4-15. Values less than 10^{-12} are noted as "NIL".

19D.4.2.1 Reactor Shutdown

The event tree for reactor shutdown is presented in Figure 19D.4-1. Reactor shutdown includes any event in which the reactor is shut down under normal operating conditions. Not all of the primary safety functions are required for this event. By definition, reactivity control is not a required function for mitigating this event. Neither is the reactor pressure control function required since there is no rapid pressure increase associated with this event.

19D.4.2.2 Non-Isolation (Turbine Trip)

The event tree for a non-isolation trip is shown in Figure 19D.4-2. Non-isolation events include any event in which the turbine is tripped and removed from the steam loop, but the condenser and feedwater remain available.

19D.4.2.3 Isolation/Loss of Feedwater

The event tree for isolation or loss of feedwater events is shown in Figure 19D.4-3. Isolation events are those events in which the reactor is isolated from the power conversion system, resulting in loss of the feedwater and the condenser. Loss of feedwater events are absorbed into this event category, as MSIV closure represents the bounding case for loss of feedwater. Although the feedwater and condenser are initially lost, there is a probability that they will be recovered.

19D.4.2.4 Loss of Offsite Power and Station Blackout Event Tree

An event tree is no longer used to calculate these values. Refer to 19D.3.1.2.4 for discussion of loss of offsite power.

19D.4.2.5 Loss of Offsite Power for 30 Minutes to Two Hours

The event tree for loss of offsite power for thirty minutes to two hours is presented in Figure 19D.4-5. This event includes any scenario for which no external power is available to the plant for two hours.

19D.4.2.6 Loss of Offsite Power for Two to Eight Hours

The event tree for loss of offsite power from two to eight hours is shown in Figure 19D.4-6. This event tree includes any scenario for which no external power is available to the plant for two to eight.

19D.4.2.7 Loss of Offsite Power for More Than Eight Hours

The event tree for loss of offsite power for more than eight hours is shown in Figure 19D.4-7. This event tree includes any scenario for which no external power is available to the plant for more than eight hours.

19D.4.2.8 Station Blackout for Less Than Two Hours

Figure 19D.4-8 presents the event tree for station blackout for less than two hours. This event tree includes any scenario for which neither external power nor station diesel or combustion turbine generator power are available to the plant for two hours following the loss of offsite power. For this situation, RCIC is the only injection system available for core cooling. The heat removal function is not impaired, since its operation is not required prior power to restoration.

19D.4.2.9 Station Blackout for Two to Eight Hours

The event tree for station blackout from two to eight hours is presented in Figure 19D.4-9. T his event tree includes any scenario for which neither external power nor station diesel or combustion turbine generator power are available to the plant for two to eight hours following the loss of offsite power. Initially RCIC is the only injection system available for core cooling. DC power for control will be available, since battery life is expected to be at least eight hours without any AC power to the battery chargers. In addition, the RCIC pump and turbine are expected to operate for at least eight hours without room coolers. Given successful RCIC operation for eight hours, the remaining injection systems become available upon the recovery of power.

19D.4.2.10 Station Blackout for More Than Eight Hours

The event tree for station blackout greater than eight hours is shown in Figure 19D.4-10. This event tree includes any scenario for which neither external power nor station diesel or combustion turbine generator power are available to the plant for more than eight hours. All sequences are conservatively assumed to lead to core damage but are sorted by accident class since timing and consequences differ.

19D.4.2.11 Inadvertent Open Relief Valve (IORV) Event Tree

The event tree for IORV is shown in Figure 19D.4-11. This event includes those scenarios which begin with one or more relief valves opening and remaining open while the reactor is under otherwise normal operating conditions. Since a stuck open relief valve will eventually result in depressurization of the reactor, the reactor pressure control function is not required.

The ABWR incorporates design features which will automatically initiate suppression pool cooling, as well as initiate automatic scram, on high suppression pool temperature. Therefore, the IORV event tree as shown represents a conservative assessment of the IORV event. Since the contribution of the IORV event to overall core damage frequency is very small, this event was not remodeled to reflect these improvements.

19D.4.2.12 Small Break LOCA Event Tree

The small break LOCA event tree is shown in Figure 19D.4-12. A small LOCA as defined under the success criteria of Subsection 19.3.1.2 is a liquid break of pipe diameter less than 1 in. or a steam pipe break diameter of less than 4 in. Similar to an IORV sequence, this event does not require the reactor pressure control function for successful mitigation.

19D.4.2.13 Medium Break LOCA Event Tree

The event tree for a medium break LOCA is shown in Figure 19D.4-13. This accident is defined as a liquid pipe break between 1 and 5 in. Break diameter is defined as 4 to 5 in. for steam passage. This sequence, also similar to an IORV, does not require the reactor pressure control safety function.

19D.4.2.14 Large Break LOCA Event Tree

The event tree for a large LOCA is shown in Figure 19D.4-14. A large LOCA is defined as a liquid or steam break on a pipe having a diameter greater than 5 in. The initiating event frequency for this event includes transfers from those sequences in other event trees where the SRVs do not open on demand thus causing large water or steam breaks. The event does not require the reactor pressure control safety function.

19D.4.2.15 ATWS Accident Sequence Event Tree

The event tree for ATWS is shown in Figure 19D.4-15. The initiating event frequency for this tree is the sum of those sequences in other event trees in which the control rods are not inserted by either the RPS or the alternate rod insertion (ARI) system. Based upon the ATWS success criteria defined in Subsection 19.3.1.2 and in light of the low frequency of ATWS initiators, a single ATWS event tree for an isolation event is adequate to conservatively assess ATWS.

Table 19D.4-1	Not Used
Table 19D.4-2	Not Used
Table 19D.4-3	Not Used
Table 19D.4-4	Not Used
Table 19D.4-5	Not Used
Table 19D.4-6	Not Used
Table 19D.4-7	Not Used
Table 19D.4-8	Not Used
Table 19D.4-9	Not Used
Table 19D.4-10	Not Used
Table 19D.4-11	Not Used
Table 19D.4-12	Not Used
Table 19D.4-13	Not Used
Table 19D.4-14	Not Used
Table 19D.4-15	Not Used
Table 19D.4-16	Not Used

Table 19D.4-17
ABWR Internal Event PRA Core Damage Frequency Summary

	Accident Class											
Init. Event	IA	IB-1	IB-2	IB-3	IC	ID	II	IIIA	IIID	IV	Tot.CDF	Percent
TM	1.84E-10	—	—	—	—	1.06E-10	—	—	—	—	2.90E-10	0.30%
TT	1.22E-11	—	—	—	—	8.28E-11	3.48E-10	—	8.92E-12	—	4.52E-10	0.46%
T/S	8.84E-10	—	—	—	—	4.70E-09	1.38E-10	—	2.66E-12	—	5.73E-09	5.84%
TE2	1.11E-09	—	—	—	—	5.85E-10	2.06E-12	—	—	—	1.70E-09	1.73%
TE8	6.70E-10	—	—	—	—	3.60E-09	5.00E-12	—	—	—	4.27E-09	4.36%
TE0	1.65E-10	—	—	—	—	9.57E-10	3.35E-10	—	—	—	1.46E-09	1.49%
BE2	—	—	—	—	—	3.47E-08	—	—	—	—	3.47E-08	35.37%
BE8	—	1.38E-08	—	—	—	—	—	—	—	—	1.38E-08	14.07%
BE0	—	—	3.32E-08	—	—	—	—	—	—	—	3.32E-08	33.88%
TI0	7.54E-12	—	—	—	—	2.62E-11	—	—	—	—	3.38E-11	0.03%
S2	—	—	—	—	—	—	—	2.90E-11	1.97E-10	—	2.26E-10	0.23%
S1	—	—	—	—	—	—	—	1.38E-11	1.27E-09	—	1.29E-09	1.31%
S0	—	—	—	—	—	—	—	—	7.51E-11	—	7.51E-11	0.08%
ATWS	—	—	—	—	—	—	2.74E-11	—	—	7.97E-10	8.24E-10	0.84%
Tot.CDF	3.04E-09	1.38E-08	3.32E-08	0.00E+00	0.00E+00	4.47E-08	8.56E-10	4.29E-11	1.56E-09	7.97E-10	9.80E-08	100.00%
Percent	3.10%	14.07%	33.88%	0.00%	0.00%	45.63%	0.87%	0.04%	1.59%	0.81%	100.00%	

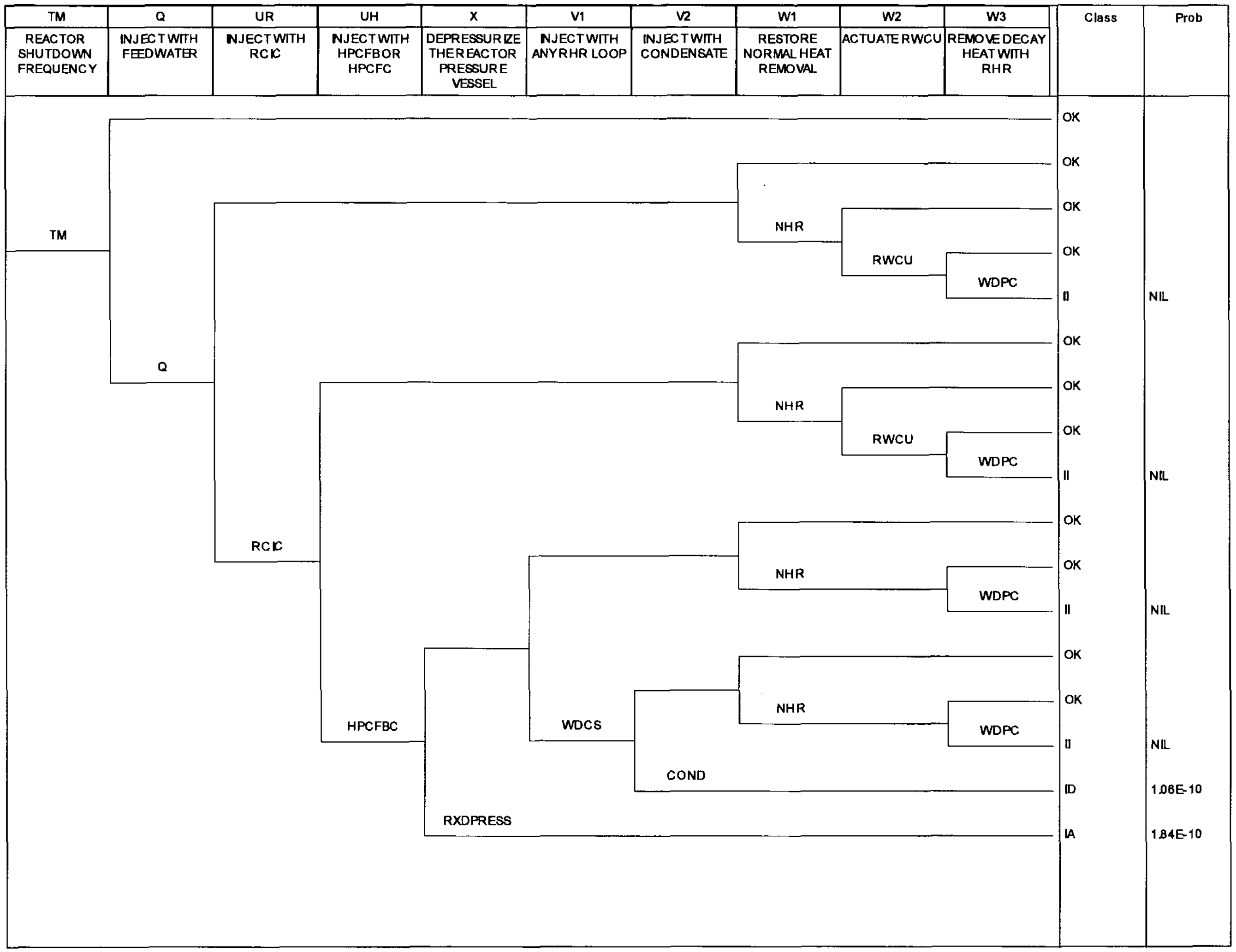


Figure 19D.4-1
Reactor Shutdown Event Tree

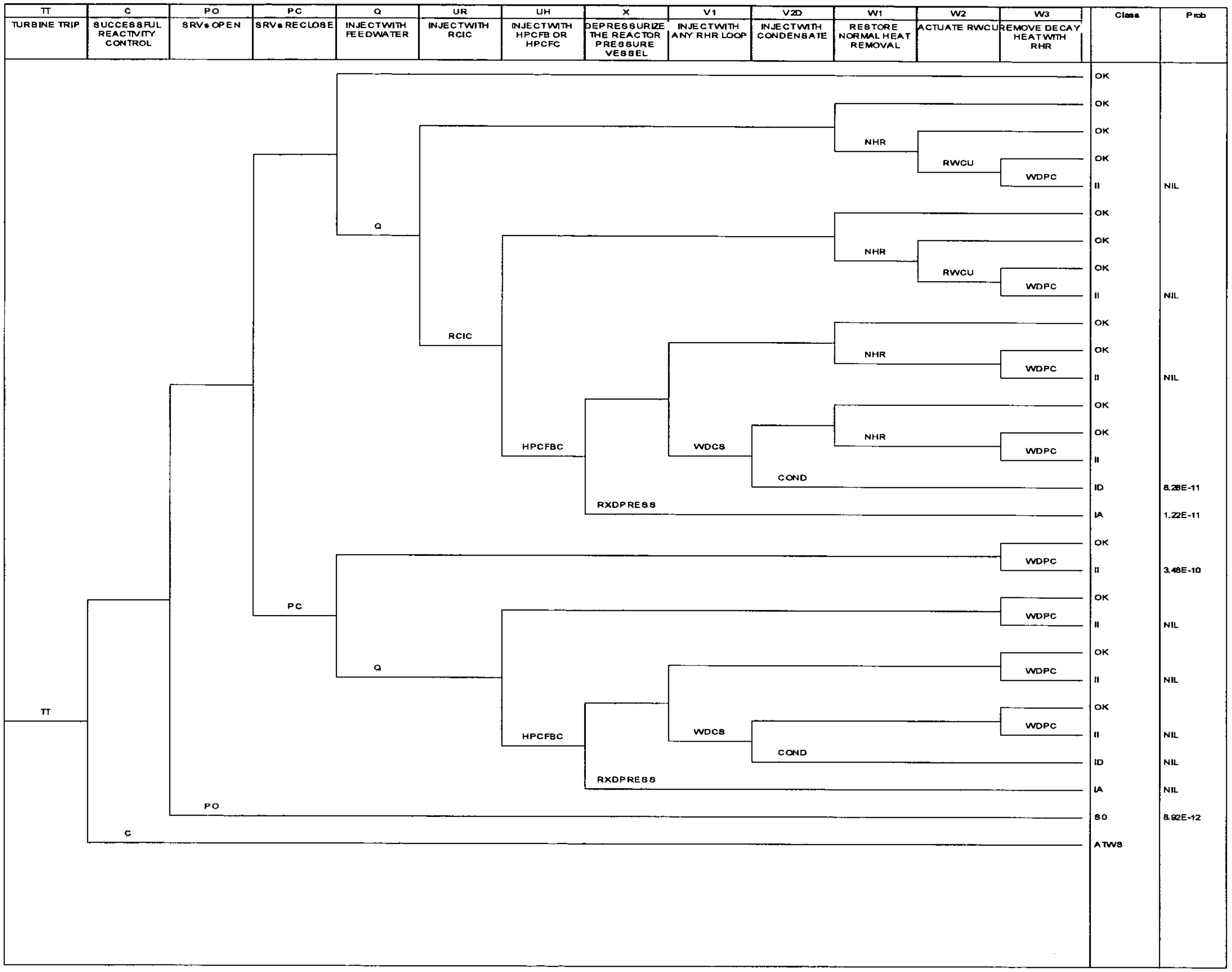


Figure 19D.4-2
Non-Isolation Event Tree

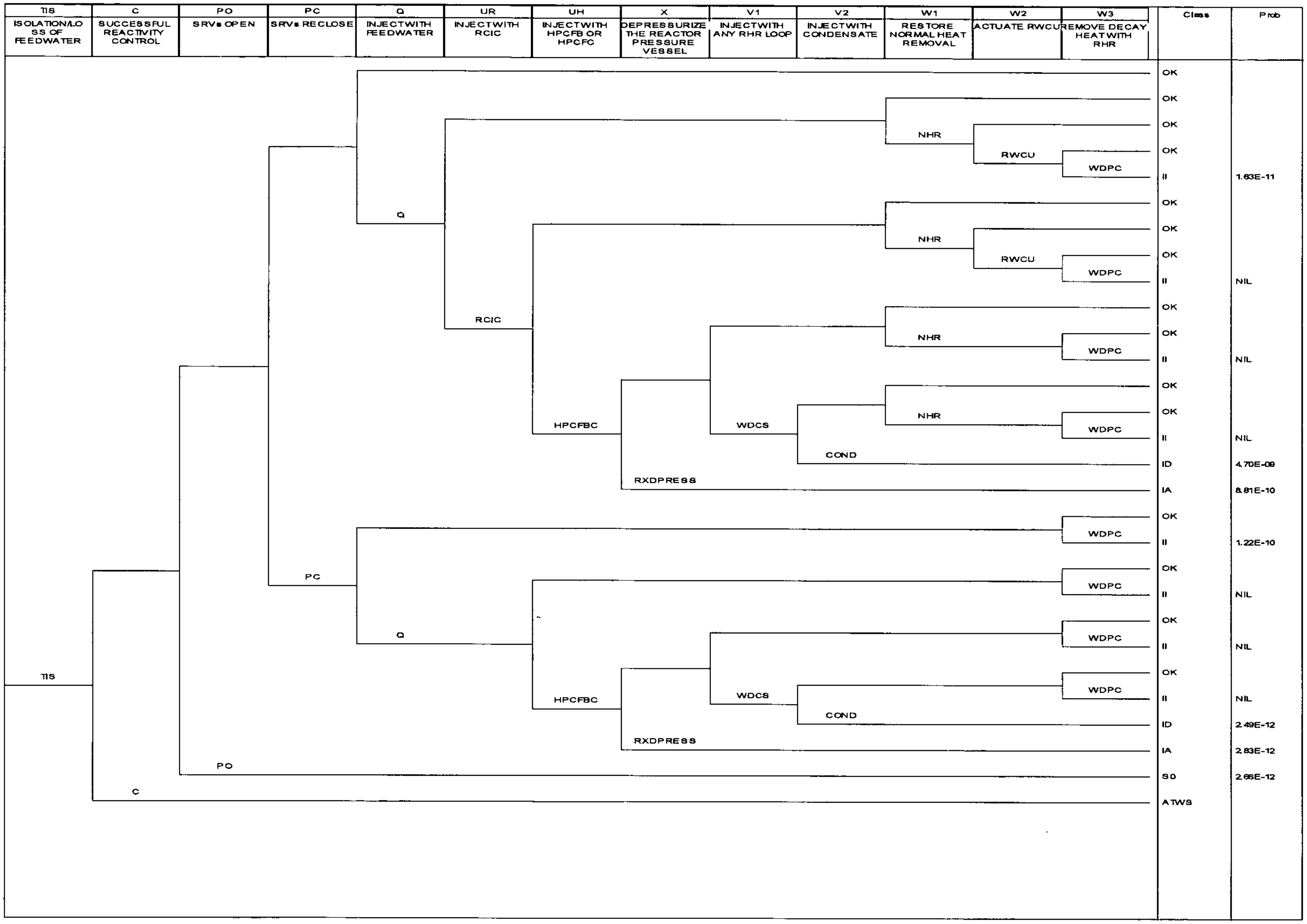


Figure 19D.4-3
Isolation/Loss of Feedwater Event Tree

Figure 19D.4-4
Not Used

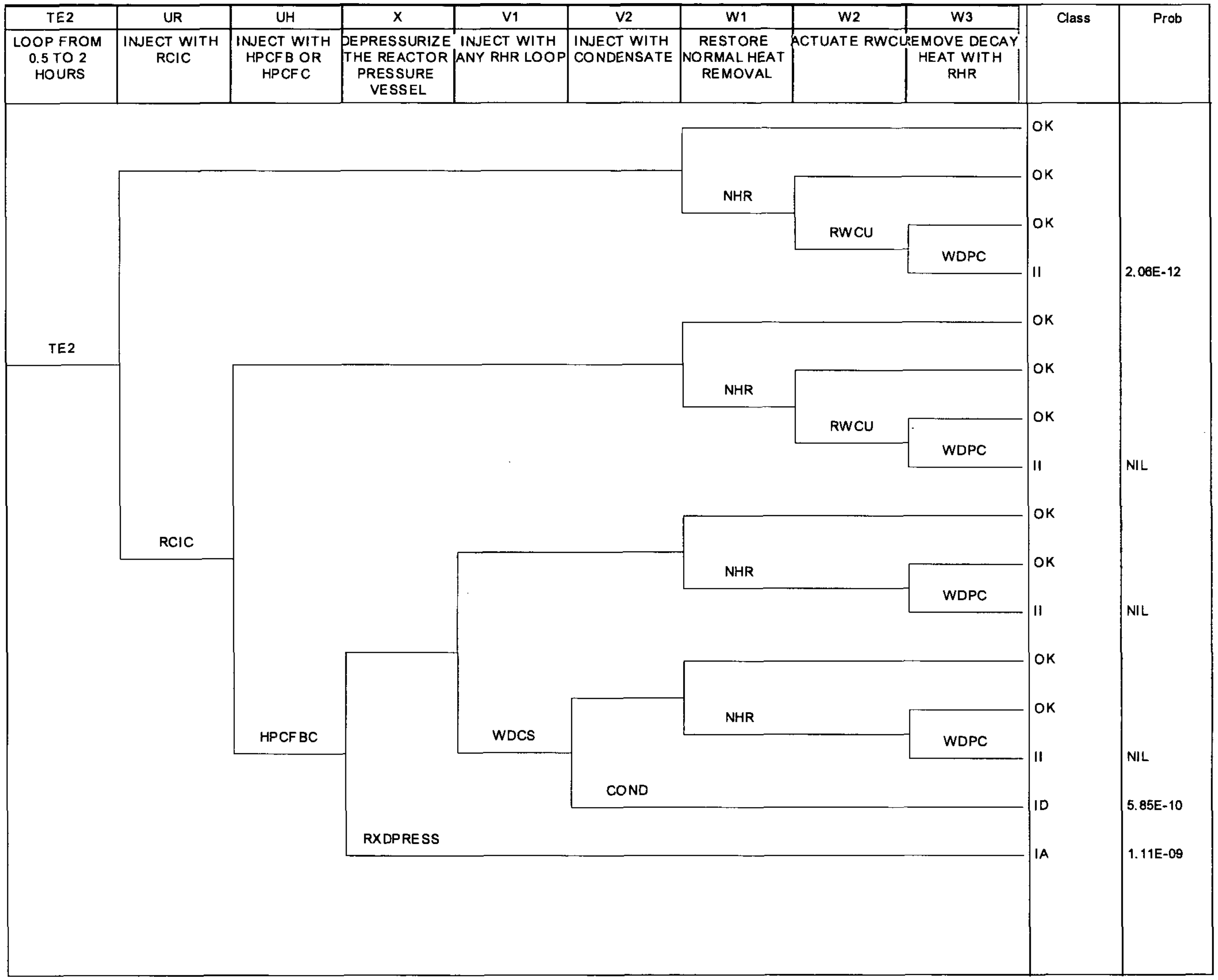


Figure 19D.4-5
Loss of Offsite Power Event Tree (Recovery time: 0.5<t<2 hr)

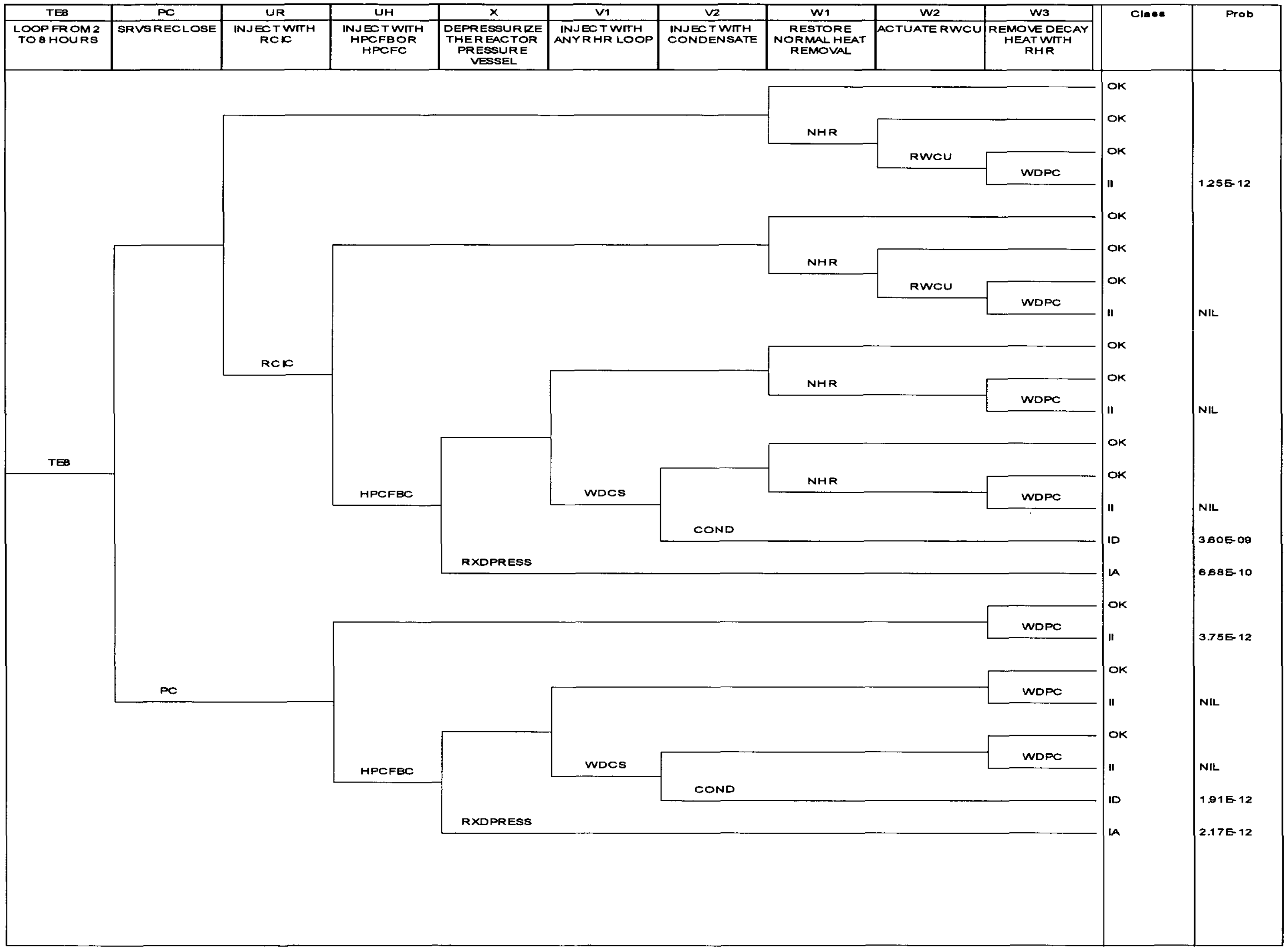


Figure 19D.4-6
Loss of Offsite Power Event Tree (Recovery time: 2<t<8 hr)

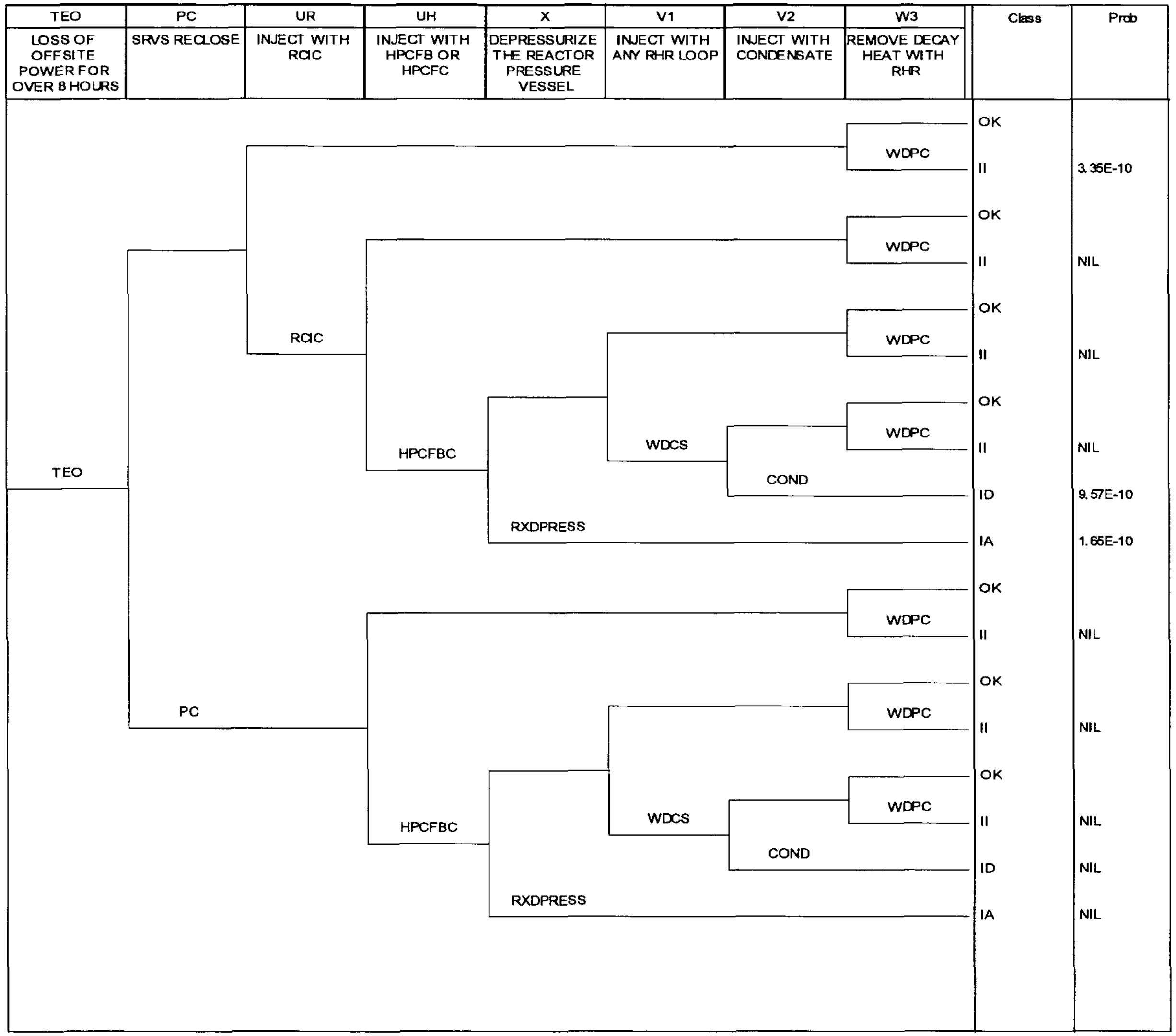


Figure 19D.4-7
Loss of Offsite Power Event Tree (Recovery time: >8 hr)

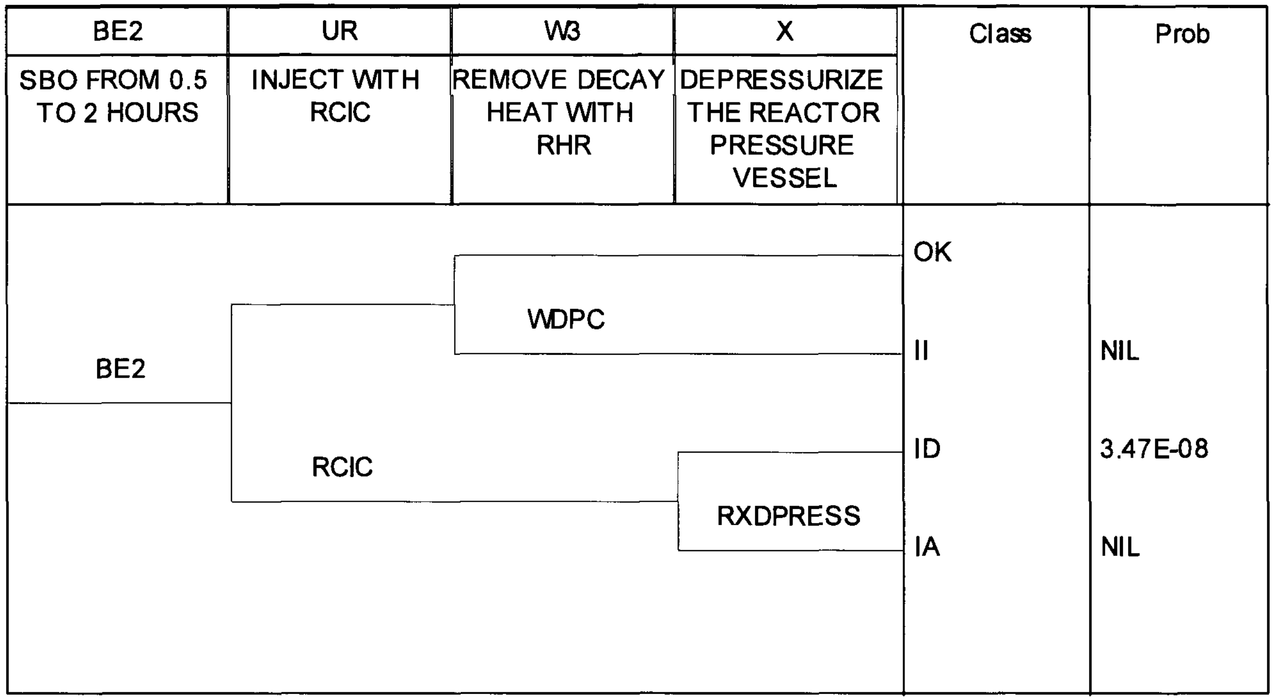


Figure 19D.4-8
Station Blackout Event Tree (Recovery time: 0.5<t<2 hr)

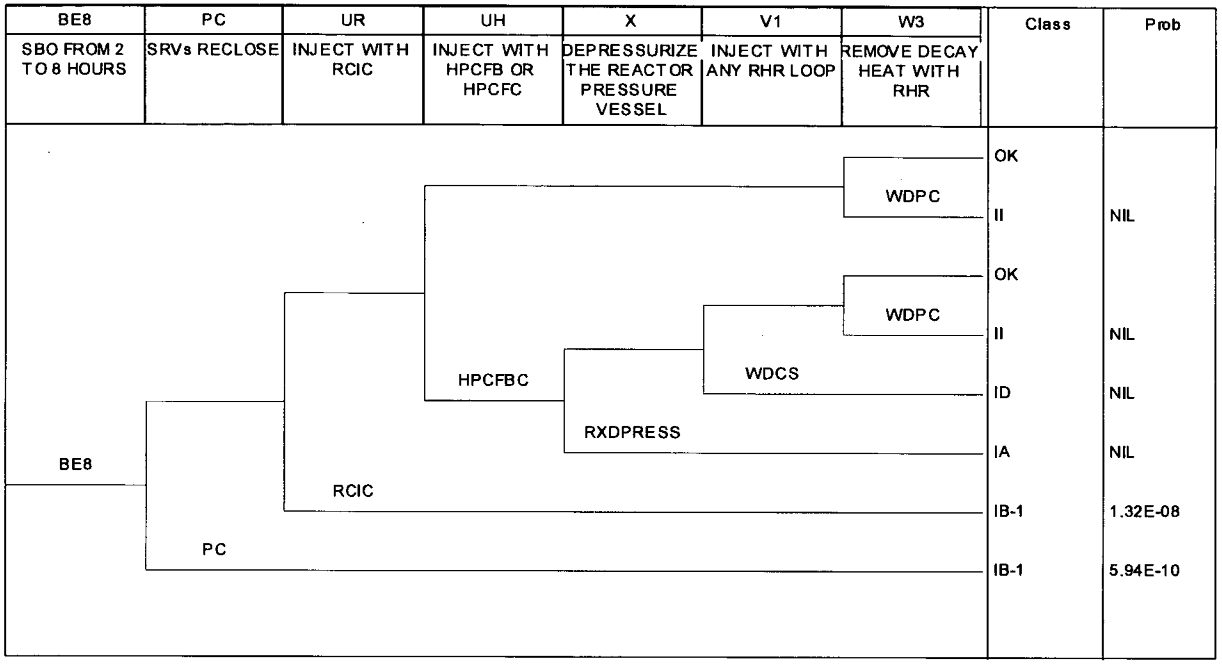


Figure 19D.4-9
Station Blackout Event Tree (Recovery time: 2<t<8 hr)

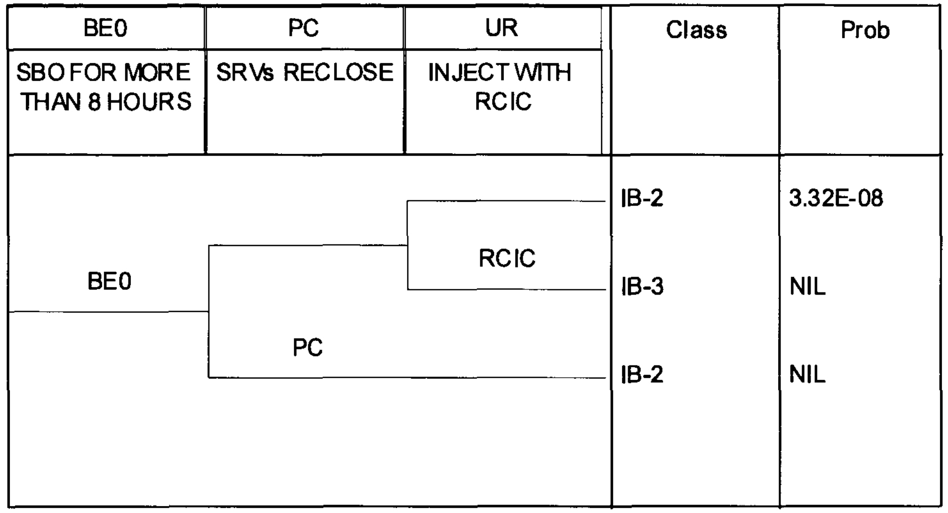


Figure 19D.4-10
Station Blackout Event Tree (Recovery time: >8 hr)

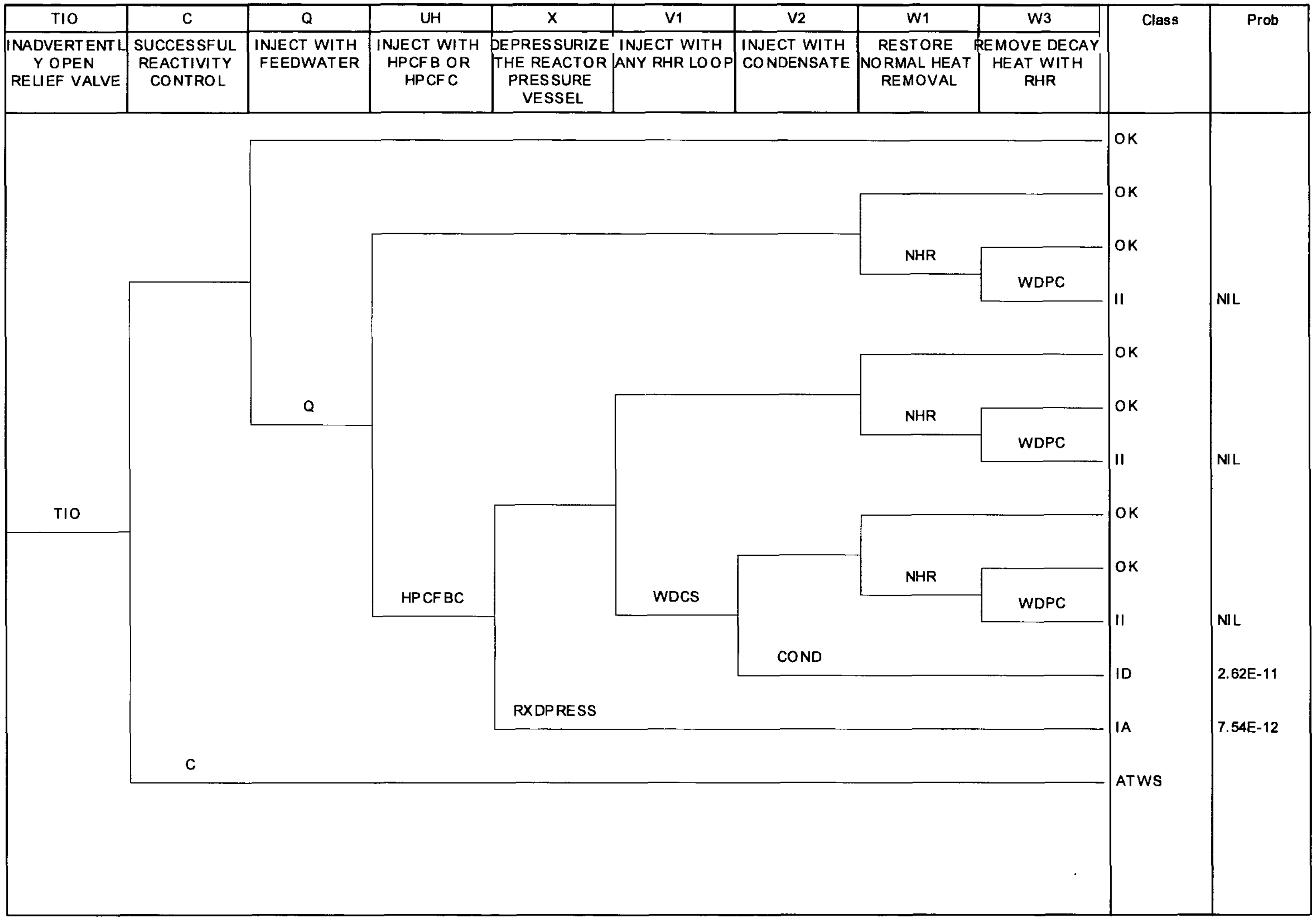


Figure 19D.4-11
Inadvertently Open Relief Valve (IORV) Event Tree

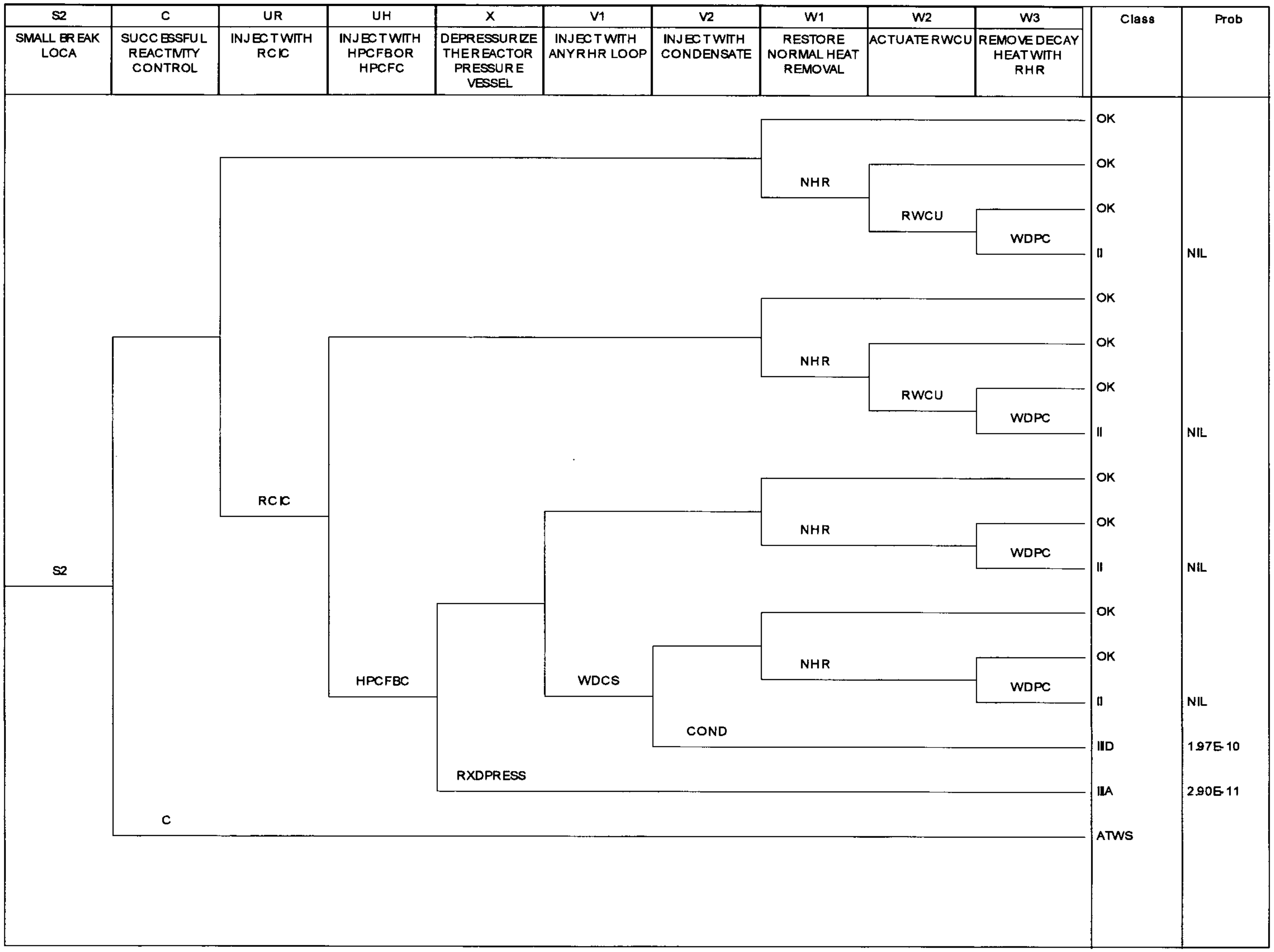


Figure 19D.4-12
Small LOCA Event Tree

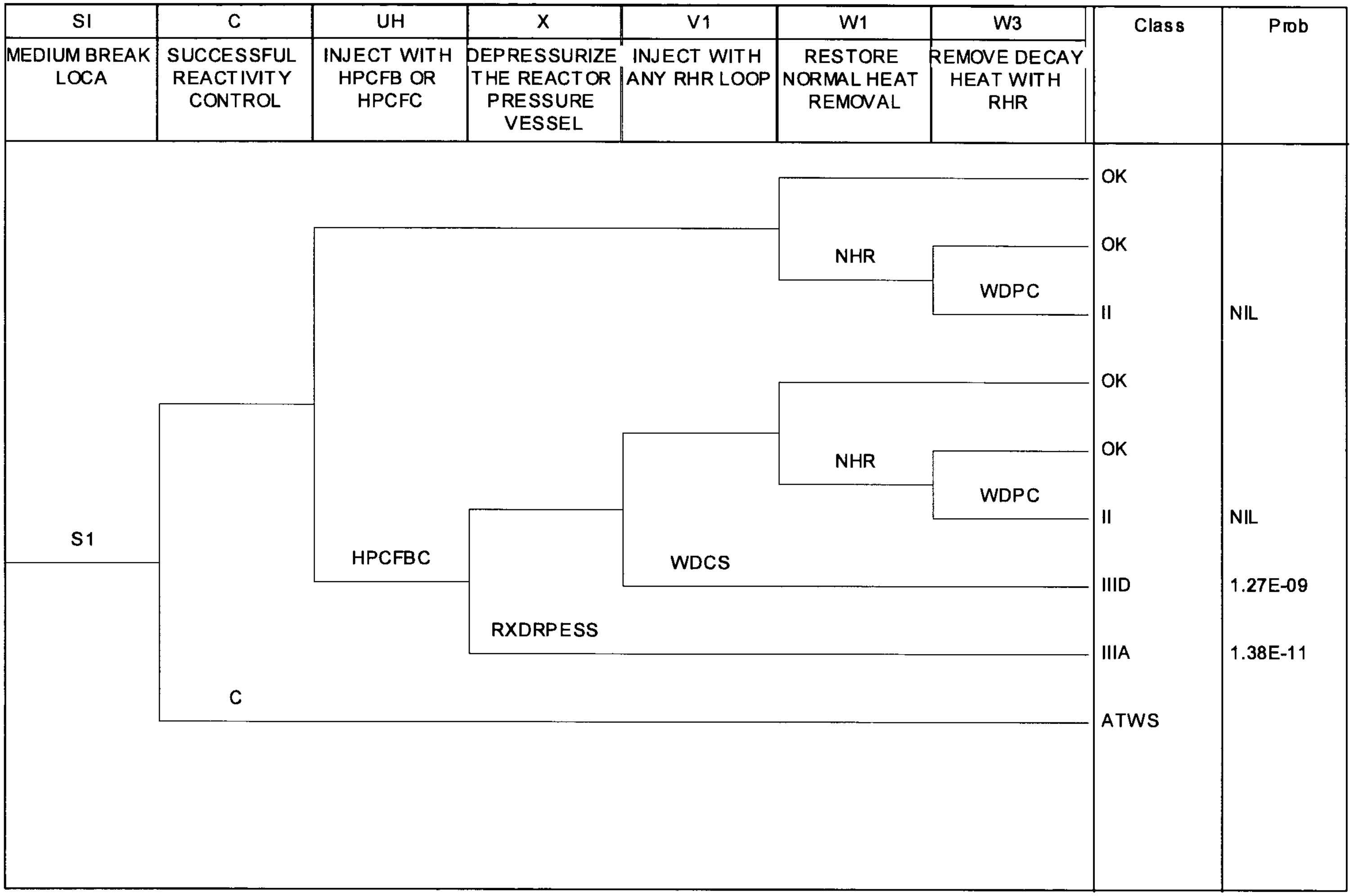


Figure 19D.4-13
Medium LOCA Event Tree

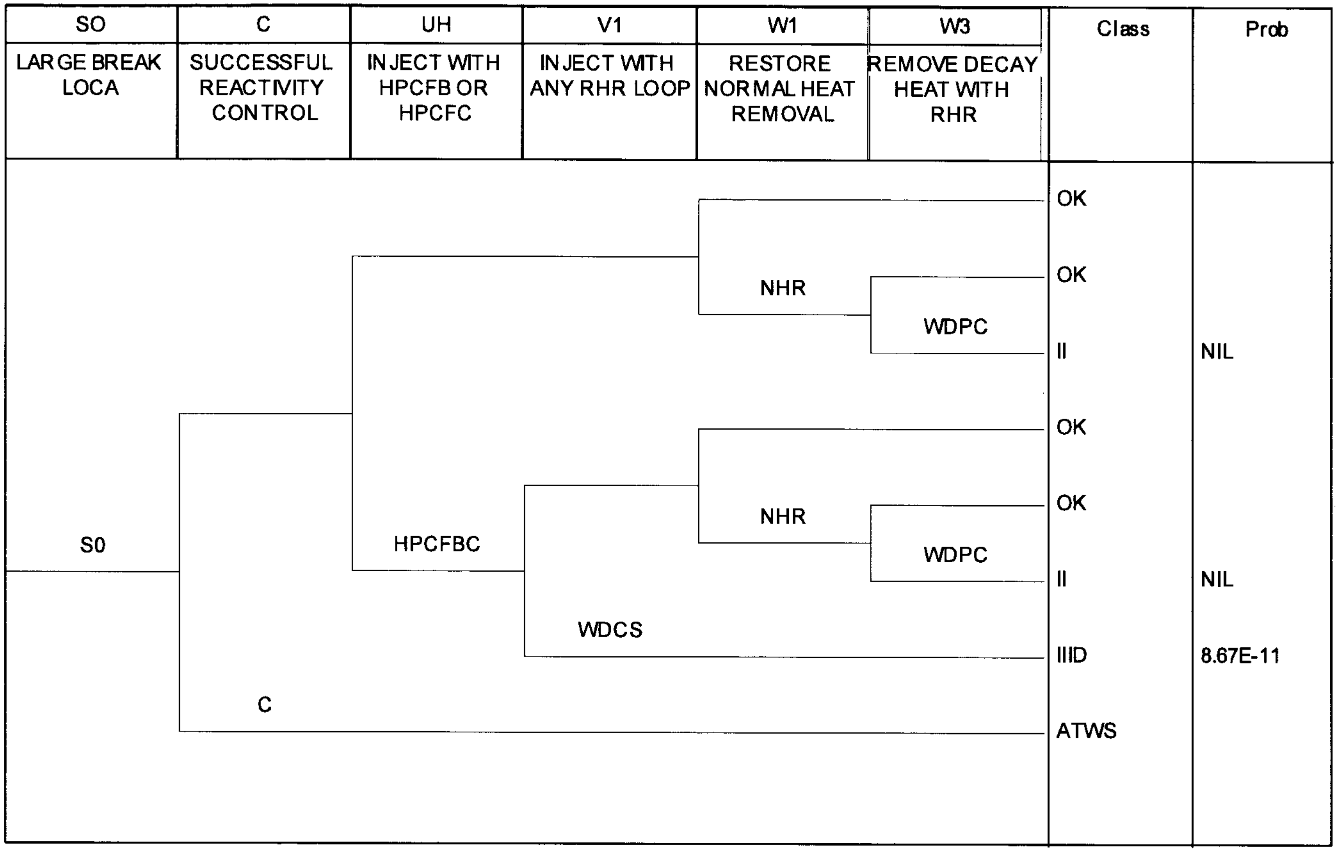


Figure 19D.4-14
Large LOCA Event Tree†

† - The value of S0SEQ3 sums Class IIID probabilities of 7.51E-11 for large break (S0) initiating events and 1.16E-11 from initiating events TT (8.92E-12 for S0 branch from Figure 19D.4-2) and TIS (2.66E-12 for S0 branch from Figure 19D.4-3). These values are shown in their rows for their respective initiating events in Table 19D.4-17.

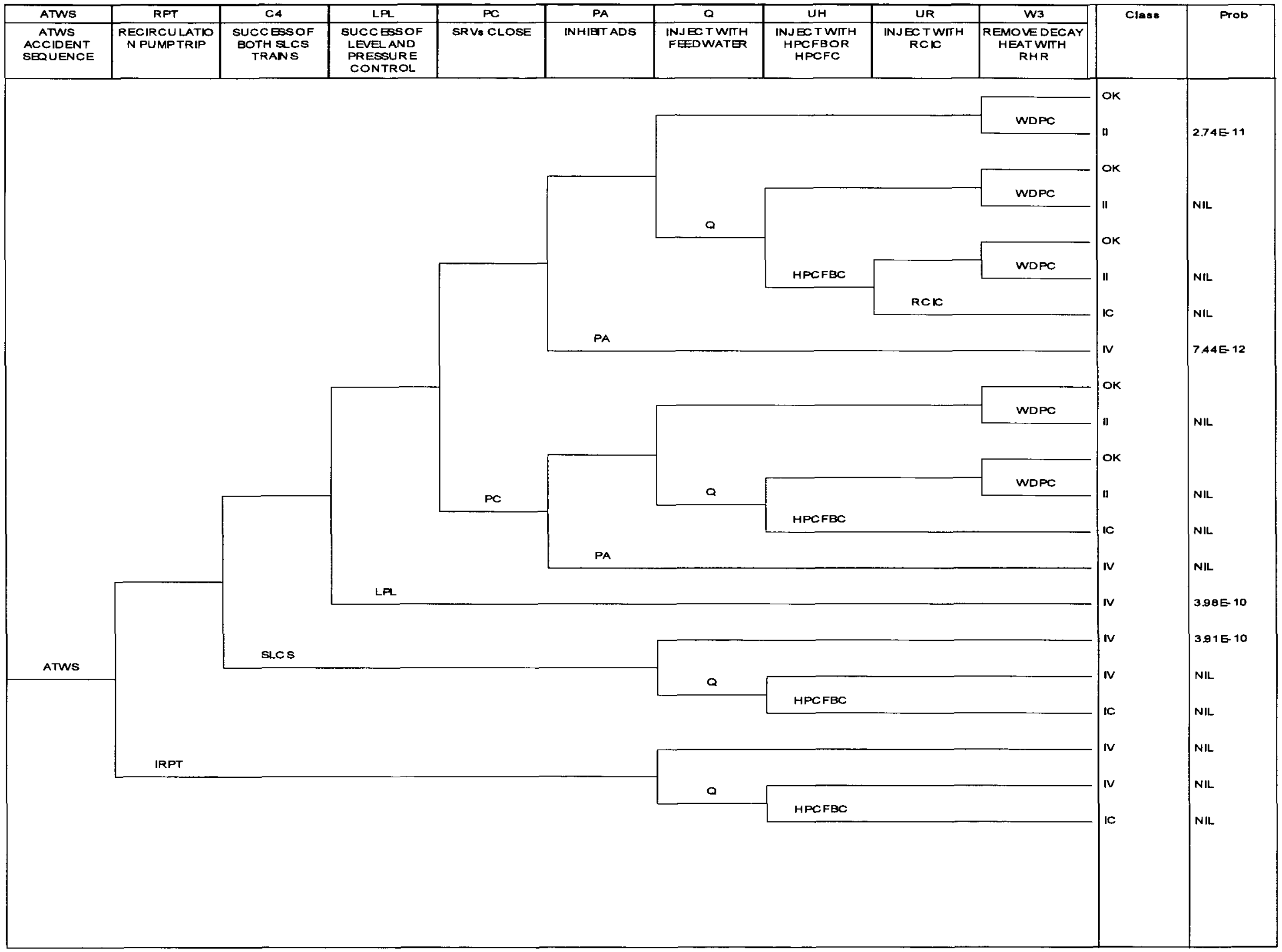


Figure 19D.4-15
ATWS Event Tree

19D.5 ABWR CONTAINMENT EVENT TREES

19D.5.1 OVERVIEW

The accident sequence event trees described in Subsection 19D.4 model the event progression for the various accident initiators, and provide the classification and frequency of accident sequences. In these event trees, the sequences which are terminated safely without core damage are designated as "OK". The event sequences which are not successfully terminated could either directly lead to core damage or in some cases could lead to containment structural failure which in turn could lead to core damage. These event sequences are "binned" into various accident classes depending upon the expected event progression, timing and mode of containment structural failure, and the amount of fission product release to the environment.

There are five basic classes (I through V), and a total of eleven classes including subclasses such as IA, IB, IC, etc. A Class IA event, for example, is a transient event with loss of high pressure water makeup systems followed by a failure to depressurize the reactor.

Generally, the event progressions for each of these classes of events are modeled in the containment event trees (CETs). The CETs model recovery actions which could prevent core damage or arrest core damage if already initiated. Where recovery actions are unsuccessful, the CETs model core melt leading to reactor vessel rupture, containment structural failure and fission product release to the environment. The CET models are based on core-melt progression analysis discussed in Subsection 19E.2. The mode and location of containment structural failure is modeled based on a study of the containment capability discussed in Appendix 19F.

There is one CET for each of the accident classes. The end states of CETs are either states with insignificant or no release (i.e., core damage prevented or core melt arrested), or states with a release path to the environment. Associated with each release path in each of the containment event trees, is a frequency of occurrence and a magnitude of fission product release. The frequencies are calculated by the CETs, and the fission product releases are evaluated using the fission product transport analysis discussed in Subsection 19E.2. The numerous release paths can be consolidated or "binned" into release categories by grouping them based on the expected timing and amount of fission product release to the environment.

The consolidated release categories and the associated frequencies are used as input to the consequence analysis discussed in Subsection 19E.3.

19D.5.2 ACCIDENT CLASSES

In Subsection 19D.4 accident event trees are developed for each of the initiators. The end states of these accident event trees are "binned" (grouped) into five basic accident classes based on similarities in the subsequent core melt event progression and the containment response. The key factors that influence the definition of the accident classes are as follows:

(1) Type of initiating event (transient, LOCA, etc.)

(2) Relative times of core melt and containment failure

(3) Suppression pool bypass status

Five basic accident classes, I through V, have been identified. A description of these five classes is provided below and is summarized in Table 19D.5-1.

(1) Class I

Class I events are transients with failure of core cooling systems. In these cases, core melt starts about one hour after event initiation and the RPV fails about one hour later. Following RPV failure, a mixture of molten core material and other metals, called corium, leaves the RPV and comes in contact with the concrete on the drywell floor. Water is supplied to the debris either through the use of active injection systems or via the passive flooder. Steam generation, and potentially, non-condensable gas generation from core concrete interaction, causes slow pressurization of the containment. If attempts to recover are not successful, this will lead to operation of the Containment Overpressure Protection System (COPS) or failure of the containment.

Event progressions for ATWS events with failure of core cooling systems are similar and are also considered as Class I events.

(2) Class II

Most Class II events are transients with successful core cooling, but with failure of the containment heat removal systems. The suppression pool heats up and the containment pressure builds up slowly until the Containment Overpressure Protection System (COPS) rupture pressure is reached in about 24 hours. This period is available to the operator to try to recover the failed systems. If the COPS fails to actuate, structural failure of the containment could occur affecting the core cooling function. Consideration of this possibility is included in the containment event trees.

Loss of core cooling leads to core melt and RPV failure. Class II core damage sequences are thus characterized by containment structural failure followed by core melt. At the time of core melt, the containment is in a failed state and the fission products are released to the atmosphere without the benefit of residence time in the containment.

Event progressions for LOCAs with successful core cooling and ATWS events with successful boron injection and successful core cooling but with failure of the containment heat removal system are similar to the event progression described for transient events and these events are also considered Class II events.

(3) Class III

LOCAs with loss of core cooling are Class III events. As in the case of Class I events, Class III events are also characterized by core melt followed by containment failure. However, because of the loss-of-coolant accident, core uncovery, core melt and RPV failure occurs faster than for Class I events.

(4) Class IV

Class IV events are ATWS events without boron injection but with core cooling available. Under these conditions, the reactor continues to produce up to 20% power. The steam produced in the reactor is routed to the suppression pool through the safety relief valves. If this situation continues unmitigated, the containment is over-pressurized leading to rupture disk opening or structural failure. As discussed under Class II events, structural failure could lead to loss of core cooling function. This, in turn, could result in core melt. In summary, Class IV events are characterized by fast containment over-pressurization followed by core melt. Following core melt, fission products are released to the environment without radioactive decay due to holdup within the containment.

(5) Class V

Class V events are events in which the suppression pool is bypassed. There are two types of Class V events. In the first, the pool is bypassed at the beginning of the event. An example of this type of event is a LOCA outside the containment. If the break is not isolated and if core cooling is unavailable, core melt will result, and the fission products will be released directly to the atmosphere without going through the suppression pool. The second type of Class V event consists of accidents in which the suppression pool is bypassed during the course of the accident. An example of this is the drywell rupture following core melt and RPV failure.

19D.5.3 ACCIDENT SUBCLASSES

19D.5.3.1 Class I Events

The accident Class I is further divided into four subclasses, IA through ID, as discussed below. A summary of the differences is provided in Table 19D.5-2.

(1) Class IA

Class IA events are characterized by high RPV pressure when the core melts. These are transient events followed by failure of high-pressure water makeup systems coupled with failure to depressurize the reactor (ADS failure, for example). The subsequent core melt event is called a high-pressure core melt. The core melt and RPV failure could result in ejection of molten corium at high pressure into the drywell, which could increase the potential for drywell failure. On the other hand, RPV failure would depressurize the reactor making the low-pressure systems available for flooding the molten core.

(2) Class IB

Class IB events are broken into three categories.

(a) Class IB-1

Class IB-1 events are station blackout events with RCIC failure. Neither core cooling nor containment heat removal is available in the beginning and the core melt starts. However, onsite power is recovered in eight hours which increases the likelihood of a core melt arrest and recovery of containment heat removal system. If core melt is not arrested and containment heat removal is not recovered, then fission products are released due to high containment pressure after 20 hours. Core melt arrest is discussed in Subsection 19D.5.8.

(b) Class IB-2

Class IB-2 events are a special class of Station Blackout events. The RCIC is available for core cooling for about eight hours, after which it is assumed to be unavailable. The suppression pool continues to heat up when RCIC is in operation. This impacts the time of containment structural failure and the time available for decay of fission products released during the accident.

(c) Class IB-3

Class IB-3 events are similar to Class IB-1 events except that onsite power is not recovered in eight hours. This leads to core melt and increased likelihood of fission product release.

(3) Class IC

Class IC events are ATWS events without boron injection coupled with loss of core cooling. Core melt occurs faster than it does for other Class I events. Conservatively, ATWS events with successful boron injection but with loss of core cooling are also included in this subclass.

(4) Class ID

Class ID includes low-pressure core melt events. These are transients followed by loss of high pressure core cooling, successful reactor vessel depressurization, and loss of low pressure core cooling. Following core melt and RPV failure, the molten core falls on the drywell floor. Unlike the Class IA event, low-pressure systems are not readily available to flood the molten core.

19D.5.3.2 Class II Events

Past analyses have shown that, as long as the core is kept covered with water, the containment response (especially the time required for containment overpressurization) is relatively independent of the type of initiating event. Therefore Class II events have not been divided into sub-classes.

19D.5.3.3 Class III Events

Theoretically, Class III events could be sub-divided like the Class I event with four classes—A, B, C and D. However, Subclasses B and C which would represent LOCA coincident with loss-of-offsite power and LOCA coincident with ATWS are events with negligible frequencies of occurrence and negligible contribution to risk. These are therefore grouped as part of Class IIIA events.

(1) Class IIIA

Class IIIA events are small or medium LOCAs with failure of high pressure coolant makeup systems followed by failure to depressurize the reactor. The low pressure coolant systems may be available but cannot inject water into the reactor because of the high reactor pressure. Core melt occurs with the reactor at high pressure. The core melt and subsequent RPV failure could result in ejection of molten corium at high pressure into the drywell, which could increase the potential for drywell failure. On the other hand, RPV failure would depressurize the reactor making the low-pressure systems available for flooding the molten core. A large LOCA is not a Class IIIA event because the break depressurizes the reactor.

(2) Class IIID

Class IIID events are LOCAs (small, medium or large) followed by failure of both the high pressure and low pressure coolant makeup systems. The reactor vessel is depressurized by the large LOCA or by the depressurization function for the small and medium LOCA. Following core melt and RPV failure, the molten core falls on the drywell floor. Unlike Class IIIA events, low pressure coolant makeup systems are not readily available to flood the molten core.

19D.5.3.4 Class IV Events

Class IV events are low probability events characterized by relatively fast containment over-pressurization and it is judged that further sub-classification of this event is not necessary.

19D.5.3.5 Class V Events

Theoretically, there could be pool bypass events associated with each of the four accident Classes I through IV. However, past PRAs have shown that frequencies of pool bypass events and their contribution to plant risk are low, and it is reasonable to group them all under one class without dividing them into subclasses.

19D.5.4 EQUIPMENT RECOVERY

Recovery of the following systems or functions has been modeled in the containment event trees:

Core Cooling
Containment Heat Removal
Onsite Power (includes diesel generators)
Offsite Powers
Equipment recovery is achieved through component repair. Typical repairs are fuse replacement, valve operator replacement, pump or motor replacement, etc.
System recovery probabilities are calculated using the exponential recovery formula:

P_f = Exponential (-T/MTTR)

where:

P_f = probability of failure to recover

T = Available time for repair

MTTR = Mean time to repair

A mean time to repair of 19 hours based on the WASH-1400 data, was assumed for the repair of most system components as long as the core and the RPV are intact. For events involving loss of offsite power or station blackout, the MTTR was based on recovery of onsite or offsite power.

For systems involving (multiple) redundant divisions of equipment, there is a potential for recovery of each of the failed divisions. For instance, if all three RHR loops failed, there is a potential that any one of them can be recovered ($1-P_f$), and the probability of failing to recover can be modeled as $P_f \times P_f \times P_f$. However, because of potential for common cause failure and limitations on the number of available operators etc., it was judged that the probability of failure to recover the failed function would be taken as half the value calculated for a single system (i.e., $0.5 \times P_f$) and not $P_f \times P_f \times P_f$.

A third type of failure considered in the fault trees and accident trees was failure of the operator to initiate the system. Failure to initiate could occur due to failure of the automatic initiation logic in combination with human error or a failure of the suppression pool temperature alarm. However, because there is a very long time to containment over-pressurization, the probability of recovering from human error is very high.

In order to determine the effects of these types of conservatisms, the fault trees were reevaluated with the appropriate nodes deleted. It was found that a recovery factor of 0.5 was appropriate.

The vast majority of remaining failure modes involved failures in the pump or valve rooms. If the pumps did not run after core damage began, then the radiation levels would be less than 0.1 Sv/h. Although this value is somewhat high, pump and valve rooms are still accessible. Therefore, the time available for RHR recovery is the time to containment over-pressurization. In order to represent the effects of radiation in the pump room, the failure to recover heat removal was multiplied by 2.

The time available for repair/recovery of each system was determined by the time in which the system had to be operating to prevent the occurrence of failure (core melt, containment overpressure, etc.). Repair times were obtained based on the core melt progression analysis discussed in Subsection 19E.2.

19D.5.5 CONTAINMENT CAPABILITY

The ABWR containment design pressure is 0.41 MPa. Past stress analyses performed for other PRAs have shown that the containments are capable of withstanding much higher pressure (typically 2-3 times the design pressure). A discussion of the ABWR containment capability is provided in Appendix 19F. The ultimate pressure capability of the ABWR containment is limited by that of the drywell head. The drywell pressure capability depends upon the temperature in the containment. At 533 K (500° F), the containment ultimate strength is evaluated to be 1.025 MPa.

19D.5.6 CONTAINMENT STRUCTURAL FAILURE MODES AND LOCATIONS

In recent years, many PRAs have focused on the issue of containment performance following a severe accident. Of special interest are events with early loss of containment structural integrity or suppression pool bypass, and events involving large releases of radioactivity. In the case of non-inerted containments, hydrogen generation and potential for subsequent hydrogen detonation are also of special interest. The ABWR containment is inerted.

19D.5.6.1 Containment Structural Failure Modes

See Appendix 19F

19D.5.6.2 Containment Failure Location & Probabilities

See Appendix 19F

19D.5.6.3 Failure Modes Explicitly Modeled in Containment Event Trees

The following containment failure modes are explicitly modeled in the CETs:

(1) Containment Over-pressurization

Containment fails in the drywell when subjected to high pressure resulting from steam and non-condensable gases.

(2) Containment Leakage

Containment seals (such as the drywell head seal) fail when subjected to a combination of high temperature and pressure [533 K (500° F) and 0.46 MPa].

(3) High Temperature Failure

When subjected to a very high temperature [e.g., greater than 644 K (700° F)], the drywell structural capacity is reduced due to reduction of material strength.

(4) Containment Failure at the Time of RPV Failure

Containment fails when the RPV fails due to factors such as direct containment heating, vapor suppression failure, missile generation, etc.

19D.5.6.4 Failures Modes Not Explicitly Modeled in Containment Event Trees

(1) Steam Explosion

In-vessel and ex-vessel steam explosions leading to containment failure are not credible events as discussed in Subsection 19E.2.3.1 and Attachment 19EB. Therefore, they are not explicitly modeled in the CETs.

(2) Hydrogen Detonation

The ABWR containment is inerted during plant operation and therefore, failure modes relating to hydrogen burning and detonation have been ruled out as having a negligible probability of occurrence. The risks associated with the small fraction of time (<1%) of ABWR plant operation when the containment is not inerted is negligible, since these are associated only with the plant startup or shutdown process, and inerting can be restarted if an accident is initiated. There is a potential for hydrogen combustion in the reactor building following the release of gases after containment structural failure. Since the containment structural failure directly results in suppression pool bypass in the ABWR CETs, special modeling of hydrogen combustion was considered unnecessary.

(3) RPV Rupture

RPV rupture, an initiating event which could potentially cause a structural failure of the ABWR containment, is judged to be a negligible contributor to risk.

(4) Basemat Penetration

Basemat penetration following core melt is not expected to result in the release of radioactive materials to the environment (Subsection 19E.2.1.3.6).

19D.5.7 SUPPRESSION POOL BYPASS

19D.5.7.1 Introduction

The magnitude of radioactive release to the environment for the severe accidents in which the suppression pool is bypassed is much higher than the severe accidents in which the release occurs through the suppression pool. Thus, suppression pool bypass paths are of special interest in BWR PRAs. This subsection discusses the various types of suppression pool bypass paths and describes how they are treated in the ABWR PRA.

Some of these bypass paths are explicitly modeled in the CETs. Others have been studied separately and found to be negligible contributors to ABWR plant risk. A summary of the various suppression pool bypass mechanisms and how they are treated in the ABWR PRA is provided in Table 19D.5-3.

19D.5.7.2 Ex-Containment LOCA

This bypass path is not modeled in the CETs.

19D.5.7.3 Failure of Isolation Valves and Pipe Ruptures

This bypass path is not modeled in the CETs.

19D.5.7.4 Failure of Drywell Vacuum Breaker

This bypass path is modeled in the CETs.

19D.5.7.5 Containment Structural Failure

The most likely structural failure of the containment occurs in the drywell. This failure mode bypasses the suppression pool and is modeled in the CETs. As discussed in Subsection 19D.5.6.3, two additional containment failure modes are modeled in the CETs. These also result in suppression pool bypass.

19D.5.7.6 Uncovery of Horizontal Vents

If after the RPV failure, the horizontal vents are uncovered due to low water level in the suppression pool, the pool will be bypassed. Calculations show that for all events other than ATWS, initial suppression pool inventory is sufficient to compensate for the evaporation loss for over 24 hours without uncovering the horizontal vents. The probability of the operator initiating suppression pool make up using normal water sources or the AC-independent water addition system, if necessary, within 24 hours is extremely high. Therefore this type of suppression pool bypass is not explicitly modeled in the CETs.

19D.5.7.7 Low Probability Bypass Events

Some events such as RPV rupture and in-vessel steam explosion, with extremely low probabilities of occurrence have the potential for causing suppression pool bypass. These are not specifically treated in the CETs because past PRAs have shown negligible contribution to risk attributable to these events. For references which provide additional details see Table 19D.5-3.

19D.5.8 CORE MELT ARREST SUCCESS CRITERIA

19D.5.8.1 Introduction

After core melt has been initiated, the process can still be arrested if the core debris is cooled with sufficient water. The success criteria for arresting core melt is described in this subsection. The analytical basis for the success criteria is developed in Subsections 19E.2.1.4.2 and 19E.2.1.4.3.

There are two ways to arrest core melt:

(1) During the early stages of an accident, core melt can be arrested prior to RPV failure,

(2) If the RPV has been breached, core melt can be arrested prior to loss of structural integrity of the containment.

In each case, a means of getting water to the corium and a means of removing heat from the containment are required. The core melt arrest success criteria is summarized in Table 19D.5-4.

19D.5.8.2 Core Melt Arrest Prior to RPV Failure

For arresting core melt within the RPV, one of the core cooling systems must be recovered within about an hour of the core melt initiation.

If the reactor is at high pressure, operation of one of the high pressure systems (HPCF B or C, RCIC, Feedwater System) must be restored.

If the reactor is at low pressure, in addition to the high pressure system, operation of one of the low pressure systems (LPFL, condensate injection) may be recovered. Alternatively, the AC-independent Water Addition System can provide sufficient core cooling.

CRD water supply when maximized and operation of both pumps is assumed to be sufficient to arrest core melt.

For removing the heat from the containment, one of the RHR loops must be available.

19D.5.8.3 Core Melt Arrest Prior to Loss of Containment Structural Integrity

Following RPV failure, core melt can be arrested by operating one of the two HPCF or any one of the low pressure systems (LPFL or condensate injection to the reactor vessel or diesel-driven fire water system). CRD water supply is assumed to be sufficient for arresting core melt when flow is maximized.

If none of the low pressure systems can be recovered in time to quench the corium on the lower drywell floor, the corium continues to heat up the lower drywell area. This melts the fusible material at the ends of pipes in the passive flooder system resulting in the transfer of the suppression pool water to the lower drywell area. This passive flooder system is described in Subsection 9.5.12. The suppression pool water quenches the molten corium and the core melt process is arrested. This process is modeled in the CETs by the node P, representing "passive mitigation". It should be noted that even after passive flooder operation, the suppression pool water level stays high enough to cover the horizontal vents.

After the core melt is arrested, it is still necessary to remove heat from the containment. Containment heat can be removed by operation of one of the RHR Systems. If the core melt is arrested but the RHR is not available, radioactivity is eventually released to the environment.

19D.5.9 Containment Release Categories

The amount of radioactive release to the environment depends upon a number of factors such as the timing of containment failure and the location of containment failure. Ideally, there is a specific radioactive release associated with each outcome of the containment event trees. However, evaluating the source terms for each event tree output is very time consuming. Therefore, the releases with similar characteristics are grouped ("binned") together to define release categories.

Detailed discussion of the binning process is provided in Subsection 19D.5.12.5.

19D.5.10 CONTAINMENT OVERPRESSURE PROTECTION

Subsection 6.2.5.2.5 describes a mitigation system called the containment overpressure protection subsystem of the atmospheric control system. This system protects the containment structural integrity and provides for controlled fission product release. If the containment nears its service level C limit, a rupture disk opens providing containment pressure relief. Since the system originates in the wetwell airspace, any fission product release will be scrubbed. The operation of COPS is indicated by "RD open" in the LCS mode of the CETs.

19D.5.11 DESCRIPTION OF CONTAINMENT EVENT TREE

19D.5.11.1 Subdivision of Accident Classes

Several of the accident subclasses were further subdivided to reduce the size of the individual trees. Each tree contains only sequences with high or low RPV pressure at the time of core damage. This subdivision was assigned to each accident subclass. Table 19D.5-5 summarizes this subdivision.

For event classes IB1, IB2, and IB3, the status of successful depressurization is not known as a direct result of the accident event trees. The depressurization system in the ABWR is automatic and does not rely on AC power. The operability of the ADS System is discussed in Subsection 19E.2.1.2.2, where it is shown that there is adequate DC power and nitrogen supply to actuate the ADS System during a blackout event. As a backup to the automatic actuation, the emergency procedure guidelines (EPGs), contained in Appendix 18A, require the operator to manually initiate the ADS System when the water level reaches the top of active fuel. Since the EPGs are symptom based, there is no differentiation between station blackout events and other events. Thus, there is no difference in the reliability of the ADS for station blackout events as compared to other transient events.

19D.5.11.2 Level 2 Results

The logic diagram shown in Figure 19D.5-3 groups the set of Level 2 sequences into source term categories (STC) based on similar sequence characteristics judged to be important to the definition of the offsite source term and consequences. Five parameters are used to define the source term categories. This grouping resulted in the definition of 53 distinct source term categories. The characteristics of each source term category are determined by the branch attributes for the pathway through the diagram. The five grouping parameters are discussed below.

19D.5.11.2.1 Initiator Code (INITCODE)

This parameter groups the sequences based on the accident sequence type. The accident sequence type definition is described in Subsection 19E.2.2 of Reference 19D.5-2. Note that sequence types NSCL (class IC) and NSCH (class IE) were of such low probability (having no accident sequences with probabilities above the value of 1E-12 attributable to these classes) that they were truncated prior to performing the Level 2 analysis and are not included in the grouping diagram. In addition, as a result of the low probability of Class IV ATWS sequences (sequence type NSRC), they were not evaluated in the Level 2 model although the Class IV frequency is shown in the logic diagram (STC 53).

19D.5.11.2.2 Core Melt Arrested In-Vessel (IV)

This parameter groups sequences based on whether late in-vessel cooling is successful in preventing vessel failure.

19D.5.11.2.3 Mode of Release (REL MODE)

This parameter groups sequences based on the mode of any fission product release from the containment. The following important characteristics are considered.

19D.5.11.2.3.1 Normal Containment Leakage

Containment pressurization is terminated, so there is no containment failure or COPS operation. These sequences have very small releases to the environment as a result of normal containment leakage.

19D.5.11.2.3.2 Rupture Disk

Operation of the COPS leads to nearly complete release of the noble gases. Other fission product releases are negligible.

19D.5.11.2.3.3 Drywell Head Failure

Long-term steam and non-condensable gas production lead to over-pressurization of the containment. The drywell head is probabilistically considered to be considerably weaker than its nominal value and fails before the COPS opens.

19D.5.11.2.3.4 Penetration Over-temperature Failure

High temperatures lead to failure of the large penetration seals in the drywell.

19D.5.11.2.3.5 Early Containment Failure

Overpressure failure of the drywell head occurs at the time of RPV failure.

19D.5.11.2.4 Pool Bypass (POOL_BP)

This parameter groups sequences based on whether radionuclides released into the drywell gas space bypass the suppression pool for fission product scrubbing. All drywell containment failure modes result in eventual pool bypass and no branching is required. For sequences without containment failure, this parameter is irrelevant. Hence, branching under this heading is only significant for the COPS release mode.

19D.5.11.2.5 Drywell Spray (SPRAY)

Operation of the drywell sprays can be effective in mitigating the release of radionuclides. However, for sequences where vessel failure has not occurred and sequences where pool bypass has not occurred, operation of the sprays is not significant since suppression pool scrubbing will effectively mitigate the radionuclide releases. Therefore, branching is only considered for sequences with pool bypass. Note that for sequences with drywell penetration high temperature failure, the drywell sprays are not operating and no branching is necessary.

<u>19D.5.11.3 Containment Event Trees for Classes I and III</u>

In order to quantitatively assess containment response, a hybrid containment safeguards/containment event tree was created for each significant accident class. A listing of top events has been created and includes a description of each event as well as the basis for its numerical value. Many events are dependent upon decomposition event trees (DET), to which the description will refer when necessary. There is a one-to-one correspondence between the nodes on the CETs and the set of DETs. Most of the nodes on the trees are system related. These nodes are described here. A few nodes deal with phenomenological uncertainties. These nodes are described in detail in the uncertainty analysis in Reference 1, Subsection 19E.2.7. Each branchpoint on the DET either assigns a probability to the event, or refers by rule to previous events on the CET or to the accident subclass. Assigned probabilities are shown on each branch. Branches which refer back to the CETs are termed sorting events. These are indicated on the tree by the symbol "<--".

19D.5.11.3.1 Operator Depressurizes Reactor (OP)

This DET, shown in Figure 19D.5-15, classifies the sequences based on the RPV pressure at the time of core damage. The only branch on the OP DET is the SUBCLASS event. Each subclass represents sequences which are either at full pressure or have been depressurized. Accident subclasses IB-1, IB-2, IB-3 are depressurized based on the accident subclass division discussed in 19D.5.11.1. All ID and IIID were depressurized in the accident event trees contained in 19D.4. Sequences in the remaining accident subclasses have not been depressurized. Accident classes II and IV have not been subdivided into high- and low-pressure subclasses since this information was not necessary for the CET analysis. This is a sorting type event quantified as 1 or 0 based on the accident subclass.

19D.5.11.3.2 Containment Heat Removal Available (CHR)

The second node of the CET, CHR, and its DET, shown in Figure 19D.5-16, check if containment heat removal was available at the beginning of the accident. Only accident subclasses IA and IIIA can have RHR available at the time of core damage. The CHR event is a sorting type event quantified as 1 or 0 based on the accident subclass. For analysis in classes IA and IIIA, the CHR event probability is based upon that of the gate WDCS in the Level 1 MOR.

19D.5.11.3.3 Core Melt Arrested in RPV (ARV)

The ARV node and DET, shown in Figure 19D.5-17, assess the probability that core damage is arrested in-vessel and vessel failure is prevented. In order to prevent RPV failure, it is assumed that an in-vessel injection source must be recovered well before the time at which the vessel would otherwise fail. The success criteria for core melt arrest in-vessel are shown in Table 19D.5-4. The probability of in-vessel core damage arrest varies for different subclasses because of differences in RPV pressure, availability of AC power, sequence timing, RHR availability and other factors. Gate ECCS is quantified and used to assess this event.

19D.5.11.3.3.1 Accident Subclass (SUBCLASS)

The first event in the DET separates the accident subclasses into groups with similar in-vessel recovery probabilities. This event is a sorting type event with an assigned probability of 0 or 1 based on the sequence subclass.

19D.5.11.3.3.2 Core Melt Arrested in RPV (ARV)

This event assigns a probability for in-vessel core melt arrest. Four cases were identified during the quantification for this event:

(1) Case 1 - Subclasses IA and IIIA. These subclasses represent high RPV pressure sequences with failure of high pressure injection. In order to arrest the core damage progression, recovery of a high-pressure injection system is required. Approximately one hour is available for system recovery. A recovery probability is calculated assuming a mean time to repair (MTTR) of 19 hours for the failed system:

Core melt arrest	0.05
No core melt arrest	0.95

(2) Case 2 - Subclass IB-2. The subclass contains station blackout sequences with RCIC operation for eight hours and with operator depressurization of the RPV. If in-vessel injection is re-established within about two hours of loss of RCIC, core damage can be arrested and vessel failure can be prevented, as discussed in Subsection 3.1.14. If power is recovered, AC powered high- or low-pressure injection systems can cool the core and prevent vessel failure. The conditional probability of recovering power in this 2-hour period is obtained from the EPRI KAG Table A2-2. A value of 0.6 is obtained by dividing the non-recovery probabilities for 8 and 10 hours. In addition, injection using the firewater addition system can also provide late in-vessel core cooling (Subsection 3.1.14). The operator is expected to monitor the availability of DC power during the blackout, so there will be approximately 10 hours of warning time before use of the firewater addition system is necessary. The firewater system is assigned a failure probability of 0.01 based on operator error probability.

This yields combined probabilities of:

Core melt arrest	0.994
No core melt arrest	0.006

(3) Case 3 - Subclasses ID and IIID. These subclasses contain low RPV pressure sequences with failure of all low pressure injection. In order to arrest the core damage progression, recovery of a high- or low-pressure injection system or operation of the firewater addition system must occur within 1 hour (Subsection 19E.2.4.2 of Reference 1). The probabilities are assigned based on failure of the operator to initiate injection:

Core melt arrest	0.9
No core melt arrest	0.1

(4) Case 4 - All Other Subclasses Excluding Class II. For all other subclasses, AC power is not available; therefore, recovery of in-vessel injection in time to prevent vessel failure is not considered:

Core melt arrest	0.0
No core melt arrest	1.0

19D.5.11.3.4 Containment Intact at RPV Failure (CI)

This node indicates if the containment survives any energetic events which occur at vessel failure. Since fuel coolant interactions were ruled out as a significant contributor to containment failure (Subsections 19E.2.3.1 and 19E.2.6.7) direct containment heating is the only contributor to this node. A representative decomposition event tree is shown in Figure 19D.5-18 for completeness. However, the descriptions of the DET events and quantification is given in the uncertainty analysis for DCH in Subsection 19EA.2.

19D.5.11.3.5 Active Injection to the Lower Drywell (LDWI)

This node and its decomposition event tree, shown in Figure 19D.5-19, assess the probability that an active injection system to supply water to the lower drywell is available at, or soon after, RPV failure. High- or low-pressure in-vessel injection systems which deliver water to the vessel after vessel failure will result in water flowing from the vessel to the lower drywell. In addition, the AC-independent firewater addition system can inject water to either the vessel or the upper drywell sprays.

19D.5.11.3.5.1 High-pressure Injection Recovered (HPI)

Recovery of a high-pressure injection system after vessel breach can supply water to the lower drywell. However, recovery of high-pressure injection following vessel breach has been conservatively neglected in this analysis.

19D.5.11.3.5.2 Accident Subclass (SUBCLASS)

This event separates the accident subclasses into groups with similar conditions for low pressure in-vessel injection availability and firewater spray operation. This event is a sorting type event which has an assigned probability of 0 or 1 based on the sequence subclass.

19D.5.11.3.5.3 Low-pressure Injection Available after RPV Failure (LPI)

This event assesses the probability that low-pressure in-vessel injection will be available after RPV failure. There were two cases identified for the quantification of this event:

(1) Case 1 - Subclasses IA and IIIA (CHR Success branch only). These subclasses represent high RPV pressure sequences with failure of the high-pressure injection system and with the RHR System available. For this group of sequences, the probability of operation of low-pressure injection after RPV failure is very high:

Low-pressure Injection Available	0.999
Low-pressure Injection Not Available	0.001

(2) Case 2 - All Other Subclasses. For all other subclasses, the RHR System is not available at the onset of core damage, nor was it available for in-vessel recovery. Consequently, for this group of sequences the probability of operation of low-pressure injection after RPV failure was conservatively set to zero:

Low-pressure Injection Available	0.0
Low-pressure Injection Not Available	1.0

19D.5.11.3.5.4 Firewater Injection to Drywell Sprays (FWS)

This event assesses the probability that the operators initiate the firewater injection system in the drywell spray mode following RPV failure. Injection via the sprays will cause the suppression pool level to increase and will eventually cause overflow into the lower drywell. Injection to the vessel provides immediate flooding of the lower drywell. Given the presence of the passive flooder (considered in the next node), there is virtually no sensitivity for the lower drywell to the use of the spray versus vessel injection mode. Therefore, in order to simplify later trees, injection is presumed to occur via the spray system and injection via the vessel is neglected.

The pumps of the AC-independent Water Addition System are continuously charging. Furthermore, the onsite firewater system can be backed up by the use of fire trucks. Therefore, failure to inject is dominated by operator error. Two cases were identified for this event:

(1) Case 1 - All Short-term Core Melt Subclasses. For this case, the operator has several hours to successfully initiate the firewater addition system. Although the firewater injection system was not operated quickly enough to arrest the core melt in the vessel, there was only a short time available for that action. Therefore, it is judged highly probable that the operator will properly operate the firewater system:

Firewater spray	0.99
No Firewater spray	0.01

(2) Case 2 - Long-term Core Melt Subclass, IB-2. In this case, somewhat more time had been available for the operator to prevent vessel breach via the firewater addition system. However, there is substantially longer time for successful firewater injection in containment:

Firewater spray	0.95
No Firewater spray	0.05

19D.5.11.3.5.5 Active Injection to Lower Drywell (LDWI)

This event has no branching. It simply summarizes the branch decision taken in the previous branches in the DET. Thus, the summary branches are:

(1) In-vessel Injection (LPI provides lower drywell injection),

(2) Firewater Spray (The firewater system injects through the drywell sprays causing the suppression pool to overflow into the drywell),

(3) No DW Injection (Active injection systems do not supply water to the lower drywell).

19D.5.11.3.6 Passive Mitigation (P)

This node, shown in Figure 19D.5-20, assesses the probability that the passive flooder system operates to cover the debris in the lower drywell with suppression pool water after RPV failure. It is assumed for this node that no active injection systems have supplied water to the lower drywell, since this makes operation of the passive flooder unnecessary and the high temperatures necessary for flooder operation will not occur if active injection operates. Since the only requirement for operation of the passive flooder is the melting of the fusible valves near the drywell floor, operation of this system is considered to be extremely likely:

Passive Mitigation	0.999
No Passive Mitigation	0.001

19D.5.11.3.7 High-temperature Failure (HTF)

This subsection describes the decomposition event tree (Figure 19D.5-21) used to assess the probability that high temperature in the upper drywell will result in seal degradation and excessive leakage through the large movable penetrations in the upper drywell. The potential for seal degradation is presumed to exist if the temperature exceeds 533 K (500°F). Two situations could lead to this condition.

High-pressure melt ejection (HPME) may entrain significant quantities of core debris into the upper drywell in cases with high RPV pressure. In this situation, operation of the upper drywell sprays is required to assure that the upper drywell temperature remains below 533 K (500°F).

For sequences where HPME does not occur, high temperatures may result if the lower drywell is not flooded and the drywell sprays do not operate.

Thus, upper drywell high temperature failures will be prevented in all cases by operation of the sprays in the upper drywell. Consequently, no branching under the HTF event heading in the CET is made if firewater injection was successful in drywell spray mode (Branch FW SPRAY in CET event LDWI) and the HTF DET is not evaluated for those sequence pathways.

19D.5.11.3.7.1 Accident Subclass (SUBCLASS)

The first event in the DET segregates the accident subclasses into two groups. For all accident subclasses except subclasses IA and IIIA when they include RHR success, operation of the drywell sprays has already been determined in CET event LDWI. For subclasses IA and IIIA that include RHR with successful LPI after RPV failure, the question of drywell spray availability was not asked in the CET event LDWI since successful lower drywell water addition was already known to exist via the RHR System.

This event is a sorting type event which has an assigned probability of 0 or 1 based on the sequence subclass.

19D.5.11.3.7.2 Operator Depressurizes Reactor (OP)

This event classifies the accident subclasses into high and low RPV pressure at RPV failure. For sequences with high RPV pressure, HPME at the time of RPV failure may result in debris entrainment into the upper drywell.

This event is a sorting type event which has an assigned probability of 0 or 1 based on the branch pathway followed under CET event heading OP.

19D.5.11.3.7.3 Mode of Active Injection to Lower Drywell (LDWI)

This event assesses whether the active injection systems have flooded the lower drywell soon after RPV failure. This event is a sorting type event which has an assigned probability of 0 or 1 based on the branch pathway followed under CET event heading LDWI.

19D.5.11.3.7.4 Drywell Sprays Operate (DW_SPRAY)

This event assesses whether the upper drywell sprays are available. Note that this event is only relevant for sequences in subclasses IA and IIIA with RHR success. For all other sequences entering this DET, failure of the drywell sprays has been previously determined in event LDWI.

(1) Case 1 - Subclasses IA and IIIA, with RHR. Sequences in these subclasses involve high RPV pressures with the RHR System available. Under these conditions, it was determined that the operation of the upper drywell sprays would be extremely likely:

Drywell Spray	0.999
No Drywell Spray	0.001

(2) Case 2 - All Other Subclasses. No consideration of these cases is necessary since operability of the drywell sprays was determined previously.

19D.5.11.3.7.5 Water Supply to Lower Drywell (LDW)

This event assesses whether the lower drywell is flooded by the active injection systems or the passive flooder after RPV failure. For sequence pathways with successful in-vessel injection (IN_VESSEL INJ) or successful use of firewater sprays under CET event heading LDWI, there will be water in the lower drywell and no branching is taken. For sequences with failure of active injection (NO DW INJECT), successful water addition to the lower drywell is determined by the success of the passive flooder (Branch PASSIVE MIT) in CET event P.

This event is a sorting type event which has an assigned probability of 0 or 1 based on the branch pathway followed under CET event heading P.

19D.5.11.3.7.6 High-temperature Failure (HTF)

This event assesses whether high-temperature failure of the large moveable penetrations in the upper drywell occurs. Four cases were identified in the quantification for this event:

(1) Case 1 - Sequences with Drywell Sprays Available. For these sequences, drywell high-temperature failure will be prevented:

No high-temperature failure	1.0
High-temperature failure	0.0

(2) Case 2 - Low-pressure Sequences with Water Supply to the Lower Drywell. For these sequences, debris entrainment to the upper drywell does not occur. The debris in the lower drywell is submerged in a pool of water. Hence, high temperatures in the upper drywell will be prevented:

No high-temperature failure	1.0
High-temperature failure	0.0

(3) Case 3 - Sequences with no Drywell Sprays and no Water Supply to the Lower Drywell. For these sequences, the debris in the lower drywell is not submerged in a pool of water. Hence, core concrete attack in the lower drywell would generate high-temperature gasses which would rise into the upper drywell. Consequently, high temperatures in the upper drywell could be expected:

No high-temperature failure	0.0
High-temperature failure	1.0

(4) Case 4 - High-pressure Sequences with no Drywell Sprays and with Water Supply to the Lower Drywell. For these high-pressure sequences, the debris in the lower drywell will be submerged in a pool of water. However, if HPME results in debris entrainment into the upper drywell then high temperatures in the upper drywell will occur since drywell sprays are not available. This event is quantified based on the probability of HPME occurring (Subsection 19EA.2.1.5 for a further discussion on the probability of HPME):

No high-temperature failure	0.2
High-temperature failure	0.8

19D.5.11.3.8 Core Debris Concrete Attack (CCI)

This node indicates if a substantial amount of core concrete attack occurs, and if so, whether the attack occurs in the presence of water. The decomposition event tree is shown in Figure 19D.5-22 for completeness; however, the descriptions of the DET events and quantification is given in the uncertainty analysis for Debris Coolability in Subsection 19EC.2.1.

19D.5.11.3.9 Pedestal Failure (PED)

This node indicates if the pedestal fails as a result of core concrete attack. If the pedestal fails, it is assumed that tipping of the vessel will lead to tearing of the containment penetrations associated with the vessel, allowing fission product release. The decomposition event tree is shown in Figure 19D.5-23 for completeness; however, the descriptions of the DET events and quantification is given in the uncertainty analysis for Debris Coolability in Subsection 19EC.2.2.

19D.5.11.3.10 RHR Recovered Prior to Fission Product Release (RCH)

This subsection describes the decomposition event tree which assesses the probability that containment heat removal (primarily RHR) is recovered prior to the release of fission products through the containment overpressure protection system or as a result of drywell head failure (Figure 19D.5-24). The probability of RHR recovery varies for different accident subclasses: for sequences with core damage terminated in-vessel, and for sequences with active injection to the lower drywell after RPV failure because of differences in the availability of AC power, sequence timing, and other factors.

19D.5.11.3.10.1 Accident Subclass (SUBCLASS)

The first event in the DET segregates the accident subclasses into groups with similar RHR recovery probabilities. This event is a sorting type event which has an assigned probability of 0 or 1 based on the sequence subclass.

19D.5.11.3.10.2 Core Melt Arrested in Vessel (ARV)

This event sorts the sequences into those with in-vessel core damage progression termination and those where RPV failure occurs. This event is a sorting type event which has an assigned probability of 0 or 1 based on the branch taken under CET event heading ARV.

19D.5.11.3.10.3 Active Injection to the Lower Drywell (L_DW_INJ)

This event classifies sequences into those with active injection into the lower drywell after RPV failure and those without active injection. This event is a sorting type event which has an assigned probability of 0 or 1 based on the branch taken under CET event heading LDWI.

19D.5.11.3.10.4 RHR Recovered Prior to Fission Product Release

Recovery of the RHR System is described in 19D.5.4. The probabilities are based on knowledge about the use of active injection, and the time available for recovery. The impact pool bypass could have on the probability of RHR recovery is discussed in 19D.5.13.

19D.5.11.3.11 Pool Bypass (POOL_BP)

This node indicates if pool bypass occurs. The decomposition event tree is shown in Figure 19D.5-25 for completeness; however, the descriptions of the DET events and quantification is given in the uncertainty analysis for Pool Bypass in Subsection 19EE.2.

19D.5.11.3.12 Late Containment Status (LCS)

This subsection describes the decomposition event tree, shown in Figures 19D.5-26 and 19D.5-27, used to assess the containment status late in the accident sequence progression. Figure 19D.5-26 is used for cases where the RHR is known to be available. Figure 19D.5-27 is used for cases where RHR is not initially available. The two trees are otherwise identical.

19D.5.11.3.12.1 Vapor Suppression Available Late (VSL)

The first event in the DET determines whether vapor suppression is effective late in the accident sequence. Vapor suppression will be failed if there is a bypass of the suppression pool. This event is a sorting type event which has an assigned probability of 0 or 1 based on the branch followed in the CET event POOL_BP.

19D.5.11.3.12.2 Type of CCI in Lower Drywell (CCI)

This event determines whether core concrete interaction (CCI) is occurring in the lower drywell. Even with effective containment heat removal available, the occurrence of CCI may result in sufficient production of non-condensable gasses to over-pressurize the containment and result in fission product release. This event is a sorting type event which has an assigned probability of 0 or 1 based on the branch followed in the CET event CCI. Dry CCI branches are not shown here because they are always presumed to have high-temperature failure (Node HTF).

19D.5.11.3.12.3 RHR Recovered Prior to Fission Product Release (RCH)

For accident classes with RHR unavailable during core damage, this event determines whether RHR is recovered prior to fission product release. This event only applies to sequences in which RHR is not initially available. This event is a sorting type event which has an assigned probability of 0 or 1 based on the branch followed in the CET event RCH.

19D.5.11.3.12.4 Mode of Drywell Spray Operation (DW_SPRAY)

This event assesses the mode of drywell spray operation. Drywell sprays mitigate the effects of loss of vapor suppression for both large and small bypass areas. Drywell spray is considered using either the RHR System or the firewater injection system. If the firewater system is used, and vapor suppression is successful, the water added to the containment will increase the suppression pool elevation. Consequently, the drywell-to-wetwell pressure differential required to clear the horizontal vents will also increase. Thus, the probability that drywell head failure will occur prior to the opening of the rupture disk will increase slightly based on uncertainties in the drywell head ultimate strength.

This event is a sorting type event which has an assigned probability of 0 or 1 based on the branch followed in the CET event LDWI.

19D.5.11.3.12.5 Containment Pressure Exceeds Rupture Disk Setpoint (C_PRESS)

This event evaluates the probability that the wetwell pressure exceeds the COPS rupture pressure of 0.72 MPa. Three cases were considered in the quantification of this event:

(1) Case 1 - Vapor Suppression OK, No CCI in Lower Drywell and RHR Available or Recovered. For this set of sequences, containment heat removal is available and no non-condensable gasses are generated from CCI. Hence, containment pressure will remain below the COPS rupture pressure:

< COPS Rupture Pressure	1.0
> COPS Rupture Pressure	0.0

(2) Case 2 - RHR not Available and not Recovered. For this set of sequences, the absence of effective containment heat removal will lead to containment pressure eventually exceeding the rupture disk setpoint:

< COPS Rupture Pressure	0.0
> COPS Rupture Pressure	1.0

(3) Case 3 - CCI in Lower Drywell and RHR Available or Recovered. For this set of sequences, containment heat removal is available. However, the generation of non-condensable gasses from CCI in the lower drywell may result in the containment pressure exceeding the COPS rupture pressure. Based on a review of MAAP calculations it is considered likely that the containment pressure will exceed the COPS rupture pressure under these conditions:

< COPS Rupture Pressure	0.1
> COPS Rupture Pressure	0.9

19D.5.11.3.12.6 Rupture Disk Opens (RD)

This event estimates the probability that the rupture disk will open prior to drywell head failure given that the containment pressure exceeds the COPS rupture pressure. The value used for this event is taken from that of the Level 1 event COPS.

(1) Case 1 - Classes I and III

For these classes, the rupture disk is not presumed to fail to open as part of the original accident sequence, and it is still available for containment pressure control:

Rupture Disk Opens	0.9999
Rupture Disk Does not Open	1.0E-4

(1) Case 1 - Class II

For Class II, the rupture disk is presumed to fail to open as part of the original accident sequence, and it is therefore unavailable for containment pressure control:

Rupture Disk Opens	0.0
Rupture Disk Does not Open	1.0

19D.5.11.3.12.7 Late Containment Status (LCS)

This event has no branching. It simply summarizes the decisions taken in previous branches. Three possible outcomes are considered: the containment is intact, the rupture disk opens, or the drywell head fails.

<u>19D.5.11.4 Decomposition Event Trees for Class II</u>

The containment event tree (CET) for Class II sequences is shown in Figure 19D.5-10. The supporting DETs are shown in Figures 19D.5-28 through 19D.5-30. This CET is substantially different from those for the Class I events. Class II consists of sequences with loss of containment heat removal (CHR) but with successful in-vessel injection. If CHR is not recovered within about 20 hours, the containment pressure will exceed the COPS rupture pressure.

The first event in the CET assesses the probability of recovery of the RHR System prior to COPS operation (or containment overpressure failure) given that RHR was not initially available (Figure 19D.5-28). If the RHR System is successfully recovered, containment pressure will decrease and the event will be terminated. This probability is estimated assuming a mean time to repair of 19 hours for the system.

The second event in the Class II CET assesses the probability that the COPS rupture disk opens prior to drywell head failure for sequences without recovery of RHR (Figure 24). The failure of the rupture disk to open is requisite for Class II classification.

The third event in the CET assesses the probability that drywell head failure will result in loss of in-vessel injection and core damage (Figure 25). A discussion of the considerations and assumptions used to estimate these event probabilities is provided below.

19D.5.11.4.1 Loss of In-vessel Injection Given Venting with COPS

The COPS is designed to vent the wetwell gas space when the wetwell pressure exceeds 0.72 MPa. As discussed below, high-suppression-pool temperatures or loss of NPSH will not threaten the ability of in-vessel injection systems to operate for an extended period of time after COPS initiation. In addition, random failures of the in-vessel injection systems during their mission time have been considered in the Level 1 analysis.

Due to these considerations, there is a negligible probability of failure of in-vessel injection given success in COPS operation.

19D.5.11.4.2 Loss of In-vessel Injection Given Containment Failure

The node CC models the probability that core cooling will be impacted following structural failure of the containment. The quantification of this node is described below.

For cases in which the core is successfully cooled but the containment is not, the containment will pressurize. If the rupture disk fails to open, the containment boundary will eventually be breached. But if core cooling is maintained, the offsite consequences of the breach will be negligible. If the containment boundary failure causes core cooling failure, the consequences would be more severe. Therefore, this potential was reviewed.

The following general areas were reviewed and are briefly discussed below:

(1) drywell head failure,

(2) high temperatures in the suppression pool,

(3) high drywell temperatures.

The most likely containment failure location is the drywell head. Drywell head failure would pressurize the relatively small volume between the head and concrete shield plugs. This could levitate some of the plugs which would then fall, potentially causing equipment damage. There is no potential for plugs falling between the reactor vessel and drywell wall because the annular space is too small. The vessel vent could be damaged but the consequences would be no worse than a small LOCA. Although unlikely, plugs could fall through the vertical equipment hatch and damage electrical equipment and/or an RHR heat exchanger. It is extremely unlikely that more than one division of core cooling would be lost as a result.

High temperatures in the suppression pool would result in increased suction temperature for core cooling pumps. However, pump performance should not be impaired because the pumps are designed for water temperatures as high as 455 K (360° F). Further, condensate storage tank water and fire tank water temperatures would not be affected.

High drywell temperatures were considered for their potential effects on SRV performance, electrical equipment, and water level instrumentation. SRV performance should not be degraded because the expected temperature/time history is less severe than the LOCA condition for which the SRVs will be qualified. There is no electrical equipment in the drywell which is required to operate to establish or maintain core cooling. Effects on water level instrument accuracy should be small since the reference and variable legs experience the same elevation drop in the drywell.

After reviewing these potential causes of core cooling loss resulting from high temperature conditions/containment failure, it was judged that the probability of core cooling loss ranged between 0.01 and 0.001. A value of 0.01 was used in the analyses for loss of conventional core cooling. In the class II sequences derived from the Level 1 PRA, firewater availability had not been considered. Firewater can be used as an additional source of water following containment failure. The firewater system is much less vulnerable to containment failure. The combined failure probability of conventional cooling and firewater is estimated to be 0.0001, but a value of 0.001 was used for conservatism.

19D.5.12 DISCUSSION OF RESULTS

19D.5.12.1 Introduction

The results of the containment event tree analyses are discussed in this subsection. To recapitulate, the accident sequence event trees described in Subsection 19D.4 identified nine accident classes.

19D.5.12.2 Core Damage Frequency

The total internal event core damage frequency (CDF) calculated from the sum of all release frequencies except STC# 49, 50, and 51, which do not have core damage, on Figure 19D.5-3 is 9.80E-8 per reactor-year. Classes I, III and IV result in core damage. Also, a negligible fraction of all class II events results in core damage. This low CDF value is attributable to the recovery of failed systems (or AC power), the ability of the ABWR RHR pumps to pump saturated water without cavitating and the ability of core cooling systems to continue to inject water to the reactor following operation of the COPS.

19D.5.12.3 Core Melt Arrest

Of the sequences resulting in core damage, 86.7% result in the core melt being arrested either in the RPV or in the containment, without significant fission product release. This means that in virtually all of the accident sequences, either radioactive material remains in the reactor vessel or is contained within the containment boundaries and not released to the environment (except through normal containment leakage). This is attributable to equipment and power recovery prior to containment failure and to "passive mitigation," i.e., flooding of the molten core from the suppression pool water when passive flooder system actuates. The frequency of core damage with significant fission product release, which includes all categories except NCL and OK, is 1.31E-8 per reactor-year.

The containment design incorporates a containment overpressure protection system which is designed to ensure that any sequence which is not arrested in the containment will have low consequences. This system consists of a line originating in the wetwell which exhausts to the plant stack. If the containment pressure rises to a level where containment integrity could be challenged, a rupture disk opens relieving the containment pressure. If there is no suppression pool bypass, the containment usually does not reach the rupture disk setpoint for about 24 hours. This ensures a late release with low magnitude. The frequency of these events is 1.21E-8, or 12% of all core damage events. The frequency of all other release events is only 9.94E-10. Thus, the upper bound for releases with the potential to be early or have high magnitude is 1.01%.

19D.5.12.4 Probability of Containment Structural Failure Due to Loss of Heat Removal

One of the goals of the ABWR design is to assure that highly reliable heat removal systems be provided to reduce the probability of containment failure by loss of heat removal.

The frequency of this sequence is 9.92E-6/yr, though core damage only occurs during these events at a frequency of 8.56E-10/yr, or in 0.0086% of Class II sequences. This incredibly low number demonstrates that the goal is met for the ABWR design. The ABWR features and other factors that contribute to this low value are:

(1) Three divisions of heat removal systems.

(2) Ability to re-establish the main condenser as a heat sink in certain accidents.

(3) Ability to remove heat using CUW heat exchanger.

(4) Long times before containment pressure reaches a value which could threaten containment integrity, which enables recovery of power and failed heat removal systems.

(5) Presence of the containment overpressure protection system.

(6) Ability of the core cooling systems to continue to maintain the core cooling function following structural failure of the containment.

19D.5.12.5 Frequencies for Radioactive Release Categories

The important release characteristics for each of the severe accident sequences are summarized in Figure 19D.5-3. The first branch of the tree identifies the initiating event for each sequence. This information is used to specify the first four letters of the severe accident sequences used for the deterministic analyses performed in Subsection 19E.2.2. Later branches identify the potential impact of other important issues such as flooder operation and mode of fission product release. Table 19D.5-7 identifies the deterministic accident sequence associated with each of the end states in Figure 19D.5-3 with a frequency of at least 1E-11. Note that all sequences with an intact containment and no rupture disk opening are assigned to class NCL (Normal Containment Leakage). Sequences with a frequency of less than 1E-11 are neglected.

The deterministic sequences are then binned according to the characteristics of the fission product release. Table 19E.3-6 indicates combination of the deterministic sequences into release bins. This combination was done by considering the timing and magnitude of the releases. Column P(i) of Table 19E.3-6 gives the probabilities associated with each of the consequence bins with frequency above 1E-10. These values are simply the result of summing all of the sequences in a given consequence bin.

STC #53 in Figure 19D.5-3 was binned with Case 9, the worst of the consequence bins. This is a very conservative assumption since the frequency associated with this sequence is the initiating event frequency for ATWS events. The assumption is made to simplify the analysis because there is a negligible effect on the consequence analysis. If this assumption impacts the risk, a containment event tree should be developed for ATWS events.

19D.5.13 SENSITIVITY OF CONTAINMENT PERFORMANCE ANALYSIS TO RHR RECOVERY ASSUMPTIONS

Deleted

19D.5.14 SENSITIVITY OF RCIC CAPABILITY DURING LOSS OF CONTAINMENT LONG TERM HEAT REMOVAL

Deleted

19D.5.15 REFERENCES

19D.5-1 Eddy, C., Establishment of Model to Evaluate Plant Specific Changes, RSC Engineers, Inc. RSC 08-06, April 2010.

19D.5-2 ABWR Standard Safety Analysis Report, Revision 5, General Electric Company.

Table 19D.5-1
Description of Accident Event Classes

Event	Boron Injected?	Core Cooling Available?	Containment Heat Removal Available?	Relative Time of Core Melt and Containment Structural Failure	Accident Class
Transient	Not Applicable (N/A)	No	Yes	Core Melts First	I
Transient	N/A	No	No	Core Melts First	I
Transient	N/A	Yes	Yes	Successful Mitigation	Plant OK
Transient	N/A	Yes	No	Containment Fails First	II
LOCA	N/A	No	Yes	Core Melts First	III
LOCA	N/A	No	No	Core Melts First	III
LOCA	N/A	Yes	Yes	Successful Mitigation	Plant OK
LOCA	N/A	Yes	No	Containment Fails First	II
ATWS	Yes	No	Yes	Core Melts First	I
ATWS	Yes	No	No	Core Melts First	I
ATWS	Yes	Yes	Yes	Successful Mitigation	Plant OK
ATWS	Yes	Yes	No	Containment Fails First	II
ATWS	No	No	Yes	Core Melts First	I
ATWS	No	No	No	Core Melts First	I
ATWS	No	Yes	Yes	Containment Fails First	IV
ATWS	No	Yes	No	Containment Fails First	IV
Containment Bypass	N/A	No	Yes or No	Containment (and suppression pool) is bypassed. Therefore, no suppression pool scrubbing of radioactive releases. Relative time of core melt and containment failure immaterial.	V
Containment Bypass	N/A	Yes	Yes or No	Successful Mitigation	Plant OK

Table 19D.5-2
Description of Accident Class I Sub-classes

Event	Boron Injected?	Reactor Pressure	Core Cooling Available?	Accident Class	Comments
All transients except certain station blackout (SBO) events	Not Applicable (N/A)	High	No	IA	Because of high reactor pressure, there is a potential for containment structural failure shortly after vessel failure.
Station blackout events	N/A	High	No	IB1	No core cooling or containment heat removal at the beginning because of absence of on-site and off-site power and RCIC failure. However, on-site power recovered in eight hours increasing the likelyhood of recovery of core cooling and containment heat removal.
Station blackout events	N/A	High	RCIC Available for the first eight hours	IB2	Sequence with core decay heat at time of core melt reduced due to RCIC operation. Also suppression pool heats up prior to core melt shortening the time to containment structural failure.
Station blackout events	N/A	High	No	IB3	No core cooling or containment heat removal.
ATWS	Yes or No	High or Low	No	IC	
All transients	N/A	Low	No	ID	

Table 19D.5-3
Treatment of Suppression Pool Bypass Mechanisms in the PRA

Suppression Pool Bypass Mechanism	How Treated in the PRA	Reference Section/ Subsection
1. Ex-Containment LOCA - High Pressure Systems - interfacing Systems LOCA	Not modeled in CETs	
2. Failure of Isolation Valves, Pipe Rupture	Not modeled in CETs	
3. Normal Containment Leakage (containment temperature < 260°C and pressure < 0.72 MPa in the wetwell)	Modeled in CETs	19E.2.4.3
4. Containment Leaks (due to high containment temperature >260°C and pressure <0.46 MPa)	Modeled in CETs	
5. Containment structural failure due to overpressure (> 0.72 MPa)	Modeled in CETs	
6. High Temperature Failure of the containment (>371°C)	Modeled in CETs	
7. Uncovery of Horizontal Vent	Not expected to occur in the first 24 hours and therefore not modeled in CETs	19D.5.7.6
8. Low Probability Events	Not modeled in CETs	
- RPV Rupture		19D.5.6.4
- In-Vessel Steam Explosion		19E.2.1.3.1
- Ex-Vessel Steam Explosion		19E.2.1.3.1
- Basemat Penetration Following Core Melt		19E.2.1.3.6
9. Vacuum breaker leakage of failure	Modeled in CET	

Table 19D.5-4
Success Criteria for Core Melt Arrest

Case	Reactor at High Pressure	Reactore at Low Pressure
Core Melt Arrest in RPV	**Core Cooling** 1 of 2 HPCF or RCIC or FW **Containment Heat Removal** 1 of 3 RHR	**Core Cooling** 1 of 2 HPCF or 1 of 3 LPFL or Condensation Injection or Fire Water Injection System **Containment Heat Removal** 1 of 3 RHR
RPV Fails But Core Melt Arrested Prior to Fission Product Release	Not applicable	**Core Cooling** 1 of 2 HPCF or 1 of 3 LPFL or Condensation Injection or Fire Water Injection System or Passive Flooder System **Containment Heat Removal** 1 of 3 RHR

Table 19D.5-5
Division of Accident Subclasses

Accident Class	Subclass	RPV Pressure	Frequency
I	IA	High	3.04E-09
	IB-1	Low	1.38E-08
	IB-2	Low	3.32E-08
	ID	Low	4.47E-08
II	II	N/A	8.56E-10
III	IIIA	High	4.29E-11
	IIID	Low	1.56E-09
IV	IV	N/A	7.97E-10

Table 19D.5-6 Not Used

Removed from study

Table 19D.5-7
Binning of Containment Event Tree Results

STC #	Deterministic Bin	
1	NCL	
4	NCL	
5	LCHPPFP	
6	LCHPFSR	
8	LCHPPBR	See Notes
10	LCHPPBD	See Notes
12	LCHP00E	
13	NCL	
14	LCLPFSR	See Notes
15	LCLPFSR	See Notes
16	NCL	
18	LCLPFSR	See Notes
19	LCLPFSD	See Notes
21	LCLPFSD	
25	NCL	
26	LCLPFSR	See Notes
28	NCL	
30	SBRCPFR	
37	NCL	
38	LBLCFSR	See Notes
40	NCL	

Notes:

Sequences 8 and 10: Releases taken for worst case scenarios from suppression pool bypass study in Attachment 19EE.

Sequence 14, 26 and 38: Sequence is arrested in vessel indicating high probability of the use of the firewater addition system.

Sequence 15: This sequence is binned with those which have releases through the rupture disk since the vessel is intact and any fission products released from the vessel will be scrubbed through the suppression pool.

Sequence 19: This sequence has bypass so the releases will not be scrubbed. The operation of sprays ensure that the release will not occur until late in the transient. Therefore, this sequence is binned with the drywell head failure sequences.

Sequence 30: No credit taken for firewater system since a long time was available to prevent core damage but the operator failed to do so.

Table 19D.5-8 Not Used

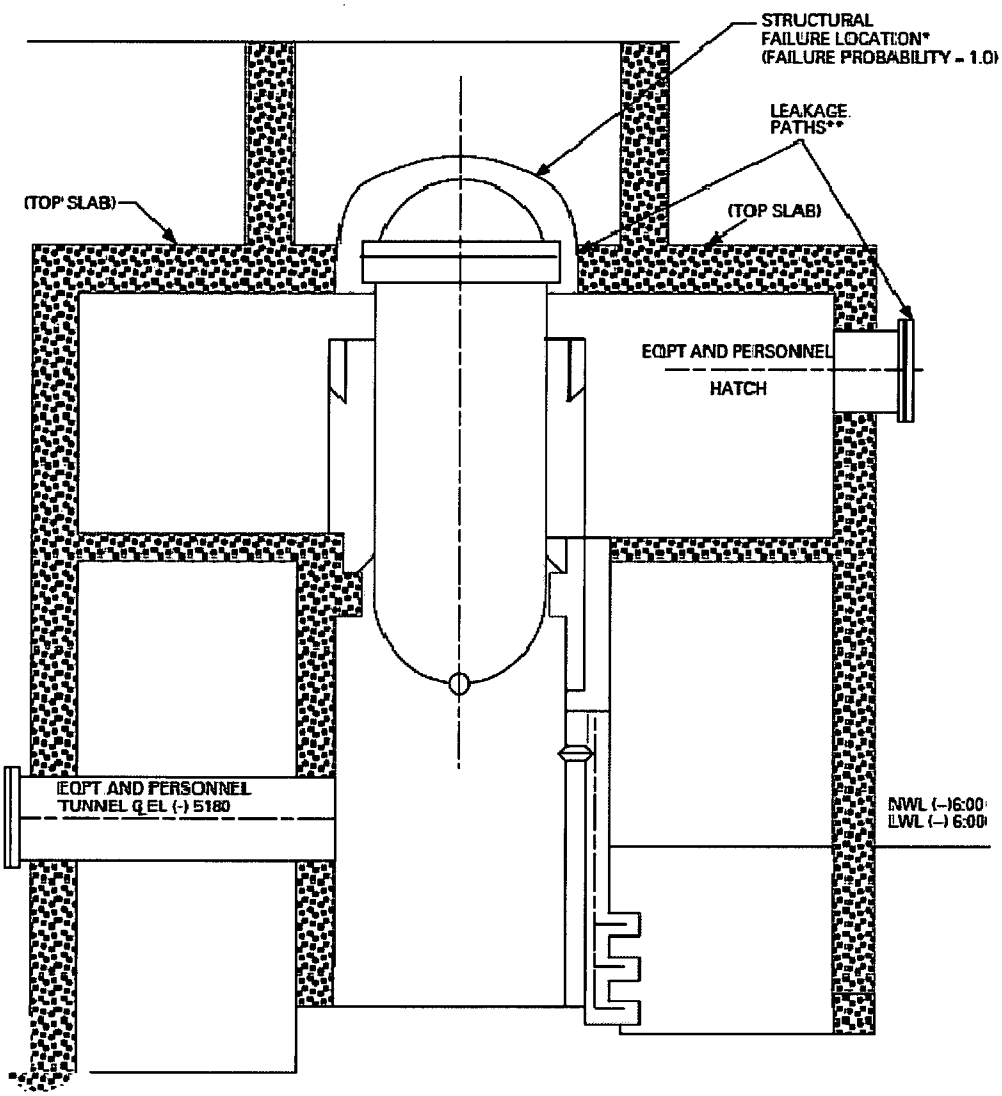
Removed from study.

Table 19D.5-9 Not Used

Removed from study.

Table 19D.5-10
Release Frequencies by Time of Release

Time of Release	Release Frequency via Rupture Disk	Release Frequency via Drywell Head
No release	8.50E-08	
> 24 hours	*Negligible*	*Negligible*
16 - 24 hours	1.10E-08	9.55E-11
8 - 16 hours	1.06E-09	*Negligible*
< 8 hours	*Negligible*	4.01E-11



* Structural failure occurs at 0.924 MPaG when temperature is 260°C.

** Leakage through drywell head, equipment hatches, personal air locks occur when temperature exceeds 260°C and pressure exceeds 0.358 MPa.

Figure 19D.5-1
ABWR Containment Failure Location and Probabilities

Figure 19D.5-2 Not Used

Removed from study.

CRITERIA	INITCODE INITIATOR CODE	IV CORE MELT ARRESTED IN-VESSEL	REL_MODE MODE OF RELEASE	PB POOL BYPASS	SPRAY DRYWELL SPRAY	STC No.
			N (NORM CNT LK)			1
		IV	R (RUP DISK)			2
			D (DW HEAD)			3
			N (NORM CNT LK)			4
	LCHP		P (PEN OT FAIL)	PB		5
				NO PB		6
			R (RUP DISK)		SPRAY	7
		NO IV		PB	NO SPRAY	8
					SPRAY	9
			D (DW HEAD)	PB	NO SPRAY	10
					SPRAY	11
			E (EARLY CF)	PB	NO SPRAY	12
			N (NORM CNT LK)			13
		IV	R (RUP DISK)			14
			D (DW HEAD)			15
			N (NORM CNT LK)			16
	LCLP		P (PEN OT FAIL)	PB		17
				NO PB		18
			R (RUP DISK)		SPRAY	19
		NO IV		PB	NO SPRAY	20
					SPRAY	21
			D (DW HEAD)	PB	NO SPRAY	22
					SPRAY	23
			E (EARLY CF)	PB	NO SPRAY	24
			N (NORM CNT LK)			25
		IV	R (RUP DISK)			26
			D (DW HEAD)			27
			N (NORM CNT LK)			28
	SBRC		P (PEN OT FAIL)	PB		29
				NO PB		30
			R (RUP DISK)		SPRAY	31
		NO IV		PB	NO SPRAY	32
					SPRAY	33
			D (DW HEAD)	PB	NO SPRAY	34
					SPRAY	35
			E (EARLY CF)	PB	NO SPRAY	36
			N (NORM CNT LK)			37
		IV	R (RUP DISK)			38
			D (DW HEAD)			39
			N (NORM CNT LK)			40
	LBLC		P (PEN OT FAIL)	PB		41
				NO PB		42
			R (RUP DISK)		SPRAY	43
		NO IV		PB	NO SPRAY	44
					SPRAY	45
			D (DW HEAD)	PB	NO SPRAY	46
					SPRAY	47
			E (EARLY CF)	PB	NO SPRAY	48
			N (NORM CNT LK)			49
		NO CORE DAMAGE	R (RUP DISK)			50
	LHRC		D (DW HEAD)			51
		CD/VF	D (DW HEAD)			52
	CLASS IV					53

Figure 19D.5-3
Source Term Category Grouping

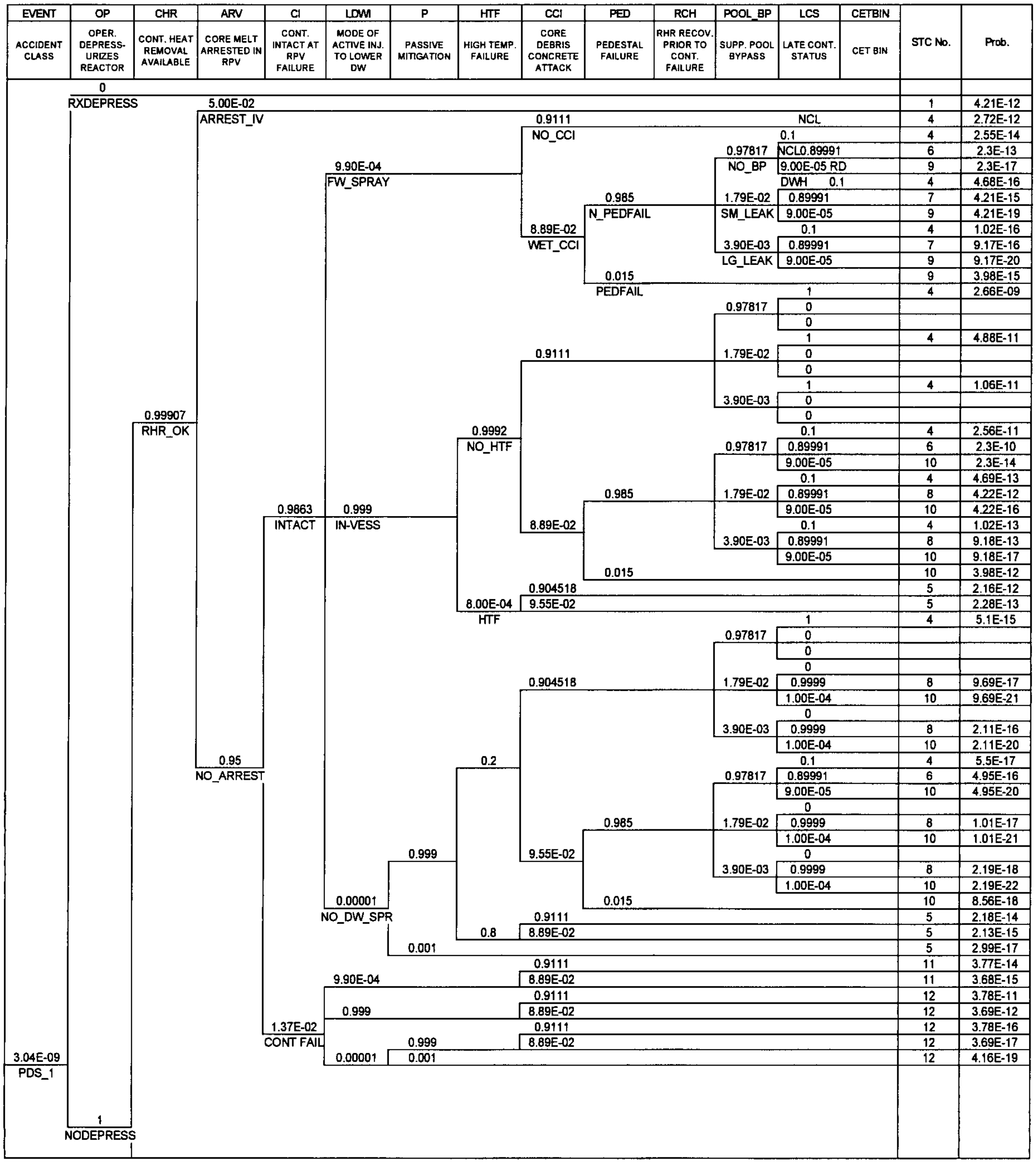


Figure 19D.5-4

PDS 1 - Containment Event Evaluation CET for Class IA Sequences

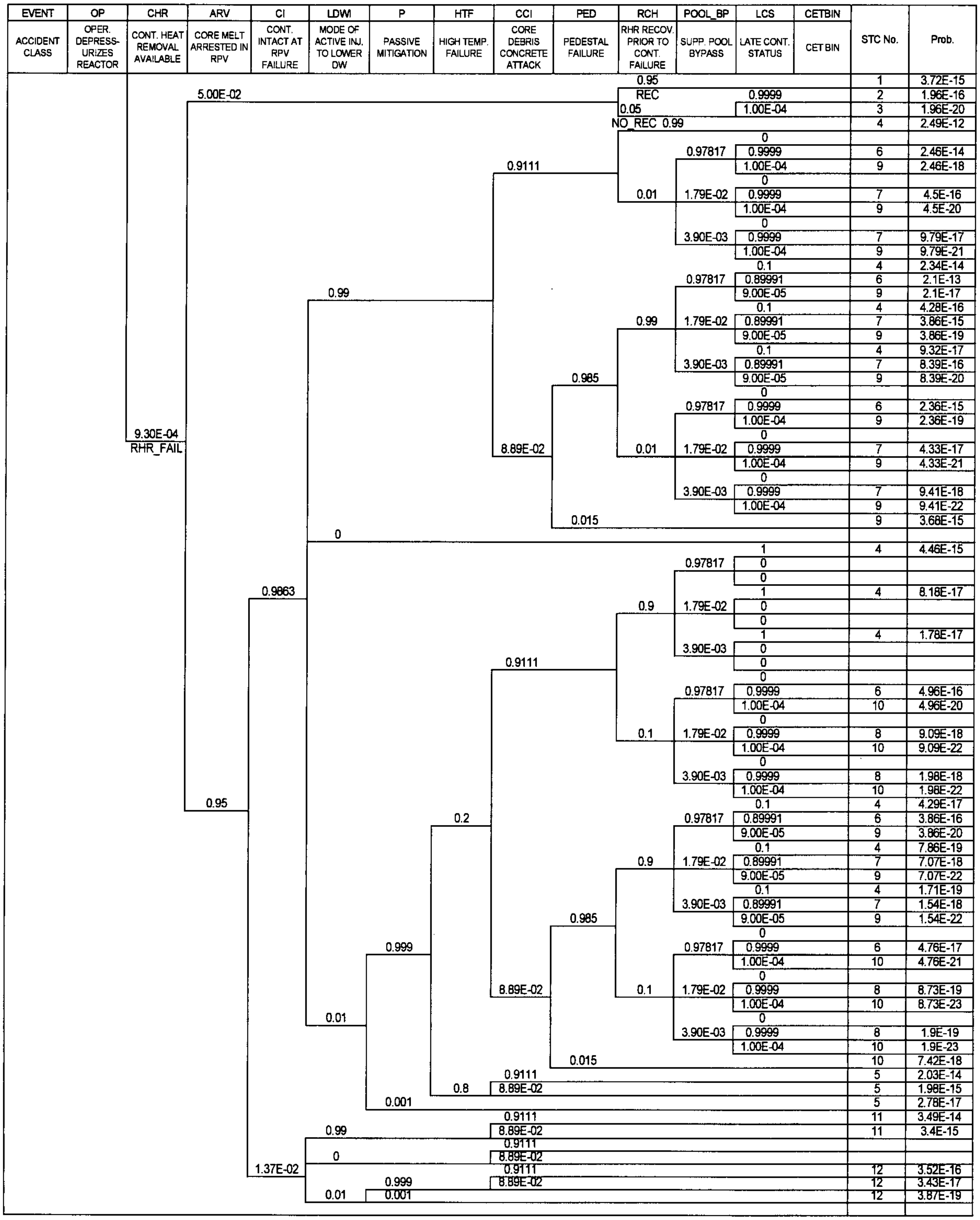


Figure 19D.5-4
PDS 1 - Containment Event Evaluation CET for Class IA Sequences, cont.

Figure 19D.5-5 Not Used

Removed from study.

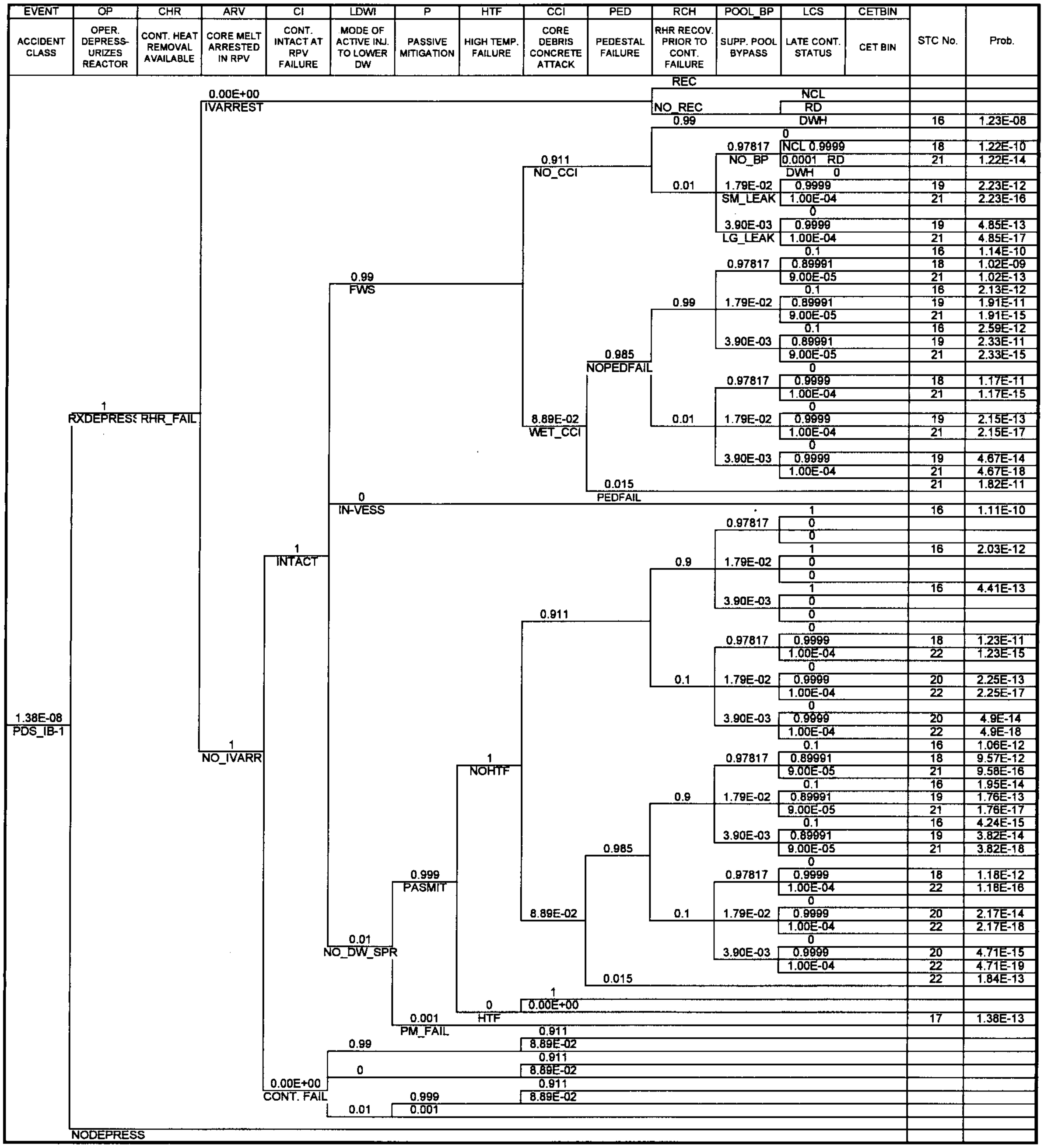


Figure 19D.5-6
PDS 2 - Containment Event Evaluation CET for Class IB-1 Sequences

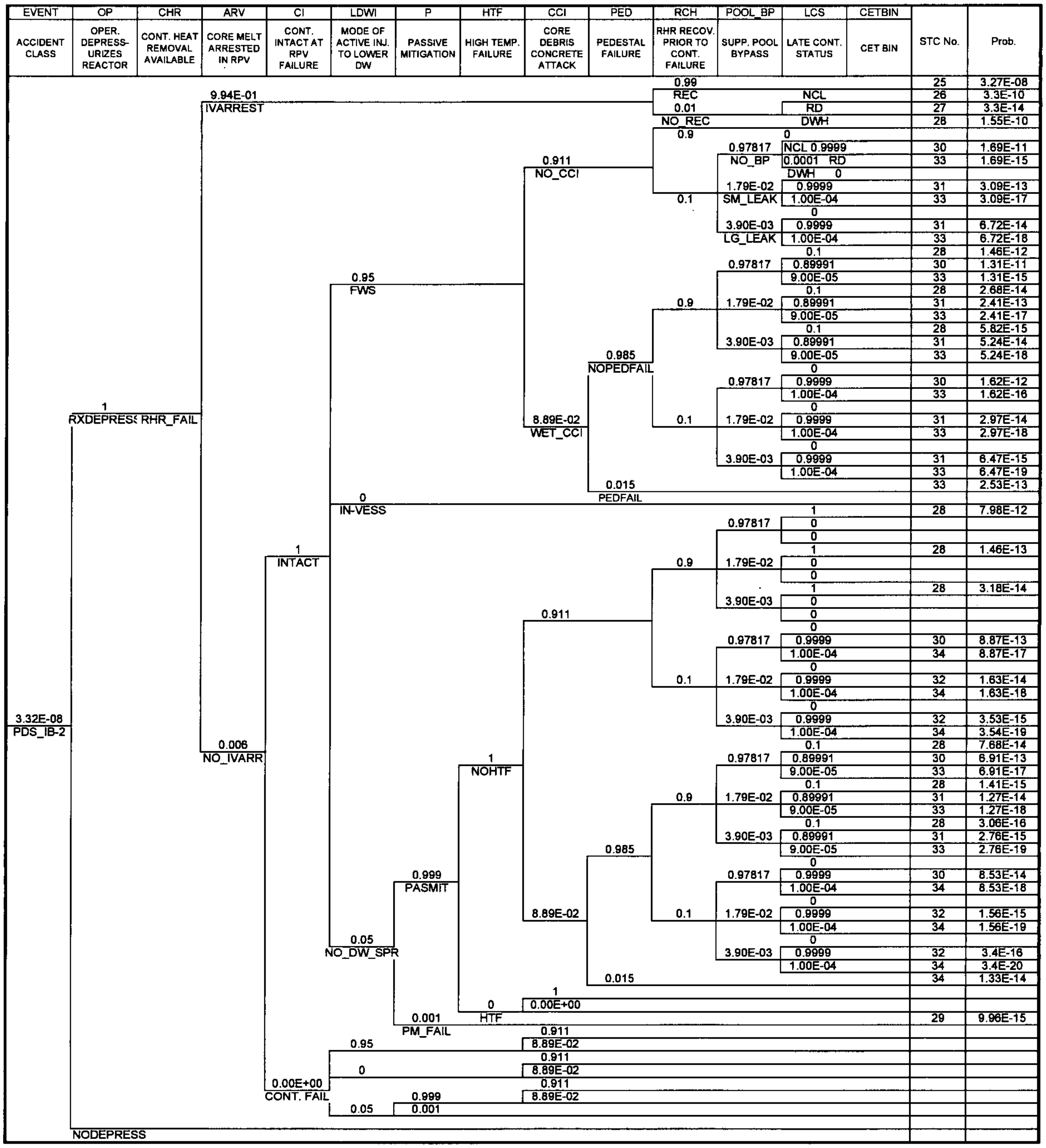


Figure 19D.5-7
PDS 3 - Containment Event Evaluation CET for Class IB-2 Sequences

Figure 19D.5-8 Not Used

Removed from study.

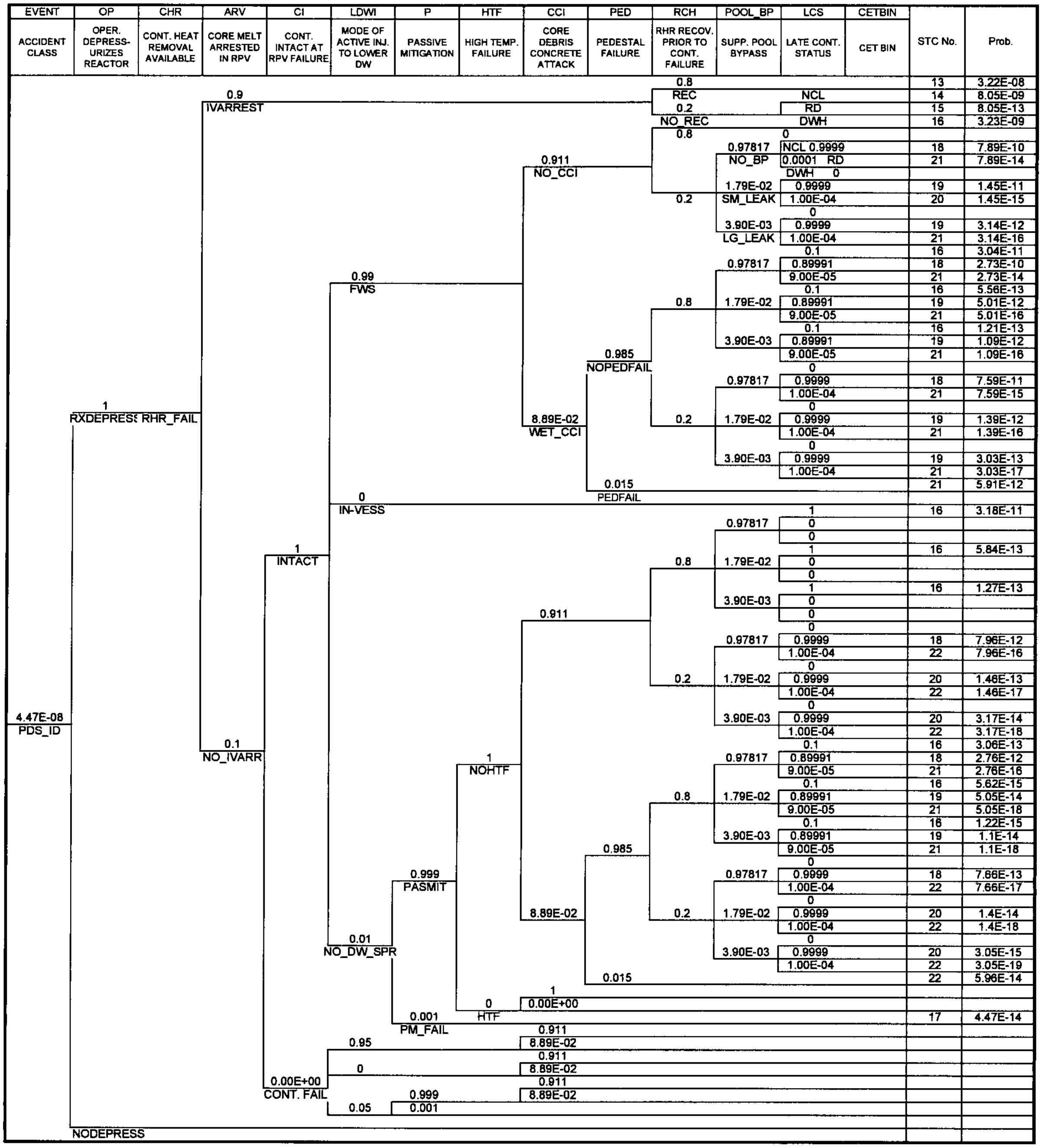


Figure 19D.5-9
PDS 4 - Containment Event Evaluation CET for Class ID Sequences

EVENT	RCH	RD	CORCOOL	STC No.	Prob.
	RHR RECOVERED PRIOR TO CONT. FAILURE	RUPTURE DISK OPENS	CONTINUED CORE COOLING		
	0.9			49	7.70E-10
8.56E-10	RHR_RECOV		CORE COOLING OK		
PDS_II		0		50	
	0.1	RD OPEN	CORE COOLING OK		
	NO_RHR_REC		0.999	51	8.55E-11
		1.00E+00	CORE COOLING OK		
		RD FAILS	0.001	52	8.56E-14
			NO CORE COOLING		

Figure 19D.5-10
PDS 5 - Containment Event Evaluation CET for Class II Sequences

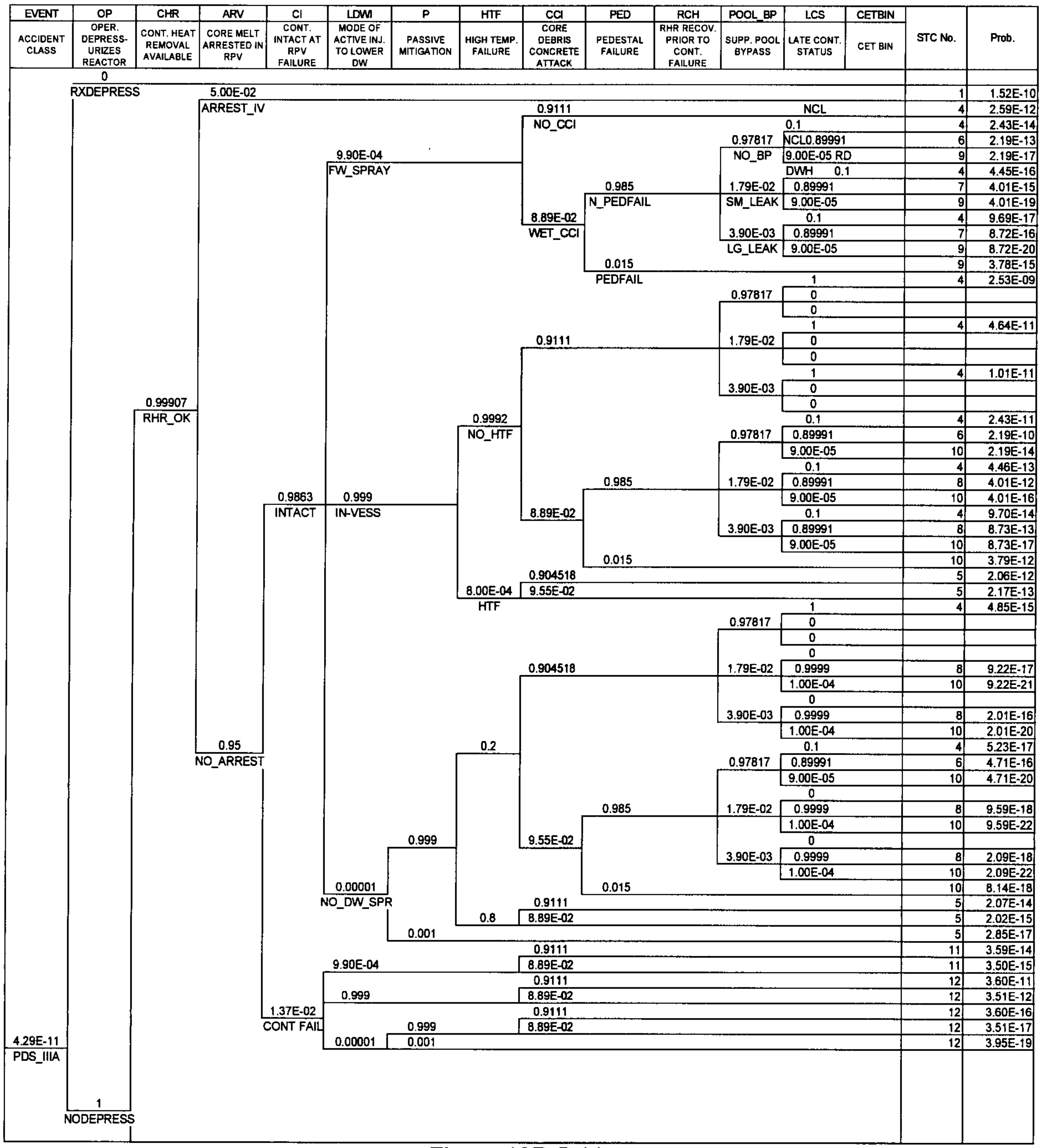


Figure 19D.5-11

PDS 6 - Containment Event Evaluation CET for Class IIIA Sequences

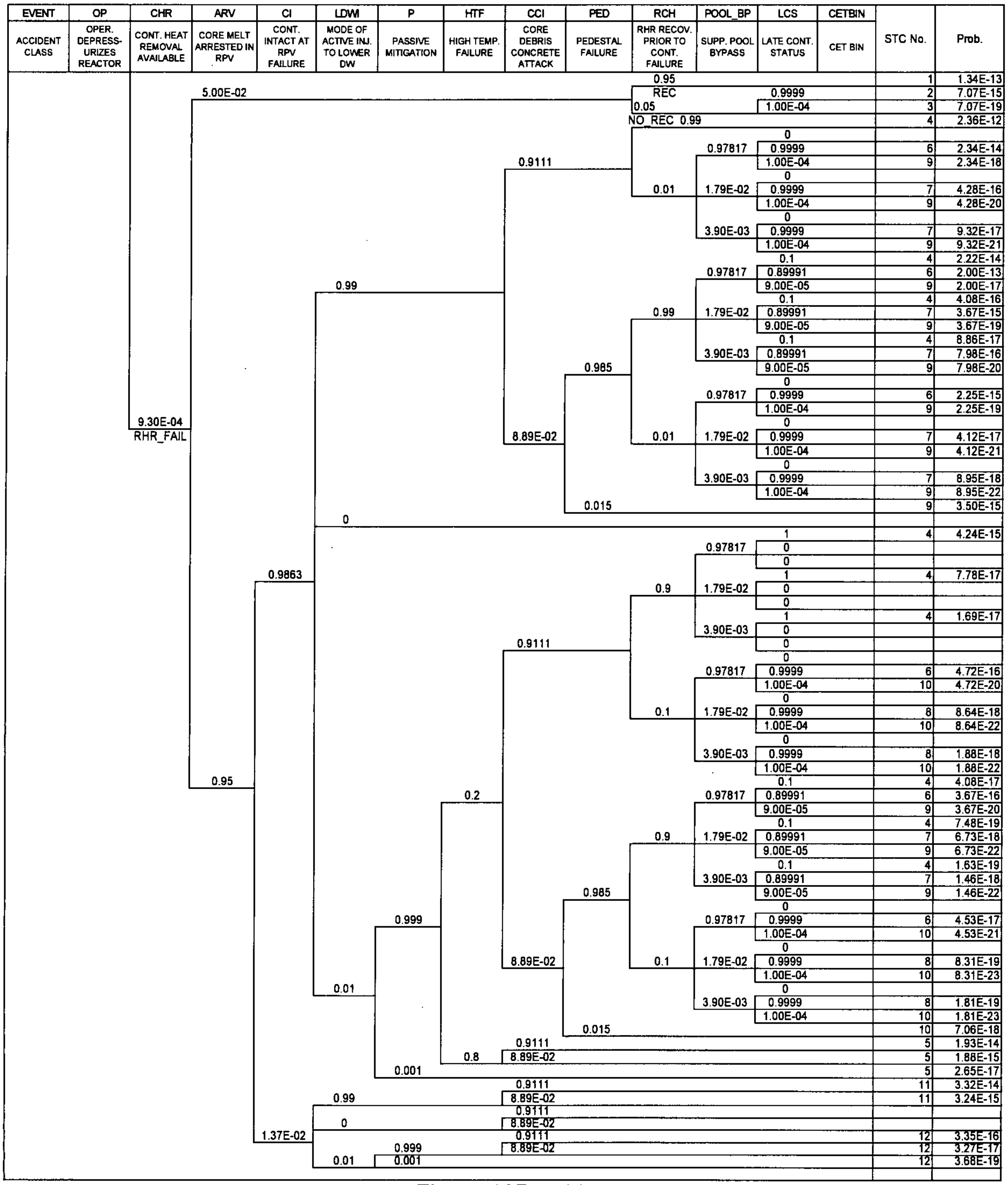


Figure 19D.5-11
PDS 6 - Containment Event Evaluation CET for Class IIIA Sequences, cont.

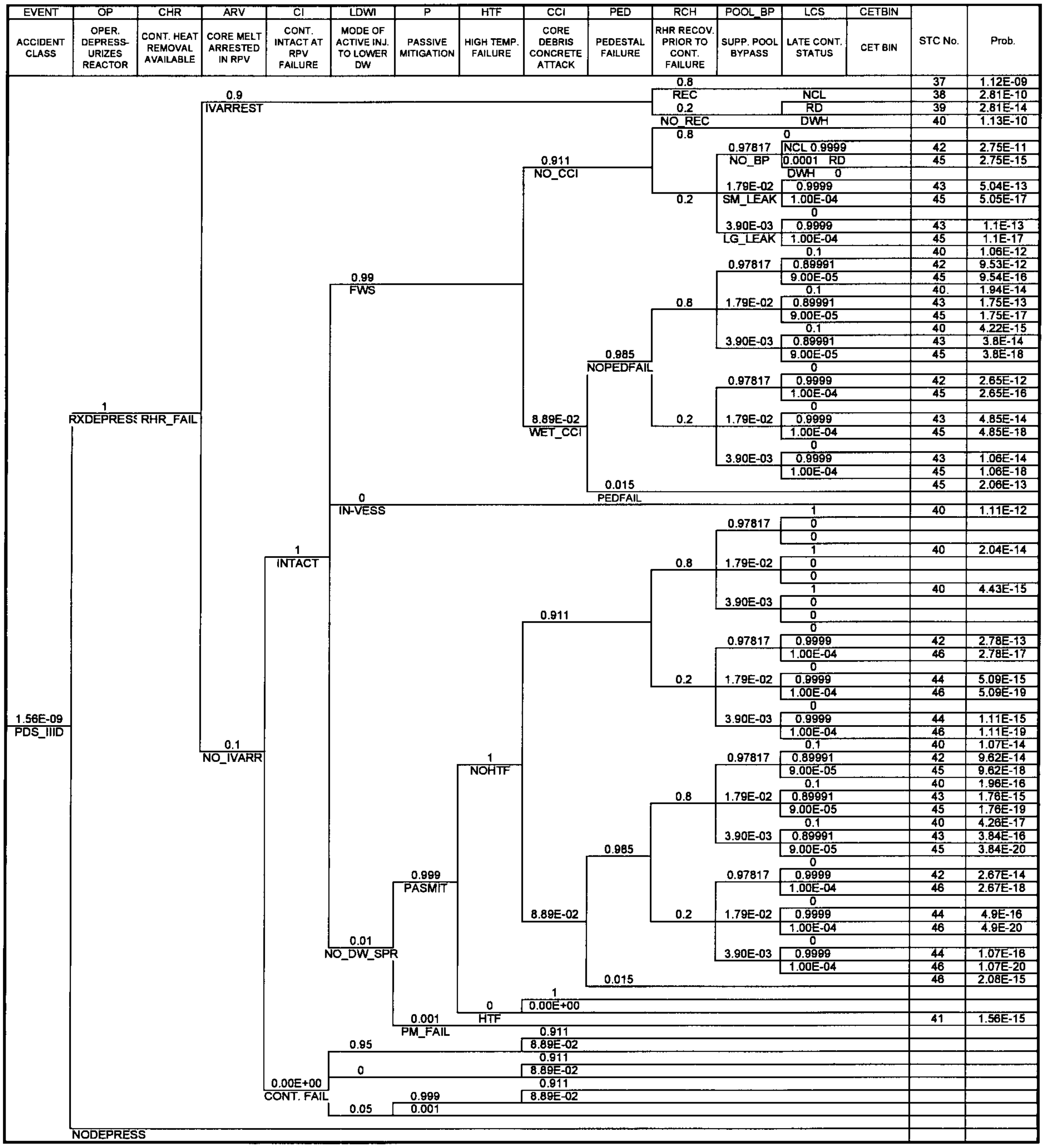


Figure 19D.5-13
PDS 7 - Containment Event Evaluation CET for Class IIID Sequences

EVENT	DUMMY	STC No.	Prob.
	TRANSFER TREE CLASS IV SEQUENCES		
7.97E-10		53	7.97E-10
PDS_IV			

Figure 19D.5-14
PDS 8 - Containment Event Evaluation CET for Class IV Sequences

EVENT	SUBCLASS	OP
	ACCIDENT SUBCLASS	OPERATOR DEPRESSURIZES REACTOR
	<–	
	IB-1, IB-2	RX DEPRESS
	<–	
	ID, IIID	RX DEPRESS
	<–	
	II, IV	N/A
	<–	
	ALL OTHERS	NOT DEPRESS

Figure 19D.5-15
Containment Event Evaluation DET for Operator Depressurizes Reactor

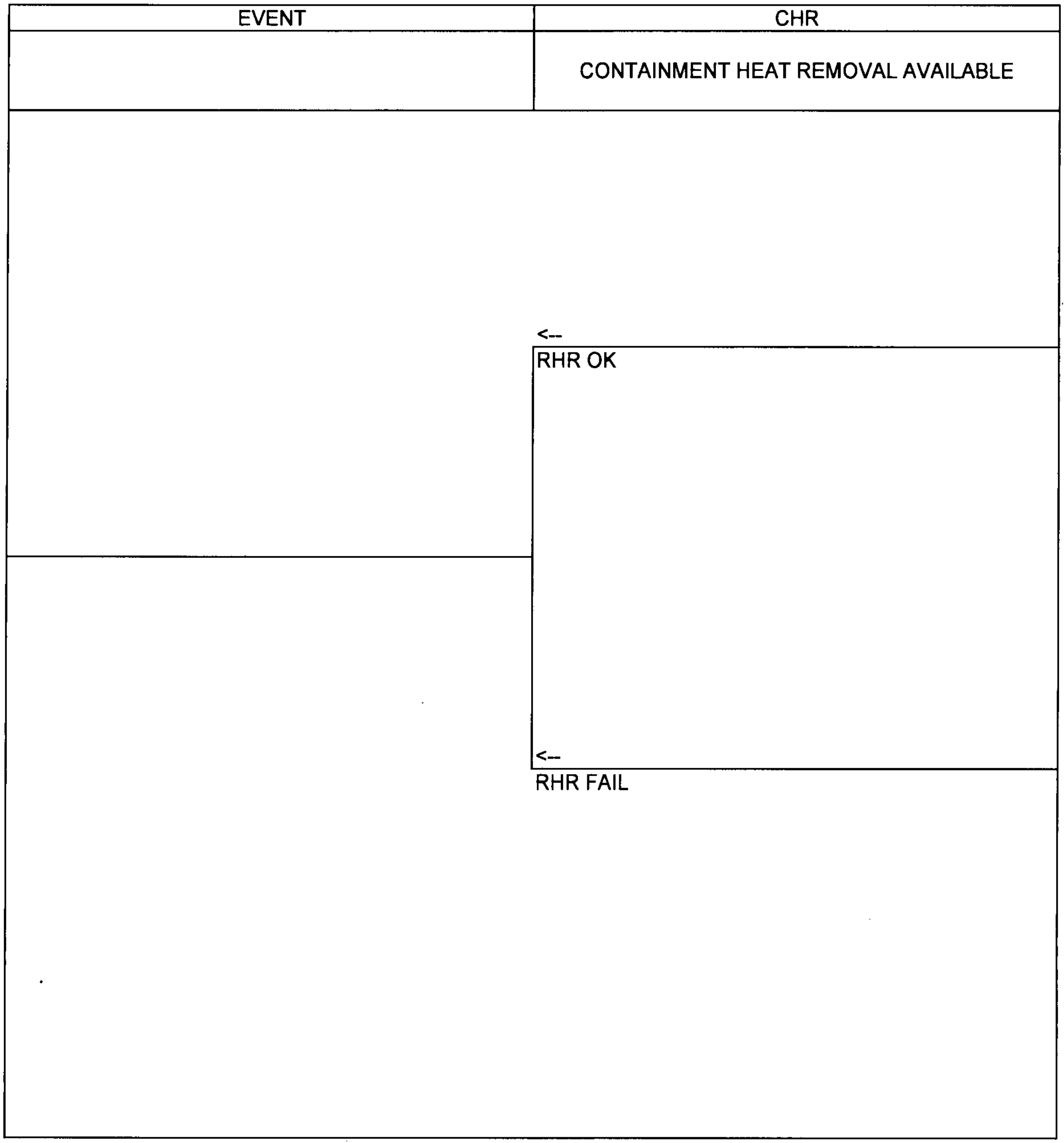


Figure 19D.5-16
Containment Event Evaluation DET for Containment Heat Removal Available

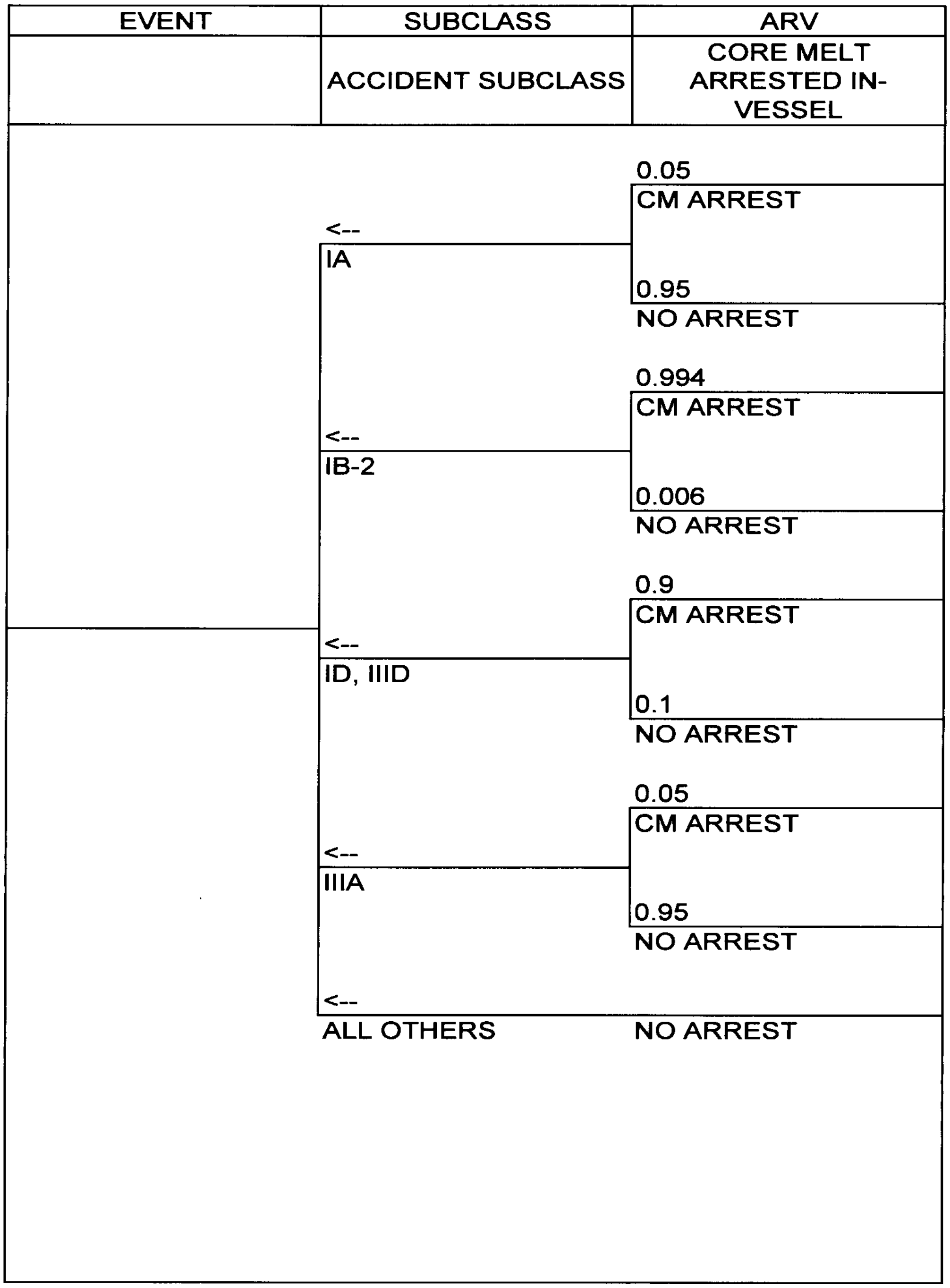


Figure 19D.5-17
Containment Event Evaluation DET for Core Melt Arrested in RPV

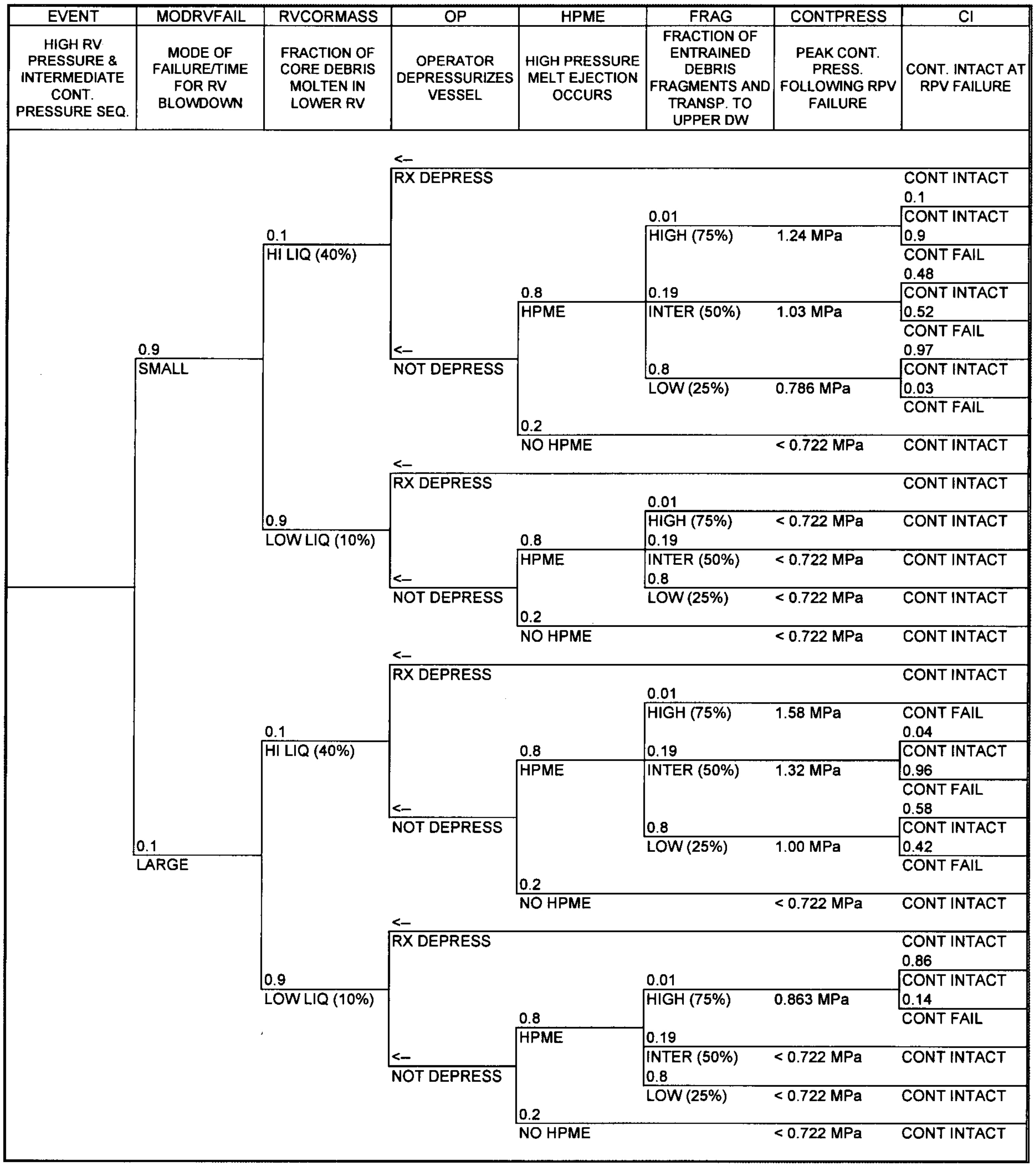


Figure 19D.5-18

Containment Event Evaluation DET for Probability of Early Containment Failure High RV Press and Low Cont Press Sequences

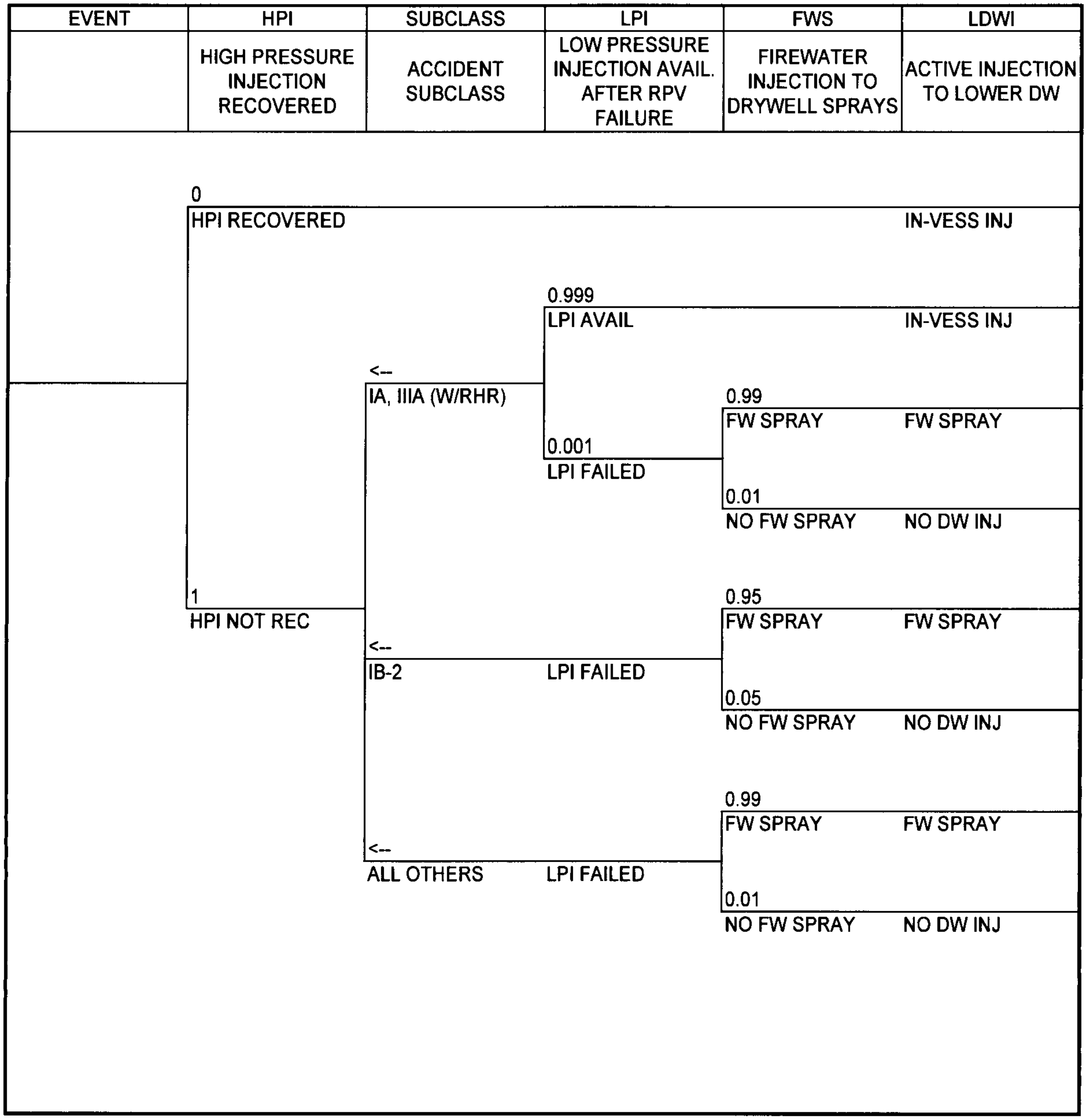


Figure 19D.5-19
Containment Event Evaluation DET for Active Injection to Lower Drywell

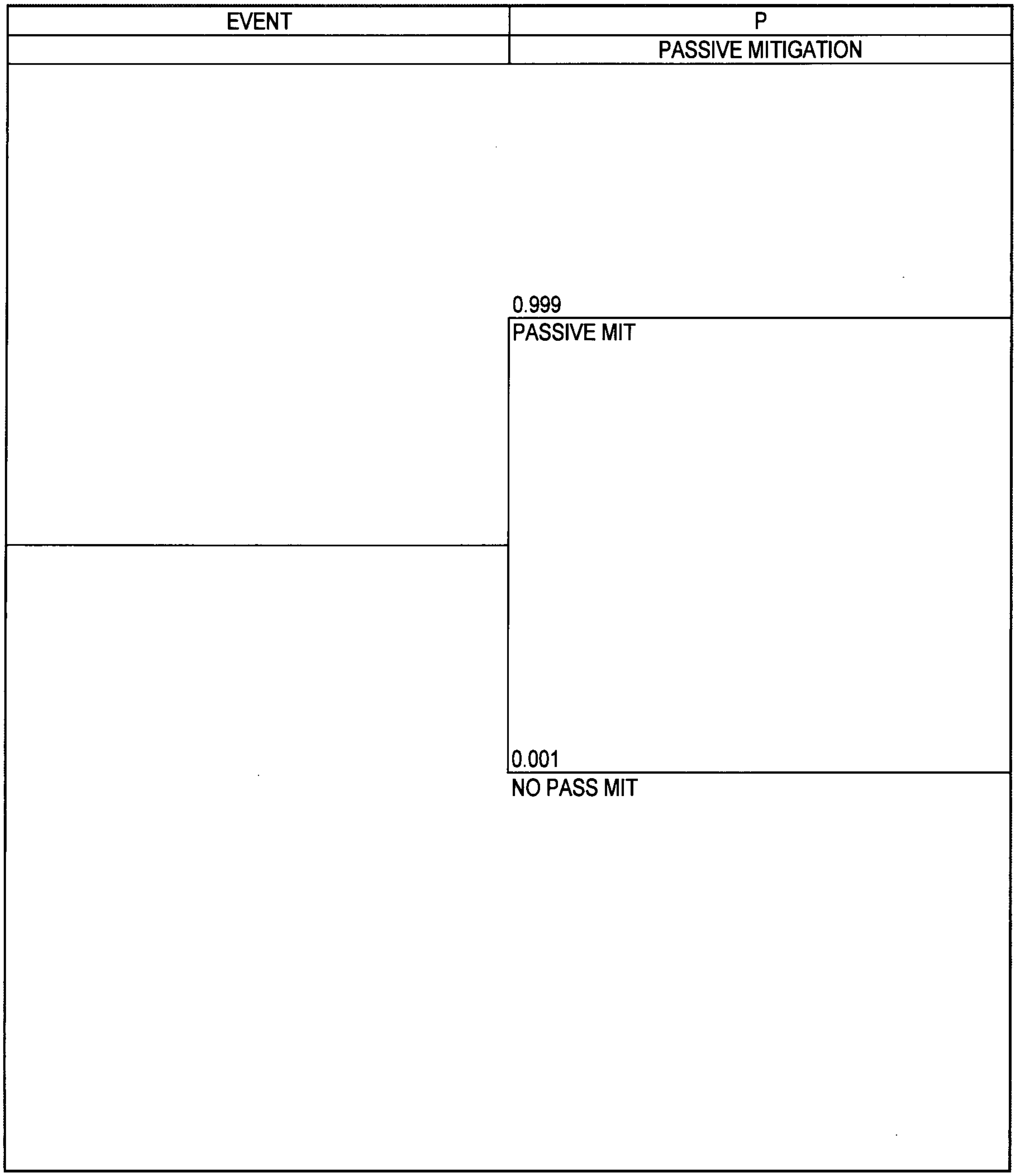


Figure 19D.5-20
Containment Event Evaluation DET for Passive Mitigation

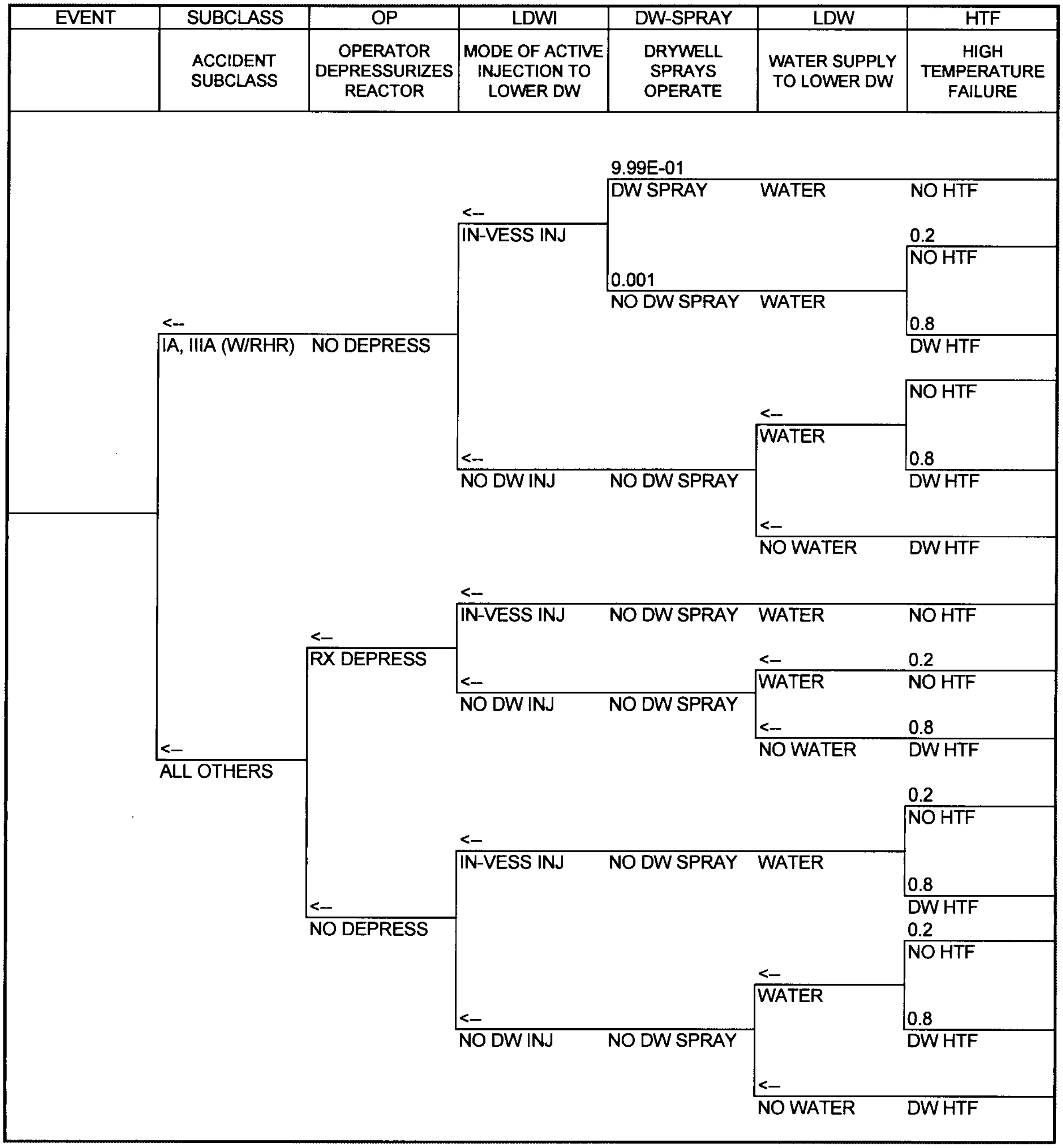


Figure 19D.5-21
Containment Event Evaluation DET for High-Temperature Failure

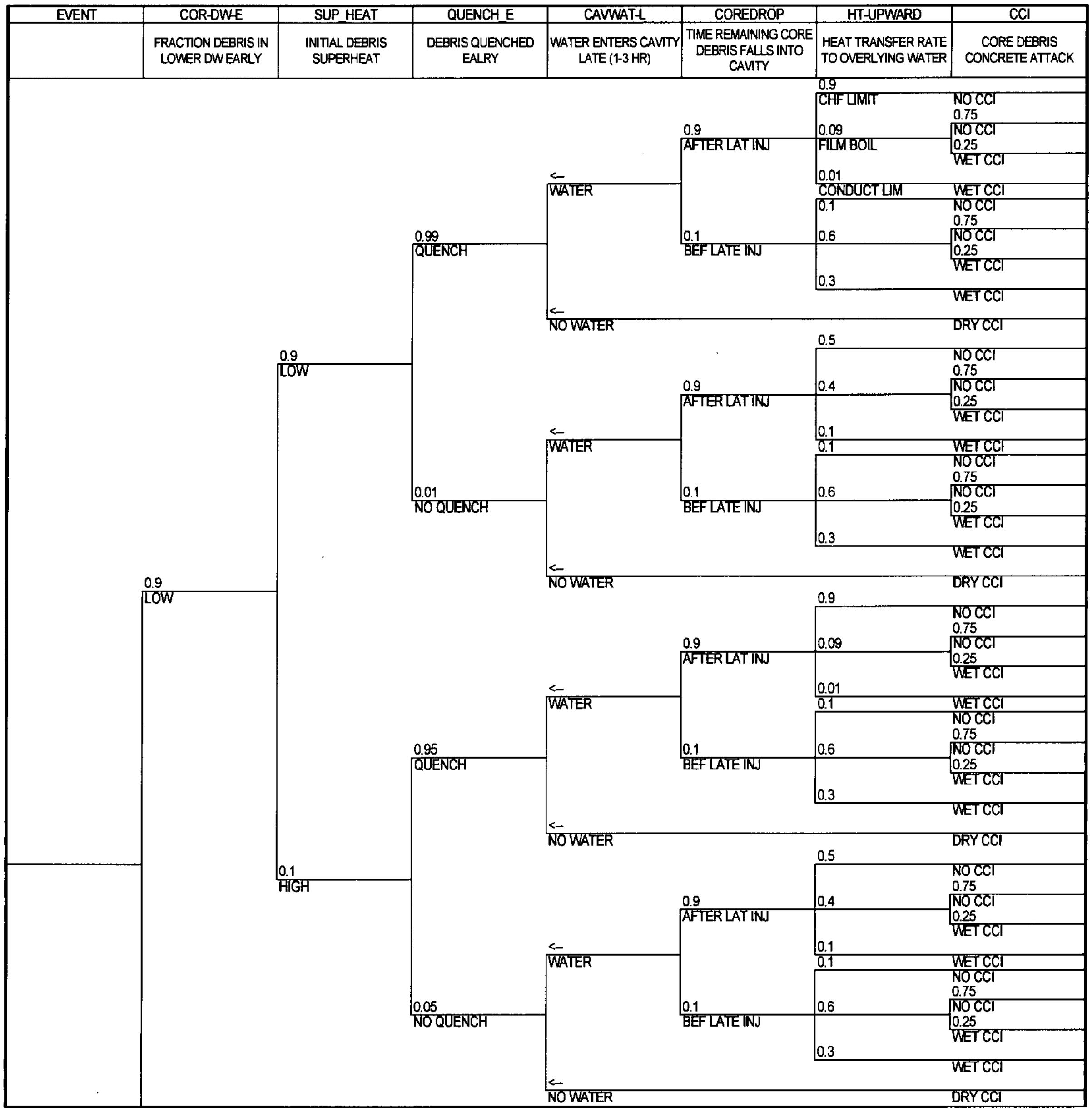


Figure 19D.5-22
Core Debris Concrete Attack DET

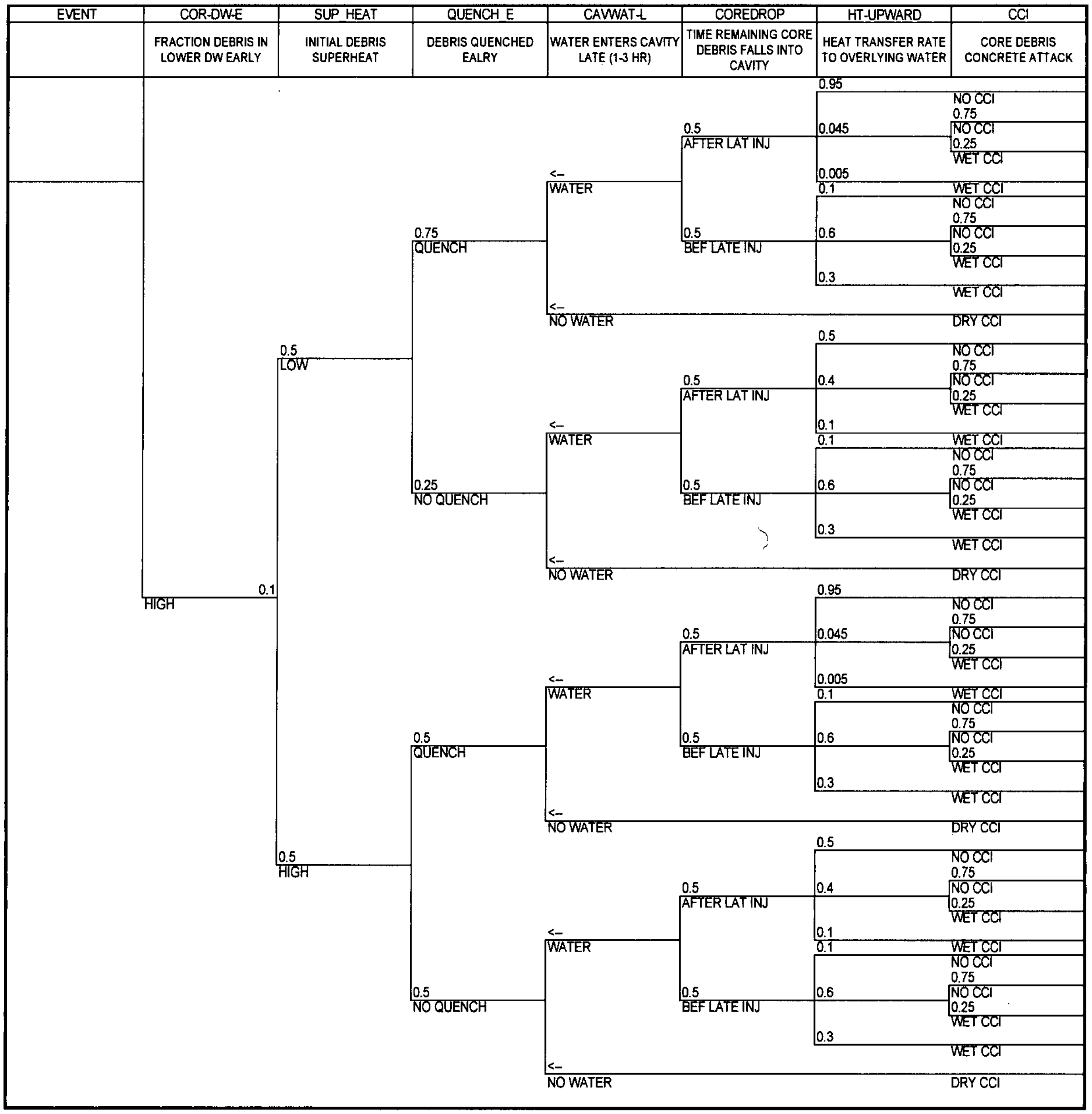


Figure 19D.5-22
Core Debris Concrete Attack DET, continued

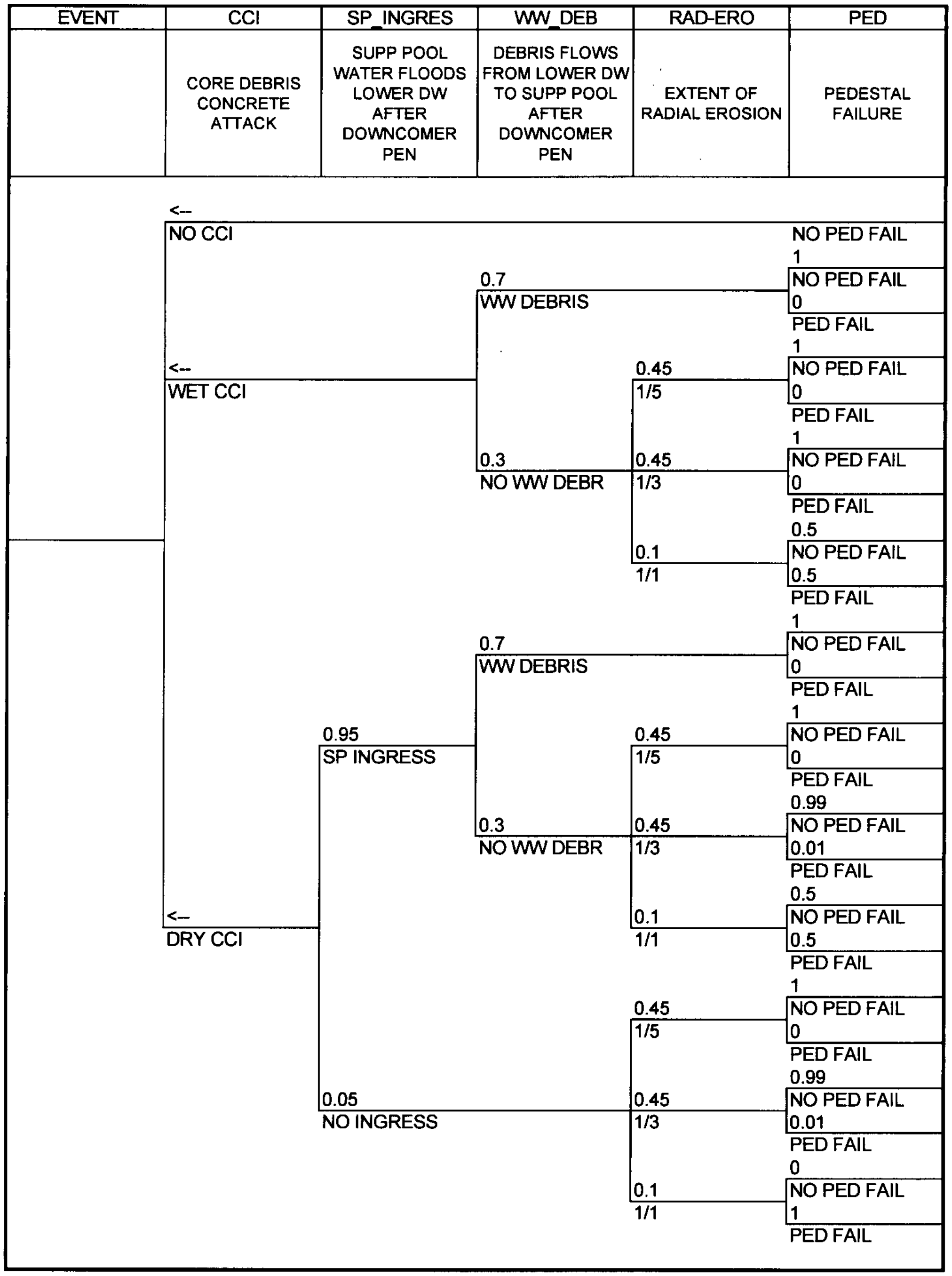


Figure 19D.5-23
Containment Event Evaluation DET for Pedestal Failure

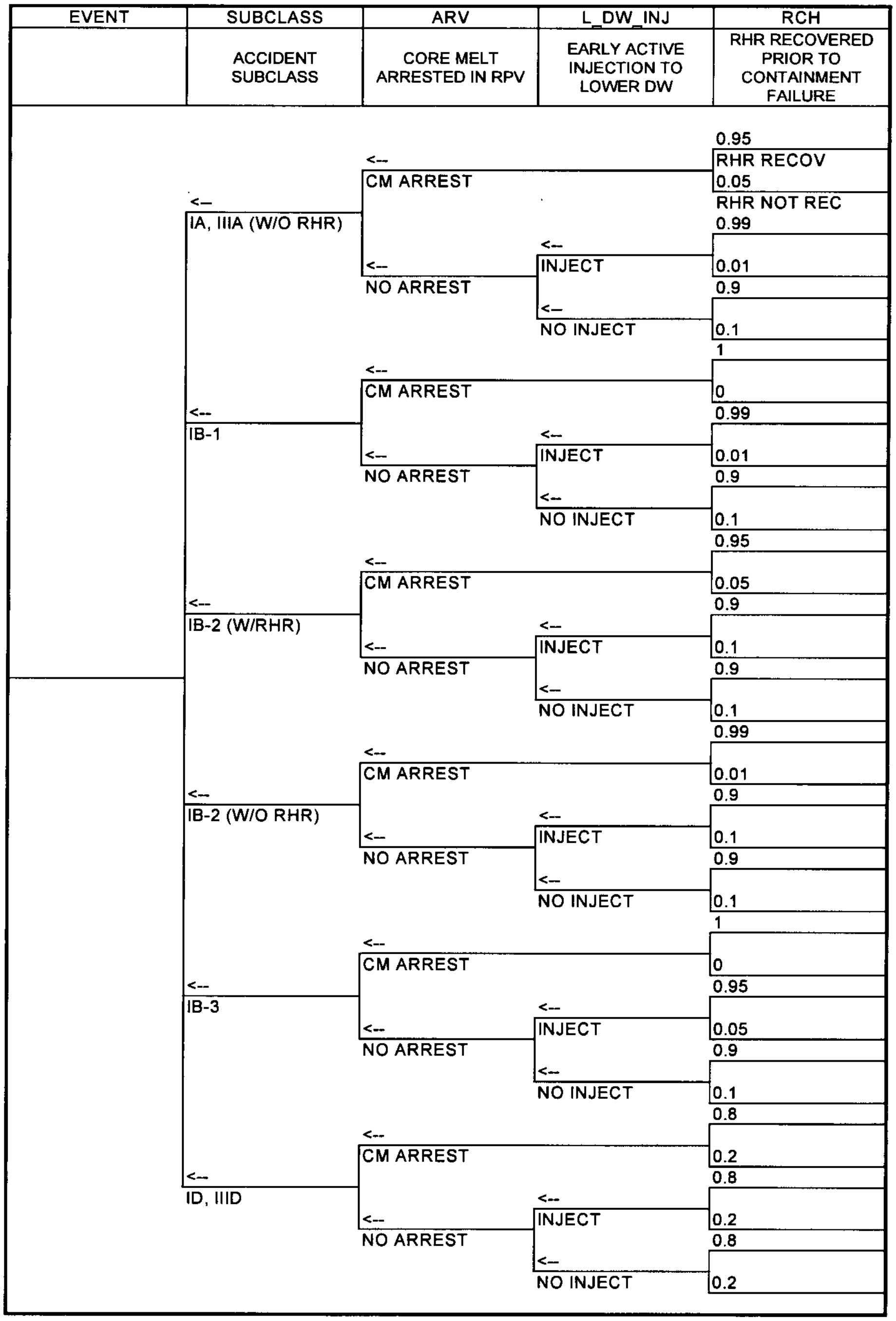


Figure 19D.5-24
Containment Event Evaluation DET for RHR Recovery Prior to Containment Struct Failure

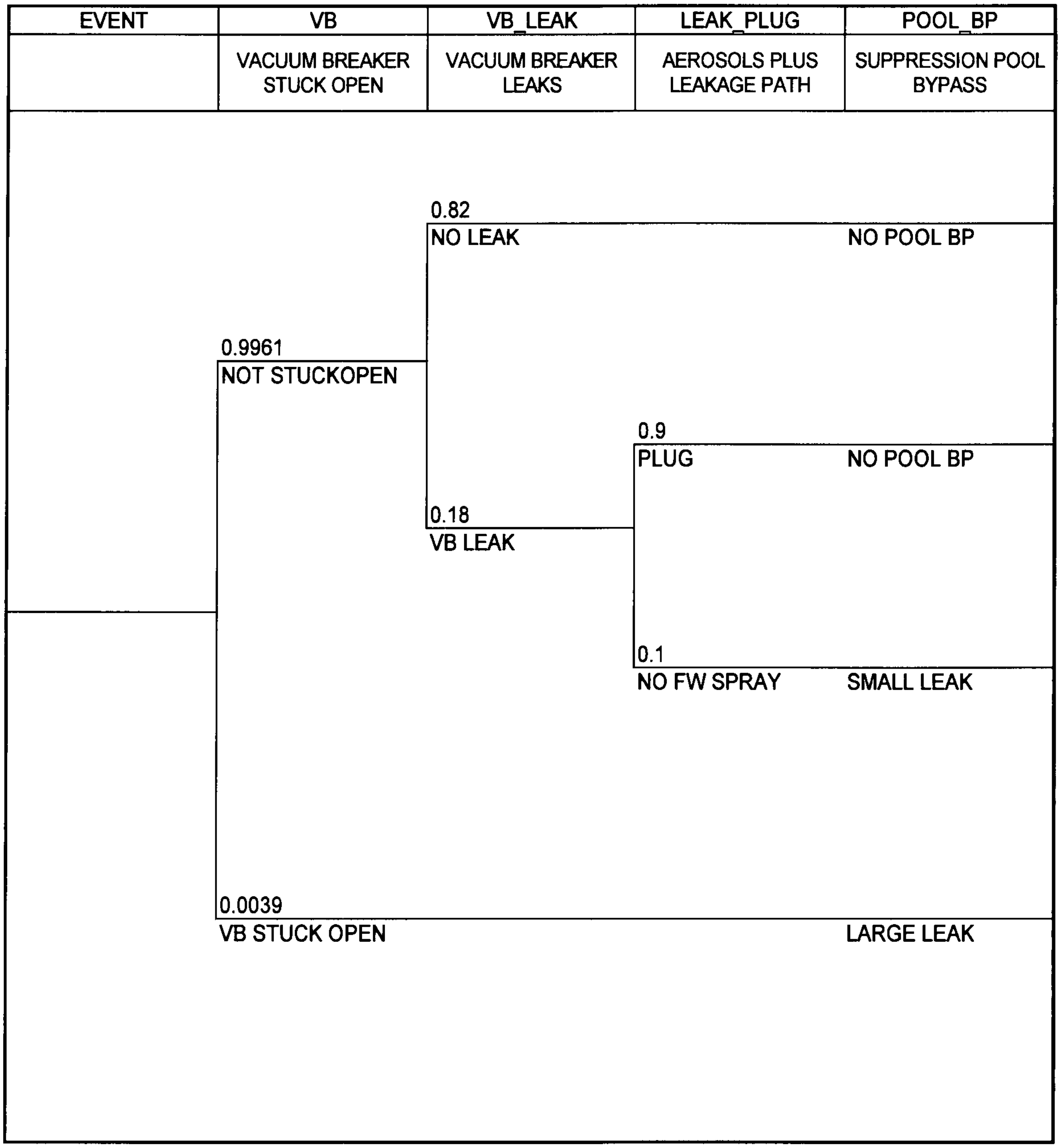


Figure 19D.5-25
Containment Event Evaluation DET for Suppression Pool Bypass

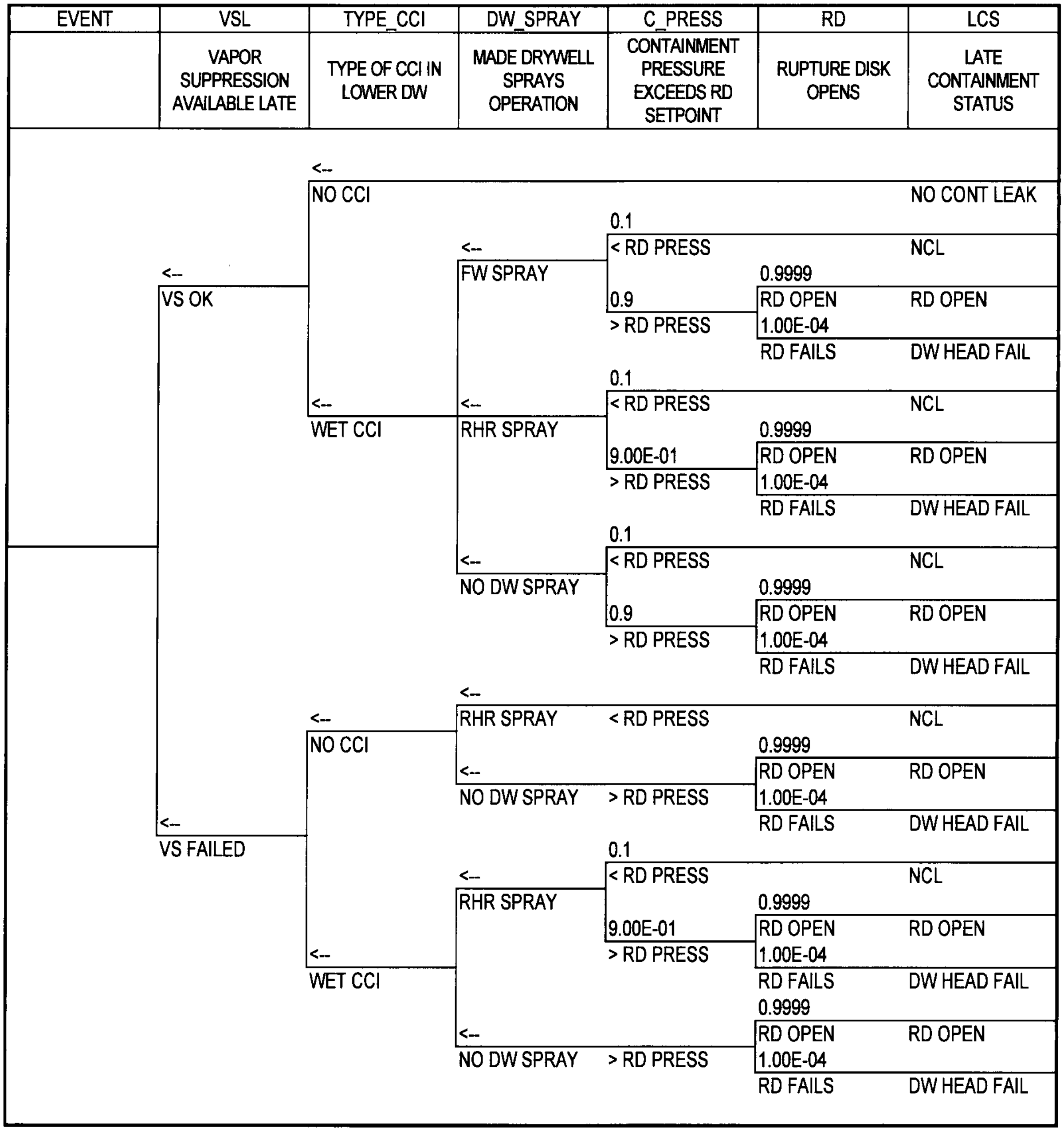


Figure 19D.5-26
Containment Event Evaluation DET for Late Containment Status for Sequences With RHR Available at Core Damage

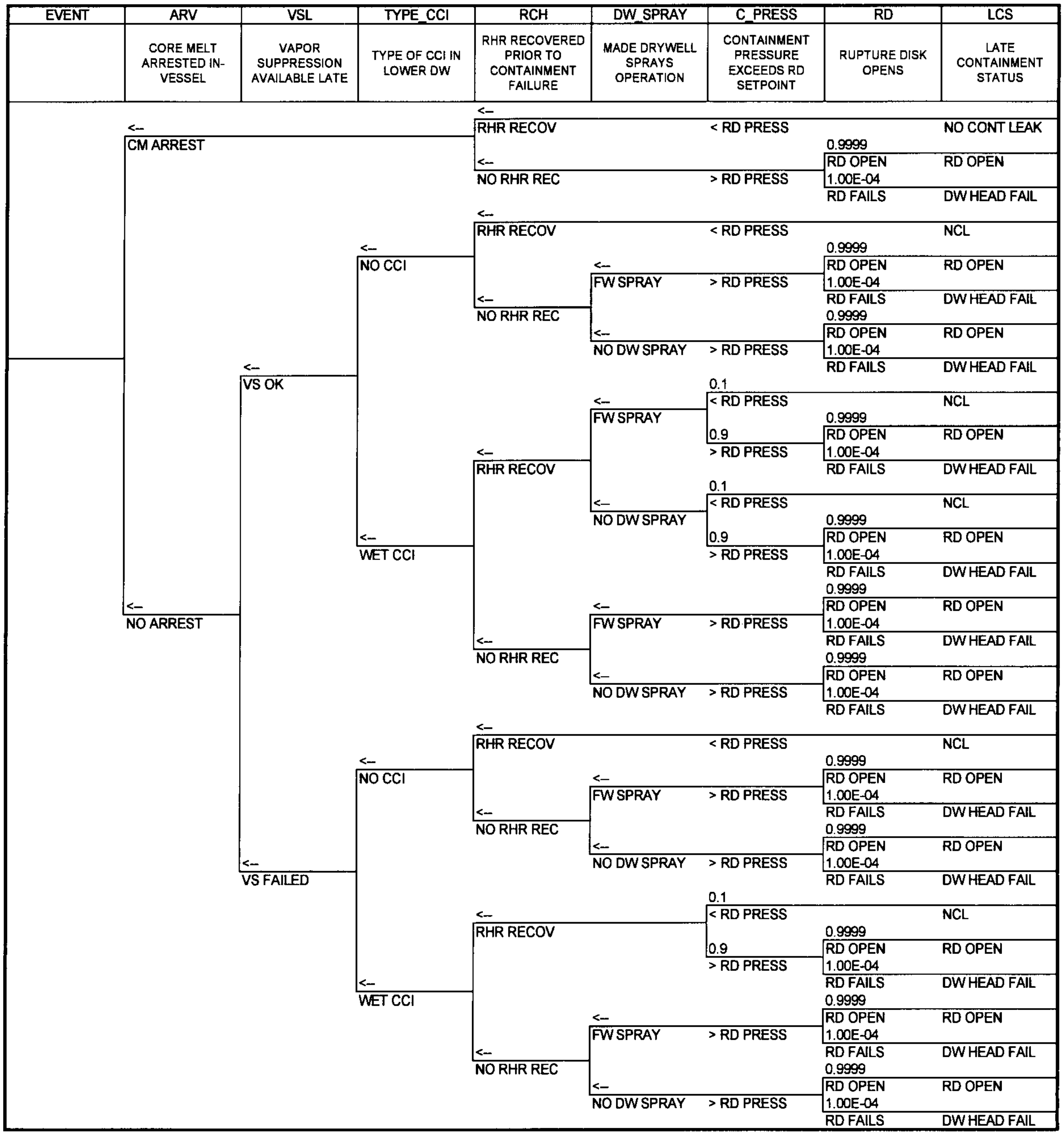


Figure 19D.5-27

Containment Event Evaluation DET for Late Containment Status for Sequences with RHR Not Available at Core Damage

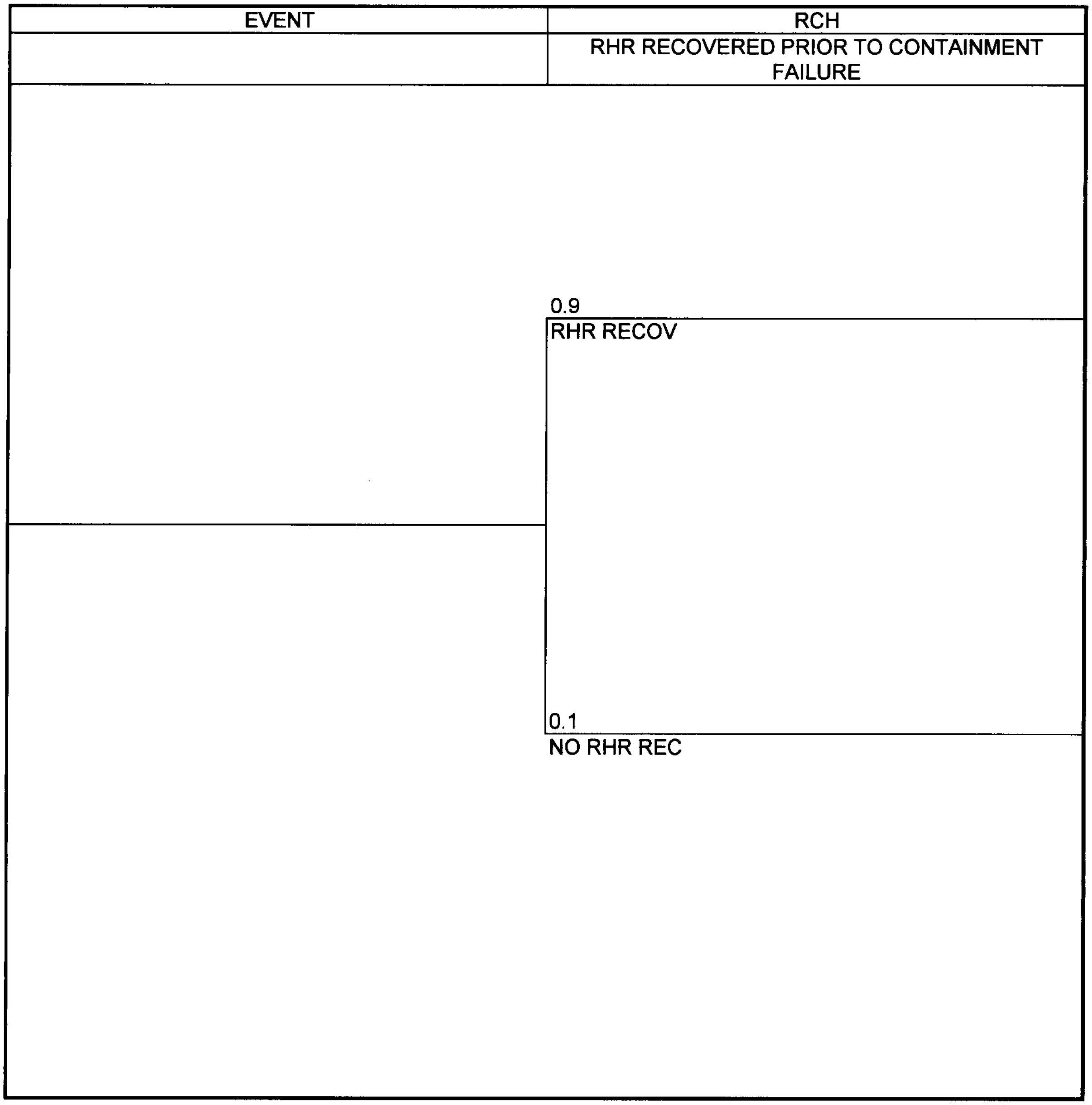


Figure 19D.5-28
Containment Event Evaluation DET for RHR Recovery Prior to Rupture Disk Setpoint Pressure (Class II)

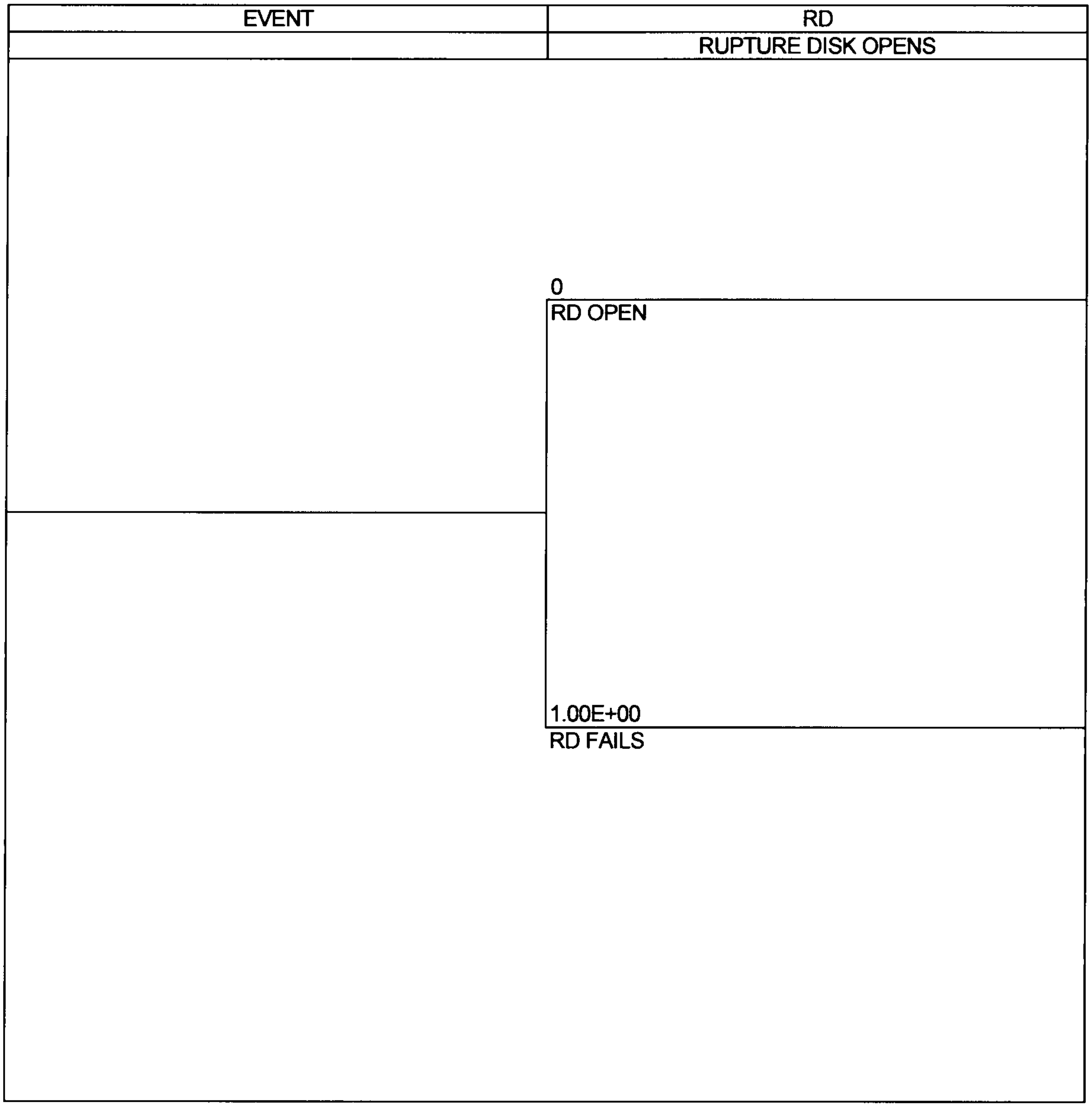


Figure 19D.5-29
Containment Event Evaluation DET for Rupture Disk Opens (Class II)

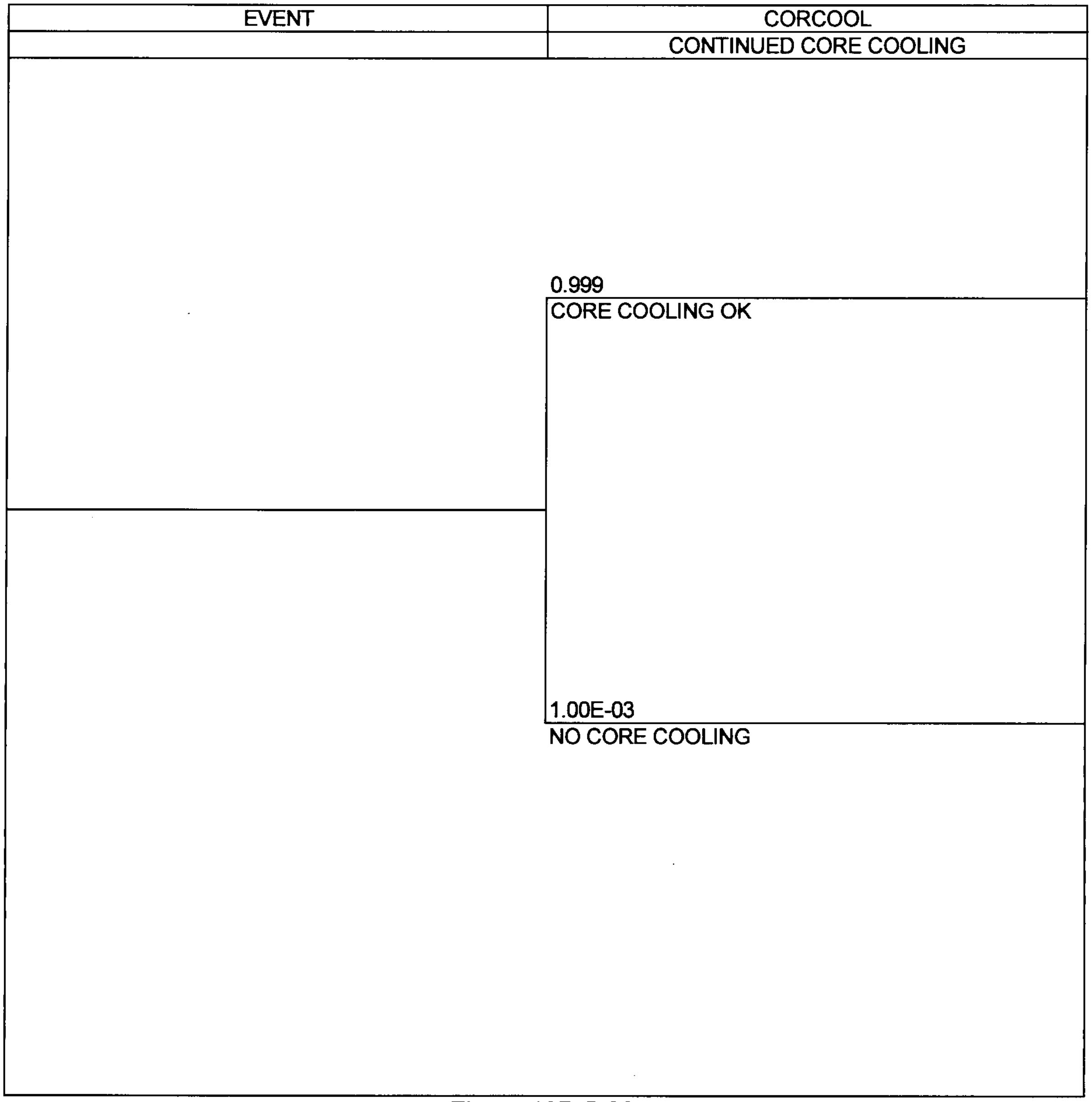


Figure 19D.5-30
Containment Event Evaluation DET for Core Cooling Recovery (Class II)

19D.6 FAULT TREES

19D.6.1 FAULT TREE ANALYSIS

19D.6.1.1 Introduction

Accident event trees described in Subsection 19D.4 identify key safety system functions required to mitigate potential consequences of postulated accident sequences. This subsection describes construction of fault trees used in the analysis to assess the failure of individual systems, as well as combinations of systems, to successfully perform these functions upon demand.

The fault trees which follow are grouped into four basic functional categories: Core Cooling (Subsection 19D.6.2); Heat Removal (Subsection 19D.6.3); Support Systems (Subsection 19D.6.4), and Reactivity Control (Subsection 19D.6.5). System trees are preceded, as appropriate, by functional fault trees illustrating the relationships between individual systems.

A brief description is provided of each system for which a fault tree has been developed and a table of basic event data used in the evaluation accompanies each tree.

19D.6.2 CORE COOLING FAULT TREE

19D.6.2.1 Core Cooling Functional Fault Tree

System fault trees developed to assess the demand unavailability of the core cooling function are presented in this subsection. Figure 19D.6-1 depicts the synthesis of these individual systems into the functional level fault tree for water injection into the reactor pressure vessel.

Detailed fault trees were developed for the High Pressure Core Flooder, Reactor Core Isolation Cooling, Automatic Depressurization, and Residual Heat Removal Systems. In addition to combining these systems, support system fault trees from Subsection 19D.6.4 were appended where indicated in each core cooling system fault tree to evaluate the overall unavailability of the core cooling function.

19D.6.2.2 Reactor Core Isolation Cooling System (RCIC)

The RCIC System provides emergency core cooling in conjunction with the high pressure (HPCF) and low pressure (RHR in LPFL mode) reactor water injection systems and, in addition, provides makeup water to the reactor vessel during transient events which are accompanied by loss of feedwater. The RCIC pump is turbine driven, with steam extracted from a main steamline and exhausted to the suppression pool.

Primary suction is taken from the condensate storage tank (CST) and secondary suction from the suppression pool. Level instrumentation provides automatic transfer of suction from the CST to the suppression pool upon receipt of either a low CST or high suppression pool water level signal. An auto-suction transfer override is also provided.

The RCIC System is initiated automatically upon receipt of either a reactor vessel low water level signal or high drywell pressure signal and can also be started, operated, and shut down manually provided initiation or shutdown signals do not exist. The turbine will shut down automatically upon receipt of a reactor vessel water Level 8 signal. The turbine automatically trips upon receipt of turbine over-speed, low pump suction, high turbine exhaust pressure, or auto-isolation signals from the leak detection system.

The RCIC fault tree is presented in Figure 19D.6-2 and applicable failure probabilities are provided in Table 19D.6-1. Evaluation of this fault tree provides the conditional probability that the RCIC System will fail to start on demand and provide rated flow for 24 hours, given the availability of DC power and that the reactor is at sufficient pressure to provide motive steam for the turbine.

19D.6.2.3 High Pressure Core Flooder (HPCF) System

The HPCF System, in conjunction with the RCIC and RHR Systems, operate to maintain a core covered condition for accidents and transients. The HPCF System consists of two high pressure loops, HPCF-B and HPCF-C, which are divisionally separated electrically and physically.

Both loops take primary suction from the condensate storage tank (CST) and secondary suction from the suppression pool. In the event the CST water level falls below a predetermined setpoint or the suppression pool water level rises above a predetermined setpoint, pump suction on either loop will automatically transfer from the CST to the suppression pool. Both system loops have suction lines that are separate from the RHR System loops.

Startup of each loop of the HPCF occurs automatically upon detection of either a low water level in the reactor vessel or high pressure in the drywell. Each loop of the system can also be placed in operation by means of a manual initiation pushbutton switch.

After automatic initiation, if the reactor vessel water level is raised to a predetermined high level, the injection valve in the discharge line will automatically close. This valve will automatically reopen if reactor water level subsequently decreases to the low initiation level. If the operator closes the injection valve before the predetermined high level is reached, the injection valve will not automatically reopen if the reactor water level subsequently decreases to the initiation level. Once the HPCF emergency mode is initiated, the system remains in this mode until manually stopped by the operator.

The HPCF fault tree is presented in Figure 19D.6-3 and applicable failure rate data are provided in Tables 19D.6-14 and 19D.6-15. This tree provides the basis for determining the conditional probability that neither HPFL loop will start on demand and run for 24 hours delivering rated flow, given a demand and that AC and DC electric power are available on their respective system buses.

19D.6.2.4 Residual Heat Removal System—Core Flooding (LPFL) Mode

The RHR System is a closed system consisting of three independent pump loops which inject water into the vessel and/or remove heat from the reactor vessel or containment.

Each of the pump loops contains the necessary piping, pumps, valves, and heat exchangers. In the LPFL mode, each loop draws water from the suppression pool and injects it into the vessel outside the core shroud via the feedwater line on one loop and via dedicated RHR return lines on two loops.

Each loop is in a single quadrant of the reactor building and receives its electric power from a bus separate from those serving the other two loops. Each bus is supplied from both on-site and offsite power sources. Each LPFL pump is initiated automatically following receipt of a high drywell pressure or low reactor water level initiation signal, and the LPFL injection valve opens on the initiation signal after a low rector pressure permissive signal is received. Each loop of the system can also be placed in operation and shut down manually from the control room. Suction is taken from the suppression pool, with the pump discharge being diverted through the minimum flow lines until the injection valve is signaled to open on low reactor pressure.

The fault tree for combined failure of all three LPFL loops is presented in Figure 19D.6-4. Tables 19D.6-14 and 19D.6-15 provide failure rate data for this system. Evaluation of this tree provides the conditional probability that the RHR System, in the injection mode, will fail to start on demand and inject water into the reactor core, given a demand and that the reactor is at low pressure.

19D.6.2.5 Automatic Depressurization System (ADS)

The Automatic Depressurization System (ADS) is a safety system designed to depressurize the reactor when the reactor is shut down and isolated from the power conversion system. Safety/relief valves are mounted on the main steam lines. They provide, in conjunction with reactor trip, overpressure protection for reactor coolant pressure boundary components. Each SRV discharge is piped to the suppression pool to permit condensation of the discharged steam in the pool. The SRVs are also used for the ADS, which provides rapid reactor depressurization for postulated accidents where large amounts of injection water at low pressure are required for core cooling.

Eight of the eighteen safety relief valves located on the main steam lines are initiated by the ADS. Three of the eight ADS valves are required for depressurization of the reactor following an isolation event. The ADS depressurizes the reactor pressure vessel to allow use of the RHR System in the core flooding mode for reactor water makeup. The system initiates automatically upon receipt of both low reactor water level and high drywell pressure signals. A high drywell pressure bypass timer will allow ADS initiation on only RPV low water level providing this signal exists longer than the bypass timer and ADS timer setpoints. The initiation signal is interlocked to prevent depressurization unless at least one RHR or HPCF pump is running. ADS may also be initiated manually by the operator. The system requires DC power for the solenoid valves and a nitrogen gas supply for the servo valves and pneumatic actuators for the ADS safety relief valves.

Nitrogen gas for the ADS function is supplied from either the atmospheric control system or two backup safety grade nitrogen gas supplies through an accumulator for short term supply. In addition, each safety relief valve has an individual nitrogen accumulator for short term supply for the relief function, making two accumulators each for the SRVs used for ADS. As a backup to the system, the eight ADS valves plus the ten non-ADS safety relief valves can be manually actuated individually to depressurize the system.

The fault tree for reactor depressurization is presented in Figure 19D.6-5 and Tables 19D.6-14 and 19D.6-15 provide the failure rate data for this system. Evaluation of this tree provides the conditional probability that reactor depressurization will not be accomplished, given a demand and that at least one of the five ECCS pumps is running.

19D.6.3 HEAT REMOVAL FAULT TREES

System fault trees presented in this subsection were developed to determine the probability of failure of the RHR System to perform successfully in each of its heat removal modes of operation, given a demand. The RHR is a closed system consisting of three independent pump loops which inject water into the vessel and/or remove heat from the reactor vessel or containment. Each RHR loop contains its own pumps, valves, heat exchangers, and necessary piping.

19D.6.3.1 RHR - Suppression Pool Cooling Mode

The RHR System in the suppression pool cooling mode provides cooling to remove heat released into the suppression pool (wetwell) as necessary, following heat dumps to the pool. During this mode of operation, water is pumped from the suppression pool, through the RHR heat exchangers, and back to the pool. This subsystem is manually activated and is shut down by operator action or initiation of the LPFL subsystem.

The fault tree for RHR System in the suppression pool cooling mode of operation is presented in Figure 19D.6-6. Failure data are provided in Tables 19D.6-14 and 19D.6-15. Evaluation of this tree provides the conditional probability that none of the three loops will be initiated and provide pool cooling given a demand.

19D.6.3.2 RHR - Shutdown Cooling Mode

The RHR System, in the shutdown cooling mode, removes decay and sensible heat from the reactor following shutdown such that refueling and servicing operations may proceed. After a normal blowdown to the main condenser, the shutdown cooling subsystem is activated to remove residual heat from the reactor vessel water to cool it to meet shutdown cooling requirements after the control rods are inserted. The subsystem then maintains or reduces this temperature.

Reactor water is cooled by pumping it directly from the reactor shutdown cooling nozzles, through the heat exchangers, and back to the vessel via feedwater on one loop and the dedicated RHR return lines on the other two loops. The subsystem is initiated and shut down by operator action.

The fault tree for the RHR System in the shutdown cooling mode of operation is presented in Figure 19D.6-7. Failure rate data used in the analysis are provided in Tables 19D.6-14 and 19D.6-15.

This tree represents the probability that none of the three loops will be initiated and provide shutdown cooling, given a demand.

19D.6.3.3 RHR - Wetwell and Drywell Spray Subsystem

The wetwell and drywell Spray Subsystem is employed to remove decay heat and condense steam in both the drywell and wetwell gas columns to prevent overpressurization of the containment.

Two of the RHR loops (B and C) provide wetwell and drywell spray cooling. This subsystem provides steam condensation and containment atmospheric cooling by pumping water from the suppression pool, through the heat exchangers, and into the wetwell and drywell spray spargers in the containment building. This subsystem is initiated and terminated by operator action. Drywell spray is enabled in the presence of high drywell pressure.

The fault trees representing the wetwell and drywell spray modes of operation of the RHR System are provided in Figures 19D.6-8a and 19D.6-8b, respectively. The database for evaluation of these fault trees is presented in Tables 19D.6-14 and 19D.6-15.

Evaluation of these trees provides the probability that neither Loop B nor Loop C will be initiated and provide drywell (or wetwell) spray cooling, given a demand.

19D.6.4 SUPPORT SYSTEM FAULT TREES

19D.6.4.1 Electric Power System

The station electrical power distribution system is designed to provide reliable power supply to the ABWR safety-related systems. This power is taken from two offsite sources.

In the event offsite sources are lost, three emergency diesel generators and four DC batteries are available onsite to meet the power requirements of the safety-related systems. The electrical power is supplied to the safety-related loads from different AC and DC buses at different voltage levels. These buses and the onsite emergency sources are arranged into three AC divisions, four DC divisions and four 120V AC uninterruptable power supply (UPS) divisions designed with a high degree of independency.

Fault trees were developed for each bus supplying essential loads. These trees are linked to the various other safety system fault trees. Events common in different trees are designated with identical acronyms to insure proper common cause failure treatment when the electrical power fault trees are linked to the various system fault trees.

The developed fault trees are presented in Figures 19D.6-9a through 19D.6-13c. Failure rates used to quantify these fault trees are presented in Tables 19D.6-14 and 19D.6-15.

19D.6.4.2 Service Water Systems

Essential equipment in the reactor building is cooled by the Reactor Building Cooling Water (RCW) System, which consists of three divisions. Each division is a closed cooling water loop which removes heat from the RHR heat exchangers, HVAC emergency cooling water system refrigerators, diesel generators, and other equipment. Heat is discharged through the RCW heat exchangers to the Reactor Service Water (RSW) System. Each RCW division has two 50% capacity motor driven pumps and three 33.5% capacity heat exchangers.

The RSW also consists of three divisions, each of which removes heat from its corresponding RCW heat exchangers and releases it to the UHS. Each division has two 50% capacity motor driven pumps which send UHS cooling water through the RCW heat exchangers.

The UHS involves the utilization of forced convection cooling towers. The following assumptions have been made about the added cooling tower fans:

- There are three divisions of fans for the UHS in each unit
- Each division contains two 50% capacity fans (accident loads)

- For non-accident cooling loads, (i.e. no suppression pool cooling), each fan is 100% capacity.
- Each forced cooling fan is interlocked with one of the RSW pumps in its Division. Each fan and its associated pump operate as a pair and receive the same start and trip signals. Therefore, like the RSW pumps, three of the forced cooling fans will normally be in operation for each unit
- One forced cooling fan in each division will normally be in standby during power operation
- The power supplies for the forced cooling fans are safety-related 480 V power supplies

A loss of the cooling capacity of these forced cooling fans is assumed to impact the ability of the UHS to remove heat and thus provide cooling for the Reactor Building Cooling Water (RBCW) System. Therefore, loss of cooling from these fans results in a loss of RBCW. The ABWR MOR will be altered to include the failure of the forced cooling fans under the service water system logic with logic being added for each of the three divisions.

In order to maintain the level of modeling complexity comparable to the complexity of the rest of the model, the following modeling conventions will be used:

Although the fans are expected to be rotated on a monthly basis, for modeling purposes, the train A, B, and C fans are assumed to be in operation and the train D, E, and F fans are assumed to be in standby. The model will not be used for applications which involve symmetry issues and as a result this simplification is acceptable.

Although each fan will typically have the same unavailability time associated with it for testing and maintenance, for modeling purposes, all unavailability was assigned to the standby fans. The model will not be used for applications which involve symmetry issues and as a result this simplification is acceptable.

During normal operation, one RCW and one RSW pump in each loop in each division and two RCW heat exchangers in each division are operating. Under these conditions, sufficient cooling capacity is available to provide seal and motor bearing cooling water for the core cooling pumps. Also, sufficient cooling capacity is available to remove heat from the RHR heat exchangers during LOCA if at least two loops are operated with all pumps and heat exchangers.

The operating and standby pumps and heat exchangers are interchanged monthly. During accident conditions, the standby pumps and heat exchangers are put into operation to provide additional cooling capacity.

The HVAC Emergency Cooling Water (HECW) System receives cooling from the RCW System through six refrigerators. This system in turn provides cooling to the three reactor building safety-related electrical equipment areas, three control building safety-related equipment areas, as well as the main control area envelope served by the control room habitability area HVAC.

The HECW System is comprised of three loops. Loop A has two pumps and two refrigerators which provide cooling to the control building Division I equipment area and the reactor building safety-related electrical equipment area.

Loops B and C have two pumps and two refrigerators each. One of these four pumps and its associated refrigerator is normally in standby mode with the pump/refrigerators rotated in and out of service equally. Loops B and C provide cooling to the reactor building safety-related equipment areas B and C as well as the control building safety-related equipment areas B and C, respectively. The loop with both refrigerators and pumps in operation provides cooling to the main control area envelope. The standby refrigerator and pump in the other loop are available to cool the main control area envelope should one of the two pumps in the operating loop fail. Each division is designed so one pump/refrigerator is sized to provide cooling to the reactor building safety-related electrical equipment area and to the control building safety-related equipment area.

The combined RCW and RSW System fault tree for each of the three divisions is presented in Figures 19D.6-14a through 19D.6-14c and applicable failure rate data are provided in Tables 19D.6-14 and 19D.6-15. The HECW System fault tree is presented in Figures 19D.6-23a through 19D.6-23c. The HECW failure rate data are included in Tables 19D.6-14 and 19D.6-15. These support system trees are combined with the various front line system and functional fault trees to evaluate core cooling and heat removal function failures.

19D.6.4.3 Instrumentation System

Each fault tree contained in this subsection represents the overall complex of instrument channels, signal logics, and transmission networks involved in generating either a reactor pressure, reactor level, or drywell pressure signal used to cause a reactor trip or to initiate the various ECCS Systems in the event of an emergency.

Fault trees were developed for each signal in each electrical division. These trees are linked to the various other safety system fault trees. Events common to a number of trees are designated with identical acronyms to insure proper common cause failure treatment when these instrumentation trees are linked to the system fault trees.

The instrumentation fault trees are presented in Figures 19D.6-15a through 19D.6-15j. Failure rate data used to evaluate these trees are provided in Tables 19D.6-14 and 19D.6-15.

There have been changes to the instrumentation and control systems architecture due to standard departures however they are not expected greatly impact the overall results of the PRA.

19D.6.5 REACTIVITY CONTROL FAULT TREES

19D.6.5.1 Reactivity Control Functional Fault Tree

System fault trees developed to determine the probability of failure to control reactivity and successfully shut down the reactor, given a demand, are presented in this subsection. The functional fault tree shown in Figure 19D.6-16 integrates each of the individual systems into the overall reactivity control function.

Fault trees were developed for the Reactor Protection System, the Control Rod Drive System, the Standby Liquid Control System, the recirculation pump trip, and alternate rod insertion in varying degrees of detail. The probability of total loss of reactivity control, given a demand, was assessed to be very low.

19D.6.5.2 Reactor Protection System (RPS)

The Reactor Protection System (RPS) is the overall complex of instrument channels, trip logics, manual controls and trip actuators that are involved in generating a reactor trip or scram. The system causes a reactor trip for situations which could result in unsafe reactor operating conditions. The RPS is a four-division system which is redundantly designed so that the failure of any single element will not interfere with a required trip.

Any single channel or division element operating falsely will not cause a trip because it will trip only one channel or only one of the two solenoids of the scram pilot valves. It combines a very high probability of operating when needed with a very low probability of operating falsely.

The Reactor Protection System is a warning and trip system implemented with software logic installed in microprocessors. The critical functions of this system are to:

(1) Make the primary decisions related to warning and trip conditions of the individual instrument channels.

(2) Make the decision for system trip (emergency reactor shutdown) based on coincidence of instrument channel trip conditions.

RPS includes detectors, switches microprocessors, solid state logic circuits, relay type contactors, relays, solid state load drivers, lamps, displays, signal transmission routes, circuits and other equipment which are required to execute the functions of the system.

The RPS fault tree is presented in Figure 19D.6-24 and applicable failure probabilities are provided in Tables 19D.6-14 and 19D.6-15. Evaluation of this fault tree provides the conditional probability that the RPS will fail to transmit a scram signal given the need.

19D.6.5.3 Control Rod Drive (CRD) System

The Control Rod Drive (CRD) System provides rapid control rod insertion (scram) so that no fuel damage results from any abnormal operating transient. An alternative method can be used to insert all the control rods in the event of a failure of the RPS logic. ARI valves can be opened by ATWS logic to vent the air header and cause hydraulic scram. The system is composed of three major elements:

(1) Electro-hydraulic fine motion control rod drive (FMCRD) mechanisms.

(2) Hydraulic control units (HCU).

(3) The control rod drive hydraulic system.

The hydraulic power required for scram is provided by high pressure water stored in the individual HCUs. Each HCU contains a nitrogen-water accumulator charged to a high pressure, and the necessary valves and components to scram two FMCRDs. A diverse means of rod insertion is to drive all the rods in simultaneously with the FMCRD motors.

The fault trees presented in this subsection address the failure of the electro-hydraulic and mechanical portions of the control rod drive system. Three fault trees were constructed on the basis of information available on the ABWR CRD System:

(1) Failure to insert an individual control rod.

(2) Failure of a hydraulic control unit.

(3) Failure of the CRD System to control reactivity.

Fault trees representing the above three situations are presented in Figures 19D.6-17, through 19D.6-19. Failure rate data are furnished in Tables 19D.6-14 and 19D.6-15.

The CRD System is also used as a backup reactor vessel makeup system. The use of CRD injection water as a makeup source is assumed to be similar to current CRD designs which allow the operation of two CRD pumps and associated components in maximized flow configuration. In this configuration, the flow injected into the reactor vessel is sufficient to maintain level for cases involving decay heat removal and when no other injection source is available. An undeveloped event is provided within the model as a recovery action addressing the use of CRD in this mode. A probability of 0.1 is chosen to represent the unreliability of both the mechanical components and the operator action to maximize flow.

19D.6.5.4 Standby Liquid Control System (SLCS)

The Standby Liquid Control System (SLCS) is a redundant diverse backup system to the rod control system. It is designed to be capable of automatically shutting down the reactor from full power operation to a cold subcritical condition without the insertion of control rods, by injecting sodium pentaborate solution into the reactor. The SLCS is automatically initiated upon receiving an anticipated transient without scram (ATWS) signal. The system can also be initiated manually through the keyboard switches in the main control room.

Redundancy is provided in the SLCS design by the inclusion of two 100% capacity parallel trains, either of which is capable of injecting the sodium pentaborate into the reactor. The boron solution is pumped from the SLCS storage tank and injected into the reactor vessel through the HPCF System sparger.

The SLCS fault tree is presented in Figure 19D.6-20 and related failure rate data are documented in Tables 19D.6-14 and 19D.6-15. Evaluation of this tree provides the conditional probability that the SLCS will fail to inject boron solution into the reactor vessel, given a demand and that the SLCS storage tank contains sufficient boron.

19D.6.5.5 Recirculation Pump Trip (RPT)

The recirculation pump trip (RPT) fault tree is presented in this subsection. During transient events when rapid power reduction is required, the Recirculation Flow Control system accepts and executes requests for RPT from external systems. The ten recirculation pumps are tripped in two stages. The first four pumps are tripped upon receipt of either a turbine control valve fast closure, turbine stop valve closure, RPV high pressure, or reactor water level 3 signal. The remaining six pumps are tripped upon receipt of a reactor water level 2 signal. For purposes of the PRA analysis, only the RPT functions associated with high pressure and reactor water level are considered.

The RPT fault tree represents the overall complex of instrument channels, signal logics, and transmission networks involved in generating reactor pressure and level signals leading to the tripping of the reactor recirculation pumps. Events common to different signals are designated with identical acronyms to insure proper common cause failure treatment in the evaluation of these trees.

The RPT fault tree is presented in Figure 19D.6-21. Failure rate data used in its evaluation are provided in Tables 19D.6-14 and 19D.6-15. The tree represents the probability that not all ten recirculation pumps will be tripped, given a demand.

19D.6.5.6 Alternate Reactivity Insertion (ARI)

The alternate reactivity insertion (ARI) fault tree is presented in this subsection. It consists of two alternate means of initiating control rod insertion as backups to the normal scram system.

The ARI fault tree represents the overall complex of instrument channels, signal logics, and transmission networks involved in generating reactor pressure and level signals leading to the initiation of alternate reactivity insertion signals. These include signals to run in the control rods with the FMCRD electric motors as well as those to the ARI valves to depressurize the scram pilot air header. Events common to different signals are designated with identical acronyms to insure proper common cause failure treatment in the evaluation of this tree. The fault tree includes the mechanical failure of the ARI valves.

The ARI fault tree is presented in Figure 19D.6-22. Failure rate data used in its evaluation are provided in Tables 19D.6-14 and 19D.6-15. The tree represents the probability that neither of the two alternate means of reactivity insertion will be initiated, given a demand.

19D.6.6 REFERENCES

19D.6-1 ABWR Standard Safety Analysis Report, Revision 5, General Electric Company.

Table 19D.6-1	Not Used
Table 19D.6-2	Not Used
Table 19D.6-3	Not Used
Table 19D.6-4	Not Used
Table 19D.6-5	Not Used
Table 19D.6-6	Not Used
Table 19D.6-7	Not Used
Table 19D.6-8	Not Used
Table 19D.6-9	Not Used
Table 19D.6-10	Not Used
Table 19D.6-11	Not Used
Table 19D.6-12	Not Used
Table 19D.6-13	Not Used

Table 19D.6-14
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ACCF004	COMMON CAUSE FAILURE OF PT SENSORS	1	N	1.36E-4	N
ACVF008A	CHECK VALVE P54-F008A FAILS TO OPEN	1	N	6.07E-5	N
ACVF008B	CHECK VALVE P54-F008B FAILS TO OPEN	1	N	6.07E-5	N
ACVF026A	CHECK VALVE B21-F026A FAILS TO OPEN	1	N	6.07E-5	N
ACVF026C	CHECK VALVE B21-F026C FAILS TO OPEN	1	N	6.07E-5	N
ACVF026F	CHECK VALVE B21-F026F FAILS TO OPEN	1	N	6.07E-5	N
ACVF026H	CHECK VALVE B21-F026H FAILS TO OPEN	1	N	6.07E-5	N
ACVF026L	CHECK VALVE B21-F026L FAILS TO OPEN	1	N	6.07E-5	N
ACVF026N	CHECK VALVE B21-F026N FAILS TO OPEN	1	N	6.07E-5	N
ACVF026R	CHECK VALVE B21-F026R FAILS TO OPEN	1	N	6.07E-5	N
ACVF026T	CHECK VALVE B21-F026T FAILS TO OPEN	1	N	6.07E-5	N
ACVF207F	CHECK VALVE P54-F207 FAILS CLOSED	1	N	6.07E-5	N

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ADSMAN	OPERATOR FAILS TO OPEN NON-ADS SRV's	1	N	2.00E-3	N
AHPT006	MISCALIBRATION OF PRESSURE SENSORS E22-PT006	1	N	2.00E-5	N
ALCVLD1L	DIVISION 1 LOGIC CARD FAILS	1	N	7.50E-5	N
ALCVLD2L	DIVISION 2 LOGIC CARD FAILS	1	N	7.50E-5	N
ARVCCFD	COMMON CAUSE FAILURE OF SRV's	1	N	4.34E-5	N
ARVMECAD	VALVE B21-F010A FAILS MECHANICALLY	1	N	7.77E-3	N
ARVMECCD	VALVE B21-F010C FAILS MECHANICALLY	1	N	7.77E-3	N
ARVMECFD	VALVE B21-F010F FAILS MECHANICALLY	1	N	7.77E-3	N
ARVMECHD	VALVE B21-F010H FAILS MECHANICALLY	1	N	7.77E-3	N
ARVMECLD	VALVE B21-F010L FAILS MECHANICALLY	1	N	7.77E-3	N
ARVMECND	VALVE B21-F010N FAILS MECHANICALLY	1	N	7.77E-3	N
ARVMECRD	VALVE B21-F010R FAILS MECHANICALLY	1	N	7.77E-3	N
ARVMECTD	VALVE B21-F010T FAILS MECHANICALLY	1	N	7.77E-3	N
ASECSNA	NON-ACTUATION OF BACK-UP N2 BY OPERATOR	1	N	1.00E-1	N
ASF101AD	FAILURE IN DIV 1 ADS SRV PILOT SOLENOID VALVE	1	N	4.23E-4	N
ASF101CD	FAILURE IN DIV 1 ADS SRV PILOT SOLENOID VALVE	1	N	4.23E-4	N

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ASF101FD	FAILURE IN DIV 1 ADS SRV PILOT SOLENOID VALVE	1	N	4.23E-4	N
ASF101HD	FAILURE IN DIV 1 ADS SRV PILOT SOLENOID VALVE	1	N	4.23E-4	N
ASF101LD	FAILURE IN DIV 1 ADS SRV PILOT SOLENOID VALVE	1	N	4.23E-4	N
ASF101ND	FAILURE IN DIV 1 ADS SRV PILOT SOLENOID VALVE	1	N	4.23E-4	N
ASF101RD	FAILURE IN DIV 1 ADS SRV PILOT SOLENOID VALVE	1	N	4.23E-4	N
ASF101TD	FAILURE IN DIV 1 ADS SRV PILOT SOLENOID VALVE	1	N	4.23E-4	N
ASF102AD	FAILURE IN DIV 2 ADS SRV PILOT SOLENOID	1	N	4.23E-4	N
ASF102CD	FAILURE IN DIV 2 ADS SRV PILOT SOLENOID	1	N	4.23E-4	N
ASF102FD	FAILURE IN DIV 2 ADS SRV PILOT SOLENOID	1	N	4.23E-4	N
ASF102HD	FAILURE IN DIV 2 ADS SRV PILOT SOLENOID	1	N	4.23E-4	N
ASF102LD	FAILURE IN DIV 2 ADS SRV PILOT SOLENOID	1	N	4.23E-4	N
ASF102ND	FAILURE IN DIV 2 ADS SRV PILOT SOLENOID	1	N	4.23E-4	N
ASF102RD	FAILURE IN DIV 2 ADS SRV PILOT SOLENOID	1	N	4.23E-4	N
ASF102TD	FAILURE IN DIV 2 ADS SRV PILOT SOLENOID	1	N	4.23E-4	N
AVF002CF	MANUAL VALVE P54-F002C FAILS TO OPEN	1	N	1.42E-4	N
AVF002DF	MANUAL VALVE P54-F002D FAILS TO OPEN	1	N	1.42E-4	N

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
BCV003B	ISOLATION CHECK VALVE B21-F003B FAILS CLOSED	1	N	6.07E-5	N
BCV004B	ISOLATION CHECK VALVE B21-F004B FAILS CLOSED	1	N	6.07E-5	N
BF004AFC	CHECK VALVE B21-F004A FAILS TO OPEN	1	N	6.07E-5	N
C001ACB	RHR CIRCUIT BREAKER FAILS TO CLOSE	1	N	1.10E-3	N
C001AMF	RHR PUMP A FAILS TO START	1	N	1.56E-3	N
C001AMOV	MANUAL OVERRIDE FAILS INITIATION SIGNAL	1	N	1.80E-4	N
C001BCB	RHR CIRCUIT BREAKER FAILS TO CLOSE	1	N	1.10E-3	N
C001BMF	RHR PUMP B FAILS TO START	1	N	1.56E-3	N
C001BMOV	MANUAL OVERRIDE FAILS INITIATION SIGNAL	1	N	1.80E-4	N
C001CCB	RHR CIRCUIT BREAKER FAILS TO CLOSE	1	N	1.10E-3	N
C001CMF	RHR PUMP C FAILS TO START	1	N	1.56E-3	N
C001CMOV	MANUAL OVERRIDE FAILS INITIATION SIGNAL	1	N	1.80E-4	N
CALN002A_A	MISCALIBRATION OF LOOP A FLOW TRANSMITTERS FT008A	1	N	5.00E-5	N
CALN002A_B	MISCALIBRATION OF LOOP B FLOW TRANSMITTERS FT008B	1	N	5.00E-5	N
CALN002A_C	MISCALIBRATION OF LOOP C FLOW TRANSMITTERS FT008C	1	N	5.00E-5	N
CCFAPRM	CCF OF APRMS	1	N	8.43E-6	N

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
CCFOLU	CCF OF OUTPUT LOGIC UNIT	1	N	3.92E-7	N
CCFRLY	CCF OF BACKUP SCRAM RELAYS	1	N	1.18E-6	N
CDWSPHDF	COMMON SPRAY SPARGER FAILURE	1	N	1.00E-4	N
CMAN	MANUAL INITIATION FAILURE	1	N	1.00E-1	N
COO1BCB	RHR CIRCUIT BREAKER FAILS TO CLOSE	1	N	1.10E-3	N
COO1CCB	RHR CIRCUIT BREAKER FAILS TO CLOSE	1	N	1.10E-3	N
CTGMANSW	CTG MANUAL DISCONNECT SWITCH OPEN	1	N	3.00E-3	N
EACEN	LOSS OF POWER	1	N	3.43E-4	N
ECB902H	INCOMING BKR 902 FAILS TO CLOSE	1	N	1.10E-3	N
ECB905H	DG BKR 905 FAILS TO CLOSE	1	N	1.10E-3	N
ECB922H	INCOMING BKR 922 FAILS TO CLOSE	1	N	1.10E-3	N
ECB925H	DG BKR 925 FAILS TO CLOSE	1	N	1.10E-3	N
ECB942H	INCOMING BKR 942 FAILS TO CLOSE	1	N	1.10E-3	N
ECB945H	DG BKR 945 FAILS TO CLOSE	1	N	1.10E-3	N
ECBCTG1	CTG DIV 1 BREAKER FAILS TO CLOSE	1	N	1.10E-3	N
ECBCTG2	CTG DIV 2 BREAKER FAILS TO CLOSE	1	N	1.10E-3	N

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ECBCTG3	CTG DIV 3 BREAKER FAILS TO CLOSE	1	N	1.10E-3	N
ECBN01H	OUTPUT TRANSFER BREAKER IN NORMAL CHARGER FAILS TO CLOSE	1	N	1.10E-3	N
ECBN02H	OUTPUT TRANSFER BREAKER IN NORMAL CHARGER FAILS TO CLOSE	1	N	1.10E-3	N
ECBN03H	OUTPUT TRANSFER BREAKER IN NORMAL CHARGER FAILS TO CLOSE	1	N	1.10E-3	N
ECBN04H	OUTPUT TRANSFER BREAKER IN NORMAL CHARGER FAILS TO CLOSE	1	N	1.10E-3	N
EDC24A	LOSS OF 24V DC POWER SUPPLY A	1	N	1.34E-5	N
EDC24B	LOSS OF 24V DC POWER SUPPLY B	1	N	1.15E-5	N
EDC24C	LOSS OF 24V DC POWER SUPPLY C	1	N	1.15E-5	N
EDCN	FAILURE OF 125V DC	1	N	1.34E-5	N
EDEU1	RECTIFIER CIRCUIT FAILURE	1	N	1.43E-6	N
EDEU2	RECTIFIER CIRCUIT FAILURE	1	N	1.43E-6	N
EDEU3	RECTIFIER CIRCUIT FAILURE	1	N	1.43E-6	N
EDEU4	RECTIFIER CIRCUIT FAILURE	1	N	1.43E-6	N
EDGCD	2 D.G.'S CCF	1	N	1.94E-3	N
EDGCDE	3 D.G.'S CCF	1	N	3.81E-4	N
EDGCE	2 D.G.'S CCF	1	N	1.94E-3	N

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
EDGCR	DG C FAILS TO LOAD AND RUN FOR FIRST HOUR	1	N	2.90E-3	N
EDGDE	2 D.G.'S CCF	1	N	1.94E-3	N
EDGDR	DG D FAILS TO LOAD AND RUN FOR FIRST HOUR	1	N	2.90E-3	N
EDGER	DG E FAILS TO LOAD AND RUN FOR FIRST HOUR	1	N	2.90E-3	N
EDGFSCD	DG FAILS TO START	1	N	2.21E-2	N
EDGFSDD	DG FAILS TO START	1	N	2.21E-2	N
EDGFSED	DG FAILS TO START	1	N	2.21E-2	N
EHU69C	OPERATOR FAILS TO TRANSFER POWER	1	N	1.00E-3	N
EHUB1	OPERATOR FAILS TO BYPASS	1	N	1.00E-3	N
EHUB2	OPERATOR FAILS TO BYPASS	1	N	1.00E-3	N
EHUB3	OPERATOR FAILS TO BYPASS	1	N	1.00E-3	N
EHUB4	OPERATOR FAILS TO BYPASS	1	N	1.00E-3	N
EHUS1AD	OPERATOR FAILS TO TRANSFER STANDBY CHARGER TO DIV 1	1	N	1.00E-3	N
EHUS1BD	OPERATOR FAILS TO TRANSFER STANDBY CHARGER TO DIV 2	1	N	1.00E-3	N
EHUS1CD	OPERATOR FAILS TO TRANSFER STANDBY CHARGER TO DIV 3	1	N	1.00E-3	N
EHUS1DD	OPERATOR FAILS TO TRANSFER STANDBY CHARGER TO DIV 4	1	N	1.00E-3	N

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ELOOP1	LOSS OF OFF-SITE LINE 1 (1,1) POWER	1	N	1.00E-2	N
ELOOP12	COMMON MODE LOSS OF BOTH OFFSITE POWER	1	N	1.00E-3	N
ELOOP2	LOSS OF OFF-SITE LINE 2 (2,2)	1	N	1.00E-2	N
EMTF1D	MANUAL BYPASS SWITCH FAILURE	1	N	1.05E-4	N
EMTF2D	MANUAL BYPASS SWITCH FAILURE	1	N	1.05E-4	N
EMTF3D	MANUAL BYPASS SWITCH FAILURE	1	N	1.05E-4	N
EMTF4D	MANUAL BYPASS SWITCH FAILURE	1	N	1.05E-4	N
F002AFC	PUMP DISCHARGE CHECK VALVE E11-F002A FAILS CLOSED	1	N	6.07E-5	N
F002BFC	PUMP DISCHARGE CHECK VALVE E11-F002B FAILS CLOSED	1	N	6.07E-5	N
F002CFC	PUMP DISCHARGE CHECK VALVE E11-F002C FAILS CLOSED	1	N	6.07E-5	N
GTURBINEFLR	CTG FAILS TO LOAD AND RUN FOR FIRST HOUR	1	N	1.87E-3	N
GTURBINEFS	CTG FAILS TO START AND LOAD	1	N	2.43E-2	N
HBMAER1	TEST VALVE E22-F005B MISPOSITIONED (CLOSED)	1	N	1.00E-2	N
HBMAER2	TEST VALVE INADVERTENTLY LEFT OPEN	1	N	1.00E-2	N
HCMAER1	TEST VALVE E22-F005C MISPOSITIONED (CLOSED)	1	N	1.00E-2	N
HCMAER2	TEST VALVE INADVERTENTLY LEFT OPEN	1	N	1.00E-2	N

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
HCV07BHP	CHECK VALVE E22-F007B FAILS CLOSED	1	N	6.07E-5	N
HCV07CHP	CHECK VALVE E22-F007C FAILS CLOSED	1	N	6.07E-5	N
HFE008CF	MISCALIBRATION OF FLOW TRANSMITTERS (CCF)	1	N	5.00E-5	N
HFELEBHX	CCF OF WATER LEVEL 8 SENSORS MISCALIBRATED (4 DIV)	1	N	2.00E-5	N
HOOBOPHL	OP. FAILS TO ATTEMPT MANUAL INITIATION AFTER 30 MIN.	1	N	1.01E-3	N
HPBMAINT	HPCF-B UNAVAIL DUE TO MAINT	1	N	2.00E-2	N
HPBMBC	PUMP B MOTOR BEARING COOLER FAILS	1	N	1.14E-4	N
HPBMSC	PUMP B MECH. SEALER COOLER FAILS	1	N	1.14E-4	N
HPCMAINT	HPCF-C UNAVAIL DUE TO MAINT	1	N	2.00E-2	N
HPCMBC	PUMP C MOTOR BEARING COOLER FAILS	1	N	1.14E-4	N
HPCMSC	PUMP C MECH. SEALER COOLER FAILS	1	N	1.14E-4	N
HPM01BDW	HPCF PUMP B FAILS TO START	1	N	1.56E-3	N
HPM01CDW	HPCF PUMP C FAILS TO START	1	N	1.56E-3	N
HPR007CF	MISCALIBRATION OF PRESSURE TRANSMITTERS (CCF)	1	N	5.00E-5	N
HSV043C	VALVE F043 FAILS TO CLOSE	1	N	4.23E-4	N
HSV044F	VALVE F044 FAILS	1	N	4.23E-4	N

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
HSV047D	SOLENOID VALVE F047 FAILS TO OPEN	1	N	4.23E-4	N
HSV48AD	SOLENOID VALVE F048A FAILS TO OPEN	1	N	4.23E-4	N
HSV48BD	SOLENOID VALVE F048B FAILS TO OPEN	1	N	4.23E-4	N
HSV49AD	SOLENOID VALVE F049A FAILS TO OPEN	1	N	4.23E-4	N
HSV49BD	SOLENOID VALVE F049B FAILS TO OPEN	1	N	4.23E-4	N
HUEROR5	OPER FAILS TO ATTEMPT MANUAL TRANSFER FROM CST TO SUPP POOL	1	N	1.00E-2	N
LCST2	CST WATER LEVEL INSUFFICIENT	1	N	6.60E-6	N
MAINLD	CCF OF MAIN SCRAM LOAD DRIVERS	1	N	3.97E-7	N
NSDRPVF	RPV WATER LEVEL INADEQUATE	1	N	1.00E-6	N
OCV001HP	PUMP DISCHARGE CHECK VALVE F004A FAILS TO OPEN	1	N	6.07E-5	N
OCV002HP	PUMP DISCHARGE CHECK VALVE F004B FAILS TO OPEN	1	N	6.07E-5	N
OCV007	CHECK VALVE C42-F007 FAILS CLOSED	1	N	6.07E-5	N
OCV008	CHECK VALVE C41-F008 FAILS CLOSED	1	N	6.07E-5	N
OHR001M	HEATER NOT IN SERVICE	1	N	1.40E-3	N
OPM001HR	SLC PUMP A C001A FAILS TO START	1	N	1.56E-3	N
OPM002HR	SLC PUMP B C001B FAILS TO START	1	N	1.56E-3	N

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
OXV002HW	GATE VALVE F005B FAILS CLOSED (NLO-FC)	1095	H	1.76E-8	H
PCASIG_A	FAILURE OF LOOP A SUPPRESSION POOL TEMP SIGNAL T53-TRS-601A & B	1	N	7.50E-5	N
PCASIG_B	FAILURE OF LOOP B SUPPRESSION POOL TEMP SIGNAL T53-TRS-601A & B	1	N	7.50E-5	N
PCASIG_C	FAILURE OF LOOP C SUPPRESSION POOL TEMP SIGNAL T53-TRS-601A & B	1	N	7.50E-5	N
Q_FANSTA	FAILURE OF RUNNING TRAIN A FORCED AIR COOLING FAN TO START	1	N	1.08E-4	N
Q_FANSTB	FAILURE OF RUNNING TRAIN B FORCED AIR COOLING FAN TO START	1	N	1.08E-4	N
Q_FANSTC	FAILURE OF RUNNING TRAIN C FORCED AIR COOLING FAN TO START	1	N	1.08E-4	N
Q_FANSTD	FAILURE OF STANDBY TRAIN D FORCED AIR COOLING FAN TO START	1	N	2.31E-3	N
Q_FANSTE	FAILURE OF STANDBY TRAIN E FORCED AIR COOLING FAN TO START	1	N	2.31E-3	N
Q_FANSTF	FAILURE OF STANDBY TRAIN F FORCED AIR COOLING FAN TO START	1	N	2.31E-3	N
RCIMAINT	RCIC UNAVAILABLE DUE TO TESTING OR MAINTENANCE	1	N	2.00E-2	N
RCV002HP	CHECK VALVE F002 FAILS TO OPEN	1	N	6.07E-5	N
RCV003HP	CHECK VALVE E51-F003 FAILS TO OPEN	1	N	6.07E-5	N
RCV005HP	OUTBOARD CHECK VALVE F005 FAILS TO OPEN	1	N	6.07E-5	N
RCV007HP	CHECK VALVE F007 FAILS TO OPEN	1	N	6.07E-5	N
RCV038HP	CHECK VALVE F038 FAILS TO OPEN	1	N	6.07E-5	N

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
REOSSMSC	ELECTRICAL OVERSPEED SENSOR MISCALIBRATED	1	N	5.00E-5	N
RFE635HX	MISCALIBRATION OF CST LEVEL SENSORS	1	N	2.00E-5	N
RFL007CF	SENSOR MISCALIBRATION	1	N	5.00E-5	N
RHRCFER	FAILURE TO MANUALLY INITIATE	1	N	1.00E-1	N
RHRDWER	FAILURE TO MANUALLY INITIATE	1	N	5.00E-1	N
RHRSDER	FAILURE TO MANUALLY INITIATE	1	N	6.00E-5	N
RHRSPER	FAILURE TO MANUALLY INITIATE	1	N	1.00E-6	N
RISOLSIG	ISOLATION SIGNAL LOGIC FAILURE	1	N	1.36E-3	N
RLU001DW	LUBRICATION SYSTEM FAILS	1	N	4.20E-3	N
RMOSSMSC	MECHANICAL OVERSPEED SENSOR MISCALIBRATED	1	N	5.00E-5	N
ROERROR3	OPERATOR FAILS TO ATTEMPT MANUAL OPENING	1	N	1.00E-2	N
ROERROR4	OPERATOR FAILS TO ATTEMPT MANUAL VALVE OPENING	1	N	1.00E-1	N
ROERROR5	VALVE F009 INADVERT LEFT OPEN	1	N	1.00E-2	N
ROOIOPHL	OPERATOR FAILS TO ATTEMPT MANUAL INITIATION	1	N	1.00E-1	N
RPM001DW	PUMP C001 FAILS TO START	1	N	8.49E-3	N
RPR005CF	SENSOR MISCALIBRATION	1	N	5.00E-5	N

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
RPR303MC	LOW SUCTION PRESSURE XMTR MISCALIBRATED	1	N	5.00E-5	N
RPR309FL	BOTH HIGH TURBINE EXHAUST PRESSURE XMTRS PIS-Z613A AND E FAIL	1	N	1.57E-6	N
RPR309MC	HIGH TURBINE EXHAUST PRESSURE XMTR MISCALIBRATED	1	N	5.00E-5	N
RSTTCOPF	OPERATOR FAILS TO RESET TRIP CIRCUIT	1	N	1.00E-2	N
SCVF024A	CHECK VALVE B21-F029A FAILS CLOSED	1	N	6.07E-5	N
SCVF024C	CHECK VALVE B21-F029C FAILS CLOSED	1	N	6.07E-5	N
SCVF024F	CHECK VALVE B21-F029F FAILS CLOSED	1	N	6.07E-5	N
SCVF024H	CHECK VALVE B21-F029H FAILS CLOSED	1	N	6.07E-5	N
SCVF024L	CHECK VALVE B21-F029L FAILS CLOSED	1	N	6.07E-5	N
SCVF024N	CHECK VALVE B21-F029N FAILS CLOSED	1	N	6.07E-5	N
SCVF024R	CHECK VALVE B21-F029R FAILS CLOSED	1	N	6.07E-5	N
SCVF024T	CHECK VALVE B21-F029T FAILS CLOSED	1	N	6.07E-5	N
SCVF209F	CHECK VALVE P54-F209 FAILS CLOSED	1	N	6.07E-5	N
SLC000SA	BORON CONCENTRATION SAMPLING FAILURE	1	N	2.00E-5	N
SLC001HE	OPERATOR FAILS TO INITIATE	1	N	1.00E-1	N
SLC001TM	SLC LOOP UNAVAILABLE DUE TO TEST AND MAINTENANCE	1	N	1.40E-3	N

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
SLC002HE	OPERATOR FAILS TO ACT	1	N	2.00E-3	N
SLC002TM	SLC LOOP B UNAVAILABLE DUE TO TEST AND MAINTENANCE	1	N	1.40E-3	N
SRVSYFD	NON-ADS SRV's FAIL TO WORK	1	N	4.23E-4	N
SSF121AD	RELIEF VALVE SOLENOID ACTUATOR FAILS	1	N	4.23E-4	N
SSF121CD	RELIEF VALVE SOLENOID ACTUATOR FAILS	1	N	4.23E-4	N
SSF121FD	RELIEF VALVE SOLENOID ACTUATOR FAILS	1	N	4.23E-4	N
SSF121HD	RELIEF VALVE SOLENOID ACTUATOR FAILS	1	N	4.23E-4	N
SSF121LD	RELIEF VALVE SOLENOID ACTUATOR FAILS	1	N	4.23E-4	N
SSF121ND	RELIEF VALVE SOLENOID ACTUATOR FAILS	1	N	4.23E-4	N
SSF121RD	RELIEF VALVE SOLENOID ACTUATOR FAILS	1	N	4.23E-4	N
SSF121TD	RELIEF VALVE SOLENOID ACTUATOR FAILS	1	N	4.23E-4	N
VOPPERRF	OPERATOR FAILS TO START PUMP	1	N	1.00E-3	N
WCVH1F	STANDBY PUMP LEG CHECK VALVE P25-F001F FAILS TO OPEN	1	N	6.07E-5	N
WCVR1DD	STANDBY PUMP CHECK VALVE P21-F001D FAILS TO OPEN	1	N	6.07E-5	N
WCVR1ED	STANDYBY PUMP CHECK VALVE P21-F001E FAILS TO OPEN	1	N	6.07E-5	N
WCVR1FD	STANDYBY PUMP CHECK VALVE P21-F001F FAILS TO OPEN	1	N	6.07E-5	N

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
WCVS1DD	STANDBY PUMP LEG CHECK VALVE P41-F001D FAILS TO OPEN	1	N	6.07E-5	N
WCVS1ED	STANDBY PUMP LEG CHECK VALVE P41-F001E FAILS TO OPEN	1	N	6.07E-5	N
WCVS1FD	STANDBY PUMP LEG CHECK VALVE P41-F001F FAILS TO OPEN	1	N	6.07E-5	N
WDAMAINT	RHR LOOP A UNAVAILABLE DUE TO MAINTENANCE	1	N	2.00E-2	N
WDAMBC	PUMP A MOTOR BEARING COOLER FAILS	1	N	1.14E-4	N
WDAMSC	PUMP A MECH. SEALER COOLER FAILS	1	N	1.14E-4	N
WDBMAINT	RHR LOOP B UNAVAILABLE DUE TO MAINTENANCE	1	N	2.00E-2	N
WDBMBC	PUMP B MOTOR BEARING COOLER FAILS	1	N	1.14E-4	N
WDBMSC	PUMP B MECH. SEALER COOLER FAILS	1	N	1.14E-4	N
WDCMAINT	RHR LOOP C UNAVAILABLE DUE TO MAINTENANCE	1	N	2.00E-2	N
WDCMBC	PUMP C MOTOR BEARING COOLER FAILS	1	N	1.14E-4	N
WDCMSC	PUMP C MECH. SEALER COOLER FAILS	1	N	1.14E-4	N
WDNPSC	REACTOR PRESSURE DROPS CAUSING SUCTION CAVITATION	1	N	1.00E-6	N
WOPERR	OPERATOR FAILS TO PERFORM INDICATED ACTION	1	N	1.00E-2	N
WPMRC1DD	STANDBY PUMP FAILS TO START	1	N	1.56E-3	N
WPMRC1FD	STANDBY PUMP FAILS TO START	1	N	1.56E-3	N

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
WPMSC1DD	STANDBY PUMP FAILS TO START	1	N	1.56E-3	N
WPMSC1ED	STANDBY PUMP FAILS TO START	1	N	1.56E-3	N
WPMSC1FD	STANDBY PUMP FAILS TO START	1	N	1.56E-3	N
WPMSTRTF	STANDBY PUMP FAILURE TO START	1	N	1.56E-3	N
ZSP100DF	SUPP POOL WATER UNAVAILABLE DUE TO POOL RUPTURE	1	N	3.00E-7	N
ZSP200DW	SP WATER UNAVAILABLE DUE TO HIGH TEMPERATURE	1	N	1.00E-6	N
AACMRFAD	RUPTURE OF N2 ACCUMULATOR B21-A003A	8760	H	9.05E-8	H
AACMRFCD	RUPTURE OF N2 ACCUMULATOR B21-A003C	8760	H	9.05E-8	H
AACMRFFD	RUPTURE OF N2 ACCUMULATOR B21-A003F	8760	H	9.05E-8	H
AACMRFHD	RUPTURE OF N2 ACCUMULATOR B21-A003H	8760	H	9.05E-8	H
AACMRFLD	RUPTURE OF N2 ACCUMULATOR B21-A003L	8760	H	9.05E-8	H
AACMRFND	RUPTURE OF N2 ACCUMULATOR B21-A003N	8760	H	9.05E-8	H
AACMRFRD	RUPTURE OF N2 ACCUMULATOR B21-A003R	8760	H	9.05E-8	H
AACMRFTD	RUPTURE OF N2 ACCUMULATOR B21-A003T	8760	H	9.05E-8	H
AACUMLEA	ACCUMULATOR HAS LEAKED EMPTY	8760	H	9.05E-8	H
AACUMLEC	ACCUMULATOR HAS LEAKED EMPTY	8760	H	9.05E-8	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
AACUMLEF	ACCUMULATOR HAS LEAKED EMPTY	8760	H	9.05E-8	H
AACUMLEH	ACCUMULATOR HAS LEAKED EMPTY	8760	H	9.05E-8	H
AACUMLEL	ACCUMULATOR HAS LEAKED EMPTY	8760	H	9.05E-8	H
AACUMLEN	ACCUMULATOR HAS LEAKED EMPTY	8760	H	9.05E-8	H
AACUMLER	ACCUMULATOR HAS LEAKED EMPTY	8760	H	9.05E-8	H
AACUMLET	ACCUMULATOR HAS LEAKED EMPTY	8760	H	9.05E-8	H
ABYTMR1	BYPASS TIMER FAILS TO TRIP VALVE	360	H	1.50E-6	H
ABYTMR2	BYPASS TIMER FAILS TO TRIP VALVE	360	H	1.50E-6	H
AHPS006B	PRESSURE SENSOR E22-PT006B FAILS	360	H	1.57E-6	H
AHPS006C	PRESSURE SENSOR E22-PT006C FAILS	360	H	1.57E-6	H
ALNBRKA	BREAK IN N2 SUPPLY LINE TO B21-F010A	8760	H	4.93E-9	H
ALNBRKB	BREAK IN N2 SUPPLY LINE TO B21-F010B	8760	H	4.93E-9	H
AMV003AD	MO VALVE P54-F003A DOES NOT OPEN	360	H	1.90E-6	H
AMV003BD	MO VALVE P54-F003B DOES NOT OPEN	360	H	1.90E-6	H
AMVF007A	NORMALLY OPEN MO VALVE P54-F007A FAILS CLOSED	360	N	1.40E-7	N
AMVF007B	NORMALLY OPEN MO VALVE P54-F007B FAILS CLOSED	360	N	1.40E-7	N

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
AMVF012A	MO VALVE P54-F012A FAILS TO CLOSE	360	H	1.90E-6	H
AMVF012B	MO VALVE P54-F012B FAILS TO CLOSE	360	H	1.90E-6	H
AMVF203F	MO VALVE P54-F203 FAILS CLOSED	360	H	1.40E-7	H
ANAFACS	NO N2 FROM ACS	360	H	1.70E-6	H
ANTMD1VL	INTERNAL TIMER FAILS TO TRIP VALVE	360	H	1.50E-6	H
ANTMD2VL	INTERNAL TIMER FAILS TO TRIP VALVE	360	H	1.50E-6	H
APF205F	PCV VALVE P54-F205 FAILS CLOSED	360	H	1.20E-6	H
APR002A	PRESSURE XMITTER P54-002A FAILS HIGH	360	H	1.57E-6	H
APR002B	PRESSURE XMITTER P54-002B FAILS HIGH	360	H	1.57E-6	H
APR004	PRESSURE XMITTER P54-004 FAILS LOW	360	H	1.57E-6	H
APR005	PRESSURE XMITTER P54-005 FAILS HIGH	360	H	1.57E-6	H
APRM1F	DIVISION 1 APRM FAILS	4	H	6.80E-6	H
APRM2F	DIVISION 2 APRM FAILS	4	H	6.80E-6	H
APRM3F	DIVISION 3 APRM FAILS	4	H	6.80E-6	H
APRM4F	DIVISION 4 APRM FAILS	4	H	6.80E-6	H
APS004A	PRESSURE SENSOR E11-PT004A FAILS	360	H	1.57E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
APS004B	PRESSURE SENSOR E11-PT004B FAILS	360	H	1.57E-6	H
APS004C	PRESSURE SENSOR E11-PT004C FAILS	360	H	1.57E-6	H
ARVVAFAD	AIR ACTUATOR TO B21-F010A FAILS	8760	H	1.96E-7	H
ARVVAFCD	AIR ACTUATOR TO B21-F010C FAILS	8760	H	1.96E-7	H
ARVVAFFD	AIR ACTUATOR TO B21-F010F FAILS	8760	H	1.96E-7	H
ARVVAFHD	AIR ACTUATOR TO B21-F010H FAILS	8760	H	1.96E-7	H
ARVVAFLD	AIR ACTUATOR TO B21-F010L FAILS	8760	H	1.96E-7	H
ARVVAFND	AIR ACTUATOR TO B21-F010N FAILS	8760	H	1.96E-7	H
ARVVAFRD	AIR ACTUATOR TO B21-F010R FAILS	8760	H	1.96E-7	H
ARVVAFTD	AIR ACTUATOR TO B21-F010T FAILS	8760	H	1.96E-7	H
ASVF101A	FAILURE IN DIV 1 ADS SRV PILOT SOLENOID VALVE	8760	H	1.96E-7	H
ASVF101C	FAILURE IN DIV 1 ADS SRV PILOT SOLENOID VALVE	8760	H	1.96E-7	H
ASVF101F	FAILURE IN DIV 1 ADS SRV PILOT SOLENOID VALVE	8760	H	1.96E-7	H
ASVF101H	FAILURE IN DIV 1 ADS SRV PILOT SOLENOID VALVE	8760	H	1.96E-7	H
ASVF101L	FAILURE IN DIV 1 ADS SRV PILOT SOLENOID VALVE	8760	H	1.96E-7	H
ASVF101N	FAILURE IN DIV 1 ADS SRV PILOT SOLENOID VALVE	8760	H	1.96E-7	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ASVF101R	FAILURE IN DIV 1 ADS SRV PILOT SOLENOID VALVE	8760	H	1.96E-7	H
ASVF101T	FAILURE IN DIV 1 ADS SRV PILOT SOLENOID VALVE	8760	H	1.96E-7	H
ASVF102A	FAILURE IN DIV 2 ADS SRV PILOT SOLENOID VALVE	8760	H	1.96E-7	H
ASVF102C	FAILURE IN DIV 2 ADS SRV PILOT SOLENOID VALVE	8760	H	1.96E-7	H
ASVF102F	FAILURE IN DIV 2 ADS SRV PILOT SOLENOID VALVE	8760	H	1.96E-7	H
ASVF102H	FAILURE IN DIV 2 ADS SRV PILOT SOLENOID VALVE	8760	H	1.96E-7	H
ASVF102L	FAILURE IN DIV 2 ADS SRV PILOT SOLENOID VALVE	8760	H	1.96E-7	H
ASVF102N	FAILURE IN DIV 2 ADS SRV PILOT SOLENOID VALVE	8760	H	1.96E-7	H
ASVF102R	FAILURE IN DIV 2 ADS SRV PILOT SOLENOID VALVE	8760	H	1.96E-7	H
ASVF102T	FAILURE IN DIV 2 ADS SRV PILOT SOLENOID VALVE	8760	H	1.96E-7	H
AVF002AF	LOCKED OPEN MANUAL VALVE P54-F002A FAILS CLOSED	360	H	1.76E-8	H
AVF002BF	LOCKED OPEN MANUAL VALVE P54-F002B FAILS CLOSED	360	H	1.76E-8	H
AVF005AD	NORMALLY OPEN PCV P54-F005A FAILS CLOSED	360	H	1.20E-6	H
AVF005BD	NORMALLY OPEN PCV P54-F005B FAILS CLOSED	360	H	1.20E-6	H
BF003AFC	TESTABLE CHECK VALVE B21-F003A FAILS TO OPEN	1095	H	5.87E-8	H
BF005AFC	MANUAL VALVE B21-F005A FAILS CLOSED (NOFC)	1095	H	1.76E-8	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
BFUSEA	VALVE A BOTTOM FUSE FAILS	8784	H	5.18E-8	H
BFUSEB	VALVE B BOTTOM FUSE FAILS	8784	H	5.18E-8	H
BS11AF	BACKUP SCRAM DIV 1 SERIES RELAY A FAILS	8784	H	6.71E-9	H
BS11GF	BACKUP SCRAM DIV 1 SERIES RELAY G FAILS	8784	H	6.71E-9	H
BS12BF	BACKUP SCRAM DIV 2 SERIES RELAY B FAILS	8784	H	6.71E-9	H
BS12EF	BACKUP SCRAM DIV 2 SERIES RELAY E FAILS	8784	H	6.71E-9	H
BS13CF	BACKUP SCRAM DIV 3 SERIES RELAY C FAILS	8784	H	6.71E-9	H
BS13HF	BACKUP SCRAM DIV 3 SERIES RELAY H FAILS	8784	H	6.71E-9	H
BS14DF	BACKUP SCRAM DIV 4 SERIES RELAY D FAILS	8784	H	6.71E-9	H
BS14FF	BACKUP SCRAM DIV 4 SERIES RELAY F FAILS	8784	H	6.71E-9	H
BXV005B	MANUAL VALVE B21-F005B FAILS CLOSED (NOFC)	1095	H	1.76E-8	H
CMUX0AH	RC&IS CHANNEL A FAILURE	332	H	1.07E-6	H
CMUX0BH	RC&IS CHANNEL B FAILURE	332	H	1.07E-6	H
DIV1MUX	DIVISION 1 TRANSMISSION NETWORK FAILURE (EMS)	59	H	1.07E-6	H
DIV2MUX	DIVISION 2 TRANSMISSION NETWORK FAILURE (EMS)	59	H	1.07E-6	H
DIV3MUX	DIVISION 3 TRANSMISSION NETWORK FAILURE (EMS)	59	H	1.07E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
DIV4MUX	DIVISION 4 TRANSMISSION NETWORK FAILURE (EMS)	59	H	1.07E-6	H
DLS001A	VALVE E11-F001A LIMIT SWITCH FAILS	1095	H	6.11E-6	H
DLS001B	VALVE E11-F001B LIMIT SWITCH FAILS	1095	H	6.11E-6	H
DLS001C	VALVE E11-F001C LIMIT SWITCH FAILS	1095	H	6.11E-6	H
DLS005A	VALVE E11-F005A LIMIT SWITCH FAILS	1095	H	6.11E-6	H
DLS005B	VALVE E11-F005B LIMIT SWITCH FAILS	1095	H	6.11E-6	H
DLS005C	VALVE E11-F005C LIMIT SWITCH FAILS	1095	H	6.11E-6	H
DLS008A	VALVE E11-F008A LIMIT SWITCH FAILS	1095	H	6.11E-6	H
DLS008B	VALVE E11-F008B LIMIT SWITCH FAILS	1095	H	6.11E-6	H
DLS008C	VALVE E11-F008C LIMIT SWITCH FAILS	1095	H	6.11E-6	H
DLS012A	VALVE E11-F012A LIMIT SWITCH FAILS	1095	H	6.11E-6	H
DLS012B	VALVE E11-F012B LIMIT SWITCH FAILS	1095	H	6.11E-6	H
DLS012C	VALVE E11-F012C LIMIT SWITCH FAILS	1095	H	6.11E-6	H
DLS018B	VALVE E11-F018B LIMIT SWITCH FAILS	1095	H	6.11E-6	H
DLS018C	VALVE E11-F018C LIMIT SWITCH FAILS	1095	H	6.11E-6	H
DLS019B	VALVE E11-F019B LIMIT SWITCH FAILS	1095	H	6.11E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
DLS019C	VALVE E11-F019C LIMIT SWITCH FAILS	1095	H	6.11E-6	H
DTM1F	DIGITAL TRIP MODULE DIVISION 1 FAILS	59	H	1.07E-6	H
DTM2F	DIGITAL TRIP MODULE DIVISION 2 FAILS	59	H	1.07E-6	H
DTM3F	DIGITAL TRIP MODULE DIVISION 3 FAILS	59	H	1.07E-6	H
DTM4F	DIGITAL TRIP MODULE DIVISION 4 FAILS	59	H	1.07E-6	H
EAC69CH	DIV 1 SWGR FAILURE	24	H	2.23E-7	H
EAC69DH	DIV 2 SWGR FAILURE	24	H	2.23E-7	H
EAC69EH	DIV 3 SWGR FAILURE	24	H	2.23E-7	H
EACL11H	DISTRIBUTION BUS FAILURE	24	H	2.23E-7	H
EACL12H	DISTRIBUTION BUS FAILURE	24	H	2.23E-7	H
EACL13H	DISTRIBUTION BUS FAILURE	24	H	2.23E-7	H
EACL14H	DISTRIBUTION BUS FAILURE	24	H	2.23E-7	H
EATSF1D	STATIC SWITCH FAILS TO TRANSFER	8784	H	5.70E-7	H
EATSF1H	STATIC TRANSFER SWITCH FAILURE	48	H	5.70E-7	H
EATSF2D	STATIC SWITCH FAILS TO TRANSFER	8784	H	5.70E-7	H
EATSF2H	STATIC TRANSFER SWITCH FAILURE	48	H	5.70E-7	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
EATSF3D	STATIC SWITCH FAILS TO TRANSFER	8784	H	5.70E-7	H
EATSF3H	STATIC TRANSFER SWITCH FAILURE	48	H	5.70E-7	H
EATSF4D	STATIC SWITCH FAILS TO TRANSFER	8784	H	5.70E-7	H
EATSF4H	STATIC TRANSFER SWITCH FAILURE	48	H	5.70E-7	H
EBC404H	480V FEEDER BREAKER 404 OPEN	7.6	H	2.88E-7	H
EBC406H	480V FEEDER BREAKER 406 OPEN	7.6	H	2.88E-7	H
EBC426H	480V FEEDER BREAKER 426 OPEN	7.6	H	2.88E-7	H
EBC434H	480V FEEDER BREAKER 434 OPEN	7.6	H	2.88E-7	H
EBCN11H	CHARGER FAILURE	55.9	H	2.43E-6	H
EBCN12H	CHARGER FAILURE	55.9	H	2.43E-6	H
EBCN13H	CHARGER FAILURE	55.9	H	2.43E-6	H
EBCN14H	CHARGER FAILURE	55.9	H	2.43E-6	H
EBCS12H	STANDBY CHARGER FAILURE	55.9	H	2.43E-6	H
EBCS34H	STANDBY CHARGER FAILURE	55.9	H	2.43E-6	H
EBS2KA	27kV BUS IPB 2KA FAILS	24	H	2.23E-7	H
EBS6C1H	SWITCHGEAR P/C 6C-1 FAILURE	24	H	2.23E-7	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
EBS6C2H	BUS P/C 6C-2 FAILURE	24	H	2.23E-7	H
EBS6D1H	SWITCHGEAR P/C 6D-1 FAILURE	24	H	2.23E-7	H
EBS6D2H	BUS P/C 6D-2 FAILURE	24	H	2.23E-7	H
EBS6E1H	SWITCHGEAR P/C 6E-1 FAILURE	24	H	2.23E-7	H
EBS6E2H	BUS P/C 6E-2 FAILURE	24	H	2.23E-7	H
EBSA4	6.9kV BUS M/C A4 FAILS	24	H	2.23E-7	H
EBSB4	6.9kV BUS M/C B4 FAILS	24	H	2.23E-7	H
EBSC4	6.9kV BUS M/C C4 FAILS	24	H	2.23E-7	H
EBSCTGC	CTG BUS FAILURE	24	H	2.23E-7	H
EBSCTGD	CTG BUS FAILURE	24	H	2.23E-7	H
EBSCTGE	CTG BUS FAILURE	24	H	2.23E-7	H
EBSE1H	ESF DIV 1 480V MCC BUS FAILURE	24	H	2.23E-7	H
EBSE2H	ESF DIV 2 480V MCC BUS FAILURE	24	H	2.23E-7	H
EBSE3H	ESF DIV 3 480V MCC BUS FAILURE	24	H	2.23E-7	H
EBSUATA	UAT NON-SEGREGATED BUS FAILURE	24	H	2.23E-7	H
EBSUATB	UAT NON-SEGREGATED BUS FAILURE	24	H	2.23E-7	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
EBSUATC	UAT NON-SEGREGATED BUS FAILURE	24	H	2.23E-7	H
EBSXX1H	ESF DIV 1 480V MCC BUS FAILURE	24	H	2.23E-7	H
EBSXX2H	ESF DIV 2 480V MCC BUS FAILURE	24	H	2.23E-7	H
EBSXX3H	ESF DIV 3 480V MCC BUS FAILURE	24	H	2.23E-7	H
EBSXX4H	ESF DIV 1 480V MCC BUS FAILURE	24	H	2.23E-7	H
EBSXX5H	ESF DIV 2 480V MCC BUS FAILURE	24	H	2.23E-7	H
EBSXX6H	ESF DIV 3 480V MCC BUS FAILURE	24	H	2.23E-7	H
EBY101H	BATTERY FAILURE	284	H	1.86E-6	H
EBY102H	BATTERY FAILURE	284	H	1.86E-6	H
EBY103H	BATTERY FAILURE	284	H	1.86E-6	H
EBY104H	BATTERY FAILURE	284	H	1.86E-6	H
ECA002H	NORMAL PREFERRED CABLE 2,4 OR 10 FAILURE	24	H	4.84E-6	H
ECA011H	CABLE 7,11 OR 12 FAILURE	24	H	4.84E-6	H
ECA021H	CABLE 21 FAILURE	24	H	4.84E-6	H
ECA022H	CABLE 22 FAILURE	24	H	4.84E-6	H
ECA023H	CABLE 23 FAILURE	24	H	4.84E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ECA024H	CABLE 24 FAILURE	24	H	4.84E-6	H
ECA025H	CABLE 25 FAILURE	24	H	4.84E-6	H
ECA026H	CABLE 26 FAILURE	24	H	4.84E-6	H
ECA040H	CABLE 40 FAILURE	24	H	4.84E-6	H
ECA041H	CABLE 41 FAILURE	24	H	4.84E-6	H
ECA042H	CABLE 42 FAILURE	24	H	4.84E-6	H
ECA043H	480V INPUT CABLE FAILURE	24	H	4.84E-6	H
ECA044H	480V FEEDER CABLE 44 FAILURE	24	H	4.84E-6	H
ECA045H	480V INPUT CABLE FAILURE	24	H	4.84E-6	H
ECA046H	480V FEEDER CABLE 46 FAILURE	24	H	4.84E-6	H
ECA047H	480V INPUT CABLE FAILURE	24	H	4.84E-6	H
ECA048H	480V FEEDER CABLE 48 FAILURE	24	H	4.84E-6	H
ECA050H	480V INPUT CABLE FAILURE	24	H	4.84E-6	H
ECA051H	480V FEEDER CABLE 51 FAILURE	24	H	4.84E-6	H
ECA054H	CABLE 54 FAILURE	24	H	4.84E-6	H
ECA055H	480V CABLE 55 FAILURE	24	H	4.84E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ECA056H	CABLE 56 FAILURE	24	H	4.84E-6	H
ECA057H	480V CABLE 57 FAILURE	24	H	4.84E-6	H
ECA058H	CABLE 58 FAILURE	24	H	4.84E-6	H
ECA059H	480V CABLE 59 FAILURE	24	H	4.84E-6	H
ECA060H	CABLE 60 FAILURE	24	H	4.84E-6	H
ECA061H	480V CABLE 61 FAILURE	24	H	4.84E-6	H
ECA062H	125V DC CABLE 62 FAILURE	24	H	4.84E-6	H
ECA063H	125V DC CABLE 63 FAILURE	24	H	4.84E-6	H
ECA064H	125V DC CABLE 64 FAILURE	24	H	4.84E-6	H
ECA065H	125V DC CABLE 65 FAILURE	24	H	4.84E-6	H
ECA1B1H	BATTERY CABLE FAILURE	24	H	4.84E-6	H
ECA1B2H	BATTERY CABLE FAILURE	24	H	4.84E-6	H
ECA1B3H	BATTERY CABLE FAILURE	24	H	4.84E-6	H
ECA1B4H	BATTERY CABLE FAILURE	24	H	4.84E-6	H
ECACTGH	CTG CABLE FAILURE	24	H	4.84E-6	H
ECADGCH	DG C CABLE FAILURE	24	H	4.84E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ECADGDH	DG D CABLE FAILURE	24	H	4.84E-6	H
ECADGEH	DG E CABLE FAILURE	24	H	4.84E-6	H
ECAN11H	CHARGER OUTPUT CABLE FAILURE	24	H	4.84E-6	H
ECAN12H	CHARGER OUTPUT CABLE FAILURE	24	H	4.84E-6	H
ECAN13H	CHARGER OUTPUT CABLE FAILURE	24	H	4.84E-6	H
ECAN14H	CHARGER OUTPUT CABLE FAILURE	24	H	4.84E-6	H
ECAS1AH	CHARGER OUTPUT CABLE FAILURE	24	H	4.84E-6	H
ECAS1BH	CHARGER OUTPUT CABLE FAILURE	24	H	4.84E-6	H
ECAS1CH	CHARGER OUTPUT CABLE FAILURE	24	H	4.84E-6	H
ECAS1DH	CHARGER OUTPUT CABLE FAILURE	24	H	4.84E-6	H
ECAXX1H	CABLE FAILURE	24	H	4.84E-6	H
ECAXX2H	CABLE FAILURE	24	H	4.84E-6	H
ECAXX3H	CABLE FAILURE	24	H	4.84E-6	H
ECAXX4H	CABLE FAILURE	24	H	4.84E-6	H
ECAXX5H	CABLE FAILURE	24	H	4.84E-6	H
ECAXX6H	CABLE FAILURE	24	H	4.84E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ECB02A1	J002A1 CIRCUIT BREAKER FAILURE	7.6	H	2.88E-7	H
ECB02B1	J002B1 CIRCUIT BREAKER FAILURE	7.6	H	2.88E-7	H
ECB02C1	J002C1 CIRCUIT BREAKER FAILURE	7.6	H	2.88E-7	H
ECB02D1	J002D1 CIRCUIT BREAKER FAILURE	7.6	H	2.88E-7	H
ECB0A1H	OUTPUT CIRCUIT BREAKER OPEN	7.6	H	2.88E-7	H
ECB0B1H	OUTPUT CIRCUIT BREAKER OPEN	7.6	H	2.88E-7	H
ECB0C1H	OUTPUT CIRCUIT BREAKER OPEN	7.6	H	2.88E-7	H
ECB0D1H	OUTPUT CIRCUIT BREAKER OPEN	7.6	H	2.88E-7	H
ECB1B1H	BATTERY OUTPUT BREAKER OPEN	7.6	H	2.88E-7	H
ECB1B2H	BATTERY OUTPUT BREAKER OPEN	7.6	H	2.88E-7	H
ECB1B3H	BATTERY OUTPUT BREAKER OPEN	7.6	H	2.88E-7	H
ECB1B4H	BATTERY OUTPUT BREAKER OPEN	7.6	H	2.88E-7	H
ECB1DCH	DIV 1 DC POWER BKR 1 FAILS OPEN	7.6	H	2.88E-7	H
ECB225AF	MCC B 225AF CIRCUIT BREAKER FAILURE	7.6	H	2.88E-7	H
ECB225BF	MCC B 225BF CIRCUIT BREAKER FAILURE	7.6	H	2.88E-7	H
ECB225CF	MCC B 225CF CIRCUIT BREAKER FAILURE	7.6	H	2.88E-7	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ECB225DF	MCC B 225DF CIRCUIT BREAKER FAILURE	7.6	H	2.88E-7	H
ECB2DCH	DIV 2 DC POWER BKR 2 FAILS OPEN	7.6	H	2.88E-7	H
ECB301H	P/C 6C-1 INCOMING BKR 301 OPEN	7.6	H	2.88E-7	H
ECB312H	SWITCHGEAR P/C 6C-1 FEED BREAKER 312 OPEN	7.6	H	2.88E-7	H
ECB331H	P/C 6C-2 INCOMING BREAKER 331 OPEN	7.6	H	2.88E-7	H
ECB341H	P/C 6D-1 INCOMING BKR 341 OPEN	7.6	H	2.88E-7	H
ECB354H	SWITCHGEAR P/C 6C-1 FEED BREAKER 354 OPEN	7.6	H	2.88E-7	H
ECB361H	P/C 6D-2 INCOMING BREAKER 361 OPEN	7.6	H	2.88E-7	H
ECB371H	P/C 6E-2 INCOMING BKR 371 OPEN	7.6	H	2.88E-7	H
ECB380H	SWITCHGEAR P/C 6E-1 FEED BREAKER 380 OPEN	7.6	H	2.88E-7	H
ECB391H	P/C 6E-2 INCOMING BREAKER 391 OPEN	7.6	H	2.88E-7	H
ECB3DCH	DIV 3 DC POWER BKR 3 FAILS OPEN	7.6	H	2.88E-7	H
ECB403H	480V FEEDER BREAKER 403 OPEN	7.6	H	2.88E-7	H
ECB405H	480V FEEDER BREAKER 405 OPEN	7.6	H	2.88E-7	H
ECB407H	DIV 1 480V MCC BREAKER 407 OPEN	7.6	H	2.88E-7	H
ECB408H	DIV 1 480V MCC BREAKER 408 OPEN	7.6	H	2.88E-7	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ECB409H	DIV 2 480V MCC BREAKER 409 OPEN	7.6	H	2.88E-7	H
ECB410H	DIV 1 480V MCC INCOMING BREAKER 410 OPEN	7.6	H	2.88E-7	H
ECB411H	DIV 2 480V MCC BREAKER 411 OPEN	7.6	H	2.88E-7	H
ECB420H	DIV 2 480V MCC INCOMING BREAKER 420 OPEN	7.6	H	2.88E-7	H
ECB423H	DIV 2 480V MCC BREAKER 423 OPEN	7.6	H	2.88E-7	H
ECB425H	480V FEEDER BREAKER 425 OPEN	7.6	H	2.88E-7	H
ECB427H	DIV 2 480V MCC BREAKER 427 OPEN	7.6	H	2.88E-7	H
ECB430H	DIV 3 480V MCC INCOMING BREAKER 430 OPEN	7.6	H	2.88E-7	H
ECB433H	480V FEEDER BREAKER 433 OPEN	7.6	H	2.88E-7	H
ECB435H	DIV 3 480V MCC BREAKER 435 OPEN	7.6	H	2.88E-7	H
ECB436H	DIV 3 480V MCC BREAKER 436 OPEN	7.6	H	2.88E-7	H
ECB901H	INCOMING BKR TO M/C E OPEN	7.6	H	2.88E-7	H
ECB904H	6.9 kV BREAKER 904 OPEN	7.6	H	2.88E-7	H
ECB911H	6.9 kV BREAKER 911 OPEN	7.6	H	2.88E-7	H
ECB921H	INCOMING BKR TO M/C F OPEN	7.6	H	2.88E-7	H
ECB924H	6.9 kV BREAKER 924 OPEN	7.6	H	2.88E-7	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ECB931H	6.9 kV BREAKER 931 OPEN	7.6	H	2.88E-7	H
ECB941H	INCOMING BKR TO M/C G OPEN	7.6	H	2.88E-7	H
ECB944H	6.9 kV BREAKER 944 OPEN	7.6	H	2.88E-7	H
ECB951H	6.9 kV BREAKER 951 OPEN	7.6	H	2.88E-7	H
ECBA4	OUTGOING BREAKER FROM M/C A4 OPEN	7.6	H	2.88E-7	H
ECBAC1H	UPS AC INPUT BREAKER OPEN	7.6	H	2.88E-7	H
ECBAC2H	UPS AC INPUT BREAKER OPEN	7.6	H	2.88E-7	H
ECBAC3H	UPS AC INPUT BREAKER OPEN	7.6	H	2.88E-7	H
ECBAC4H	UPS AC INPUT BREAKER OPEN	7.6	H	2.88E-7	H
ECBB4	OUTGOING BREAKER FROM M/C B4 OPEN	7.6	H	2.88E-7	H
ECBBU1H	UPS BACKUP AC INPUT BREAKER OPEN	7.6	H	2.88E-7	H
ECBBU2H	UPS BACKUP AC INPUT BREAKER OPEN	7.6	H	2.88E-7	H
ECBBU3H	UPS BACKUP AC INPUT BREAKER OPEN	7.6	H	2.88E-7	H
ECBBU4H	UPS BACKUP AC INPUT BREAKER OPEN	7.6	H	2.88E-7	H
ECBC4	OUTGOING BREAKER FROM M/C C4 OPEN	7.6	H	2.88E-7	H
ECBD11H	DIV 1 125V DC BREAKER D11 OPEN	7.6	H	2.88E-7	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ECBD12H	125V DC DISTRIBUTION PANEL BKR D12 OPEN	7.6	H	2.88E-7	H
ECBD21H	DIV 2 125V DC BREAKER D21 OPEN	7.6	H	2.88E-7	H
ECBD22H	125V DC DISTRIBUTION PANEL BKR D22 OPEN	7.6	H	2.88E-7	H
ECBD31H	DIV 3 125V DC BREAKER D31 OPEN	7.6	H	2.88E-7	H
ECBD32H	125V DC DISTRIBUTION PANEL BKR D32 OPEN	7.6	H	2.88E-7	H
ECBD41H	DIV 4 125V DC BREAKER D41 OPEN	7.6	H	2.88E-7	H
ECBDC1H	UPS DC INPUT BREAKER OPEN	7.6	H	2.88E-7	H
ECBDC2H	UPS DC INPUT BREAKER OPEN	7.6	H	2.88E-7	H
ECBDC3H	UPS DC INPUT BREAKER OPEN	7.6	H	2.88E-7	H
ECBDC4H	UPS DC INPUT BREAKER OPEN	7.6	H	2.88E-7	H
ECBX11H	DIV 1 480V MCC INCOMING BREAKER OPEN	7.6	H	2.88E-7	H
ECBX12H	DIV 2 480V MCC INCOMING BREAKER OPEN	7.6	H	2.88E-7	H
ECBX13H	DIV 3 480V MCC INCOMING BREAKER OPEN	7.6	H	2.88E-7	H
ECBX14H	DIV 1 480V MCC INCOMING BREAKER OPEN	7.6	H	2.88E-7	H
ECBX15H	DIV 2 480V MCC INCOMING BREAKER OPEN	7.6	H	2.88E-7	H
ECBX16H	DIV 3 480V MCC INCOMING BREAKER OPEN	7.6	H	2.88E-7	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ECBXX1H	SWITCHGEAR P/C 6C-2 FEED BREAKER OPEN	7.6	H	2.88E-7	H
ECBXX2H	SWITCHGEAR P/C 6D-2 FEED BREAKER OPEN	7.6	H	2.88E-7	H
ECBXX3H	SWITCHGEAR P/C 6E-2 FEED BREAKER OPEN	7.6	H	2.88E-7	H
ECBXX4H	SWGR P/C 6C-1 FEEDER BREAKER OPEN	7.6	H	2.88E-7	H
ECBXX5H	SWGR P/C 6D-1 FEEDER BREAKER OPEN	7.6	H	2.88E-7	H
ECBXX6H	SWGR P/C 6E-1 FEEDER BREAKER OPEN	7.6	H	2.88E-7	H
ECL902H	BKR CONTROL LOGIC FAILURE	365	H	1.07E-6	H
ECL922H	BKR CONTROL LOGIC FAILURE	365	H	1.07E-6	H
ECL942H	BKR CONTROL LOGIC FAILURE	365	H	1.07E-6	H
EDGC	DG C FAILS TO RUN	23	H	1.09E-3	H
EDGD	DG D FAILS TO RUN	23	H	1.09E-3	H
EDGE	DG E FAILS TO RUN	23	H	1.09E-3	H
EDP101H	FAILURE OF DIV 1 DISTRIBUTION PANEL	24	H	2.23E-7	H
EDP102H	FAILURE OF DIV 2 DISTRIBUTION PANEL	24	H	2.23E-7	H
EDP103H	FAILURE OF DIV 3 DISTRIBUTION PANEL	24	H	2.23E-7	H
EDP104H	FAILURE OF DIV 4 DISTRIBUTION PANEL	24	H	2.23E-7	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
EIVOF1H	INVERTER FAILURE	132	H	1.11E-5	H
EIVOF2H	INVERTER FAILURE	132	H	1.11E-5	H
EIVOF3H	INVERTER FAILURE	132	H	1.11E-5	H
EIVOF4H	INVERTER FAILURE	132	H	1.11E-5	H
ELCDGCH	DG C CONTROL FAILURE	365	H	1.07E-6	H
ELCDGDH	DG D CONTROL FAILURE	365	H	1.07E-6	H
ELCDGEH	DG E CONTROL FAILURE	365	H	1.07E-6	H
ELNK1F	EMS/DTM LINK FOR DIVISION 1 FAILS	59	H	1.07E-6	H
ELNK2F	EMS/DTM LINK FOR DIVISION 2 FAILS	59	H	1.07E-6	H
ELNK3F	EMS/DTM LINK FOR DIVISION 3 FAILS	59	H	1.07E-6	H
ELNK4F	EMS/DTM LINK FOR DIVISION 4 FAILS	59	H	1.07E-6	H
ETR6C1H	TRANSFORMER T6C1 FAILS	60	H	1.09E-6	H
ETR6C2H	TRANSFORMER T6C2 FAILURE	60	H	1.09E-6	H
ETR6D1H	TRANSFORMER T6D1 FAILS	60	H	1.09E-6	H
ETR6D2H	TRANSFORMER T6D2 FAILURE	60	H	1.09E-6	H
ETR6E1H	TRANSFORMER T6E1 FAILS	60	H	1.09E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ETR6E2H	TRANSFORMER T6E2 FAILURE	60	H	1.09E-6	H
ETRSU1H	MAIN POWER TRANSFORMER FAILS	60	H	1.09E-6	H
ETRSU2H	RESERVE AUXILIARY TRANSFORMER FAILURE	60	H	1.09E-6	H
ETRU11H	BACKUP TRANSFORMER FAILURE	60	H	1.09E-6	H
ETRU12H	BACKUP TRANSFORMER FAILURE	60	H	1.09E-6	H
ETRU13H	BACKUP TRANSFORMER FAILURE	60	H	1.09E-6	H
ETRU14H	BACKUP TRANSFORMER FAILURE	60	H	1.09E-6	H
ETRUATA	UNIT AUXILIARY TRANSFORMER A FAILURE	60	H	1.09E-6	H
ETRUATB	UNIT AUXILIARY TRANSFORMER B FAILURE	60	H	1.09E-6	H
ETRUATC	UNIT AUXILIARY TRANSFORMER C FAILURE	60	H	1.09E-6	H
F001AFC	PUMP SUCTION VALVE E11-F001A FAILS CLOSED (NOFC)	1095	H	1.40E-7	H
F001AFO	PUMP SUCTION VALVE E11-F001A FAILS OPEN (NOFO)	1095	H	1.90E-6	H
F001BFC	PUMP SUCTION VALVE E11-F001B FAILS CLOSED (NOFC)	1095	H	1.40E-7	H
F001BFO	PUMP SUCTION VALVE E11-F001B FAILS OPEN (NOFO)	1095	H	1.90E-6	H
F001CFC	PUMP SUCTION VALVE E11-F001C FAILS CLOSED (NOFC)	1095	H	1.40E-7	H
F001CFO	PUMP SUCTION VALVE E11-F001C FAILS OPEN (NOFO)	1095	H	1.90E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
F003AFC	MANUAL VALVE E11-F003A FAILS CLOSED	1095	H	1.76E-8	H
F003BFC	MANUAL VALVE E11-F003B FAILS CLOSED	1095	H	1.76E-8	H
F003CFC	MANUAL VALVE E11-F003C FAILS CLOSED	1095	H	1.76E-8	H
F004AFC	VALVE E11-F004A FAILS CLOSED (NOFC)	1095	H	1.40E-7	H
F004BFC	VALVE E11-F004B FAILS CLOSED (NOFC)	1095	H	1.40E-7	H
F004CFC	VALVE E11-F004C FAILS CLOSED (NOFC)	1095	H	1.40E-7	H
F005AFO	INJECTION VALVE E11-F005A FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F005AMF	MECHANICAL FAILURE OF INJECTION VALVE E11-F005A (NCFC)	1095	H	1.90E-6	H
F005BFO	INJECTION VALVE E11-F005B FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F005BMF	MECHANICAL FAILURE OF INJECTION VALVE E11-F005B (NCFC)	1095	H	1.90E-6	H
F005CFO	INJECTION VALVE E11-F005C FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F005CMF	MECHANICAL FAILURE OF INJECTION VALVE E11-F005C (NCFC)	1095	H	1.90E-6	H
F006AMF	MECHANICAL FAILURE OF TESTABLE CHECK VALVE E11-F006A	1095	H	5.87E-8	H
F006BMF	MECHANICAL FAILURE OF TESTABLE CHECK VALVE E11-F006B	1095	N	5.87E-8	N
F006CMF	MECHANICAL FAILURE OF TESTABLE CHECK VALVE E11-F006C	1095	N	5.87E-8	N
F008A	SUPP POOL DISCH. VALVE E11-F008A FAILS CLOSED (NCFC)	1095	H	1.90E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
F008AFC	VALVE E11-F008A FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F008B	SUPP POOL DISCH. VALVE E11-F008B FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
F008BFC	VALVE E11-F008B FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F008C	SUPP POOL DISCH. VALVE E11-F008C FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
F008CFC	VALVE E11-F008C FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F009AFC	MANUAL VALVE E11-F009A FAILS TO REMAIN OPEN	1095	H	1.76E-8	H
F009BFC	MANUAL VALVE E11-F009B FAILS TO REMAIN OPEN	1095	H	1.76E-8	H
F009CFC	MANUAL VALVE E11-F009C FAILS TO REMAIN OPEN	1095	H	1.76E-8	H
F010AFC	VALVE E11-F010A FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
F010BFC	VALVE E11-F010B FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
F010CFC	VALVE E11-F010C FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
F011AFC	VALVE E11-F011A FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
F011BFC	VALVE E11-F011B FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
F011CFC	VALVE E11-F011C FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
F012AFC	RPV SUCTION LINE ISOLATION VALVE E11-F012A FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
F012AFO	RPV SUCTION LINE ISOLATION VALVE E11-F012A FAILS OPEN (NCFO)	1095	H	1.40E-7	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
F012BFC	RPV SUCTION LINE ISOLATION VALVE E11-F012B FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
F012BFO	RPV SUCTION LINE ISOLATION VALVE E11-F012B FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F012CFC	RPV SUCTION LINE ISOLATION VALVE E11-F012C FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
F012CFO	RPV SUCTION LINE ISOLATION VALVE E11-F012C FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F013AFO	HX TUBE SIDE BYPASS VALVE E11-F013A FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F013BFO	HX TUBE SIDE BYPASS VALVE E11-F013B FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F013CFO	HX TUBE SIDE BYPASS VALVE E11-F013C FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F014BFO	FUEL POOL VALVE E11-F014B FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F014CFO	FUEL POOL VALVE E11-F014C FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F015BFO	FUEL POOL VALVE E11-F015B FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F015CFO	FUEL POOL VALVE E11-F015C FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F017BFC	DW SPRAY VALVE E11-F017B FAILS CLOSED	1095	H	1.90E-6	H
F017BFO	DW SPRAY VALVE E11-F017B FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F017CFC	DW SPRAY VALVE E11-F017C FAILS CLOSED	1095	H	1.90E-6	H
F017CFO	DW SPRAY VALVE E11-F017C FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F018BFC	DW SPRAY VALVE E11-F018B FAILS CLOSED	1095	H	1.90E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
F018BFO	DW SPRAY VALVE E11-F018B FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F018CFC	DW SPRAY VALVE E11-F018C FAILS CLOSED	1095	H	1.90E-6	H
F018CFO	DW SPRAY VALVE E11-F018C FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F019BFC	DW SPRAY VALVE E11-F019B FAILS CLOSED	1095	H	1.90E-6	H
F019BFO	WW SPRAY VALVE E11-F019B FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F019CFC	DW SPRAY VALVE E11-F019C FAILS CLOSED	1095	H	1.90E-6	H
F019CFO	WW SPRAY VALVE E11-F019C FAILS OPEN (NCFO)	1095	H	1.40E-7	H
F021AFO	MINIMUM FLOW VALVE E11-F021A FAILS OPEN (NOFO)	1095	H	1.90E-6	H
F021BFO	MINIMUM FLOW VALVE E11-F021B FAILS OPEN (NOFO)	1095	H	1.90E-6	H
F021CFO	MINIMUM FLOW VALVE E11-F021C FAILS OPEN (NOFO)	1095	H	1.90E-6	H
GTURBINEFR	CTG FAILS TO RUN	23	H	8.48E-4	H
HCOI12	OPTICAL ISOLATOR DIV 1 TO 2 FAILS	59	H	1.15E-6	H
HCOI13	OPTICAL ISOLATOR DIV 1 TO 3 FAILS	59	H	1.15E-6	H
HCOI14	OPTICAL ISOLATOR DIV 1 TO 4 FAILS	59	H	1.15E-6	H
HCOI21	OPTICAL ISOLATOR DIV 2 TO 1 FAILS	59	H	1.15E-6	H
HCOI23	OPTICAL ISOLATOR DIV 2 TO 3 FAILS	59	H	1.15E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
HCOI24	OPTICAL ISOLATOR DIV 2 TO 4 FAILS	59	H	1.15E-6	H
HCOI31	OPTICAL ISOLATOR DIV 3 TO 1 FAILS	59	H	1.15E-6	H
HCOI32	OPTICAL ISOLATOR DIV 3 TO 2 FAILS	59	H	1.15E-6	H
HCOI34	OPTICAL ISOLATOR DIV 3 TO 4 FAILS	59	H	1.15E-6	H
HCOI41	OPTICAL ISOLATOR DIV 4 TO 1 FAILS	59	H	1.15E-6	H
HCOI42	OPTICAL ISOLATOR DIV 4 TO 2 FAILS	59	H	1.15E-6	H
HCOI43	OPTICAL ISOLATOR DIV 4 TO 3 FAILS	59	H	1.15E-6	H
HCV02BHP	CHECK VALVE E22-F002B FAILS CLOSED	1095	H	5.87E-8	H
HCV02CHP	CHECK VALVE E22-F002C FAILS CLOSED	1095	H	5.87E-8	H
HCV04BHP	TESTABLE CHECK VALVE E22-F004B FAILS TO OPEN	1095	H	5.87E-8	H
HCV04CHP	TESTABLE CHECK VALVE E22-F004C FAILS TO OPEN	1095	H	5.87E-8	H
HCV21BHP	PUMP DISCHARGE CHECK VALVE E22-F021B FAILS TO OPEN	1095	H	5.87E-8	H
HCV21CHP	PUMP DISCHARGE CHECK VALVE E22-F021C FAILS TO OPEN	1095	H	5.87E-8	H
HFE008BH	FLOW XMITTER E22-FT008B-2 FAILS LOW	1095	H	1.38E-7	H
HFE008CH	FLOW XMITTER E22-FT008C-2 FAILS LOW	1095	H	1.38E-7	H
HFLSPBHW	SUCTION STRAINER E22-D003B PLUGGED	1095	H	5.92E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
HFLSPCHW	SUCTION STRAINER E22-D003C PLUGGED	1095	H	5.92E-6	H
HLECSTAH	CST LEVEL SENSOR P13-LS-Z601A FAILS	4	H	4.25E-7	H
HLECSTBH	CST LEVEL SENSOR P13-LS-Z601B FAILS	4	H	4.25E-7	H
HLECSTCH	CST LEVEL SENSOR P13-LS-Z601C FAILS	4	H	4.25E-7	H
HLECSTDH	CST LEVEL SENSOR P13-LS-Z601D FAILS	4	H	4.25E-7	H
HMO01BDR	HPCF PUMP B FAILS TO RUN AFTER FIRST HOUR	23	H	7.05E-6	H
HMO01BDR1	HPCF PUMP B FAILS TO RUN DURING FIRST HOUR	1	H	9.69E-6	H
HMO01CDR	HPCF PUMP C FAILS TO RUN AFTER FIRST HOUR	23	H	7.05E-6	H
HMO01CDR1	HPCF PUMP C FAILS TO RUN DURING FIRST HOUR	1	H	9.69E-6	H
HMV06BHP	PUMP SUCTION VALVE E22-F006B FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
HMV06CHP	PUMP SUCTION VALVE E22-F006C FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
HMV08BHI	TEST VALVE E22-F008B FAILS TO CLOSE	1095	H	1.90E-6	H
HMV08BHO	VALVE E22-F008B FAILS OPEN (NCFO)	1095	H	1.40E-7	H
HMV08CHI	TEST VALVE E22-F008C FAILS TO CLOSE	1095	H	1.90E-6	H
HMV08CHO	VALVE E22-F008C FAILS OPEN (NCFO)	1095	H	1.40E-7	H
HMV09BHI	TEST VALVE E22-F009B FAILS TO CLOSE	1095	H	1.90E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
HMV09BHO	VALVE E22-F009B FAILS OPEN (NCFO)	1095	H	1.40E-7	H
HMV09CHI	TEST VALVE E22-F009C FAILS TO CLOSE	1095	H	1.90E-6	H
HMV09CHO	VALVE E22-F009C FAILS OPEN (NCFO)	1095	H	1.40E-7	H
HMV10BHO	VALVE E22-F010B FAILS TO CLOSE	1095	H	1.90E-6	H
HMV10BHW	VALVE E22-F010B FAILS TO OPEN	1095	H	1.90E-6	H
HMV10CHO	VALVE E22-F010C FAILS TO CLOSE	1095	H	1.90E-6	H
HMV10CHW	VALVE E22-F010C FAILS TO OPEN	1095	H	1.90E-6	H
HMV14BHW	VALVE E22-F003B FAILS CLOSED (NCFC-VALVE BODY STUCK)	1095	H	1.90E-6	H
HMV14CHW	VALVE E22-F003C FAILS CLOSED (NCFC-VALVE BODY STUCK)	1095	H	1.90E-6	H
HPBRAC	PUMP B ROOM A.C. UNIT FAILS	24	H	9.40E-5	H
HPCRAC	PUMP C ROOM A.C. UNIT FAILS	24	H	9.40E-5	H
HPCSPARG	SPARGERS PLUGGED	1095	H	1.50E-6	H
HPR007BH	PRESSURE XMITTER E22-PT007B FAILS LOW	4	H	1.57E-6	H
HPR007CH	PRESSURE XMITTER E22-PT007C FAILS LOW	4	H	1.57E-6	H
HSS000HW	SPARGERS PLUGGED	1095	H	1.50E-6	H
HXA	HEAT EXCHANGER PLUGS DURING OPERATION	24	H	3.60E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
HXA1	HEAT EXCHANGER LEAKS DURING OPERATION	24	H	2.10E-6	H
HXB	HEAT EXCHANGER PLUGS DURING OPERATION	24	H	3.60E-6	H
HXB1	HEAT EXCHANGER LEAKS DURING OPERATION	24	H	2.10E-6	H
HXC	HEAT EXCHANGER PLUGS DURING OPERATION	24	H	3.60E-6	H
HXC1	HEAT EXCHANGER LEAKS DURING OPERATION	24	H	2.10E-6	H
HXV01BHQ	VALVE E22-F001B FAILS CLOSED (NOFC)	1095	H	1.40E-7	H
HXV01CHQ	VALVE E22-F001C FAILS CLOSED (NOFC)	1095	H	1.40E-7	H
HXV16BHQ	VALVE E22-F005B FAILS TO REMAIN OPEN	1095	H	1.76E-8	H
HXV16CHQ	VALVE E22-F005C FAILS TO REMAIN OPEN	1095	H	1.76E-8	H
IBYP1	BYPASS UNIT DIVISION 1 FAILS	370	H	1.07E-6	H
IBYP2	BYPASS UNIT DIVISION 2 FAILS	370	H	1.07E-6	H
IBYP3	BYPASS UNIT DIVISION 3 FAILS	370	H	1.07E-6	H
IDMC01H	RFC DMC A FAILURE	332	H	1.07E-6	H
IDMC02H	RFC DMC B FAILURE	332	H	1.07E-6	H
IDMC03H	RFC DMC C FAILURE	332	H	1.07E-6	H
IDTM1	DIGITAL TRIP MODULE FAILS DIVISION 1	59	H	1.07E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
IDTM2	DIGITAL TRIP MODULE FAILS DIVISION 2	59	H	1.07E-6	H
IDTM3	DIGITAL TRIP MODULE FAILS DIVISION 3	59	H	1.07E-6	H
IDTM4	DIGITAL TRIP MODULE FAILS DIVISION 4	59	H	1.07E-6	H
IIN012H	DIV 1 TO 2 DTM TO SLU TRANSMISSION FAILS	59	H	1.15E-6	H
IIN013H	DIV 1 TO 3 DTM TO SLU TRANSMISSION FAILS	59	H	1.15E-6	H
IIN021H	DIV 2 TO 1 DTM TO SLU TRANSMISSION FAILS	59	H	1.15E-6	H
IIN023H	DIV 2 TO 3 DTM TO SLU TRANSMISSION FAILS	59	H	1.15E-6	H
IIN031H	DIV 3 TO 1 DTM TO SLU TRANSMISSION FAILS	59	H	1.15E-6	H
IIN032H	DIV 3 TO 2 DTM TO SLU TRANSMISSION FAILS	59	H	1.15E-6	H
IIN041H	DIV 4 TO 1 DTM TO SLU TRANSMISSION FAILS	59	H	1.15E-6	H
IIN042H	DIV 4 TO 2 DTM TO SLU TRANSMISSION FAILS	59	H	1.15E-6	H
IIN043H	DIV 4 TO 3 DTM TO SLU TRANSMISSION FAILS	59	H	1.15E-6	H
ILC001H	DIV 1 REMOTE MULTIPLEXING UNIT FAILS	59	H	1.07E-6	H
ILC002H	DIV 2 REMOTE MULTIPLEXING UNIT FAILS	59	H	1.07E-6	H
ILC003H	DIV 3 REMOTE MULTIPLEXING UNIT FAILS	59	H	1.07E-6	H
ILC004H	DIV 4 REMOTE MULTIPLEXING UNIT FAILS	59	H	1.07E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ILE011H	RPV LEVEL SENSOR FAILURE B21-LT003A	4	H	4.25E-7	H
ILE012H	RPV LEVEL SENSOR FAILURE B21-LT003B	4	H	4.25E-7	H
ILE013H	RPV LEVEL SENSOR FAILURE B21-LT003C	4	H	4.25E-7	H
ILE014H	RPV LEVEL SENSOR FAILURE B21-LT003D	4	H	4.25E-7	H
ILE021H	RPV LEVEL SENSOR FAILURE B21-LT003E	4	H	4.25E-7	H
ILE022H	RPV LEVEL SENSOR FAILURE B21-LT003F	4	H	4.25E-7	H
ILE023H	RPV LEVEL SENSOR FAILURE B21-LT003H	4	H	4.25E-7	H
ILE024H	RPV LEVEL SENSOR FAILURE B21-LT003G	4	H	4.25E-7	H
ILE031H	LEVEL 3 SENSOR B21-LT001A FAILS	4	H	4.25E-7	H
ILE032H	LEVEL 3 SENSOR B21-LT001B FAILS	4	H	4.25E-7	H
ILE033H	LEVEL 3 SENSOR B21-LT001C FAILS	4	H	4.25E-7	H
ILINK11	SLU/EMS LINK FOR DIVISION 1 SLU 1 FAILS	4.25	H	1.07E-6	H
ILINK11H	DIV 1 TO 1 DTM TO SLU TRANSMISSION FAILS	4.25	H	1.07E-6	H
ILINK12	SLU/EMS LINK FOR DIVISION 2 SLU 1 FAILS	4.25	H	1.07E-6	H
ILINK13	SLU/EMS LINK FOR DIVISION 3 SLU 1 FAILS	4.25	H	1.07E-6	H
ILINK21	SLU/EMS LINK FOR DIVISION 1 SLU 2 FAILS	4.25	H	1.07E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ILINK22	SLU/EMS LINK FOR DIVISION 2 SLU 2 FAILS	4.25	H	1.07E-6	H
ILINK22H	DIV 2 TO 2 DTM TO SLU TRANSMISSION FAILS	4.25	H	1.07E-6	H
ILINK23	SLU/EMS LINK FOR DIVISION 3 SLU 2 FAILS	4.25	H	1.07E-6	H
ILINK33H	DIV 3 TO 3 DTM TO SLU TRANSMISSION FAILS	4.25	H	1.07E-6	H
ILINK41	EMS/DTM LINK FOR DIVISION 1 FAILS	4.25	H	1.07E-6	H
ILINK42	EMS/DTM LINK FOR DIVISION 2 FAILS	4.25	H	1.07E-6	H
ILINK43	EMS/DTM LINK FOR DIVISION 3 FAILS	4.25	H	1.07E-6	H
ILINK44	EMS/DTM LINK FOR DIVISION 4 FAILS	4.25	H	1.07E-6	H
IMUX0AH	CHANNEL A MUX FAILURE	332	H	1.07E-6	H
IMUX0BH	CHANNEL B MUX FAILURE	332	H	1.07E-6	H
IMUX0CH	CHANNEL C MUX FAILURE	332	H	1.07E-6	H
IPR001H	RPV PRESSURE SENSOR B21-PT008A FAILS	4	H	1.57E-6	H
IPR002H	RPV PRESSURE SENSOR B21-PT008B FAILS	4	H	1.57E-6	H
IPR003H	RPV PRESSURE SENSOR B21-PT008C FAILS	4	H	1.57E-6	H
IPRDW1H	DW PRESSURE SENSOR FAILURE B21-PT025A	4	H	1.57E-6	H
IPRDW2H	DW PRESSURE SENSOR FAILURE B21-PT025B	4	H	1.57E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
IPRDW3H	DW PRESSURE SENSOR FAILURE B21-PT025C	4	H	1.57E-6	H
IPRDW4H	DW PRESSURE SENSOR FAILURE B21-PT025D	4	H	1.57E-6	H
IREEIA	RELAY EI-A FAILS TO CLOSE	1	H	1.45E-5	H
IREEIB	RELAY EI-B FAILS TO CLOSE	1	H	1.45E-5	H
IREEIC	RELAY EI-B FAILS TO CLOSE	1	H	1.45E-5	H
IRMP01H	DIV 1 REMOTE MULTIPLEXING UNIT FAILS	332	H	1.07E-6	H
IRMP02H	DIV 2 REMOTE MULTIPLEXING UNIT FAILS	332	H	1.07E-6	H
IRMP03H	DIV 3 REMOTE MULTIPLEXING UNIT FAILS	332	H	1.07E-6	H
IRMU01A	DIV 1 REMOTE MULTIPLEXING UNIT FAILS	332	H	1.07E-6	H
IRMU01B	DIV 2 REMOTE MULTIPLEXING UNIT FAILS	332	H	1.07E-6	H
IRMU01C	DIV 3 REMOTE MULTIPLEXING UNIT FAILS	332	H	1.07E-6	H
IRMU11	1ST ESF REMOTE MULTIPLEXING UNIT DIV 1 FAILS	59	H	1.07E-6	H
IRMU12	1ST ESF REMOTE MULTIPLEXING UNIT DIV 2 FAILS	59	H	1.07E-6	H
IRMU13	1ST ESF REMOTE MULTIPLEXING UNIT DIV 3 FAILS	59	H	1.07E-6	H
IRMU21	2ND ESF REMOTE MULTIPLEXING UNIT DIV 1 FAILS	59	H	1.07E-6	H
IRMU22	2ND ESF REMOTE MULTIPLEXING UNIT DIV 2 FAILS	59	H	1.07E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
IRMU23	2ND ESF REMOTE MULTIPLEXING UNIT DIV 3 FAILS	59	H	1.07E-6	H
IRMUDWP1	DIV 1 REMOTE MULTIPLEXING UNIT FAILS	59	H	1.07E-6	H
IRMUDWP2	DIV 2 REMOTE MULTIPLEXING UNIT FAILS	59	H	1.07E-6	H
IRMUDWP3	DIV 3 REMOTE MULTIPLEXING UNIT FAILS	59	H	1.07E-6	H
IRMUDWP4	DIV 4 REMOTE MULTIPLEXING UNIT FAILS	59	H	1.07E-6	H
IRMULV11	DIV 1 REMOTE MULTIPLEXING UNIT FAILS	59	H	1.07E-6	H
IRMULV12	DIV 2 REMOTE MULTIPLEXING UNIT FAILS	59	H	1.07E-6	H
IRMULV13	DIV 3 REMOTE MULTIPLEXING UNIT FAILS	59	H	1.07E-6	H
IRMULV14	DIV 4 REMOTE MULTIPLEXING UNIT FAILS	59	H	1.07E-6	H
IRMULV21	DIV 1 REMOTE MULTIPLEXING UNIT FAILS	59	H	1.07E-6	H
IRMULV22	DIV 2 REMOTE MULTIPLEXING UNIT FAILS	59	H	1.07E-6	H
IRMULV23	DIV 3 REMOTE MULTIPLEXING UNIT FAILS	59	H	1.07E-6	H
IRMULV24	DIV 4 REMOTE MULTIPLEXING UNIT FAILS	59	H	1.07E-6	H
ISLU11	SLU 1 FAILS DIVISION 1	59	H	1.07E-6	H
ISLU12	SLU 1 FAILS DIVISION 2	59	H	1.07E-6	H
ISLU13	SLU 1 FAILS DIVISION 3	59	H	1.07E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
ISLU21	SLU 2 FAILS DIVISION 1	59	H	1.07E-6	H
ISLU22	SLU 2 FAILS DIVISION 2	59	H	1.07E-6	H
ISLU23	SLU 2 FAILS DIVISION 3	59	H	1.07E-6	H
LNK1F	DTM TO TLU DIV 1 LINK FAILS	59	H	1.07E-6	H
LNK2F	DTM TO TLU DIV 2 LINK FAILS	59	H	1.07E-6	H
LNK3F	DTM TO TLU DIV 3 LINK FAILS	59	H	1.07E-6	H
LNK4F	DTM TO TLU DIV 4 LINK FAILS	59	H	1.07E-6	H
OFL000HW	PLUGGED SUCTION LINE FROM TANK	1095	H	1.70E-6	H
OFTM	FLUX 3 MIN TIMER FAILS	1095	H	1.50E-6	H
OHR001HW	HEATER B001 FAILS	1095	H	8.00E-6	H
OHR002HW	HEATER B002 FAILS	1095	H	8.00E-6	H
OLS012	LIMIT SWITCH C41-LS012 FAILS	1095	H	6.11E-6	H
OLS01A	LIMIT SWITCH C41-LS001A FAILS	1095	H	6.11E-6	H
OLS01B	LIMIT SWITCH C41-LS001B FAILS	1095	H	6.11E-6	H
OLS602A	LEVEL SWITCH C41-LS602A FAILS LOW	4	H	4.25E-7	H
OLS602B	LEVEL SWITCH C41-LS602B FAILS LOW	4	H	4.25E-7	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
OLS602C	LEVEL SWITCH C41-LS602C FAILS LOW	4	H	4.25E-7	H
OLS602D	LEVEL SWITCH C41-LS602D FAILS LOW	4	H	4.25E-7	H
OLU1F	OUPUT LOGIC UNIT DIVISION 1 FAILS	370	H	1.07E-6	H
OLU2F	OUPUT LOGIC UNIT DIVISION 2 FAILS	370	H	1.07E-6	H
OLU3F	OUPUT LOGIC UNIT DIVISION 3 FAILS	370	H	1.07E-6	H
OLU4F	OUPUT LOGIC UNIT DIVISION 4 FAILS	370	H	1.07E-6	H
OMV001HW	SUCTION VALVE C41-F001A FAILS (NCFC)	1095	H	1.90E-6	H
OMV002HW	SUCTION VALVE C41-F001B FAILS (NCFC)	1095	H	1.90E-6	H
OMV003HW	MOTOR OPERATED VALVE F006A FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
OMV004HW	MOTOR OPERATED VALVE F006B FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
OMV012	TEST VALVE C41-F012 FAILS OPEN LCFO	1095	H	1.40E-7	H
OPM001HW	SLC PUMP A C001A FAILS TO RUN AFTER FIRST HOUT	1.5	H	7.05E-6	H
OPM001HW1	SLC PUMP A C001A FAILS TO RUN DURING FIRST HOUR	1	H	9.69E-6	H
OPM002HW	SLC PUMP B C001B FAILS TO RUN AFTER FIRST HOUR	1.5	H	7.05E-6	H
OPM002HW1	SLC PUMP B C001B FAILS TO RUN DURING FIRST HOUR	1	H	9.69E-6	H
OPP000HF	LEAK OR PIPE RUPTURE IN TANK PIPE	1095	H	4.93E-9	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
OPTMA	10 SECOND PUMP TIMER A FAILS	1095	H	1.50E-6	H
OPTMB	10 SECOND PUMP TIMER B FAILS	1095	H	1.50E-6	H
OSW000HW	C.R. LEVEL INDICATOR C41-LS601 FAILS TO WARN OPERATOR	4	H	4.25E-7	H
OTK000HW	TANK A001 FAILS	1095	H	7.43E-8	H
OTS002HW	SLC TANK TIS002 FAILS HIGH	4	H	1.17E-6	H
OTS003HW	SLC TANK TIS003 FAILS	4	H	1.17E-6	H
OTS006HW	B001 HEATER SURFACE TIS006 FAILS HIGH	4	H	1.17E-6	H
OXV001HW	GATE VALVE F005A FAILS CLOSED (NLO-FC)	1095	H	1.76E-8	H
OXV003HW	GATE VALVE F002A FAILS CLOSED (NLO-FC)	1095	H	1.76E-8	H
OXV004HW	GATE VALVE F002B FAILS CLOSED (NLO-FC)	1095	H	1.76E-8	H
P11AF	MAIN SCRAM GRP1 DIV 1 PARL LD-A FAILS	370	H	1.07E-6	H
P12BF	MAIN SCRAM GRP1 DIV 2 PARL LD-B FAILS	370	H	1.07E-6	H
P13CF	MAIN SCRAM GRP1 DIV 3 PARL LD-C FAILS	370	H	1.07E-6	H
P14DF	MAIN SCRAM GRP1 DIV 4 PARL LD-D FAILS	370	H	1.07E-6	H
P21AF	MAIN SCRAM GRP2 DIV 1 PARL LD-A FAILS	370	H	1.07E-6	H
P22BF	MAIN SCRAM GRP2 DIV 2 PARL LD-B FAILS	370	H	1.07E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
P23CF	MAIN SCRAM GRP2 DIV 3 PARL LD-C FAILS	370	H	1.07E-6	H
P24DF	MAIN SCRAM GRP2 DIV 4 PARL LD-D FAILS	370	H	1.07E-6	H
P31AF	MAIN SCRAM GRP3 DIV 1 PARL LD-A FAILS	370	H	1.07E-6	H
P32BF	MAIN SCRAM GRP3 DIV 2 PARL LD-B FAILS	370	H	1.07E-6	H
P33CF	MAIN SCRAM GRP3 DIV 3 PARL LD-C FAILS	370	H	1.07E-6	H
P34DF	MAIN SCRAM GRP3 DIV 4 PARL LD-D FAILS	370	H	1.07E-6	H
P41AF	MAIN SCRAM GRP4 DIV 1 PARL LD-A FAILS	370	H	1.07E-6	H
P42BF	MAIN SCRAM GRP4 DIV 2 PARL LD-B FAILS	370	H	1.07E-6	H
P43CF	MAIN SCRAM GRP4 DIV 3 PARL LD-C FAILS	370	H	1.07E-6	H
P44DF	MAIN SCRAM GRP4 DIV 4 PARL LD-D FAILS	370	H	1.07E-6	H
PPP101	RPV PRESSURE SENSOR FAILURE B21-PT007A	4	H	1.57E-6	H
PPP102	RPV PRESSURE SENSOR FAILURE B21-PT007B	4	H	1.57E-6	H
PPP103	RPV PRESSURE SENSOR FAILURE B21-PT007C	4	H	1.57E-6	H
PPP104	RPV PRESSURE SENSOR FAILURE B21-PT007D	4	H	1.57E-6	H
Q_FANDRUN	FAILURE OF STANDBY TRAIN D FORCED AIR COOLING FAN TO RUN AFTER FIRST HOUR	23	H	4.49E-5	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
Q_FANDRUN1	FAILURE OF STANDBY TRAIN D FORCED AIR COOLING FAN TO RUN DURING FIRST HOUR	1	H	1.65E-3	H
Q_FANERUN	FAILURE OF STANDBY TRAIN E FORCED AIR COOLING FAN TO RUN AFTER FIRST HOUR	23	H	4.49E-5	H
Q_FANERUN1	FAILURE OF STANDBY TRAIN E FORCED AIR COOLING FAN TO RUN DURING FIRST HOUR	1	H	1.65E-3	H
Q_FANFRUN	FAILURE OF STANDBY TRAIN F FORCED AIR COOLING FAN TO RUN AFTER FIRST HOUR	23	H	4.49E-5	H
Q_FANFRUN1	FAILURE OF STANDBY TRAIN F FORCED AIR COOLING FAN TO RUN DURING FIRST HOUR	1	H	1.65E-3	H
Q_FANRUNA2	FAILURE OF RUNNING TRAIN A FORCED AIR COOLING FAN TO RUN	8760	H	5.95E-7	H
Q_FANRUNB2	FAILURE OF RUNNING TRAIN B FORCED AIR COOLING FAN TO RUN	8760	H	5.95E-7	H
Q_FANRUNC2	FAILURE OF RUNNING TRAIN C FORCED AIR COOLING FAN TO RUN	8760	H	5.95E-7	H
RFL007BF	FLOW SENSOR E51-FT007-2 FAILS	1095	H	1.38E-7	H
RFLOSPHW	SUCTION STRAINER E51-D002 PLUGGED	1095	H	5.92E-6	H
RLS037HW	VALVE F037 LIMIT SWITCH FAILS	1095	H	6.11E-6	H
RLS039HW	LIMIT SWITCH FOR VALVE F039 FAILS	1095	H	6.11E-6	H
RLS045HW	LIMIT SWITCH FOR VALVE F045 FAILS	1095	H	6.11E-6	H
RLSTVSHW	TURBINE TRIP / THROTTLE VALVE LIMIT SWITCH FAILS	1095	H	6.11E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
RMV001HQ	CST ISOLATION VALVE F001 FAILS CLOSED (NOFC)	1095	H	1.40E-7	H
RMV002	ISOLATION VALVE F002 FAILS TO CLOSE (NOFO)	1095	H	1.90E-6	H
RMV003	ISOLATION VALVE F003 FAILS TO CLOSE (NOFO)	1095	H	1.90E-6	H
RMV004HP	VALVE F004 FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
RMV006HP	PUMP SUCTION VALVE F006 FAILS TO OPEN (NCFC)	1095	H	1.90E-6	H
RMV008HO	VALVE F008 FAILS OPEN (NCFO)	1095	H	1.40E-7	H
RMV009HO	VALVE F009 FAILS OPEN (NCFO)	1095	H	1.40E-7	H
RMV011HN	VALVE E51-F011 FAILS NOFO	1095	H	1.90E-6	H
RMV011HP	VALVE E51-F011 FAILS CLOSED NCFC	1095	H	1.90E-6	H
RMV035HQ	ISOLATION VALVE F035 FAILS CLOSED (NOFC)	1095	H	1.40E-7	H
RMV036HQ	ISOLATION VALVE F036 FAILS CLOSED (NOFC)	1095	H	1.40E-7	H
RMV037HR	F037 FAILS CLOSED (NCFC)	1095	H	1.90E-6	H
RMV039HQ	ISOLATION VALVE F039 FAILS CLOSED (NOFC)	1095	H	1.40E-7	H
RMV08HD1	VALVE F008 FAILS OPEN (NOFO)	1095	H	1.90E-6	H
RMV09HD1	VALVE F009 FAILS OPEN (NOFO)	1095	H	1.90E-6	H
RPM001FR1	PUMP C001 FAILS TO RUN FOR FIRST HOUR	1	H	1.15E-4	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
RPM001FR2	PUMP C001 FAILS TO RUN AFTER FIRST HOUR	23	H	6.98E-5	H
RPR005BF	PRESSURE SENSOR E51-PIS-Z605 FAILS	4	H	1.57E-6	H
RPR303FL	LOW SUCTION PRESSURE XMTR PIS-Z602 FAILURE	4	H	1.57E-6	H
RRMCOOL1	RCIC PUMP ROOM A.C. UNIT FAILS	24	H	9.40E-5	H
RTU001DH	TURBINE MECHANICAL FAILURE	370	H	1.00E-5	H
S11AF	MAIN SCRAM GRP1 DIV 1 SERS LD-A FAILS	370	H	1.07E-6	H
S12BF	MAIN SCRAM GRP1 DIV 2 SERS LD-B FAILS	370	H	1.07E-6	H
S13CF	MAIN SCRAM GRP1 DIV 3 SERS LD-C FAILS	370	H	1.07E-6	H
S14DF	MAIN SCRAM GRP1 DIV 4 SERS LD-D FAILS	370	H	1.07E-6	H
S21AF	MAIN SCRAM GRP2 DIV 1 SERS LD-A FAILS	370	H	1.07E-6	H
S22BF	MAIN SCRAM GRP2 DIV 2 SERS LD-B FAILS	370	H	1.07E-6	H
S23CF	MAIN SCRAM GRP2 DIV 3 SERS LD-C FAILS	370	H	1.07E-6	H
S24DF	MAIN SCRAM GRP2 DIV 4 SERS LD-D FAILS	370	H	1.07E-6	H
S31AF	MAIN SCRAM GRP3 DIV 1 SERS LD-A FAILS	370	H	1.07E-6	H
S32BF	MAIN SCRAM GRP3 DIV 2 SERS LD-B FAILS	370	H	1.07E-6	H
S33CF	MAIN SCRAM GRP3 DIV 3 SERS LD-C FAILS	370	H	1.07E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
S34DF	MAIN SCRAM GRP3 DIV 4 SERS LD-D FAILS	370	H	1.07E-6	H
S41AF	MAIN SCRAM GRP4 DIV 1 SERS LD-A FAILS	370	H	1.07E-6	H
S42BF	MAIN SCRAM GRP4 DIV 2 SERS LD-B FAILS	370	H	1.07E-6	H
S43CF	MAIN SCRAM GRP4 DIV 3 SERS LD-C FAILS	370	H	1.07E-6	H
S44DF	MAIN SCRAM GRP4 DIV 4 SERS LD-D FAILS	370	H	1.07E-6	H
SACMRPAD	RUPTURE OF SRV N2 ACCUMULATOR B21-A004A	8760	H	9.05E-8	H
SACMRPCD	RUPTURE OF SRV N2 ACCUMULATOR B21-A004N	8760	H	9.05E-8	H
SACMRPFD	RUPTURE OF SRV N2 ACCUMULATOR B21-A004F	8760	H	9.05E-8	H
SACMRPHD	RUPTURE OF SRV N2 ACCUMULATOR B21-A004H	8760	H	9.05E-8	H
SACMRPLD	RUPTURE OF SRV N2 ACCUMULATOR B21-A004L	8760	H	9.05E-8	H
SACMRPND	RUPTURE OF SRV N2 ACCUMULATOR B21-A004N	8760	H	9.05E-8	H
SACMRPRD	RUPTURE OF SRV N2 ACCUMULATOR B21-A004R	8760	H	9.05E-8	H
SACMRPTD	RUPTURE OF SRV N2 ACCUMULATOR B21-A004T	8760	H	9.05E-8	H
SACUMLEA	SRV A ACCUMULATOR HAS LEAKED EMPTY	8760	H	9.05E-8	H
SACUMLEC	SRV C ACCUMULATOR HAS LEAKED EMPTY	8760	H	9.05E-8	H
SACUMLEF	SRV F ACCUMULATOR HAS LEAKED EMPTY	8760	H	9.05E-8	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
SACUMLEH	SRV H ACCUMULATOR HAS LEAKED EMPTY	8760	H	9.05E-8	H
SACUMLEL	SRV L ACCUMULATOR HAS LEAKED EMPTY	8760	H	9.05E-8	H
SACUMLEN	SRV N ACCUMULATOR HAS LEAKED EMPTY	8760	H	9.05E-8	H
SACUMLER	SRV R ACCUMULATOR HAS LEAKED EMPTY	8760	H	9.05E-8	H
SACUMLET	SRV T ACCUMULATOR HAS LEAKED EMPTY	8760	H	9.05E-8	H
SLNBRK	BREAK IN N2 SUPPLY LINES	8760	H	4.93E-9	H
SMVF200F	MO VALVE P54-F200 FAILS CLOSED (NOFC)	360	H	1.40E-7	H
SSVF121A	RELIEF VALVE FAILS	8760	H	3.60E-6	H
SSVF121C	RELIEF VALVE FAILS	8760	H	3.60E-6	H
SSVF121F	RELIEF VALVE FAILS	8760	H	3.60E-6	H
SSVF121H	RELIEF VALVE FAILS	8760	H	3.60E-6	H
SSVF121L	RELIEF VALVE FAILS	8760	H	3.60E-6	H
SSVF121N	RELIEF VALVE FAILS	8760	H	3.60E-6	H
SSVF121R	RELIEF VALVE FAILS	8760	H	3.60E-6	H
SSVF121T	RELIEF VALVE FAILS	8760	H	3.60E-6	H
TFUSEA	VALVE A TOP FUSE FAILS	8784	H	5.18E-8	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
TFUSEB	VALVE B TOP FUSE FAILS	8784	H	5.18E-8	H
TLU1F	TRIP LOGIC UNIT DIVISION 1 FAILS	59	H	1.07E-6	H
TLU2F	TRIP LOGIC UNIT DIVISION 2 FAILS	59	H	1.07E-6	H
TLU3F	TRIP LOGIC UNIT DIVISION 3 FAILS	59	H	1.07E-6	H
TLU4F	TRIP LOGIC UNIT DIVISION 4 FAILS	59	H	1.07E-6	H
VF016A	TCV P25-F016A FAILS CLOSED (NOFC)	24	H	1.40E-7	H
VF016B	TCV P25-F016B FAILS CLOSED (NOFC)	24	H	1.40E-7	H
VF016C	TCV P25-F016C FAILS CLOSED (NOFC)	24	H	1.40E-7	H
VF022A	TCV P25-F022A FAILS CLOSED (NOFC)	24	H	1.40E-7	H
VF022B	TCV P25-F022B FAILS CLOSED (NOFC)	24	H	1.40E-7	H
VF022C	TCV P25-F022C FAILS CLOSED (NOFC)	24	H	1.40E-7	H
VPR007A	DIFF PRESSURE TRANSMITTER P25-DPT-007A FAILS	4	H	1.57E-6	H
VPR007B	DIFF PRESSURE TRANSMITTER P25-DPT-007B FAILS	4	H	1.57E-6	H
VPR007C	DIFF PRESSURE TRANSMITTER P25-DPT-007C FAILS	4	H	1.57E-6	H
VPRXXXA	DIFF PRESSURE TRANSMITTER A FAILS	4	H	1.57E-6	H
VPRXXXB	DIFF PRESSURE TRANSMITTER B FAILS	4	H	1.57E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
VPRXXXC	DIFF PRESSURE TRANSMITTER C FAILS	4	H	1.57E-6	H
VPRXXXD	DIFF PRESSURE TRANSMITTER D FAILS	4	H	1.57E-6	H
VPRXXXE	DIFF PRESSURE TRANSMITTER E FAILS	4	H	1.57E-6	H
VPRXXXF	DIFF PRESSURE TRANSMITTER F FAILS	4	H	1.57E-6	H
VPVF012A	PCV P25-F012A FAILS FULL OPEN (NOFO)	24	H	1.90E-6	H
VPVF012B	PCV P25-F012B FAILS FULL OPEN (NOFO)	24	H	1.90E-6	H
VPVF012C	PCV P25-F012C FAILS FULL OPEN (NOFO)	24	H	1.90E-6	H
VTE005A	TEMP SENSOR P25-TE-005A FAILS	4	H	1.17E-6	H
VTE005B	TEMP SENSOR P25-TE-005B FAILS	4	H	1.17E-6	H
VTE005C	TEMP SENSOR P25-TE-005C FAILS	4	H	1.17E-6	H
W013AFC	FAILURE OF HX INLET VALVE P21-F013A (NCFC)	1095	H	1.90E-6	H
W013BFC	FAILURE OF HX INLET VALVE P21-F013B (NCFC)	1095	H	1.90E-6	H
W013CFC	FAILURE OF HX INLET VALVE P21-F013C (NCFC)	1095	H	1.90E-6	H
WAVR6AH	TEMPERATURE CONTROL VALVE P21-F006A FAILS CLOSED (NOFC)	24	H	1.40E-7	H
WAVR6BH	TEMPERATURE CONTROL VALVE P21-F006B FAILS CLOSED (NOFC)	24	H	1.40E-7	H
WAVR6CH	TEMPERATURE CONTROL VALVE P21-F006C FAILS CLOSED (NOFC)	24	H	1.40E-7	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
WDALUB	RHR PUMP A FAILS TO RUN AFTER FIRST HOUR	23	H	7.05E-6	H
WDALUB1	RHR PUMP A FAILS TO RUN DURING FIRST HOUR	1	H	9.69E-6	H
WDAN2LF	FLOW SIGNAL FROM FT008A FAILS	1095	H	1.38E-7	H
WDARAC	RHR-A PUMP ROOM A.C. UNIT FAILS	24	H	9.40E-5	H
WDASTRN	STRAINER E11-D001A PLUGGED	1095	H	5.92E-6	H
WDBLUB	RHR PUMP B FAILS TO RUN AFTER FIRST HOUR	23	H	7.05E-6	H
WDBLUB1	RHR PUMP B FAILS TO RUN DURING FIRST HOUR	1	H	9.69E-6	H
WDBN2LF	FLOW SIGNAL FROM FT008B FAILS	1095	H	1.38E-7	H
WDBRAC	RHR-B PUMP ROOM A.C. UNIT FAILS	24	H	9.40E-5	H
WDBSTRN	STRAINER E11-D001B PLUGGED	1095	H	5.92E-6	H
WDCLUB	RHR PUMP C FAILS TO RUN AFTER FIRST HOUR	23	H	7.05E-6	H
WDCLUB1	RHR PUMP C FAILS TO RUN DURING FIRST HOUR	1	H	9.69E-6	H
WDCN2LF	FLOW SIGNAL FROM FT008C FAILS	1095	H	1.38E-7	H
WDCRAC	RHR-C PUMP ROOM A.C. UNIT FAILS	24	H	9.40E-5	H
WDCSTRN	STRAINER E11-D001C PLUGGED	1095	H	5.92E-6	H
WFLD1AH	STRAINER P41-D001A PLUGS	24	H	5.92E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
WFLD1BH	STRAINER P41-D001B PLUGS	24	H	5.92E-6	H
WFLD1CH	STRAINER P41-D001C PLUGS	24	H	5.92E-6	H
WFLD1DH	STRAINER P41-D001D PLUGS	24	H	5.92E-6	H
WFLD1EH	STRAINER P41-D001E PLUGS	24	H	5.92E-6	H
WFLD1FH	STRAINER P41-D001F PLUGS	24	H	5.92E-6	H
WHEB1AH	ACTIVE HEAT EXCHANGER PLUGS	24	H	3.60E-6	H
WHEB1AH1	ACTIVE HEAT EXCHANGER LEAKS	24	H	2.10E-6	H
WHEB1BH	ACTIVE HEAT EXCHANGER PLUGS	24	H	3.60E-6	H
WHEB1BH1	ACTIVE HEAT EXCHANGER FAILURE	24	H	2.10E-6	H
WHEB1CH	ACTIVE HEAT EXCHANGER PLUGS	24	H	3.60E-6	H
WHEB1CH1	ACTIVE HEAT EXCHANGER LEAKS	24	H	2.10E-6	H
WHEB1DH	ACTIVE HEAT EXCHANGER PLUGS	24	H	3.60E-6	H
WHEB1DH1	ACTIVE HEAT EXCHANGER LEAKS	24	H	2.10E-6	H
WHEB1EH	ACTIVE HEAT EXCHANGER PLUGS	24	H	3.60E-6	H
WHEB1EH1	ACTIVE HEAT EXCHANGER LEAKS	24	H	2.10E-6	H
WHEB1FH	ACTIVE HEAT EXCHANGER PLUGS	24	H	3.60E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
WHEB1FH1	ACTIVE HEAT EXCHANGER LEAKS	24	H	2.10E-6	H
WHEB1GH	STANDBY HEAT EXCHANGER PLUGS	24	H	3.60E-6	H
WHEB1GH1	STANDBY HEAT EXCHANGER LEAKS	24	H	2.10E-6	H
WHEB1GM	STANDBY HEAT EXCHANGER IN MAINTENANCE	9.5	H	7.62E-3	H
WHEB1HH	STANDBY HEAT EXCHANGER PLUGS	24	H	3.60E-6	H
WHEB1HH1	STANDBY HEAT EXCHANGER LEAKS	24	H	2.10E-6	H
WHEB1HM	STANDBY HEAT EXCHANGER IN MAINTENANCE	9.5	H	7.62E-3	H
WHEB1JH	STANDBY HEAT EXCHANGER PLUGS	24	H	3.60E-6	H
WHEB1JH1	STANDBY HEAT EXCHANGER LEAKS	24	H	2.10E-6	H
WHEB1JM	STANDBY HEAT EXCHANGER IN MAINTENANCE	9.5	H	7.62E-3	H
WMV13AH	VALVE P41-F013A FAILS CLOSED (NOFC)	24	H	1.40E-7	H
WMV13BH	VALVE P41-F013B FAILS CLOSED (NOFC)	24	H	1.40E-7	H
WMV13CH	VALVE P41-F013C FAILS CLOSED (NOFC)	24	H	1.40E-7	H
WMV13DH	VALVE P41-F013D FAILS CLOSED (NCFC)	360	H	1.90E-6	H
WMV13EH	VALVE P41-F013E FAILS CLOSED (NCFC)	360	H	1.90E-6	H
WMV13FH	VALVE P41-F013F FAILS CLOSED (NCFC)	360	H	1.90E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
WMVR4AH	SEPARATION VALVE P21-F004A FAILS CLOSED (NOFC)	24	H	1.40E-7	H
WMVR4BH	SEPARATION VALVE P21-F004B FAILS CLOSED (NOFC)	24	H	1.40E-7	H
WMVR4CH	SEPARATION VALVE P21-F004C FAILS CLOSED (NOFC)	24	H	1.40E-7	H
WMVR4DH	SEPARATION VALVE P21-F004D FAILS CLOSED (NOFC)	24	H	1.40E-7	H
WMVR4EH	SEPARATION VALVE P21-F004E FAILS CLOSED (NOFC)	24	H	1.40E-7	H
WMVR4FH	SEPARATION VALVE P21-F004F FAILS CLOSED (NOFC)	24	H	1.40E-7	H
WMVR4GH	SEPARATION VALVE P21-F004G FAILS TO OPEN (NCFC)	360	H	1.90E-6	H
WMVR4HH	SEPARATION VALVE P21-F004H FAILS TO OPEN (NCFC)	360	H	1.90E-6	H
WMVR4JH	SEPARATION VALVE P21-F004J FAILS TO OPEN (NCFC)	360	H	1.90E-6	H
WMVS3AH	SEPARATION VALVE P41-F003A FAILS (NOFC)	24	H	1.40E-7	H
WMVS3BH	SEPARATION VALVE P41-F003B FAILS (NOFC)	24	H	1.40E-7	H
WMVS3CH	SEPARATION VALVE P41-F003C FAILS (NOFC)	24	H	1.40E-7	H
WMVS3DH	SEPARATION VALVE P41-F003D FAILS (NOFC)	24	H	1.40E-7	H
WMVS3EH	SEPARATION VALVE P41-F003E FAILS (NOFC)	24	H	1.40E-7	H
WMVS3FH	SEPARATION VALVE P41-F003F FAILS (NOFC)	24	H	1.40E-7	H
WMVS3GH	SEPARATION VALVE F003G FOR THE STANDBY HX FAILS CLOSED (NCFC)	360	H	1.90E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
WMVS3HH	SEPARATION VALVE F003H FOR THE STANDBY HX FAILS CLOSED (NCFC)	360	H	1.90E-6	H
WMVS3JH	SEPARATION VALVE F003J FOR THE STANDBY HX FAILS CLOSED (NCFC)	360	H	1.90E-6	H
WMVS4AH	VALVE P41-F004A FAILS CLOSED (NOFC)	24	H	1.40E-7	H
WMVS4BH	VALVE P41-F004B FAILS CLOSED (NOFC)	24	H	1.40E-7	H
WMVS4CH	VALVE P41-F004C FAILS CLOSED (NOFC)	24	H	1.40E-7	H
WMVS4DH	VALVE P41-F004D FAILS CLOSED (NOFC)	360	H	1.40E-7	H
WMVS4EH	VALVE P41-F004E FAILS CLOSED (NOFC)	360	H	1.40E-7	H
WMVS4FH	VALVE P41-F004F FAILS CLOSED (NOFC)	360	H	1.40E-7	H
WMVS5AH	SEPARATION VALVE P41-F005A FAILS (NOFC)	24	H	1.40E-7	H
WMVS5BH	SEPARATION VALVE P41-F005B FAILS (NOFC)	24	H	1.40E-7	H
WMVS5CH	SEPARATION VALVE P41-F005C FAILS (NOFC)	24	H	1.40E-7	H
WMVS5DH	SEPARATION VALVE P41-F005D FAILS (NOFC)	24	H	1.40E-7	H
WMVS5EH	SEPARATION VALVE P41-F005E FAILS (NOFC)	24	H	1.40E-7	H
WMVS5FH	SEPARATION VALVE P41-F005F FAILS (NOFC)	24	H	1.40E-7	H
WMVS5GH	SEPARATION VALVE F005G FOR THE STANDBY HX FAILS CLOSED (NCFC)	360	H	1.90E-6	H
WMVS5HH	SEPARATION VALVE F005H FOR THE STANDBY HX FAILS CLOSED (NCFC)	360	H	1.90E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
WMVS5JH	SEPARATION VALVE F005J FOR THE STANDBY HX FAILS CLOSED (NCFC)	360	H	1.90E-6	H
WPMHC1A	PUMP FAILURE	24	H	6.27E-6	H
WPMHC1B	PUMP FAILURE	24	H	6.27E-6	H
WPMHC1C	PUMP FAILURE	24	H	6.27E-6	H
WPMHC1D	PUMP FAILURE	24	H	6.27E-6	H
WPMHC1E	PUMP FAILURE	24	H	6.27E-6	H
WPMHC1F	PUMP FAILURE	24	H	6.27E-6	H
WPMMNTF	STANDBY PUMP IN MAINTENANCE	9.5	H	9.70E-6	H
WPMRC1AH	RUNNING PUMP FAILS DURING MISSION	24	H	6.27E-6	H
WPMRC1BH	RUNNING PUMP FAILS DURING MISSION	24	H	6.27E-6	H
WPMRC1CH	RUNNING PUMP FAILS DURING MISSION	24	H	6.27E-6	H
WPMRC1DH	STANDBY PUMP FAILS TO RUN AFTER FIRST HOUR	23	H	7.05E-6	H
WPMRC1DH1	STANDBY PUMP FAILS TO RUN DURING FIRST HOUR	1	H	9.69E-6	H
WPMRC1DM	STANDBY PUMP IN MAINTENANCE	9.5	H	9.70E-6	H
WPMRC1ED	STANDBY PUMP FAILS TO START	1	N	1.56E-3	N
WPMRC1EH	STANDBY PUMP FAILS TO RUN AFTER FIRST HOUR	23	H	7.05E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
WPMRC1EH1	STANDBY PUMP FAILS TO RUN DURING FIRST HOUR	1	H	9.69E-6	H
WPMRC1EM	STANDBY PUMP IN MAINTENANCE	9.5	H	9.70E-6	H
WPMRC1FH	STANDBY PUMP FAILS TO RUN AFTER FIRST HOUR	23	H	7.05E-6	H
WPMRC1FH1	STANDBY PUMP FAILS TO RUN DURING FIRST HOUR	1	H	9.69E-6	H
WPMRC1FM	STANDBY PUMP IN MAINTENANCE	9.5	H	9.70E-6	H
WPMSC1AH	PUMP FAILS	24	H	6.27E-6	H
WPMSC1BH	PUMP FAILS	24	H	6.27E-6	H
WPMSC1CH	PUMP FAILS	24	H	6.27E-6	H
WPMSC1DH	STANDBY PUMP FAILS TO RUN AFTER FIRST HOUR	23	H	7.05E-6	H
WPMSC1DH1	STANDBY PUMP FAILS TO RUN DURING FIRST HOUR	1	H	9.69E-6	H
WPMSC1DM	STANDBY PUMP IN MAINTENANCE	9.5	H	9.70E-6	H
WPMSC1EH	STANDBY PUMP FAILS TO RUN AFTER FIRST HOUR	23	H	7.05E-6	H
WPMSC1EH1	STANDBY PUMP FAILS TO RUN DURING FIRST HOUR	1	H	9.69E-6	H
WPMSC1EM	STANDBY PUMP IN MAINTENANCE	9.5	H	9.70E-6	H
WPMSC1FH	STANDBY PUMP FAILS TO RUN AFTER START	23	H	7.05E-6	H
WPMSC1FH1	STANDBY PUMP FAILS TO RUN DURING FIRST HOUR	1	H	9.69E-6	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
WPMSC1FM	STANDBY PUMP IN MAINTENANCE	9.5	H	9.70E-6	H
WPPHEAH	PIPE RUPTURED	24	H	4.93E-9	H
WPPHEBH	PIPE RUPTURED	24	H	4.93E-9	H
WPPHECH	PIPE RUPTURED	24	H	4.93E-9	H
WPPREAH	PIPE RUPTURED	24	H	4.93E-9	H
WPPREBH	PIPE RUPTURED	24	H	4.93E-9	H
WPPRECH	PIPE RUPTURED	24	H	4.93E-9	H
WPPSWSAH	PIPE RUPTURED	24	H	4.93E-9	H
WPPSWSBH	PIPE RUPTURED	24	H	4.93E-9	H
WPPSWSCH	PIPE RUPTURED	24	H	4.93E-9	H
WPV025A	MO-PCV P21-F025A FAILS CLOSED	24	H	1.40E-7	H
WPV025B	MO-PCV P21-F025B FAILS CLOSED	24	H	1.40E-7	H
WPV025C	MO-PCV P21-F025C FAILS CLOSED	24	H	1.40E-7	H
WPV025D	MO-PCV P21-F025D FAILS CLOSED	24	H	1.40E-7	H
WPV025E	MO-PCV P21-F025E FAILS CLOSED	24	H	1.40E-7	H
WPV025F	MO-PCV P21-F025F FAILS CLOSED	24	H	1.40E-7	H

Table 19D.6-14 (continued)
Component Failure Rate Data

Name	Description	Mission Duration	Units	Failure Rate	Units
WRFD1A	REFRIGERATOR FAILURE	24	H	8.46E-5	H
WRFD1B	REFRIGERATOR FAILURE	24	H	8.46E-5	H
WRFD1C	REFRIGERATOR FAILURE	24	H	8.46E-5	H
WRFD1D	REFRIGERATOR FAILURE	24	H	8.46E-5	H
WRFD1E	REFRIGERATOR FAILURE	24	H	8.46E-5	H
WRFD1F	REFRIGERATOR FAILURE	24	H	8.46E-5	H
WRFMNTF	STANDBY REFRIGERATOR IN MAINTENANCE	1.963	H	3.00E-5	H
WTE005A	TEMPERATURE ELEMENT P21-TE-005A FAILS LOW	4	H	1.17E-6	H
WTE005B	TEMPERATURE ELEMENT P21-TE-005B FAILS LOW	4	H	1.17E-6	H
WTE005C	TEMPERATURE ELEMENT P21-TE-005C FAILS LOW	4	H	1.17E-6	H
WTE052	TEMP SENSOR U41-TE-052 FAILS	4	H	1.17E-6	H
WTE056	TEMP SENSOR U41-TE-056 FAILS	4	H	1.17E-6	H
WTE060	TEMP SENSOR U41-TE-060 FAILS	4	H	1.17E-6	H
WTE113A	TEMP SENSOR U41-TE-113A FAILS	4	H	1.17E-6	H
WTE113B	TEMP SENSOR U41-TE-113B FAILS	4	H	1.17E-6	H
WTE113C	TEMP SENSOR U41-TE-113C FAILS	4	H	1.17E-6	H

Table 19D.6 15
Component Failure Probabilities

Name	Description	Probability
2DPCCCFABC	RHR SUPPRESSION POOL COOLING A, B, AND C TRAIN LEVEL CCF	1.39E-3
3HCULOC	LOCATION	4.00E-8
3HCURAND	RANDOM	4.00E-8
55RANROD	FAILURE OF 55 RANDOM RODS	1.00E-9
AN2BAFFE	1ST FIVE BOTTLES OF N2 EMPTY	0.00E+00
AN2BBFFE	1ST FIVE BOTTLES OF N2 EMPTY	0.00E+00
AN2BCSFE	2ND FIVE BOTTLES OF N2 EMPTY	0.00E+00
AN2BDSFE	2ND FIVE BOTTLES OF N2 EMPTY	0.00E+00
BALLSP	BALL SPINDLE BREAKS OR DISTORTS	1.00E-6
BB	BOLTS BREAK	1.00E-8
BCVP	BALL CHECK VALVE PLUGS	1.00E-6
BNJAMOB	BALL NUT JAMS OR BREAKS	1.00E-6
C	FAILURE TO CONTROL ACTIVITY WITH RODS	1.00E-8
CCFBYP	CCF SLU BYPASS UNIT	2.45E-6
CCFDTM	CCF OF DIGITAL TRIP UNITS	6.31E-8

Table 19D.6 15 (continued)
Component Failure Probabilities

Name	Description	Probability
CCFMUX	CCF OF TRANSMISSION NETWORK (EMS)	6.31E-8
CCFS3A	RPV PRESSURE SENSORS CCF	1.26E-6
CCFTLU	CCF SYSTEM LOGIC UNIT FAILS	6.31E-8
CHCUF	CORRESPONDING HCU's FAIL	8.00E-8
CLCCCFH	ROD CONTROL & INFORMATION SYSTEM CCF	1.74E-7
CMF	CORRESPONDING MOTORS FAIL	3.16E-6
COND	CONDENSATE PUMPS NOT AVAILABLE	4.32E-2
COPS	FAILURE OF CONTAINMENT OVERPRESSURE PROTECTION SYSTEM	1.00E-4
CRD	FAILURE OF CONTROL ROD DRIVE INJECTION TO MAINTAIN LEVEL	1.00E-1
CRDGTB	CRD GUIDE TUBE BREAKS	1.00E-6
E1	FAIL TO RECOVER OSP IN 30 MINUTES	0.731
E2D	FAIL TO RECOVER 1 DG IN 2 HOURS	0.648
E2O	FAIL TO RECOVER OSP IN 2 HOURS	0.318
E8D	FAIL TO RECOVER 1 DG IN 8 HOURS	0.296
E8O	FAIL TO RECOVER OSP IN 8 HOURS	0.0672
EAC695C	ALL DIESELS & COMBUSTION TG FAIL	4.65E-5

Table 19D.6 15 (continued)
Component Failure Probabilities

Name	Description	Probability
EBY1CCF	BATTERY CCF	5.28E-6
EDC11H	UPS DC THYRISTER SWITCH FAILURE	1.43E-6
EDC12H	UPS DC THYRISTER SWITCH FAILURE	1.43E-6
EDC13H	UPS DC THYRISTER SWITCH FAILURE	1.43E-6
EDC14H	UPS DC THYRISTER SWITCH FAILURE	1.43E-6
EMBDNE	E-M BREAK DOES NOT ENERGIZE	1.00E-4
FARFLOC	LOCATION	1.00E-8
FARFRAND	RANDOM	1.00E-9
FOID	FOREIGN OBJECT IN DRIVE	1.00E-6
FW	FEEDWATER PUMPS NOT AVAILABLE	5.00E-2
G523	RPV SUCTION LINE ISOLATION VALVES FAIL OPEN (NCFO)	0.00E+00
HCLB	CHARGING WATER LINE BREAKS	1.00E-6
HCULB	HCU LINE BREAK	1.00E-6
HLNCSTL1	INSTRUMENT LINE BREAK	2.05E-3
HLNCSTL2	INSTRUMENT LINE BREAK	2.05E-3
HLNCSTL3	INSTRUMENT LINE BREAK	2.05E-3

Table 19D.6 15 (continued)
Component Failure Probabilities

Name	Description	Probability
HLNCSTL4	INSTRUMENT LINE BREAK	2.05E-3
HOLLOP	HOLLOW PISTON BREAKS OR DISTORTS	1.00E-6
HPCFCCFBC	HPCF B AND C TRAIN LEVEL CCF	1.85E-3
HPWLB	PURGE WATER LINE BREAKS	1.00E-6
HSVCCF	COMMON CAUSE FAILURE OF ARI SOLENOID VALVES	1.92E-6
ICV101	INADVERTENT CLOSURE OF VALVE 101	1.00E-6
ICV140	INADVERTENT CLOSURE OF VALVE 140	1.00E-6
IDMCCCF	RFC COMMON CAUSE FAILURE	1.66E-6
ILCCCFH	CCF REMOTE MULTIPLEXING UNITS	6.31E-8
ILECCFH	NARROW RANGE LEVEL SENSORS CCF	2.01E-7
ILEPVCH	RPV LEVEL SENSORS CCF	2.01E-7
ILN0D1H	INSTRUMENT LINE BREAK	2.05E-3
ILN0D2H	INSTRUMENT LINE BREAK	2.05E-3
ILN0D3H	INSTRUMENT LINE BREAK	2.05E-3
ILN0D4H	INSTRUMENT LINE BREAK	2.05E-3
ILN0V1H	INSTRUMENT LINE BREAK	2.05E-3

Table 19D.6 15 (continued)
Component Failure Probabilities

Name	Description	Probability
ILN0V2H	INSTRUMENT LINE BREAK	2.05E-3
ILN0V3H	INSTRUMENT LINE BREAK	2.05E-3
ILN0V4H	INSTRUMENT LINE BREAK	2.05E-3
IPOWERA	ELECTRIC POWER A FAILURE	4.20E-5
IPOWERB	ELECTRIC POWER B FAILURE	4.20E-5
IPOWERC	ELECTRIC POWER C FAILURE	4.20E-5
IPRDWCH	DW PRESSURE SENSORS CCF	1.26E-6
LFFSS	LATCH FINGERS FAIL TO SUSTAIN SCRAM	1.00E-6
LNSP	LOSS OF NON-SAFETY POWER TO FMCRD MOTOR	3.43E-4
LPL	LEVEL AND PRESSURE CONTROL FAILURE	1.98E-2
LVLFAIL	WATER LEVEL CONTROL FAILURE	1.00E-2
MF	MOTOR FAILS	3.80E-4
NEDC	LOSS OF 125V DC NON-DIVISIONAL POWER GROUP A	1.34E-5
NEDCB	LOSS OF 125V DC NON-DIVISIONAL POWER GROUP B	1.33E-5
NHR	FAILURE TO RESTORE NORMAL HEAT REMOVAL	1.00E-2
PC_ATWS	FAILURE OF SRVS TO RECLOSE (PC_ATWS)	1.00E-1

Table 19D.6 15 (continued)
Component Failure Probabilities

Name	Description	Probability
PC_OTHER	FAILURE OF SRVS TO RECLOSE (PC_Other)	3.00E-3
PDF	LOSS OF 6.9 kV DIV 1 AND PIP POWER SUPPLY	3.16E-6
PDISVLA_2	RPV DISCHARGE FAILURE (SD MODE)	0.00E+00
PF	POWER SUPPLY FAILS	3.16E-4
PO	FAILURE OF SRVS TO OPEN	1.00E-6
PO1	SRVS FAIL TO OPEN (AFTER SCRAM)	1.00E-6
PO2	SRVS FAIL TO OPEN (NO SCRAM)	1.00E-4
Q_FANDUNAVAIL	STANDBY TRAIN D FORCED AIR COOLING FAM UNAVAILABLE FOR TEST OR MAINT	1.86E-3
Q_FANEUNAVAIL	STANDBY TRAIN E FORCED AIR COOLING FAN UNAVAILABLE FOR TEST OR MAINT	1.86E-3
Q_FANFUNAVAIL	STANDBY TRAIN F FORCED AIR COOLING FAN UNAVAILABLE FOR TEST OR MAINT	1.86E-3
Q_OTHER	FAILURE TO INJECT WITH FEEDWATER (Q_Other)	2.54E-3
Q_TIS	FAILURE TO INJECT WITH FEEDWATER (Q_TIS)	4.02E-1
RWCU	FAILURE TO ACTUATE RWCU	0.1
SANPC	SCRAM ACCUMULATOR NOT PROPERLY CHARGED	1.00E-6
SCB	SPLINE CONNECTION BREAKS	1.00E-8

Table 19D.6 15 (continued)
Component Failure Probabilities

Name	Description	Probability
TSTINPRB	TEST IN PROGRESS	6.85E-4
TSTINPRC	TEST IN PROGRESS	6.85E-4
V115BCVL	CHECK VALVE 115 LEAKS	1.00E-6
V126FTO	SCRAM VALVE 126 FAILS MECHANICALLY	1.00E-6
V138BCVL	CHECK VALVE 138 BREAKS	1.00E-6
V139FTO	VALVE 139 FAILS TO OPEN	1.00E-6
WDCSCCFABC	RHR CORE FLOOD A, B, AND C TRAIN LEVEL CCF	8.85E-4
WPMPCCFRCWA	RBCW CCFS WITHIN A DIVISION (A)	1.16E-5
WPMPCCFRCWAB	RBCW CCFS BETWEEN DIV. A & B	4.94E-6
WPMPCCFRCWABC	RBCW CCFS BETWEEN DIV. A, B, & C	4.55E-6
WPMPCCFRCWAC	RBCW CCFS BETWEEN DIV. A & C	4.94E-6
WPMPCCFRCWB	RBCW CCFS WITHIN A DIVISION (B)	1.16E-5
WPMPCCFRCWBC	RBCW CCFS BETWEEN DIV. B & C	4.94E-6
WPMPCCFRCWC	RBCW CCFS WITHIN A DIVISION (C)	1.16E-5
WPMPCCFRSWA	RSW CCFs WITHIN A DIVISION (A)	1.16E-5
WPMPCCFRSWAB	RSW CCFs BETWEEN DIV. A & B	4.94E-6

Table 19D.6 15 (continued)
Component Failure Probabilities

Name	Description	Probability
WPMPCCFRSWABC	RSW CCFs BETWEEN DIV. A, B & C	4.55E-6
WPMPCCFRSWAC	RSW CCFs BETWEEN DIV. A & C	4.94E-6
WPMPCCFRSWB	RSW CCFs WITHIN A DIVISION (B)	1.16E-5
WPMPCCFRSWBC	RSW CCFs BETWEEN DIV. B & C	4.94E-6
WPMPCCFRSWC	RSW CCFs WITHIN A DIVISION (C)	1.16E-5

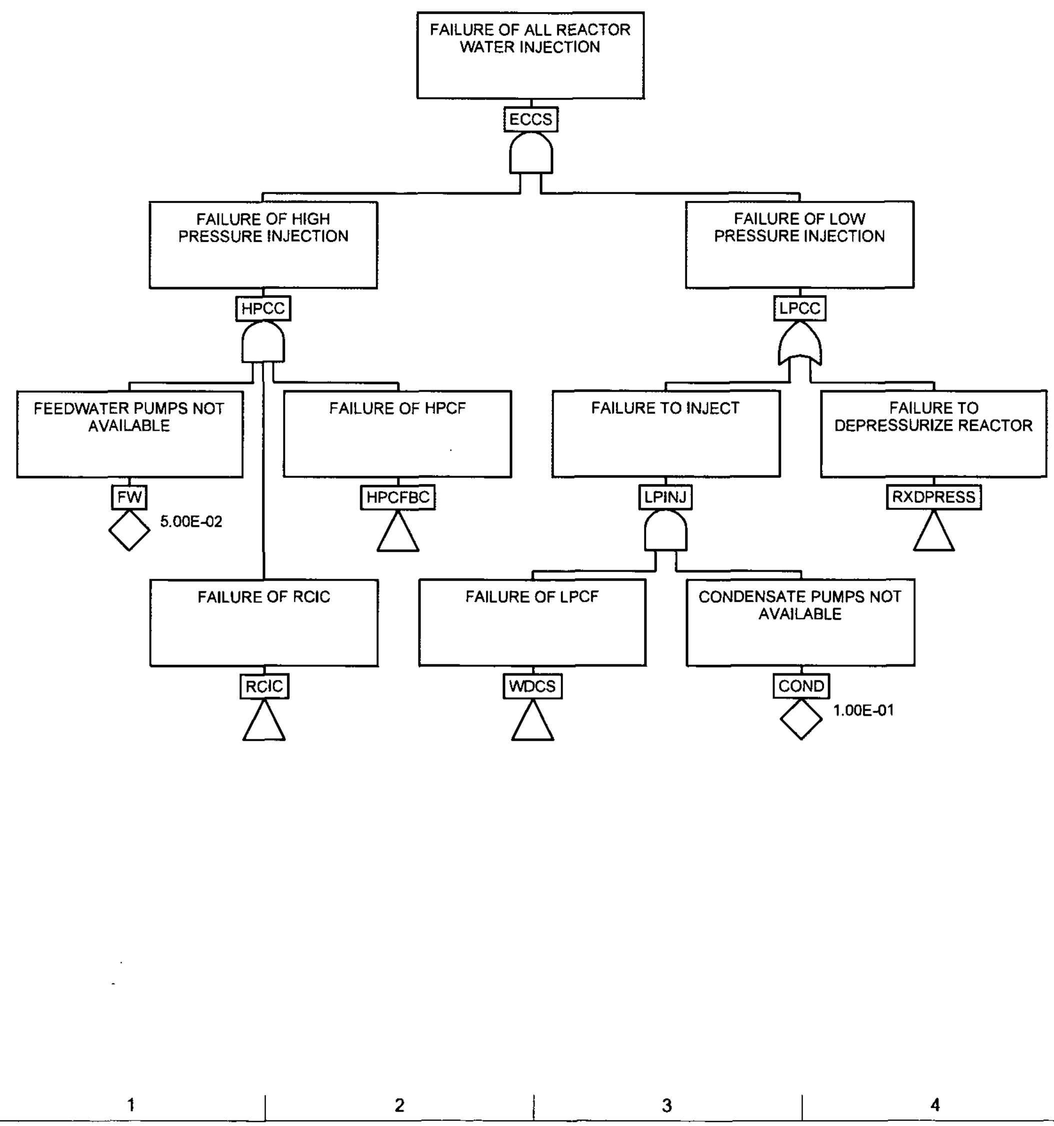
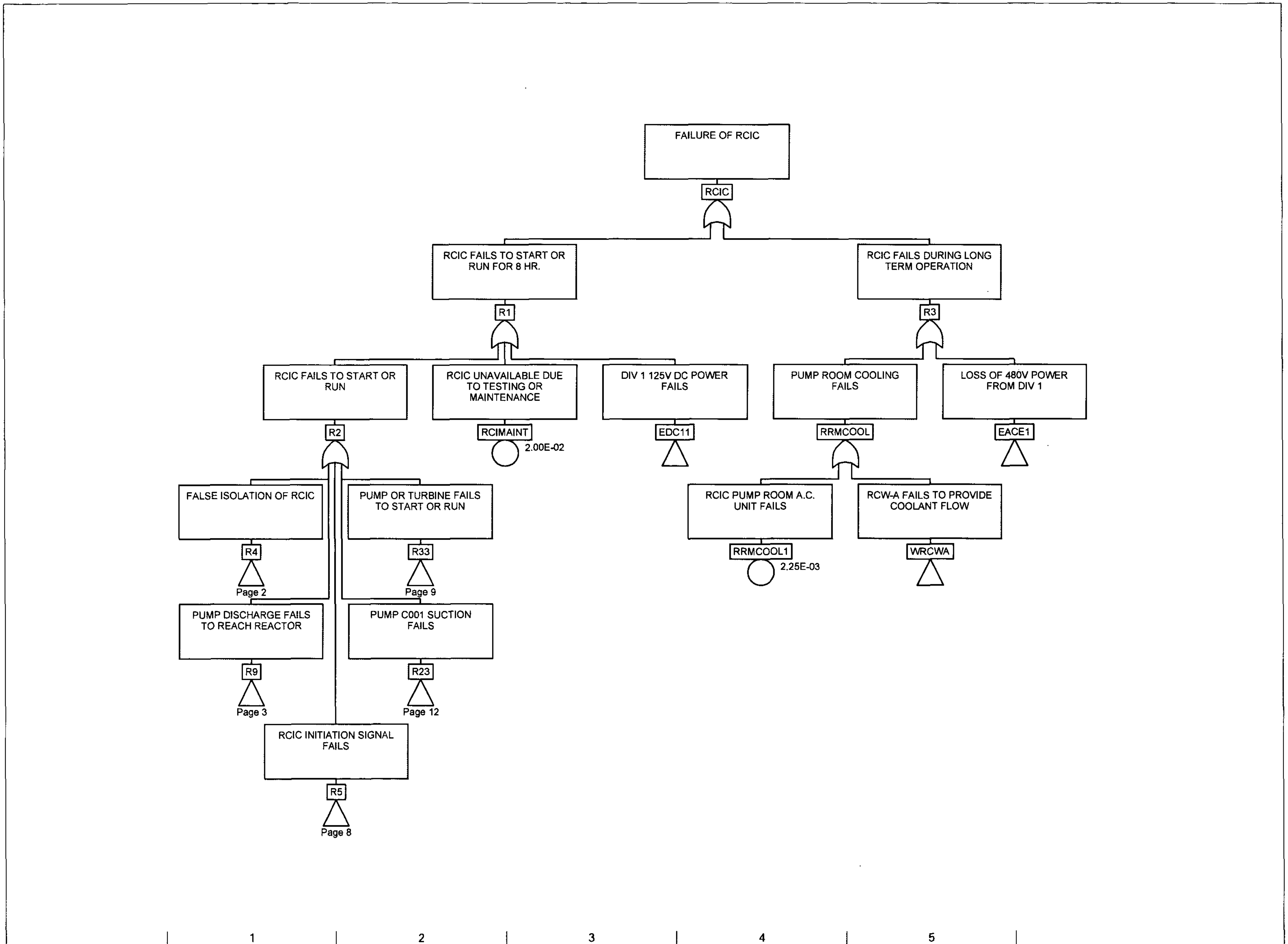


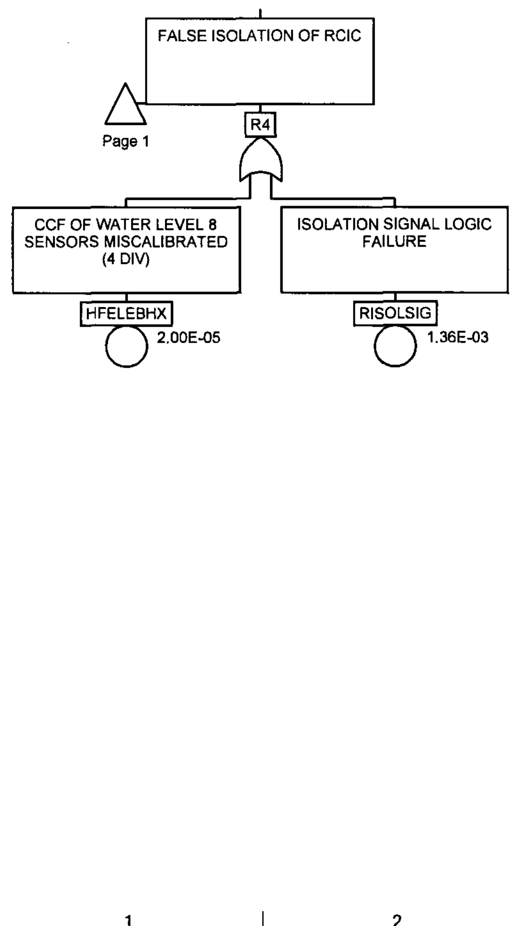
Figure 19D.6-1 Emergency Core Cooling System Fault Tree

Name	Page	Zone	Name	Page	Zone
COND	1	4			
ECCS	1	2			
FW	1	1			
HPCC	1	2			
HPCFBC	1	2			
LPCC	1	4			
LPINJ	1	3			
RCIC	1	2			
RXDPRESS	1	4			
WDCS	1	3			

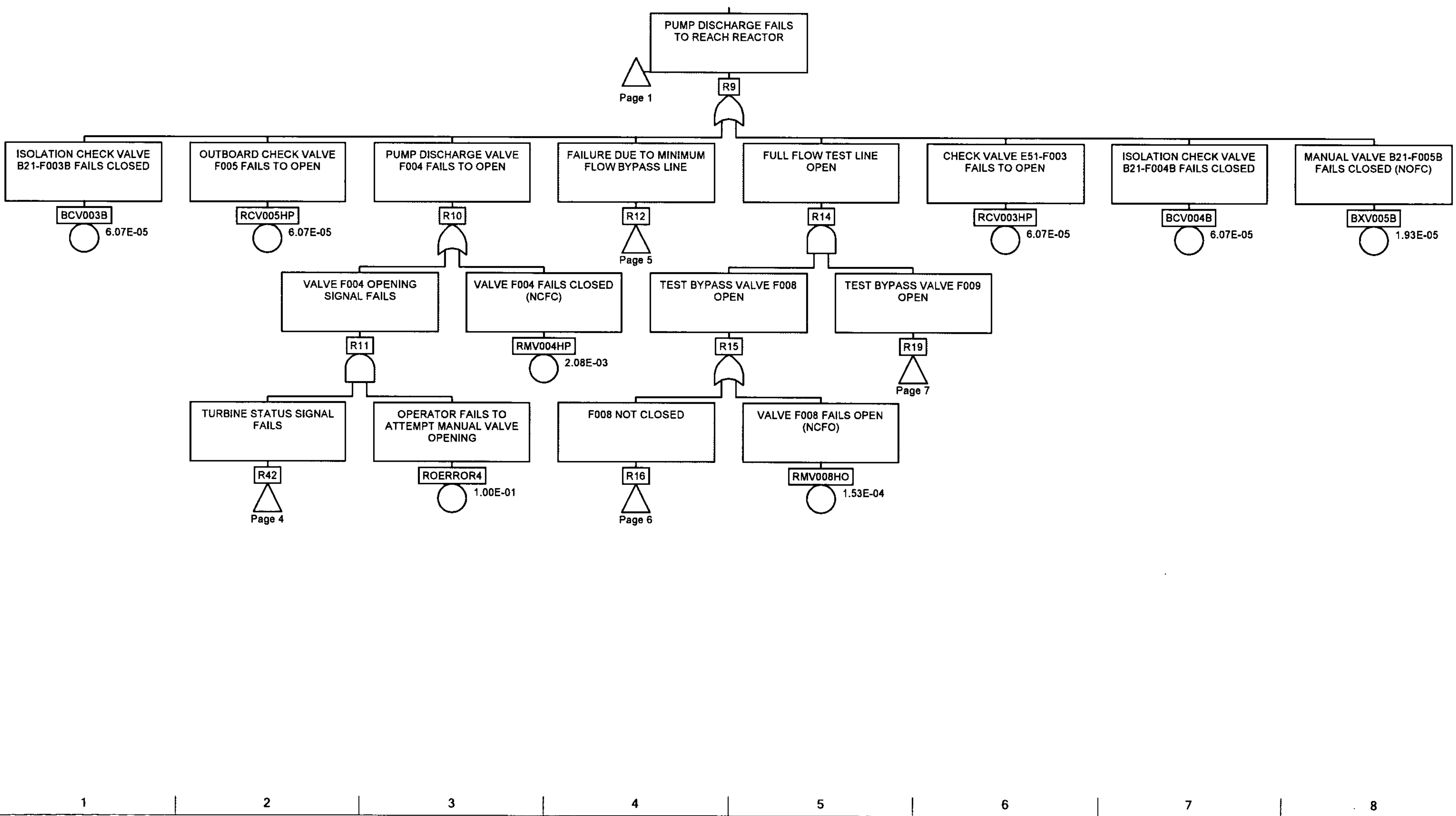
Figure 19D.6-1 Emergency Core Cooling System Fault Tree

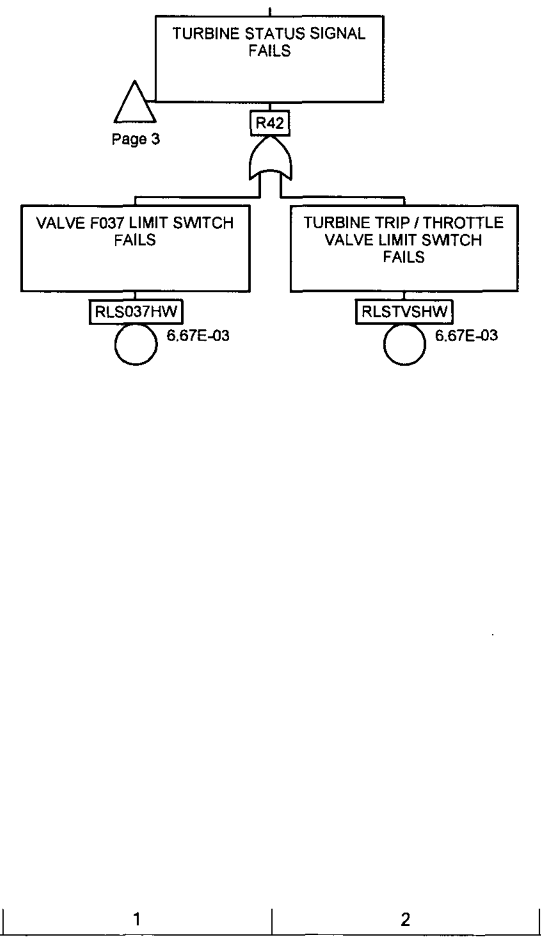


19D.6-2 Reactor Core Isolation Cooling System Fault Tree

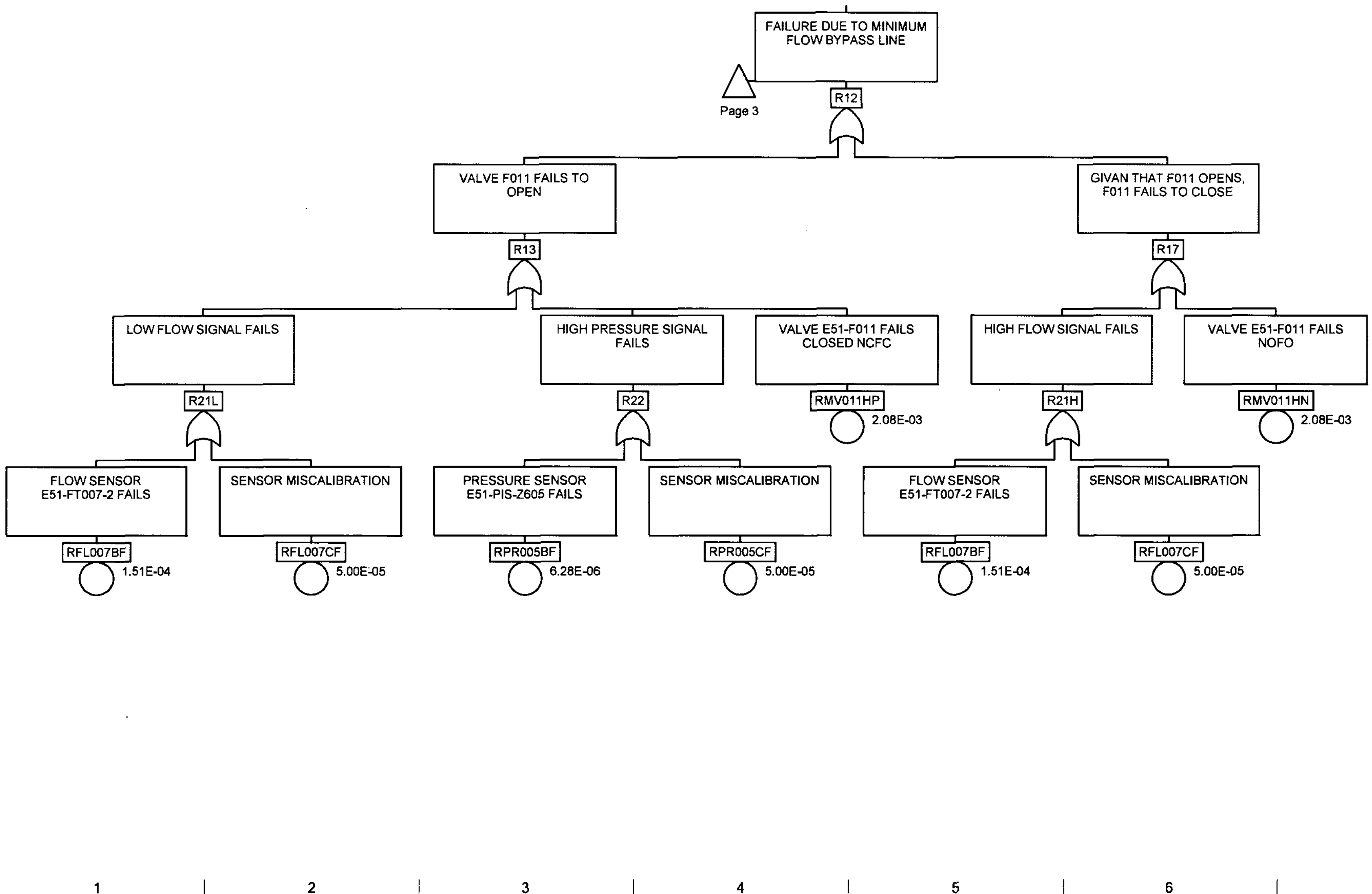


19D.6-2 Reactor Core Isolation Cooling System Fault Tree

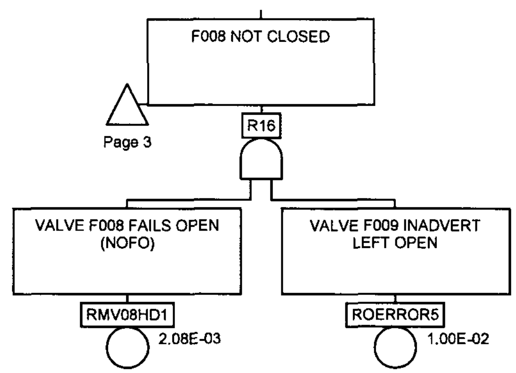

PUMP DISCHARGE FAILS TO REACH REACTOR
Page 1
R9
ISOLATION CHECK VALVE B21-F003B FAILS CLOSED
BCV003B
6.07E-05
OUTBOARD CHECK VALVE F005 FAILS TO OPEN
RCV005HP
6.07E-05
PUMP DISCHARGE VALVE F004 FAILS TO OPEN
R10
FAILURE DUE TO MINIMUM FLOW BYPASS LINE
R12
Page 5
FULL FLOW TEST LINE OPEN
R14
CHECK VALVE E51-F003 FAILS TO OPEN
RCV003HP
6.07E-05
ISOLATION CHECK VALVE B21-F004B FAILS CLOSED
BCV004B
6.07E-05
MANUAL VALVE B21-F005B FAILS CLOSED (NOFC)
BXV005B
1.93E-05
VALVE F004 OPENING SIGNAL FAILS
R11
VALVE F004 FAILS CLOSED (NCFC)
RMV004HP
2.08E-03
TEST BYPASS VALVE F008 OPEN
R15
TEST BYPASS VALVE F009 OPEN
R19
Page 7
TURBINE STATUS SIGNAL FAILS
R42
Page 4
OPERATOR FAILS TO ATTEMPT MANUAL VALVE OPENING
ROERROR4
1.00E-01
F008 NOT CLOSED
R16
Page 6
VALVE F008 FAILS OPEN (NCFO)
RMV008HO
1.53E-04
1
2
3
4
5
6
7
8



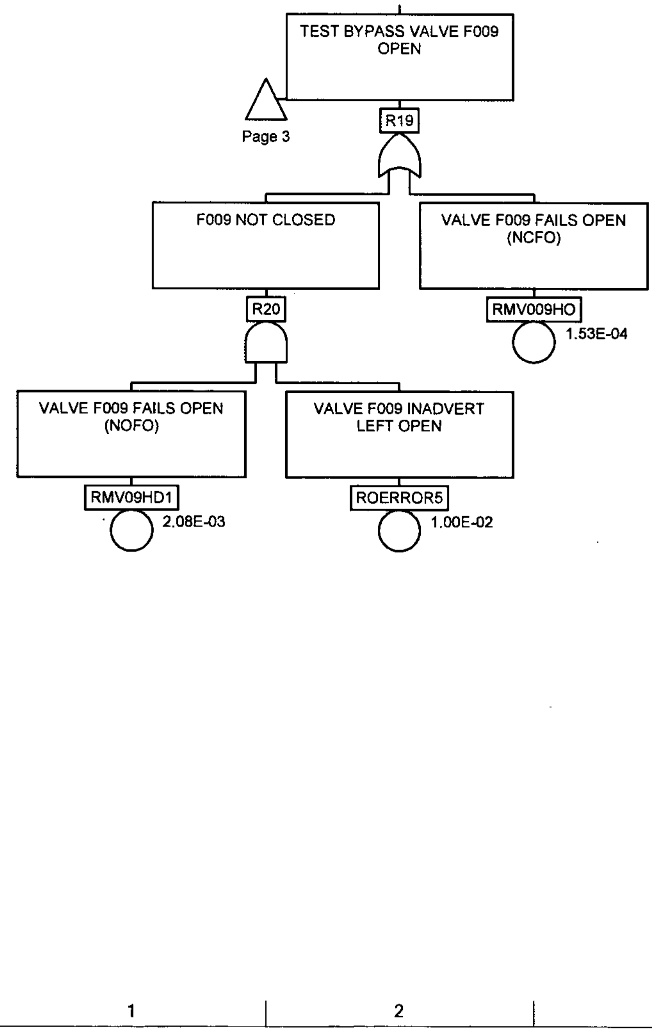
19D.6-2 Reactor Core Isolation Cooling System Fault Tree

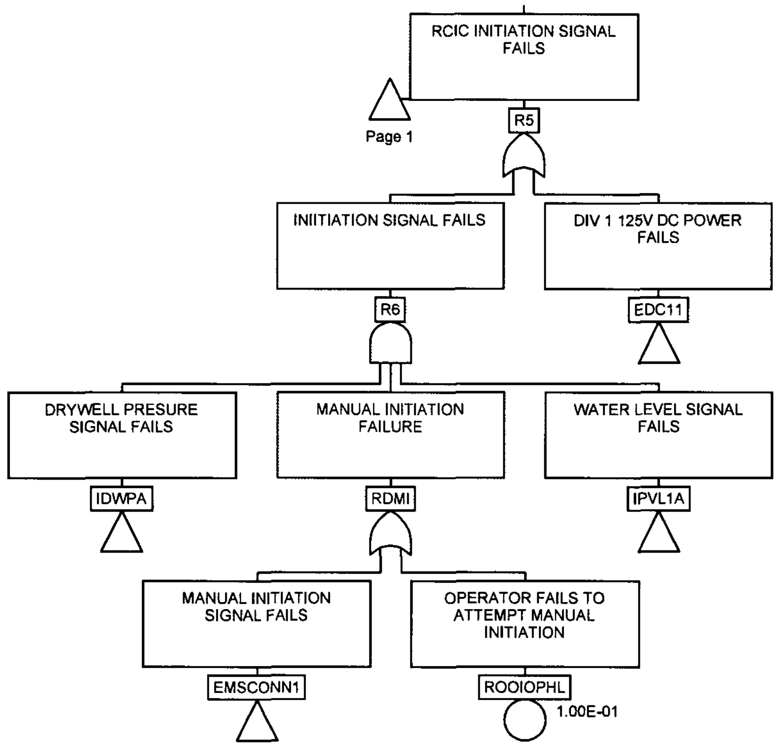


19D.6-2 Reactor Core Isolation Cooling System Fault Tree

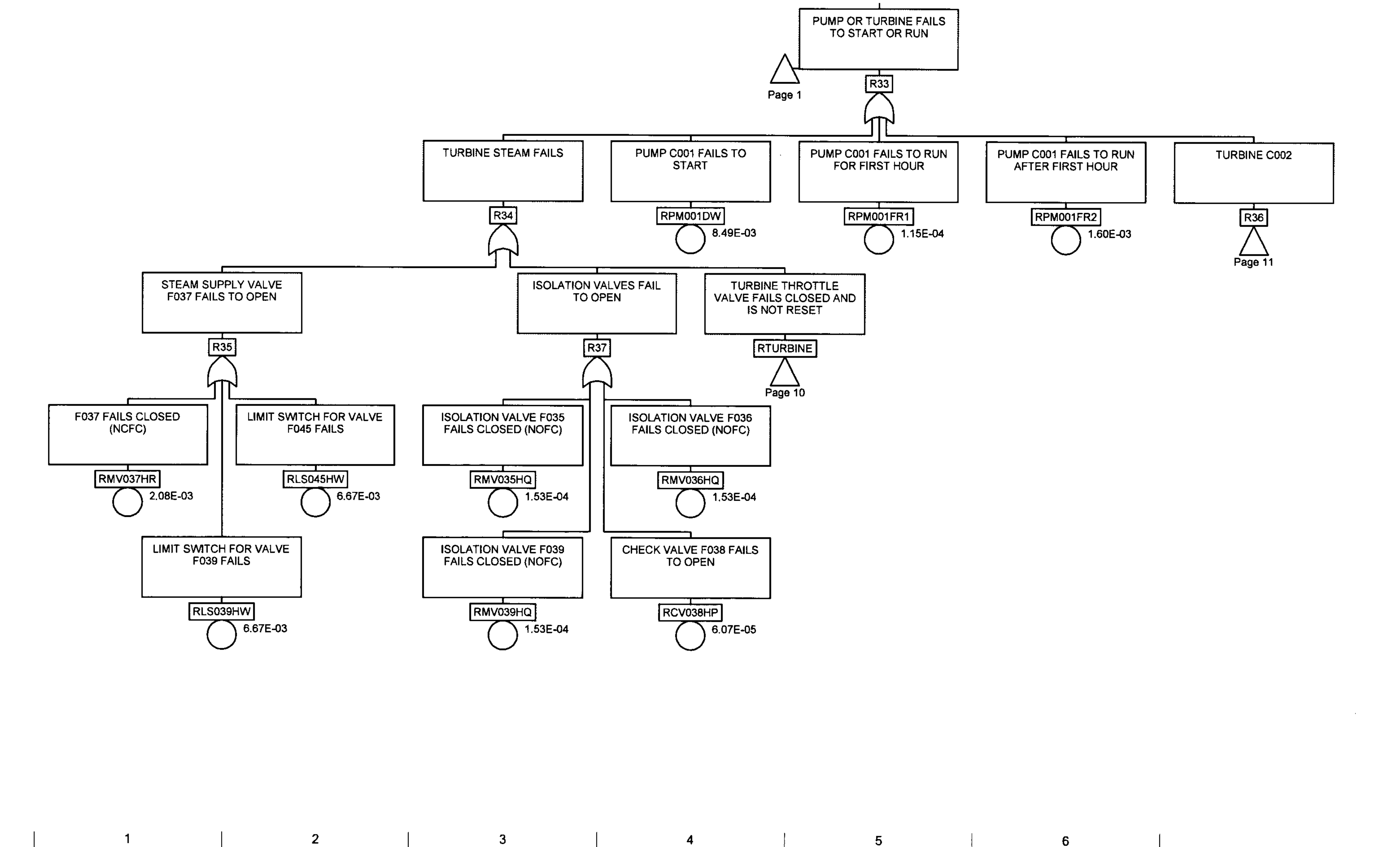

F008 NOT CLOSED
Page 3
R16
VALVE F008 FAILS OPEN (NOFO)
VALVE F009 INADVERT LEFT OPEN
RMV08HD1
2.08E-03
ROERROR5
1.00E-02

1	2

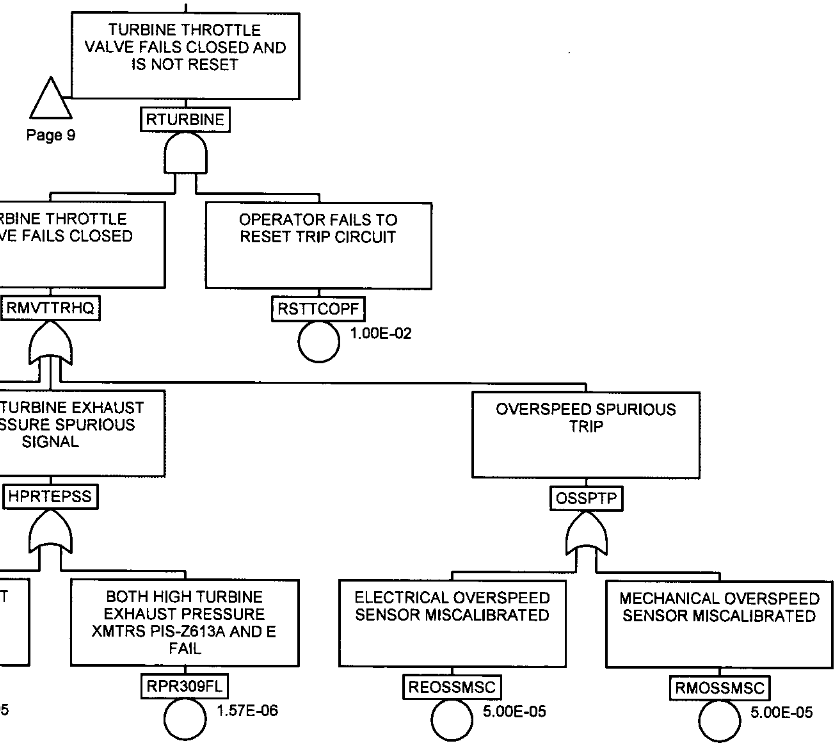

TEST BYPASS VALVE F009 OPEN
Page 3
R19
F009 NOT CLOSED
VALVE F009 FAILS OPEN (NCFO)
R20
RMV009HO
1.53E-04
VALVE F009 FAILS OPEN (NOFO)
VALVE F009 INADVERT LEFT OPEN
RMV09HD1
2.08E-03
ROERROR5
1.00E-02
1
2



19D.6-2 Reactor Core Isolation Cooling System Fault Tree

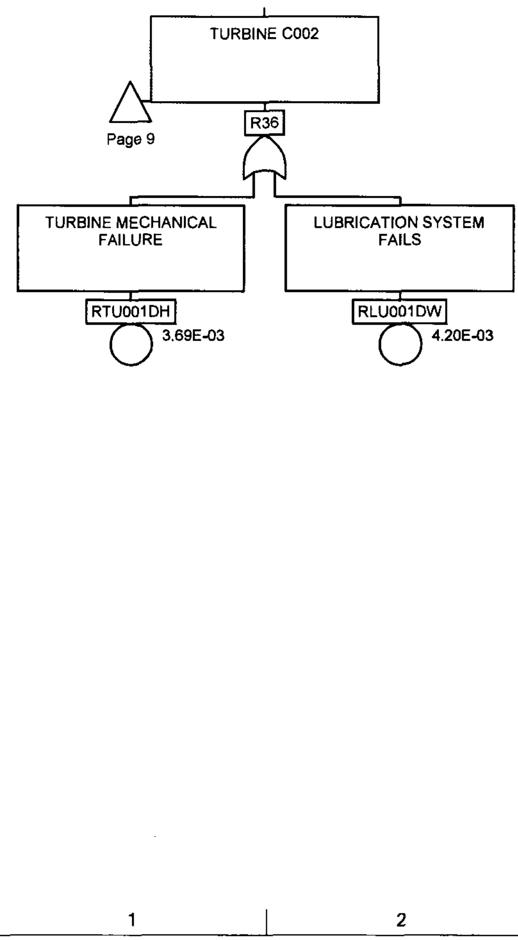


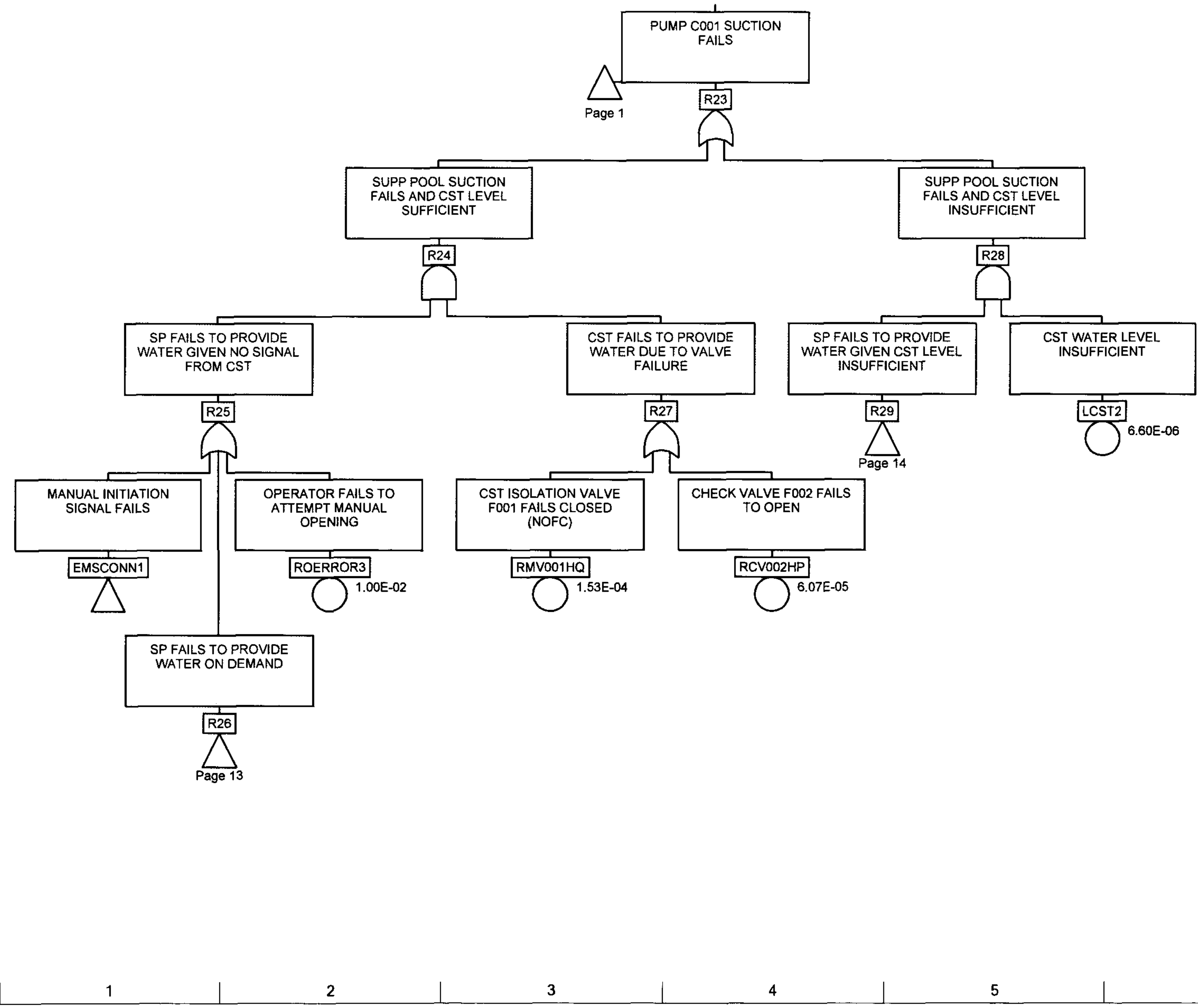
19D.6-2 Reactor Core Isolation Cooling System Fault Tree



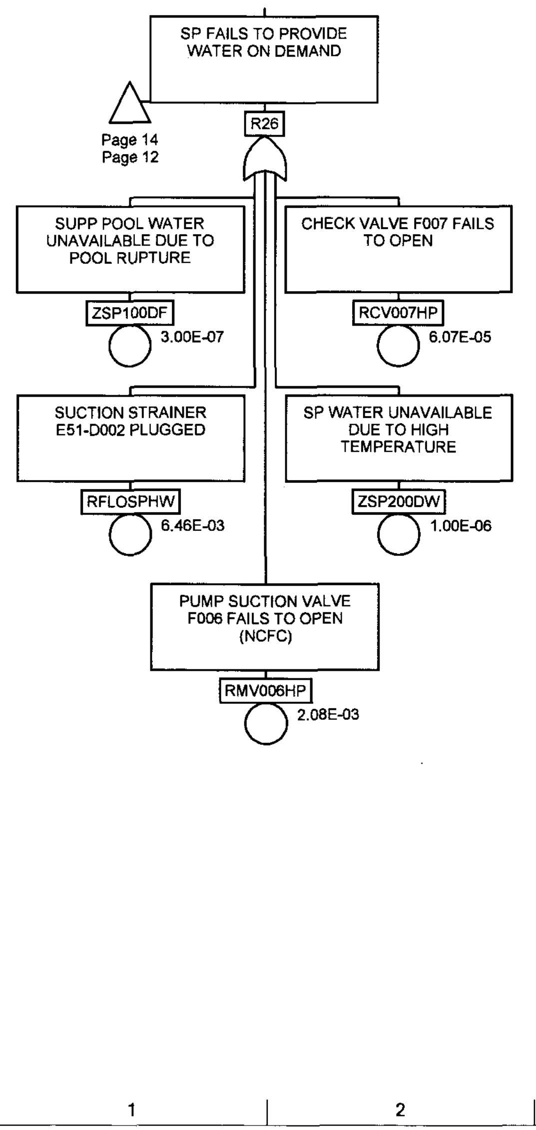
1 2 3 4 5 6

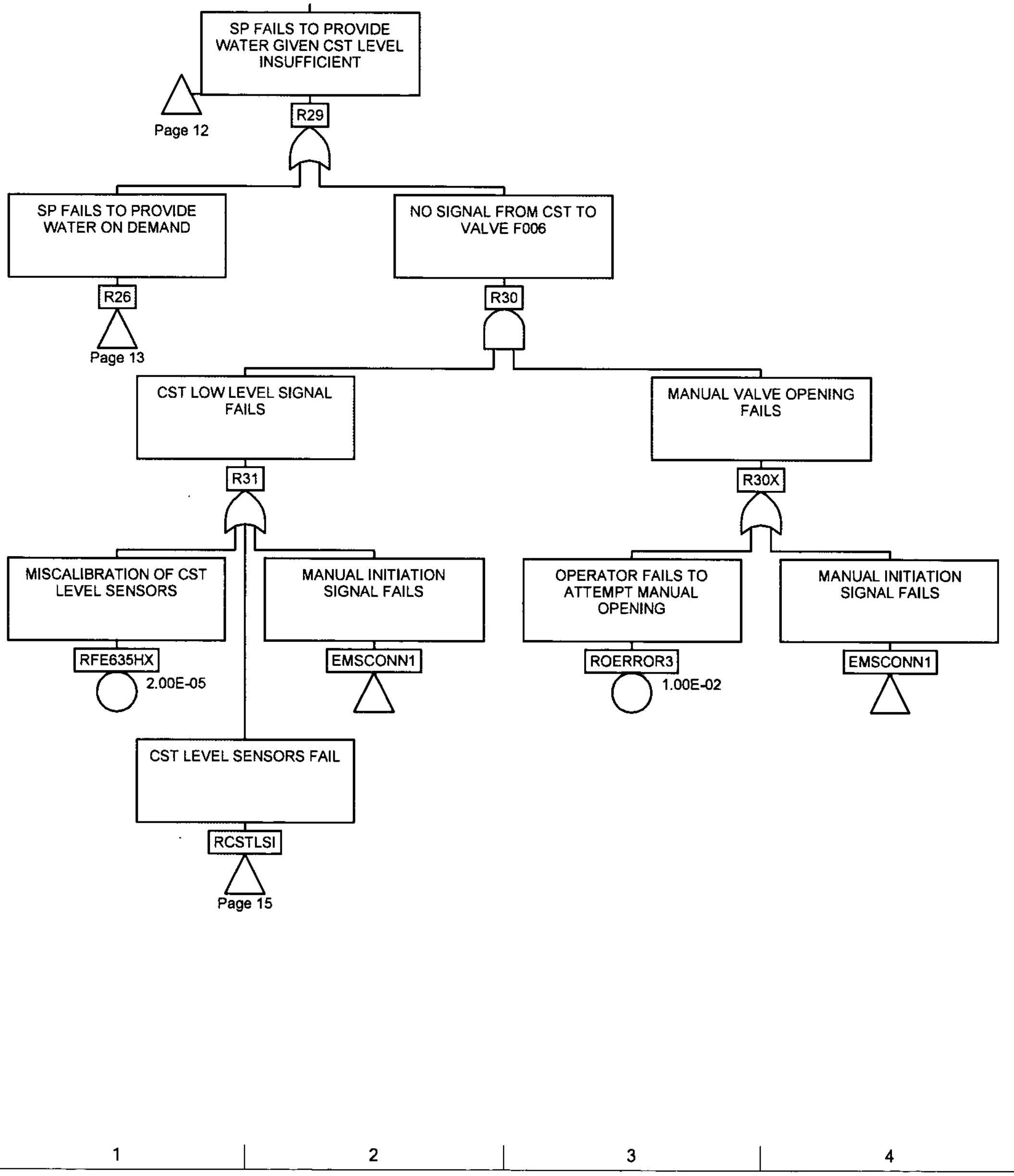
19D.6-2 Reactor Core Isolation Cooling System Fault Tree


TURBINE C002
Page 9
R36
TURBINE MECHANICAL FAILURE
RTU001DH
3.69E-03
LUBRICATION SYSTEM FAILS
RLU001DW
4.20E-03
1
2

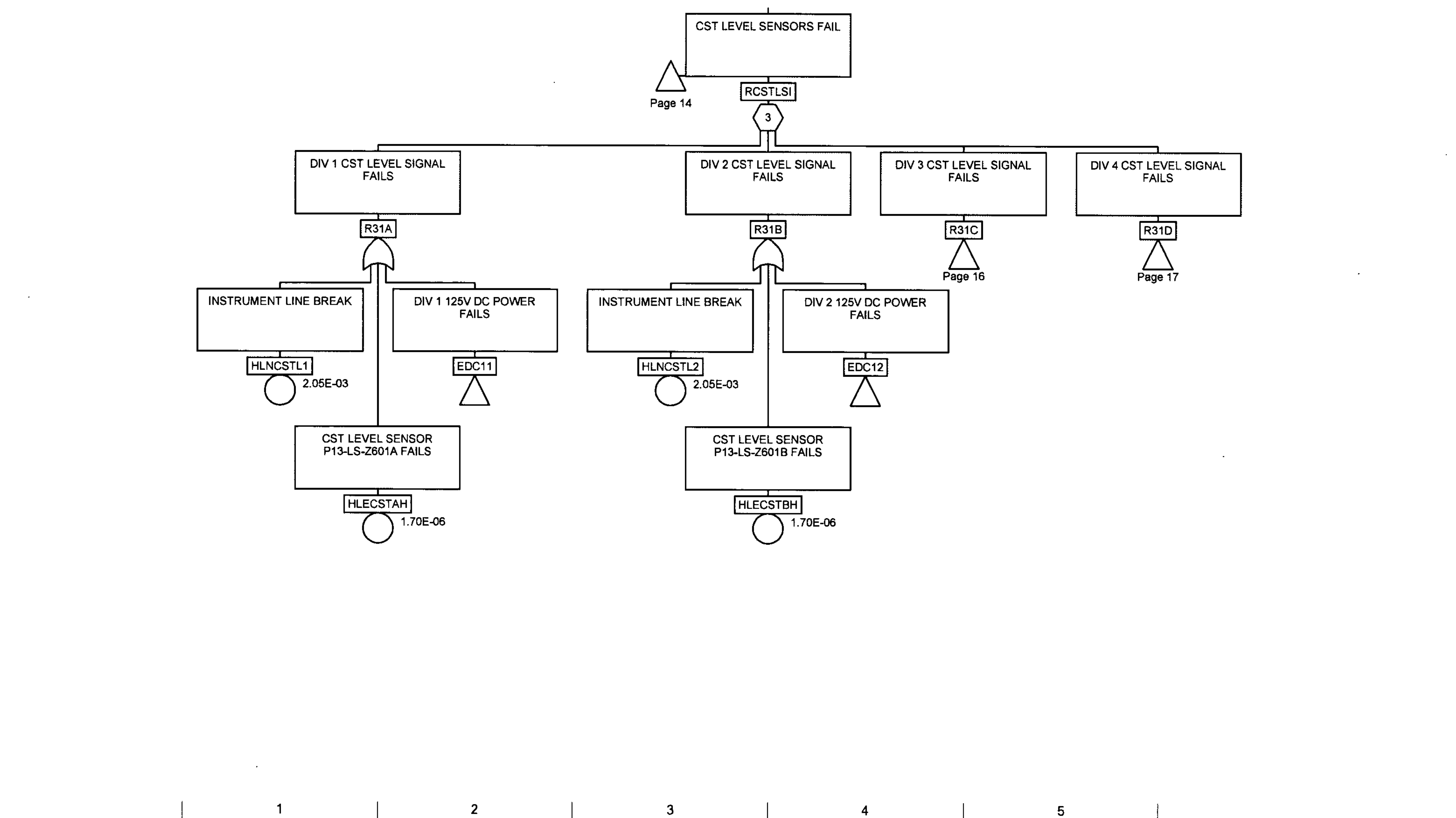


19D.6-2 Reactor Core Isolation Cooling System Fault Tree

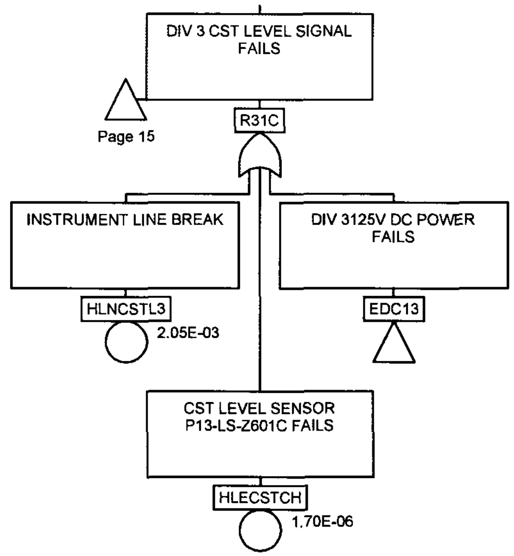

SP FAILS TO PROVIDE WATER ON DEMAND
Page 14
Page 12
R26
SUPP POOL WATER UNAVAILABLE DUE TO POOL RUPTURE
ZSP100DF
3.00E-07
CHECK VALVE F007 FAILS TO OPEN
RCV007HP
6.07E-05
SUCTION STRAINER E51-D002 PLUGGED
RFLOSPHW
6.46E-03
SP WATER UNAVAILABLE DUE TO HIGH TEMPERATURE
ZSP200DW
1.00E-06
PUMP SUCTION VALVE F006 FAILS TO OPEN (NCFC)
RMV006HP
2.08E-03
1
2



19D.6-2 Reactor Core Isolation Cooling System Fault Tree

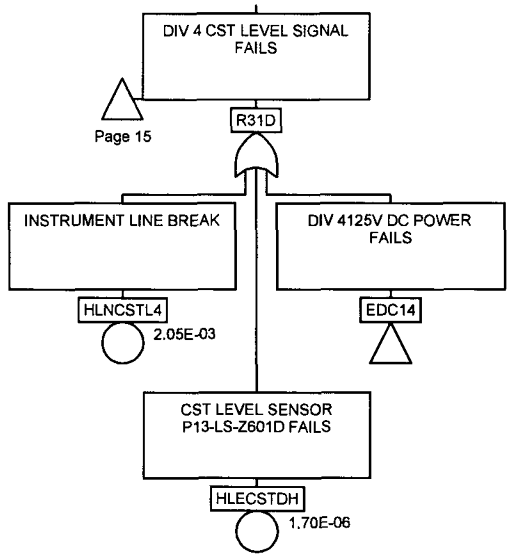


CST LEVEL SENSORS FAIL
Page 14
RCSTLSI
3
DIV 1 CST LEVEL SIGNAL FAILS
R31A
DIV 2 CST LEVEL SIGNAL FAILS
R31B
DIV 3 CST LEVEL SIGNAL FAILS
R31C
Page 16
DIV 4 CST LEVEL SIGNAL FAILS
R31D
Page 17
INSTRUMENT LINE BREAK
HLNCSTL1
2.05E-03
DIV 1 125V DC POWER FAILS
EDC11
INSTRUMENT LINE BREAK
HLNCSTL2
2.05E-03
DIV 2 125V DC POWER FAILS
EDC12
CST LEVEL SENSOR P13-LS-Z601A FAILS
HLECSTAH
1.70E-06
CST LEVEL SENSOR P13-LS-Z601B FAILS
HLECSTBH
1.70E-06
1
2
3
4
5



DIV 3 CST LEVEL SIGNAL FAILS
Page 15
R31C
INSTRUMENT LINE BREAK
DIV 3125V DC POWER FAILS
HLNCSTL3
2.05E-03
EDC13
CST LEVEL SENSOR P13-LS-Z601C FAILS
HLECSTCH
1.70E-06

1 2



DIV 4 CST LEVEL SIGNAL FAILS
Page 15
R31D
INSTRUMENT LINE BREAK
DIV 4125V DC POWER FAILS
HLNCSTL4
2.05E-03
EDC14
CST LEVEL SENSOR P13-LS-Z601D FAILS
HLECSTDH
1.70E-06

1 2

Name	Page	Zone	Name	Page	Zone
BCV003B	3	1	R23	12	4
BCV004B	3	7	R24	12	2
BXV005B	3	8	R25	12	2
EACE1	1	6	R26	12	2
EDC11	1	4	R26	13	2
EDC11	8	3	R26	14	1
EDC11	15	2	R27	12	4
EDC12	15	4	R28	12	5
EDC13	16	2	R29	12	5
EDC14	17	2	R29	14	2
EMSCONN1	8	2	R3	1	5
EMSCONN1	12	1	R30	14	2
EMSCONN1	14	2	R30X	14	4
EMSCONN1	14	4	R31	14	2
HFELEBHX	2	1	R31A	15	2
HLECSTAH	15	2	R31B	15	4
HLECSTBH	15	4	R31C	15	5
HLECSTCH	16	2	R31C	16	2
HLECSTDH	17	2	R31D	15	6
HLNCSTL1	15	1	R31D	17	2
HLNCSTL2	15	3	R33	1	2
HLNCSTL3	16	1	R33	9	5
HLNCSTL4	17	1	R34	9	3
HPRTEPSS	10	4	R35	9	2
IDWPA	8	1	R36	9	7
IPVL1A	8	3	R36	11	2
LCST2	12	6	R37	9	4
LSPSS	10	2	R4	1	1
OSSPTP	10	6	R4	2	2
R1	1	2	R42	3	2
R10	3	3	R42	4	2
R11	3	3	R5	1	2
R12	3	4	R5	8	3
R12	5	4	R6	8	2
R13	5	3	R9	1	1
R14	3	5	R9	3	5
R15	3	5	RCIC	1	4
R16	3	4	RCIMAINT	1	3
R16	6	2	RCSTLSI	14	2
R17	5	6	RCSTLSI	15	3
R19	3	6	RCV002HP	12	4
R19	7	2	RCV003HP	3	6
R2	1	2	RCV005HP	3	2
R20	7	2	RCV007HP	13	2
R21H	5	6	RCV038HP	9	4
R21L	5	2	RDMI	8	2
R22	5	4	REOSSMSC	10	5
R23	1	2	RFE635HX	14	1

Name	Page	Zone	Name	Page	Zone
RFL007BF	5	1	ZSP100DF	13	1
RFL007BF	5	5	ZSP200DW	13	2
RFL007CF	5	2			
RFL007CF	5	6			
RFLOSPHW	13	1			
RISOLSIG	2	2			
RLS037HW	4	1			
RLS039HW	9	2			
RLS045HW	9	2			
RLSTVSHW	4	2			
RLU001DW	11	2			
RMOSSMSC	10	6			
RMV001HQ	12	3			
RMV004HP	3	4			
RMV006HP	13	2			
RMV008HO	3	5			
RMV009HO	7	3			
RMV011HN	5	7			
RMV011HP	5	5			
RMV035HQ	9	3			
RMV036HQ	9	4			
RMV037HR	9	1			
RMV039HQ	9	3			
RMV08HD1	6	1			
RMV09HD1	7	1			
RMVTTRHQ	10	3			
ROERROR3	12	2			
ROERROR3	14	3			
ROERROR4	3	3			
ROERROR5	6	2			
ROERROR5	7	2			
ROOIOPHL	8	3			
RPM001DW	9	4			
RPM001FR1	9	5			
RPM001FR2	9	6			
RPR005BF	5	3			
RPR005CF	5	4			
RPR303FL	10	2			
RPR303MC	10	1			
RPR309FL	10	4			
RPR309MC	10	3			
RRMCOOL	1	5			
RRMCOOL1	1	4			
RSTTCOPF	10	4			
RTU001DH	11	1			
RTURBINE	9	5			
RTURBINE	10	4			
WRCWA	1	5			

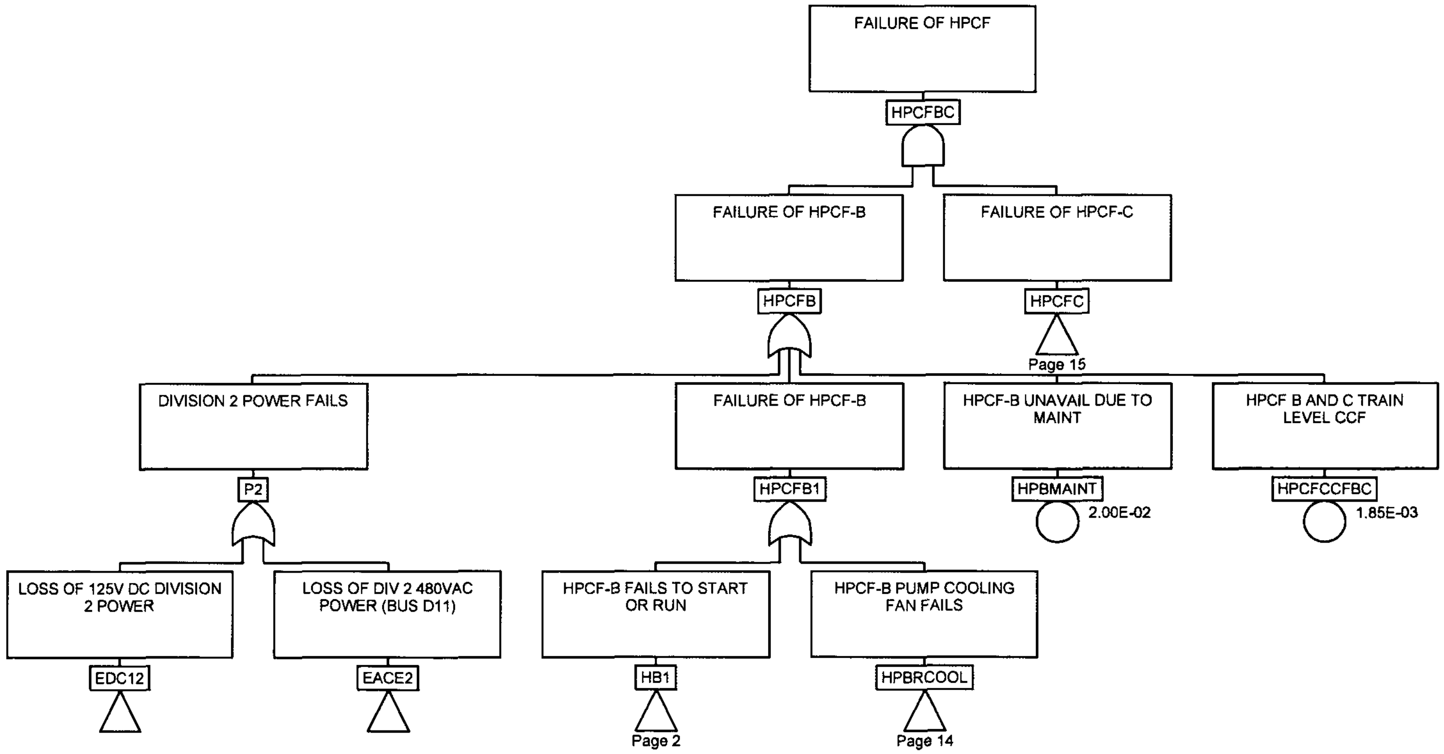


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

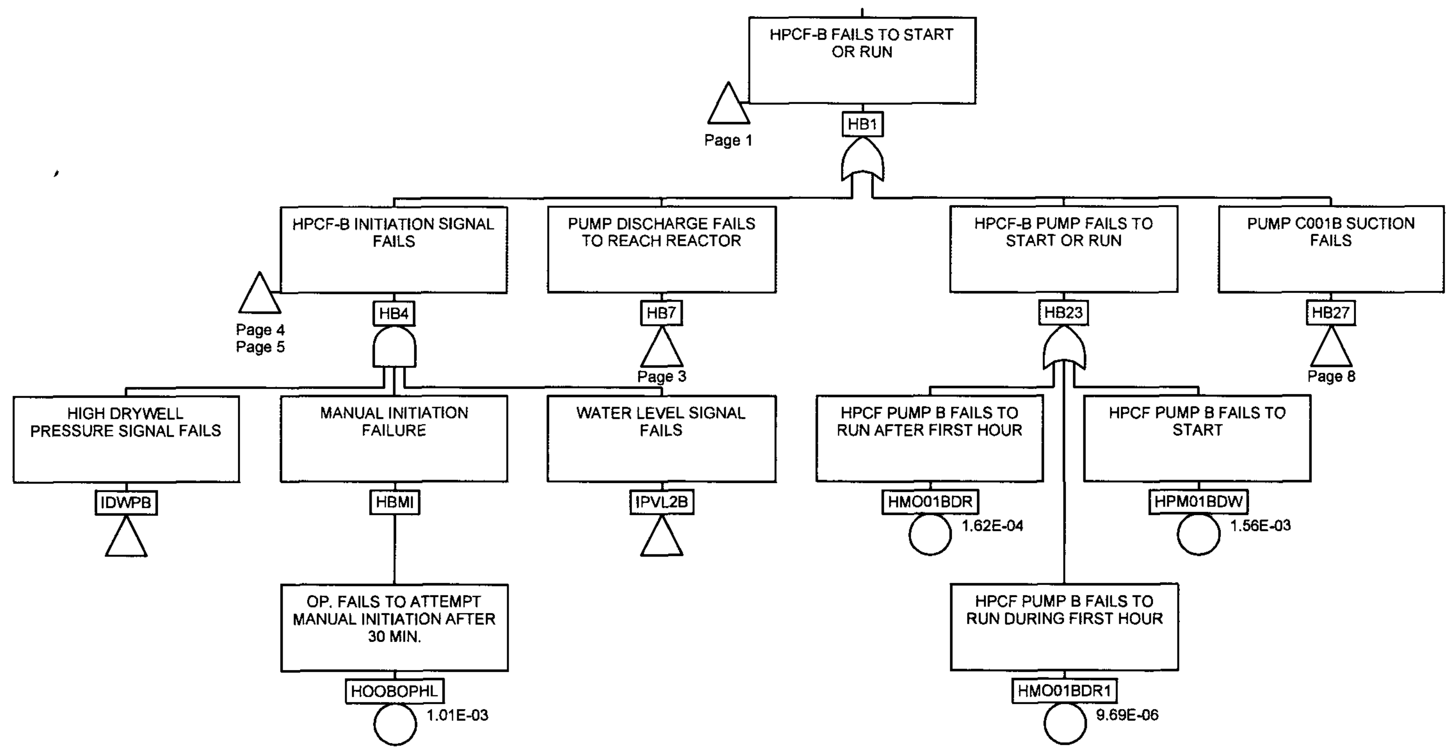


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

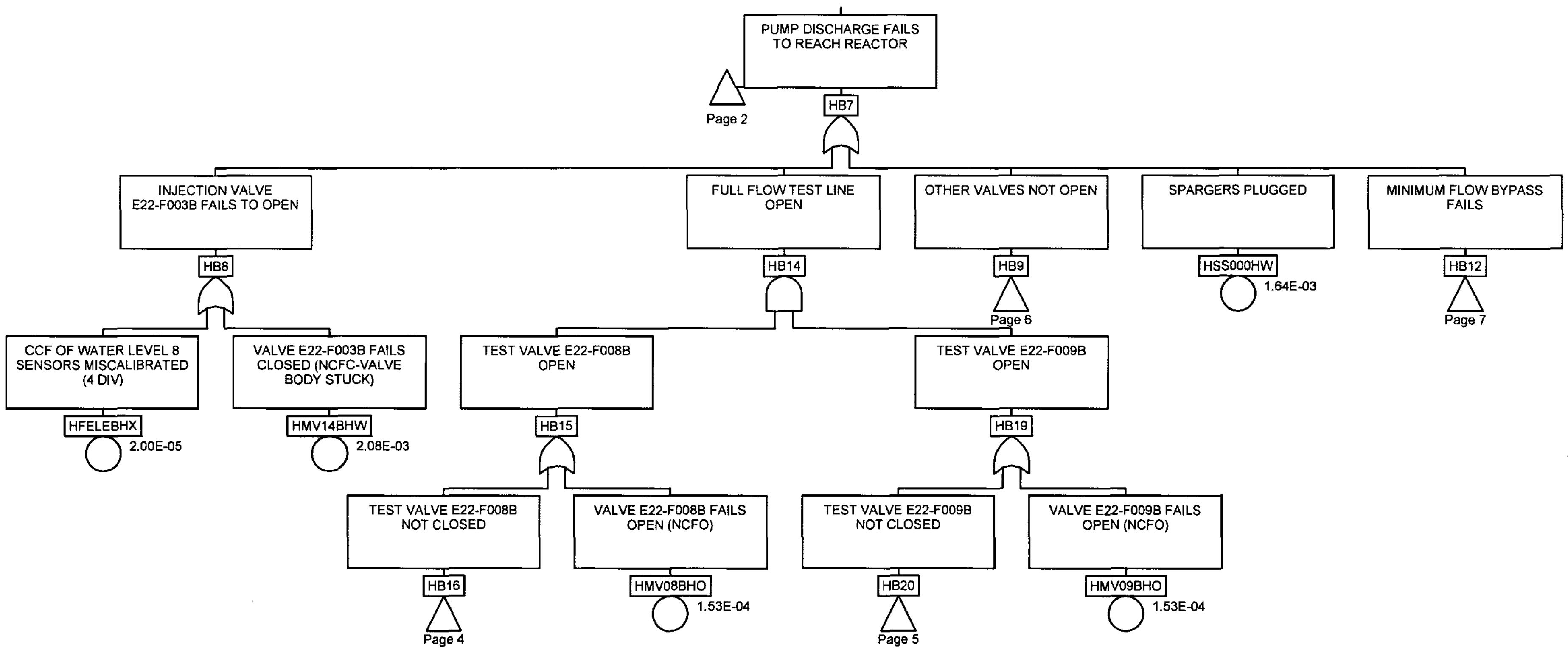


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

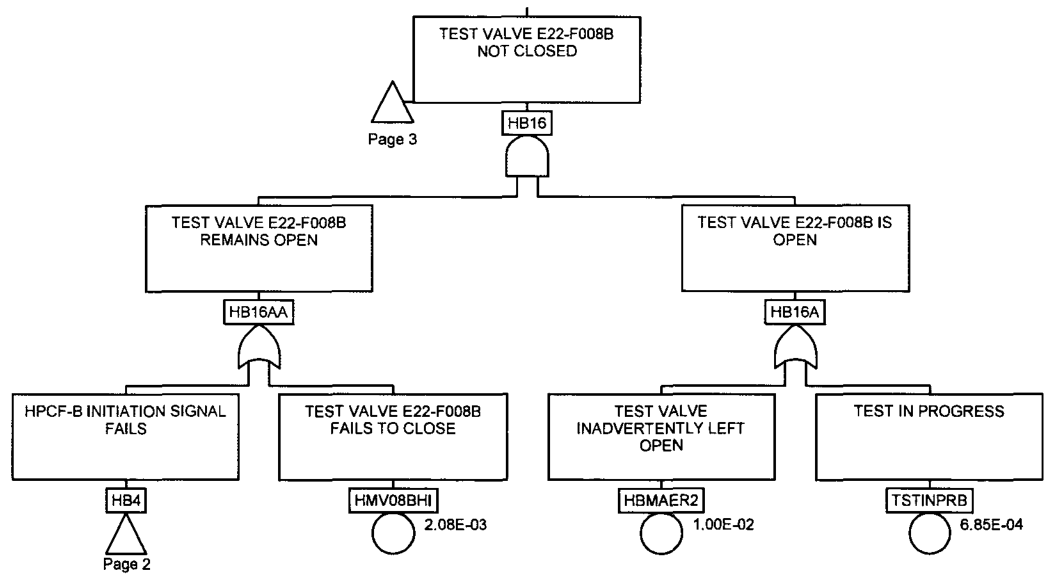
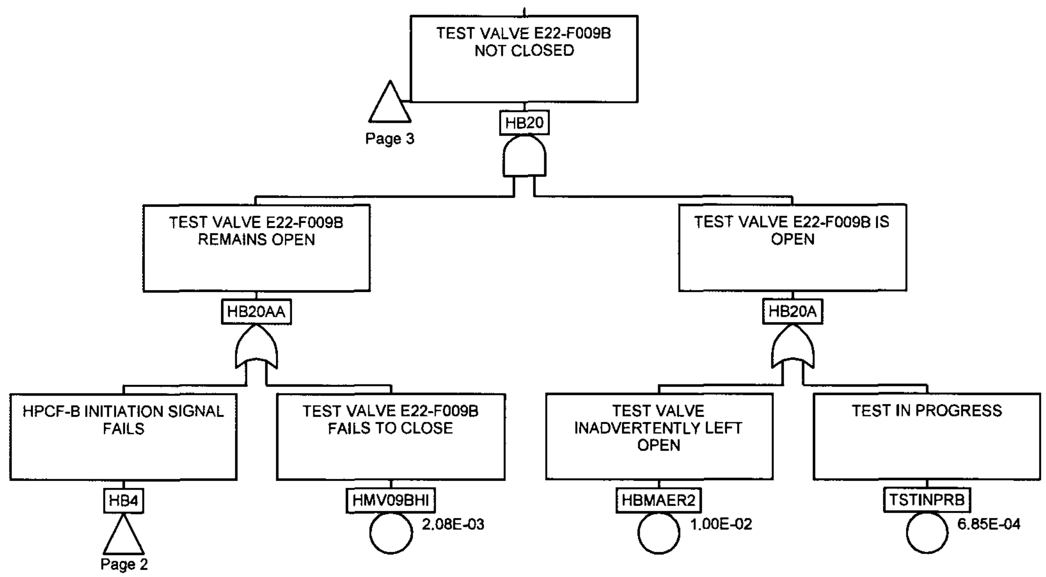


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree


TEST VALVE E22-F009B NOT CLOSED
Page 3
HB20
TEST VALVE E22-F009B REMAINS OPEN
TEST VALVE E22-F009B IS OPEN
HB20AA
HB20A
HPCF-B INITIATION SIGNAL FAILS
TEST VALVE E22-F009B FAILS TO CLOSE
TEST VALVE INADVERTENTLY LEFT OPEN
TEST IN PROGRESS
HB4
Page 2
HMV09BHI
2.08E-03
HBMAER2
1.00E-02
TSTINPRB
6.85E-04

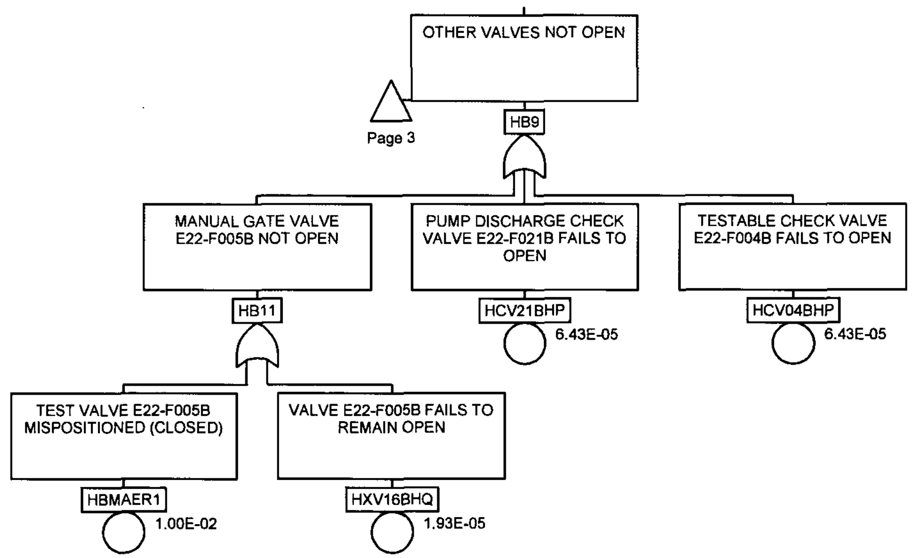


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

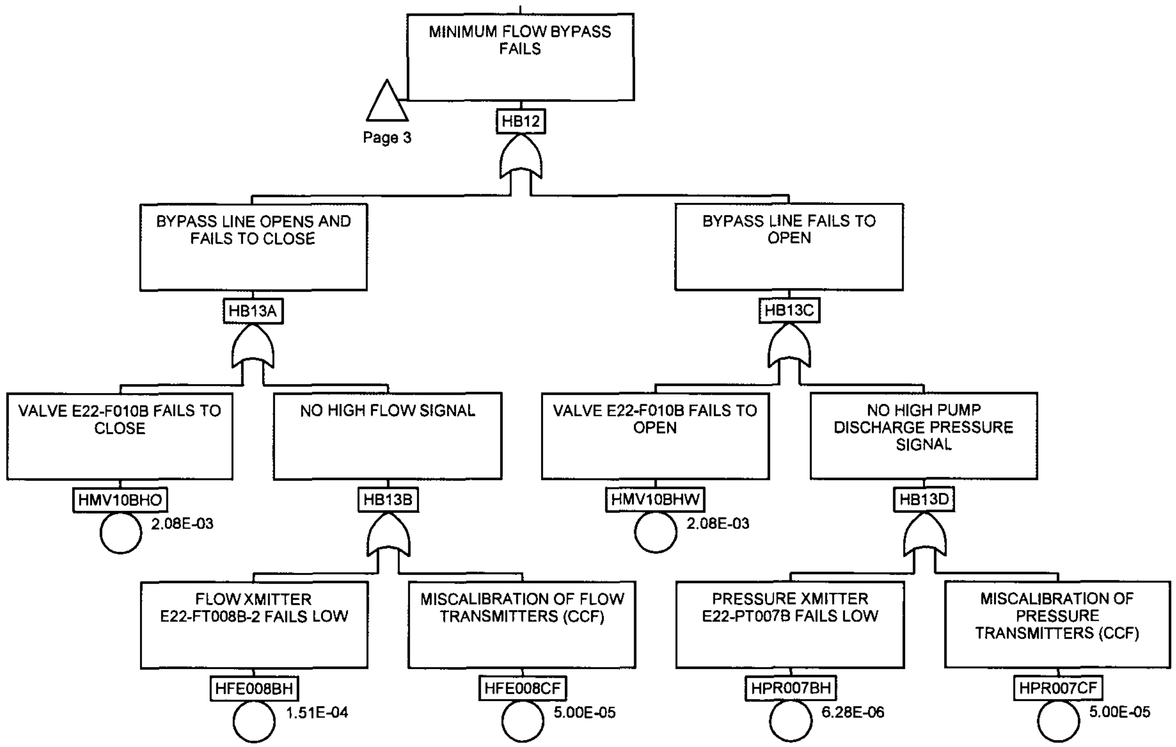


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

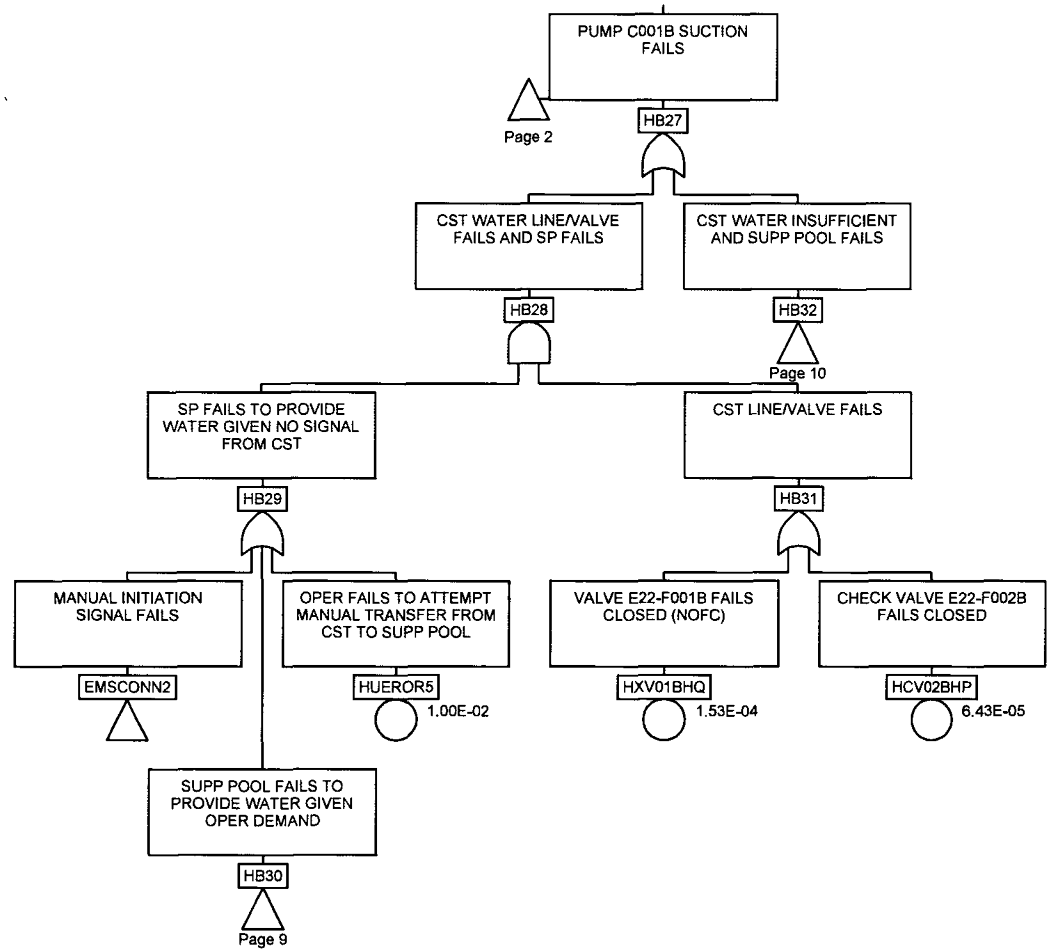


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

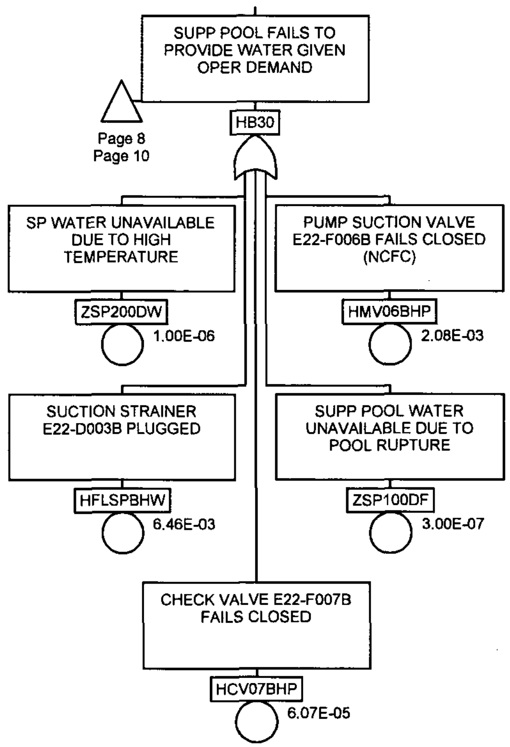


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

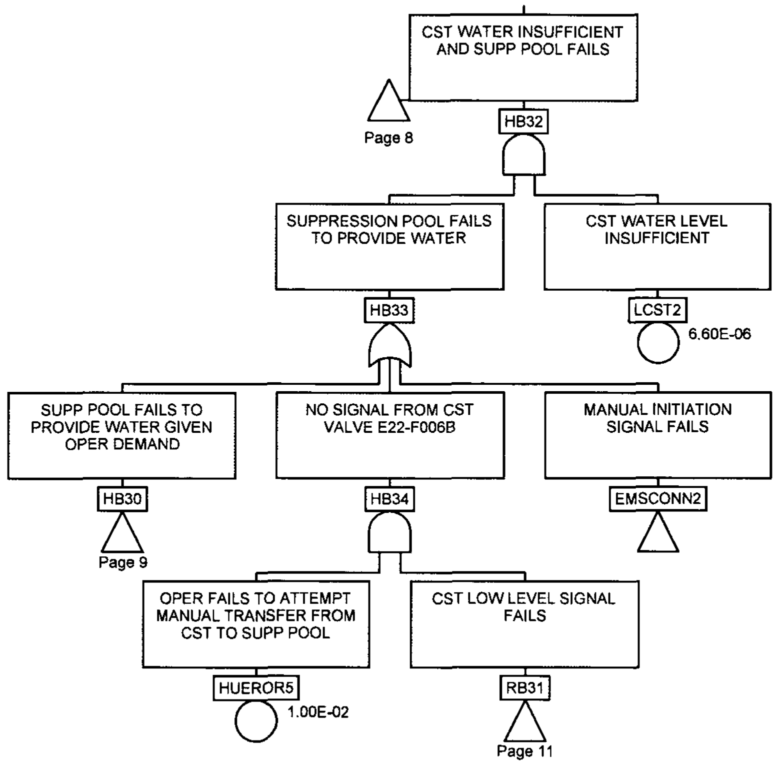


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

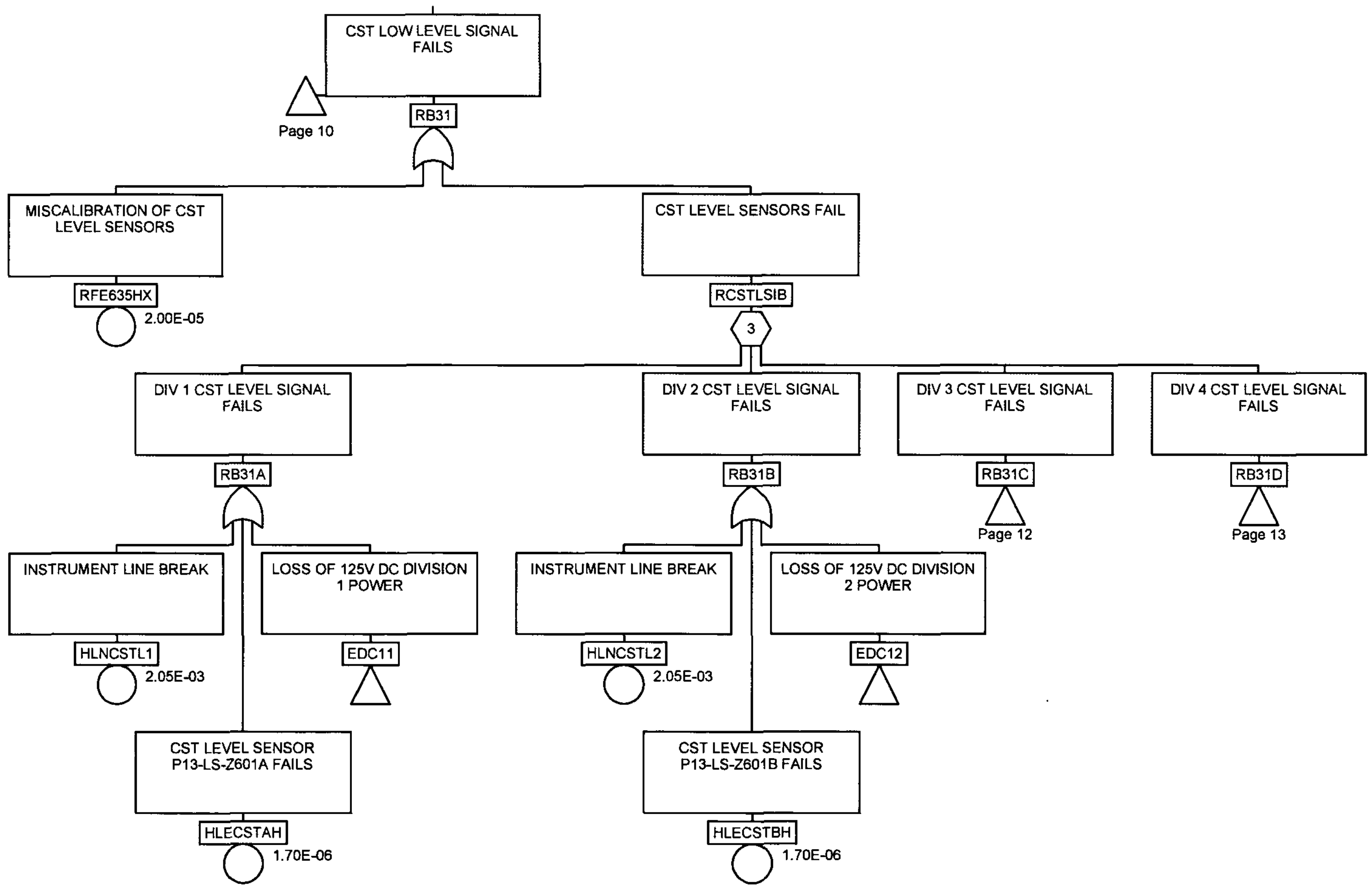


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

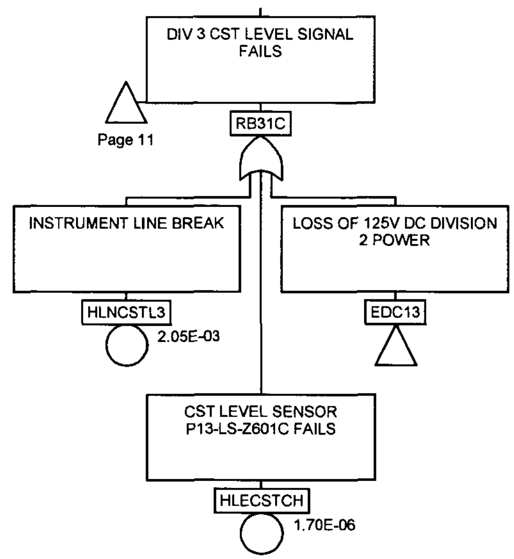


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

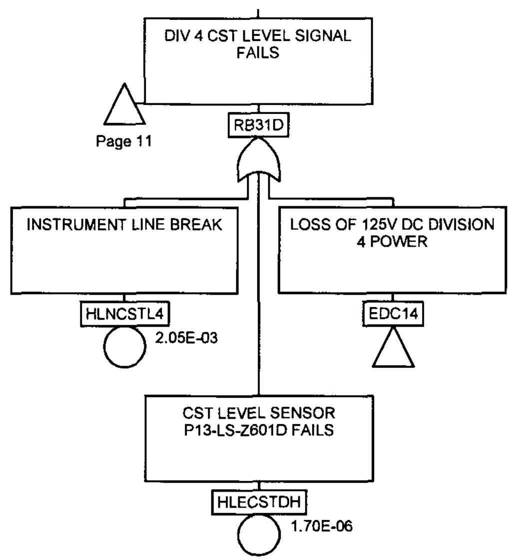


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

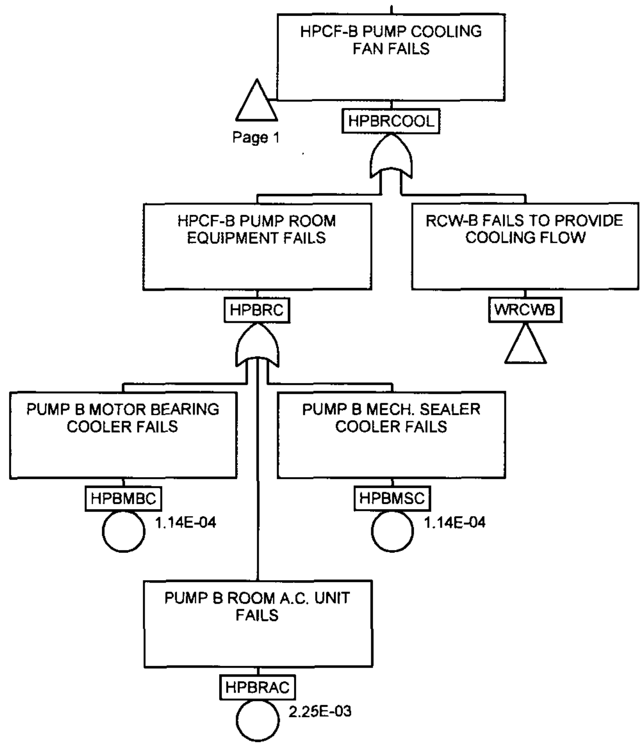


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

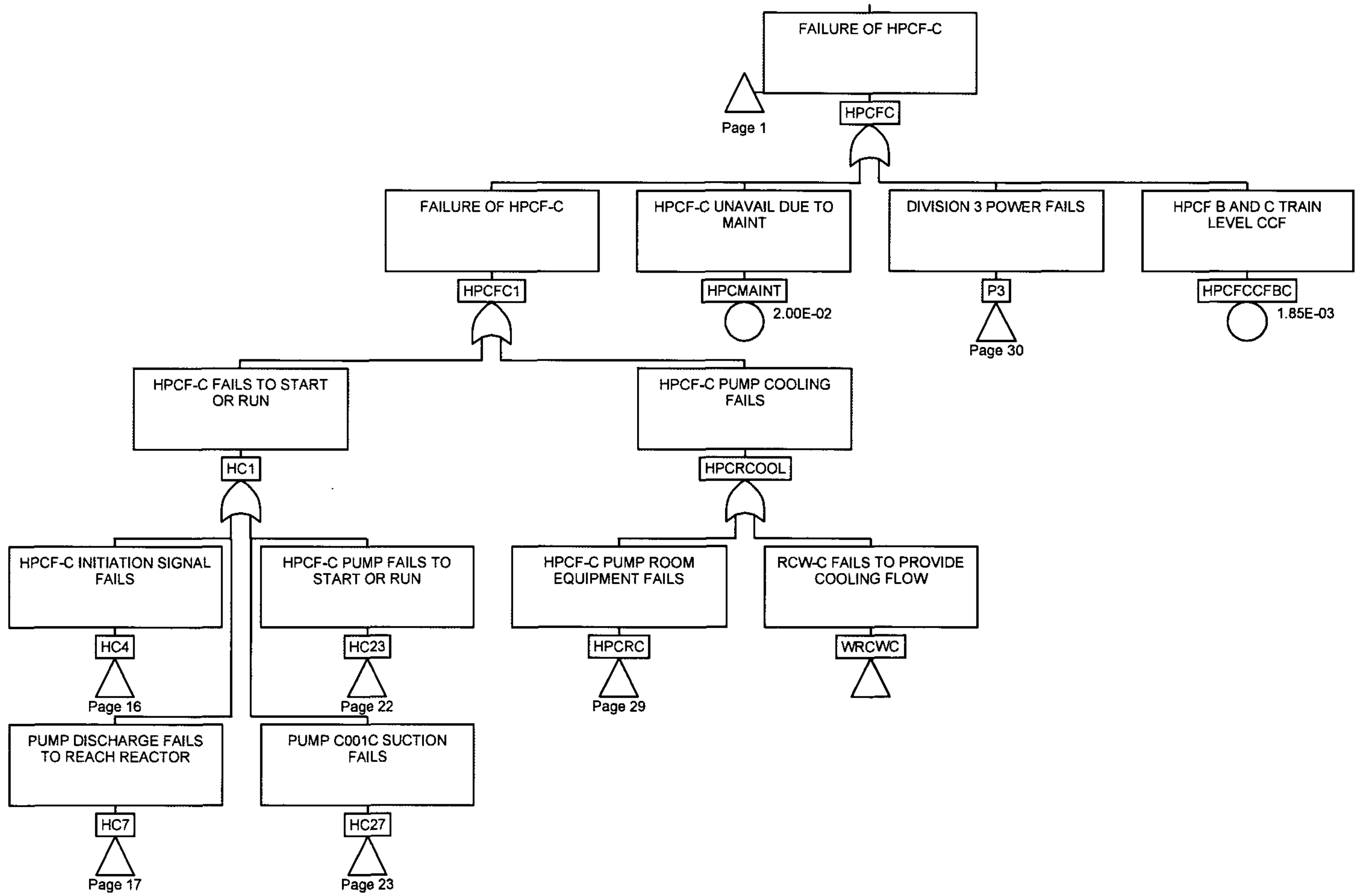


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

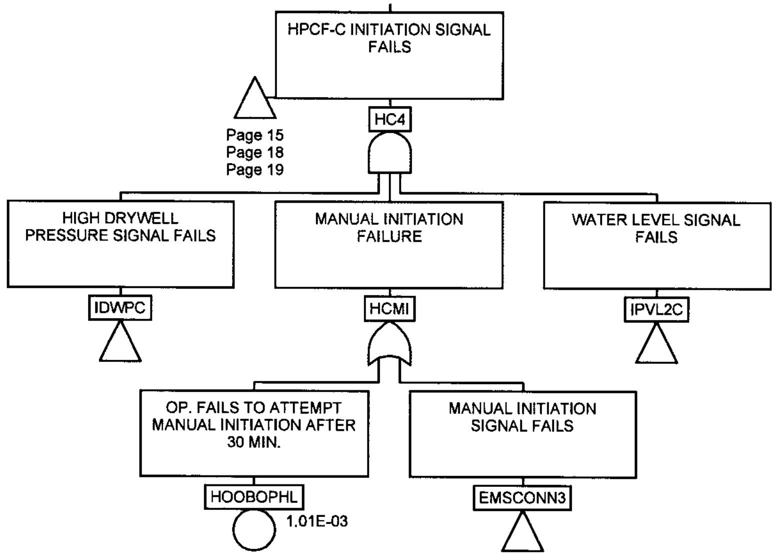
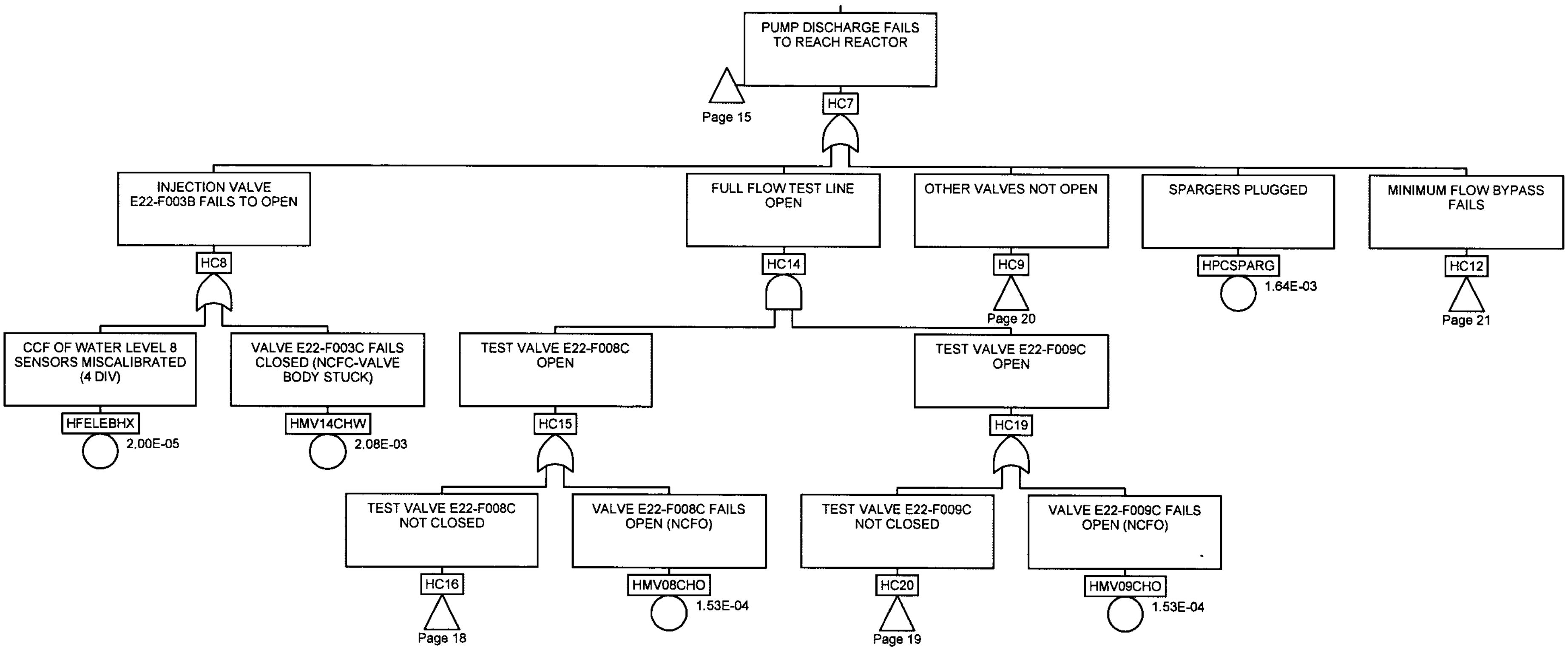


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree



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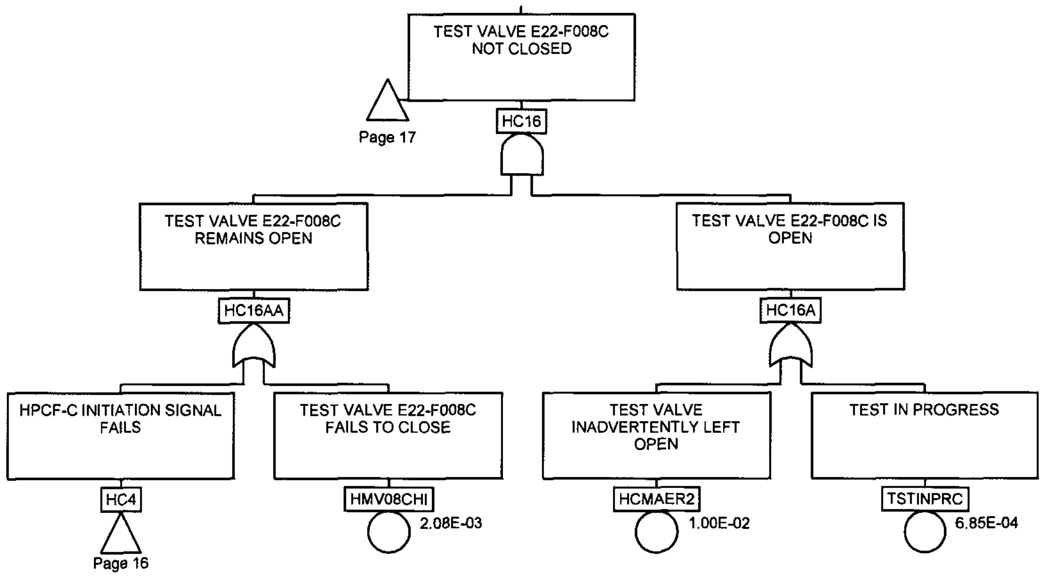

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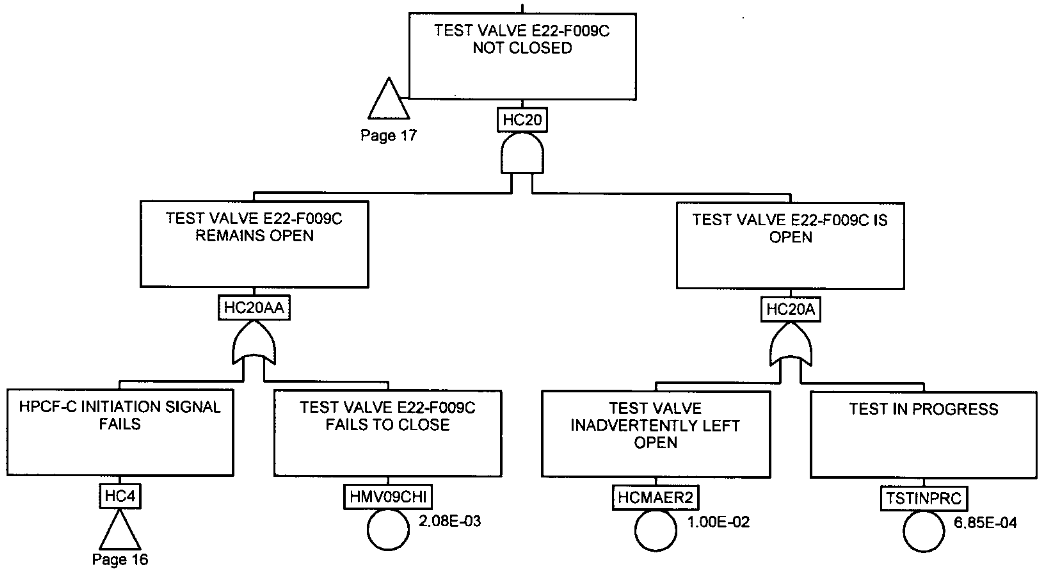

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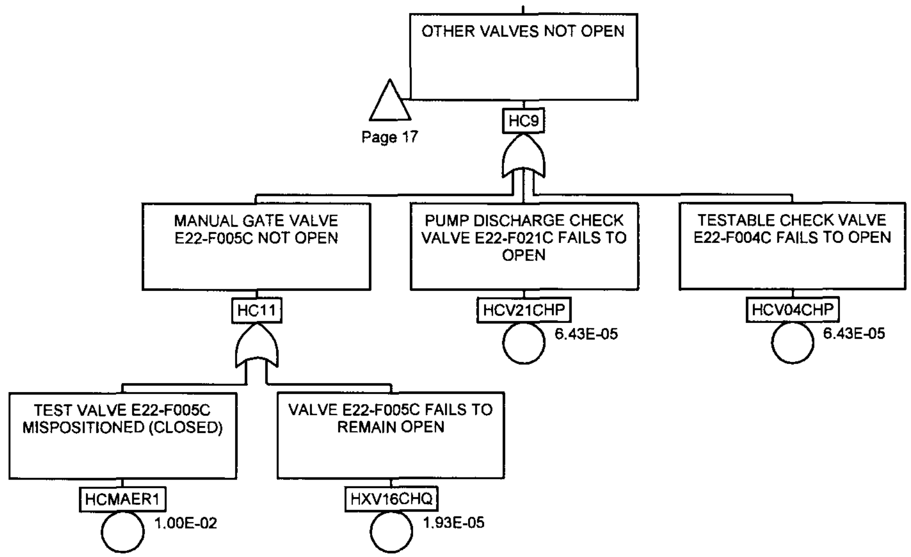

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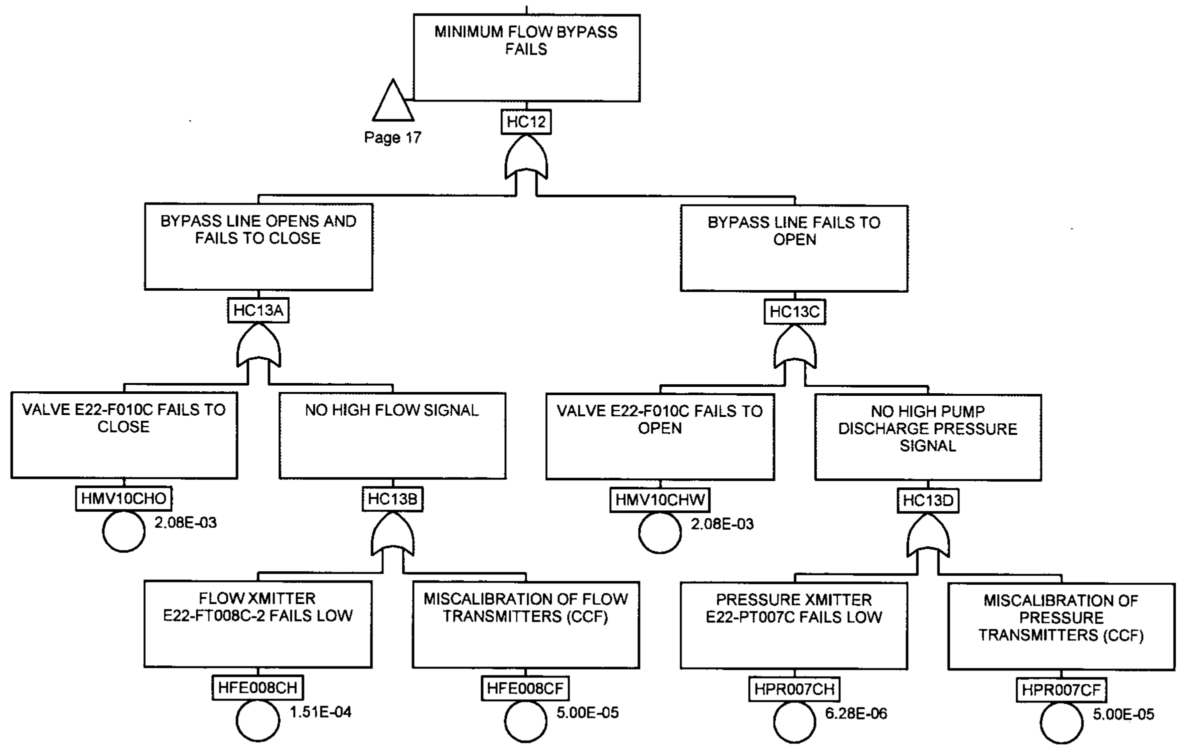

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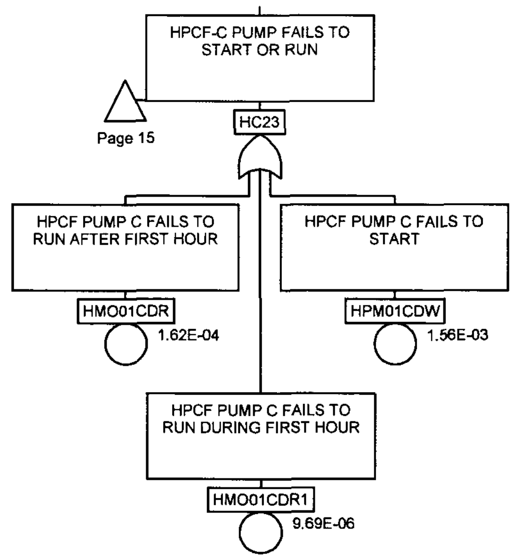

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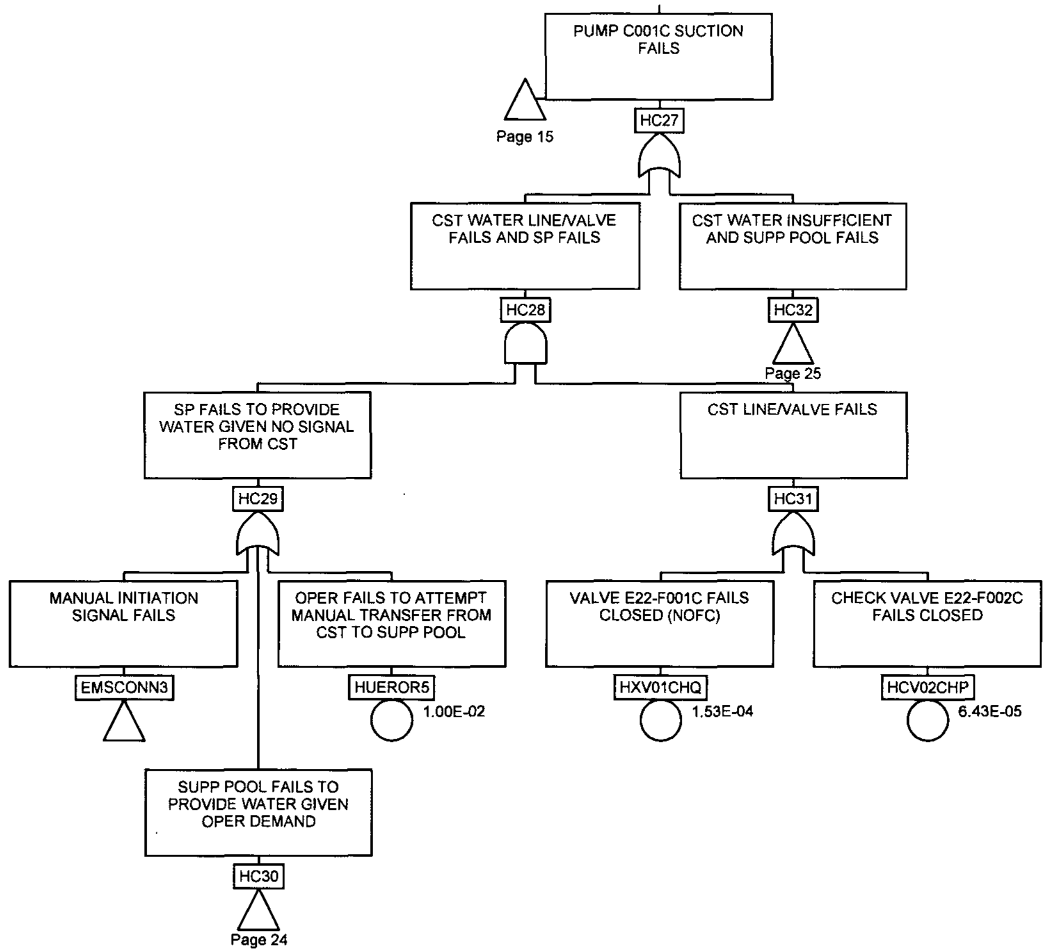


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

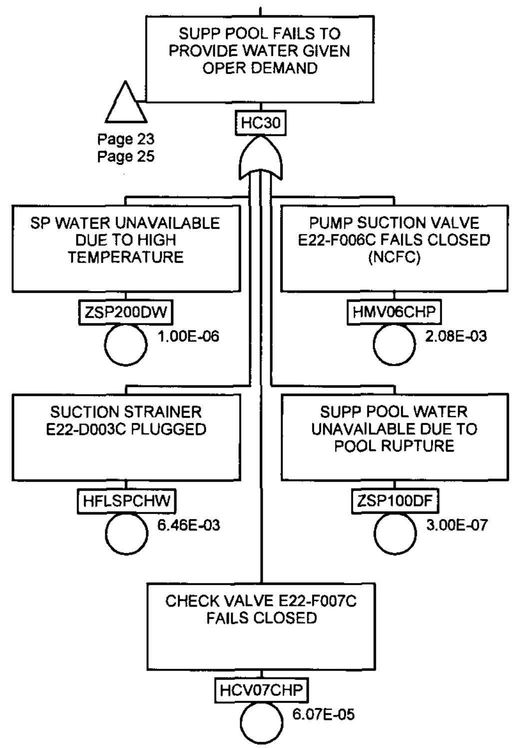


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

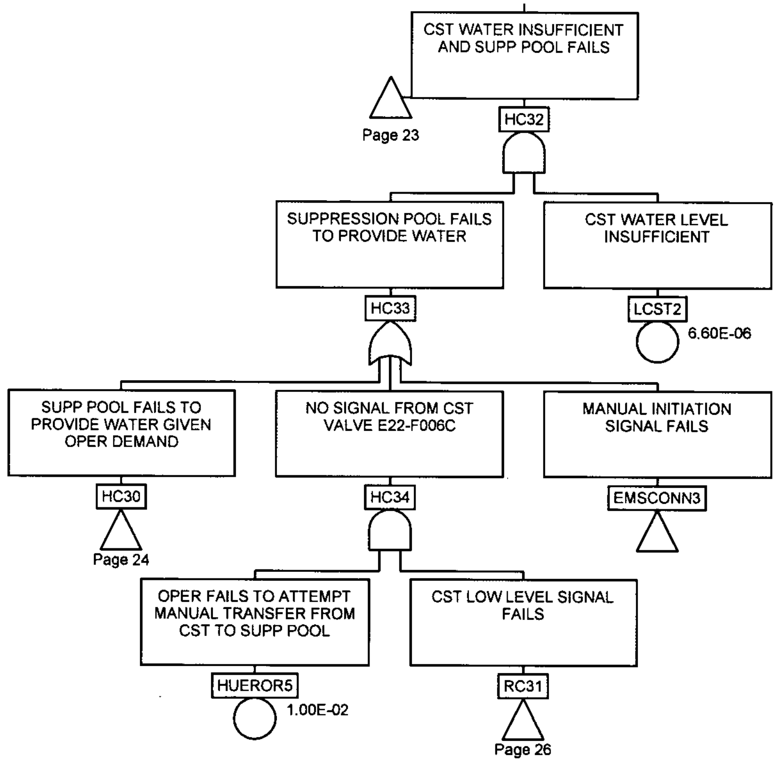


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

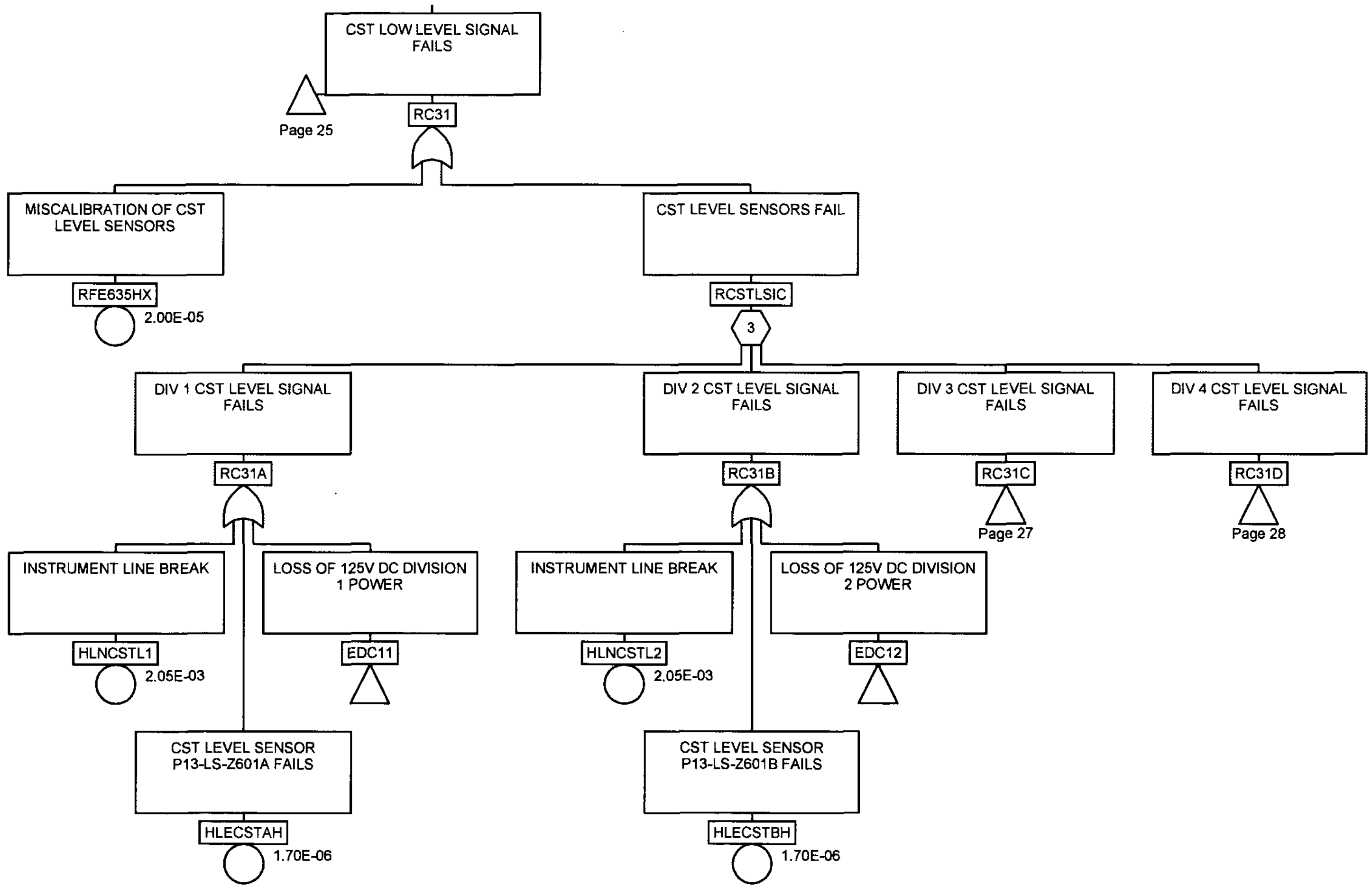


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

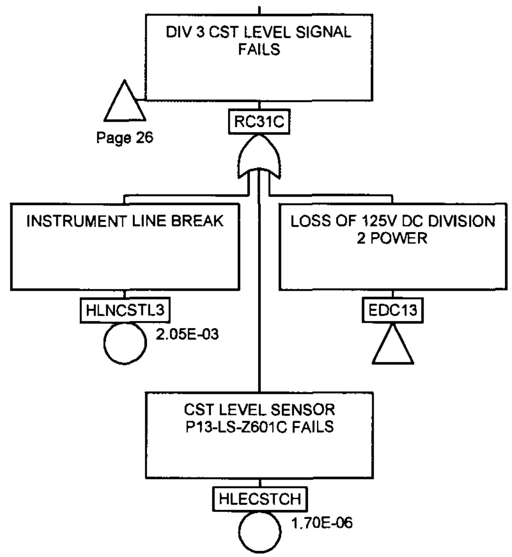


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

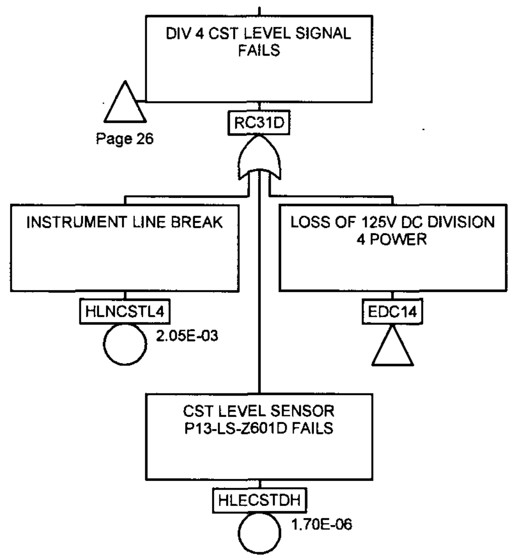


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

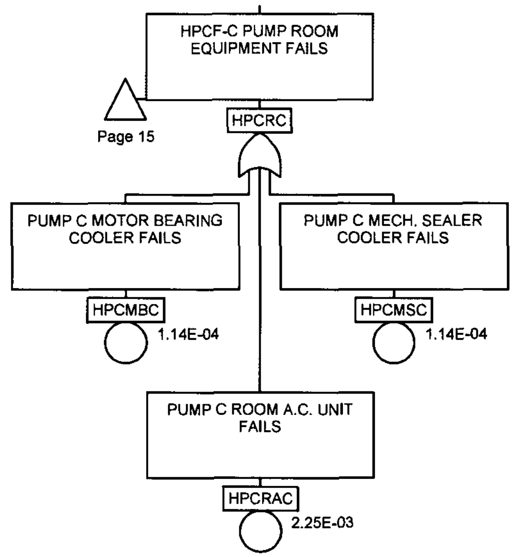


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree

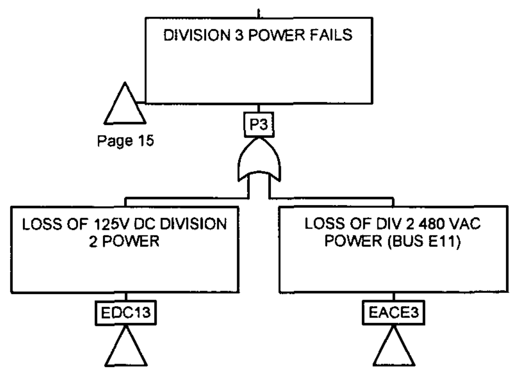
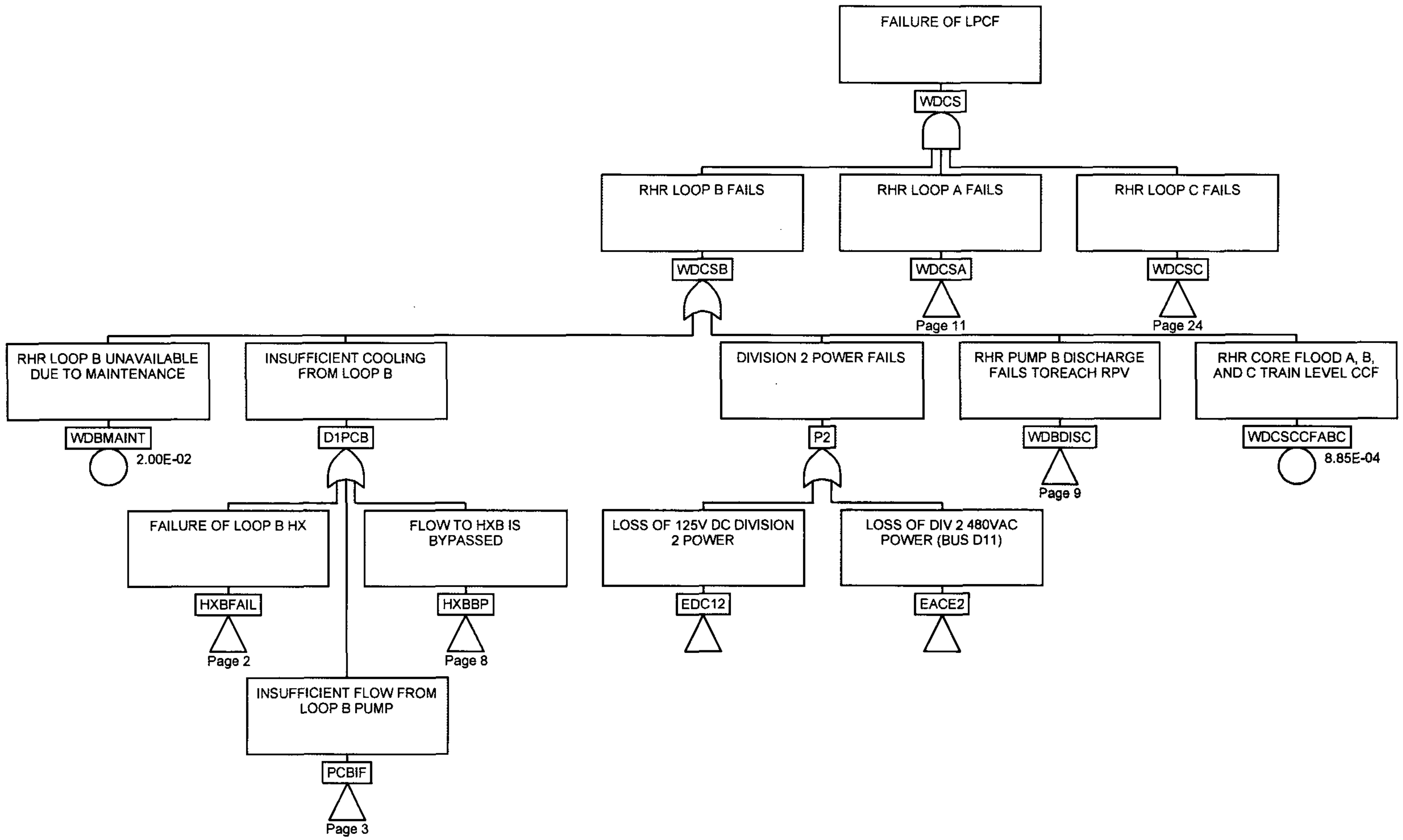
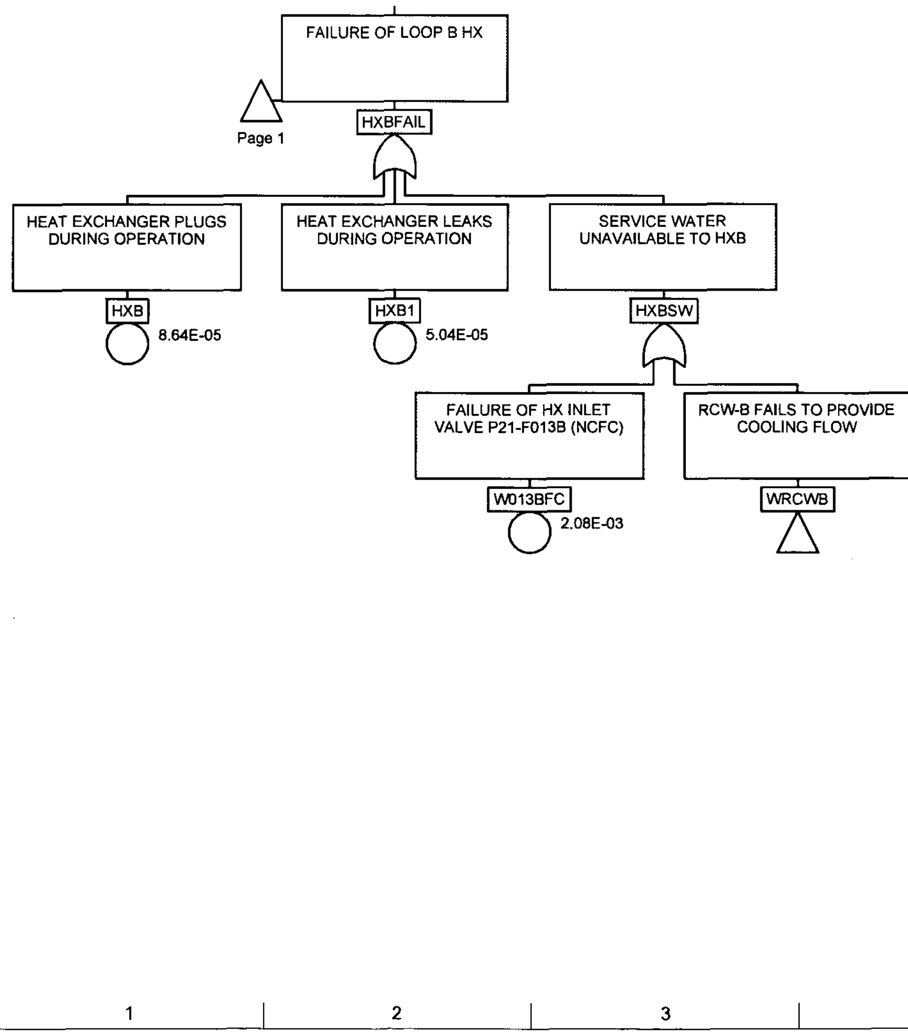


Figure 19D.6-3 High Pressure Core Flooder System Fault Tree



	1	2	3	4	5	

Figure 19D.6-4 RHR Core Flooding Fault Tree



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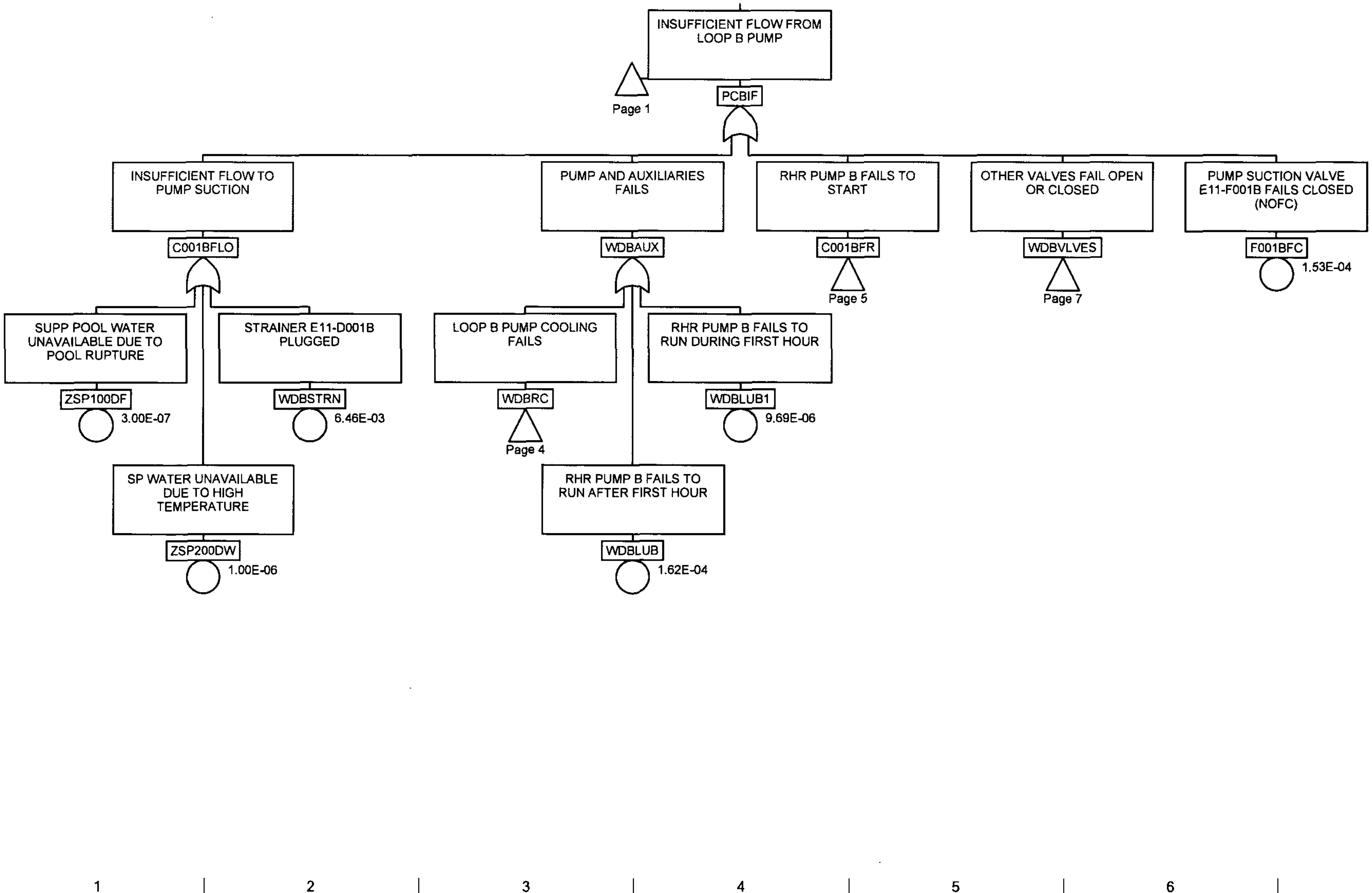
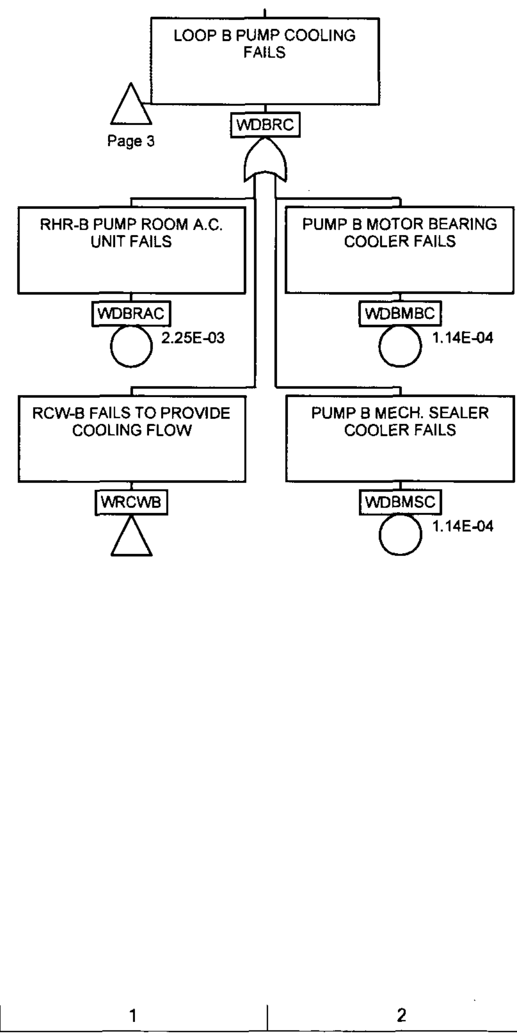


Figure 19D.6-4 RHR Core Flooding Fault Tree



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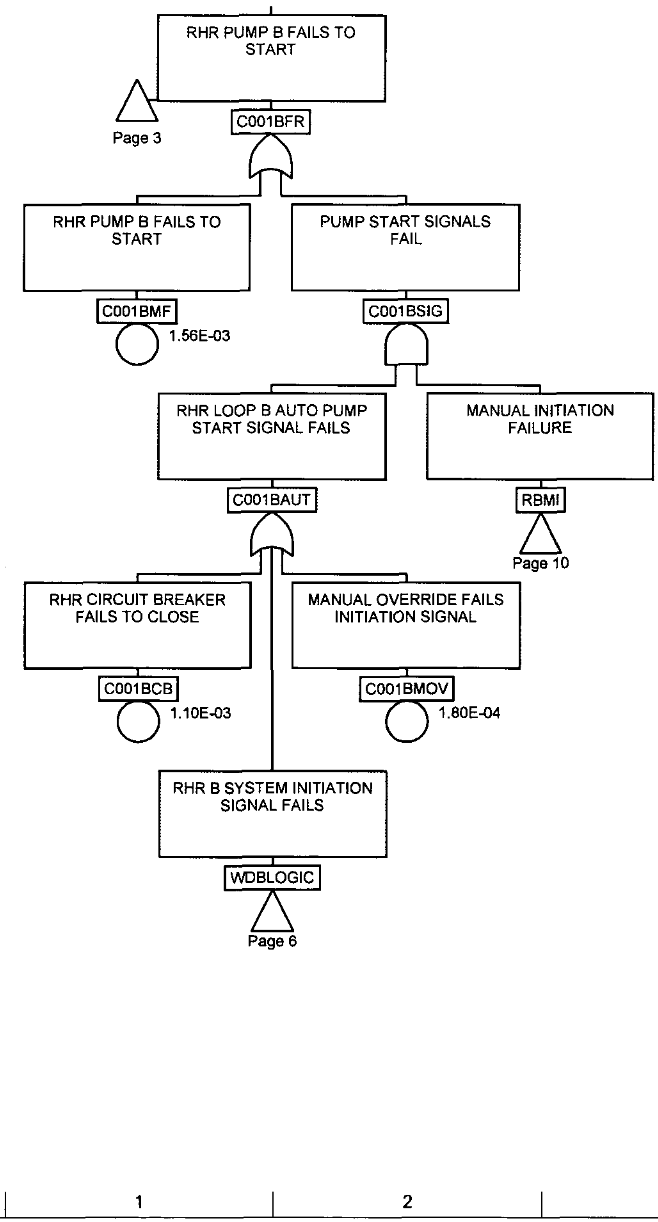

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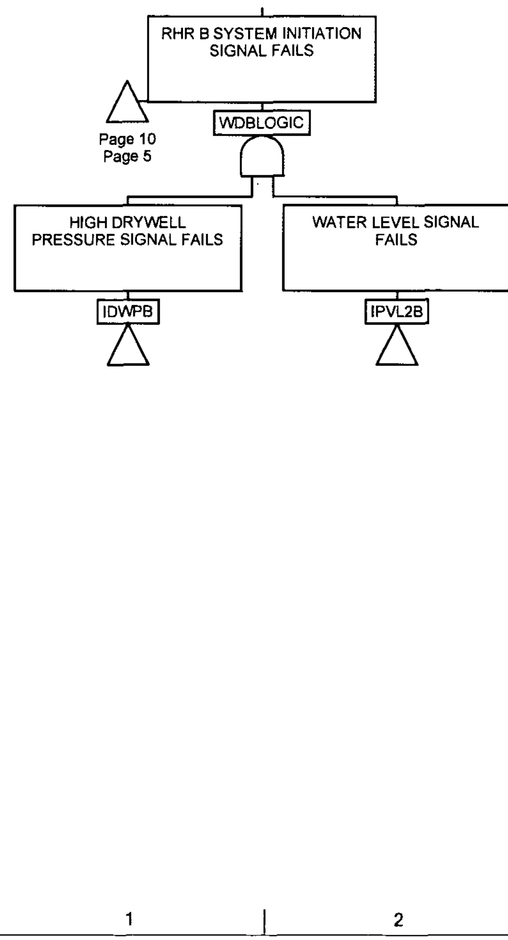


Figure 19D.6-4 RHR Core Flooding Fault Tree

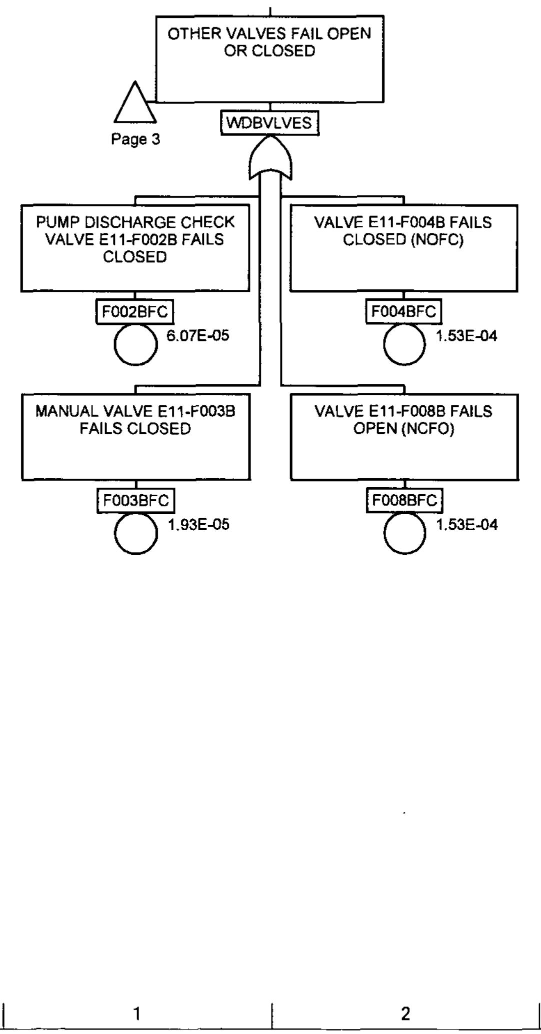


Figure 19D.6-4 RHR Core Flooding Fault Tree

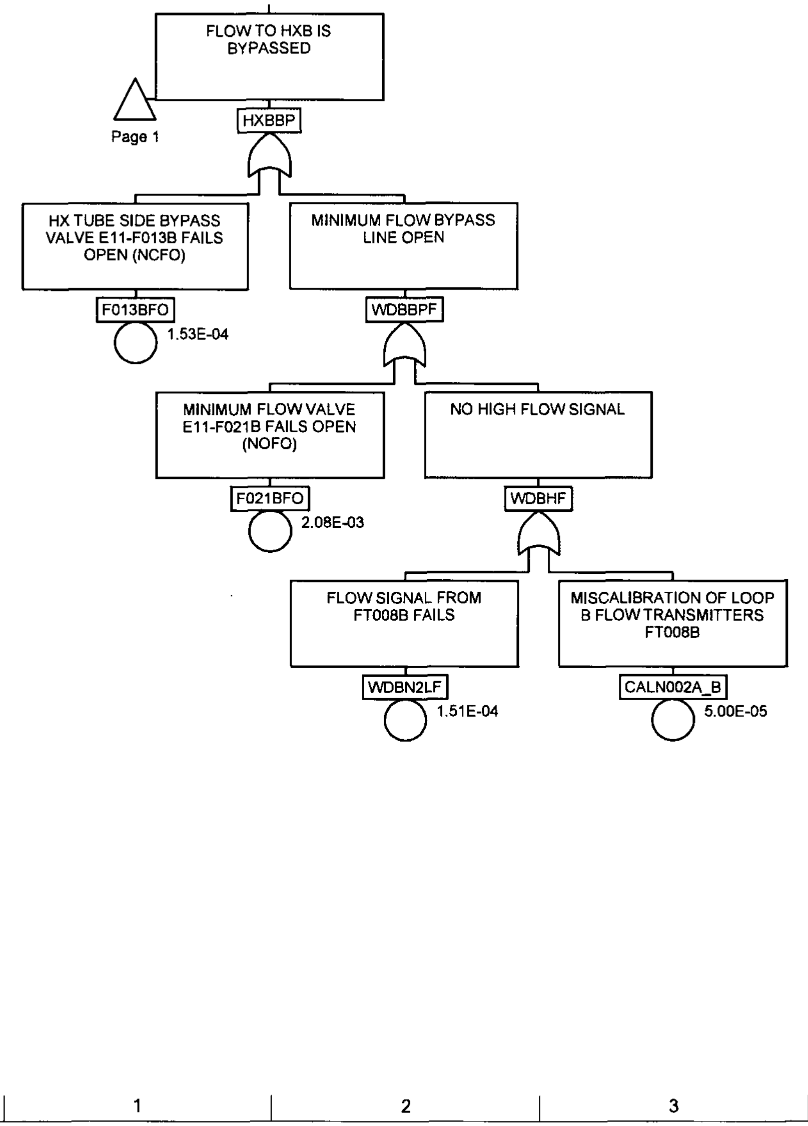


Figure 19D.6-4 RHR Core Flooding Fault Tree

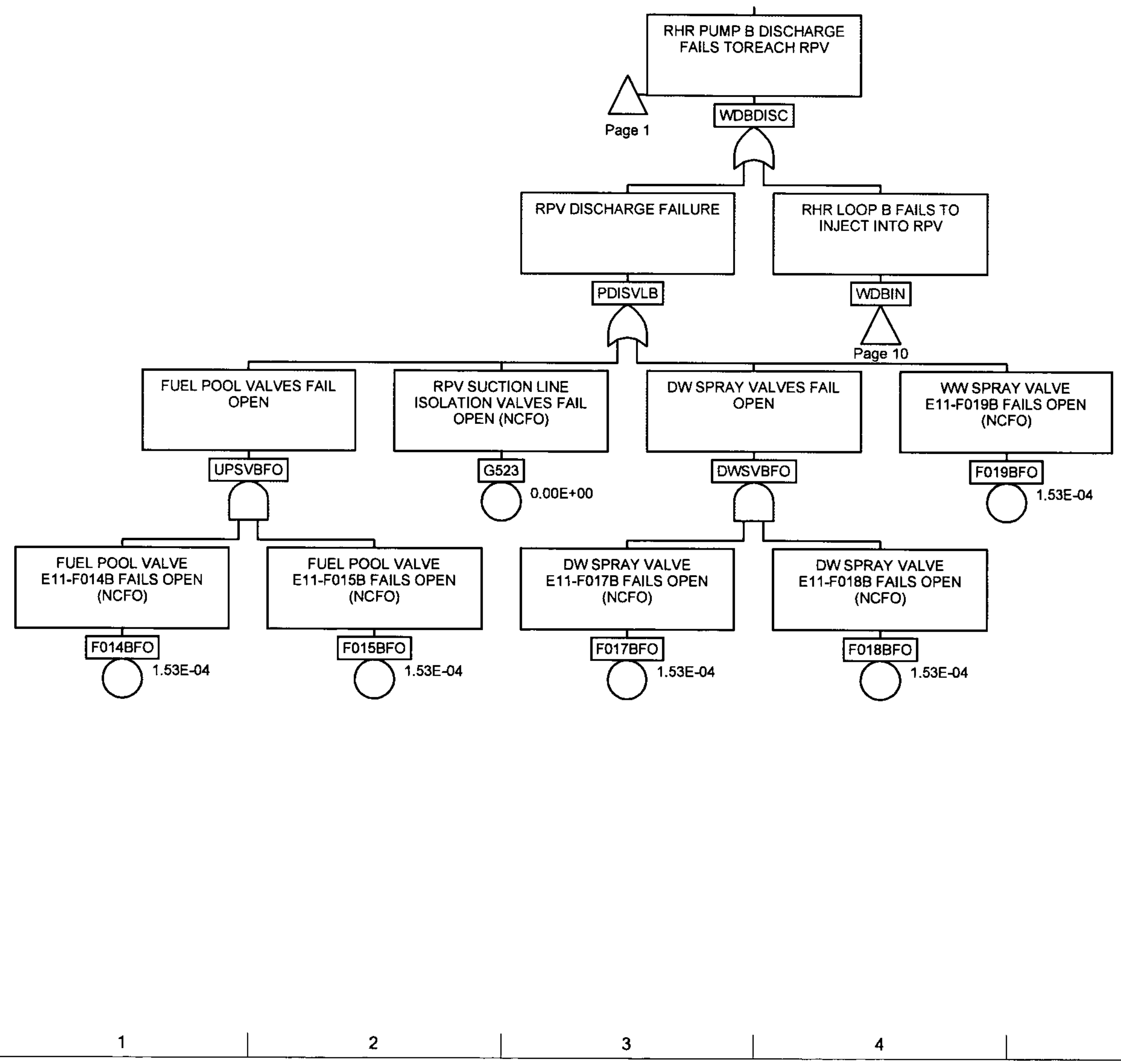
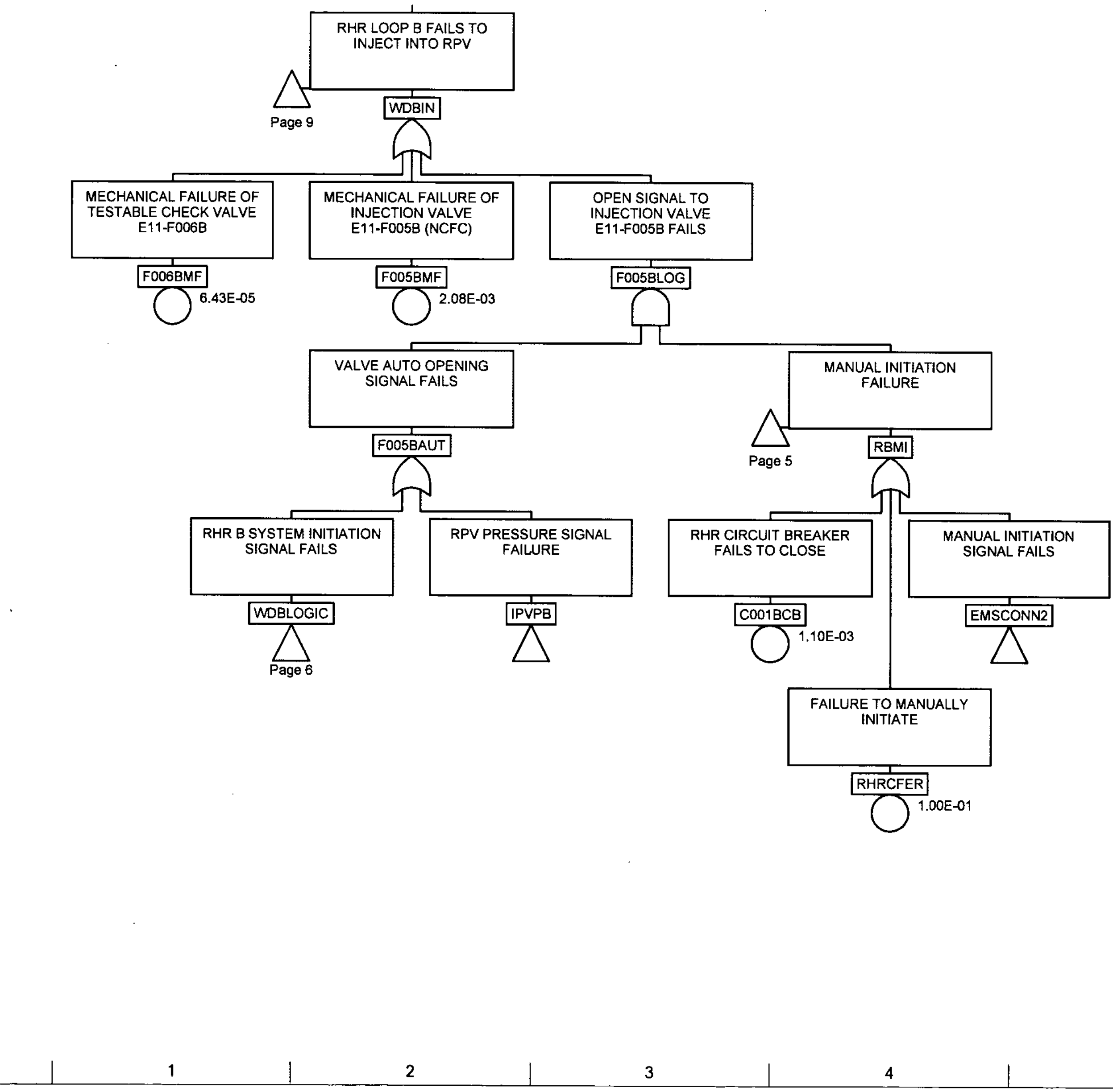


Figure 19D.6-4 RHR Core Flooding Fault Tree



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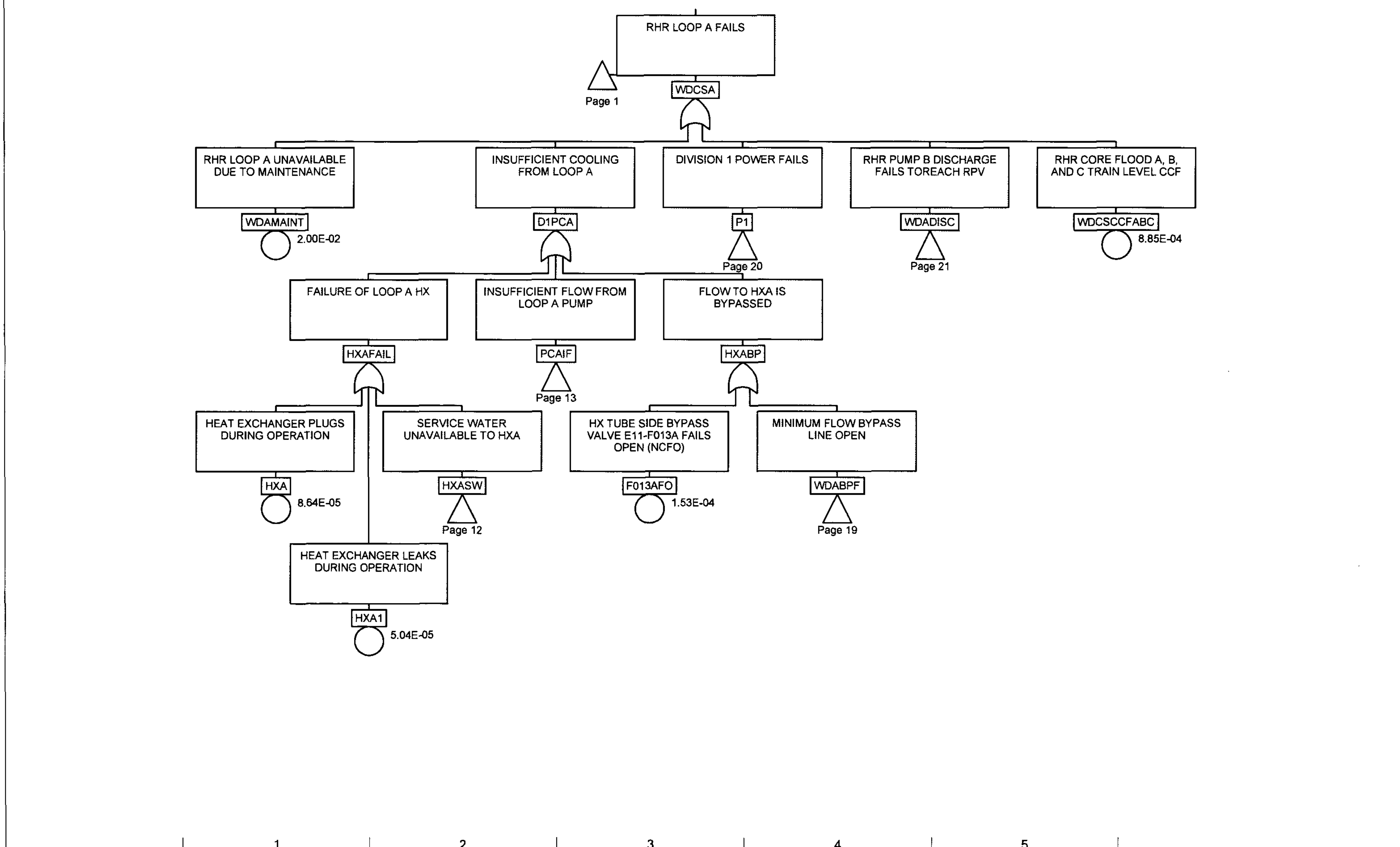

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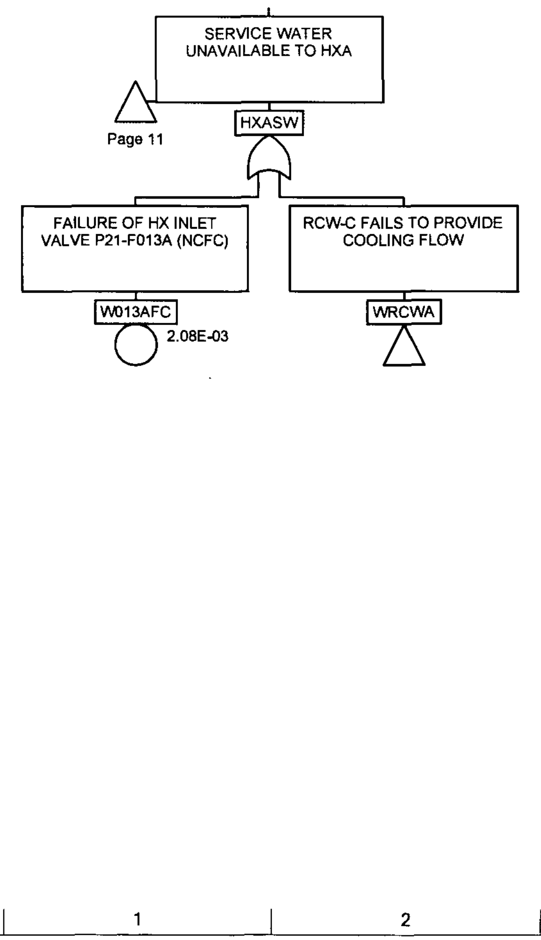

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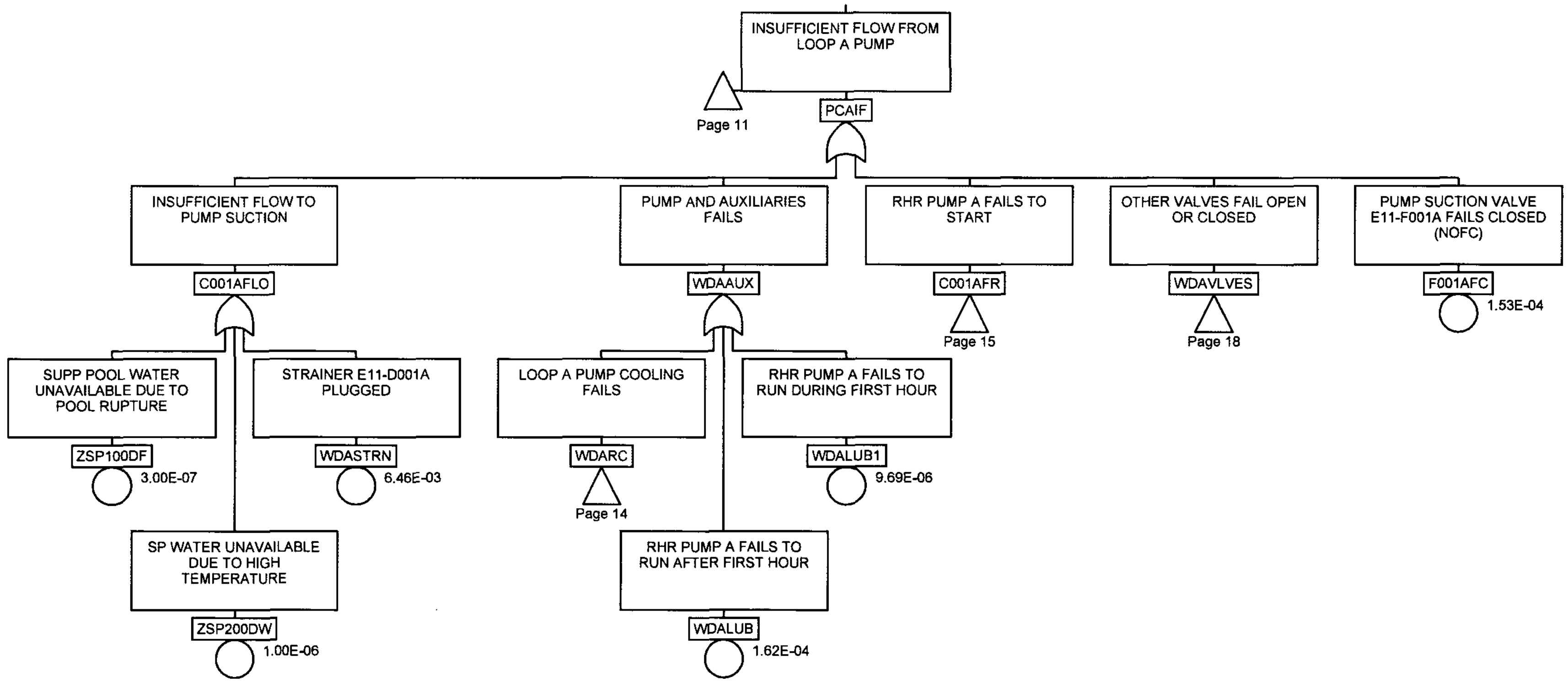

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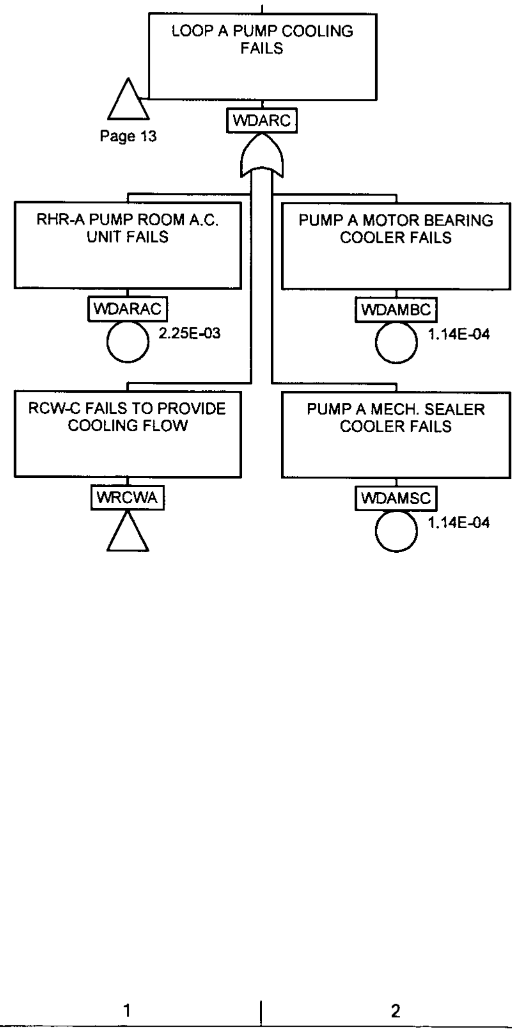

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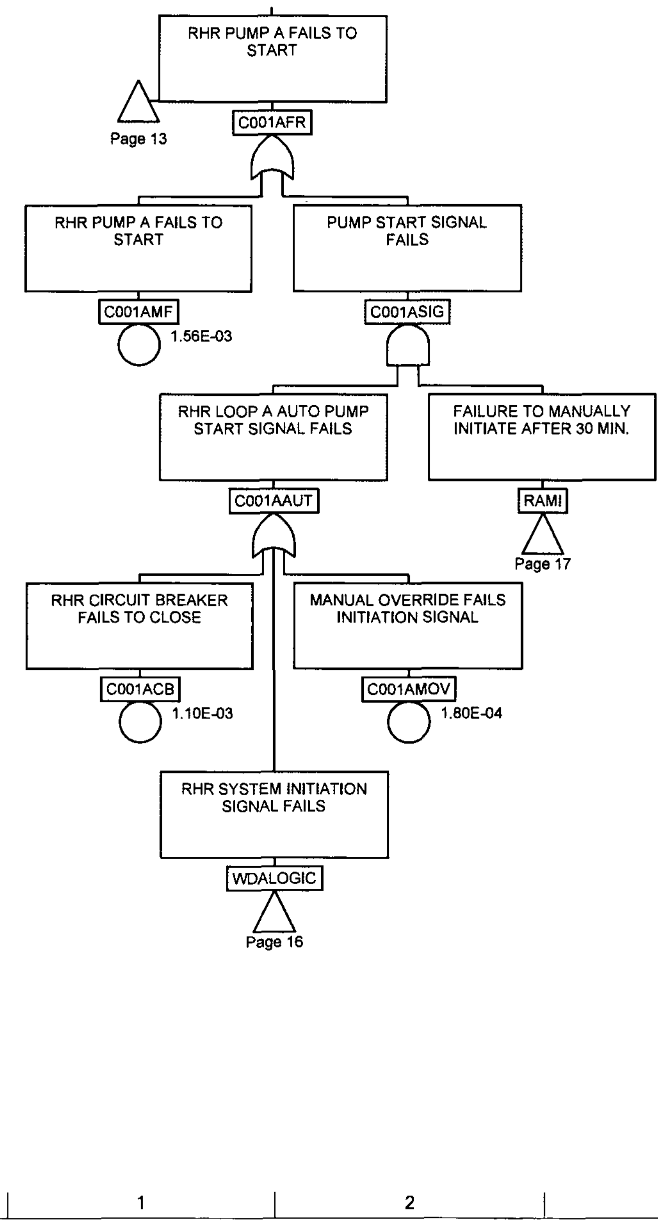

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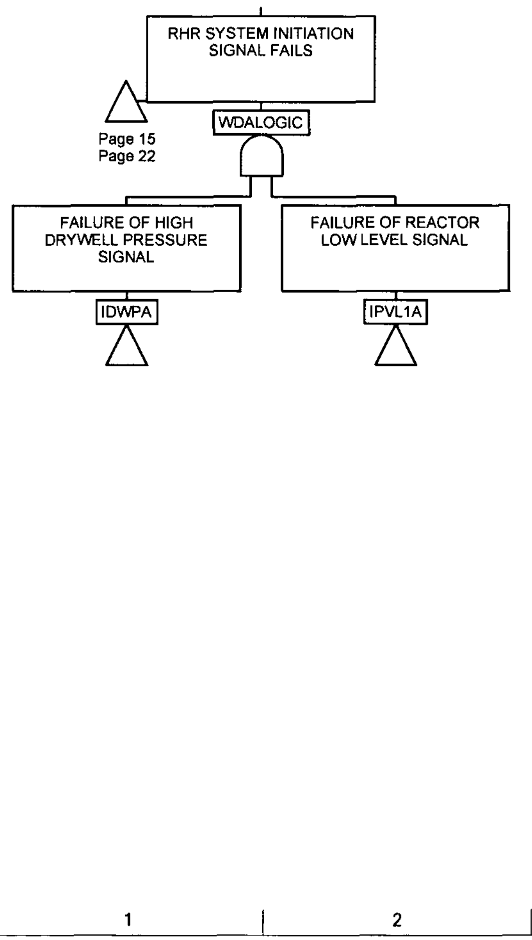

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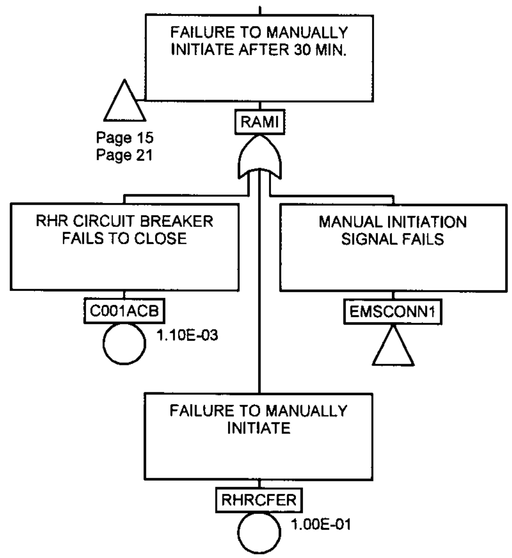

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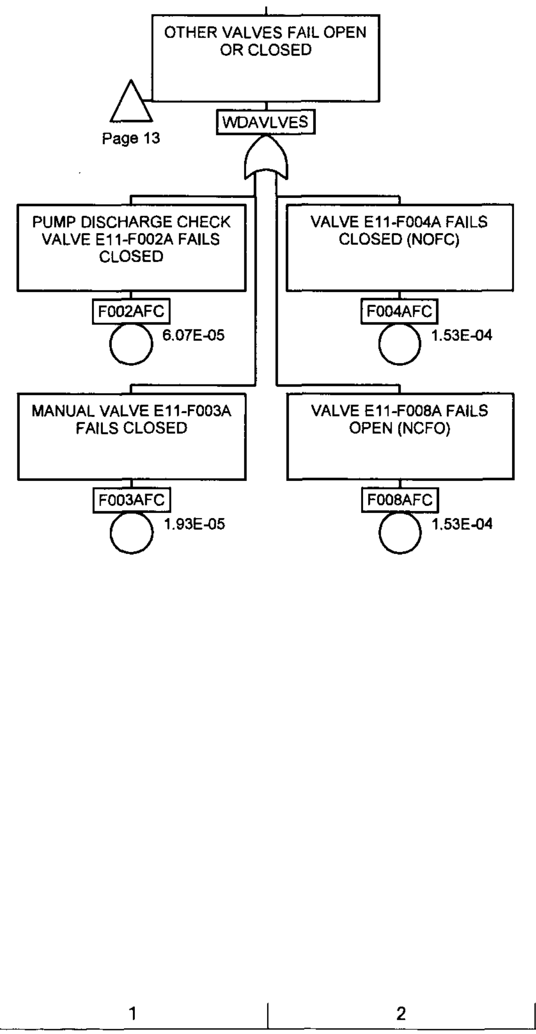


Figure 19D.6-4 RHR Core Flooding Fault Tree

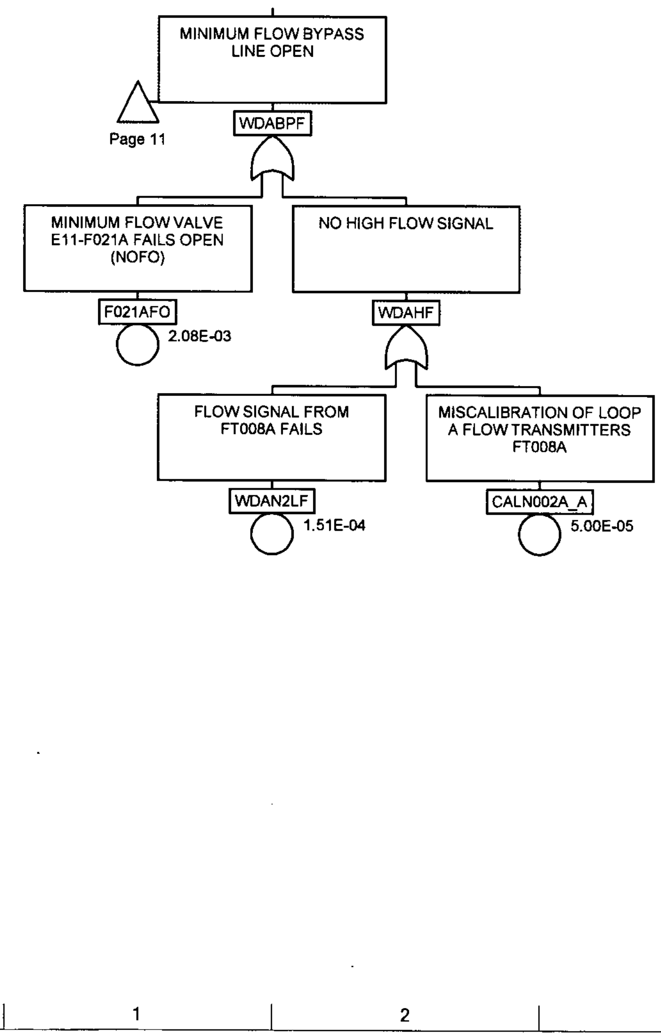
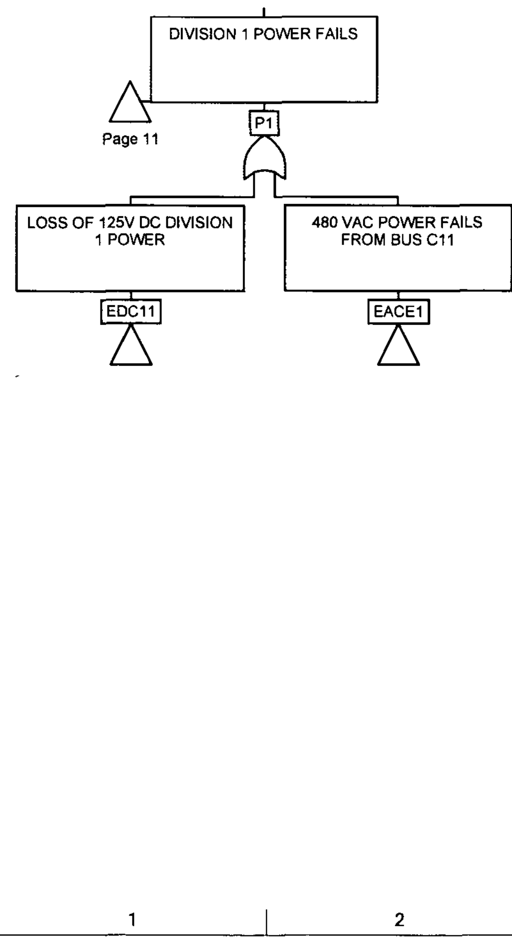


Figure 19D.6-4 RHR Core Flooding Fault Tree



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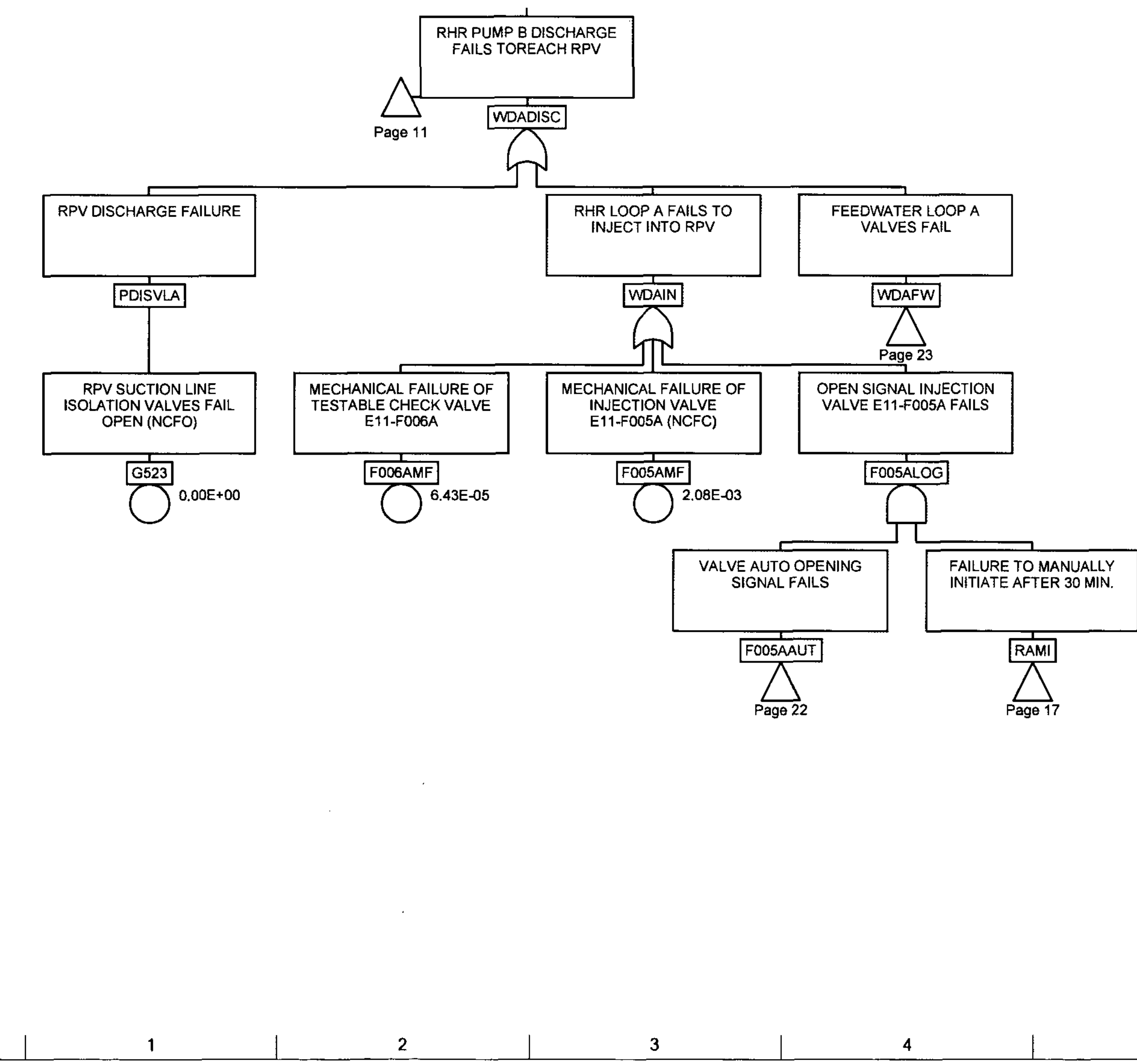

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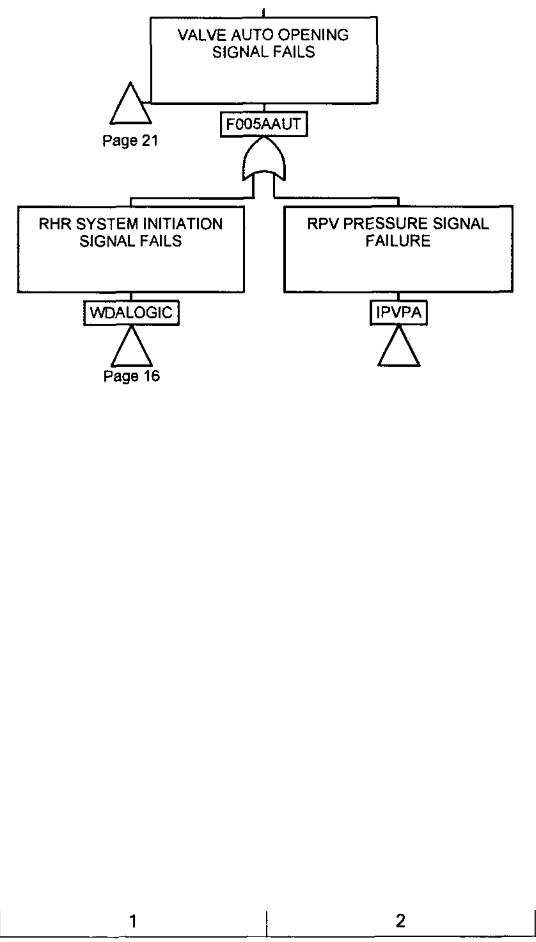

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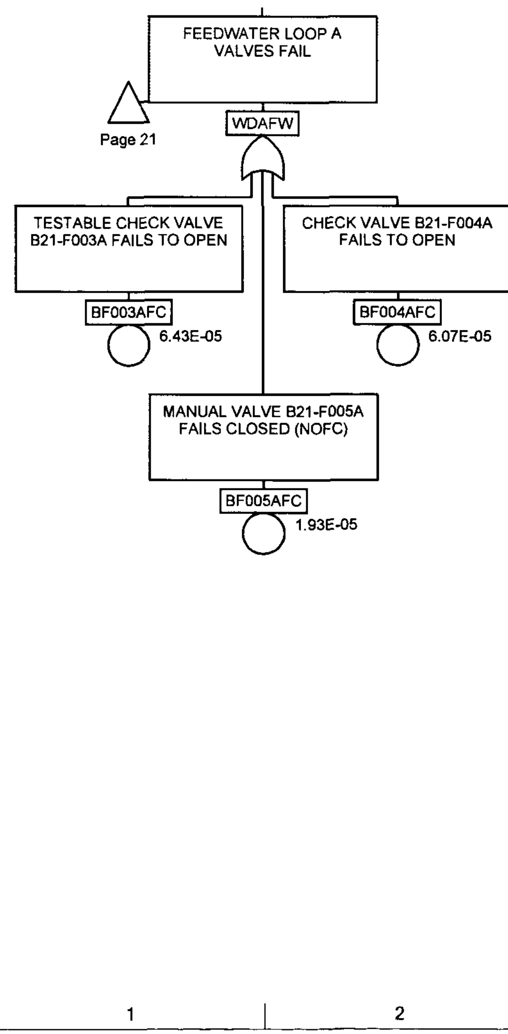


Figure 19D.6-4 RHR Core Flooding Fault Tree

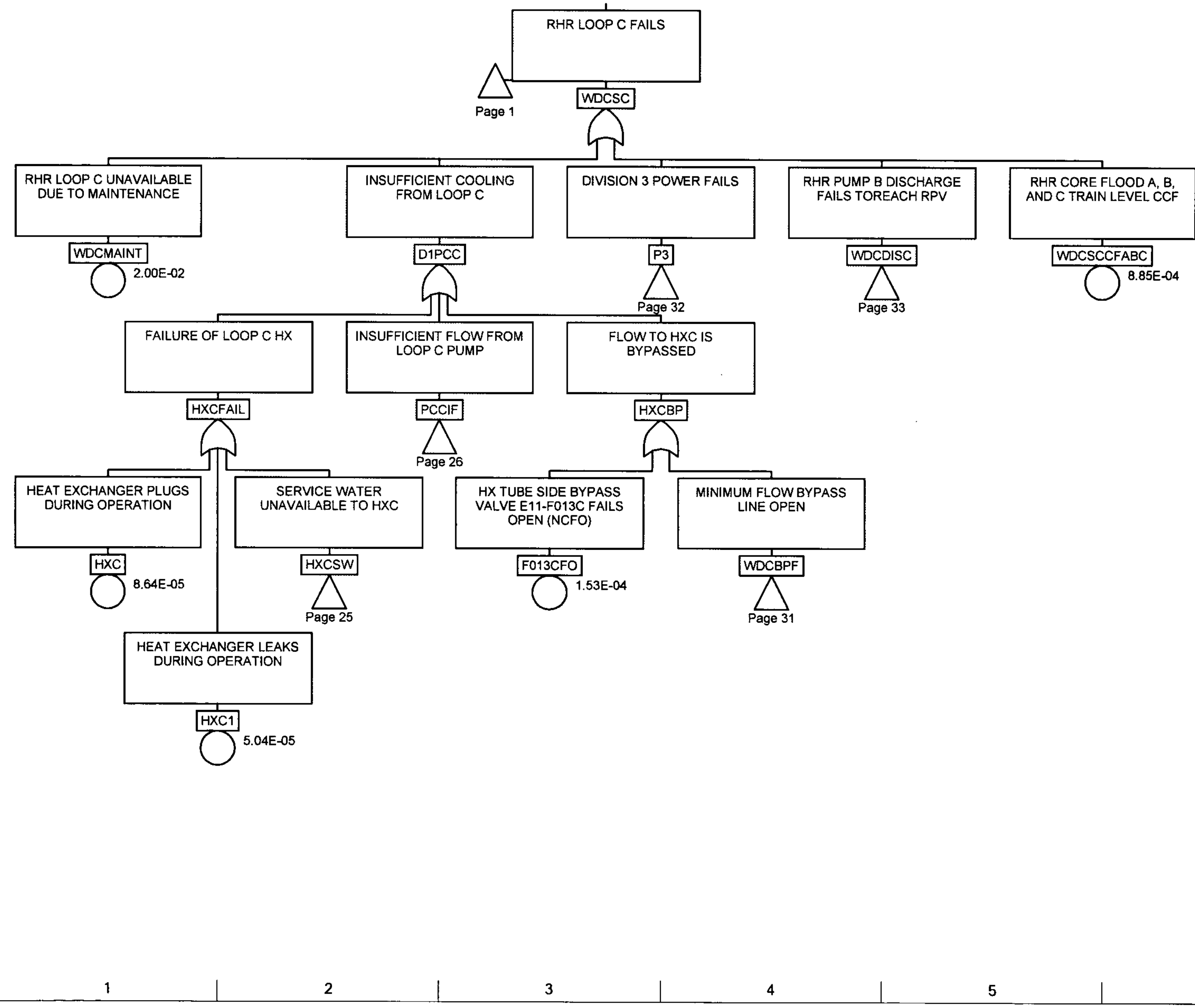


Figure 19D.6-4 RHR Core Flooding Fault Tree

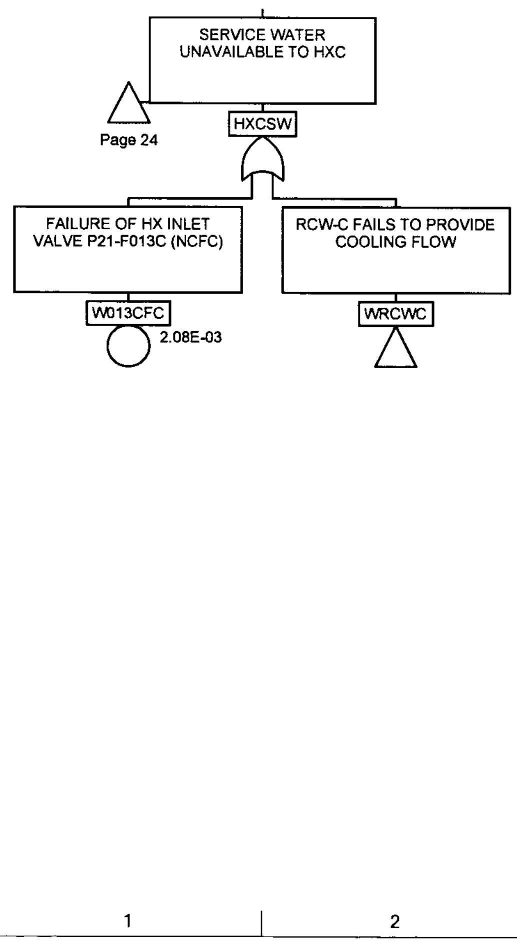


Figure 19D.6-4 RHR Core Flooding Fault Tree

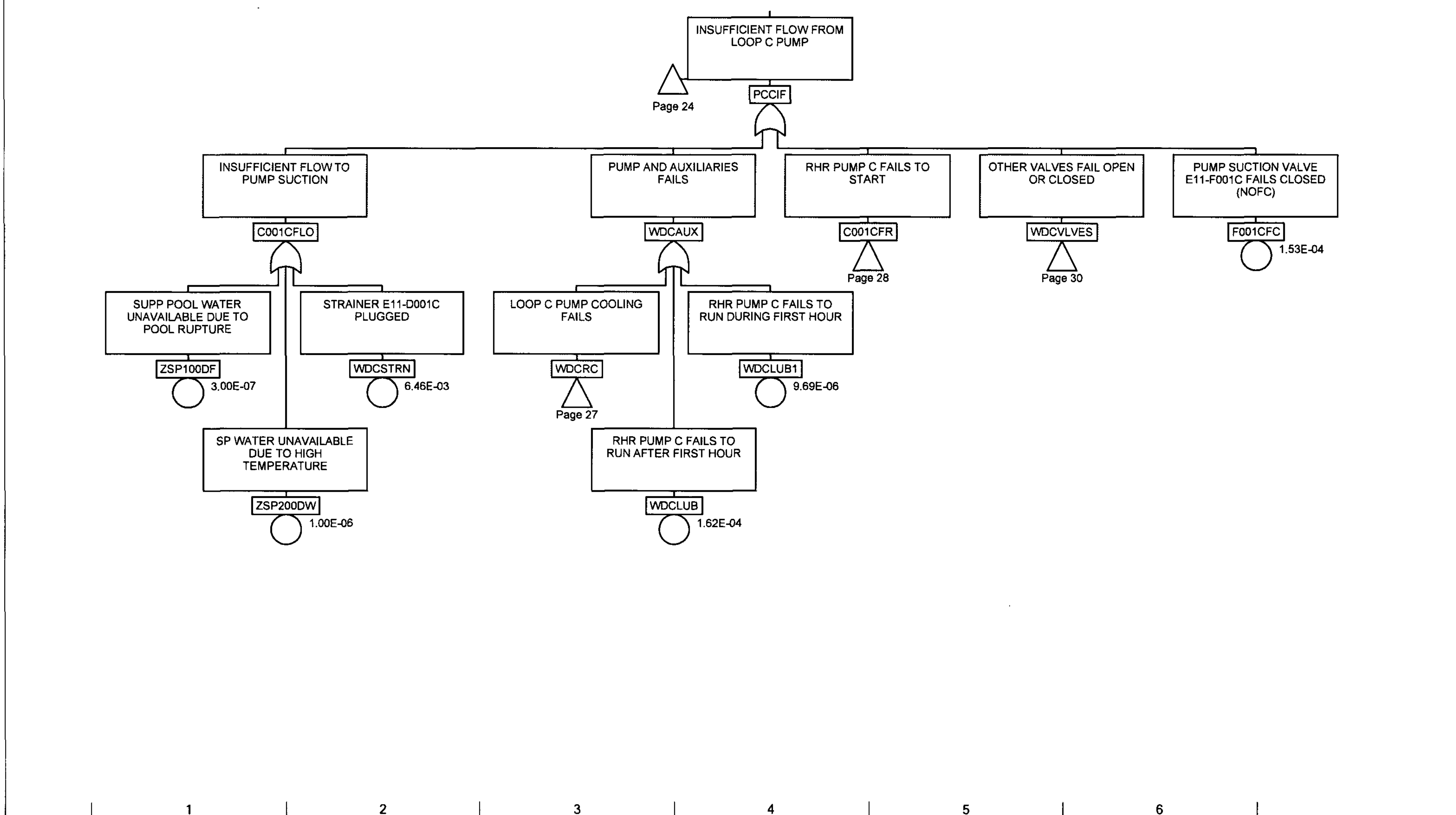


Figure 19D.6-4 RHR Core Flooding Fault Tree

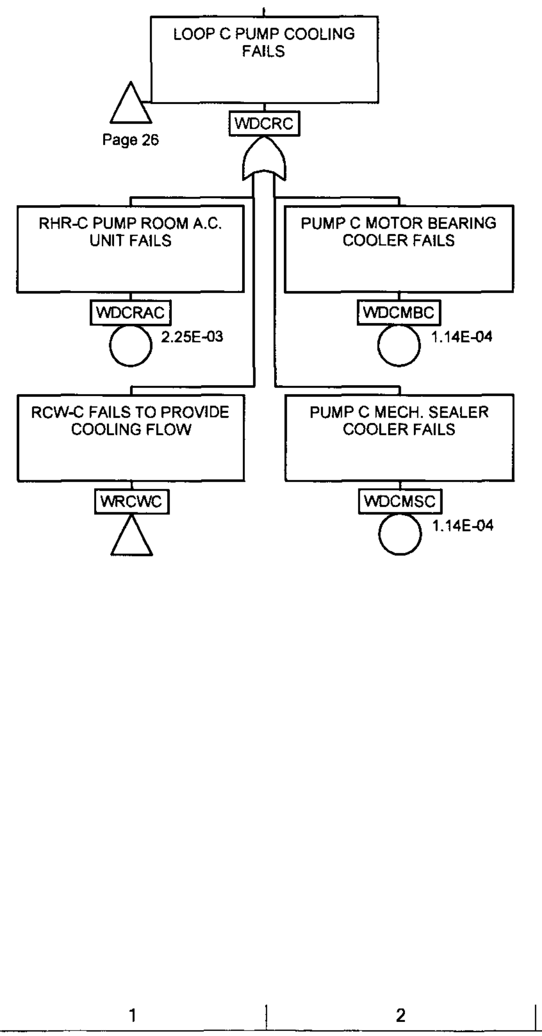


Figure 19D.6-4 RHR Core Flooding Fault Tree

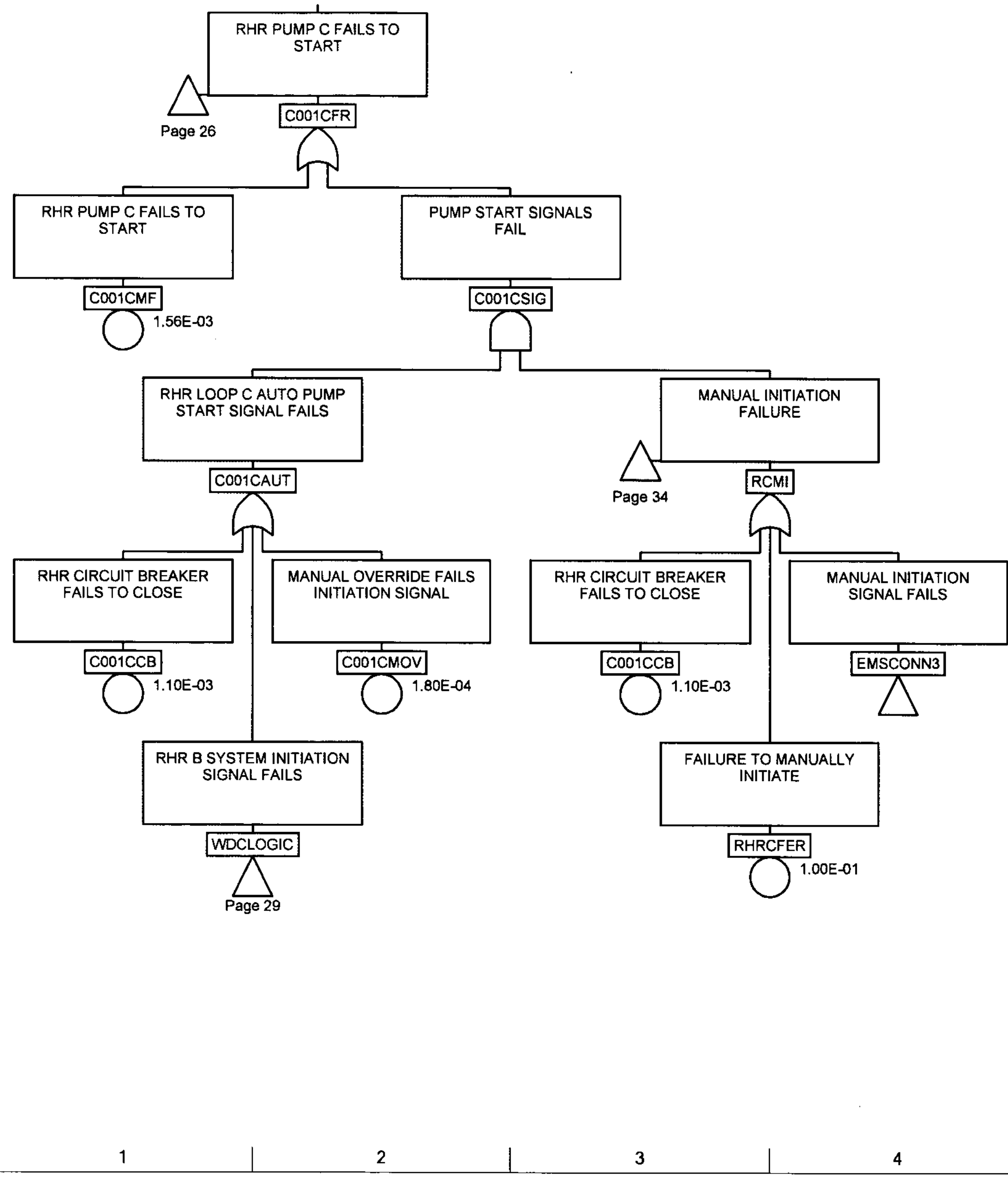
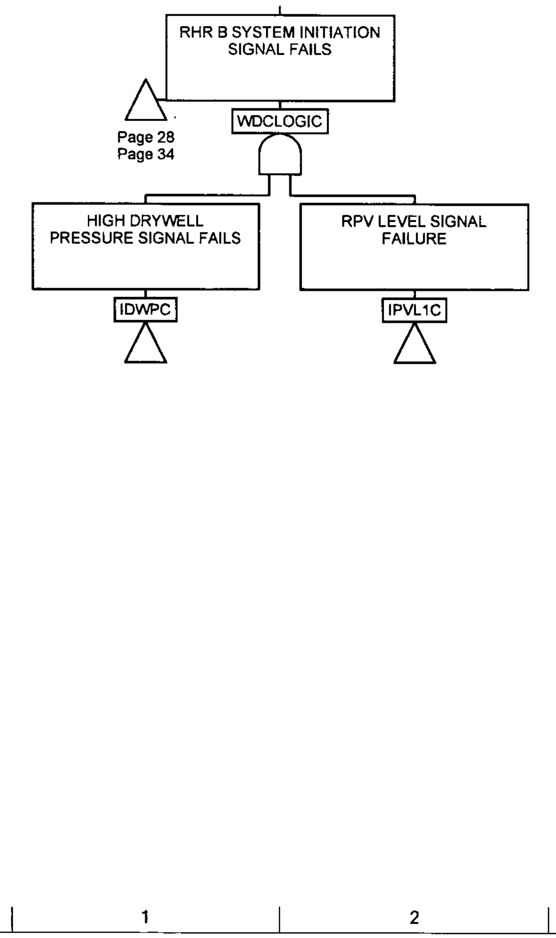


Figure 19D.6-4 RHR Core Flooding Fault Tree



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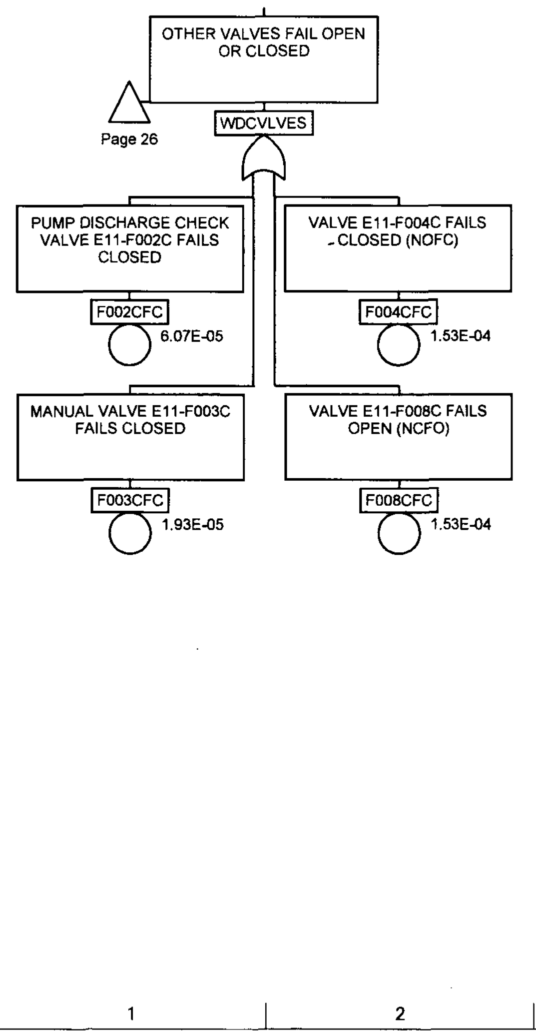

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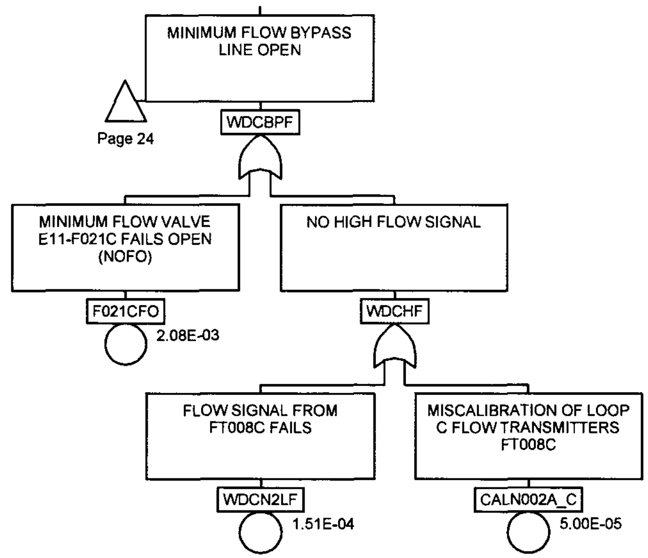

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Figure 19D.6-4 RHR Core Flooding Fault Tree

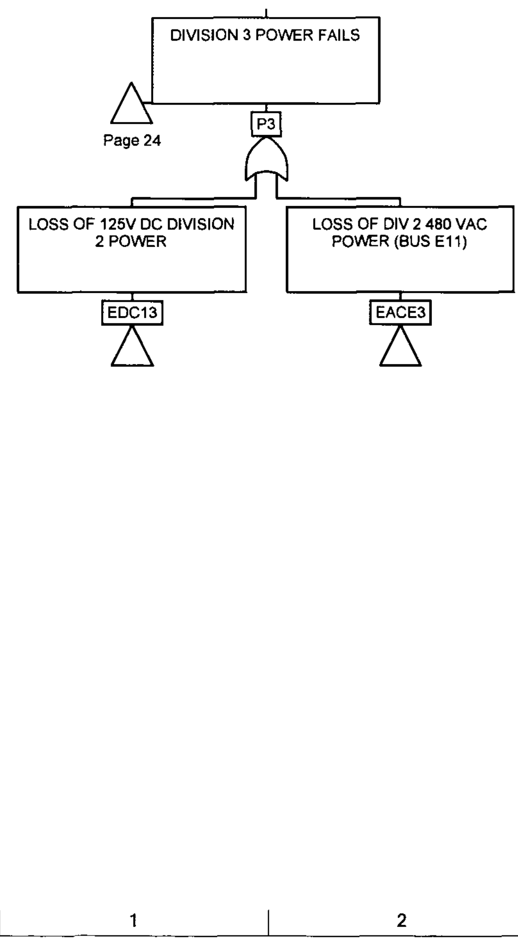

Figure 19D.6-4 RHR Core Flooding Fault Tree

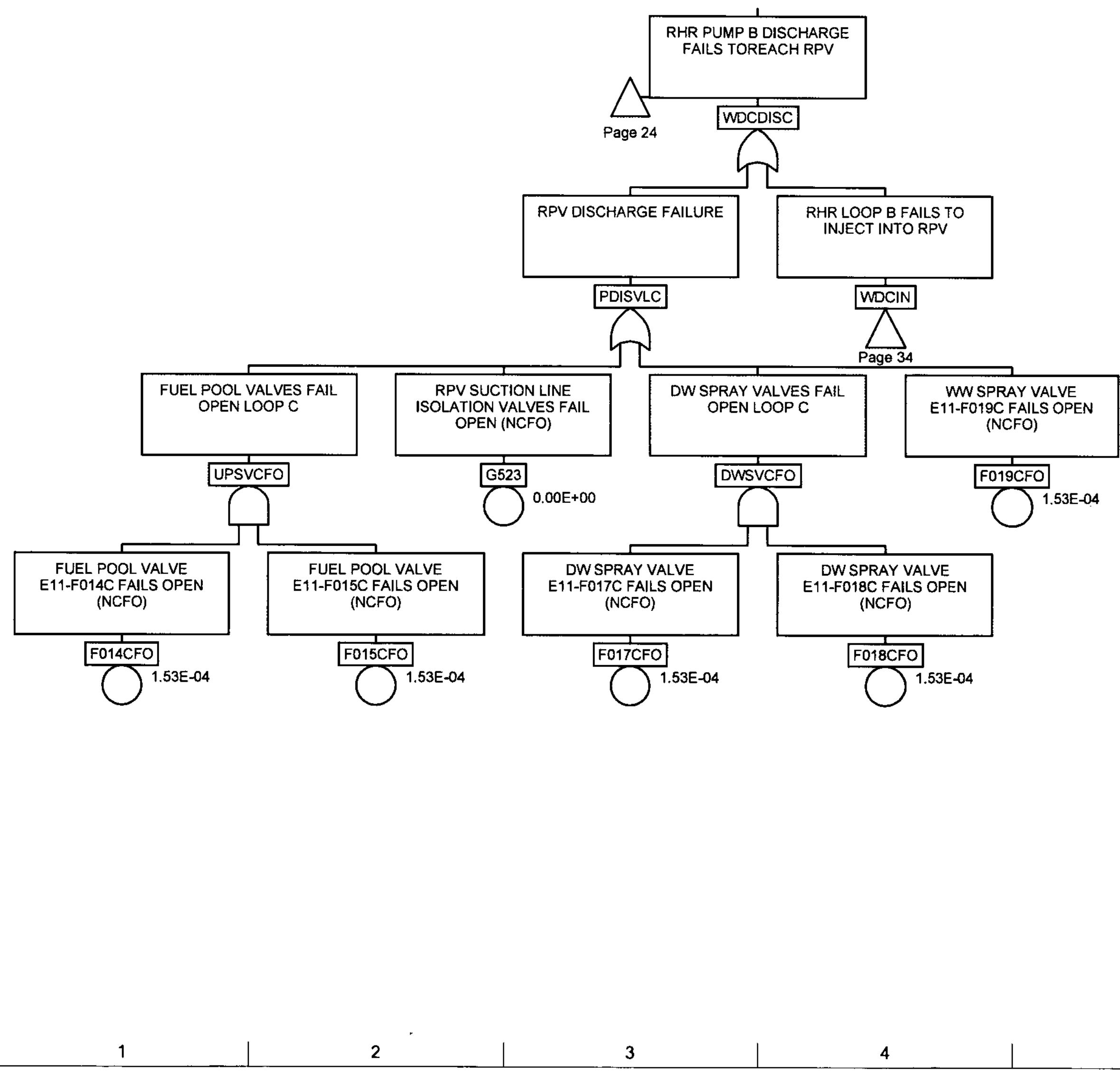

Figure 19D.6-4 RHR Core Flooding Fault Tree

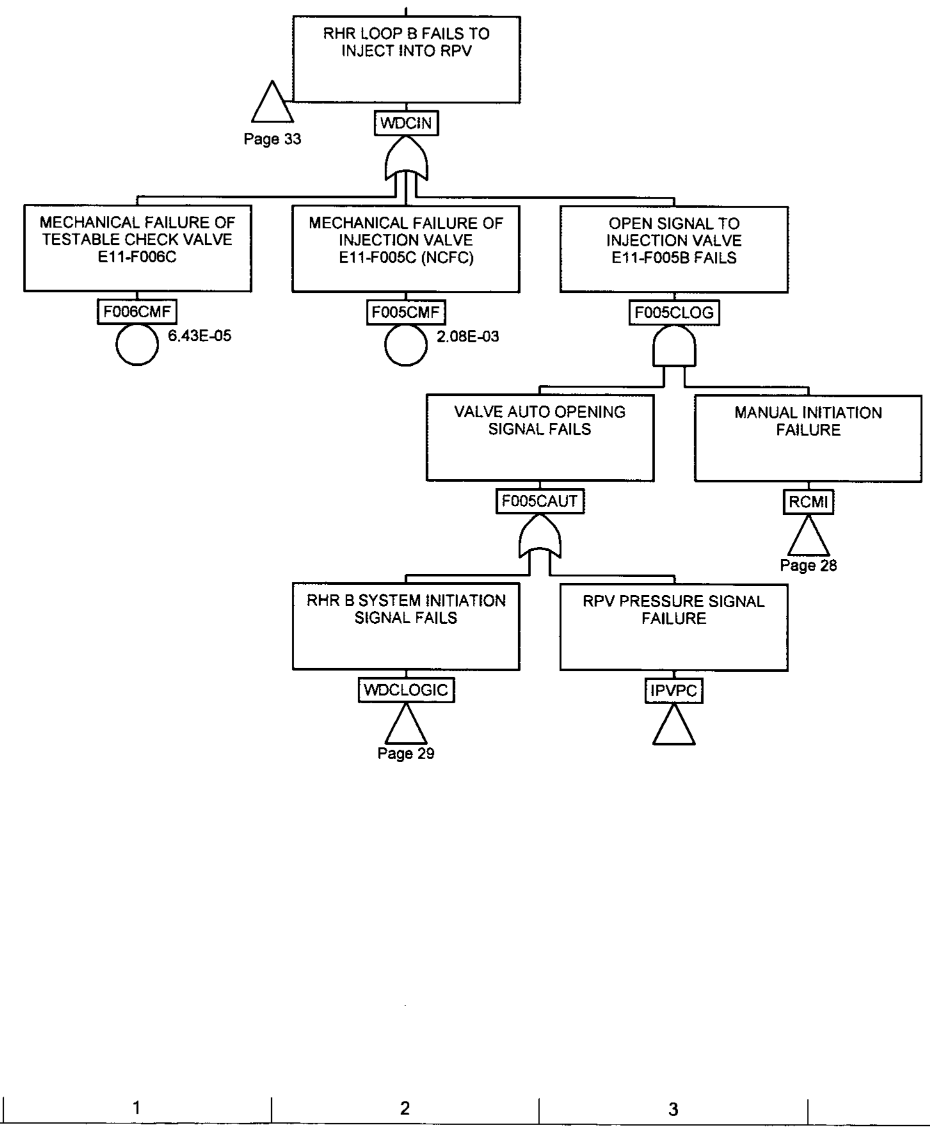

Figure 19D.6-4 RHR Core Flooding Fault Tree

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BF004AFC	23	2	F001CFC	26	7
BF005AFC	23	2	F002AFC	18	1
C001AAUT	15	2	F002BFC	7	1
C001ACB	15	1	F002CFC	30	1
C001ACB	17	1	F003AFC	18	1
C001AFLO	13	2	F003BFC	7	1
C001AFR	13	5	F003CFC	30	1
C001AFR	15	2	F004AFC	18	2
C001AMF	15	1	F004BFC	7	2
C001AMOV	15	2	F004CFC	30	2
C001ASIG	15	2	F005AAUT	21	4
C001BAUT	5	2	F005AAUT	22	2
C001BCB	5	1	F005ALOG	21	4
C001BCB	10	4	F005AMF	21	3
C001BFLO	3	2	F005BAUT	10	2
C001BFR	3	5	F005BLOG	10	3
C001BFR	5	2	F005BMF	10	2
C001BMF	5	1	F005CAUT	34	3
C001BMOV	5	2	F005CLOG	34	3
C001BSIG	5	2	F005CMF	34	2
C001CAUT	28	2	F006AMF	21	2
C001CCB	28	1	F006BMF	10	1
C001CCB	28	3	F006CMF	34	1
C001CFLO	26	2	F008AFC	18	2
C001CFR	26	5	F008BFC	7	2
C001CFR	28	2	F008CFC	30	2
C001CMF	28	1	F013AFO	11	3
C001CMOV	28	2	F013BFO	8	1
C001CSIG	28	2	F013CFO	24	3
CALN002A_A	19	3	F014BFO	9	1
CALN002A_B	8	3	F014CFO	33	1
CALN002A_C	31	3	F015BFO	9	2
D1PCA	11	2	F015CFO	33	2
D1PCB	1	2	F017BFO	9	3
D1PCC	24	2	F017CFO	33	3
DWSVBFO	9	4	F018BFO	9	4
DWSVCFO	33	4	F018CFO	33	4
EACE1	20	2	F019BFO	9	5
EACE2	1	5	F019CFO	33	5
EACE3	32	2	F021AFO	19	1
EDC11	20	1	F021BFO	8	2
EDC12	1	4	F021CFO	31	1
EDC13	32	1	G523	9	3
EMSCONN1	17	2	G523	21	1
EMSCONN2	10	5	G523	33	3
EMSCONN3	28	4	HXA	11	1
F001AFC	13	7	HXA1	11	2

Figure 19D.6-4 RHR Core Flooding Fault Tree

Name	Page	Zone	Name	Page	Zone
HXABP	11	4	RHRCFER	17	2
HXAFAIL	11	2	RHRCFER	28	4
HXASW	11	2	UPSVBFO	9	2
HXASW	12	2	UPSVCFO	33	2
HXB	2	1	W013AFC	12	1
HXB1	2	2	W013BFC	2	3
HXBBP	1	3	W013CFC	25	1
HXBBP	8	2	WDAAUX	13	4
HXBFAIL	1	2	WDABPF	11	4
HXBFAIL	2	2	WDABPF	19	2
HXBSW	2	3	WDADISC	11	4
HXC	24	1	WDADISC	21	3
HXC1	24	2	WDAFW	21	4
HXCBP	24	4	WDAFW	23	2
HXCFAIL	24	2	WDAHF	19	2
HXCSW	24	2	WDAIN	21	3
HXCSW	25	2	WDALOGIC	15	2
IDWPA	16	1	WDALOGIC	16	2
IDWPB	6	1	WDALOGIC	22	1
IDWPC	29	1	WDALUB	13	4
IPVL1A	16	2	WDALUB1	13	4
IPVL1C	29	2	WDAMAINT	11	1
IPVL2B	6	2	WDAMBC	14	2
IPVPA	22	2	WDAMSC	14	2
IPVPB	10	3	WDAN2LF	19	2
IPVPC	34	3	WDARAC	14	1
P1	11	3	WDARC	13	3
P1	20	2	WDARC	14	2
P2	1	4	WDASTRN	13	2
P3	24	3	WDAVLVES	13	6
P3	32	2	WDAVLVES	18	2
PCAIF	11	3	WDBAUX	3	4
PCAIF	13	4	WDBBPF	8	2
PCBIF	1	2	WDBDISC	1	5
PCBIF	3	4	WDBDISC	9	3
PCCIF	24	3	WDBHF	8	3
PCCIF	26	4	WDBIN	9	4
PDISVLA	21	1	WDBIN	10	2
PDISVLB	9	3	WDBLOGIC	5	2
PDISVLC	33	3	WDBLOGIC	6	2
RAMI	15	3	WDBLOGIC	10	2
RAMI	17	2	WDBLUB	3	4
RAMI	21	5	WDBLUB1	3	4
RBMI	5	3	WDBMAINT	1	1
RBMI	10	4	WDBMBC	4	2
RCMI	28	4	WDBMSC	4	2
RCMI	34	4	WDBN2LF	8	2
RHRCFER	10	4	WDBRAC	4	1

Name	Page	Zone	Name	Page	Zone
WDBRC	3	3	ZSP200DW	26	2
WDBRC	4	2			
WDBSTRN	3	2			
WDBVLVES	3	6			
WDBVLVES	7	2			
WDCAUX	26	4			
WDCBPF	24	4			
WDCBPF	31	2			
WDCDISC	24	4			
WDCDISC	33	3			
WDCHF	31	2			
WDCIN	33	4			
WDCIN	34	2			
WDCLOGIC	28	2			
WDCLOGIC	29	2			
WDCLOGIC	34	2			
WDCLUB	26	4			
WDCLUB1	26	4			
WDCMAINT	24	1			
WDCMBC	27	2			
WDCMSC	27	2			
WDCN2LF	31	2			
WDCRAC	27	1			
WDCRC	26	3			
WDCRC	27	2			
WDCS	1	4			
WDCSA	1	4			
WDCSA	11	3			
WDCSB	1	3			
WDCSC	1	5			
WDCSC	24	3			
WDCSCCFABC	1	6			
WDCSCCFABC	11	5			
WDCSCCFABC	24	5			
WDCSTRN	26	2			
WDCVLVES	26	6			
WDCVLVES	30	2			
WRCWA	12	2			
WRCWA	14	1			
WRCWB	2	4			
WRCWB	4	1			
WRCWC	25	2			
WRCWC	27	1			
ZSP100DF	3	1			
ZSP100DF	13	1			
ZSP100DF	26	1			
ZSP200DW	3	2			
ZSP200DW	13	2			

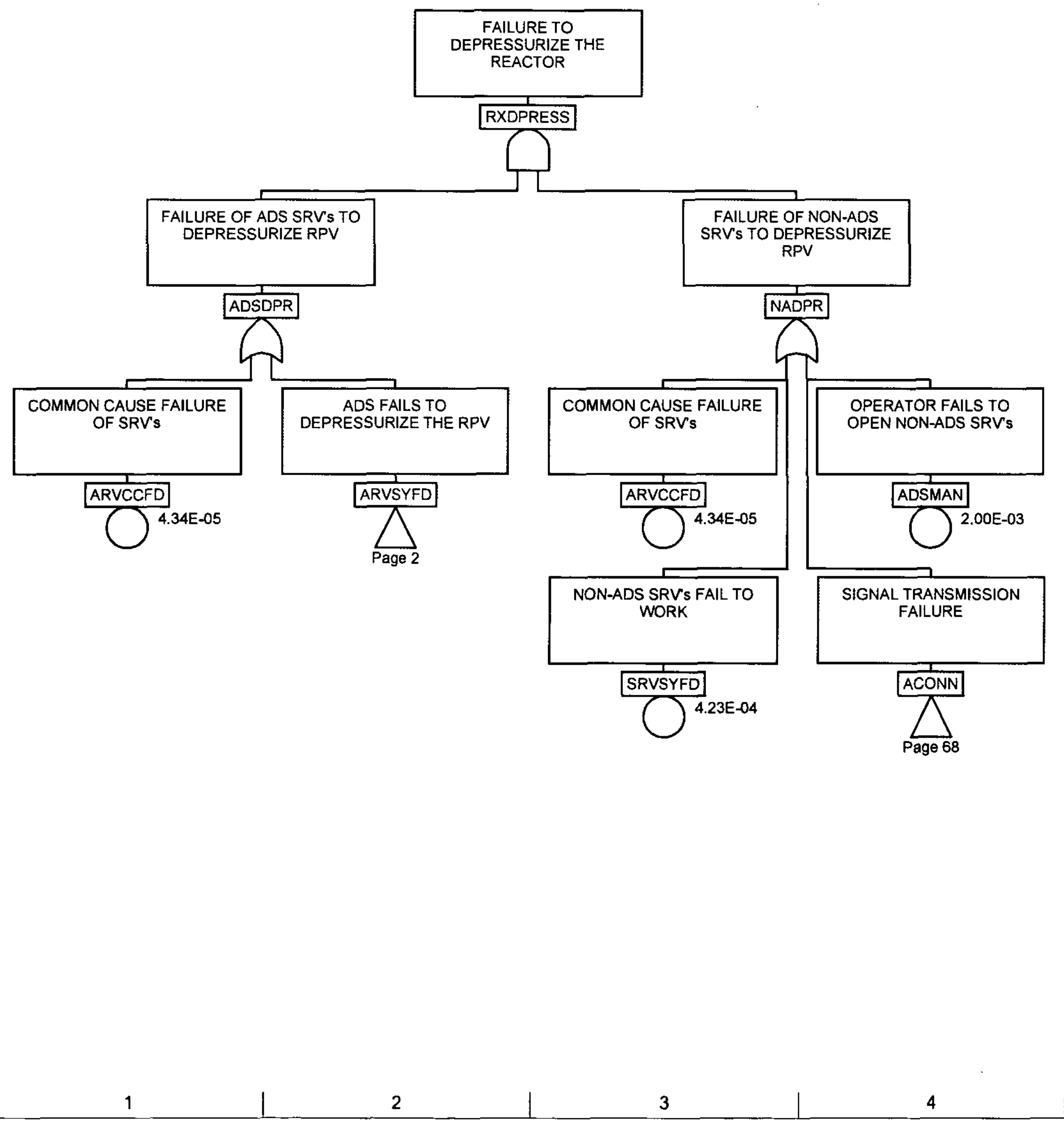


Figure 19D.6-5 Reactor Depressurization System Fault Tree

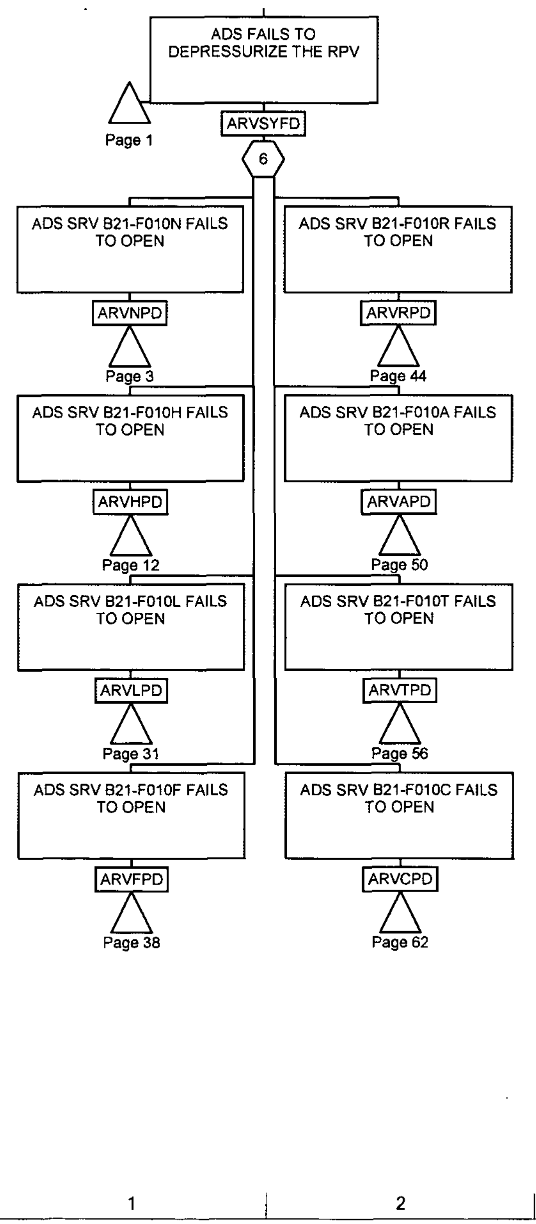


Figure 19D.6-5 Reactor Depressurization System Fault Tree

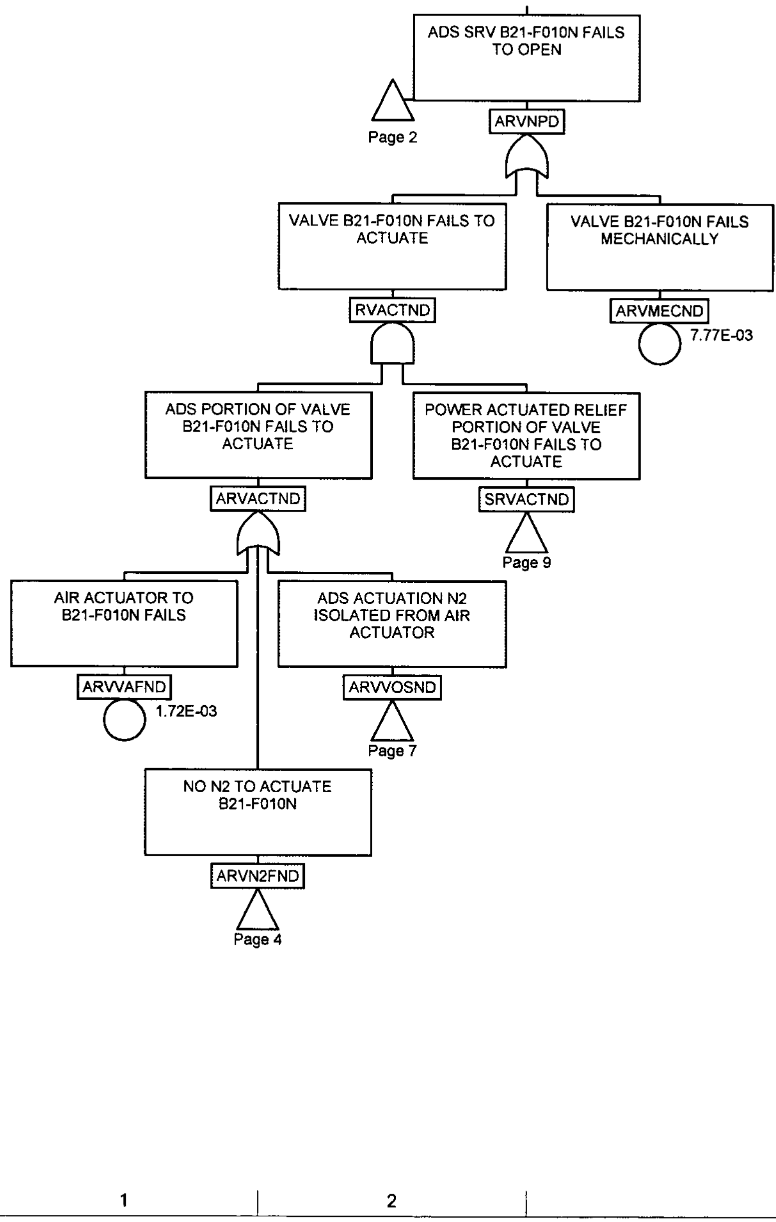
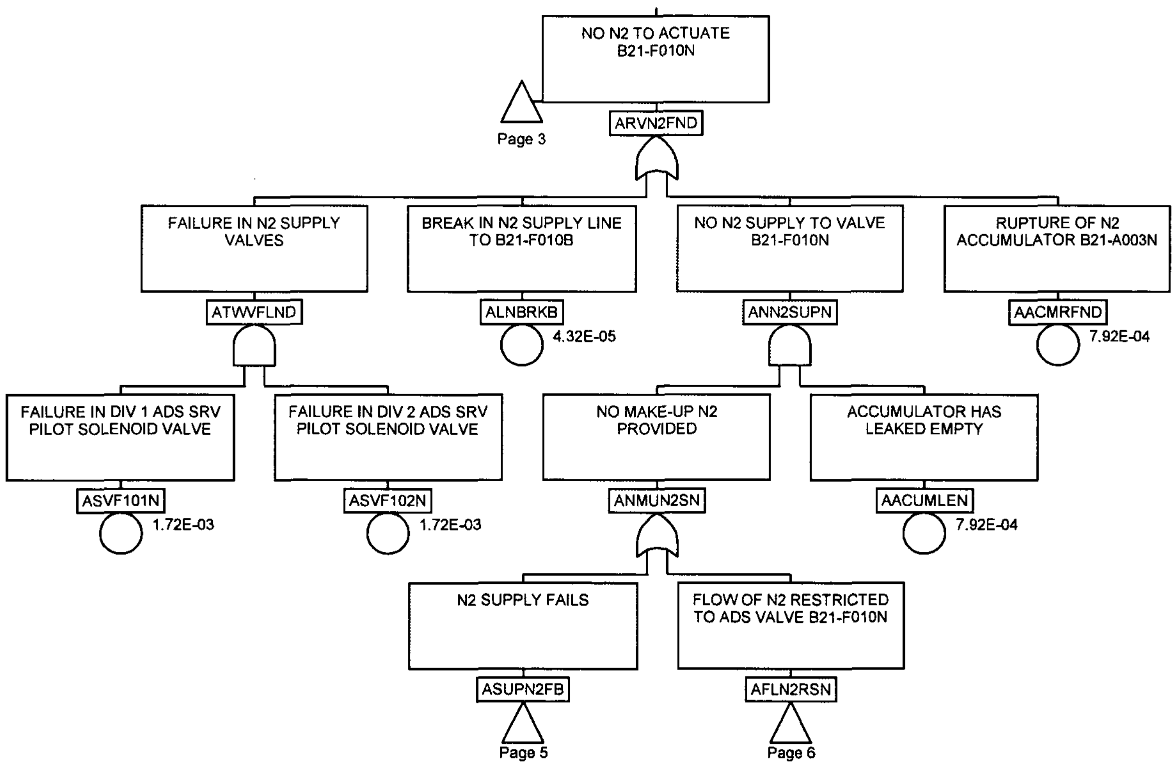


Figure 19D.6-5 Reactor Depressurization System Fault Tree



1	2	3	4

Figure 19D.6-5 Reactor Depressurization System Fault Tree

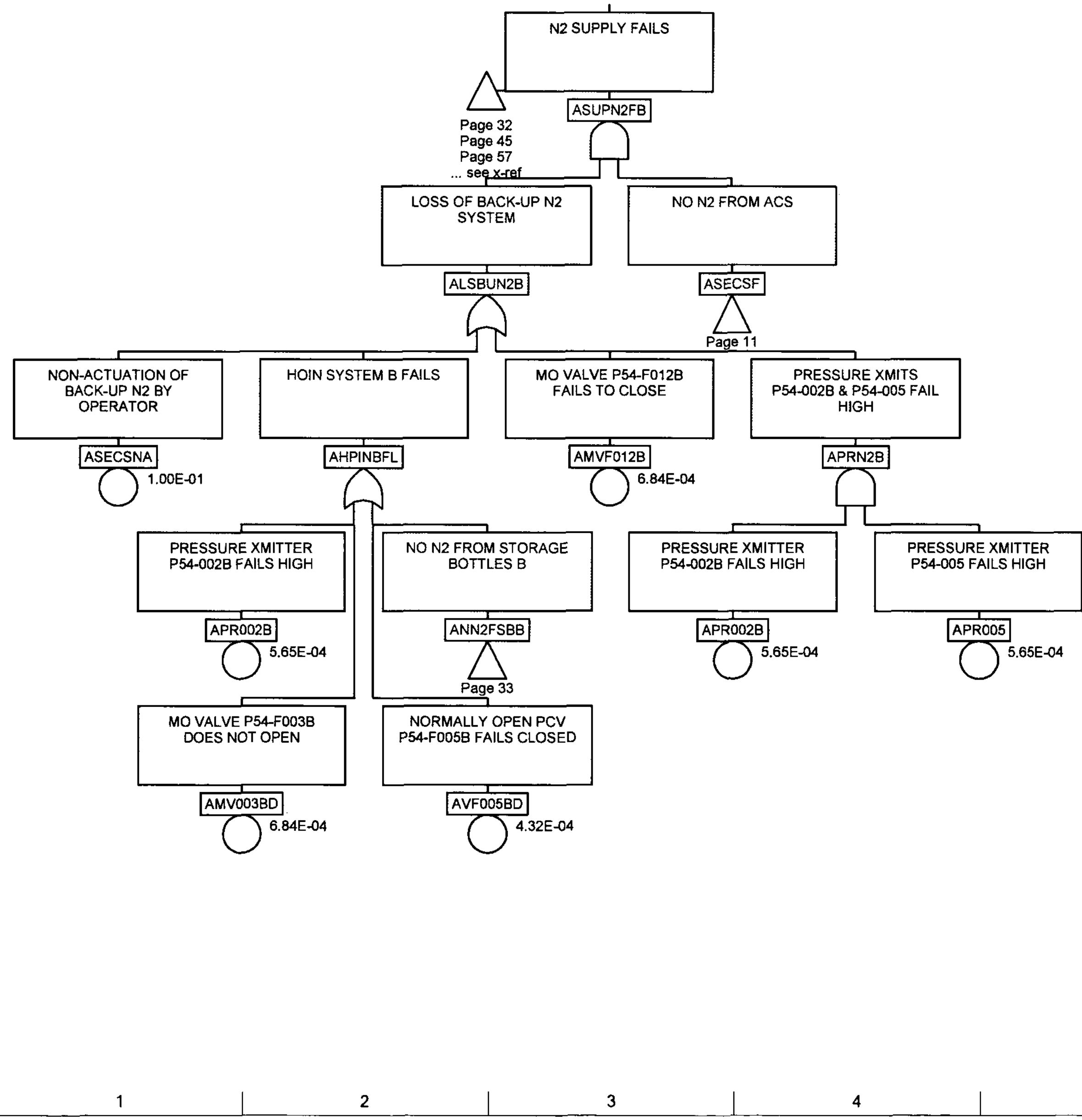
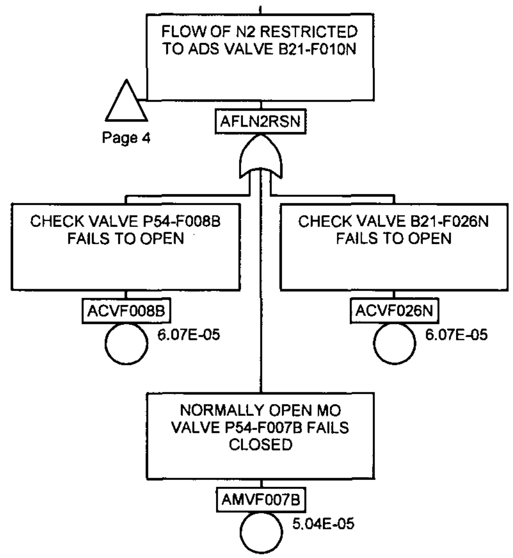


Figure 19D.6-5 Reactor Depressurization System Fault Tree



1 | 2

Figure 19D.6-5 Reactor Depressurization System Fault Tree

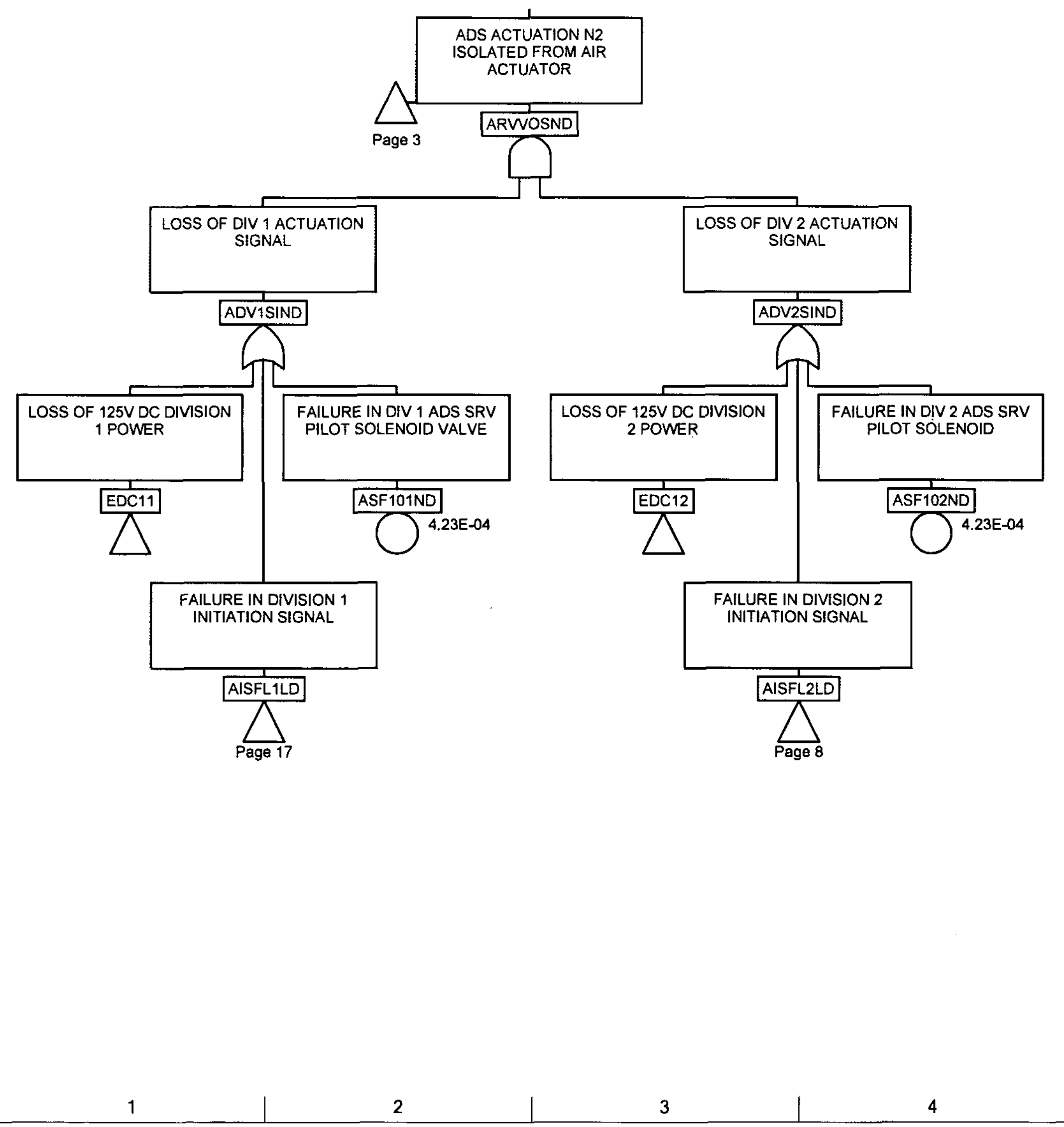


Figure 19D.6-5 Reactor Depressurization System Fault Tree

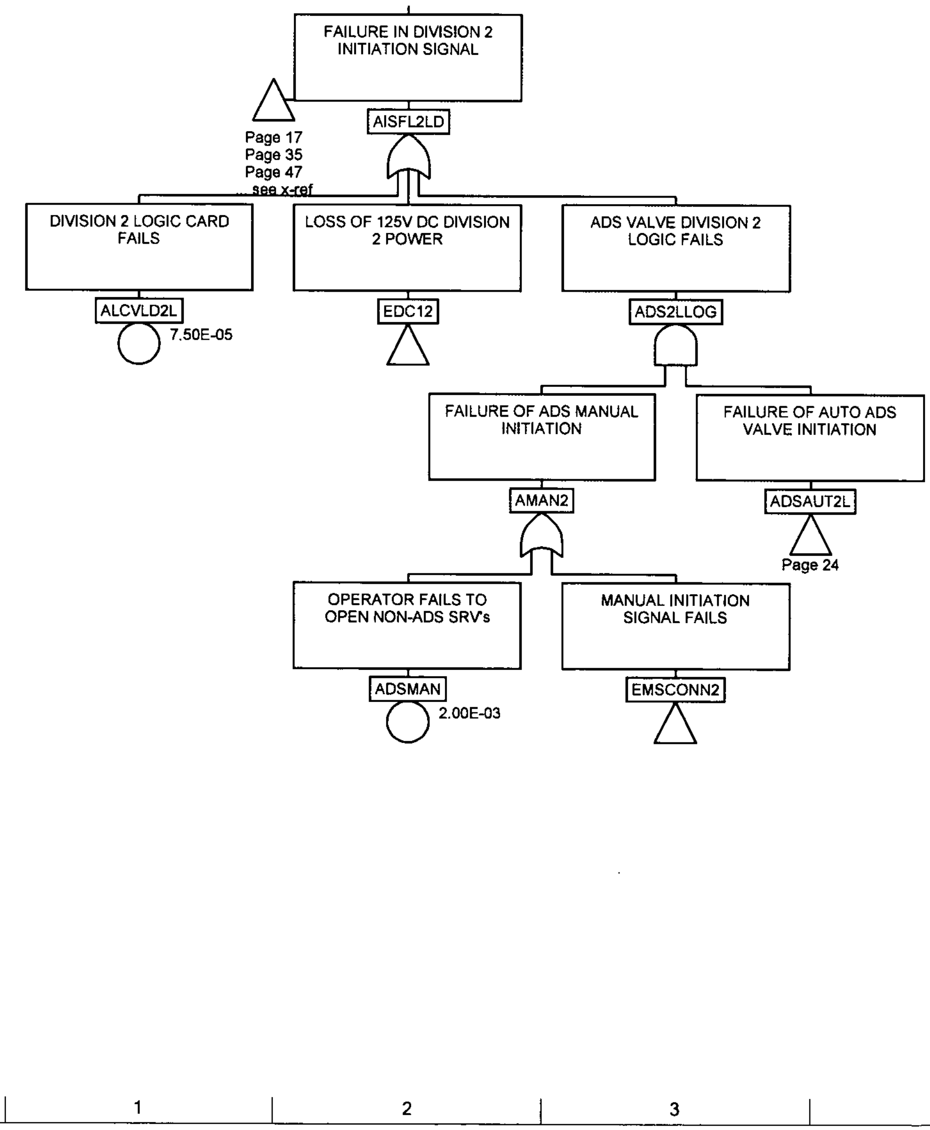


Figure 19D.6-5 Reactor Depressurization System Fault Tree

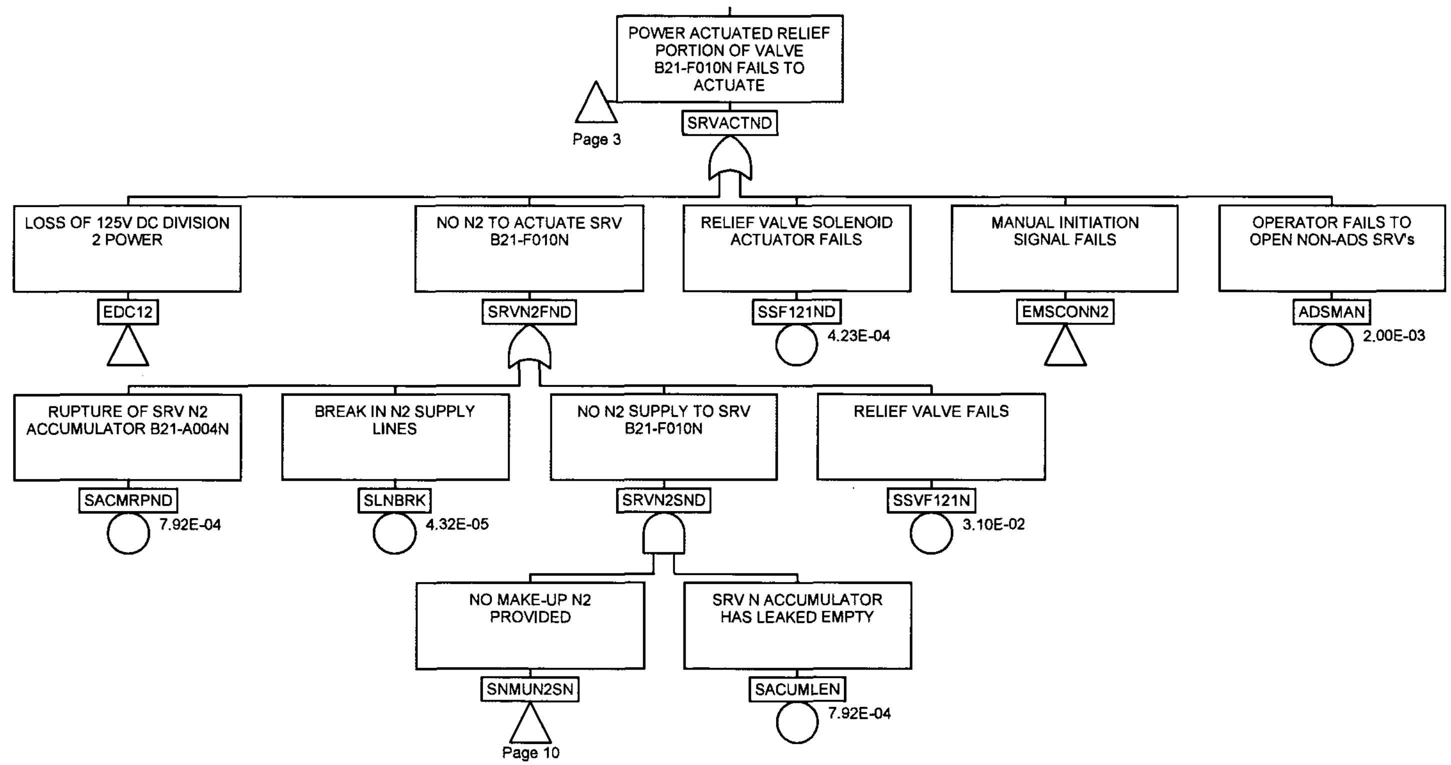


Figure 19D.6-5 Reactor Depressurization System Fault Tree

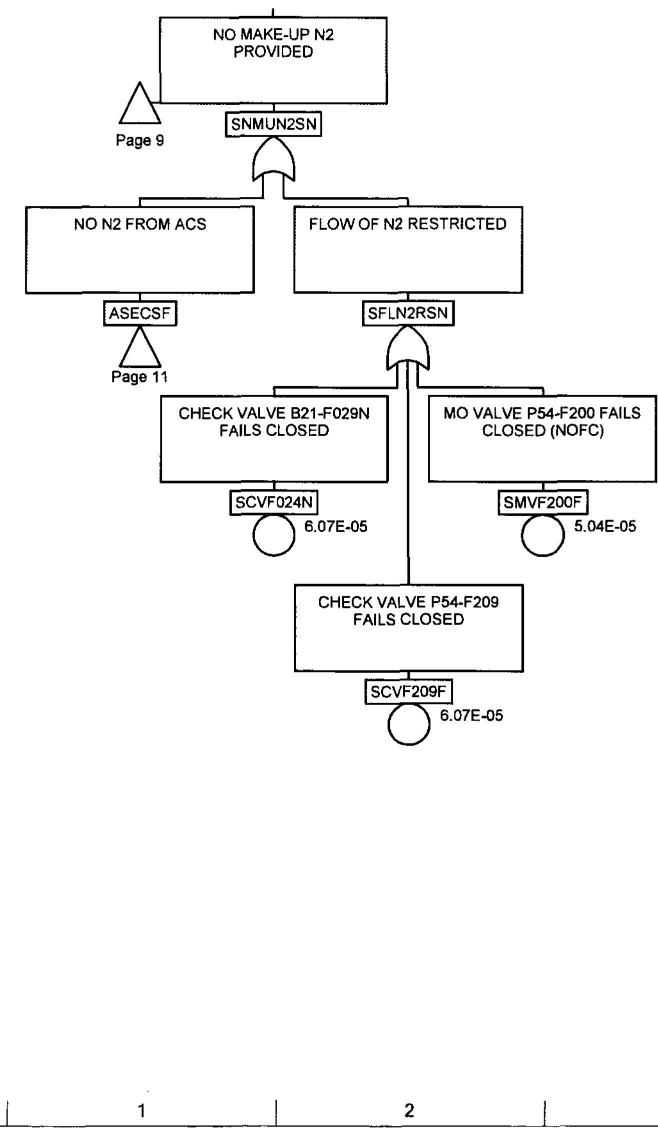


Figure 19D.6-5 Reactor Depressurization System Fault Tree

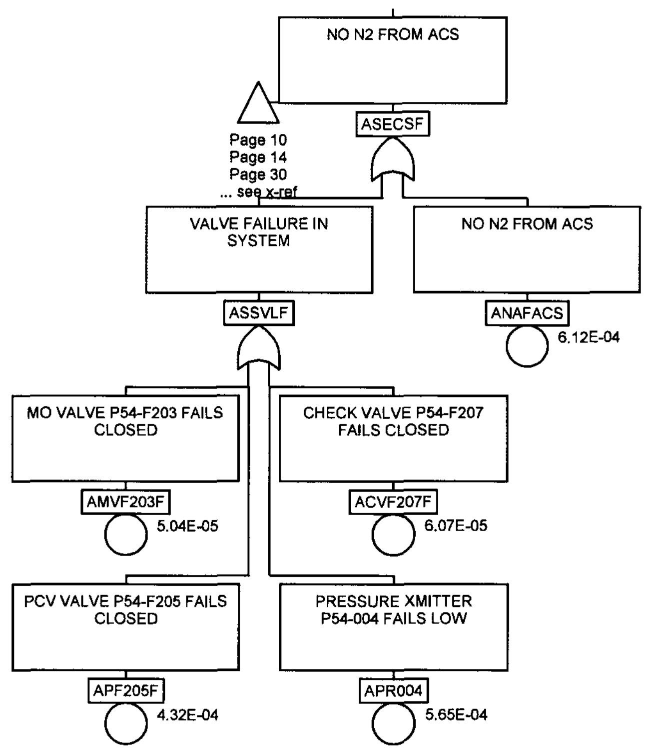


Figure 19D.6-5 Reactor Depressurization System Fault Tree

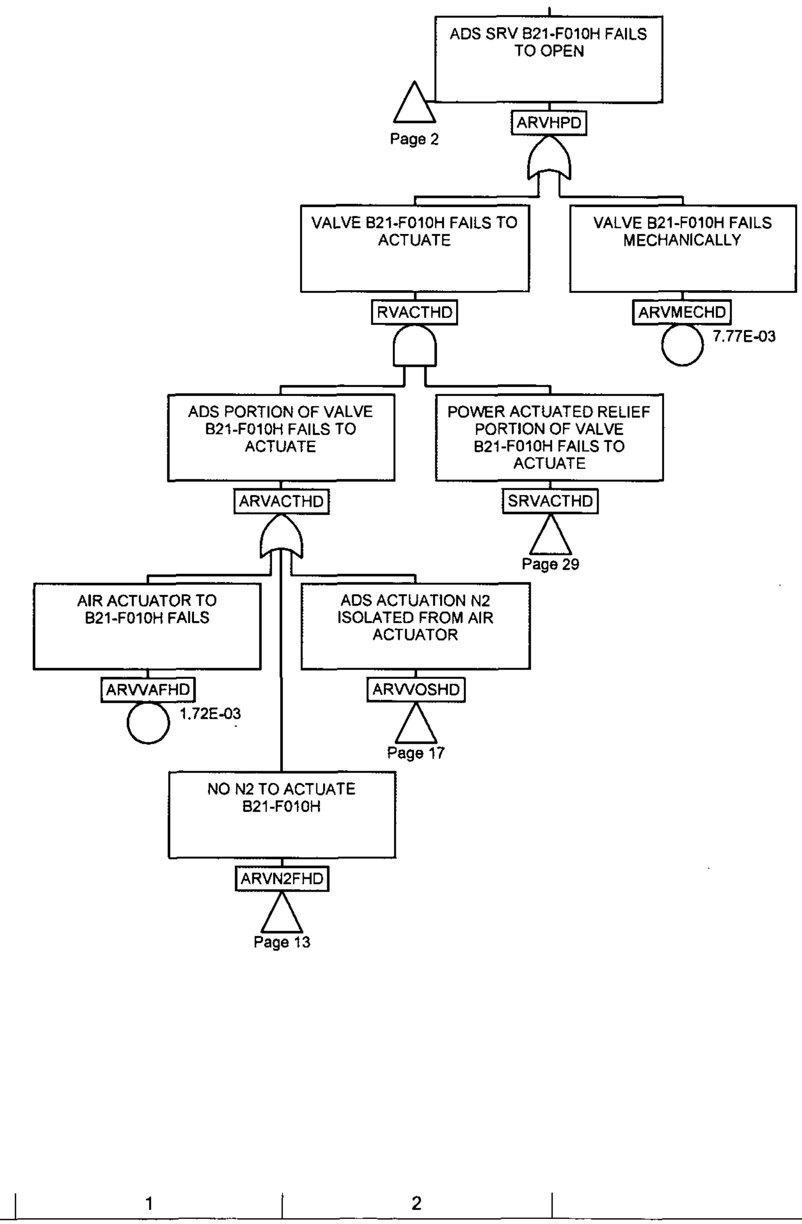


Figure 19D.6-5 Reactor Depressurization System Fault Tree

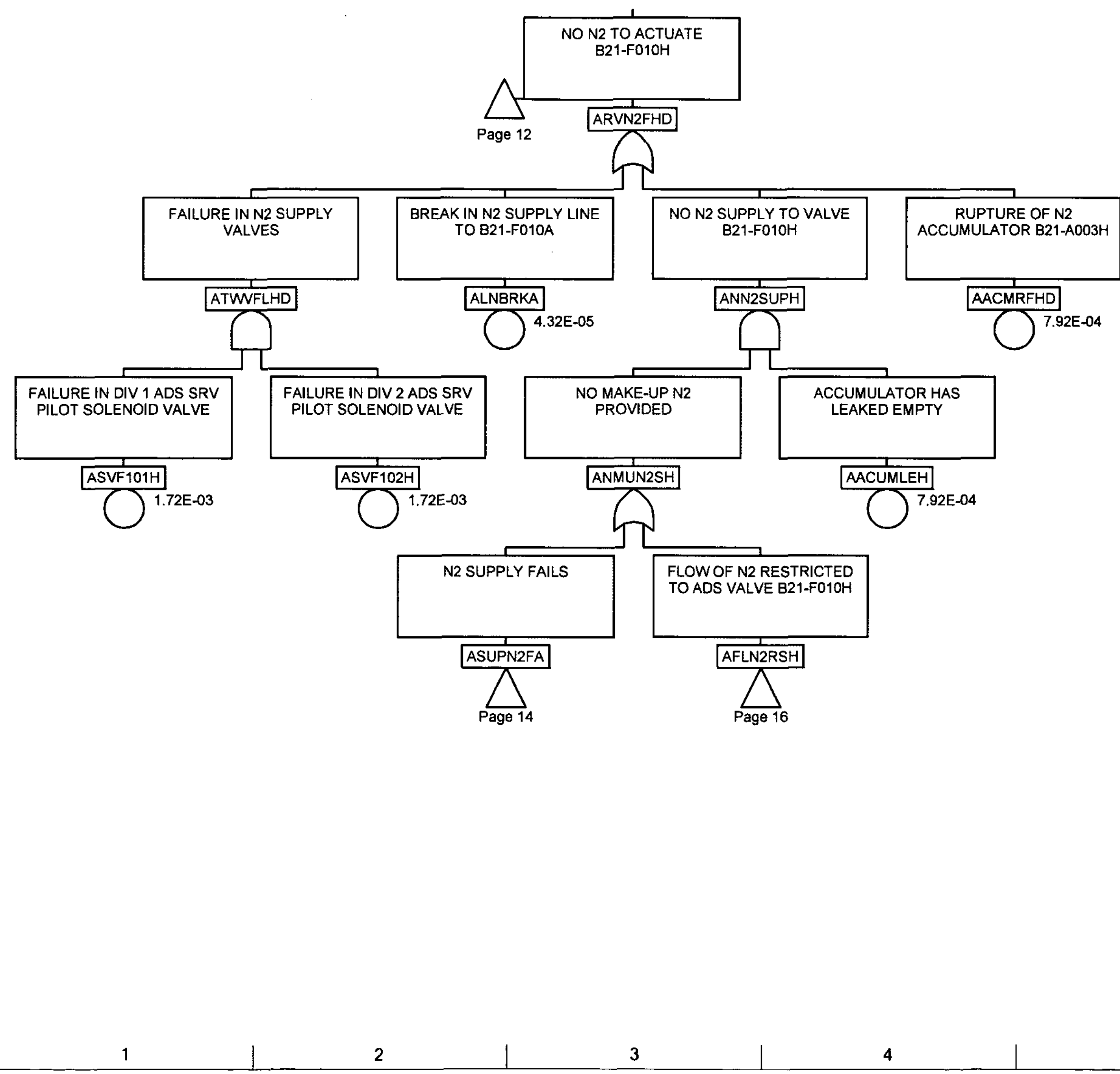


Figure 19D.6-5 Reactor Depressurization System Fault Tree

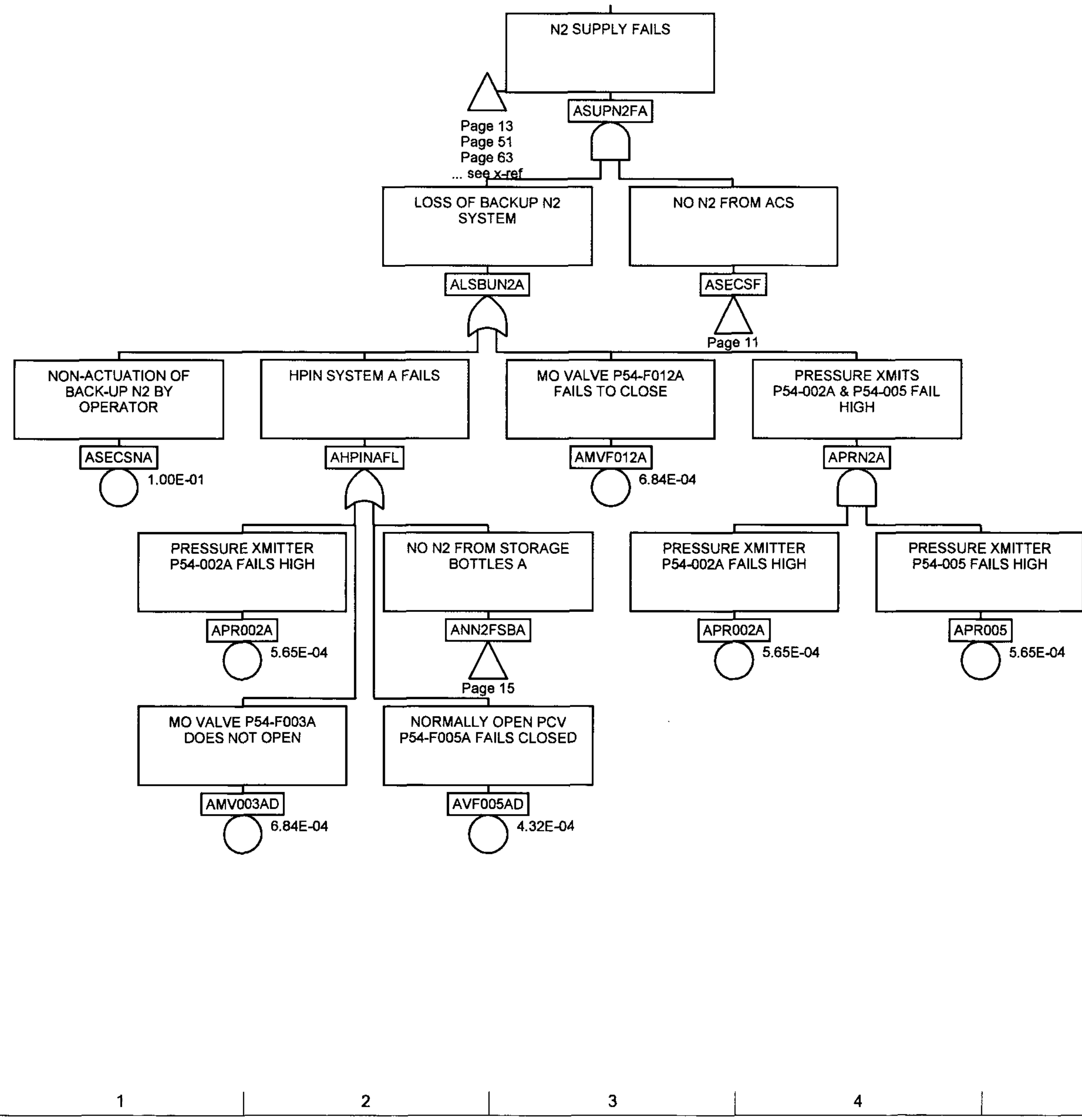


Figure 19D.6-5 Reactor Depressurization System Fault Tree

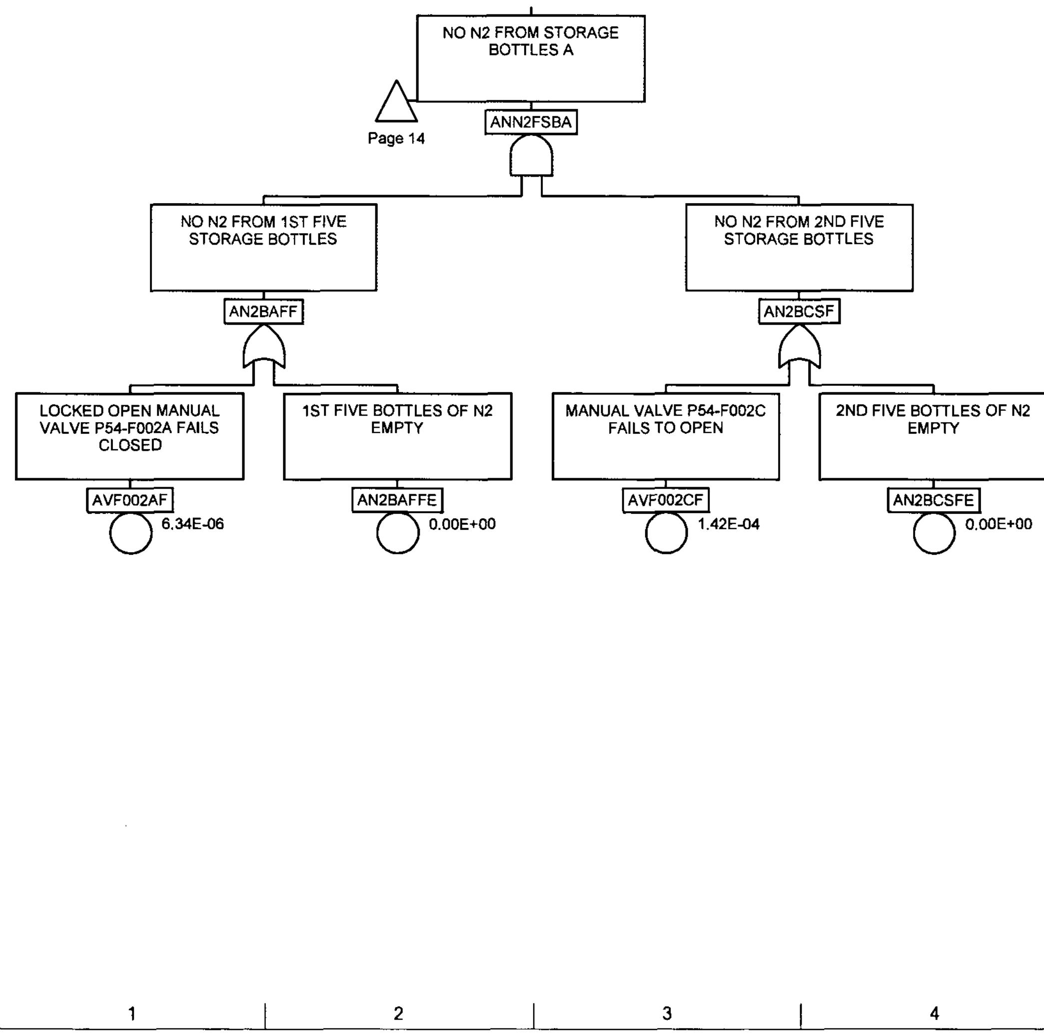


Figure 19D.6-5 Reactor Depressurization System Fault Tree

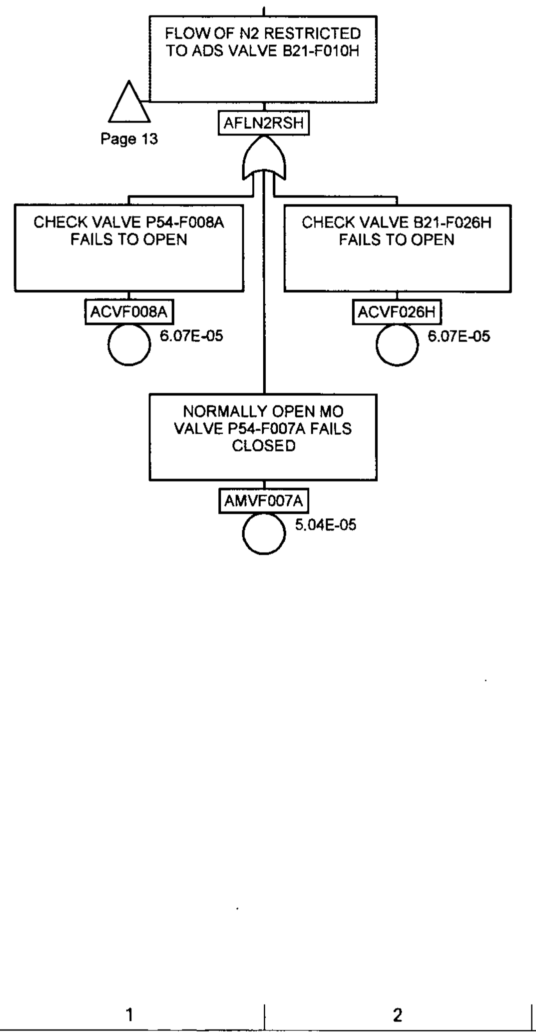


Figure 19D.6-5 Reactor Depressurization System Fault Tree

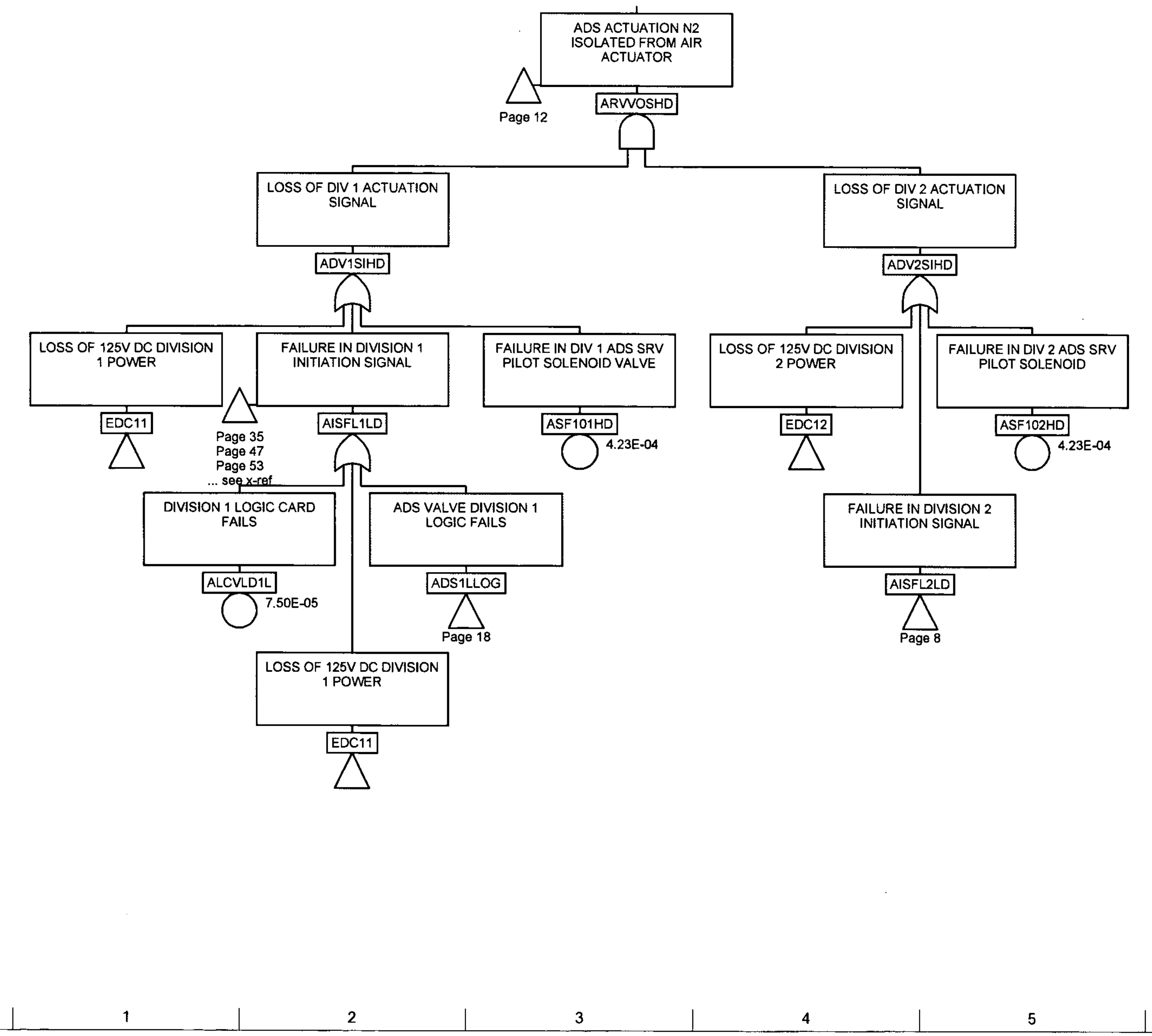


Figure 19D.6-5 Reactor Depressurization System Fault Tree

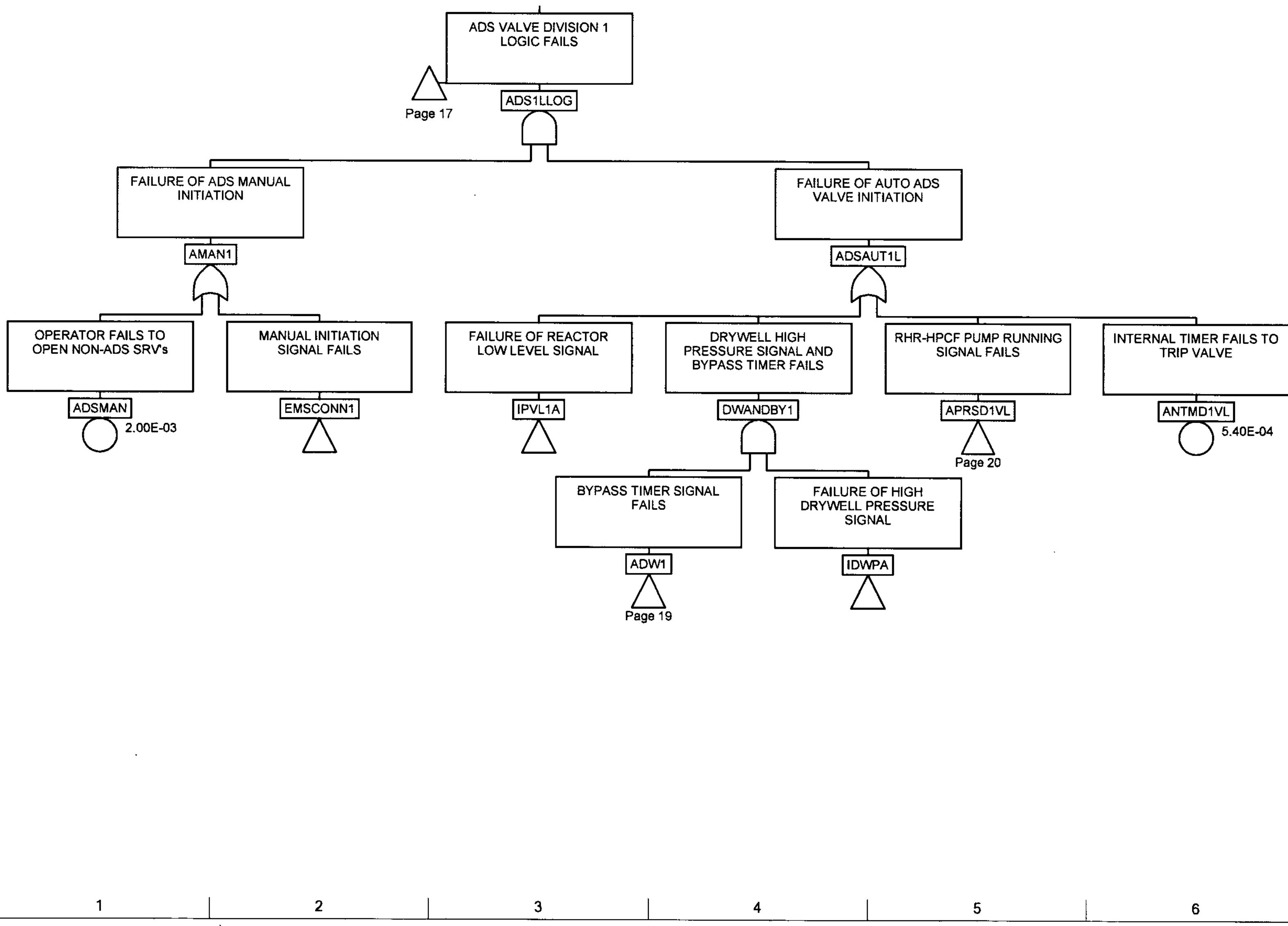


Figure 19D.6-5 Reactor Depressurization System Fault Tree

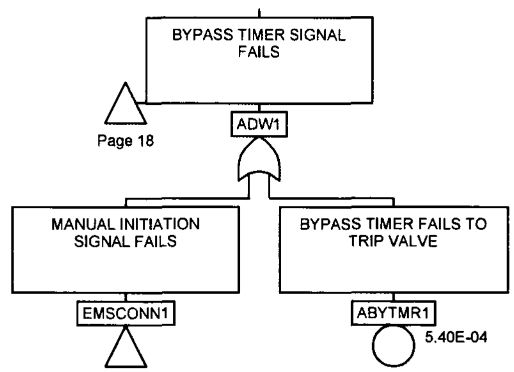


Figure 19D.6-5 Reactor Depressurization System Fault Tree

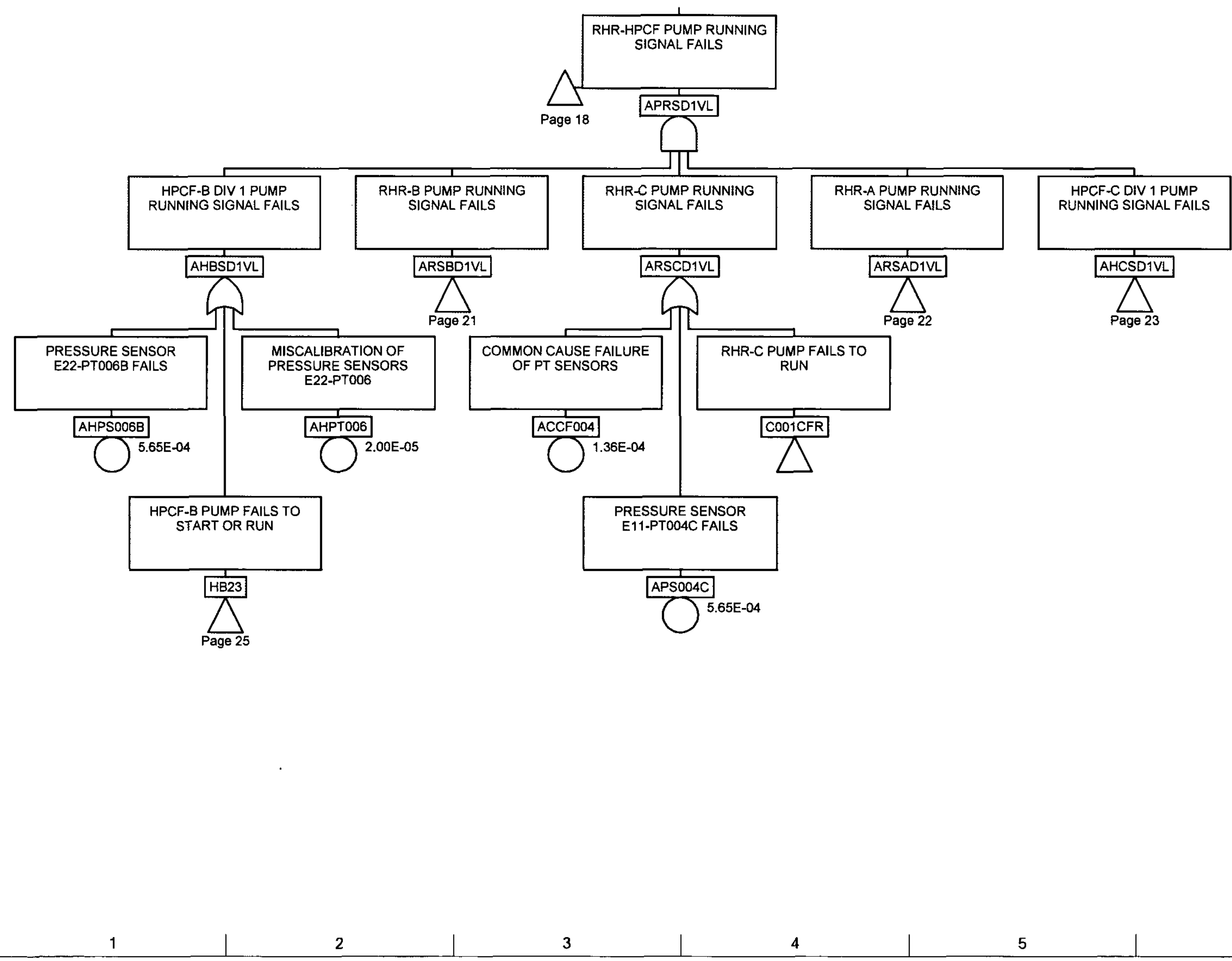


Figure 19D.6-5 Reactor Depressurization System Fault Tree

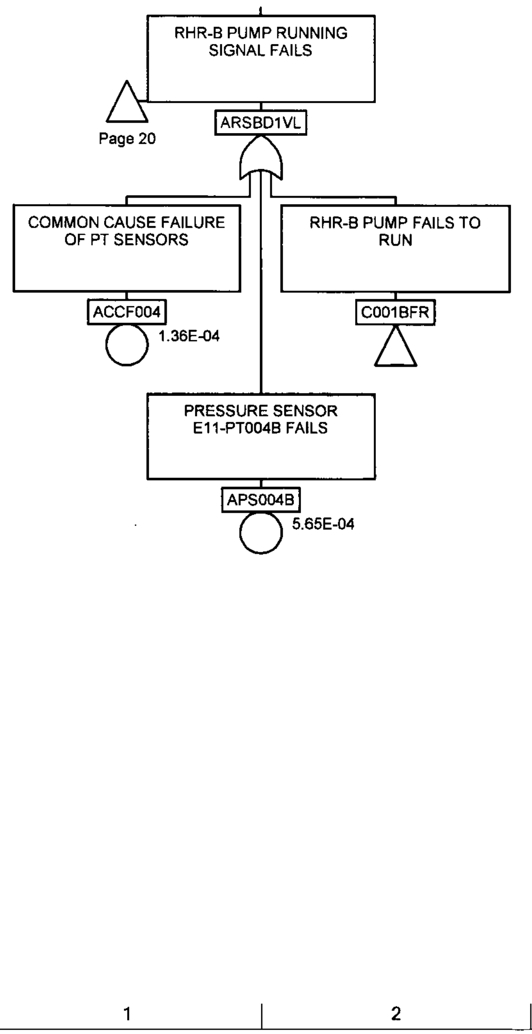
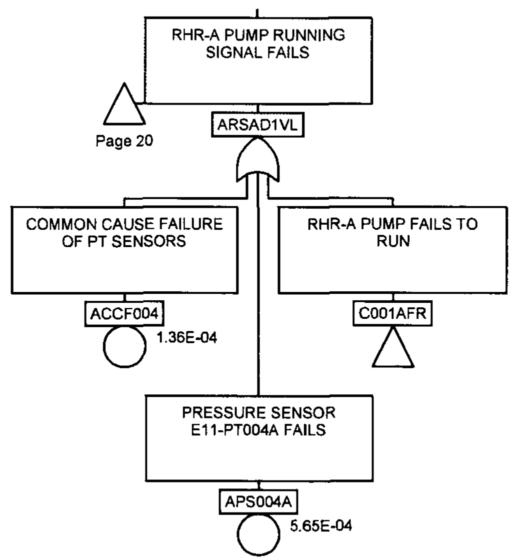


Figure 19D.6-5 Reactor Depressurization System Fault Tree



1 2

Figure 19D.6-5 Reactor Depressurization System Fault Tree

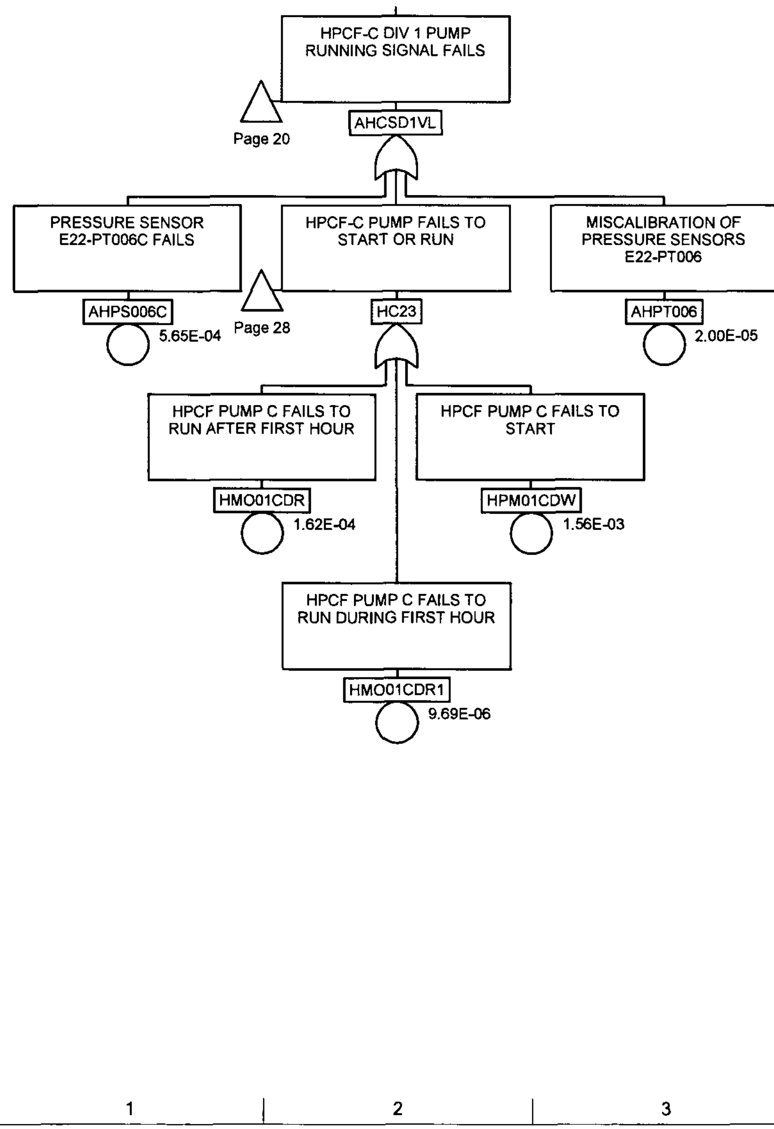


Figure 19D.6-5 Reactor Depressurization System Fault Tree

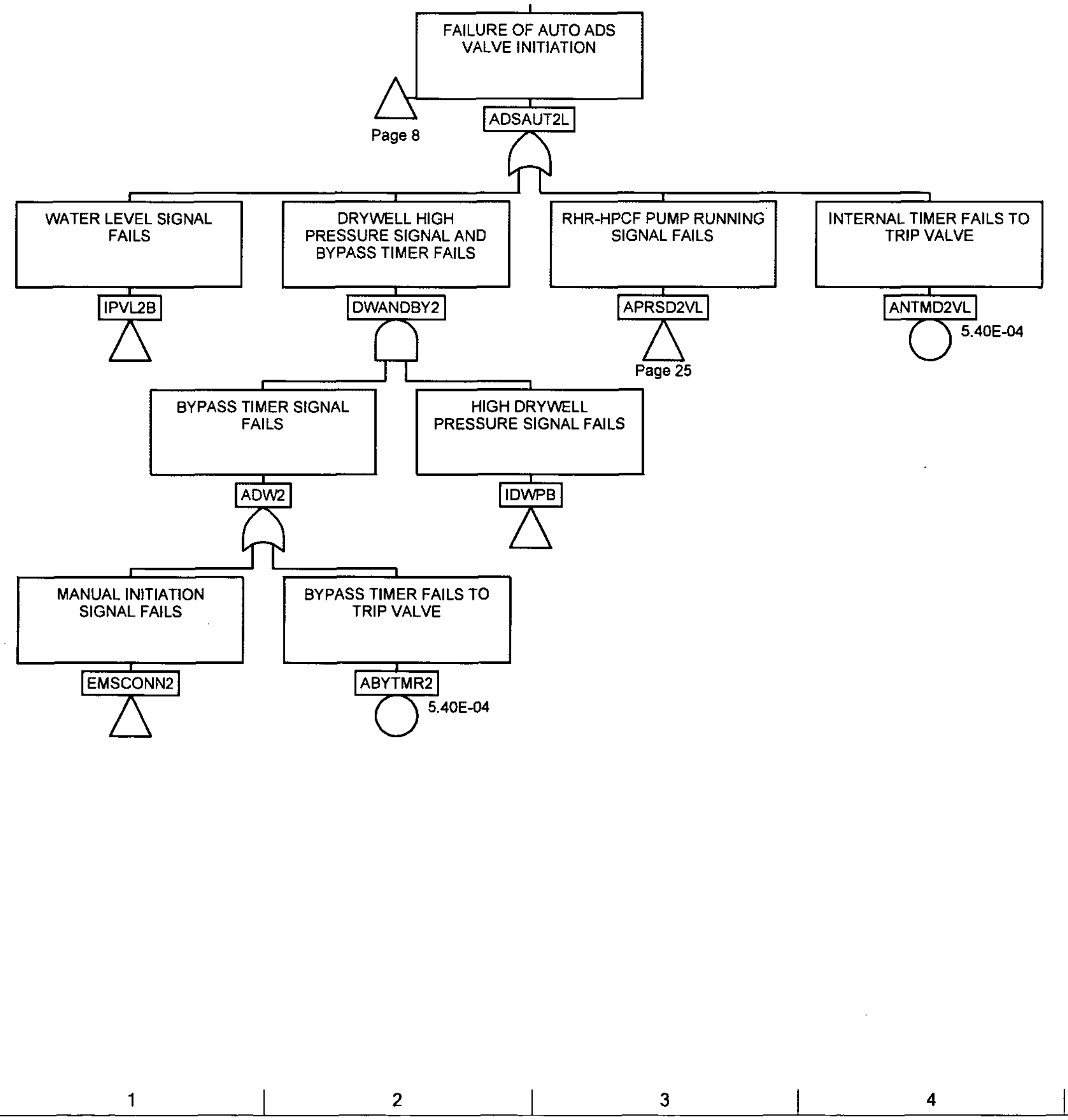


Figure 19D.6-5 Reactor Depressurization System Fault Tree

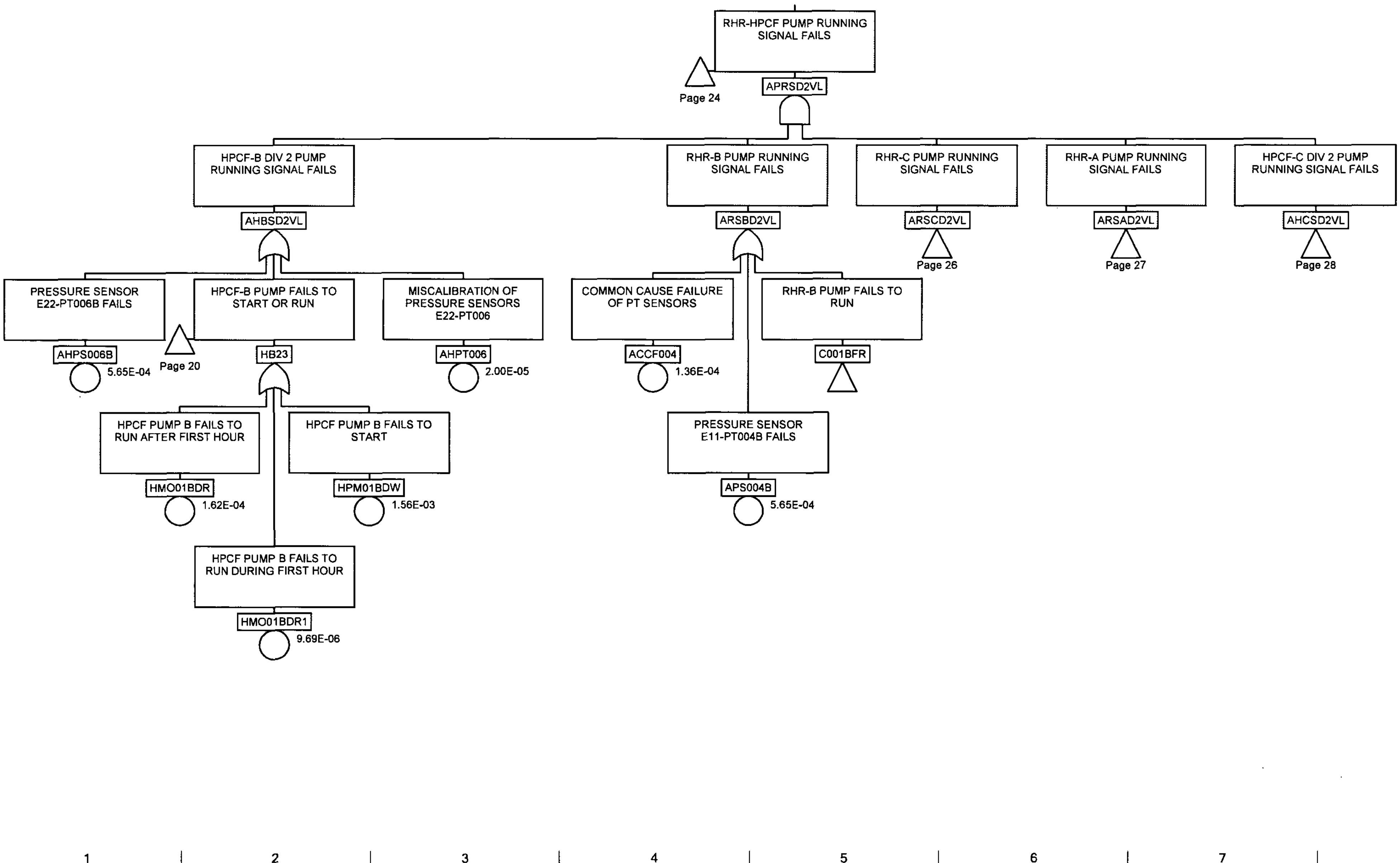


Figure 19D.6-5 Reactor Depressurization System Fault Tree

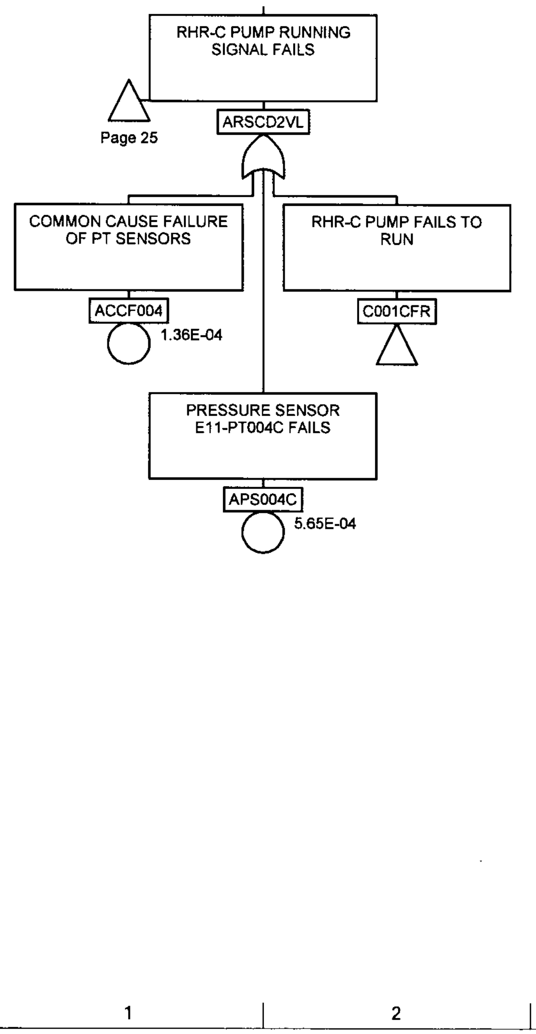
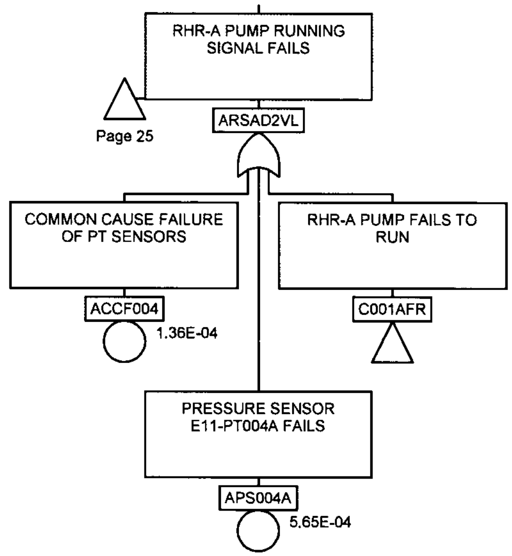


Figure 19D.6-5 Reactor Depressurization System Fault Tree


RHR-A PUMP RUNNING SIGNAL FAILS
Page 25
ARSAD2VL
COMMON CAUSE FAILURE OF PT SENSORS
RHR-A PUMP FAILS TO RUN
ACCF004
1.36E-04
C001AFR
PRESSURE SENSOR E11-PT004A FAILS
APS004A
5.65E-04

1 2

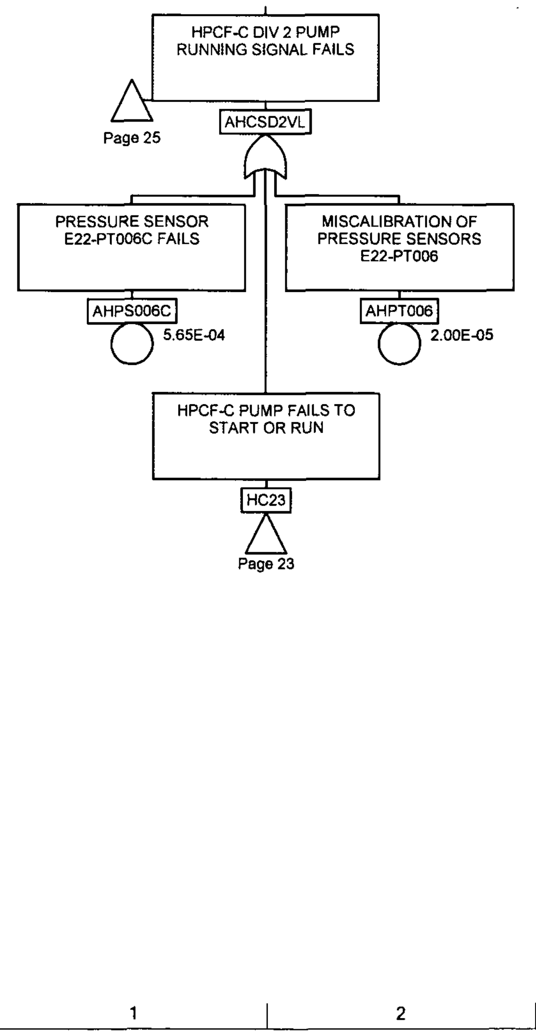


Figure 19D.6-5 Reactor Depressurization System Fault Tree

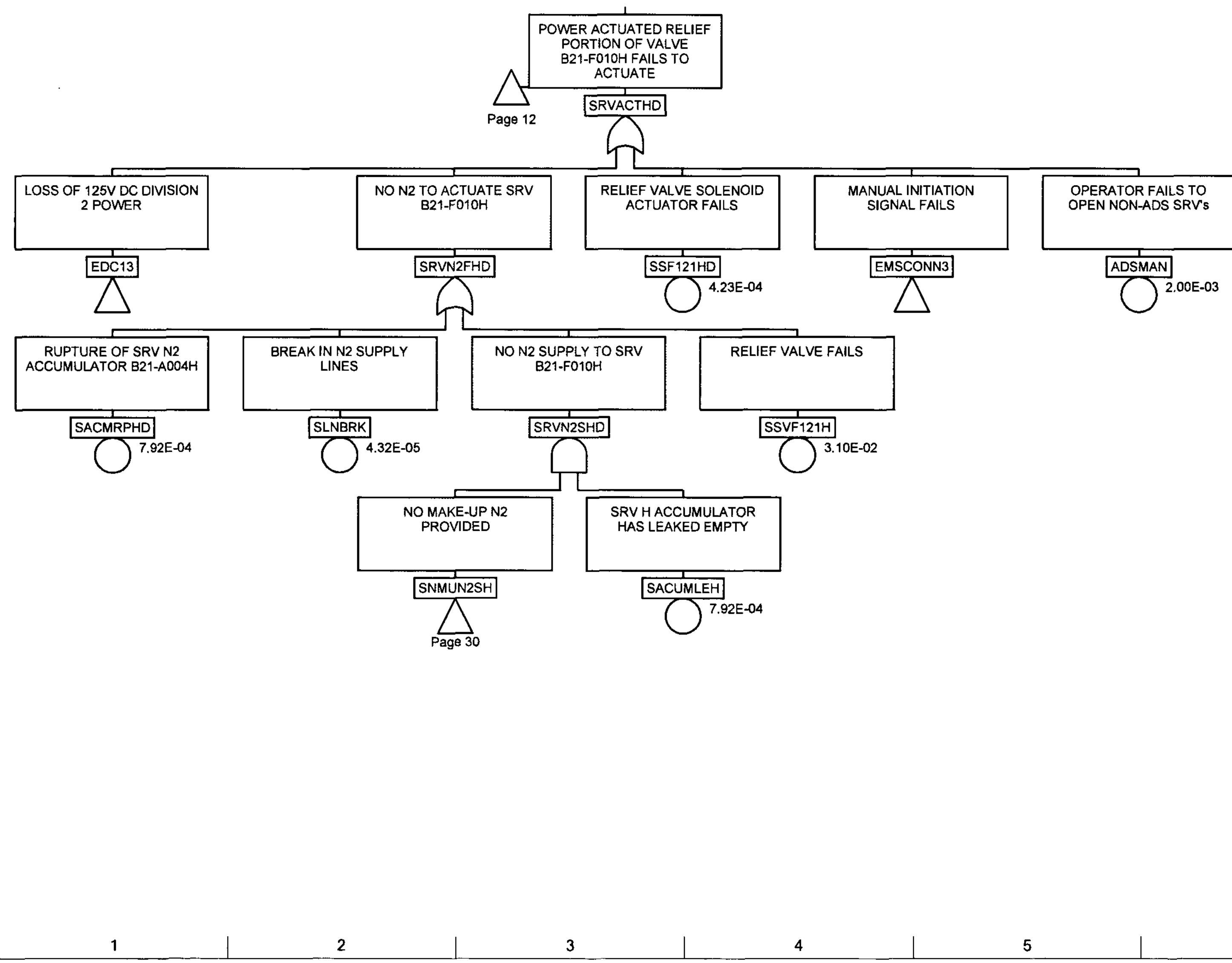
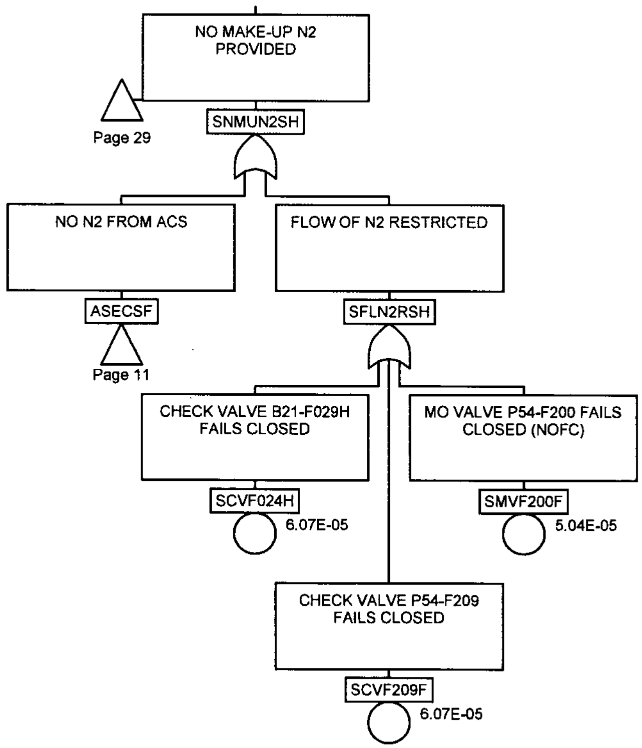


Figure 19D.6-5 Reactor Depressurization System Fault Tree



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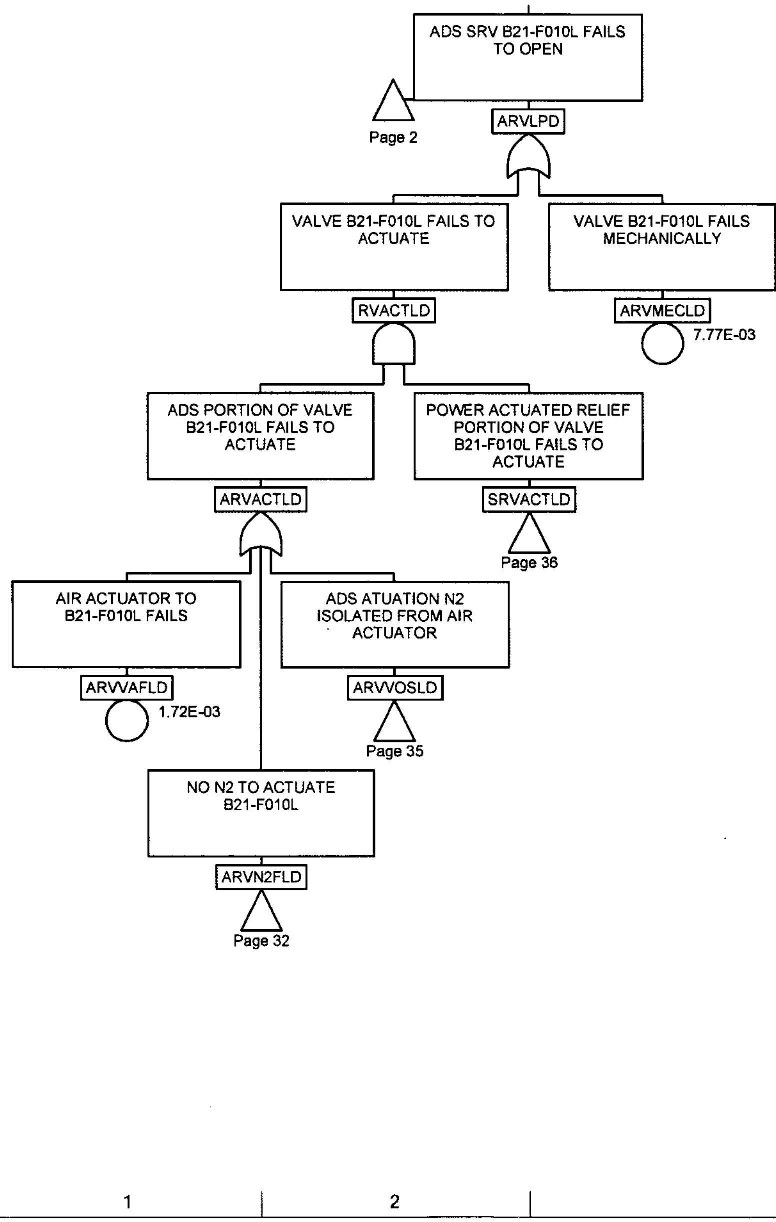


Figure 19D.6-5 Reactor Depressurization System Fault Tree

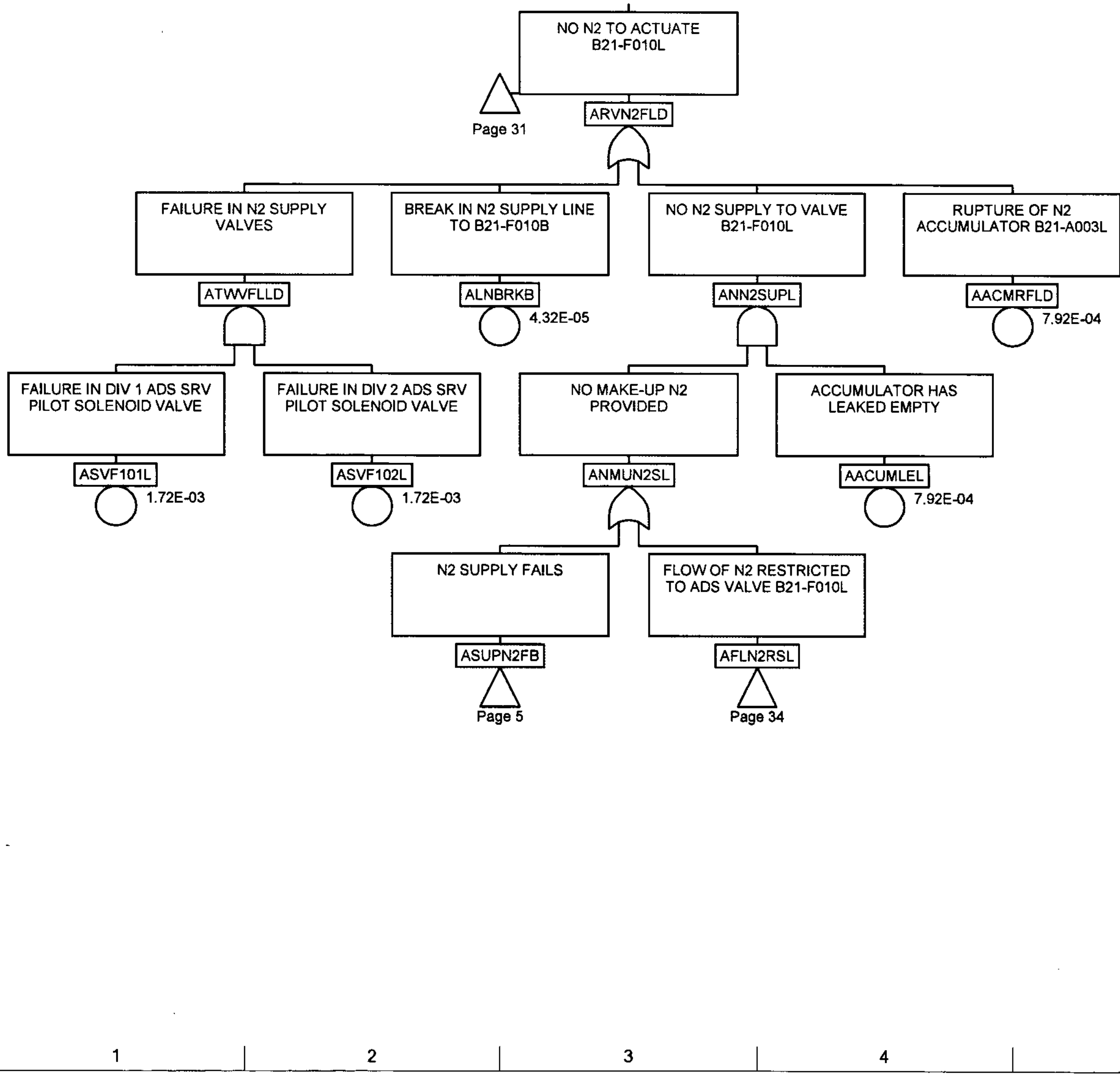
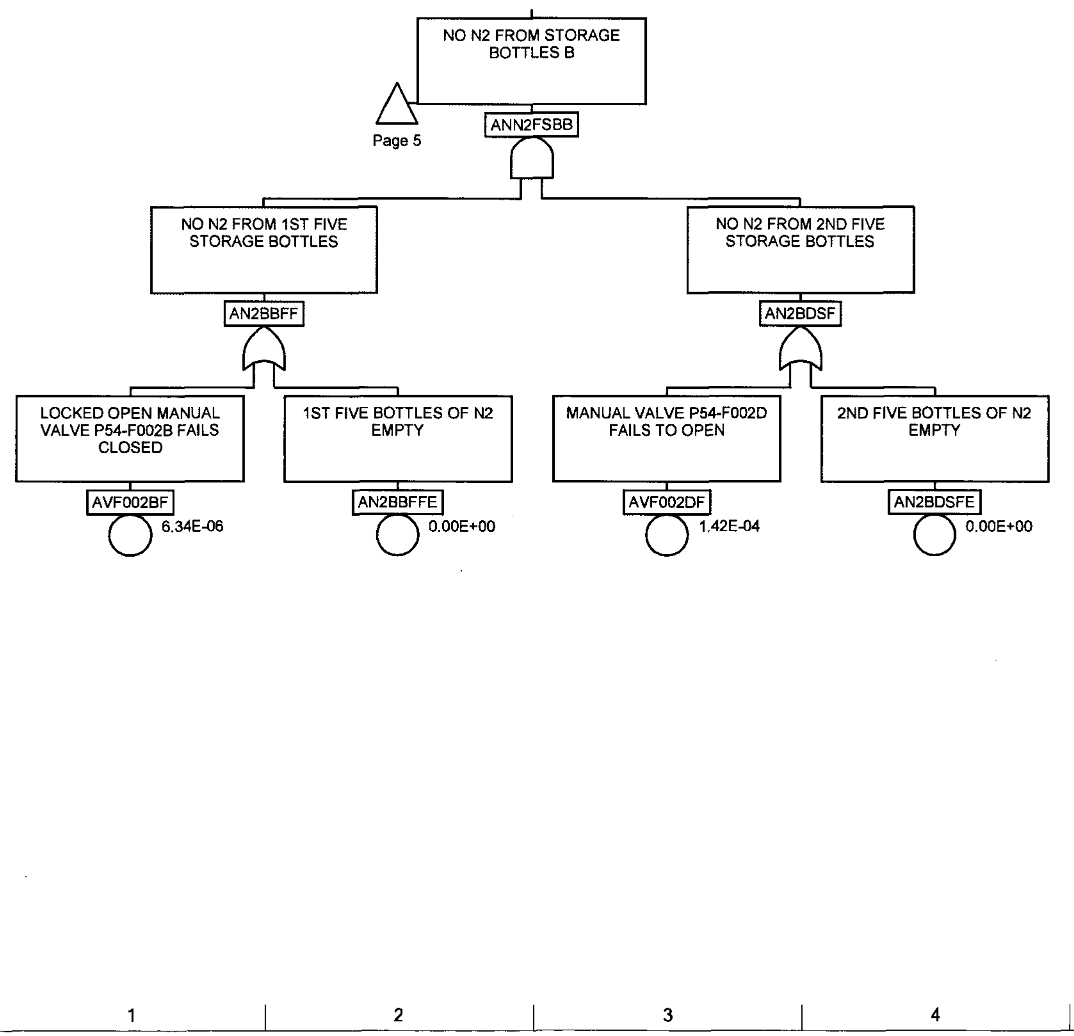


Figure 19D.6-5 Reactor Depressurization System Fault Tree



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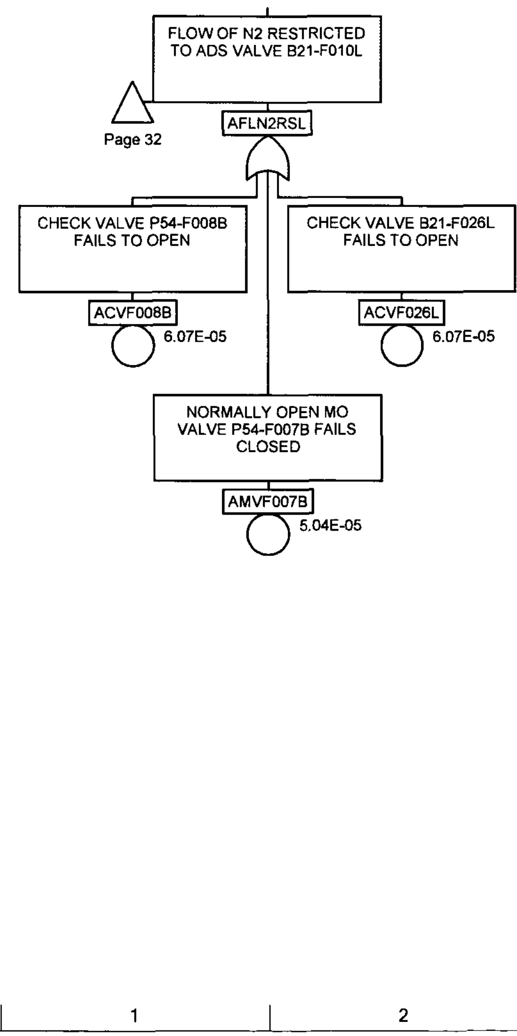
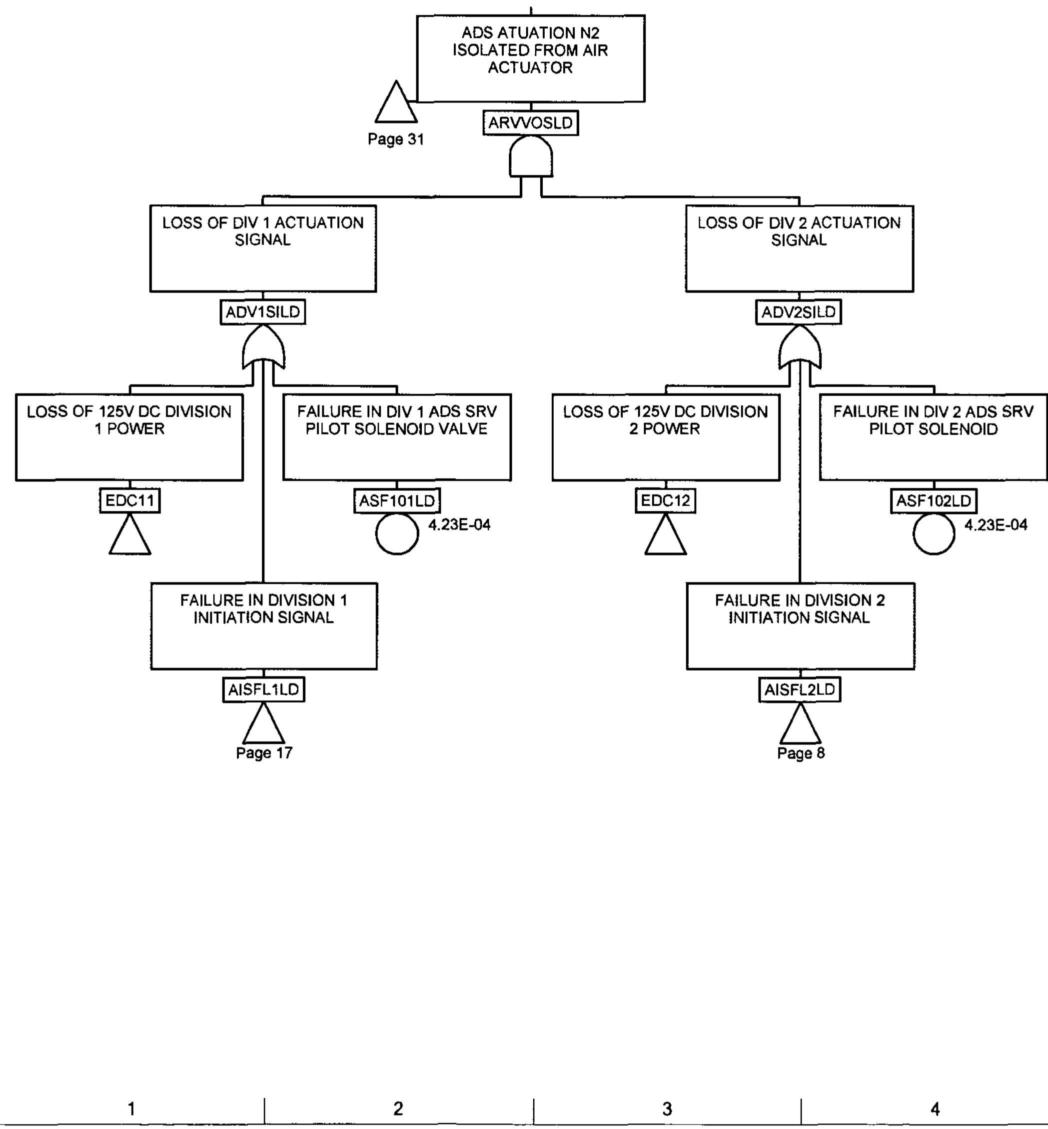


Figure 19D.6-5 Reactor Depressurization System Fault Tree



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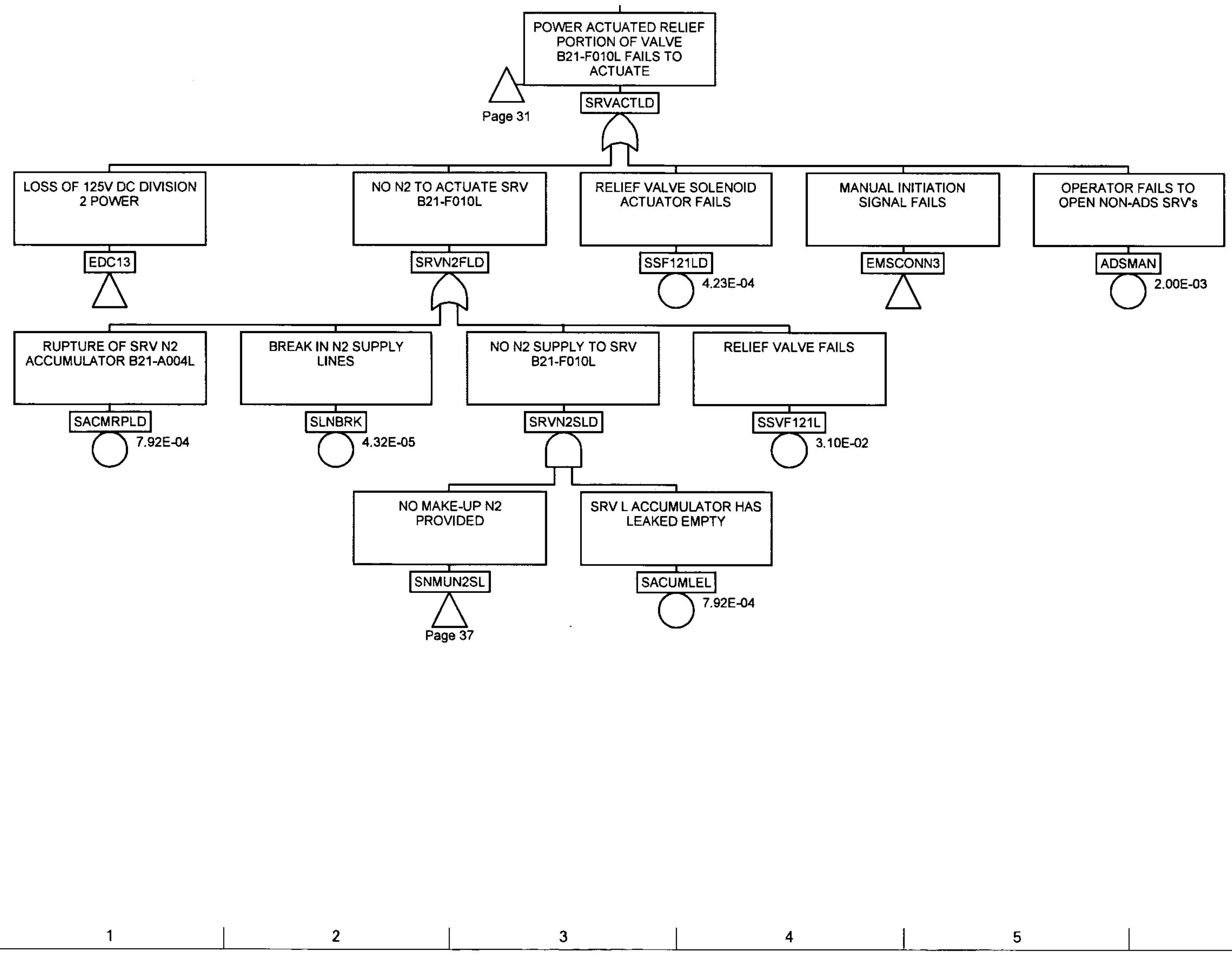

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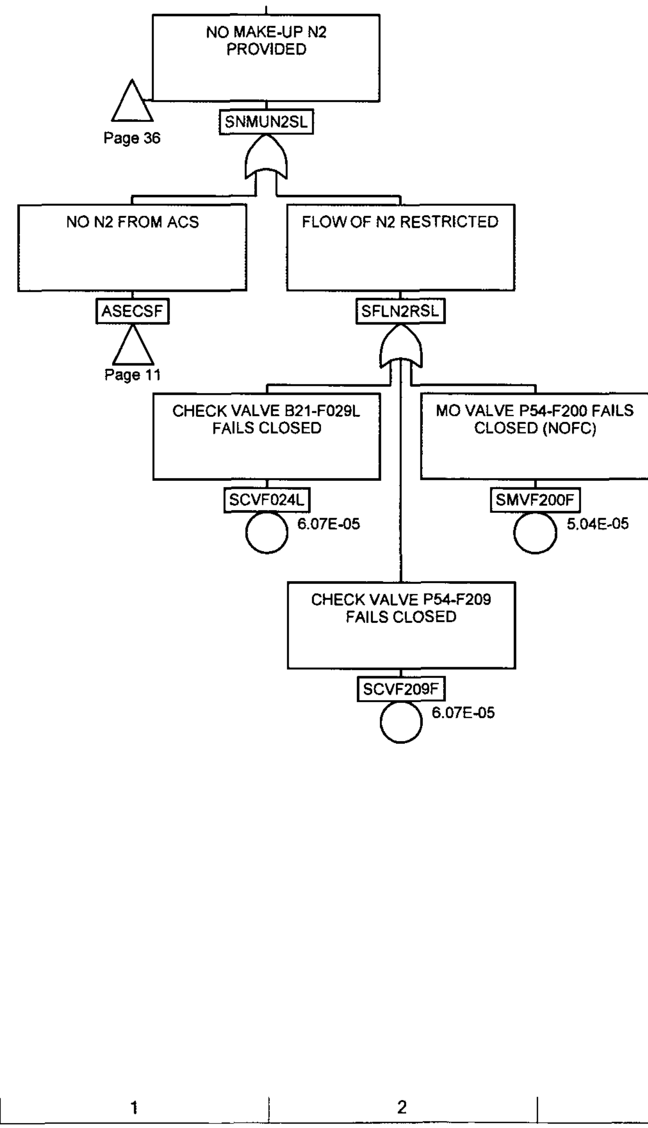

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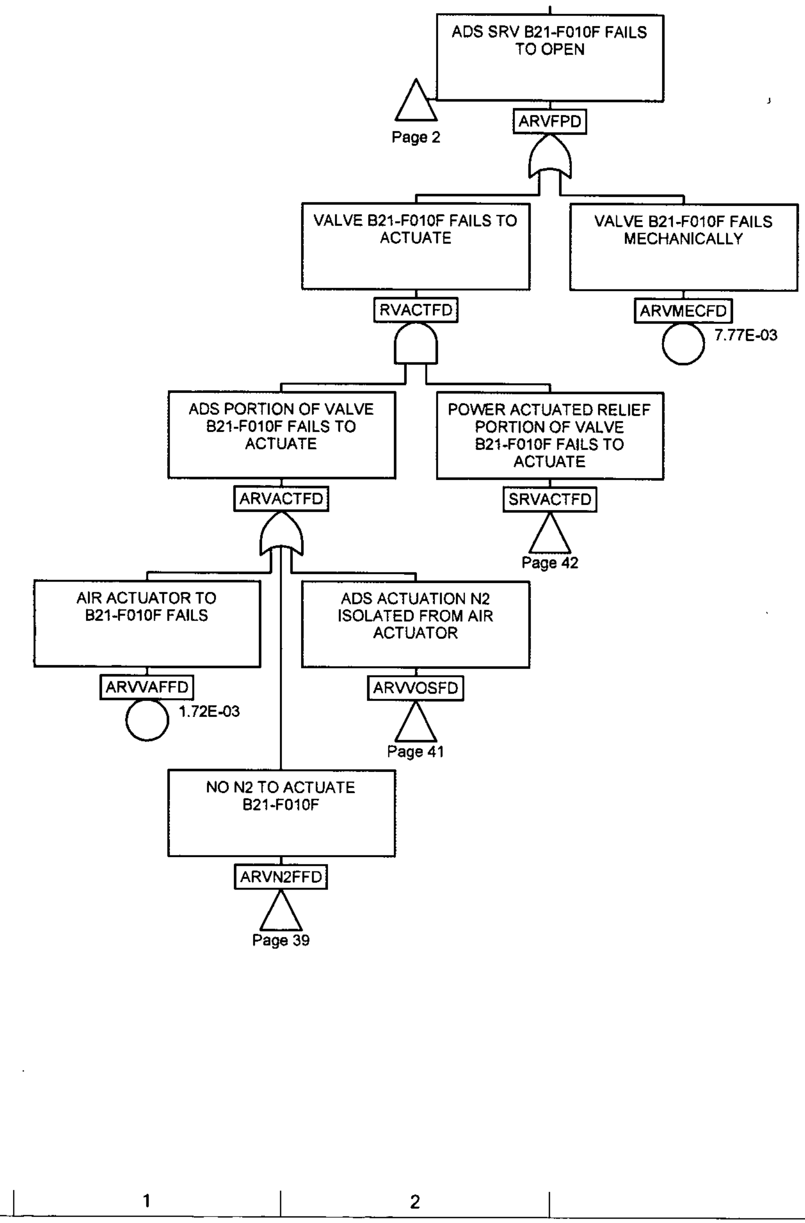

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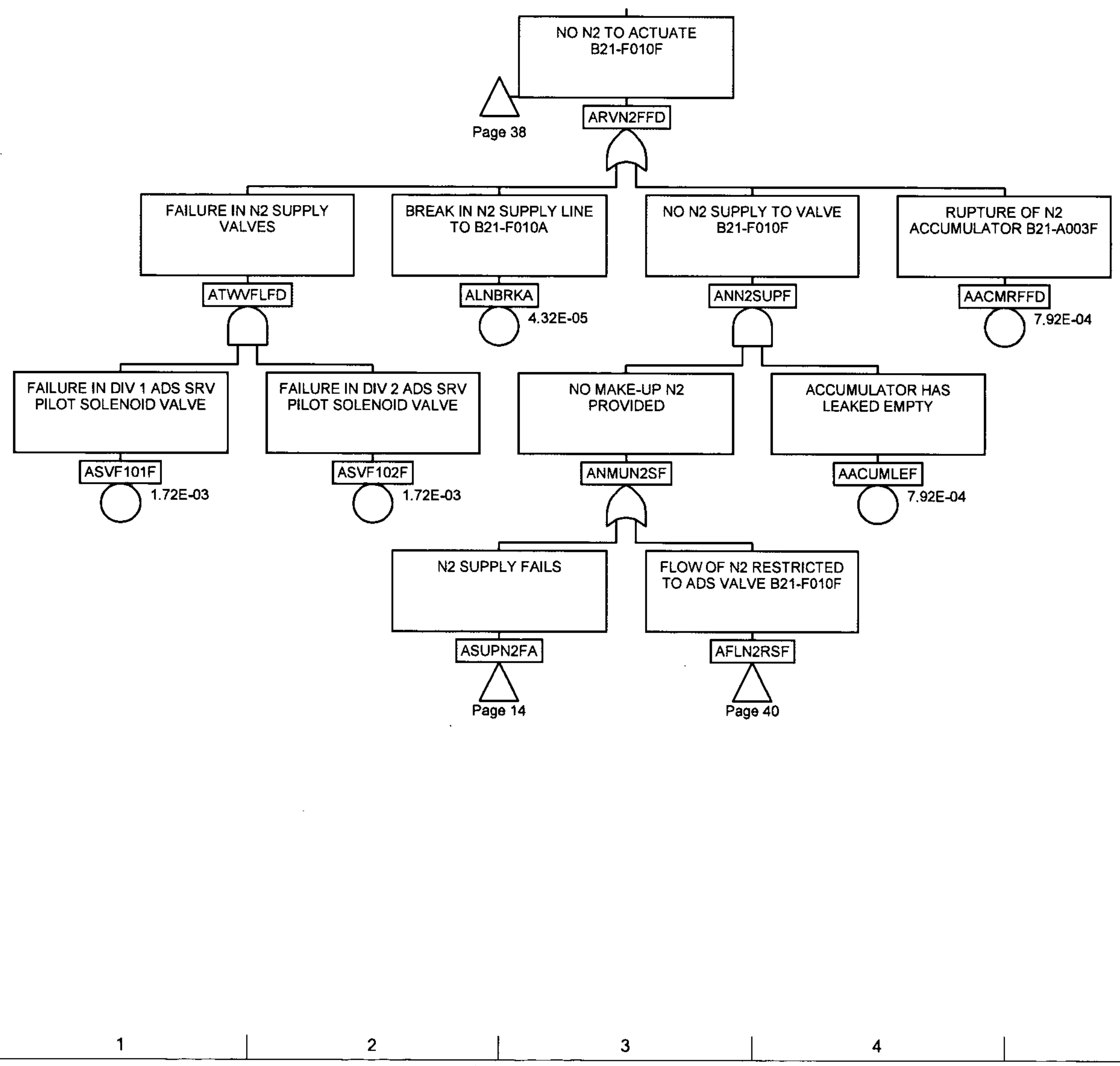

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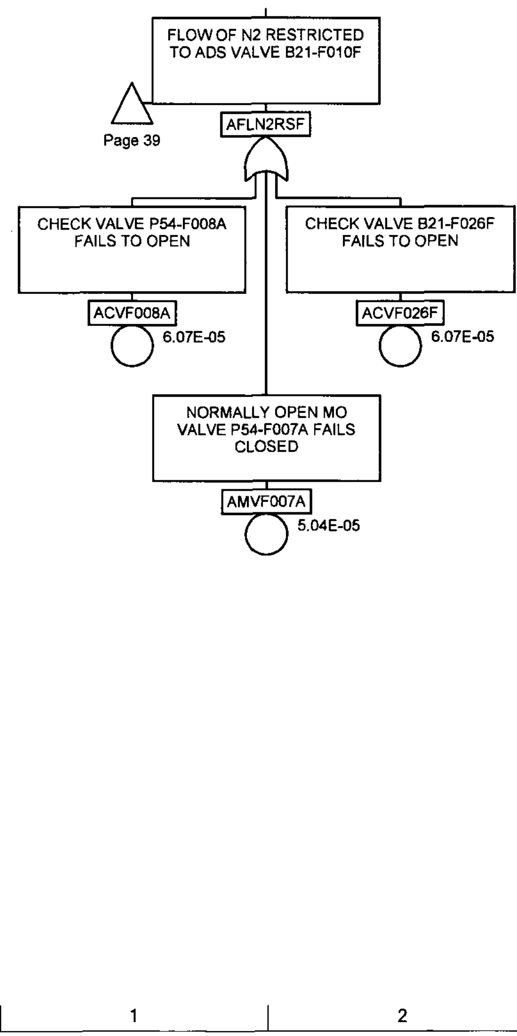

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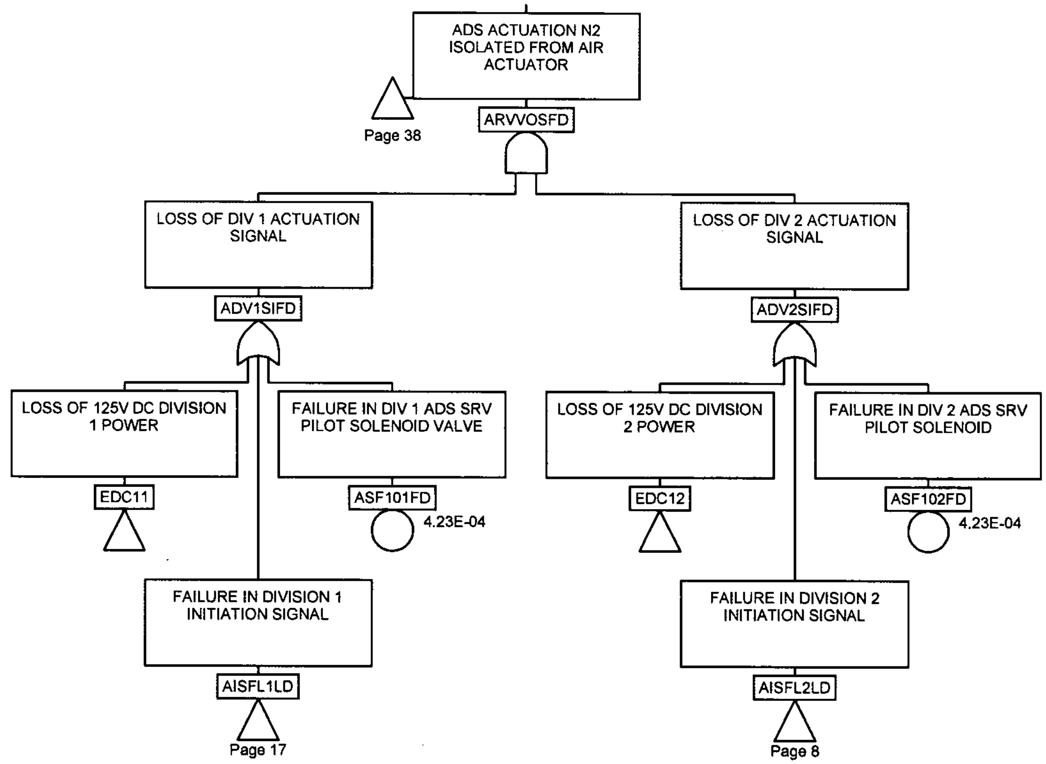
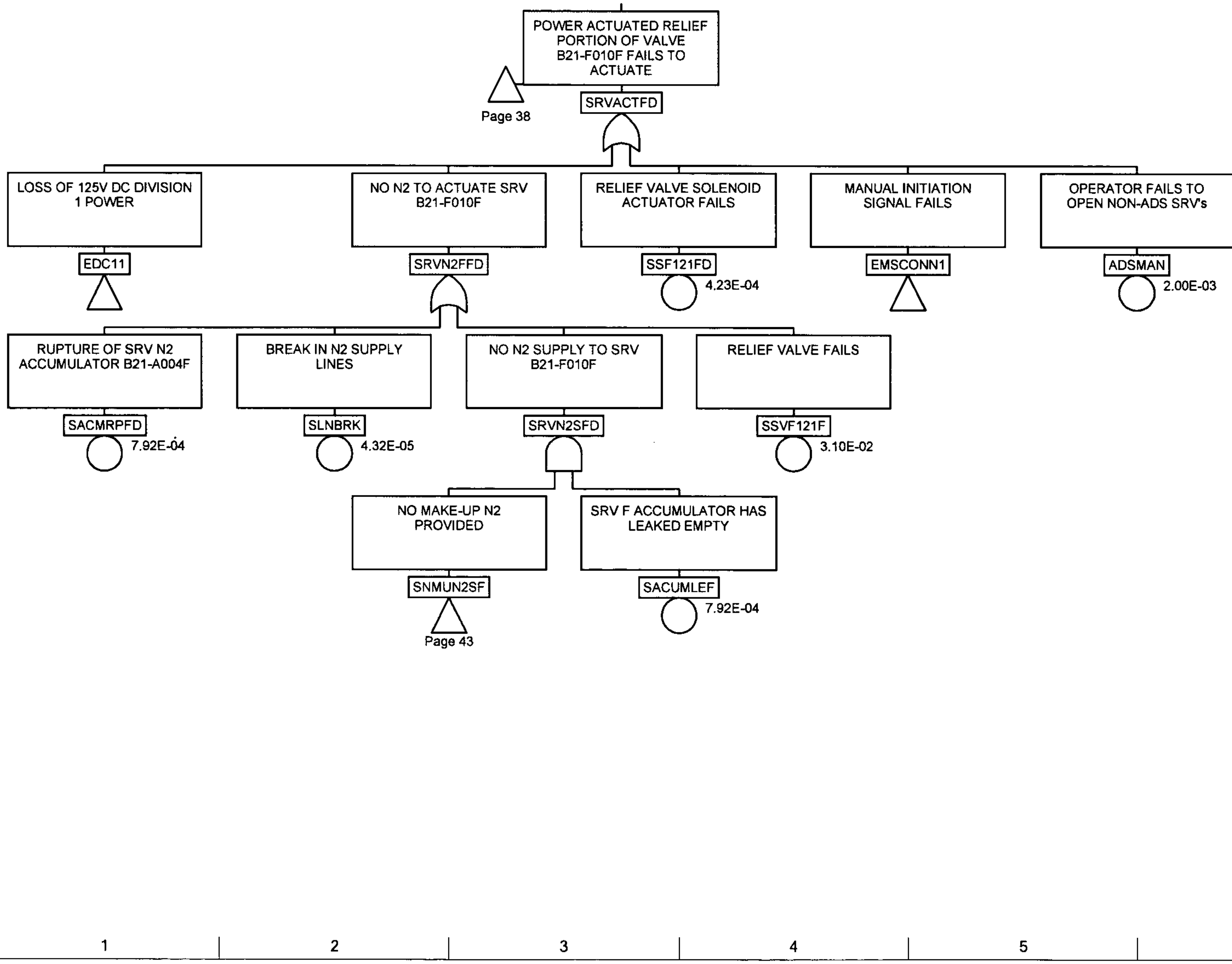


1	2	3	4

Figure 19D.6-5 Reactor Depressurization System Fault Tree



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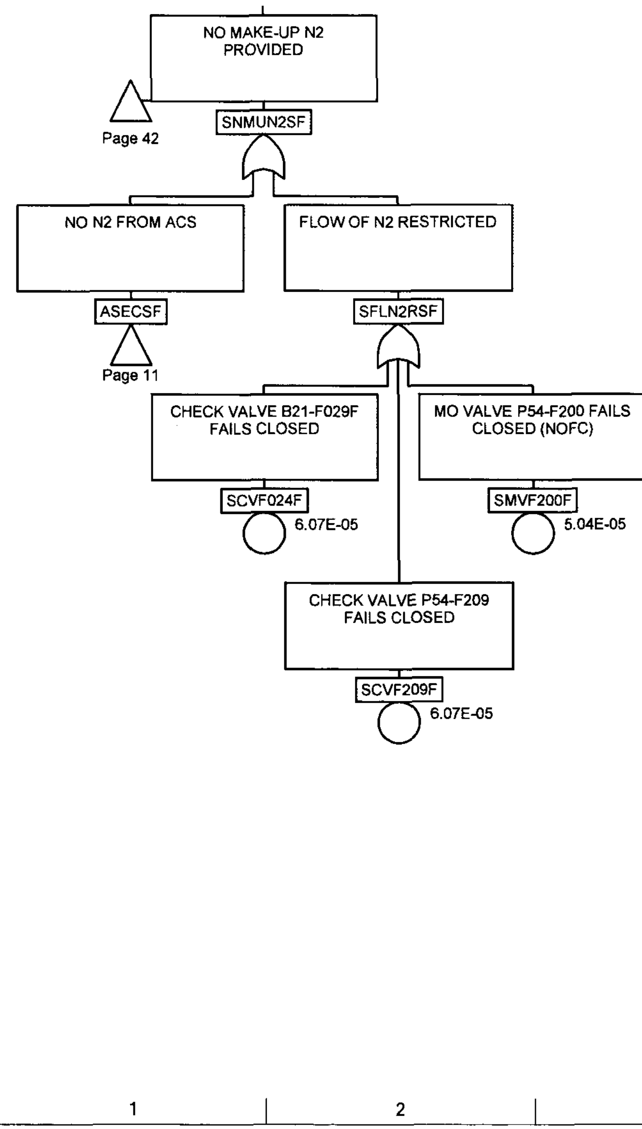


Figure 19D.6-5 Reactor Depressurization System Fault Tree

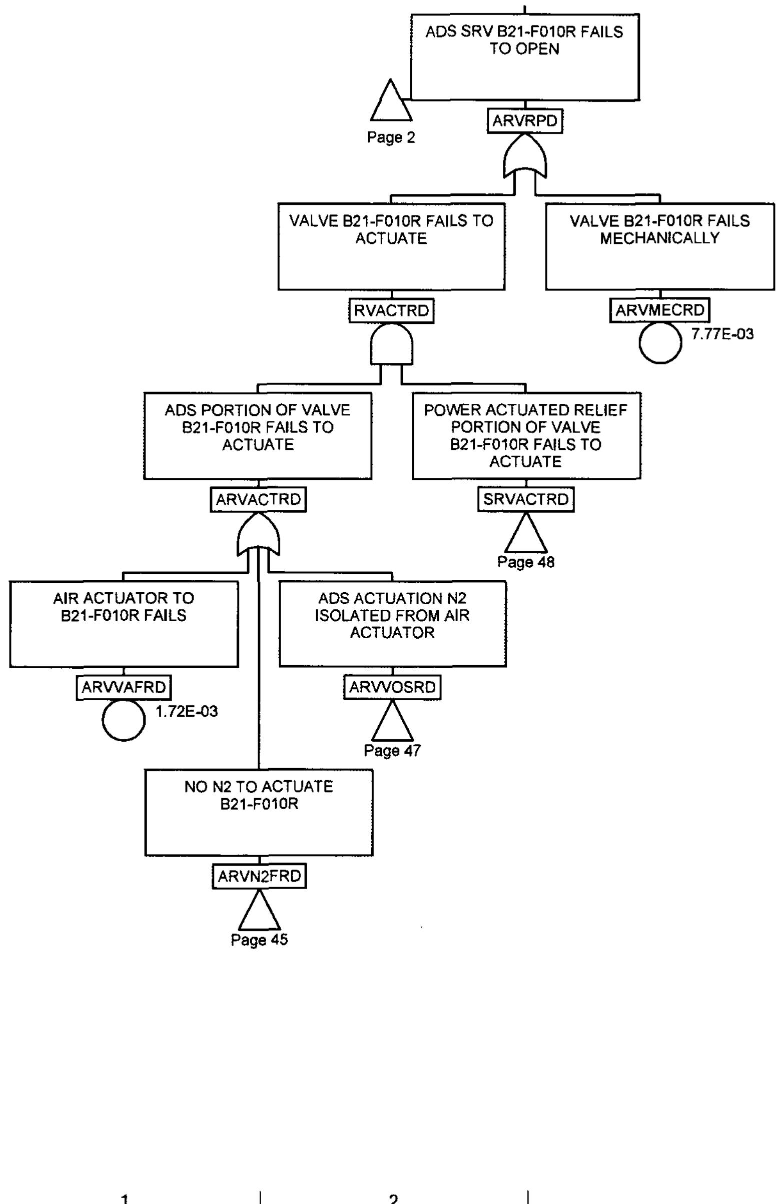
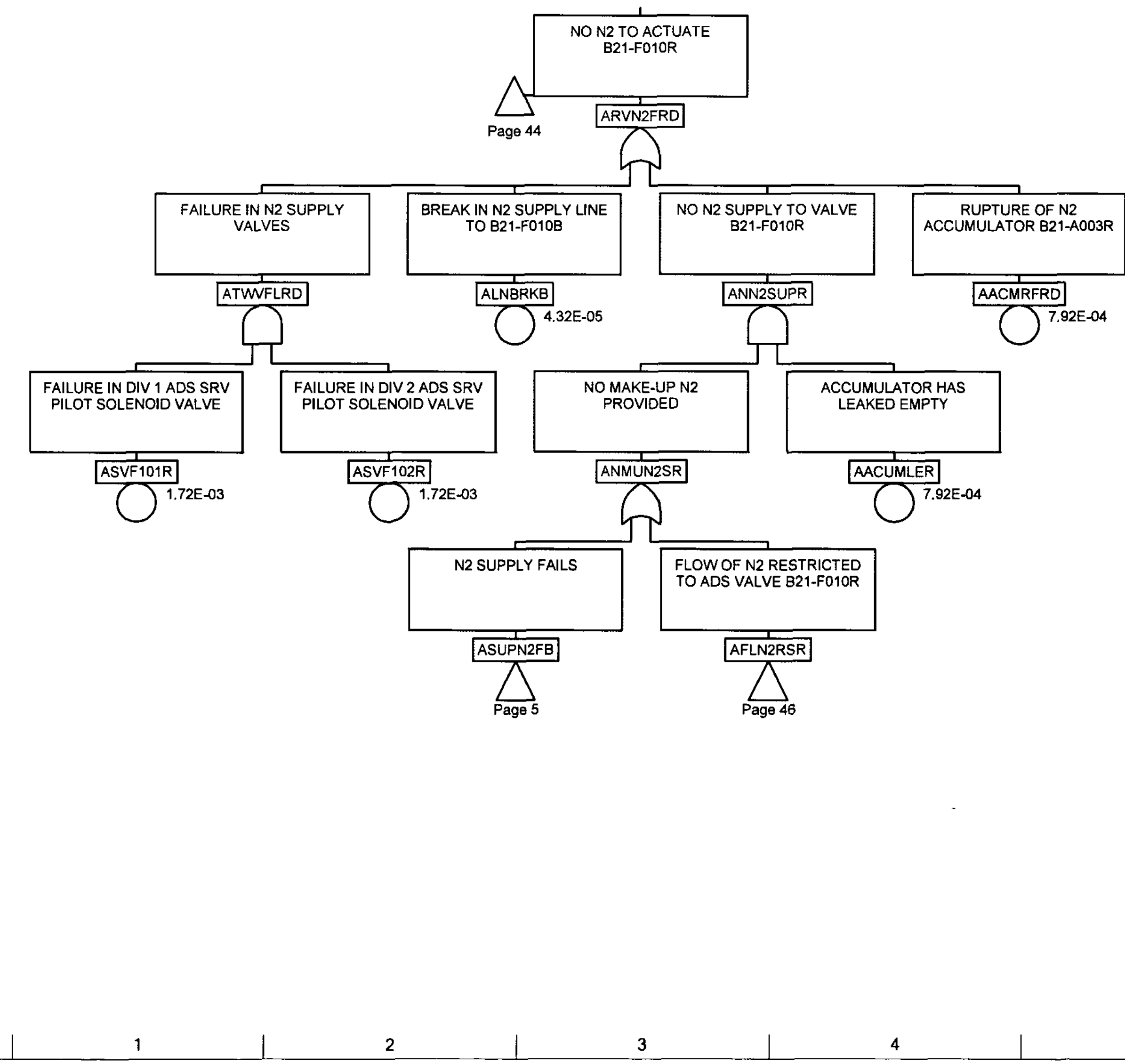


Figure 19D.6-5 Reactor Depressurization System Fault Tree



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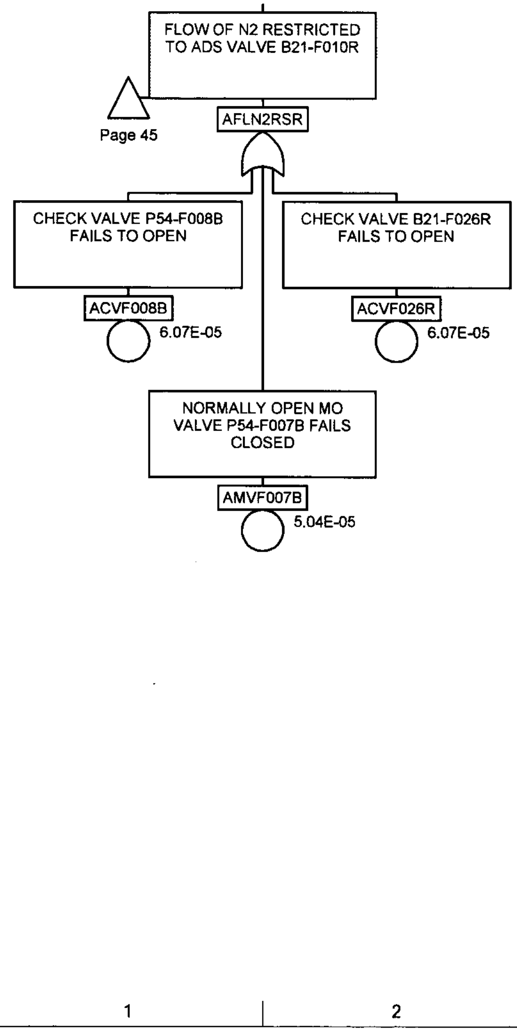


Figure 19D.6-5 Reactor Depressurization System Fault Tree

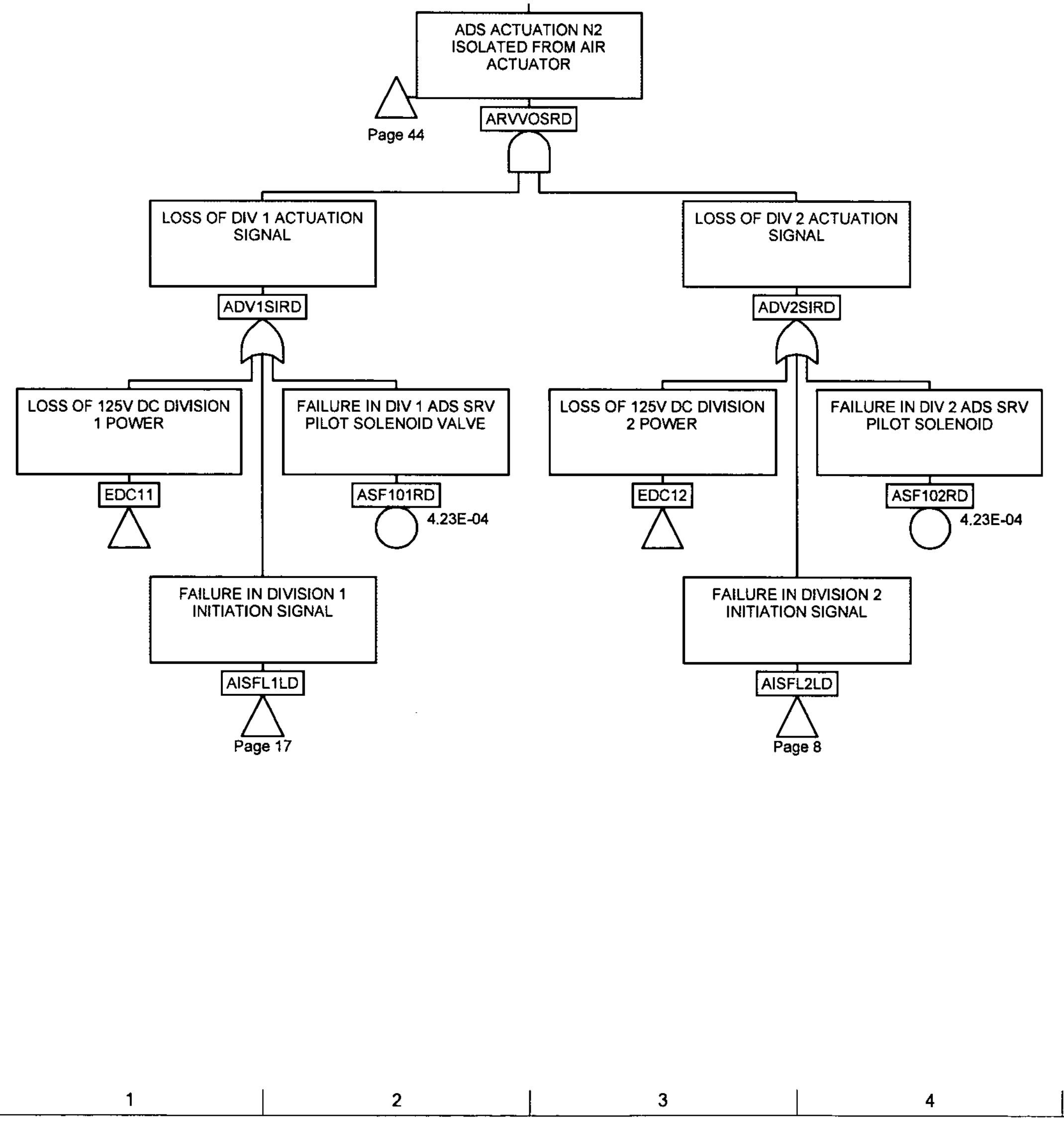
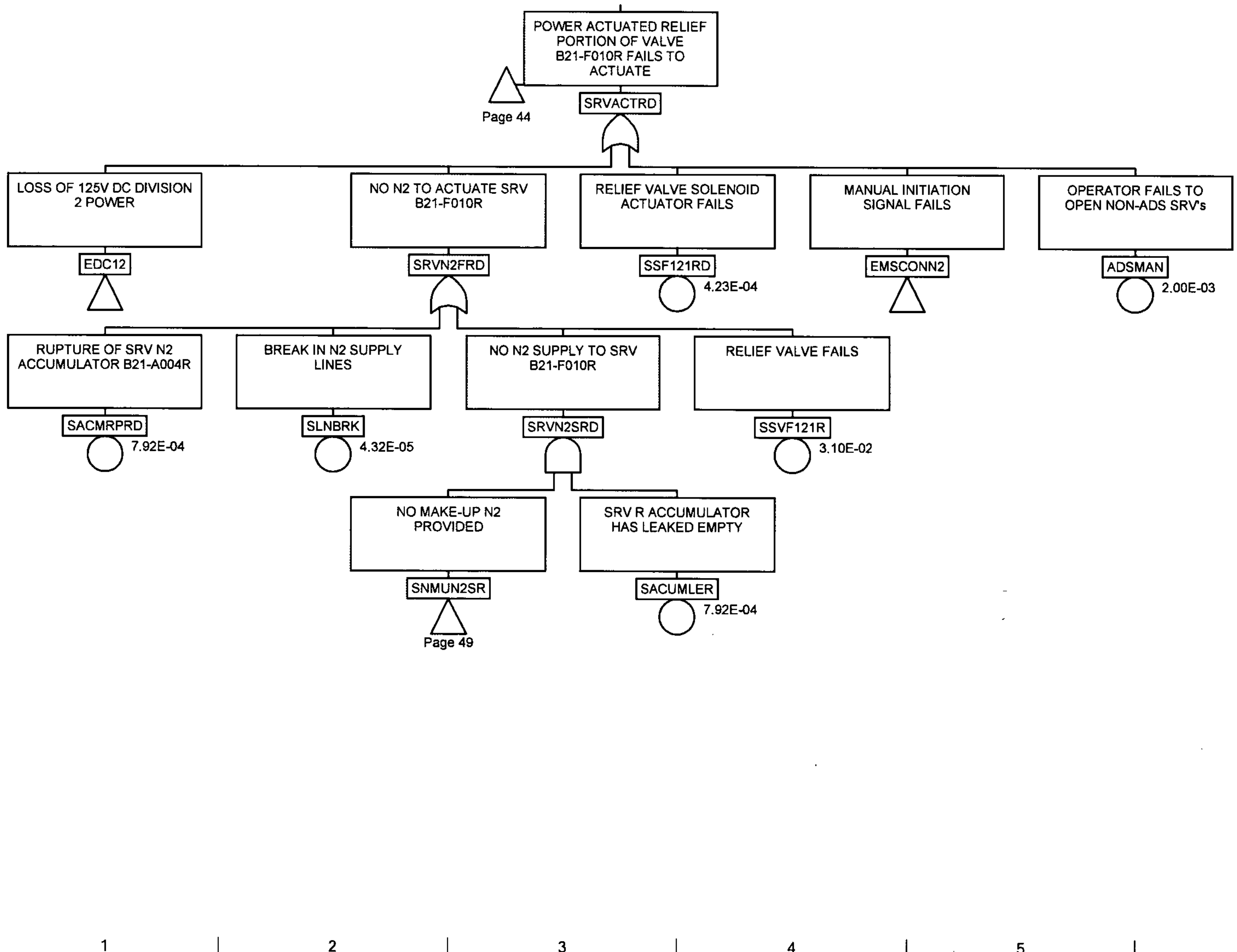


Figure 19D.6-5 Reactor Depressurization System Fault Tree


POWER ACTUATED RELIEF PORTION OF VALVE B21-F010R FAILS TO ACTUATE
SRVACTRD
Page 44
LOSS OF 125V DC DIVISION 2 POWER
EDC12
NO N2 TO ACTUATE SRV B21-F010R
SRVN2FRD
RELIEF VALVE SOLENOID ACTUATOR FAILS
SSF121RD
4.23E-04
MANUAL INITIATION SIGNAL FAILS
EMSCONN2
OPERATOR FAILS TO OPEN NON-ADS SRV's
ADSMAN
2.00E-03
RUPTURE OF SRV N2 ACCUMULATOR B21-A004R
SACMRPRD
7.92E-04
BREAK IN N2 SUPPLY LINES
SLNBRK
4.32E-05
NO N2 SUPPLY TO SRV B21-F010R
SRVN2SRD
RELIEF VALVE FAILS
SSVF121R
3.10E-02
NO MAKE-UP N2 PROVIDED
SNMUN2SR
Page 49
SRV R ACCUMULATOR HAS LEAKED EMPTY
SACUMLER
7.92E-04
1
2
3
4
5

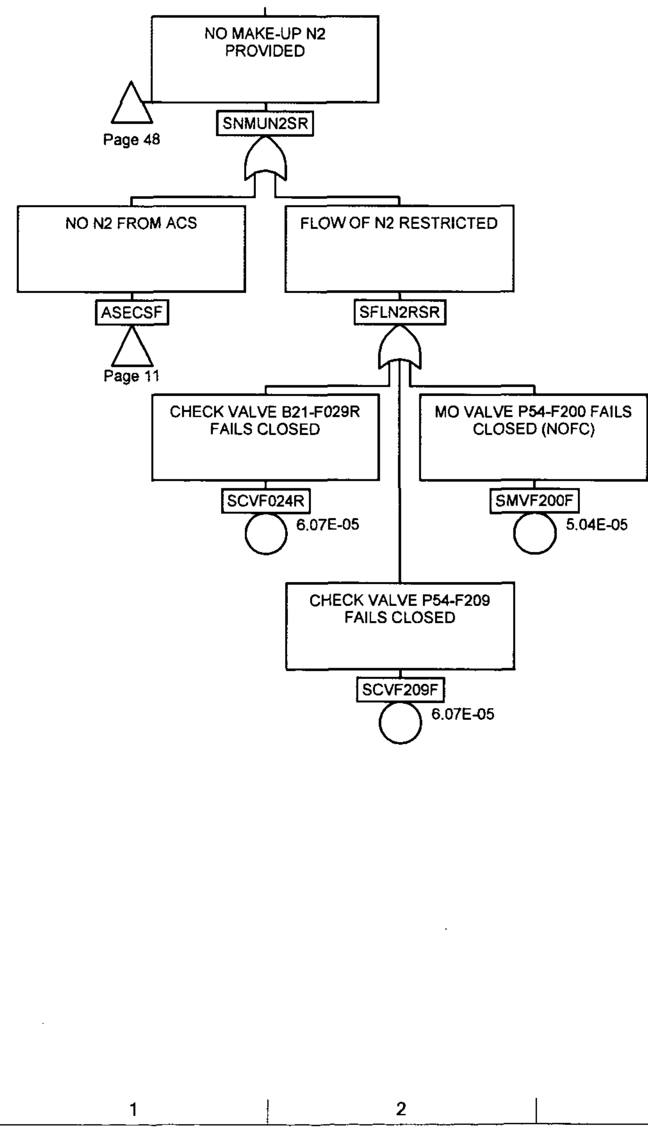


Figure 19D.6-5 Reactor Depressurization System Fault Tree

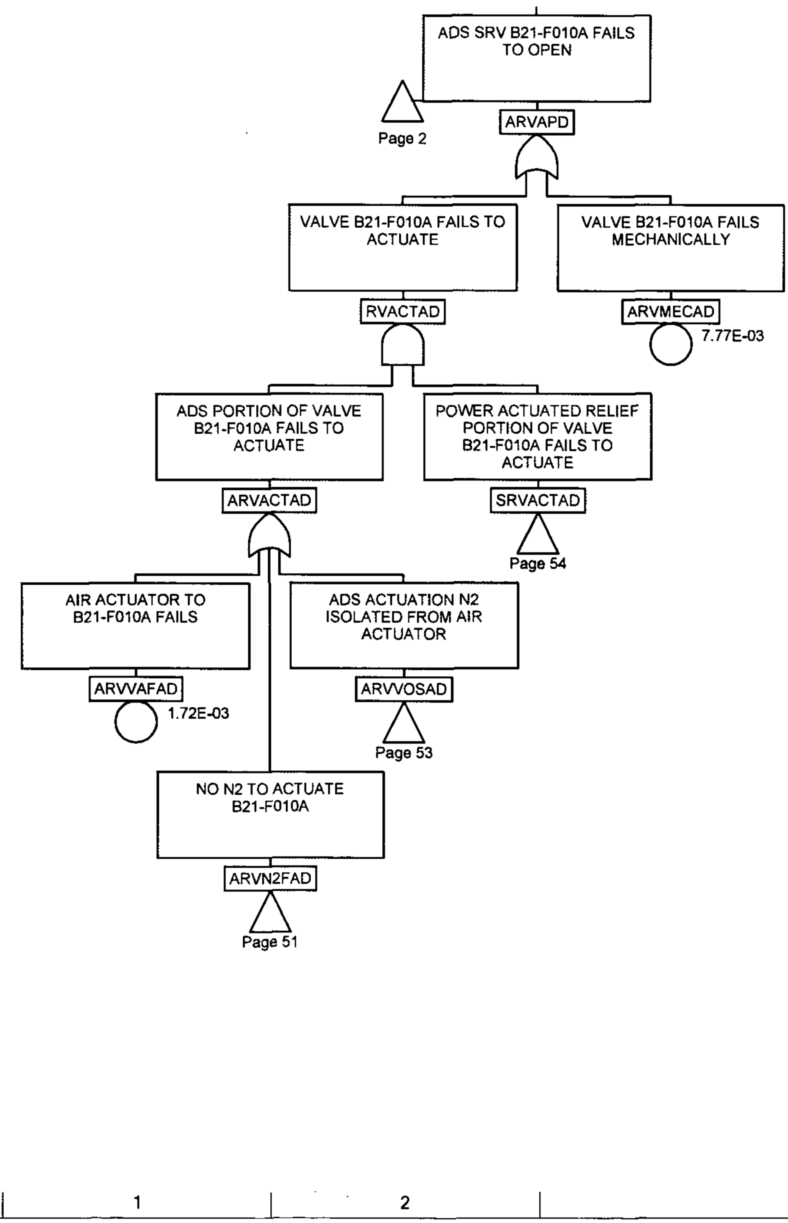


Figure 19D.6-5 Reactor Depressurization System Fault Tree

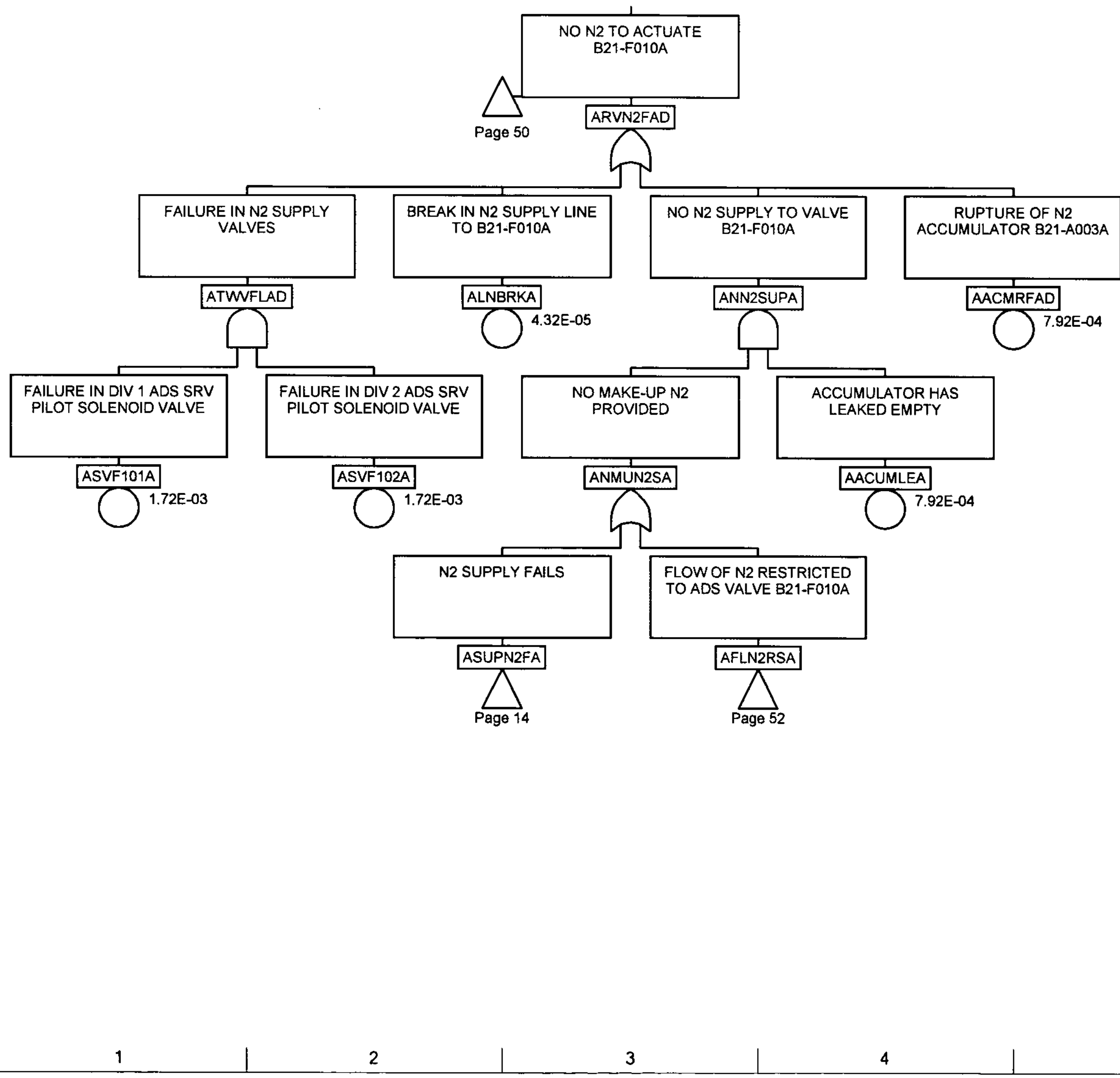


Figure 19D.6-5 Reactor Depressurization System Fault Tree

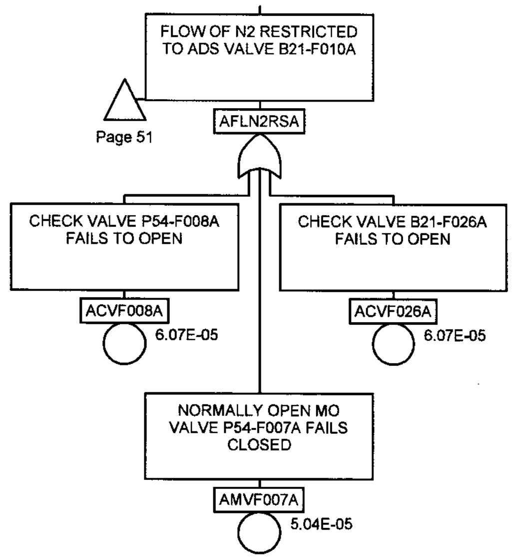


Figure 19D.6-5 Reactor Depressurization System Fault Tree

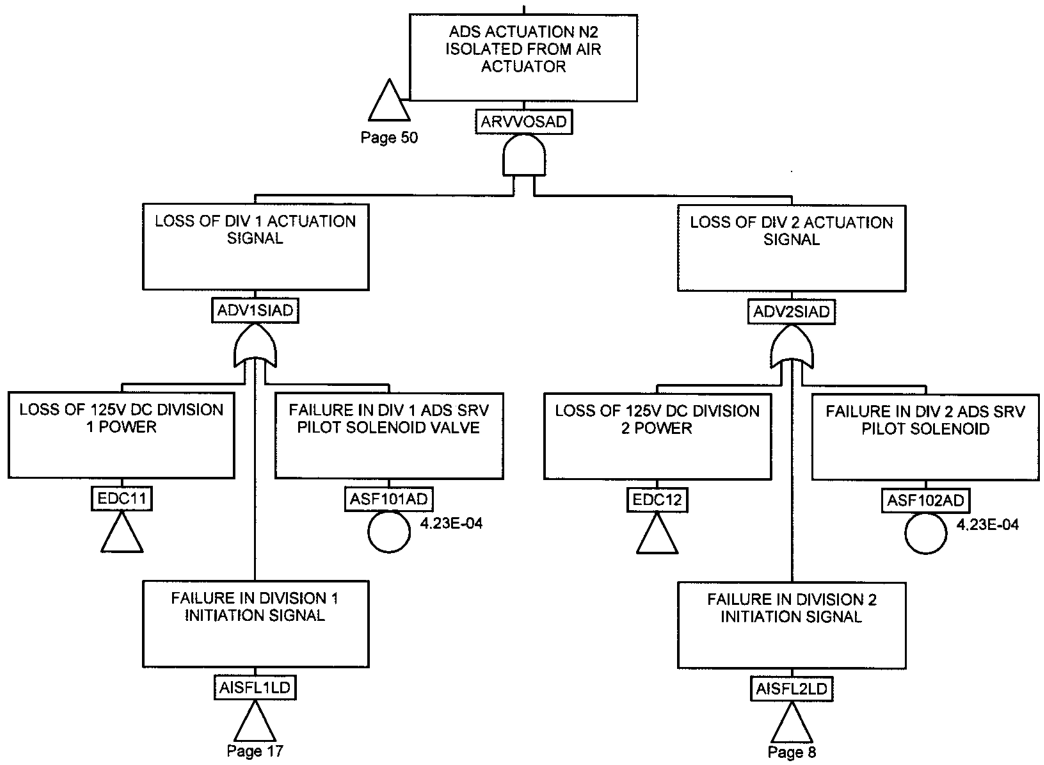
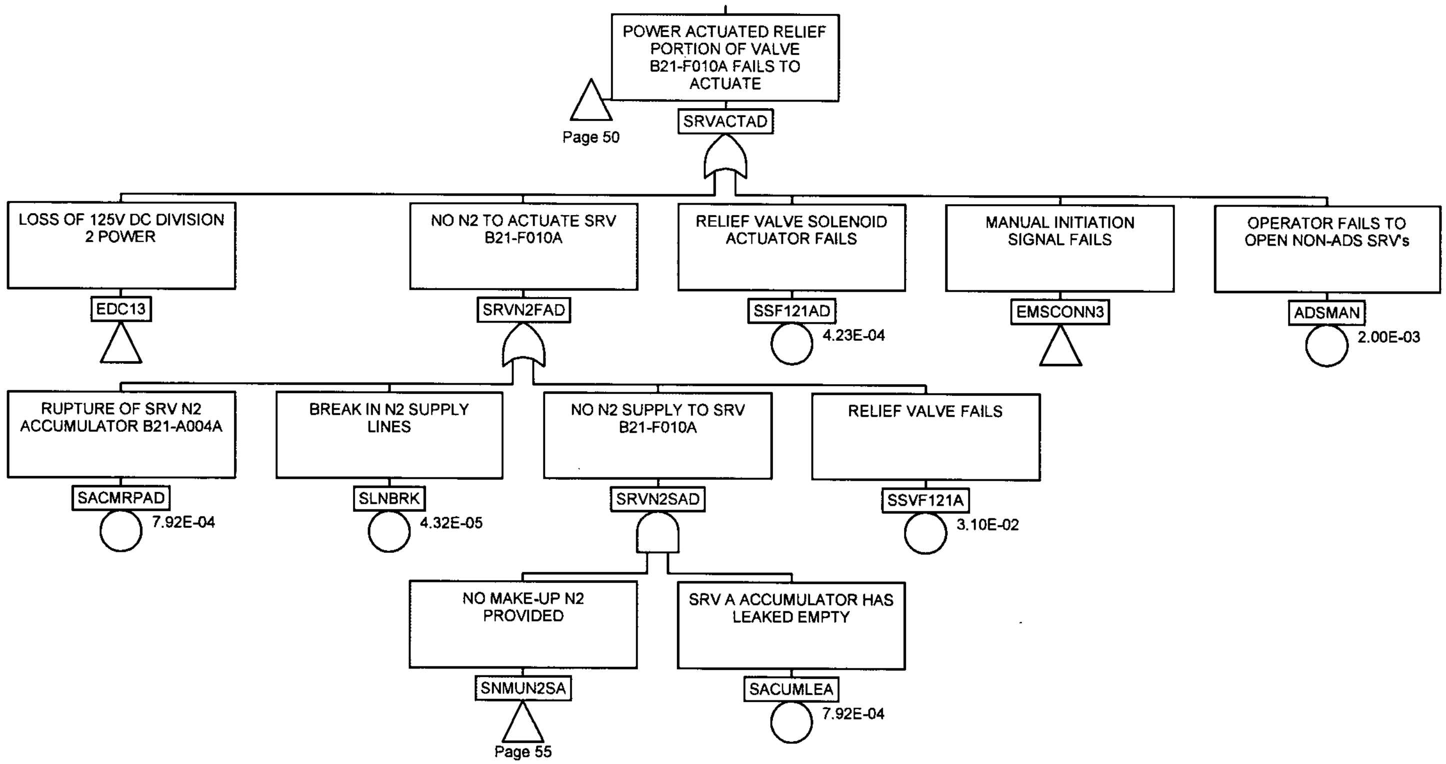


Figure 19D.6-5 Reactor Depressurization System Fault Tree



1 2 3 4 5

Figure 19D.6-5 Reactor Depressurization System Fault Tree

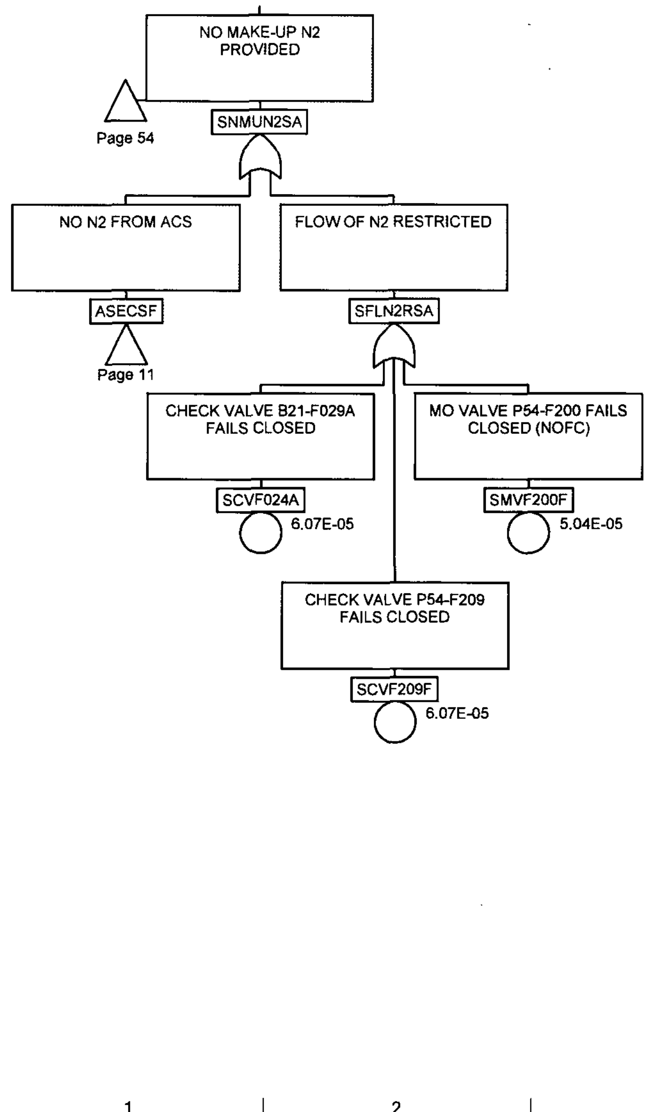


Figure 19D.6-5 Reactor Depressurization System Fault Tree

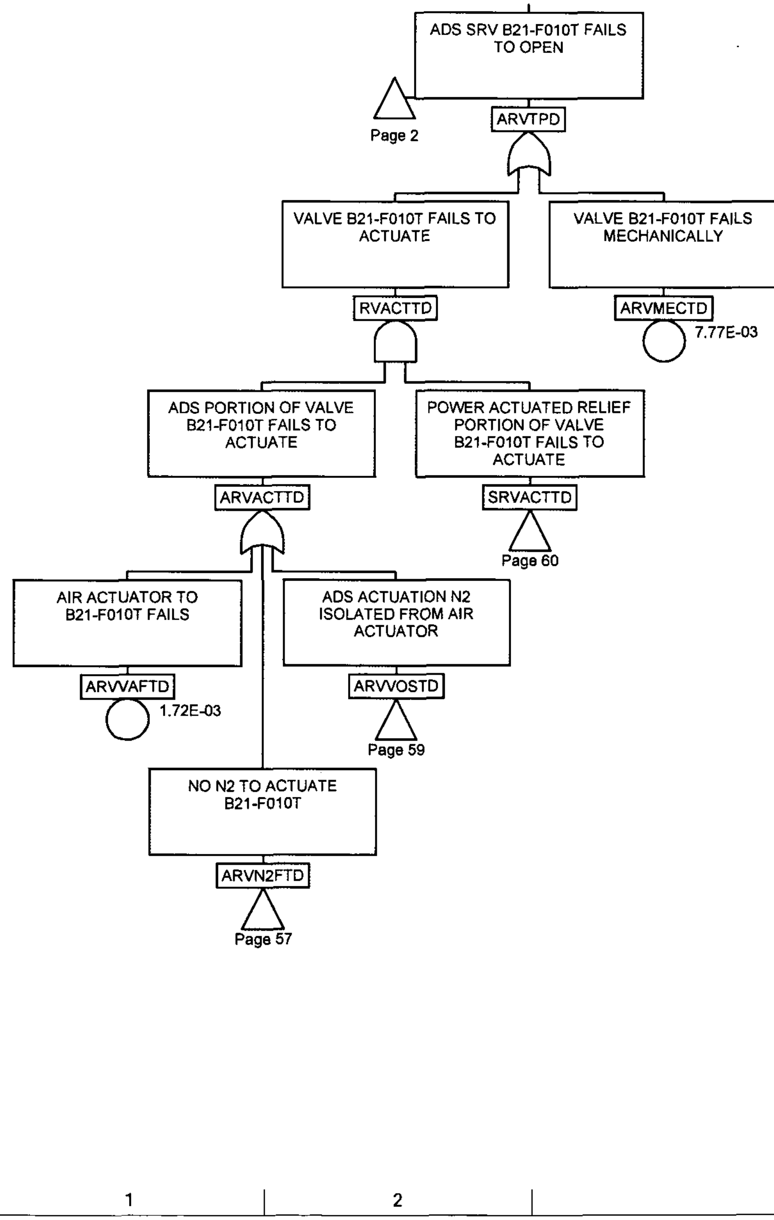


Figure 19D.6-5 Reactor Depressurization System Fault Tree

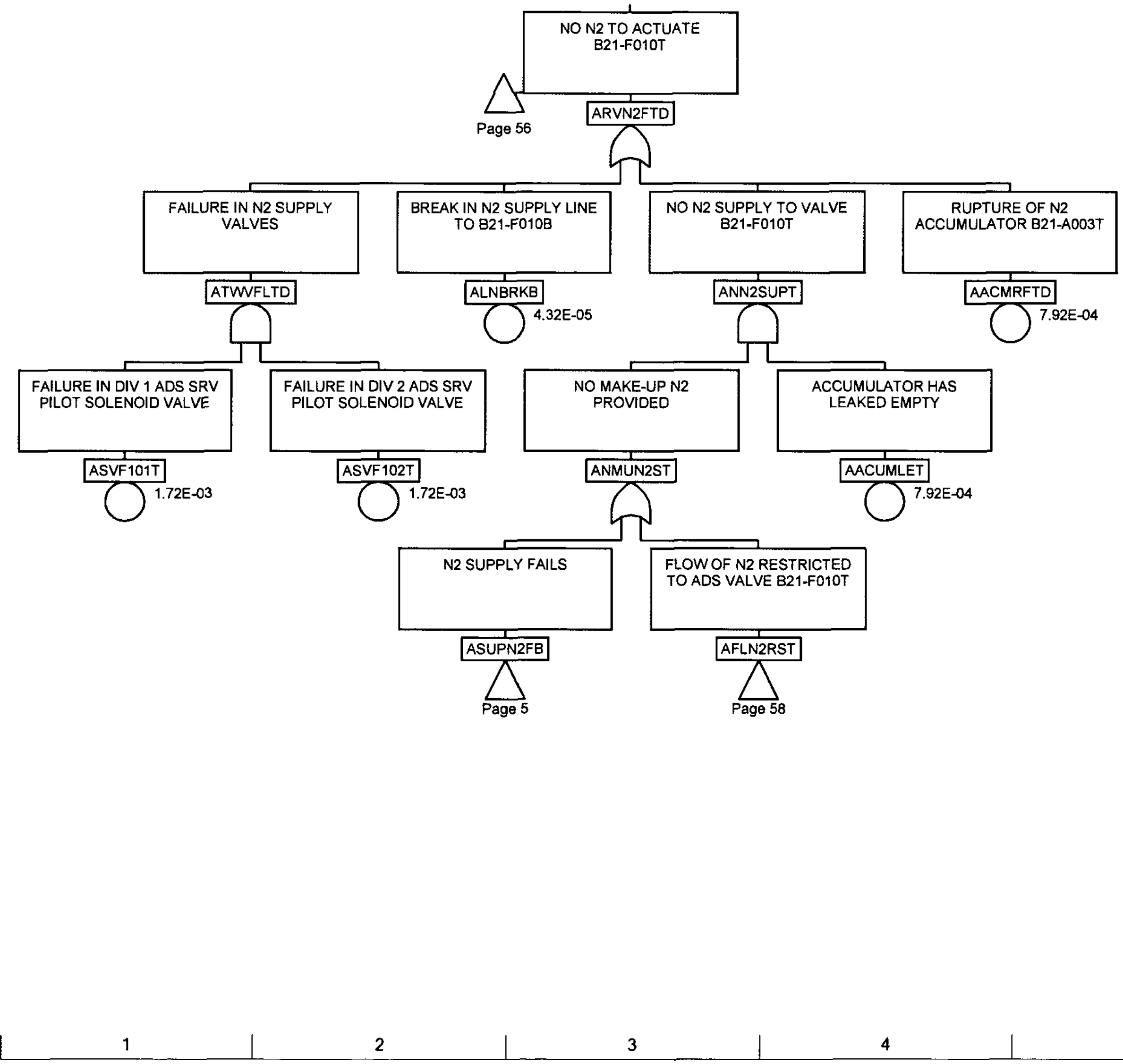


Figure 19D.6-5 Reactor Depressurization System Fault Tree

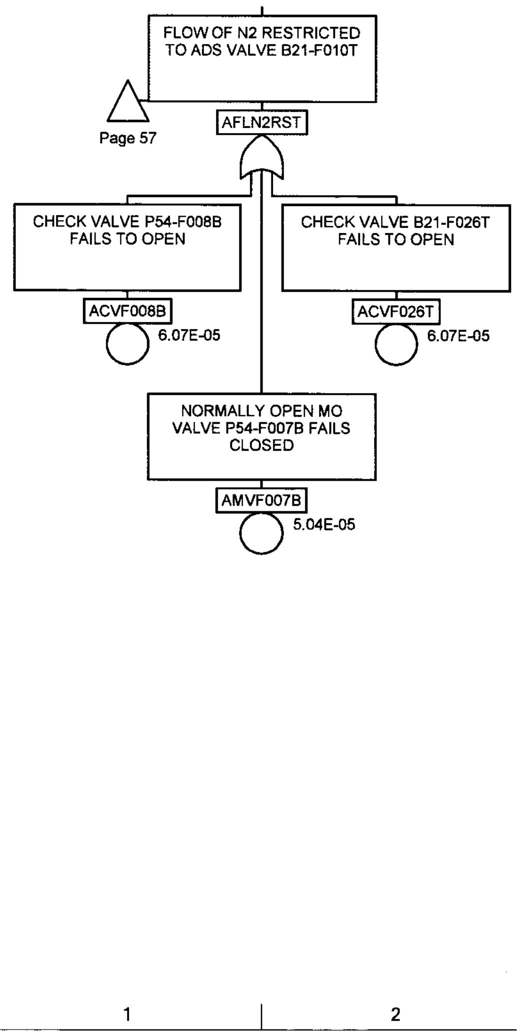


Figure 19D.6-5 Reactor Depressurization System Fault Tree

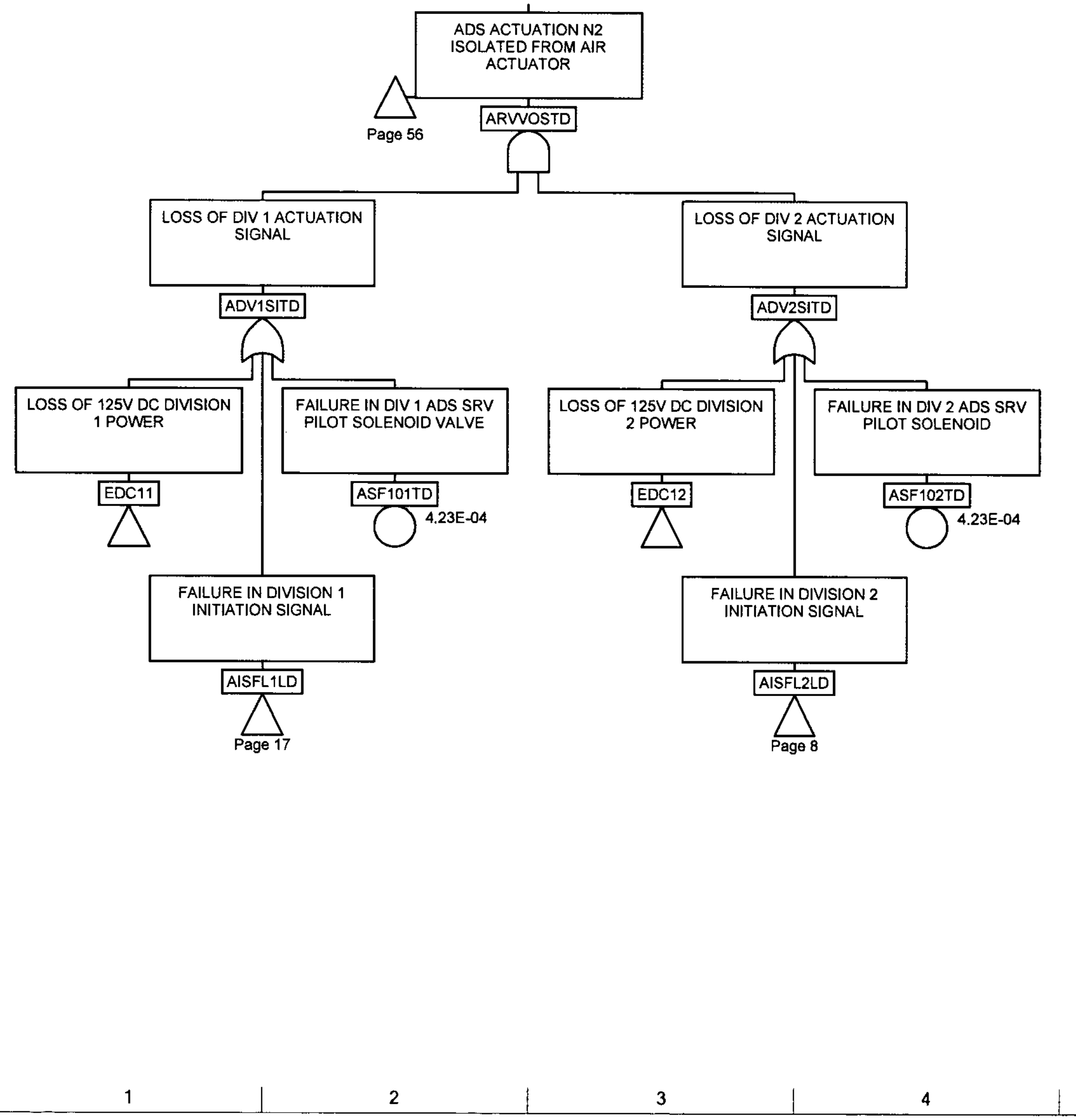
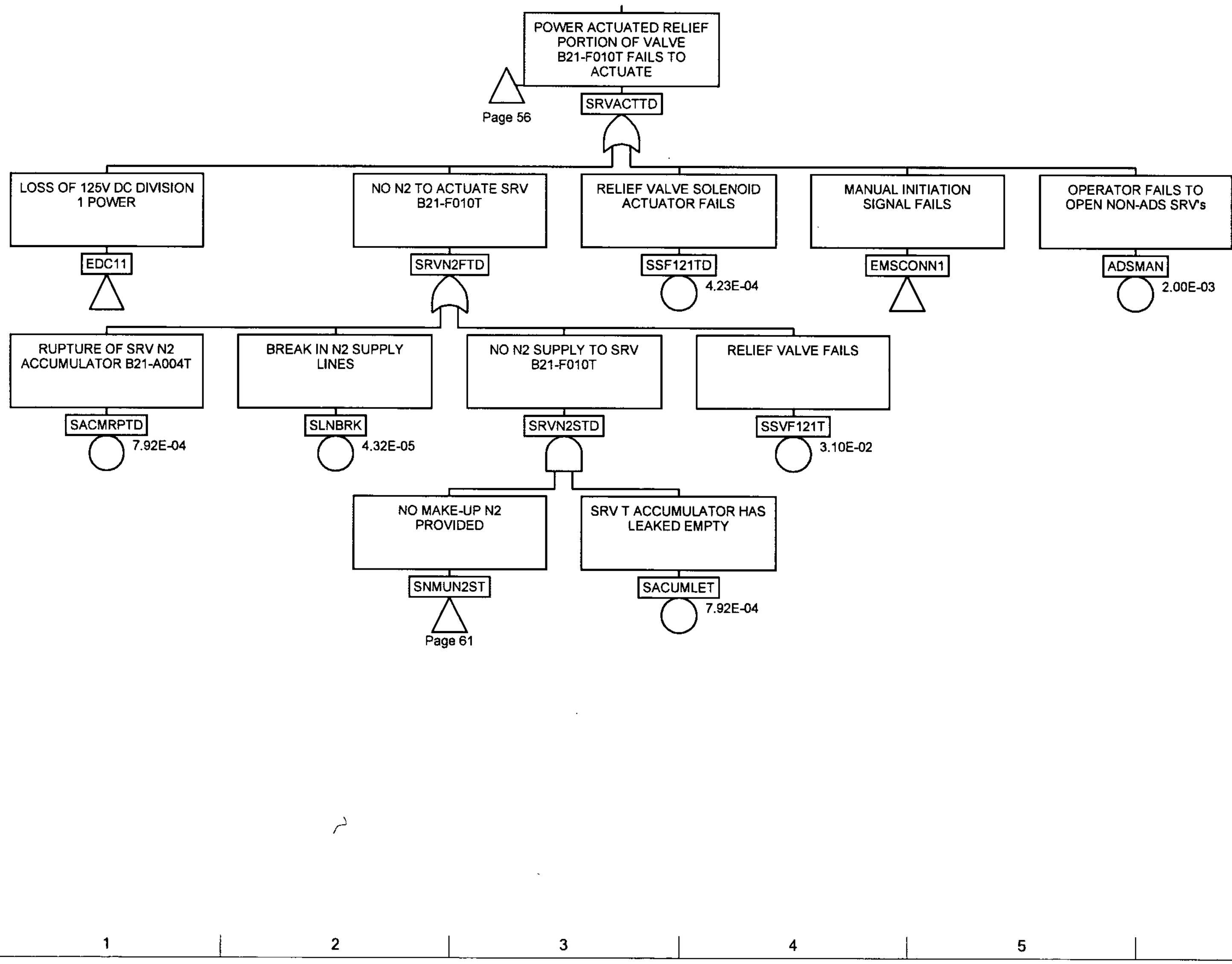


Figure 19D.6-5 Reactor Depressurization System Fault Tree


POWER ACTUATED RELIEF PORTION OF VALVE B21-F010T FAILS TO ACTUATE
SRVACTTD
Page 56
LOSS OF 125V DC DIVISION 1 POWER
EDC11
NO N2 TO ACTUATE SRV B21-F010T
SRVN2FTD
RELIEF VALVE SOLENOID ACTUATOR FAILS
SSF121TD
4.23E-04
MANUAL INITIATION SIGNAL FAILS
EMSCONN1
OPERATOR FAILS TO OPEN NON-ADS SRV's
ADSMAN
2.00E-03
RUPTURE OF SRV N2 ACCUMULATOR B21-A004T
SACMRPTD
7.92E-04
BREAK IN N2 SUPPLY LINES
SLNBRK
4.32E-05
NO N2 SUPPLY TO SRV B21-F010T
SRVN2STD
RELIEF VALVE FAILS
SSVF121T
3.10E-02
NO MAKE-UP N2 PROVIDED
SNMUN2ST
Page 61
SRV T ACCUMULATOR HAS LEAKED EMPTY
SACUMLET
7.92E-04
1
2
3
4
5

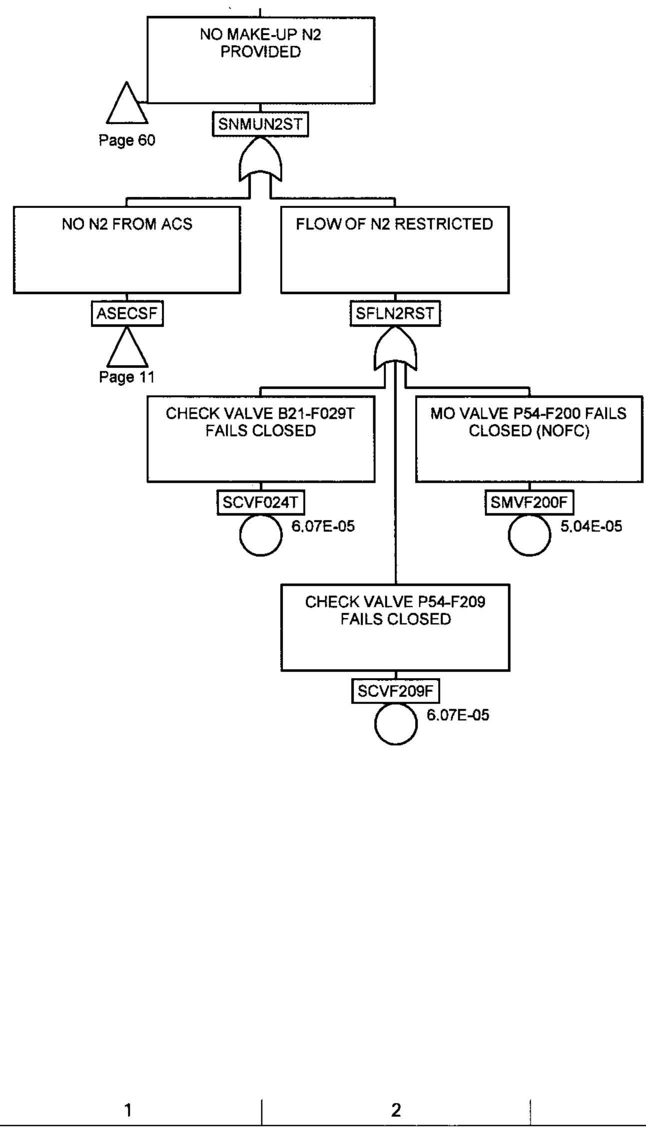


Figure 19D.6-5 Reactor Depressurization System Fault Tree

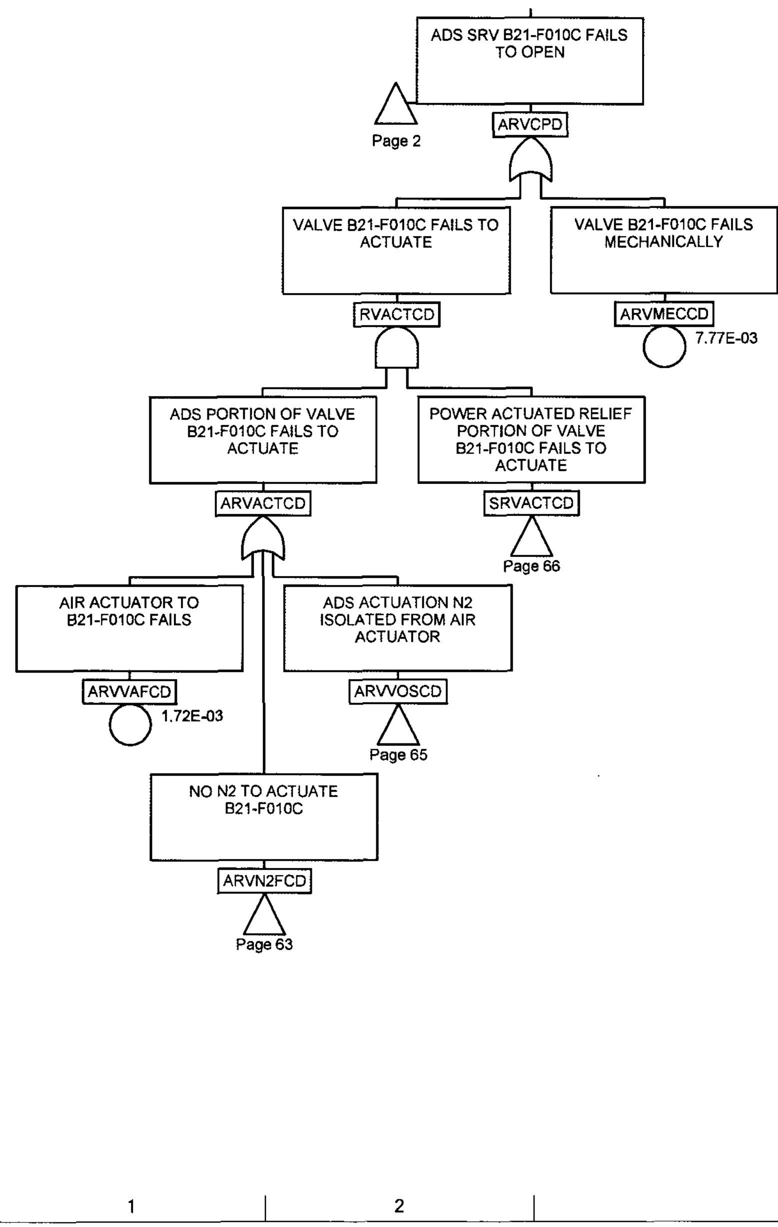


Figure 19D.6-5 Reactor Depressurization System Fault Tree

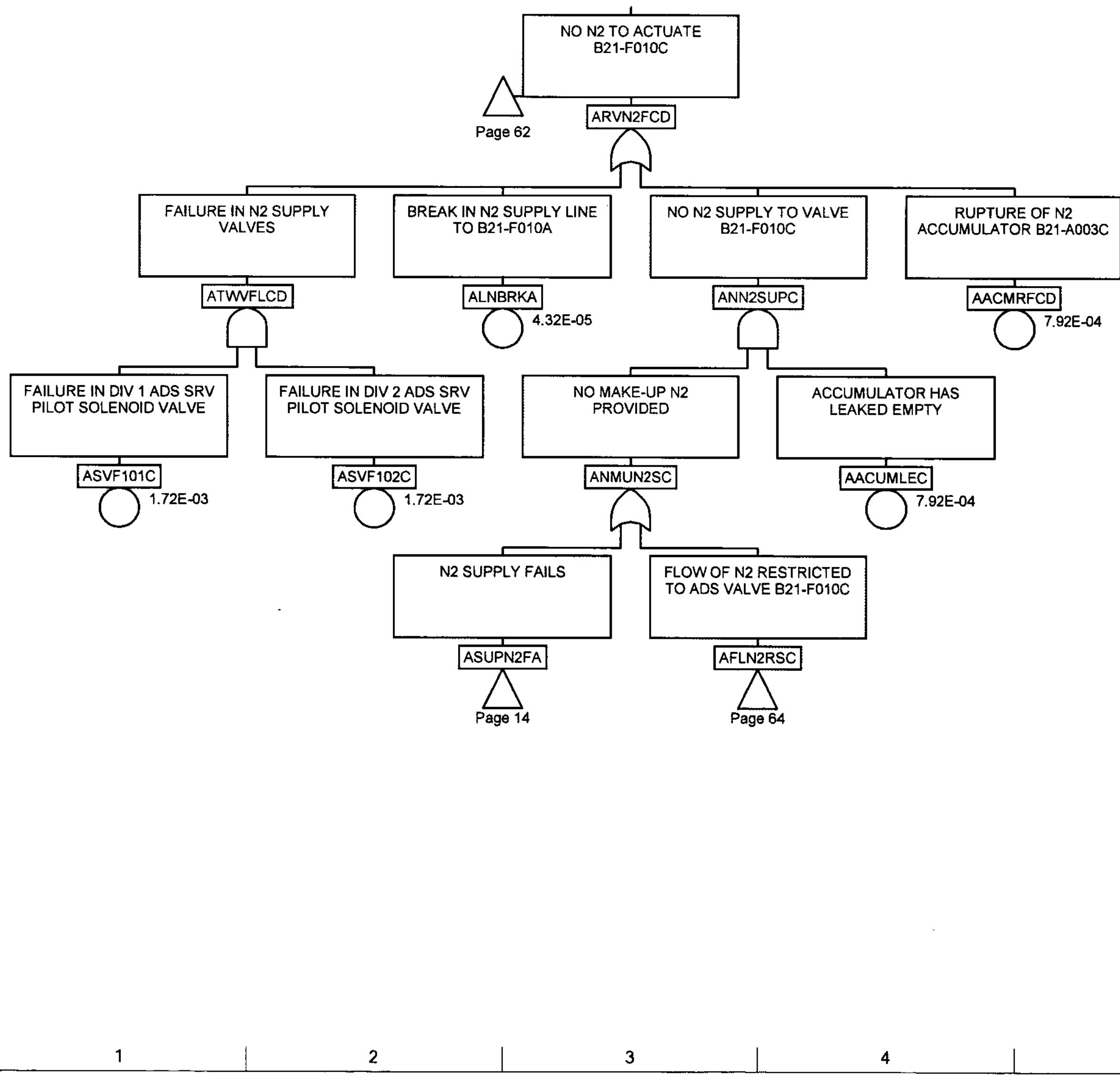


Figure 19D.6-5 Reactor Depressurization System Fault Tree

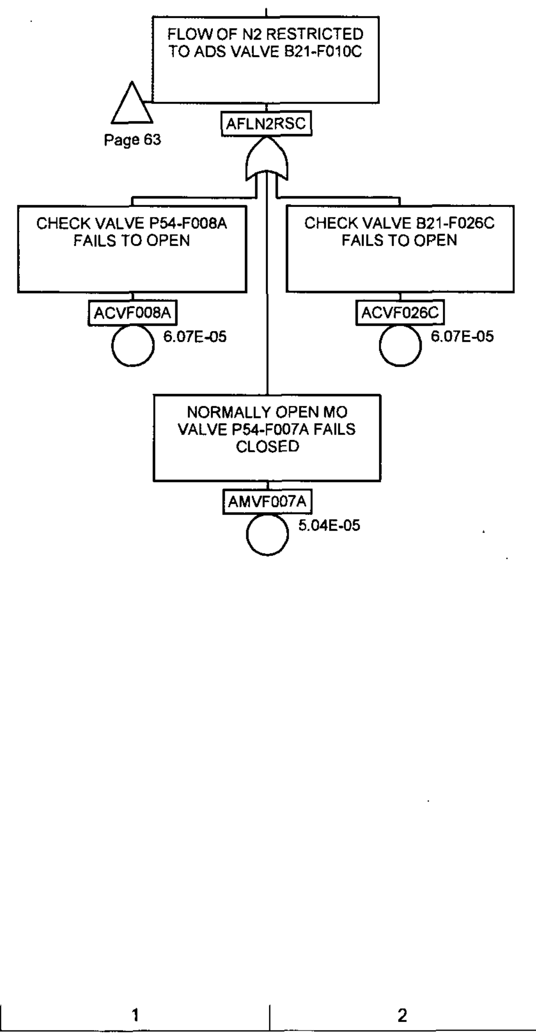
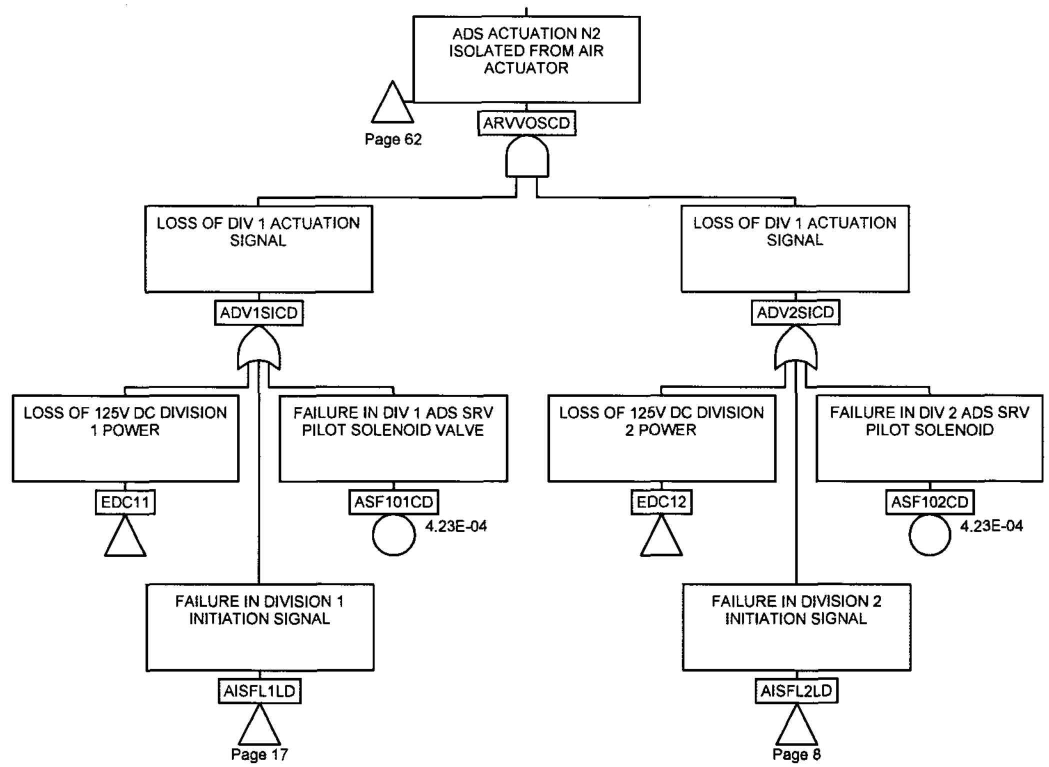


Figure 19D.6-5 Reactor Depressurization System Fault Tree



1	2	3	4

Figure 19D.6-5 Reactor Depressurization System Fault Tree

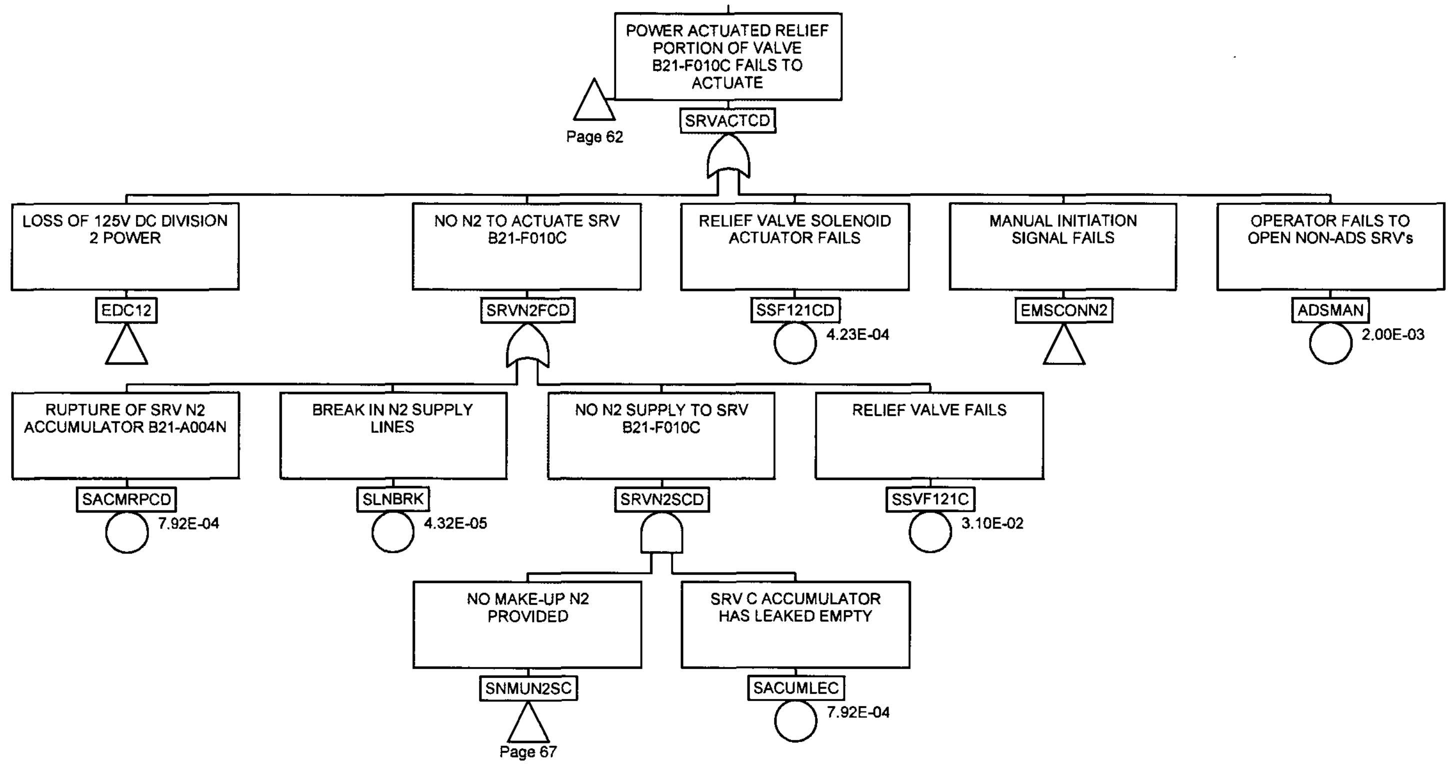


Figure 19D.6-5 Reactor Depressurization System Fault Tree

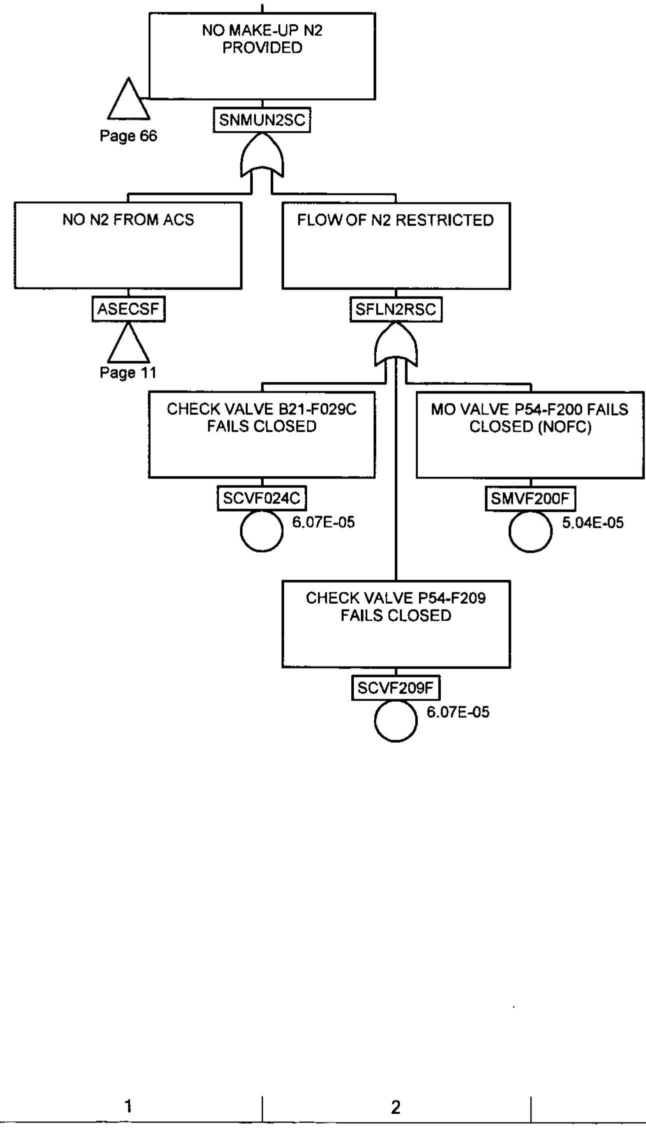
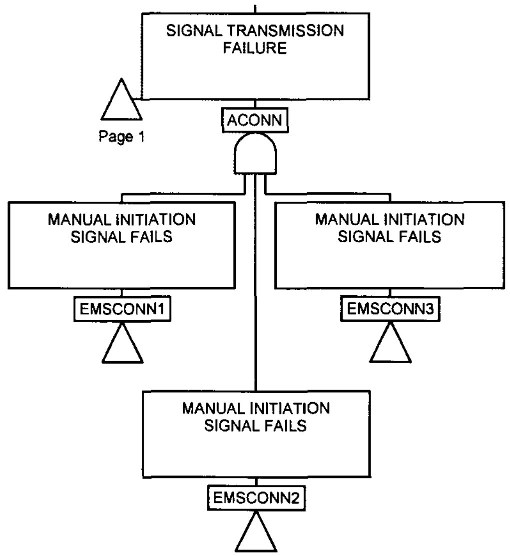


Figure 19D.6-5 Reactor Depressurization System Fault Tree



1 2

Figure 19D.6-5 Reactor Depressurization System Fault Tree

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AACMRFCD	63	5	ADSDPR	1	2
AACMRFFD	39	5	ADSMAN	1	4
AACMRFHD	13	5	ADSMAN	8	2
AACMRFLD	32	5	ADSMAN	9	6
AACMRFND	4	5	ADSMAN	18	1
AACMRFRD	45	5	ADSMAN	29	6
AACMRFTD	57	5	ADSMAN	36	6
AACUMLEA	51	4	ADSMAN	42	6
AACUMLEC	63	4	ADSMAN	48	6
AACUMLEF	39	4	ADSMAN	54	6
AACUMLEH	13	4	ADSMAN	60	6
AACUMLEL	32	4	ADSMAN	66	6
AACUMLEN	4	4	ADV1SIAD	53	2
AACUMLER	45	4	ADV1SICD	65	2
AACUMLET	57	4	ADV1SIFD	41	2
ABYTMR1	19	2	ADV1SIHD	17	2
ABYTMR2	24	2	ADV1SILD	35	2
ACCF004	20	3	ADV1SIND	7	2
ACCF004	21	1	ADV1SIRD	47	2
ACCF004	22	1	ADV1SITD	59	2
ACCF004	25	4	ADV2SIAD	53	4
ACCF004	26	1	ADV2SICD	65	4
ACCF004	27	1	ADV2SIFD	41	4
ACONN	1	4	ADV2SIHD	17	5
ACONN	68	2	ADV2SILD	35	4
ACVF008A	16	1	ADV2SIND	7	4
ACVF008A	40	1	ADV2SIRD	47	4
ACVF008A	52	1	ADV2SITD	59	4
ACVF008A	64	1	ADW1	18	4
ACVF008B	6	1	ADW1	19	2
ACVF008B	34	1	ADW2	24	2
ACVF008B	46	1	AFLN2RSA	51	4
ACVF008B	58	1	AFLN2RSA	52	2
ACVF026A	52	2	AFLN2RSC	63	4
ACVF026C	64	2	AFLN2RSC	64	2
ACVF026F	40	2	AFLN2RSF	39	4
ACVF026H	16	2	AFLN2RSF	40	2
ACVF026L	34	2	AFLN2RSH	13	4
ACVF026N	6	2	AFLN2RSH	16	2
ACVF026R	46	2	AFLN2RSL	32	4
ACVF026T	58	2	AFLN2RSL	34	2
ACVF207F	11	2	AFLN2RSN	4	4
ADS1LLOG	17	3	AFLN2RSN	6	2
ADS1LLOG	18	3	AFLN2RSR	45	4
ADS2LLOG	8	3	AFLN2RSR	46	2
ADSAUT1L	18	5	AFLN2RST	57	4
ADSAUT2L	8	4	AFLN2RST	58	2

Name	Page	Zone	Name	Page	Zone
AHBSD1VL	20	2	AMV003BD	5	2
AHBSD2VL	25	2	AMVF007A	16	2
AHCSD1VL	20	6	AMVF007A	40	2
AHCSD1VL	23	2	AMVF007A	52	2
AHCSD2VL	25	8	AMVF007A	64	2
AHCSD2VL	28	2	AMVF007B	6	2
AHPINAFL	14	2	AMVF007B	34	2
AHPINBFL	5	2	AMVF007B	46	2
AHPS006B	20	1	AMVF007B	58	2
AHPS006B	25	1	AMVF012A	14	3
AHPS006C	23	1	AMVF012B	5	3
AHPS006C	28	1	AMVF203F	11	1
AHPT006	20	2	AN2BAFF	15	2
AHPT006	23	3	AN2BAFFE	15	2
AHPT006	25	3	AN2BBFF	33	2
AHPT006	28	2	AN2BBFFE	33	2
AISFL1LD	7	2	AN2BCSF	15	4
AISFL1LD	17	2	AN2BCSFE	15	4
AISFL1LD	35	2	AN2BDSF	33	4
AISFL1LD	41	2	AN2BDSFE	33	4
AISFL1LD	47	2	ANAFACS	11	3
AISFL1LD	53	2	ANMUN2SA	51	3
AISFL1LD	59	2	ANMUN2SC	63	3
AISFL1LD	65	2	ANMUN2SF	39	3
AISFL2LD	7	4	ANMUN2SH	13	3
AISFL2LD	8	2	ANMUN2SL	32	3
AISFL2LD	17	5	ANMUN2SN	4	3
AISFL2LD	35	4	ANMUN2SR	45	3
AISFL2LD	41	4	ANMUN2ST	57	3
AISFL2LD	47	4	ANN2FSBA	14	3
AISFL2LD	53	4	ANN2FSBA	15	2
AISFL2LD	59	4	ANN2FSBB	5	3
AISFL2LD	65	4	ANN2FSBB	33	2
ALCVLD1L	17	2	ANN2SUPA	51	4
ALCVLD2L	8	1	ANN2SUPC	63	4
ALNBRKA	13	3	ANN2SUPF	39	4
ALNBRKA	39	3	ANN2SUPH	13	4
ALNBRKA	51	3	ANN2SUPL	32	4
ALNBRKA	63	3	ANN2SUPN	4	4
ALNBRKB	4	3	ANN2SUPR	45	4
ALNBRKB	32	3	ANN2SUPT	57	4
ALNBRKB	45	3	ANTMD1VL	18	6
ALNBRKB	57	3	ANTMD2VL	24	4
ALSBUN2A	14	3	APF205F	11	1
ALSBUN2B	5	3	APR002A	14	2
AMAN1	18	2	APR002A	14	4
AMAN2	8	3	APR002B	5	2
AMV003AD	14	2	APR002B	5	4

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APR004	11	2	ARVMECHD	12	3
APR005	5	5	ARVMECLD	31	3
APR005	14	5	ARVMECND	3	3
APRN2A	14	4	ARVMECRD	44	3
APRN2B	5	4	ARVMECTD	56	3
APRSD1VL	18	5	ARVN2FAD	50	2
APRSD1VL	20	3	ARVN2FAD	51	3
APRSD2VL	24	3	ARVN2FCD	62	2
APRSD2VL	25	5	ARVN2FCD	63	3
APS004A	22	2	ARVN2FFD	38	2
APS004A	27	2	ARVN2FFD	39	3
APS004B	21	2	ARVN2FHD	12	2
APS004B	25	5	ARVN2FHD	13	3
APS004C	20	4	ARVN2FLD	31	2
APS004C	26	2	ARVN2FLD	32	3
ARSAD1VL	20	5	ARVN2FND	3	2
ARSAD1VL	22	2	ARVN2FND	4	3
ARSAD2VL	25	7	ARVN2FRD	44	2
ARSAD2VL	27	2	ARVN2FRD	45	3
ARSBD1VL	20	3	ARVN2FTD	56	2
ARSBD1VL	21	2	ARVN2FTD	57	3
ARSBD2VL	25	5	ARVNPD	2	1
ARSCD1VL	20	4	ARVNPD	3	2
ARSCD2VL	25	6	ARVRPD	2	2
ARSCD2VL	26	2	ARVRPD	44	2
ARVACTAD	50	2	ARVSYFD	1	2
ARVACTCD	62	2	ARVSYFD	2	2
ARVACTFD	38	2	ARVTPD	2	2
ARVACTHD	12	2	ARVTPD	56	2
ARVACTLD	31	2	ARVVAFAD	50	1
ARVACTND	3	2	ARVVAFCD	62	1
ARVACTRD	44	2	ARVVAFFD	38	1
ARVACTTD	56	2	ARVVAFHD	12	1
ARVAPD	2	2	ARVVAFLD	31	1
ARVAPD	50	2	ARVVAFND	3	1
ARVCCFD	1	1	ARVVAFRD	44	1
ARVCCFD	1	3	ARVVAFTD	56	1
ARVCPD	2	2	ARVVOSAD	50	2
ARVCPD	62	2	ARVVOSAD	53	2
ARVFPD	2	1	ARVVOSCD	62	2
ARVFPD	38	2	ARVVOSCD	65	2
ARVHPD	2	1	ARVVOSFD	38	2
ARVHPD	12	2	ARVVOSFD	41	2
ARVLPD	2	1	ARVVOSHD	12	2
ARVLPD	31	2	ARVVOSHD	17	3
ARVMECAD	50	3	ARVVOSLD	31	2
ARVMECCD	62	3	ARVVOSLD	35	2
ARVMECFD	38	3	ARVVOSND	3	2

Figure 19D.6-5 Reactor Depressurization System Fault Tree

Name	Page	Zone	Name	Page	Zone
ARVVOSND	7	2	ASVF101H	13	1
ARVVOSRD	44	2	ASVF101L	32	1
ARVVOSRD	47	2	ASVF101N	4	1
ARVVOSTD	56	2	ASVF101R	45	1
ARVVOSTD	59	2	ASVF101T	57	1
ASECSF	5	4	ASVF102A	51	2
ASECSF	10	1	ASVF102C	63	2
ASECSF	11	2	ASVF102F	39	2
ASECSF	14	4	ASVF102H	13	2
ASECSF	30	1	ASVF102L	32	2
ASECSF	37	1	ASVF102N	4	2
ASECSF	43	1	ASVF102R	45	2
ASECSF	49	1	ASVF102T	57	2
ASECSF	55	1	ATWVFLAD	51	2
ASECSF	61	1	ATWVFLCD	63	2
ASECSF	67	1	ATWVFLFD	39	2
ASECSNA	5	1	ATWVFLHD	13	2
ASECSNA	14	1	ATWVFLLD	32	2
ASF101AD	53	2	ATWVFLND	4	2
ASF101CD	65	2	ATWVFLRD	45	2
ASF101FD	41	2	ATWVFLTD	57	2
ASF101HD	17	3	AVF002AF	15	1
ASF101LD	35	2	AVF002BF	33	1
ASF101ND	7	2	AVF002CF	15	3
ASF101RD	47	2	AVF002DF	33	3
ASF101TD	59	2	AVF005AD	14	3
ASF102AD	53	4	AVF005BD	5	3
ASF102CD	65	4	C001AFR	22	2
ASF102FD	41	4	C001AFR	27	2
ASF102HD	17	5	C001BFR	21	2
ASF102LD	35	4	C001BFR	25	5
ASF102ND	7	4	C001CFR	20	4
ASF102RD	47	4	C001CFR	26	2
ASF102TD	59	4	DWANDBY1	18	4
ASSVLF	11	2	DWANDBY2	24	2
ASUPN2FA	13	3	EDC11	7	1
ASUPN2FA	14	3	EDC11	17	1
ASUPN2FA	39	3	EDC11	17	2
ASUPN2FA	51	3	EDC11	35	1
ASUPN2FA	63	3	EDC11	41	1
ASUPN2FB	4	3	EDC11	42	1
ASUPN2FB	5	3	EDC11	47	1
ASUPN2FB	32	3	EDC11	53	1
ASUPN2FB	45	3	EDC11	59	1
ASUPN2FB	57	3	EDC11	60	1
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ASVF101C	63	1	EDC12	7	3
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Name	Page	Zone	Name	Page	Zone
EDC12	9	1	RVACTND	3	2
EDC12	17	4	RVACTRD	44	2
EDC12	35	3	RVACTTD	56	2
EDC12	41	3	RXDPRESS	1	2
EDC12	47	3	SACMRPAD	54	1
EDC12	48	1	SACMRPCD	66	1
EDC12	53	3	SACMRPFD	42	1
EDC12	59	3	SACMRPHD	29	1
EDC12	65	3	SACMRPLD	36	1
EDC12	66	1	SACMRPND	9	1
EDC13	29	1	SACMRPRD	48	1
EDC13	36	1	SACMRPTD	60	1
EDC13	54	1	SACUMLEA	54	4
EMSCONN1	18	2	SACUMLEC	66	4
EMSCONN1	19	1	SACUMLEF	42	4
EMSCONN1	42	5	SACUMLEH	29	4
EMSCONN1	60	5	SACUMLEL	36	4
EMSCONN1	68	1	SACUMLEN	9	4
EMSCONN2	8	3	SACUMLER	48	4
EMSCONN2	9	5	SACUMLET	60	4
EMSCONN2	24	1	SCVF024A	55	2
EMSCONN2	48	5	SCVF024C	67	2
EMSCONN2	66	5	SCVF024F	43	2
EMSCONN2	68	2	SCVF024H	30	2
EMSCONN3	29	5	SCVF024L	37	2
EMSCONN3	36	5	SCVF024N	10	2
EMSCONN3	54	5	SCVF024R	49	2
EMSCONN3	68	2	SCVF024T	61	2
HB23	20	2	SCVF209F	10	2
HB23	25	2	SCVF209F	30	2
HC23	23	2	SCVF209F	37	2
HC23	28	2	SCVF209F	43	2
HMO01BDR	25	2	SCVF209F	49	2
HMO01BDR1	25	2	SCVF209F	55	2
HMO01CDR	23	2	SCVF209F	61	2
HMO01CDR1	23	2	SCVF209F	67	2
HPM01BDW	25	3	SFLN2RSA	55	2
HPM01CDW	23	3	SFLN2RSC	67	2
IDWPA	18	5	SFLN2RSF	43	2
IDWPB	24	3	SFLN2RSH	30	2
IPVL1A	18	3	SFLN2RSL	37	2
IPVL2B	24	1	SFLN2RSN	10	2
NADPR	1	4	SFLN2RSR	49	2
RVACTAD	50	2	SFLN2RST	61	2
RVACTCD	62	2	SLNBRK	9	2
RVACTFD	38	2	SLNBRK	29	2
RVACTHD	12	2	SLNBRK	36	2
RVACTLD	31	2	SLNBRK	42	2

Figure 19D.6-5 Reactor Depressurization System Fault Tree

Name	Page	Zone	Name	Page	Zone
SLNBRK	48	2	SRVN2FLD	36	3
SLNBRK	54	2	SRVN2FND	9	3
SLNBRK	60	2	SRVN2FRD	48	3
SLNBRK	66	2	SRVN2FTD	60	3
SMVF200F	10	3	SRVN2SAD	54	3
SMVF200F	30	3	SRVN2SCD	66	3
SMVF200F	37	3	SRVN2SFD	42	3
SMVF200F	43	3	SRVN2SHD	29	3
SMVF200F	49	3	SRVN2SLD	36	3
SMVF200F	55	3	SRVN2SND	9	3
SMVF200F	61	3	SRVN2SRD	48	3
SMVF200F	67	3	SRVN2STD	60	3
SNMUN2SA	54	3	SRVSYFD	1	3
SNMUN2SA	55	2	SSF121AD	54	4
SNMUN2SC	66	3	SSF121CD	66	4
SNMUN2SC	67	2	SSF121FD	42	4
SNMUN2SF	42	3	SSF121HD	29	4
SNMUN2SF	43	2	SSF121LD	36	4
SNMUN2SH	29	3	SSF121ND	9	4
SNMUN2SH	30	2	SSF121RD	48	4
SNMUN2SL	36	3	SSF121TD	60	4
SNMUN2SL	37	2	SSVF121A	54	4
SNMUN2SN	9	3	SSVF121C	66	4
SNMUN2SN	10	2	SSVF121F	42	4
SNMUN2SR	48	3	SSVF121H	29	4
SNMUN2SR	49	2	SSVF121L	36	4
SNMUN2ST	60	3	SSVF121N	9	4
SNMUN2ST	61	2	SSVF121R	48	4
SRVACTAD	50	3	SSVF121T	60	4
SRVACTAD	54	3			
SRVACTCD	62	3			
SRVACTCD	66	3			
SRVACTFD	38	3			
SRVACTFD	42	3			
SRVACTHD	12	3			
SRVACTHD	29	3			
SRVACTLD	31	3			
SRVACTLD	36	3			
SRVACTND	3	3			
SRVACTND	9	3			
SRVACTRD	44	3			
SRVACTRD	48	3			
SRVACTTD	56	3			
SRVACTTD	60	3			
SRVN2FAD	54	3			
SRVN2FCD	66	3			
SRVN2FFD	42	3			
SRVN2FHD	29	3			

Figure 19D.6-5 Reactor Depressurization System Fault Tree

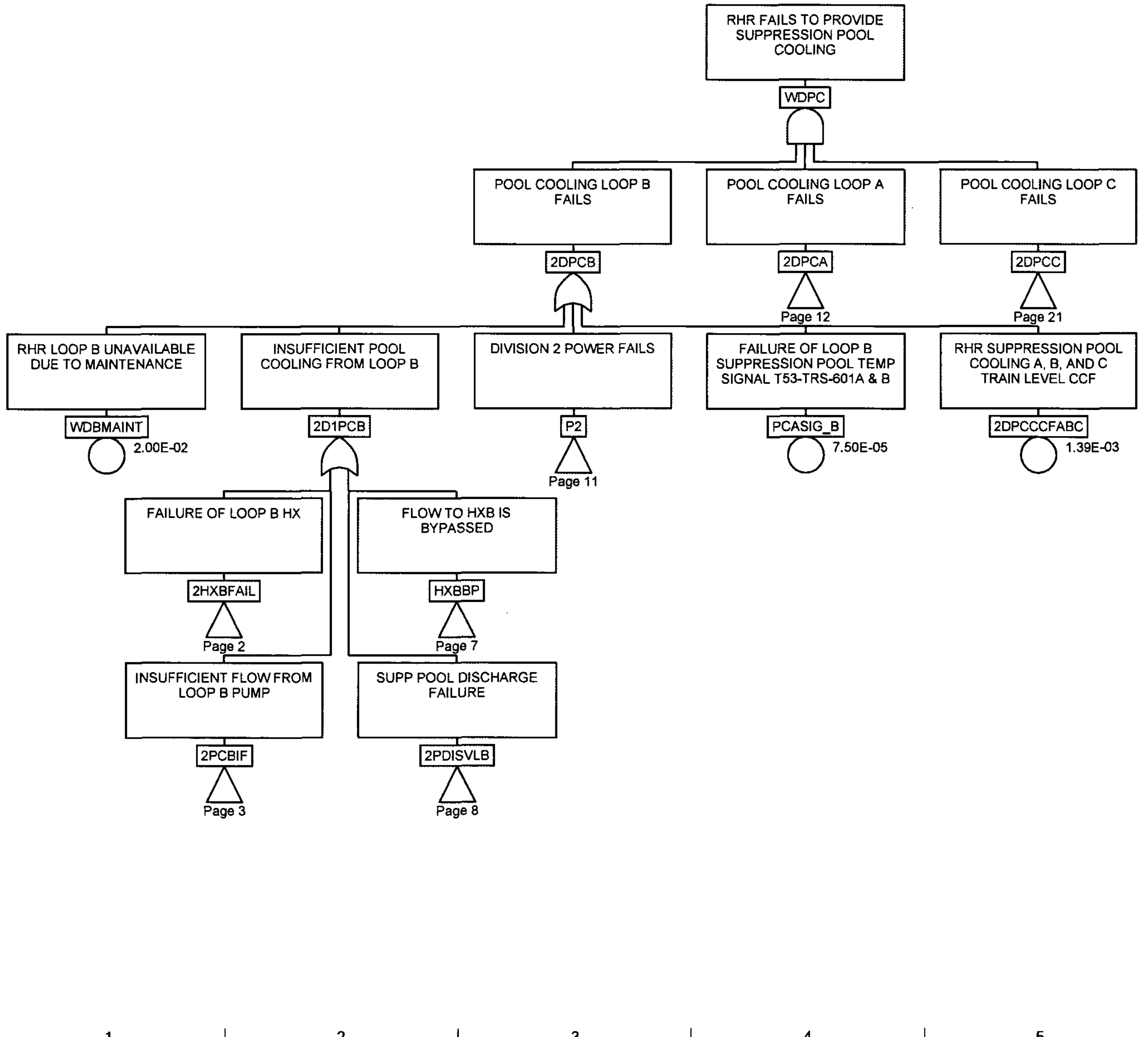


Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree

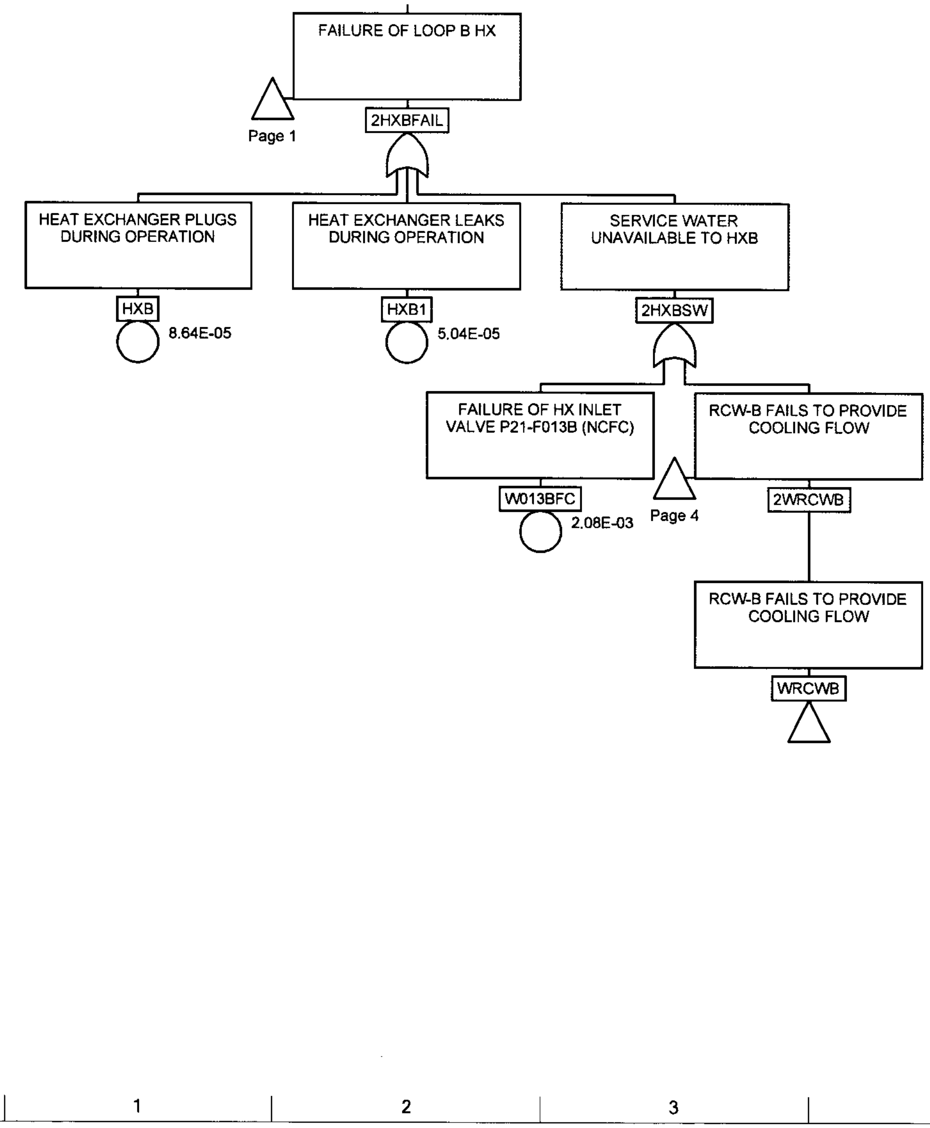


Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree

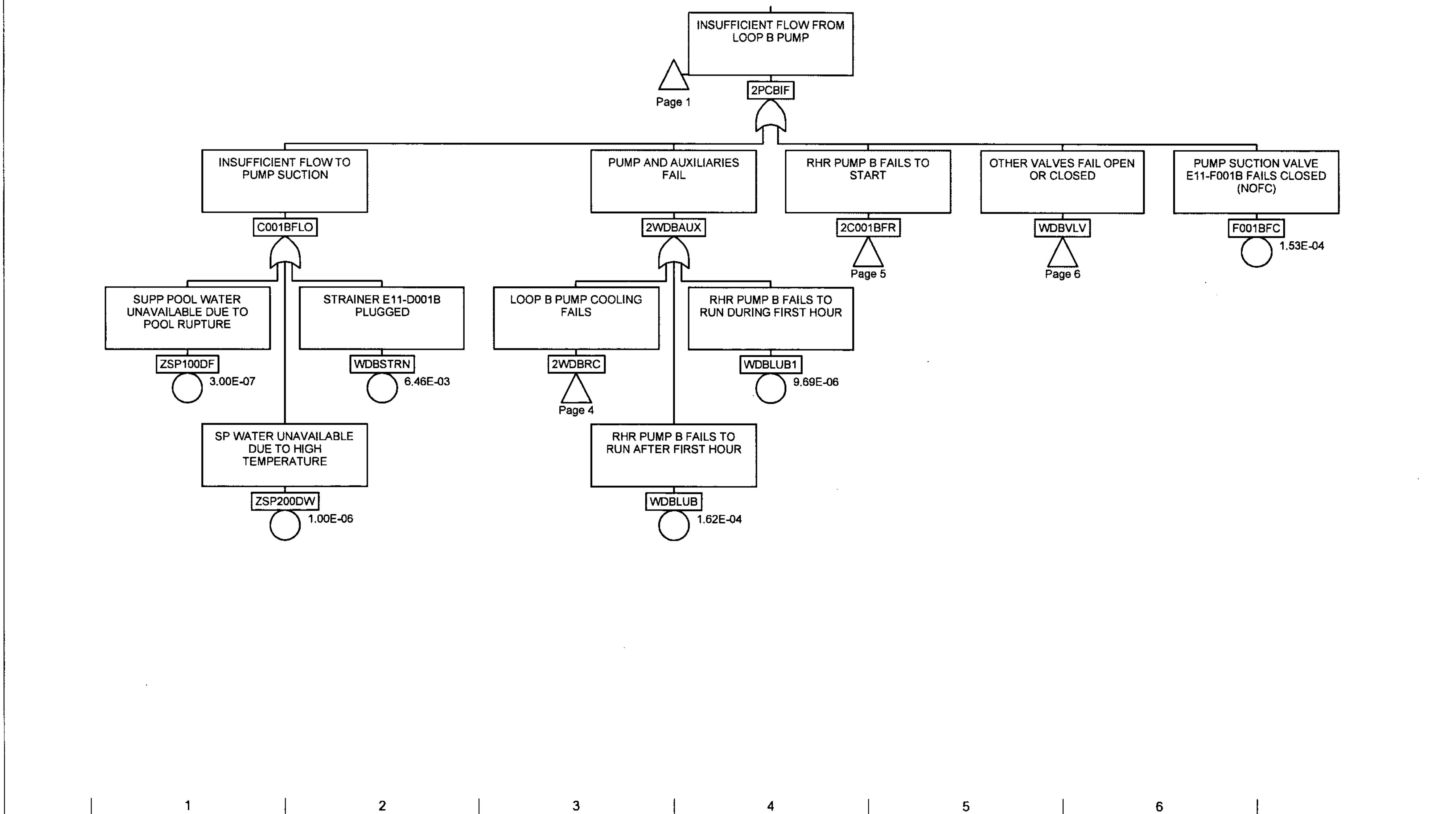


Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree

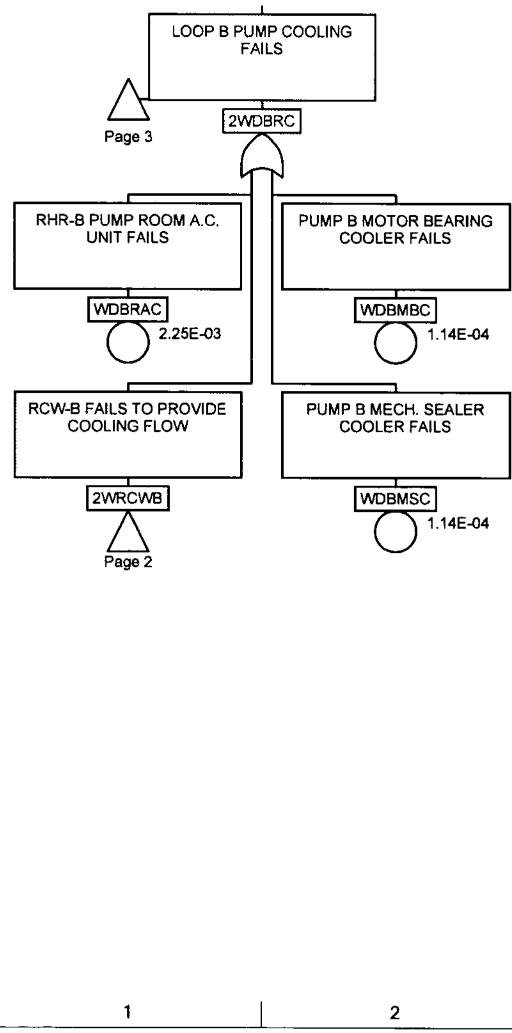


Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree

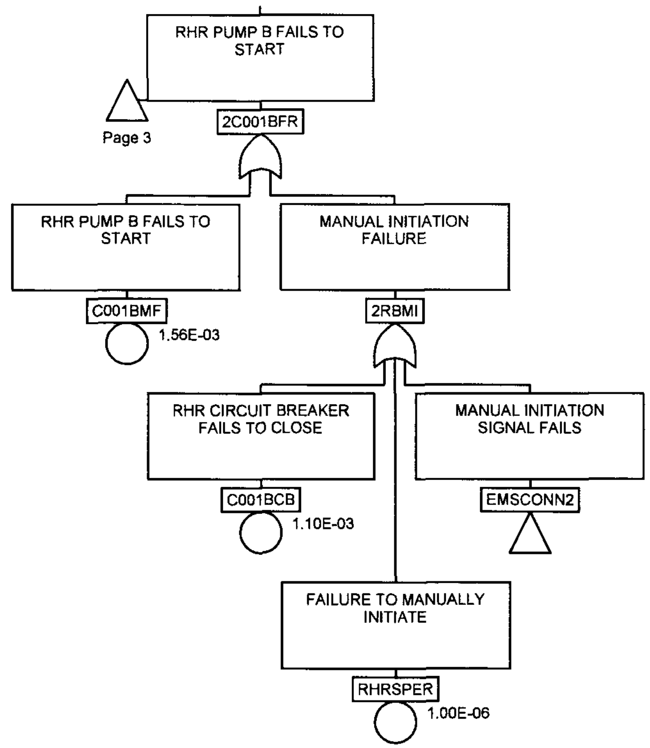


Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree

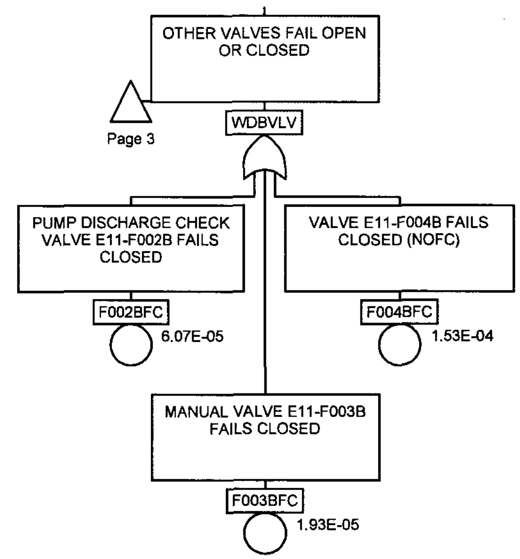
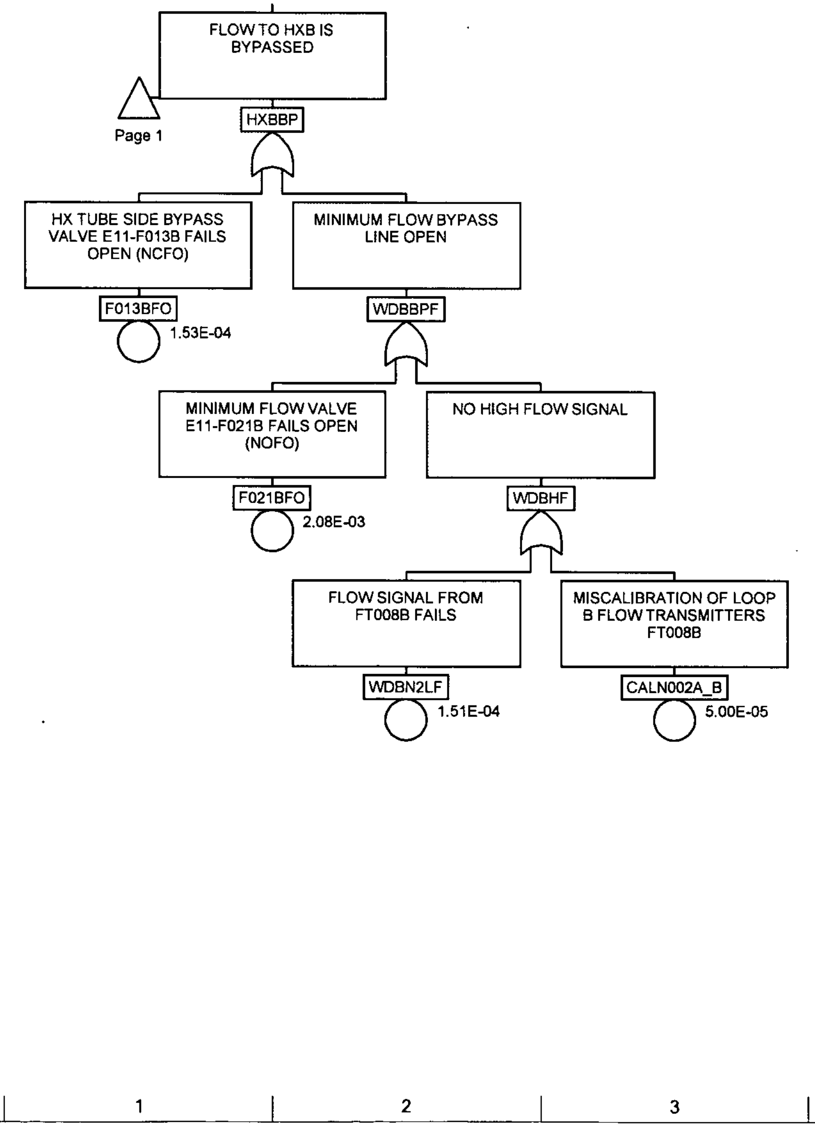


Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree



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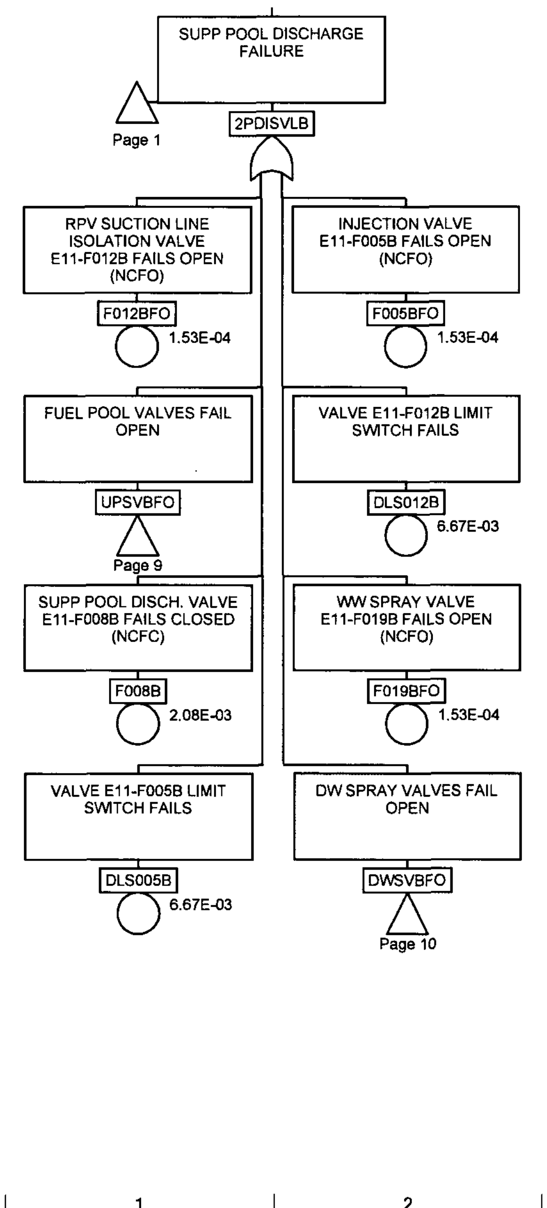

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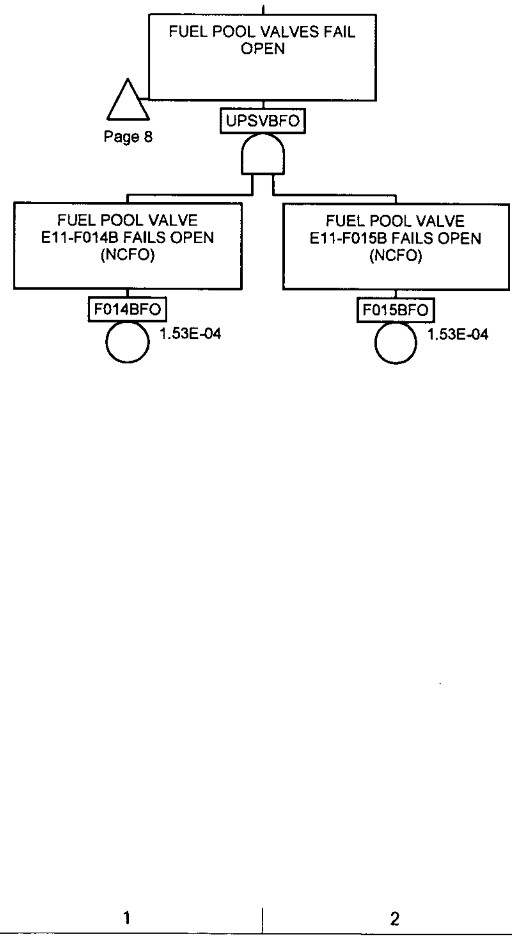

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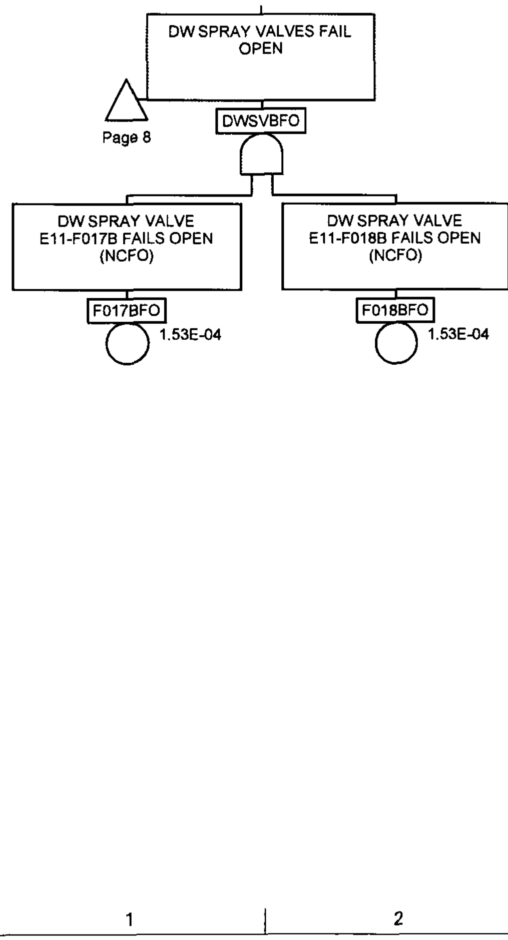

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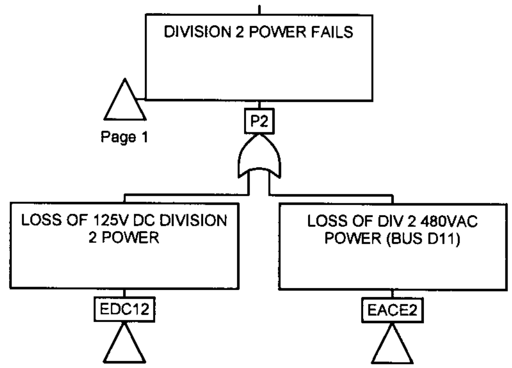

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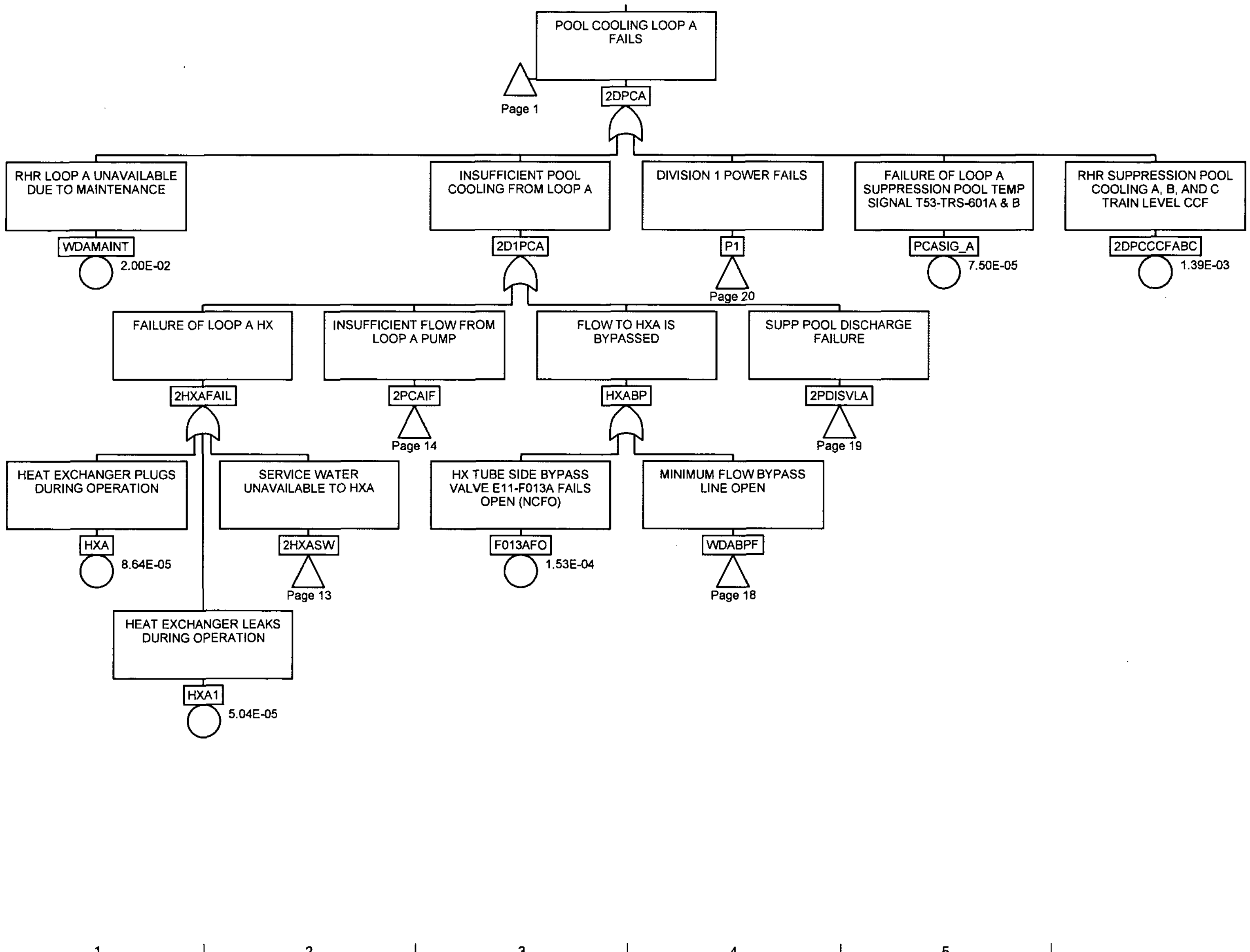
Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree



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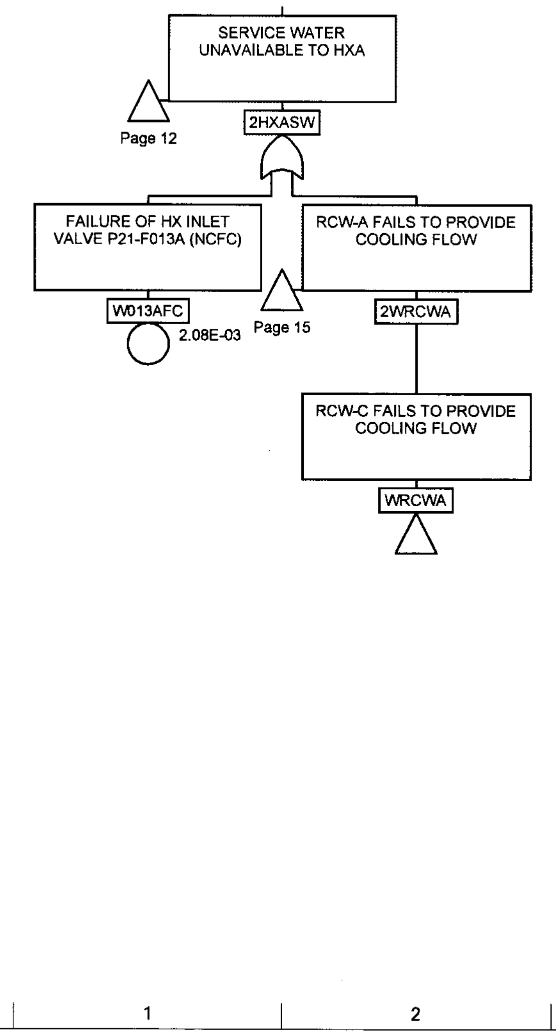

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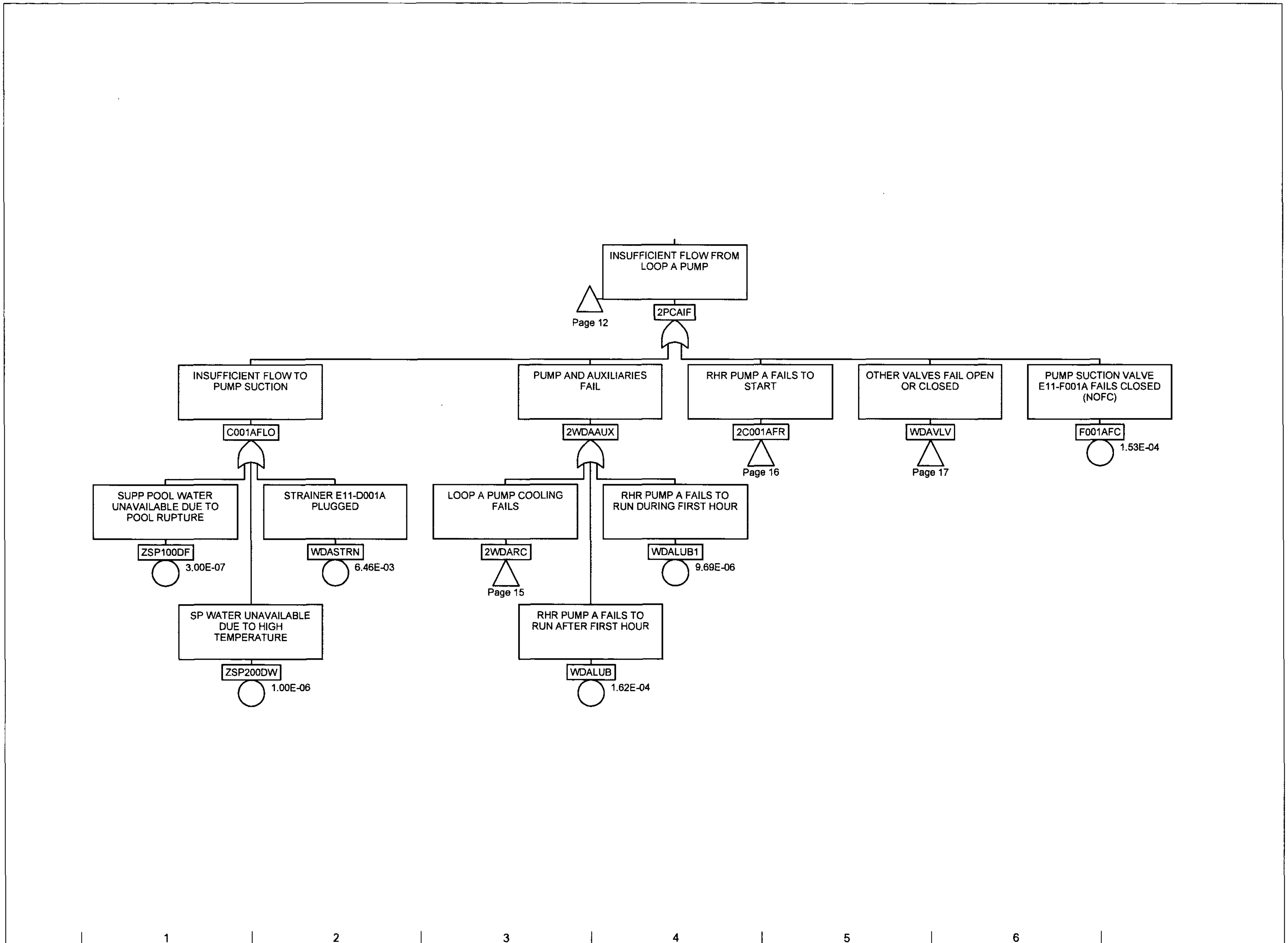
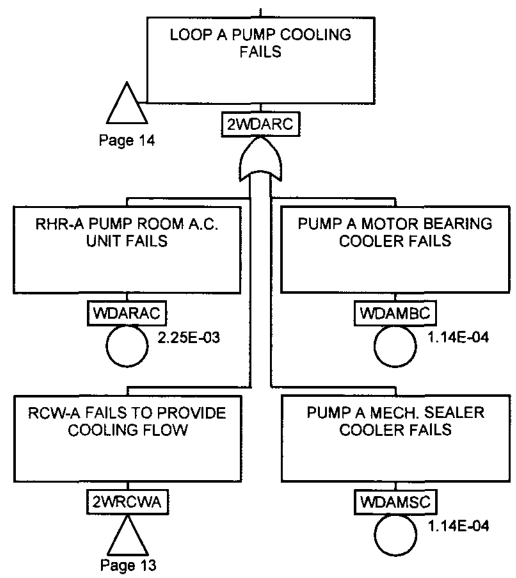


Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree



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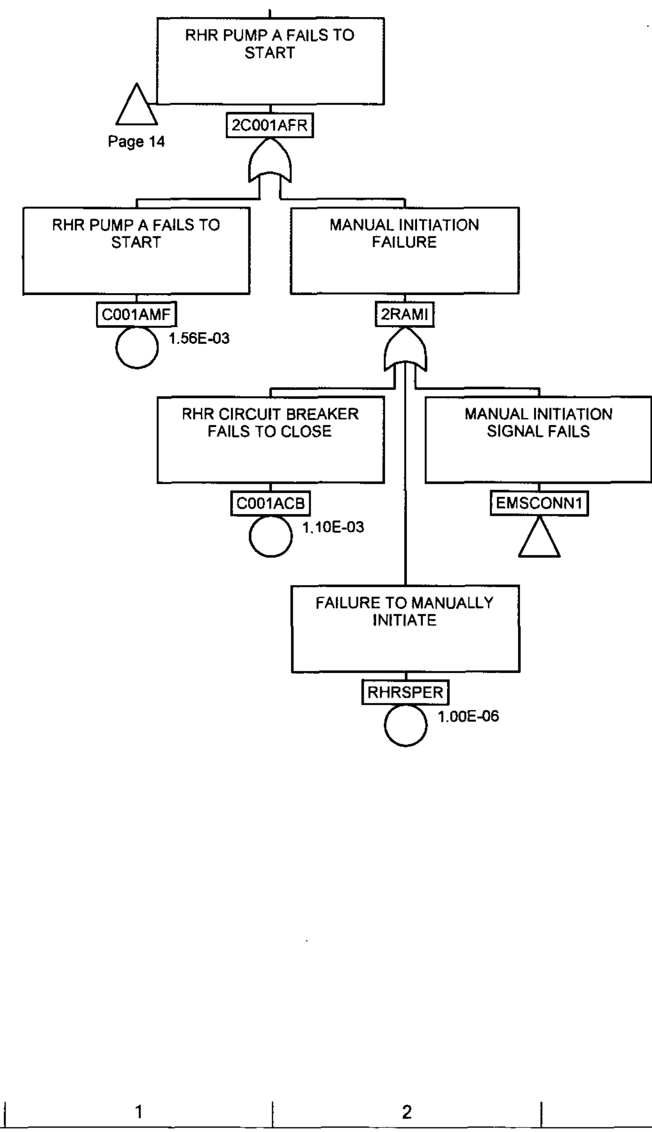

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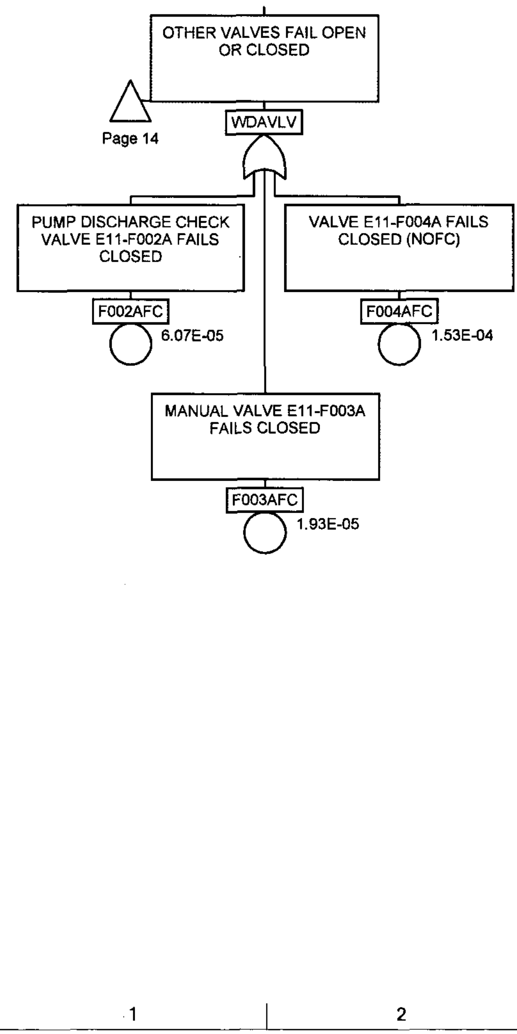

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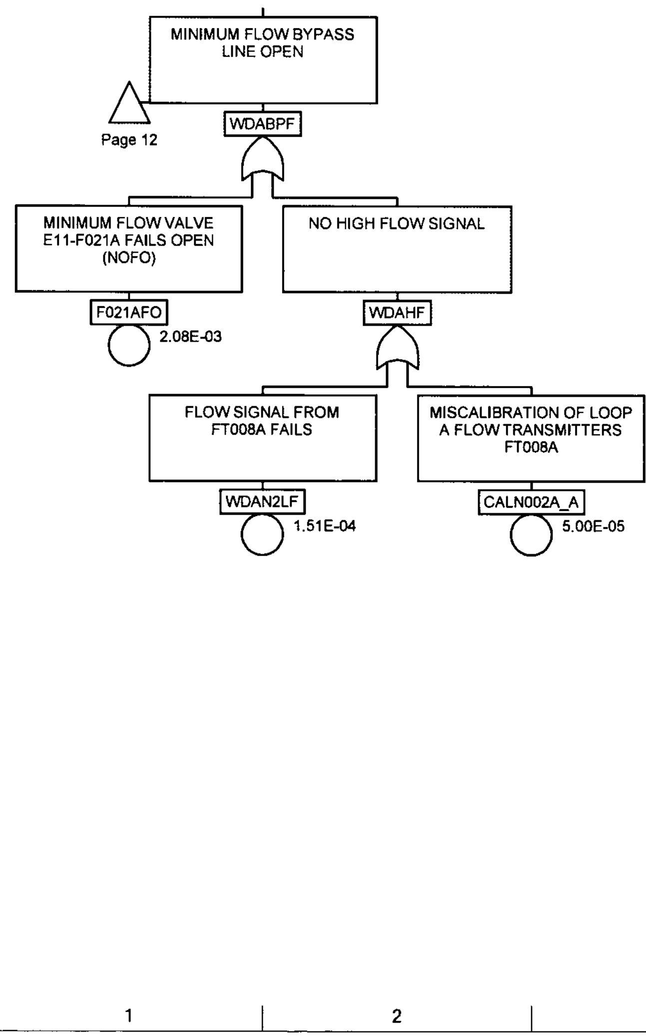

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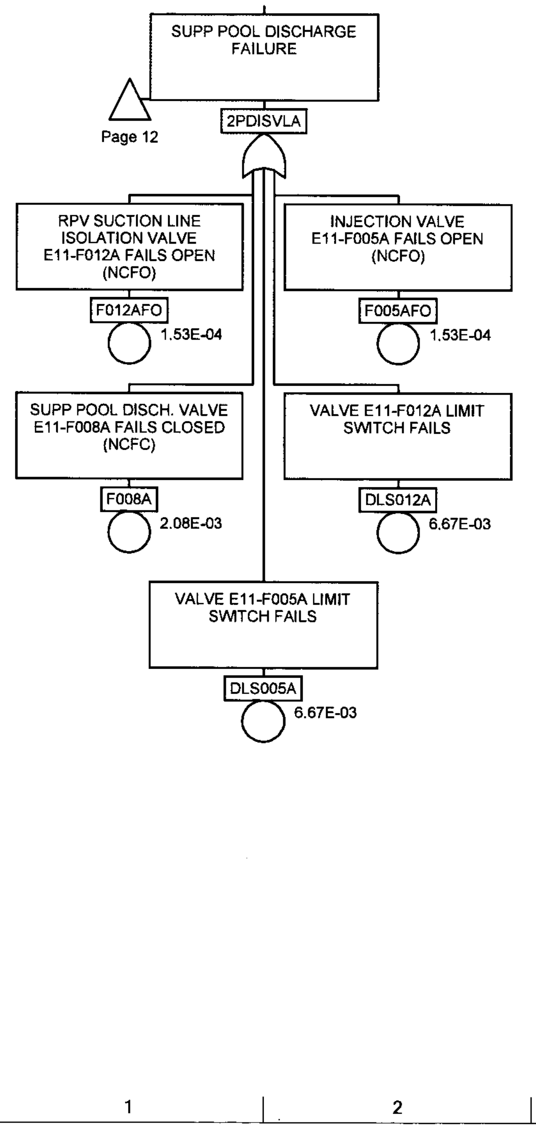

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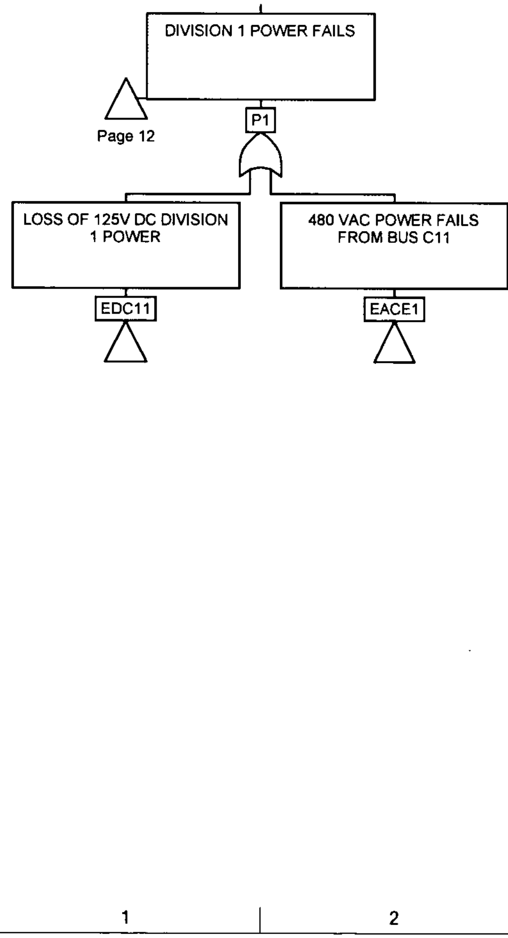

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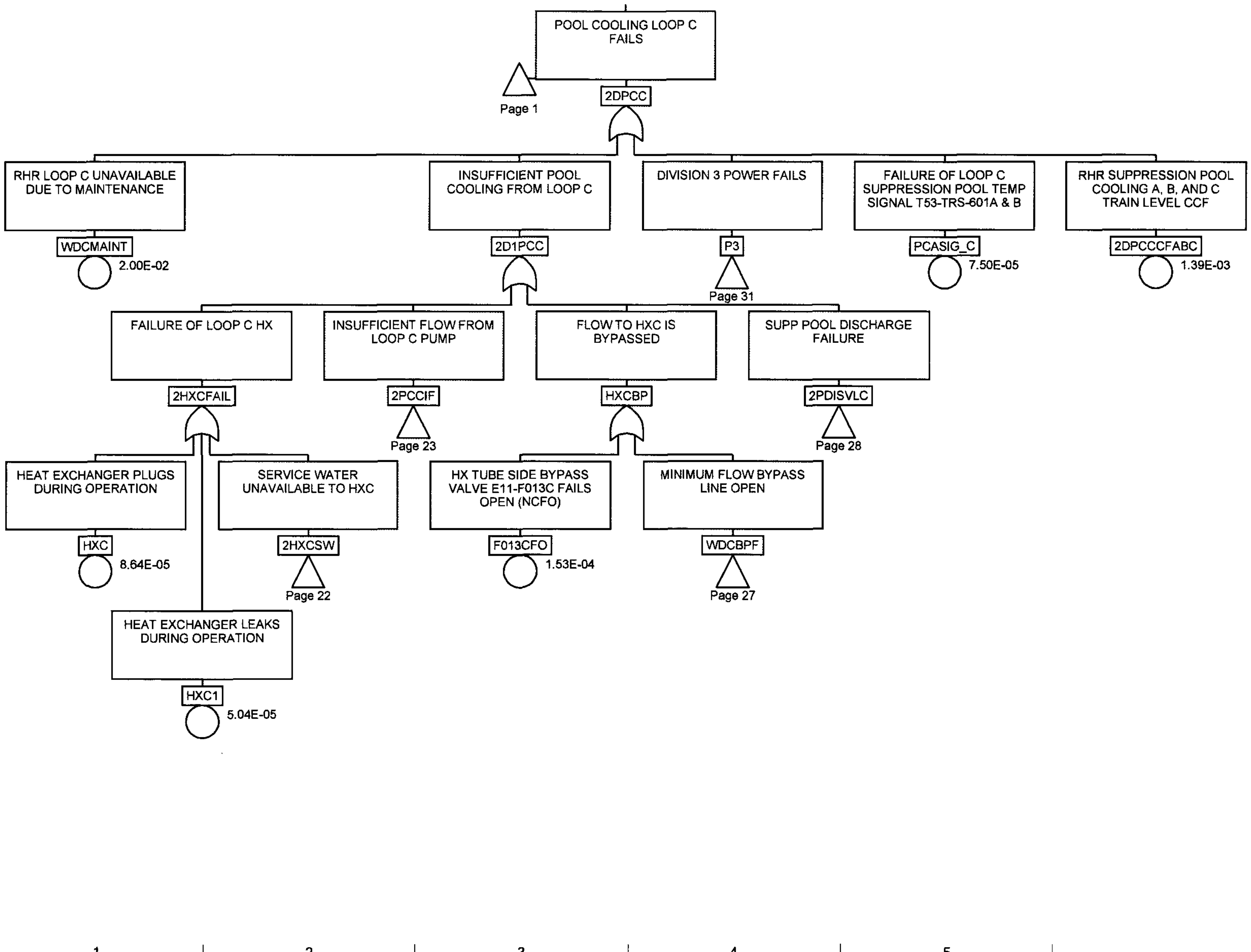


Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree

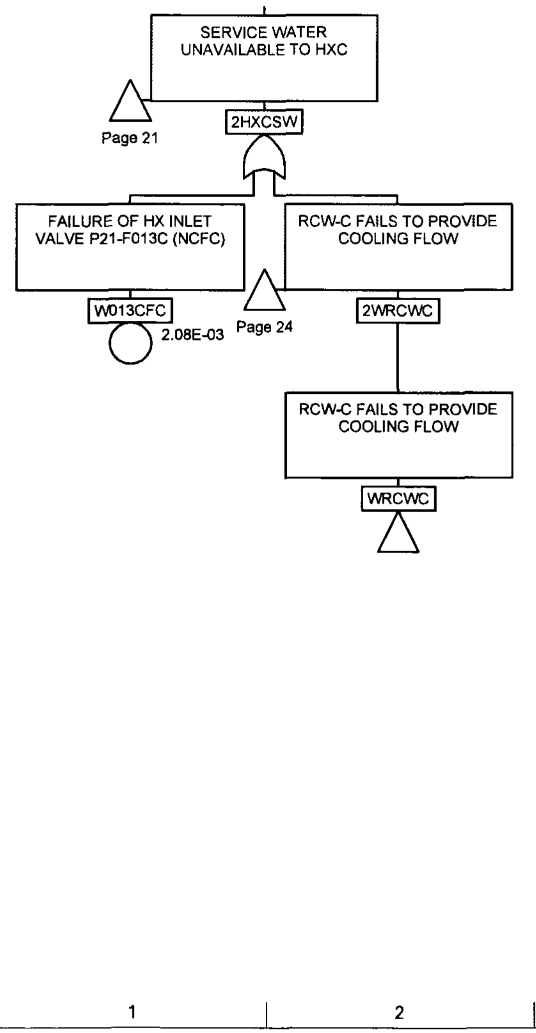
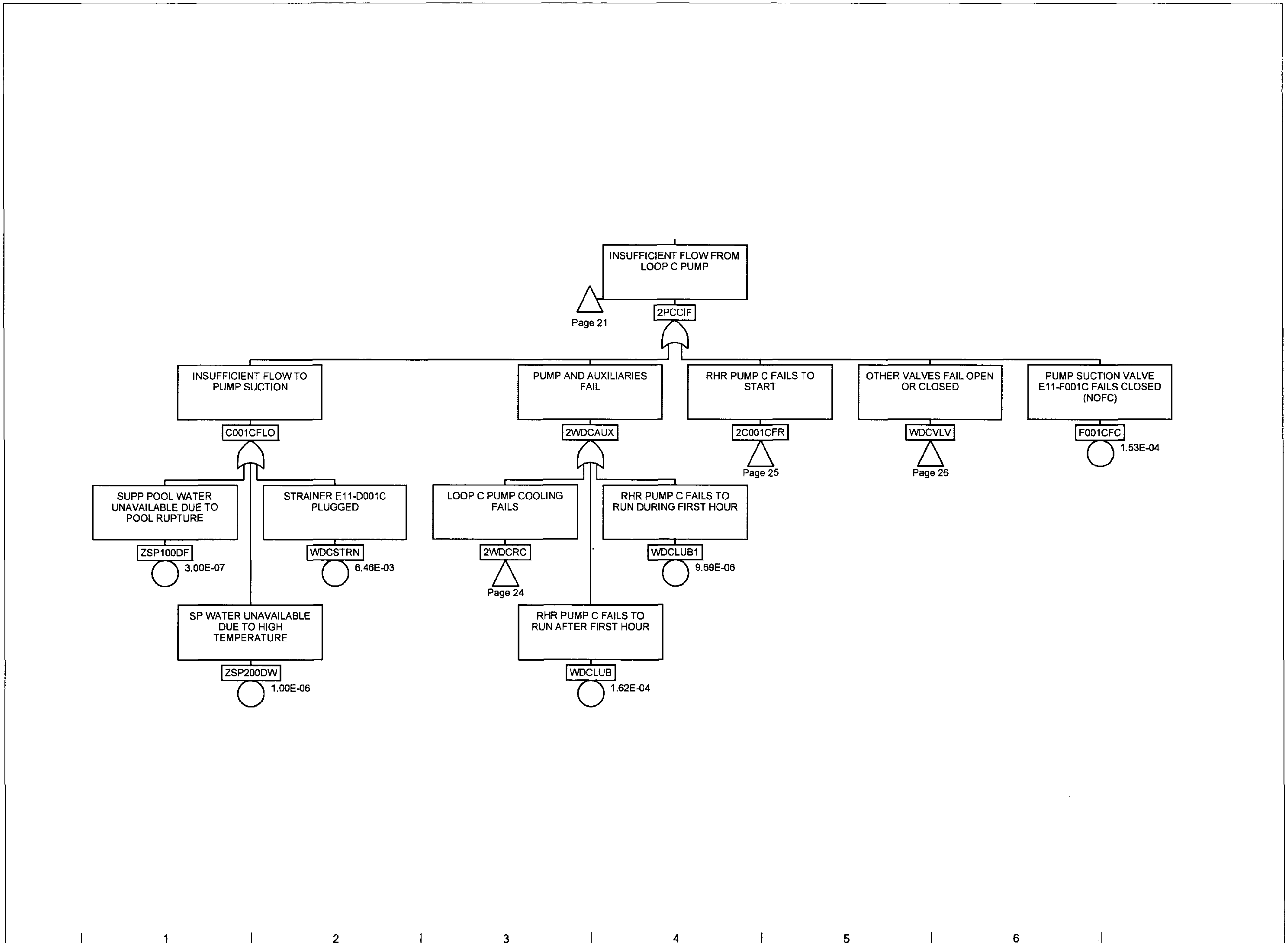


Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree



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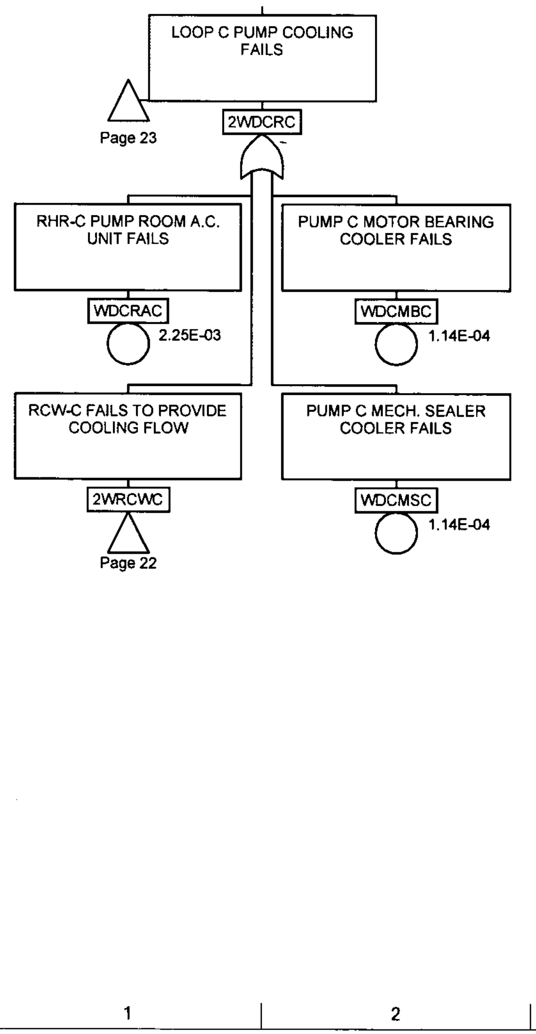


Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree

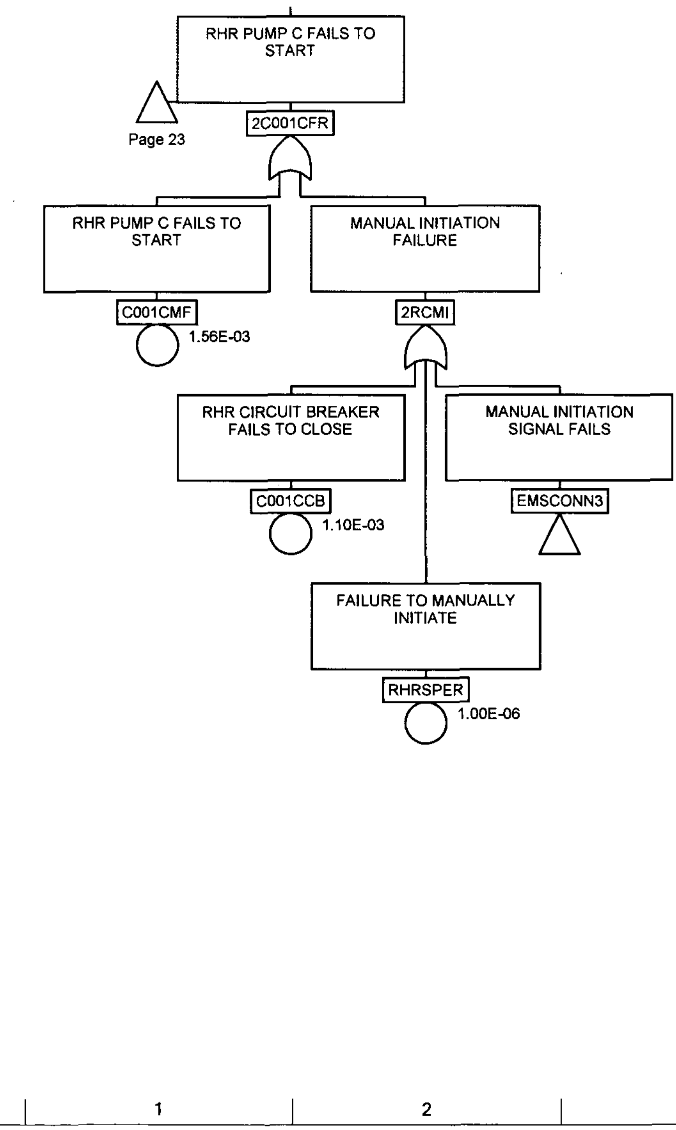


Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree

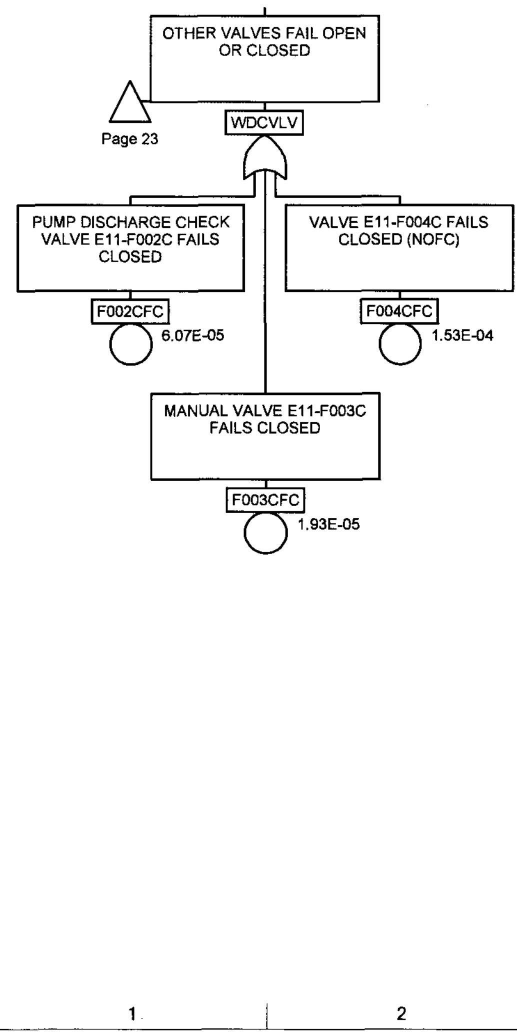


Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree

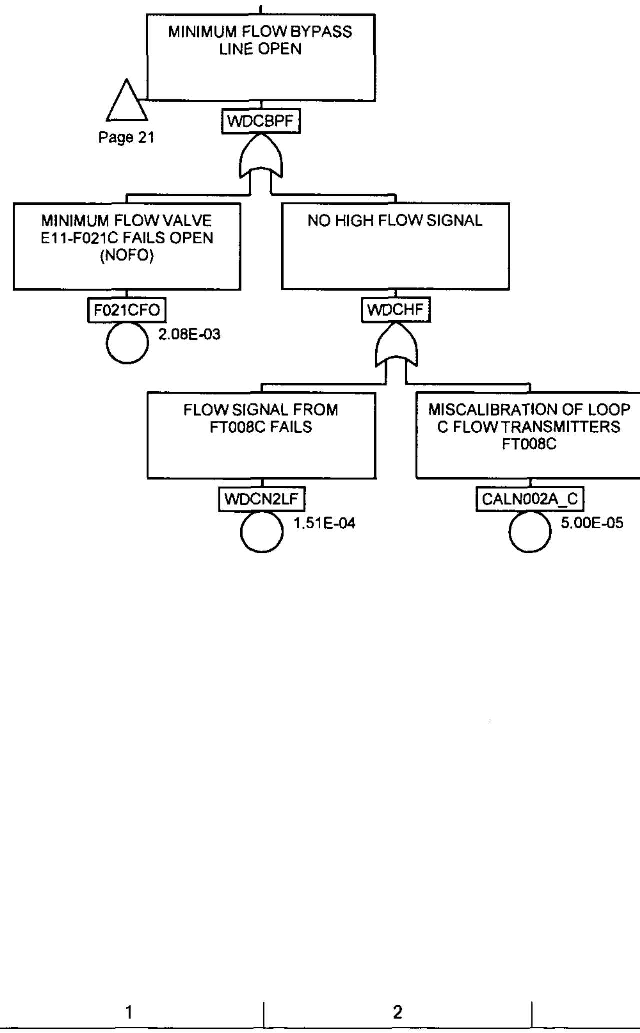
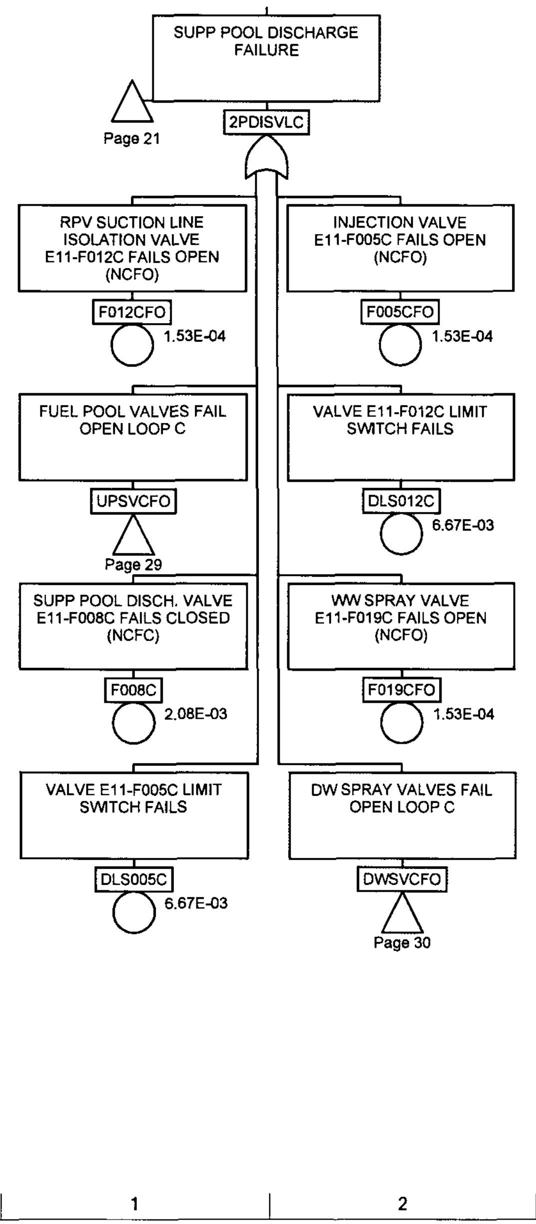


Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree



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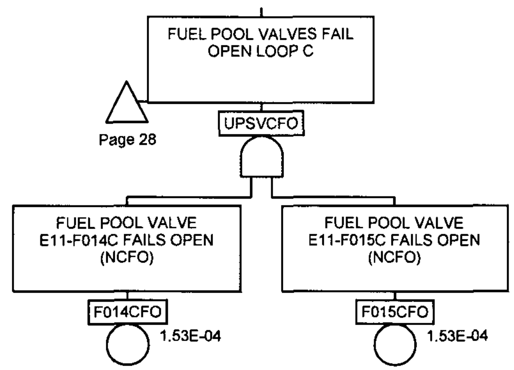

Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree

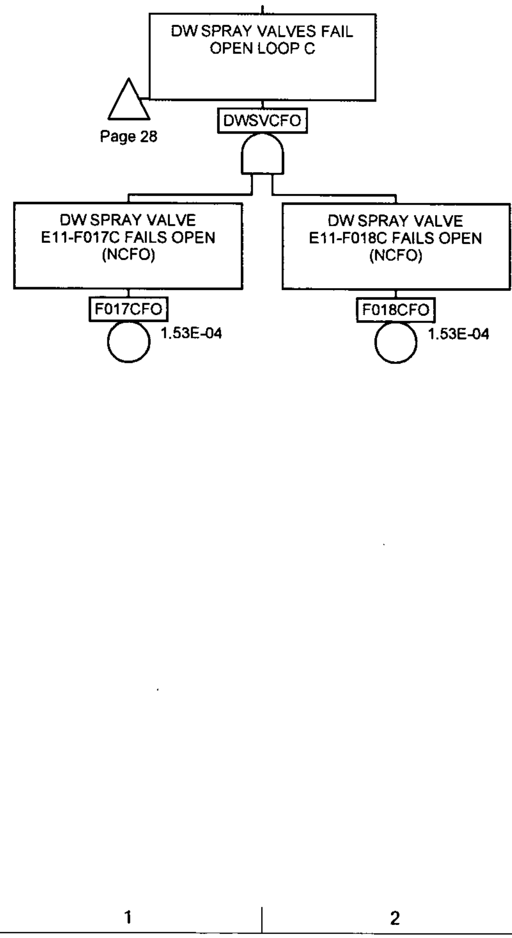

Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree

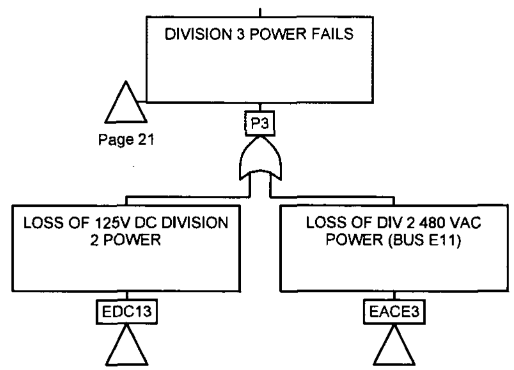

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Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree

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2C001AFR	16	2	2WDCRC	24	2
2C001BFR	3	5	2WRCWA	13	2
2C001BFR	5	2	2WRCWA	15	1
2C001CFR	23	5	2WRCWB	2	4
2C001CFR	25	2	2WRCWB	4	1
2D1PCA	12	3	2WRCWC	22	2
2D1PCB	1	2	2WRCWC	24	1
2D1PCC	21	3	C001ACB	16	2
2DPCA	1	4	C001AFLO	14	2
2DPCA	12	3	C001AMF	16	1
2DPCB	1	3	C001BCB	5	2
2DPCC	1	5	C001BFLO	3	2
2DPCC	21	3	C001BMF	5	1
2DPCCCFABC	1	5	C001CCB	25	2
2DPCCCFABC	12	6	C001CFLO	23	2
2DPCCCFABC	21	6	C001CMF	25	1
2HXAFAIL	12	2	CALN002A_A	18	3
2HXASW	12	2	CALN002A_B	7	3
2HXASW	13	2	CALN002A_C	27	3
2HXBFAIL	1	2	DLS005A	19	2
2HXBFAIL	2	2	DLS005B	8	1
2HXBSW	2	3	DLS005C	28	1
2HXCFAIL	21	2	DLS012A	19	2
2HXCSW	21	2	DLS012B	8	2
2HXCSW	22	2	DLS012C	28	2
2PCAIF	12	3	DWSVBFO	8	2
2PCAIF	14	4	DWSVBFO	10	2
2PCBIF	1	2	DWSVCFO	28	2
2PCBIF	3	4	DWSVCFO	30	2
2PCCIF	21	3	EACE1	20	2
2PCCIF	23	4	EACE2	11	2
2PDISVLA	12	5	EACE3	31	2
2PDISVLA	19	2	EDC11	20	1
2PDISVLB	1	3	EDC12	11	1
2PDISVLB	8	2	EDC13	31	1
2PDISVLC	21	5	EMSCONN1	16	3
2PDISVLC	28	2	EMSCONN2	5	3
2RAMI	16	2	EMSCONN3	25	3
2RBMI	5	2	F001AFC	14	7
2RCMI	25	2	F001BFC	3	7
2WDAAUX	14	4	F001CFC	23	7
2WDARC	14	3	F002AFC	17	1
2WDARC	15	2	F002BFC	6	1
2WDBAUX	3	4	F002CFC	26	1
2WDBRC	3	3	F003AFC	17	2
2WDBRC	4	2	F003BFC	6	2
2WDCAUX	23	4	F003CFC	26	2

Name	Page	Zone	Name	Page	Zone
F004AFC	17	2	RHRSPER	16	2
F004BFC	6	2	RHRSPER	25	2
F004CFC	26	2	UPSVBFO	8	1
F005AFO	19	2	UPSVBFO	9	2
F005BFO	8	2	UPSVCFO	28	1
F005CFO	28	2	UPSVCFO	29	2
F008A	19	1	W013AFC	13	1
F008B	8	1	W013BFC	2	3
F008C	28	1	W013CFC	22	1
F012AFO	19	1	WDABPF	12	4
F012BFO	8	1	WDABPF	18	2
F012CFO	28	1	WDAHF	18	2
F013AFO	12	3	WDALUB	14	4
F013BFO	7	1	WDALUB1	14	4
F013CFO	21	3	WDAMAINT	12	1
F014BFO	9	1	WDAMBC	15	2
F014CFO	29	1	WDAMSC	15	2
F015BFO	9	2	WDAN2LF	18	2
F015CFO	29	2	WDARAC	15	1
F017BFO	10	1	WDASTRN	14	2
F017CFO	30	1	WDAVLV	14	6
F018BFO	10	2	WDAVLV	17	2
F018CFO	30	2	WDBBPF	7	2
F019BFO	8	2	WDBHF	7	3
F019CFO	28	2	WDBLUB	3	4
F021AFO	18	1	WDBLUB1	3	4
F021BFO	7	2	WDBMAINT	1	1
F021CFO	27	1	WDBMBC	4	2
HXA	12	1	WDBMSC	4	2
HXA1	12	2	WDBN2LF	7	2
HXABP	12	4	WDBRAC	4	1
HXB	2	1	WDBSTRN	3	2
HXB1	2	2	WDBVLV	3	6
HXBBP	1	3	WDBVLV	6	2
HXBBP	7	2	WDCBPF	21	4
HXC	21	1	WDCBPF	27	2
HXC1	21	2	WDCHF	27	2
HXCBP	21	4	WDCLUB	23	4
P1	12	4	WDCLUB1	23	4
P1	20	2	WDCMAINT	21	1
P2	1	3	WDCMBC	24	2
P2	11	2	WDCMSC	24	2
P3	21	4	WDCN2LF	27	2
P3	31	2	WDCRAC	24	1
PCASIG_A	12	5	WDCSTRN	23	2
PCASIG_B	1	4	WDCVLV	23	6
PCASIG_C	21	5	WDCVLV	26	2
RHRSPER	5	2	WDPC	1	4

Figure 19D.6-6 RHR Suppression Pool Cooling Mode Fault Tree

Name	Page	Zone	Name	Page	Zone
WRCWA	13	2			
WRCWB	2	4			
WRCWC	22	2			
ZSP100DF	3	1			
ZSP100DF	14	1			
ZSP100DF	23	1			
ZSP200DW	3	2			
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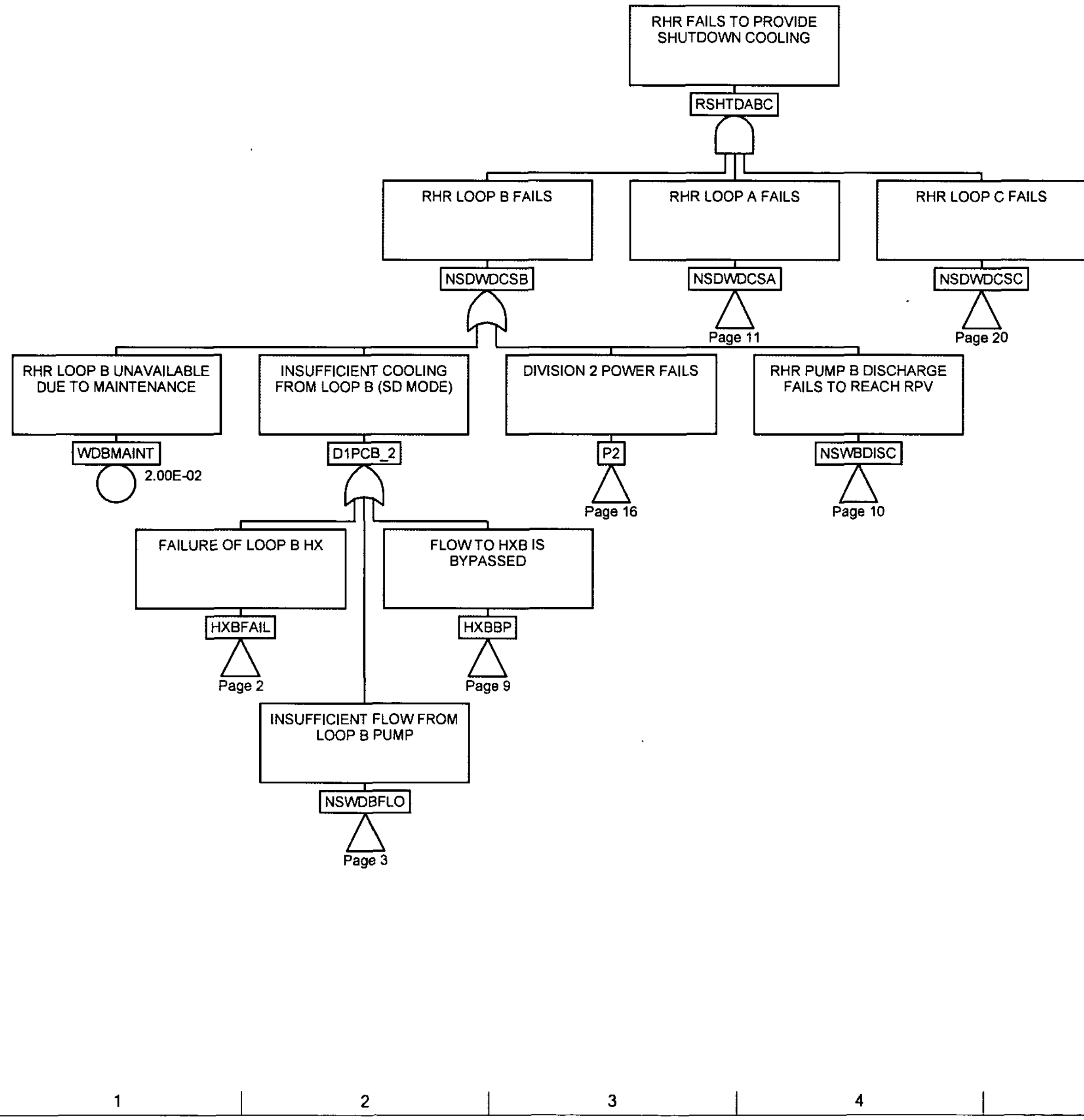


Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

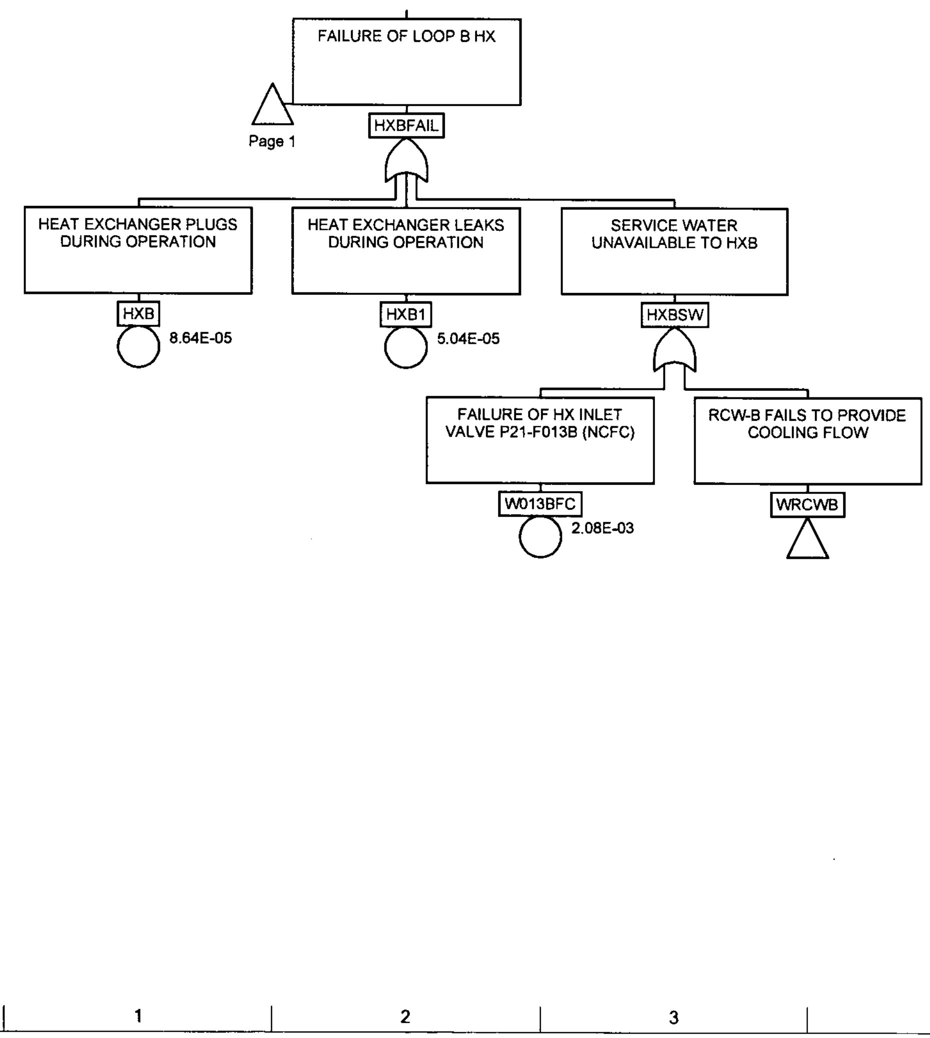


Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

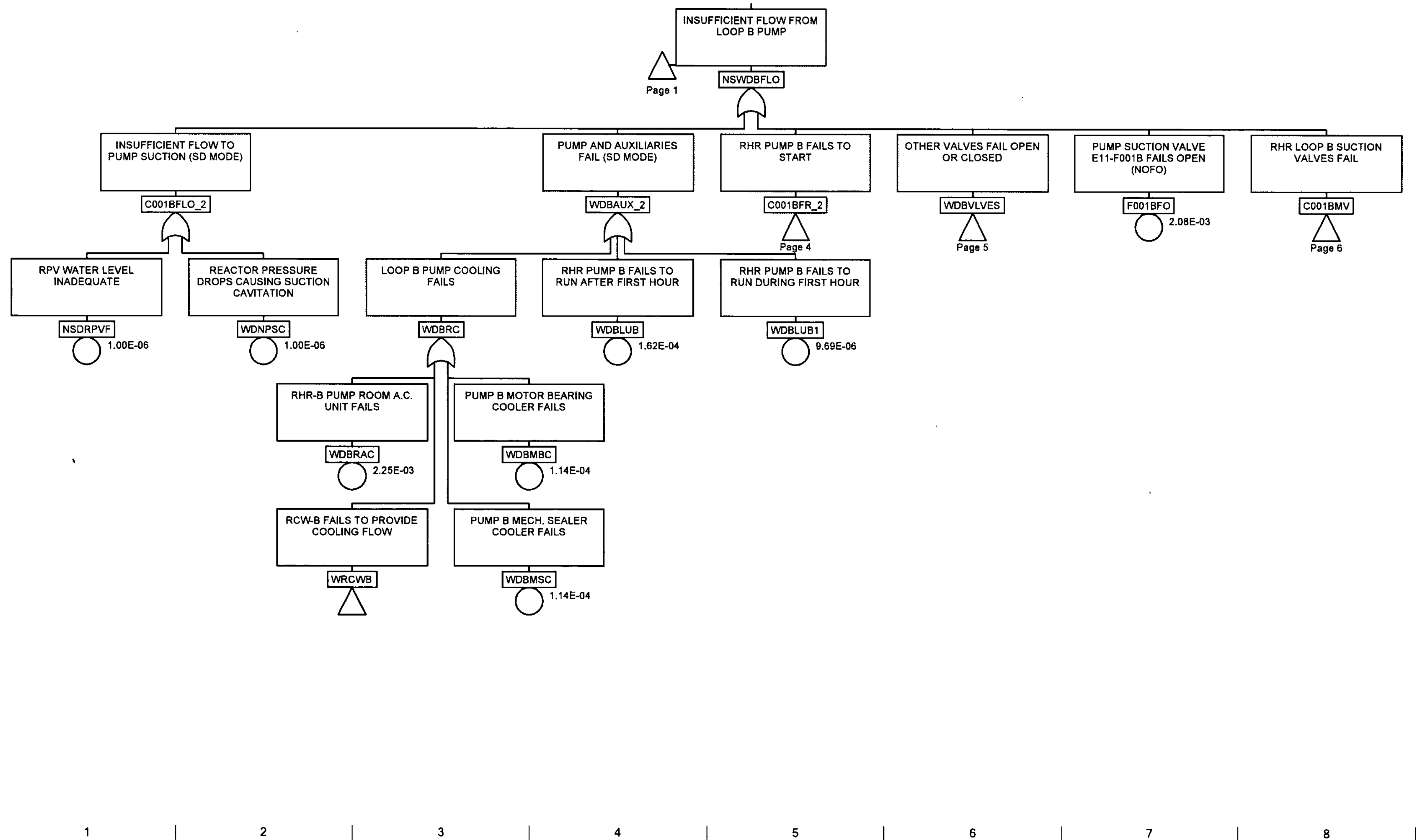
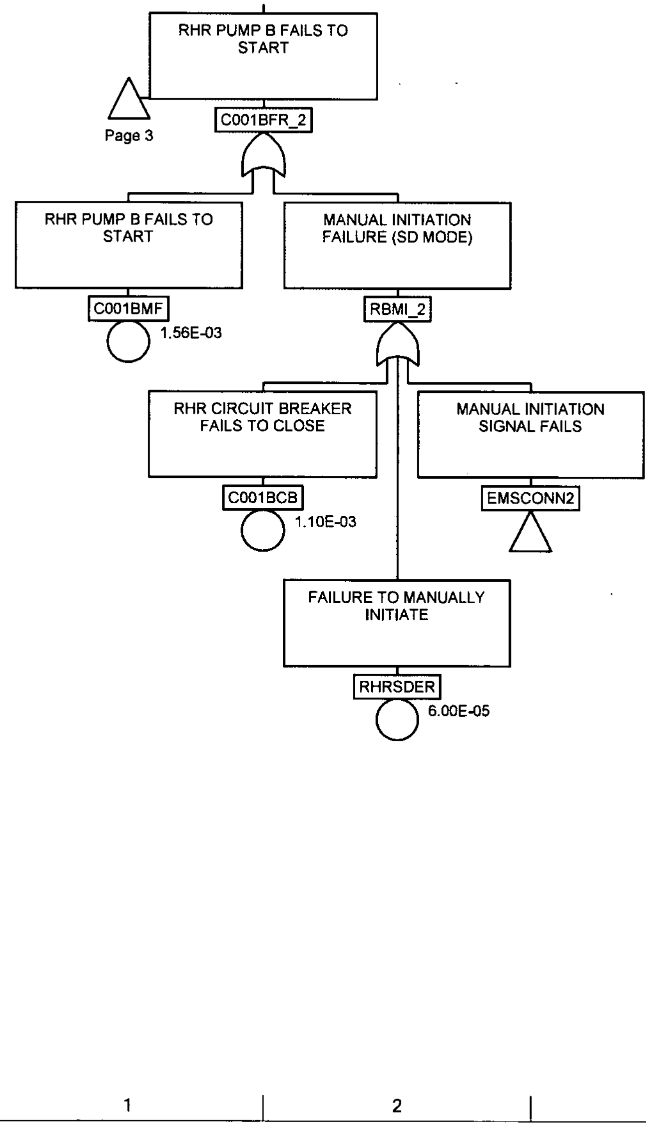


Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree



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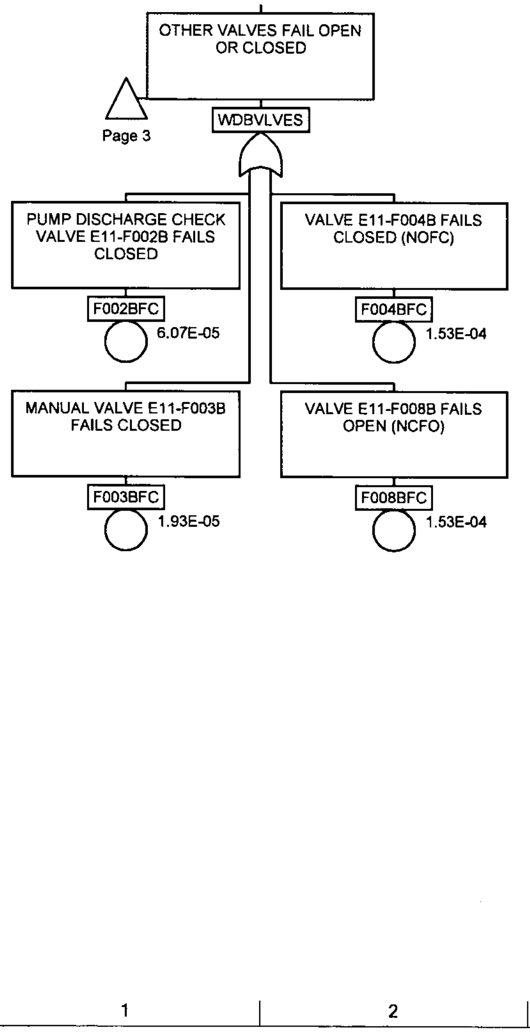

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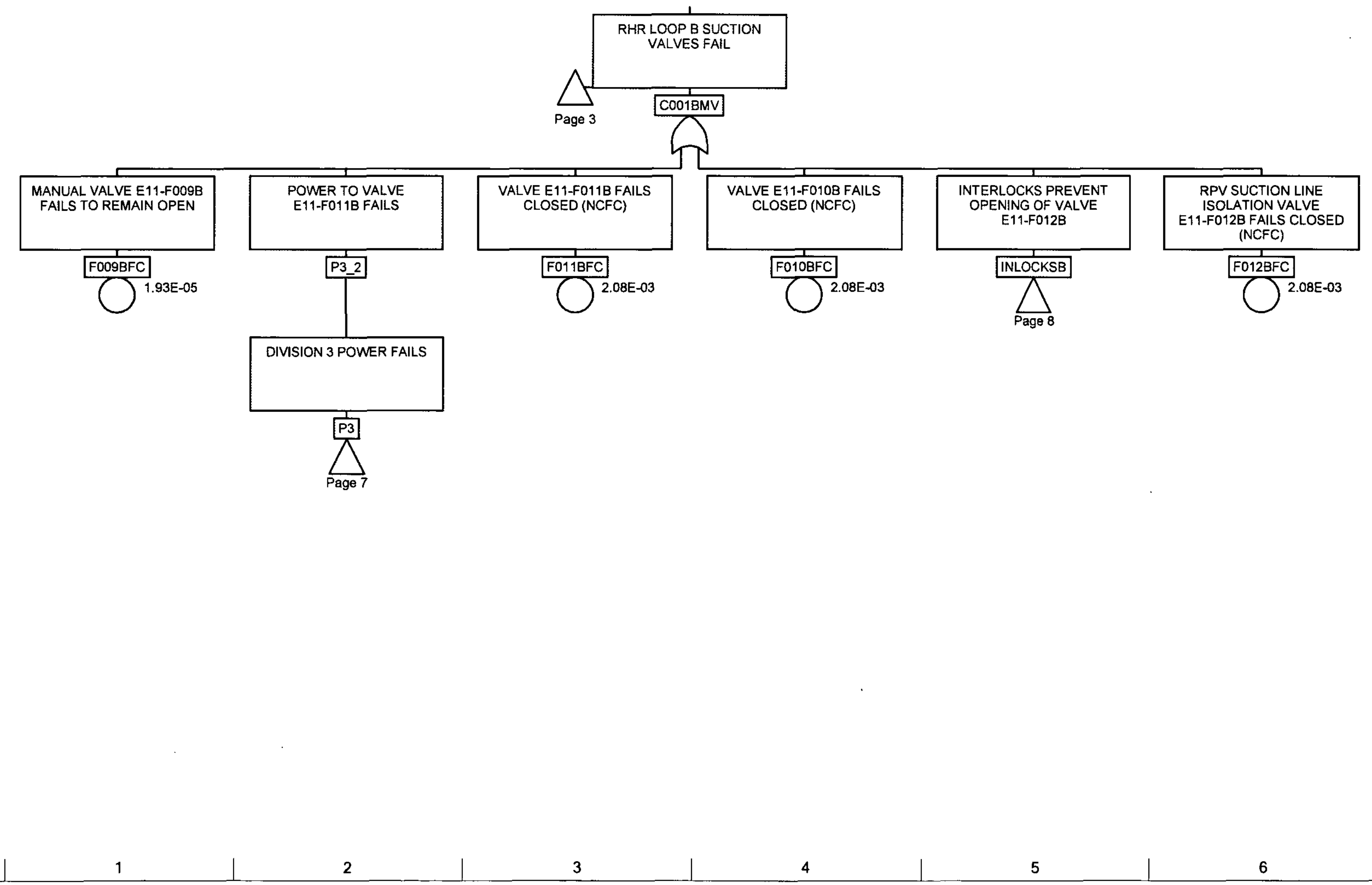
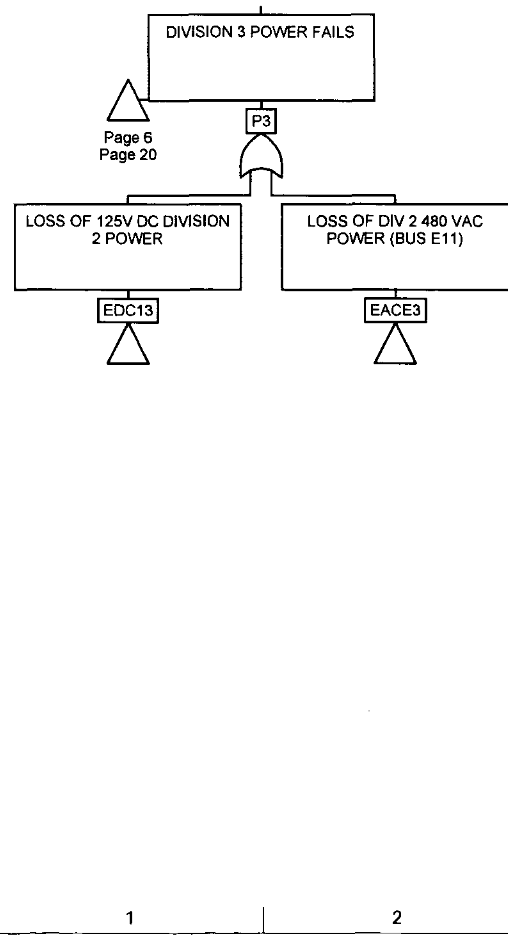


Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree



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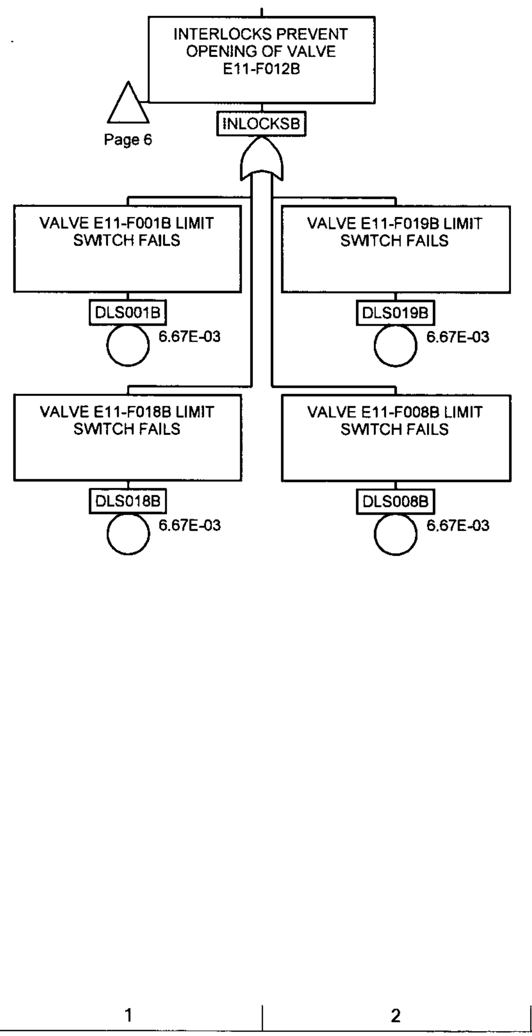

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

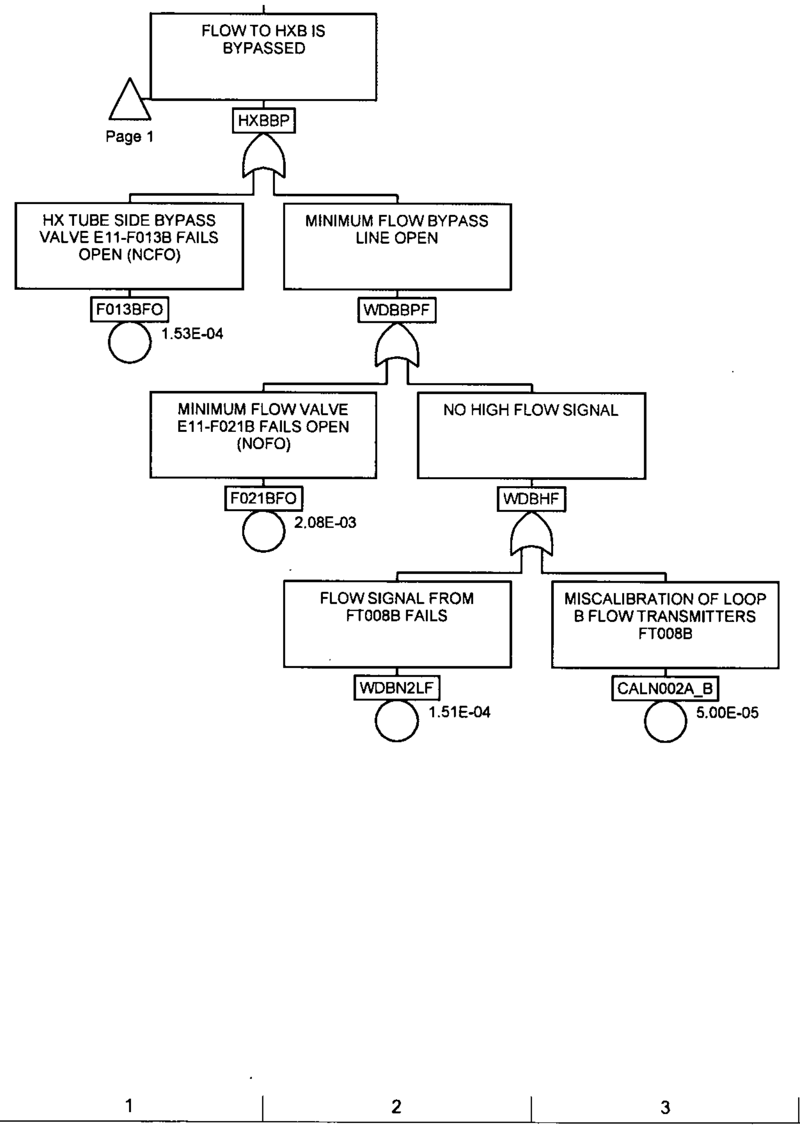

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

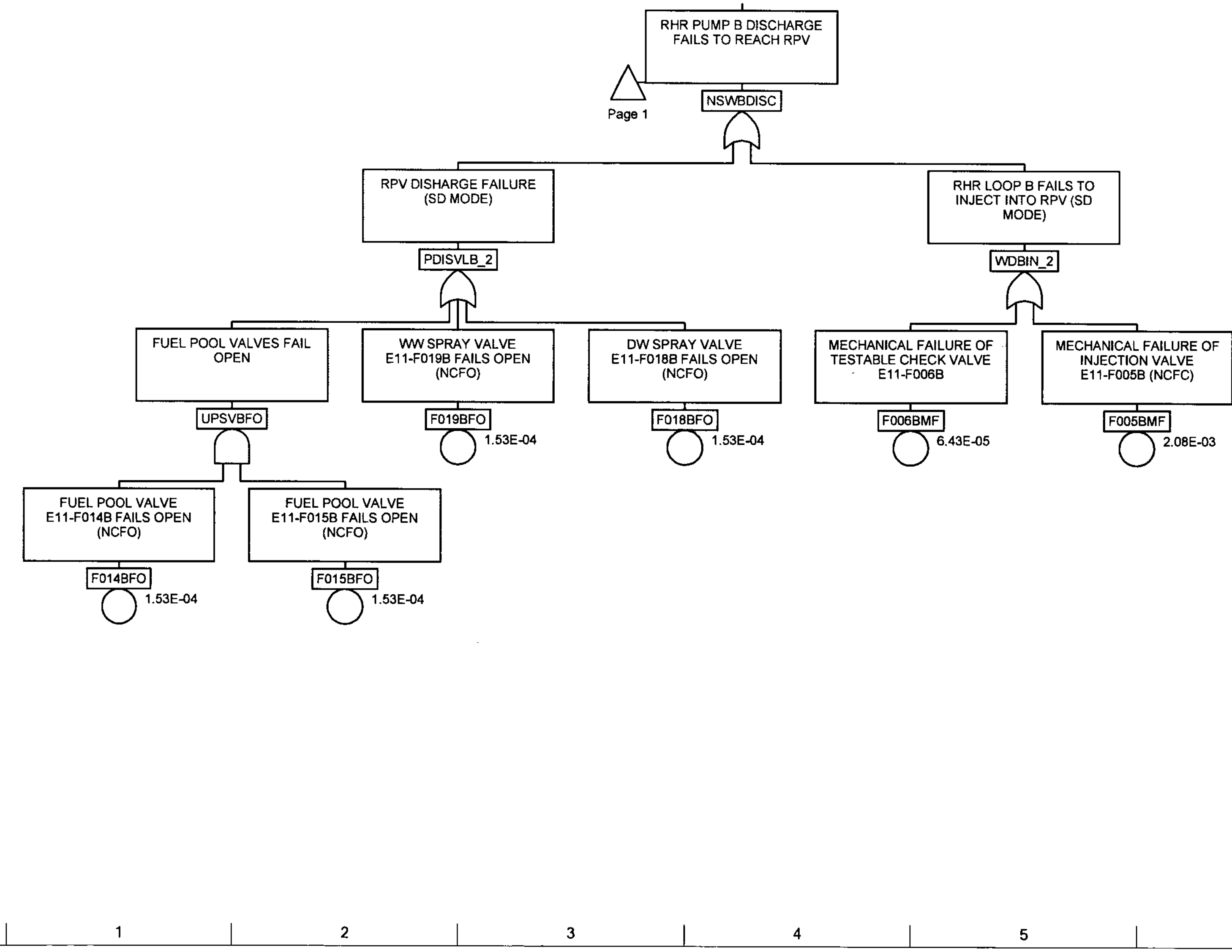

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

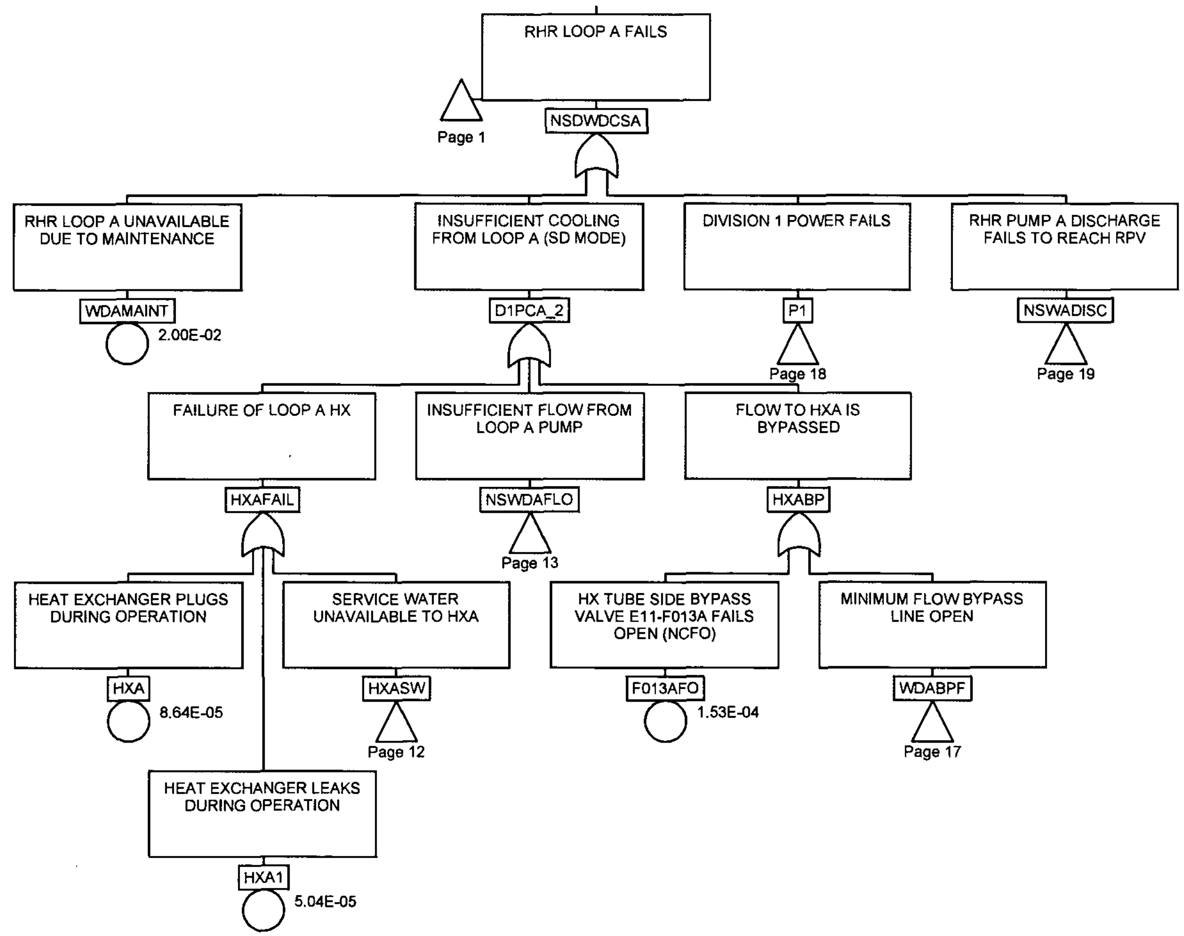

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

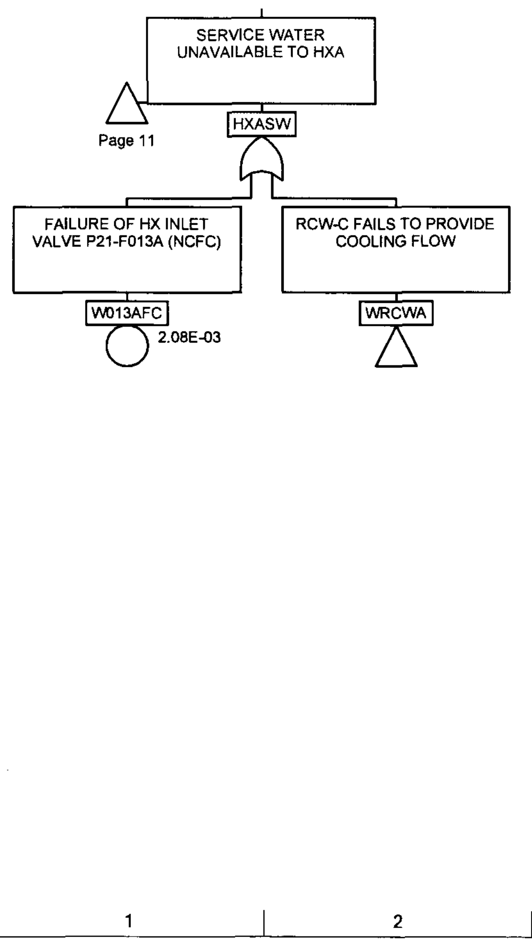

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

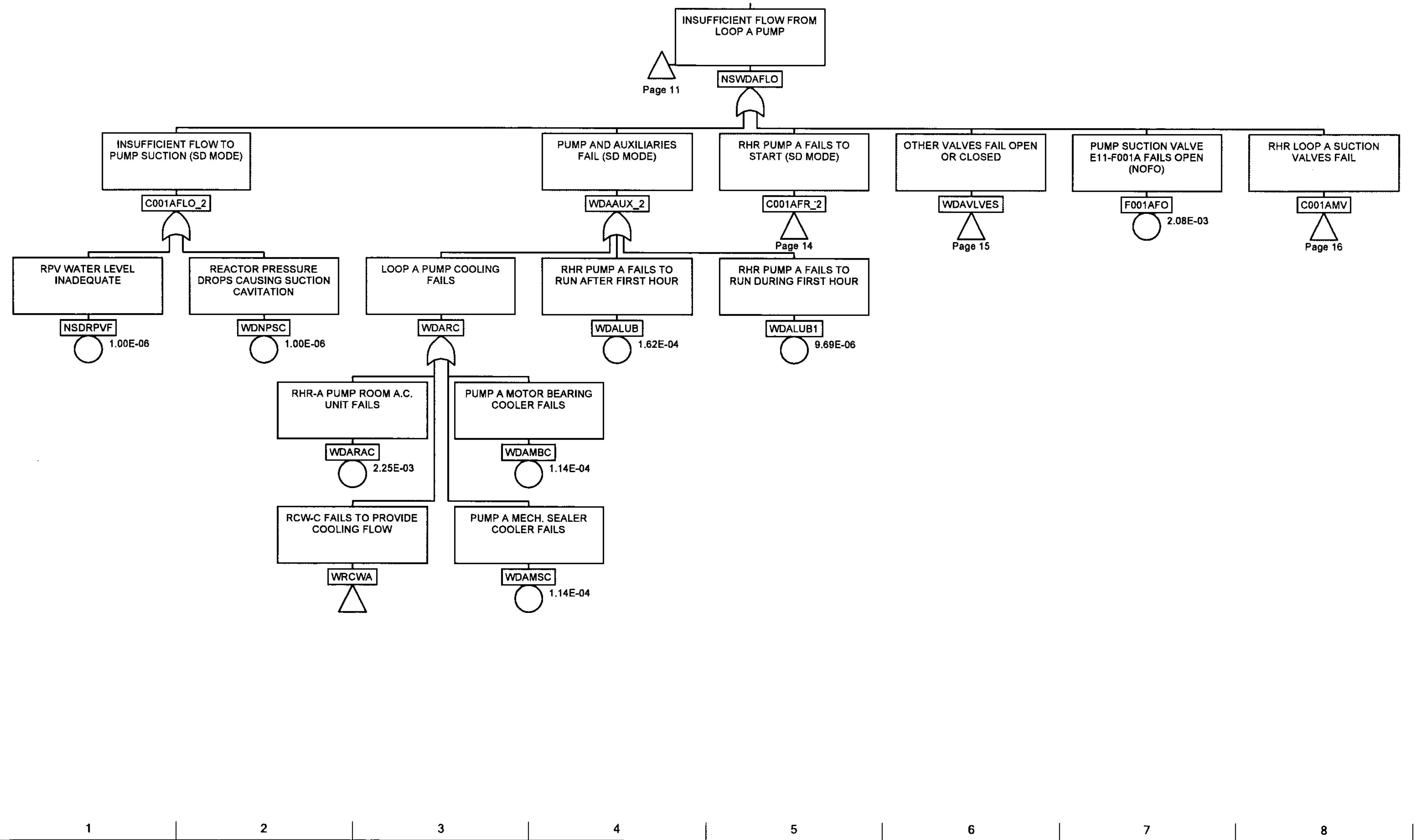

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

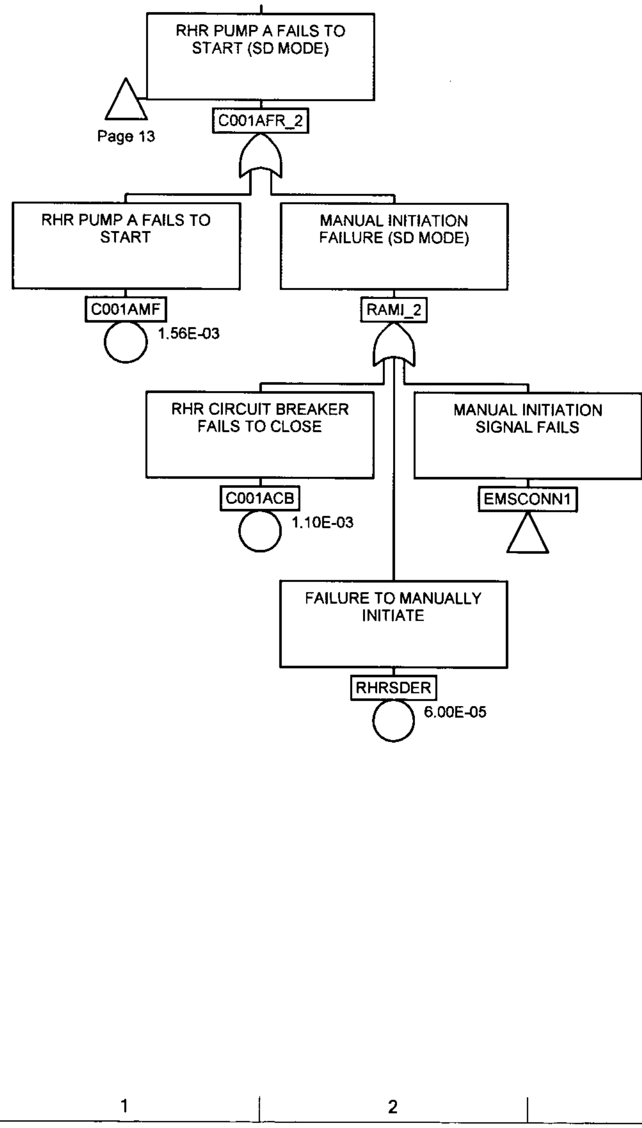

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

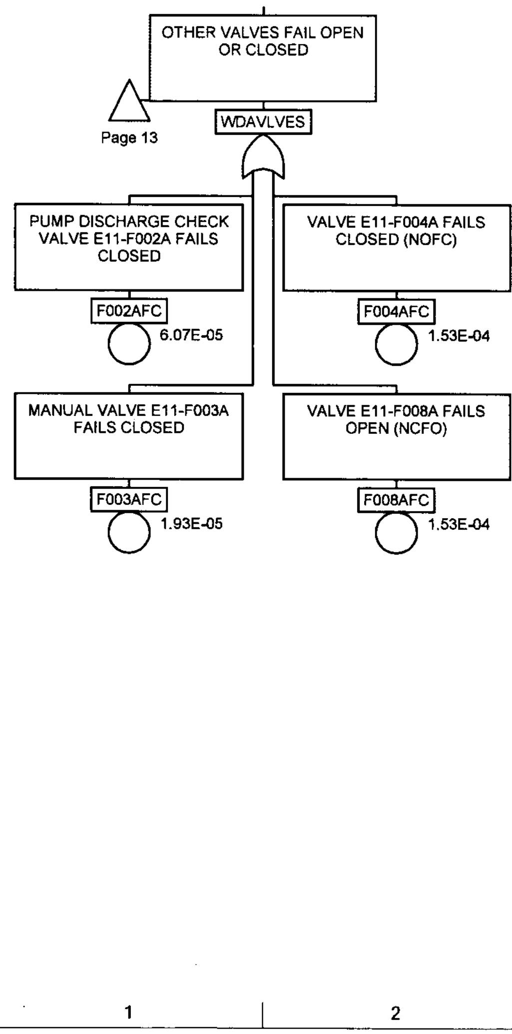

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

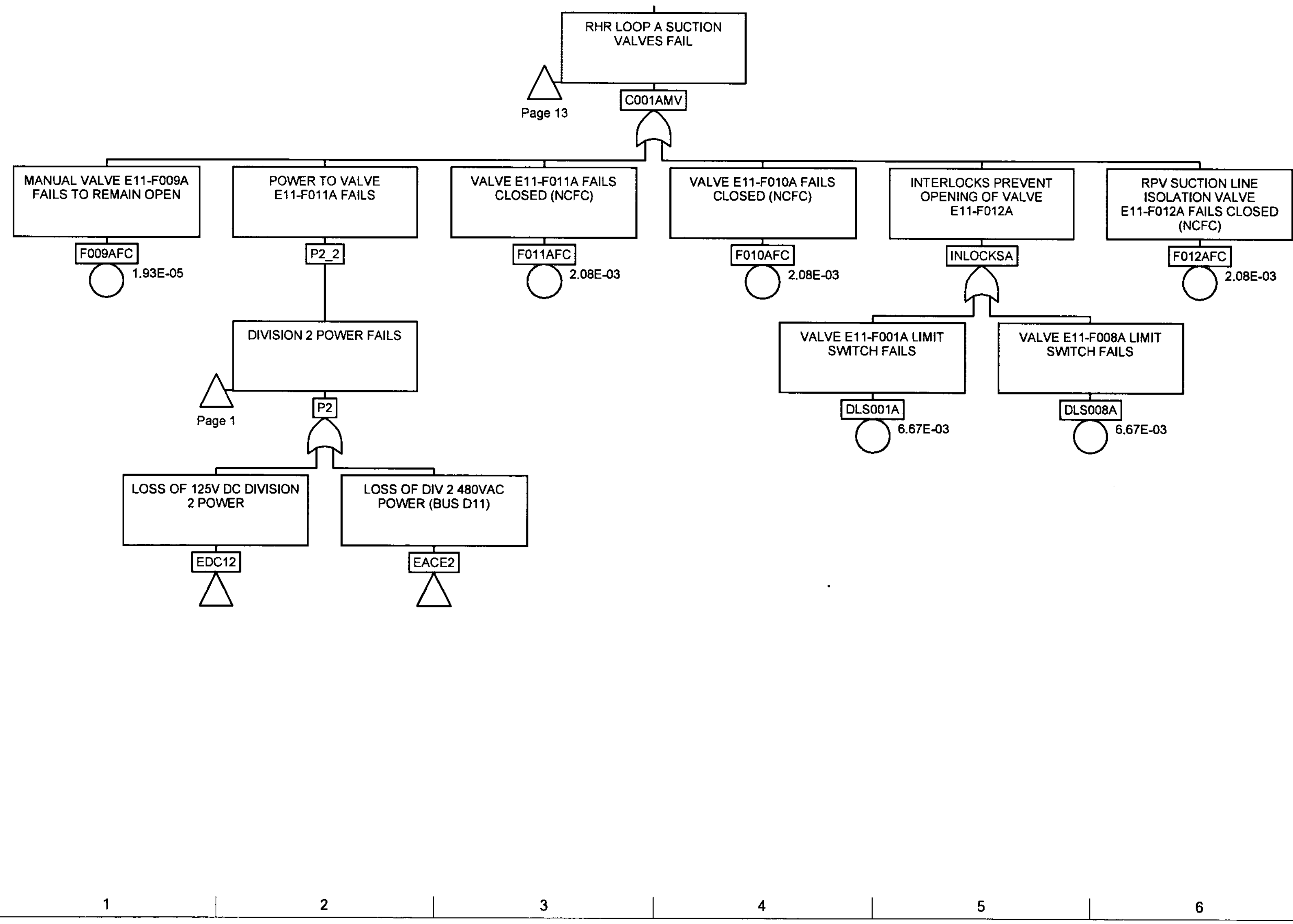

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

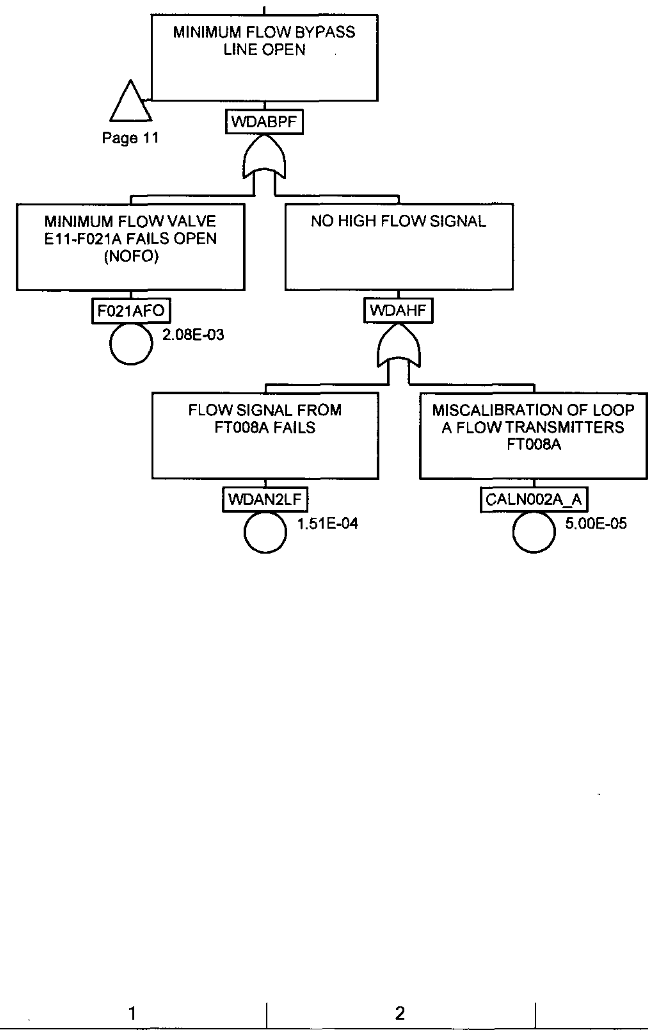

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

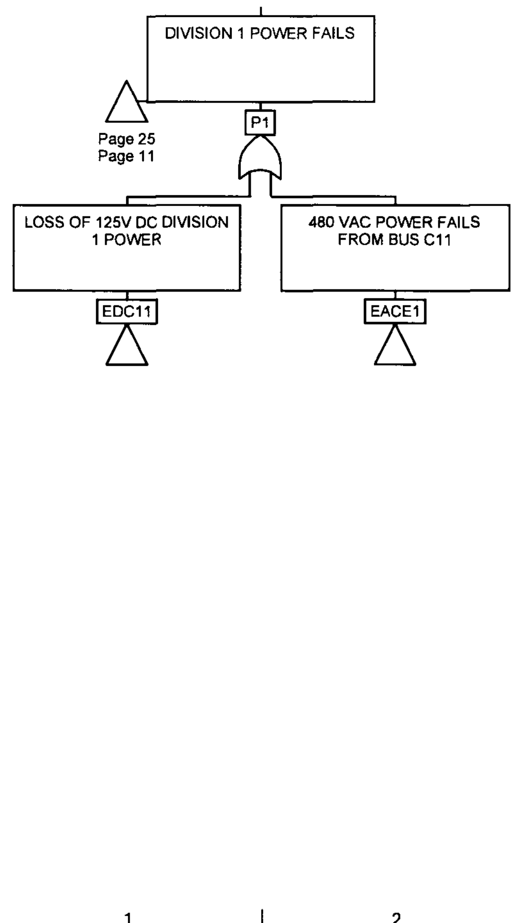

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

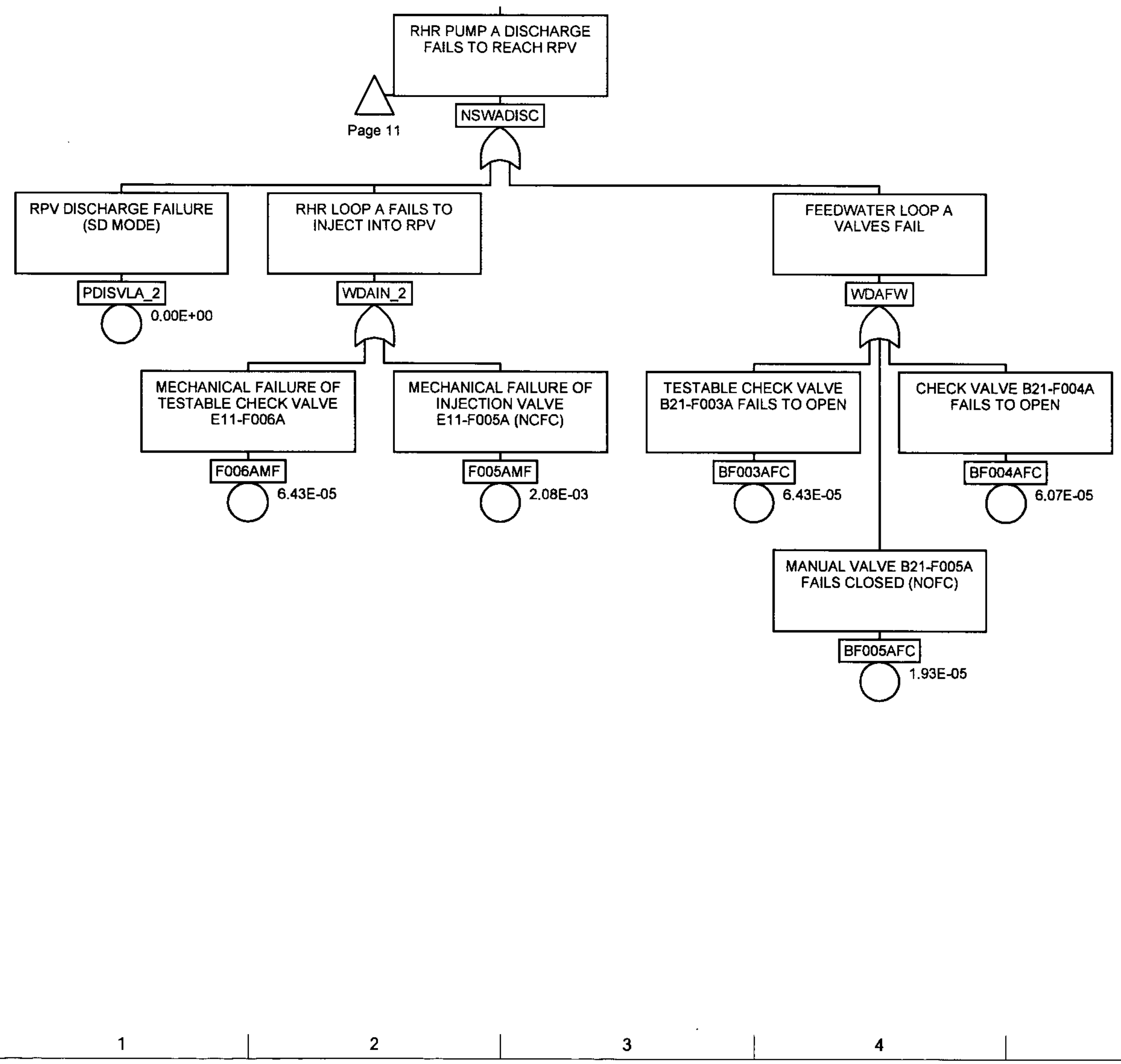

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

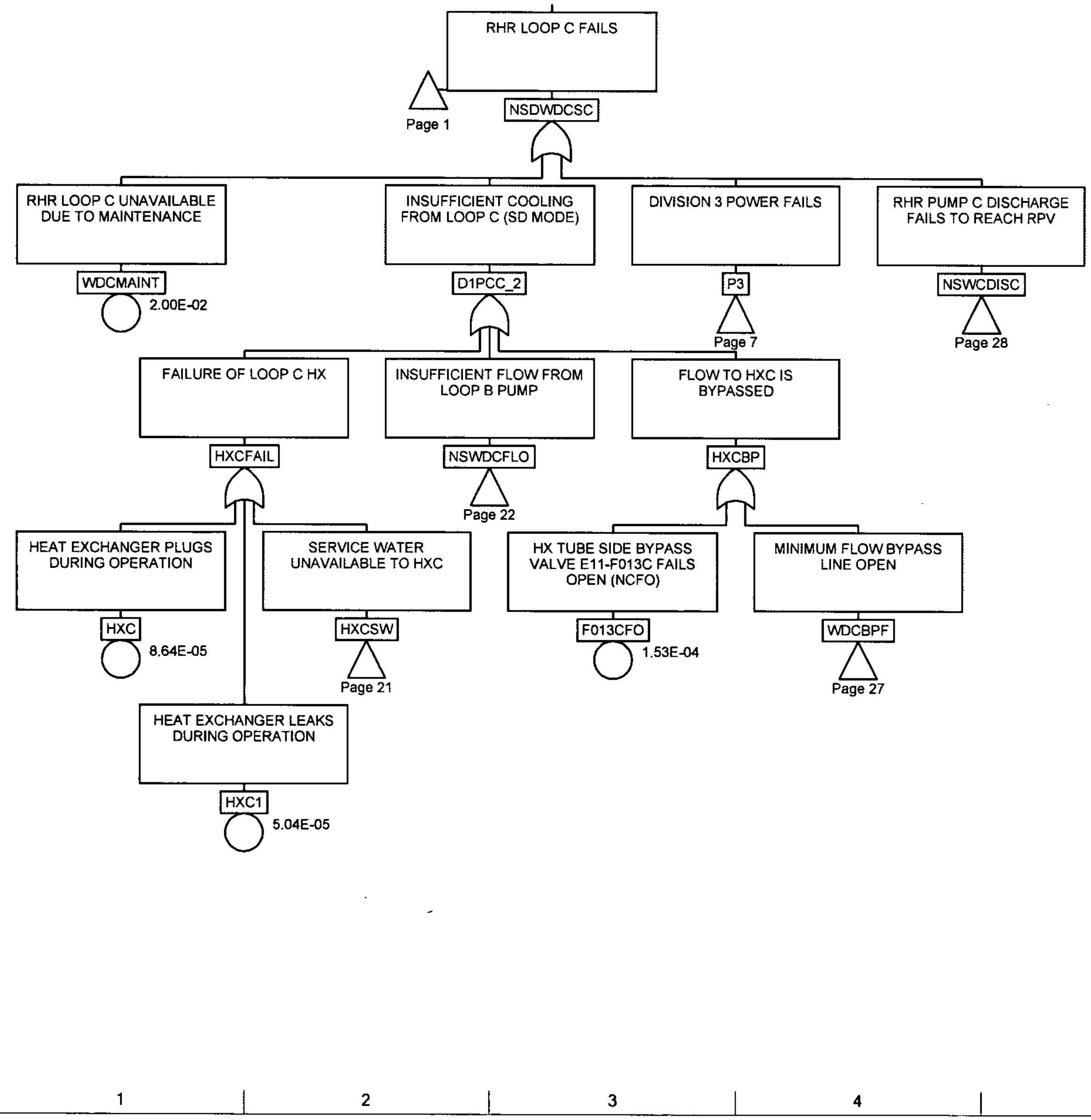

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

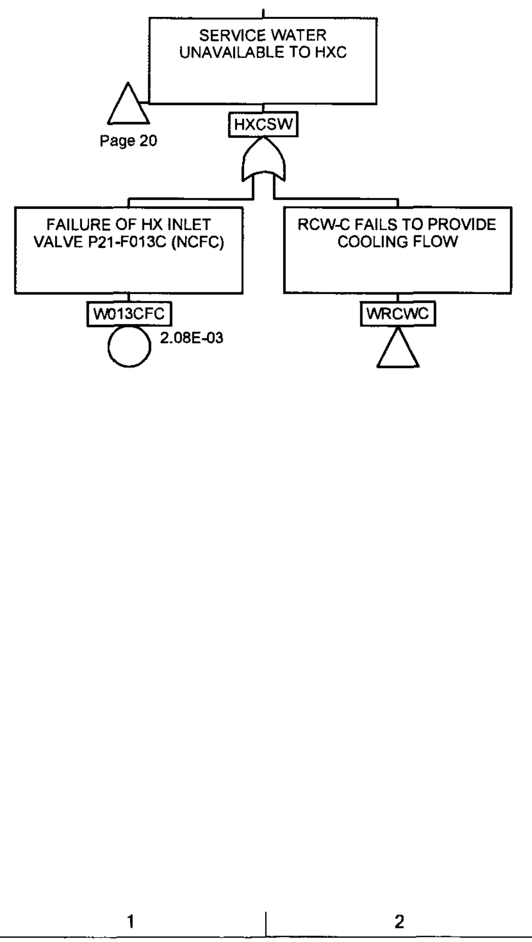

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

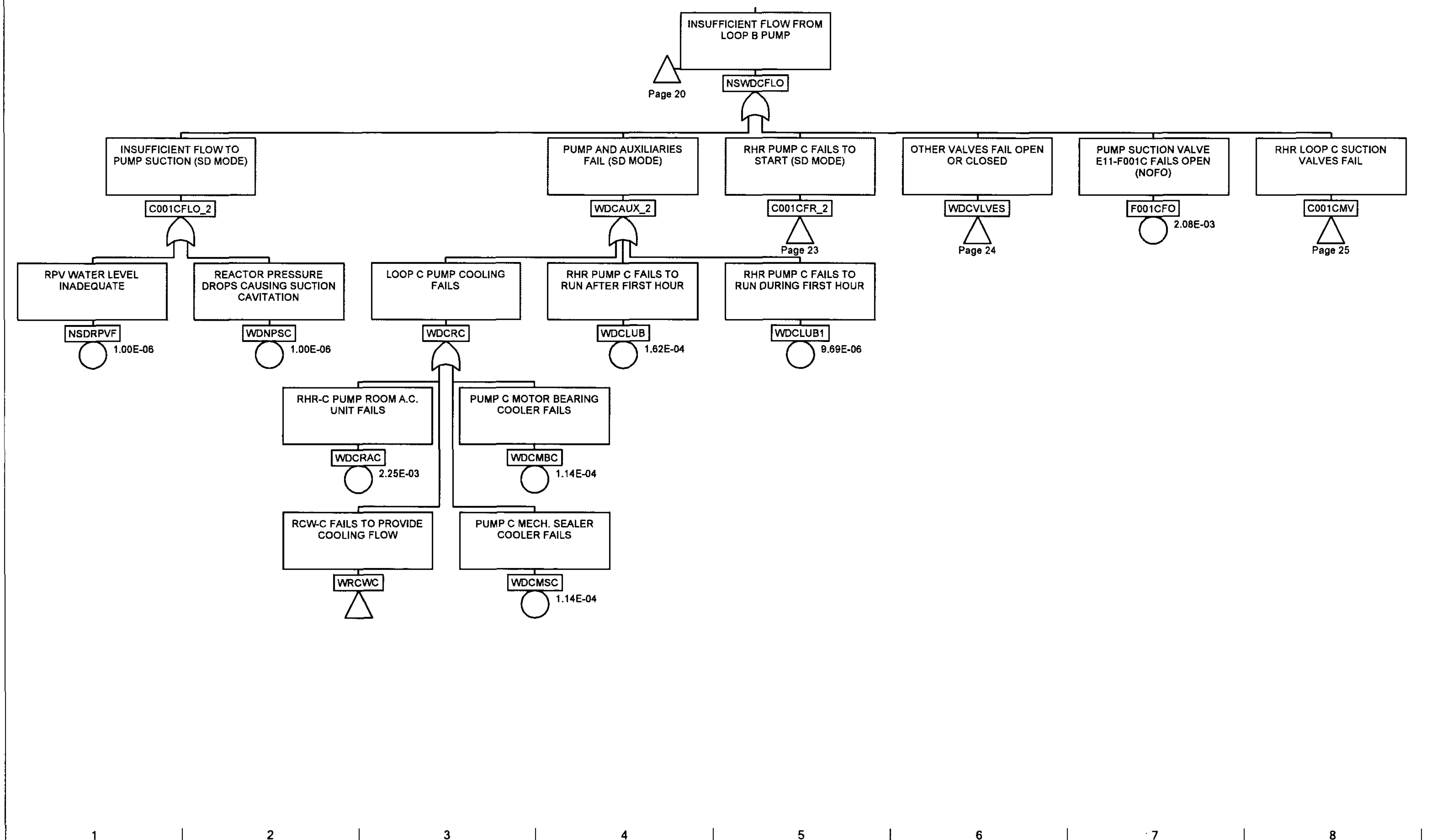

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

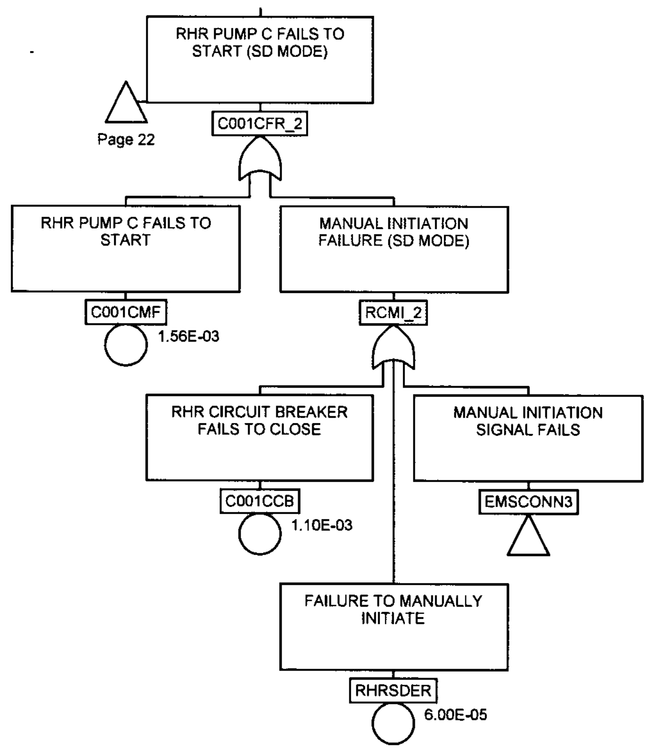

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

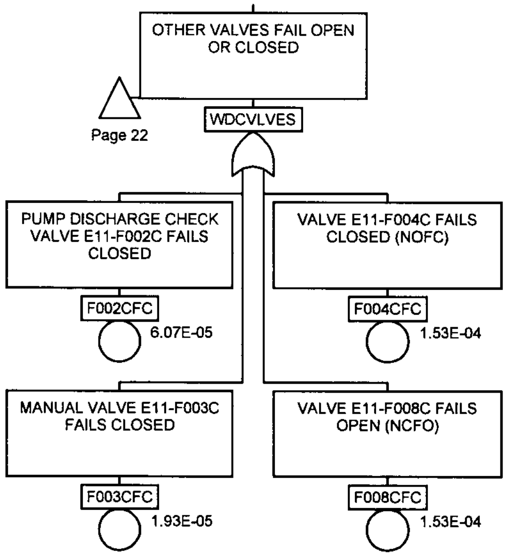

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

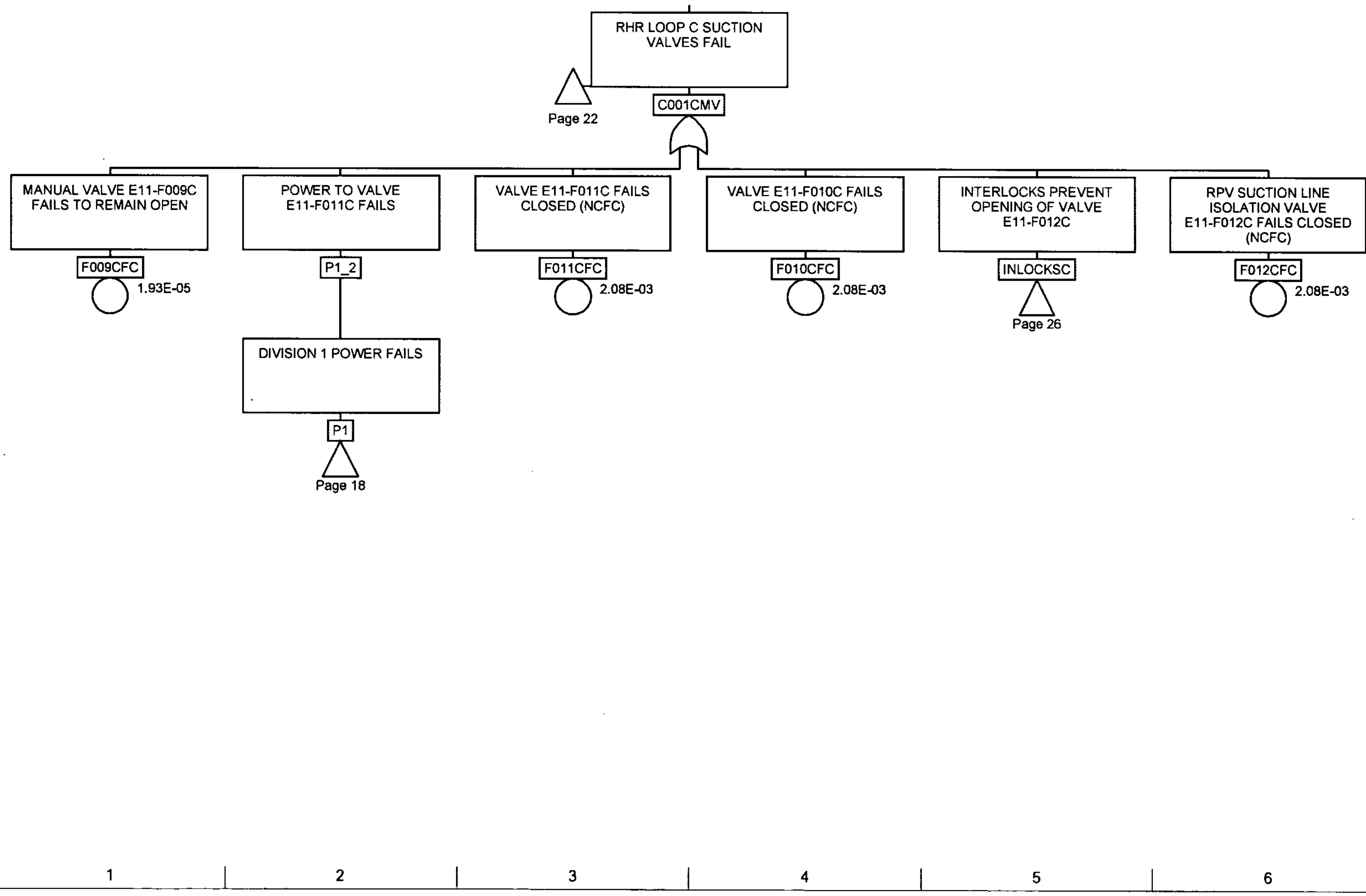

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

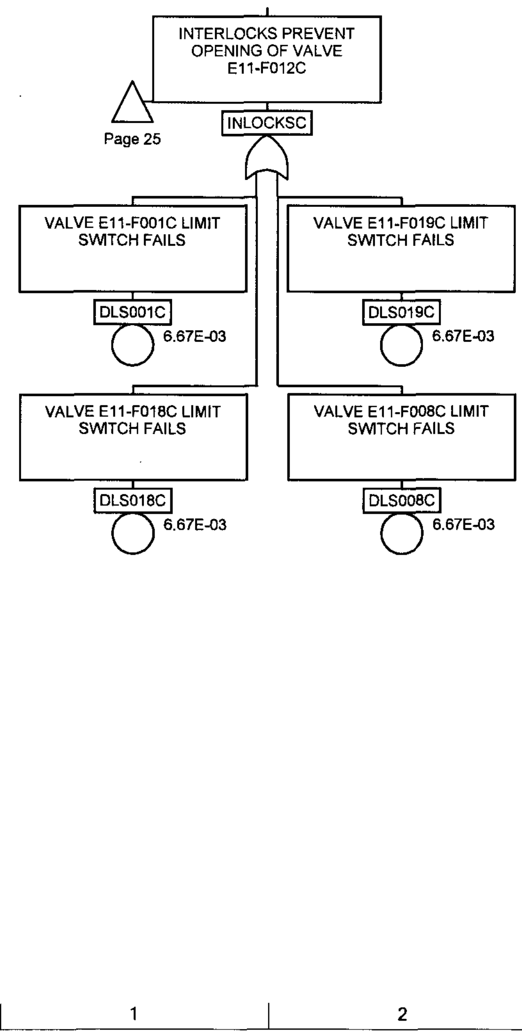

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

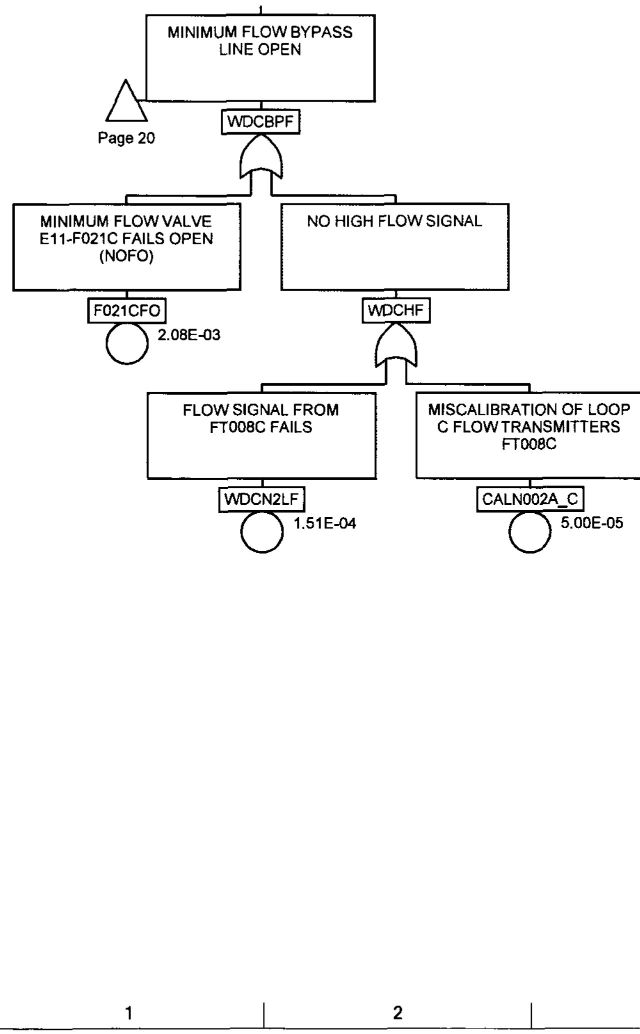

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

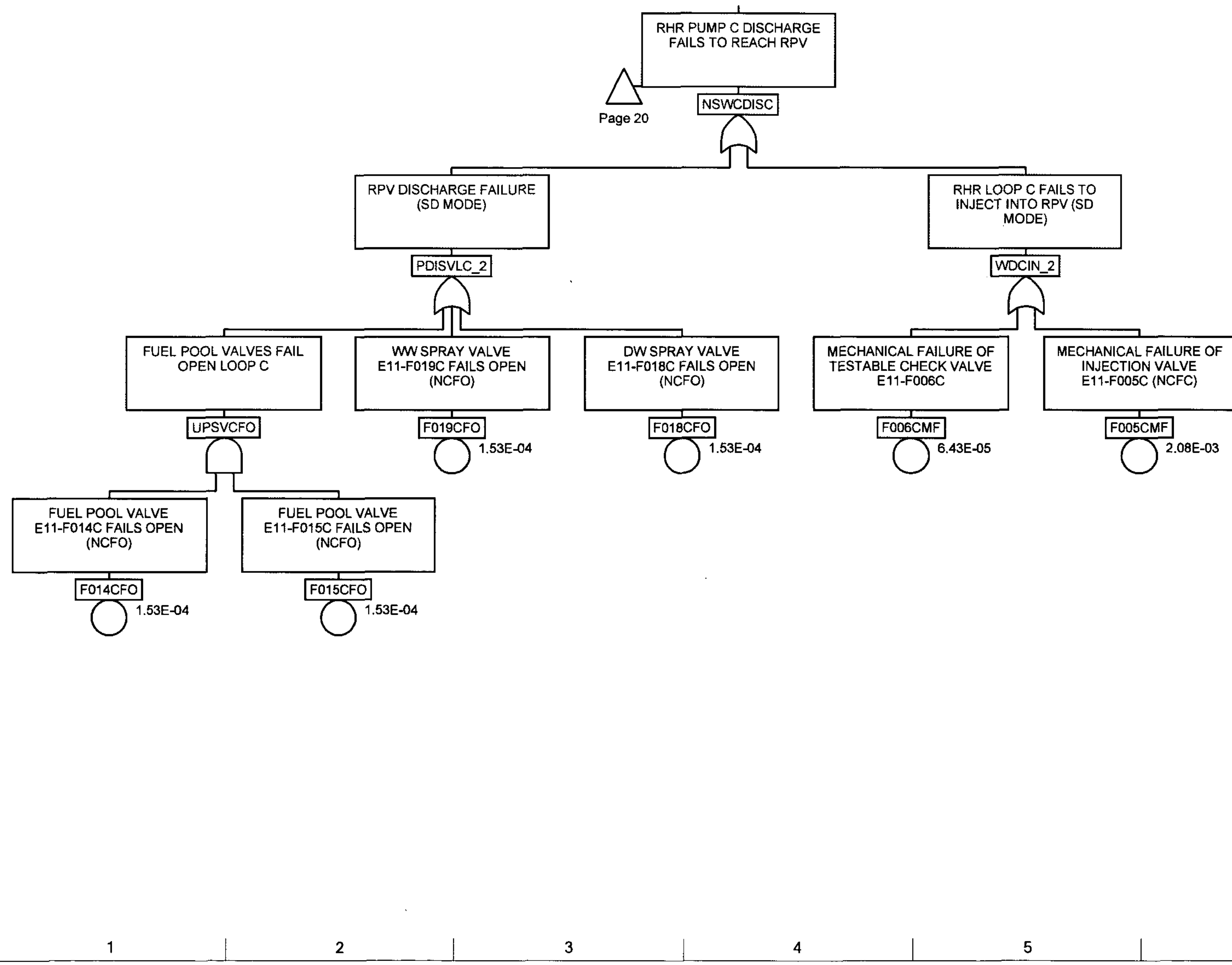

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

Name	Page	Zone	Name	Page	Zone
BF003AFC	19	4	EMSCONN3	23	3
BF004AFC	19	5	F001AFO	13	7
BF005AFC	19	4	F001BFO	3	7
C001ACB	14	2	F001CFO	22	7
C001AFLO_2	13	2	F002AFC	15	1
C001AFR_2	13	5	F002BFC	5	1
C001AFR_2	14	2	F002CFC	24	1
C001AMF	14	1	F003AFC	15	1
C001AMV	13	8	F003BFC	5	1
C001AMV	16	4	F003CFC	24	1
C001BCB	4	2	F004AFC	15	2
C001BFLO_2	3	2	F004BFC	5	2
C001BFR_2	3	5	F004CFC	24	2
C001BFR_2	4	2	F005AMF	19	3
C001BMF	4	1	F005BMF	10	6
C001BMV	3	8	F005CMF	28	6
C001BMV	6	4	F006AMF	19	2
C001CCB	23	2	F006BMF	10	5
C001CFLO_2	22	2	F006CMF	28	5
C001CFR_2	22	5	F008AFC	15	2
C001CFR_2	23	2	F008BFC	5	2
C001CMF	23	1	F008CFC	24	2
C001CMV	22	8	F009AFC	16	1
C001CMV	25	4	F009BFC	6	1
CALN002A_A	17	3	F009CFC	25	1
CALN002A_B	9	3	F010AFC	16	4
CALN002A_C	27	3	F010BFC	6	4
D1PCA_2	11	2	F010CFC	25	4
D1PCB_2	1	2	F011AFC	16	3
D1PCC_2	20	2	F011BFC	6	3
DLS001A	16	5	F011CFC	25	3
DLS001B	8	1	F012AFC	16	6
DLS001C	26	1	F012BFC	6	6
DLS008A	16	6	F012CFC	25	6
DLS008B	8	2	F013AFO	11	3
DLS008C	26	2	F013BFO	9	1
DLS018B	8	1	F013CFO	20	3
DLS018C	26	1	F014BFO	10	1
DLS019B	8	2	F014CFO	28	1
DLS019C	26	2	F015BFO	10	2
EACE1	18	2	F015CFO	28	2
EACE2	16	3	F018BFO	10	4
EACE3	7	2	F018CFO	28	4
EDC11	18	1	F019BFO	10	3
EDC12	16	2	F019CFO	28	3
EDC13	7	1	F021AFO	17	1
EMSCONN1	14	3	F021BFO	9	2
EMSCONN2	4	3	F021CFO	27	1

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

Name	Page	Zone	Name	Page	Zone
HXA	11	1	P2	1	3
HXA1	11	2	P2	16	2
HXABP	11	4	P2_2	16	2
HXAFAIL	11	2	P3	6	2
HXASW	11	2	P3	7	2
HXASW	12	2	P3	20	3
HXB	2	1	P3_2	6	2
HXB1	2	2	PDISVLA_2	19	1
HXBBP	1	3	PDISVLB_2	10	2
HXBBP	9	2	PDISVLC_2	28	2
HXBFAIL	1	2	RAMI_2	14	2
HXBFAIL	2	2	RBMI_2	4	2
HXBSW	2	3	RCMI_2	23	2
HXC	20	1	RHRSDER	4	2
HXC1	20	2	RHRSDER	14	2
HXCBP	20	4	RHRSDER	23	2
HXCFAIL	20	2	RSHTDABC	1	3
HXCSW	20	2	UPSVBFO	10	2
HXCSW	21	2	UPSVCFO	28	2
INLOCKSA	16	5	W013AFC	12	1
INLOCKSB	6	5	W013BFC	2	3
INLOCKSB	8	2	W013CFC	21	1
INLOCKSC	25	5	WDAAUX_2	13	4
INLOCKSC	26	2	WDABPF	11	4
NSDRPVF	3	1	WDABPF	17	2
NSDRPVF	13	1	WDAFW	19	4
NSDRPVF	22	1	WDAHF	17	2
NSDWDCSA	1	4	WDAIN_2	19	2
NSDWDCSA	11	3	WDALUB	13	4
NSDWDCSB	1	3	WDALUB1	13	5
NSDWDCSC	1	5	WDAMAINT	11	1
NSDWDCSC	20	3	WDAMBC	13	4
NSWADISC	11	4	WDAMSC	13	4
NSWADISC	19	2	WDAN2LF	17	2
NSWBDISC	1	4	WDARAC	13	3
NSWBDISC	10	4	WDARC	13	3
NSWCDISC	20	4	WDAVLVES	13	6
NSWCDISC	28	4	WDAVLVES	15	2
NSWDAFLO	11	3	WDBAUX_2	3	4
NSWDAFLO	13	5	WDBBPF	9	2
NSWDBFLO	1	2	WDBHF	9	3
NSWDBFLO	3	5	WDBIN_2	10	5
NSWDCFLO	20	3	WDBLUB	3	4
NSWDCFLO	22	5	WDBLUB1	3	5
P1	11	3	WDBMAINT	1	1
P1	18	2	WDBMBC	3	4
P1	25	2	WDBMSC	3	4
P1_2	25	2	WDBN2LF	9	2

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

Name	Page	Zone	Name	Page	Zone
WDBRAC	3	3			
WDBRC	3	3			
WDBVLVES	3	6			
WDBVLVES	5	2			
WDCAUX_2	22	4			
WDCBPF	20	4			
WDCBPF	27	2			
WDCHF	27	2			
WDCIN_2	28	5			
WDCLUB	22	4			
WDCLUB1	22	5			
WDCMAINT	20	1			
WDCMBC	22	4			
WDCMSC	22	4			
WDCN2LF	27	2			
WDCRAC	22	3			
WDCRC	22	3			
WDCVLVES	22	6			
WDCVLVES	24	2			
WDNPSC	3	2			
WDNPSC	13	2			
WDNPSC	22	2			
WRCWA	12	2			
WRCWA	13	3			
WRCWB	2	4			
WRCWB	3	3			
WRCWC	21	2			
WRCWC	22	3			

Figure 19D.6-7 RHR Shutdown Cooling Mode Fault Tree

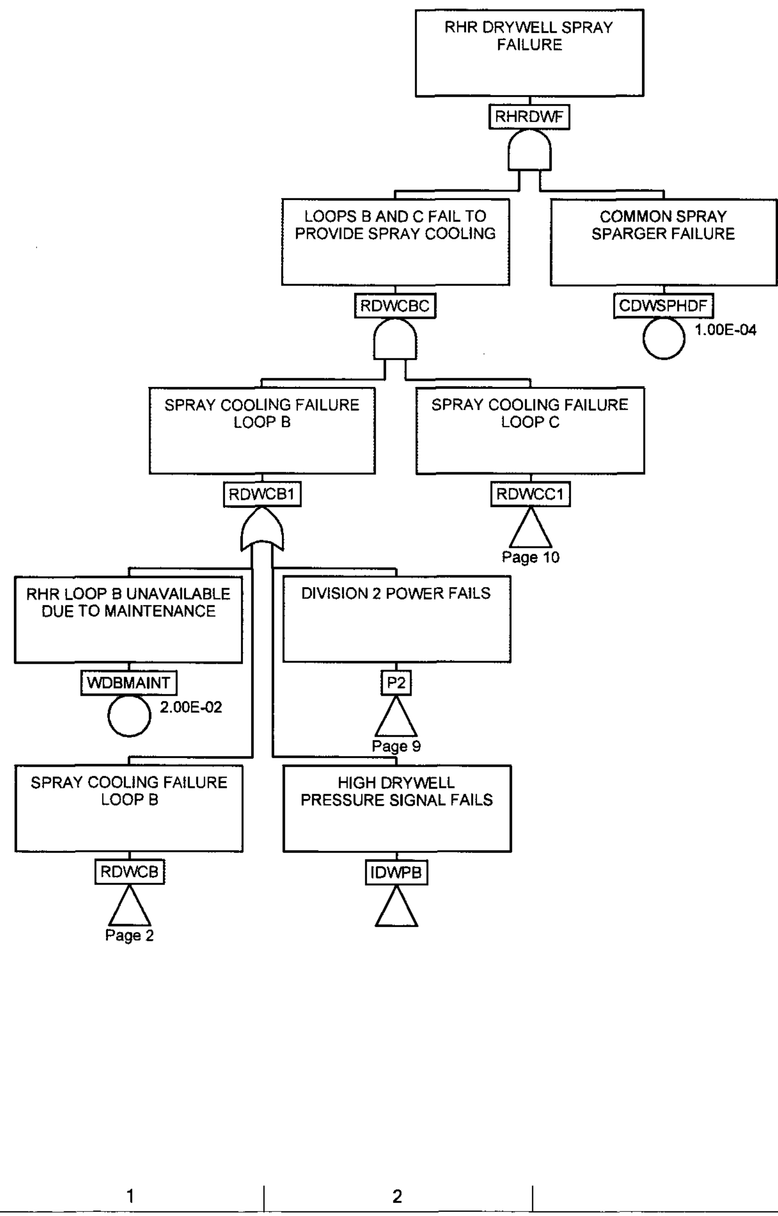

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

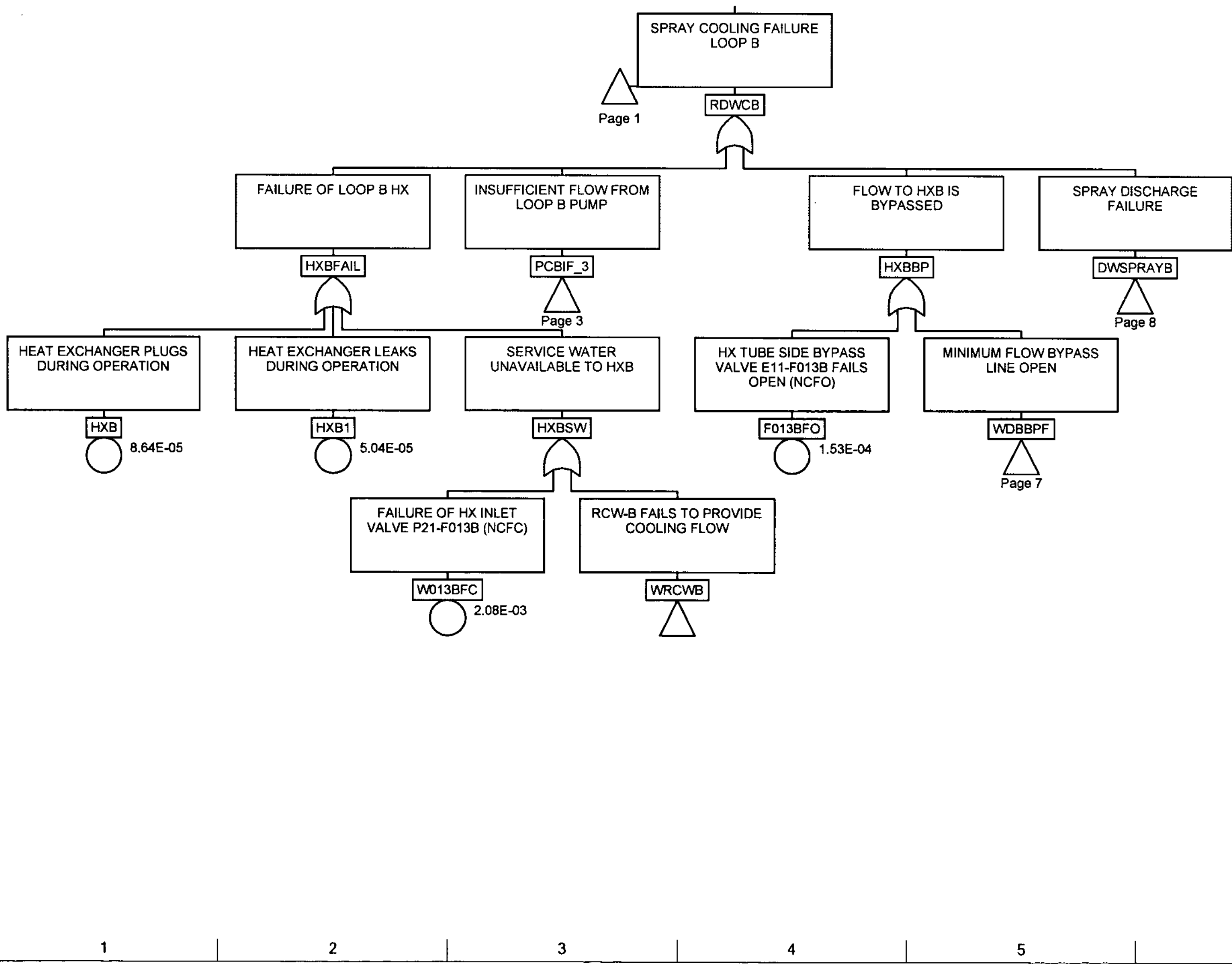

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

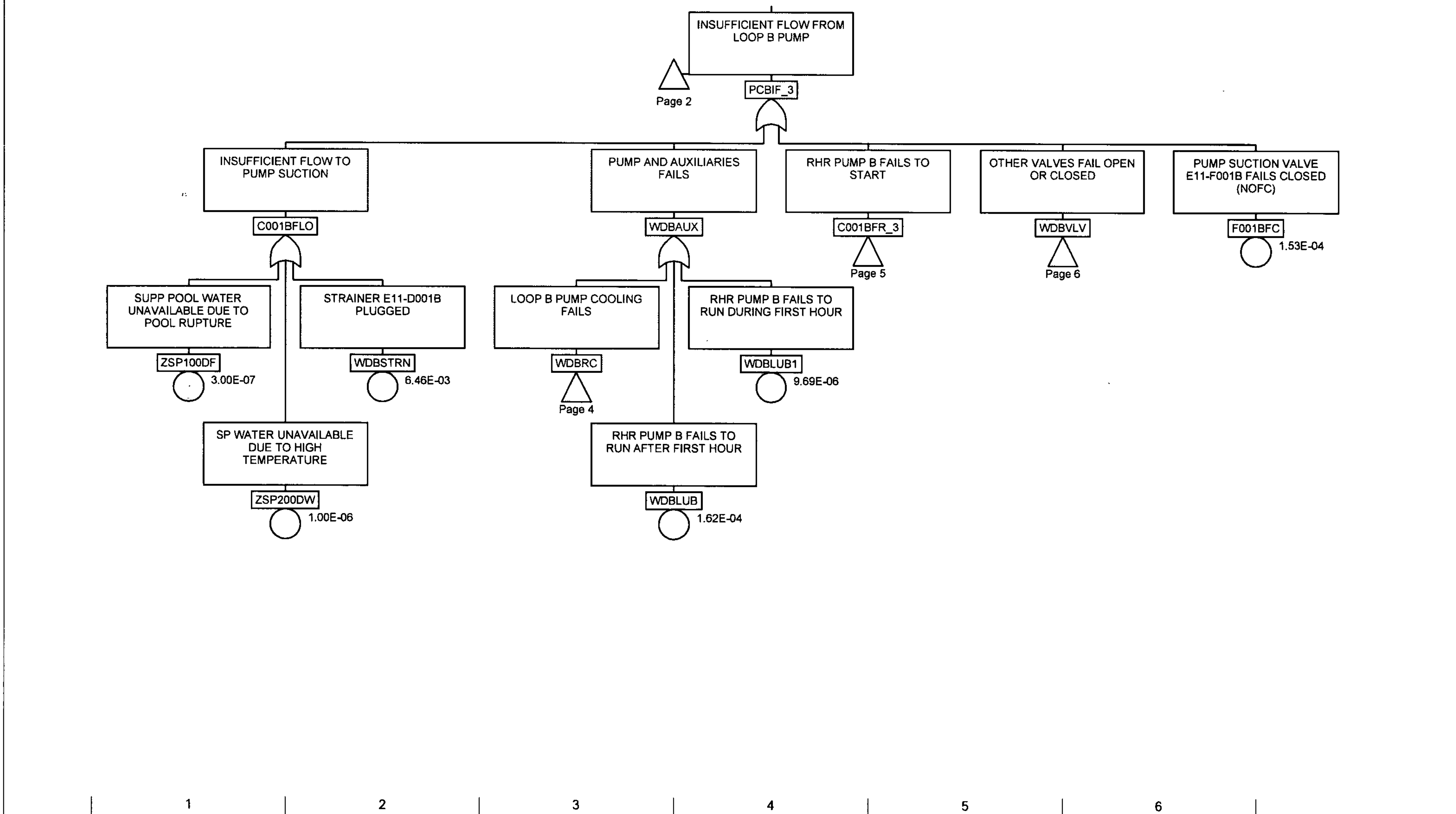

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

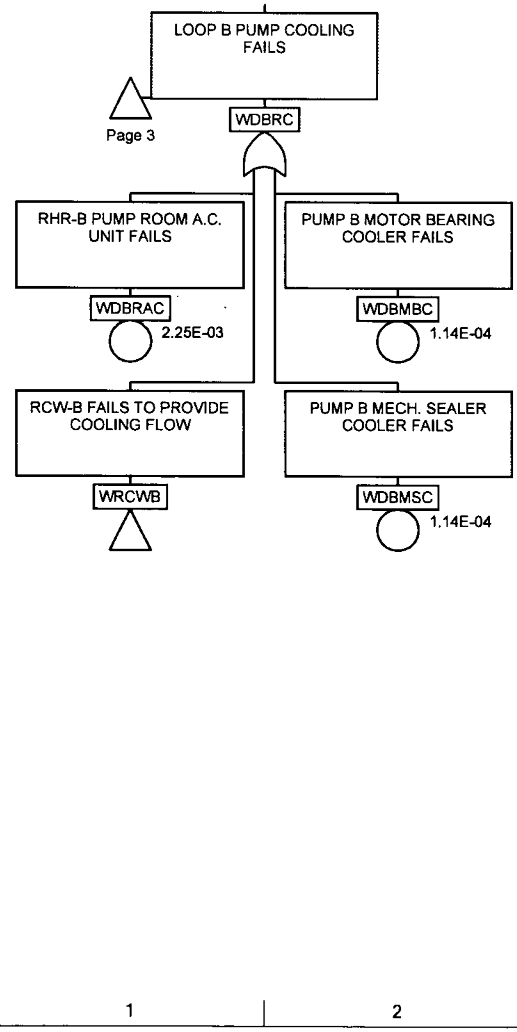

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

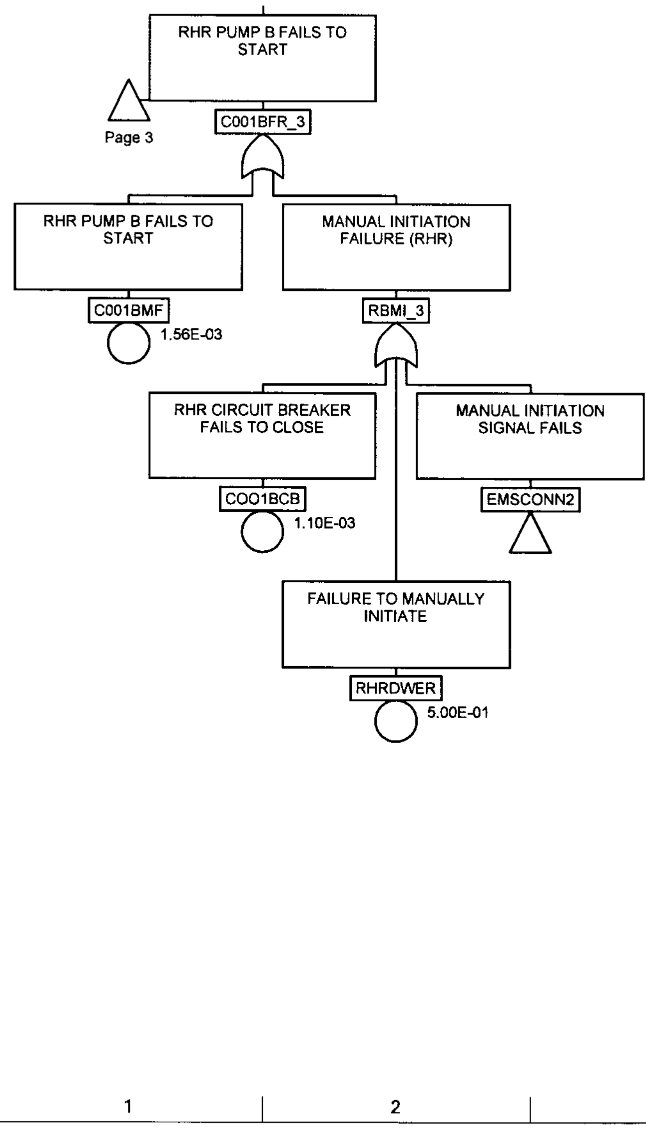

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

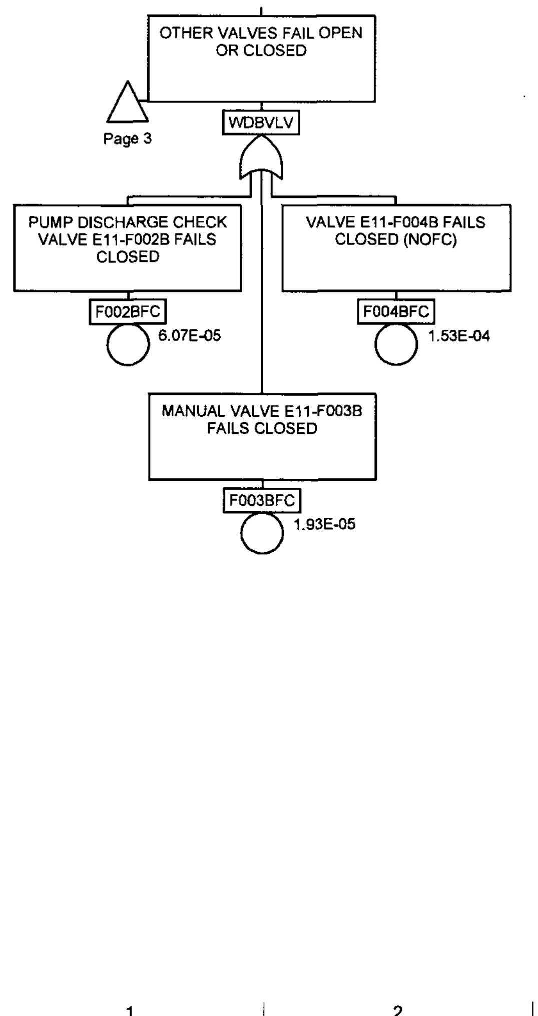

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

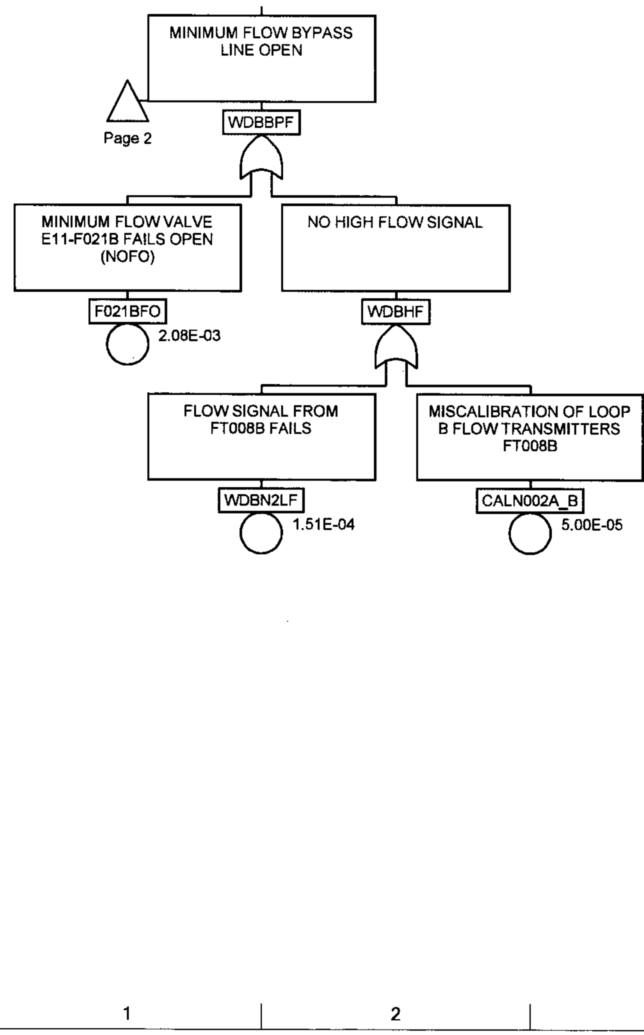

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

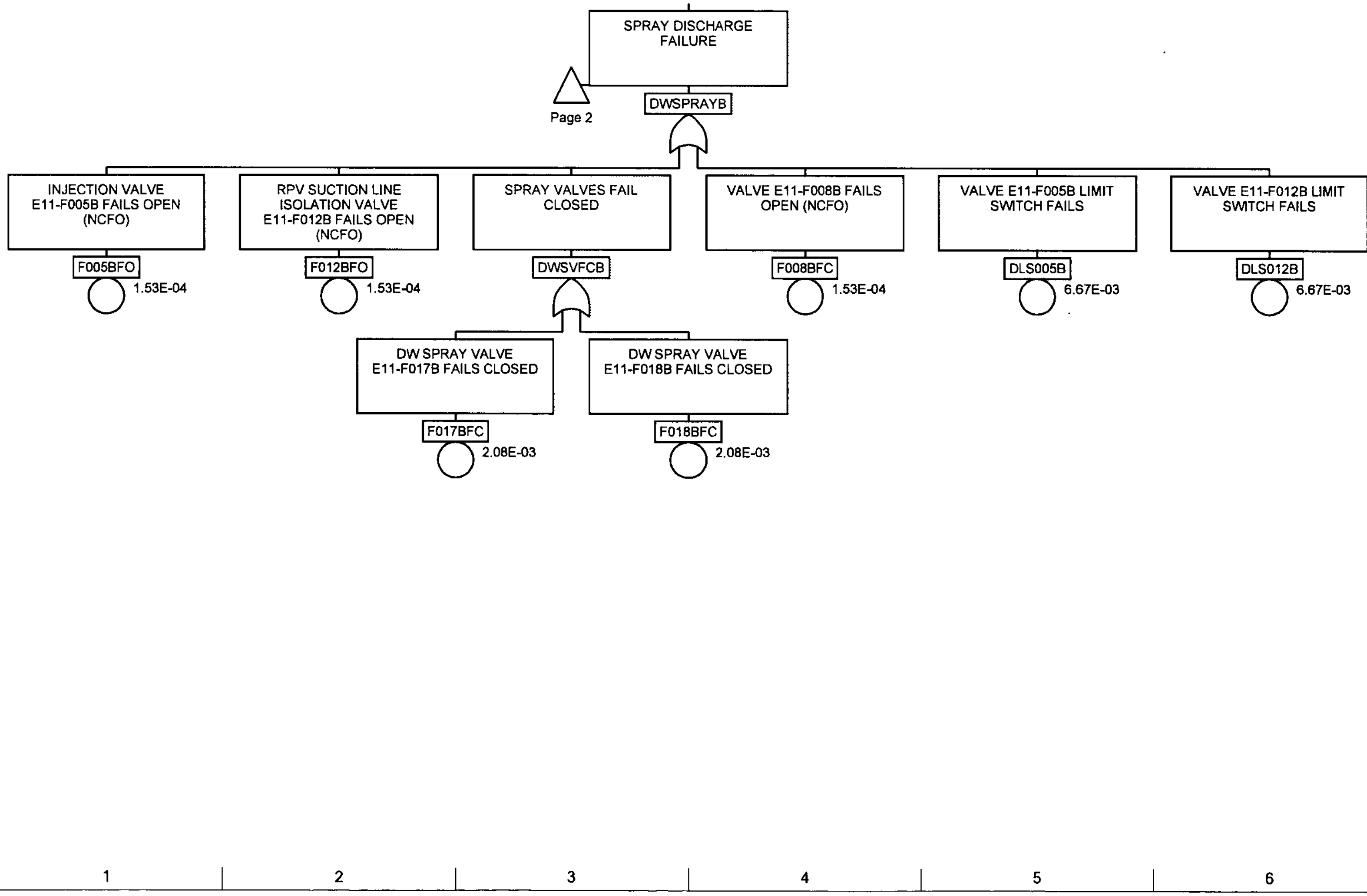

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

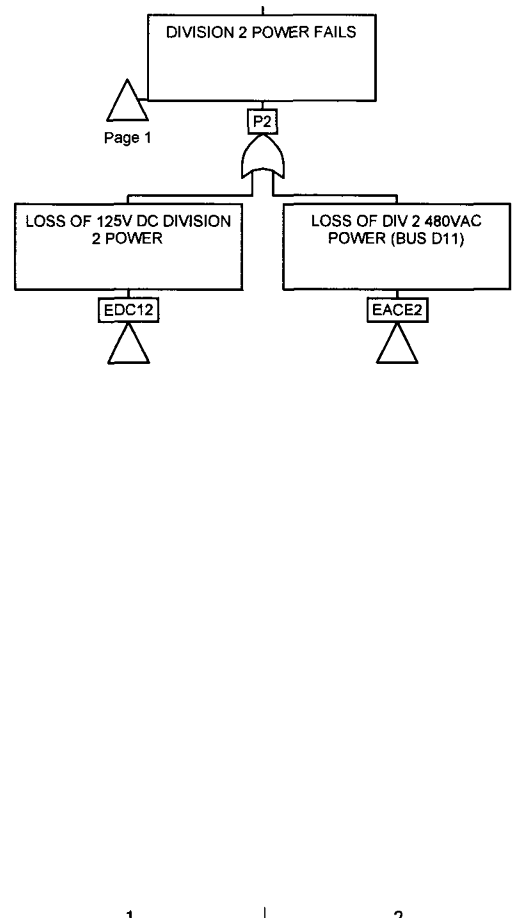

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

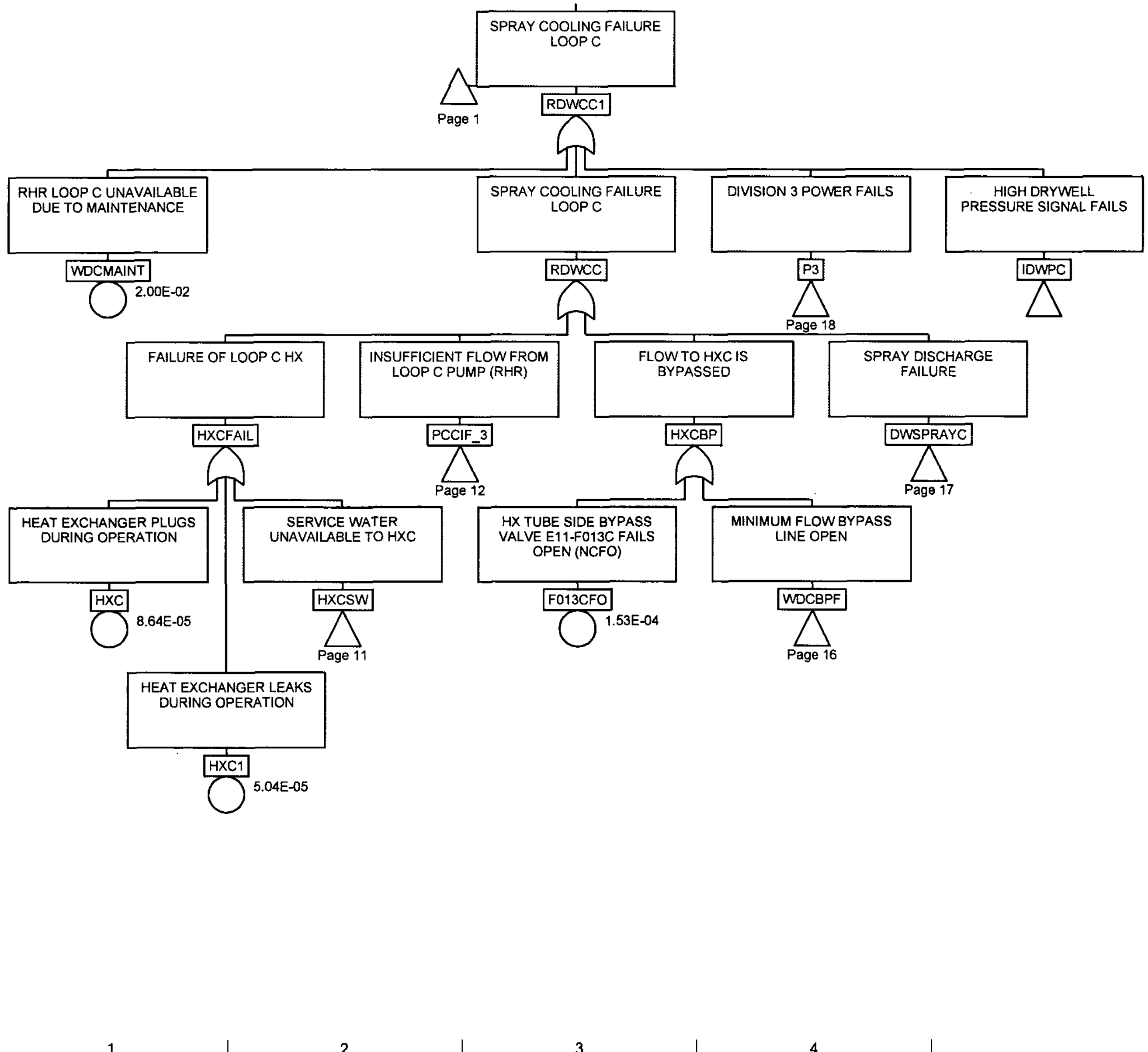

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

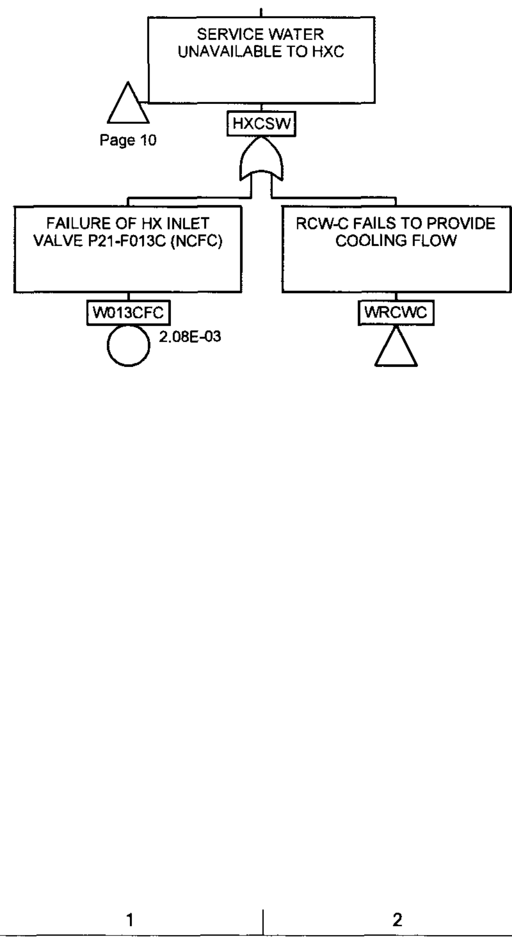

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

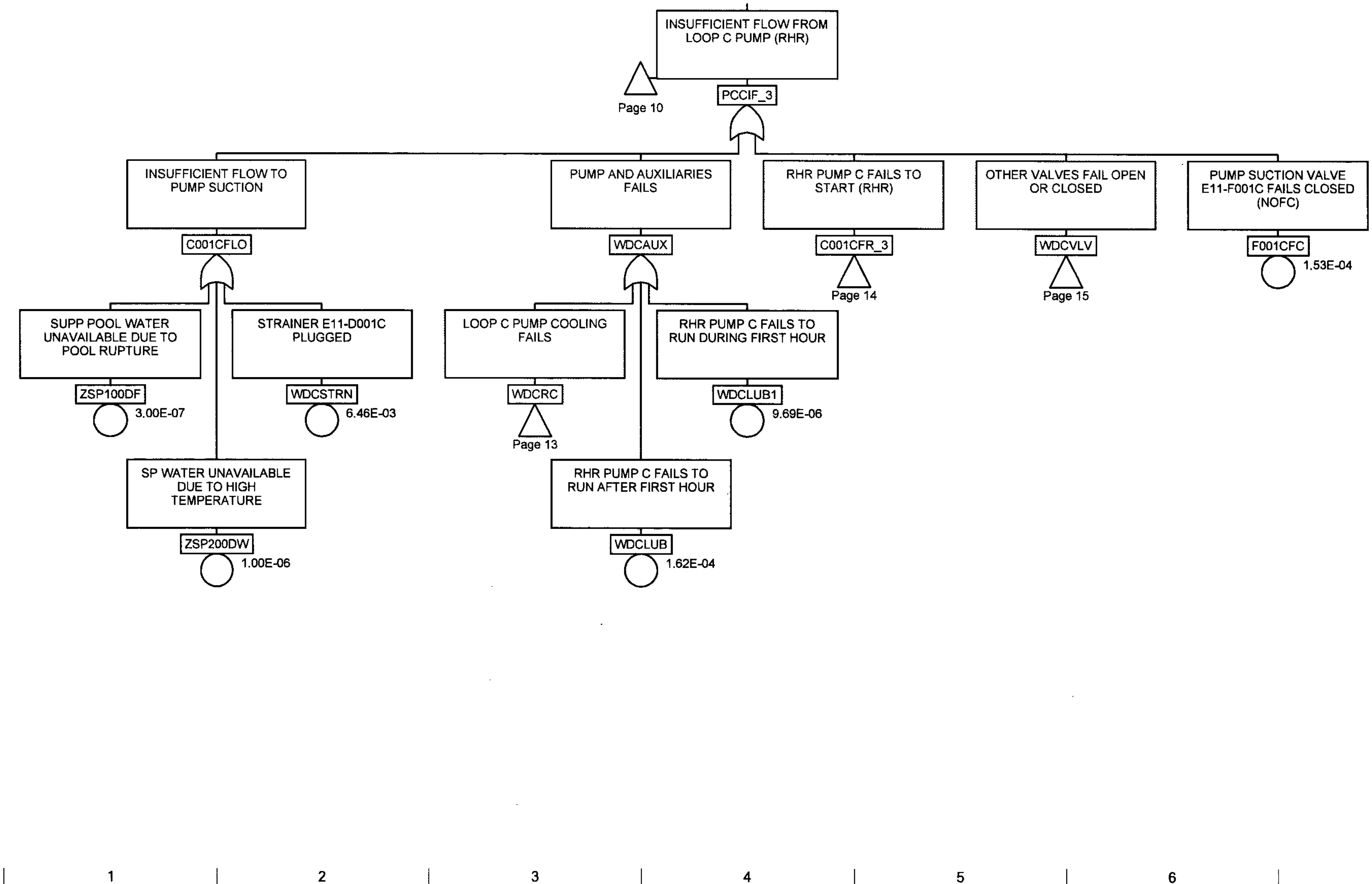

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

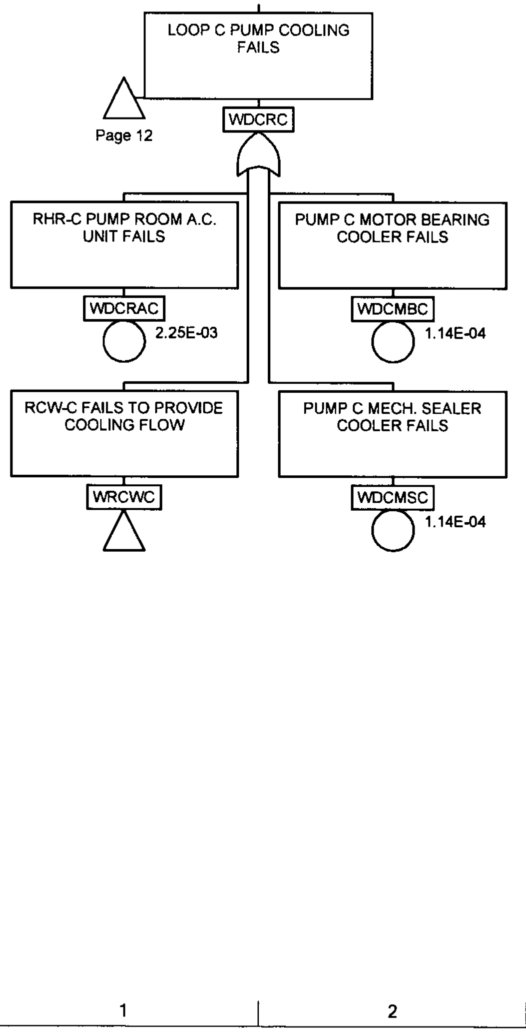

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

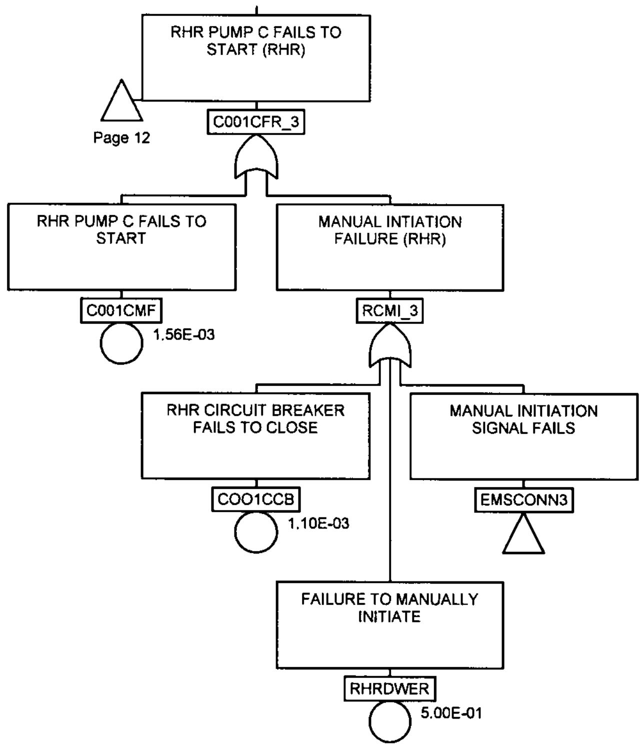

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

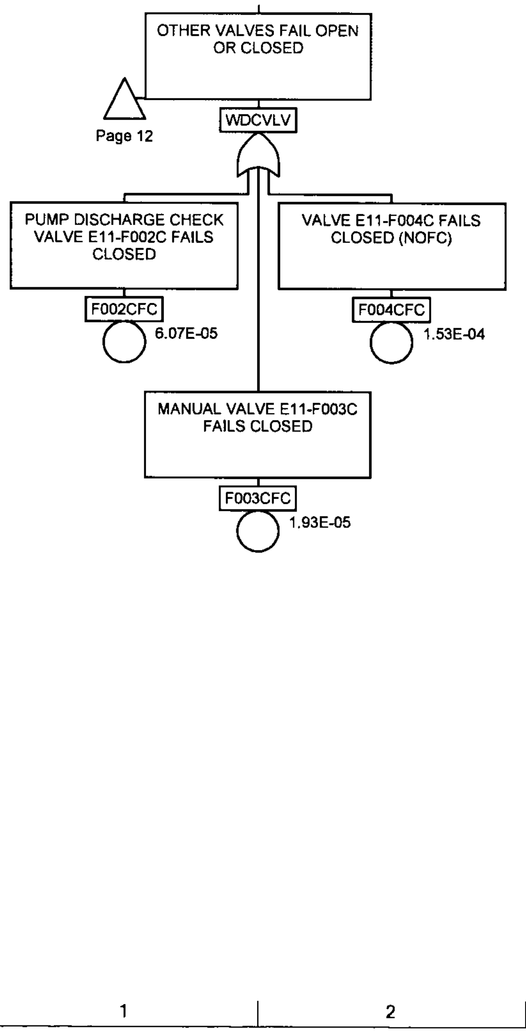

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

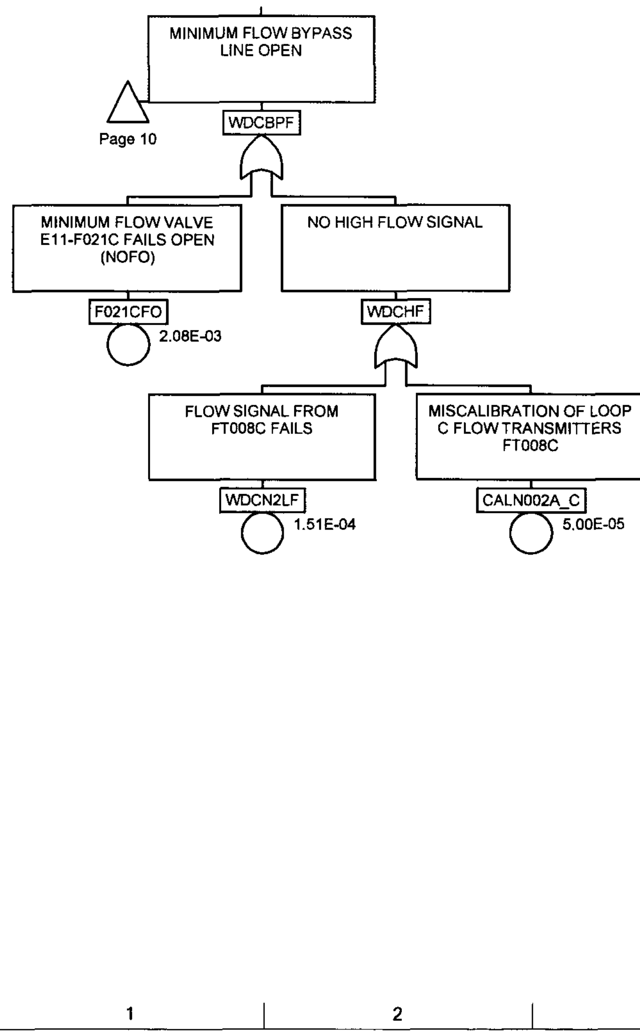

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

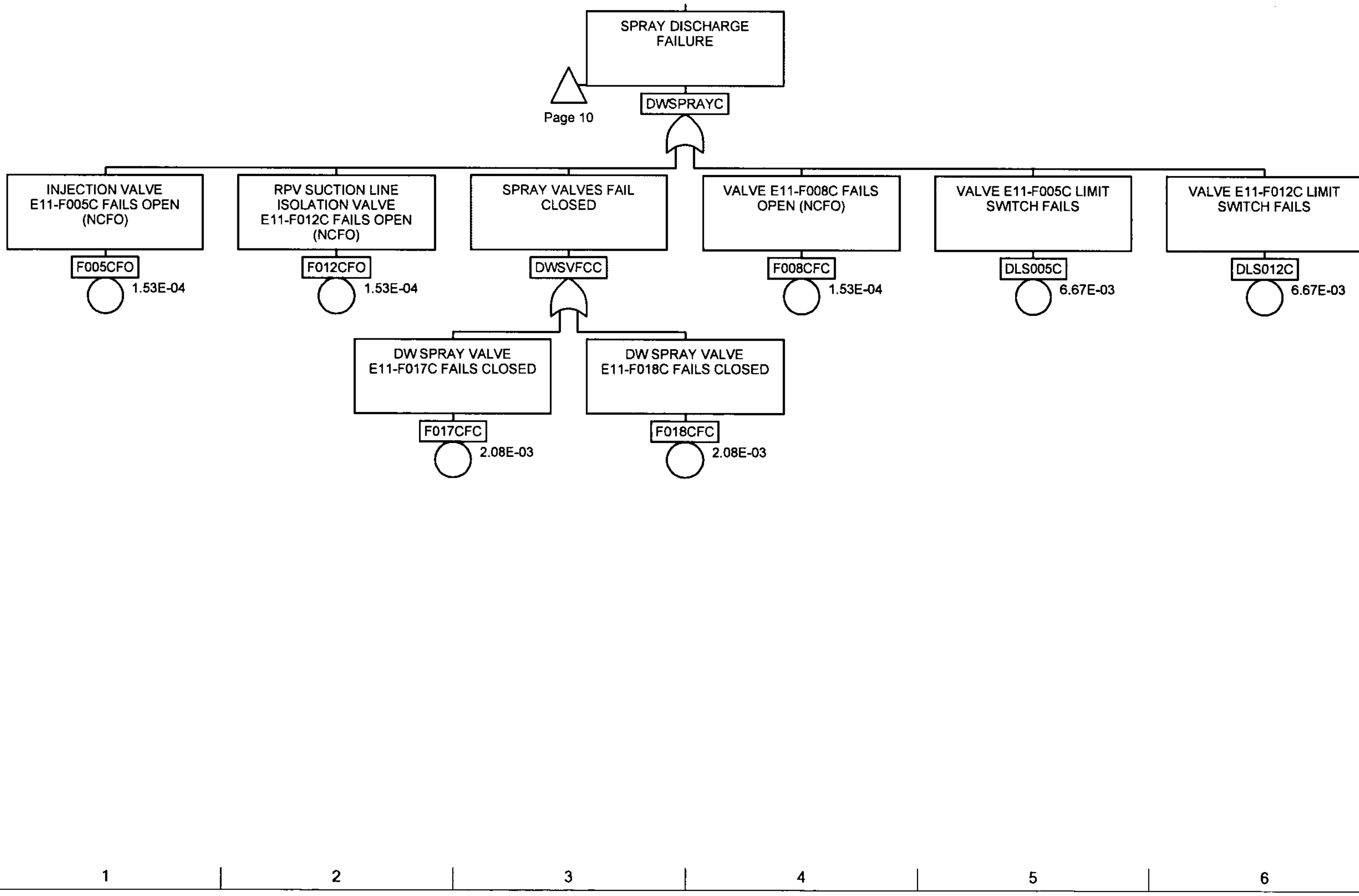

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

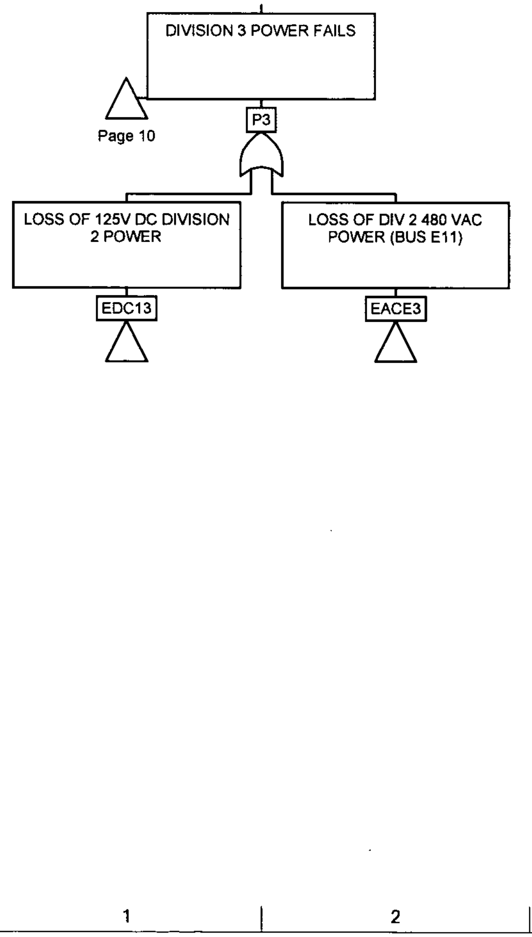

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

Name	Page	Zone	Name	Page	Zone
C001BFLO	3	2	F018CFC	17	4
C001BFR_3	3	5	F021BFO	7	1
C001BFR_3	5	2	F021CFO	16	1
C001BMF	5	1	HXB	2	1
C001CFLO	12	2	HXB1	2	2
C001CFR_3	12	5	HXBBP	2	5
C001CFR_3	14	2	HXBFAIL	2	2
C001CMF	14	1	HXBSW	2	3
CALN002A_B	7	3	HXC	10	1
CALN002A_C	16	3	HXC1	10	2
CDWSPHDF	1	3	HXCBP	10	4
COO1BCB	5	2	HXCFAIL	10	2
COO1CCB	14	2	HXCSW	10	2
DLS005B	8	5	HXCSW	11	2
DLS005C	17	5	IDWPB	1	2
DLS012B	8	6	IDWPC	10	5
DLS012C	17	6	P2	1	2
DWSPRAYB	2	6	P2	9	2
DWSPRAYB	8	4	P3	10	4
DWSPRAYC	10	5	P3	18	2
DWSPRAYC	17	4	PCBIF_3	2	3
DWSVFCB	8	3	PCBIF_3	3	4
DWSVFCC	17	3	PCCIF_3	10	3
EACE2	9	2	PCCIF_3	12	4
EACE3	18	2	RBMI_3	5	2
EDC12	9	1	RCMI_3	14	2
EDC13	18	1	RDWCB	1	1
EMSCONN2	5	3	RDWCB	2	4
EMSCONN3	14	3	RDWCB1	1	2
F001BFC	3	7	RDWCBC	1	2
F001CFC	12	7	RDWCC	10	3
F002BFC	6	1	RDWCC1	1	3
F002CFC	15	1	RDWCC1	10	3
F003BFC	6	2	RHRDWER	5	2
F003CFC	15	2	RHRDWER	14	2
F004BFC	6	2	RHRDWF	1	2
F004CFC	15	2	W013BFC	2	3
F005BFO	8	1	W013CFC	11	1
F005CFO	17	1	WDBAUX	3	4
F008BFC	8	4	WDBBPF	2	5
F008CFC	17	4	WDBBPF	7	2
F012BFO	8	2	WDBHF	7	2
F012CFO	17	2	WDBLUB	3	4
F013BFO	2	4	WDBLUB1	3	4
F013CFO	10	3	WDBMAINT	1	1
F017BFC	8	3	WDBMBC	4	2
F017CFC	17	3	WDBMSC	4	2
F018BFC	8	4	WDBN2LF	7	2

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

Name	Page	Zone	Name	Page	Zone
WDBRAC	4	1			
WDBRC	3	3			
WDBRC	4	2			
WDBSTRN	3	2			
WDBVLV	3	6			
WDBVLV	6	2			
WDCAUX	12	4			
WDCBPF	10	4			
WDCBPF	16	2			
WDCHF	16	2			
WDCLUB	12	4			
WDCLUB1	12	4			
WDCMAINT	10	1			
WDCMBC	13	2			
WDCMSC	13	2			
WDCN2LF	16	2			
WDCRAC	13	1			
WDCRC	12	3			
WDCRC	13	2			
WDCSTRN	12	2			
WDCVLV	12	6			
WDCVLV	15	2			
WRCWB	2	4			
WRCWB	4	1			
WRCWC	11	2			
WRCWC	13	1			
ZSP100DF	3	1			
ZSP100DF	12	1			
ZSP200DW	3	2			
ZSP200DW	12	2			

Figure 19D.6-8A RHR Drywell Spray System Fault Tree

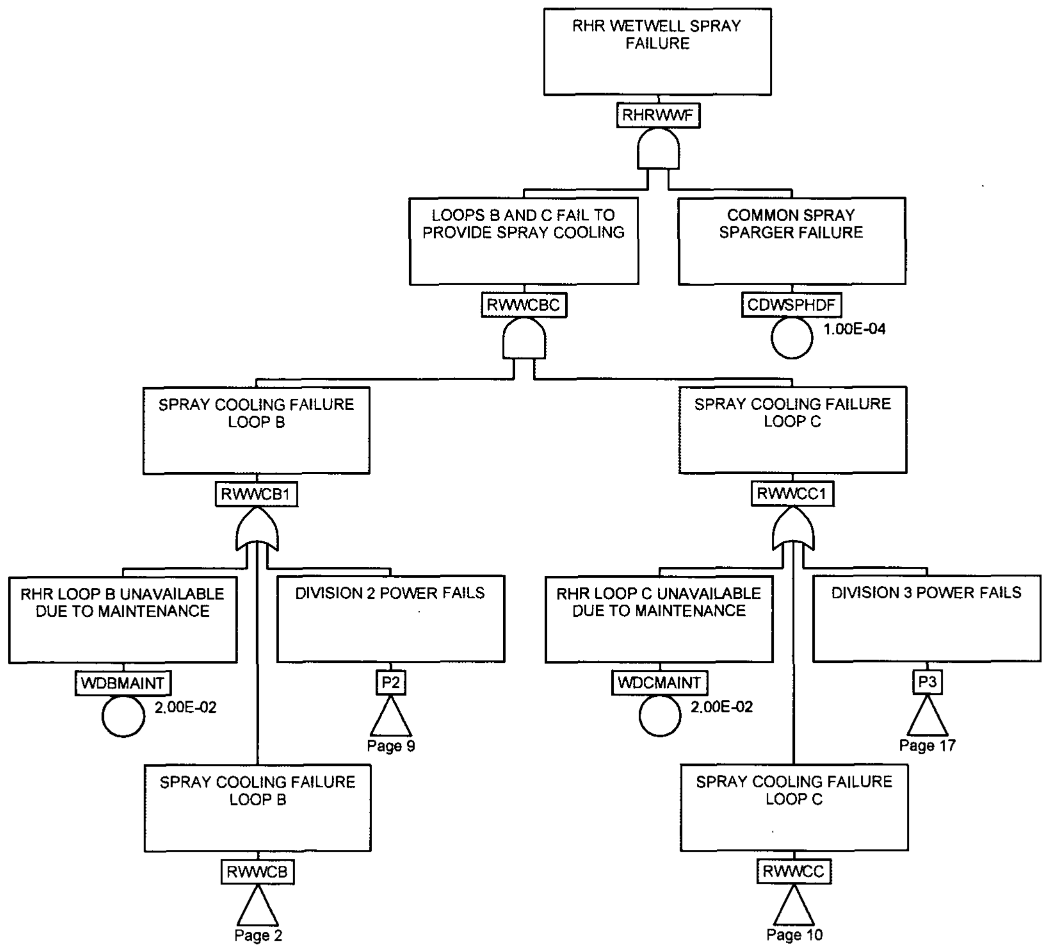

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

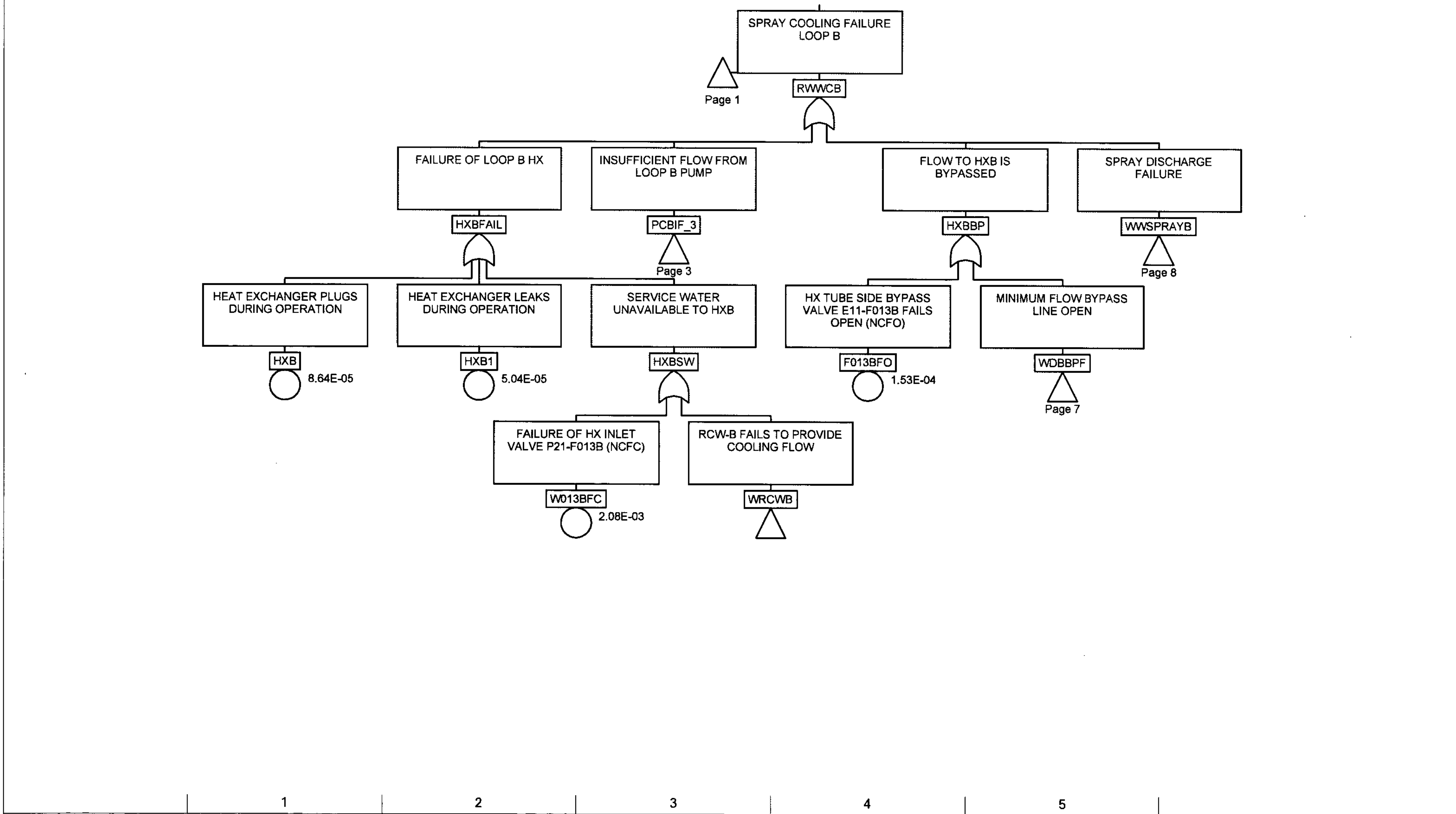

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

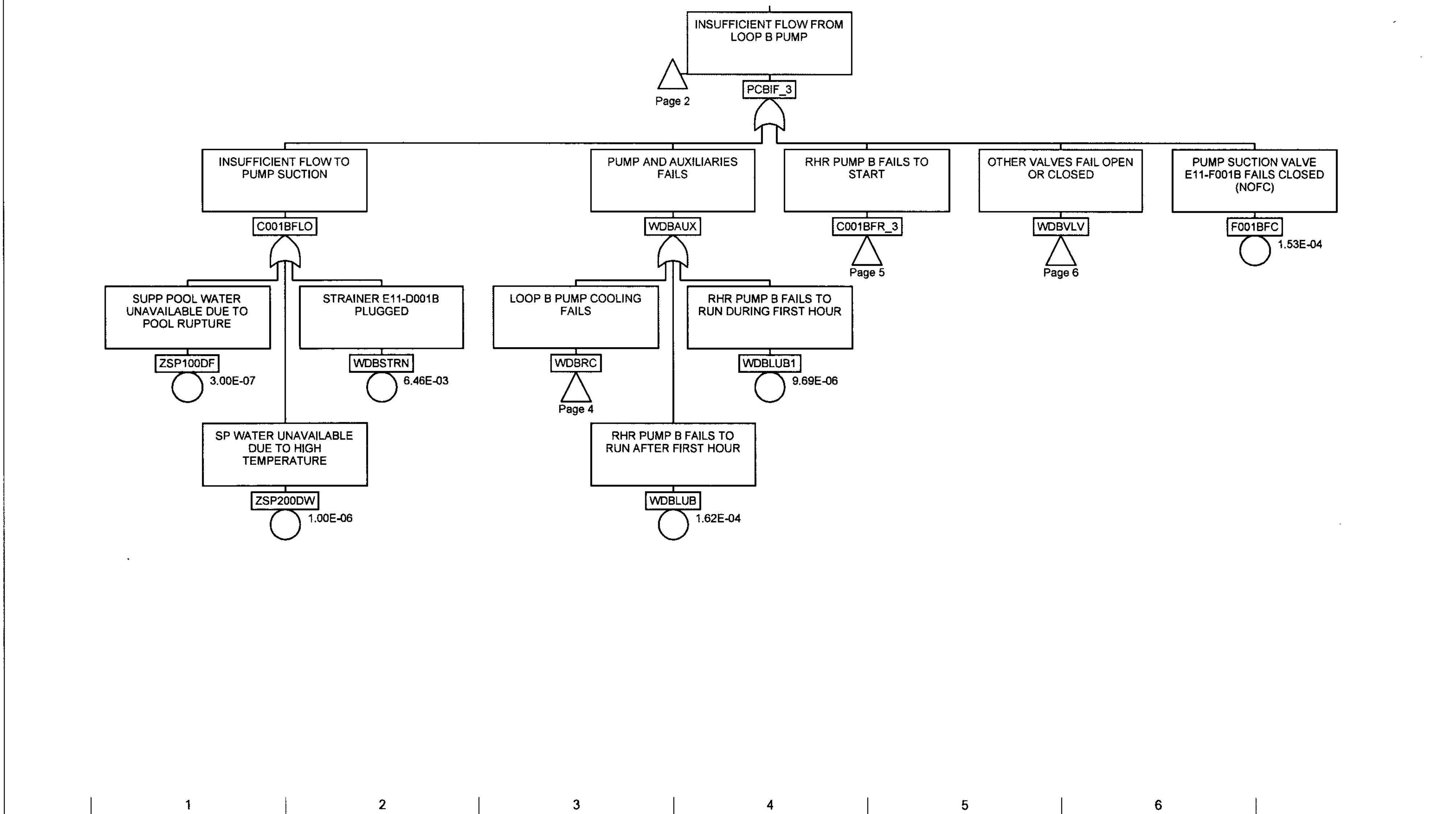

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

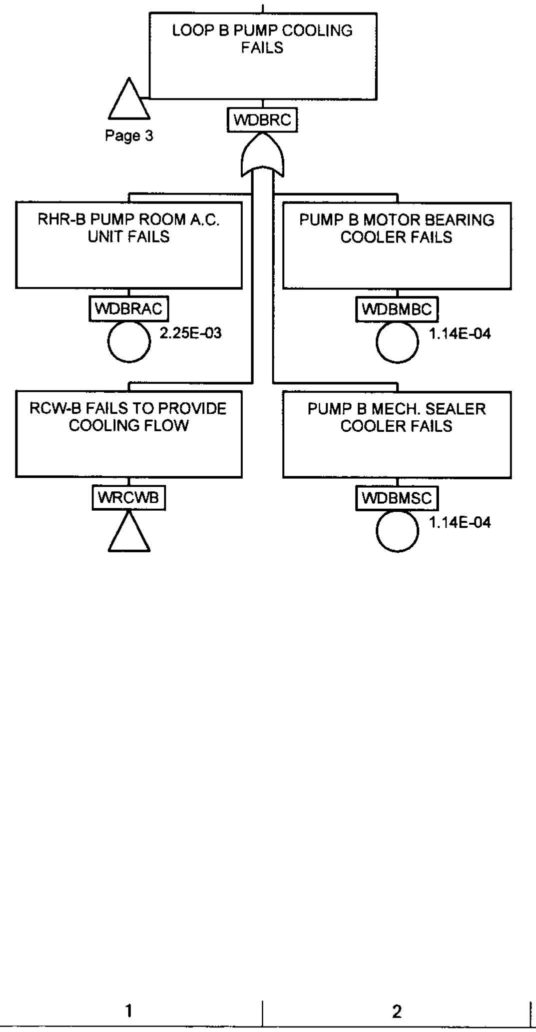

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

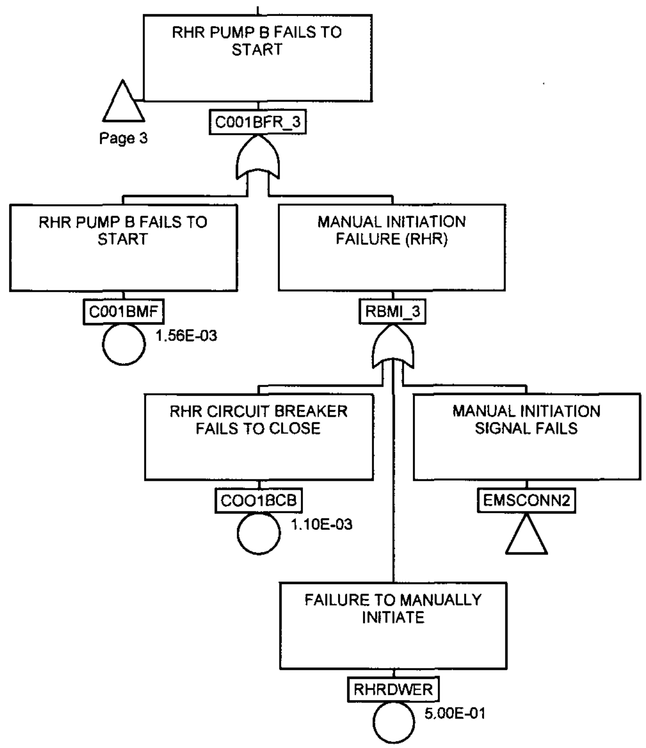

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

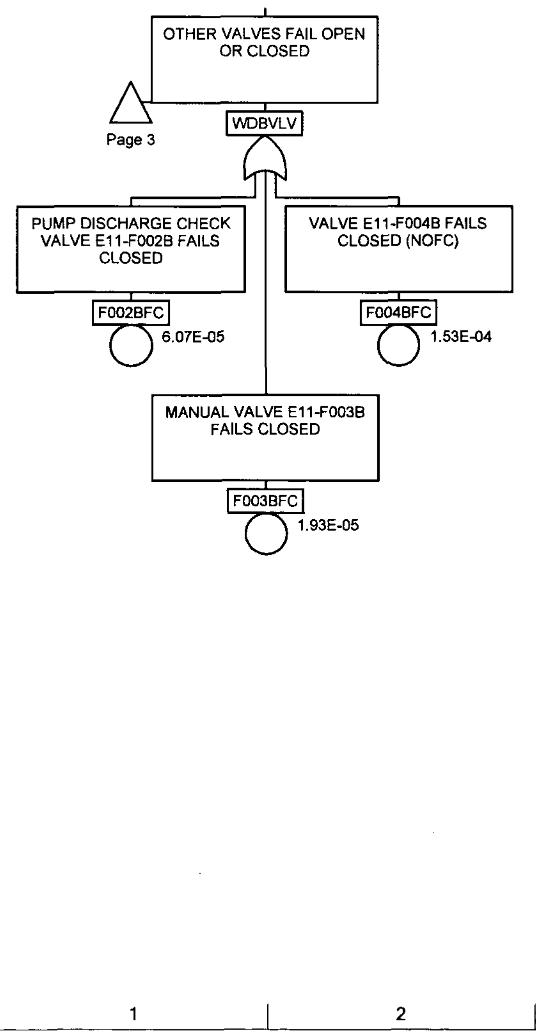

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

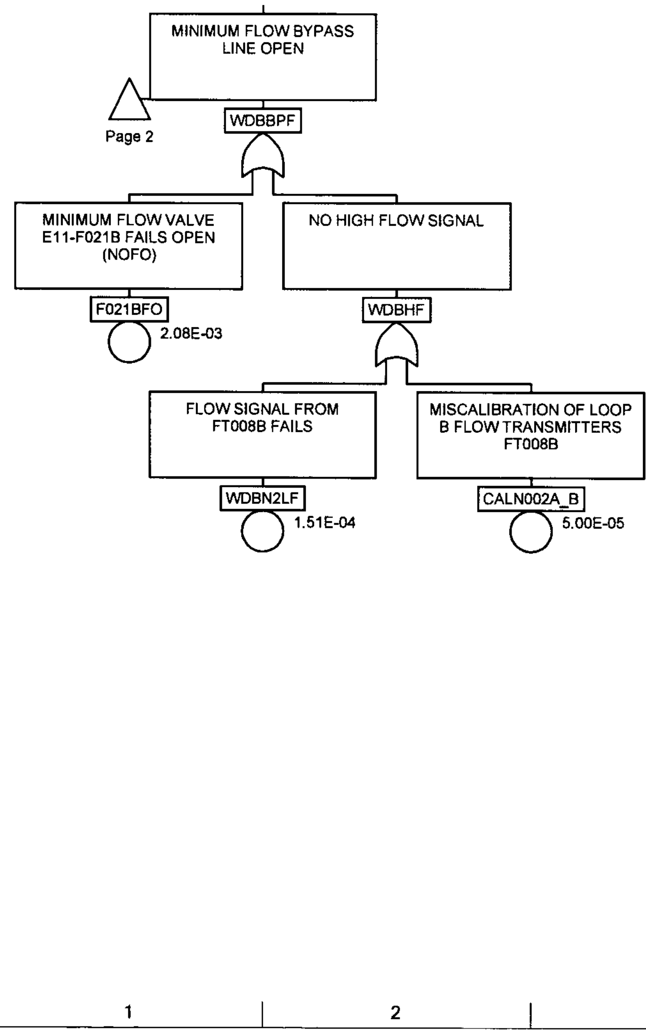

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

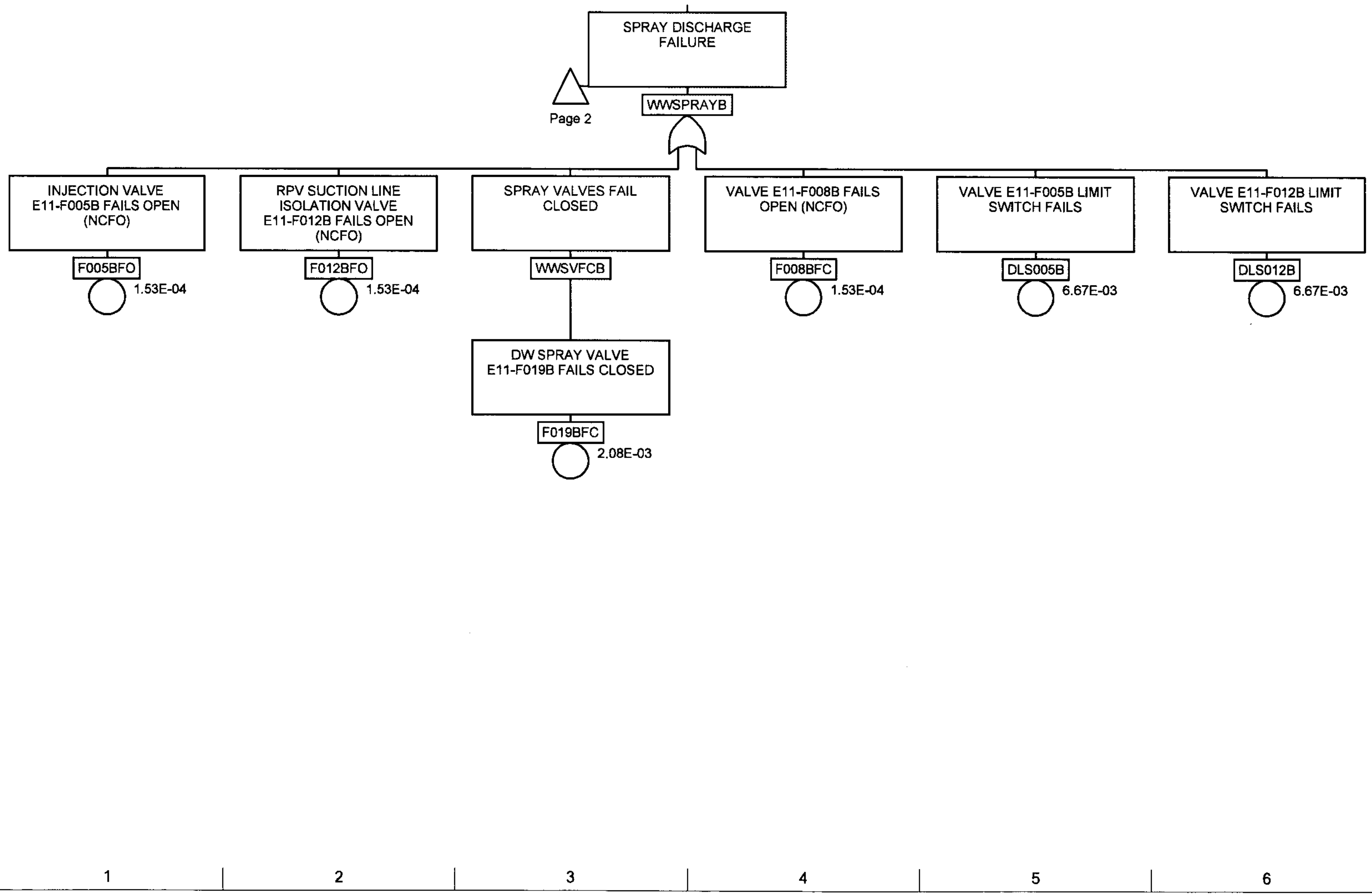

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

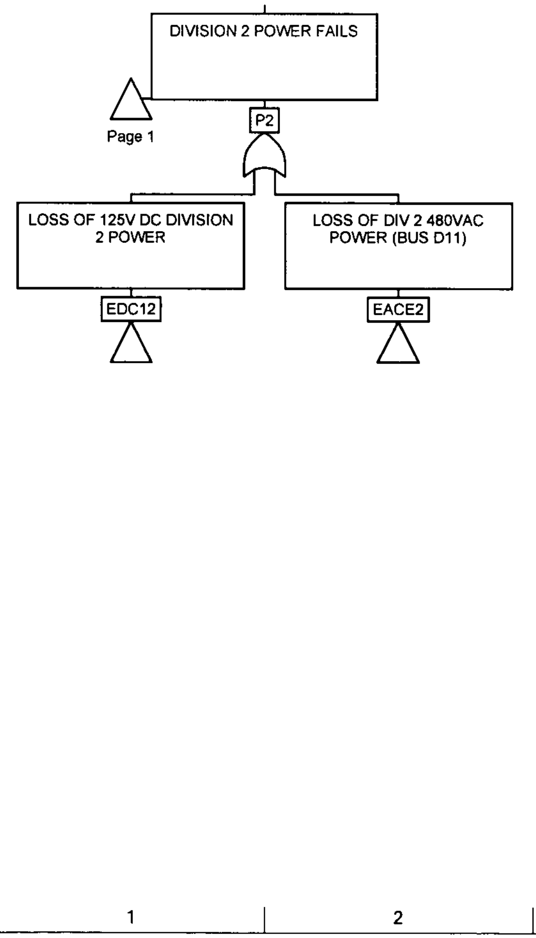

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

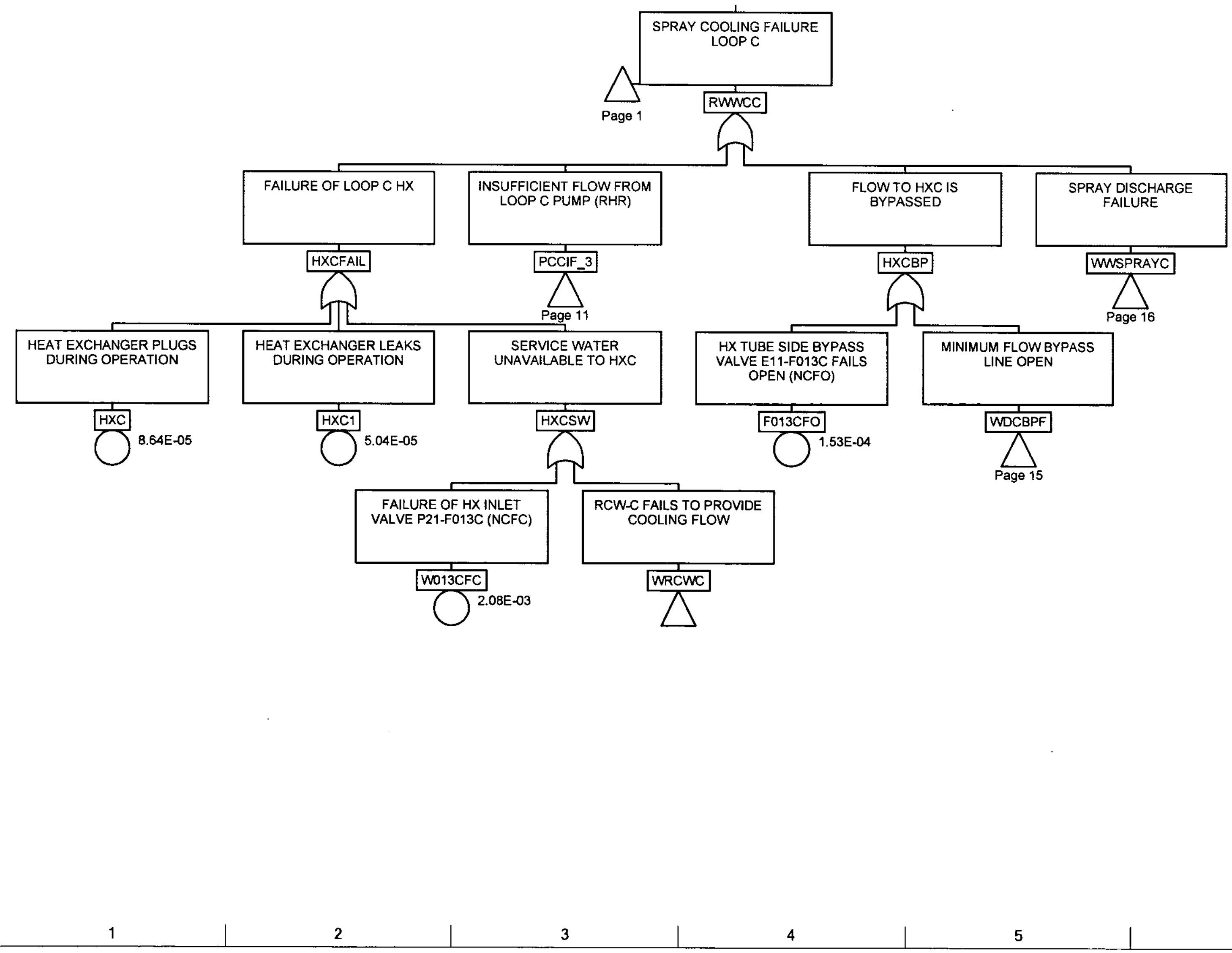

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

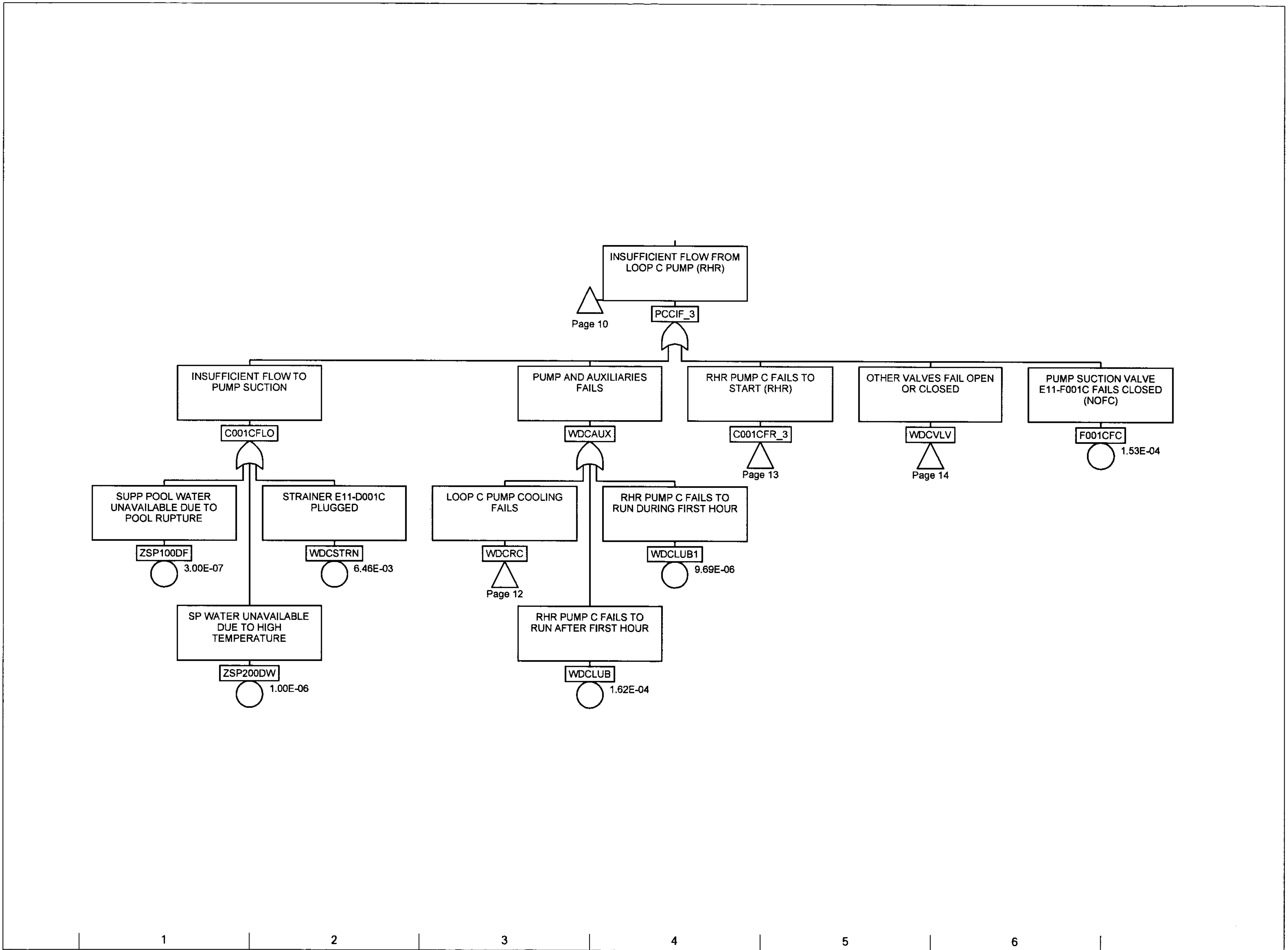

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

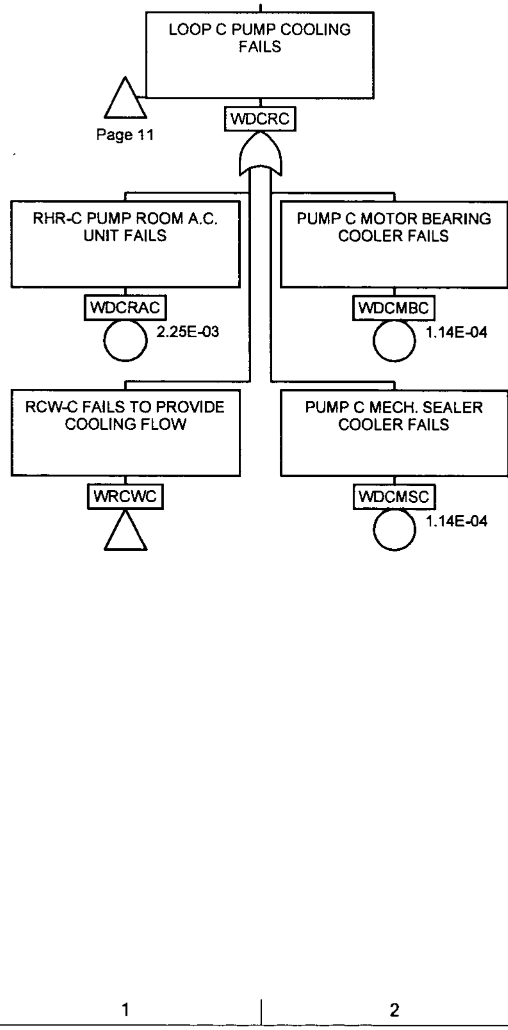

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

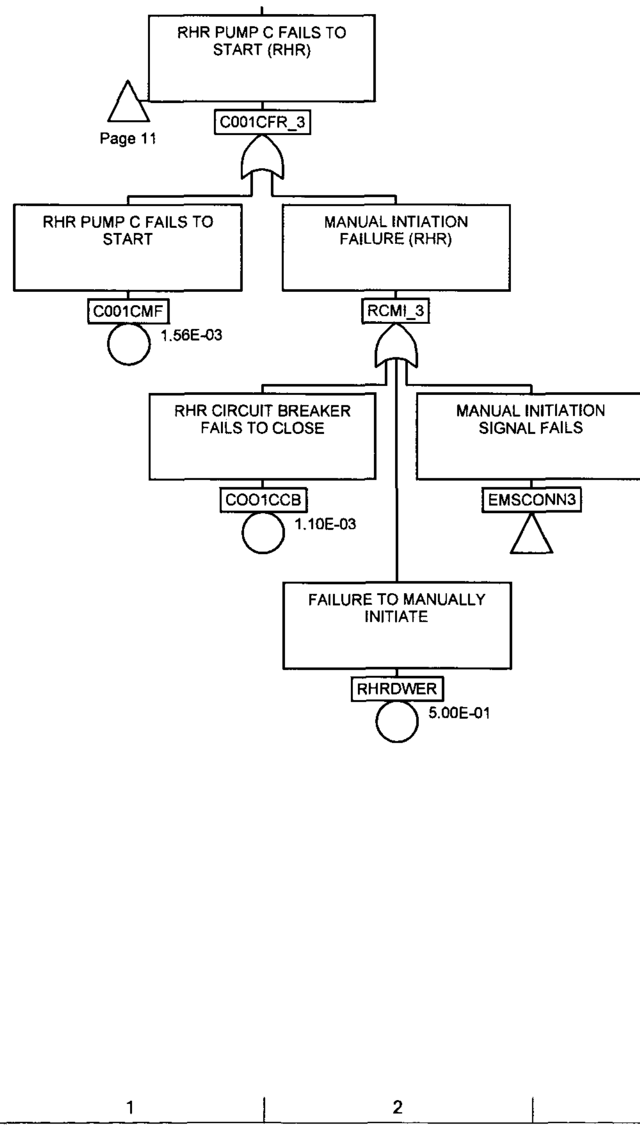

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

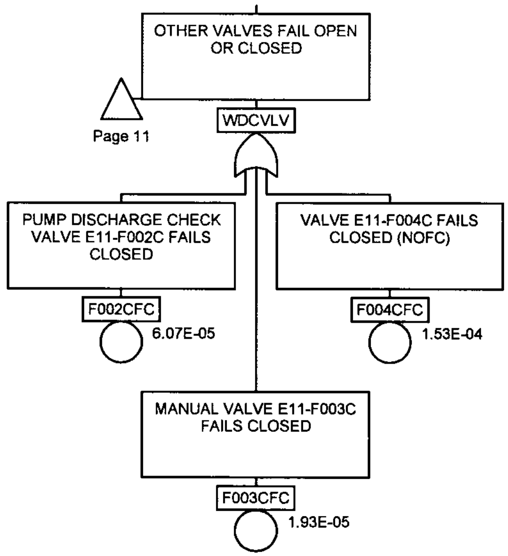

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Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

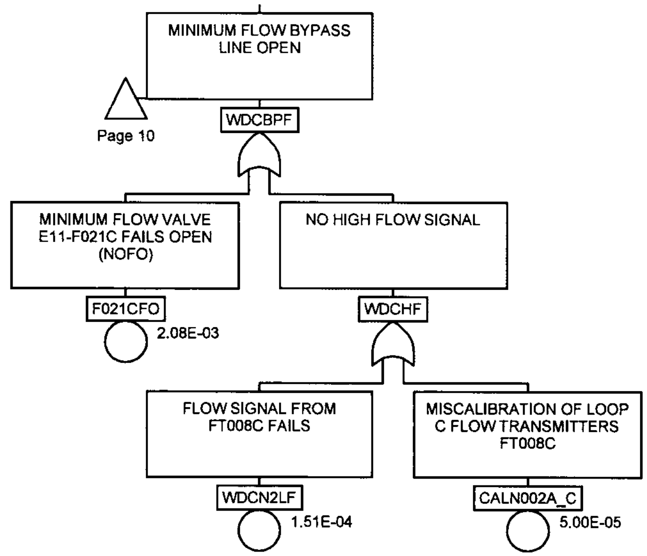

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

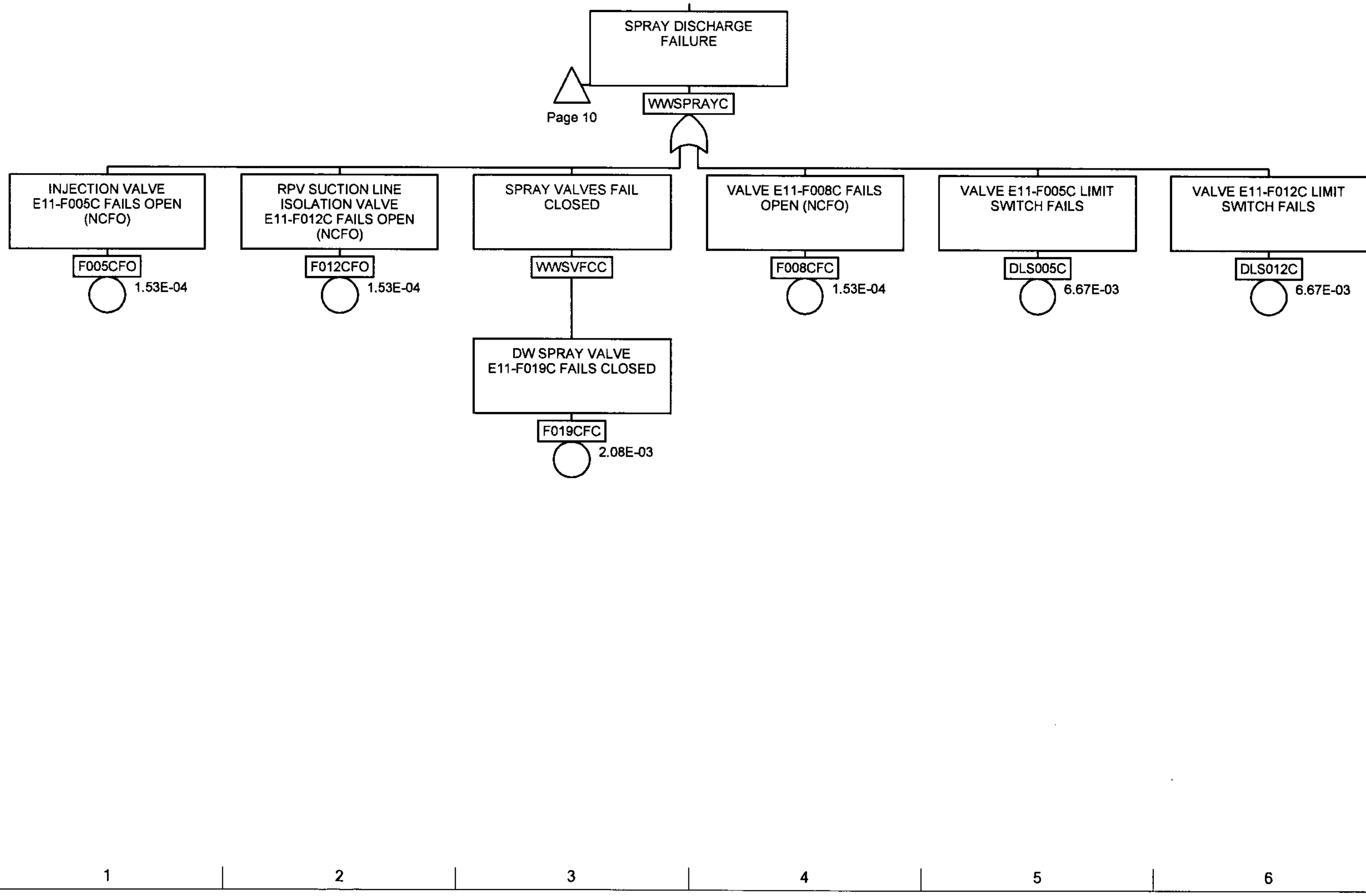

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

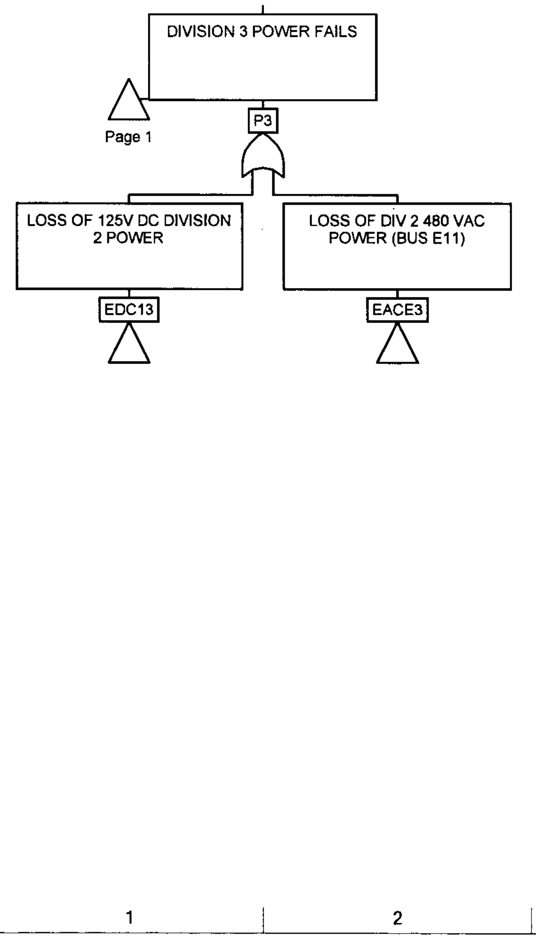

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

Name	Page	Zone	Name	Page	Zone
C001BFLO	3	2	HXC	10	1
C001BFR_3	3	5	HXC1	10	2
C001BFR_3	5	2	HXCBP	10	5
C001BMF	5	1	HXCFAIL	10	2
C001CFLO	11	2	HXCSW	10	3
C001CFR_3	11	5	P2	1	2
C001CFR_3	13	2	P2	9	2
C001CMF	13	1	P3	1	4
CALN002A_B	7	3	P3	17	2
CALN002A_C	15	3	PCBIF_3	2	3
CDWSPHDF	1	3	PCBIF_3	3	4
COO1BCB	5	2	PCCIF_3	10	3
COO1CCB	13	2	PCCIF_3	11	4
DLS005B	8	5	RBMI_3	5	2
DLS005C	16	5	RCMI_3	13	2
DLS012B	8	6	RHRDWER	5	2
DLS012C	16	6	RHRDWER	13	2
EACE2	9	2	RHRWWF	1	3
EACE3	17	2	RWWCB	1	2
EDC12	9	1	RWWCB	2	4
EDC13	17	1	RWWCB1	1	2
EMSCONN2	5	3	RWWCBC	1	2
EMSCONN3	13	3	RWWCC	1	4
F001BFC	3	7	RWWCC	10	4
F001CFC	11	7	RWWCC1	1	4
F002BFC	6	1	W013BFC	2	3
F002CFC	14	1	W013CFC	10	3
F003BFC	6	2	WDBAUX	3	4
F003CFC	14	2	WDBBPF	2	5
F004BFC	6	2	WDBBPF	7	2
F004CFC	14	2	WDBHF	7	2
F005BFO	8	1	WDBLUB	3	4
F005CFO	16	1	WDBLUB1	3	4
F008BFC	8	4	WDBMAINT	1	1
F008CFC	16	4	WDBMBC	4	2
F012BFO	8	2	WDBMSC	4	2
F012CFO	16	2	WDBN2LF	7	2
F013BFO	2	4	WDBRAC	4	1
F013CFO	10	4	WDBRC	3	3
F019BFC	8	3	WDBRC	4	2
F019CFC	16	3	WDBSTRN	3	2
F021BFO	7	1	WDBVLV	3	6
F021CFO	15	1	WDBVLV	6	2
HXB	2	1	WDCAUX	11	4
HXB1	2	2	WDCBPF	10	5
HXBBP	2	5	WDCBPF	15	2
HXBFAIL	2	2	WDCHF	15	2
HXBSW	2	3	WDCLUB	11	4

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

Name	Page	Zone	Name	Page	Zone
WDCLUB1	11	4			
WDCMAINT	1	3			
WDCMBC	12	2			
WDCMSC	12	2			
WDCN2LF	15	2			
WDCRAC	12	1			
WDCRC	11	3			
WDCRC	12	2			
WDCSTRN	11	2			
WDCVLV	11	6			
WDCVLV	14	2			
WRCWB	2	4			
WRCWB	4	1			
WRCWC	10	4			
WRCWC	12	1			
WWSPRAYB	2	6			
WWSPRAYB	8	4			
WWSPRAYC	10	6			
WWSPRAYC	16	4			
WWSVFCB	8	3			
WWSVFCC	16	3			
ZSP100DF	3	1			
ZSP100DF	11	1			
ZSP200DW	3	2			
ZSP200DW	11	2			

Figure 19D.6-8B RHR Wetwell Spray System Fault Tree

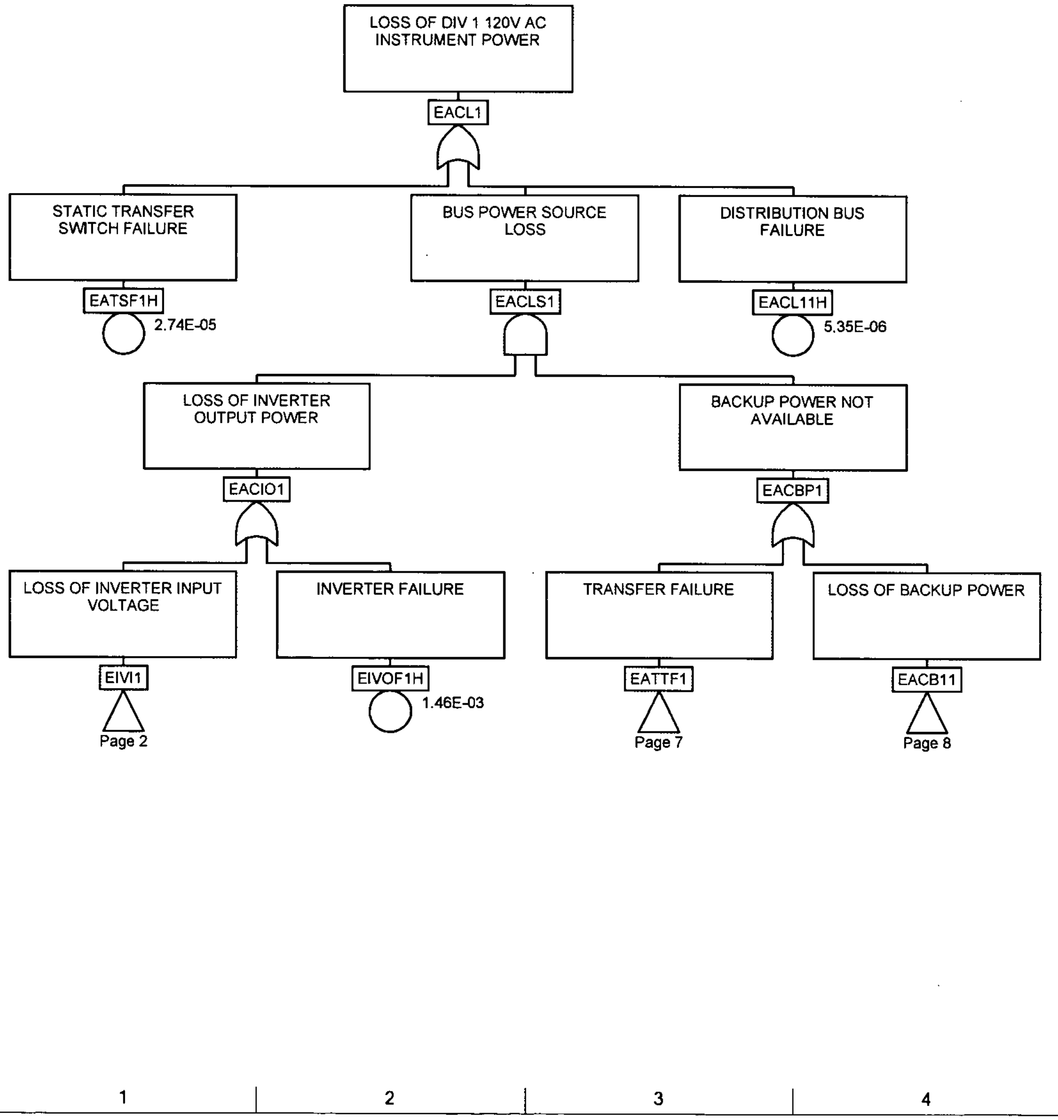

Figure 19D.6-9A Electric Power-120 VAC Fault Tree Division 1

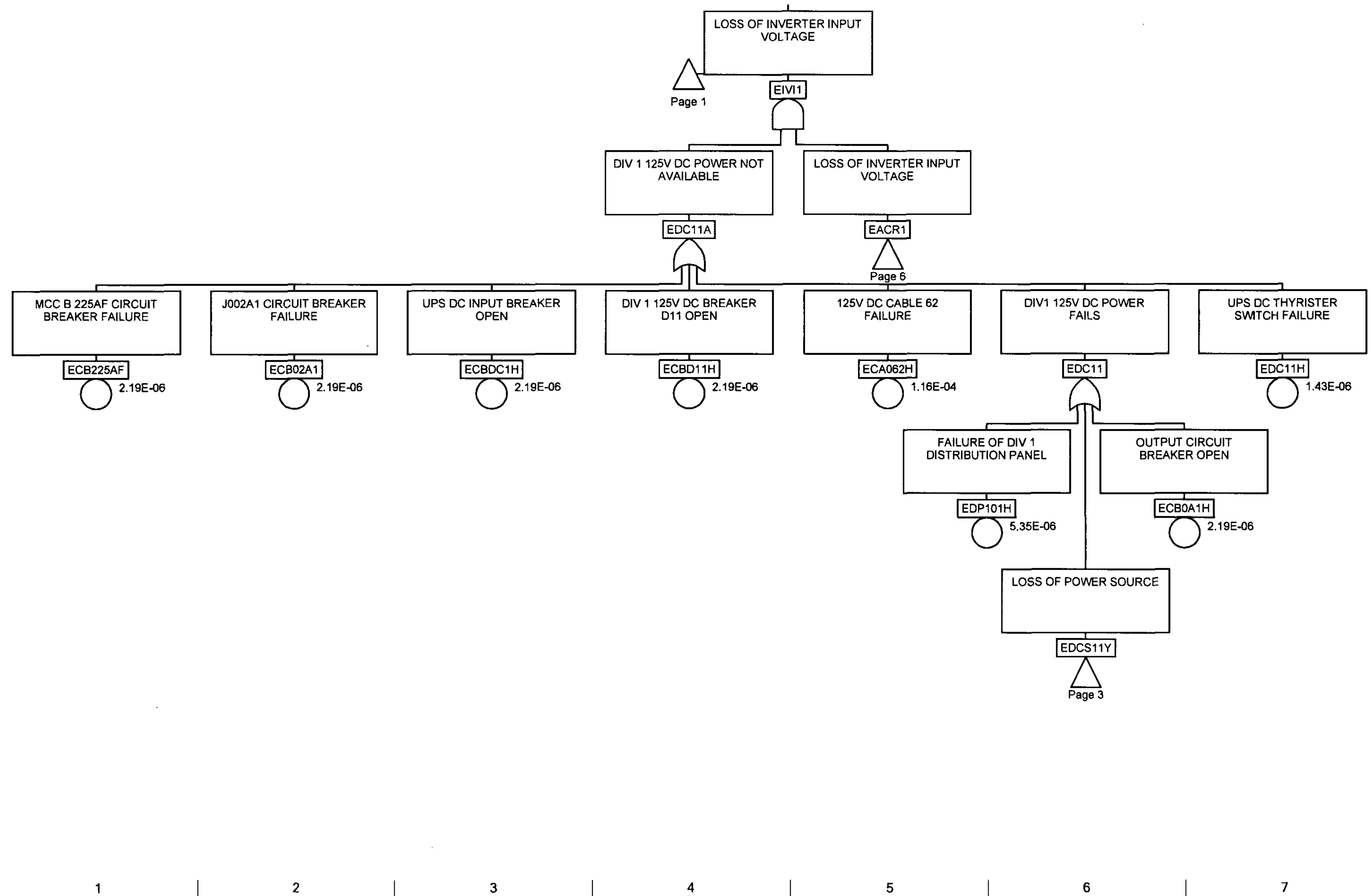

Figure 19D.6-9A Electric Power-120 VAC Fault Tree Division 1

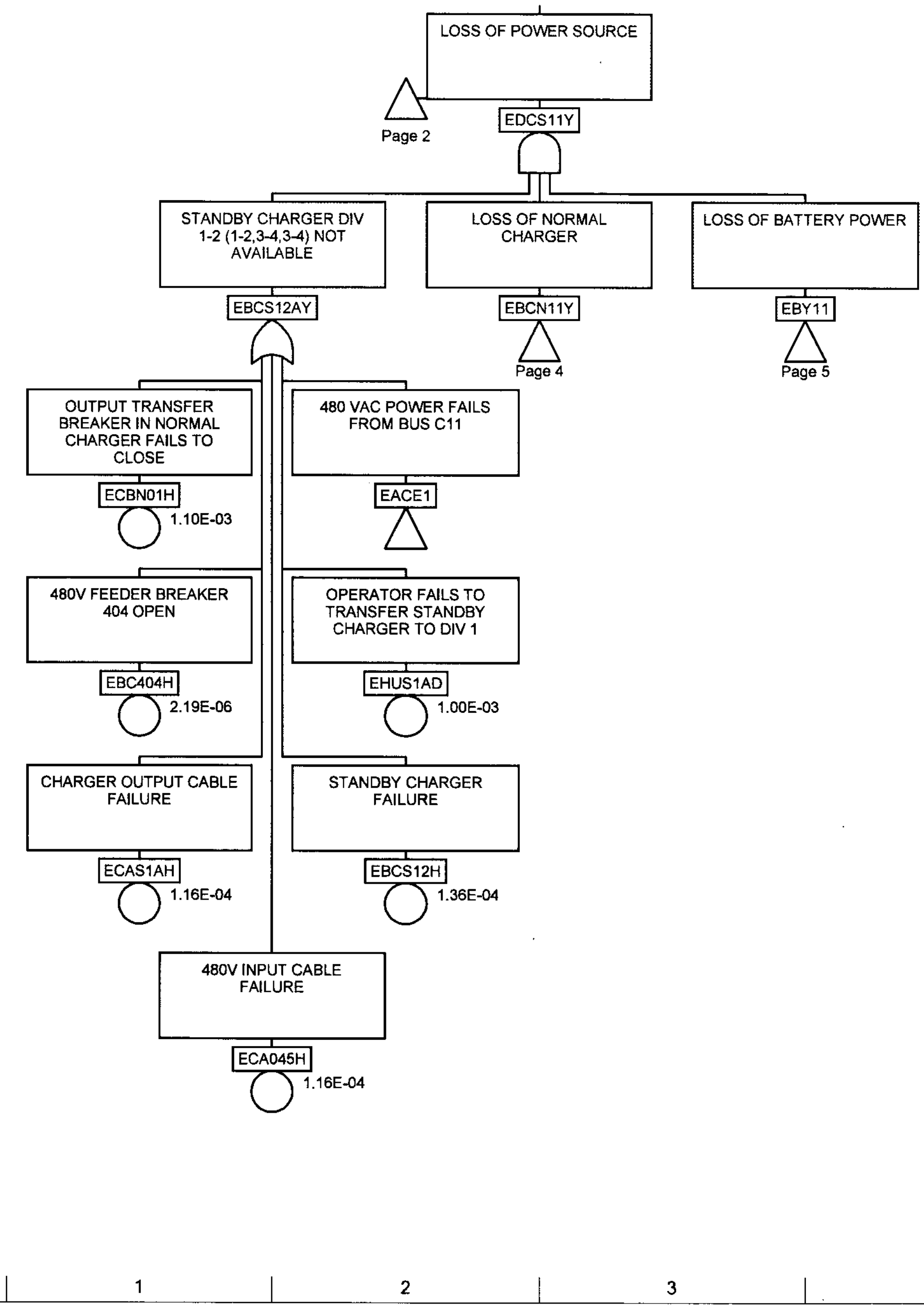

Figure 19D.6-9A Electric Power-120 VAC Fault Tree Division 1

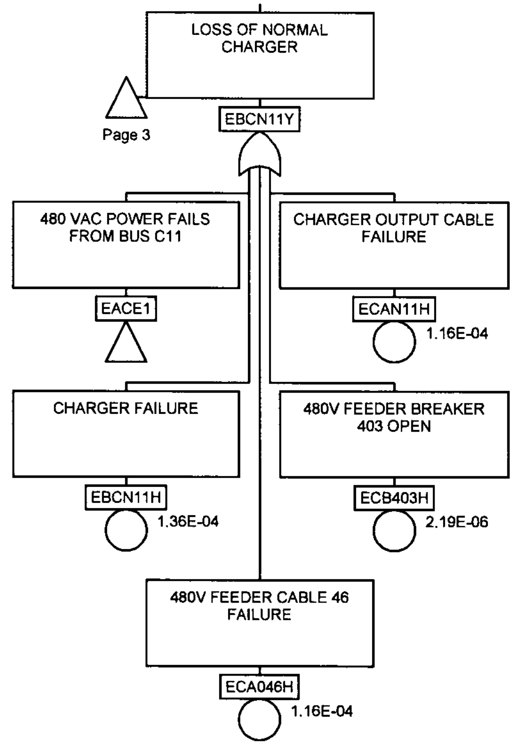

Figure 19D.6-9A Electric Power-120 VAC Fault Tree Division 1

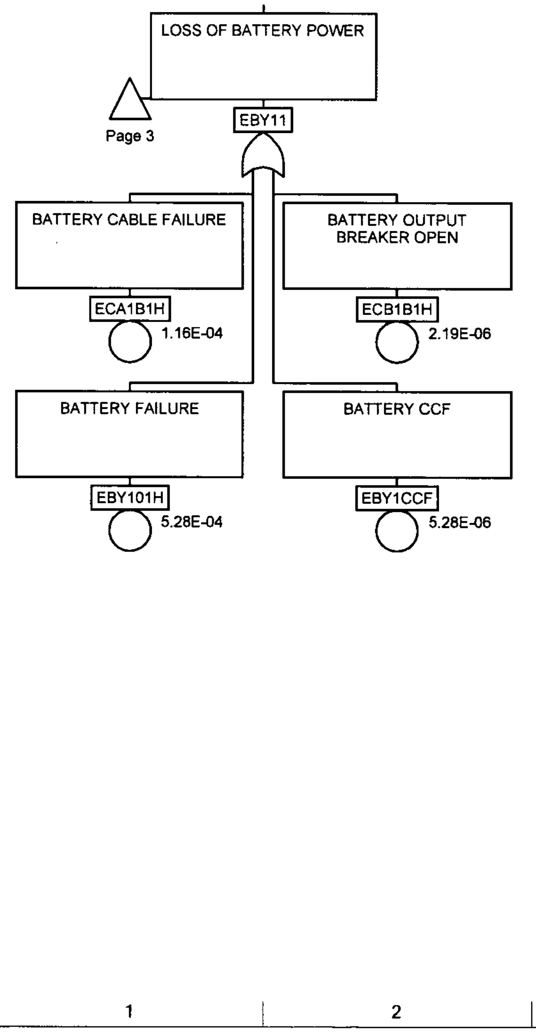

Figure 19D.6-9A Electric Power-120 VAC Fault Tree Division 1

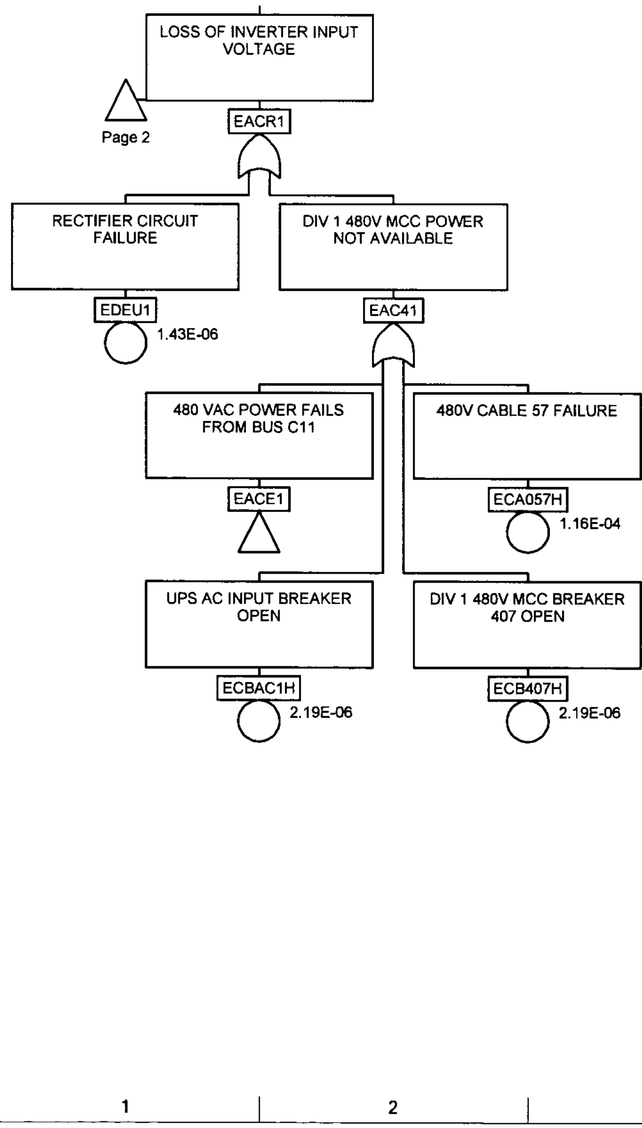

Figure 19D.6-9A Electric Power-120 VAC Fault Tree Division 1

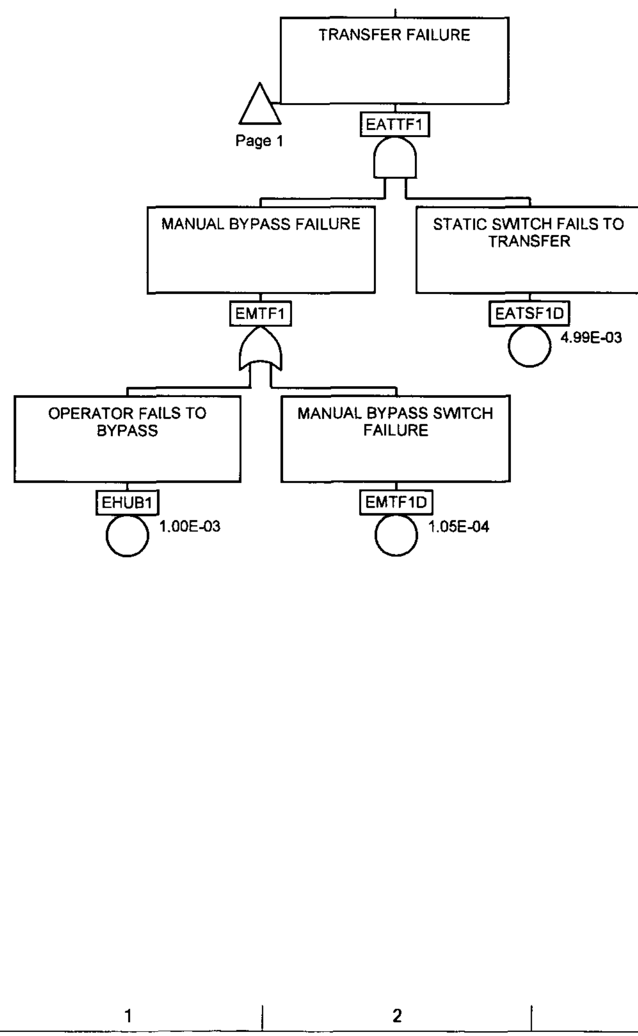

Figure 19D.6-9A Electric Power-120 VAC Fault Tree Division 1

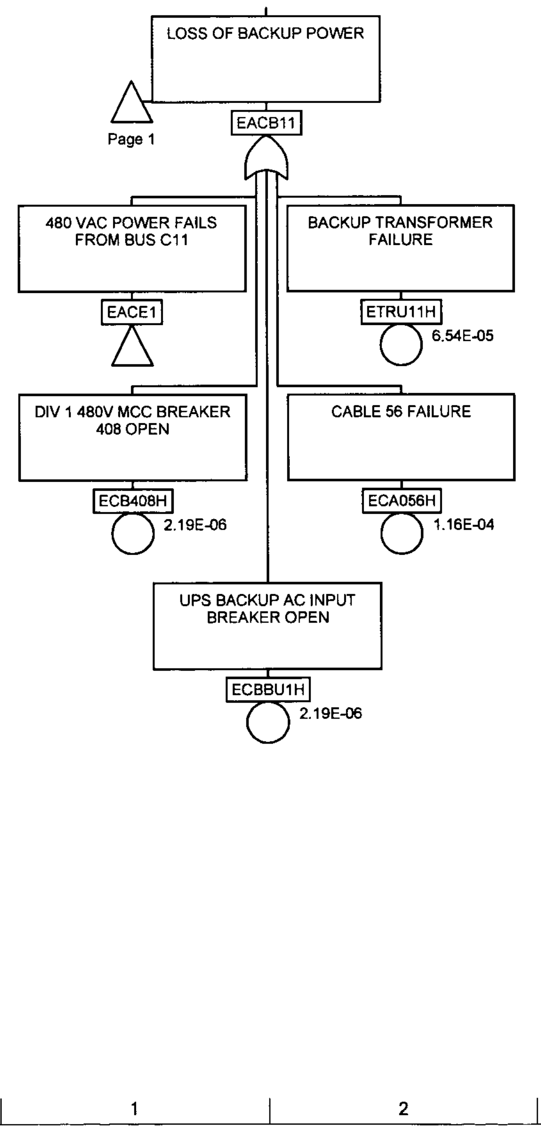

Figure 19D.6-9A Electric Power-120 VAC Fault Tree Division 1

Name	Page	Zone
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EACB11	1	4
EACB11	8	2
EACBP1	1	4
EACE1	3	2
EACE1	4	1
EACE1	6	2
EACE1	8	1
EACIO1	1	2
EACL1	1	2
EACL11H	1	3
EACLS1	1	2
EACR1	2	5
EACR1	6	2
EATSF1D	7	3
EATSF1H	1	1
EATTF1	1	3
EATTF1	7	2
EBC404H	3	1
EBCN11H	4	1
EBCN11Y	3	3
EBCN11Y	4	2
EBCS12AY	3	2
EBCS12H	3	2
EBY101H	5	1
EBY11	3	4
EBY11	5	2
EBY1CCF	5	2
ECA045H	3	2
ECA046H	4	2
ECA056H	8	2
ECA057H	6	3
ECA062H	2	5
ECA1B1H	5	1
ECAN11H	4	2
ECAS1AH	3	1
ECB02A1	2	2
ECB0A1H	2	7
ECB1B1H	5	2
ECB225AF	2	1
ECB403H	4	2
ECB407H	6	3
ECB408H	8	1
ECBAC1H	6	2
ECBBU1H	8	2
ECBD11H	2	4
ECBDC1H	2	3
ECBN01H	3	1
EDC11	2	6
EDC11A	2	4
EDC11H	2	7
EDCS11Y	2	6
EDCS11Y	3	2
EDEU1	6	1
EDP101H	2	6
EHUB1	7	1
EHUS1AD	3	2
EIVI1	1	1
EIVI1	2	5
EIVOF1H	1	2
EMTF1	7	2
EMTF1D	7	2
ETRU11H	8	2

Figure 19D.6-9A Electric Power-120 VAC Fault Tree Division 1

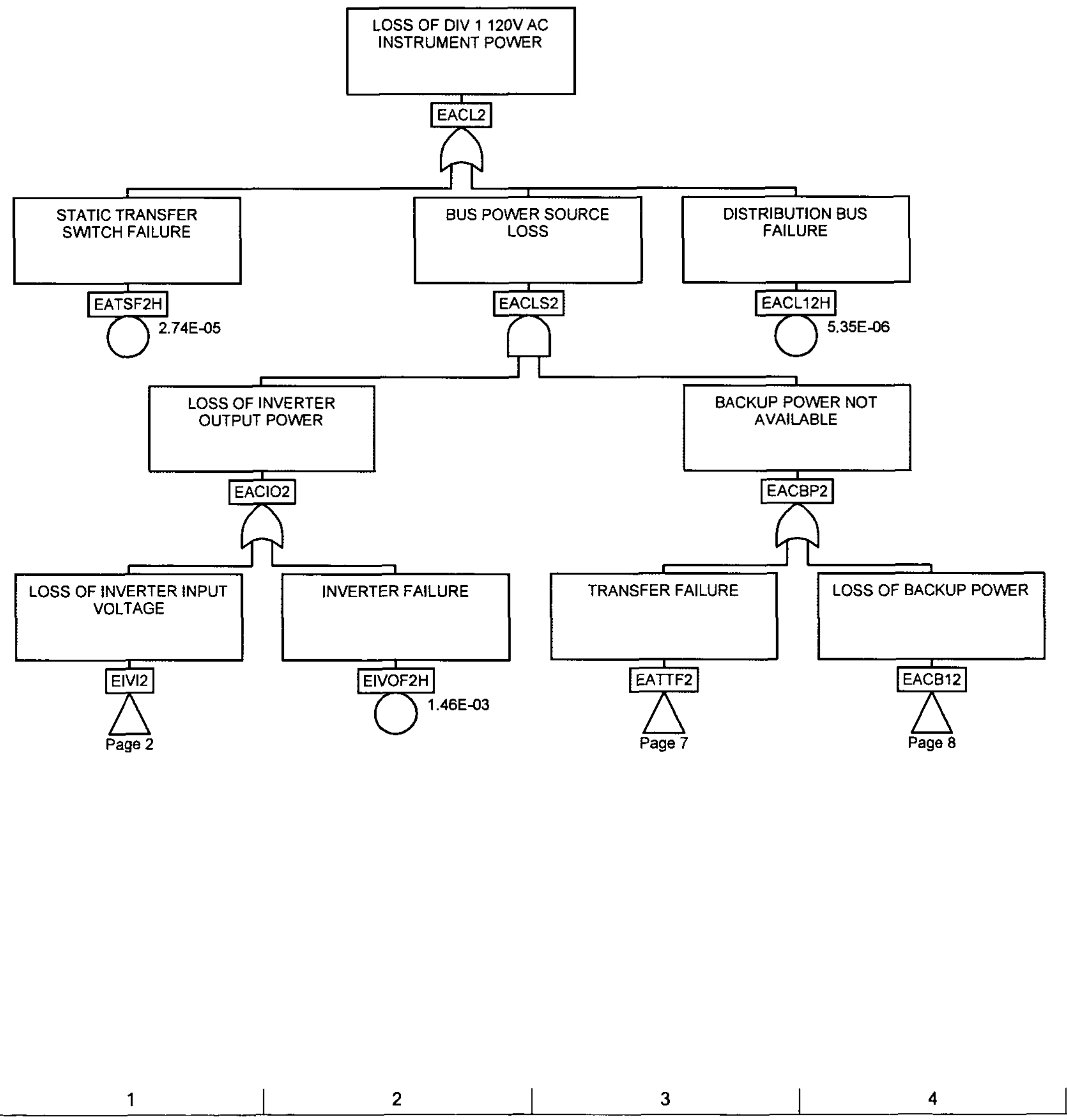

Figure 19D.6-9B Electric Power-120V AC Fault Tree Division 2

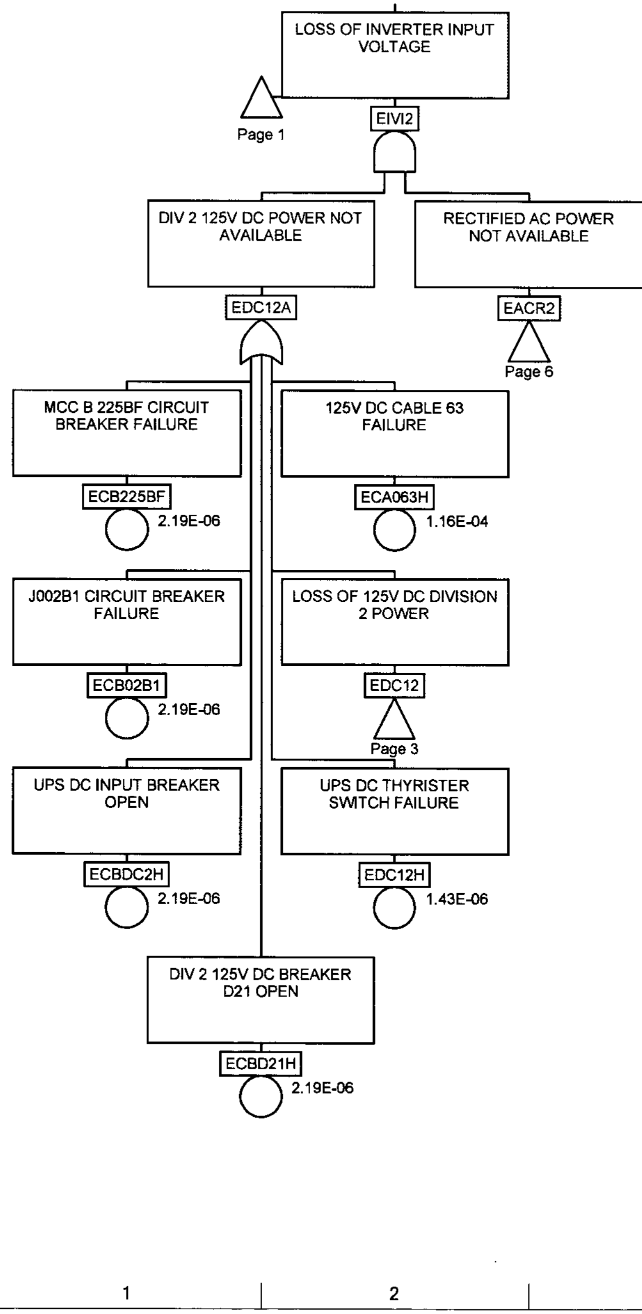

Figure 19D.6-9B Electric Power-120V AC Fault Tree Division 2

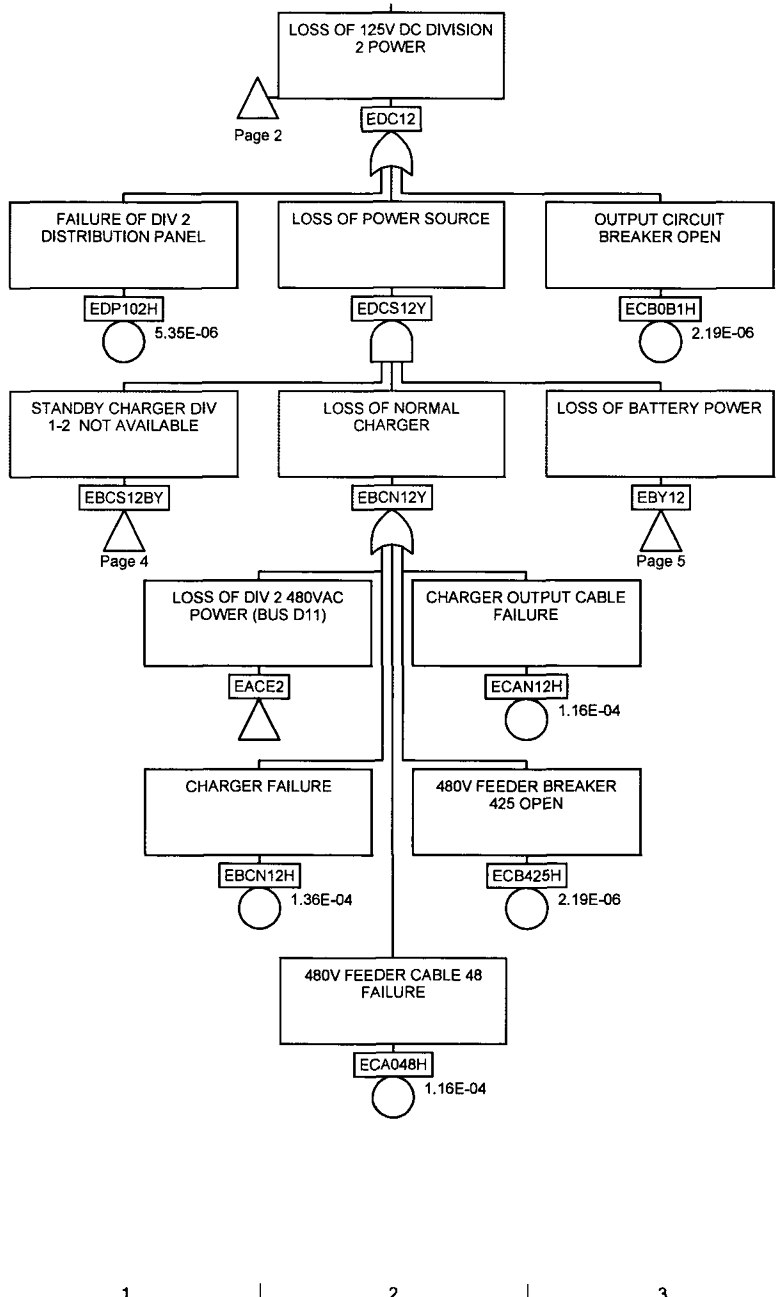

Figure 19D.6-9B Electric Power-120V AC Fault Tree Division 2

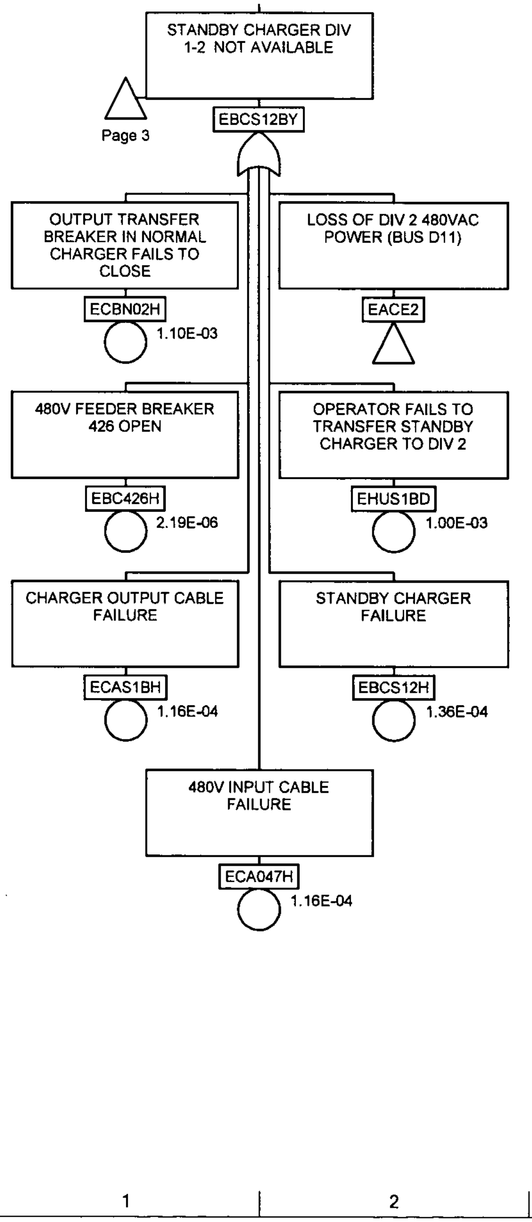

Figure 19D.6-9B Electric Power-120V AC Fault Tree Division 2

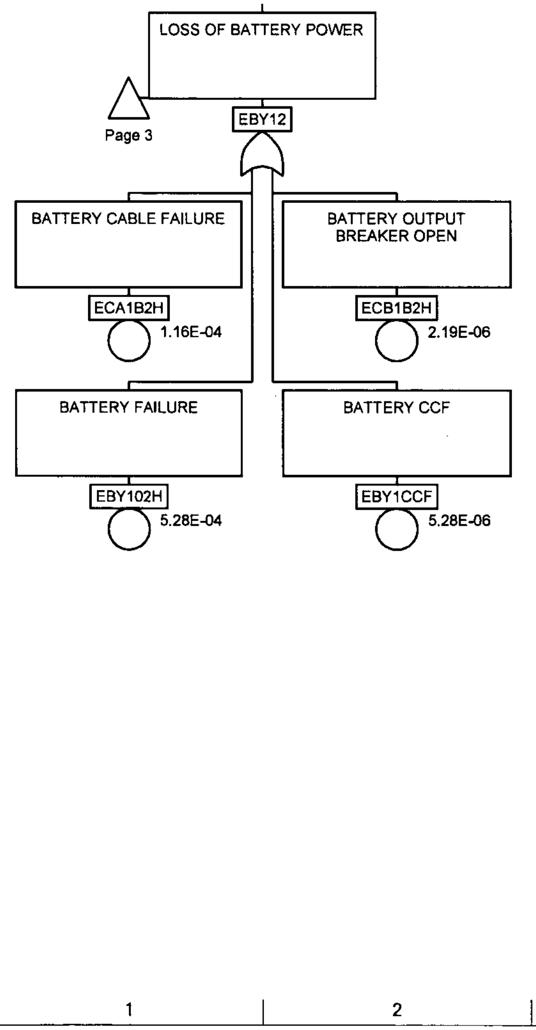

Figure 19D.6-9B Electric Power-120V AC Fault Tree Division 2

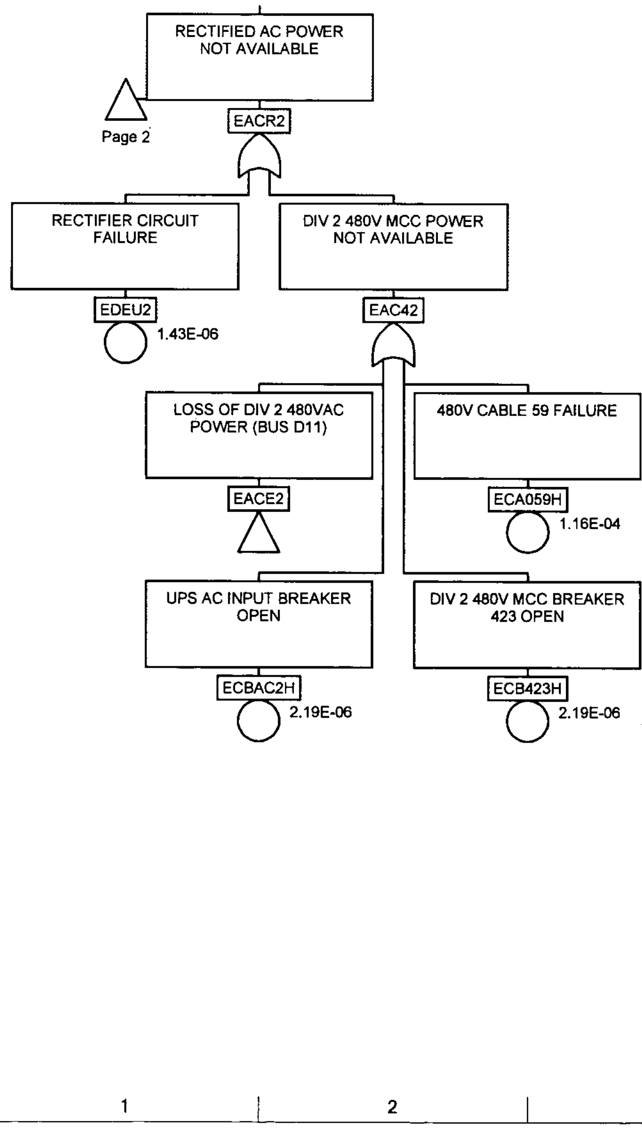

Figure 19D.6-9B Electric Power-120V AC Fault Tree Division 2

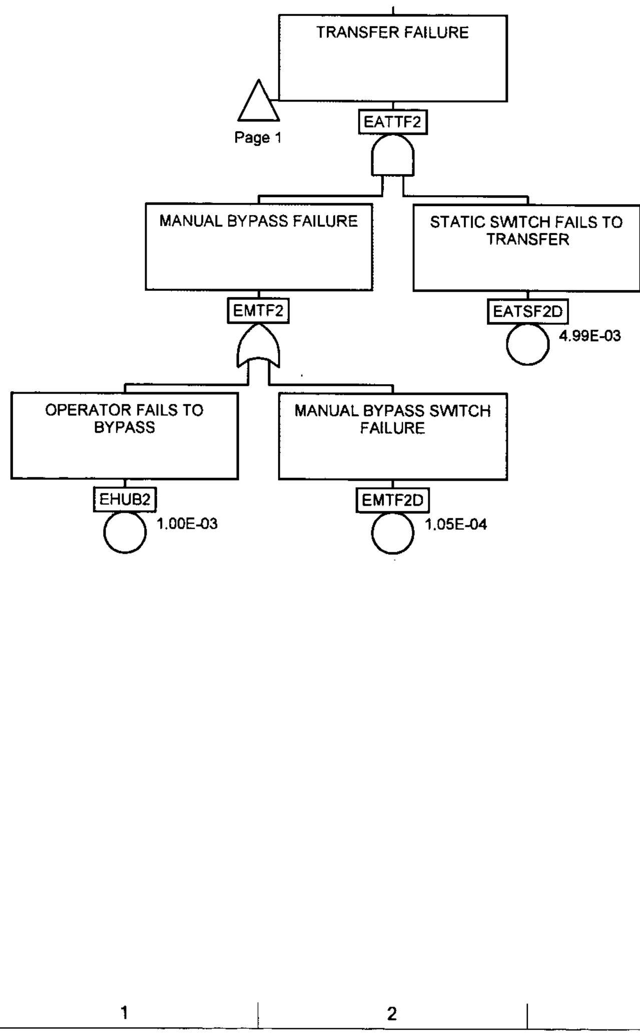

Figure 19D.6-9B Electric Power-120V AC Fault Tree Division 2

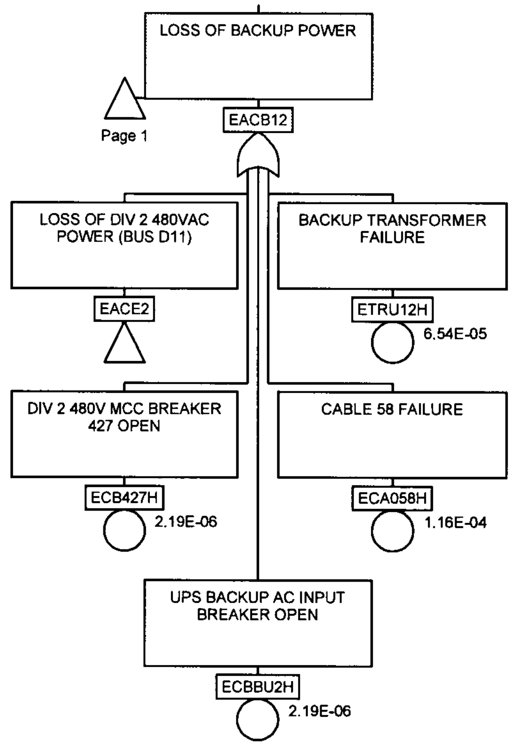

Figure 19D.6-9B Electric Power-120V AC Fault Tree Division 2

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EAC42	6	2	EDC12	2	2
EACB12	1	4	EDC12	3	2
EACB12	8	2	EDC12A	2	2
EACBP2	1	4	EDC12H	2	2
EACE2	3	2	EDCS12Y	3	2
EACE2	4	2	EDEU2	6	1
EACE2	6	2	EDP102H	3	1
EACE2	8	1	EHUB2	7	1
EACIO2	1	2	EHUS1BD	4	2
EACL12H	1	3	EIVI2	1	1
EACL2	1	2	EIVI2	2	2
EACLS2	1	2	EIVOF2H	1	2
EACR2	2	3	EMTF2	7	2
EACR2	6	2	EMTF2D	7	2
EATSF2D	7	3	ETRU12H	8	2
EATSF2H	1	1			
EATTF2	1	3			
EATTF2	7	2			
EBC426H	4	1			
EBCN12H	3	2			
EBCN12Y	3	2			
EBCS12BY	3	1			
EBCS12BY	4	2			
EBCS12H	4	2			
EBY102H	5	1			
EBY12	3	3			
EBY12	5	2			
EBY1CCF	5	2			
ECA047H	4	2			
ECA048H	3	2			
ECA058H	8	2			
ECA059H	6	3			
ECA063H	2	2			
ECA1B2H	5	1			
ECAN12H	3	3			
ECAS1BH	4	1			
ECB02B1	2	1			
ECB0B1H	3	3			
ECB1B2H	5	2			
ECB225BF	2	1			
ECB423H	6	3			
ECB425H	3	3			
ECB427H	8	1			
ECBAC2H	6	2			
ECBBU2H	8	2			
ECBD21H	2	2			
ECBDC2H	2	1			
ECBN02H	4	1			

Figure 19D.6-9B Electric Power-120V AC Fault Tree Division 2

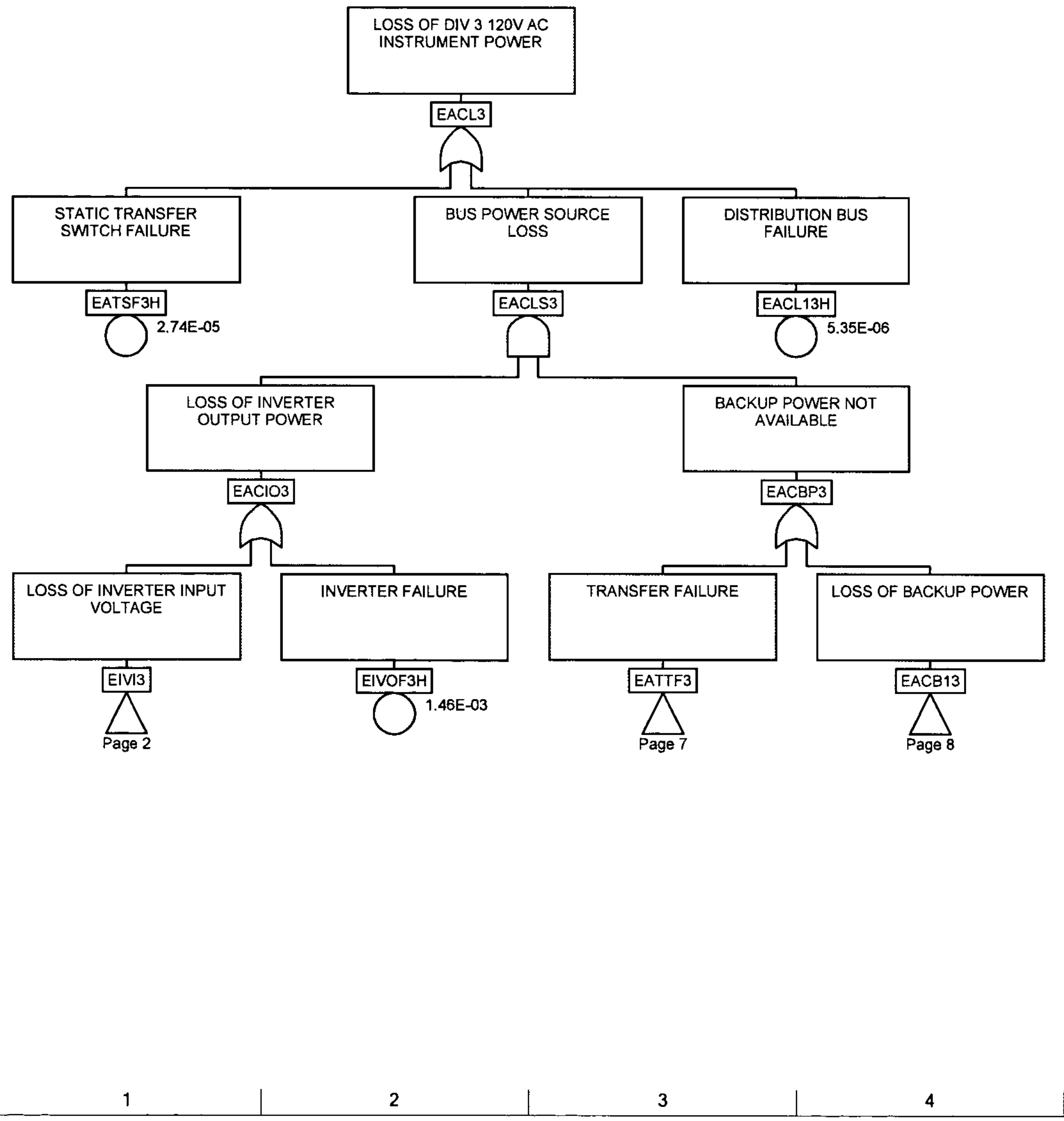


Figure 19D.6-9C Electric Power-120V AC Fault Tree Division 3

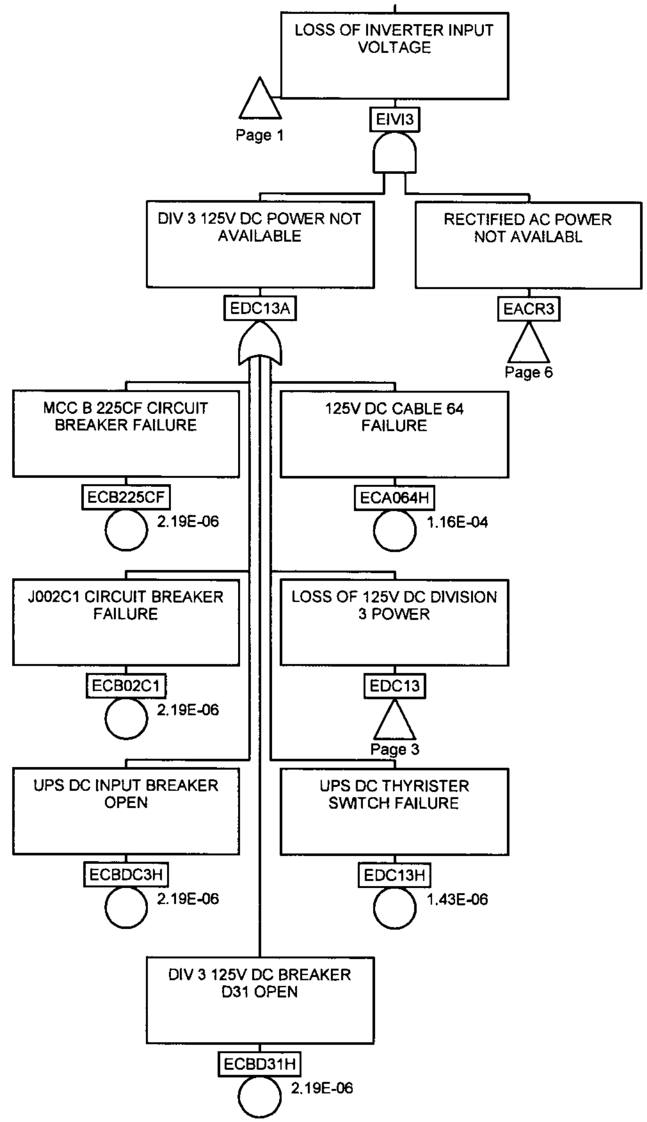


Figure 19D.6-9C Electric Power-120V AC Fault Tree Division 3

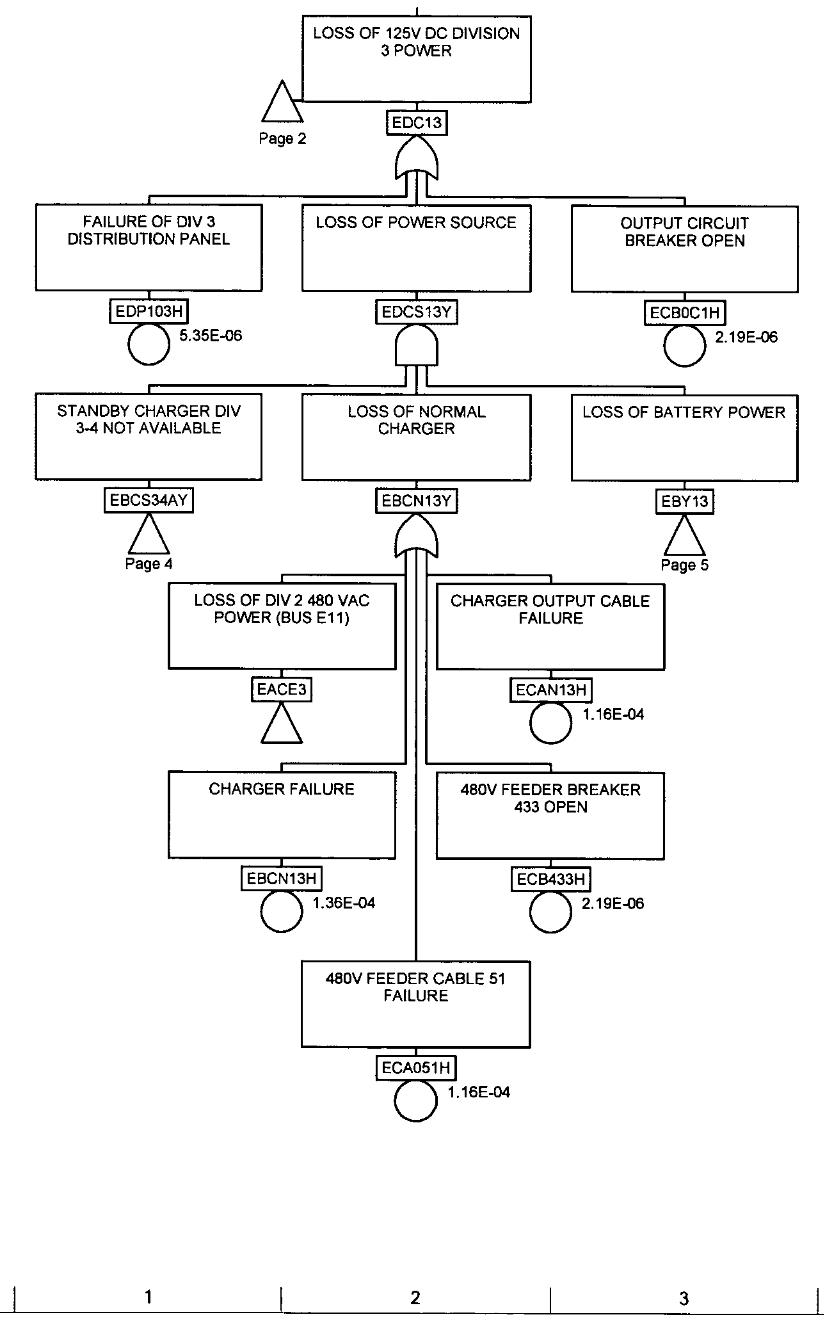


Figure 19D.6-9C Electric Power-120V AC Fault Tree Division 3

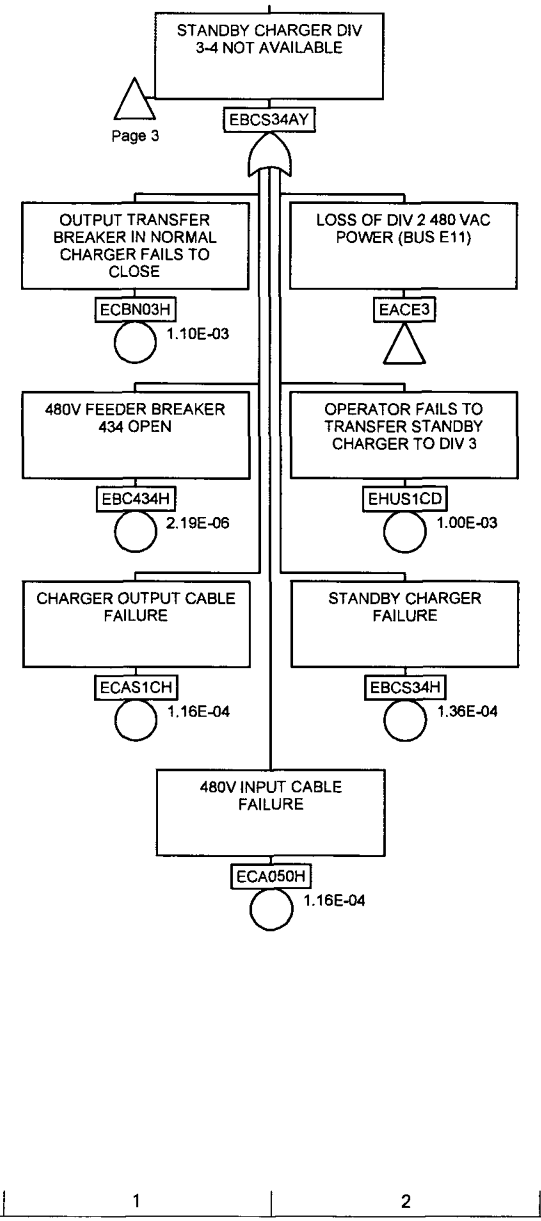


Figure 19D.6-9C Electric Power-120V AC Fault Tree Division 3

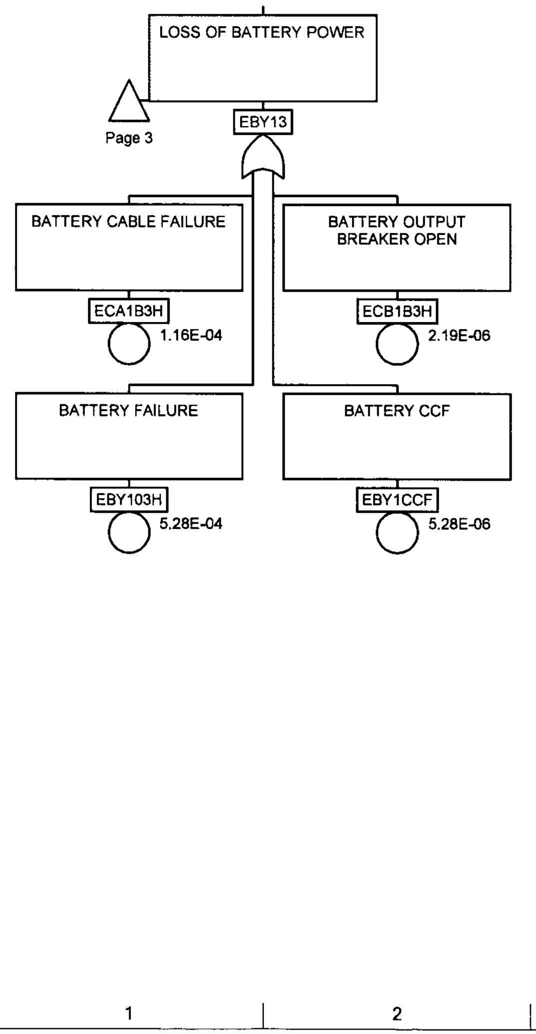


Figure 19D.6-9C Electric Power-120V AC Fault Tree Division 3

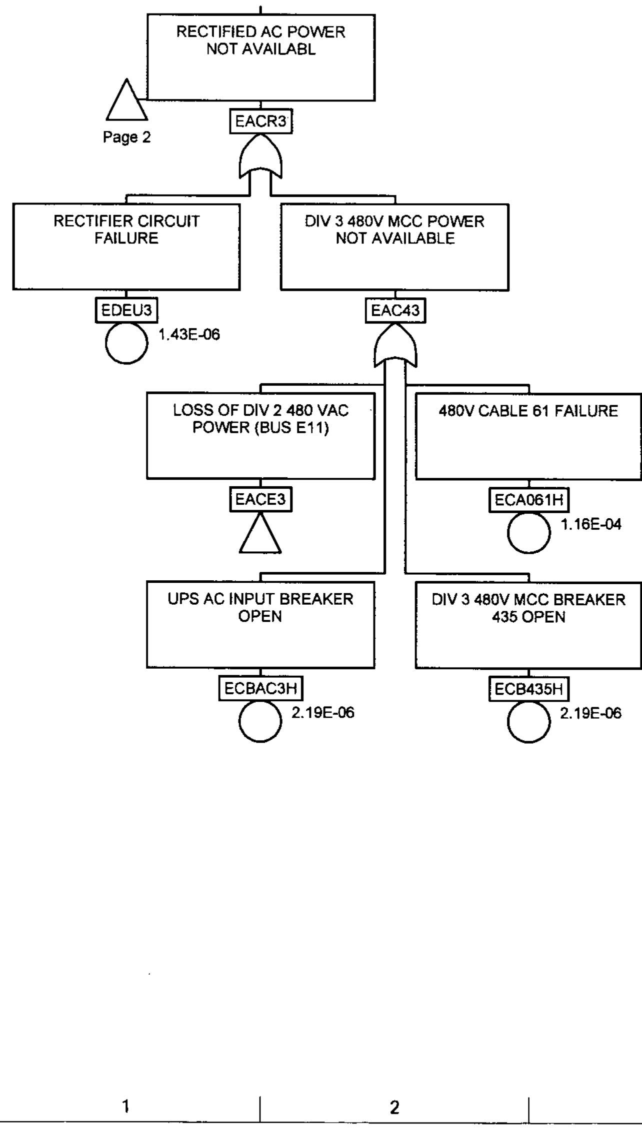


Figure 19D.6-9C Electric Power-120V AC Fault Tree Division 3

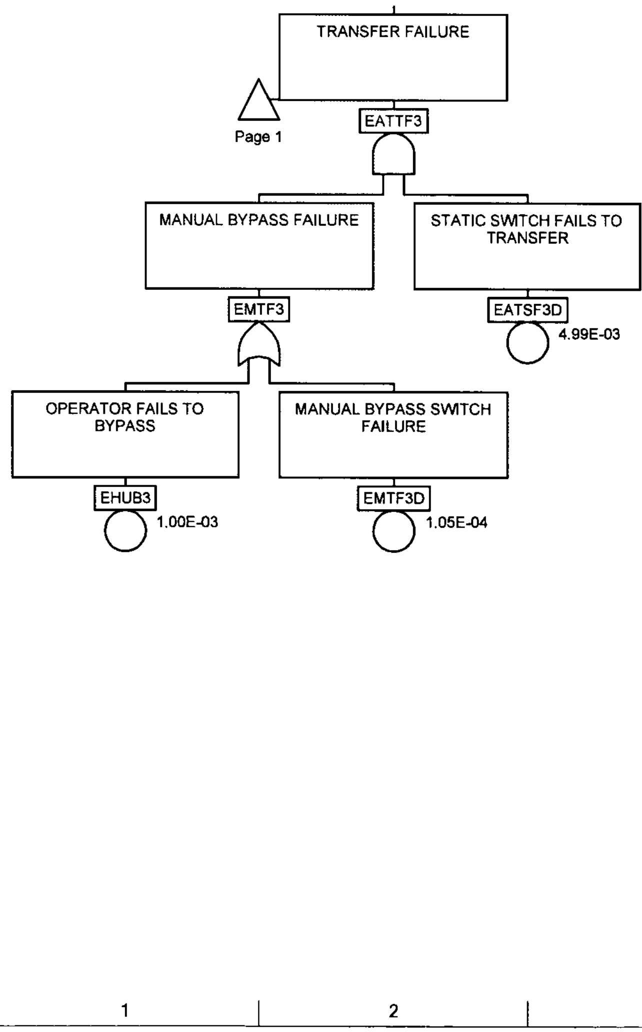


Figure 19D.6-9C Electric Power-120V AC Fault Tree Division 3

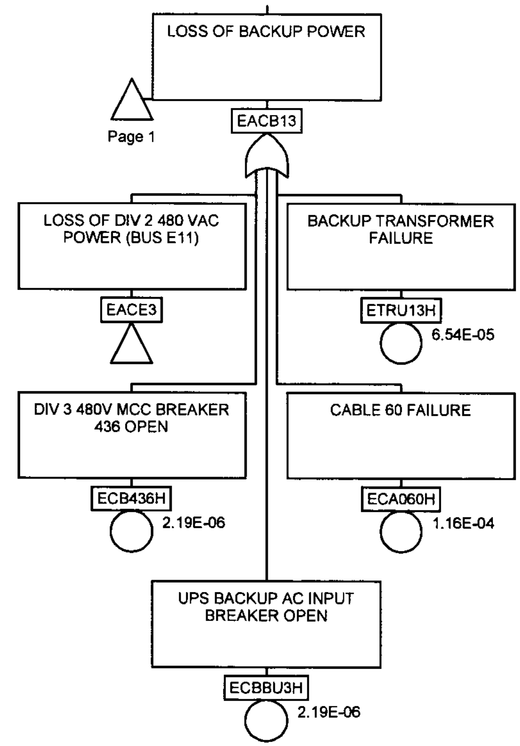


Figure 19D.6-9C Electric Power-120V AC Fault Tree Division 3

Figure 19D.6-9C Electric Power-120V AC Fault Tree Division 3

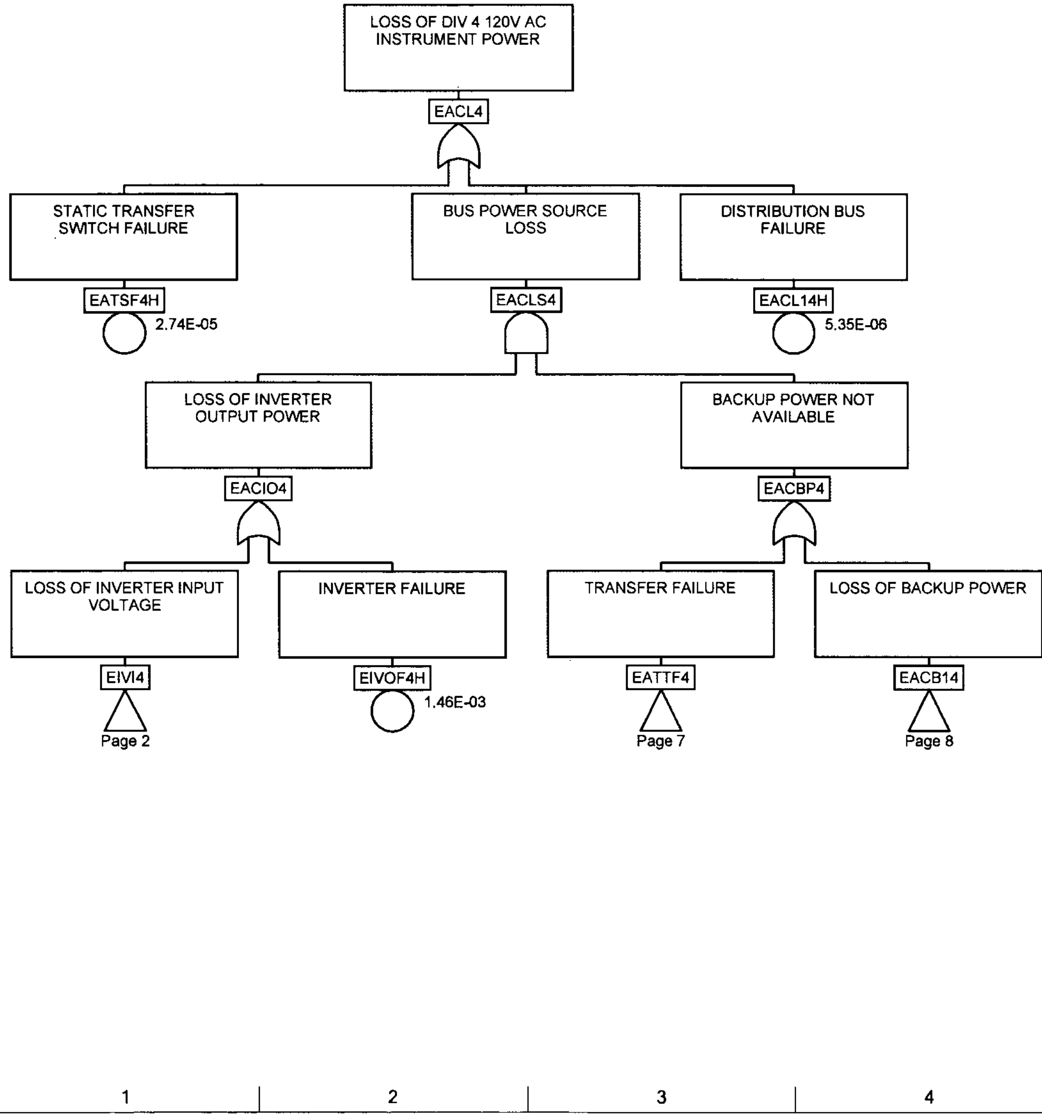


Figure 19D.6-9D Electric Power-120V AC Fault Tree Division 4

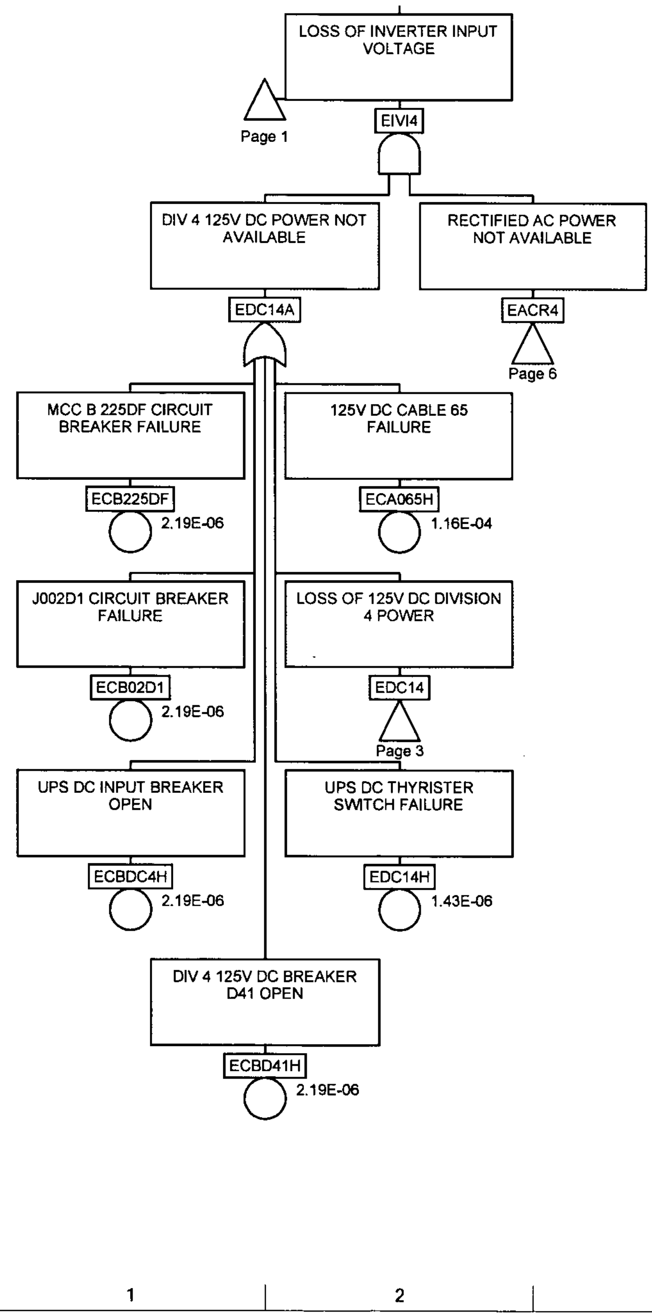


Figure 19D.6-9D Electric Power-120V AC Fault Tree Division 4

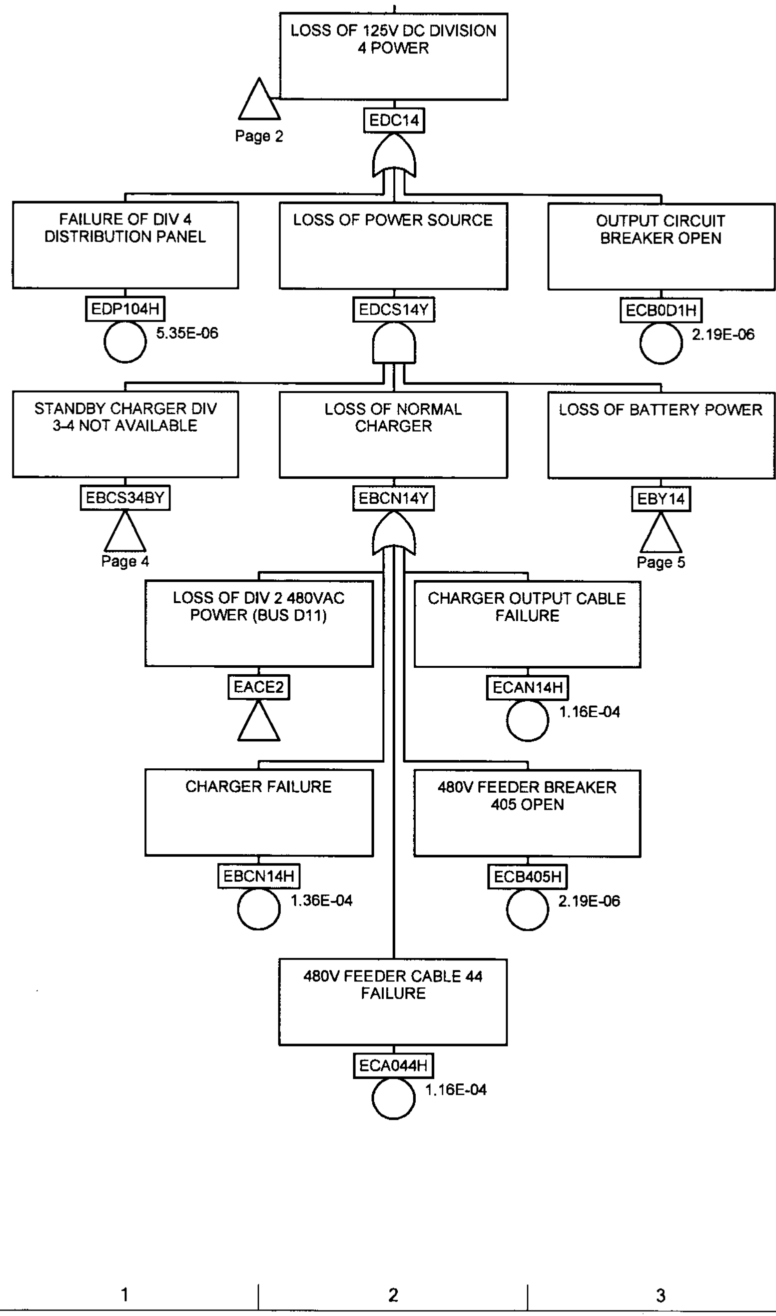


Figure 19D.6-9D Electric Power-120V AC Fault Tree Division 4

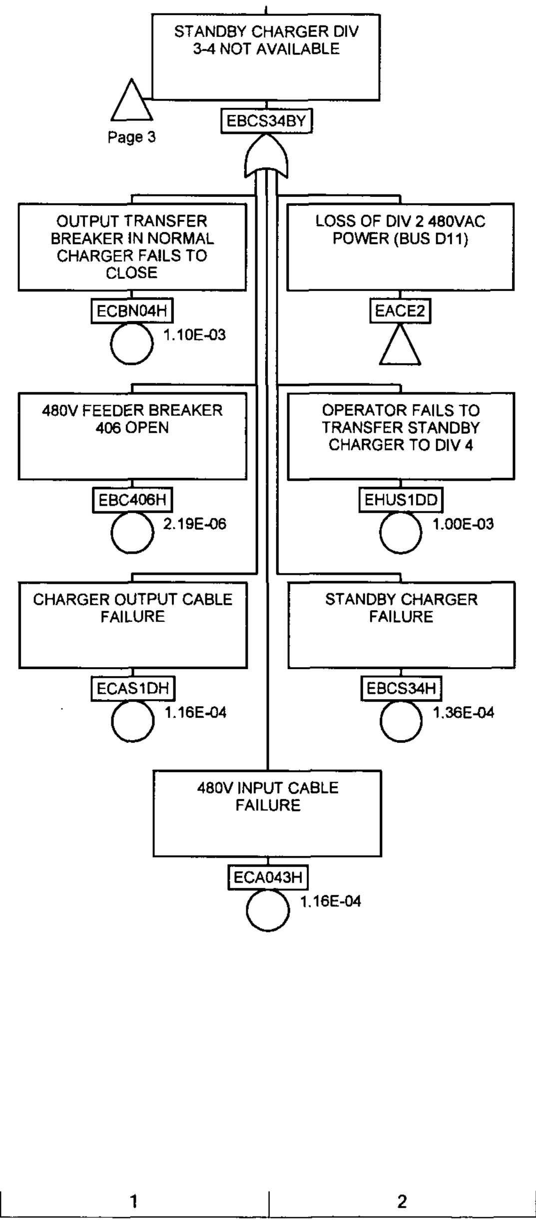


Figure 19D.6-9D Electric Power-120V AC Fault Tree Division 4

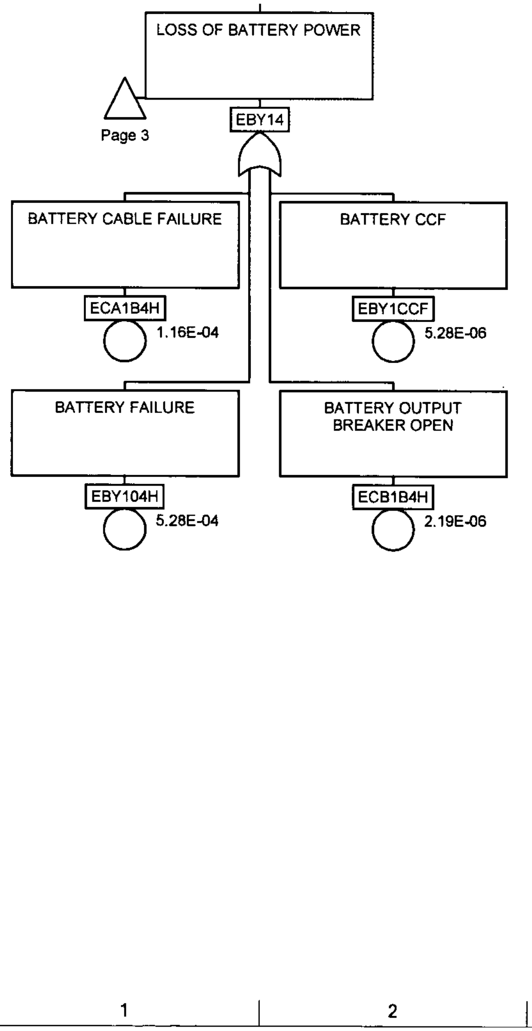


Figure 19D.6-9D Electric Power-120V AC Fault Tree Division 4

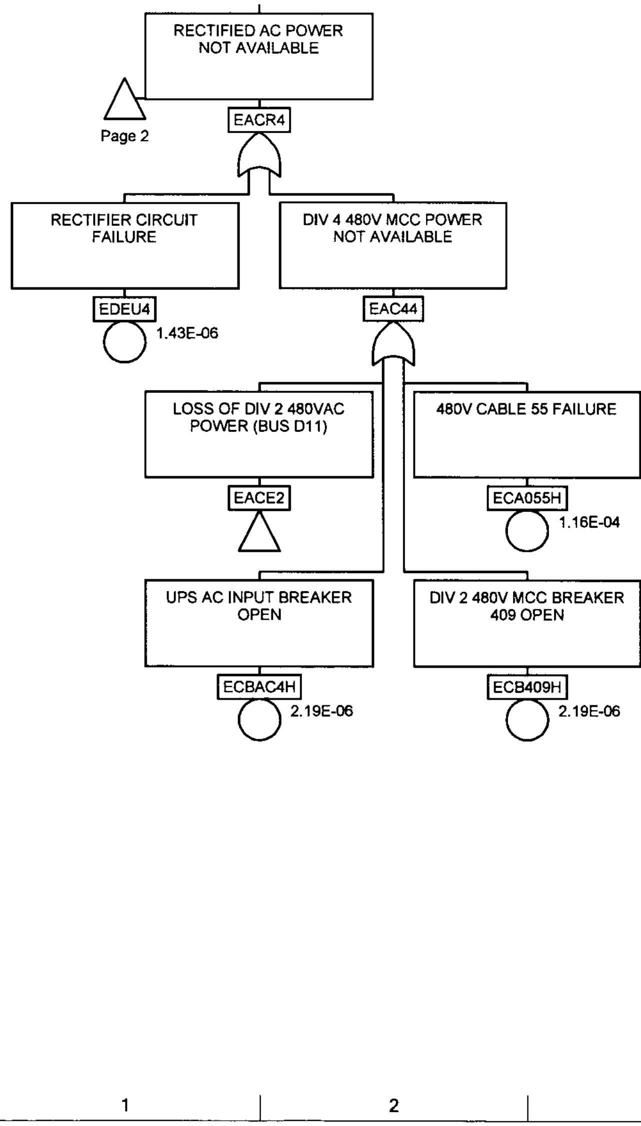


Figure 19D.6-9D Electric Power-120V AC Fault Tree Division 4

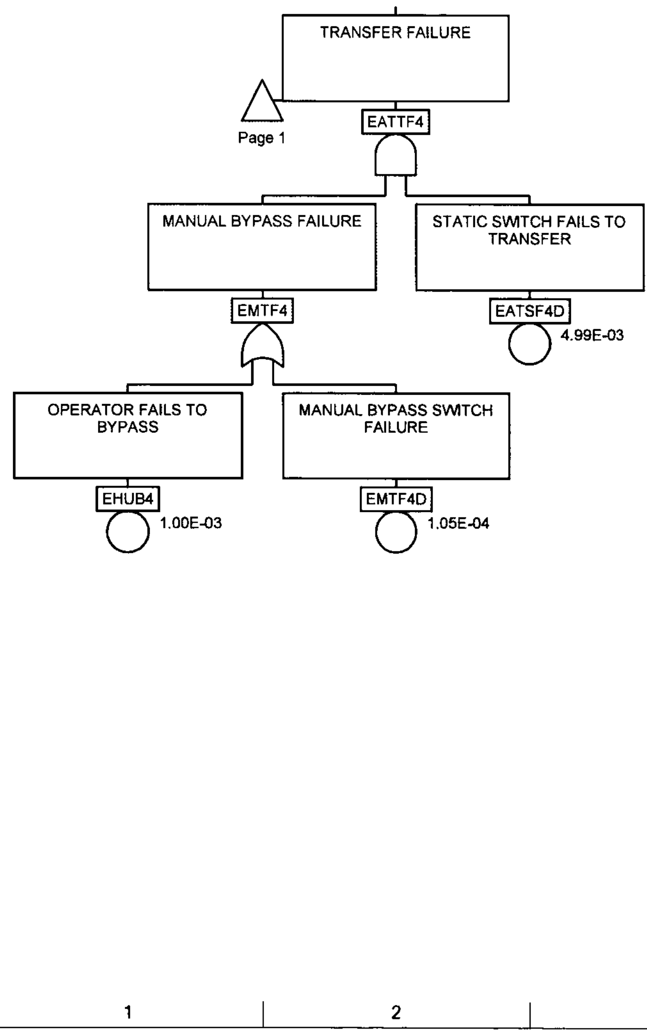


Figure 19D.6-9D Electric Power-120V AC Fault Tree Division 4

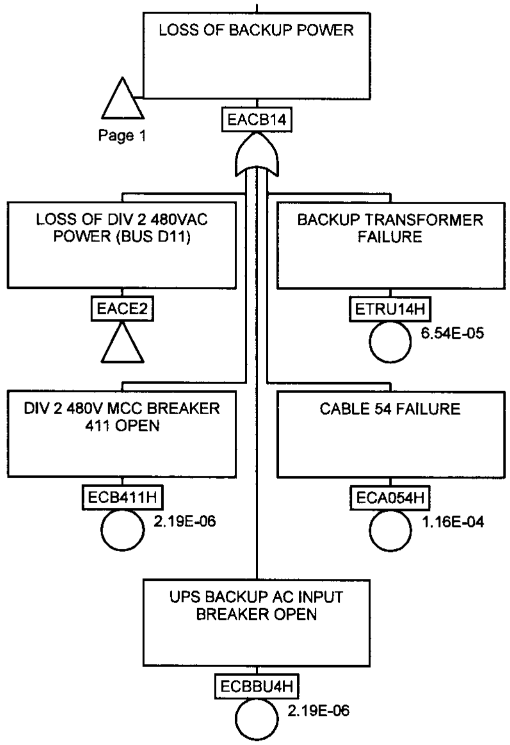


Figure 19D.6-9D Electric Power-120V AC Fault Tree Division 4

Name	Page	Zone	Name	Page	Zone
EAC44	6	2	EDC14	2	2
EACB14	1	4	EDC14	3	2
EACB14	8	2	EDC14A	2	2
EACBP4	1	4	EDC14H	2	2
EACE2	3	2	EDCS14Y	3	2
EACE2	4	2	EDEU4	6	1
EACE2	6	2	EDP104H	3	1
EACE2	8	1	EHUB4	7	1
EACIO4	1	2	EHUS1DD	4	2
EACL14H	1	3	EIVI4	1	1
EACL4	1	2	EIVI4	2	2
EACLS4	1	2	EIVOF4H	1	2
EACR4	2	3	EMTF4	7	2
EACR4	6	2	EMTF4D	7	2
EATSF4D	7	3	ETRU14H	8	2
EATSF4H	1	1			
EATTF4	1	3			
EATTF4	7	2			
EBC406H	4	1			
EBCN14H	3	2			
EBCN14Y	3	2			
EBCS34BY	3	1			
EBCS34BY	4	2			
EBCS34H	4	2			
EBY104H	5	1			
EBY14	3	3			
EBY14	5	2			
EBY1CCF	5	2			
ECA043H	4	2			
ECA044H	3	2			
ECA054H	8	2			
ECA055H	6	3			
ECA065H	2	2			
ECA1B4H	5	1			
ECAN14H	3	3			
ECAS1DH	4	1			
ECB02D1	2	1			
ECB0D1H	3	3			
ECB1B4H	5	2			
ECB225DF	2	1			
ECB405H	3	3			
ECB409H	6	3			
ECB411H	8	1			
ECBAC4H	6	2			
ECBBU4H	8	2			
ECBD41H	2	2			
ECBDC4H	2	1			
ECBN04H	4	1			

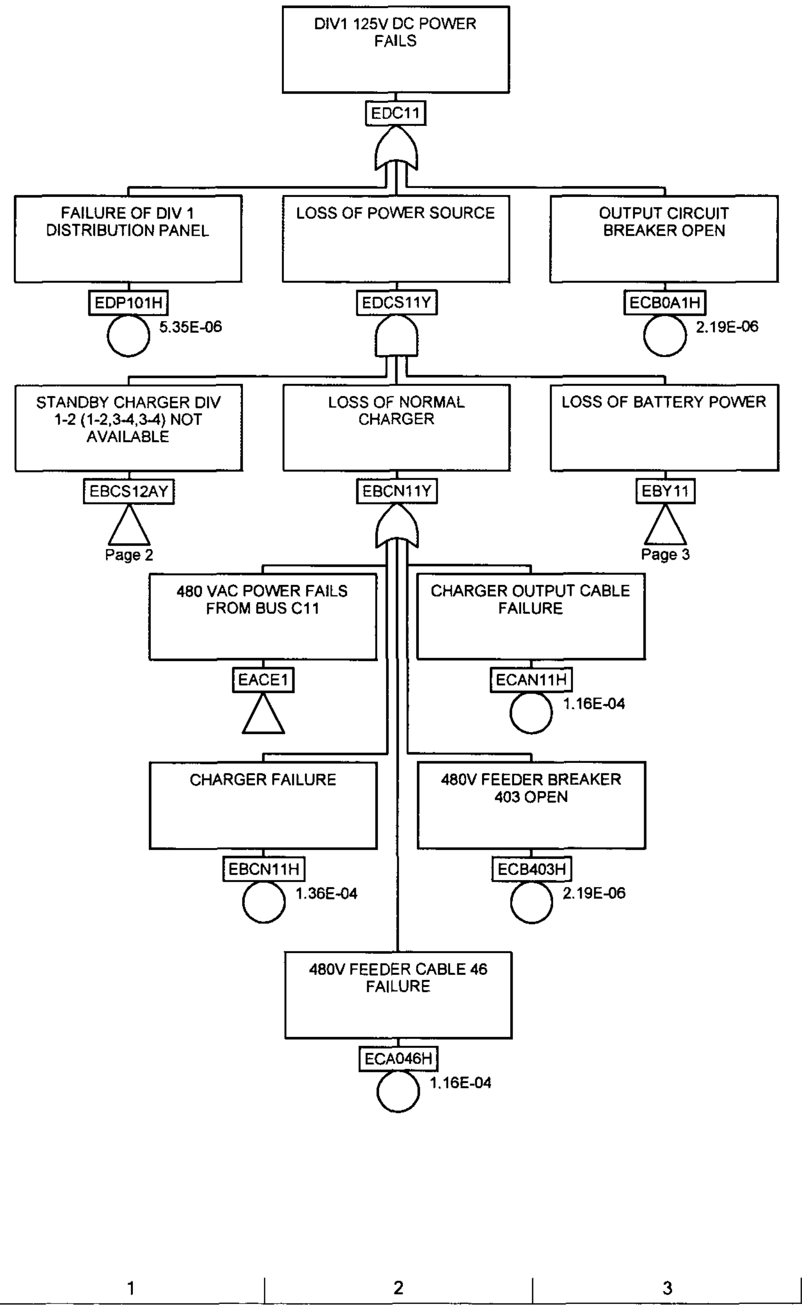


Figure 19D.6-10A Electric Power-125V DC Fault Tree Division 1

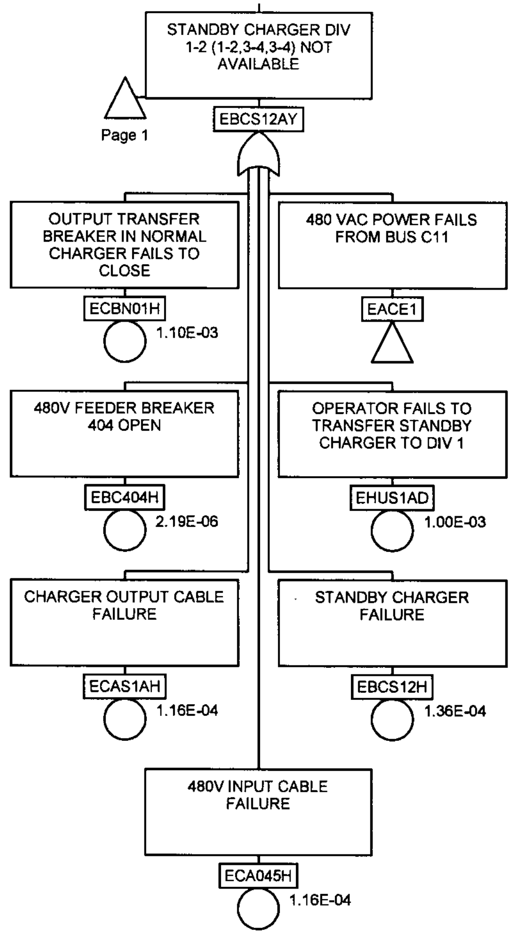


Figure 19D.6-10A Electric Power-125V DC Fault Tree Division 1

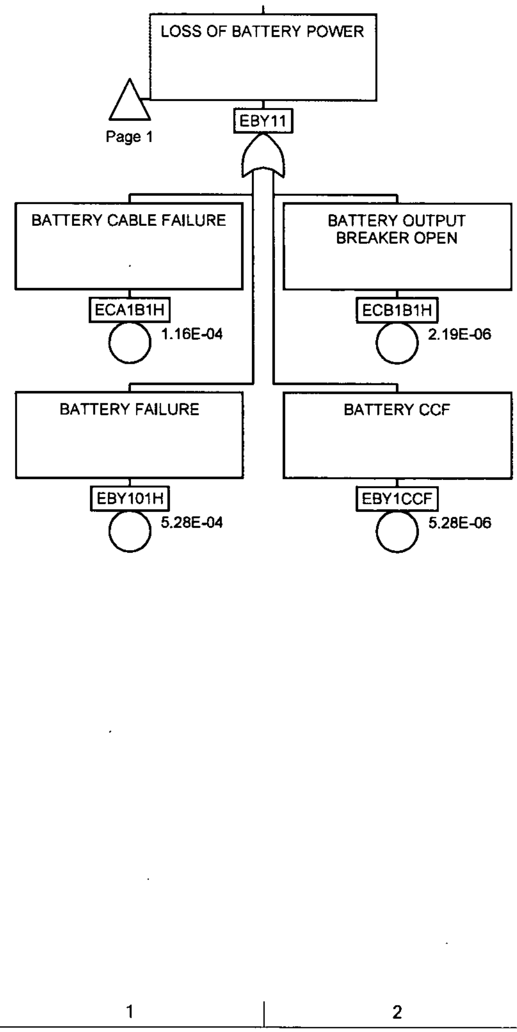


Figure 19D.6-10A Electric Power-125V DC Fault Tree Division 1

Name	Page	Zone	Name	Page	Zone
EACE1	1	2			
EACE1	2	2			
EBC404H	2	1			
EBCN11H	1	2			
EBCN11Y	1	2			
EBCS12AY	1	1			
EBCS12AY	2	2			
EBCS12H	2	2			
EBY101H	3	1			
EBY11	1	3			
EBY11	3	2			
EBY1CCF	3	2			
ECA045H	2	2			
ECA046H	1	2			
ECA1B1H	3	1			
ECAN11H	1	3			
ECAS1AH	2	1			
ECB0A1H	1	3			
ECB1B1H	3	2			
ECB403H	1	3			
ECBN01H	2	1			
EDC11	1	2			
EDCS11Y	1	2			
EDP101H	1	1			
EHUS1AD	2	2			

Figure 19D.6-10A Electric Power-125V DC Fault Tree Division 1

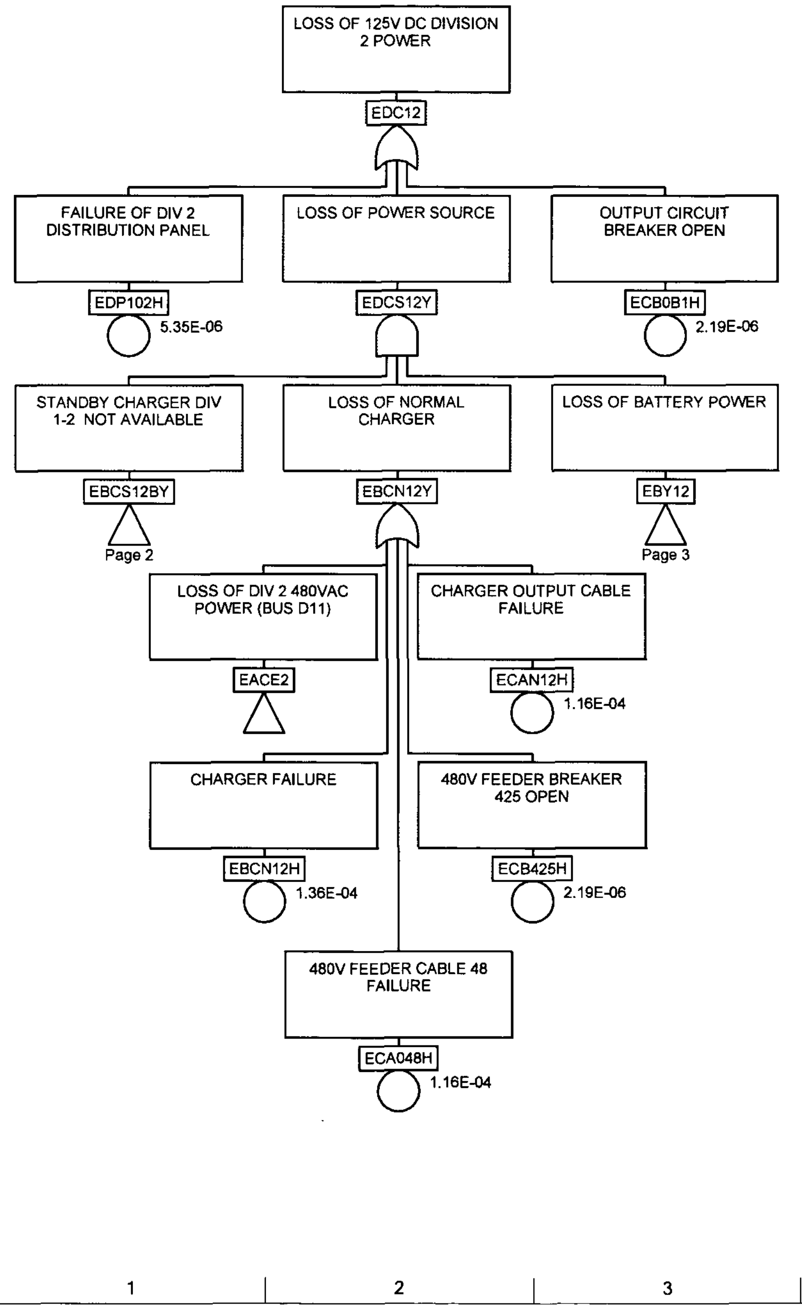


Figure 19D.6-10B Electric Power-125V DC Fault Tree Division 2

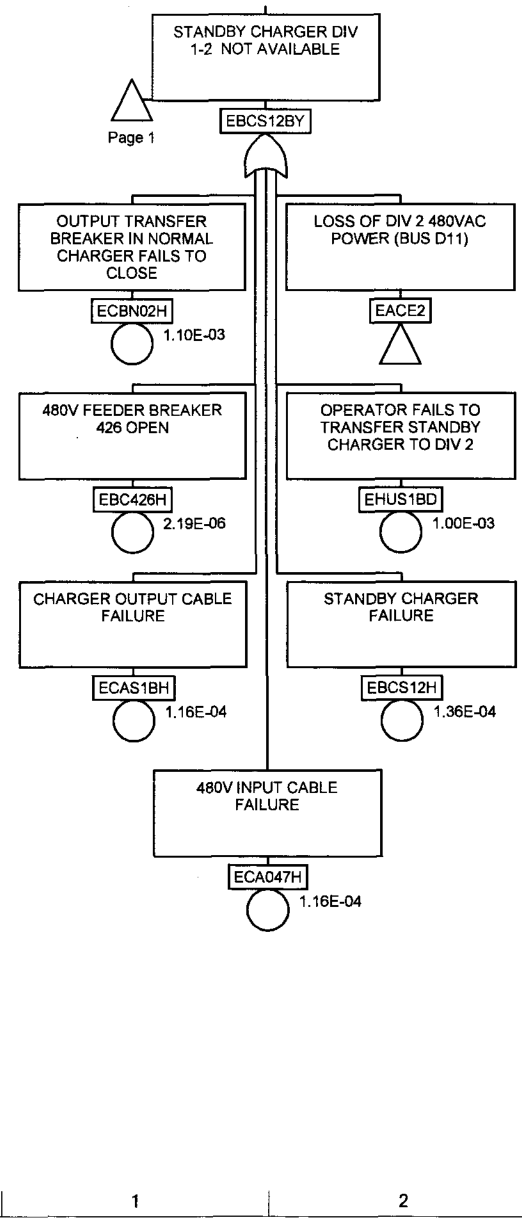


Figure 19D.6-10B Electric Power-125V DC Fault Tree Division 2

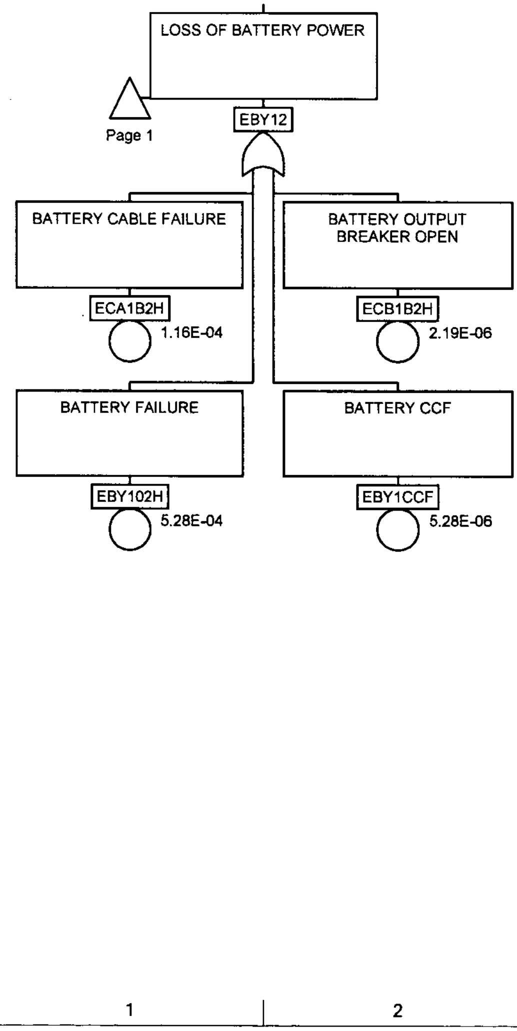
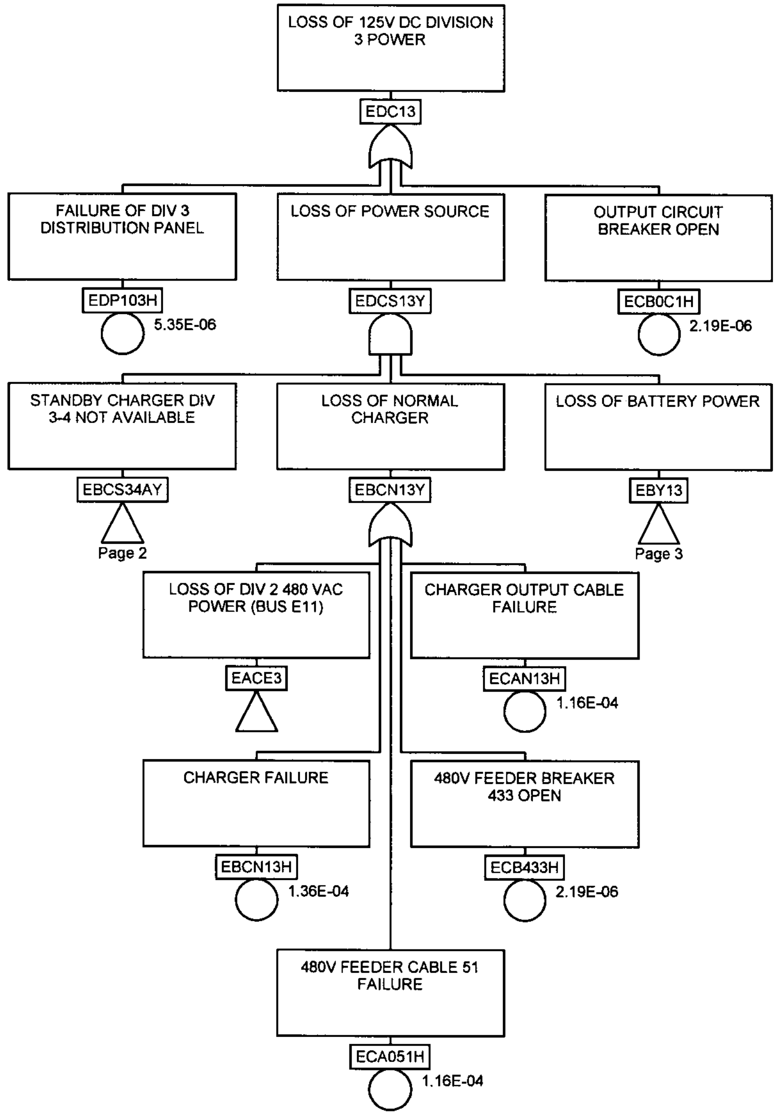


Figure 19D.6-10B Electric Power-125V DC Fault Tree Division 2

Name	Page	Zone	Name	Page	Zone
EACE2	1	2			
EACE2	2	2			
EBC426H	2	1			
EBCN12H	1	2			
EBCN12Y	1	2			
EBCS12BY	1	1			
EBCS12BY	2	2			
EBCS12H	2	2			
EBY102H	3	1			
EBY12	1	3			
EBY12	3	2			
EBY1CCF	3	2			
ECA047H	2	2			
ECA048H	1	2			
ECA1B2H	3	1			
ECAN12H	1	3			
ECAS1BH	2	1			
ECB0B1H	1	3			
ECB1B2H	3	2			
ECB425H	1	3			
ECBN02H	2	1			
EDC12	1	2			
EDCS12Y	1	2			
EDP102H	1	1			
EHUS1BD	2	2			



1 2 3

Figure 19D.6-10C Electric Power-125 VDC Fault Tree Division 3

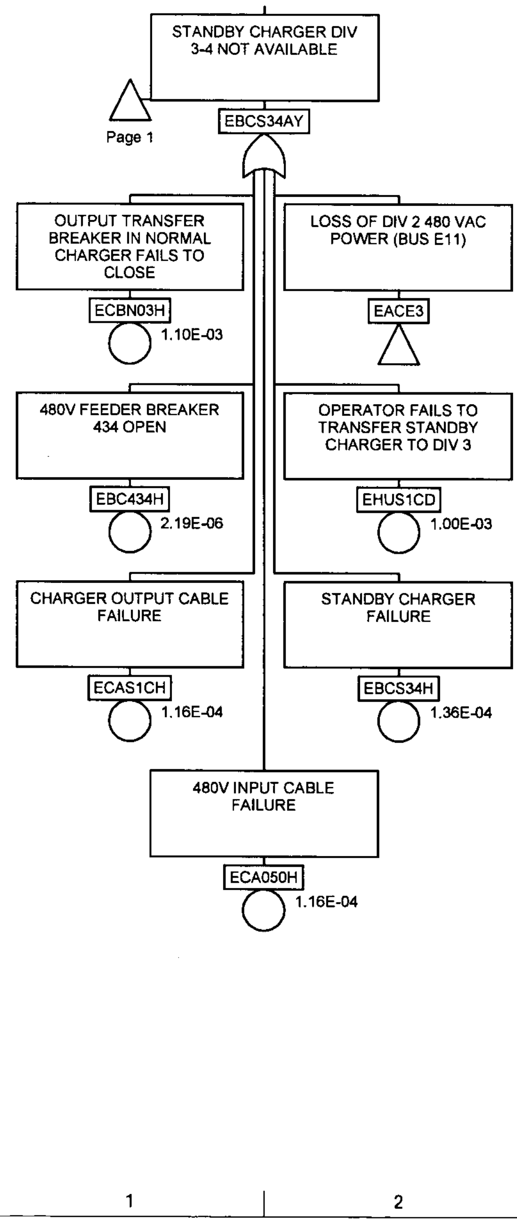


Figure 19D.6-10C Electric Power-125 VDC Fault Tree Division 3

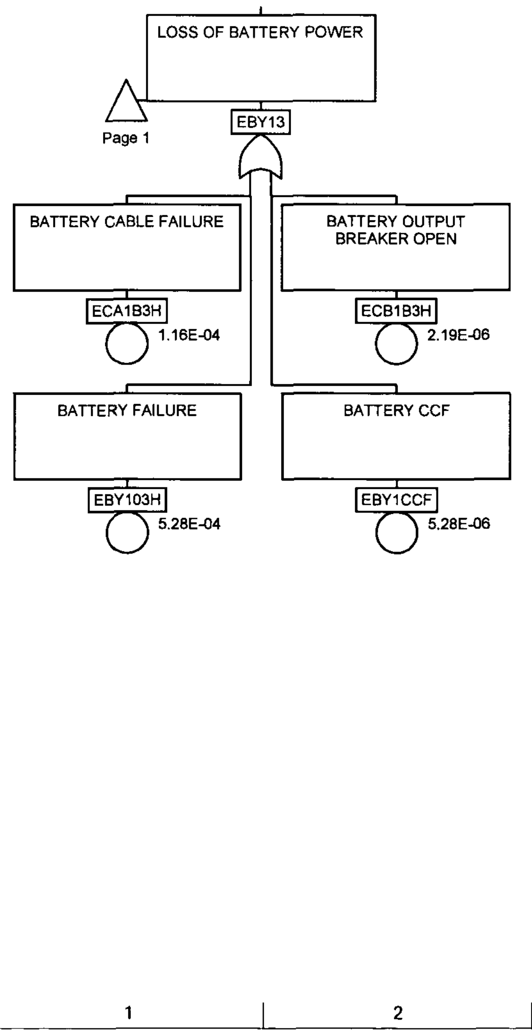


Figure 19D.6-10C Electric Power-125 VDC Fault Tree Division 3

Name	Page	Zone	Name	Page	Zone
EACE3	1	2			
EACE3	2	2			
EBC434H	2	1			
EBCN13H	1	2			
EBCN13Y	1	2			
EBCS34AY	1	1			
EBCS34AY	2	2			
EBCS34H	2	2			
EBY103H	3	1			
EBY13	1	3			
EBY13	3	2			
EBY1CCF	3	2			
ECA050H	2	2			
ECA051H	1	2			
ECA1B3H	3	1			
ECAN13H	1	3			
ECAS1CH	2	1			
ECB0C1H	1	3			
ECB1B3H	3	2			
ECB433H	1	3			
ECBN03H	2	1			
EDC13	1	2			
EDCS13Y	1	2			
EDP103H	1	1			
EHUS1CD	2	2			

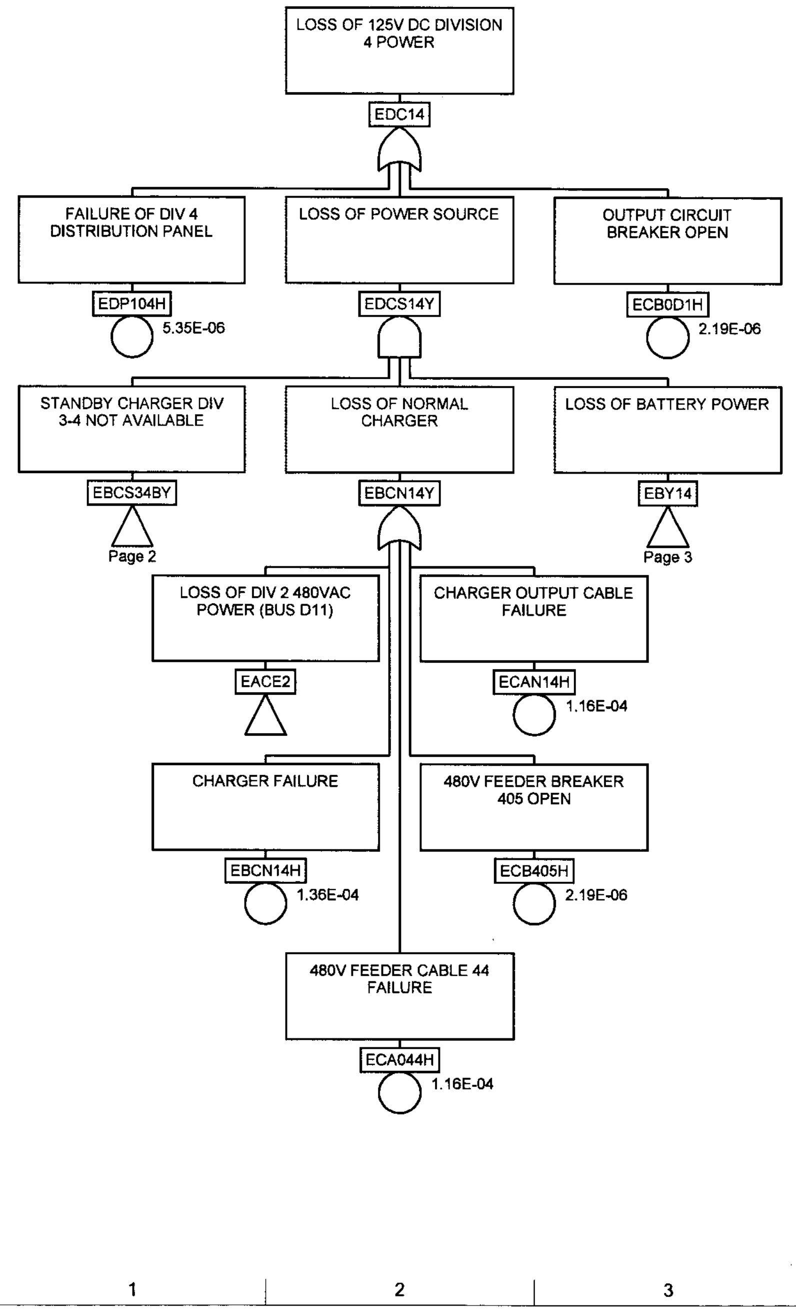


Figure 19D.6-10D Electric Power-125V DC Fault Tree Division 4

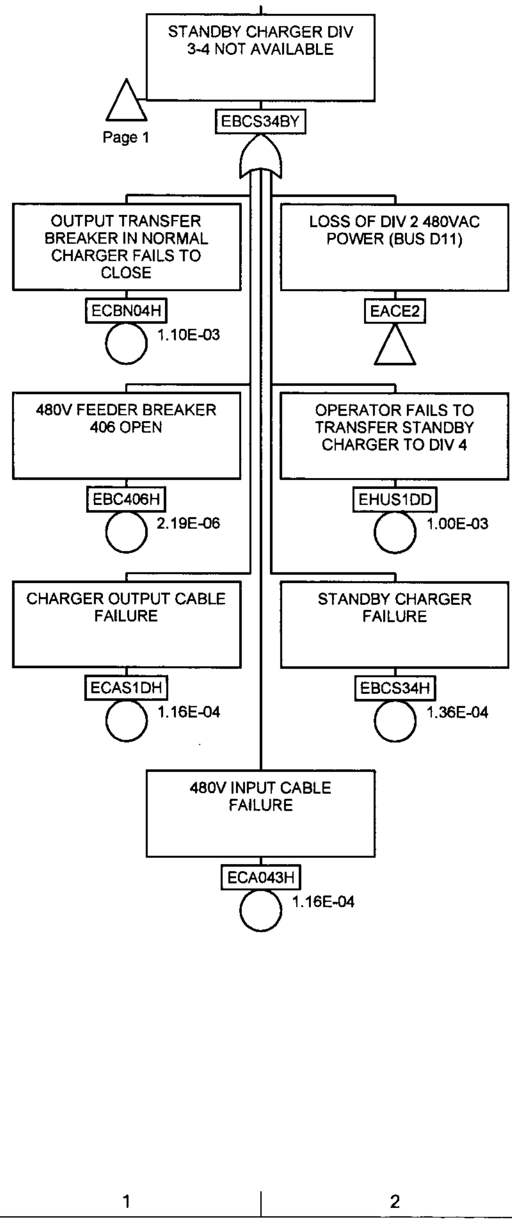


Figure 19D.6-10D Electric Power-125V DC Fault Tree Division 4

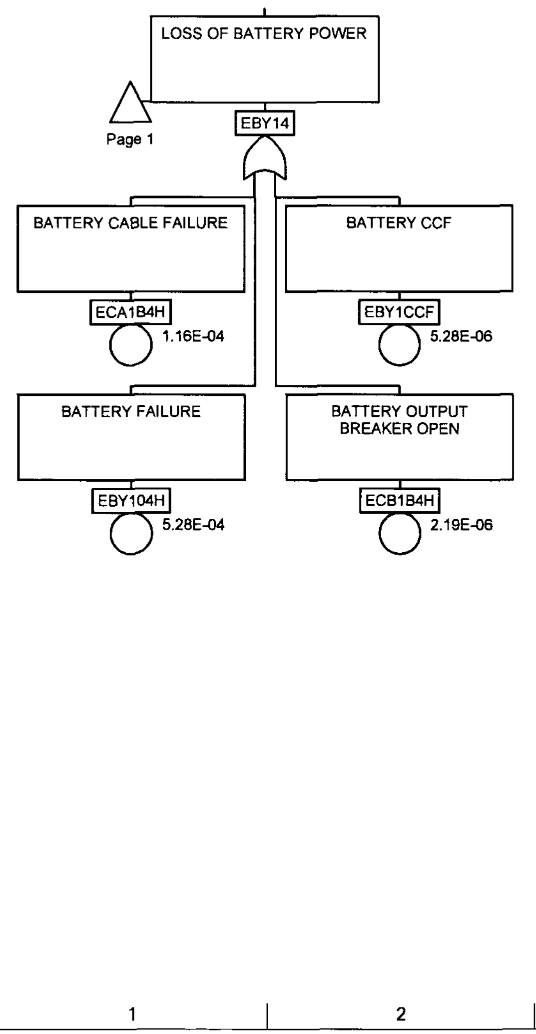


Figure 19D.6-10D Electric Power-125V DC Fault Tree Division 4

Name	Page	Zone	Name	Page	Zone
EACE2	1	2			
EACE2	2	2			
EBC406H	2	1			
EBCN14H	1	2			
EBCN14Y	1	2			
EBCS34BY	1	1			
EBCS34BY	2	2			
EBCS34H	2	2			
EBY104H	3	1			
EBY14	1	3			
EBY14	3	2			
EBY1CCF	3	2			
ECA043H	2	2			
ECA044H	1	2			
ECA1B4H	3	1			
ECAN14H	1	3			
ECAS1DH	2	1			
ECB0D1H	1	3			
ECB1B4H	3	2			
ECB405H	1	3			
ECBN04H	2	1			
EDC14	1	2			
EDCS14Y	1	2			
EDP104H	1	1			
EHUS1DD	2	2			

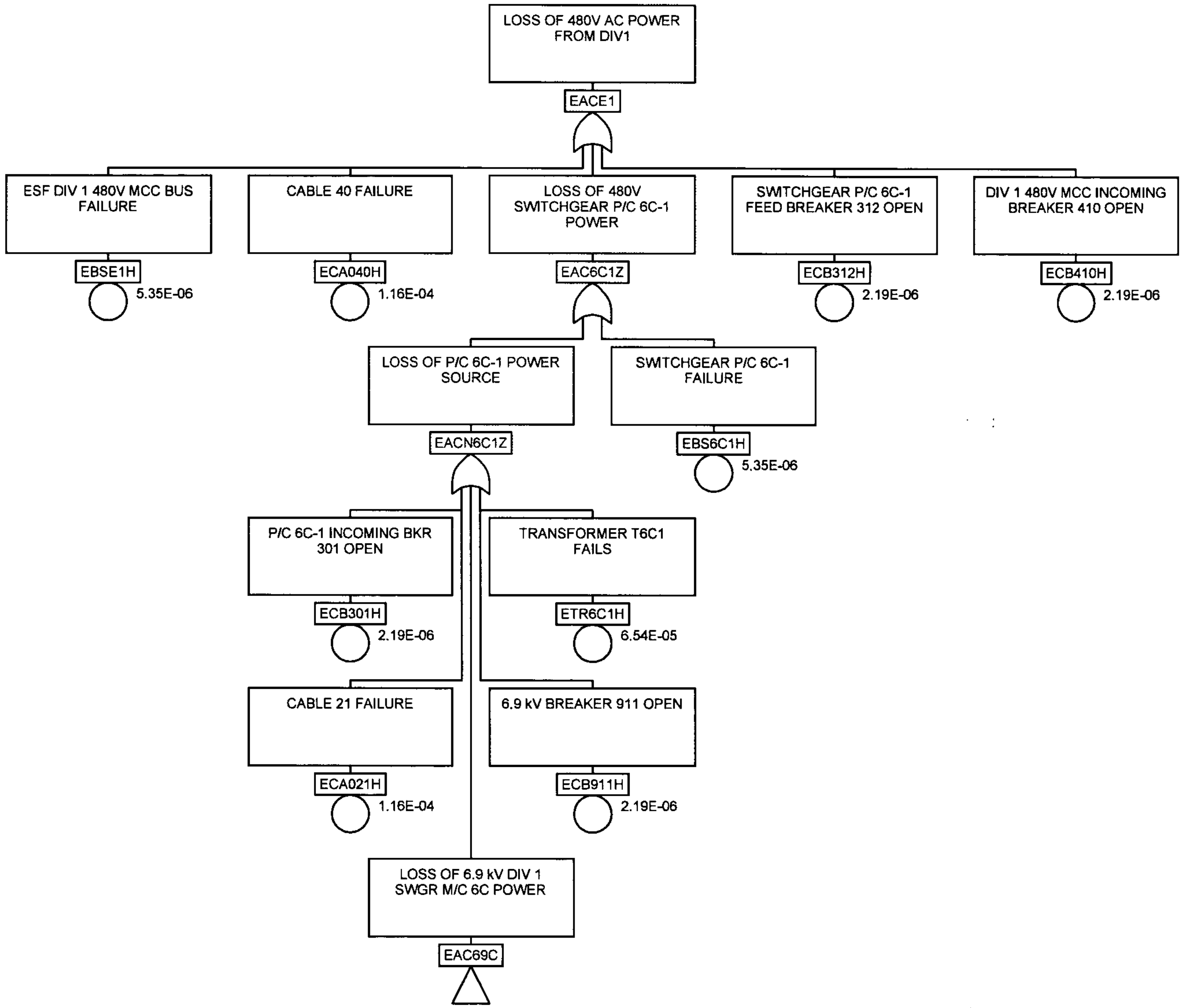


Figure 19D.6-11A Electric Power-480V MCC Fault Tree Division 1

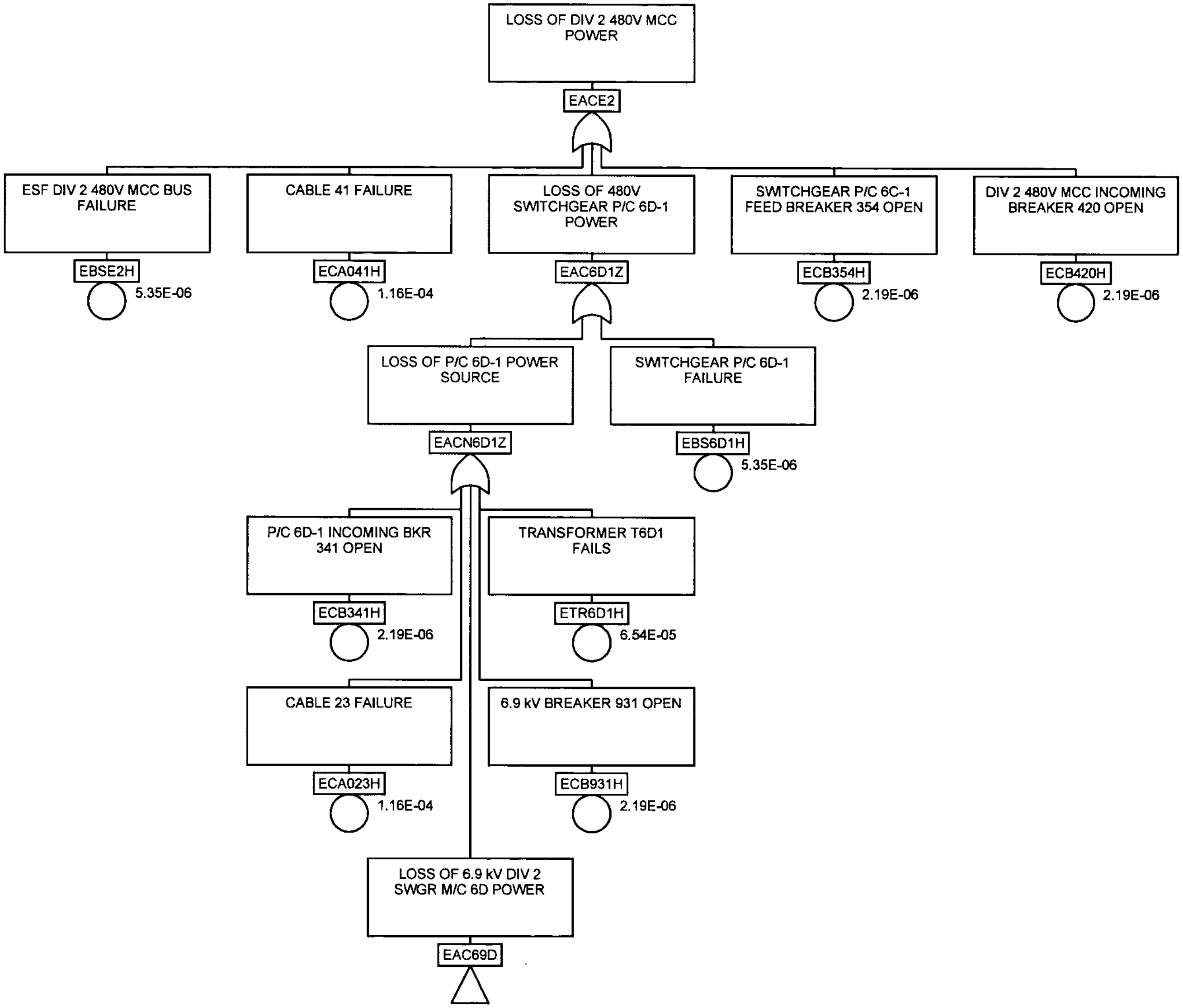


Figure 19D.6-11B Electric Power- 480V MCC Fault Tree Division 2

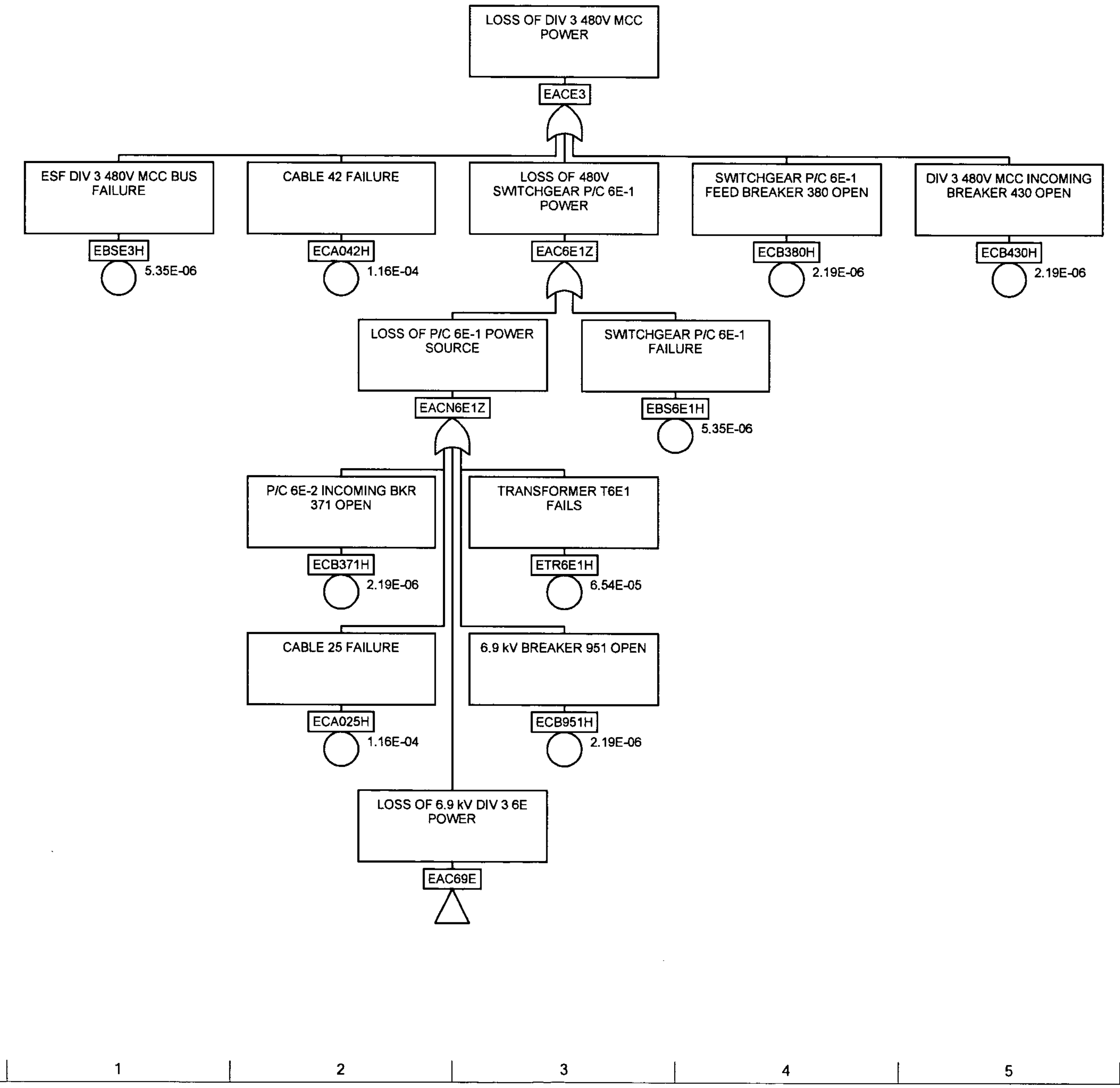


Figure 19D.6-11C Electric Power-480V MCC Fault Tree Division 3

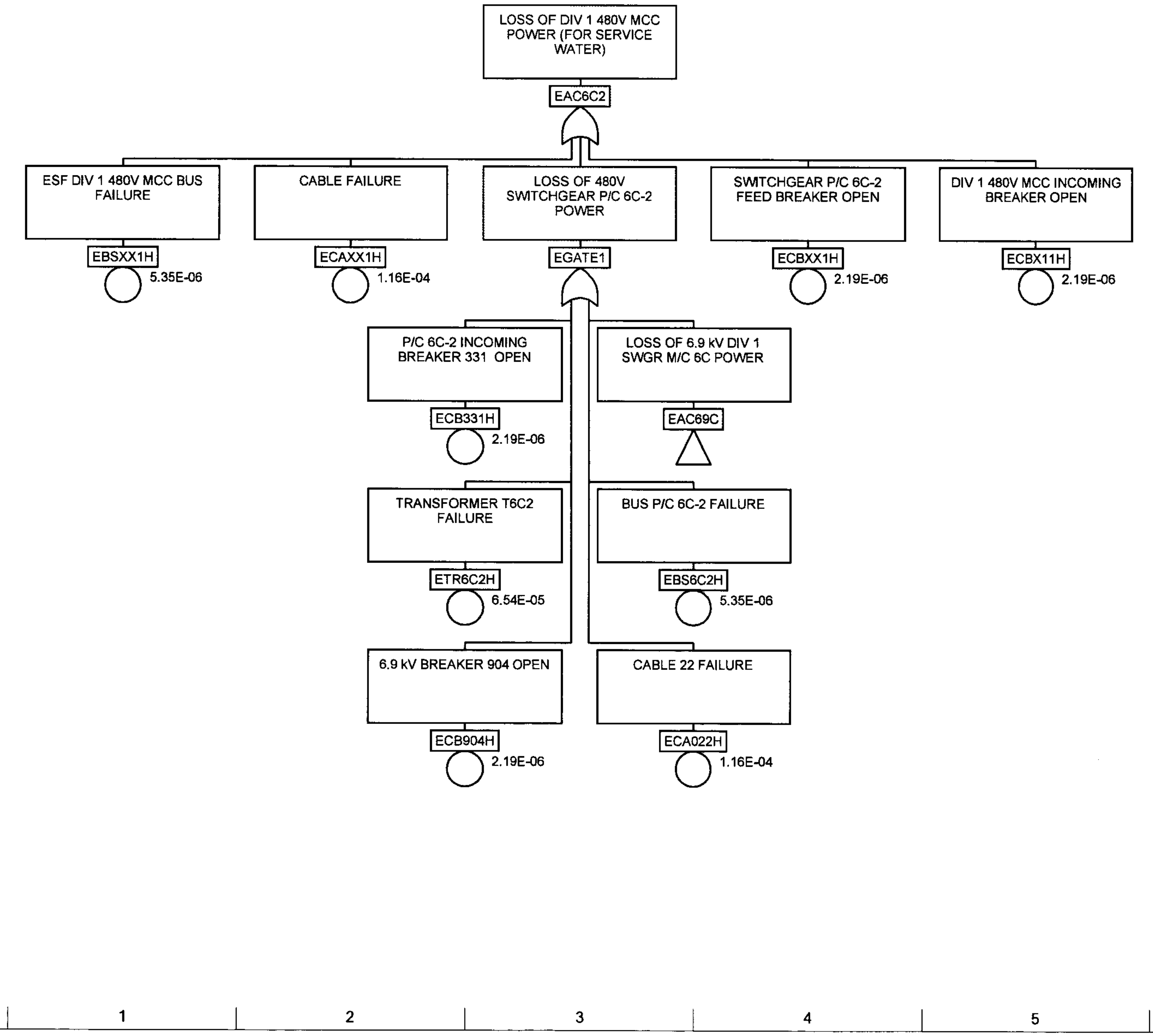


Figure 19D.6-12A Electric Power-480V Switchgear Fault Tree Division 1

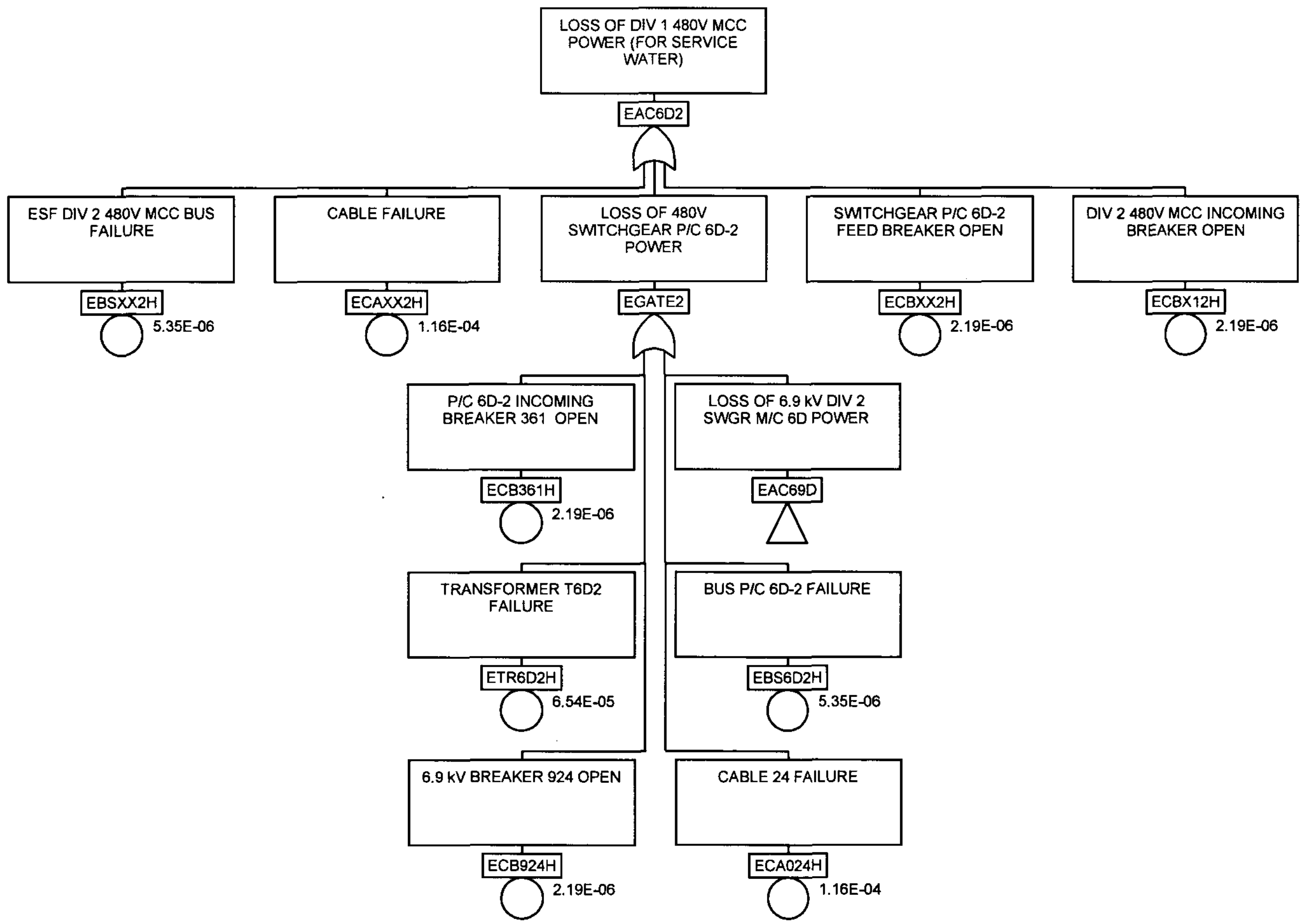


Figure 19D.6-12B Electric Power-480V Switchgear Fault Tree Division 2

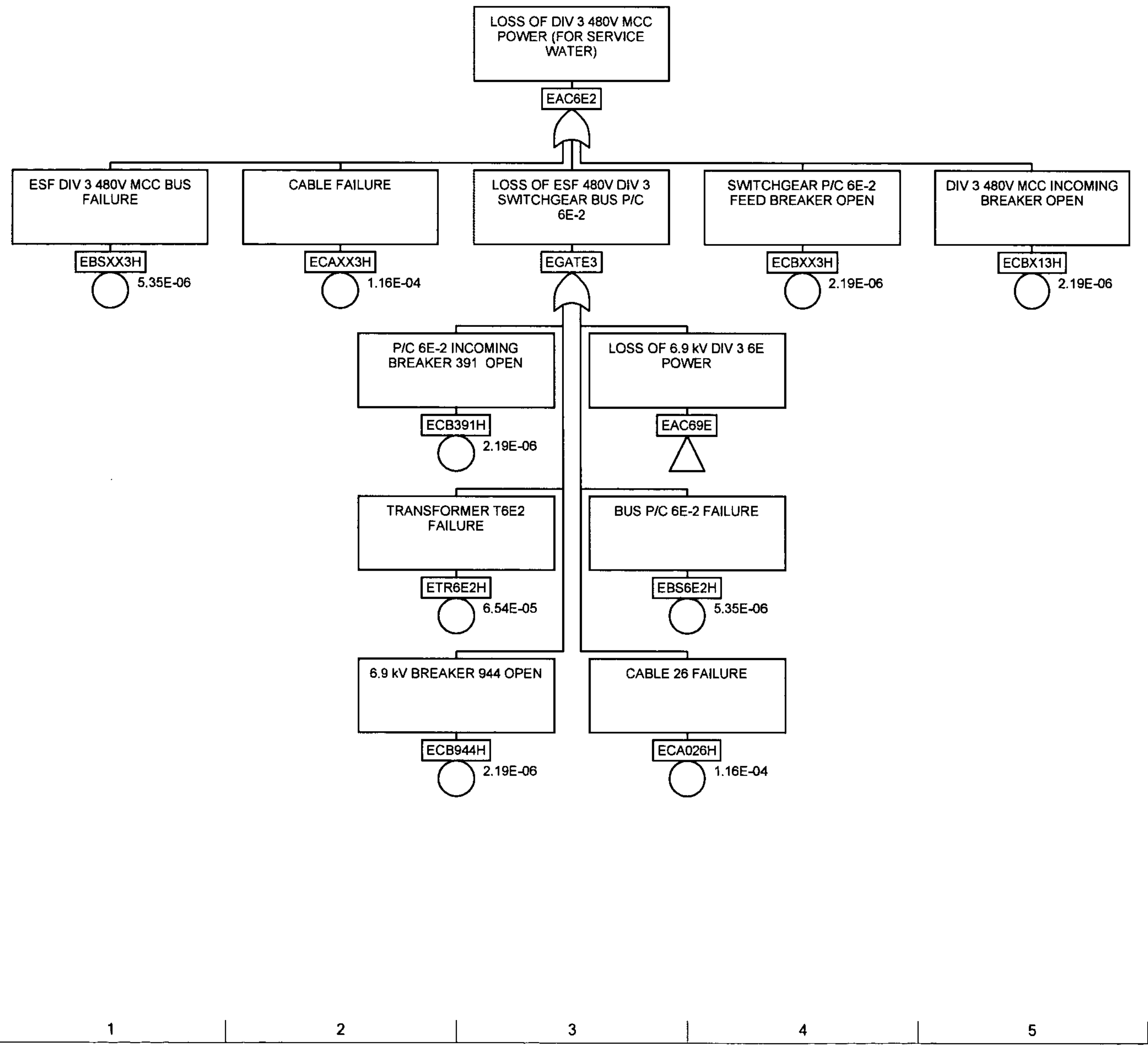


Figure 19D.6-12C Electric Power-480V Switchgear Fault Tree Division 3

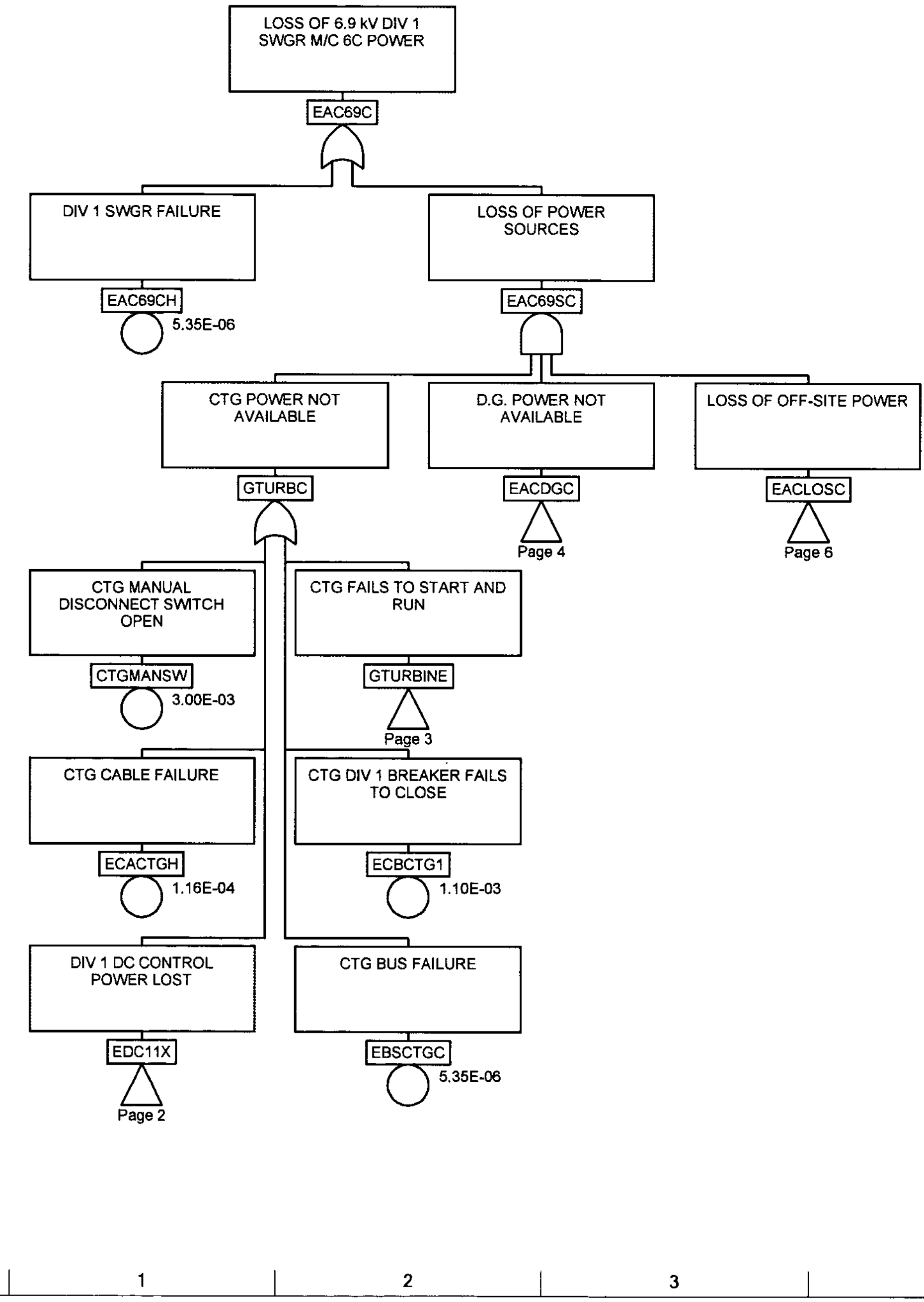


Figure 19D.6-13A Electric Power-6.9kV Switchgear Fault Tree Division 1

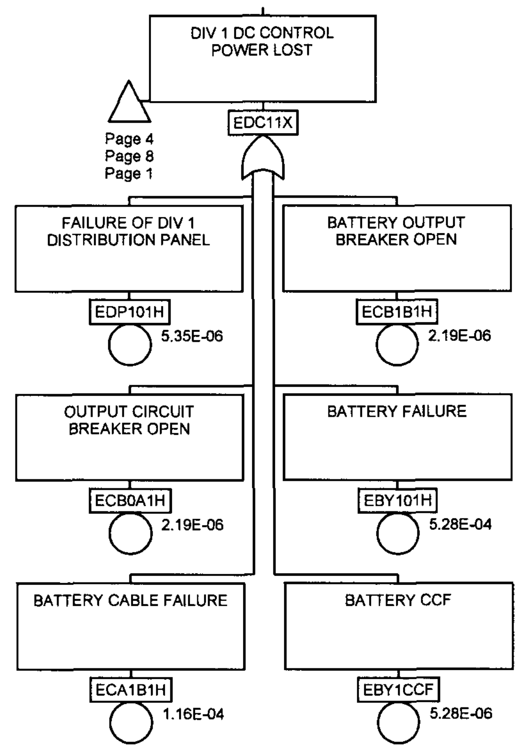
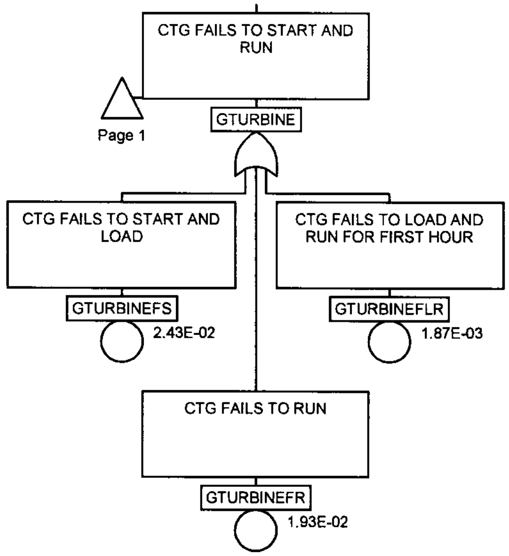


Figure 19D.6-13A Electric Power-6.9kV Switchgear Fault Tree Division 1



1 2

Figure 19D.6-13A Electric Power-6.9kV Switchgear Fault Tree Division 1

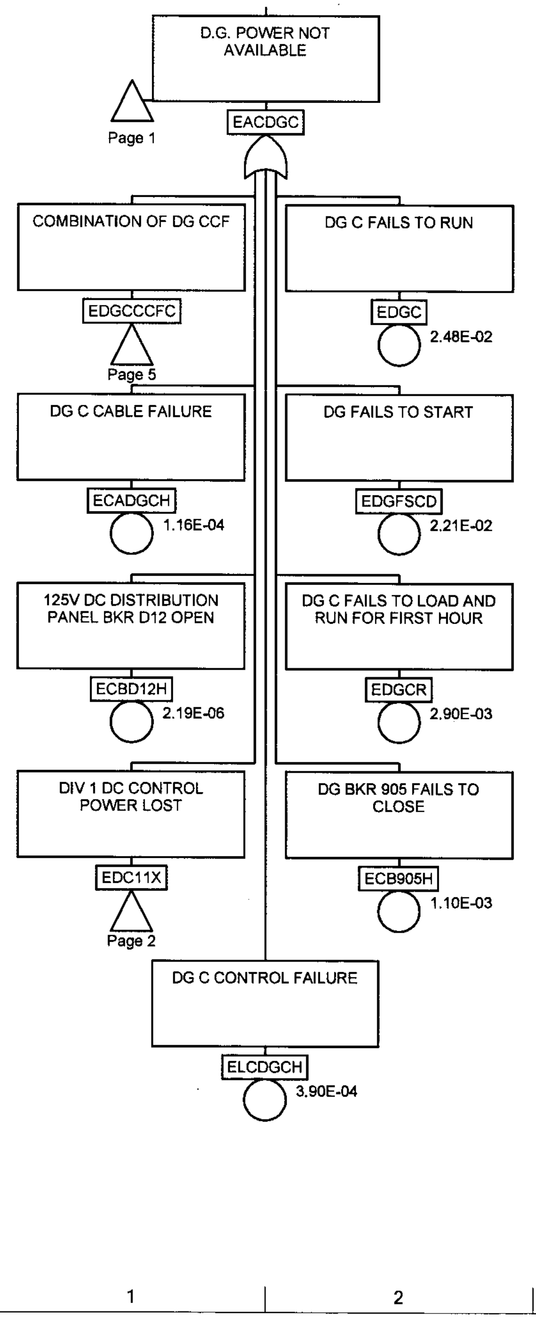


Figure 19D.6-13A Electric Power-6.9kV Switchgear Fault Tree Division 1

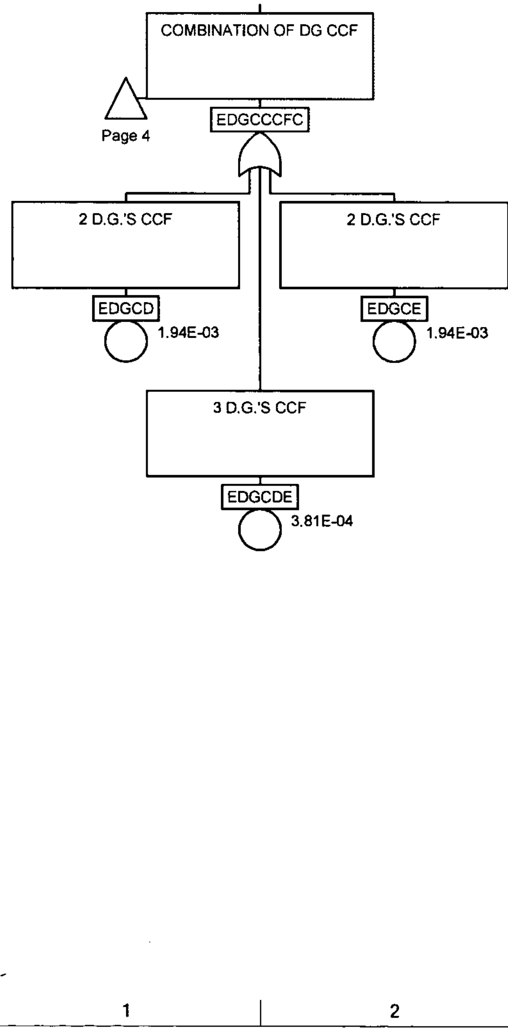


Figure 19D.6-13A Electric Power-6.9kV Switchgear Fault Tree Division 1

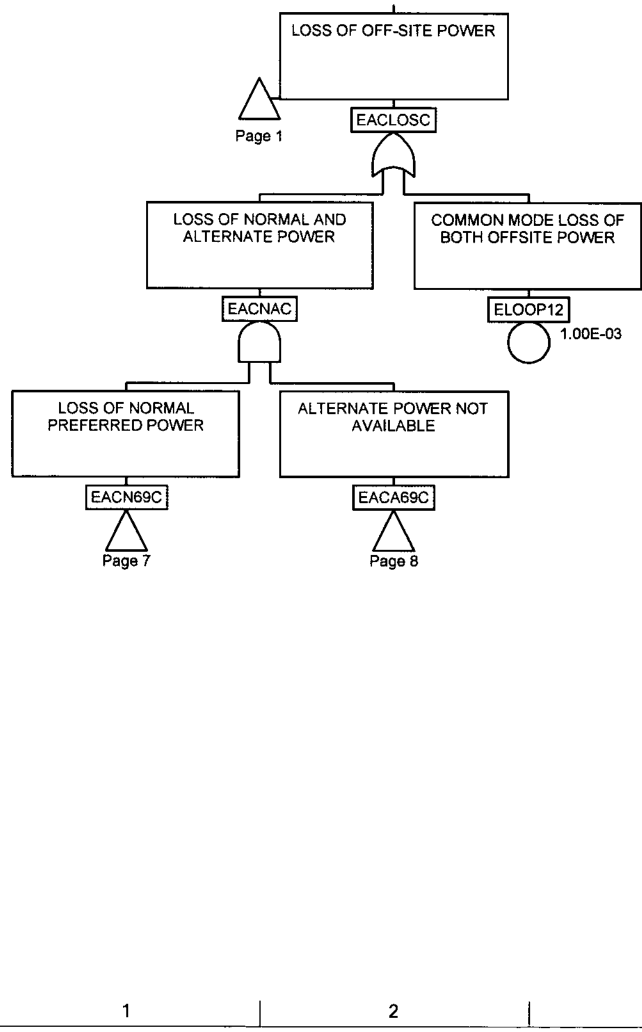


Figure 19D.6-13A Electric Power-6.9kV Switchgear Fault Tree Division 1

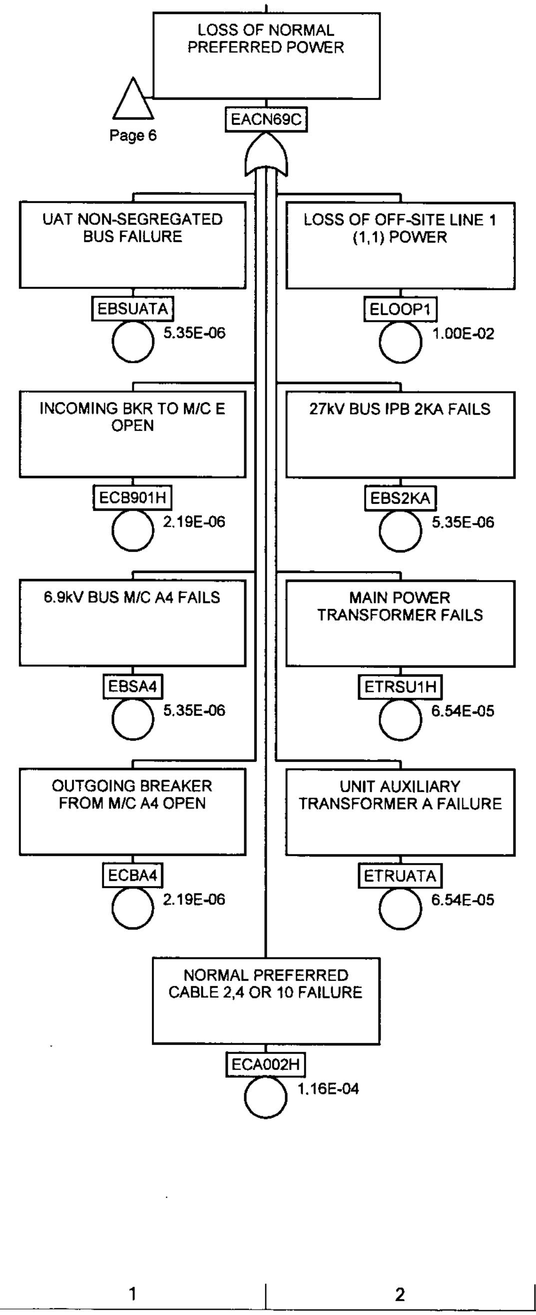


Figure 19D.6-13A Electric Power-6.9kV Switchgear Fault Tree Division 1

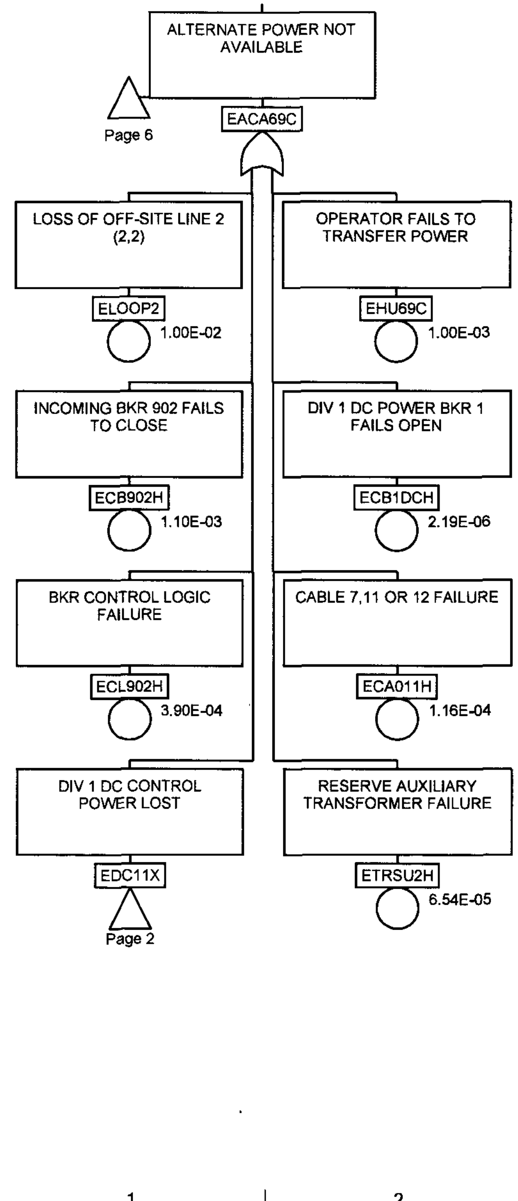


Figure 19D.6-13A Electric Power-6.9kV Switchgear Fault Tree Division 1

Name	Page	Zone
CTGMANSW	1	1
EAC69C	1	2
EAC69CH	1	1
EAC69SC	1	2
EACA69C	6	2
EACA69C	8	2
EACDGC	1	3
EACDGC	4	2
EACLOSC	1	4
EACLOSC	6	2
EACN69C	6	1
EACN69C	7	2
EACNAC	6	2
EBS2KA	7	2
EBSA4	7	1
EBSCTGC	1	2
EBSUATA	7	1
EBY101H	2	2
EBY1CCF	2	2
ECA002H	7	2
ECA011H	8	2
ECA1B1H	2	1
ECACTGH	1	1
ECADGCH	4	1
ECB0A1H	2	1
ECB1B1H	2	2
ECB1DCH	8	2
ECB901H	7	1
ECB902H	8	1
ECB905H	4	2
ECBA4	7	1
ECBCTG1	1	2
ECBD12H	4	1
ECL902H	8	1
EDC11X	1	1
EDC11X	2	2
EDC11X	4	1
EDC11X	8	1
EDGC	4	2
EDGCCCFC	4	1
EDGCCCFC	5	2
EDGCD	5	1
EDGCDE	5	2
EDGCE	5	2
EDGCR	4	2
EDGFSCD	4	2
EDP101H	2	1
EHU69C	8	2

Name	Page	Zone
ELCDGCH	4	2
ELOOP1	7	2
ELOOP12	6	3
ELOOP2	8	1
ETRSU1H	7	2
ETRSU2H	8	2
ETRUATA	7	2
GTURBC	1	2
GTURBINE	1	2
GTURBINE	3	2
GTURBINEFLR	3	2
GTURBINEFR	3	2
GTURBINEFS	3	1

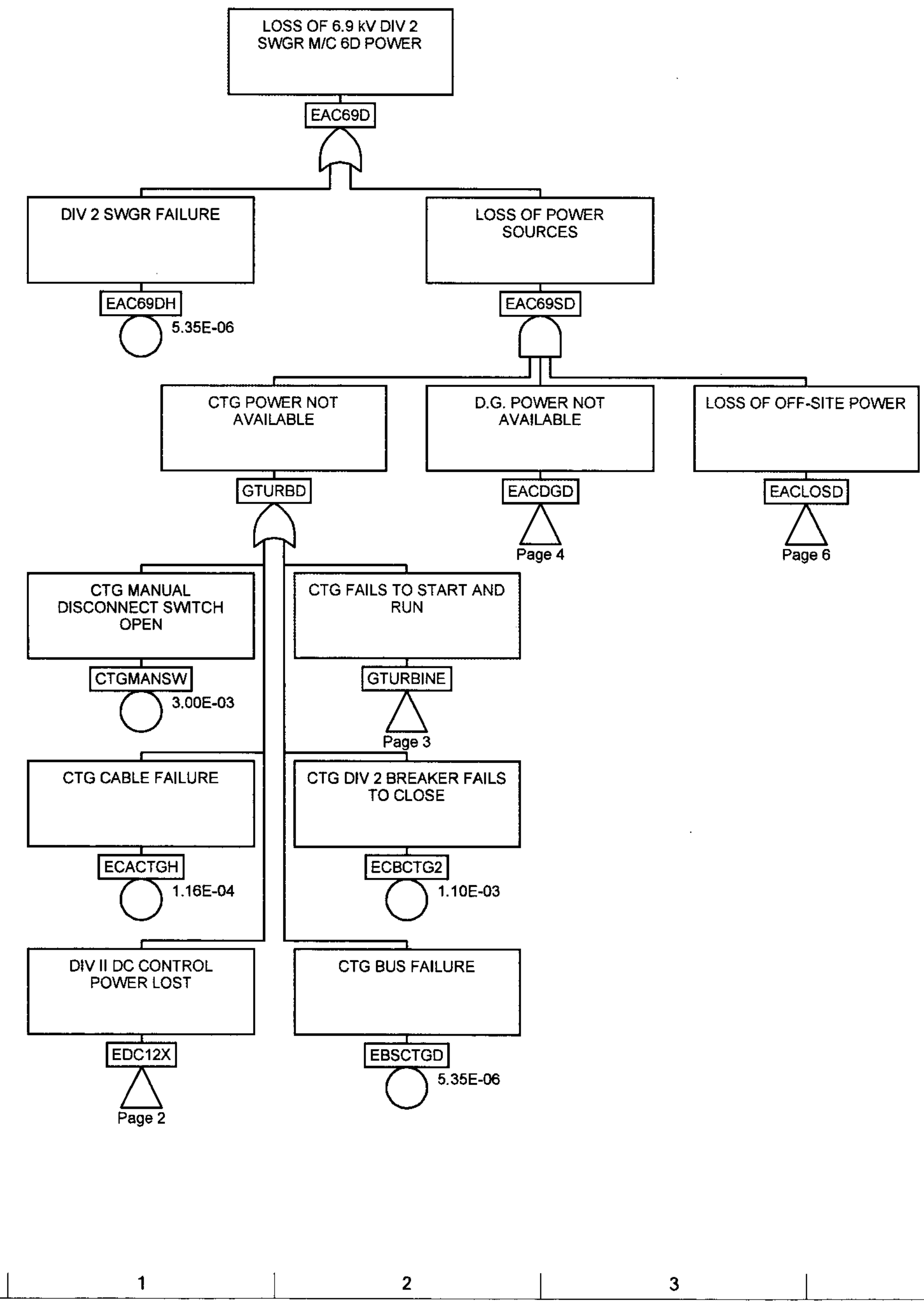


Figure 19D.6-13B Electric Power-6.9kV Switchgear Fault Tree Division 2

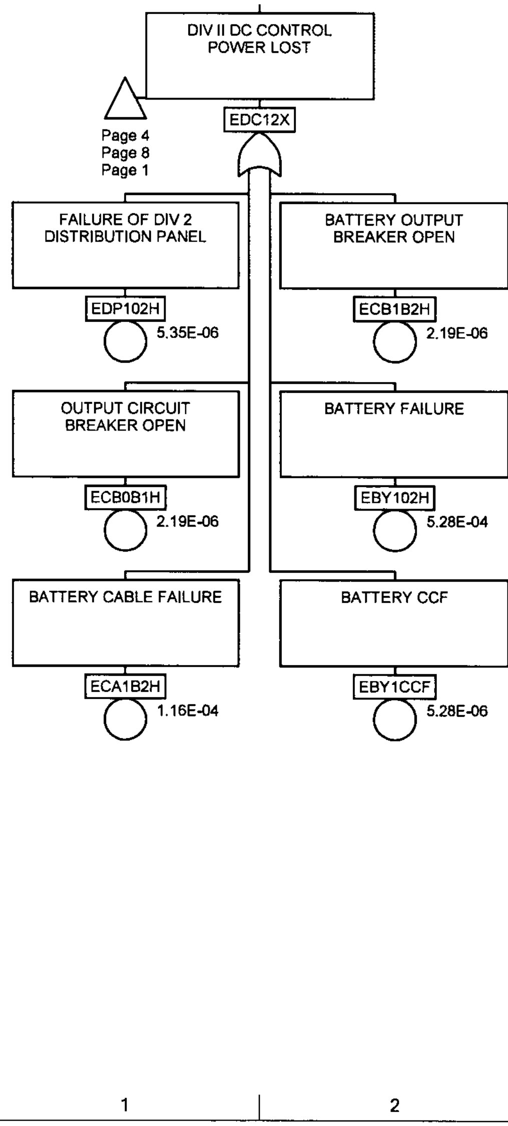


Figure 19D.6-13B Electric Power-6.9kV Switchgear Fault Tree Division 2

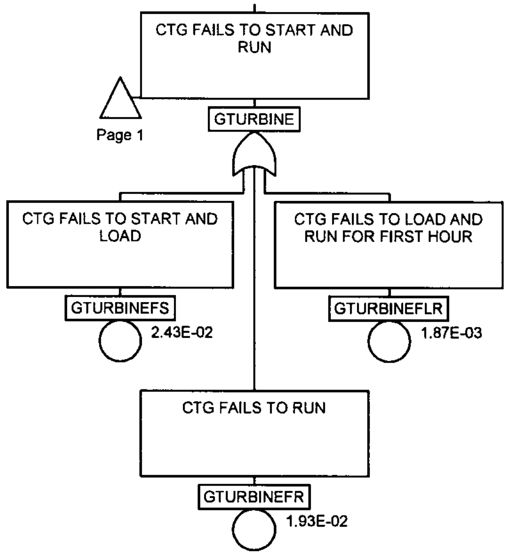


Figure 19D.6-13B Electric Power-6.9kV Switchgear Fault Tree Division 2

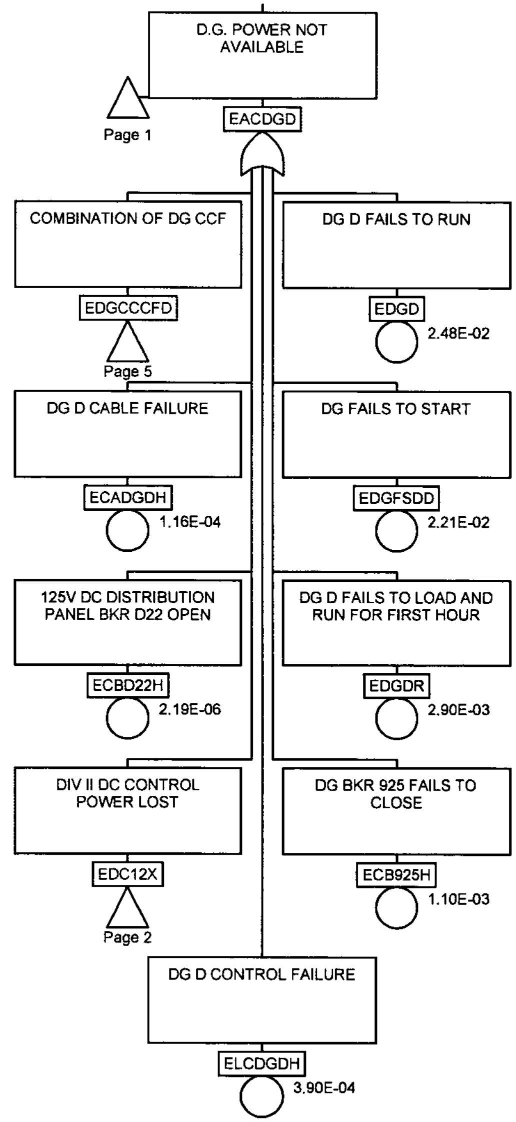
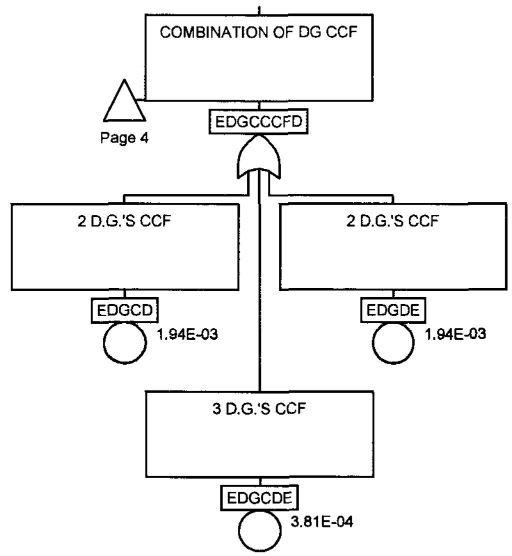


Figure 19D.6-13B Electric Power-6.9kV Switchgear Fault Tree Division 2


COMBINATION OF DG CCF
Page 4
EDGCCCFD
2 D.G.'S CCF
2 D.G.'S CCF
EDGCD
1.94E-03
EDGDE
1.94E-03
3 D.G.'S CCF
EDGCDE
3.81E-04

1 2

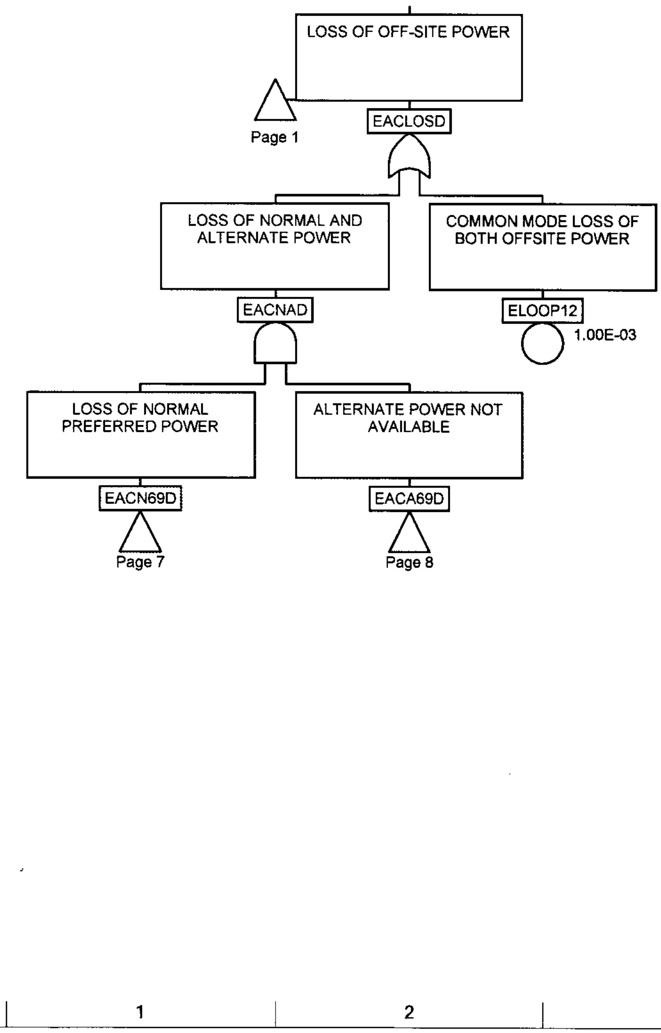


Figure 19D.6-13B Electric Power-6.9kV Switchgear Fault Tree Division 2

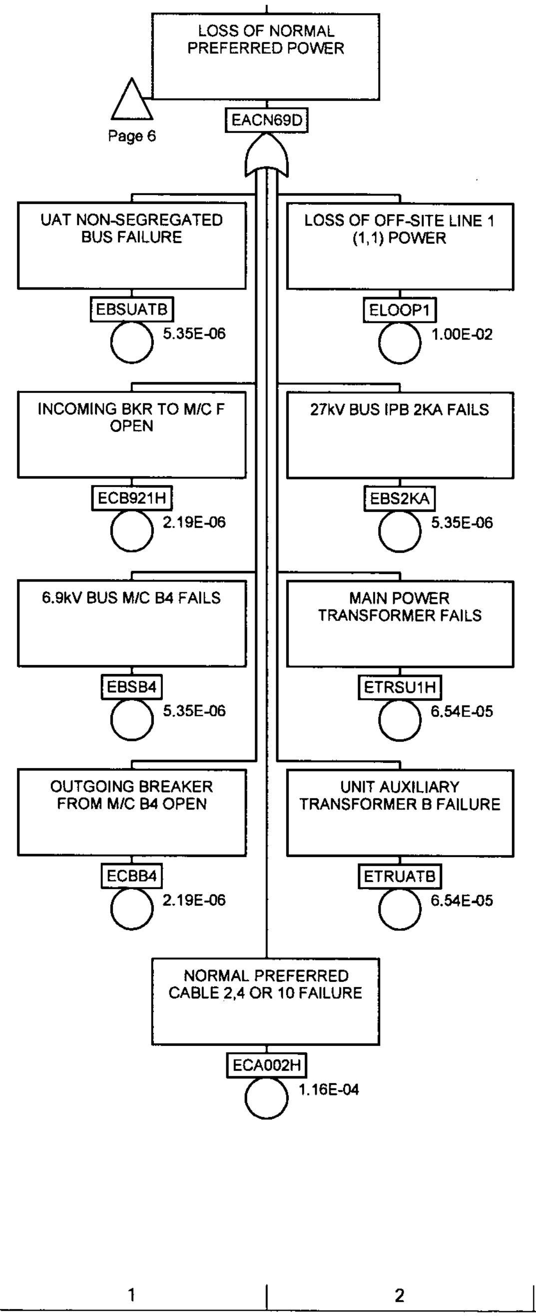


Figure 19D.6-13B Electric Power-6.9kV Switchgear Fault Tree Division 2

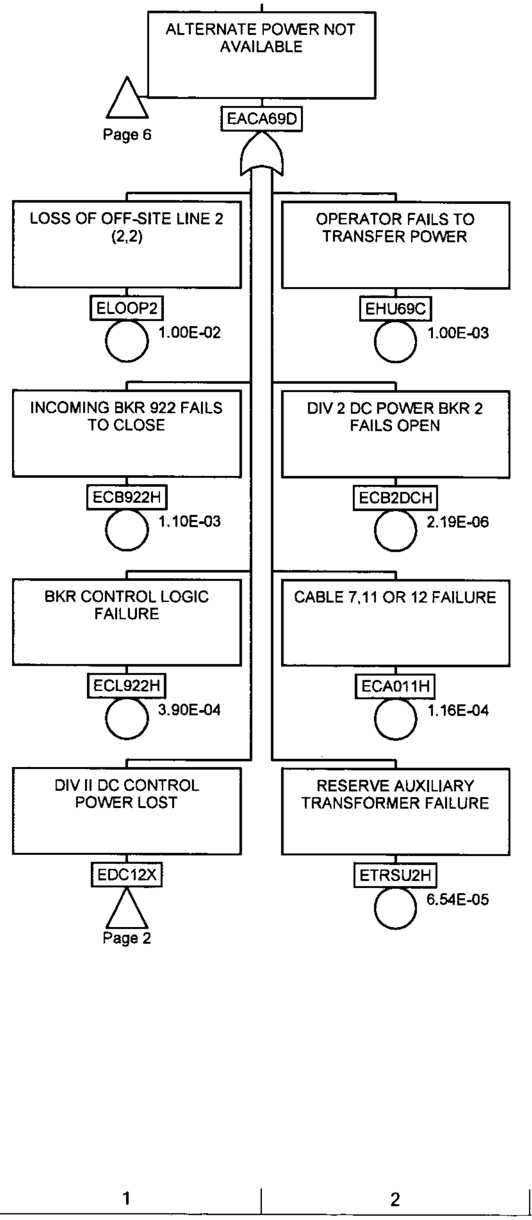


Figure 19D.6-13B Electric Power-6.9kV Switchgear Fault Tree Division 2

Name	Page	Zone	Name	Page	Zone
CTGMANSW	1	1	ELCDGDH	4	2
EAC69D	1	2	ELOOP1	7	2
EAC69DH	1	1	ELOOP12	6	3
EAC69SD	1	2	ELOOP2	8	1
EACA69D	6	2	ETRSU1H	7	2
EACA69D	8	2	ETRSU2H	8	2
EACDGD	1	3	ETRUATB	7	2
EACDGD	4	2	GTURBD	1	2
EACLOSD	1	4	GTURBINE	1	2
EACLOSD	6	2	GTURBINE	3	2
EACN69D	6	1	GTURBINEFLR	3	2
EACN69D	7	2	GTURBINEFR	3	2
EACNAD	6	2	GTURBINEFS	3	1
EBS2KA	7	2			
EBSB4	7	1			
EBSCTGD	1	2			
EBSUATB	7	1			
EBY102H	2	2			
EBY1CCF	2	2			
ECA002H	7	2			
ECA011H	8	2			
ECA1B2H	2	1			
ECACTGH	1	1			
ECADGDH	4	1			
ECB0B1H	2	1			
ECB1B2H	2	2			
ECB2DCH	8	2			
ECB921H	7	1			
ECB922H	8	1			
ECB925H	4	2			
ECBB4	7	1			
ECBCTG2	1	2			
ECBD22H	4	1			
ECL922H	8	1			
EDC12X	1	1			
EDC12X	2	2			
EDC12X	4	1			
EDC12X	8	1			
EDGCCCFD	4	1			
EDGCCCFD	5	2			
EDGCD	5	1			
EDGCDE	5	2			
EDGD	4	2			
EDGDE	5	2			
EDGDR	4	2			
EDGFSDD	4	2			
EDP102H	2	1			
EHU69C	8	2			

Figure 19D.6-13B Electric Power-6.9kV Switchgear Fault Tree Division 2

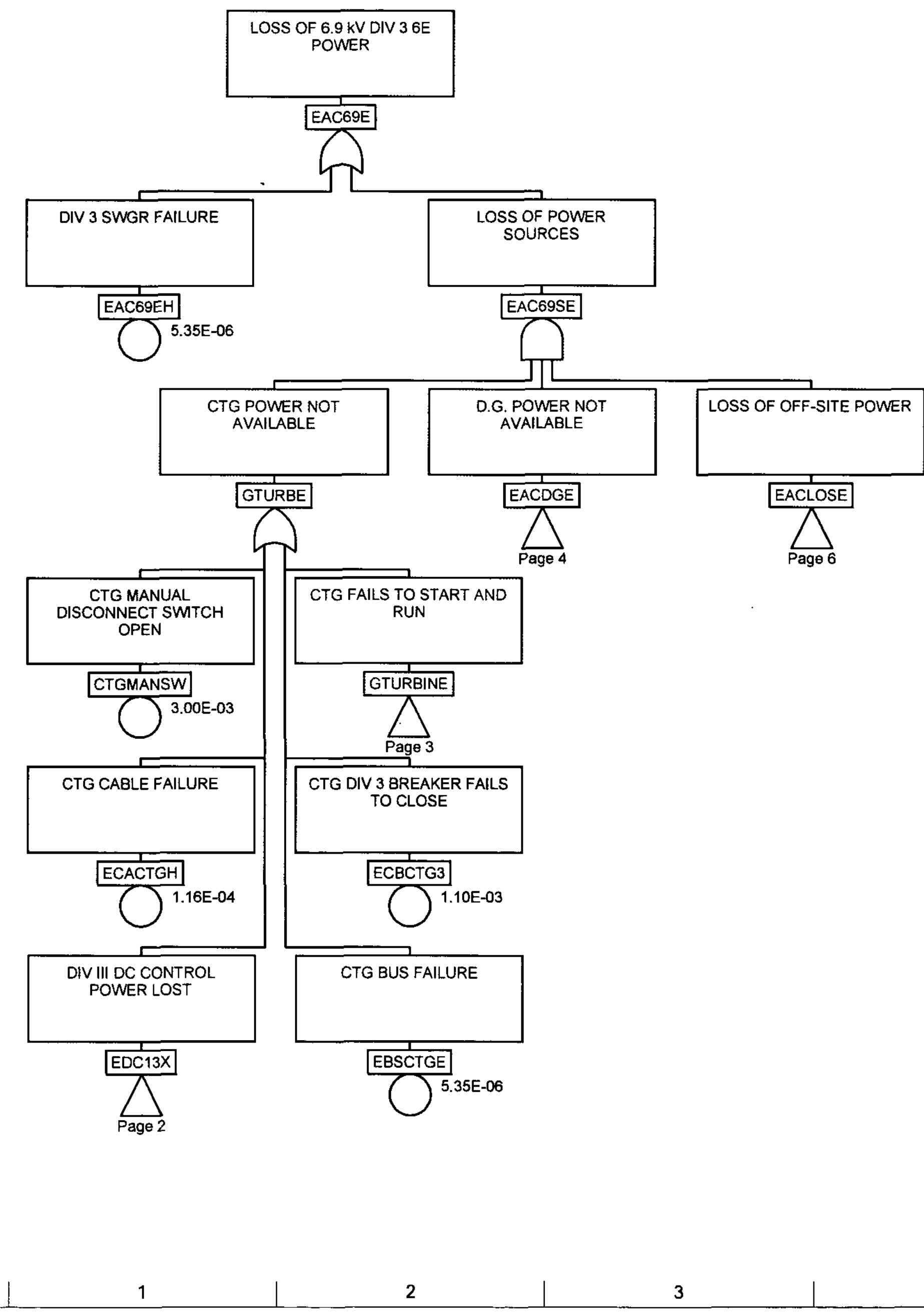


Figure 19D.6-13C Electric Power-6.9kV Switchgear Fault Tree Division 3

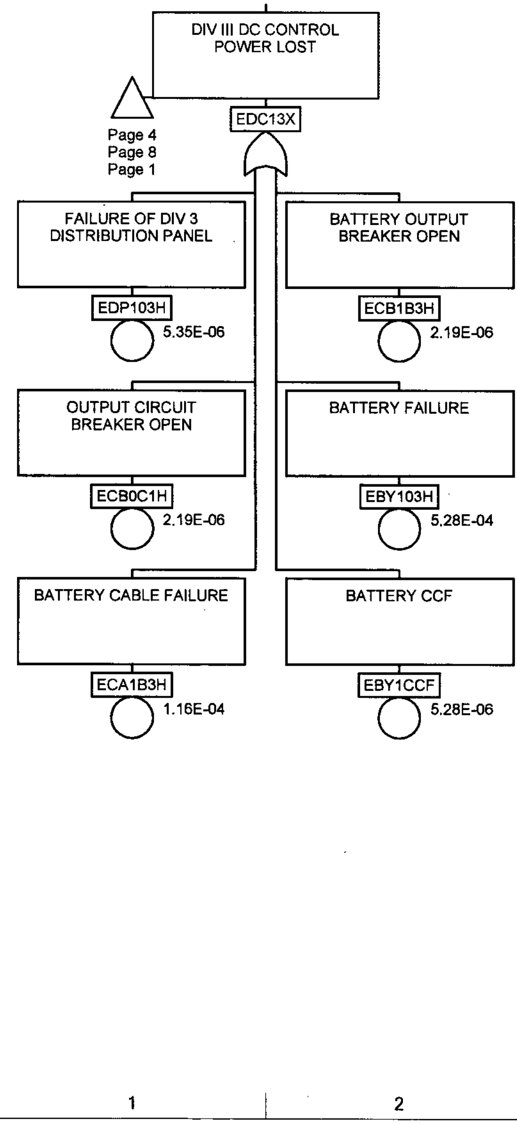


Figure 19D.6-13C Electric Power-6.9kV Switchgear Fault Tree Division 3

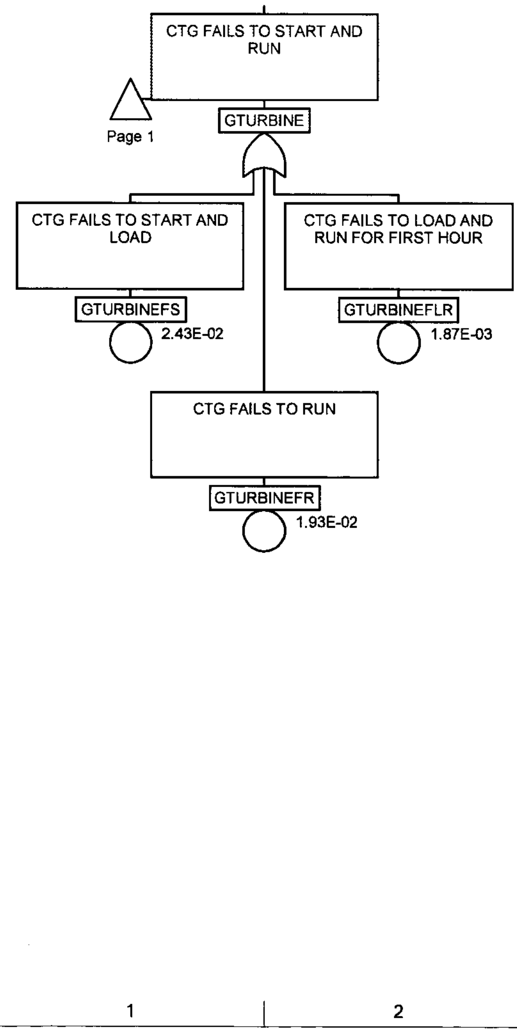


Figure 19D.6-13C Electric Power-6.9kV Switchgear Fault Tree Division 3

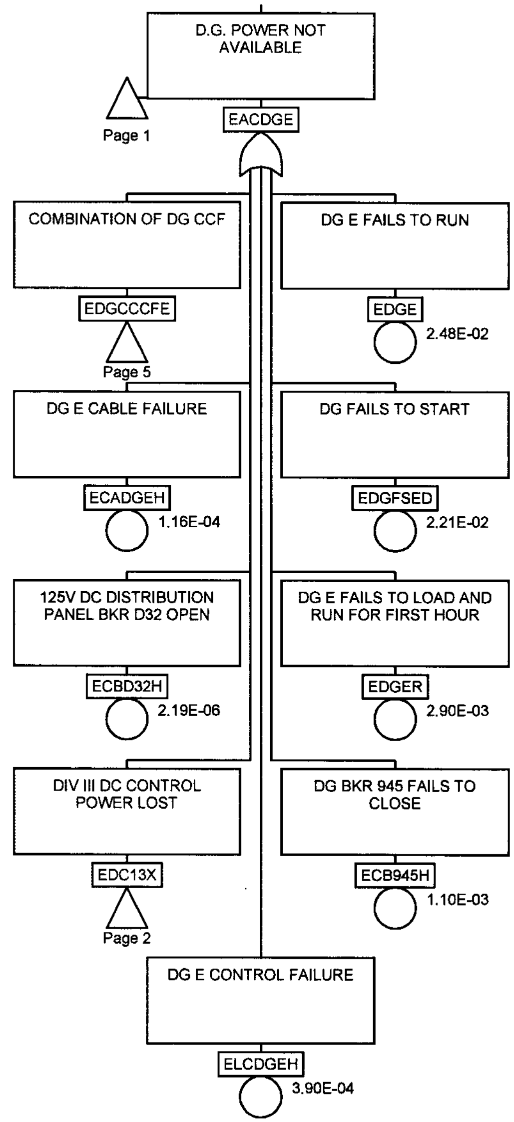


Figure 19D.6-13C Electric Power-6.9kV Switchgear Fault Tree Division 3

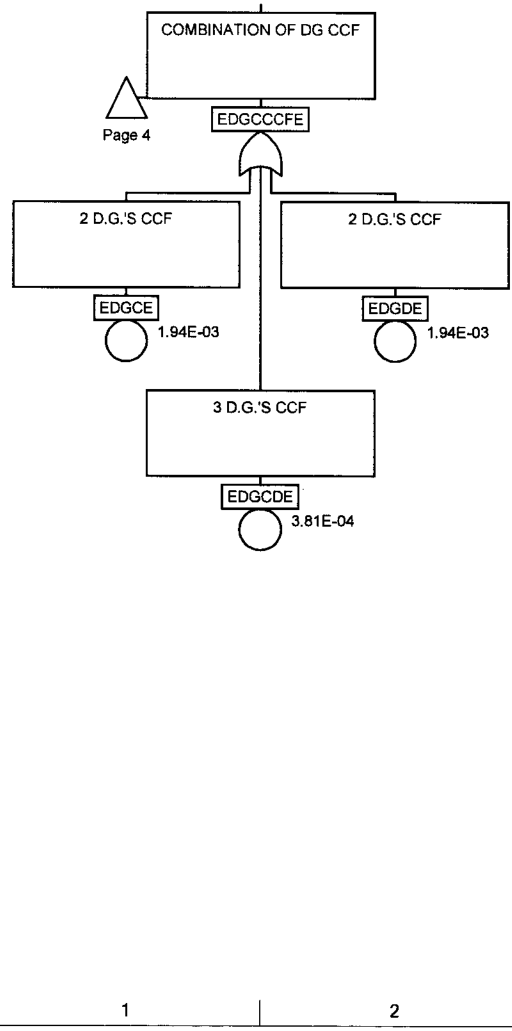


Figure 19D.6-13C Electric Power-6.9kV Switchgear Fault Tree Division 3

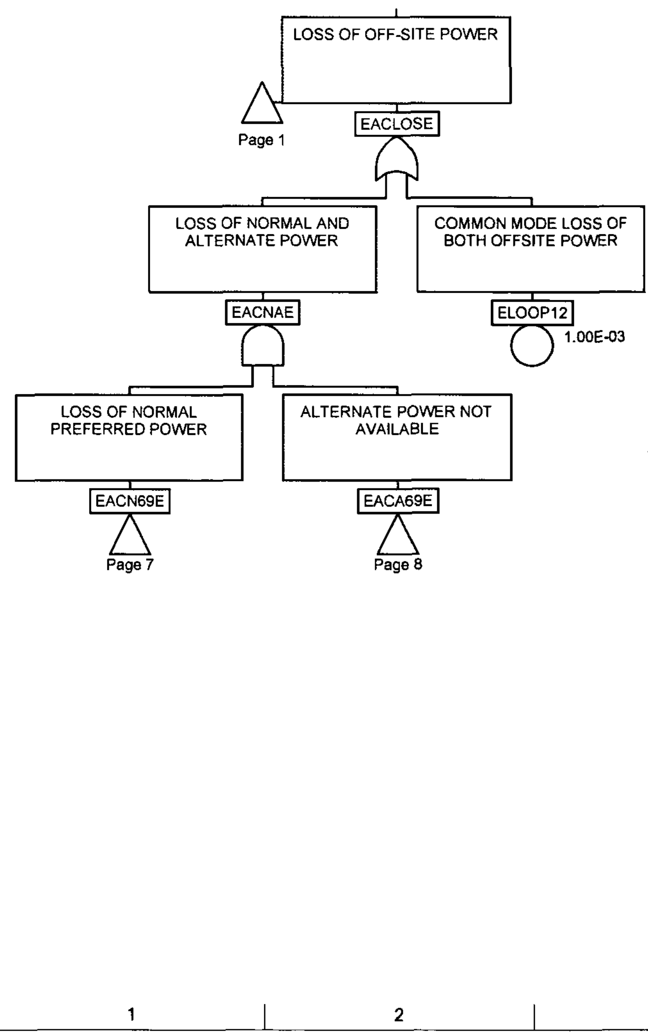


Figure 19D.6-13C Electric Power-6.9kV Switchgear Fault Tree Division 3

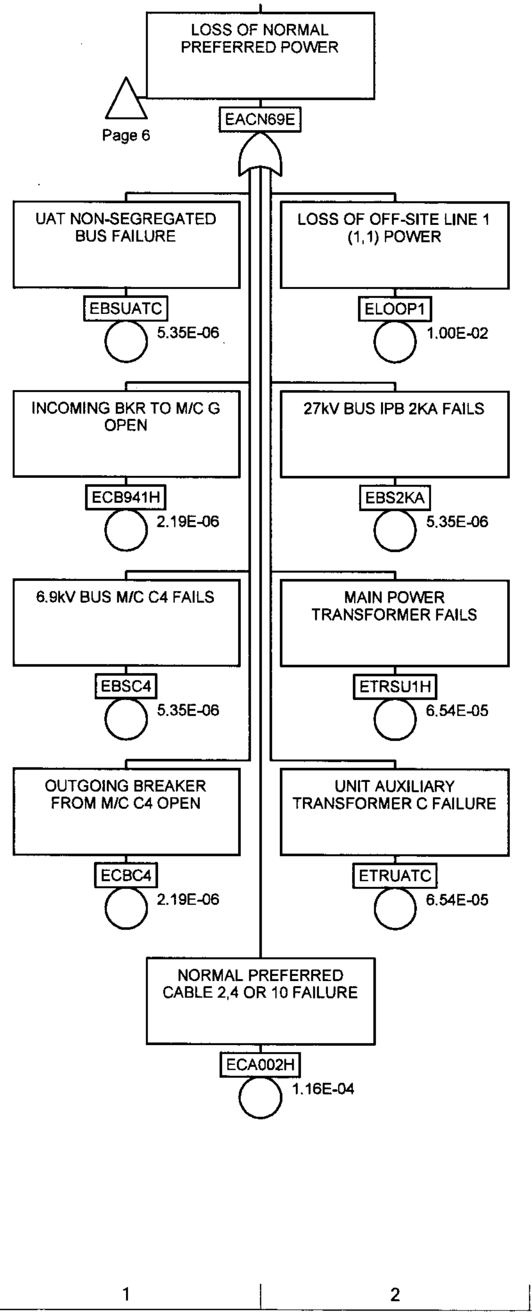


Figure 19D.6-13C Electric Power-6.9kV Switchgear Fault Tree Division 3

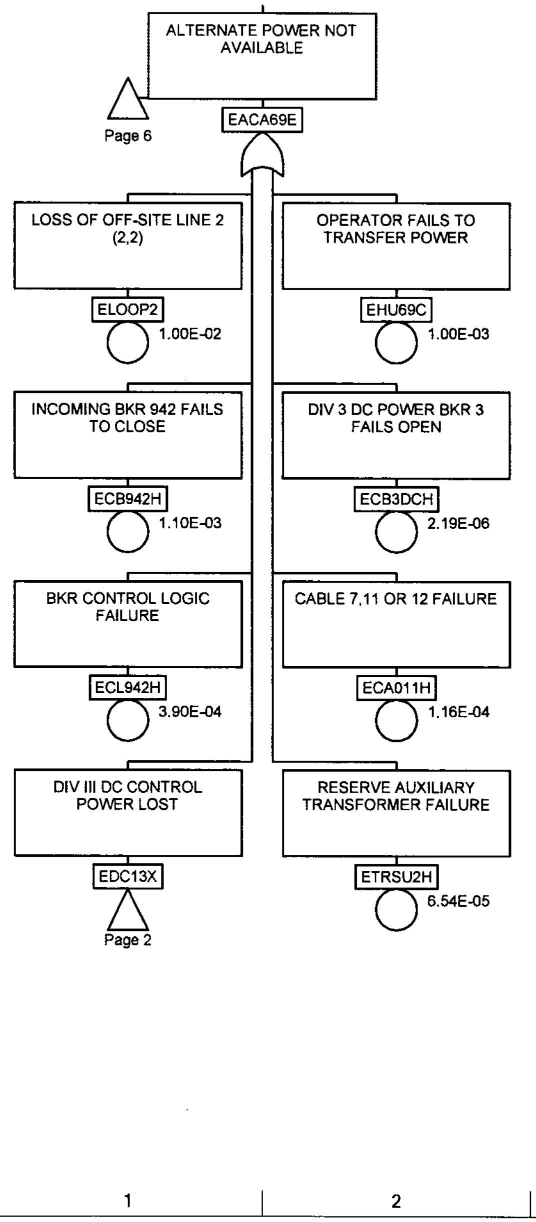


Figure 19D.6-13C Electric Power-6.9kV Switchgear Fault Tree Division 3

Name	Page	Zone	Name	Page	Zone
CTGMANSW	1	1	ELCDGEH	4	2
EAC69E	1	2	ELOOP1	7	2
EAC69EH	1	1	ELOOP12	6	3
EAC69SE	1	2	ELOOP2	8	1
EACA69E	6	2	ETRSU1H	7	2
EACA69E	8	2	ETRSU2H	8	2
EACDGE	1	3	ETRUATC	7	2
EACDGE	4	2	GTURBE	1	2
EACLOSE	1	4	GTURBINE	1	2
EACLOSE	6	2	GTURBINE	3	2
EACN69E	6	1	GTURBINEFLR	3	2
EACN69E	7	2	GTURBINEFR	3	2
EACNAE	6	2	GTURBINEFS	3	1
EBS2KA	7	2			
EBSC4	7	1			
EBSCTGE	1	2			
EBSUATC	7	1			
EBY103H	2	2			
EBY1CCF	2	2			
ECA002H	7	2			
ECA011H	8	2			
ECA1B3H	2	1			
ECACTGH	1	1			
ECADGEH	4	1			
ECB0C1H	2	1			
ECB1B3H	2	2			
ECB3DCH	8	2			
ECB941H	7	1			
ECB942H	8	1			
ECB945H	4	2			
ECBC4	7	1			
ECBCTG3	1	2			
ECBD32H	4	1			
ECL942H	8	1			
EDC13X	1	1			
EDC13X	2	2			
EDC13X	4	1			
EDC13X	8	1			
EDGCCCFE	4	1			
EDGCCCFE	5	2			
EDGCDE	5	2			
EDGCE	5	1			
EDGDE	5	2			
EDGE	4	2			
EDGER	4	2			
EDGFSED	4	2			
EDP103H	2	1			
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Figure 19D.6-13C Electric Power-6.9kV Switchgear Fault Tree Division 3

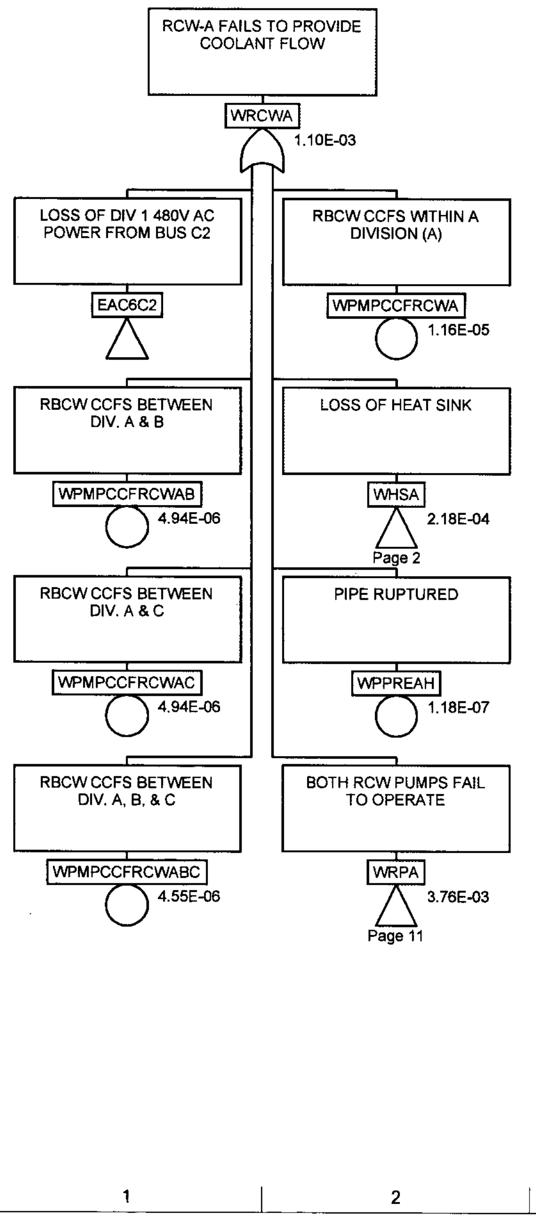

Figure 19D.6-14A Reactor Building Cooling and Service Water Fault Tree (A)

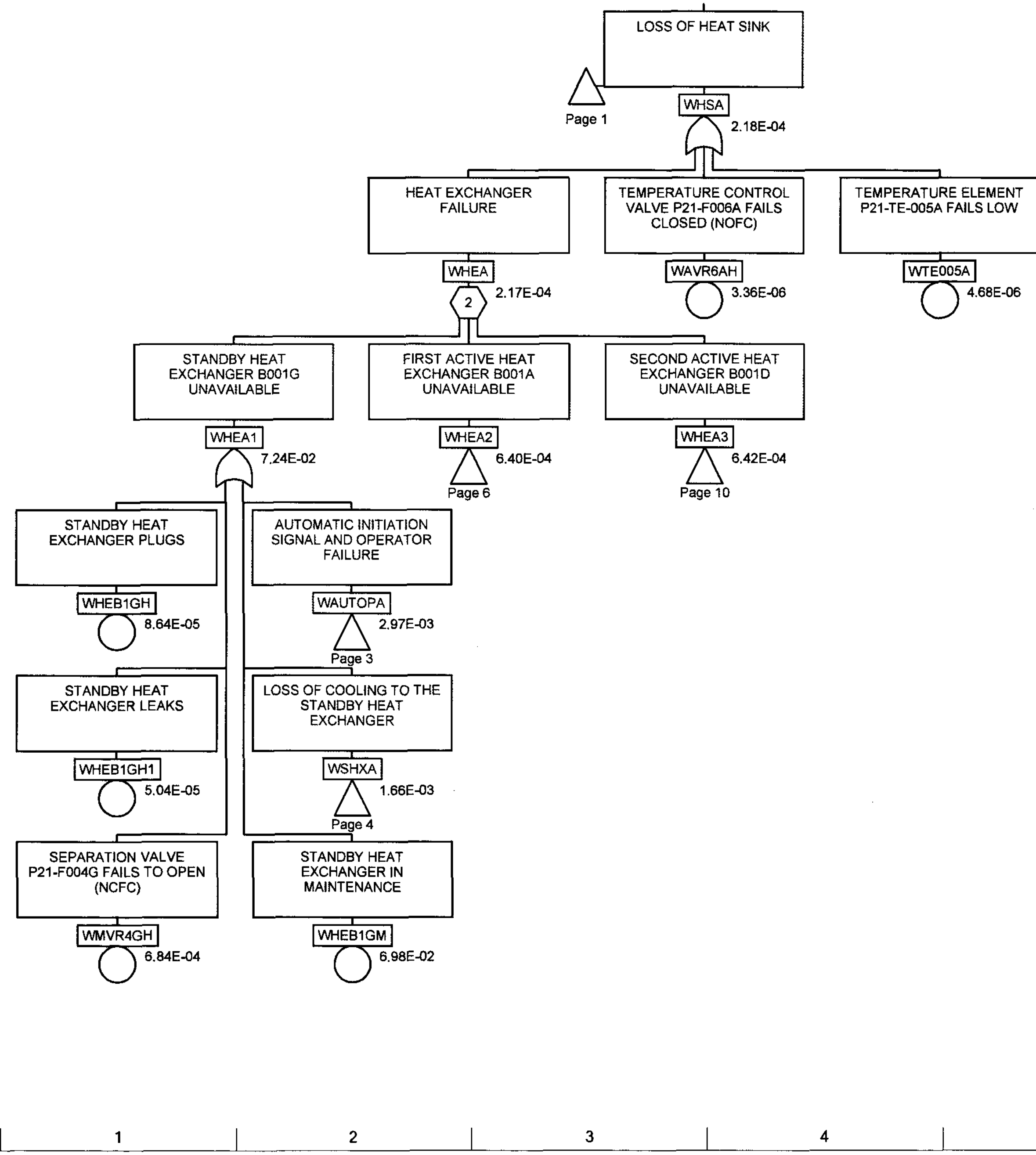

Figure 19D.6-14A Reactor Building Cooling and Service Water Fault Tree (A)

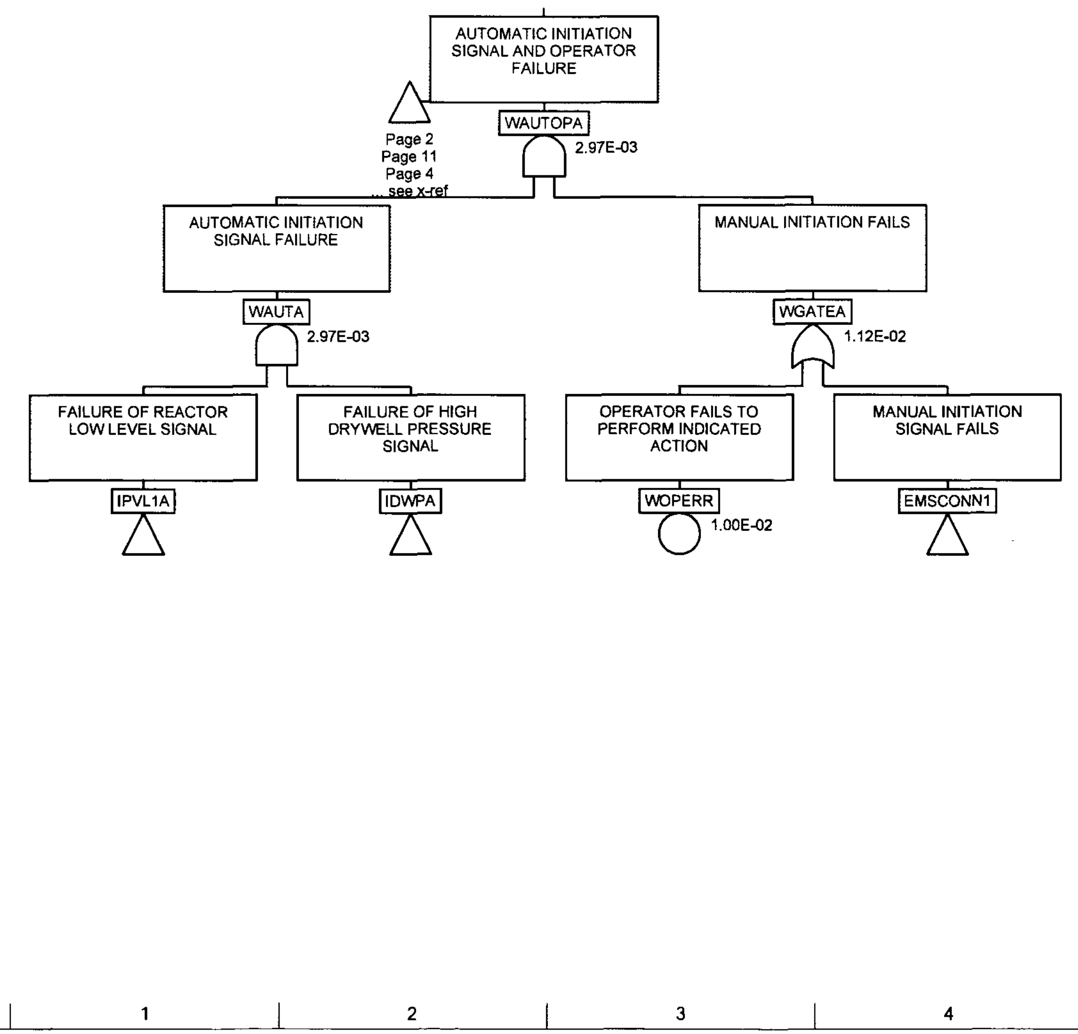

Figure 19D.6-14A Reactor Building Cooling and Service Water Fault Tree (A)

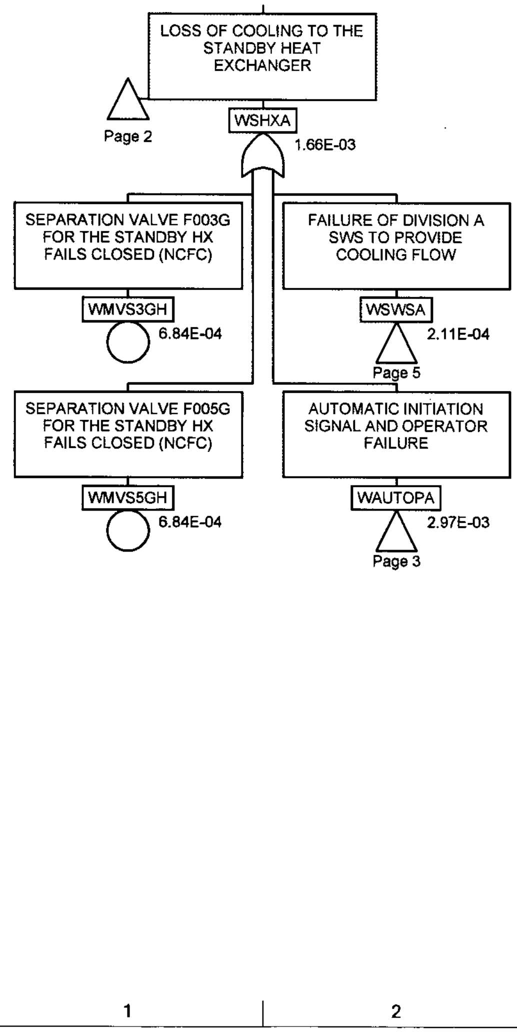

Figure 19D.6-14A Reactor Building Cooling and Service Water Fault Tree (A)

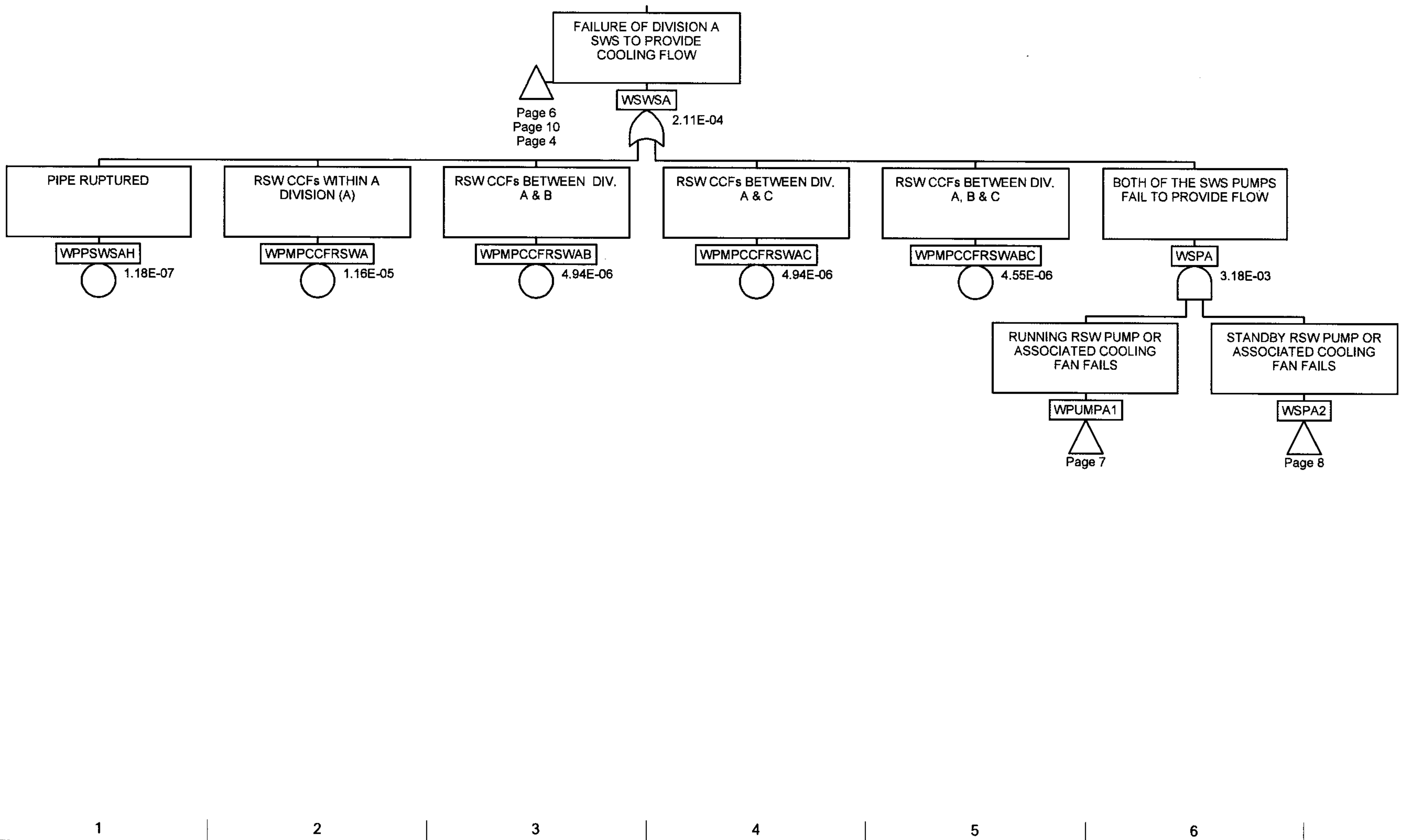

Figure 19D.6-14A Reactor Building Cooling and Service Water Fault Tree (A)

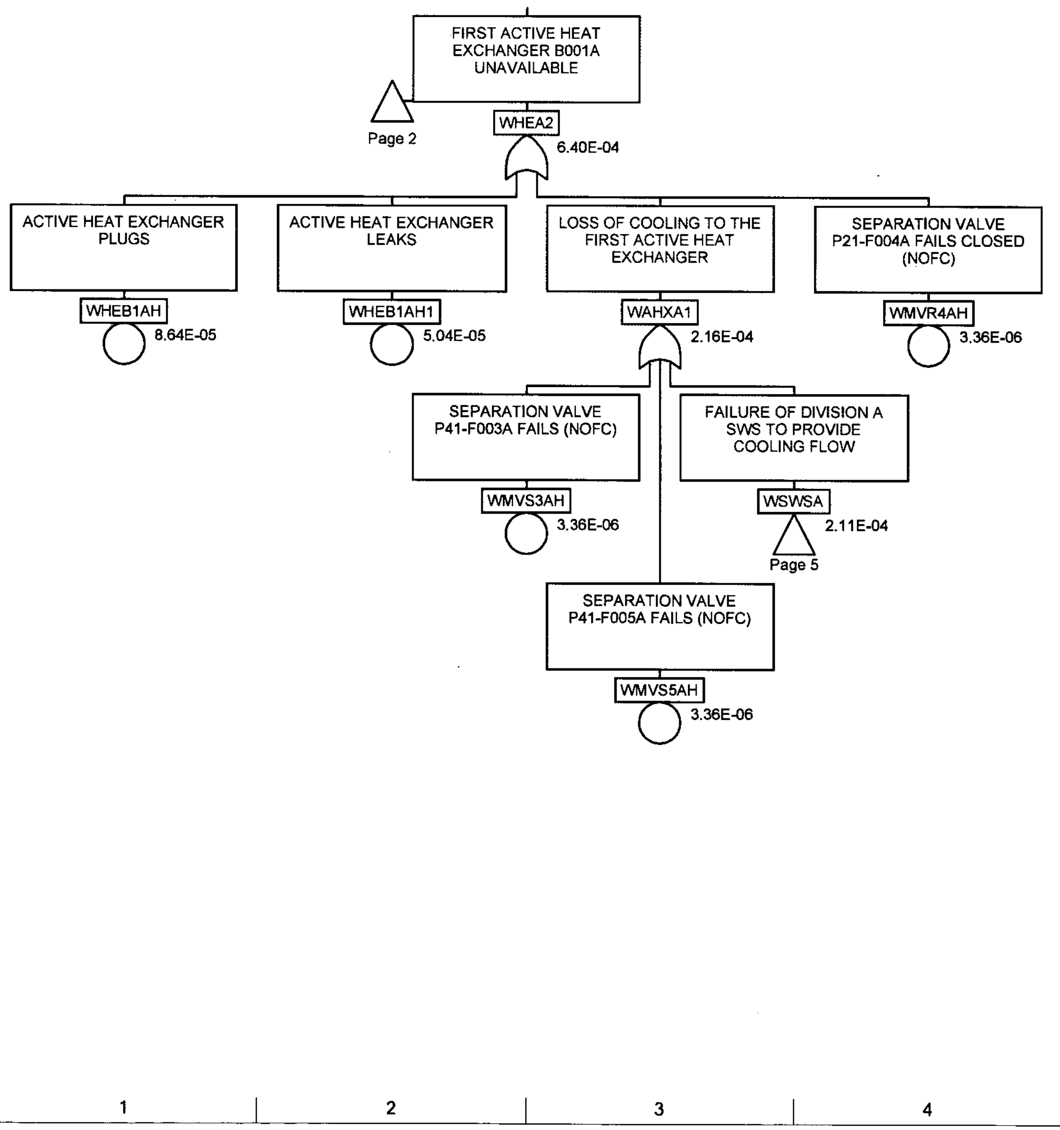

Figure 19D.6-14A Reactor Building Cooling and Service Water Fault Tree (A)

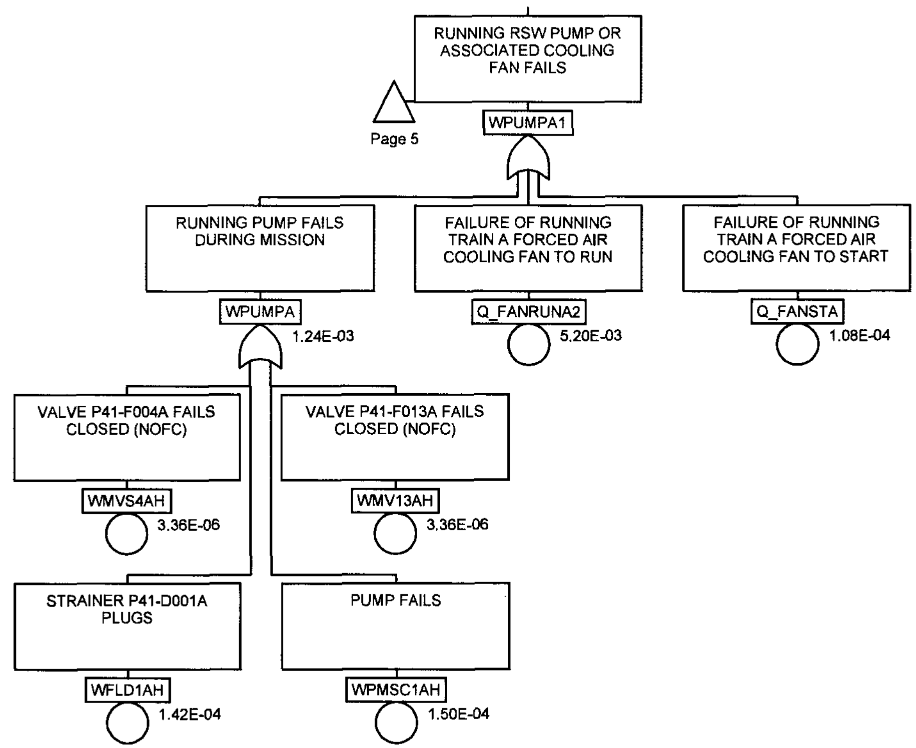

Figure 19D.6-14A Reactor Building Cooling and Service Water Fault Tree (A)

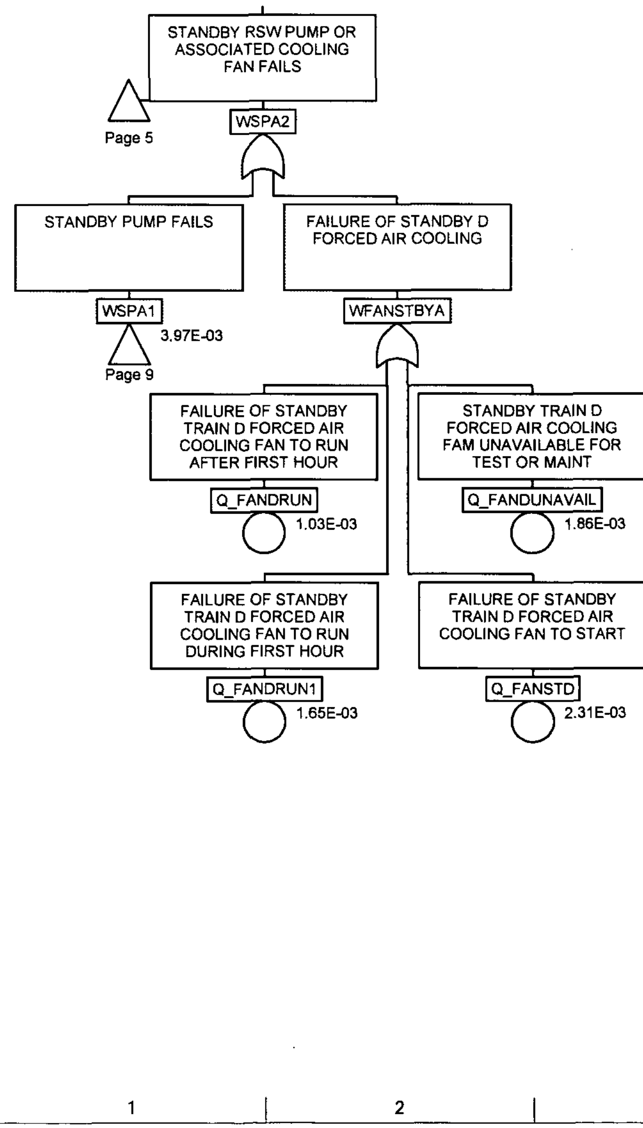

Figure 19D.6-14A Reactor Building Cooling and Service Water Fault Tree (A)

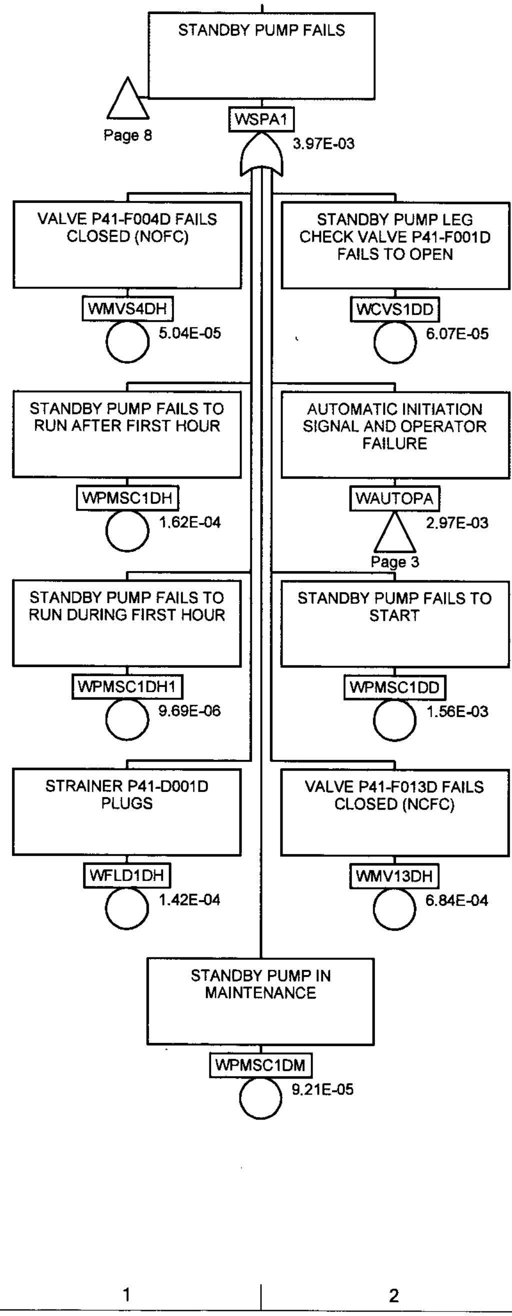

Figure 19D.6-14A Reactor Building Cooling and Service Water Fault Tree (A)

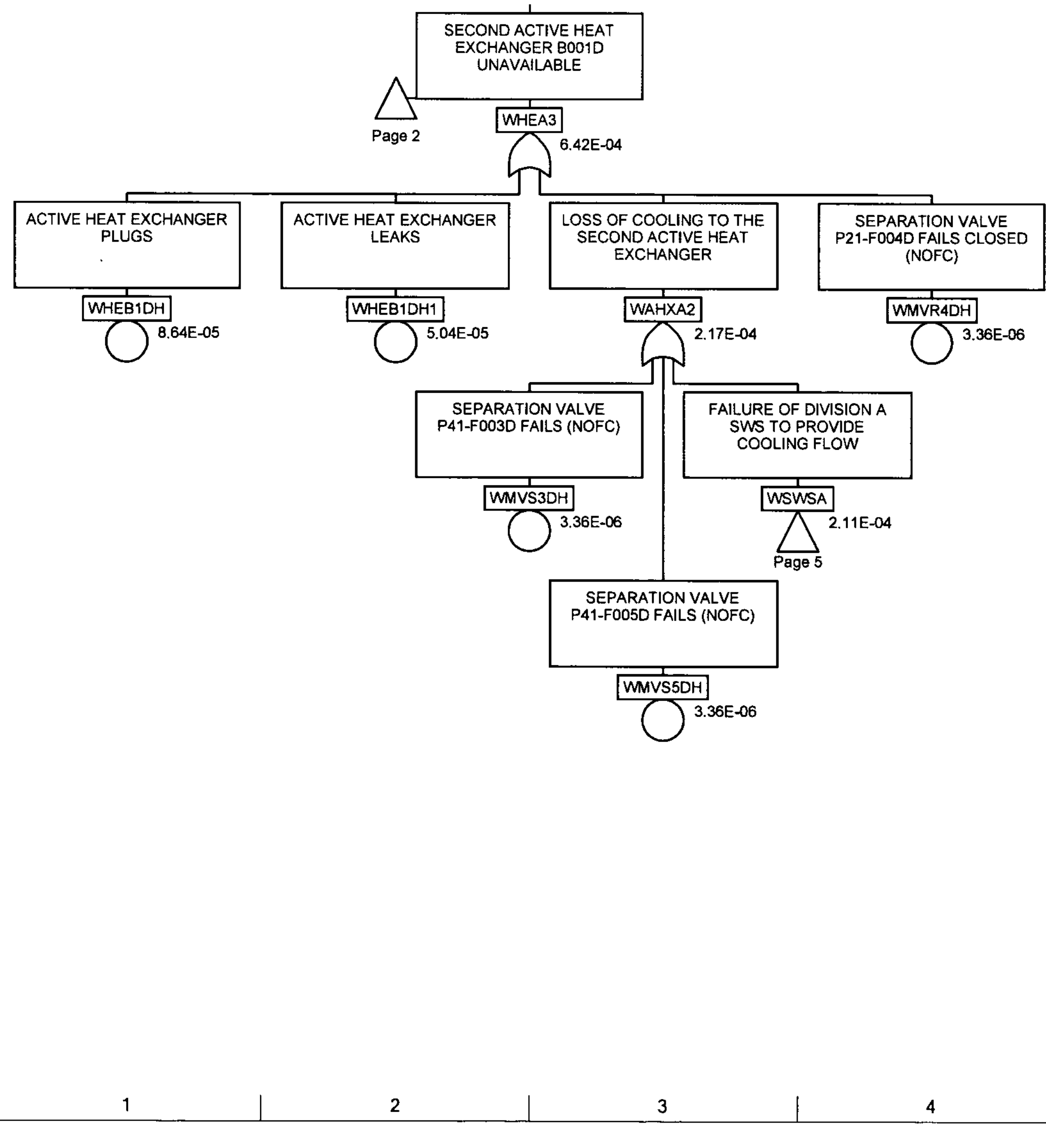

Figure 19D.6-14A Reactor Building Cooling and Service Water Fault Tree (A)

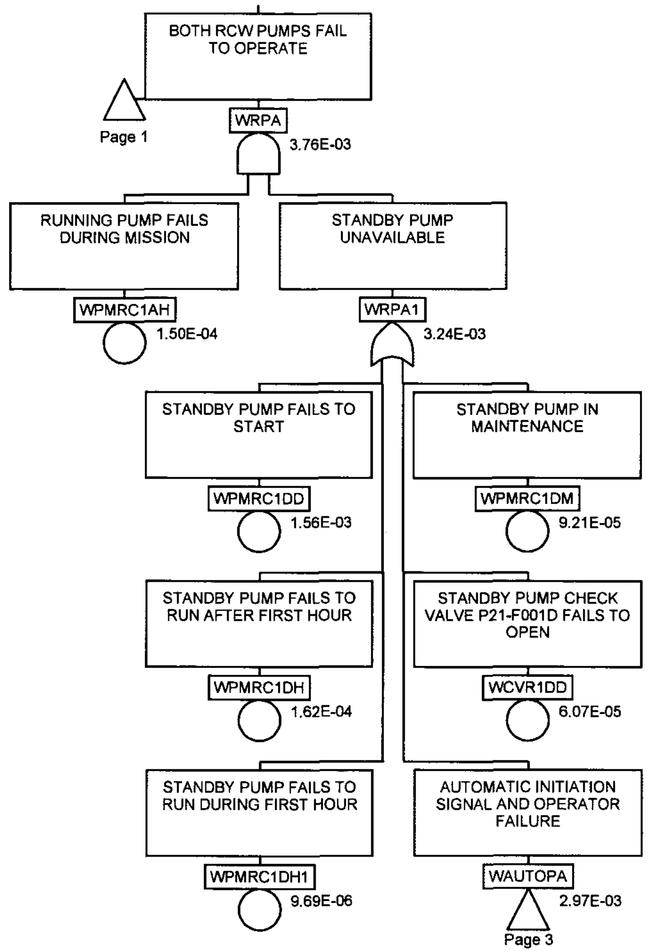

Figure 19D.6-14A Reactor Building Cooling and Service Water Fault Tree (A)

Name	Page	Zone	Name	Page	Zone
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IDWPA	3	2	WMVS5AH	6	3
IPVL1A	3	1	WMVS5DH	10	3
Q_FANDRUN	8	2	WMVS5GH	4	1
Q_FANDRUN1	8	2	WOPERR	3	3
Q_FANDUNAVAIL	8	3	WPMPCCFRCWA	1	2
Q_FANRUNA2	7	3	WPMPCCFRCWAB	1	1
Q_FANSTA	7	4	WPMPCCFRCWABC	1	1
Q_FANSTD	8	3	WPMPCCFRCWAC	1	1
WAHXA1	6	3	WPMPCCFRSWA	5	2
WAHXA2	10	3	WPMPCCFRSWAB	5	3
WAUTA	3	2	WPMPCCFRSWABC	5	5
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WAUTOPA	9	2	WPMRC1DH	11	2
WAUTOPA	11	3	WPMRC1DH1	11	2
WAVR6AH	2	3	WPMRC1DM	11	3
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WCVS1DD	9	2	WPMSC1DD	9	2
WFANSTBYA	8	2	WPMSC1DH	9	1
WFLD1AH	7	1	WPMSC1DH1	9	1
WFLD1DH	9	1	WPMSC1DM	9	2
WGATEA	3	4	WPPREAH	1	2
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WHEA1	2	2	WPUMPA	7	2
WHEA2	2	3	WPUMPA1	5	6
WHEA2	6	3	WPUMPA1	7	2
WHEA3	2	4	WRCWA	1	2
WHEA3	10	3	WRPA	1	2
WHEB1AH	6	1	WRPA	11	2
WHEB1AH1	6	2	WRPA1	11	2
WHEB1DH	10	1	WSHXA	2	2
WHEB1DH1	10	2	WSHXA	4	2
WHEB1GH	2	1	WSPA	5	6
WHEB1GH1	2	1	WSPA1	8	1
WHEB1GM	2	2	WSPA1	9	2
WHSA	1	2	WSPA2	5	7
WHSA	2	3	WSPA2	8	2
WMV13AH	7	2	WSWSA	4	2
WMV13DH	9	2	WSWSA	5	4
WMVR4AH	6	4	WSWSA	6	4
WMVR4DH	10	4	WSWSA	10	4
WMVR4GH	2	1	WTE005A	2	4
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Figure 19D.6-14A Reactor Building Cooling and Service Water Fault Tree (A)

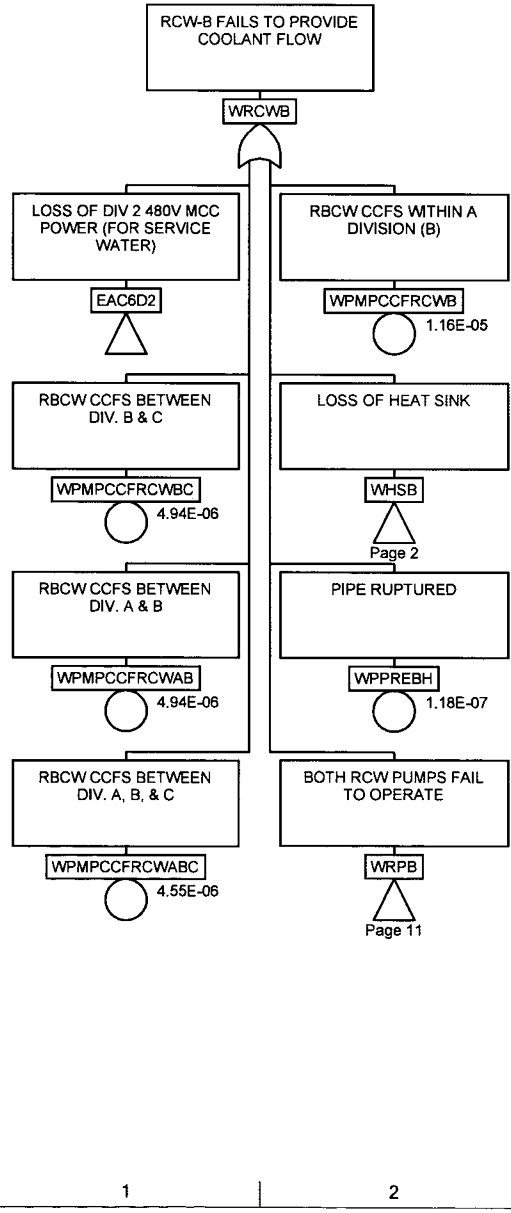

Figure 19D.6-14B Reactor Building Cooling and Service Water Fault Tree (B)

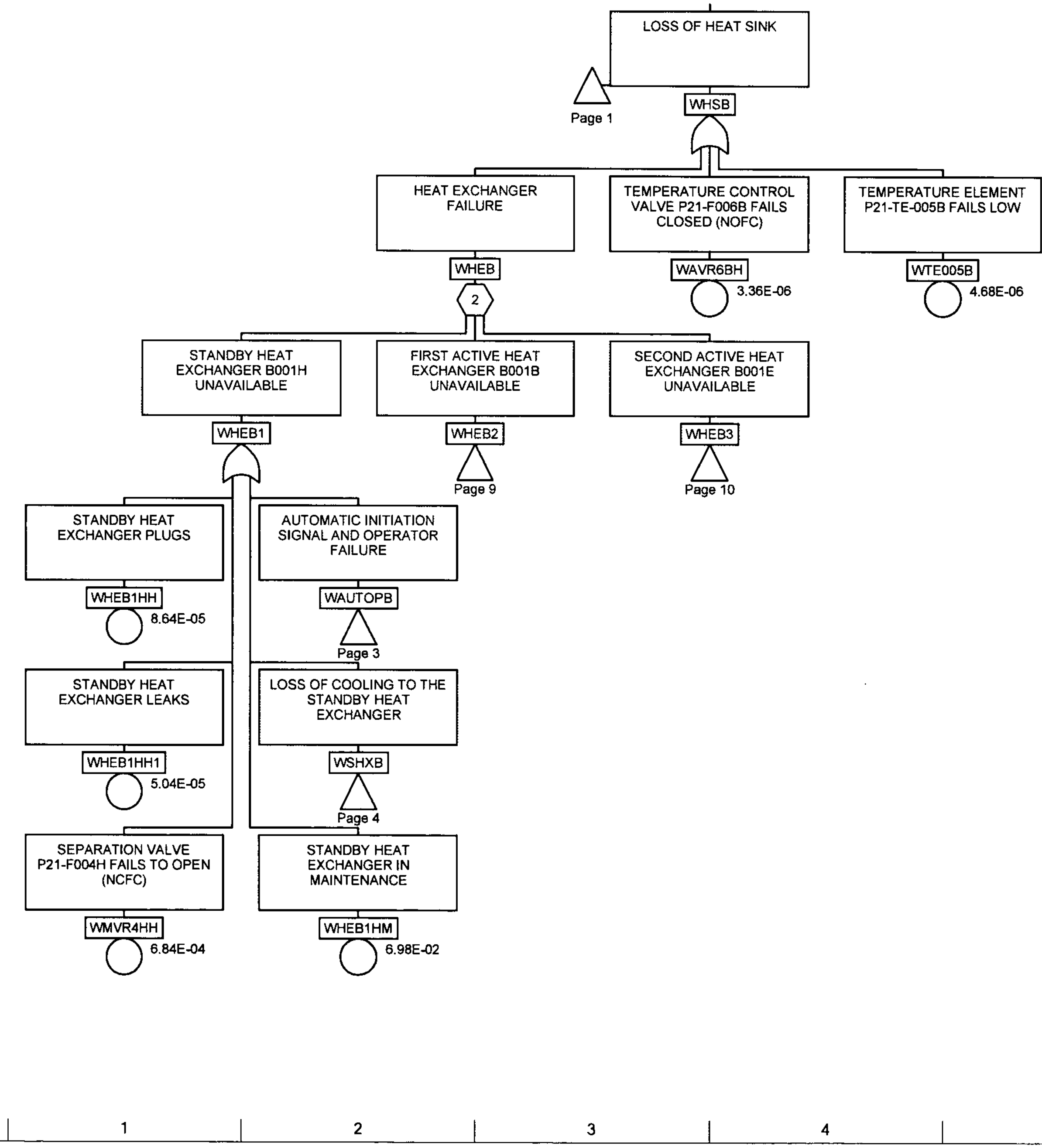

Figure 19D.6-14B Reactor Building Cooling and Service Water Fault Tree (B)

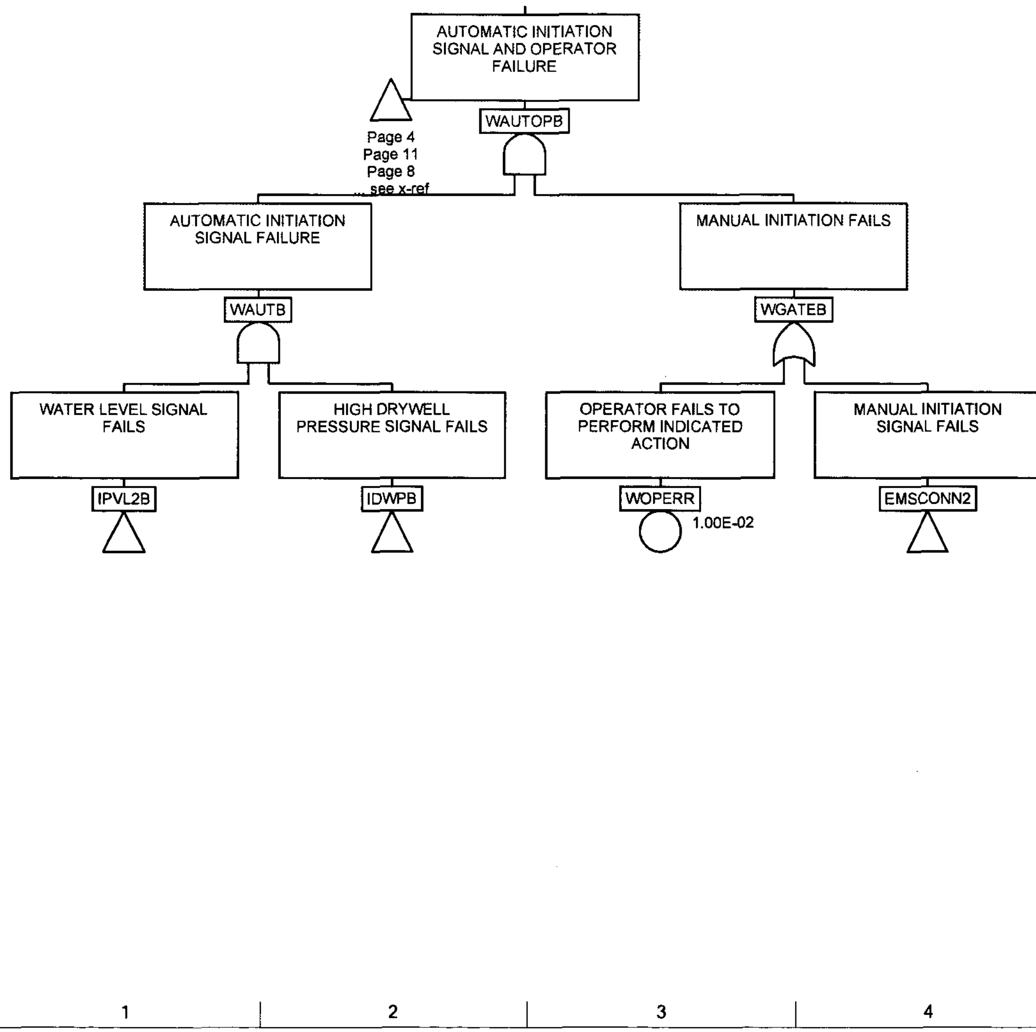

Figure 19D.6-14B Reactor Building Cooling and Service Water Fault Tree (B)

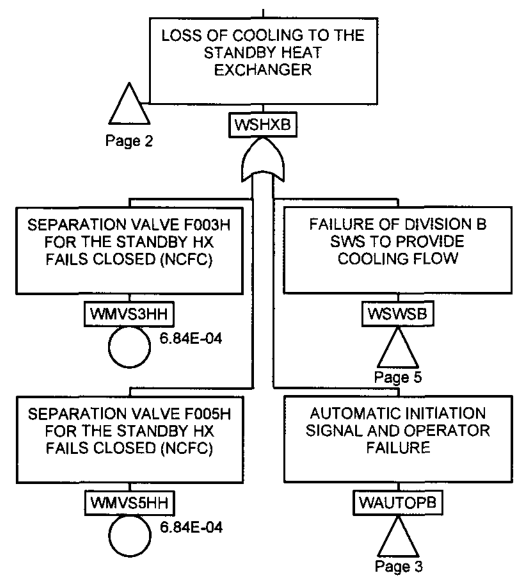

Figure 19D.6-14B Reactor Building Cooling and Service Water Fault Tree (B)

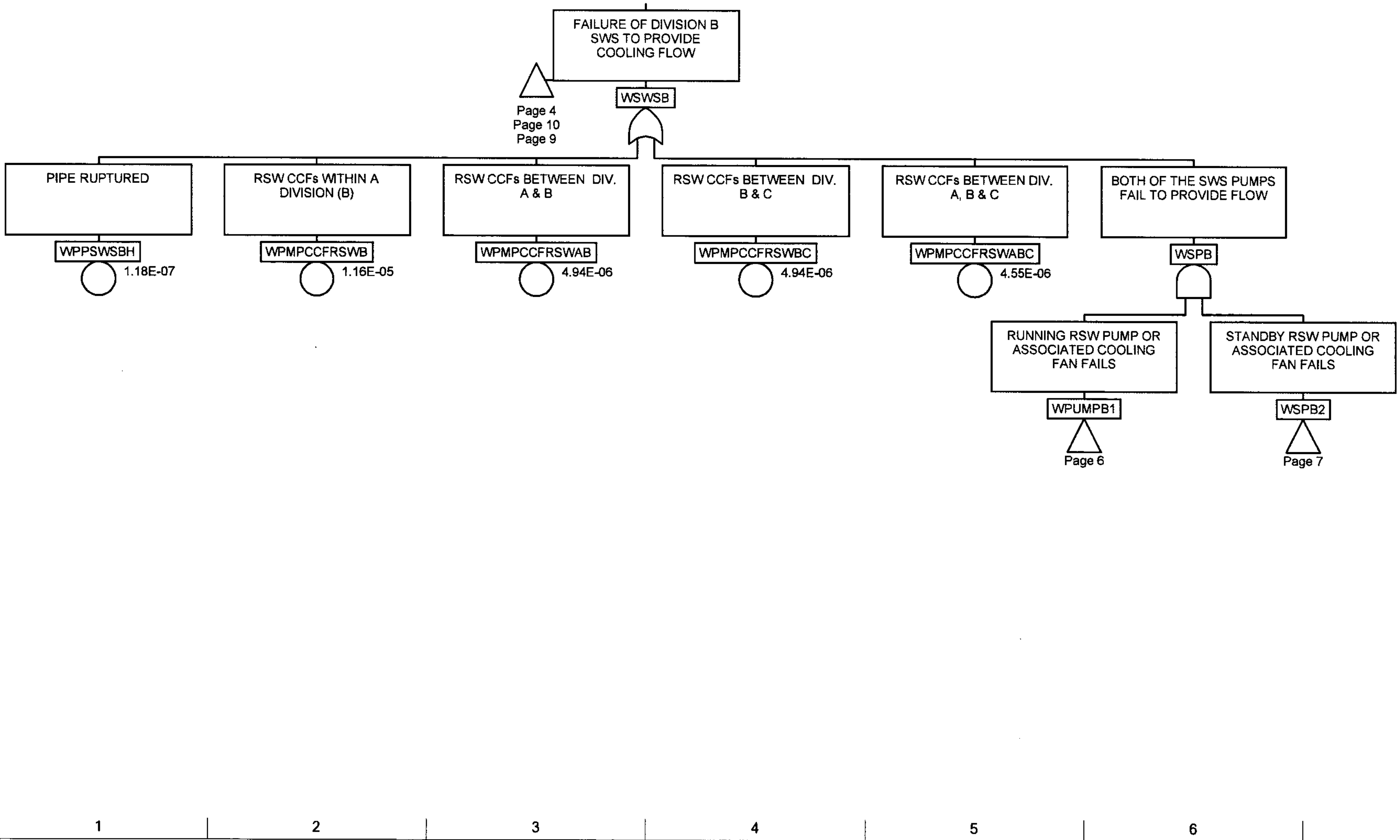

Figure 19D.6-14B Reactor Building Cooling and Service Water Fault Tree (B)

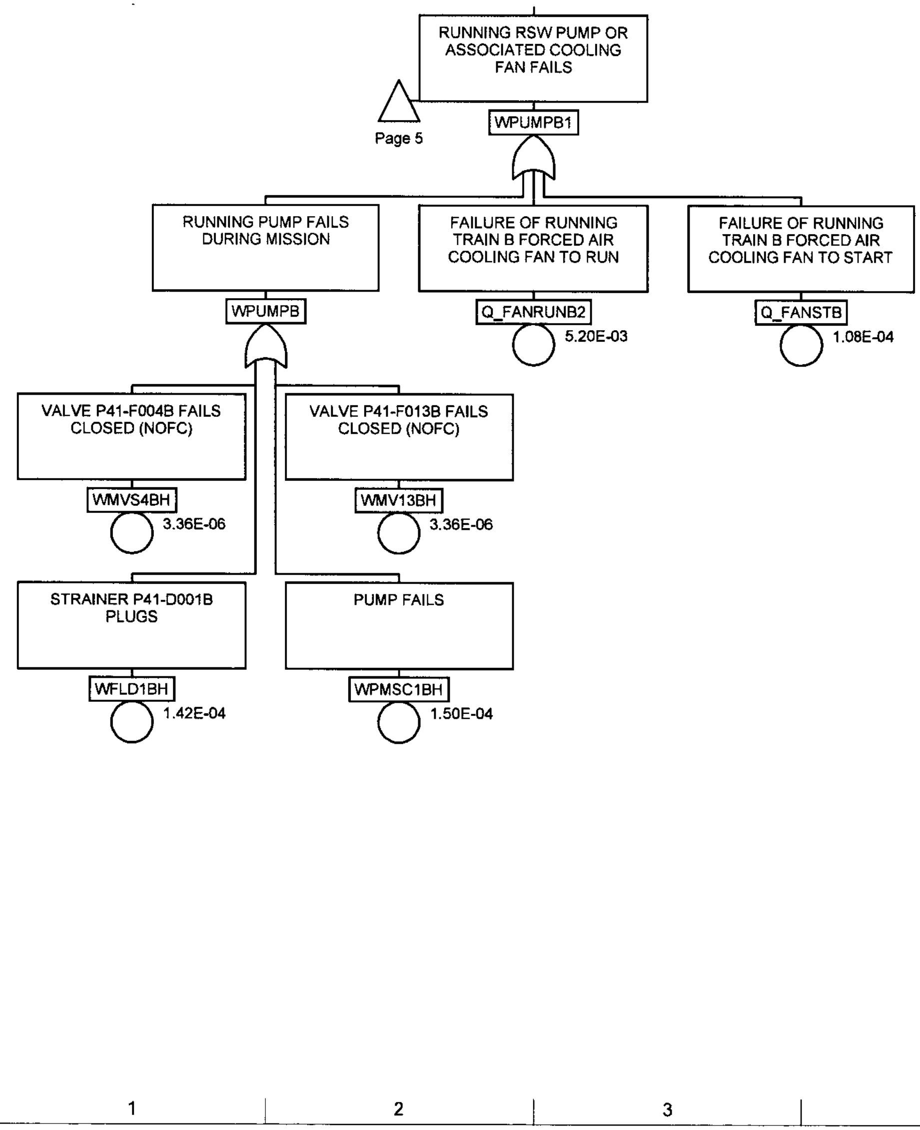

Figure 19D.6-14B Reactor Building Cooling and Service Water Fault Tree (B)

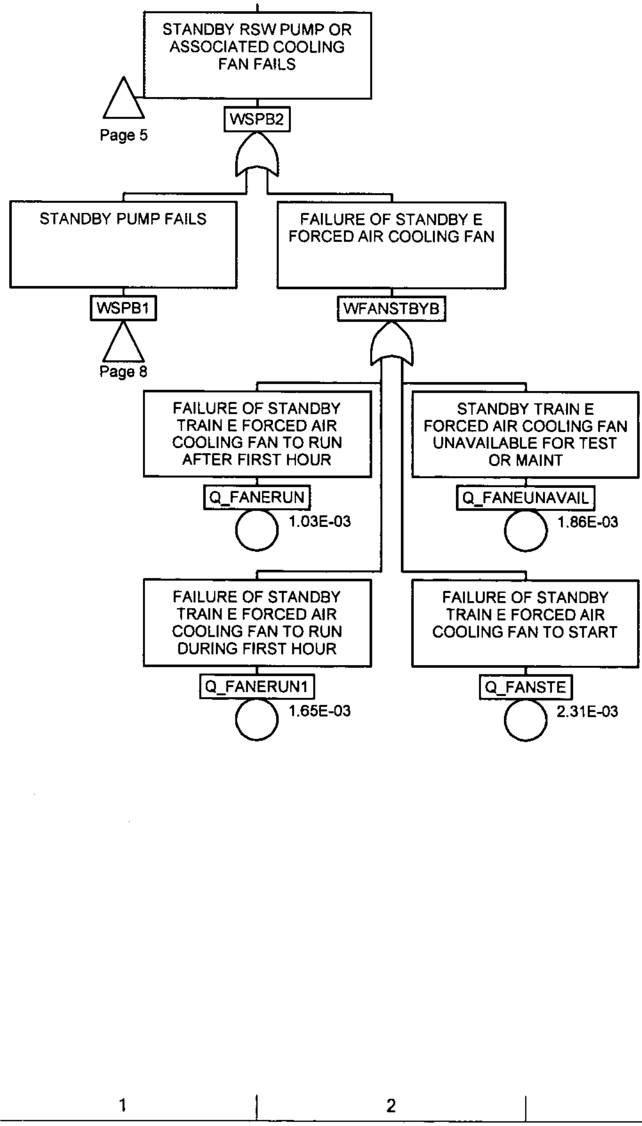

Figure 19D.6-14B Reactor Building Cooling and Service Water Fault Tree (B)

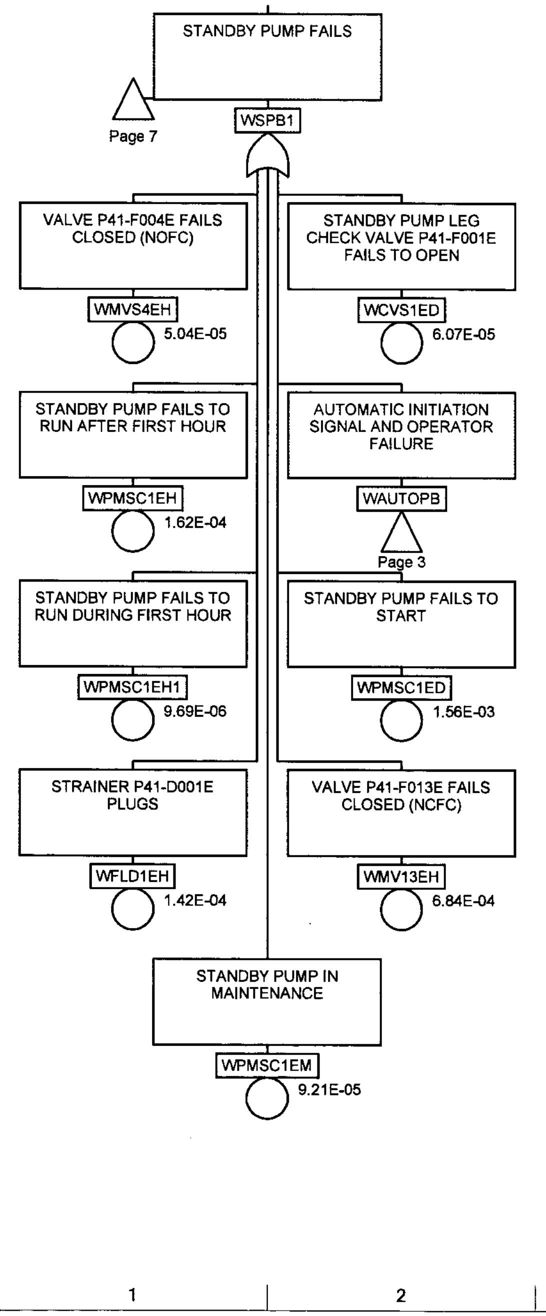

Figure 19D.6-14B Reactor Building Cooling and Service Water Fault Tree (B)

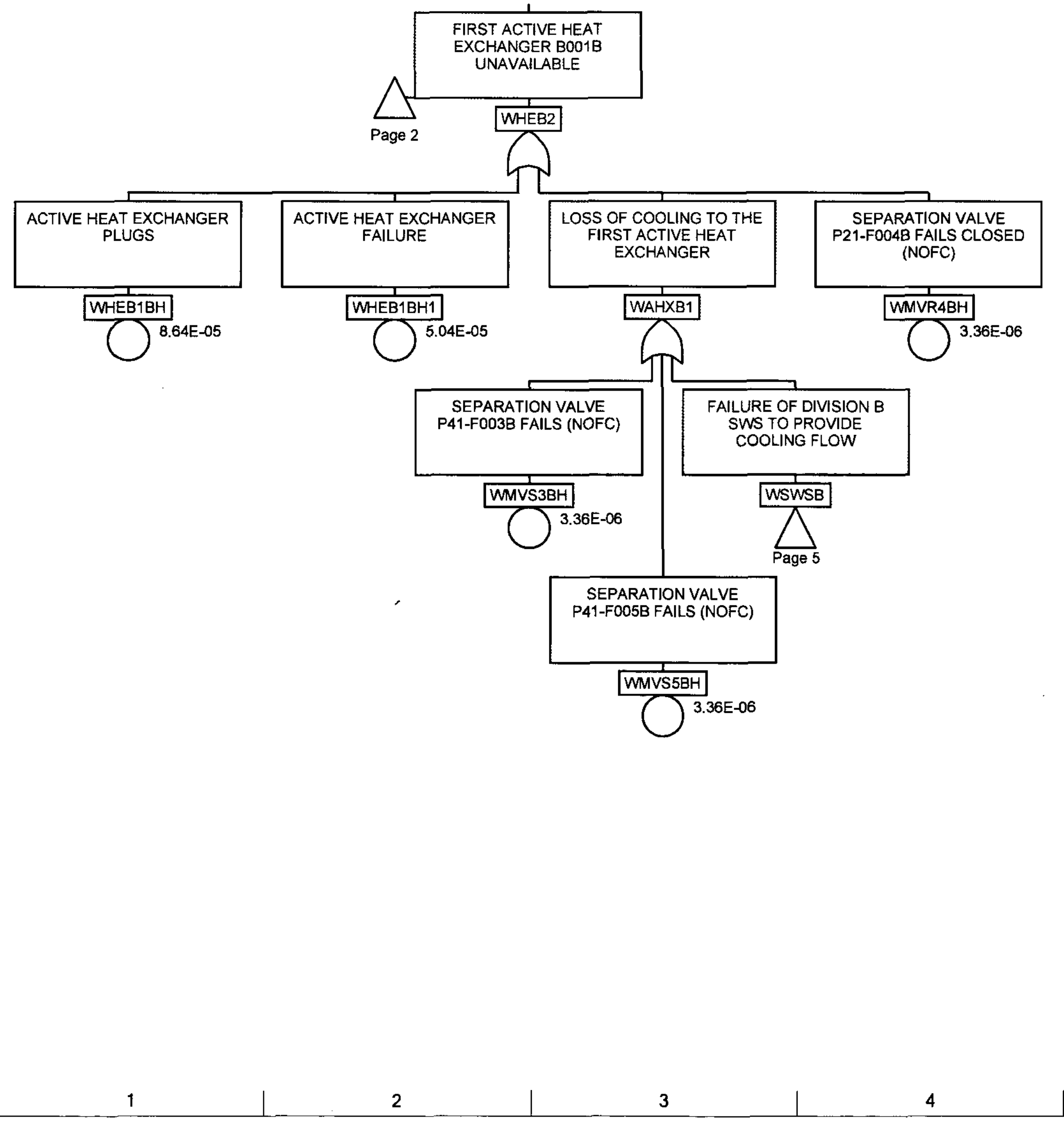

Figure 19D.6-14B Reactor Building Cooling and Service Water Fault Tree (B)

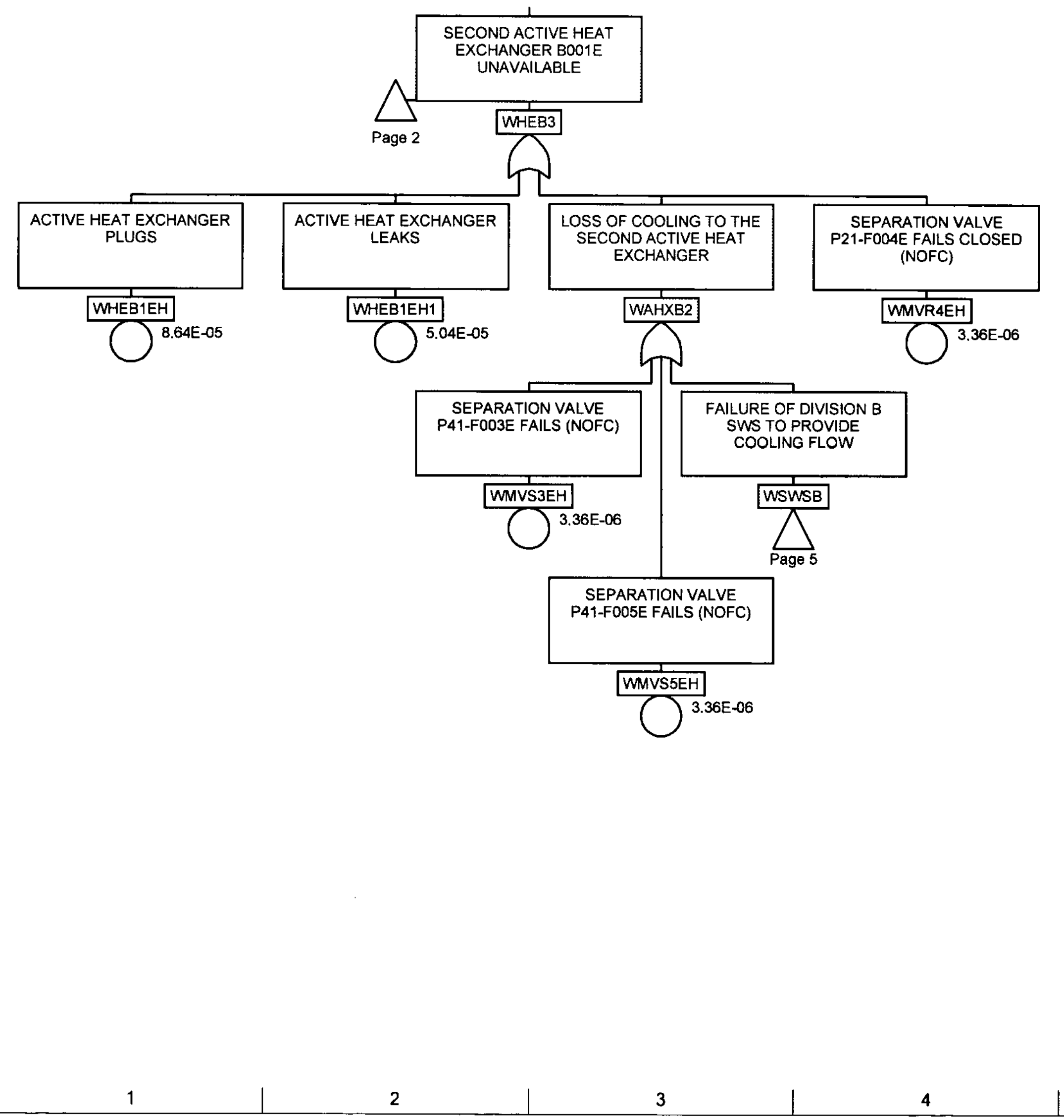

Figure 19D.6-14B Reactor Building Cooling and Service Water Fault Tree (B)

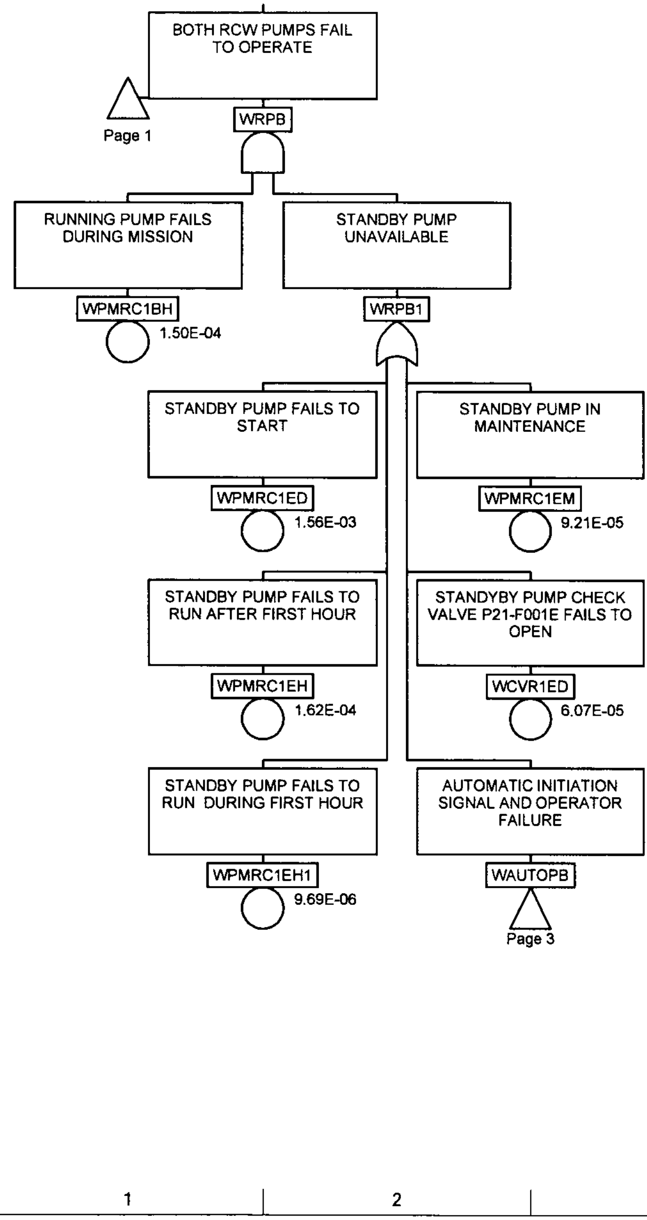

Figure 19D.6-14B Reactor Building Cooling and Service Water Fault Tree (B)

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IPVL2B	3	1
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Q_FANERUN1	7	2
Q_FANEUNAVAIL	7	3
Q_FANRUNB2	6	3
Q_FANSTB	6	4
Q_FANSTE	7	3
WAHXB1	9	3
WAHXB2	10	3
WAUTB	3	2
WAUTOPB	2	2
WAUTOPB	3	2
WAUTOPB	4	2
WAUTOPB	8	2
WAUTOPB	11	3
WAVR6BH	2	3
WCVR1ED	11	3
WCVS1ED	8	2
WFANSTBYB	7	2
WFLD1BH	6	1
WFLD1EH	8	1
WGATEB	3	4
WHEB	2	2
WHEB1	2	2
WHEB1BH	9	1
WHEB1BH1	9	2
WHEB1EH	10	1
WHEB1EH1	10	2
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WMVS5BH	9	3
WMVS5EH	10	3
WMVS5HH	4	1
WOPERR	3	3
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WPMPCCFRCWABC	1	1
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WPMPCCFRCWBC	1	1
WPMPCCFRSWAB	5	3
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WPMRC1EH	11	2
WPMRC1EH1	11	2
WPMRC1EM	11	3
WPMSC1BH	6	2
WPMSC1ED	8	2
WPMSC1EH	8	1
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WPMSC1EM	8	2
WPPREBH	1	2
WPPSWSBH	5	1
WPUMPB	6	2
WPUMPB1	5	6
WPUMPB1	6	2
WRCWB	1	2
WRPB	1	2
WRPB	11	2
WRPB1	11	2
WSHXB	2	2
WSHXB	4	2
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WSPB1	8	2
WSPB2	5	7
WSPB2	7	2
WSWSB	4	2
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Figure 19D.6-14B Reactor Building Cooling and Service Water Fault Tree (B)

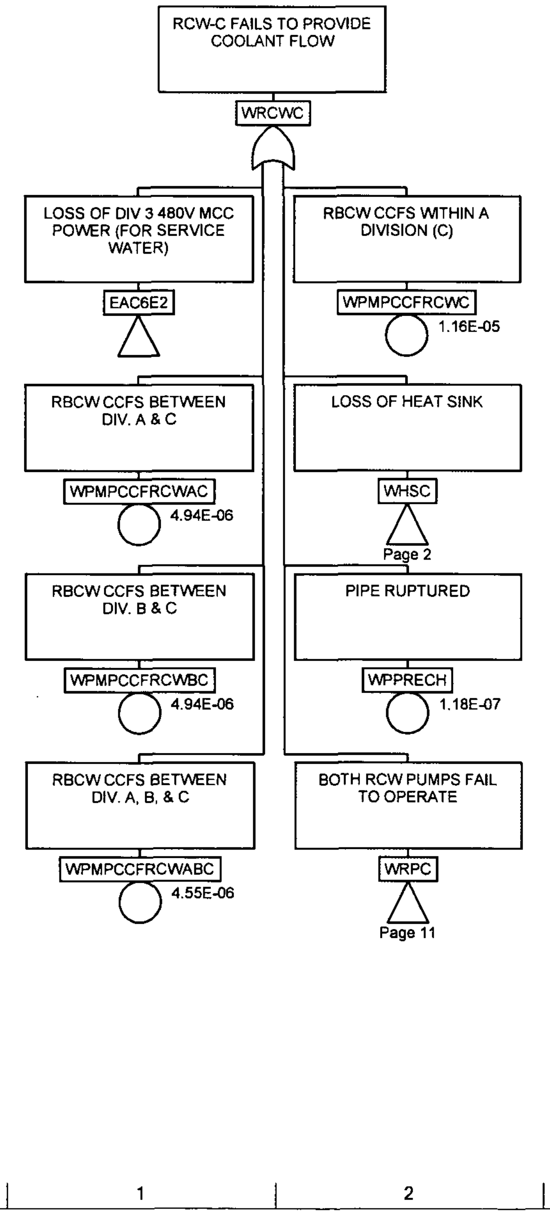

Figure 19D.6-14C Reactor Building Cooling and Service Water Fault Trees (C)

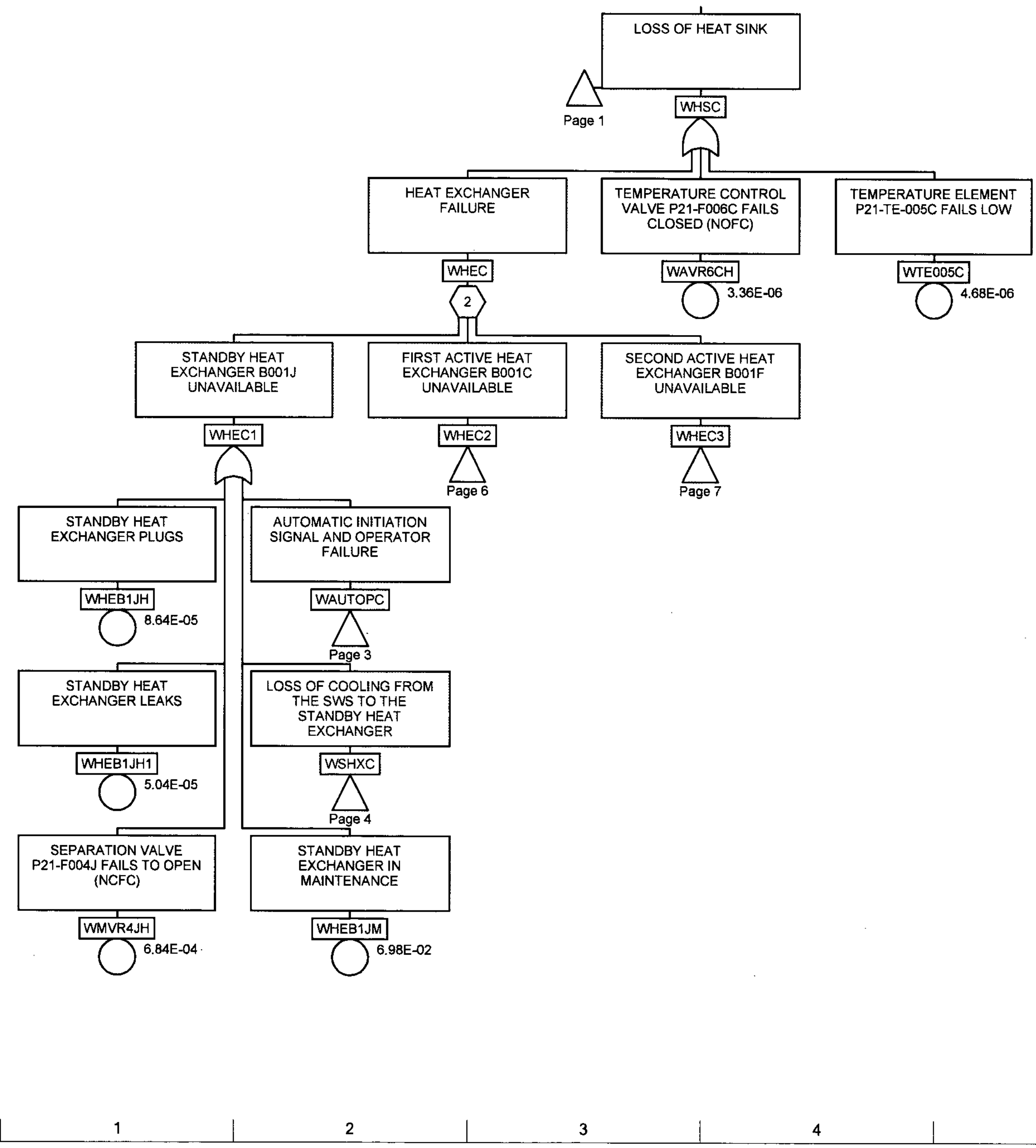

Figure 19D.6-14C Reactor Building Cooling and Service Water Fault Trees (C)

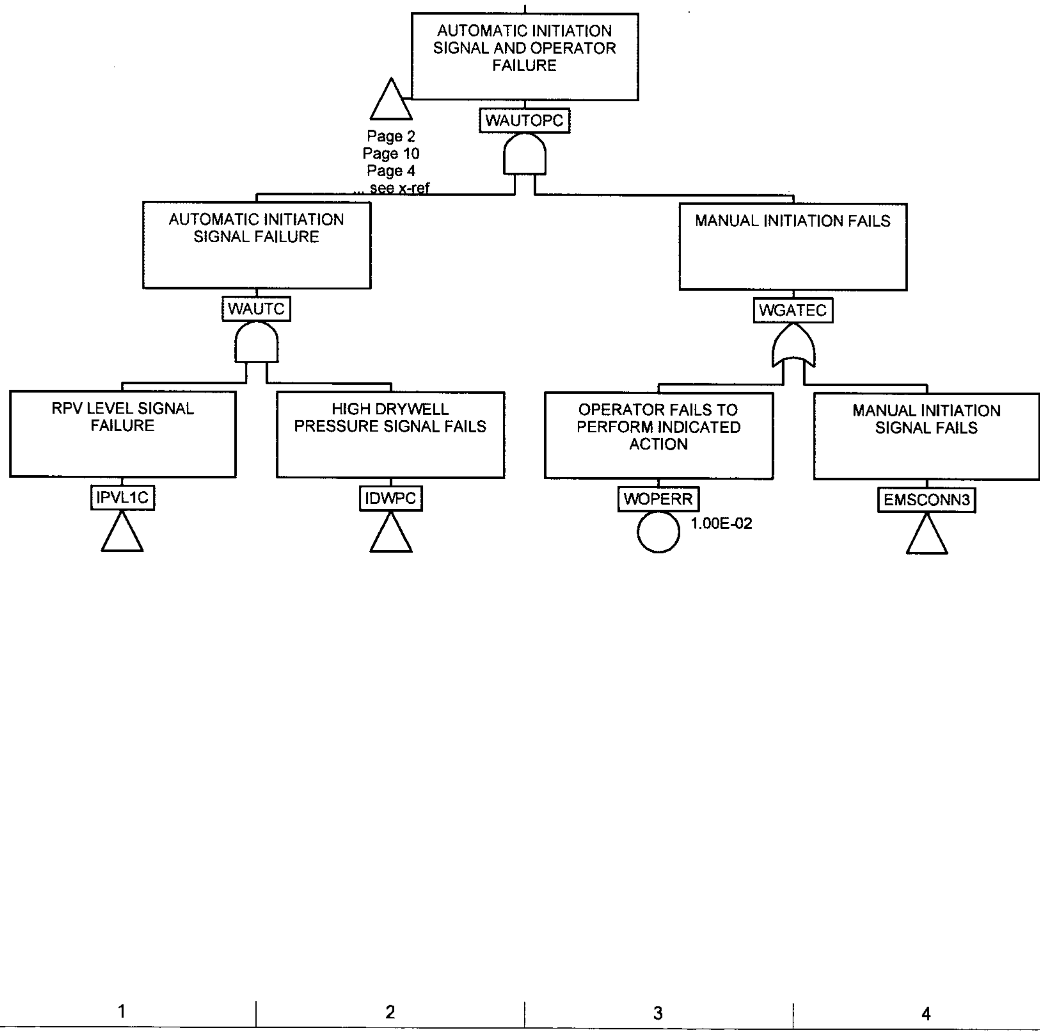

Figure 19D.6-14C Reactor Building Cooling and Service Water Fault Trees (C)

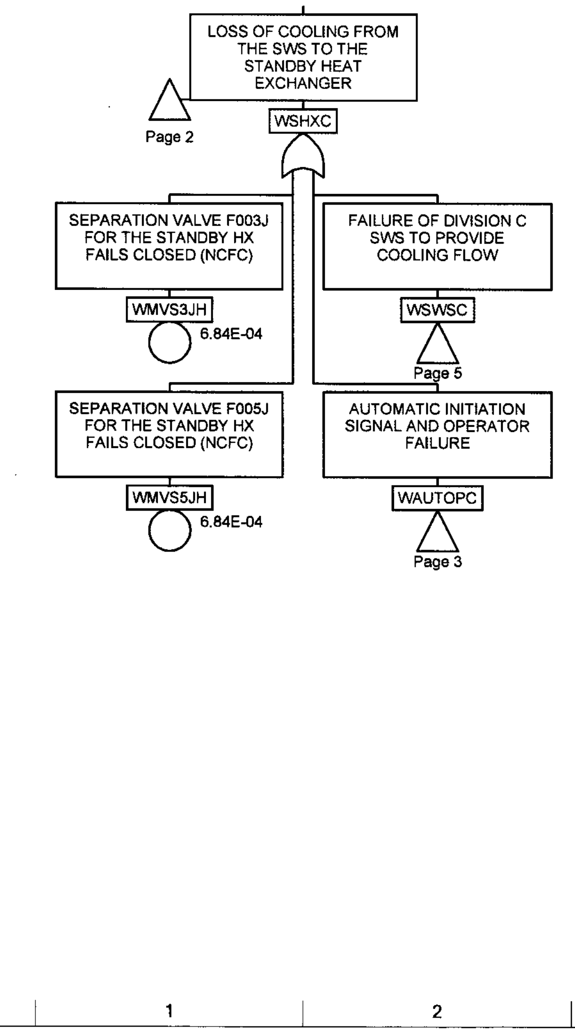

Figure 19D.6-14C Reactor Building Cooling and Service Water Fault Trees (C)

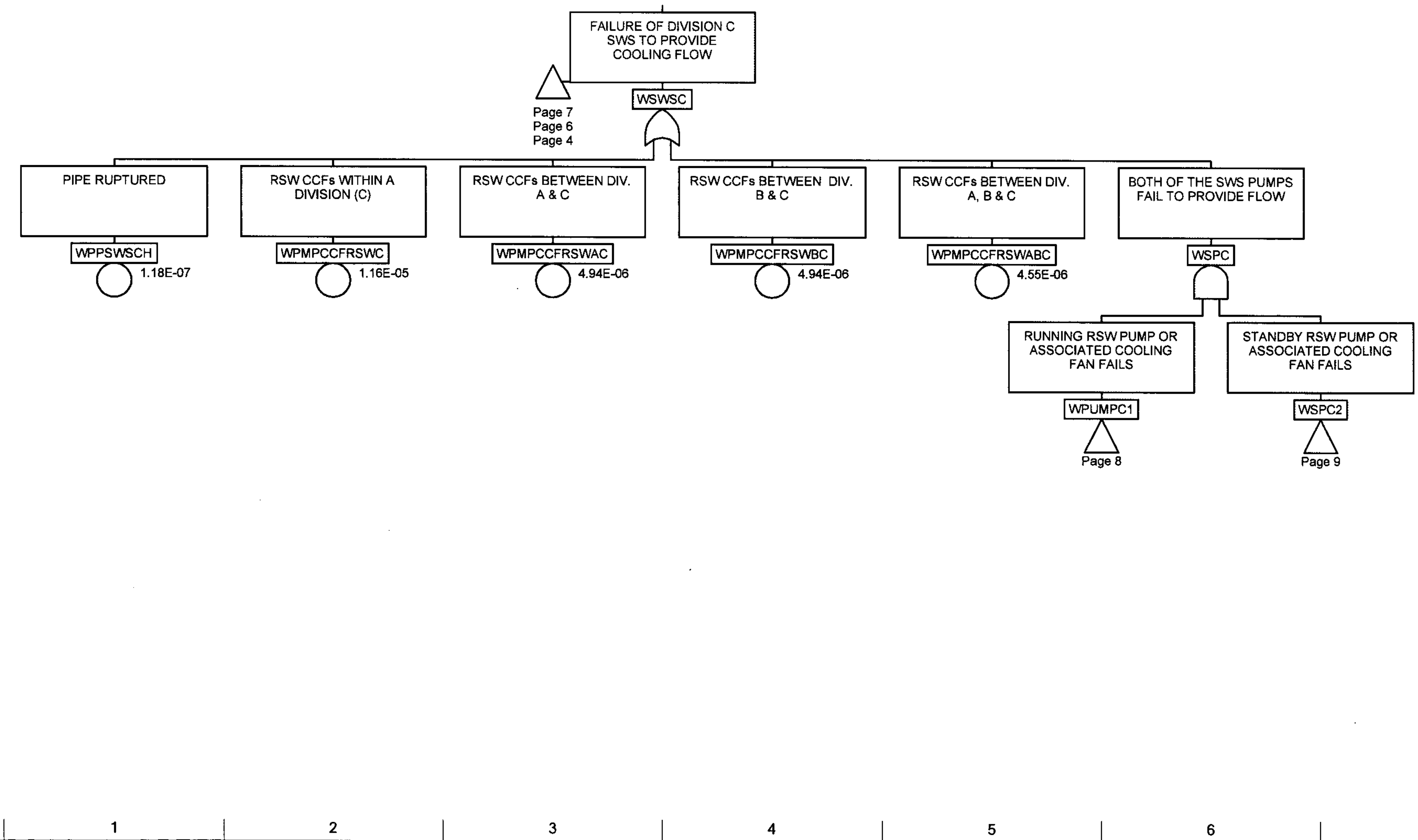

Figure 19D.6-14C Reactor Building Cooling and Service Water Fault Trees (C)

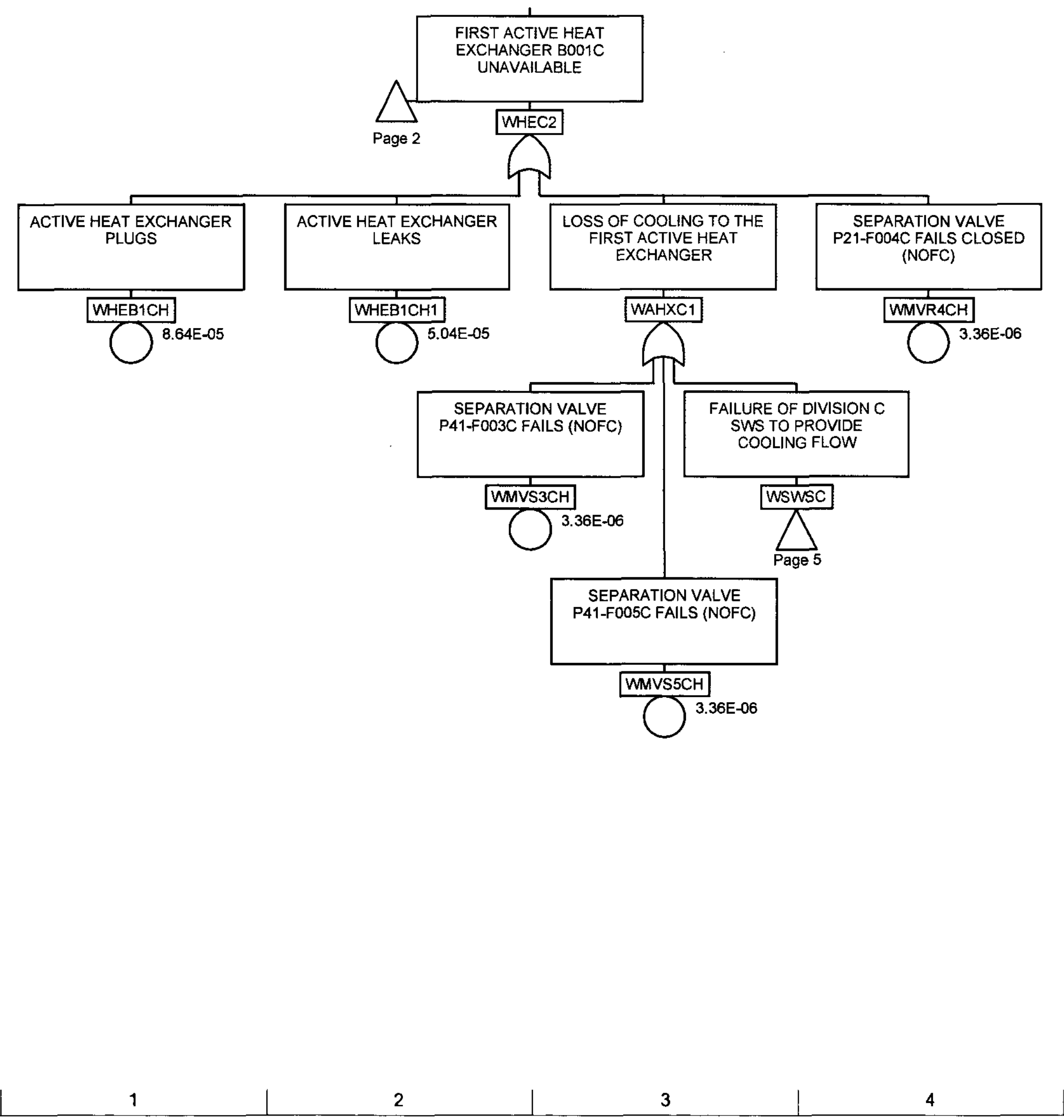

Figure 19D.6-14C Reactor Building Cooling and Service Water Fault Trees (C)

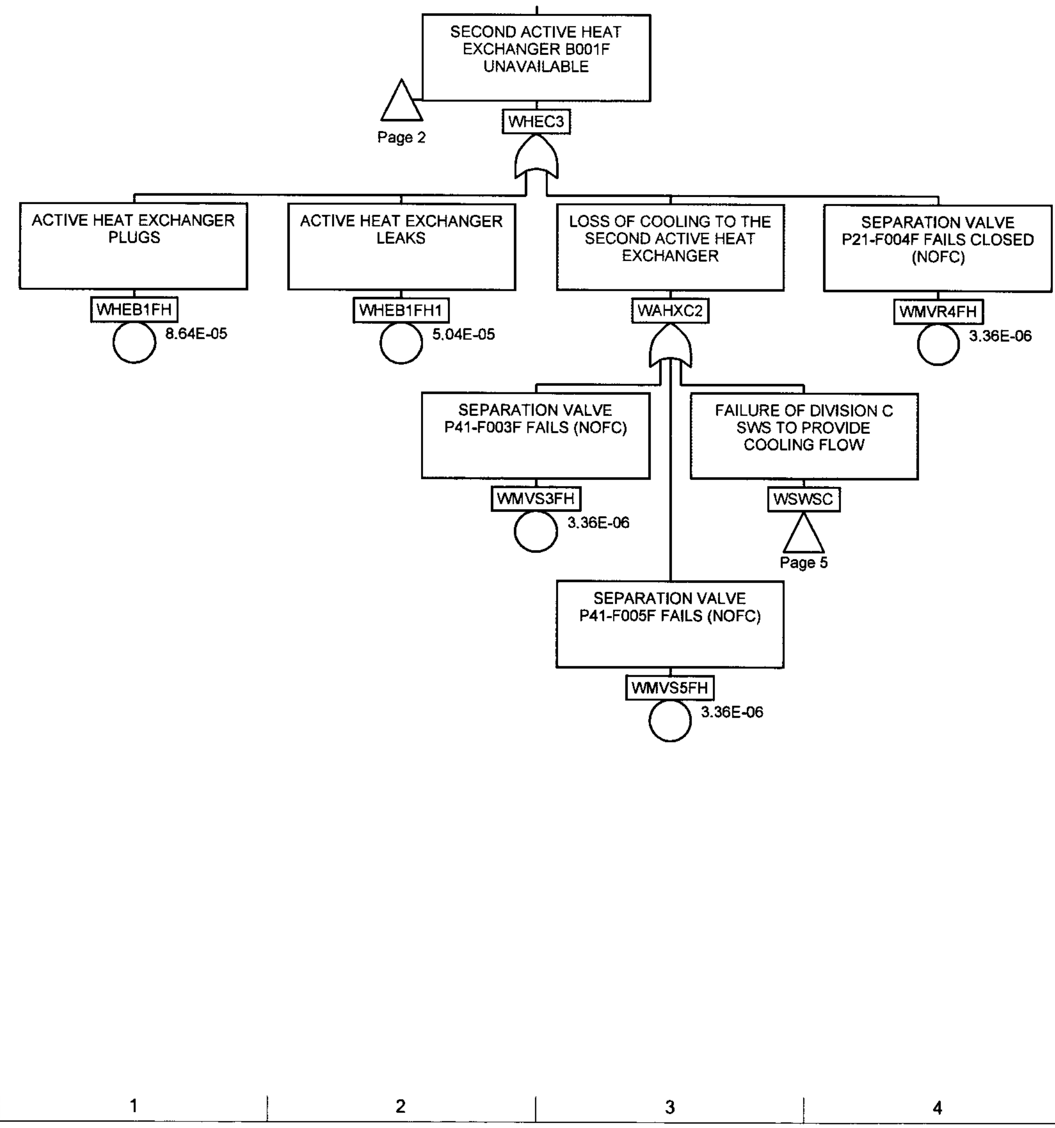

Figure 19D.6-14C Reactor Building Cooling and Service Water Fault Trees (C)

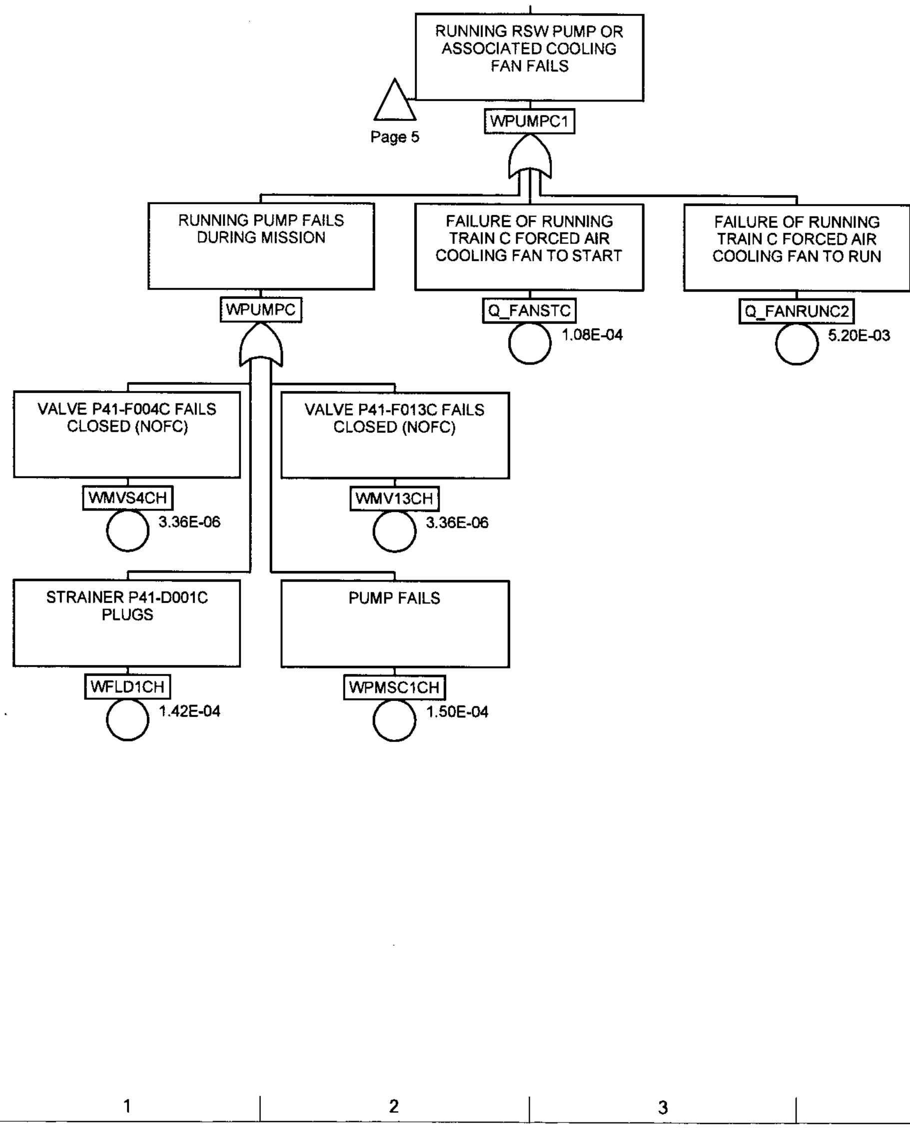

Figure 19D.6-14C Reactor Building Cooling and Service Water Fault Trees (C)

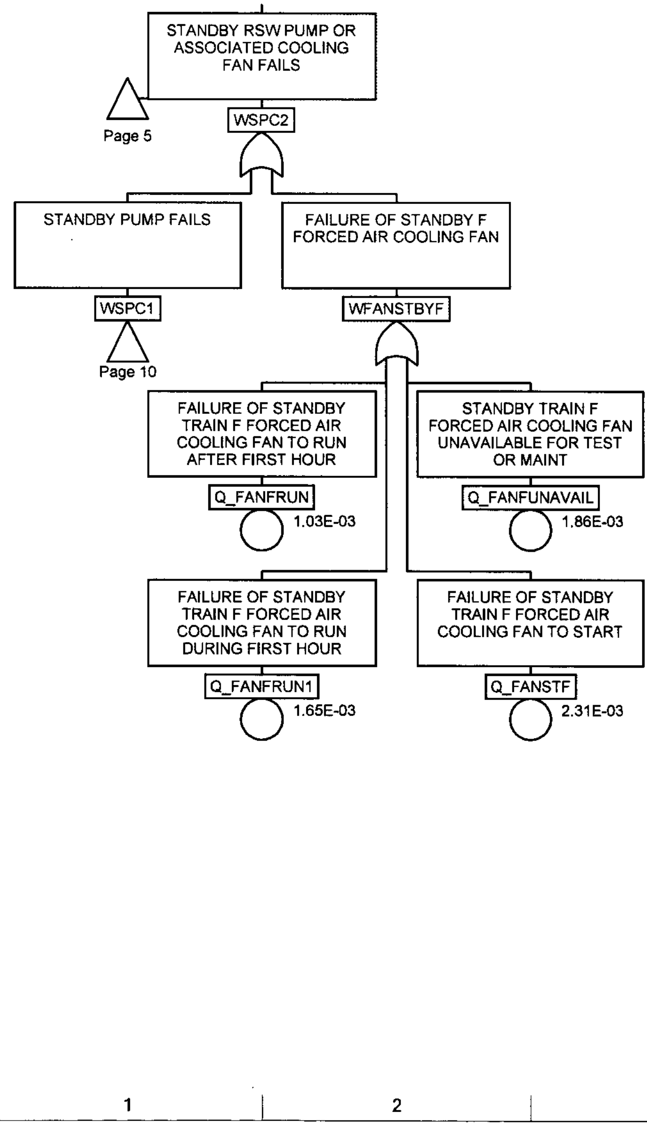

Figure 19D.6-14C Reactor Building Cooling and Service Water Fault Trees (C)

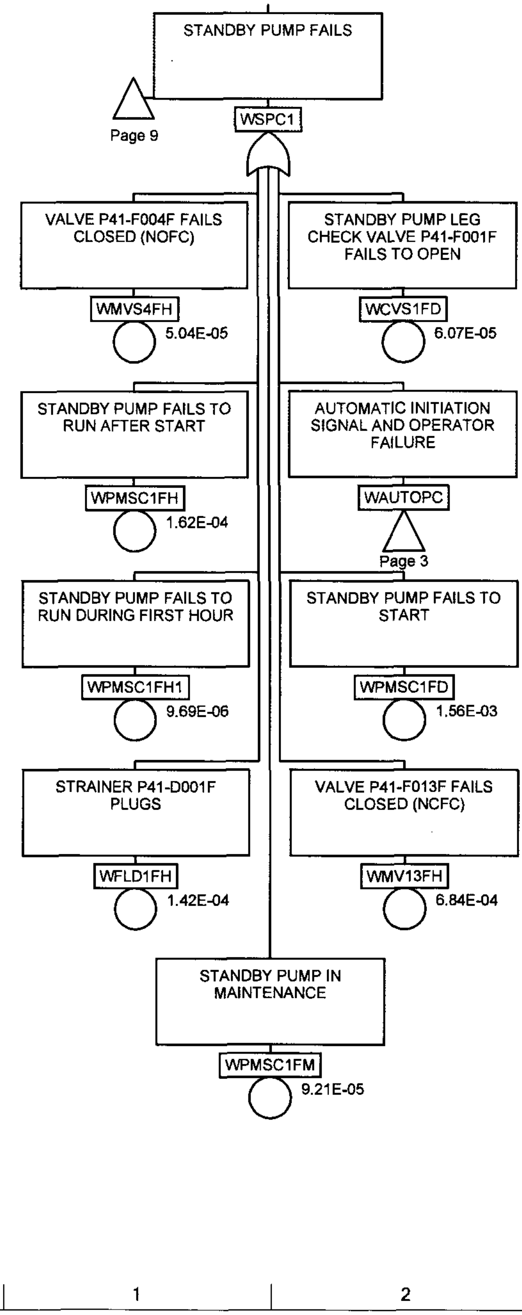

Figure 19D.6-14C Reactor Building Cooling and Service Water Fault Trees (C)

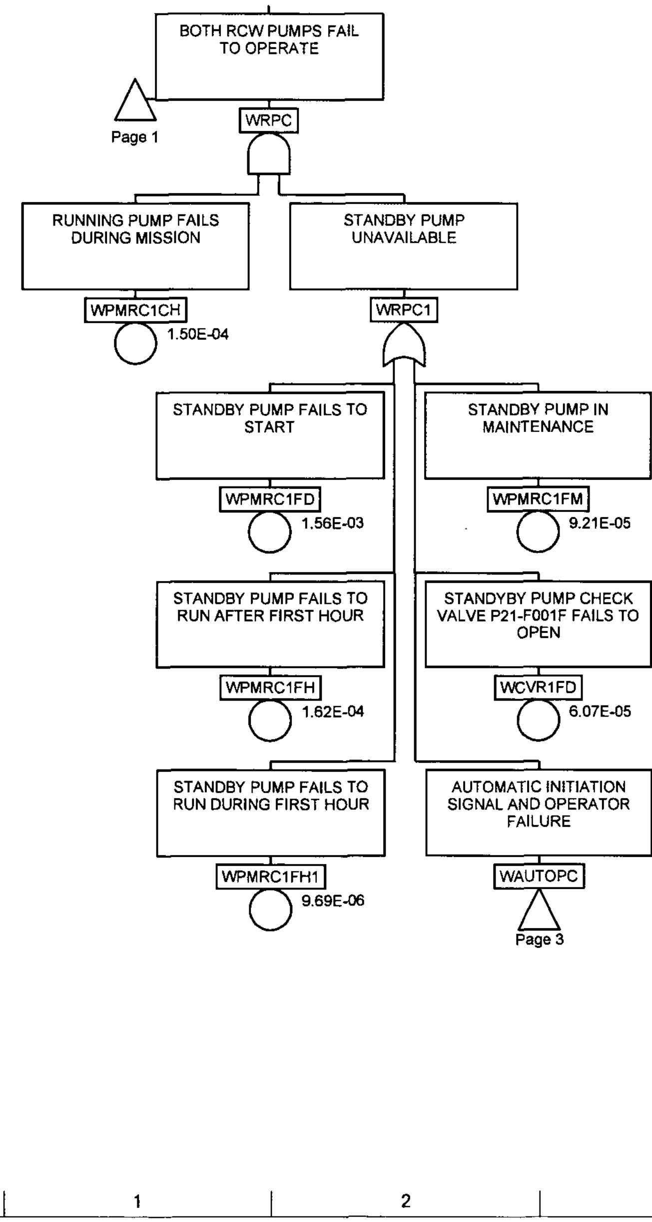

Figure 19D.6-14C Reactor Building Cooling and Service Water Fault Trees (C)

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EMSCONN3	3	4	WMVS4FH	10	1
IDWPC	3	2	WMVS5CH	6	3
IPVL1C	3	1	WMVS5FH	7	3
Q_FANFRUN	9	2	WMVS5JH	4	1
Q_FANFRUN1	9	2	WOPERR	3	3
Q_FANFUNAVAIL	9	3	WPMPCCFRCWABC	1	1
Q_FANRUNC2	8	4	WPMPCCFRCWAC	1	1
Q_FANSTC	8	3	WPMPCCFRCWBC	1	1
Q_FANSTF	9	3	WPMPCCFRCWC	1	2
WAHXC1	6	3	WPMPCCFRSWABC	5	5
WAHXC2	7	3	WPMPCCFRSWAC	5	3
WAUTC	3	2	WPMPCCFRSWBC	5	4
WAUTOPC	2	2	WPMPCCFRSWC	5	2
WAUTOPC	3	2	WPMRC1CH	11	1
WAUTOPC	4	2	WPMRC1FD	11	2
WAUTOPC	10	2	WPMRC1FH	11	2
WAUTOPC	11	3	WPMRC1FH1	11	2
WAVR6CH	2	3	WPMRC1FM	11	3
WCVR1FD	11	3	WPMSC1CH	8	2
WCVS1FD	10	2	WPMSC1FD	10	2
WFANSTBYF	9	2	WPMSC1FH	10	1
WFLD1CH	8	1	WPMSC1FH1	10	1
WFLD1FH	10	1	WPMSC1FM	10	2
WGATEC	3	4	WPPRECH	1	2
WHEB1CH	6	1	WPPSWSCH	5	1
WHEB1CH1	6	2	WPUMPC	8	2
WHEB1FH	7	1	WPUMPC1	5	6
WHEB1FH1	7	2	WPUMPC1	8	2
WHEB1JH	2	1	WRCWC	1	2
WHEB1JH1	2	1	WRPC	1	2
WHEB1JM	2	2	WRPC	11	2
WHEC	2	2	WRPC1	11	2
WHEC1	2	2	WSHXC	2	2
WHEC2	2	3	WSHXC	4	2
WHEC2	6	3	WSPC	5	6
WHEC3	2	4	WSPC1	9	1
WHEC3	7	3	WSPC1	10	2
WHSC	1	2	WSPC2	5	7
WHSC	2	3	WSPC2	9	2
WMV13CH	8	2	WSWSC	4	2
WMV13FH	10	2	WSWSC	5	4
WMVR4CH	6	4	WSWSC	6	4
WMVR4FH	7	4	WSWSC	7	4
WMVR4JH	2	1	WTE005C	2	4
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Figure 19D.6-14C Reactor Building Cooling and Service Water Fault Trees (C)

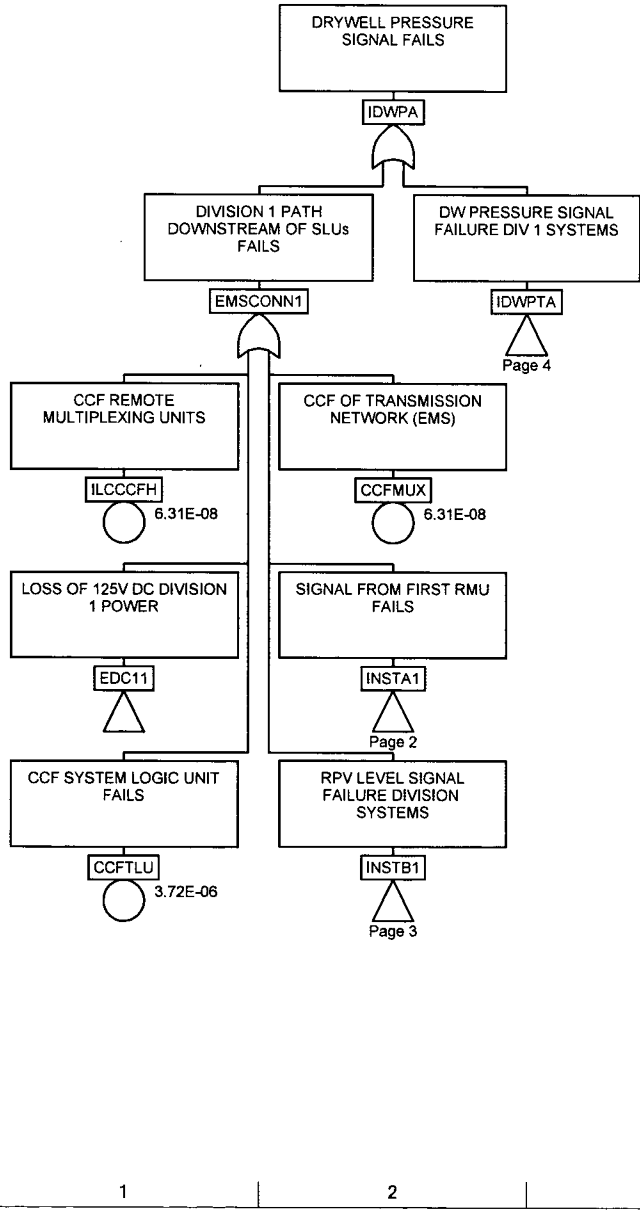

Figure 19D.6-15A High Drywell Pressure Instrumentation System Fault Tree Div. 1

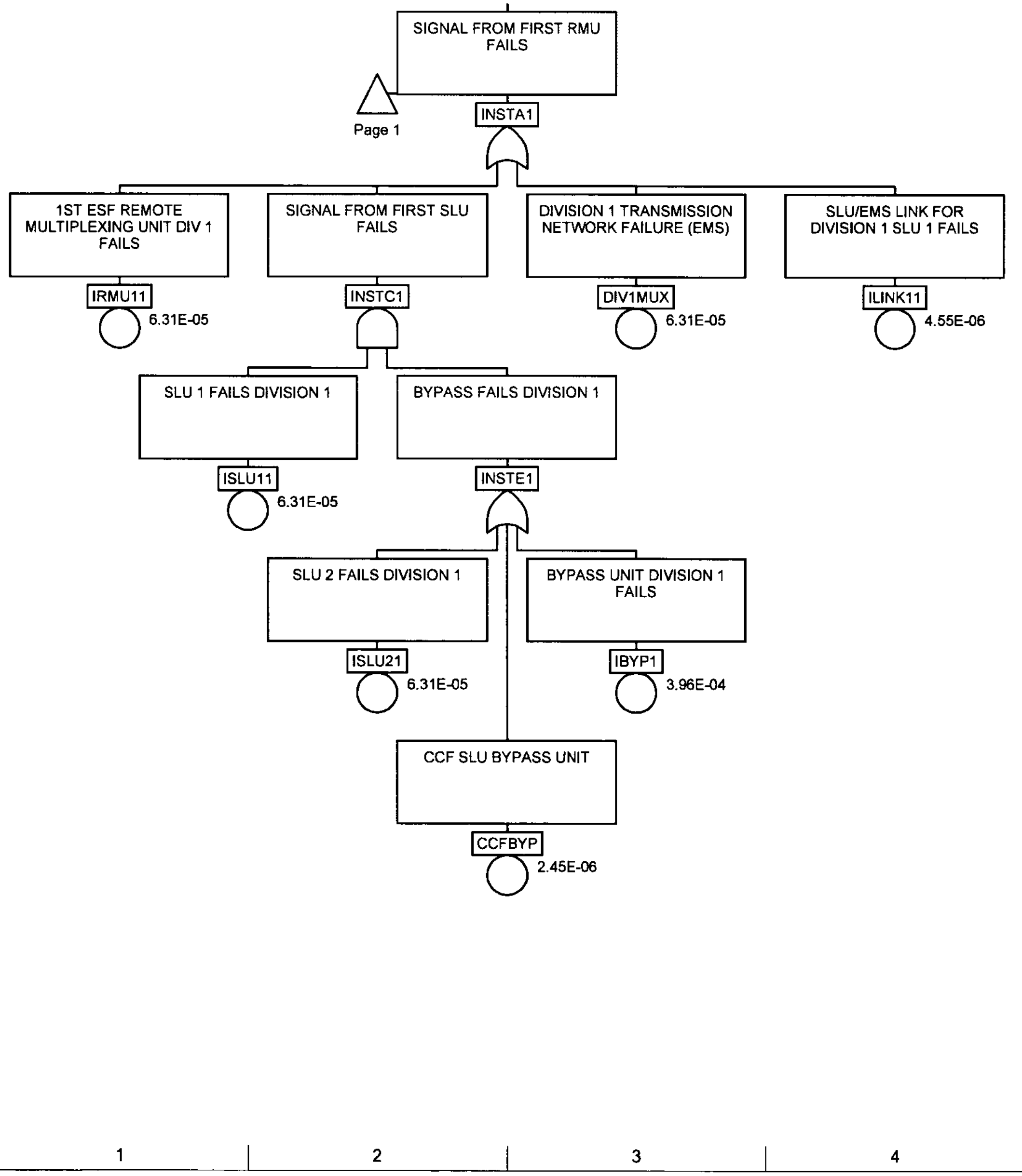

Figure 19D.6-15A High Drywell Pressure Instrumentation System Fault Tree Div. 1

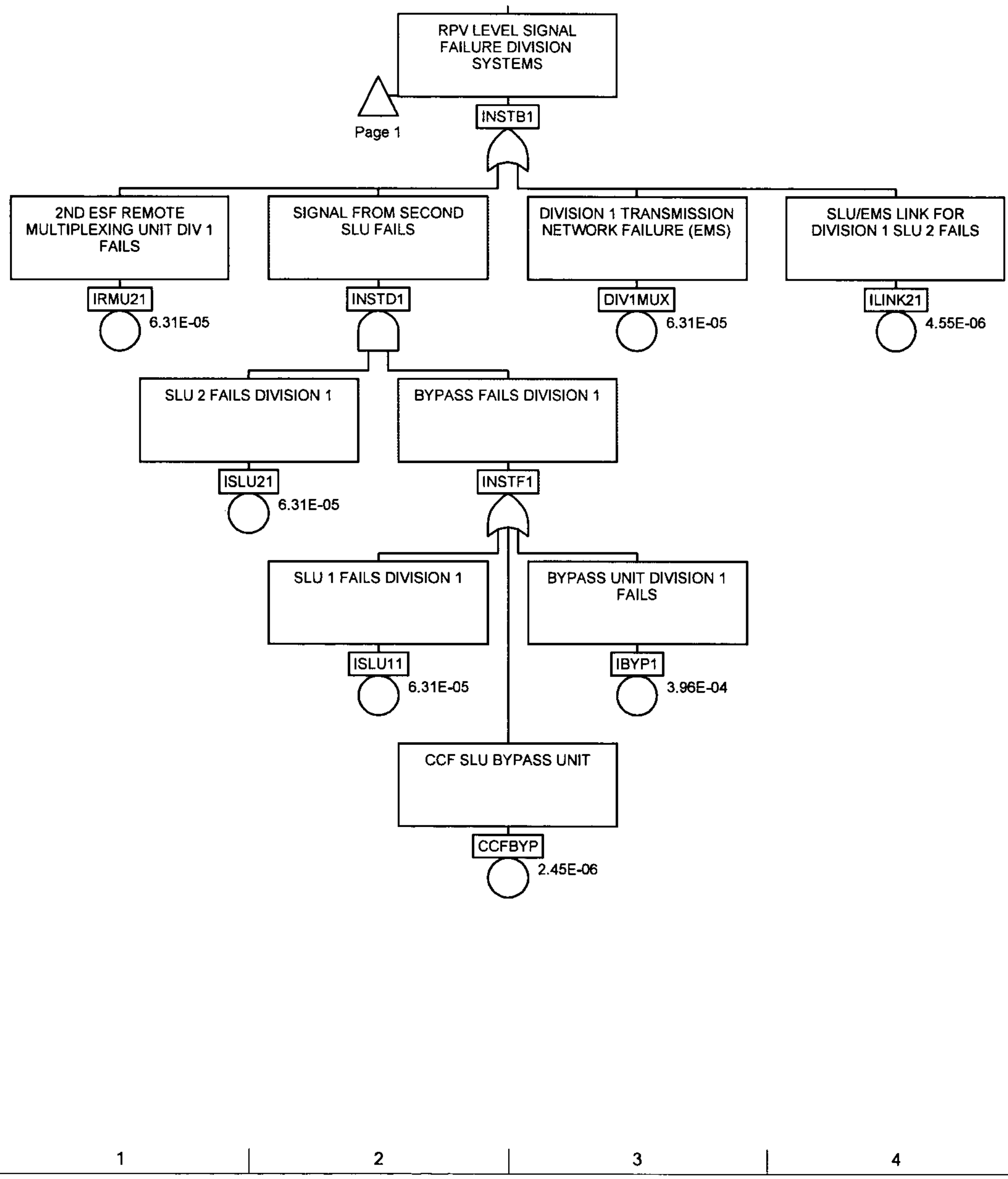

Figure 19D.6-15A High Drywell Pressure Instrumentation System Fault Tree Div. 1

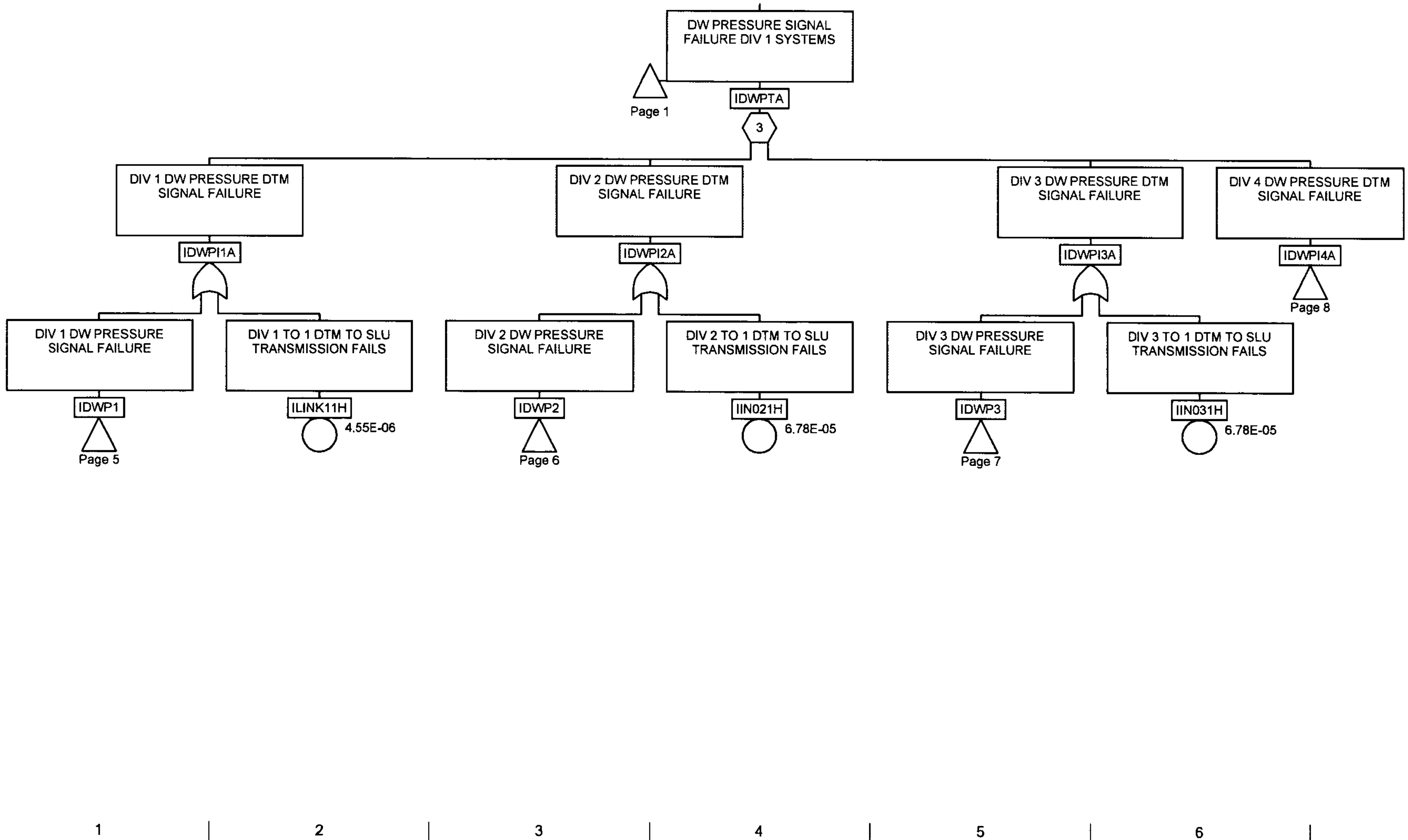

Figure 19D.6-15A High Drywell Pressure Instrumentation System Fault Tree Div. 1

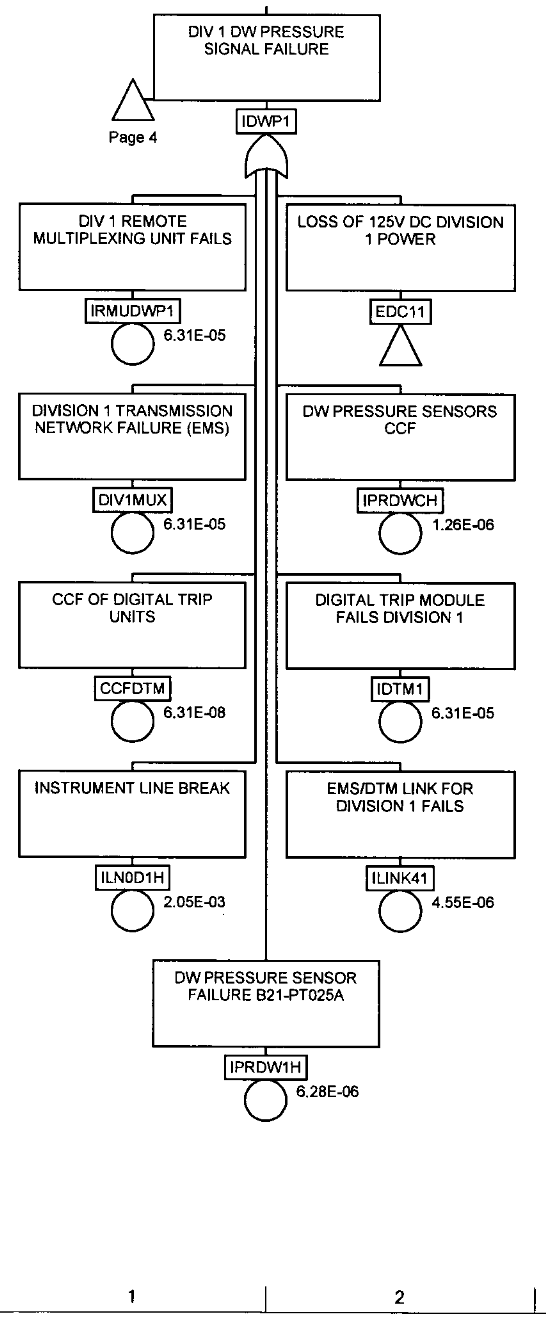

Figure 19D.6-15A High Drywell Pressure Instrumentation System Fault Tree Div. 1

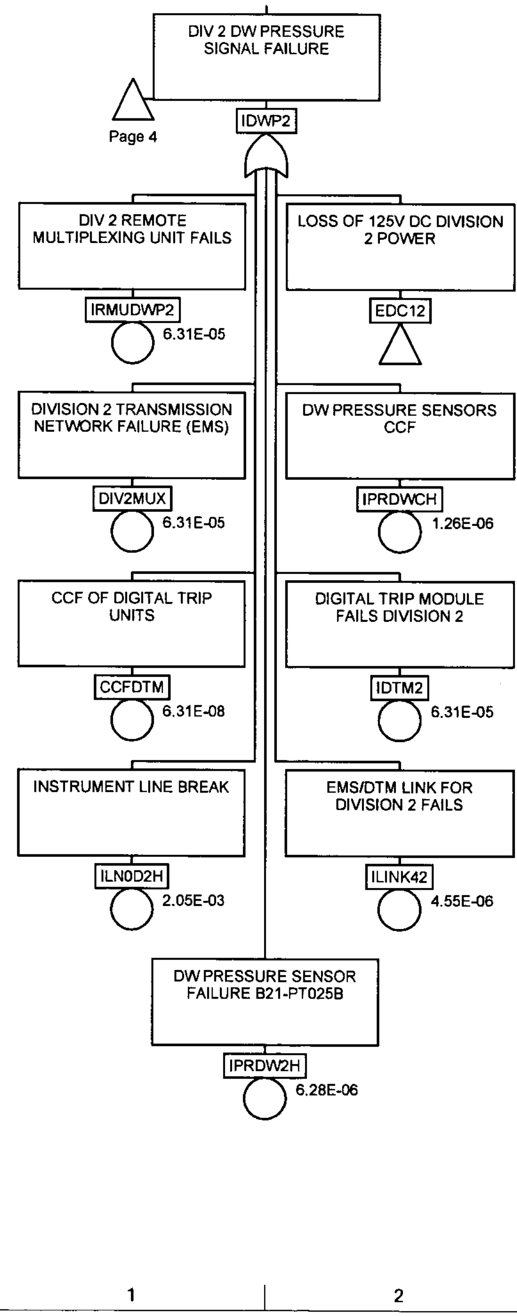

Figure 19D.6-15A High Drywell Pressure Instrumentation System Fault Tree Div. 1

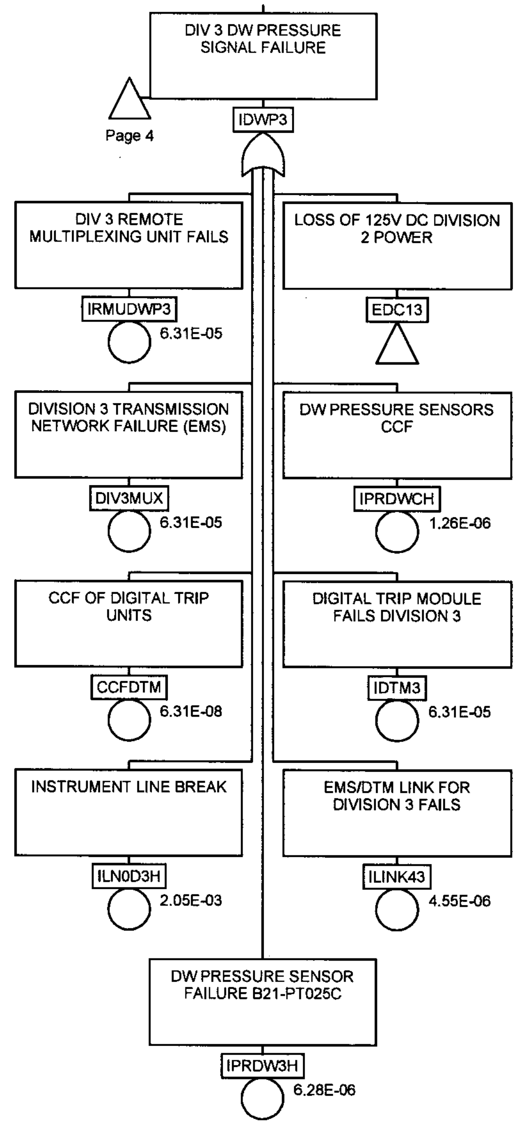

Figure 19D.6-15A High Drywell Pressure Instrumentation System Fault Tree Div. 1

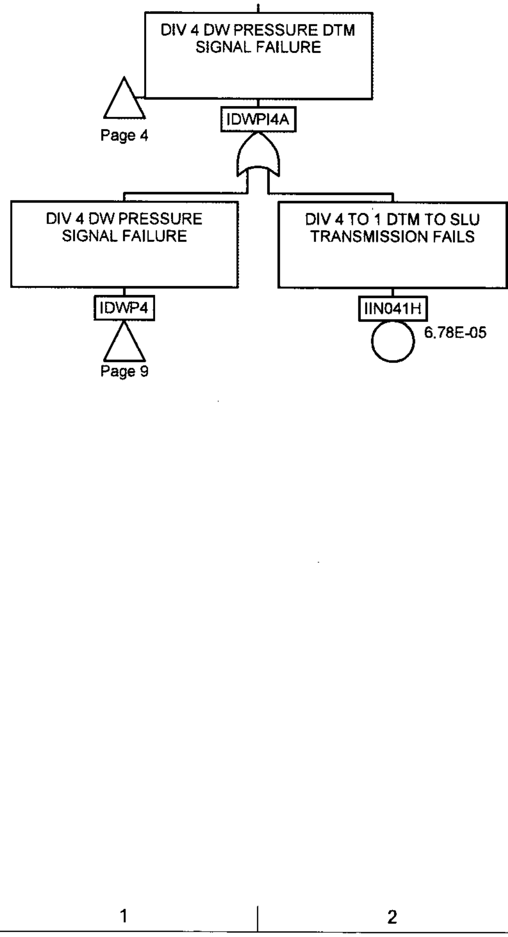

Figure 19D.6-15A High Drywell Pressure Instrumentation System Fault Tree Div. 1

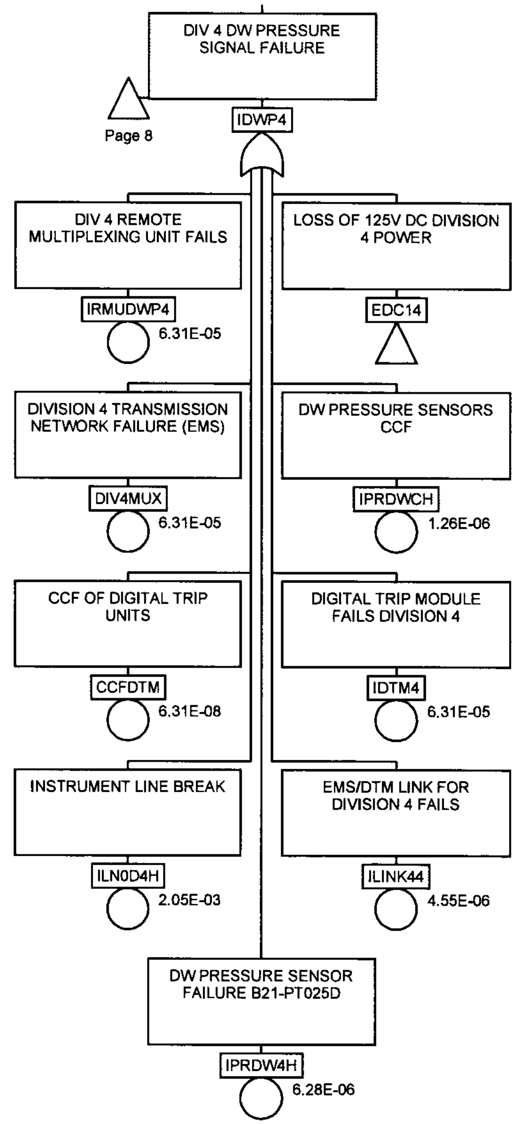

Figure 19D.6-15A High Drywell Pressure Instrumentation System Fault Tree Div. 1

Name	Page	Zone	Name	Page	Zone
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CCFDTM	5	1	ILINK42	6	2
CCFDTM	6	1	ILINK43	7	2
CCFDTM	7	1	ILINK44	9	2
CCFDTM	9	1	ILN0D1H	5	1
CCFMUX	1	2	ILN0D2H	6	1
CCFTLU	1	1	ILN0D3H	7	1
DIV1MUX	2	3	ILN0D4H	9	1
DIV1MUX	3	3	INSTA1	1	2
DIV1MUX	5	1	INSTA1	2	3
DIV2MUX	6	1	INSTB1	1	2
DIV3MUX	7	1	INSTB1	3	3
DIV4MUX	9	1	INSTC1	2	2
EDC11	1	1	INSTD1	3	2
EDC11	5	2	INSTE1	2	3
EDC12	6	2	INSTF1	3	3
EDC13	7	2	IPRDW1H	5	2
EDC14	9	2	IPRDW2H	6	2
EMSCONN1	1	2	IPRDW3H	7	2
IBYP1	2	3	IPRDW4H	9	2
IBYP1	3	3	IPRDWCH	5	2
IDTM1	5	2	IPRDWCH	6	2
IDTM2	6	2	IPRDWCH	7	2
IDTM3	7	2	IPRDWCH	9	2
IDTM4	9	2	IRMU11	2	1
IDWP1	4	1	IRMU21	3	1
IDWP1	5	2	IRMUDWP1	5	1
IDWP2	4	3	IRMUDWP2	6	1
IDWP2	6	2	IRMUDWP3	7	1
IDWP3	4	5	IRMUDWP4	9	1
IDWP3	7	2	ISLU11	2	2
IDWP4	8	1	ISLU11	3	2
IDWP4	9	2	ISLU21	2	2
IDWPA	1	2	ISLU21	3	2
IDWPI1A	4	2			
IDWPI2A	4	4			
IDWPI3A	4	6			
IDWPI4A	4	7			
IDWPI4A	8	2			
IDWPTA	1	3			
IDWPTA	4	4			
IIN021H	4	4			
IIN031H	4	6			
IIN041H	8	2			
ILCCCFH	1	1			
ILINK11	2	4			
ILINK11H	4	2			

Figure 19D.6-15A High Drywell Pressure Instrumentation System Fault Tree Div. 1

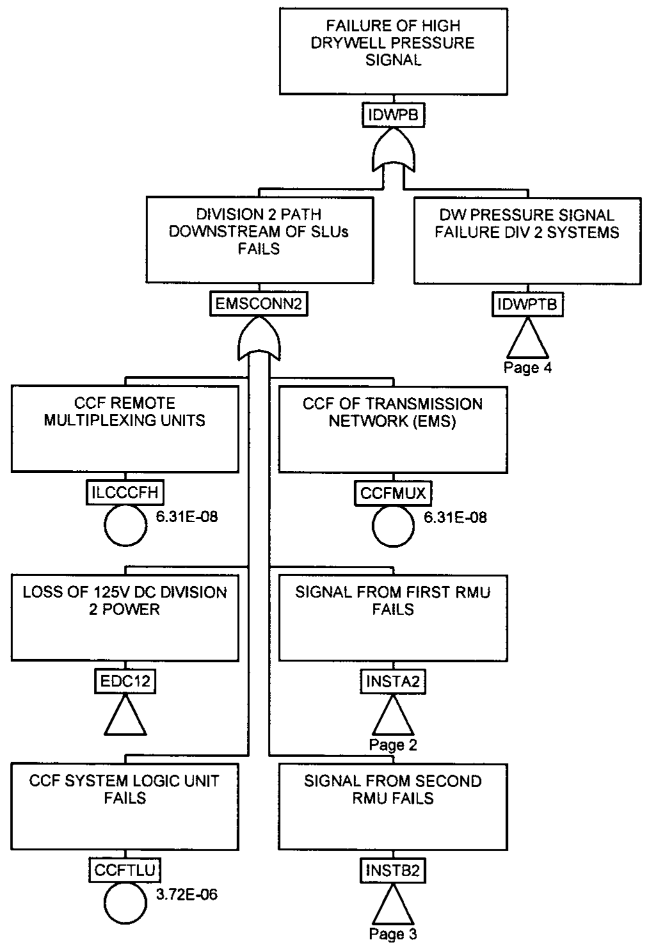

Figure 19D.6-15B High Drywell Pressure Instrumentation System Fault Tree Div. 2

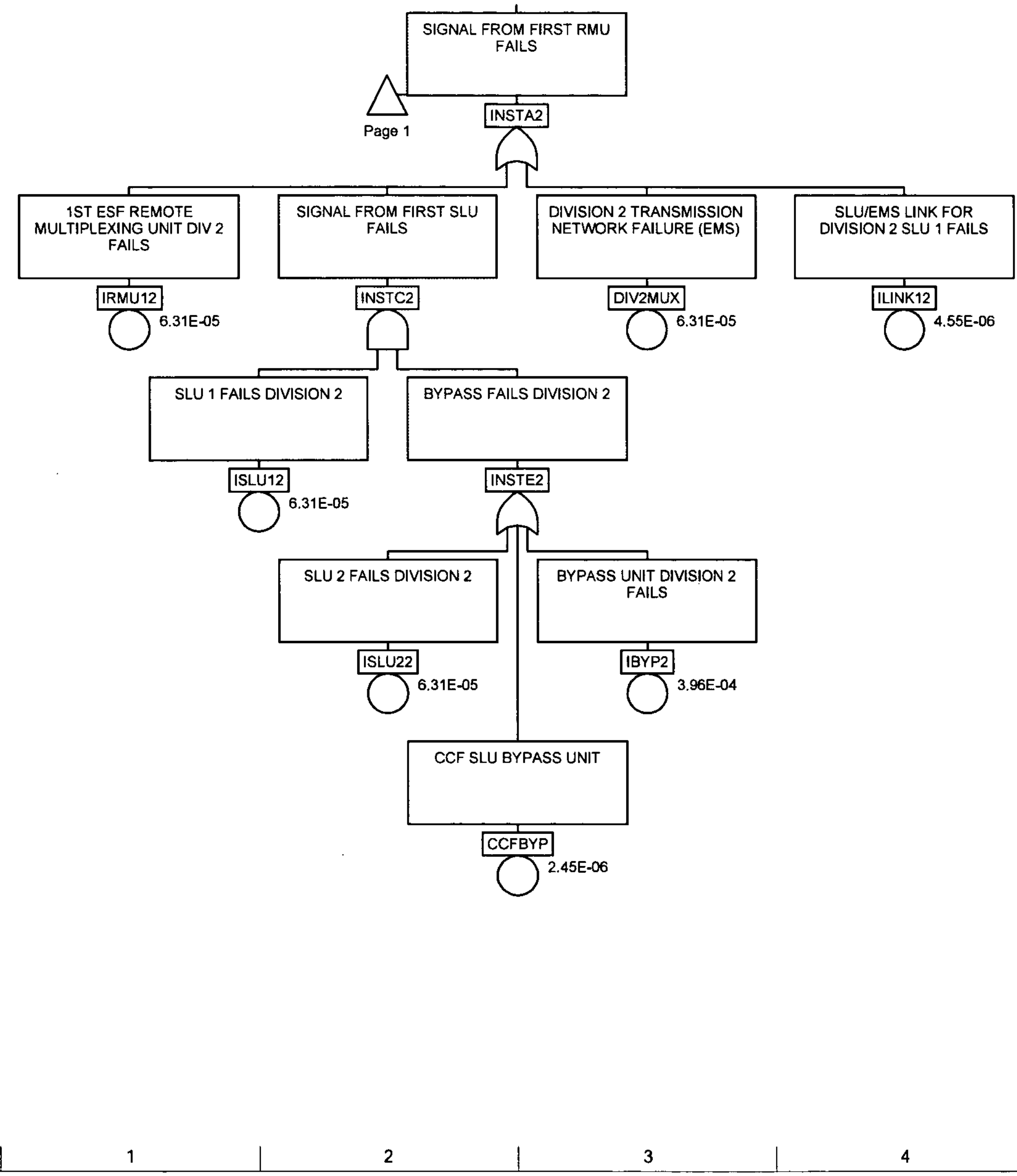

Figure 19D.6-15B High Drywell Pressure Instrumentation System Fault Tree Div. 2

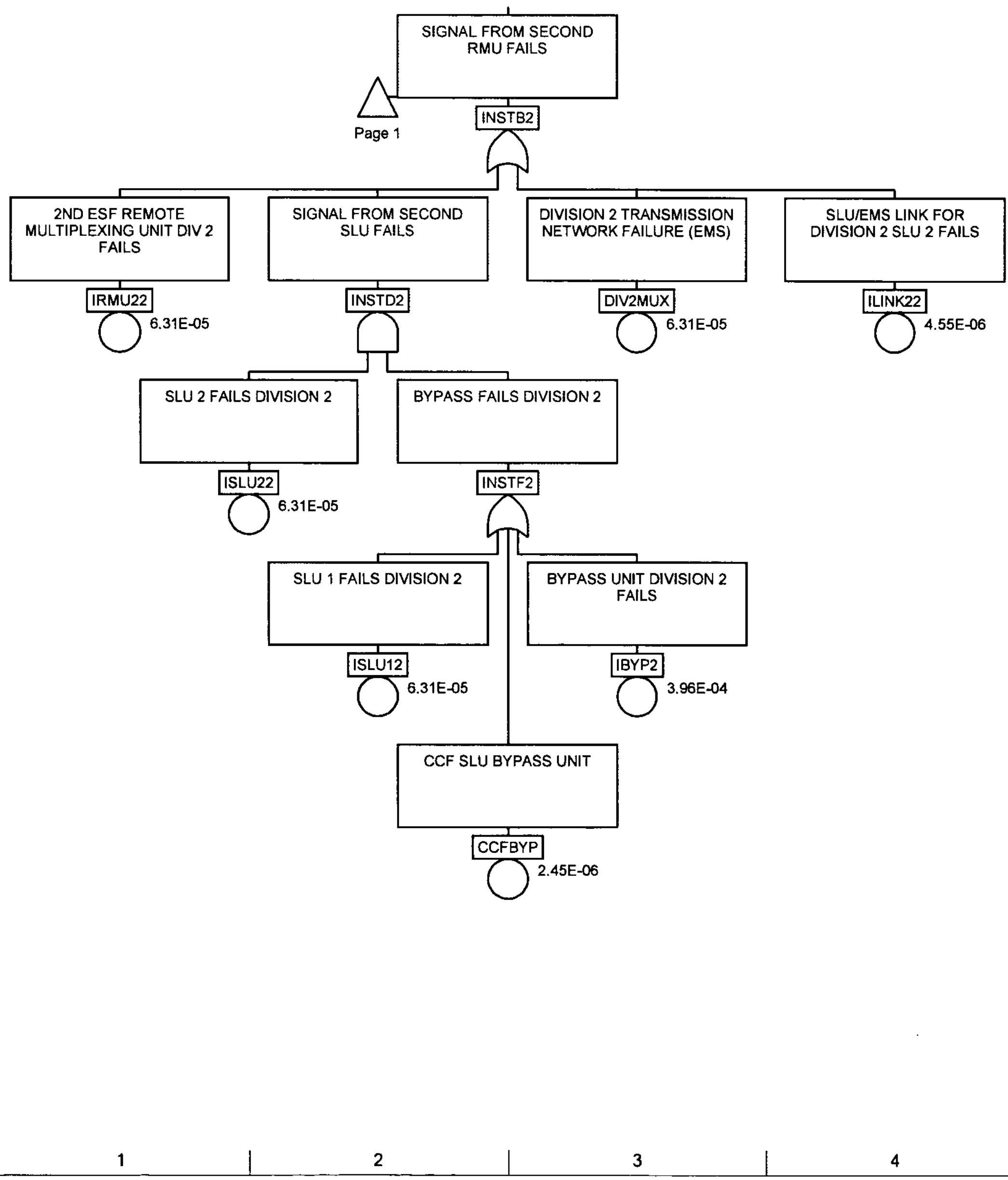

Figure 19D.6-15B High Drywell Pressure Instrumentation System Fault Tree Div. 2

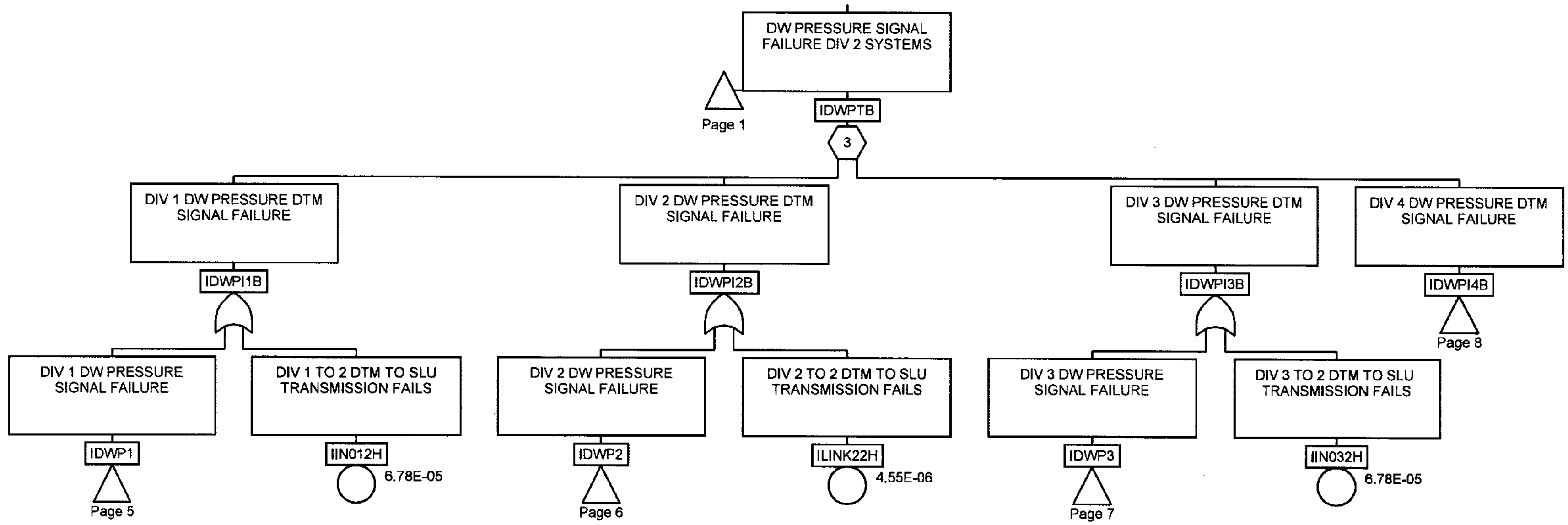

Figure 19D.6-15B High Drywell Pressure Instrumentation System Fault Tree Div. 2

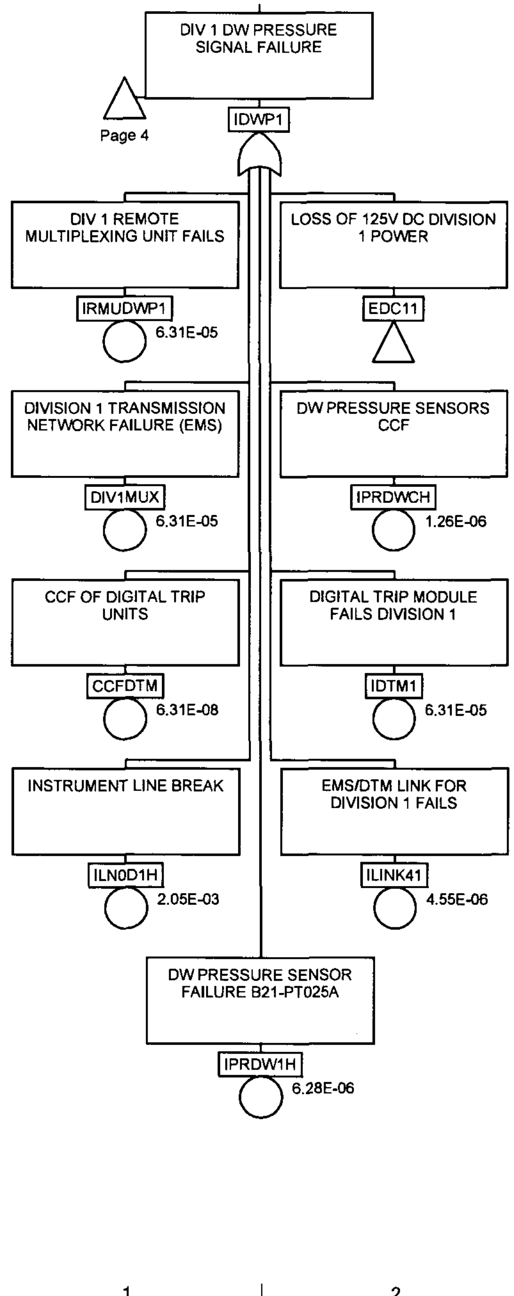

Figure 19D.6-15B High Drywell Pressure Instrumentation System Fault Tree Div. 2

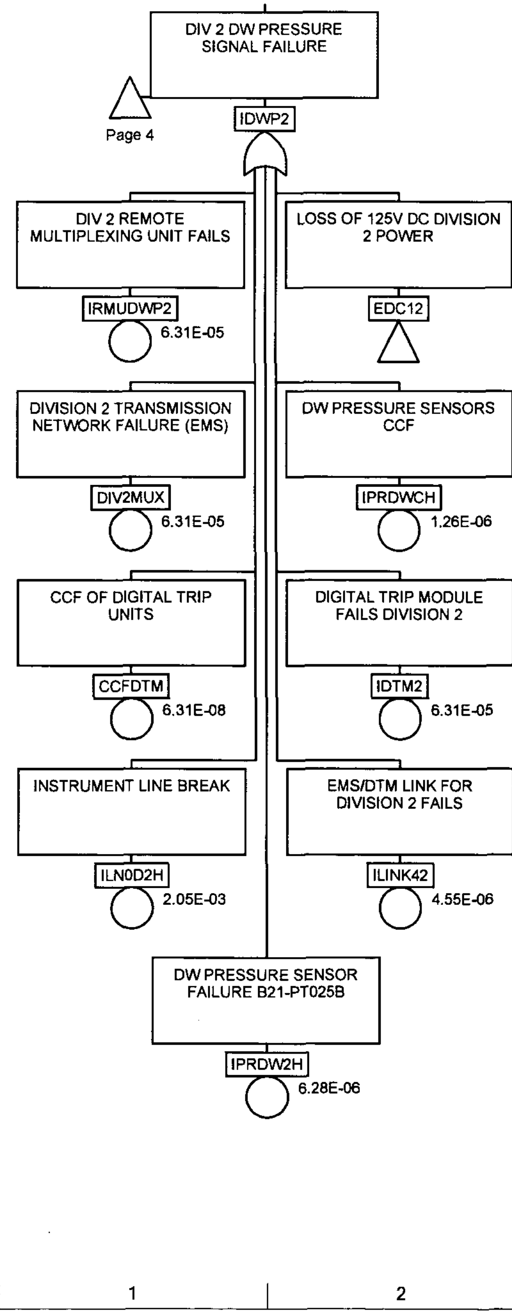

Figure 19D.6-15B High Drywell Pressure Instrumentation System Fault Tree Div. 2

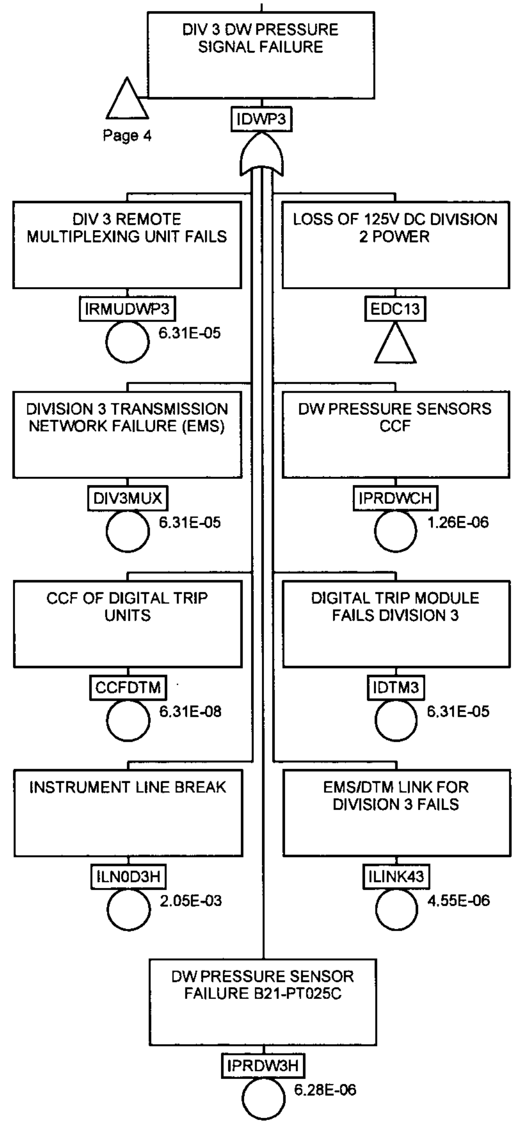

Figure 19D.6-15B High Drywell Pressure Instrumentation System Fault Tree Div. 2

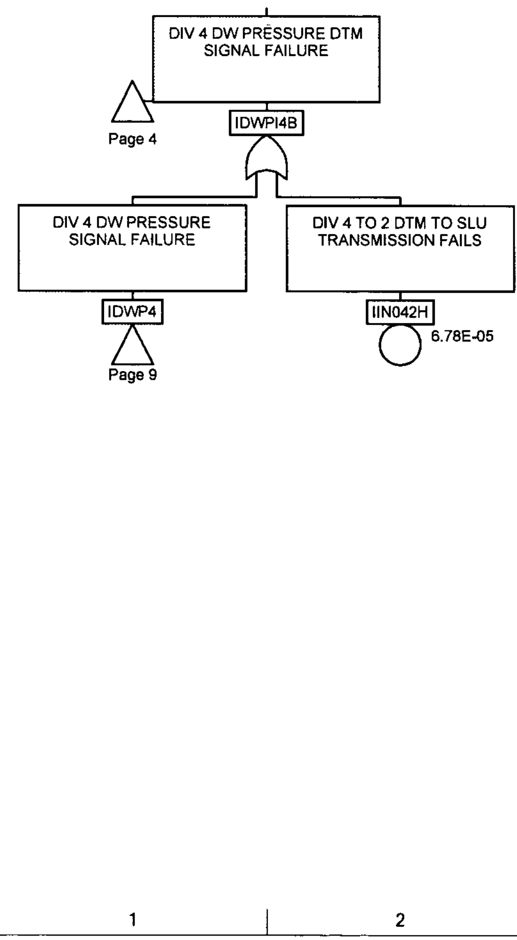

Figure 19D.6-15B High Drywell Pressure Instrumentation System Fault Tree Div. 2

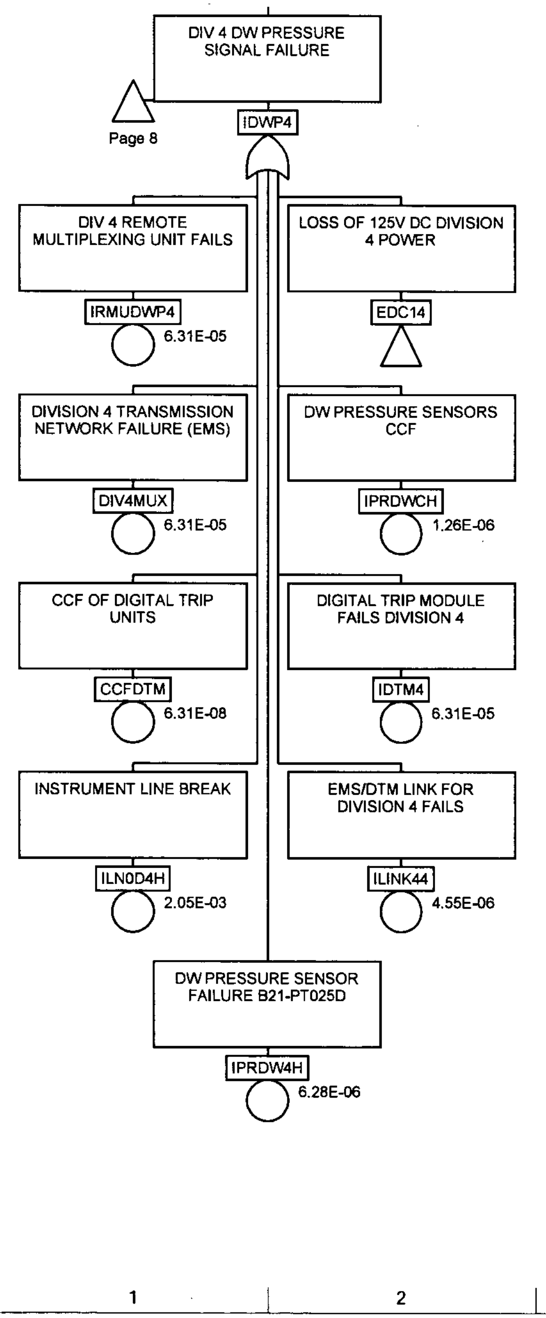

Figure 19D.6-15B High Drywell Pressure Instrumentation System Fault Tree Div. 2

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CCFDTM	6	1
CCFDTM	7	1
CCFDTM	9	1
CCFMUX	1	2
CCFTLU	1	1
DIV1MUX	5	1
DIV2MUX	2	3
DIV2MUX	3	3
DIV2MUX	6	1
DIV3MUX	7	1
DIV4MUX	9	1
EDC11	5	2
EDC12	1	1
EDC12	6	2
EDC13	7	2
EDC14	9	2
EMSCONN2	1	2
IBYP2	2	3
IBYP2	3	3
IDTM1	5	2
IDTM2	6	2
IDTM3	7	2
IDTM4	9	2
IDWP1	4	1
IDWP1	5	2
IDWP2	4	3
IDWP2	6	2
IDWP3	4	5
IDWP3	7	2
IDWP4	8	1
IDWP4	9	2
IDWPB	1	2
IDWPI1B	4	2
IDWPI2B	4	4
IDWPI3B	4	6
IDWPI4B	4	7
IDWPI4B	8	2
IDWPTB	1	3
IDWPTB	4	4
IIN012H	4	2
IIN032H	4	6
IIN042H	8	2
ILCCCFH	1	1
ILINK12	2	4
ILINK22	3	4
ILINK22H	4	4
ILINK41	5	2
ILINK42	6	2
ILINK43	7	2
ILINK44	9	2
ILN0D1H	5	1
ILN0D2H	6	1
ILN0D3H	7	1
ILN0D4H	9	1
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INSTE2	2	3
INSTF2	3	3
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IPRDW2H	6	2
IPRDW3H	7	2
IPRDW4H	9	2
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IPRDWCH	6	2
IPRDWCH	7	2
IPRDWCH	9	2
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IRMU22	3	1
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IRMUDWP2	6	1
IRMUDWP3	7	1
IRMUDWP4	9	1
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ISLU22	2	2
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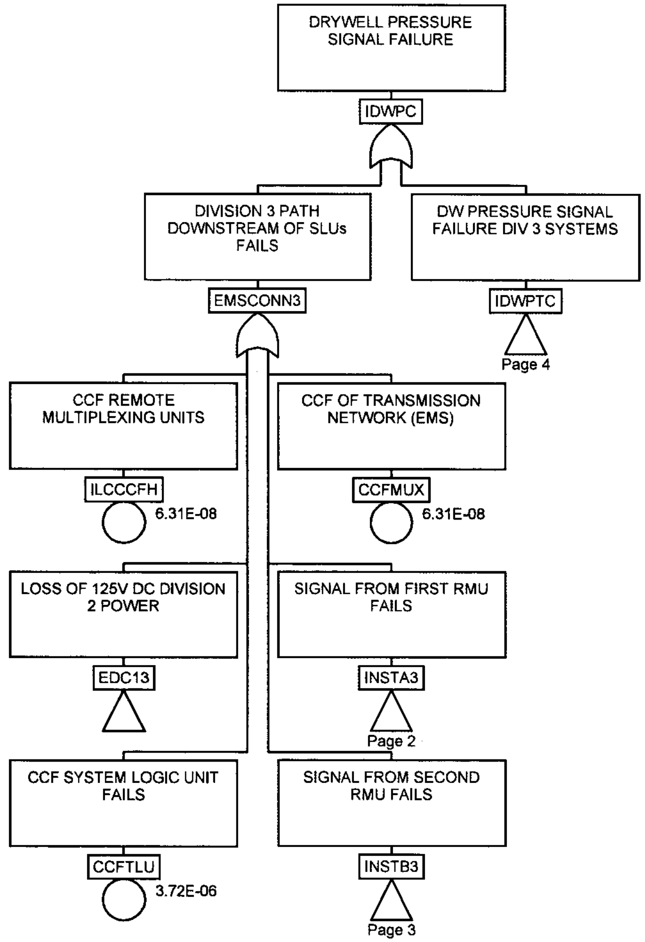

Figure 19D.6-15C High Drywell Pressure Instrumentation System Fault Tree Div. 3

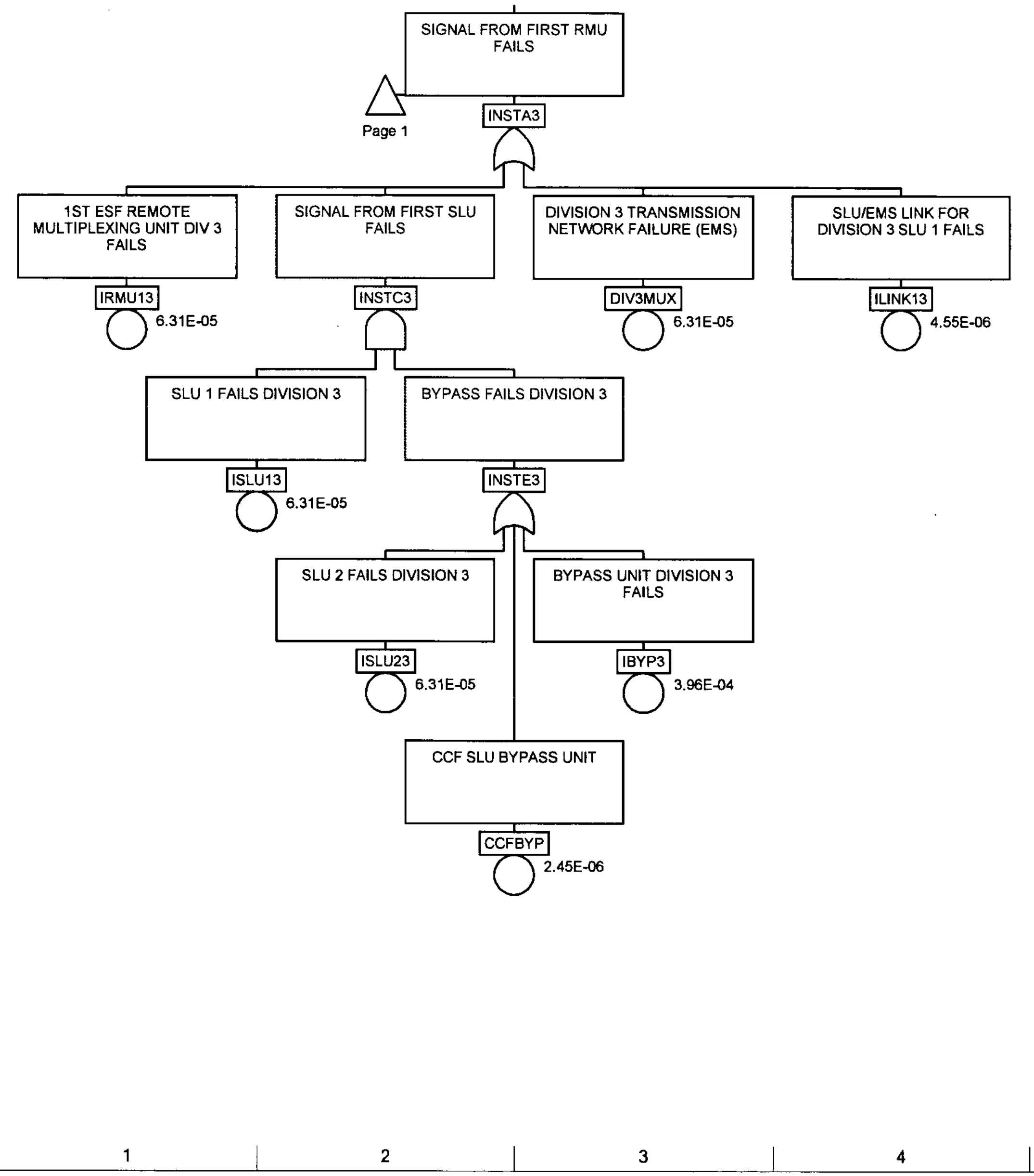

Figure 19D.6-15C High Drywell Pressure Instrumentation System Fault Tree Div. 3

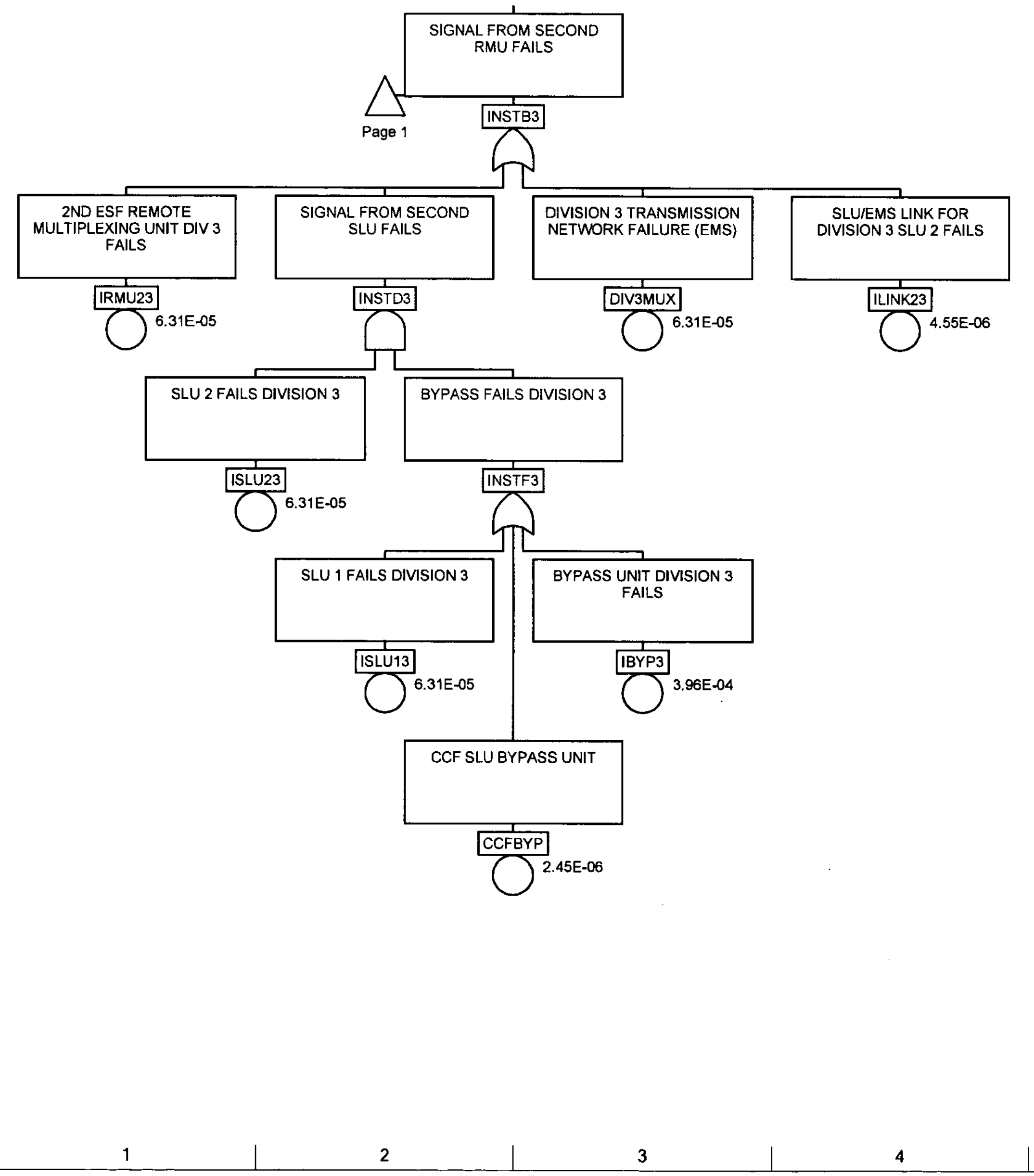

Figure 19D.6-15C High Drywell Pressure Instrumentation System Fault Tree Div. 3

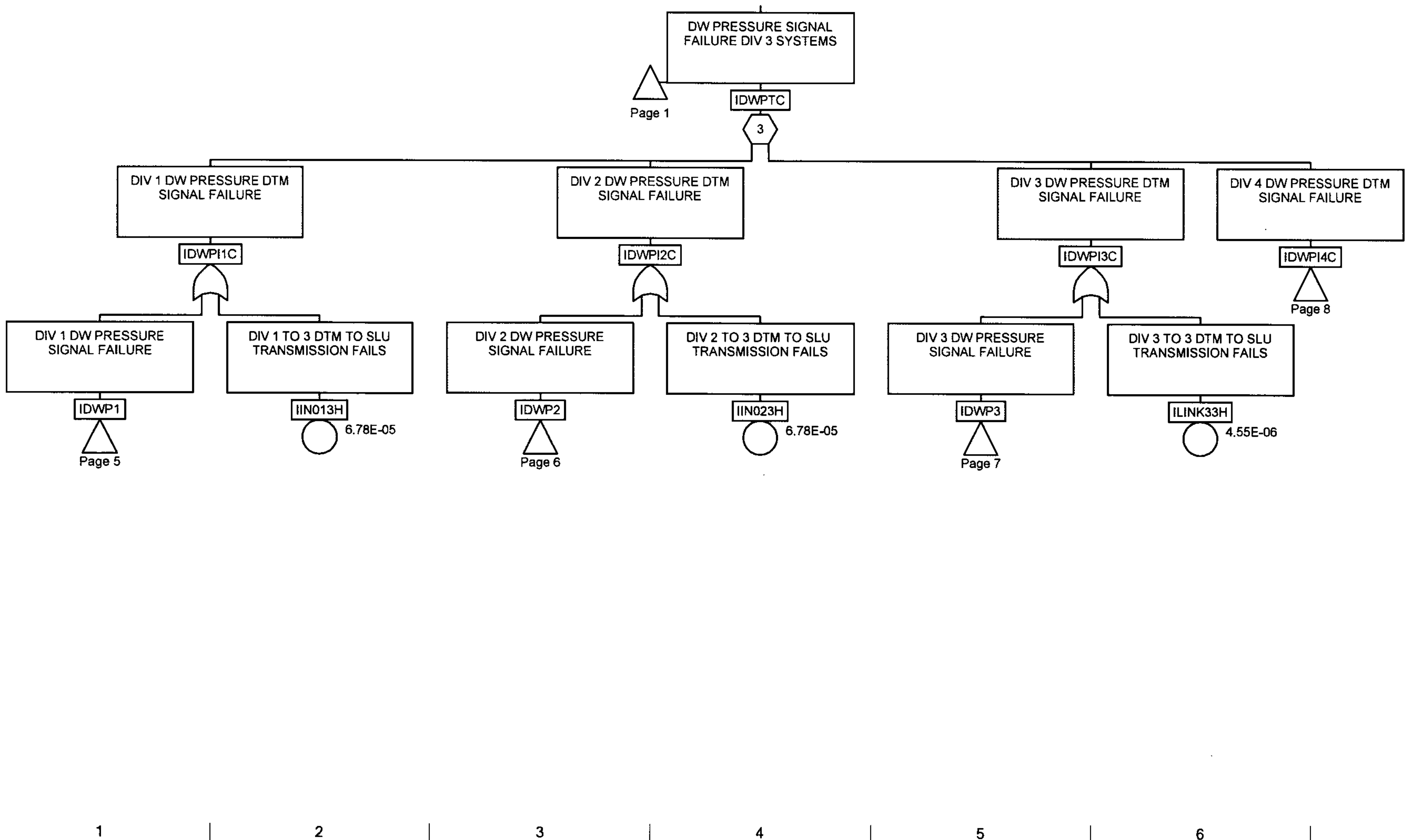

Figure 19D.6-15C High Drywell Pressure Instrumentation System Fault Tree Div. 3

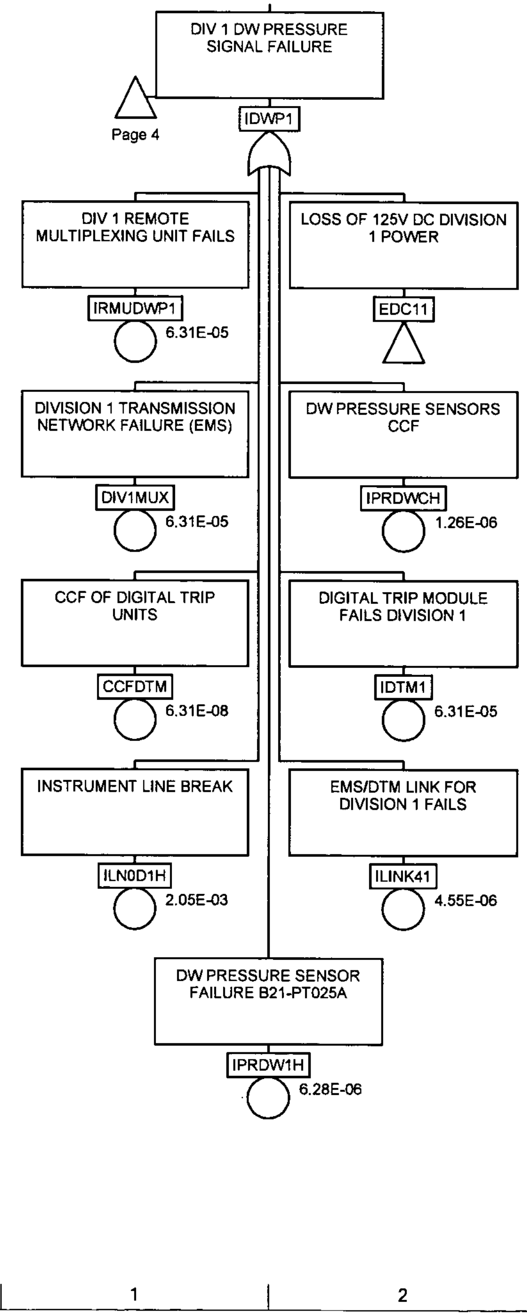

Figure 19D.6-15C High Drywell Pressure Instrumentation System Fault Tree Div. 3

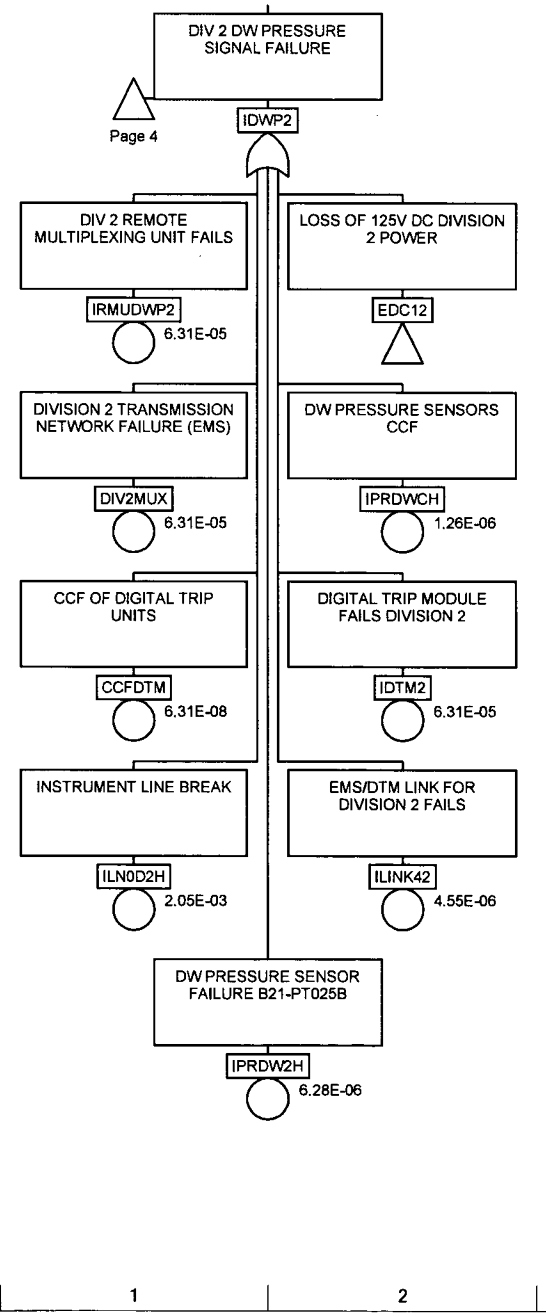

Figure 19D.6-15C High Drywell Pressure Instrumentation System Fault Tree Div. 3

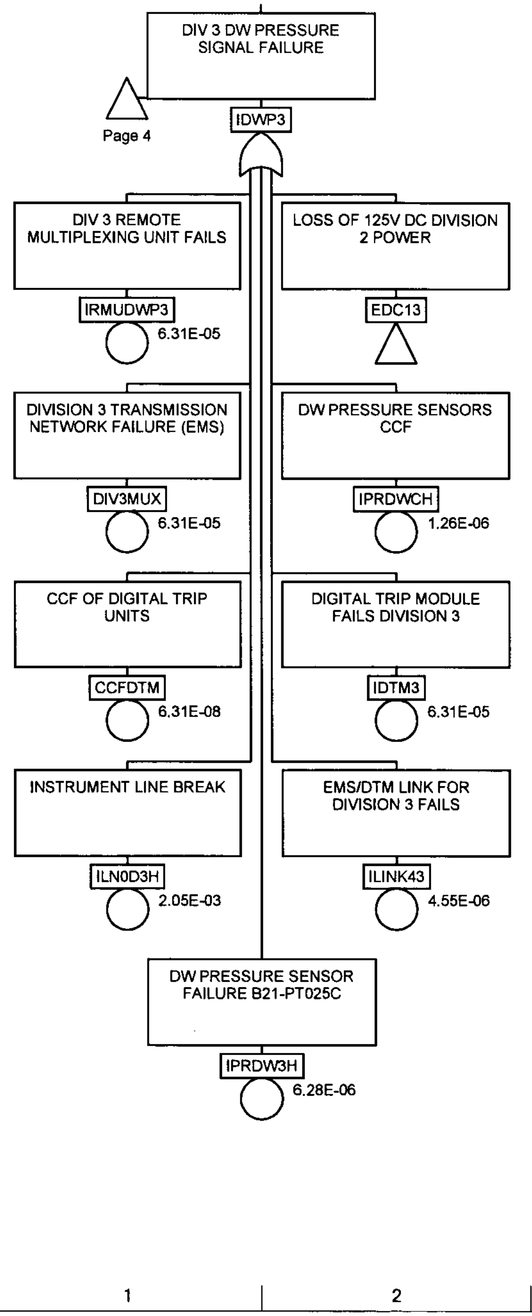

Figure 19D.6-15C High Drywell Pressure Instrumentation System Fault Tree Div. 3

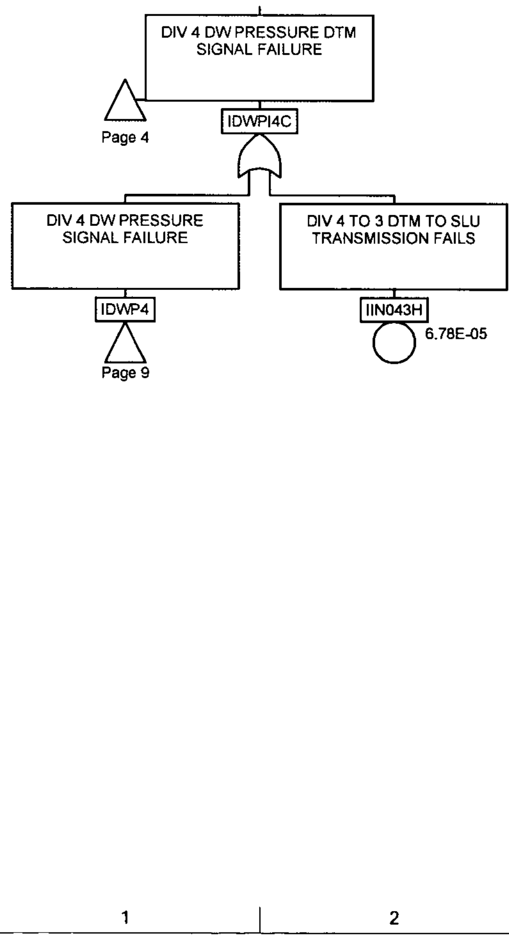

Figure 19D.6-15C High Drywell Pressure Instrumentation System Fault Tree Div. 3

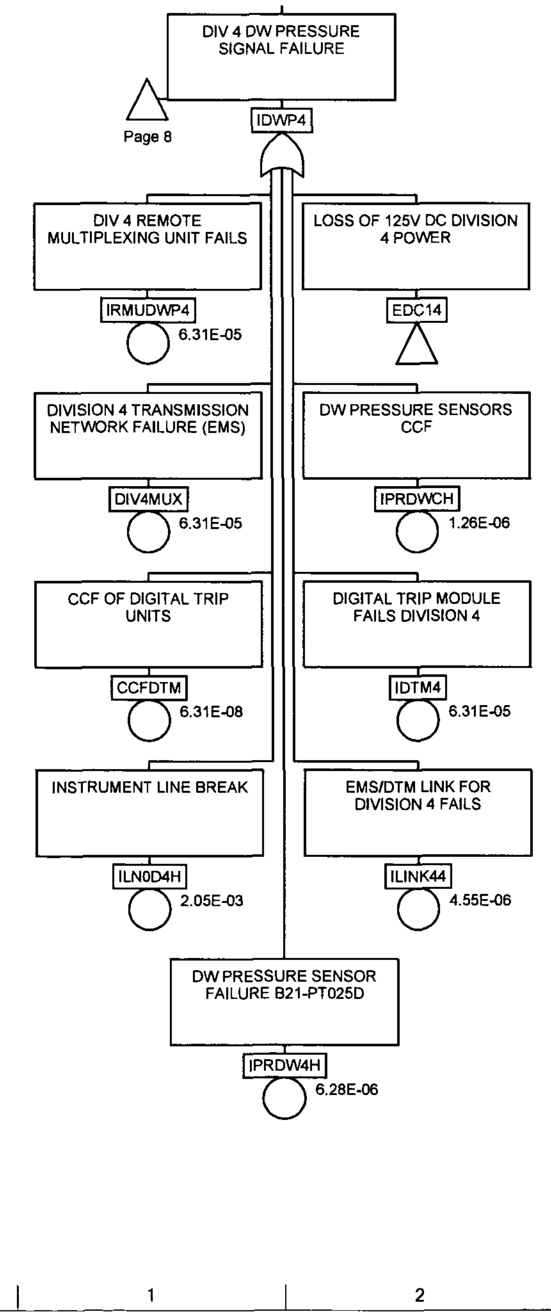

Figure 19D.6-15C High Drywell Pressure Instrumentation System Fault Tree Div. 3

Name	Page	Zone	Name	Page	Zone
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CCFDTM	5	1	ILINK42	6	2
CCFDTM	6	1	ILINK43	7	2
CCFDTM	7	1	ILINK44	9	2
CCFDTM	9	1	ILN0D1H	5	1
CCFMUX	1	2	ILN0D2H	6	1
CCFTLU	1	1	ILN0D3H	7	1
DIV1MUX	5	1	ILN0D4H	9	1
DIV2MUX	6	1	INSTA3	1	2
DIV3MUX	2	3	INSTA3	2	3
DIV3MUX	3	3	INSTB3	1	2
DIV3MUX	7	1	INSTB3	3	3
DIV4MUX	9	1	INSTC3	2	2
EDC11	5	2	INSTD3	3	2
EDC12	6	2	INSTE3	2	3
EDC13	1	1	INSTF3	3	3
EDC13	7	2	IPRDW1H	5	2
EDC14	9	2	IPRDW2H	6	2
EMSCONN3	1	2	IPRDW3H	7	2
IBYP3	2	3	IPRDW4H	9	2
IBYP3	3	3	IPRDWCH	5	2
IDTM1	5	2	IPRDWCH	6	2
IDTM2	6	2	IPRDWCH	7	2
IDTM3	7	2	IPRDWCH	9	2
IDTM4	9	2	IRMU13	2	1
IDWP1	4	1	IRMU23	3	1
IDWP1	5	2	IRMUDWP1	5	1
IDWP2	4	3	IRMUDWP2	6	1
IDWP2	6	2	IRMUDWP3	7	1
IDWP3	4	5	IRMUDWP4	9	1
IDWP3	7	2	ISLU13	2	2
IDWP4	8	1	ISLU13	3	2
IDWP4	9	2	ISLU23	2	2
IDWPC	1	2	ISLU23	3	2
IDWPI1C	4	2			
IDWPI2C	4	4			
IDWPI3C	4	6			
IDWPI4C	4	7			
IDWPI4C	8	2			
IDWPTC	1	3			
IDWPTC	4	4			
IIN013H	4	2			
IIN023H	4	4			
IIN043H	8	2			
ILCCCFH	1	1			
ILINK13	2	4			
ILINK23	3	4			

Figure 19D.6-15C High Drywell Pressure Instrumentation System Fault Tree Div. 3

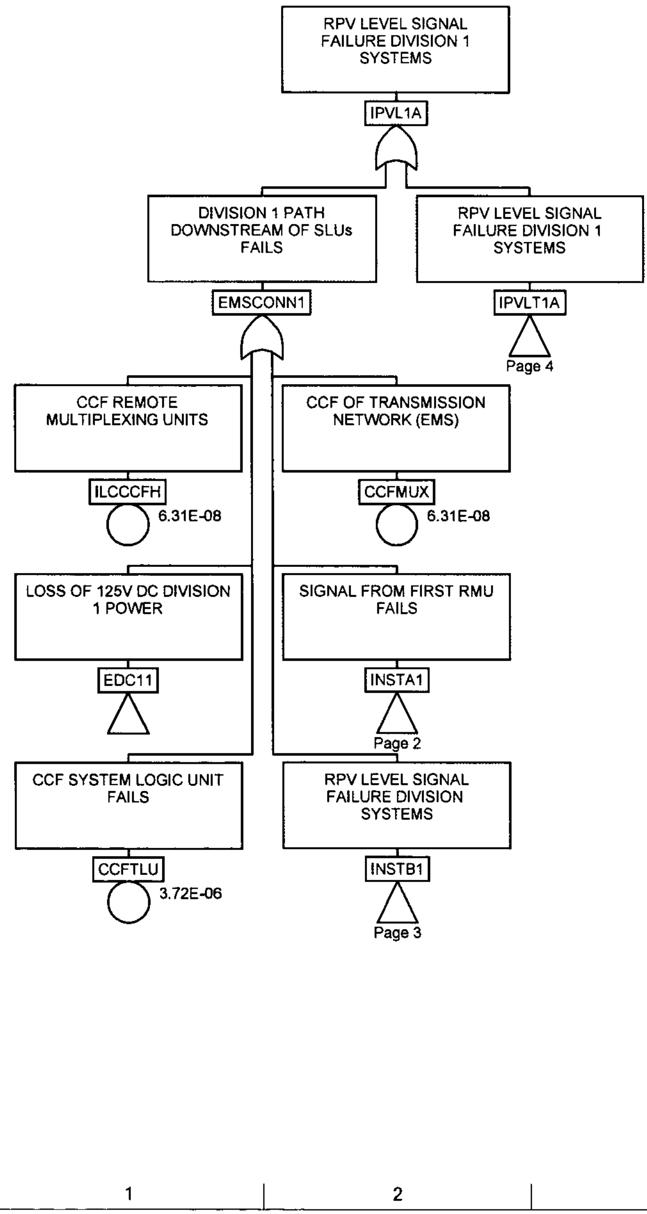


Figure 19D.6-15D Low Reactor Pressure Vessel Group 1 Fault Tree Division 1

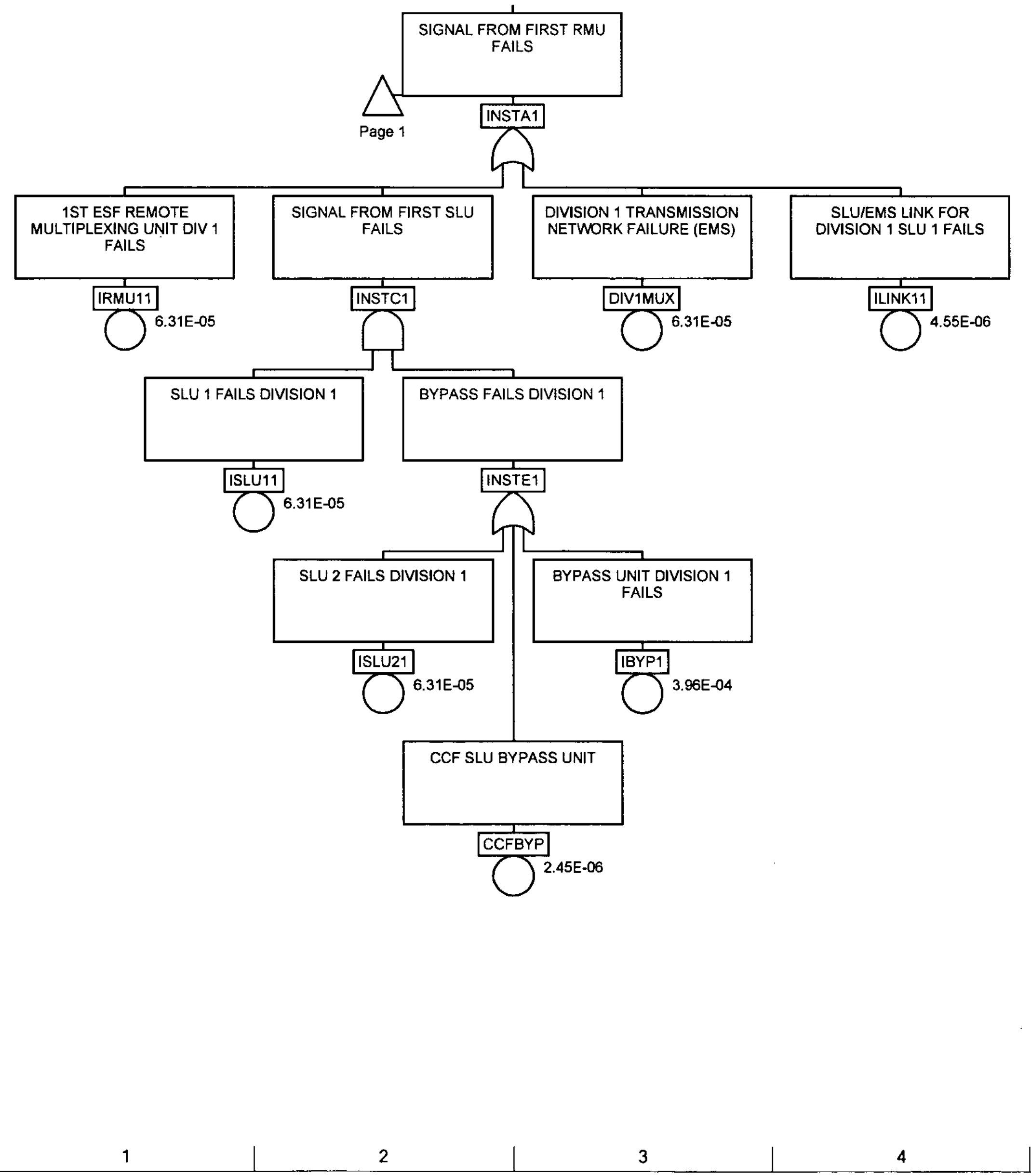


Figure 19D.6-15D Low Reactor Pressure Vessel Group 1 Fault Tree Division 1

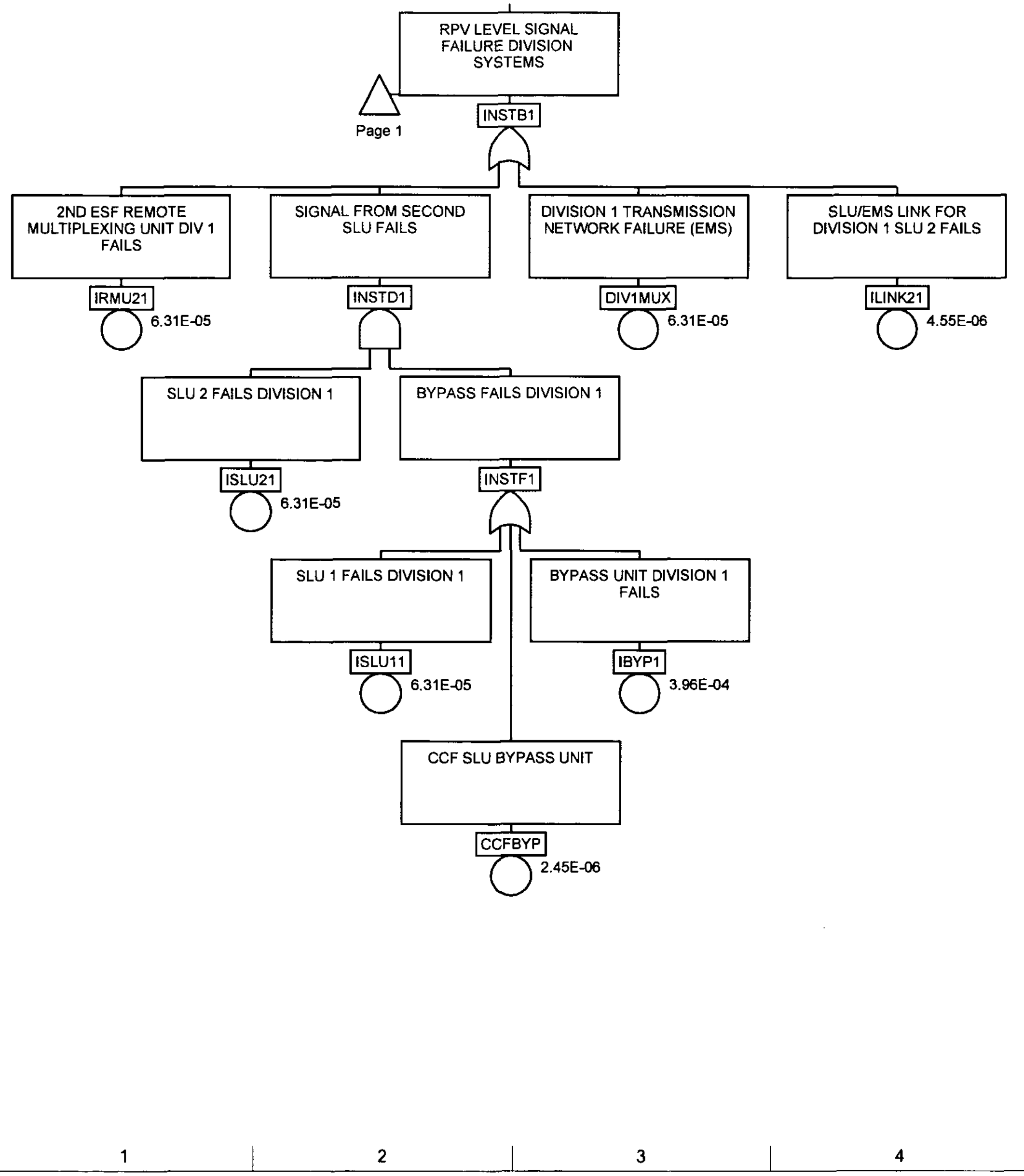


Figure 19D.6-15D Low Reactor Pressure Vessel Group 1 Fault Tree Division 1

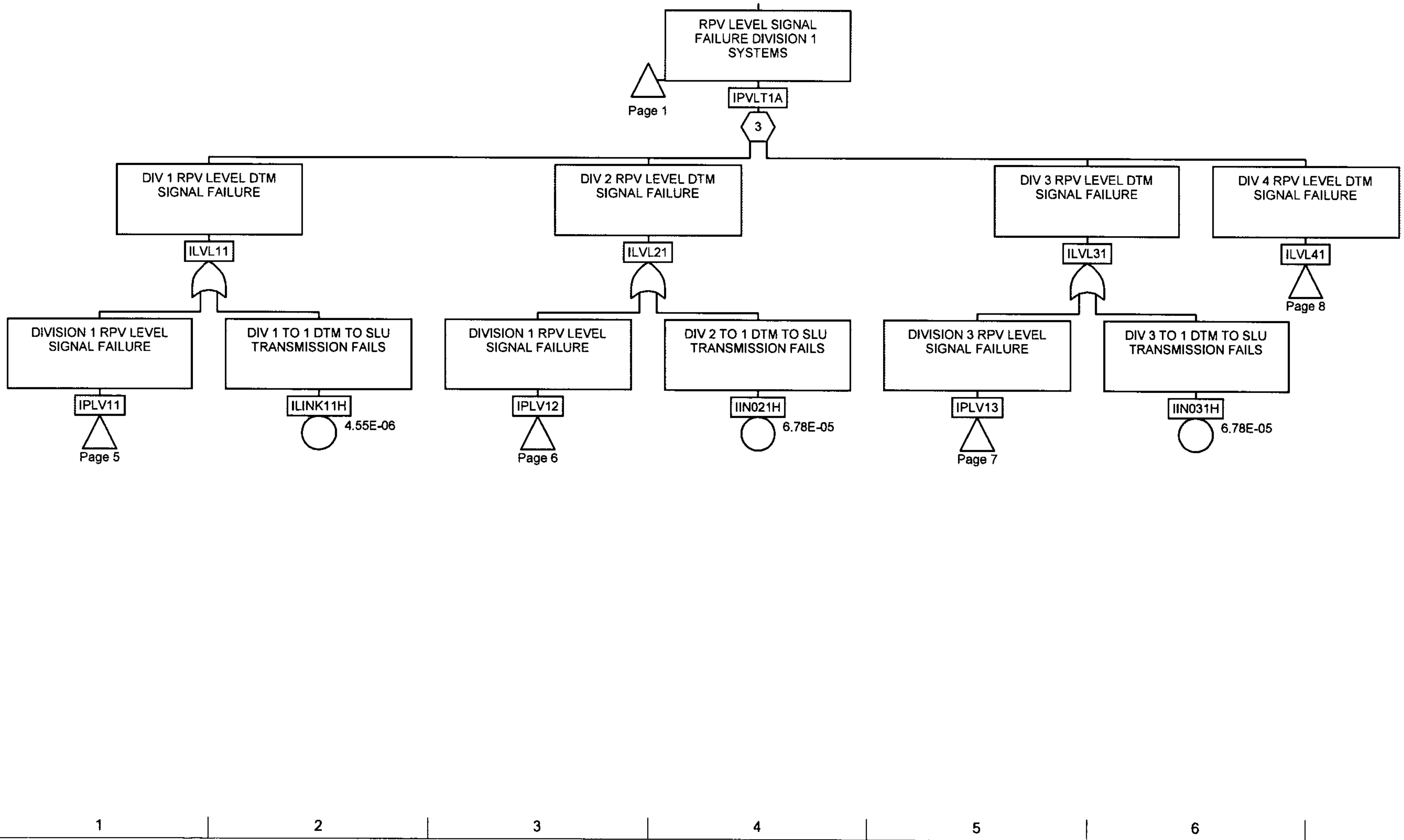


Figure 19D.6-15D Low Reactor Pressure Vessel Group 1 Fault Tree Division 1

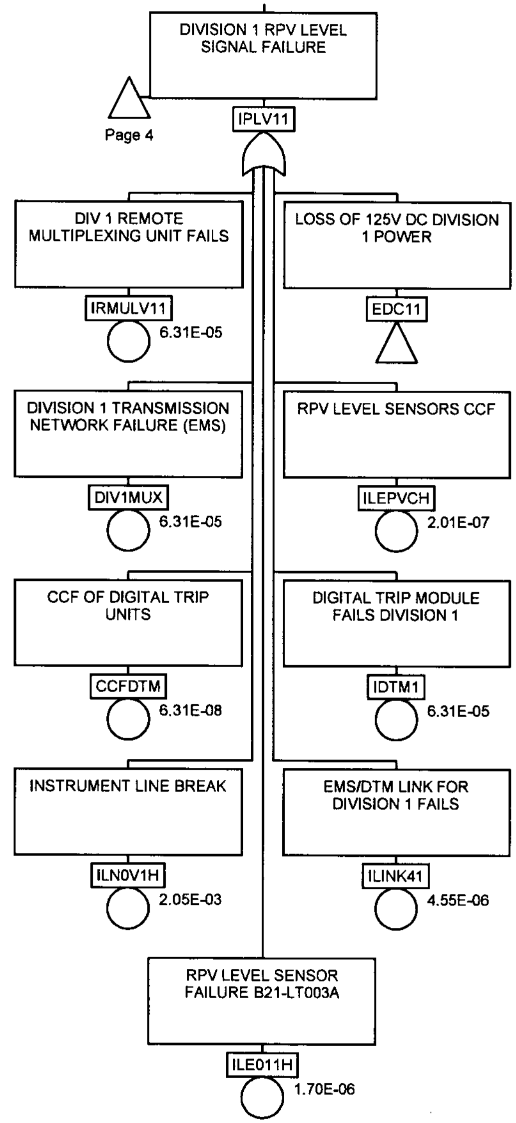


Figure 19D.6-15D Low Reactor Pressure Vessel Group 1 Fault Tree Division 1

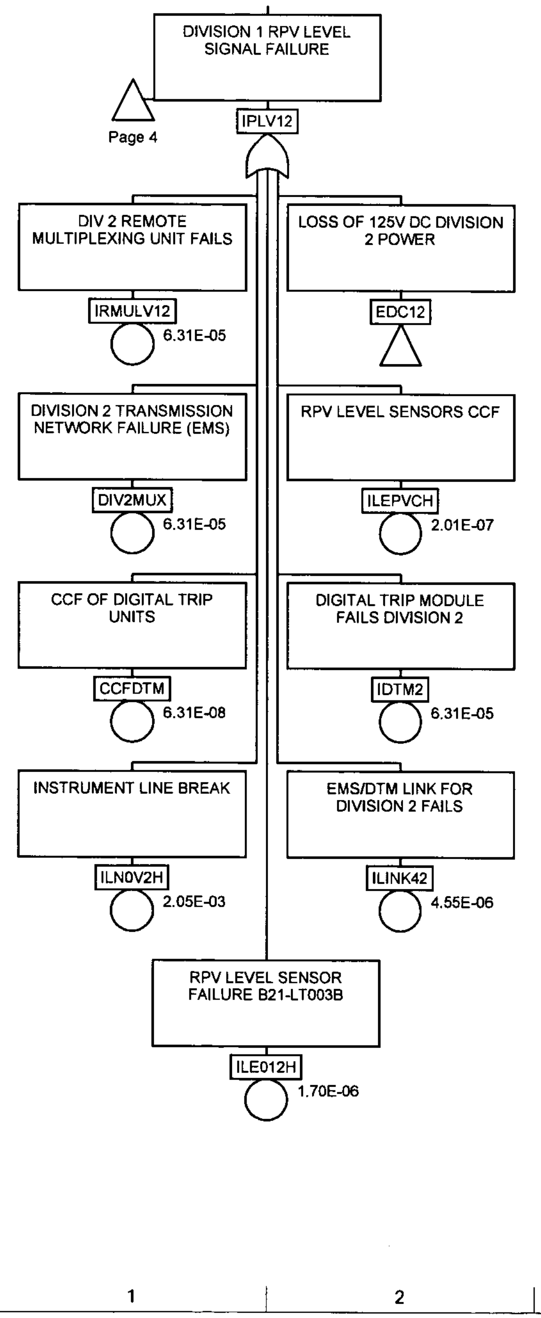


Figure 19D.6-15D Low Reactor Pressure Vessel Group 1 Fault Tree Division 1

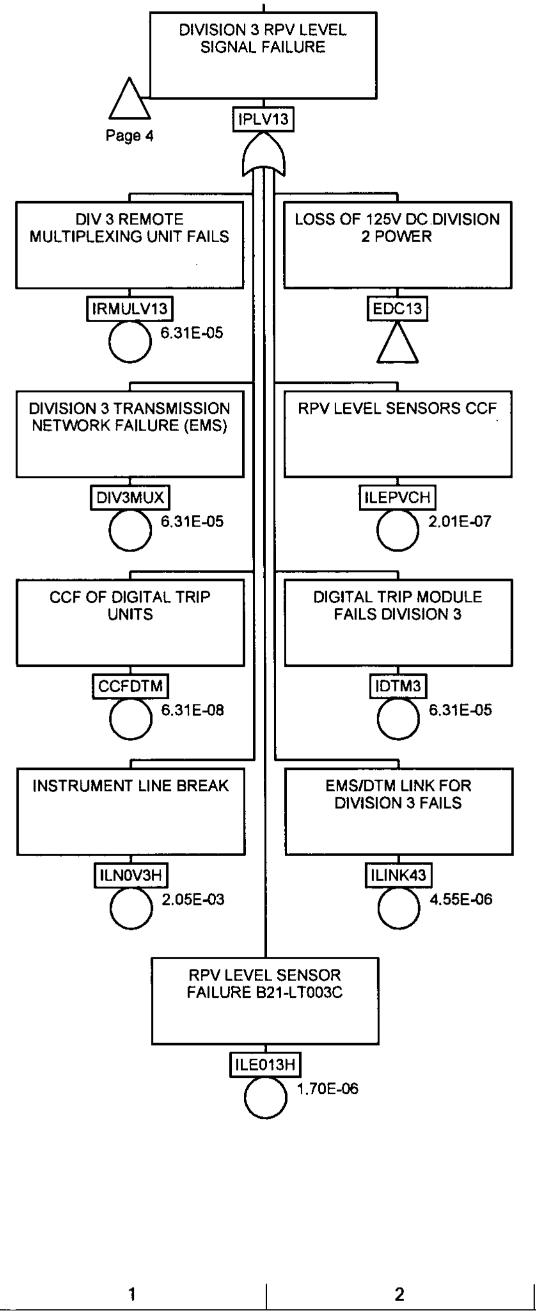
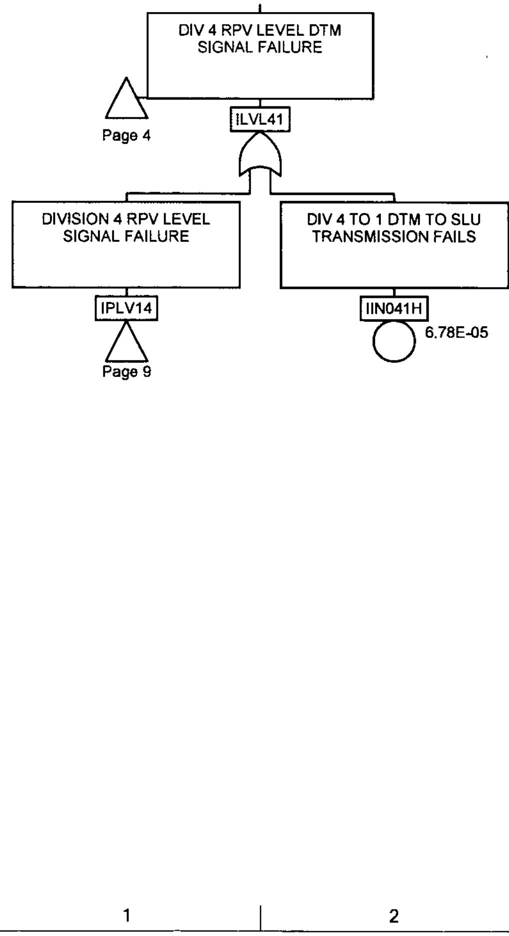


Figure 19D.6-15D Low Reactor Pressure Vessel Group 1 Fault Tree Division 1



DIV 4 RPV LEVEL DTM SIGNAL FAILURE
Page 4
ILVL41
DIVISION 4 RPV LEVEL SIGNAL FAILURE
DIV 4 TO 1 DTM TO SLU TRANSMISSION FAILS
IPLV14
IIN041H
6.78E-05
Page 9
1
2

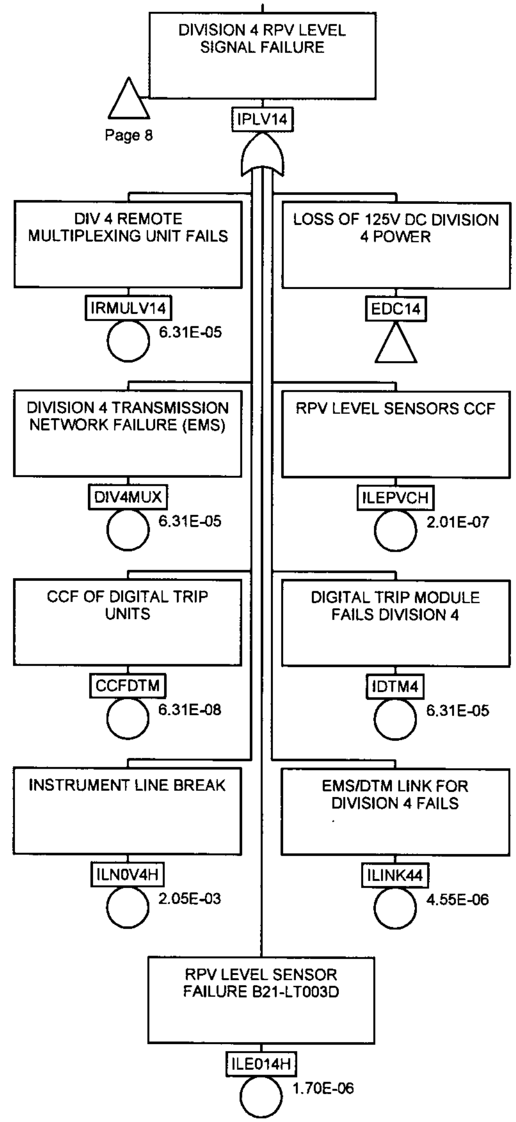


Figure 19D.6-15D Low Reactor Pressure Vessel Group 1 Fault Tree Division 1

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CCFDTM	5	1	ILVL21	4	4
CCFDTM	6	1	ILVL31	4	6
CCFDTM	7	1	ILVL41	4	7
CCFDTM	9	1	ILVL41	8	2
CCFMUX	1	2	INSTA1	1	2
CCFTLU	1	1	INSTA1	2	3
DIV1MUX	2	3	INSTB1	1	2
DIV1MUX	3	3	INSTB1	3	3
DIV1MUX	5	1	INSTC1	2	2
DIV2MUX	6	1	INSTD1	3	2
DIV3MUX	7	1	INSTE1	2	3
DIV4MUX	9	1	INSTF1	3	3
EDC11	1	1	IPLV11	4	1
EDC11	5	2	IPLV11	5	2
EDC12	6	2	IPLV12	4	3
EDC13	7	2	IPLV12	6	2
EDC14	9	2	IPLV13	4	5
EMSCONN1	1	2	IPLV13	7	2
IBYP1	2	3	IPLV14	8	1
IBYP1	3	3	IPLV14	9	2
IDTM1	5	2	IPVL1A	1	2
IDTM2	6	2	IPVLT1A	1	3
IDTM3	7	2	IPVLT1A	4	4
IDTM4	9	2	IRMU11	2	1
IIN021H	4	4	IRMU21	3	1
IIN031H	4	6	IRMULV11	5	1
IIN041H	8	2	IRMULV12	6	1
ILCCCFH	1	1	IRMULV13	7	1
ILE011H	5	2	IRMULV14	9	1
ILE012H	6	2	ISLU11	2	2
ILE013H	7	2	ISLU11	3	2
ILE014H	9	2	ISLU21	2	2
ILEPVCH	5	2	ISLU21	3	2
ILEPVCH	6	2			
ILEPVCH	7	2			
ILEPVCH	9	2			
ILINK11	2	4			
ILINK11H	4	2			
ILINK21	3	4			
ILINK41	5	2			
ILINK42	6	2			
ILINK43	7	2			
ILINK44	9	2			
ILN0V1H	5	1			
ILN0V2H	6	1			
ILN0V3H	7	1			

Figure 19D.6-15D Low Reactor Pressure Vessel Group 1 Fault Tree Division 1

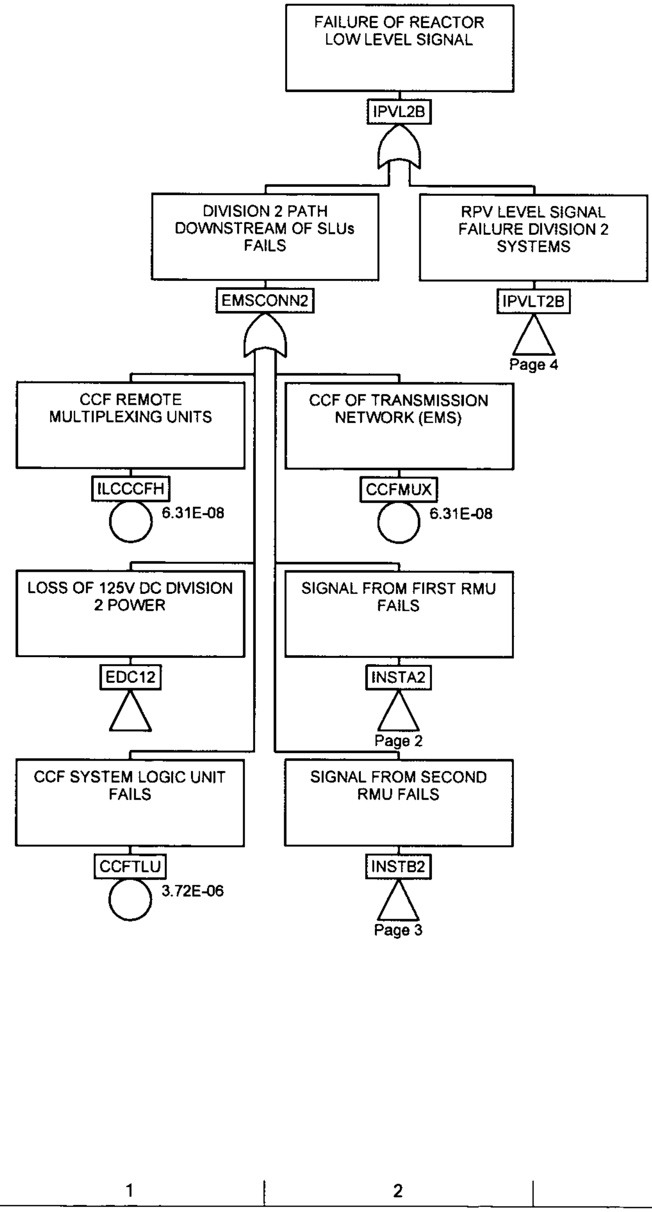

Figure 19D.6-15E Low Reactor Pressure Vessel Group 2 Fault Tree Division 2

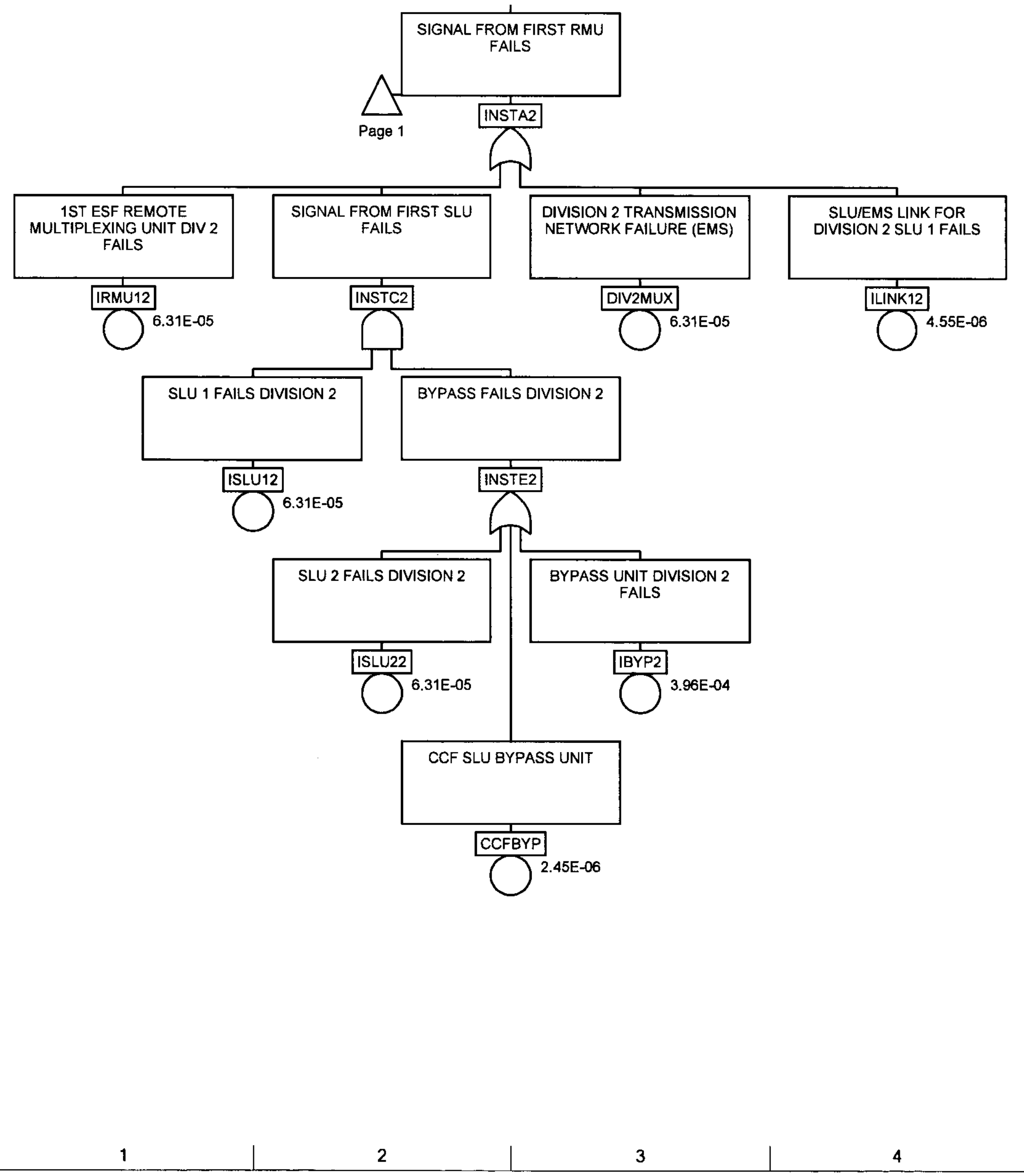

Figure 19D.6-15E Low Reactor Pressure Vessel Group 2 Fault Tree Division 2

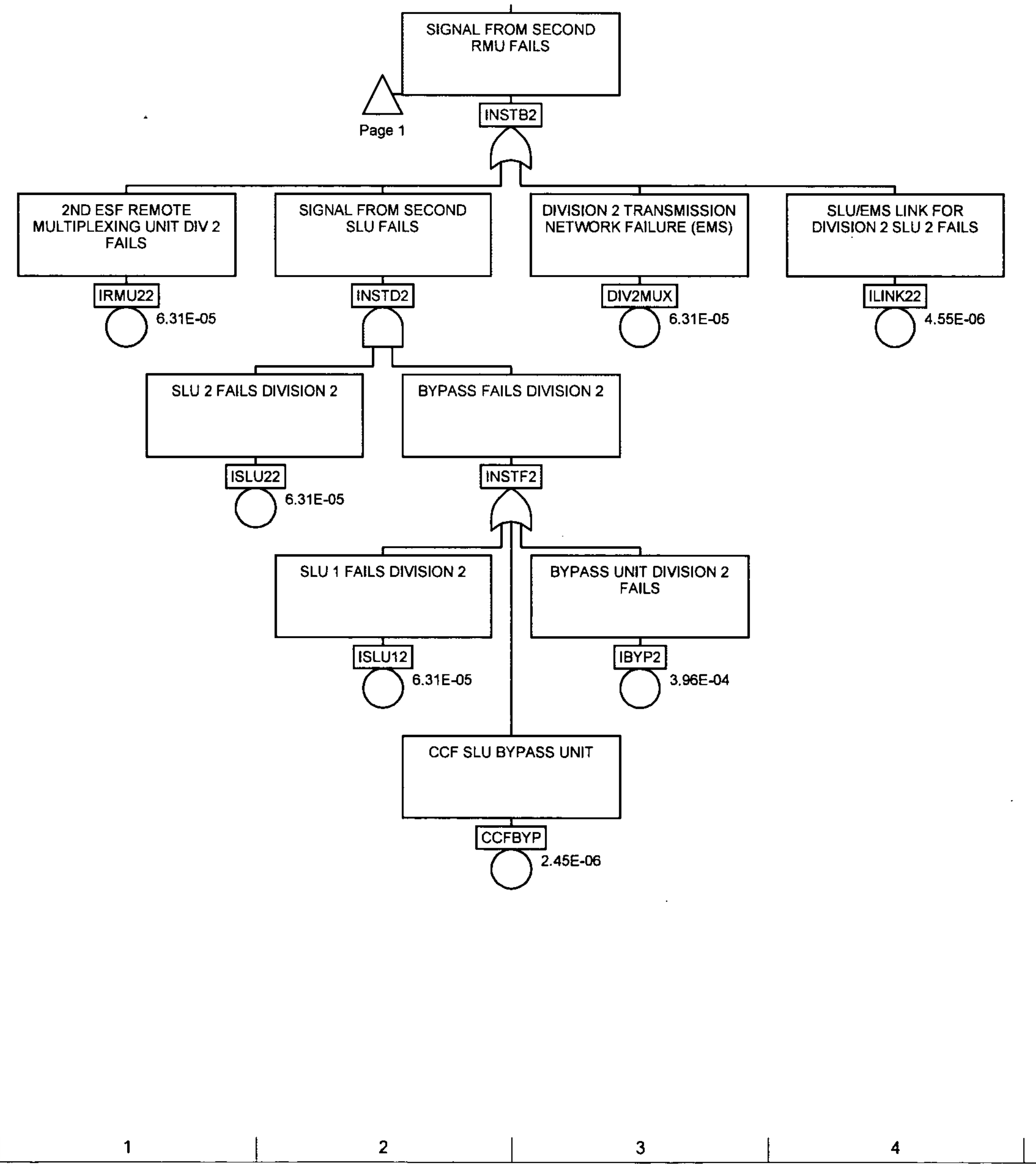

Figure 19D.6-15E Low Reactor Pressure Vessel Group 2 Fault Tree Division 2

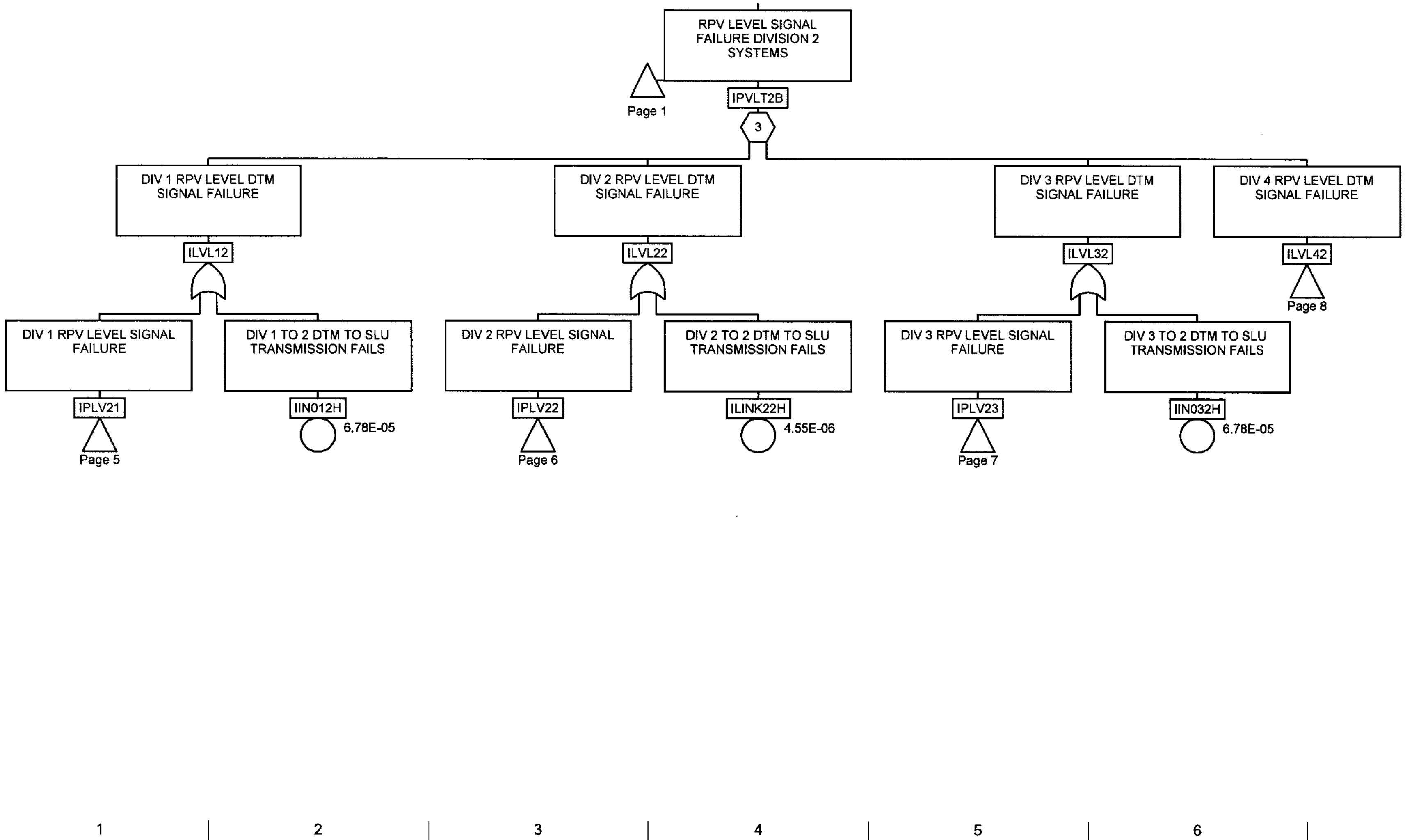

Figure 19D.6-15E Low Reactor Pressure Vessel Group 2 Fault Tree Division 2

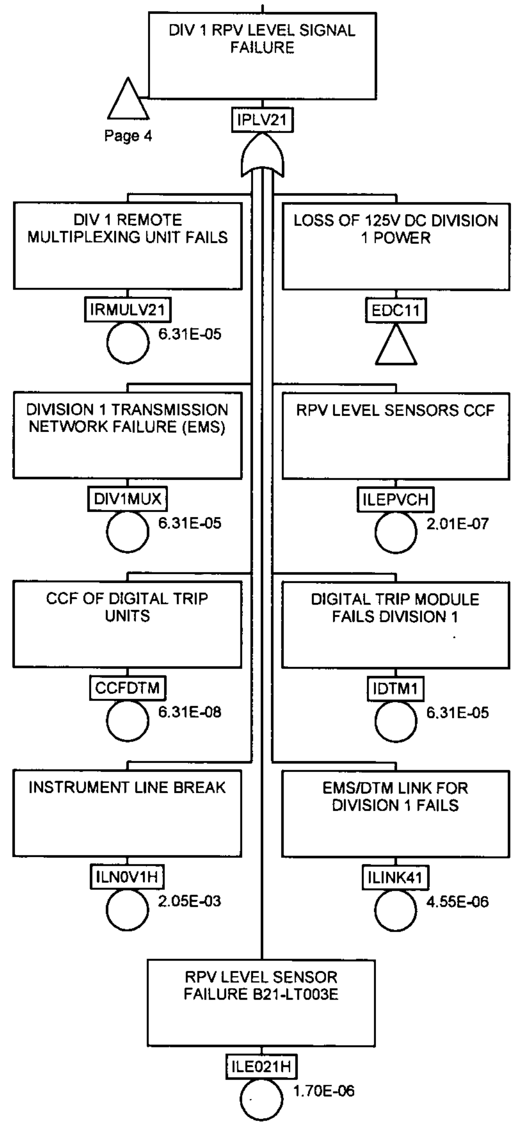

Figure 19D.6-15E Low Reactor Pressure Vessel Group 2 Fault Tree Division 2

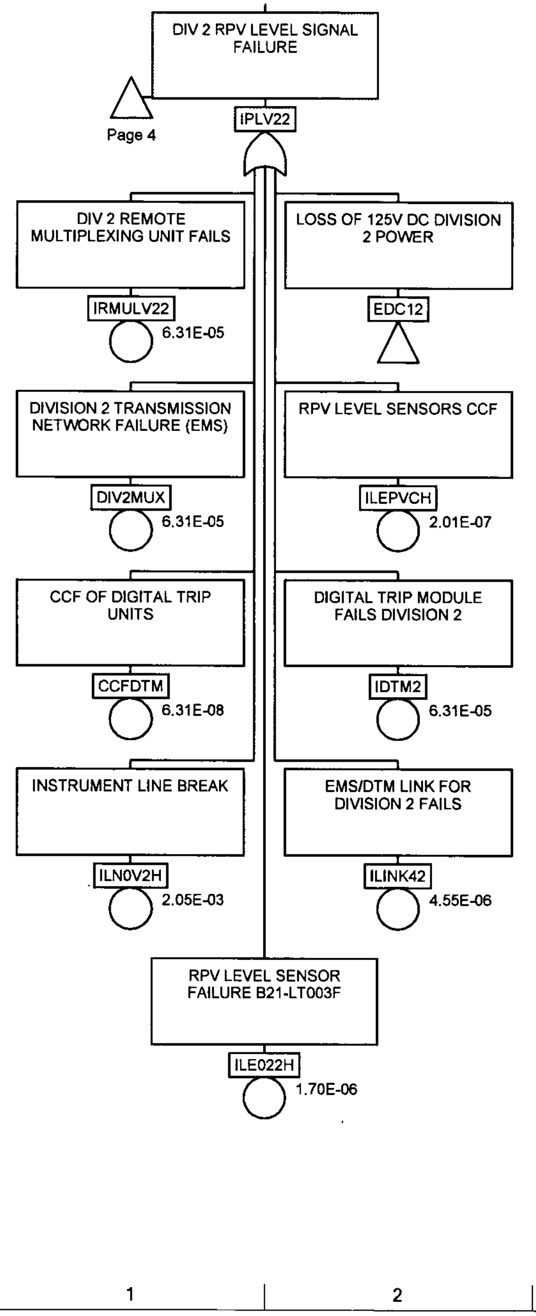

Figure 19D.6-15E Low Reactor Pressure Vessel Group 2 Fault Tree Division 2

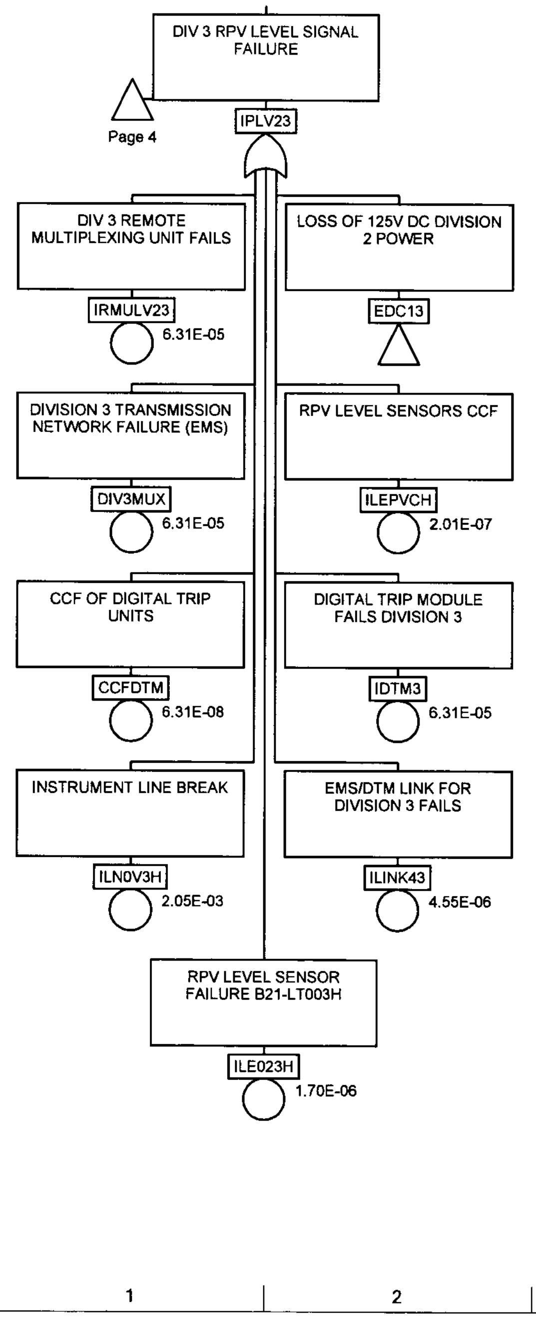

Figure 19D.6-15E Low Reactor Pressure Vessel Group 2 Fault Tree Division 2

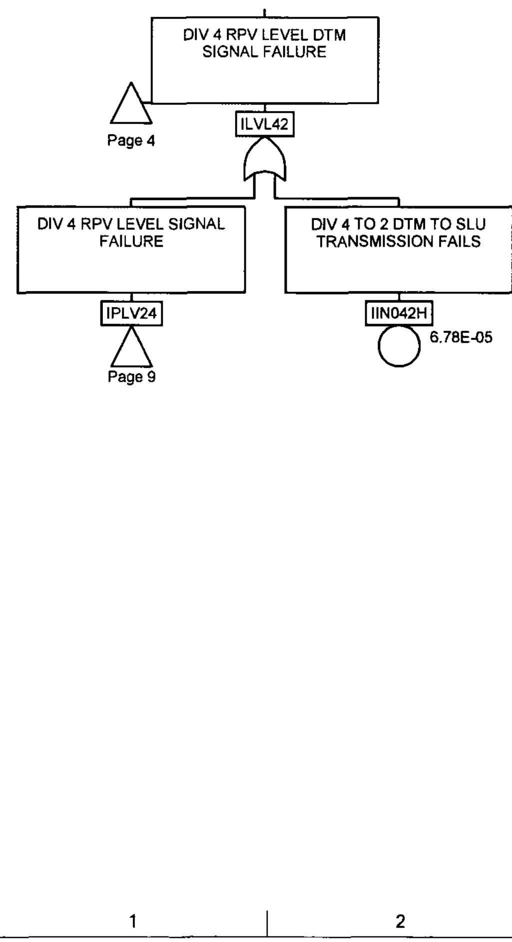

Figure 19D.6-15E Low Reactor Pressure Vessel Group 2 Fault Tree Division 2

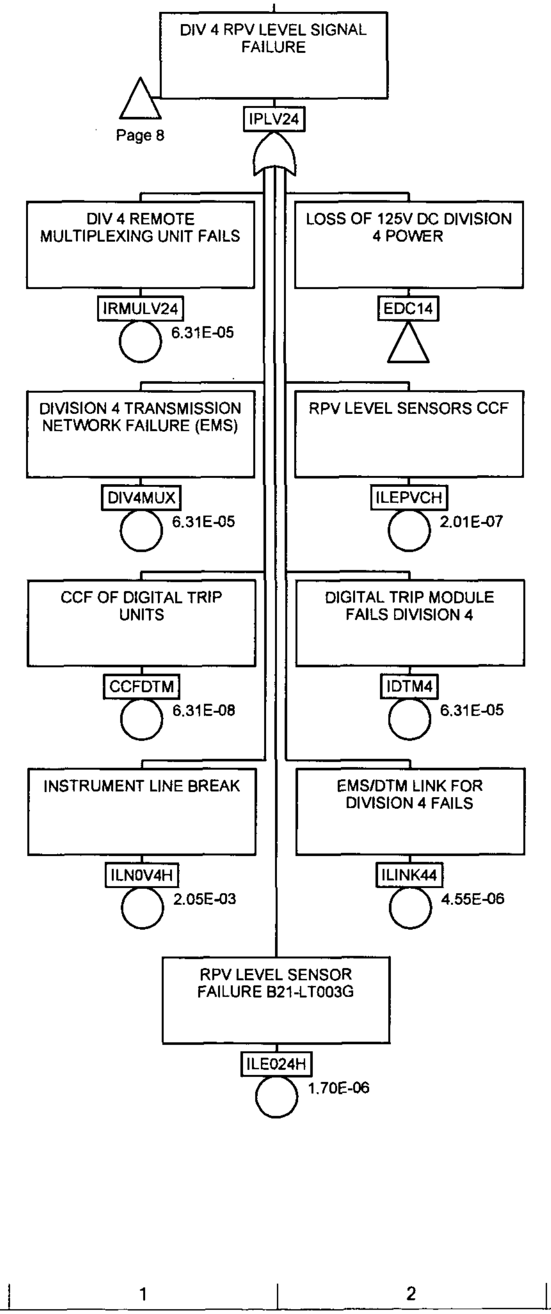

Figure 19D.6-15E Low Reactor Pressure Vessel Group 2 Fault Tree Division 2

Name	Page	Zone	Name	Page	Zone
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CCFDTM	5	1	ILVL22	4	4
CCFDTM	6	1	ILVL32	4	6
CCFDTM	7	1	ILVL42	4	7
CCFDTM	9	1	ILVL42	8	2
CCFMUX	1	2	INSTA2	1	2
CCFTLU	1	1	INSTA2	2	3
DIV1MUX	5	1	INSTB2	1	2
DIV2MUX	2	3	INSTB2	3	3
DIV2MUX	3	3	INSTC2	2	2
DIV2MUX	6	1	INSTD2	3	2
DIV3MUX	7	1	INSTE2	2	3
DIV4MUX	9	1	INSTF2	3	3
EDC11	5	2	IPLV21	4	1
EDC12	1	1	IPLV21	5	2
EDC12	6	2	IPLV22	4	3
EDC13	7	2	IPLV22	6	2
EDC14	9	2	IPLV23	4	5
EMSCONN2	1	2	IPLV23	7	2
IBYP2	2	3	IPLV24	8	1
IBYP2	3	3	IPLV24	9	2
IDTM1	5	2	IPVL2B	1	2
IDTM2	6	2	IPVLT2B	1	3
IDTM3	7	2	IPVLT2B	4	4
IDTM4	9	2	IRMU12	2	1
IIN012H	4	2	IRMU22	3	1
IIN032H	4	6	IRMULV21	5	1
IIN042H	8	2	IRMULV22	6	1
ILCCCFH	1	1	IRMULV23	7	1
ILE021H	5	2	IRMULV24	9	1
ILE022H	6	2	ISLU12	2	2
ILE023H	7	2	ISLU12	3	2
ILE024H	9	2	ISLU22	2	2
ILEPVCH	5	2	ISLU22	3	2
ILEPVCH	6	2			
ILEPVCH	7	2			
ILEPVCH	9	2			
ILINK12	2	4			
ILINK22	3	4			
ILINK22H	4	4			
ILINK41	5	2			
ILINK42	6	2			
ILINK43	7	2			
ILINK44	9	2			
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ILN0V2H	6	1			
ILN0V3H	7	1			

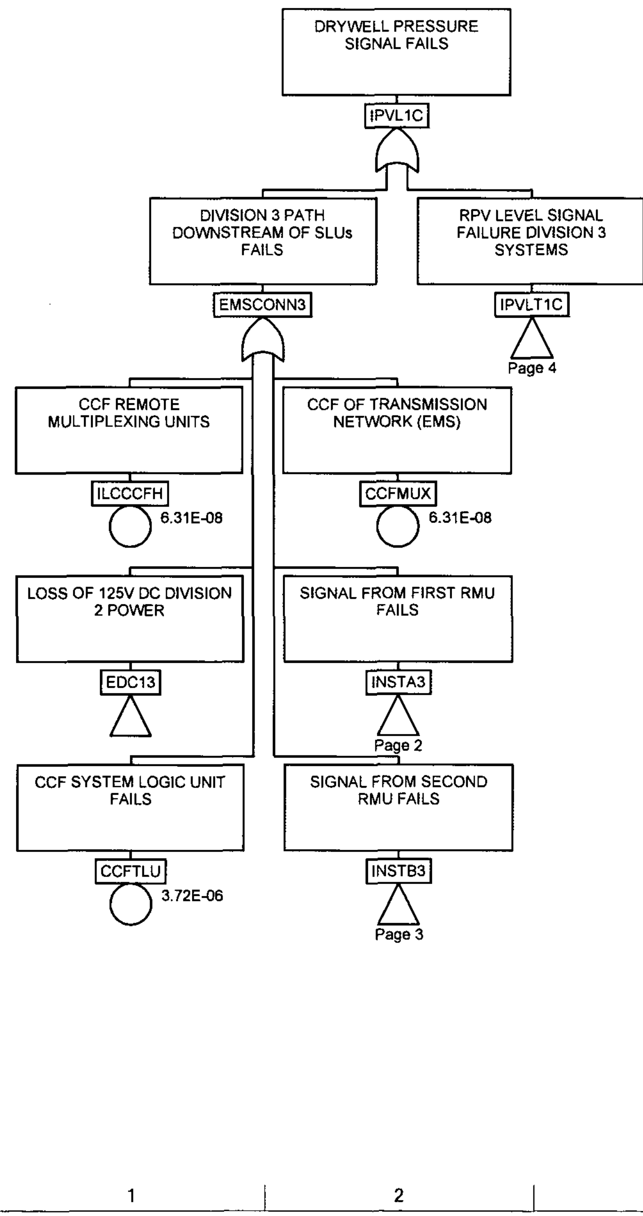


Figure 19D.6-15F Low Reactor Pressure Vessel Group 1 Fault Tree Division 3

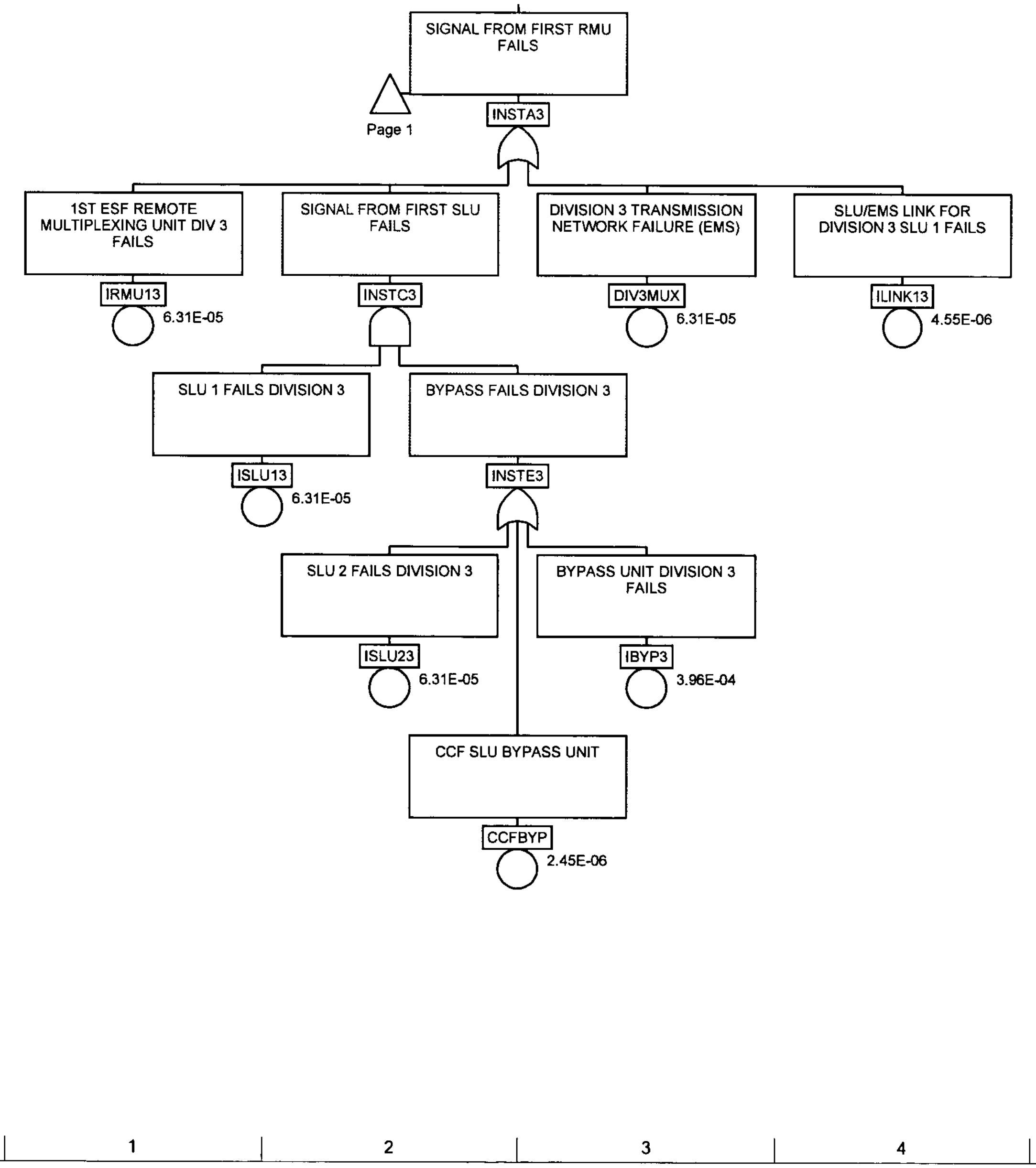


Figure 19D.6-15F Low Reactor Pressure Vessel Group 1 Fault Tree Division 3

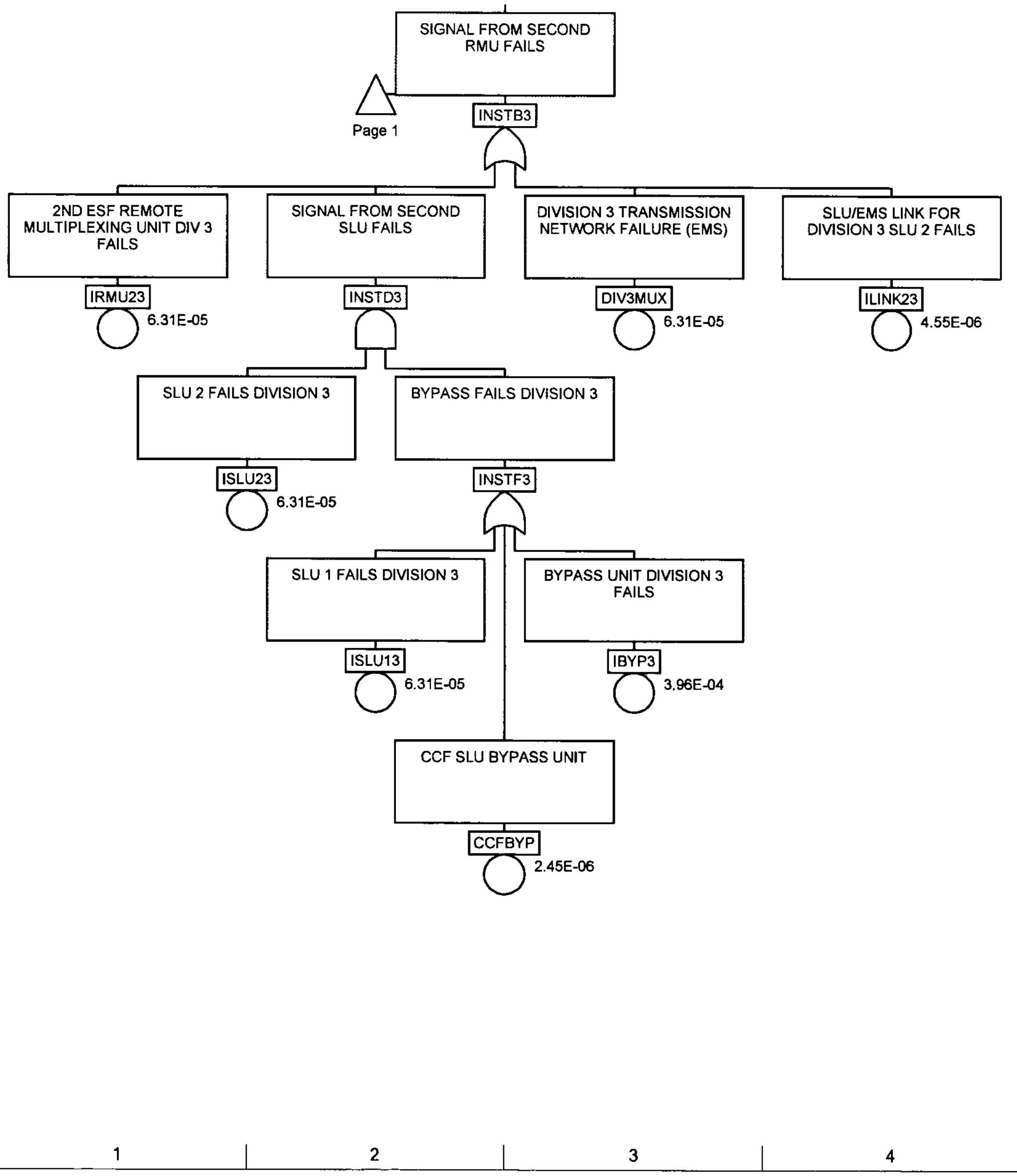


Figure 19D.6-15F Low Reactor Pressure Vessel Group 1 Fault Tree Division 3

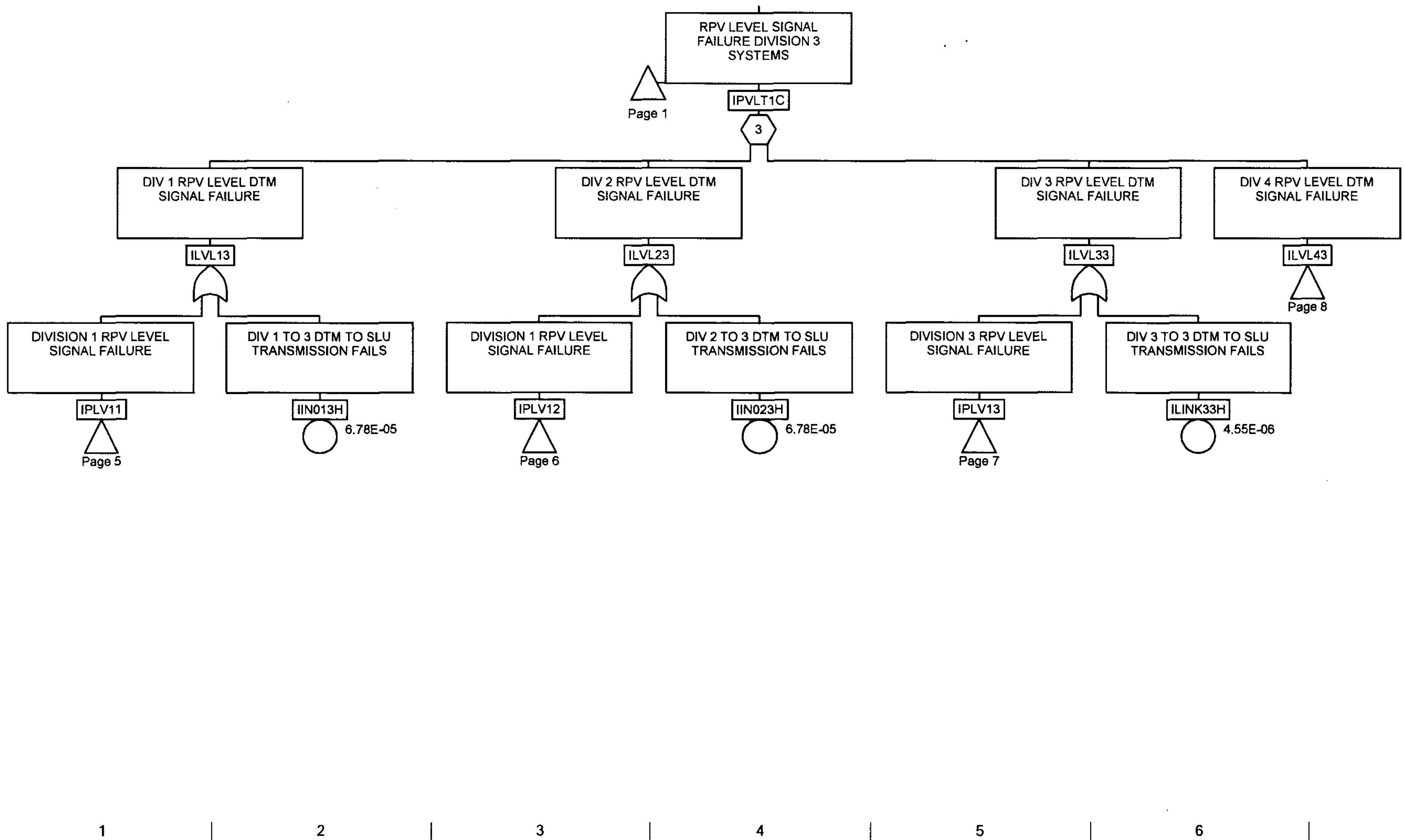


Figure 19D.6-15F Low Reactor Pressure Vessel Group 1 Fault Tree Division 3

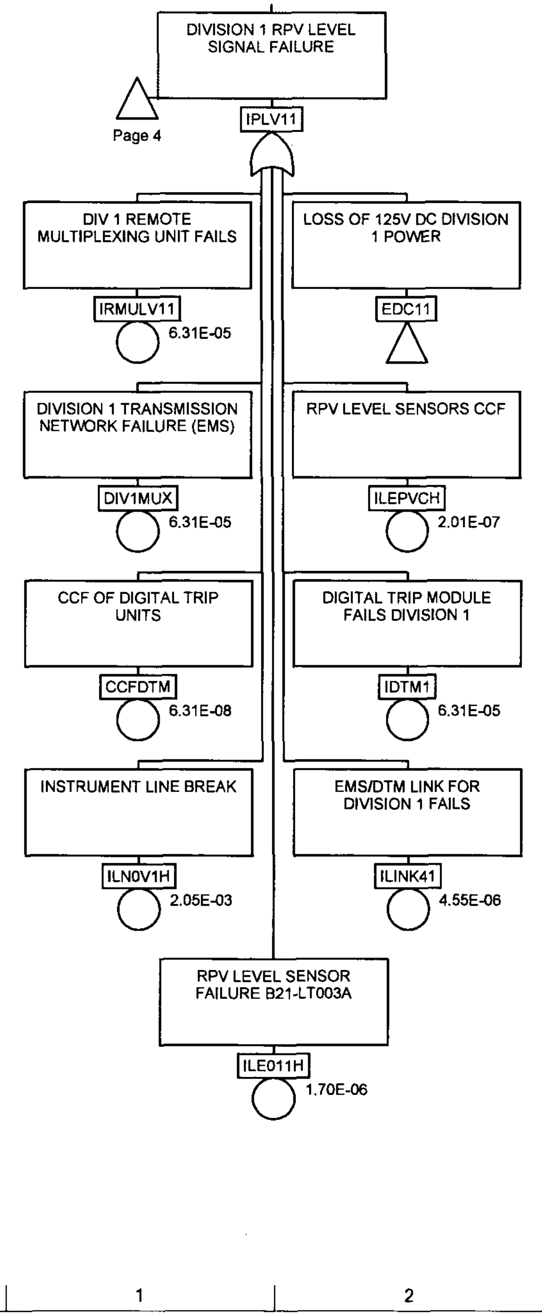
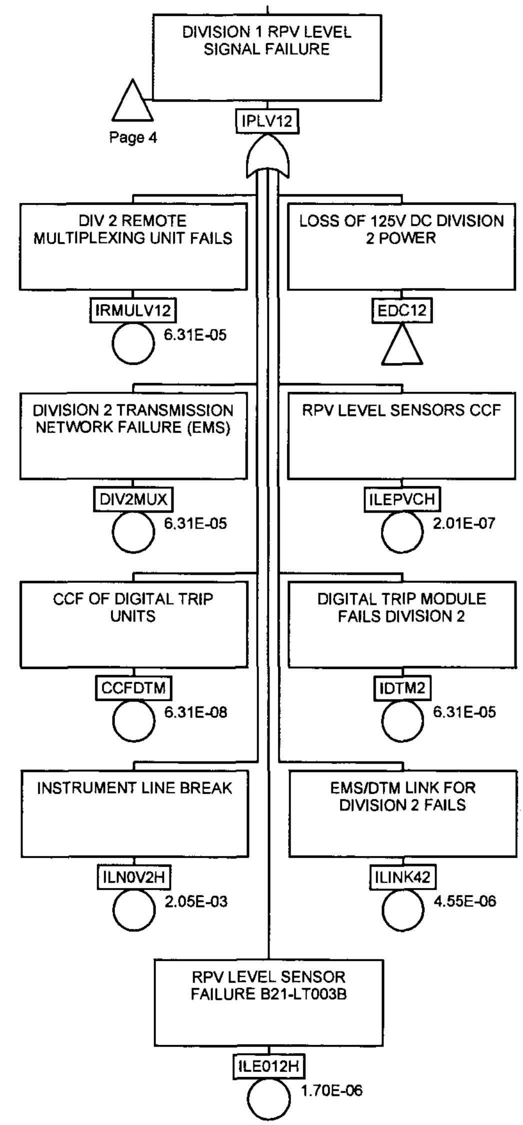


Figure 19D.6-15F Low Reactor Pressure Vessel Group 1 Fault Tree Division 3



1 | 2

Figure 19D.6-15F Low Reactor Pressure Vessel Group 1 Fault Tree Division 3

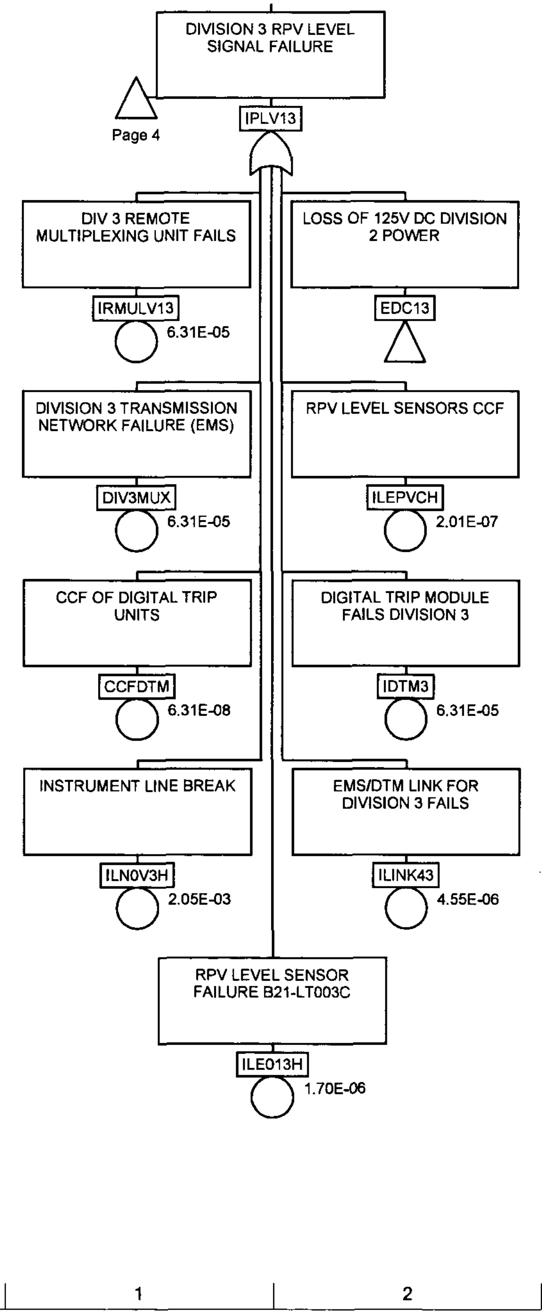


Figure 19D.6-15F Low Reactor Pressure Vessel Group 1 Fault Tree Division 3

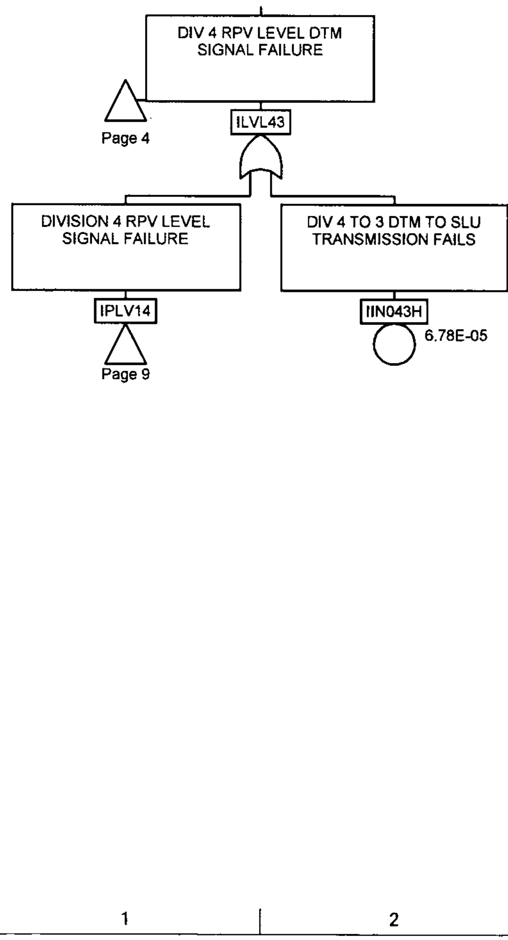


Figure 19D.6-15F Low Reactor Pressure Vessel Group 1 Fault Tree Division 3

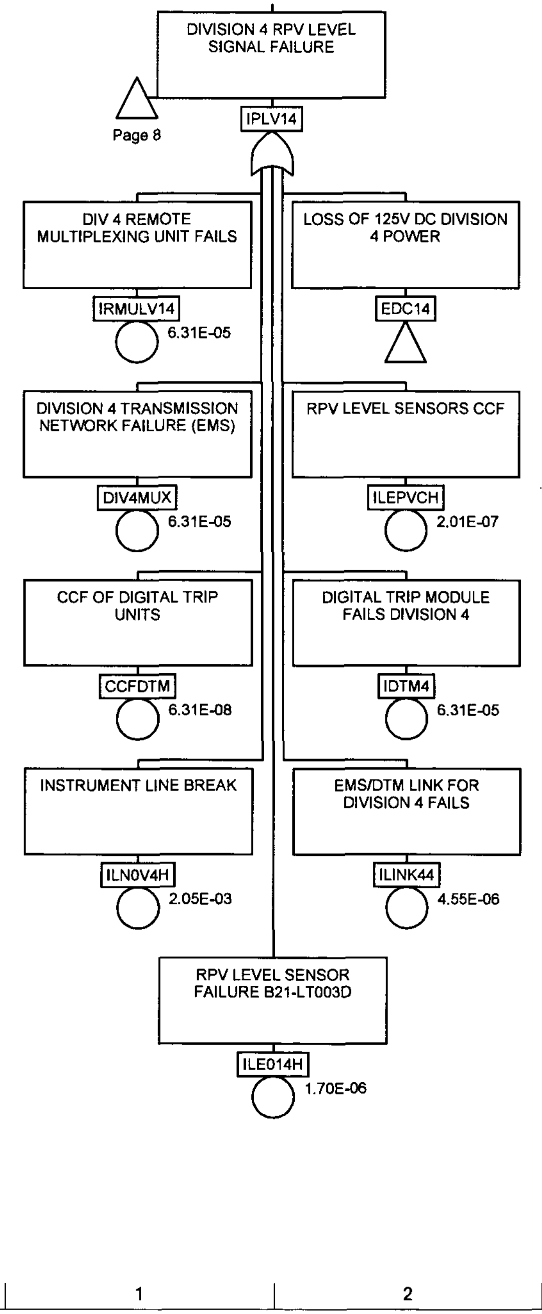


Figure 19D.6-15F Low Reactor Pressure Vessel Group 1 Fault Tree Division 3

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CCFBYP	3	3	ILVL13	4	2
CCFDTM	5	1	ILVL23	4	4
CCFDTM	6	1	ILVL33	4	6
CCFDTM	7	1	ILVL43	4	7
CCFDTM	9	1	ILVL43	8	2
CCFMUX	1	2	INSTA3	1	2
CCFTLU	1	1	INSTA3	2	3
DIV1MUX	5	1	INSTB3	1	2
DIV2MUX	6	1	INSTB3	3	3
DIV3MUX	2	3	INSTC3	2	2
DIV3MUX	3	3	INSTD3	3	2
DIV3MUX	7	1	INSTE3	2	3
DIV4MUX	9	1	INSTF3	3	3
EDC11	5	2	IPLV11	4	1
EDC12	6	2	IPLV11	5	2
EDC13	1	1	IPLV12	4	3
EDC13	7	2	IPLV12	6	2
EDC14	9	2	IPLV13	4	5
EMSCONN3	1	2	IPLV13	7	2
IBYP3	2	3	IPLV14	8	1
IBYP3	3	3	IPLV14	9	2
IDTM1	5	2	IPVL1C	1	2
IDTM2	6	2	IPVLT1C	1	3
IDTM3	7	2	IPVLT1C	4	4
IDTM4	9	2	IRMU13	2	1
IIN013H	4	2	IRMU23	3	1
IIN023H	4	4	IRMULV11	5	1
IIN043H	8	2	IRMULV12	6	1
ILCCCFH	1	1	IRMULV13	7	1
ILE011H	5	2	IRMULV14	9	1
ILE012H	6	2	ISLU13	2	2
ILE013H	7	2	ISLU13	3	2
ILE014H	9	2	ISLU23	2	2
ILEPVCH	5	2	ISLU23	3	2
ILEPVCH	6	2			
ILEPVCH	7	2			
ILEPVCH	9	2			
ILINK13	2	4			
ILINK23	3	4			
ILINK33H	4	6			
ILINK41	5	2			
ILINK42	6	2			
ILINK43	7	2			
ILINK44	9	2			
ILN0V1H	5	1			
ILN0V2H	6	1			
ILN0V3H	7	1			

Figure 19D.6-15F Low Reactor Pressure Vessel Group 1 Fault Tree Division 3

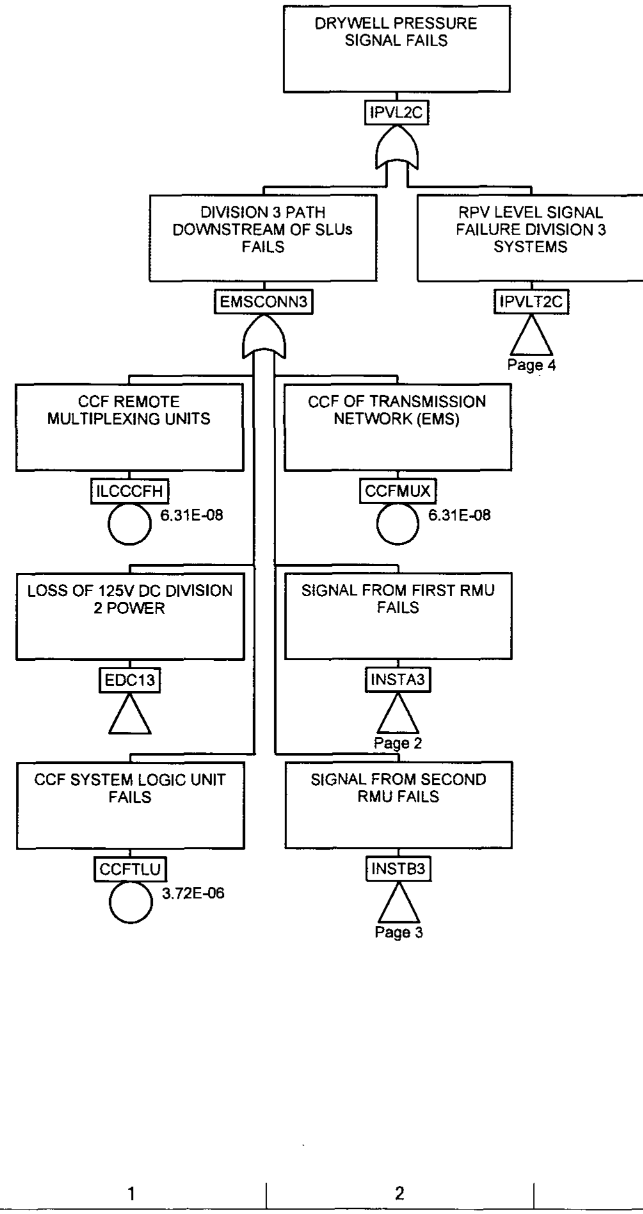

Figure 19D.6-15G Low Reactor Pressure Vessel Group 2 Fault Tree Division 3

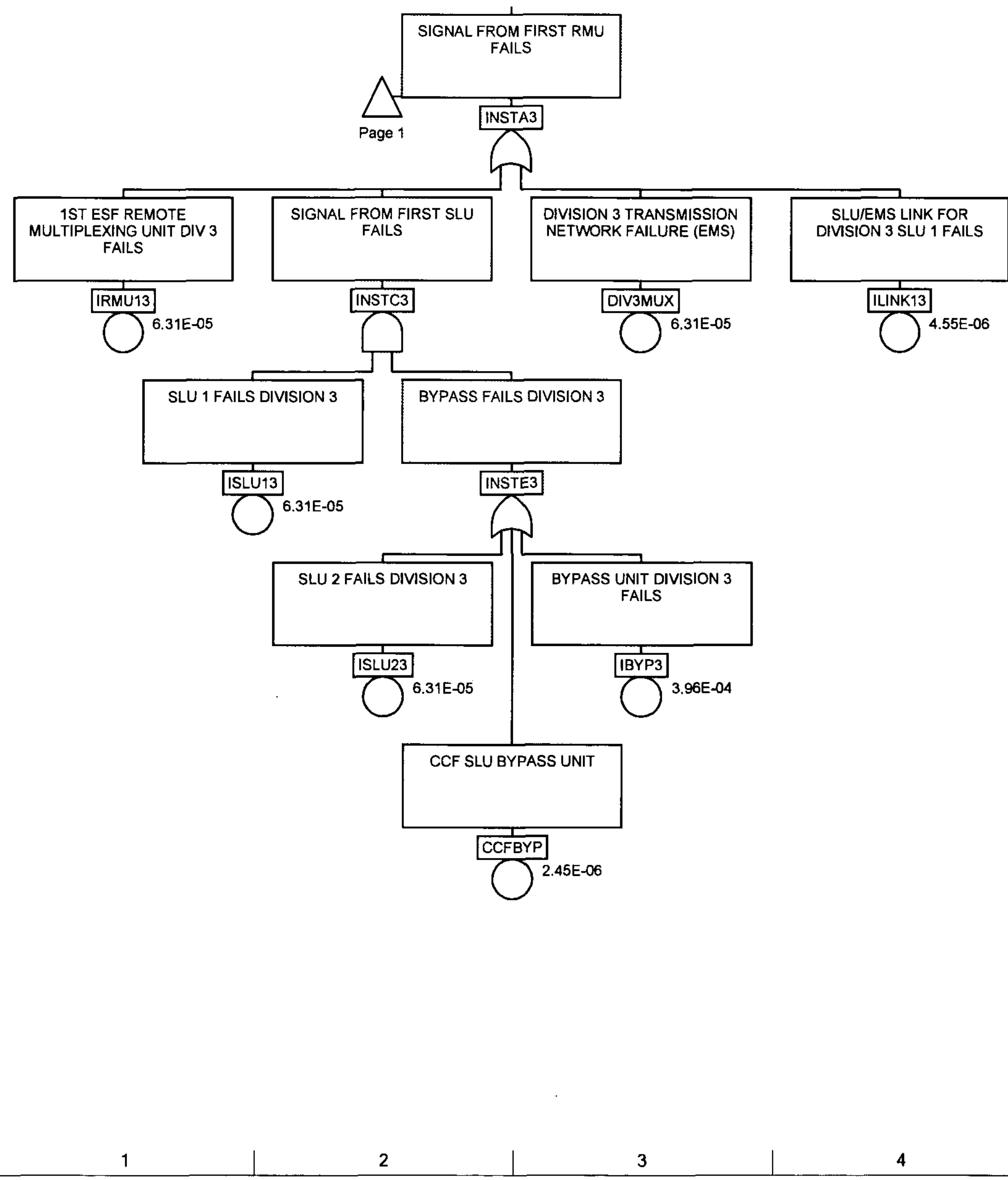

Figure 19D.6-15G Low Reactor Pressure Vessel Group 2 Fault Tree Division 3

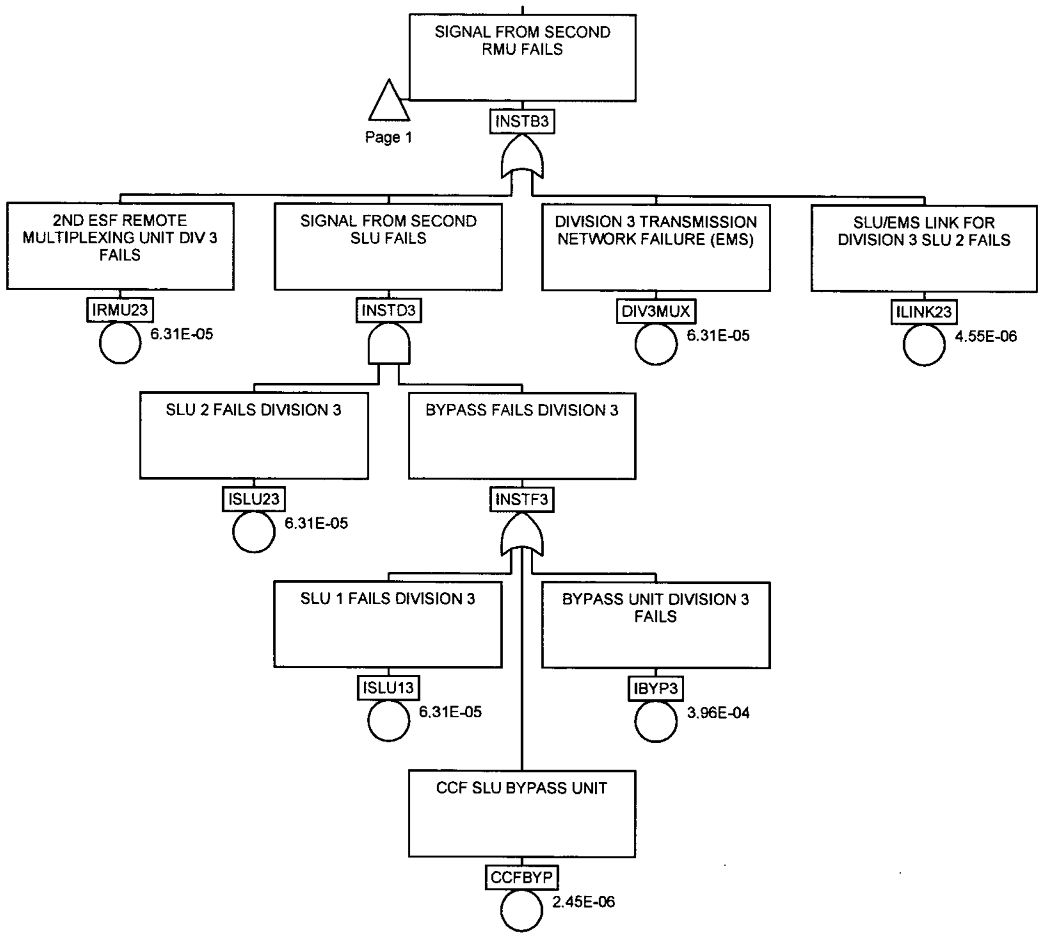

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Figure 19D.6-15G Low Reactor Pressure Vessel Group 2 Fault Tree Division 3

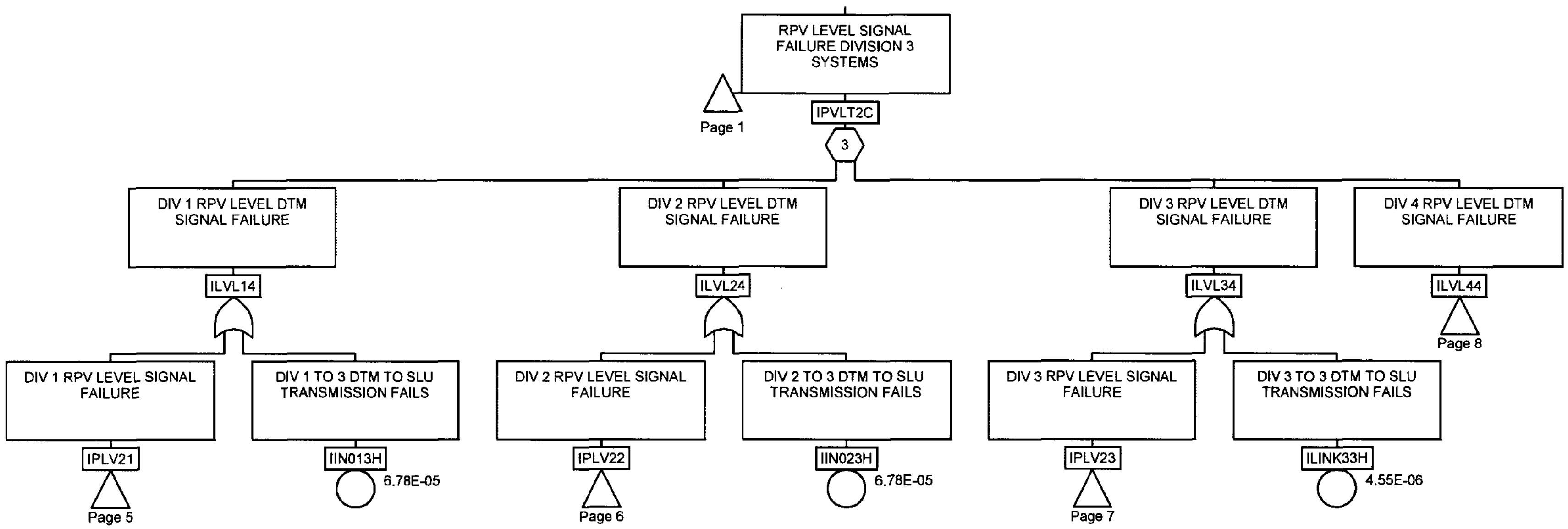

Figure 19D.6-15G Low Reactor Pressure Vessel Group 2 Fault Tree Division 3

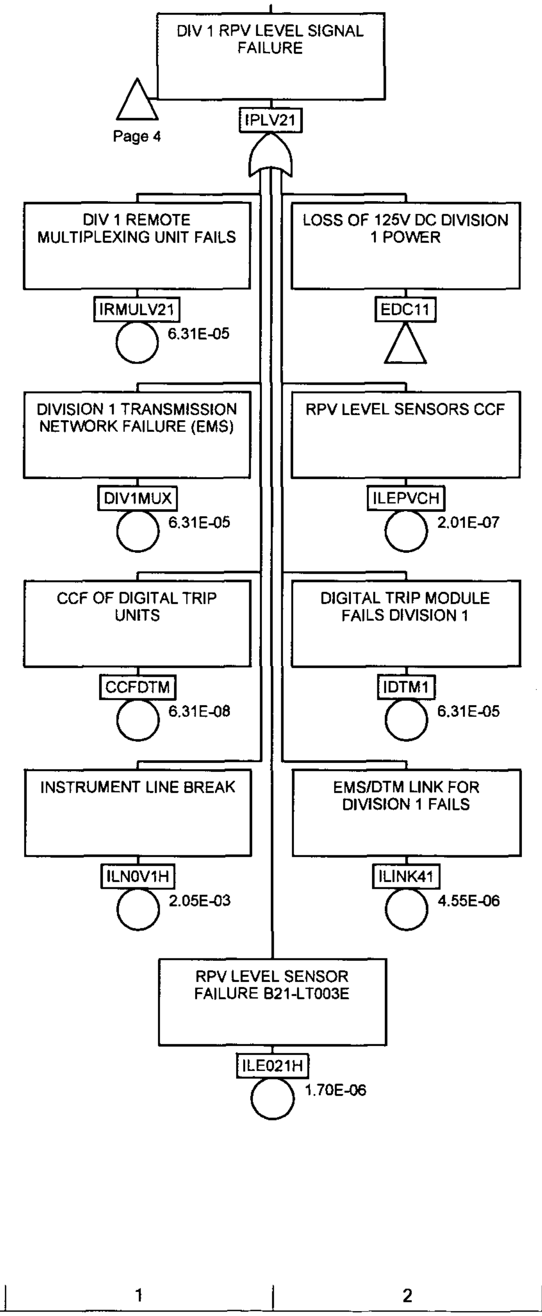

Figure 19D.6-15G Low Reactor Pressure Vessel Group 2 Fault Tree Division 3

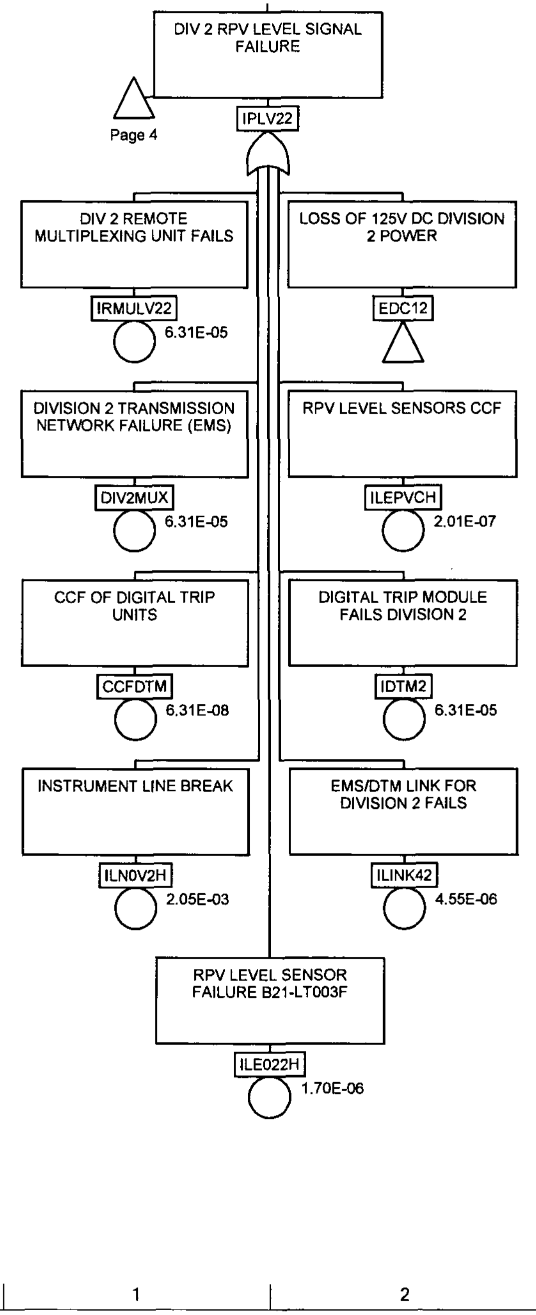

Figure 19D.6-15G Low Reactor Pressure Vessel Group 2 Fault Tree Division 3

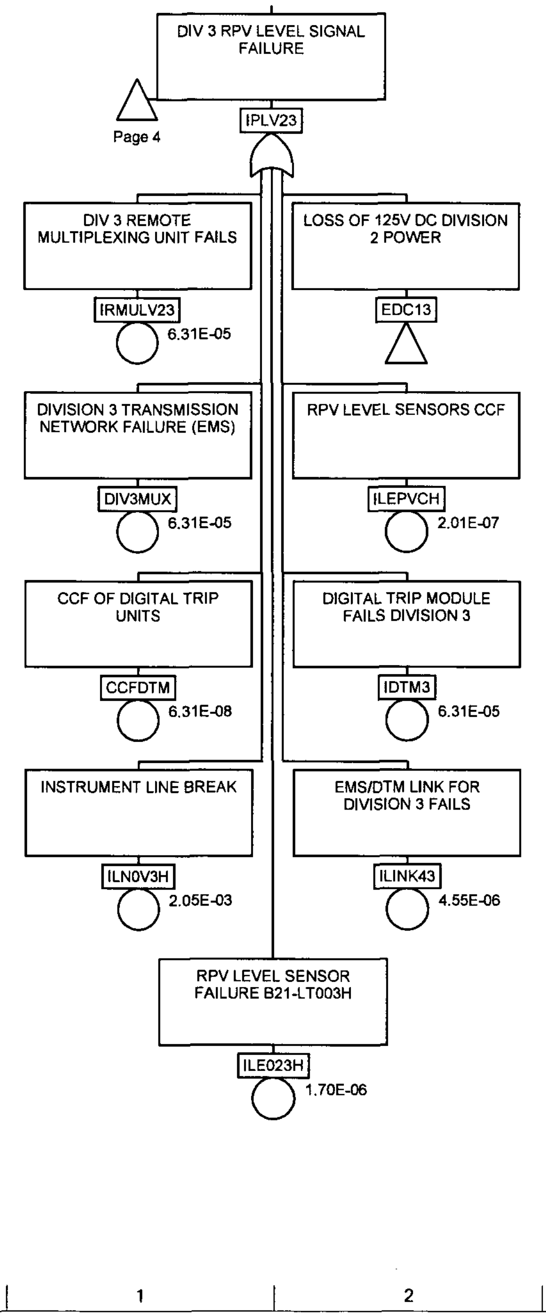

Figure 19D.6-15G Low Reactor Pressure Vessel Group 2 Fault Tree Division 3

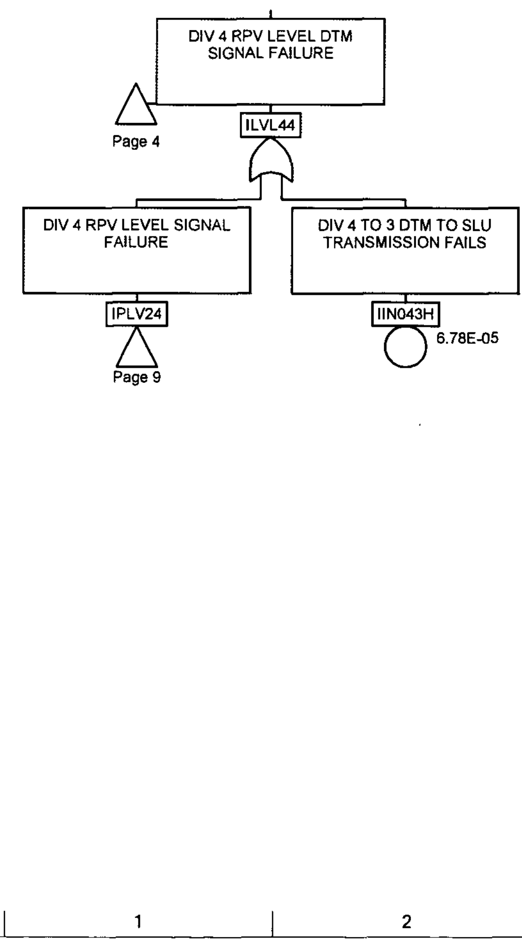

Figure 19D.6-15G Low Reactor Pressure Vessel Group 2 Fault Tree Division 3

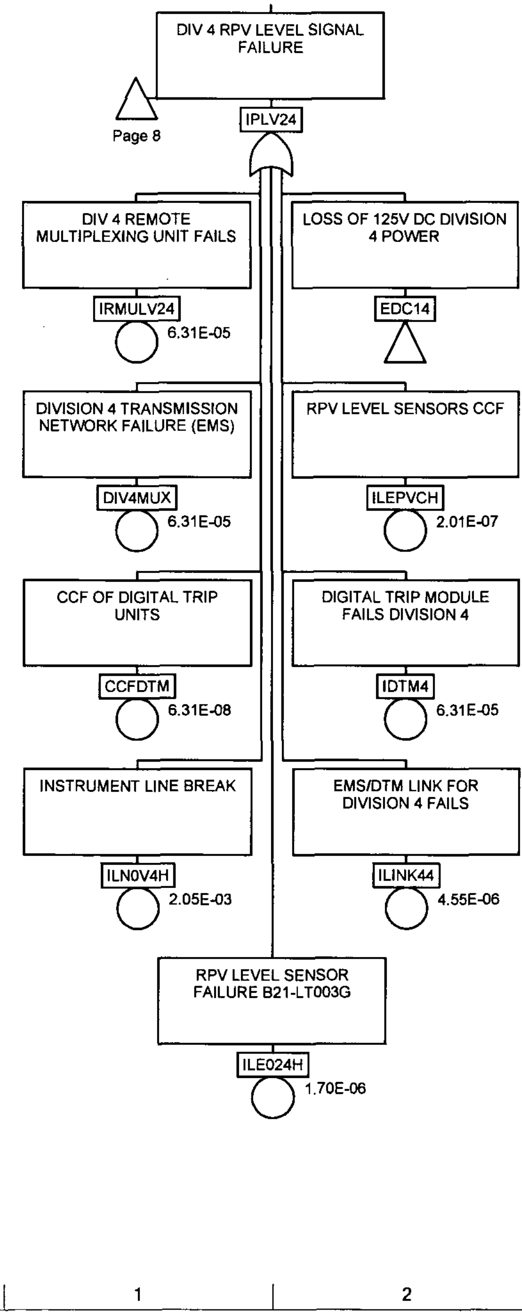

Figure 19D.6-15G Low Reactor Pressure Vessel Group 2 Fault Tree Division 3

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CCFDTM	5	1	ILVL24	4	4
CCFDTM	6	1	ILVL34	4	6
CCFDTM	7	1	ILVL44	4	7
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CCFMUX	1	2	INSTA3	1	2
CCFTLU	1	1	INSTA3	2	3
DIV1MUX	5	1	INSTB3	1	2
DIV2MUX	6	1	INSTB3	3	3
DIV3MUX	2	3	INSTC3	2	2
DIV3MUX	3	3	INSTD3	3	2
DIV3MUX	7	1	INSTE3	2	3
DIV4MUX	9	1	INSTF3	3	3
EDC11	5	2	IPLV21	4	1
EDC12	6	2	IPLV21	5	2
EDC13	1	1	IPLV22	4	3
EDC13	7	2	IPLV22	6	2
EDC14	9	2	IPLV23	4	5
EMSCONN3	1	2	IPLV23	7	2
IBYP3	2	3	IPLV24	8	1
IBYP3	3	3	IPLV24	9	2
IDTM1	5	2	IPVL2C	1	2
IDTM2	6	2	IPVLT2C	1	3
IDTM3	7	2	IPVLT2C	4	4
IDTM4	9	2	IRMU13	2	1
IIN013H	4	2	IRMU23	3	1
IIN023H	4	4	IRMULV21	5	1
IIN043H	8	2	IRMULV22	6	1
ILCCCFH	1	1	IRMULV23	7	1
ILE021H	5	2	IRMULV24	9	1
ILE022H	6	2	ISLU13	2	2
ILE023H	7	2	ISLU13	3	2
ILE024H	9	2	ISLU23	2	2
ILEPVCH	5	2	ISLU23	3	2
ILEPVCH	6	2			
ILEPVCH	7	2			
ILEPVCH	9	2			
ILINK13	2	4			
ILINK23	3	4			
ILINK33H	4	6			
ILINK41	5	2			
ILINK42	6	2			
ILINK43	7	2			
ILINK44	9	2			
ILN0V1H	5	1			
ILN0V2H	6	1			
ILN0V3H	7	1			

Figure 19D.6-15G Low Reactor Pressure Vessel Group 2 Fault Tree Division 3

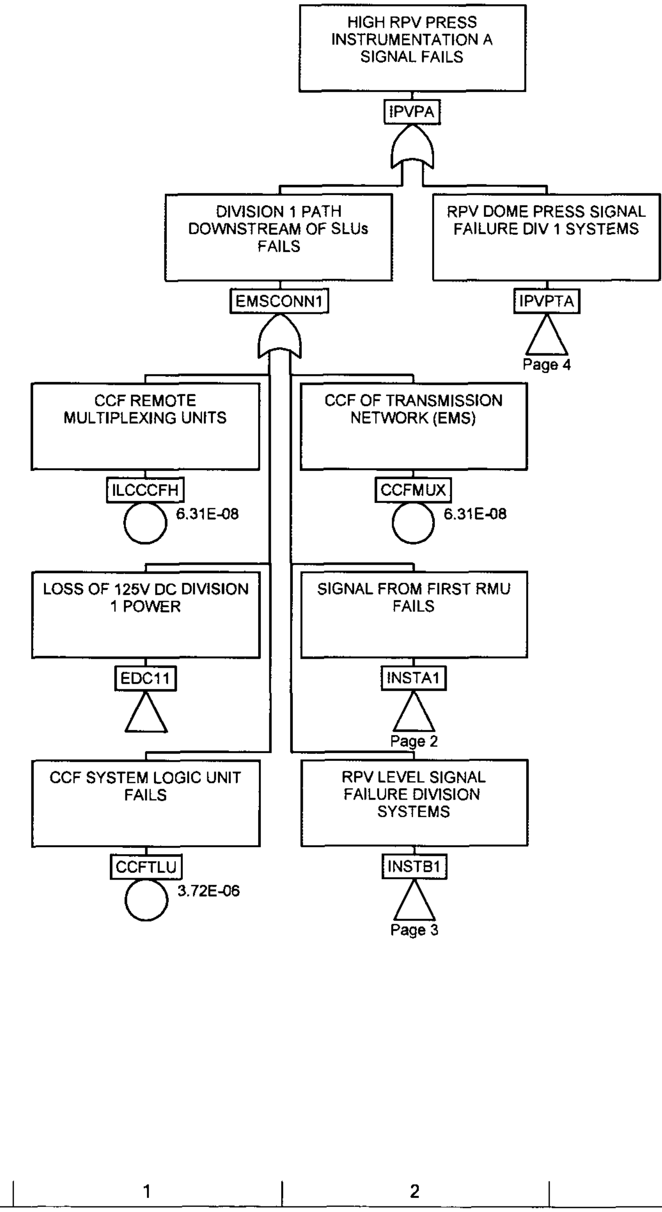

Figure 19D.6-15H High Reactor Pressure Vessel Pressure System Fault Tree Div. 1

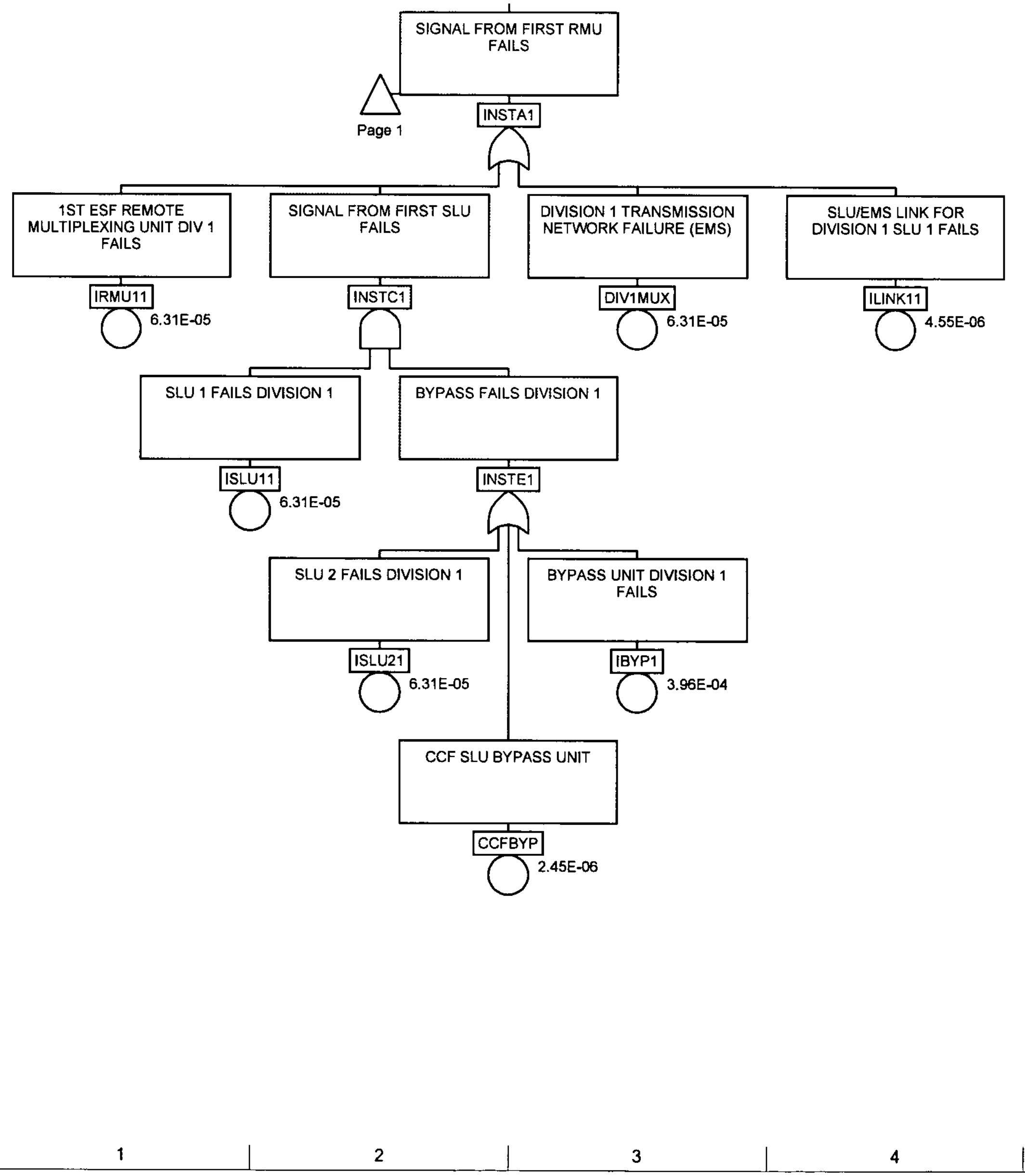

Figure 19D.6-15H High Reactor Pressure Vessel Pressure System Fault Tree Div. 1

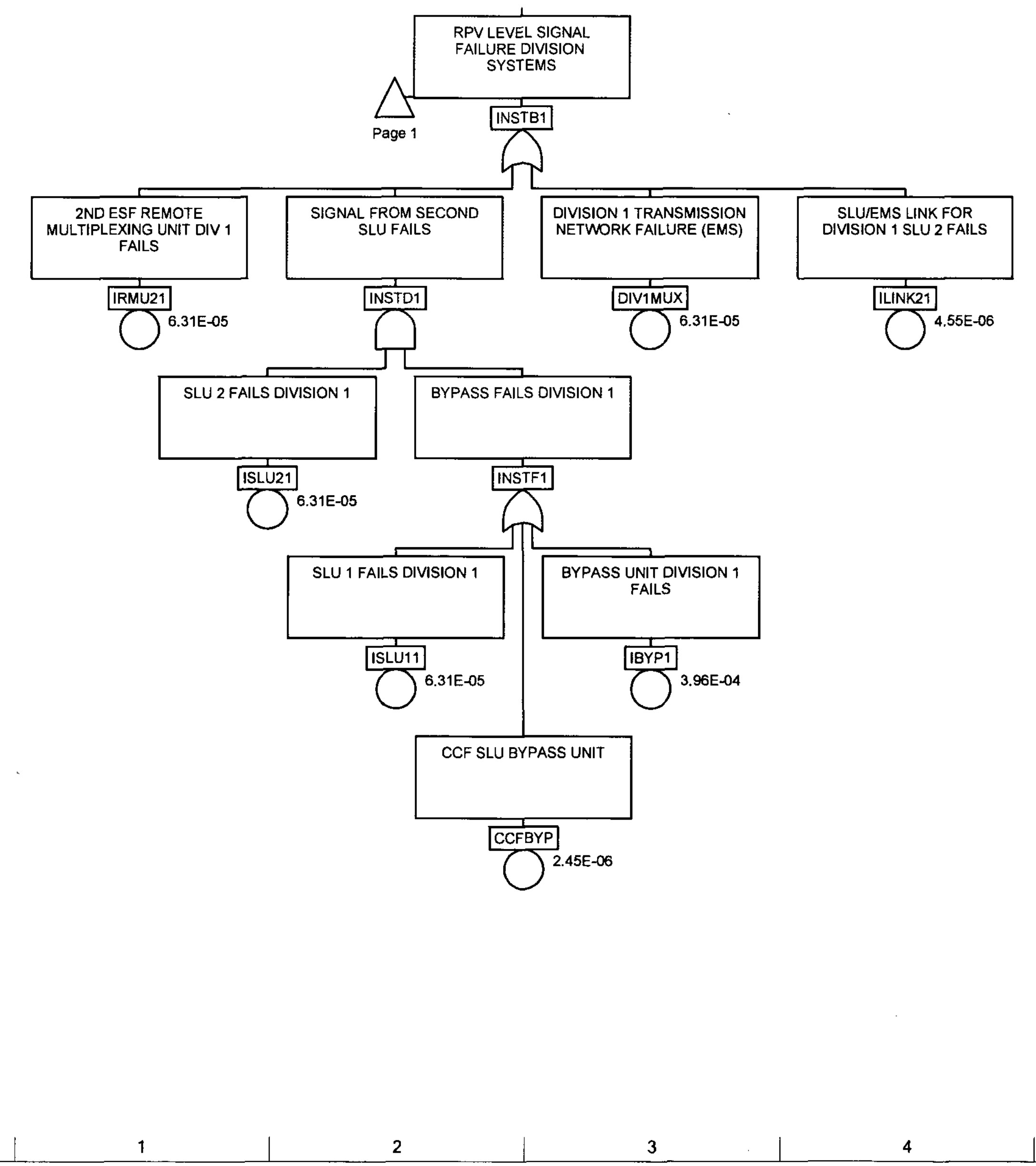

Figure 19D.6-15H High Reactor Pressure Vessel Pressure System Fault Tree Div. 1

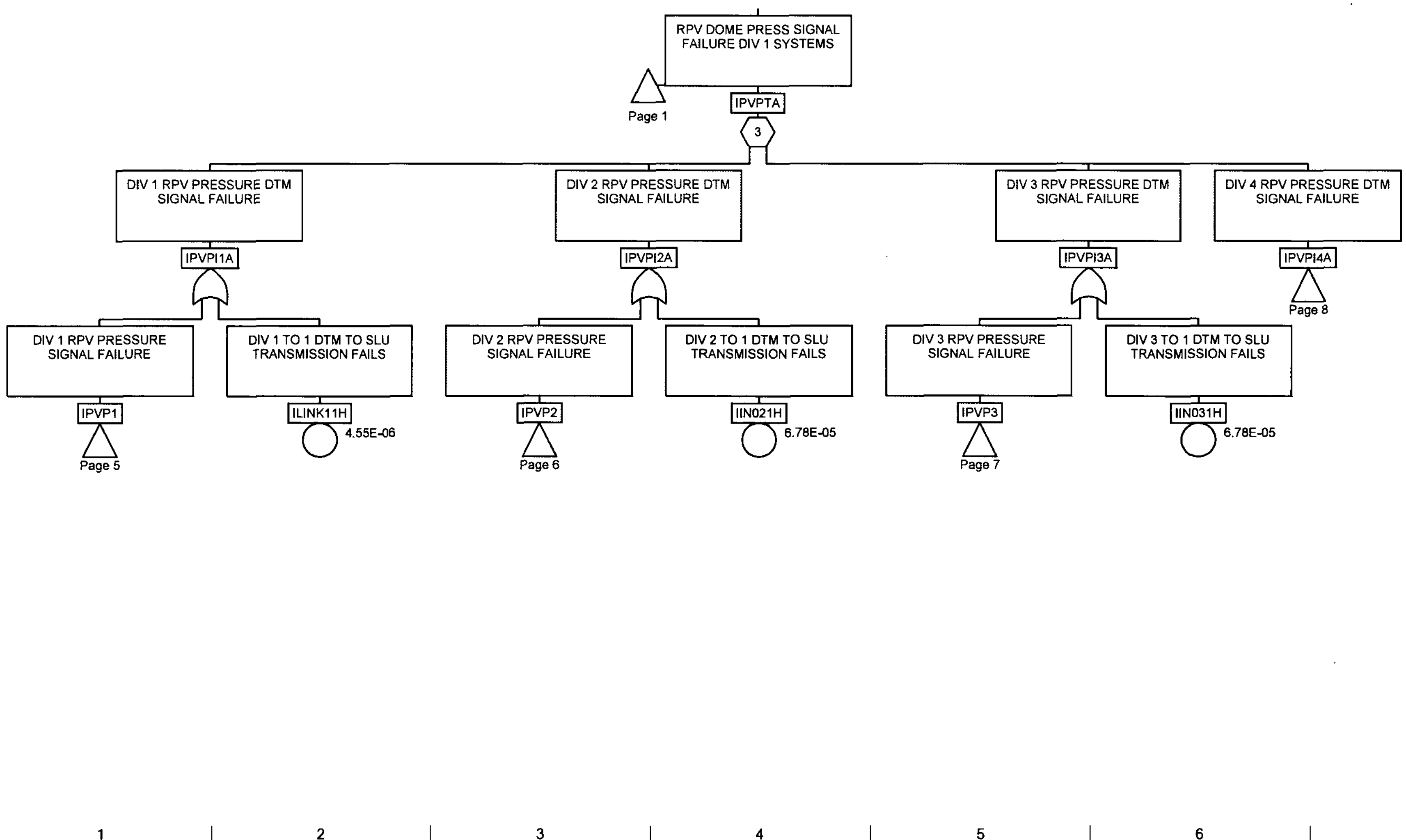

Figure 19D.6-15H High Reactor Pressure Vessel Pressure System Fault Tree Div. 1

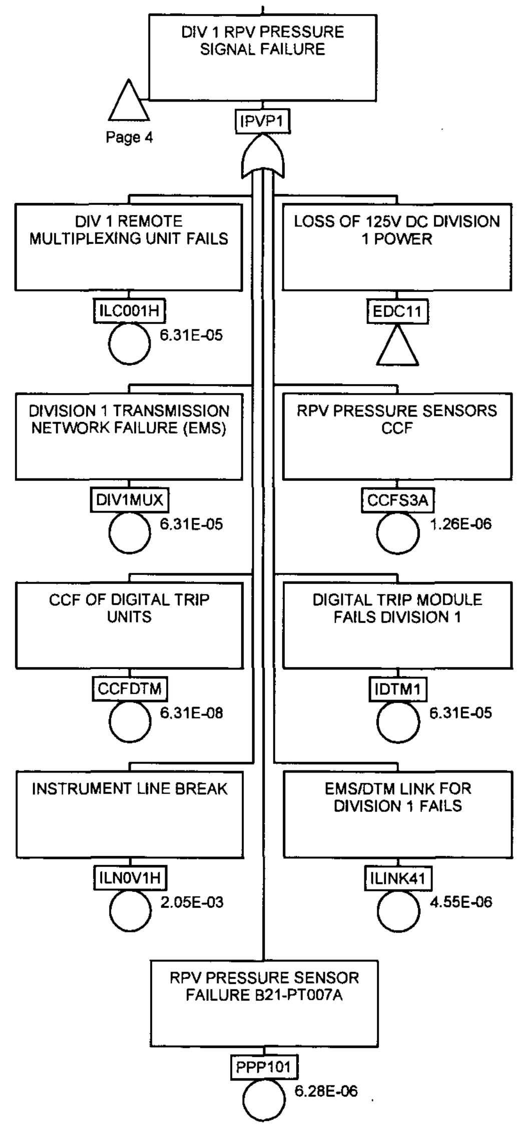

Figure 19D.6-15H High Reactor Pressure Vessel Pressure System Fault Tree Div. 1

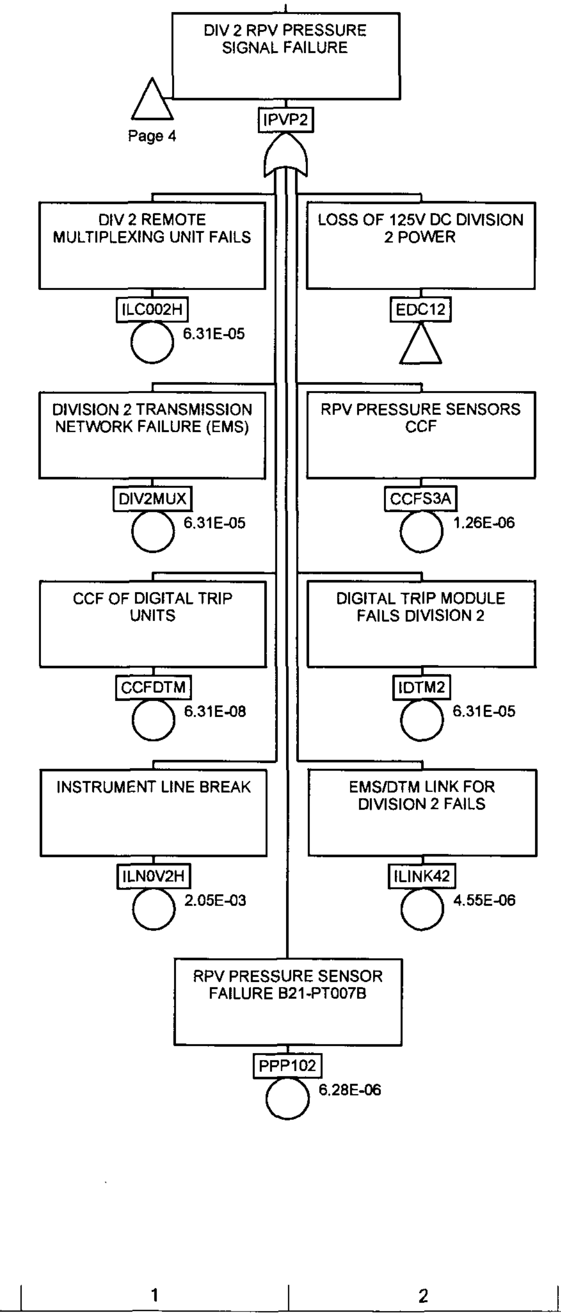

Figure 19D.6-15H High Reactor Pressure Vessel Pressure System Fault Tree Div. 1

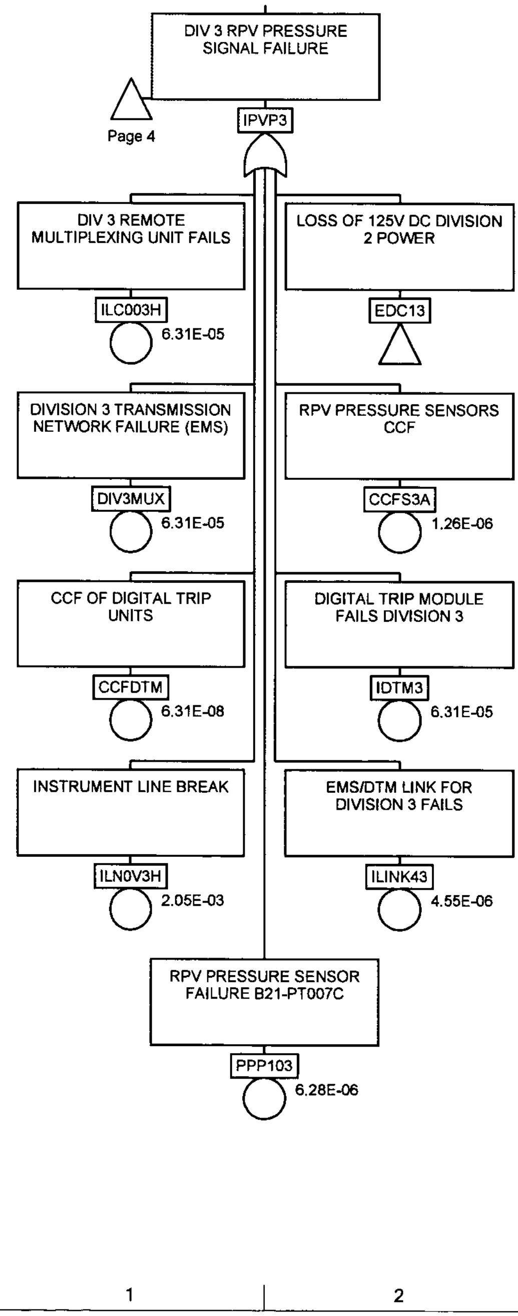

Figure 19D.6-15H High Reactor Pressure Vessel Pressure System Fault Tree Div. 1

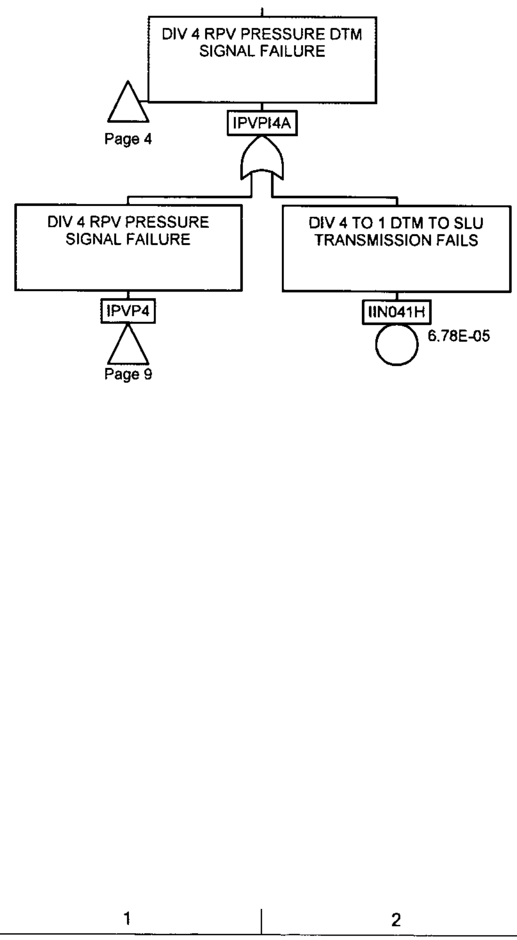

Figure 19D.6-15H High Reactor Pressure Vessel Pressure System Fault Tree Div. 1

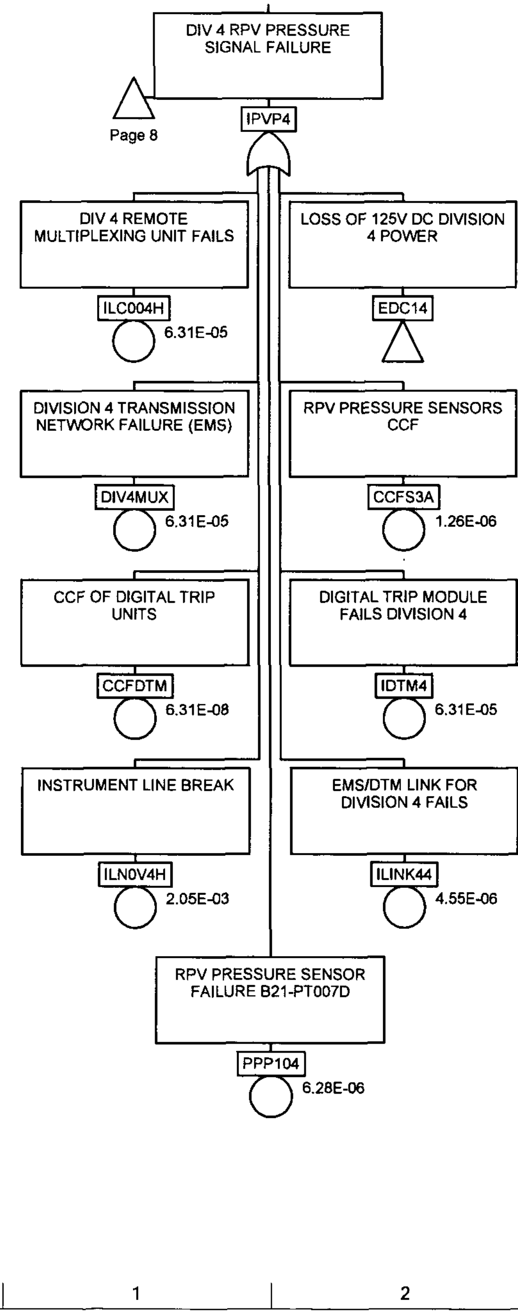

Figure 19D.6-15H High Reactor Pressure Vessel Pressure System Fault Tree Div. 1

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CCFBYP	3	3	INSTA1	1	2
CCFDTM	5	1	INSTA1	2	3
CCFDTM	6	1	INSTB1	1	2
CCFDTM	7	1	INSTB1	3	3
CCFDTM	9	1	INSTC1	2	2
CCFMUX	1	2	INSTD1	3	2
CCFS3A	5	2	INSTE1	2	3
CCFS3A	6	2	INSTF1	3	3
CCFS3A	7	2	IPVP1	4	1
CCFS3A	9	2	IPVP1	5	2
CCFTLU	1	1	IPVP2	4	3
DIV1MUX	2	3	IPVP2	6	2
DIV1MUX	3	3	IPVP3	4	5
DIV1MUX	5	1	IPVP3	7	2
DIV2MUX	6	1	IPVP4	8	1
DIV3MUX	7	1	IPVP4	9	2
DIV4MUX	9	1	IPVPA	1	2
EDC11	1	1	IPVPI1A	4	2
EDC11	5	2	IPVPI2A	4	4
EDC12	6	2	IPVPI3A	4	6
EDC13	7	2	IPVPI4A	4	7
EDC14	9	2	IPVPI4A	8	2
EMSCONN1	1	2	IPVPTA	1	3
IBYP1	2	3	IPVPTA	4	4
IBYP1	3	3	IRMU11	2	1
IDTM1	5	2	IRMU21	3	1
IDTM2	6	2	ISLU11	2	2
IDTM3	7	2	ISLU11	3	2
IDTM4	9	2	ISLU21	2	2
IIN021H	4	4	ISLU21	3	2
IIN031H	4	6	PPP101	5	2
IIN041H	8	2	PPP102	6	2
ILC001H	5	1	PPP103	7	2
ILC002H	6	1	PPP104	9	2
ILC003H	7	1			
ILC004H	9	1			
ILCCCFH	1	1			
ILINK11	2	4			
ILINK11H	4	2			
ILINK21	3	4			
ILINK41	5	2			
ILINK42	6	2			
ILINK43	7	2			
ILINK44	9	2			
ILN0V1H	5	1			
ILN0V2H	6	1			
ILN0V3H	7	1			

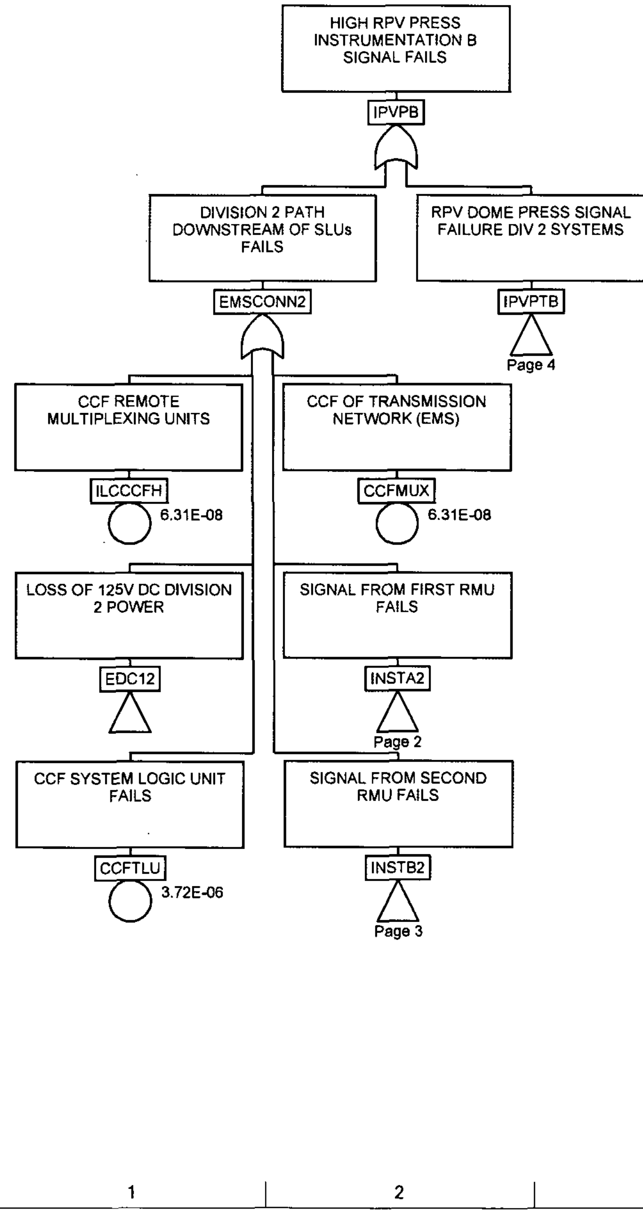


Figure 19D.6-15I High Reactor Pressure Vessel Pressure System Fault Tree Div. 2

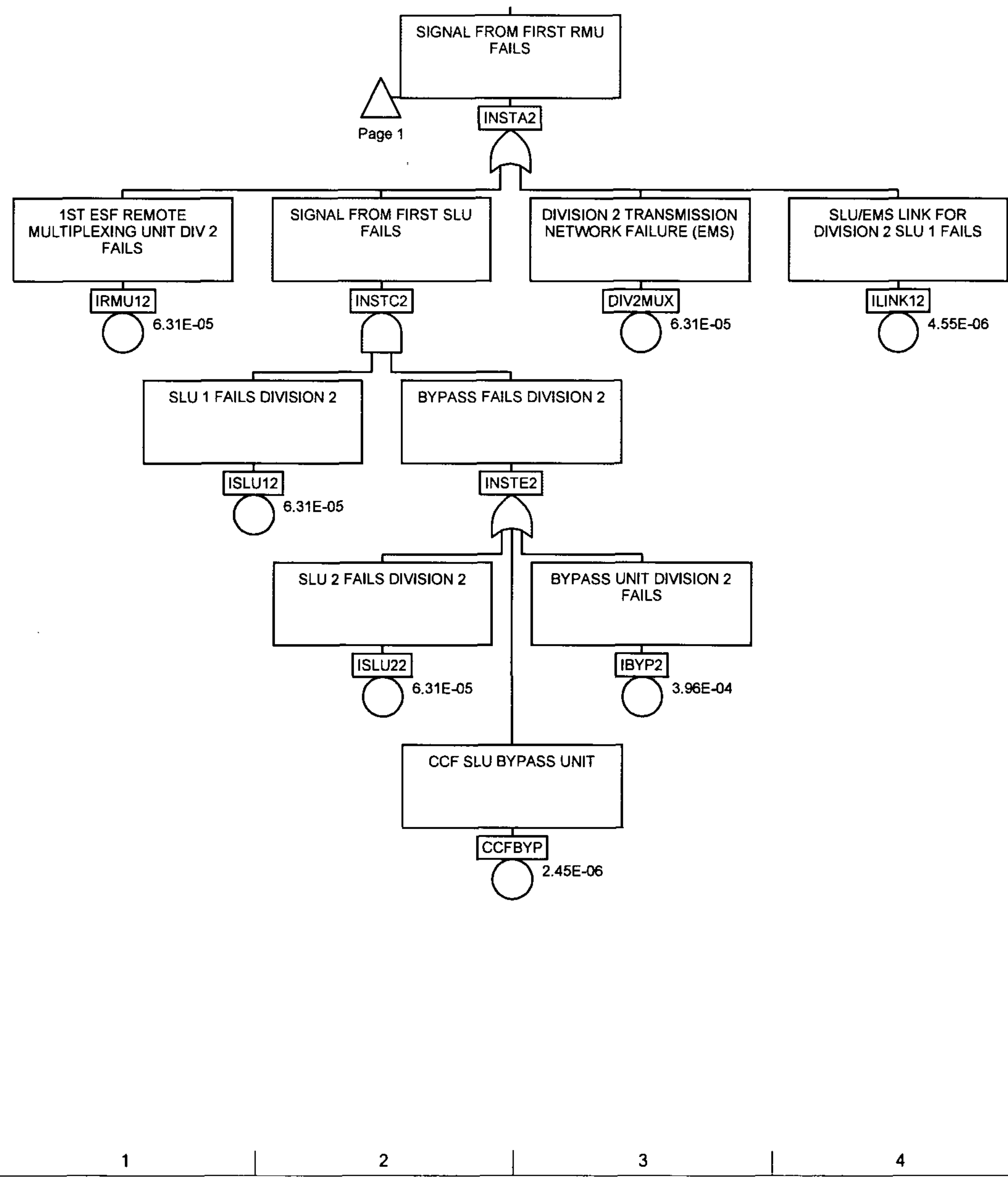


Figure 19D.6-15I High Reactor Pressure Vessel Pressure System Fault Tree Div. 2

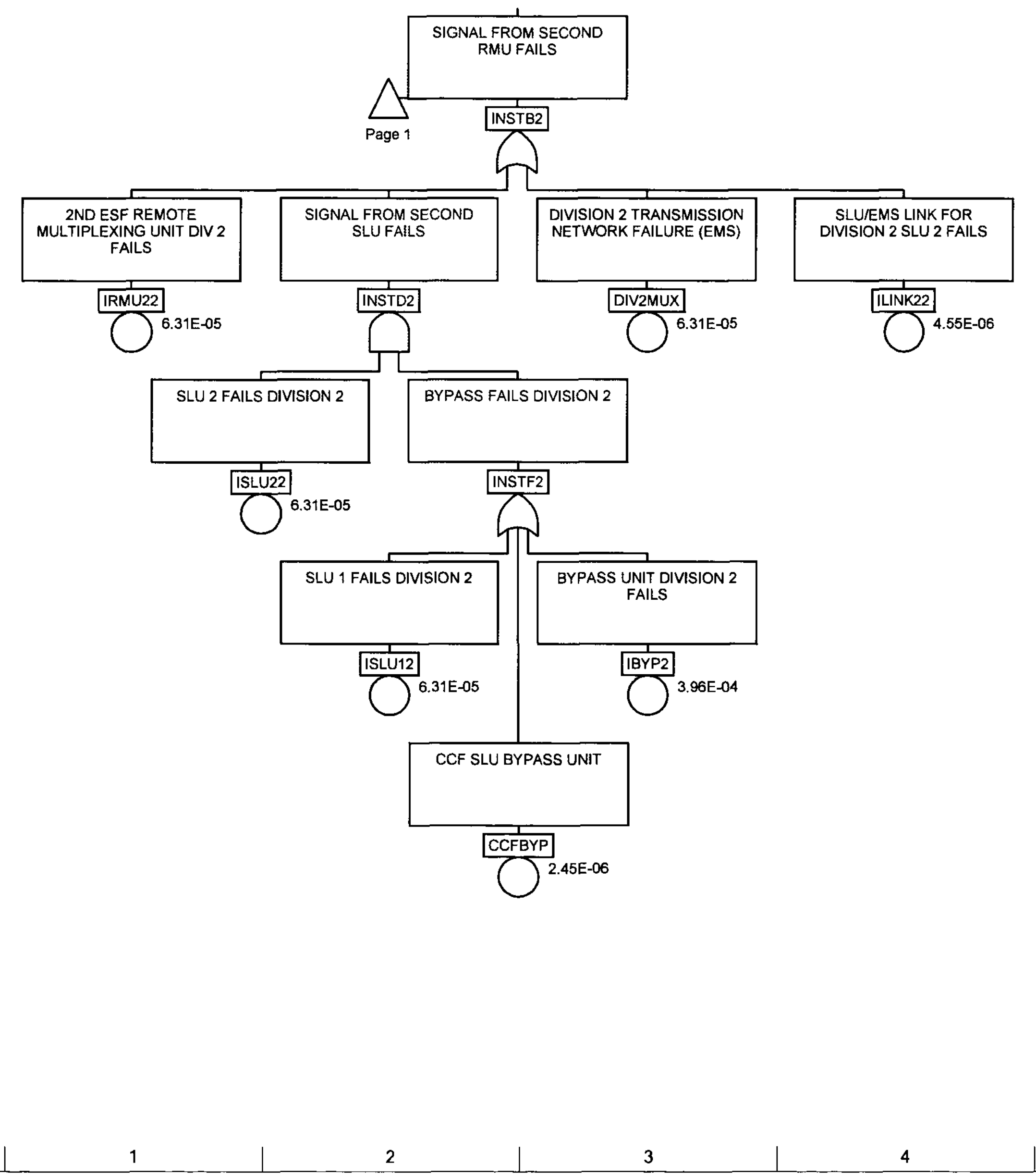


Figure 19D.6-15I High Reactor Pressure Vessel Pressure System Fault Tree Div. 2

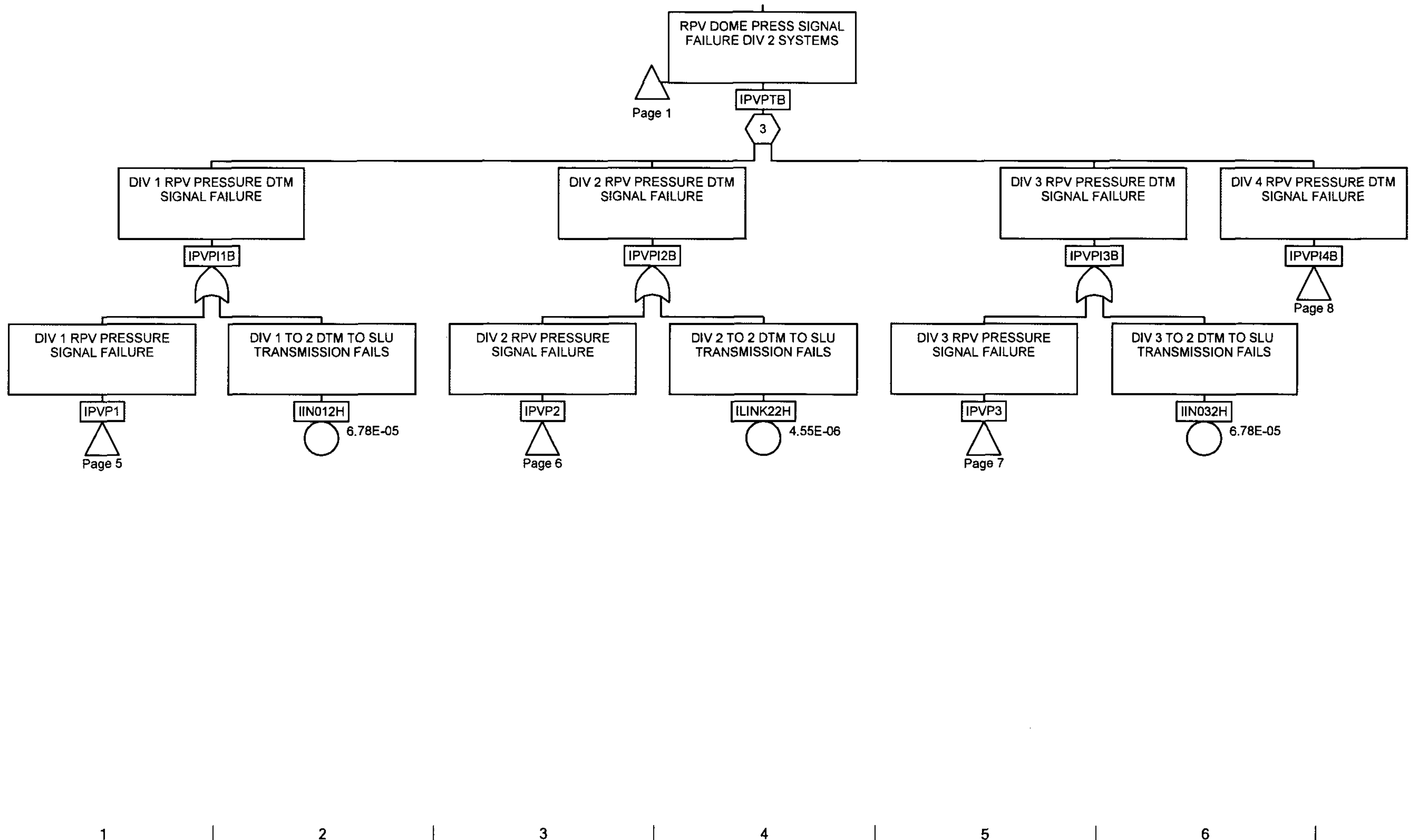


Figure 19D.6-15I High Reactor Pressure Vessel Pressure System Fault Tree Div. 2

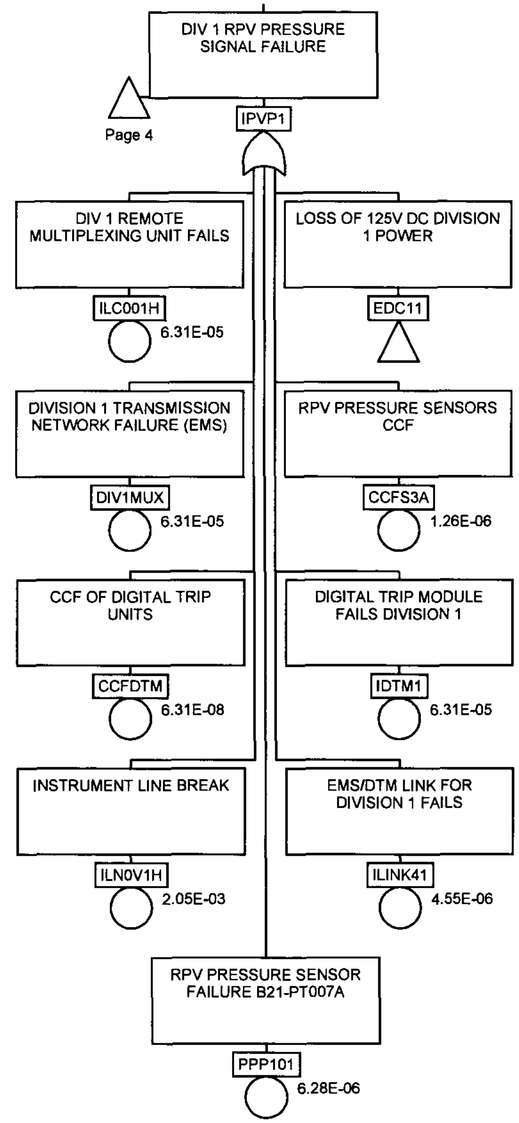


Figure 19D.6-15I High Reactor Pressure Vessel Pressure System Fault Tree Div. 2

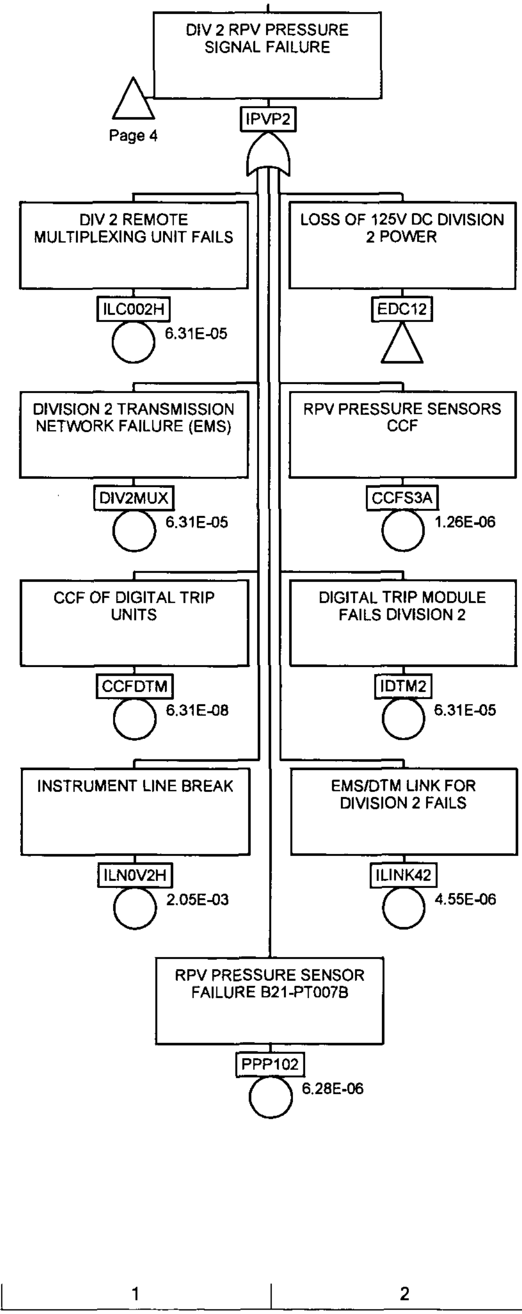


Figure 19D.6-15I High Reactor Pressure Vessel Pressure System Fault Tree Div. 2

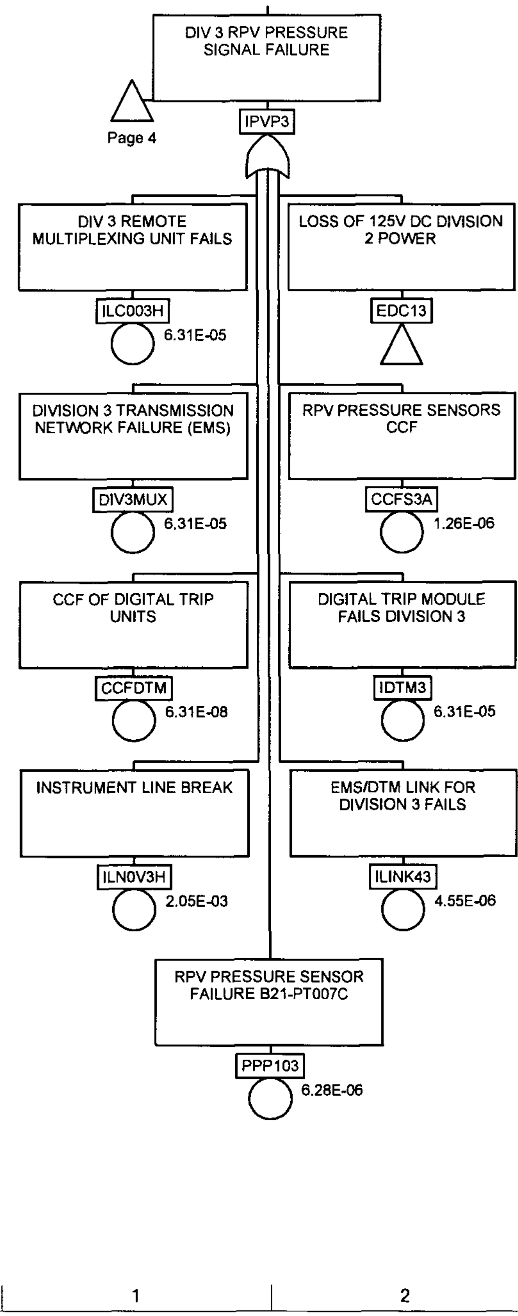


Figure 19D.6-15I High Reactor Pressure Vessel Pressure System Fault Tree Div. 2

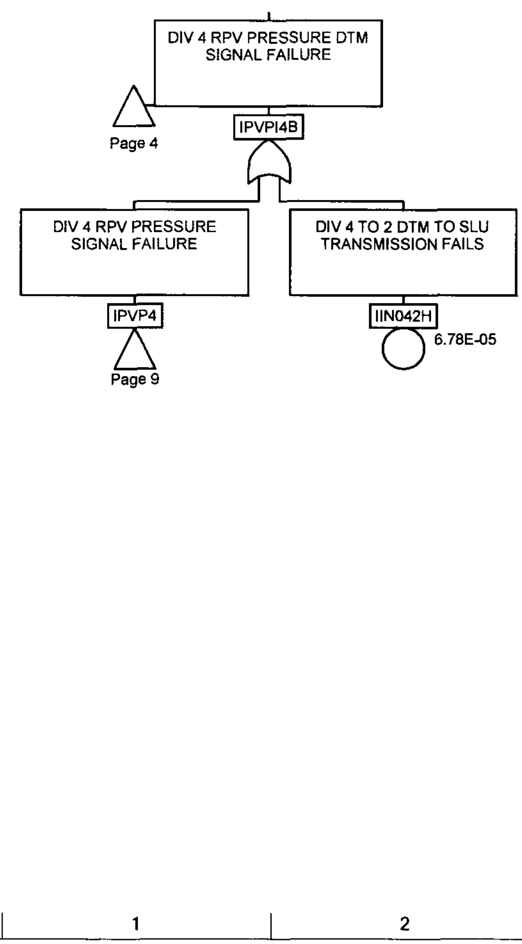


Figure 19D.6-15I High Reactor Pressure Vessel Pressure System Fault Tree Div. 2

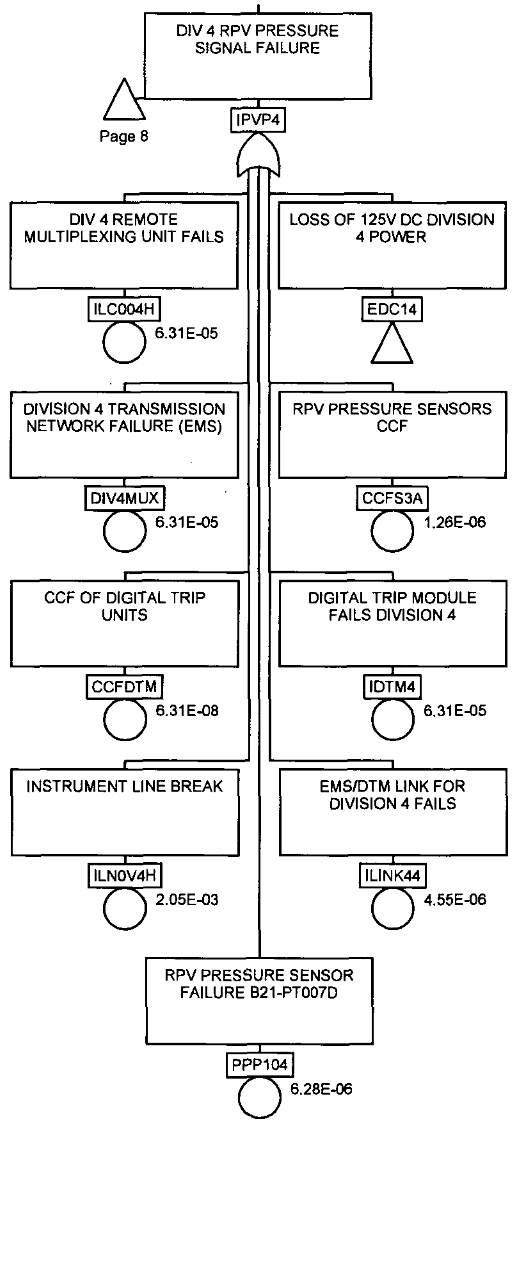


Figure 19D.6-15I High Reactor Pressure Vessel Pressure System Fault Tree Div. 2

Name	Page	Zone
CCFBYP	2	3
CCFBYP	3	3
CCFDTM	5	1
CCFDTM	6	1
CCFDTM	7	1
CCFDTM	9	1
CCFMUX	1	2
CCFS3A	5	2
CCFS3A	6	2
CCFS3A	7	2
CCFS3A	9	2
CCFTLU	1	1
DIV1MUX	5	1
DIV2MUX	2	3
DIV2MUX	3	3
DIV2MUX	6	1
DIV3MUX	7	1
DIV4MUX	9	1
EDC11	5	2
EDC12	1	1
EDC12	6	2
EDC13	7	2
EDC14	9	2
EMSCONN2	1	2
IBYP2	2	3
IBYP2	3	3
IDTM1	5	2
IDTM2	6	2
IDTM3	7	2
IDTM4	9	2
IIN012H	4	2
IIN032H	4	6
IIN042H	8	2
ILC001H	5	1
ILC002H	6	1
ILC003H	7	1
ILC004H	9	1
ILCCCFH	1	1
ILINK12	2	4
ILINK22	3	4
ILINK22H	4	4
ILINK41	5	2
ILINK42	6	2
ILINK43	7	2
ILINK44	9	2
ILN0V1H	5	1
ILN0V2H	6	1
ILN0V3H	7	1
ILN0V4H	9	1
INSTA2	1	2
INSTA2	2	3
INSTB2	1	2
INSTB2	3	3
INSTC2	2	2
INSTD2	3	2
INSTE2	2	3
INSTF2	3	3
IPVP1	4	1
IPVP1	5	2
IPVP2	4	3
IPVP2	6	2
IPVP3	4	5
IPVP3	7	2
IPVP4	8	1
IPVP4	9	2
IPVPB	1	2
IPVPI1B	4	2
IPVPI2B	4	4
IPVPI3B	4	6
IPVPI4B	4	7
IPVPI4B	8	2
IPVPTB	1	3
IPVPTB	4	4
IRMU12	2	1
IRMU22	3	1
ISLU12	2	2
ISLU12	3	2
ISLU22	2	2
ISLU22	3	2
PPP101	5	2
PPP102	6	2
PPP103	7	2
PPP104	9	2

Figure 19D.6-15I High Reactor Pressure Vessel Pressure System Fault Tree Div. 2

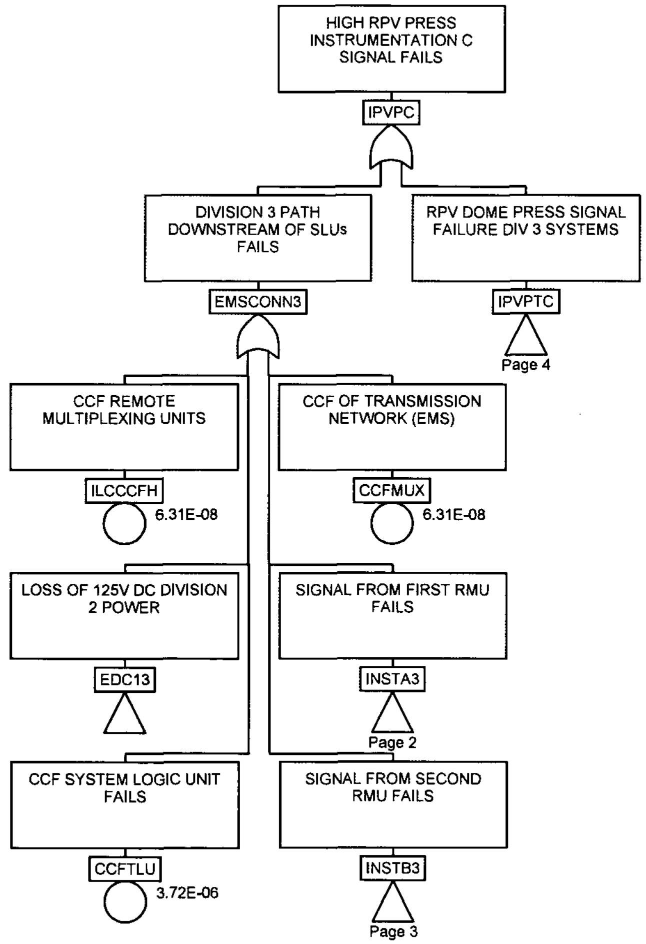


Figure 19D.6-15J High Reactor Pressure Vessel Pressure System Fault Tree Div. 3

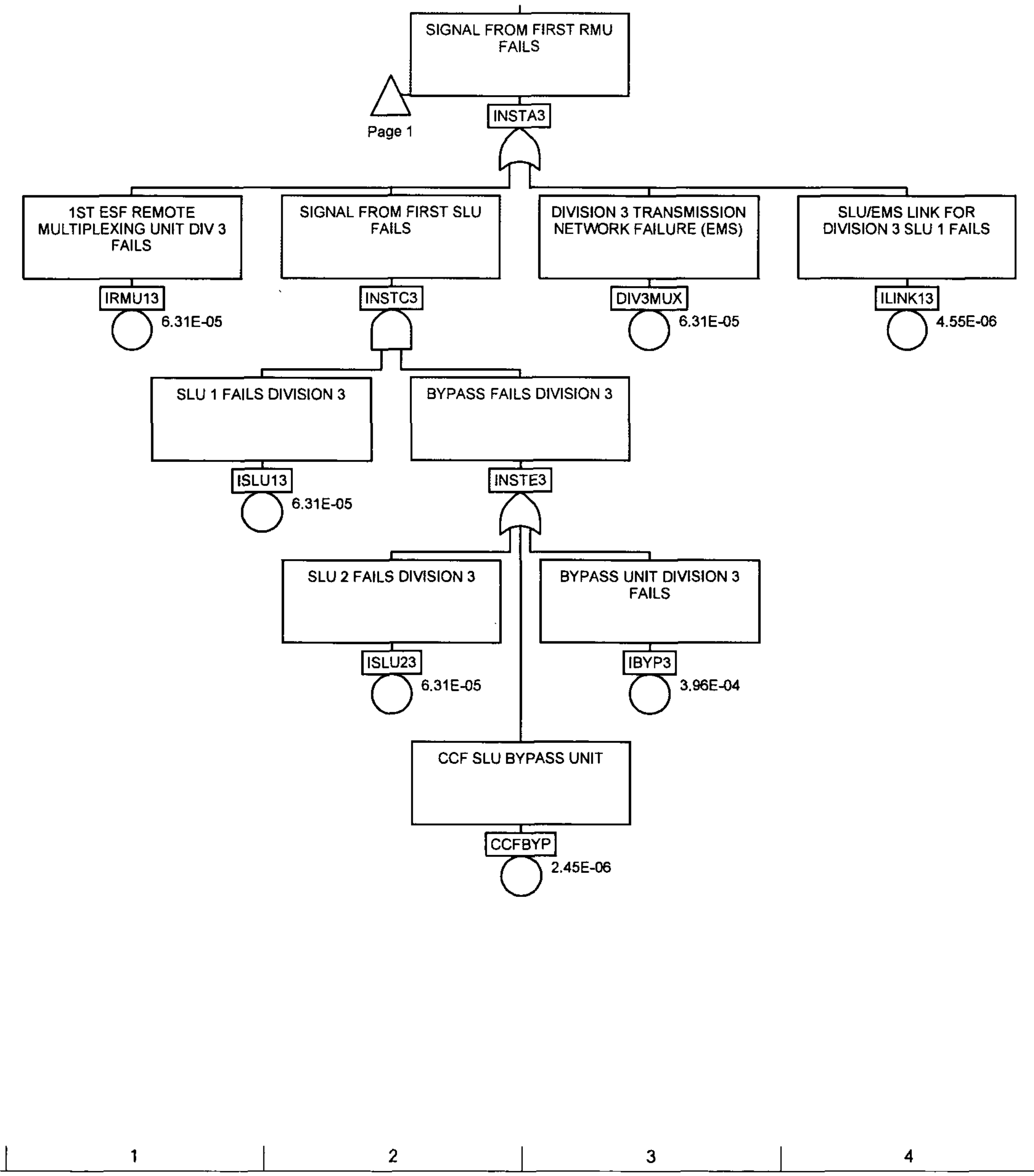


Figure 19D.6-15J High Reactor Pressure Vessel Pressure System Fault Tree Div. 3

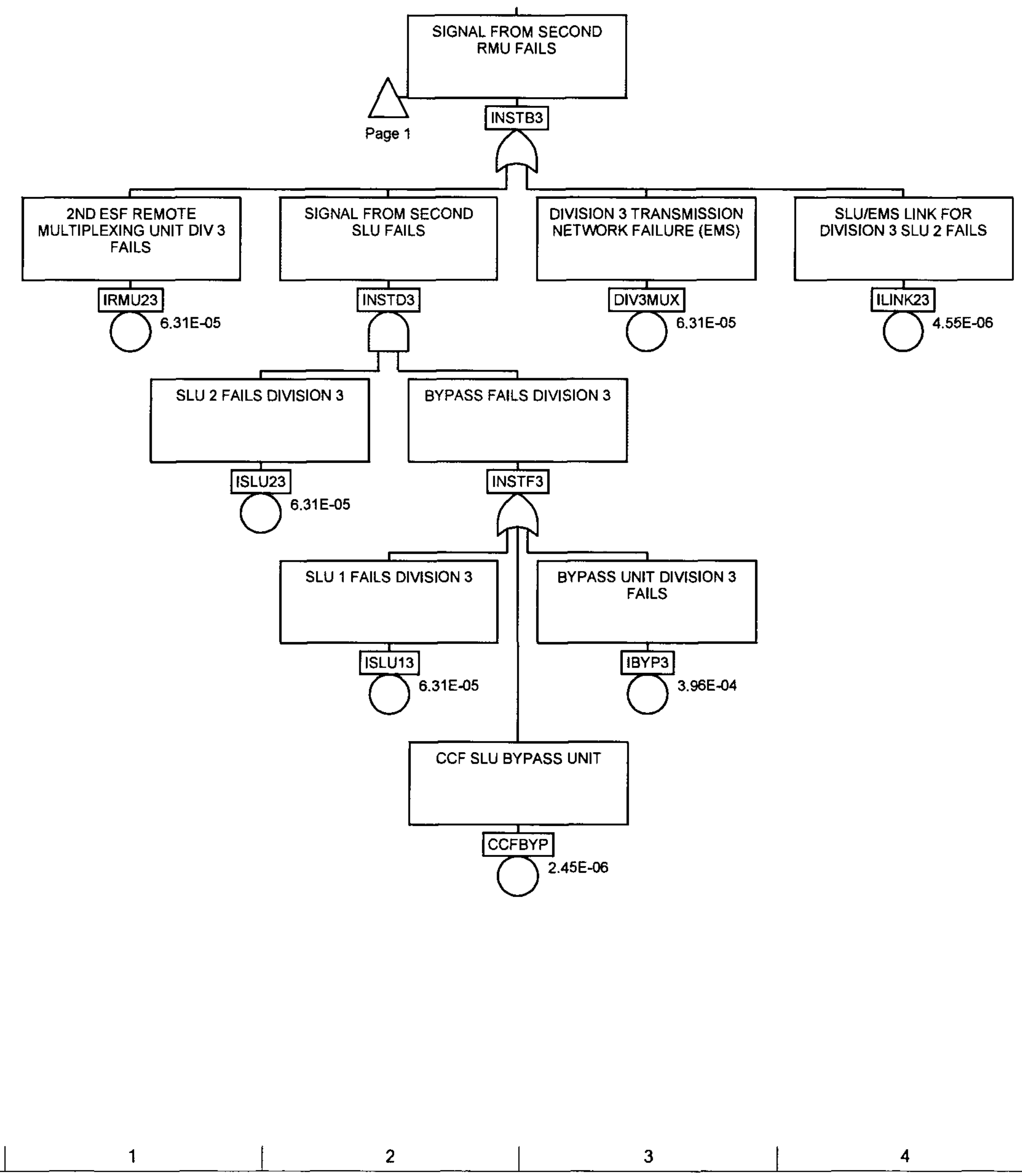


Figure 19D.6-15J High Reactor Pressure Vessel Pressure System Fault Tree Div. 3

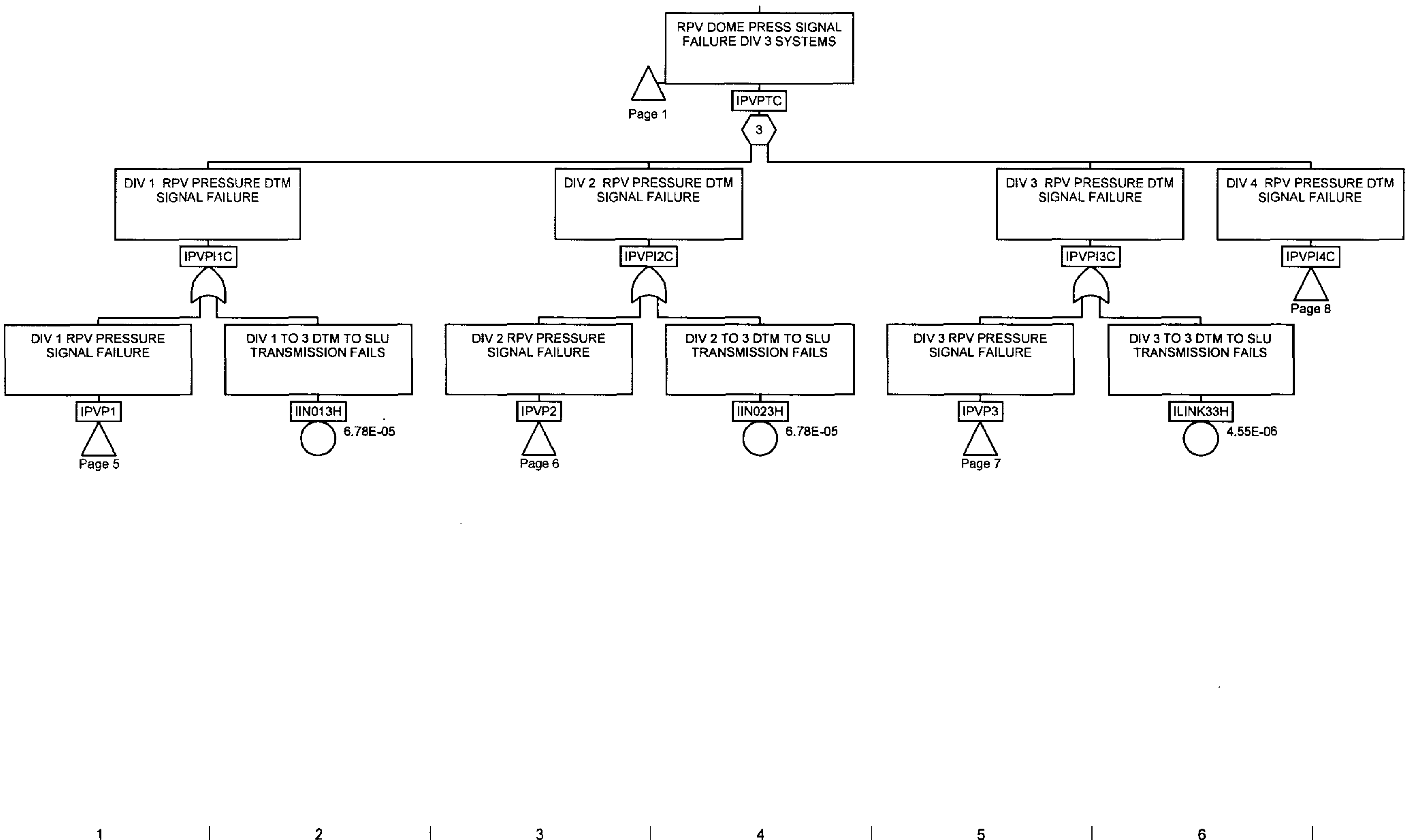


Figure 19D.6-15J High Reactor Pressure Vessel Pressure System Fault Tree Div. 3

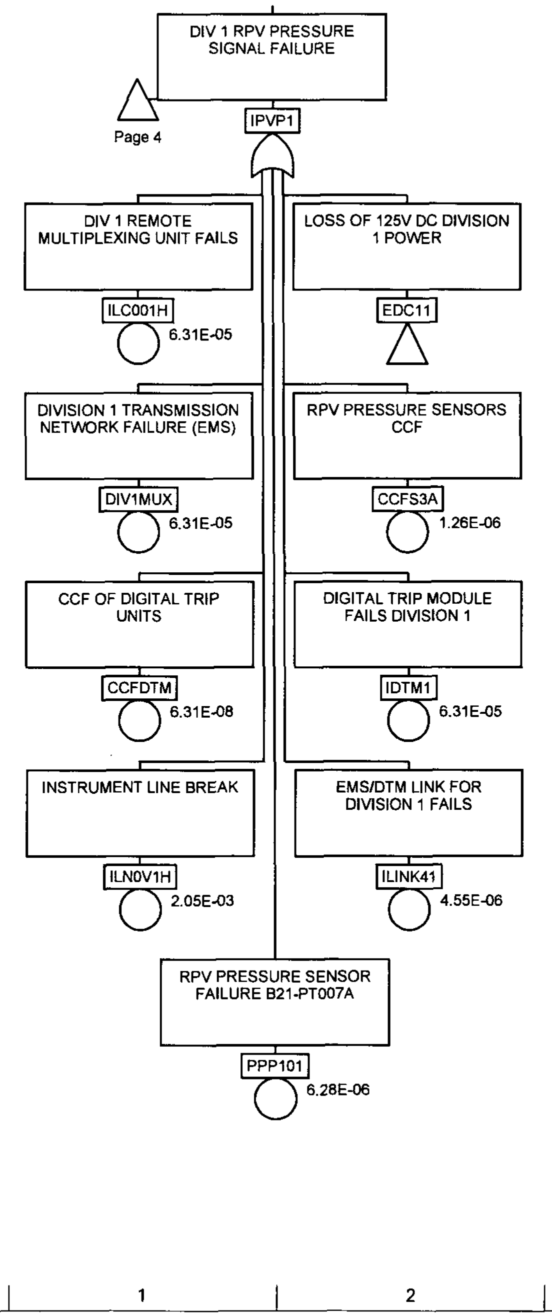


Figure 19D.6-15J High Reactor Pressure Vessel Pressure System Fault Tree Div. 3

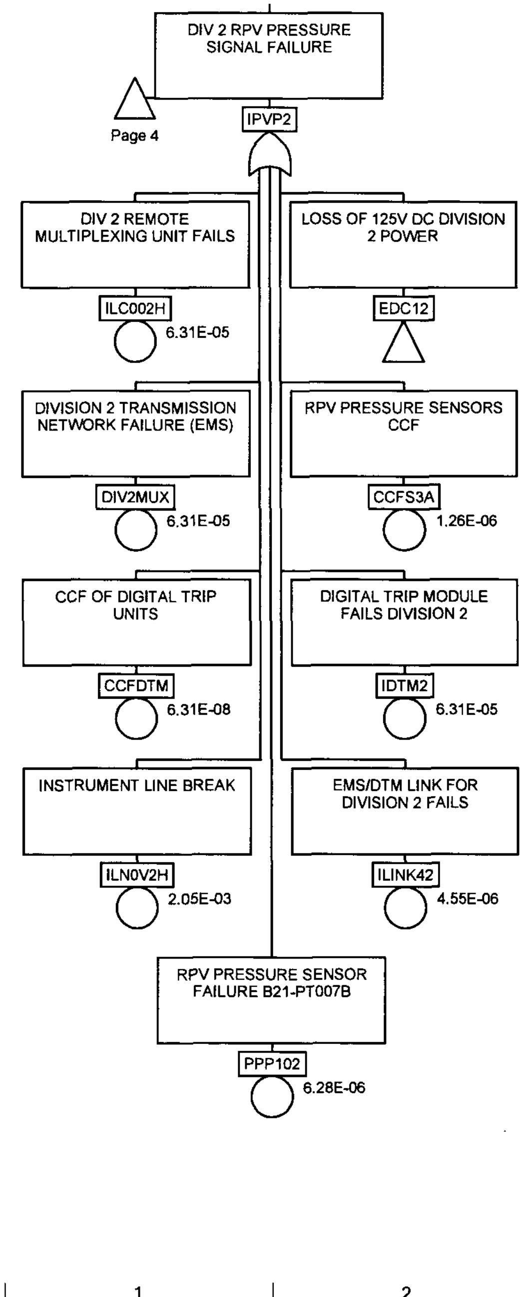


Figure 19D.6-15J High Reactor Pressure Vessel Pressure System Fault Tree Div. 3

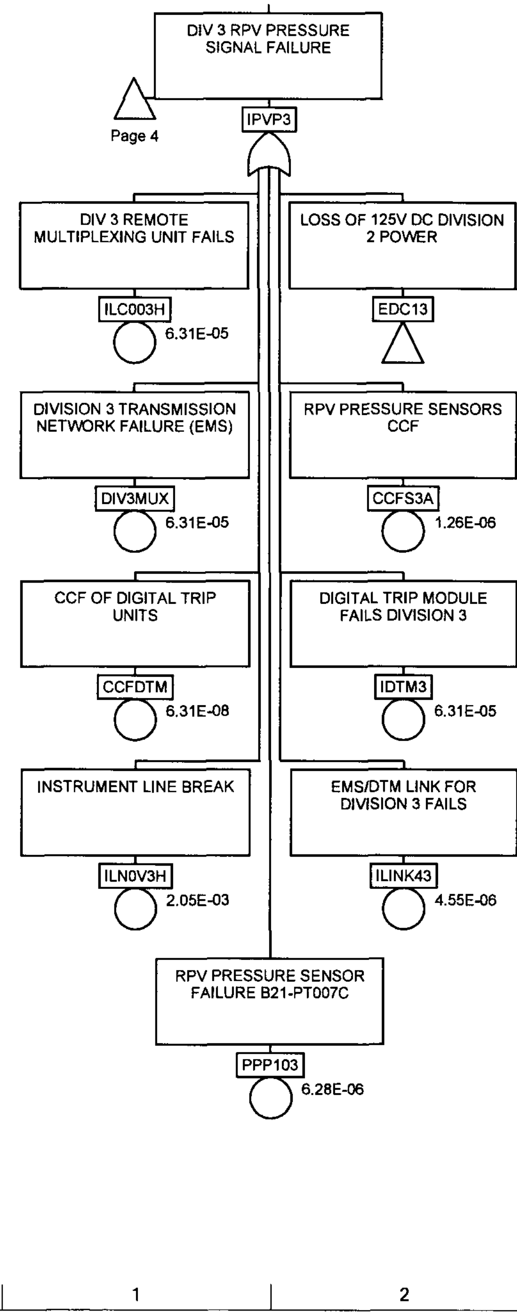
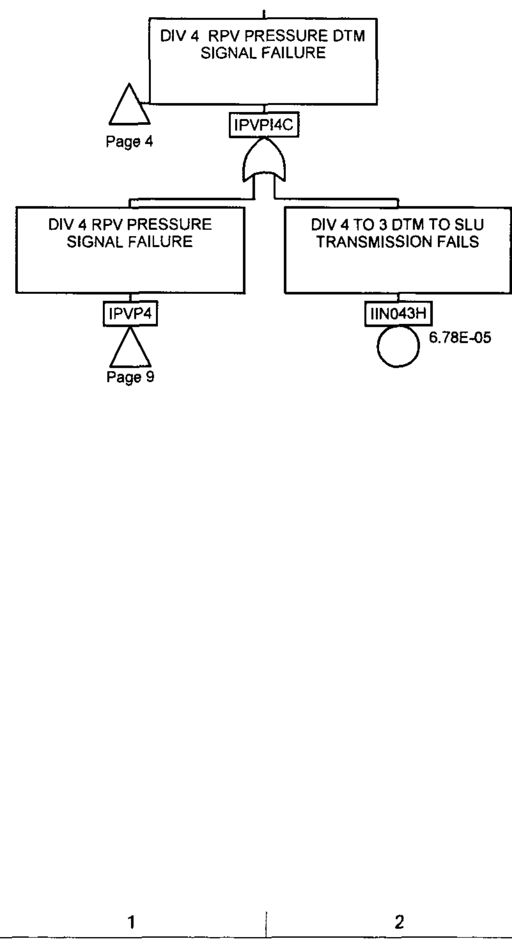


Figure 19D.6-15J High Reactor Pressure Vessel Pressure System Fault Tree Div. 3


DIV 4 RPV PRESSURE DTM SIGNAL FAILURE
Page 4
IPVPI4C
DIV 4 RPV PRESSURE SIGNAL FAILURE
DIV 4 TO 3 DTM TO SLU TRANSMISSION FAILS
IPVP4
IIN043H
6.78E-05
Page 9
1
2

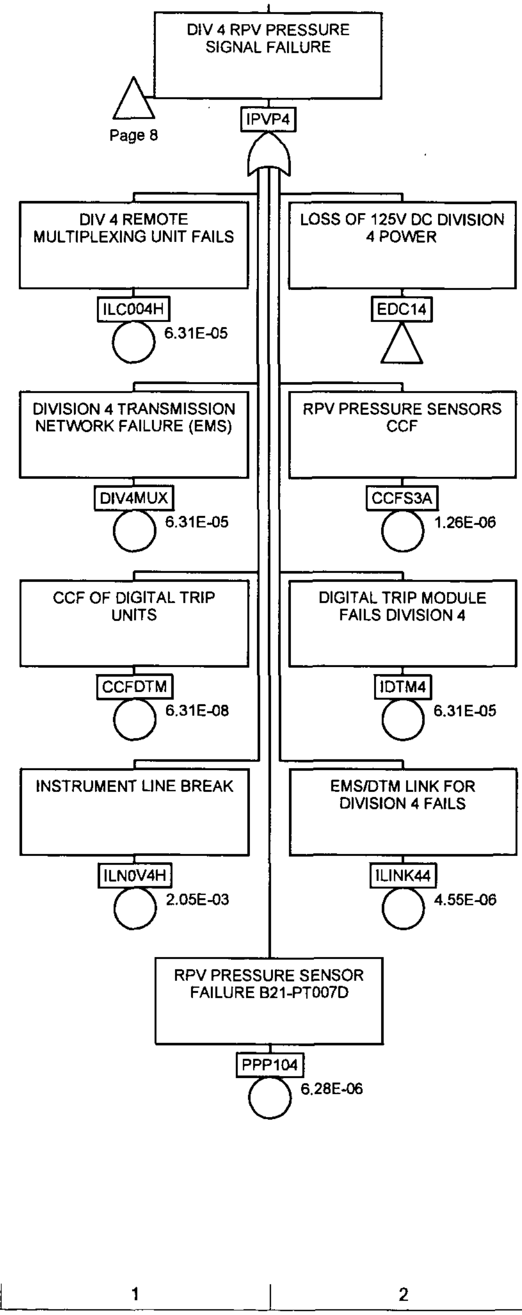


Figure 19D.6-15J High Reactor Pressure Vessel Pressure System Fault Tree Div. 3

Name	Page	Zone	Name	Page	Zone
CCFBYP	2	3	ILN0V4H	9	1
CCFBYP	3	3	INSTA3	1	2
CCFDTM	5	1	INSTA3	2	3
CCFDTM	6	1	INSTB3	1	2
CCFDTM	7	1	INSTB3	3	3
CCFDTM	9	1	INSTC3	2	2
CCFMUX	1	2	INSTD3	3	2
CCFS3A	5	2	INSTE3	2	3
CCFS3A	6	2	INSTF3	3	3
CCFS3A	7	2	IPVP1	4	1
CCFS3A	9	2	IPVP1	5	2
CCFTLU	1	1	IPVP2	4	3
DIV1MUX	5	1	IPVP2	6	2
DIV2MUX	6	1	IPVP3	4	5
DIV3MUX	2	3	IPVP3	7	2
DIV3MUX	3	3	IPVP4	8	1
DIV3MUX	7	1	IPVP4	9	2
DIV4MUX	9	1	IPVPC	1	2
EDC11	5	2	IPVPI1C	4	2
EDC12	6	2	IPVPI2C	4	4
EDC13	1	1	IPVPI3C	4	6
EDC13	7	2	IPVPI4C	4	7
EDC14	9	2	IPVPI4C	8	2
EMSCONN3	1	2	IPVPTC	1	3
IBYP3	2	3	IPVPTC	4	4
IBYP3	3	3	IRMU13	2	1
IDTM1	5	2	IRMU23	3	1
IDTM2	6	2	ISLU13	2	2
IDTM3	7	2	ISLU13	3	2
IDTM4	9	2	ISLU23	2	2
IIN013H	4	2	ISLU23	3	2
IIN023H	4	4	PPP101	5	2
IIN043H	8	2	PPP102	6	2
ILC001H	5	1	PPP103	7	2
ILC002H	6	1	PPP104	9	2
ILC003H	7	1			
ILC004H	9	1			
ILCCCFH	1	1			
ILINK13	2	4			
ILINK23	3	4			
ILINK33H	4	6			
ILINK41	5	2			
ILINK42	6	2			
ILINK43	7	2			
ILINK44	9	2			
ILN0V1H	5	1			
ILN0V2H	6	1			
ILN0V3H	7	1			

Figure 19D.6-15J High Reactor Pressure Vessel Pressure System Fault Tree Div. 3

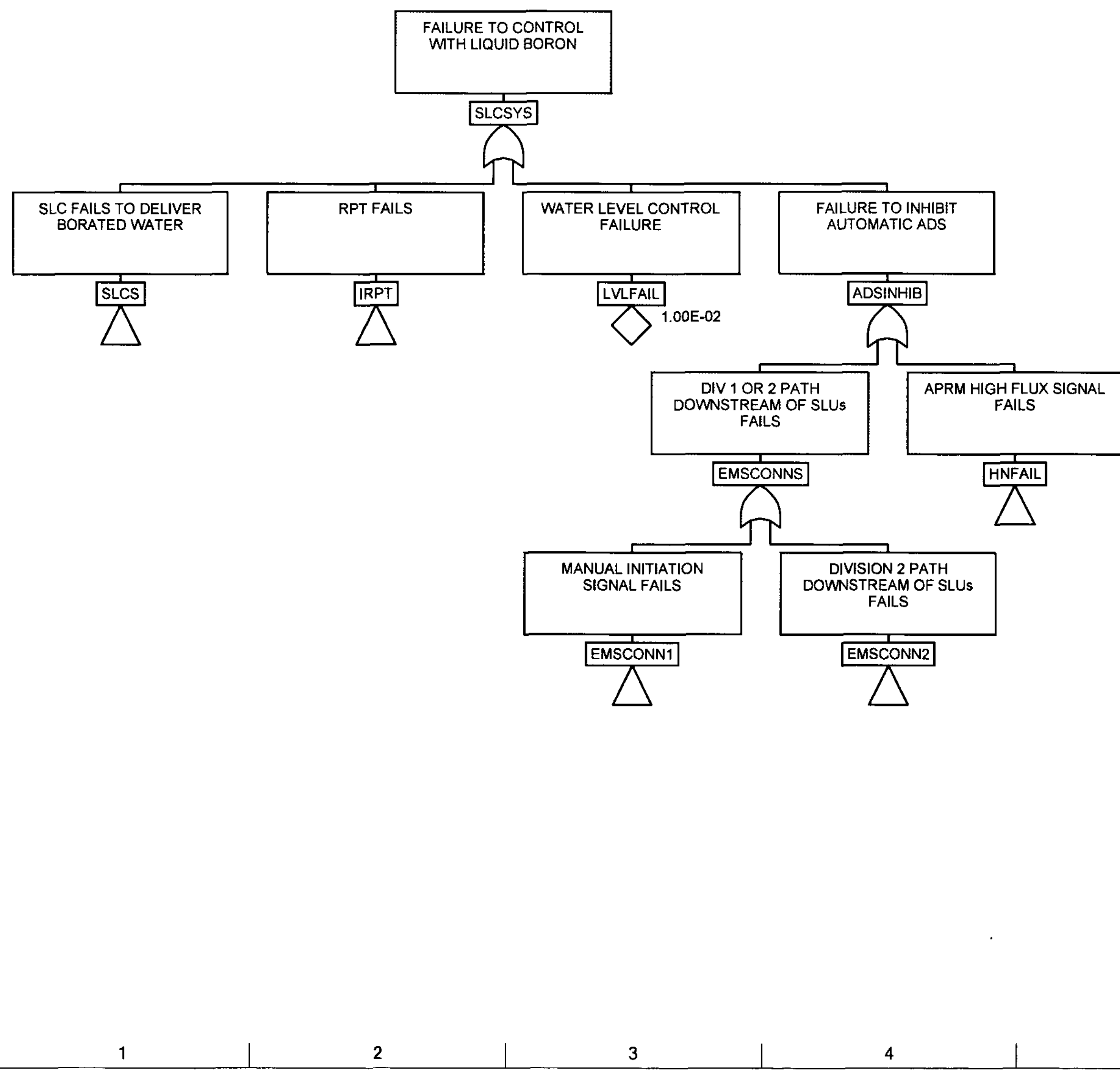


Figure 19D.6-16A Reactivity Control With Boron Functional Fault Tree

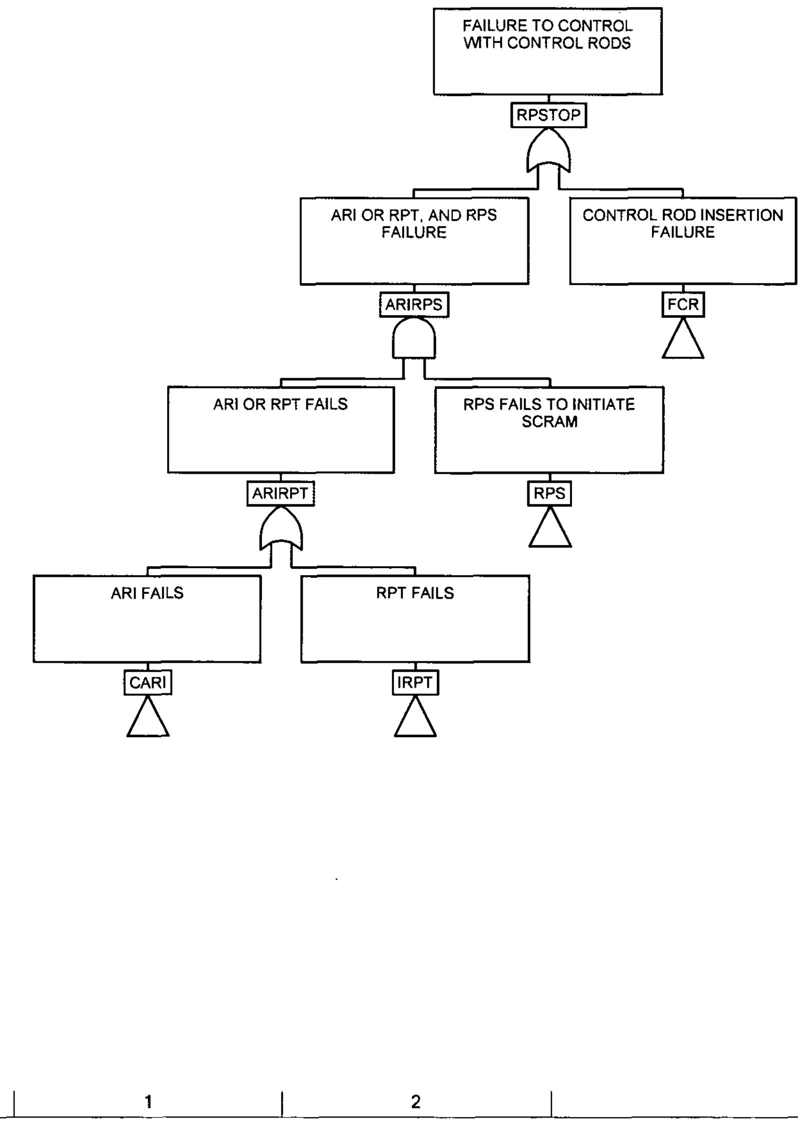


Figure 19D.6-16B Reactivity Control With Rods Fault Tree

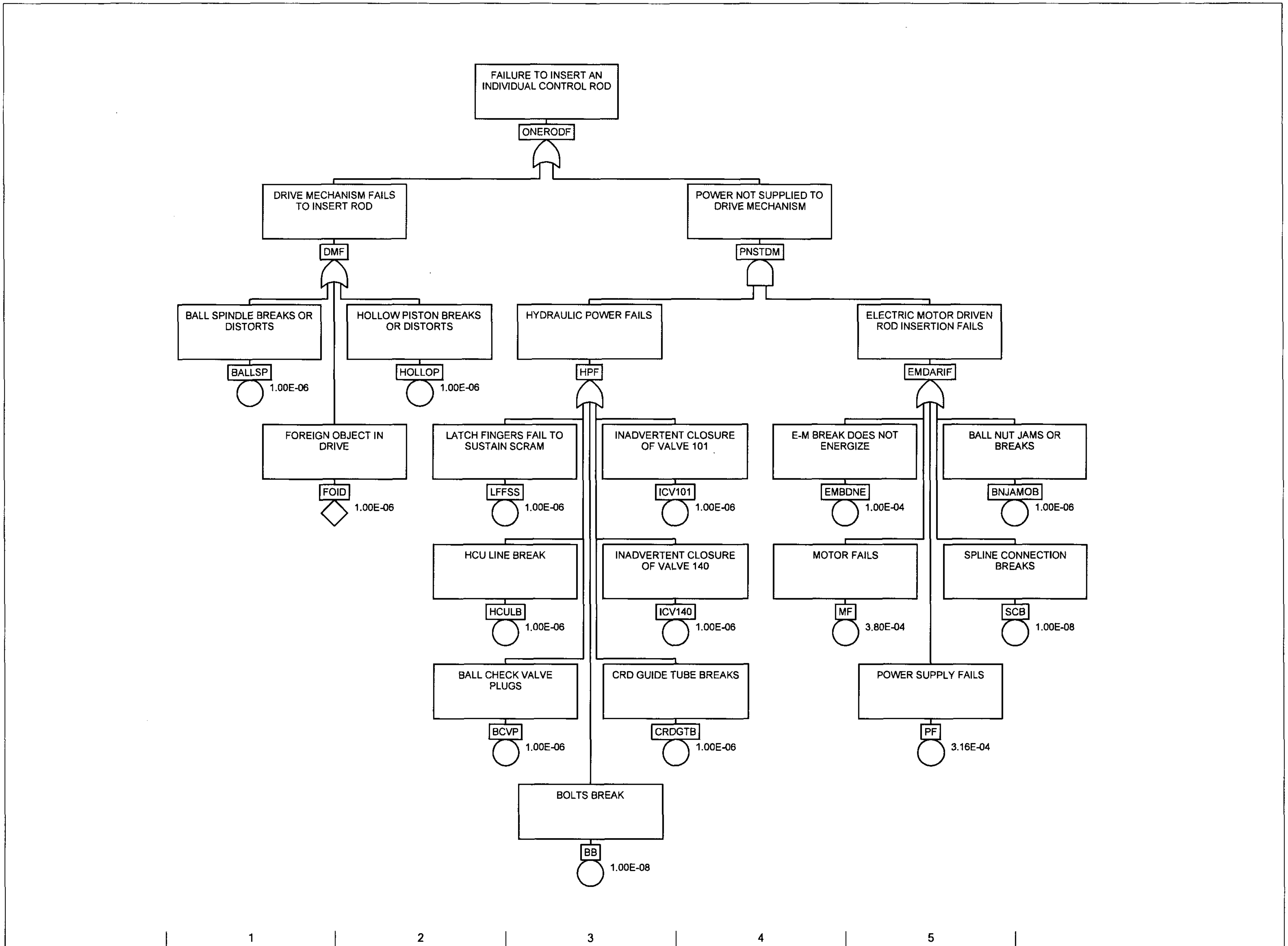


Figure 19D.6-17 Single Control Rod Drive Fault Tree

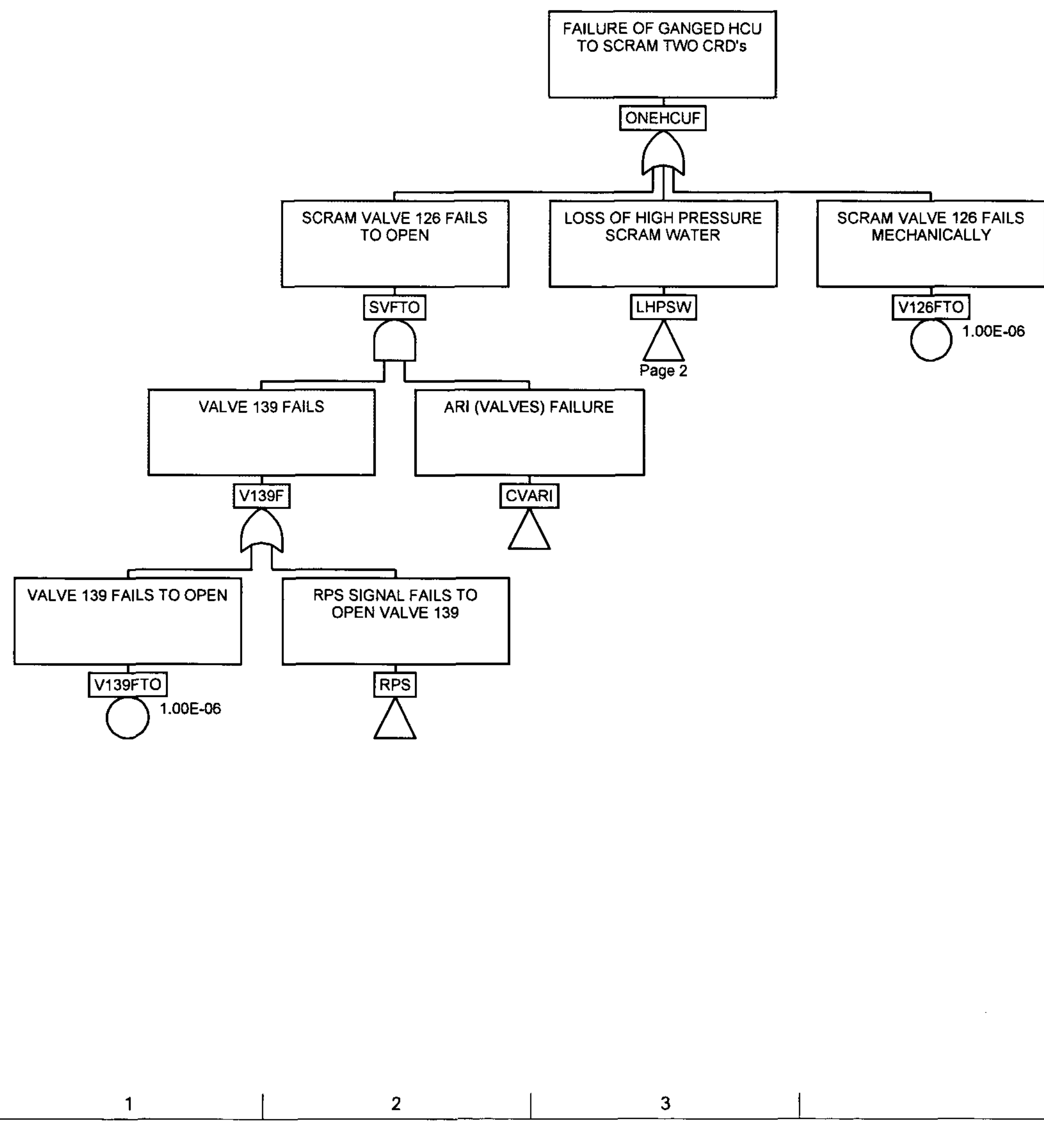


Figure 19D.6-18 Hydraulic Control Unit Fault Tree

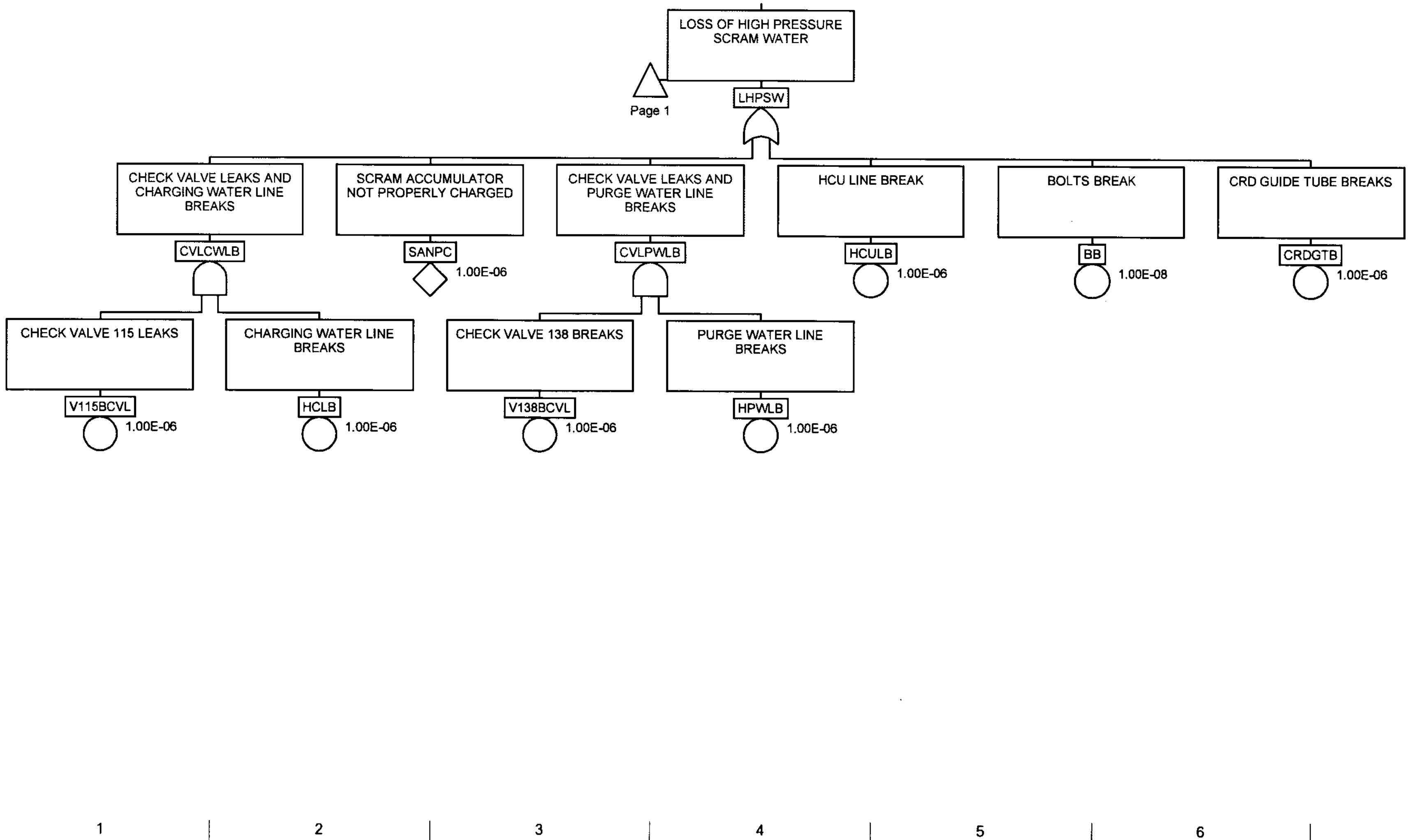


Figure 19D.6-18 Hydraulic Control Unit Fault Tree

Name	Page	Zone	Name	Page	Zone
BB	2	6			
CRDGTB	2	7			
CVARI	1	3			
CVLCWLB	2	2			
CVLPWLB	2	4			
HCLB	2	2			
HCULB	2	5			
HPWLB	2	4			
LHPSW	1	3			
LHPSW	2	4			
ONEHCUF	1	3			
RPS	1	2			
SANPC	2	3			
SVFTO	1	2			
V115BCVL	2	1			
V126FTO	1	4			
V138BCVL	2	3			
V139F	1	2			
V139FTO	1	1			

Figure 19D.6-18 Hydraulic Control Unit Fault Tree

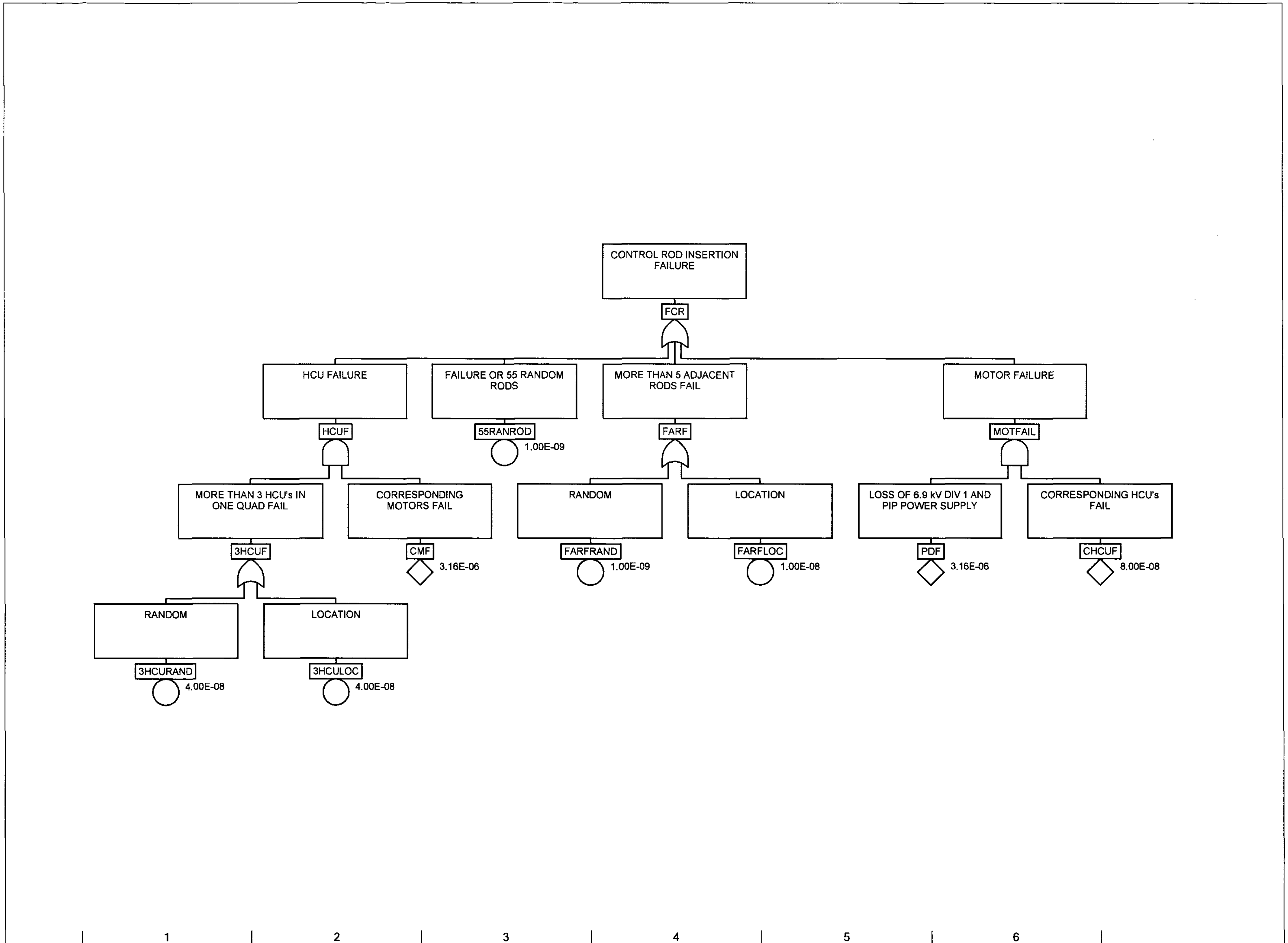


Figure 19D.6-19 Control Rod Drive System Fault Tree

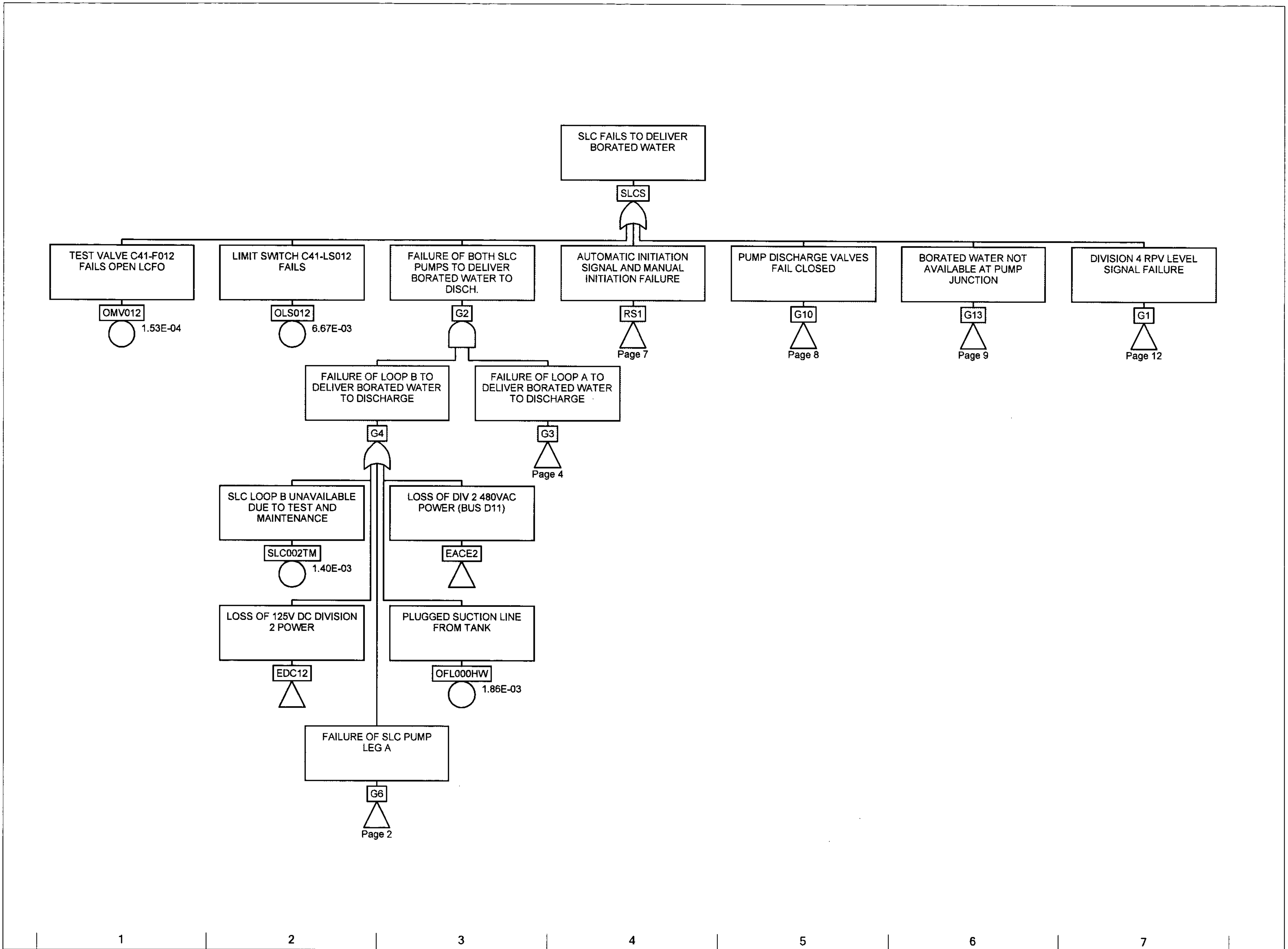


Figure 19D.6-20 Standby Liquid Control System Fault Tree

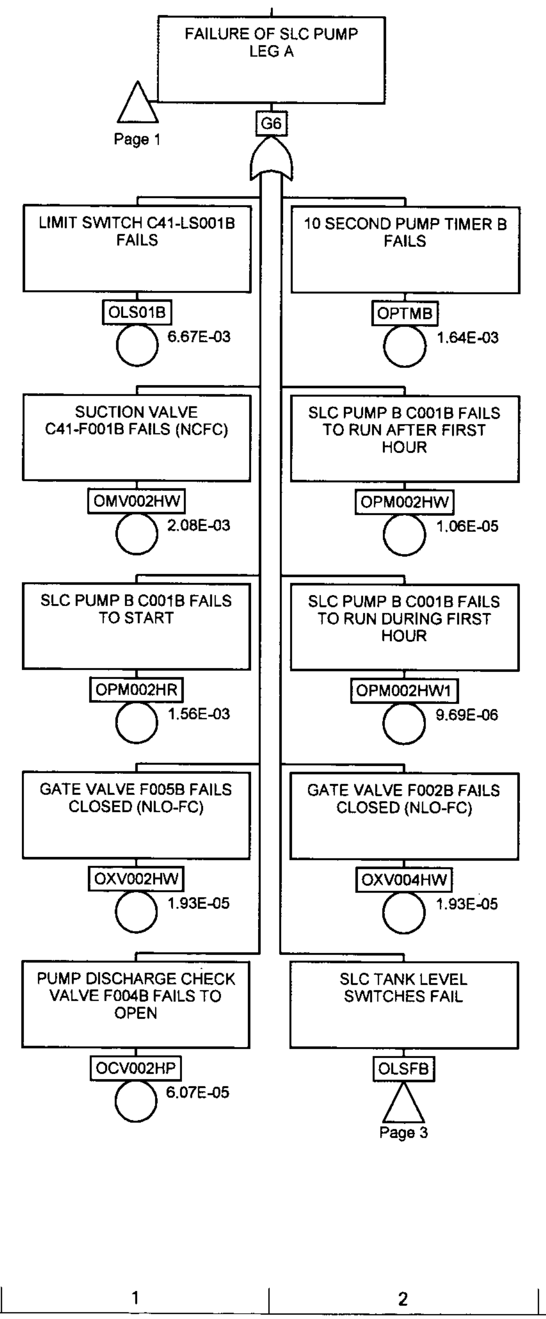


Figure 19D.6-20 Standby Liquid Control System Fault Tree

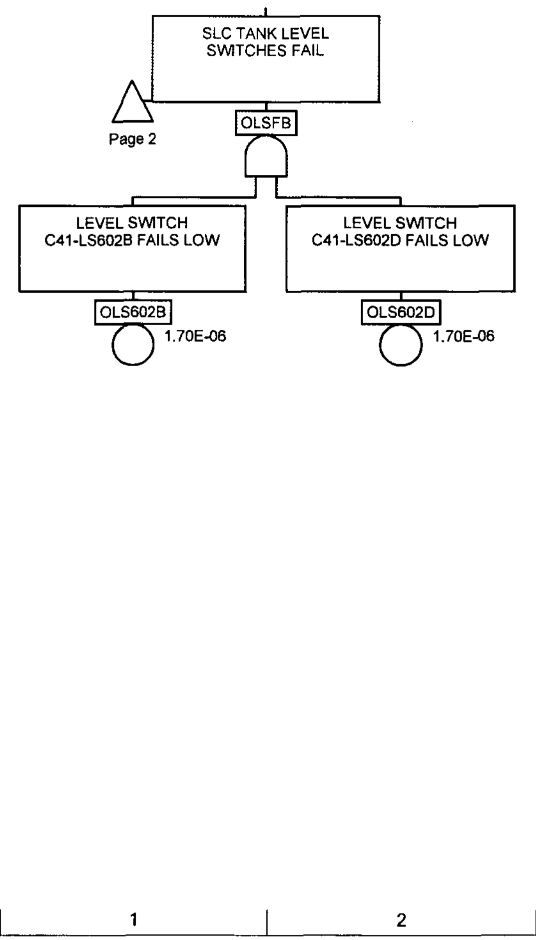


Figure 19D.6-20 Standby Liquid Control System Fault Tree

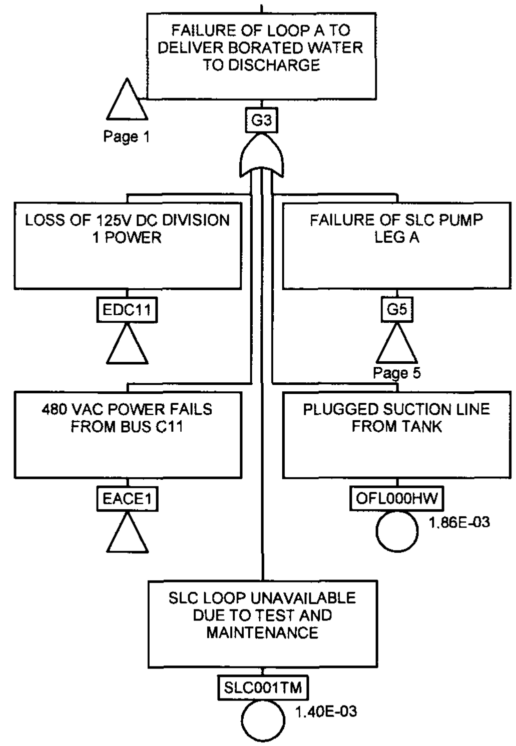


Figure 19D.6-20 Standby Liquid Control System Fault Tree

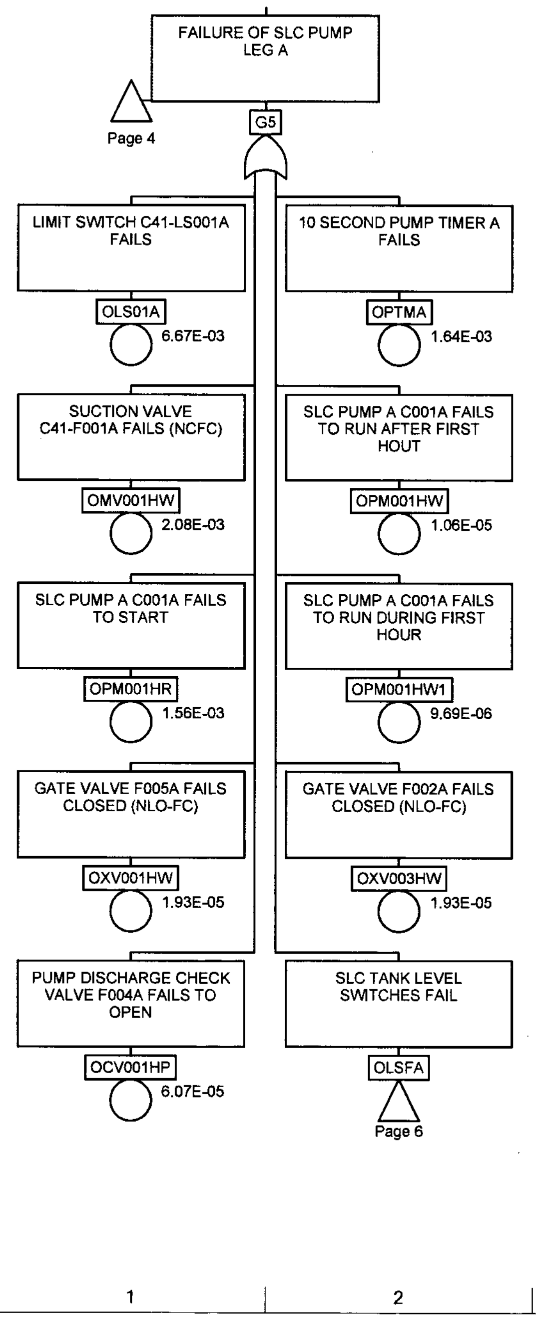


Figure 19D.6-20 Standby Liquid Control System Fault Tree

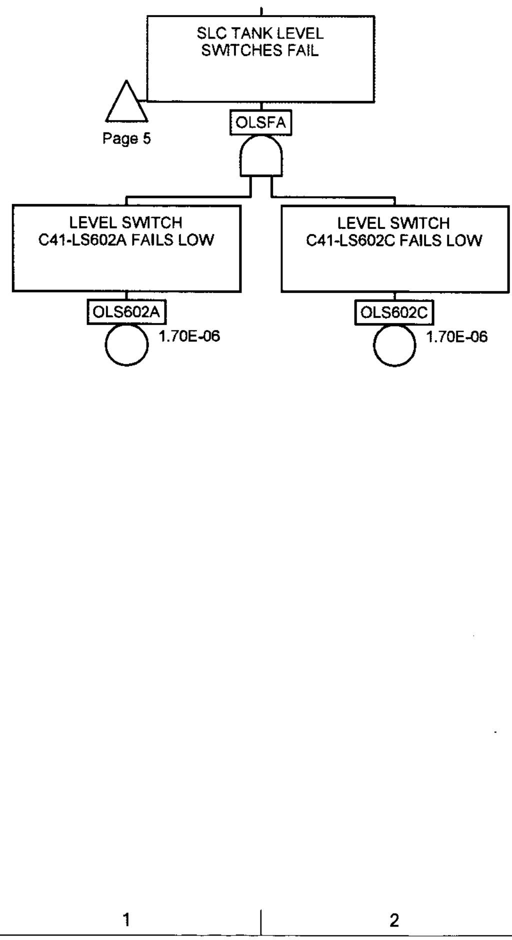


Figure 19D.6-20 Standby Liquid Control System Fault Tree

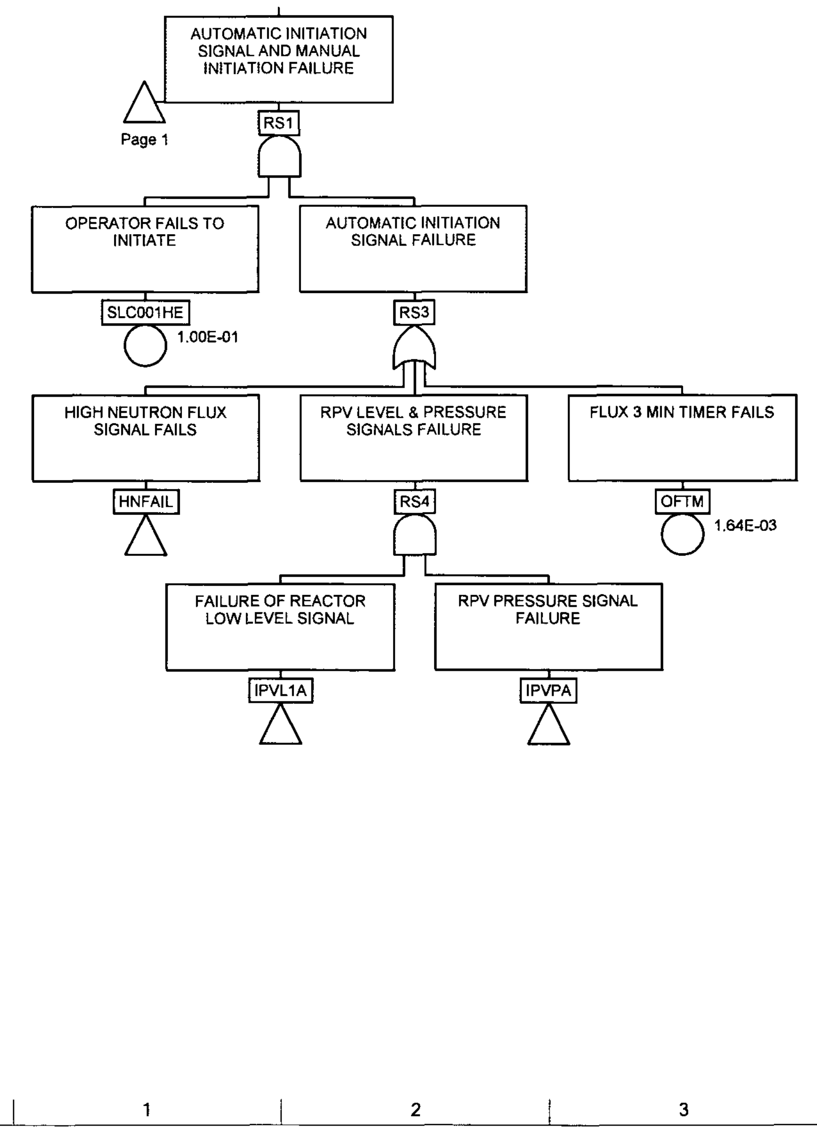


Figure 19D.6-20 Standby Liquid Control System Fault Tree

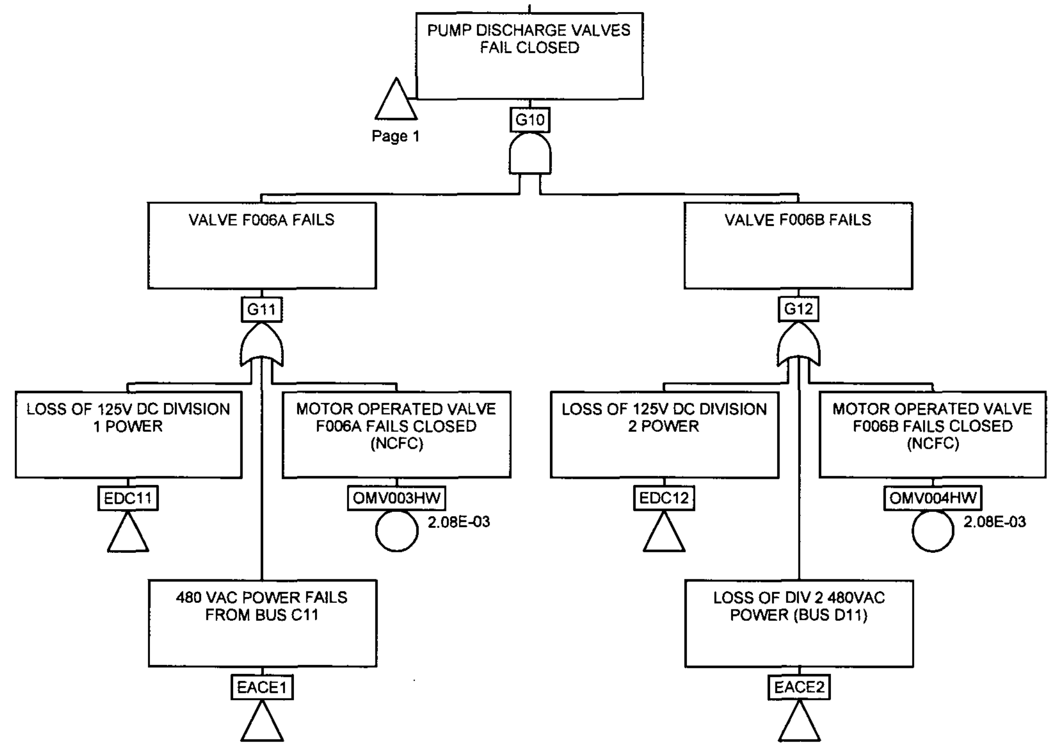


Figure 19D.6-20 Standby Liquid Control System Fault Tree

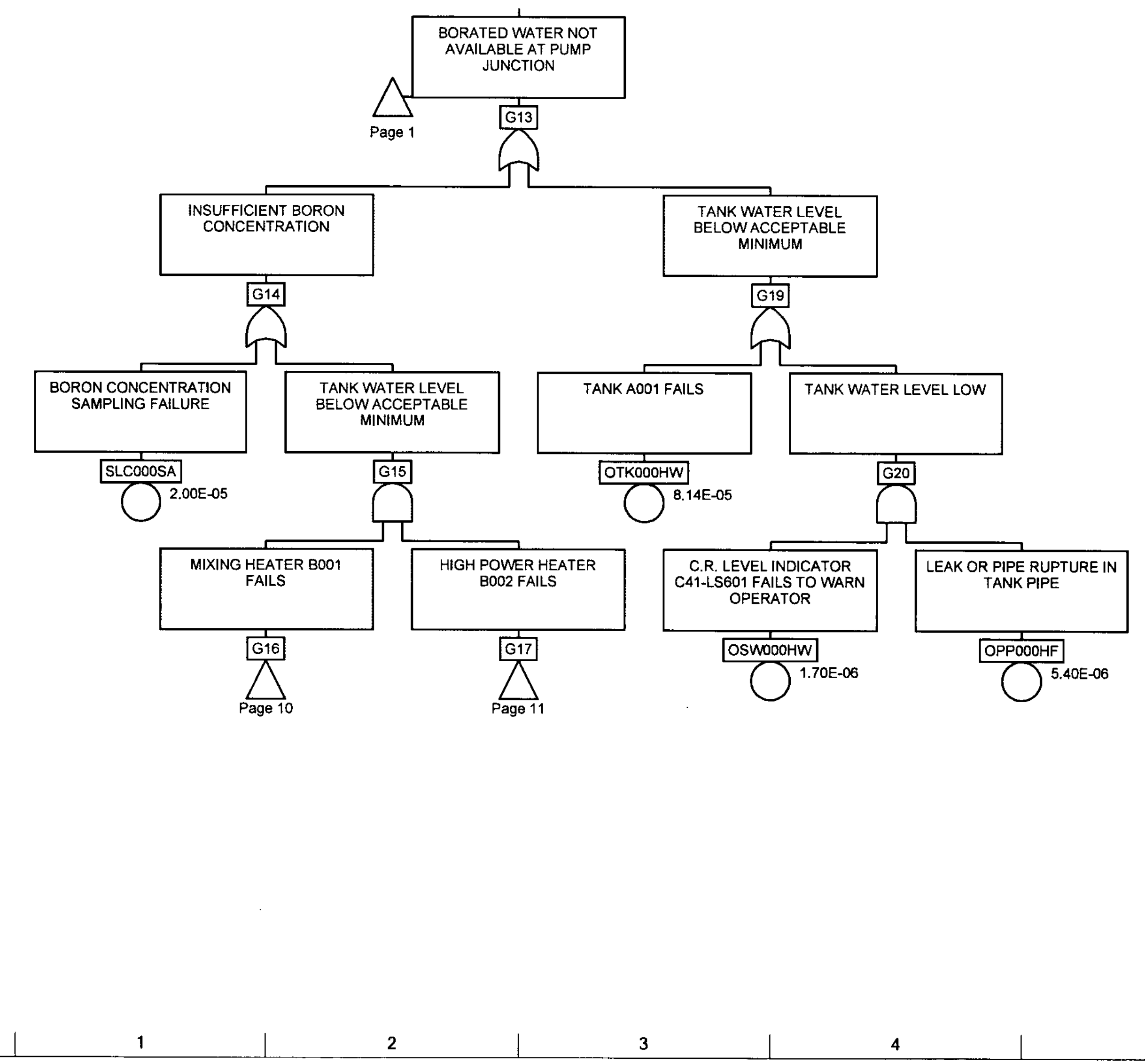


Figure 19D.6-20 Standby Liquid Control System Fault Tree

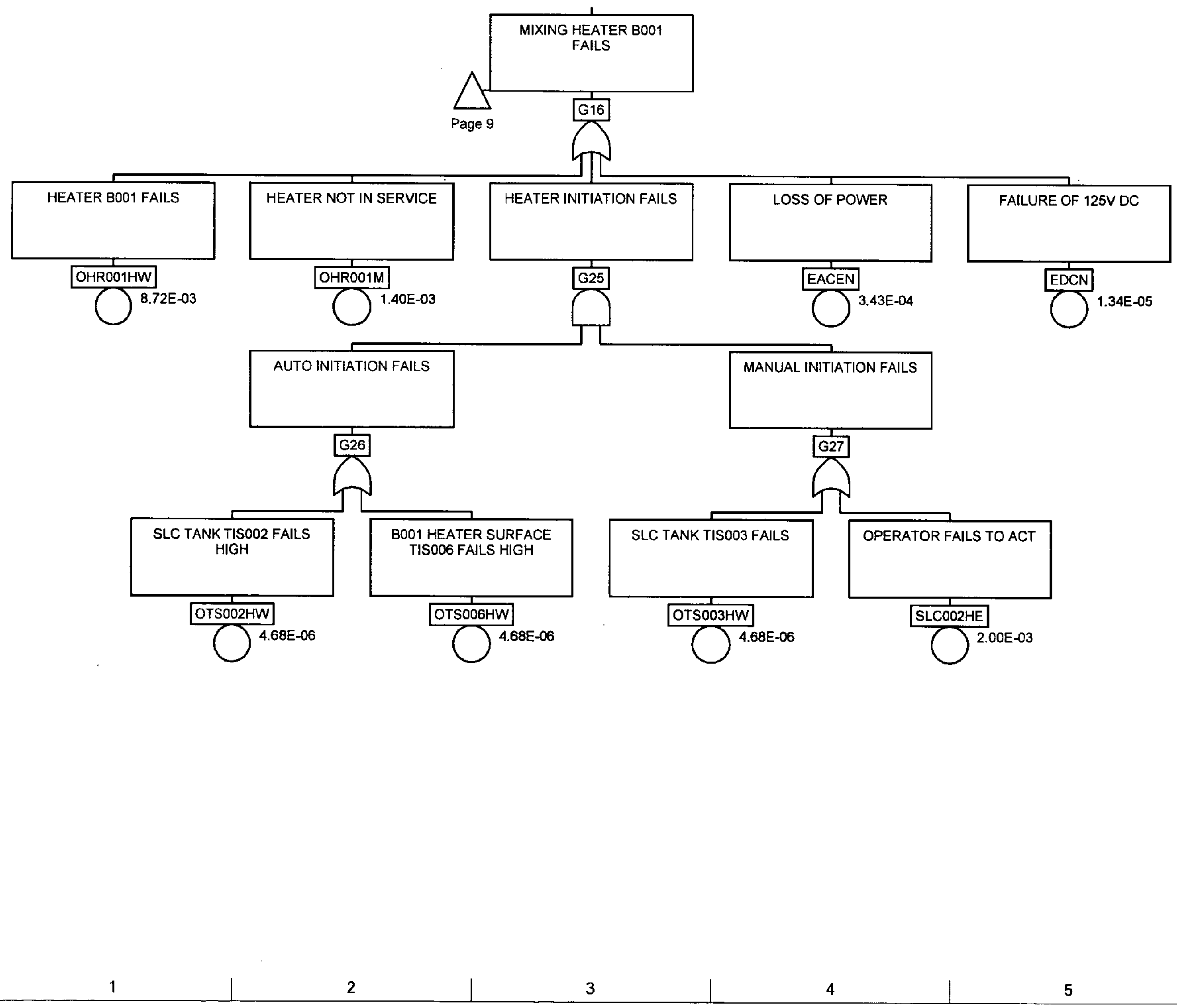


Figure 19D.6-20 Standby Liquid Control System Fault Tree

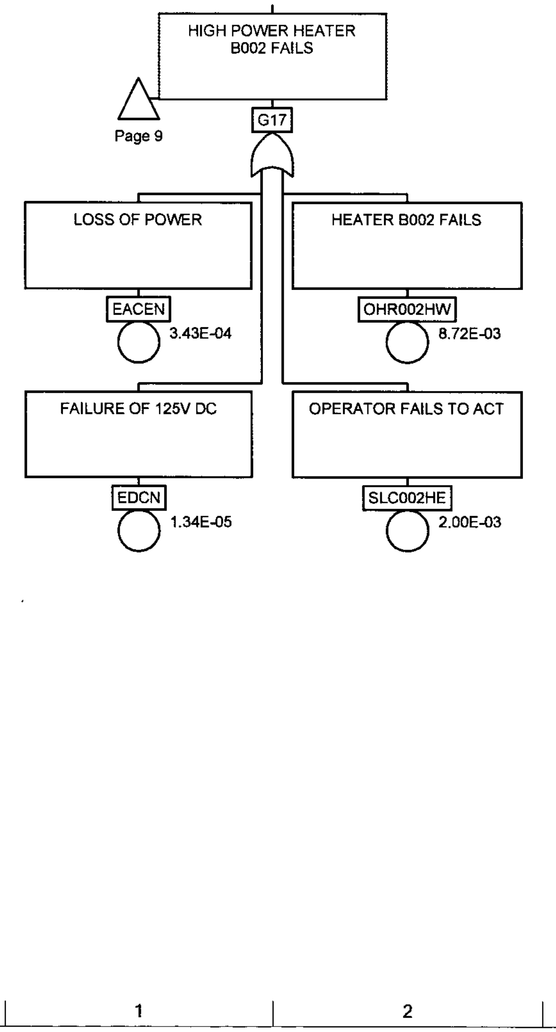


Figure 19D.6-20 Standby Liquid Control System Fault Tree

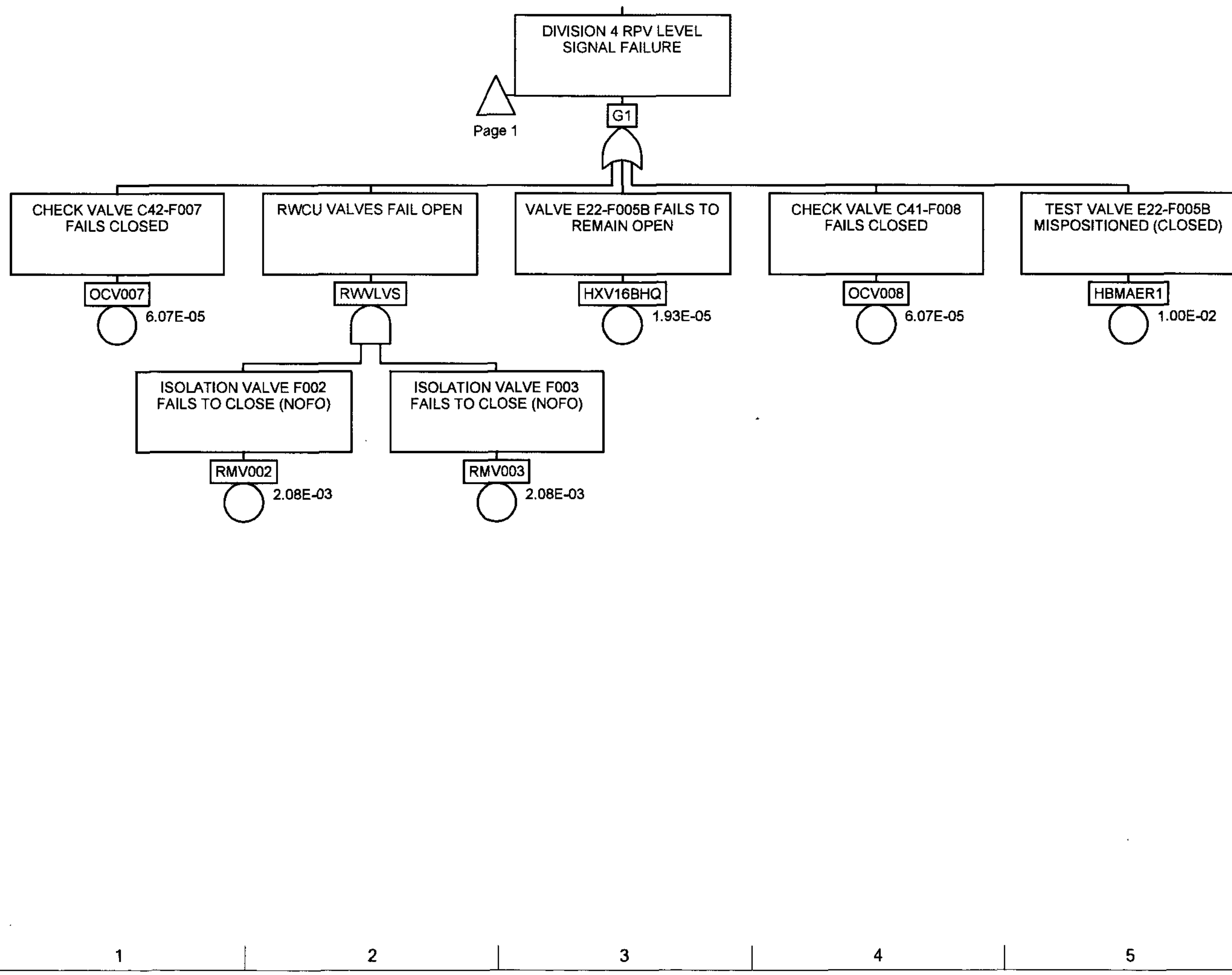


Figure 19D.6-20 Standby Liquid Control System Fault Tree

Name	Page	Zone	Name	Page	Zone
EACE1	4	1	OFL000HW	1	3
EACE1	8	2	OFL000HW	4	2
EACE2	1	3	OFTM	7	3
EACE2	8	4	OHR001HW	10	1
EACEN	10	4	OHR001M	10	2
EACEN	11	1	OHR002HW	11	2
EDC11	4	1	OLS012	1	2
EDC11	8	1	OLS01A	5	1
EDC12	1	2	OLS01B	2	1
EDC12	8	3	OLS602A	6	1
EDCN	10	5	OLS602B	3	1
EDCN	11	1	OLS602C	6	2
G1	1	7	OLS602D	3	2
G1	12	3	OLSFA	5	2
G10	1	5	OLSFA	6	2
G10	8	2	OLSFB	2	2
G11	8	2	OLSFB	3	2
G12	8	4	OMV001HW	5	1
G13	1	6	OMV002HW	2	1
G13	9	2	OMV003HW	8	2
G14	9	2	OMV004HW	8	4
G15	9	2	OMV012	1	1
G16	9	2	OPM001HR	5	1
G16	10	3	OPM001HW	5	2
G17	9	3	OPM001HW1	5	2
G17	11	2	OPM002HR	2	1
G19	9	4	OPM002HW	2	2
G2	1	3	OPM002HW1	2	2
G20	9	4	OPP000HF	9	5
G25	10	3	OPTMA	5	2
G26	10	2	OPTMB	2	2
G27	10	4	OSW000HW	9	4
G3	1	4	OTK000HW	9	3
G3	4	2	OTS002HW	10	2
G4	1	3	OTS003HW	10	4
G5	4	2	OTS006HW	10	3
G5	5	2	OXV001HW	5	1
G6	1	3	OXV002HW	2	1
G6	2	2	OXV003HW	5	2
HBMAER1	12	5	OXV004HW	2	2
HNFAIL	7	1	RMV002	12	2
HXV16BHQ	12	3	RMV003	12	3
IPVL1A	7	2	RS1	1	4
IPVPA	7	3	RS1	7	2
OCV001HP	5	1	RS3	7	2
OCV002HP	2	1	RS4	7	2
OCV007	12	1	RWVLVS	12	2
OCV008	12	4	SLC000SA	9	1

Figure 19D.6-20 Standby Liquid Control System Fault Tree

Name	Page	Zone	Name	Page	Zone
SLC001HE	7	1			
SLC001TM	4	2			
SLC002HE	10	5			
SLC002HE	11	2			
SLC002TM	1	2			
SLCS	1	4			

Figure 19D.6-20 Standby Liquid Control System Fault Tree

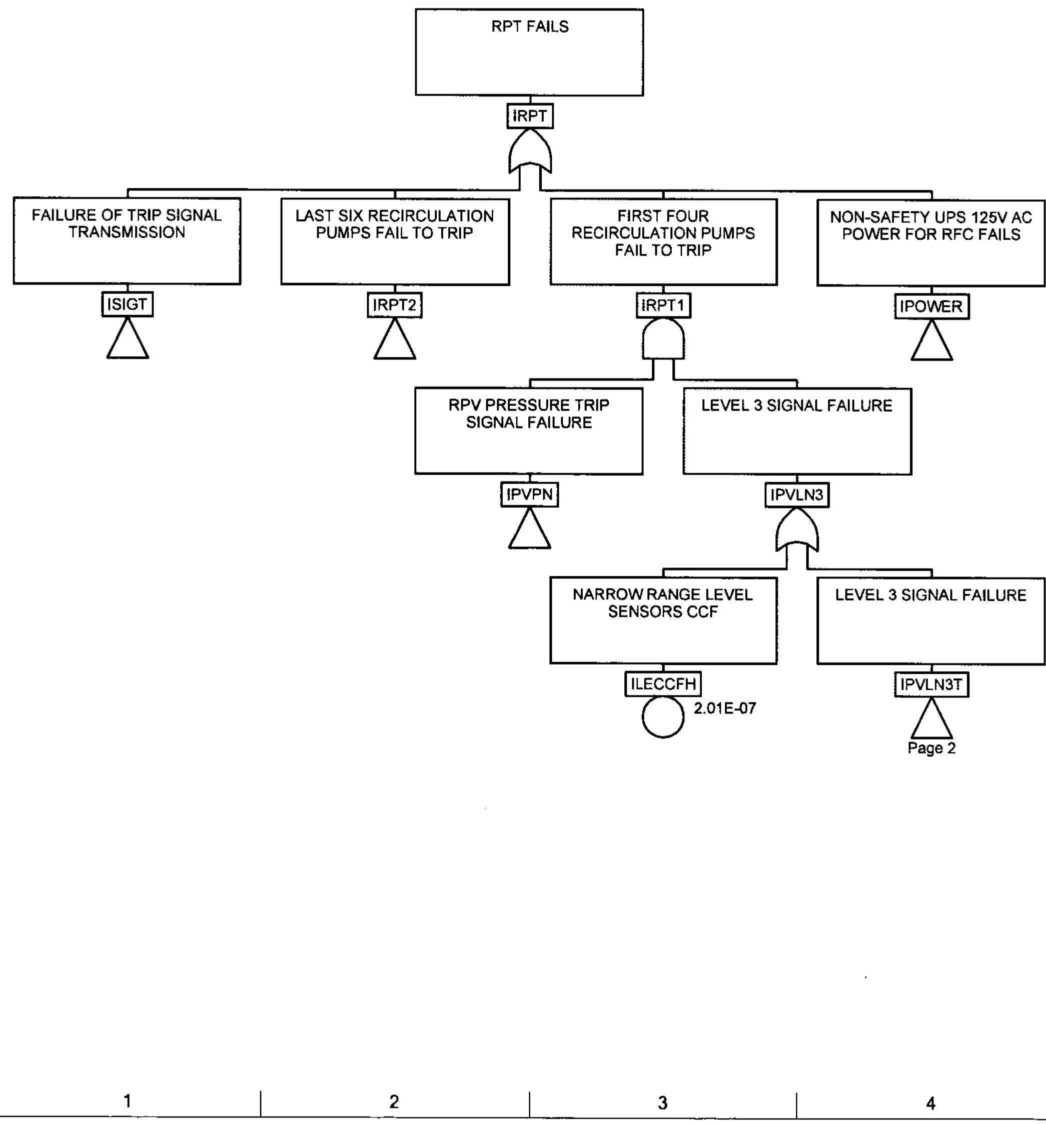


Figure 19D.6-21 Recirculation Pump Trip Fault Tree

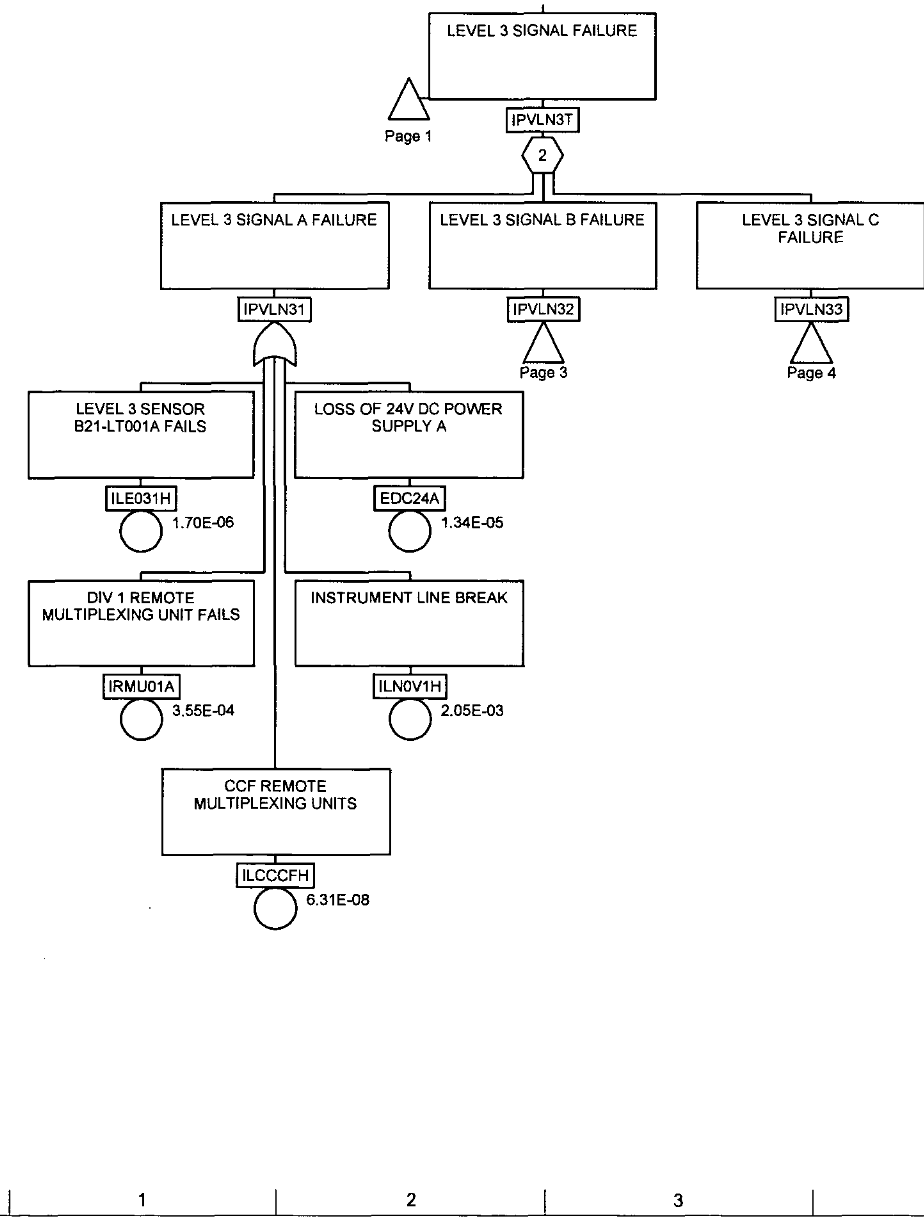


Figure 19D.6-21 Recirculation Pump Trip Fault Tree

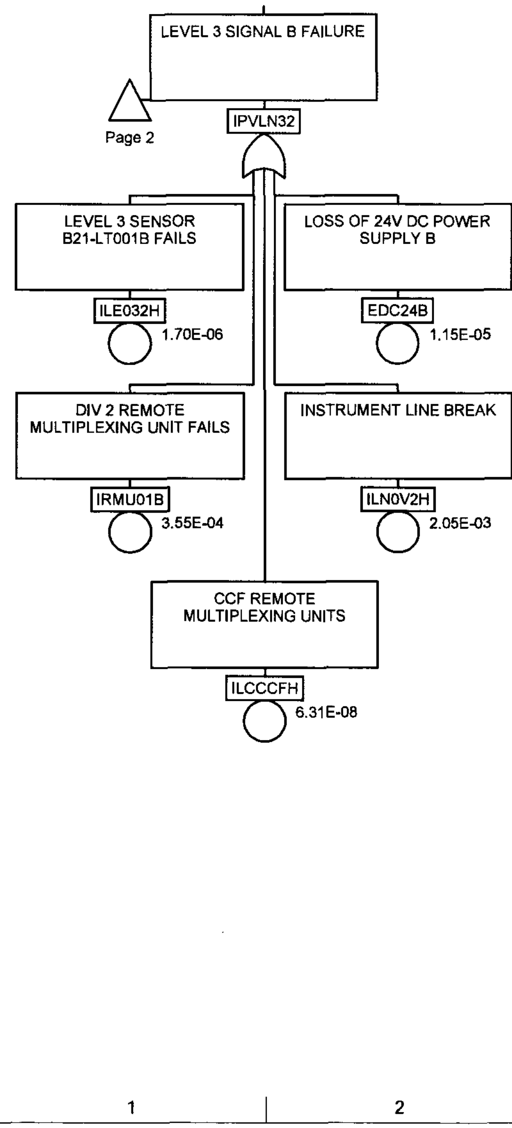


Figure 19D.6-21 Recirculation Pump Trip Fault Tree

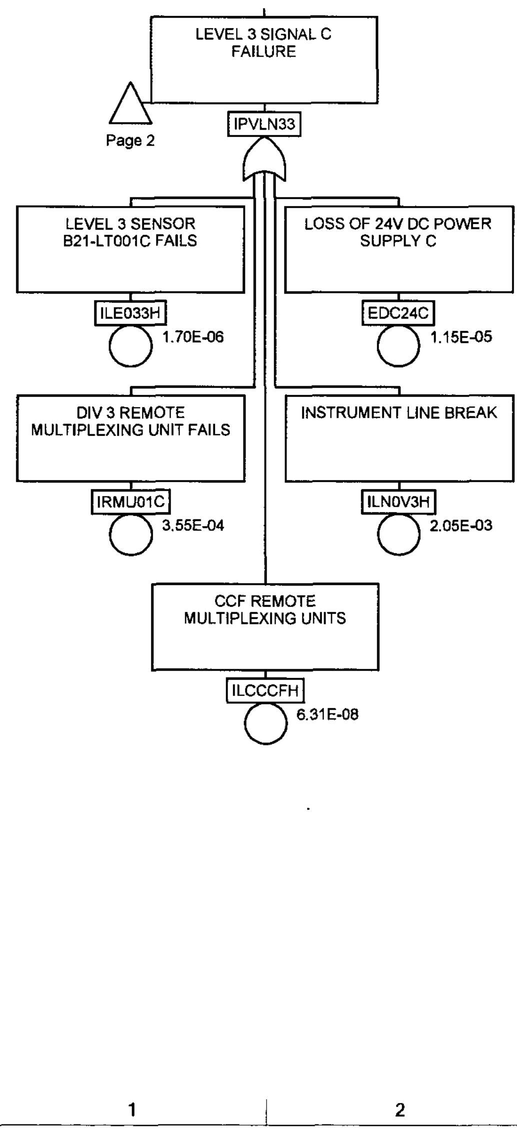


Figure 19D.6-21 Recirculation Pump Trip Fault Tree

Name	Page	Zone	Name	Page	Zone
EDC24A	2	2			
EDC24B	3	2			
EDC24C	4	2			
ILCCCFH	2	2			
ILCCCFH	3	2			
ILCCCFH	4	2			
ILE031H	2	1			
ILE032H	3	1			
ILE033H	4	1			
ILECCFH	1	3			
ILN0V1H	2	2			
ILN0V2H	3	2			
ILN0V3H	4	2			
IPOWER	1	4			
IPVLN3	1	4			
IPVLN31	2	2			
IPVLN32	2	3			
IPVLN32	3	2			
IPVLN33	2	4			
IPVLN33	4	2			
IPVLN3T	1	4			
IPVLN3T	2	2			
IPVPN	1	3			
IRMU01A	2	1			
IRMU01B	3	1			
IRMU01C	4	1			
IRPT	1	3			
IRPT1	1	3			
IRPT2	1	2			
ISIGT	1	1			

Figure 19D.6-21 Recirculation Pump Trip Fault Tree

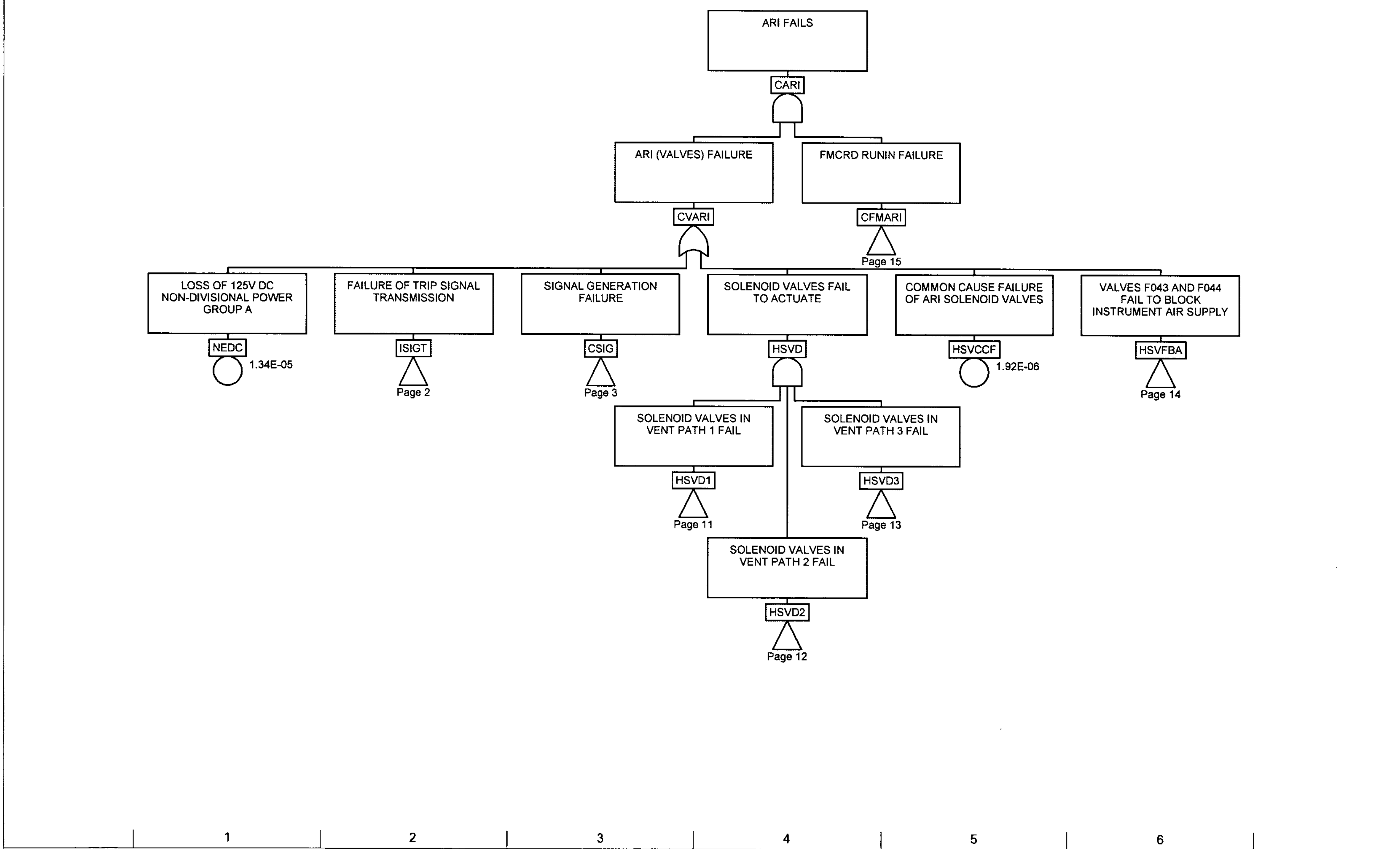


Figure 19D.6-22 Alternate Rod Insertion Fault Tree

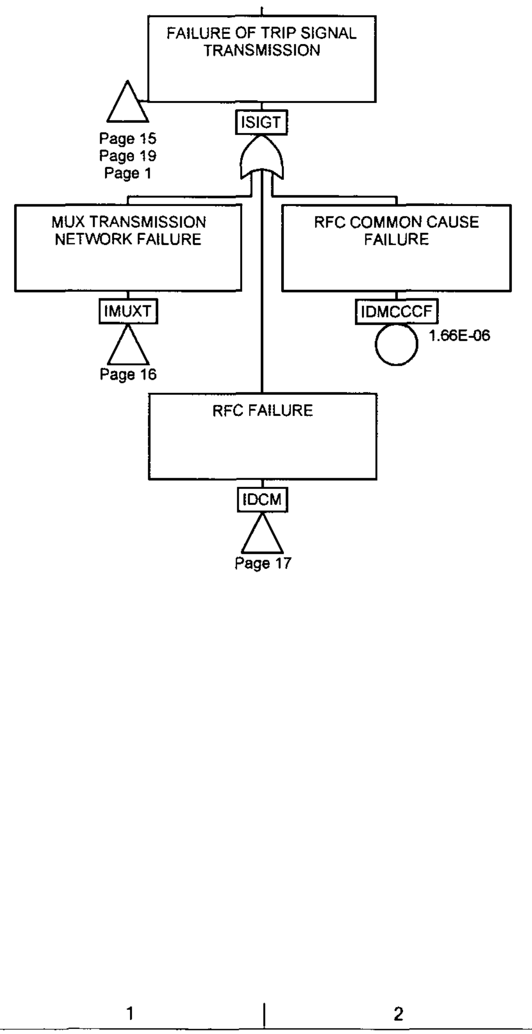


Figure 19D.6-22 Alternate Rod Insertion Fault Tree

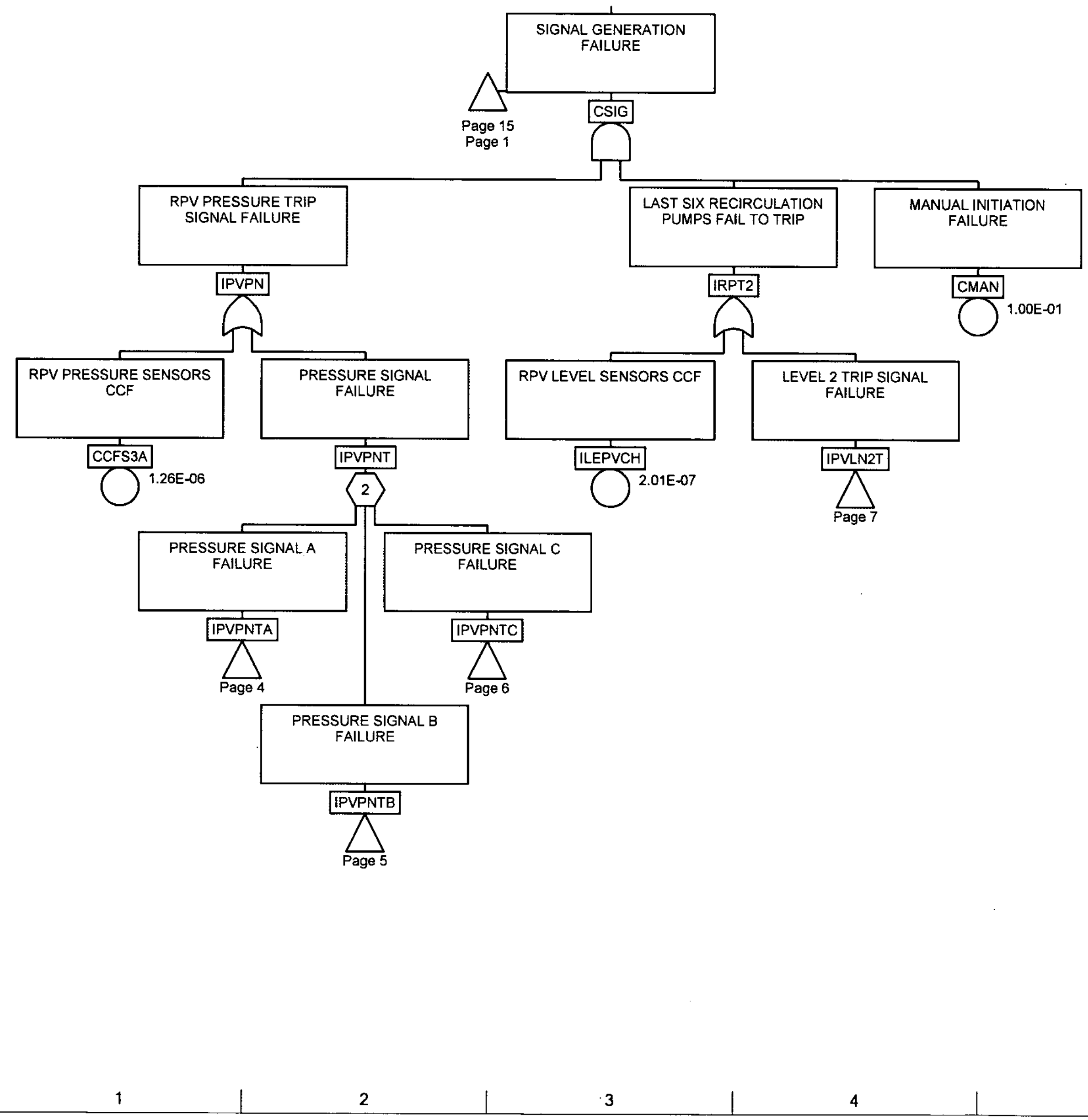


Figure 19D.6-22 Alternate Rod Insertion Fault Tree

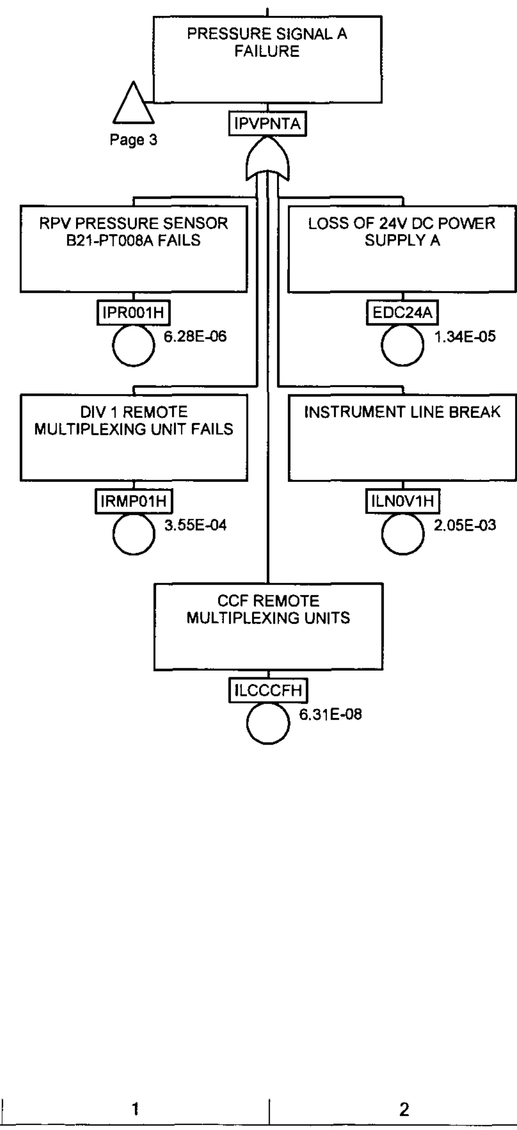


Figure 19D.6-22 Alternate Rod Insertion Fault Tree

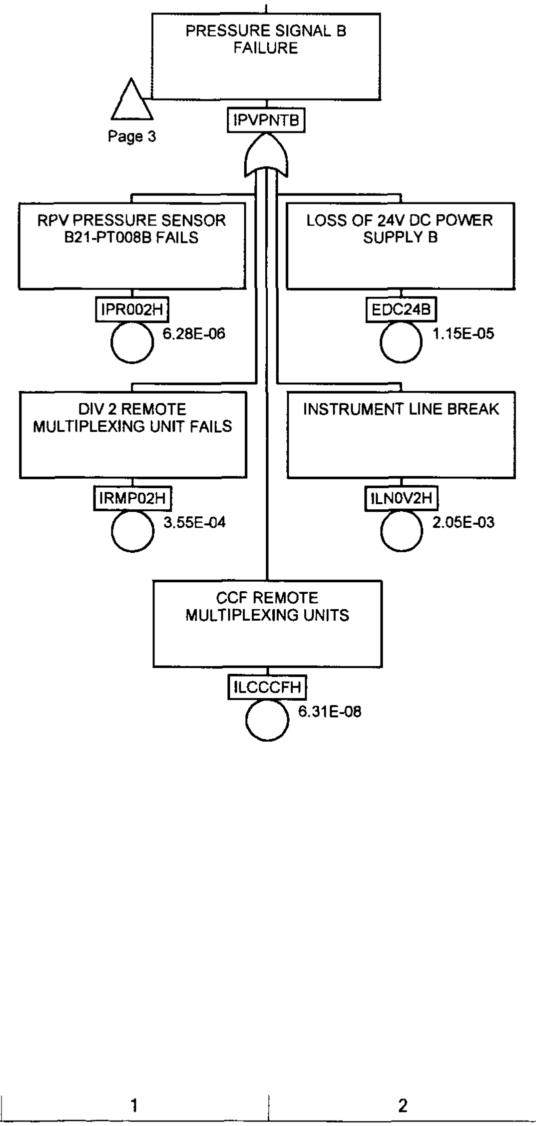


Figure 19D.6-22 Alternate Rod Insertion Fault Tree

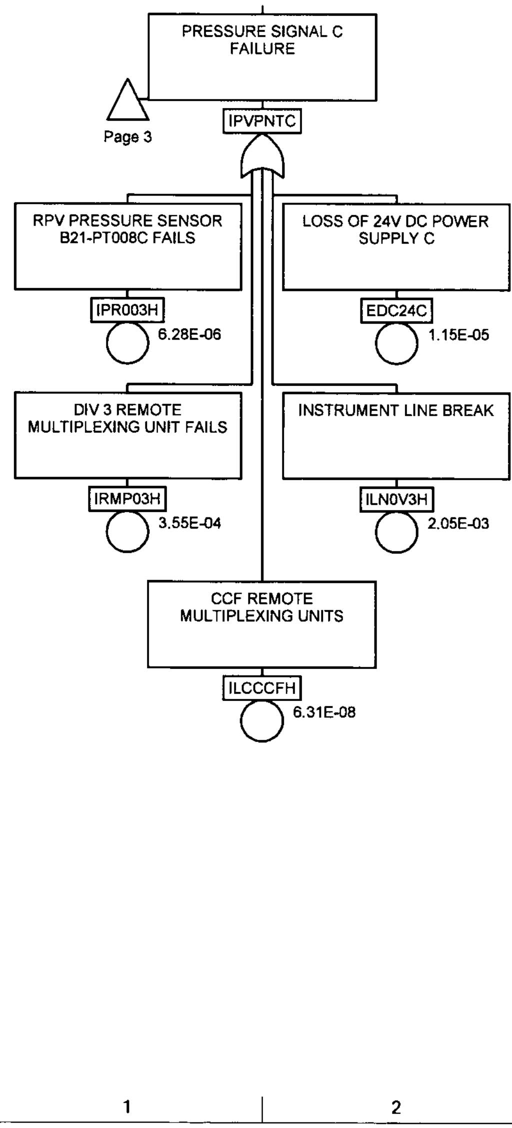


Figure 19D.6-22 Alternate Rod Insertion Fault Tree

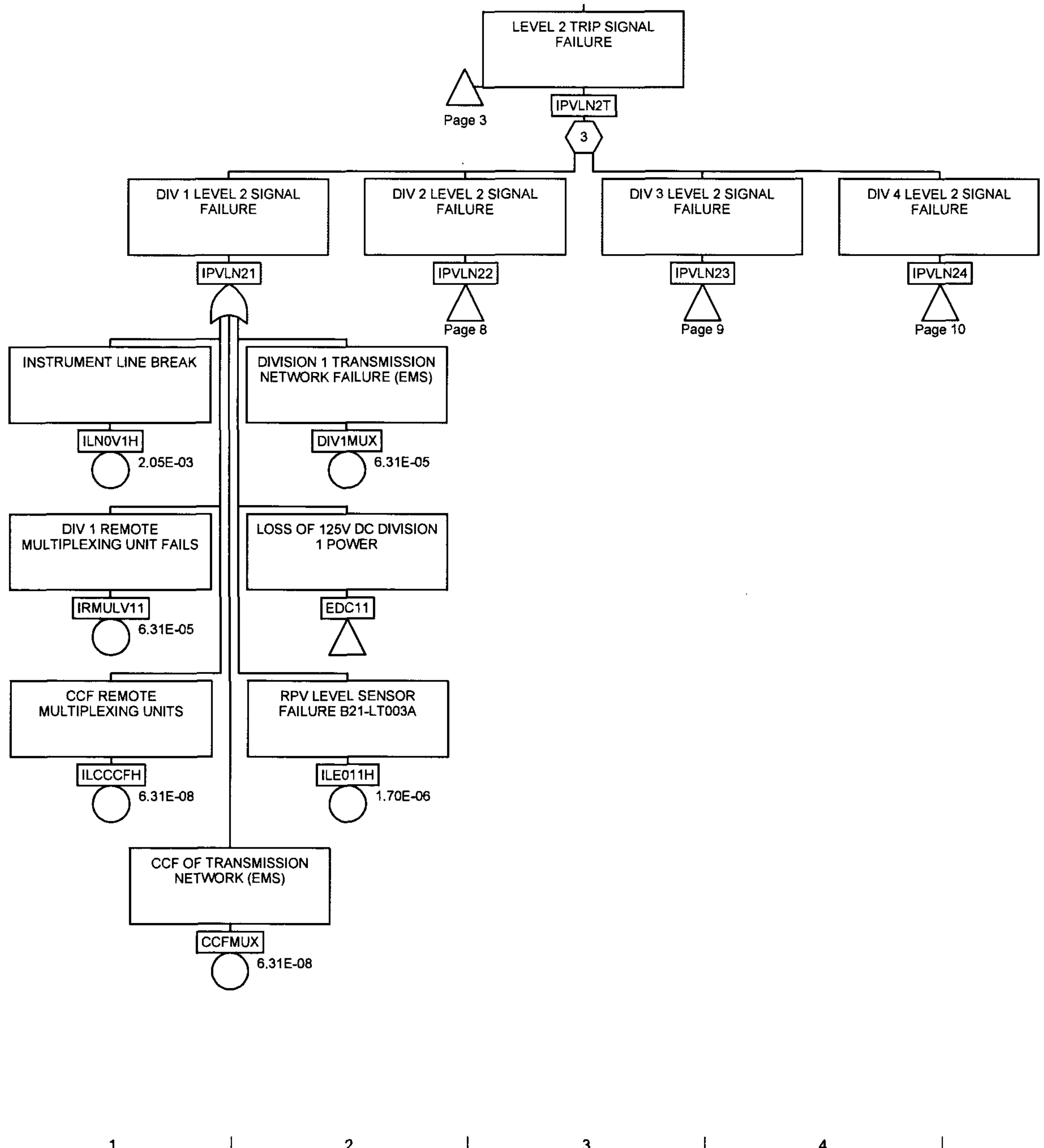


Figure 19D.6-22 Alternate Rod Insertion Fault Tree

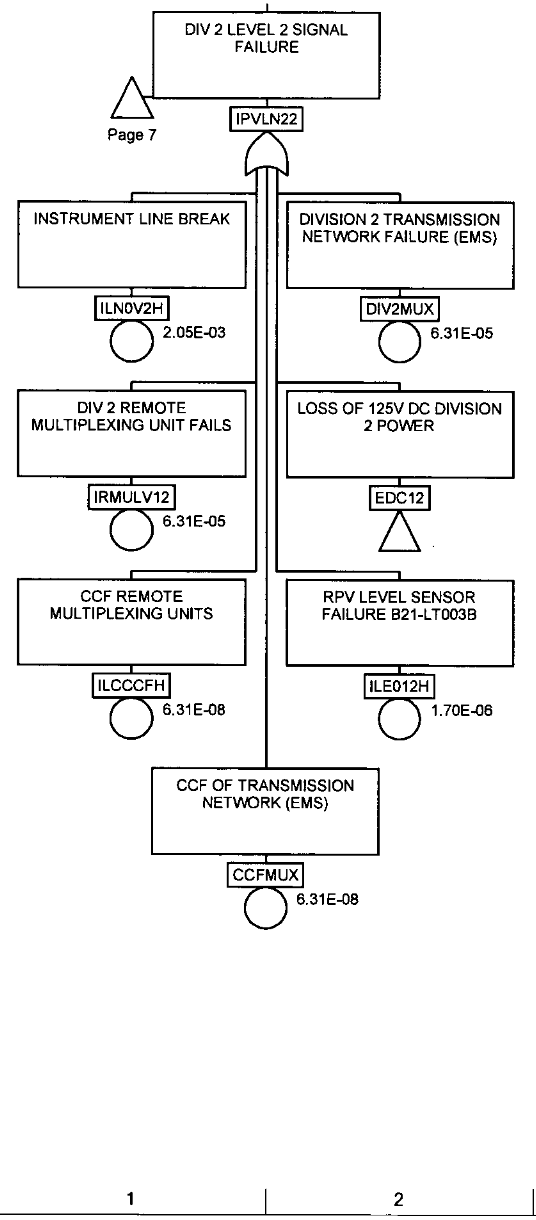


Figure 19D.6-22 Alternate Rod Insertion Fault Tree

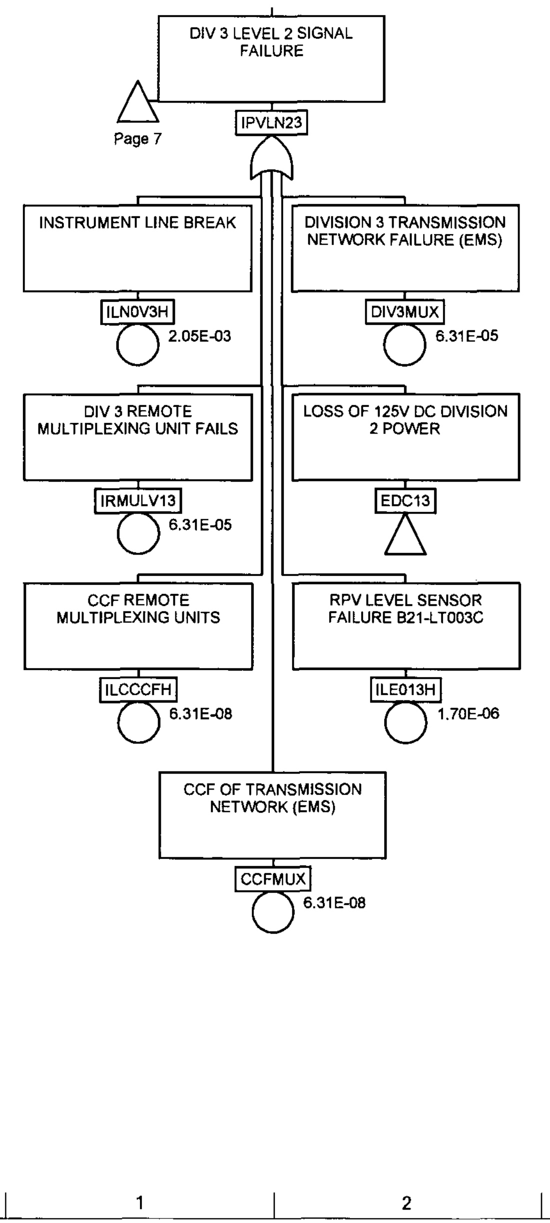


Figure 19D.6-22 Alternate Rod Insertion Fault Tree

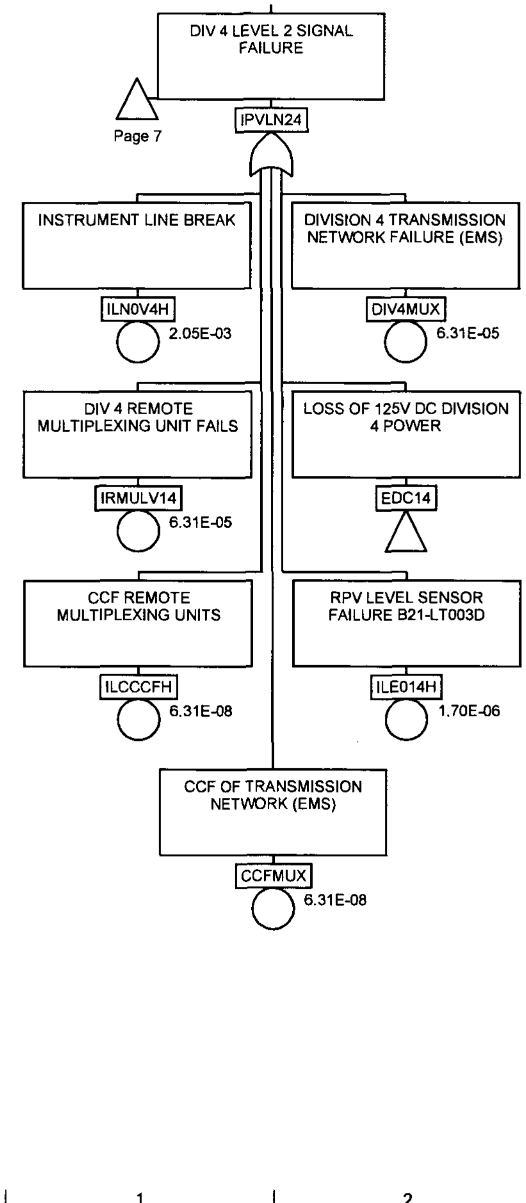


Figure 19D.6-22 Alternate Rod Insertion Fault Tree

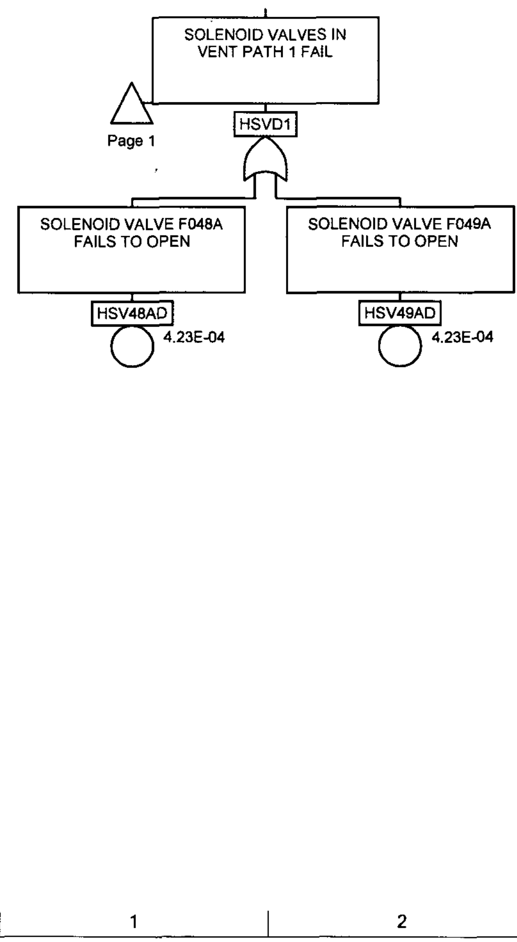


Figure 19D.6-22 Alternate Rod Insertion Fault Tree

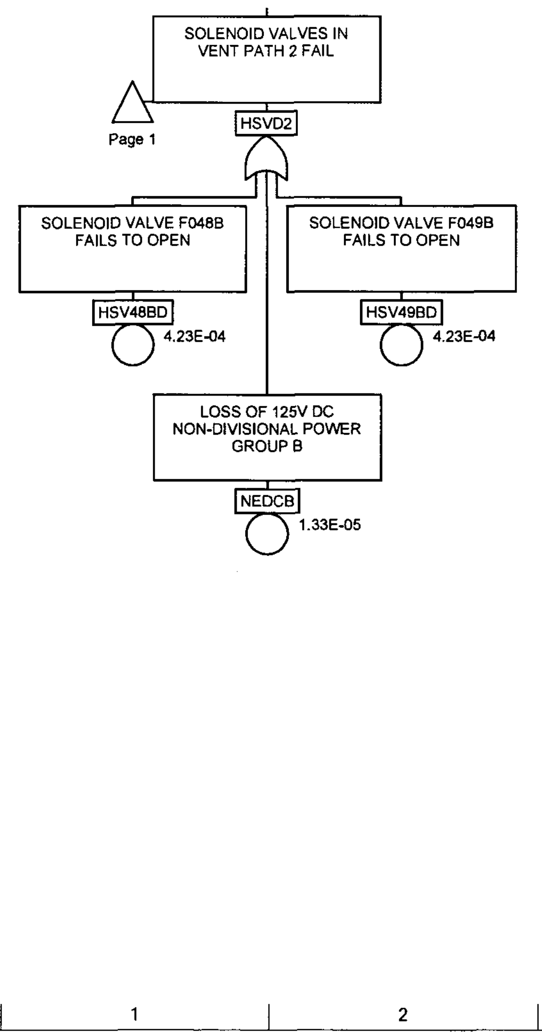


Figure 19D.6-22 Alternate Rod Insertion Fault Tree

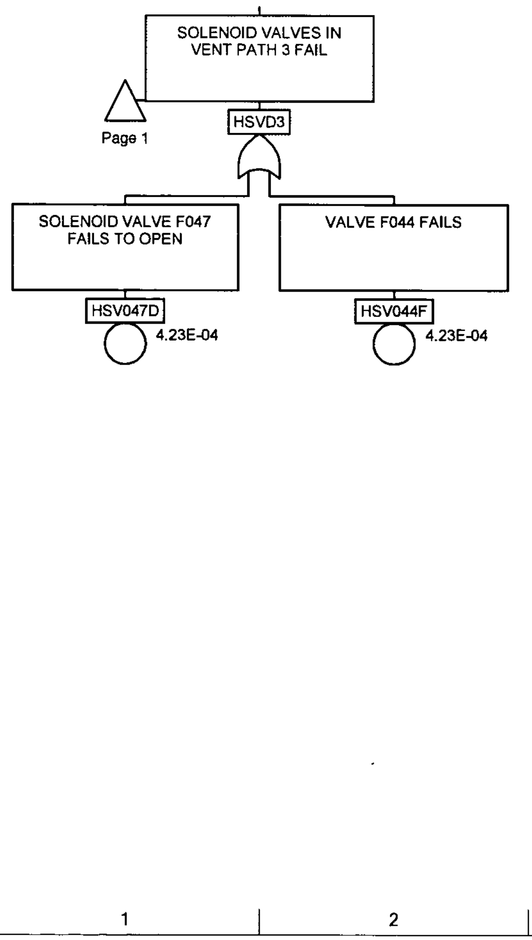


Figure 19D.6-22 Alternate Rod Insertion Fault Tree

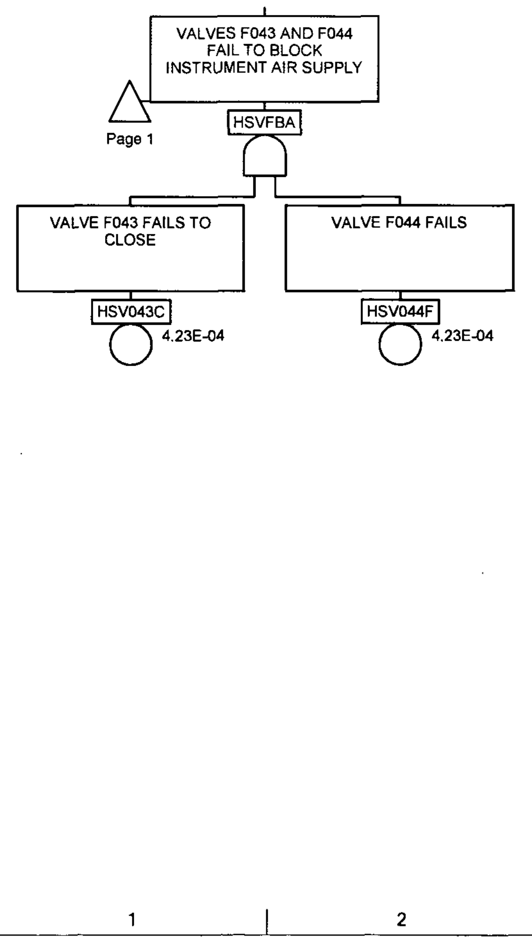


Figure 19D.6-22 Alternate Rod Insertion Fault Tree

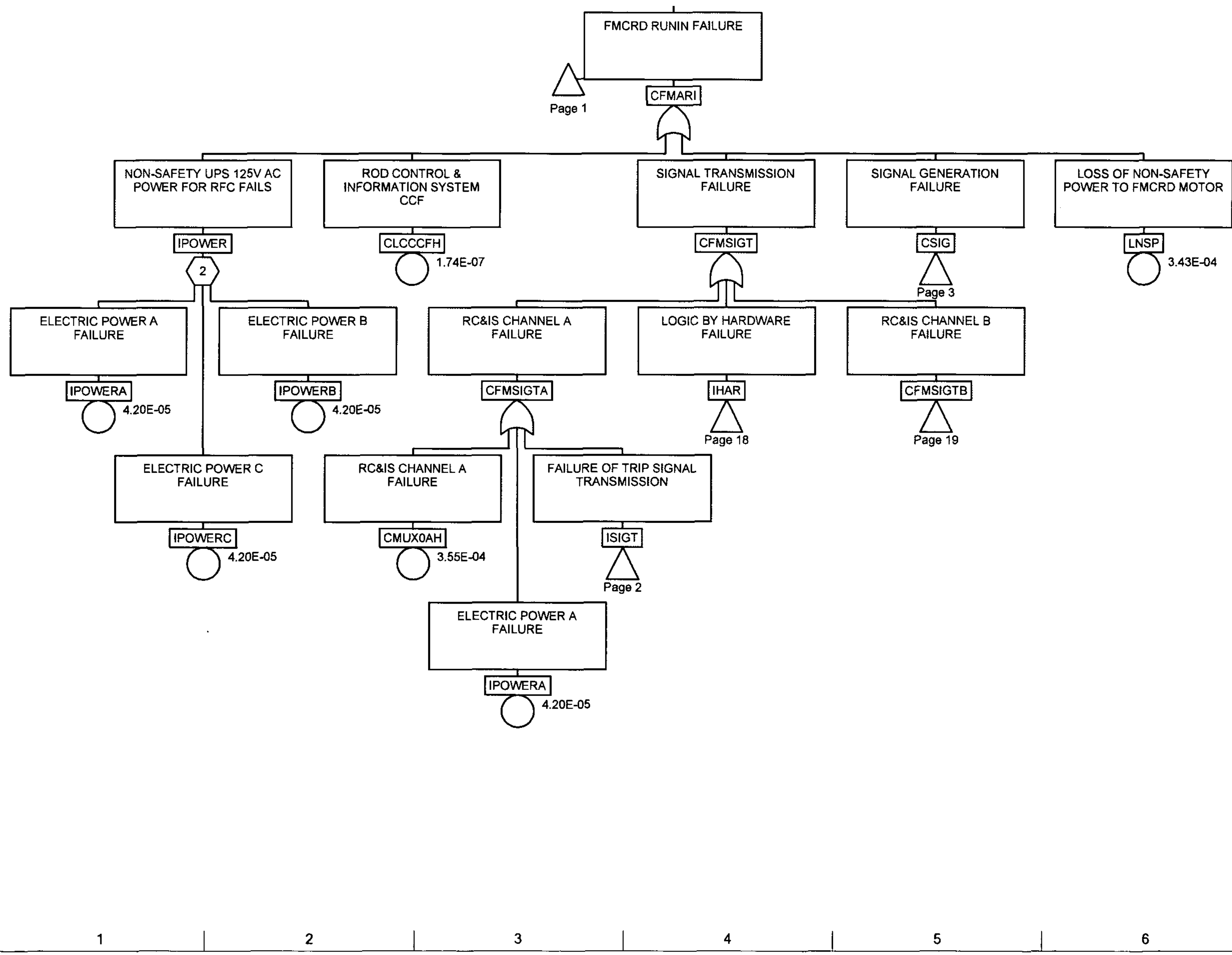


Figure 19D.6-22 Alternate Rod Insertion Fault Tree

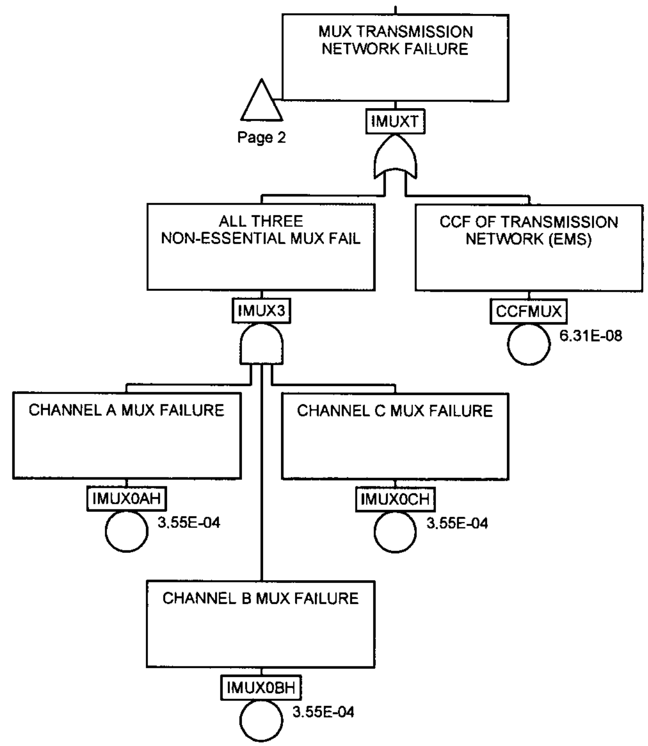


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Figure 19D.6-22 Alternate Rod Insertion Fault Tree

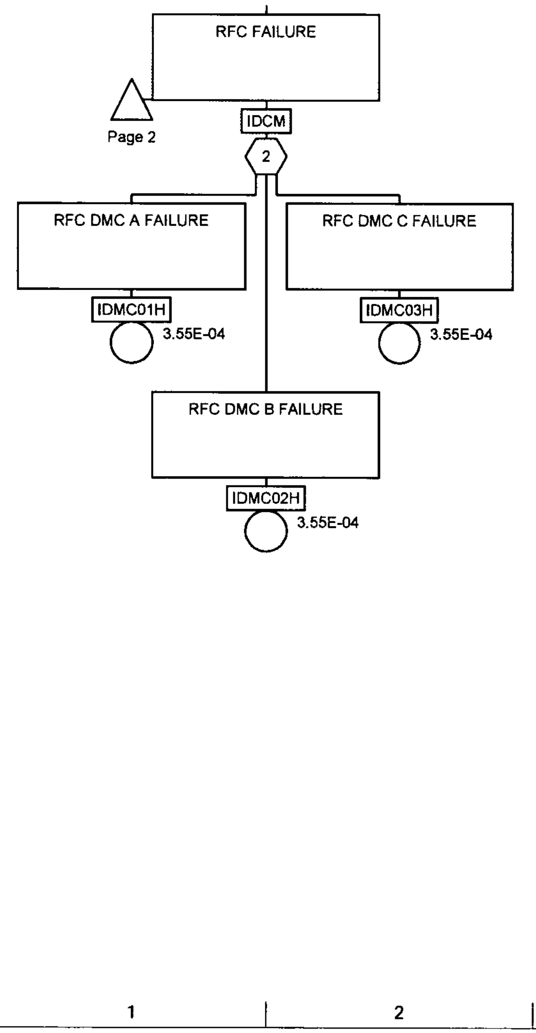
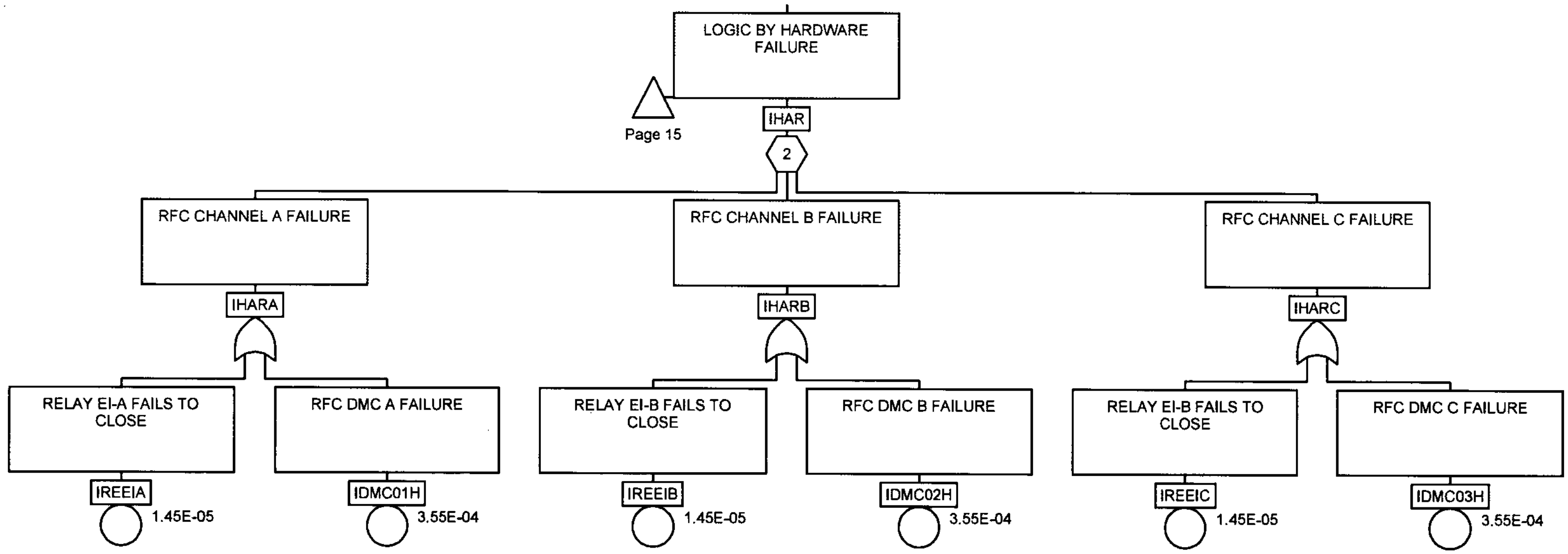


Figure 19D.6-22 Alternate Rod Insertion Fault Tree



1	2	3	4	5	6

Figure 19D.6-22 Alternate Rod Insertion Fault Tree

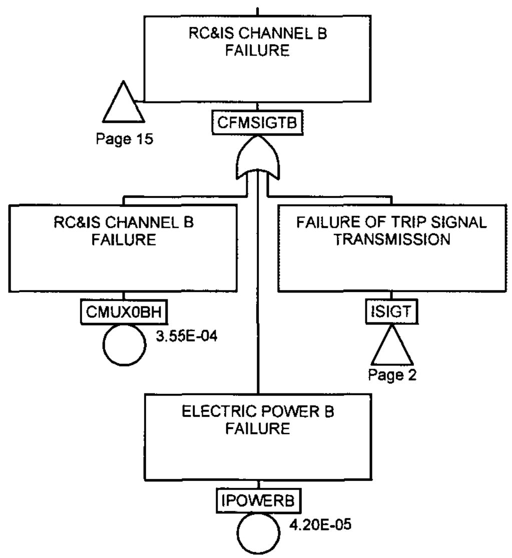


Figure 19D.6-22 Alternate Rod Insertion Fault Tree

Name	Page	Zone	Name	Page	Zone
CARI	1	4	HSVFBA	1	6
CCFMUX	7	2	HSVFBA	14	2
CCFMUX	8	2	IDCM	2	2
CCFMUX	9	2	IDCM	17	2
CCFMUX	10	2	IDMC01H	17	1
CCFMUX	16	3	IDMC01H	18	2
CCFS3A	3	1	IDMC02H	17	2
CFMARI	1	5	IDMC02H	18	4
CFMARI	15	4	IDMC03H	17	2
CFMSIGT	15	4	IDMC03H	18	6
CFMSIGTA	15	3	IDMCCCF	2	2
CFMSIGTB	15	5	IHAR	15	4
CFMSIGTB	19	2	IHAR	18	3
CLCCCFH	15	3	IHARA	18	2
CMAN	3	5	IHARB	18	4
CMUX0AH	15	3	IHARC	18	6
CMUX0BH	19	1	ILCCCFH	4	2
CSIG	1	3	ILCCCFH	5	2
CSIG	3	3	ILCCCFH	6	2
CSIG	15	5	ILCCCFH	7	1
CVARI	1	4	ILCCCFH	8	1
DIV1MUX	7	2	ILCCCFH	9	1
DIV2MUX	8	2	ILCCCFH	10	1
DIV3MUX	9	2	ILE011H	7	2
DIV4MUX	10	2	ILE012H	8	2
EDC11	7	2	ILE013H	9	2
EDC12	8	2	ILE014H	10	2
EDC13	9	2	ILEPVCH	3	3
EDC14	10	2	ILN0V1H	4	2
EDC24A	4	2	ILN0V1H	7	1
EDC24B	5	2	ILN0V2H	5	2
EDC24C	6	2	ILN0V2H	8	1
HSV043C	14	1	ILN0V3H	6	2
HSV044F	13	2	ILN0V3H	9	1
HSV044F	14	2	ILN0V4H	10	1
HSV047D	13	1	IMUX0AH	16	1
HSV48AD	11	1	IMUX0BH	16	2
HSV48BD	12	1	IMUX0CH	16	2
HSV49AD	11	2	IMUX3	16	2
HSV49BD	12	2	IMUXT	2	1
HSVCCF	1	5	IMUXT	16	2
HSVD	1	4	IPOWER	15	2
HSVD1	1	4	IPOWERA	15	1
HSVD1	11	2	IPOWERA	15	3
HSVD2	1	4	IPOWERB	15	2
HSVD2	12	2	IPOWERB	19	2
HSVD3	1	5	IPOWERC	15	2
HSVD3	13	2	IPR001H	4	1

Figure 19D.6-22 Alternate Rod Insertion Fault Tree

Name	Page	Zone	Name	Page	Zone
IPR002H	5	1			
IPR003H	6	1			
IPVLN21	7	2			
IPVLN22	7	3			
IPVLN22	8	2			
IPVLN23	7	4			
IPVLN23	9	2			
IPVLN24	7	5			
IPVLN24	10	2			
IPVLN2T	3	4			
IPVLN2T	7	3			
IPVPN	3	2			
IPVPNT	3	2			
IPVPNTA	3	2			
IPVPNTA	4	2			
IPVPNTB	3	2			
IPVPNTB	5	2			
IPVPNTC	3	3			
IPVPNTC	6	2			
IREEIA	18	1			
IREEIB	18	3			
IREEIC	18	5			
IRMP01H	4	1			
IRMP02H	5	1			
IRMP03H	6	1			
IRMULV11	7	1			
IRMULV12	8	1			
IRMULV13	9	1			
IRMULV14	10	1			
IRPT2	3	4			
ISIGT	1	2			
ISIGT	2	2			
ISIGT	15	4			
ISIGT	19	2			
LNSP	15	6			
NEDC	1	1			
NEDCB	12	2			

Figure 19D.6-22 Alternate Rod Insertion Fault Tree

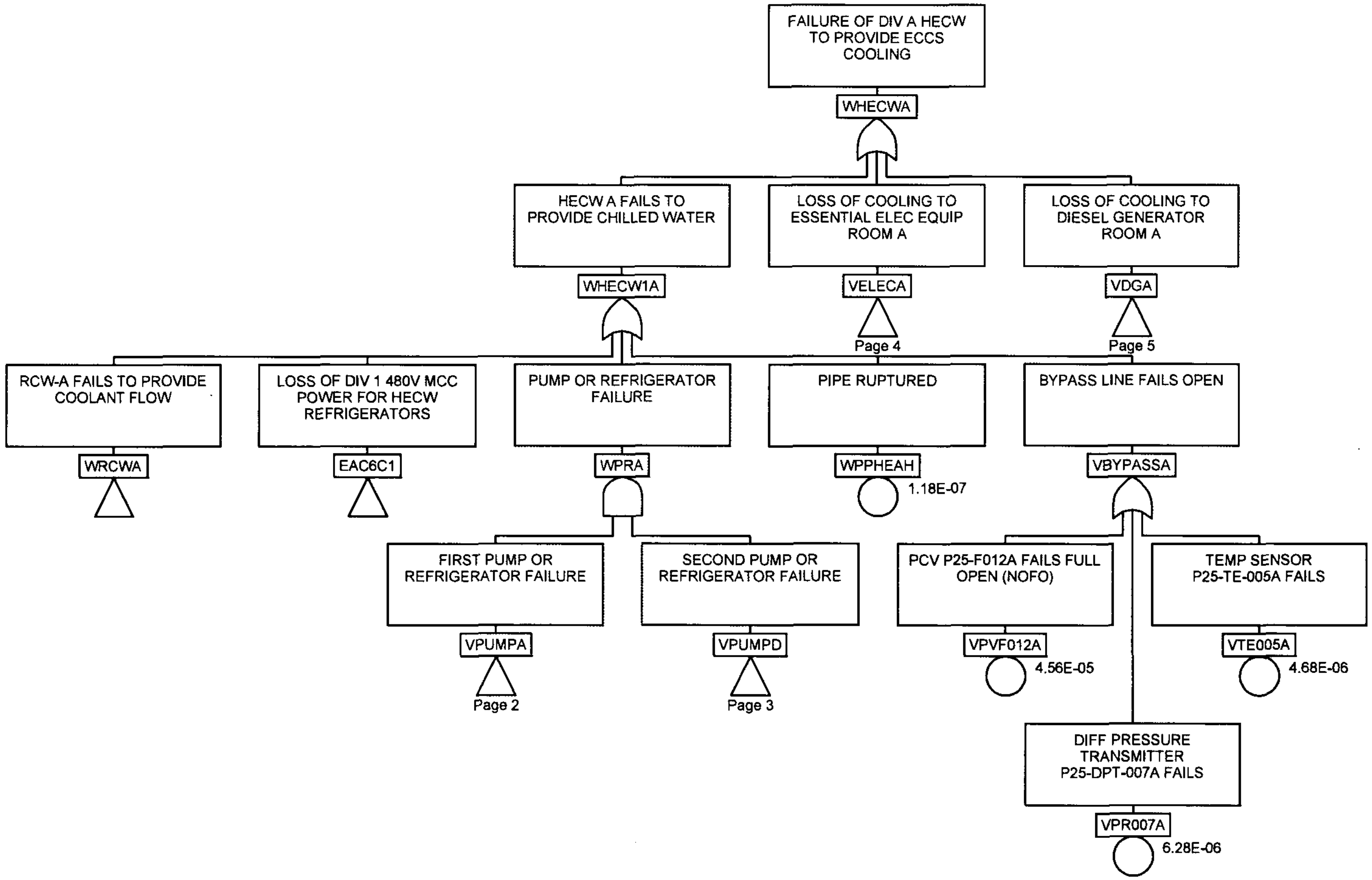


Figure 19D.6-23A HVAC Emergency Cooling Water Fault Tree Train A

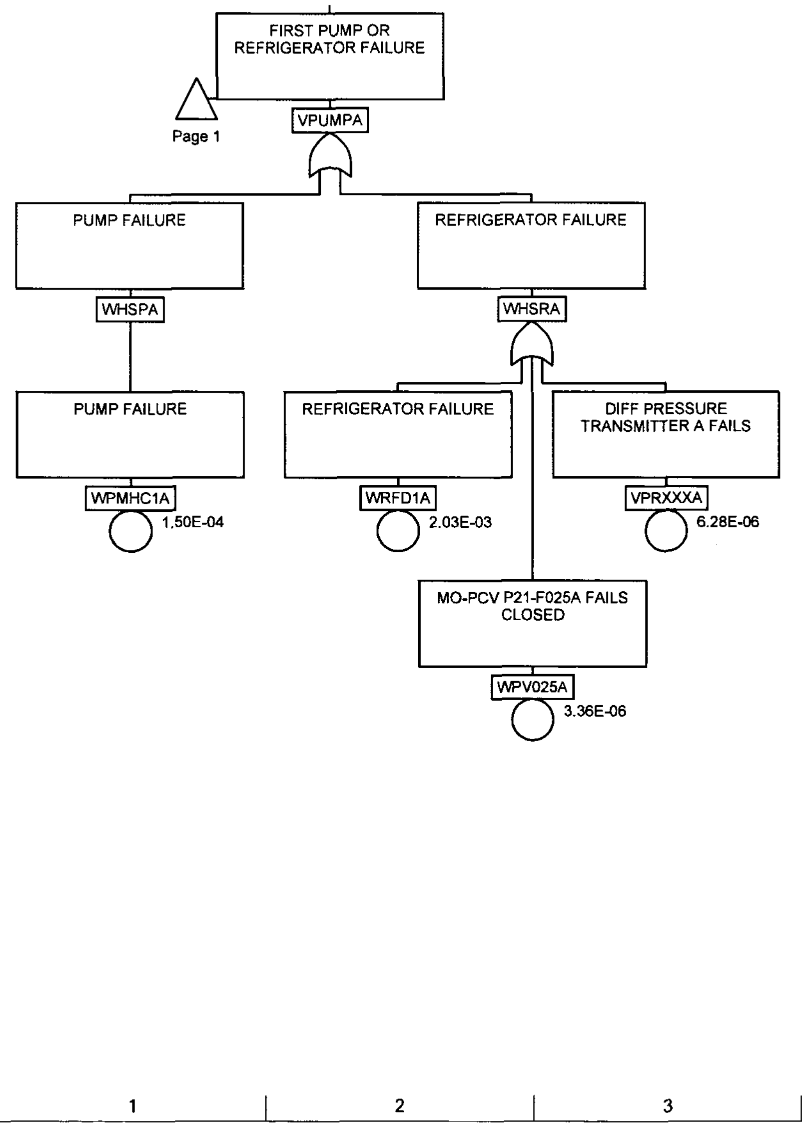


Figure 19D.6-23A HVAC Emergency Cooling Water Fault Tree Train A

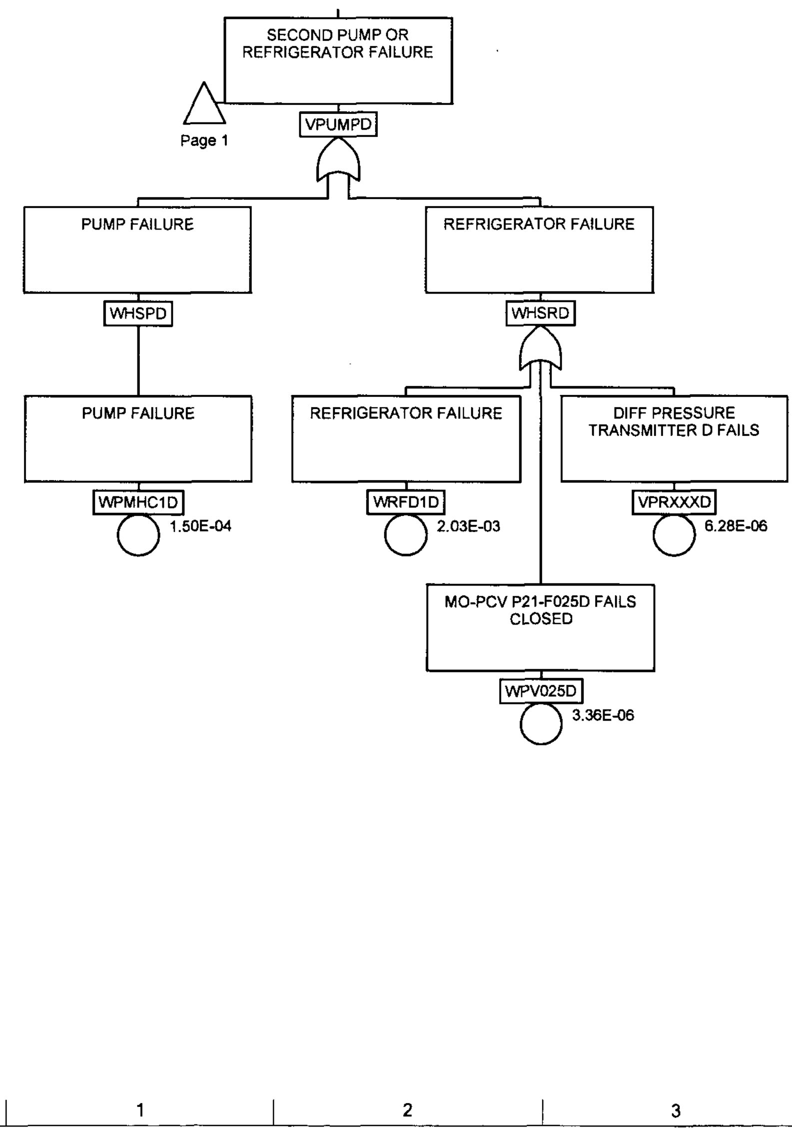


Figure 19D.6-23A HVAC Emergency Cooling Water Fault Tree Train A

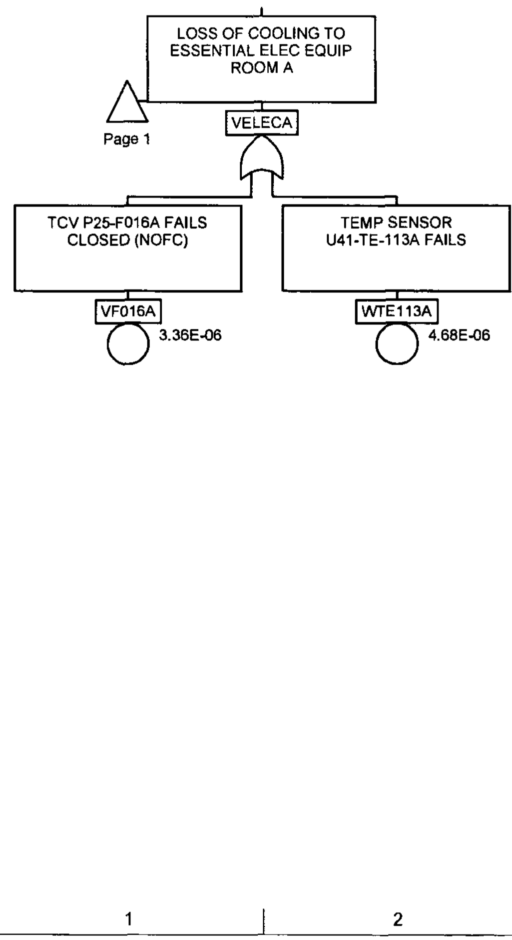


Figure 19D.6-23A HVAC Emergency Cooling Water Fault Tree Train A

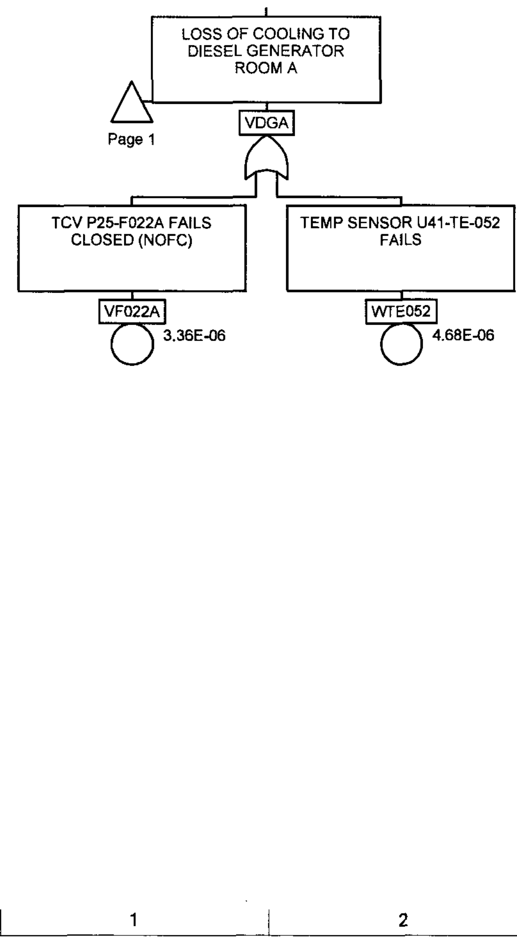


Figure 19D.6-23A HVAC Emergency Cooling Water Fault Tree Train A

Name	Page	Zone	Name	Page	Zone
EAC6C1	1	2			
VBYPASSA	1	5			
VDGA	1	5			
VDGA	5	2			
VELECA	1	4			
VELECA	4	2			
VF016A	4	1			
VF022A	5	1			
VPR007A	1	5			
VPRXXXA	2	3			
VPRXXXD	3	3			
VPUMPA	1	3			
VPUMPA	2	2			
VPUMPD	1	4			
VPUMPD	3	2			
VPVF012A	1	5			
VTE005A	1	6			
WHECW1A	1	3			
WHECWA	1	4			
WHSPA	2	1			
WHSPD	3	1			
WHSRA	2	3			
WHSRD	3	3			
WPMHC1A	2	1			
WPMHC1D	3	1			
WPPHEAH	1	4			
WPRA	1	3			
WPV025A	2	3			
WPV025D	3	3			
WRCWA	1	1			
WRFD1A	2	2			
WRFD1D	3	2			
WTE052	5	2			
WTE113A	4	2			

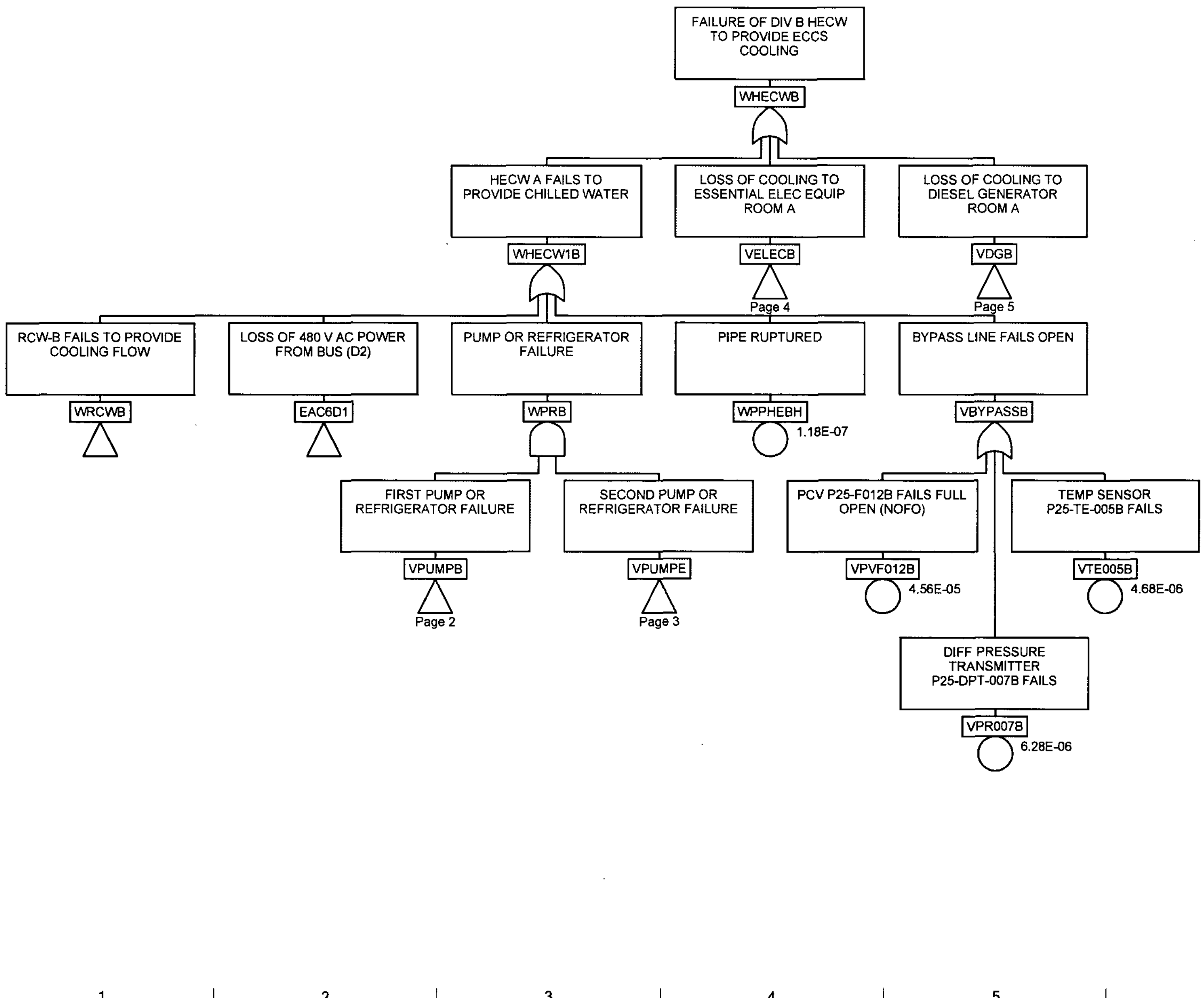


Figure 19D.6-23B HVAC Emergency Cooling Water Fault Tree Train B

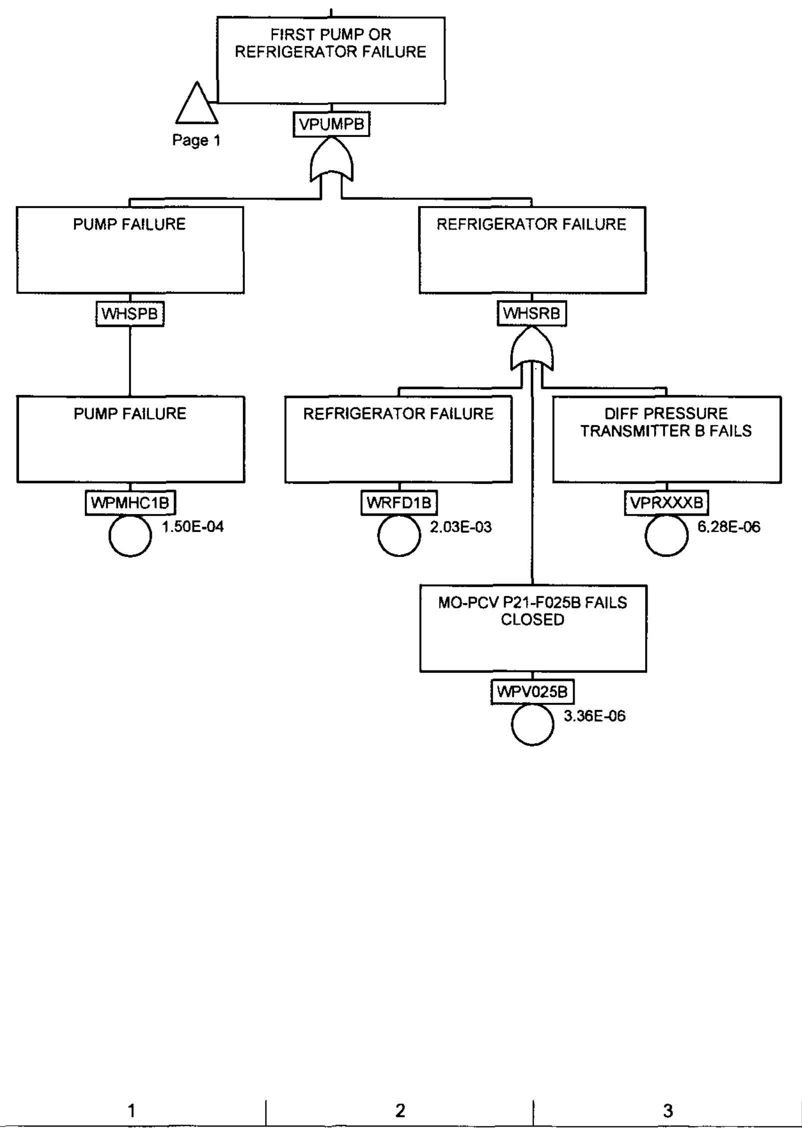


Figure 19D.6-23B HVAC Emergency Cooling Water Fault Tree Train B

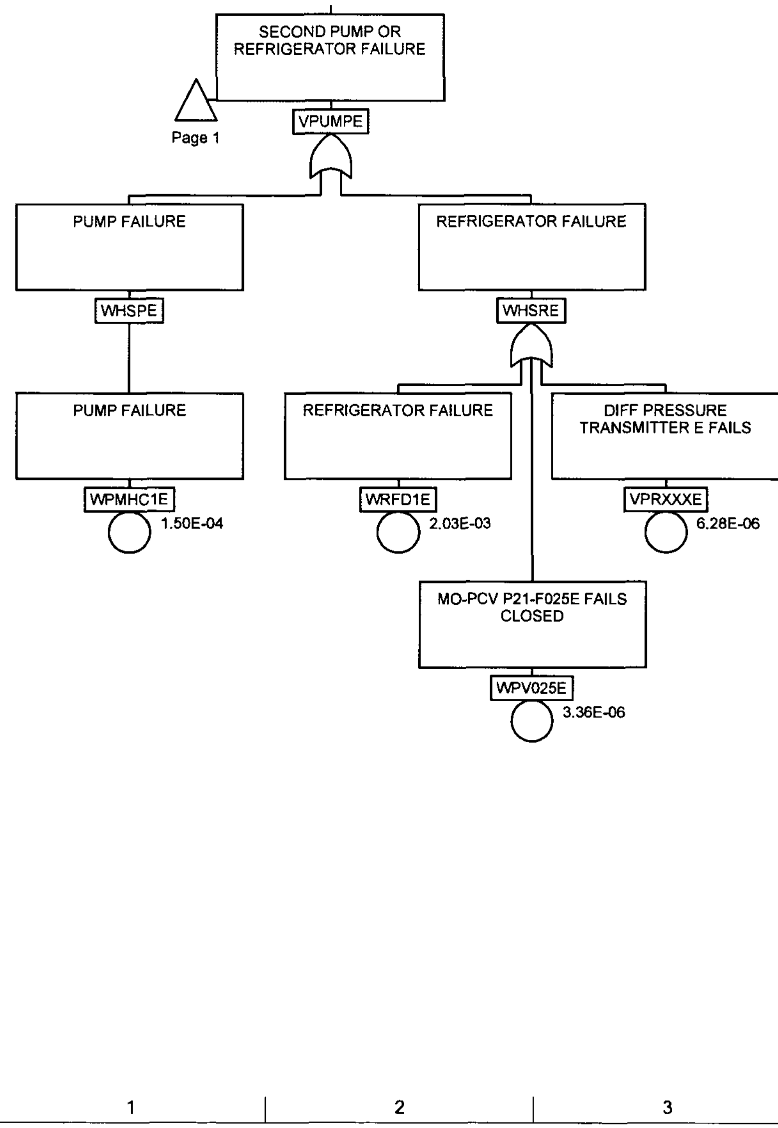


Figure 19D.6-23B HVAC Emergency Cooling Water Fault Tree Train B

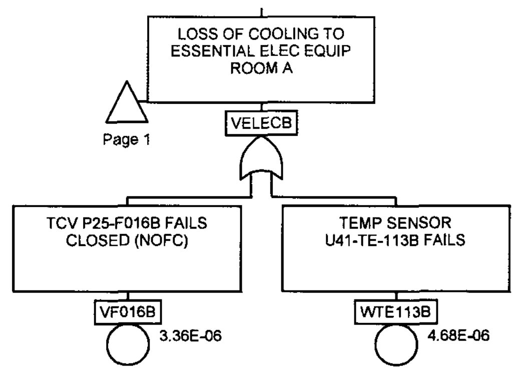
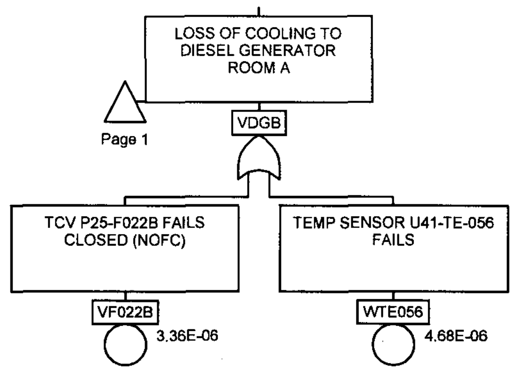


Figure 19D.6-23B HVAC Emergency Cooling Water Fault Tree Train B



1 2

Figure 19D.6-23B HVAC Emergency Cooling Water Fault Tree Train B

Name	Page	Zone	Name	Page	Zone
EAC6D1	1	2			
VBYPASSB	1	5			
VDGB	1	5			
VDGB	5	2			
VELECB	1	4			
VELECB	4	2			
VF016B	4	1			
VF022B	5	1			
VPR007B	1	5			
VPRXXXB	2	3			
VPRXXXE	3	3			
VPUMPB	1	3			
VPUMPB	2	2			
VPUMPE	1	4			
VPUMPE	3	2			
VPVF012B	1	5			
VTE005B	1	6			
WHECW1B	1	3			
WHECWB	1	4			
WHSPB	2	1			
WHSPE	3	1			
WHSRB	2	3			
WHSRE	3	3			
WPMHC1B	2	1			
WPMHC1E	3	1			
WPPHEBH	1	4			
WPRB	1	3			
WPV025B	2	3			
WPV025E	3	3			
WRCWB	1	1			
WRFD1B	2	2			
WRFD1E	3	2			
WTE056	5	2			
WTE113B	4	2			

Figure 19D.6-23B HVAC Emergency Cooling Water Fault Tree Train B

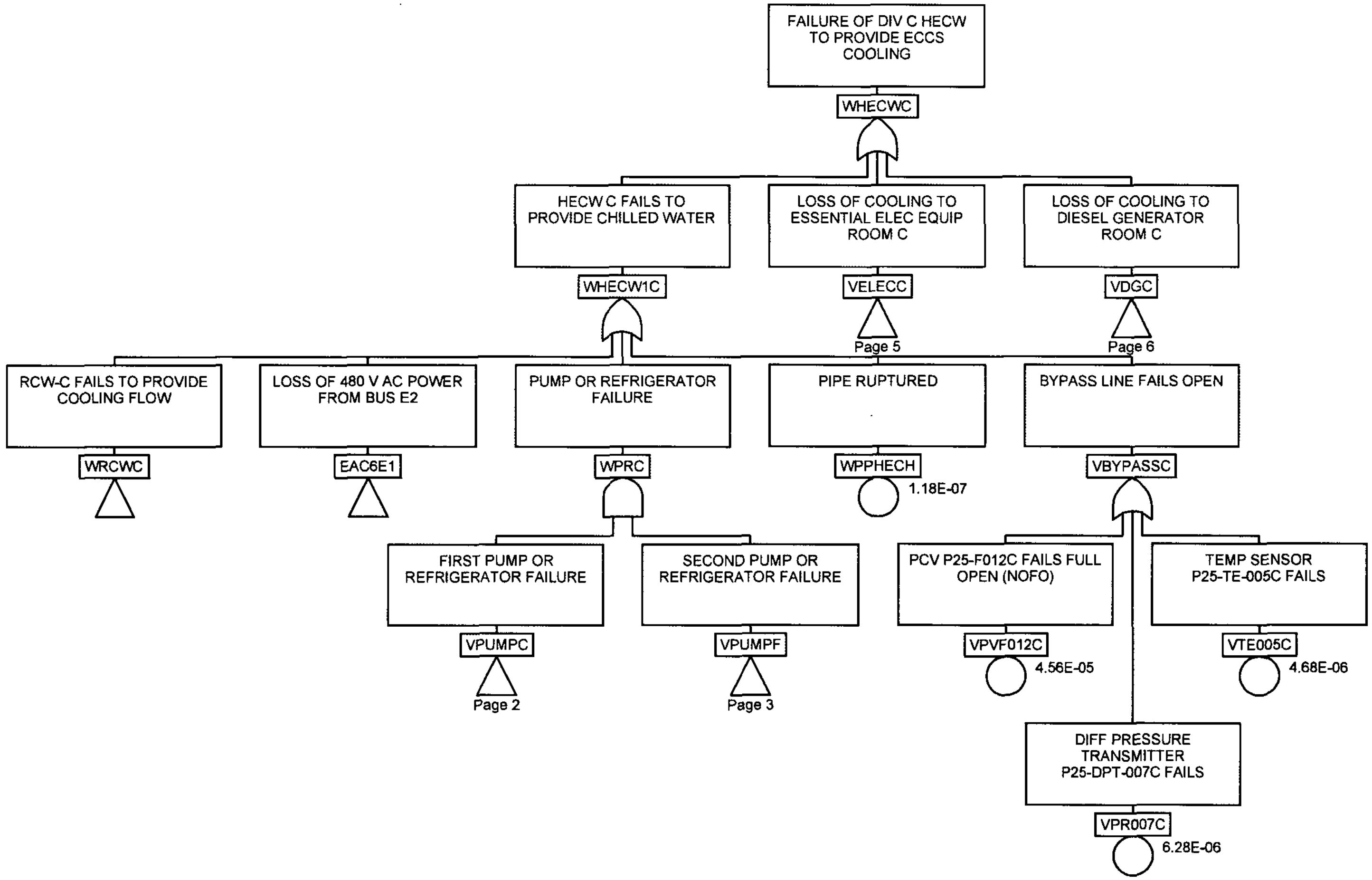


Figure 19D.6-23C HVAC Emergency Cooling Water Fault Tree Train C

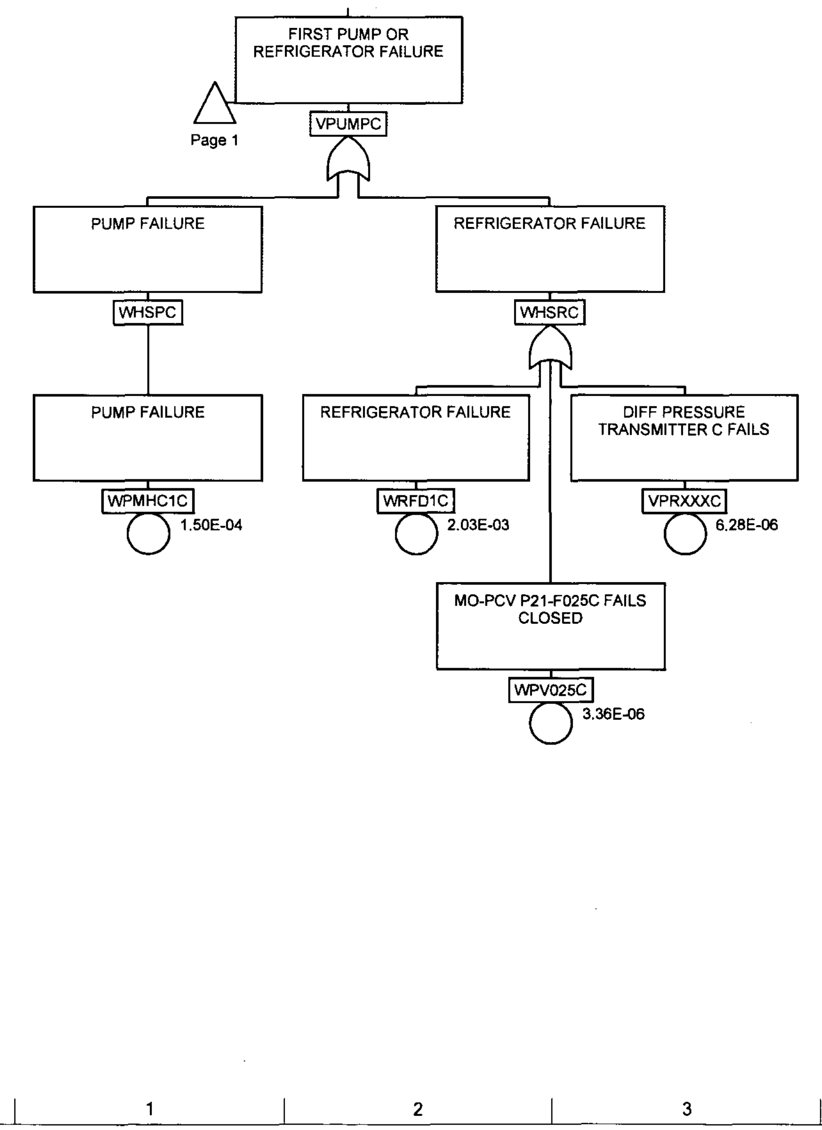


Figure 19D.6-23C HVAC Emergency Cooling Water Fault Tree Train C

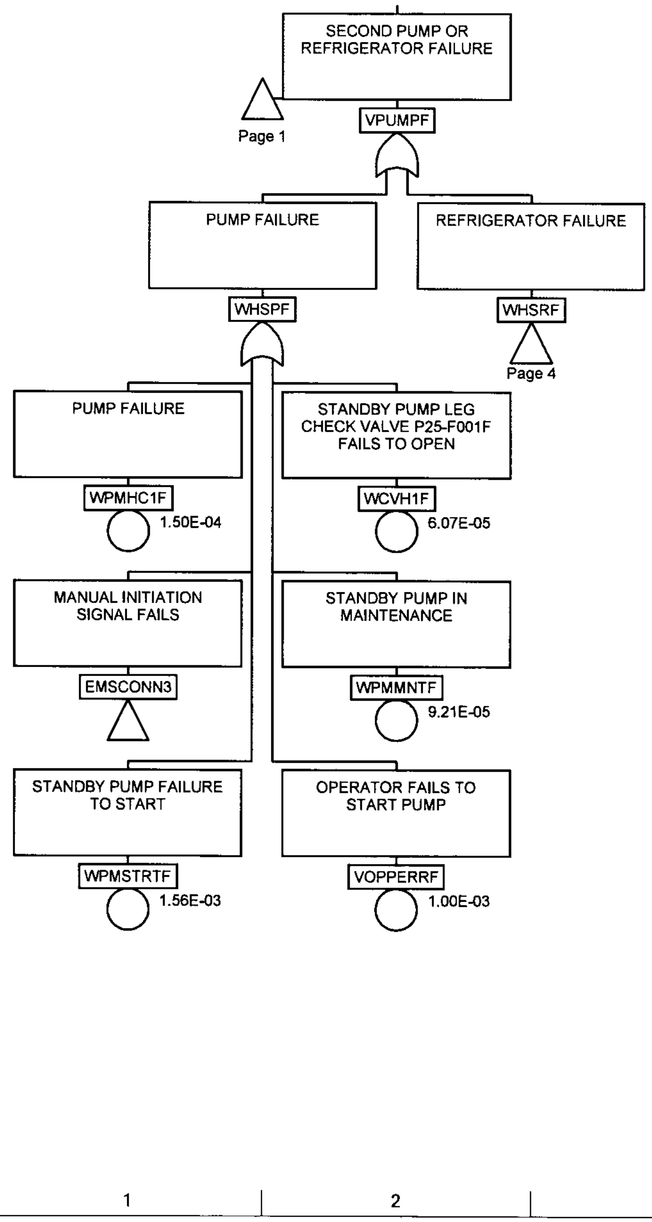


Figure 19D.6-23C HVAC Emergency Cooling Water Fault Tree Train C

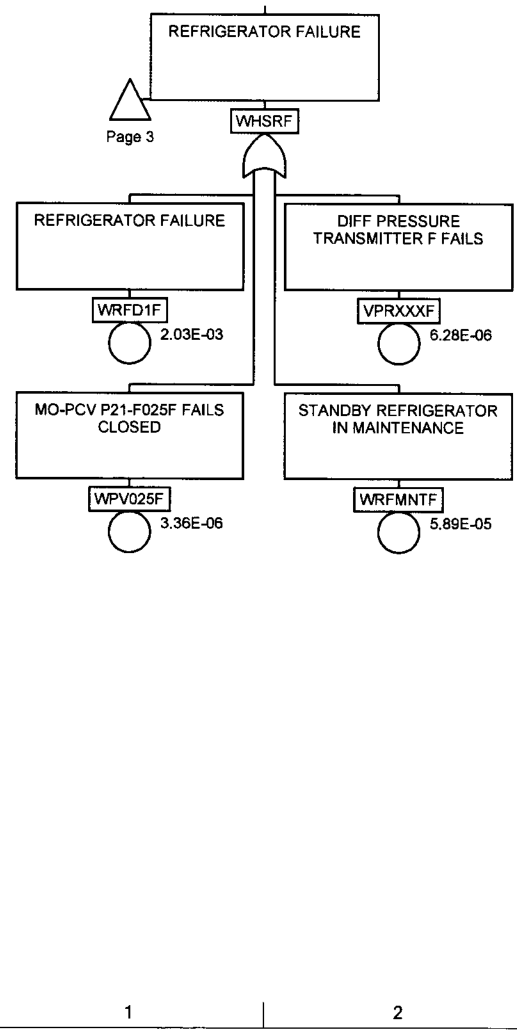


Figure 19D.6-23C HVAC Emergency Cooling Water Fault Tree Train C

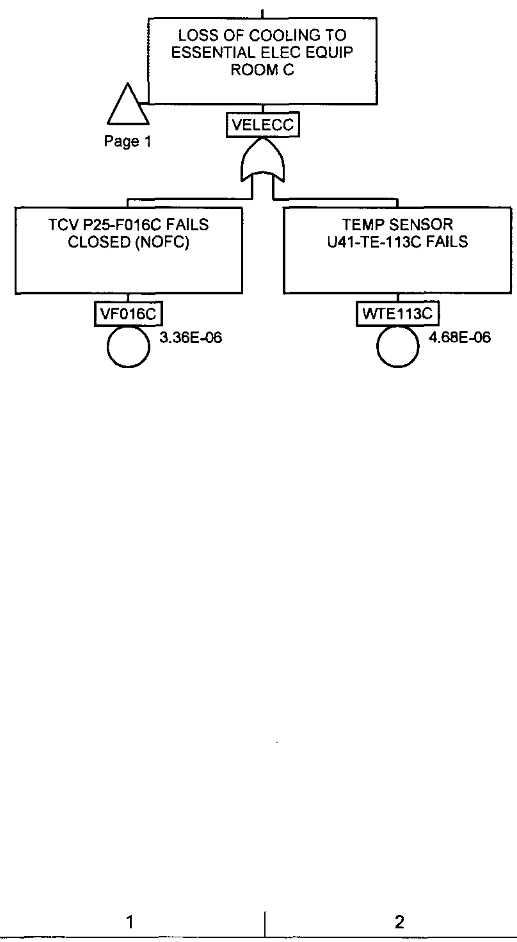


Figure 19D.6-23C HVAC Emergency Cooling Water Fault Tree Train C

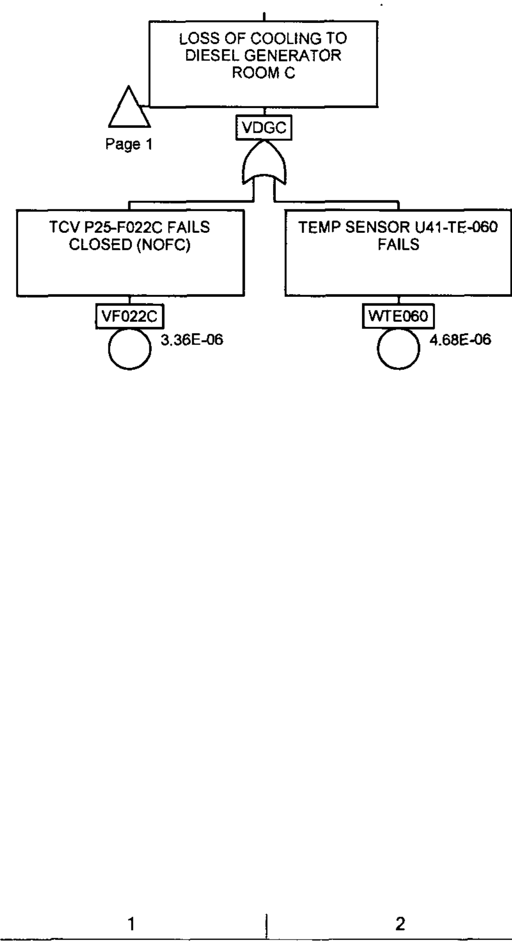


Figure 19D.6-23C HVAC Emergency Cooling Water Fault Tree Train C

Name	Page	Zone	Name	Page	Zone
EAC6E1	1	2			
EMSCONN3	3	1			
VBYPASSC	1	5			
VDGC	1	5			
VDGC	6	2			
VELECC	1	4			
VELECC	5	2			
VF016C	5	1			
VF022C	6	1			
VOPPERRF	3	2			
VPR007C	1	5			
VPRXXXC	2	3			
VPRXXXF	4	2			
VPUMPC	1	3			
VPUMPC	2	2			
VPUMPF	1	4			
VPUMPF	3	2			
VPVF012C	1	5			
VTE005C	1	6			
WCVH1F	3	2			
WHECW1C	1	3			
WHECWC	1	4			
WHSPC	2	1			
WHSPF	3	2			
WHSRC	2	3			
WHSRF	3	3			
WHSRF	4	2			
WPMHC1C	2	1			
WPMHC1F	3	1			
WPMMNTF	3	2			
WPMSTRTF	3	1			
WPPHECH	1	4			
WPRC	1	3			
WPV025C	2	3			
WPV025F	4	1			
WRCWC	1	1			
WRFD1C	2	2			
WRFD1F	4	1			
WRFMNTF	4	2			
WTE060	6	2			
WTE113C	5	2			

Figure 19D.6-23C HVAC Emergency Cooling Water Fault Tree Train C

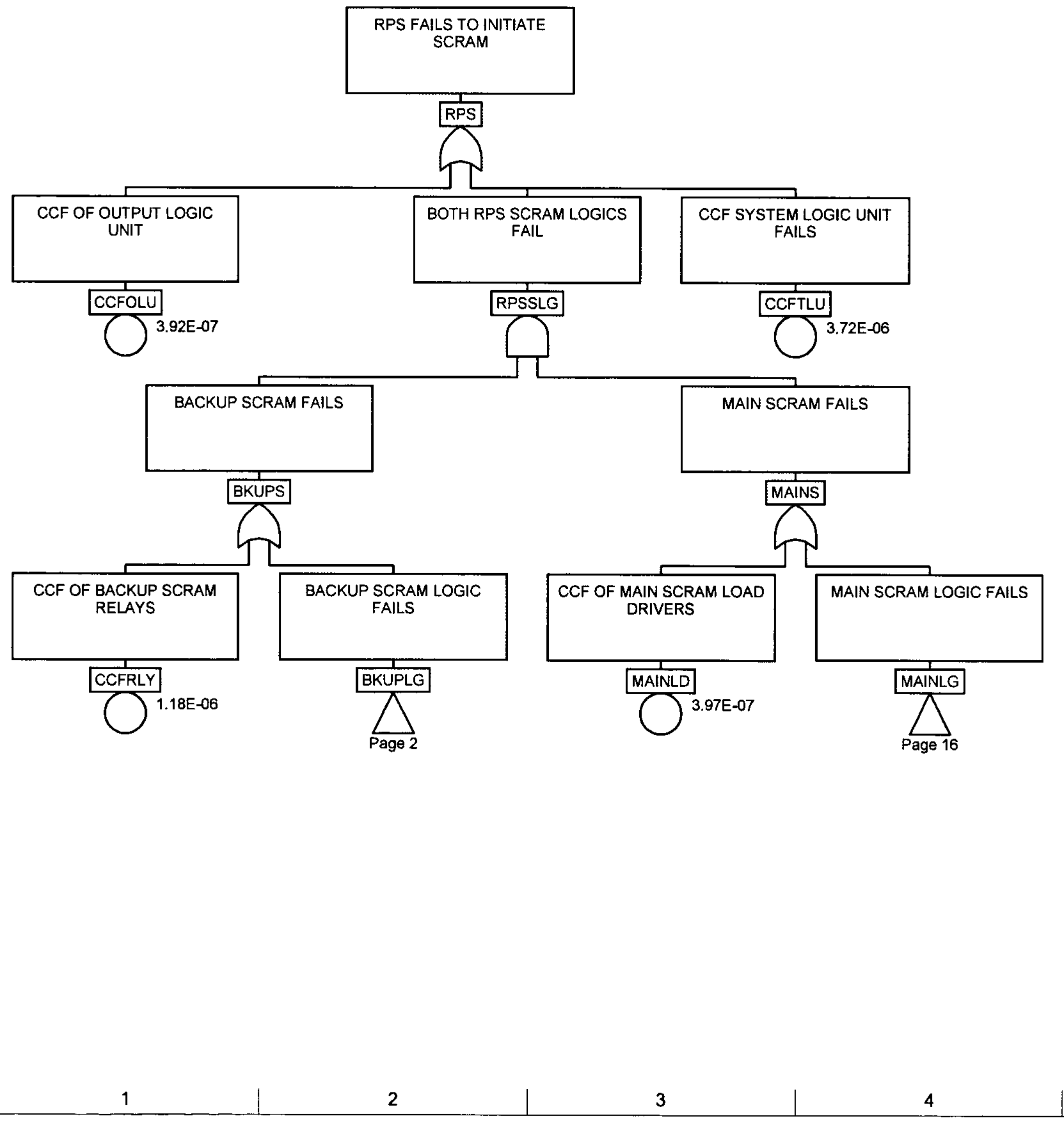


Figure 19D.6-24 Reactor Protection System Fault Tree

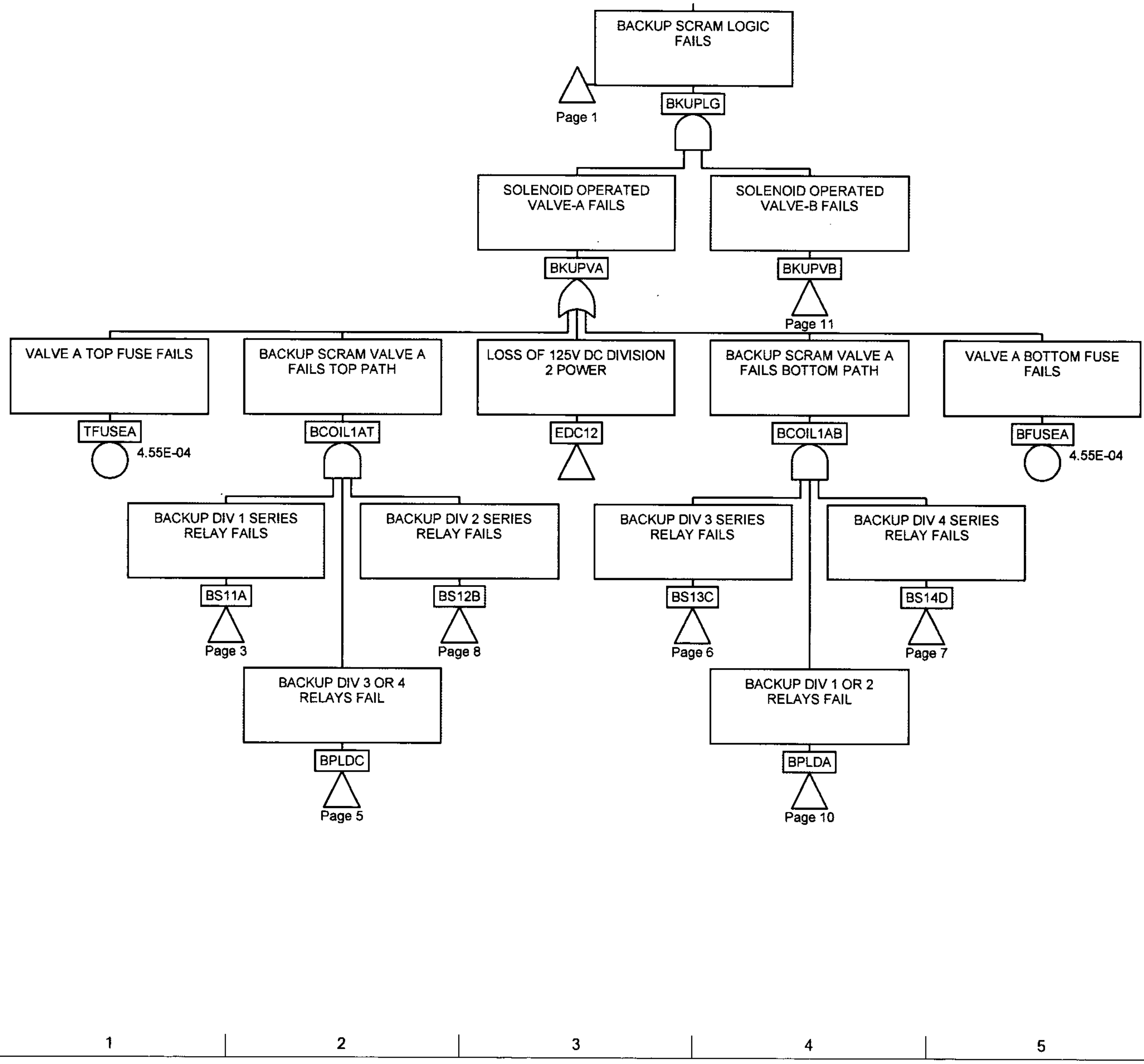


Figure 19D.6-24 Reactor Protection System Fault Tree

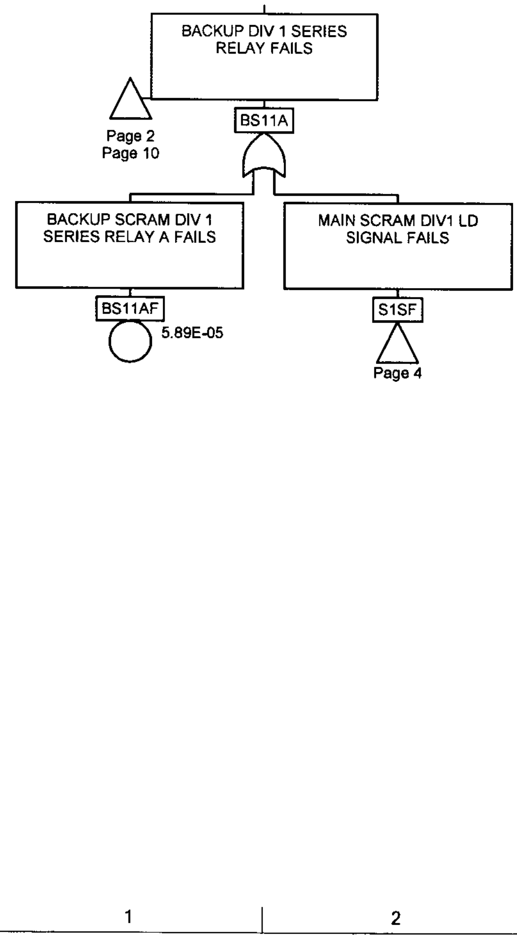
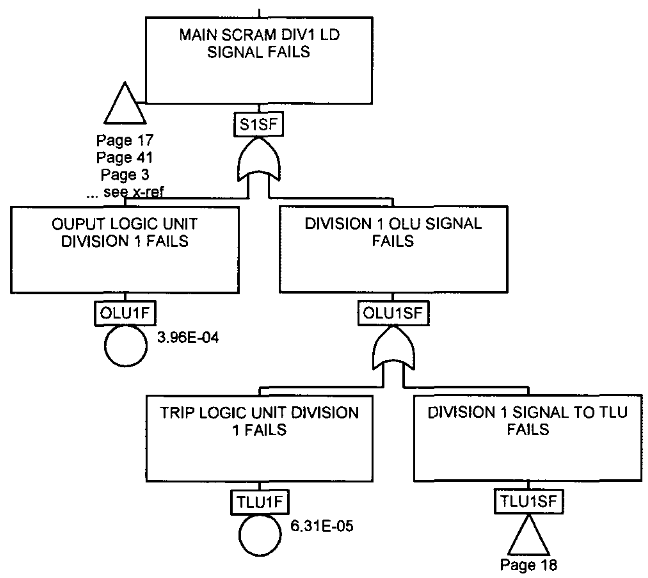


Figure 19D.6-24 Reactor Protection System Fault Tree


MAIN SCRAM DIV1 LD SIGNAL FAILS
S1SF
Page 17
Page 41
Page 3
... see x-ref
OUPUT LOGIC UNIT DIVISION 1 FAILS
OLU1F
3.96E-04
DIVISION 1 OLU SIGNAL FAILS
OLU1SF
TRIP LOGIC UNIT DIVISION 1 FAILS
TLU1F
6.31E-05
DIVISION 1 SIGNAL TO TLU FAILS
TLU1SF
Page 18
1
2

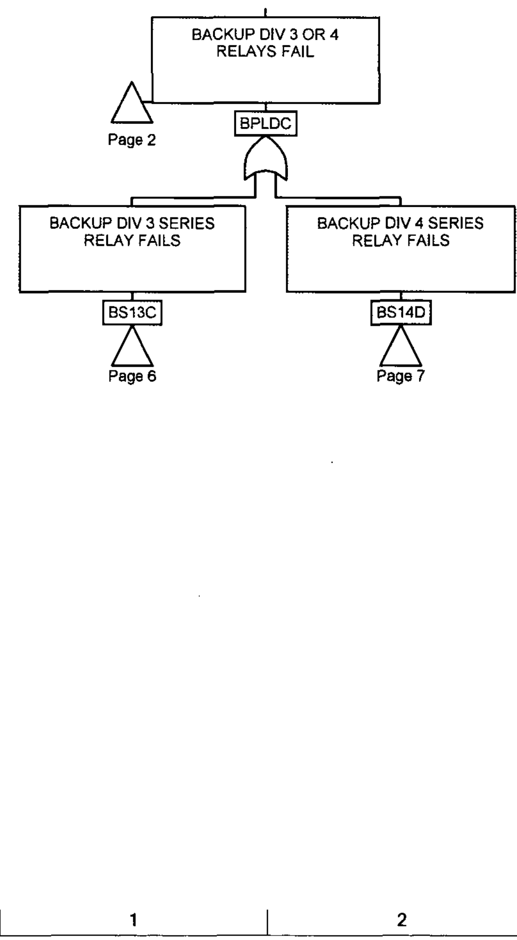
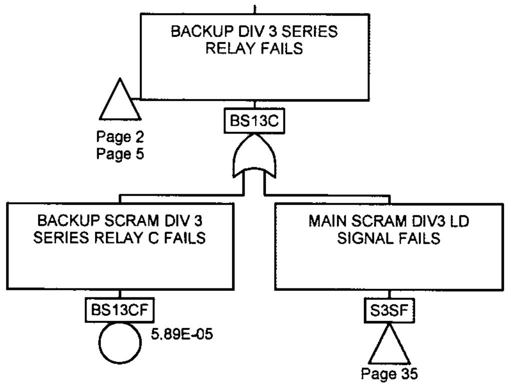


Figure 19D.6-24 Reactor Protection System Fault Tree


BACKUP DIV 3 SERIES RELAY FAILS
Page 2
Page 5
BS13C
BACKUP SCRAM DIV 3 SERIES RELAY C FAILS
MAIN SCRAM DIV3 LD SIGNAL FAILS
BS13CF
5.89E-05
S3SF
Page 35

1	2

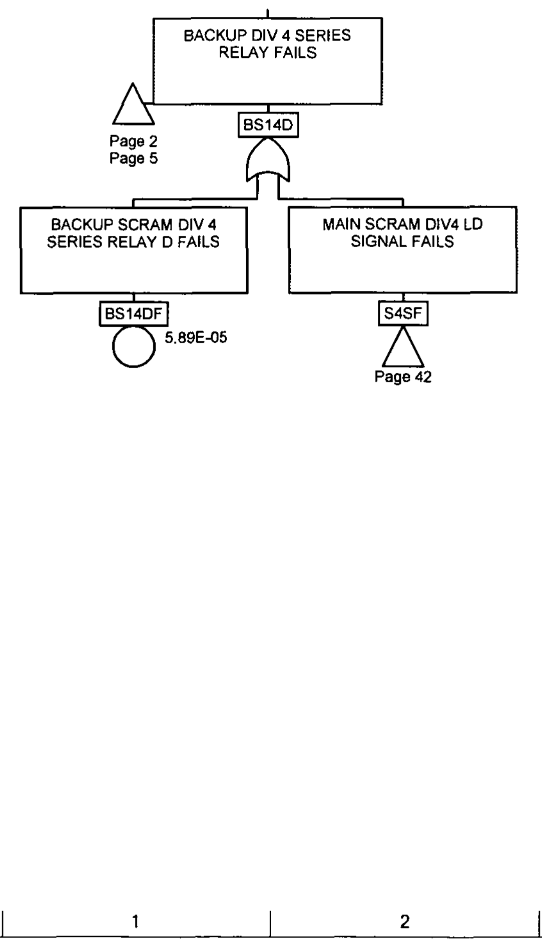


Figure 19D.6-24 Reactor Protection System Fault Tree

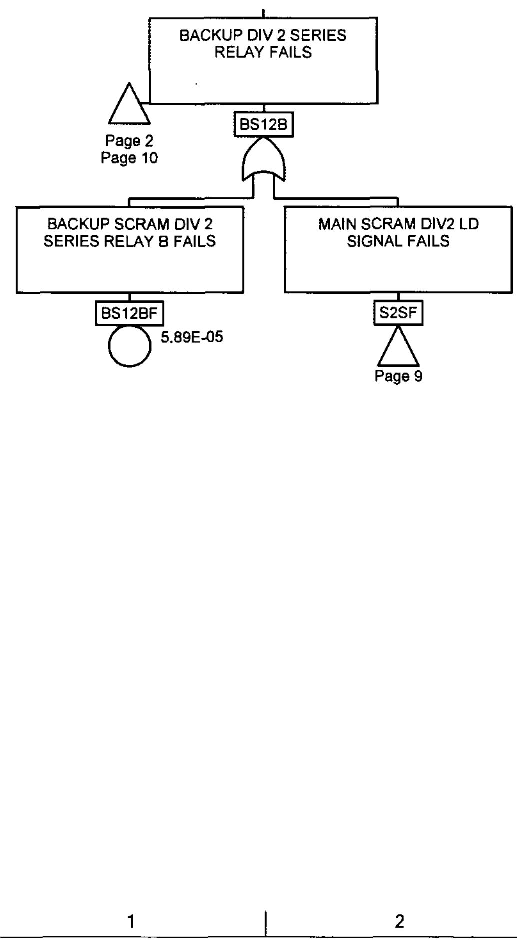


Figure 19D.6-24 Reactor Protection System Fault Tree

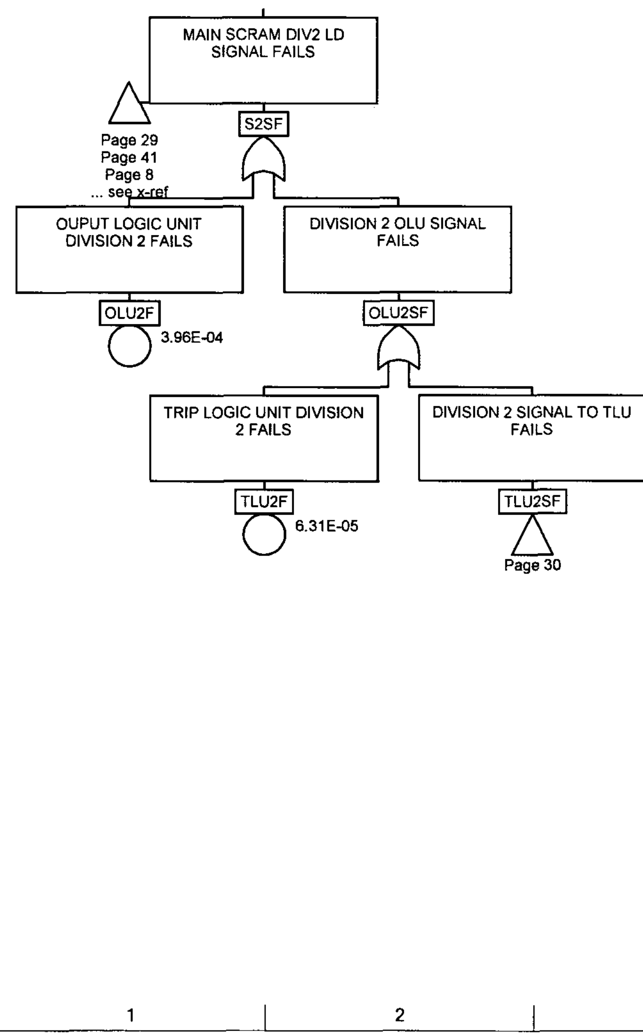


Figure 19D.6-24 Reactor Protection System Fault Tree

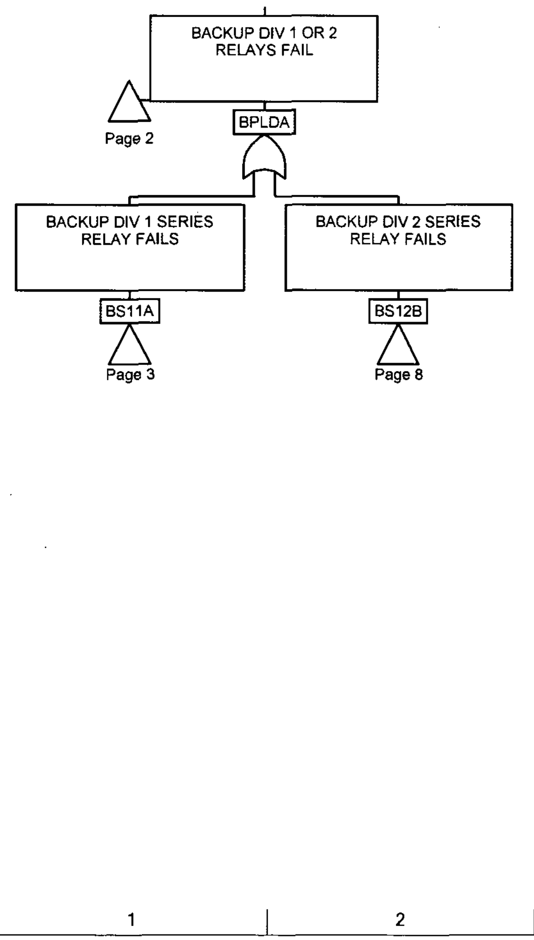


Figure 19D.6-24 Reactor Protection System Fault Tree

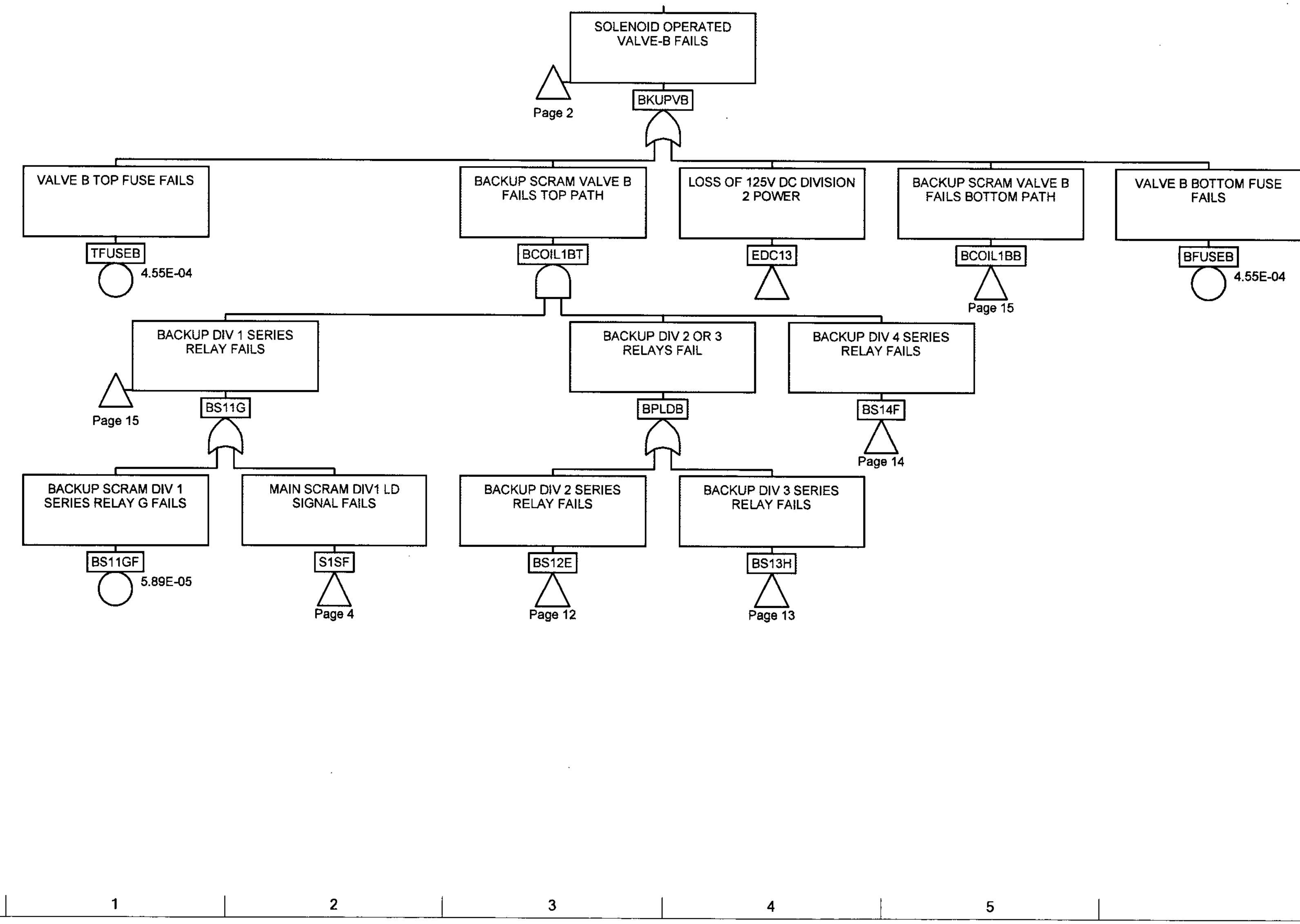


Figure 19D.6-24 Reactor Protection System Fault Tree

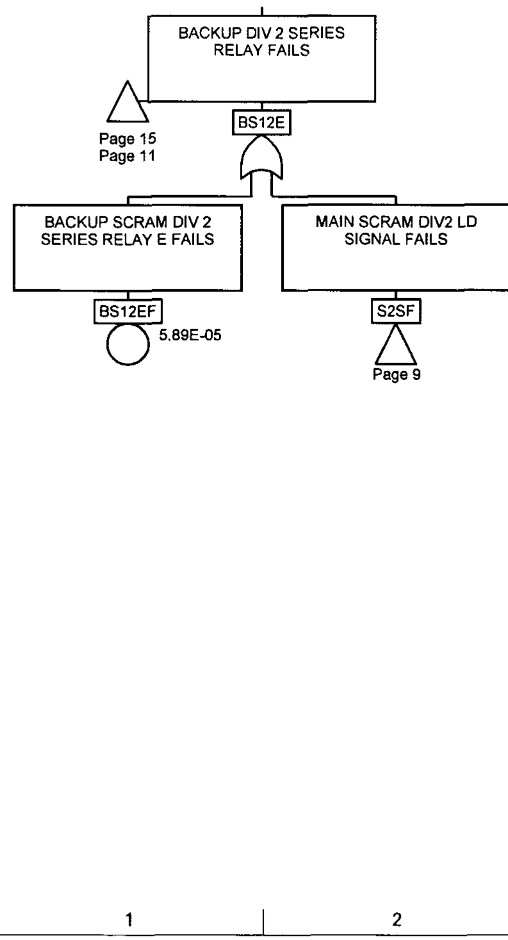


Figure 19D.6-24 Reactor Protection System Fault Tree

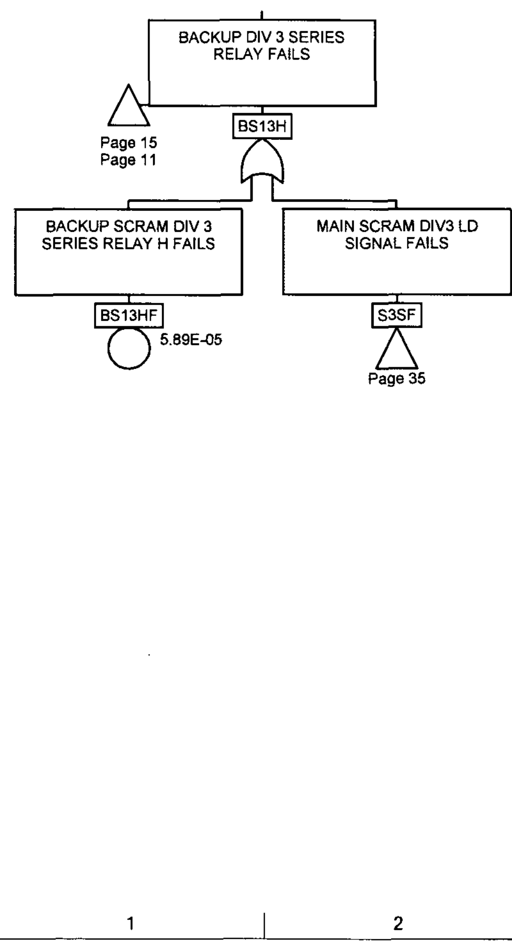
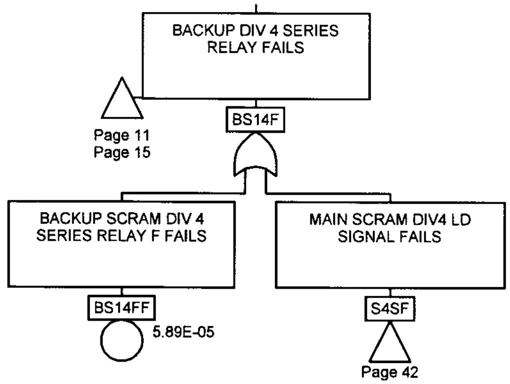


Figure 19D.6-24 Reactor Protection System Fault Tree



1 2

Figure 19D.6-24 Reactor Protection System Fault Tree

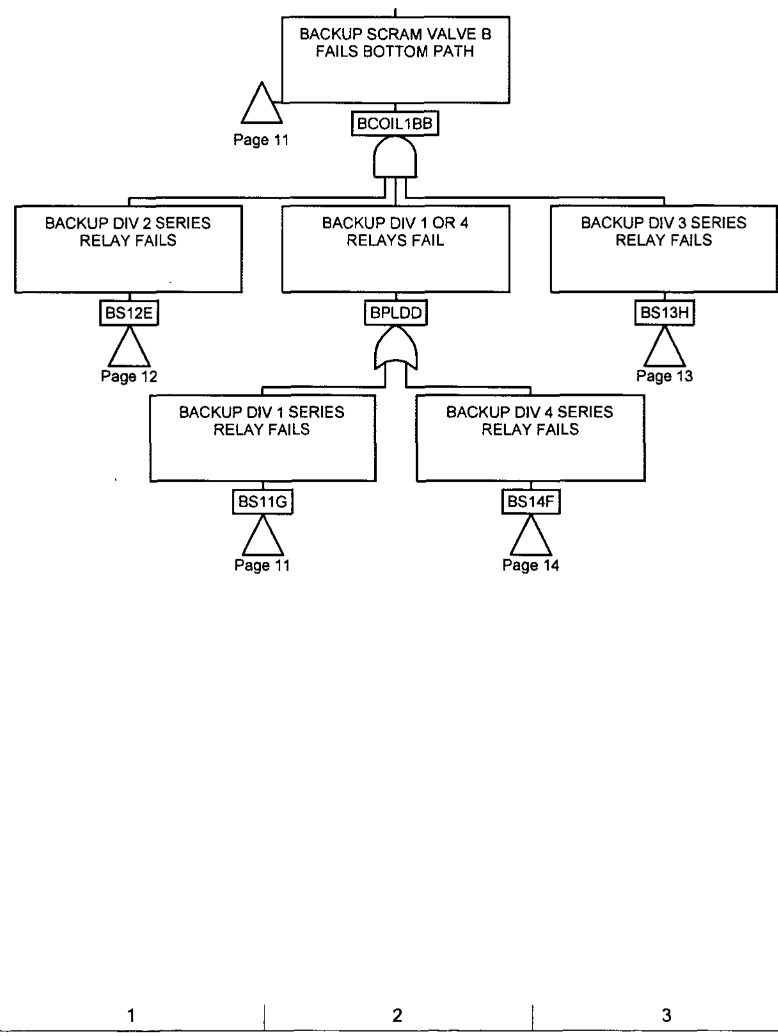


Figure 19D.6-24 Reactor Protection System Fault Tree

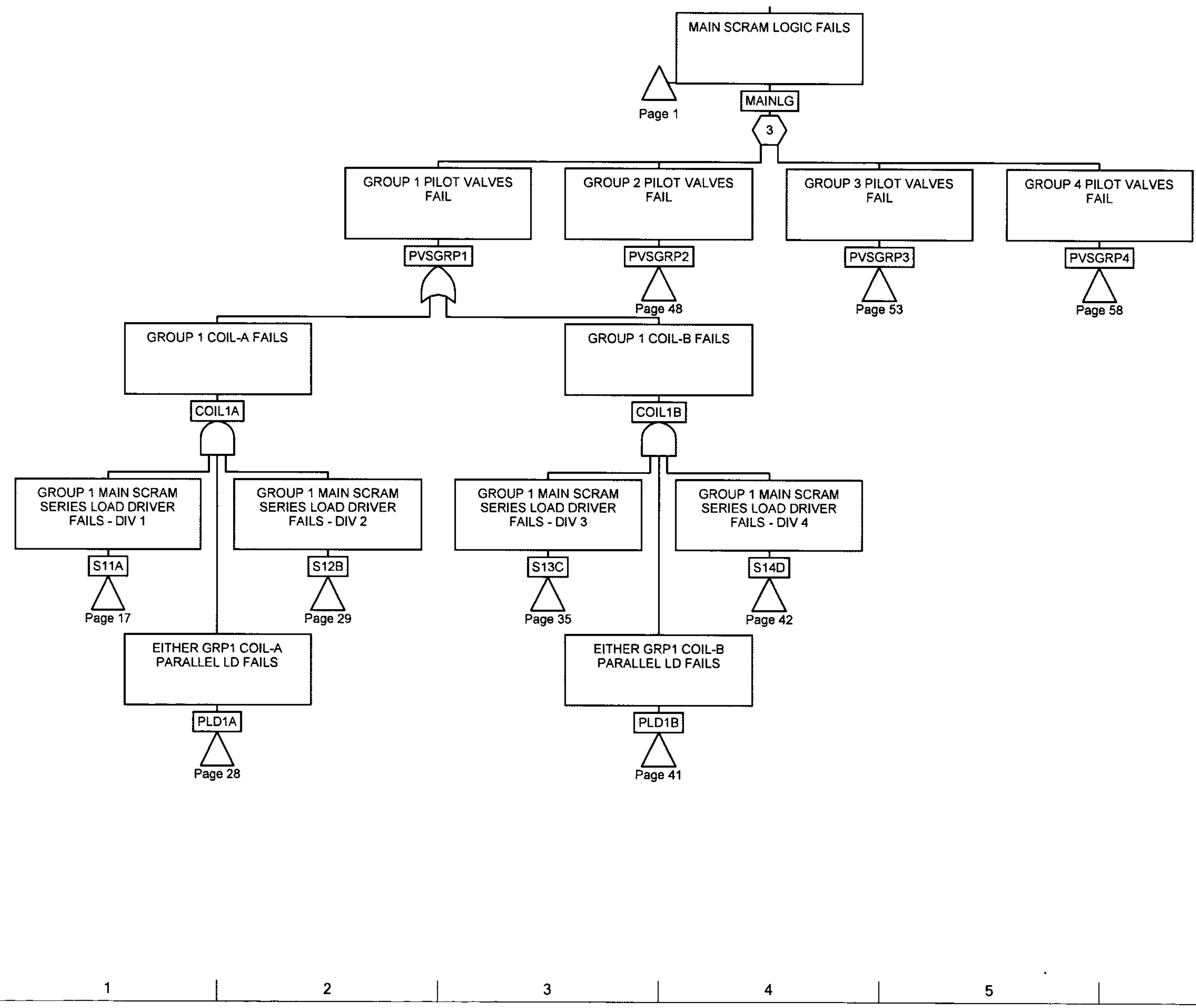


Figure 19D.6-24 Reactor Protection System Fault Tree

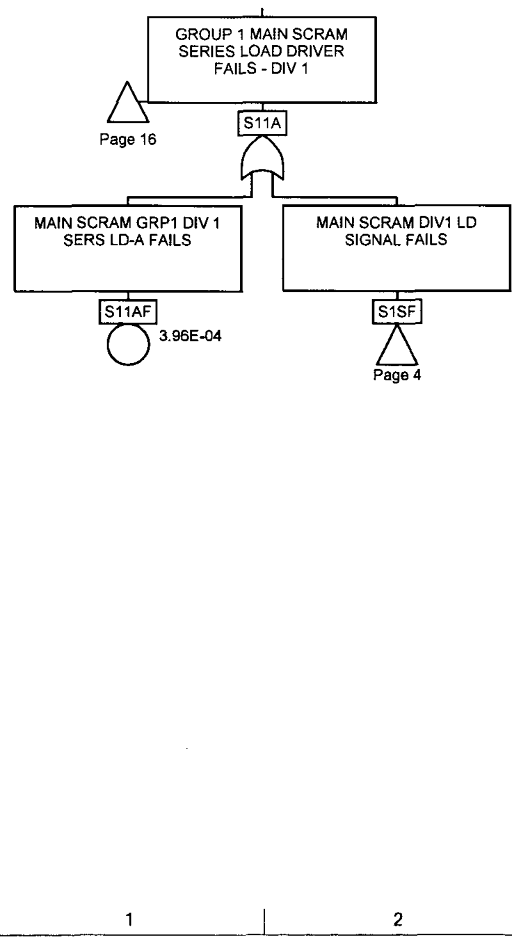


Figure 19D.6-24 Reactor Protection System Fault Tree

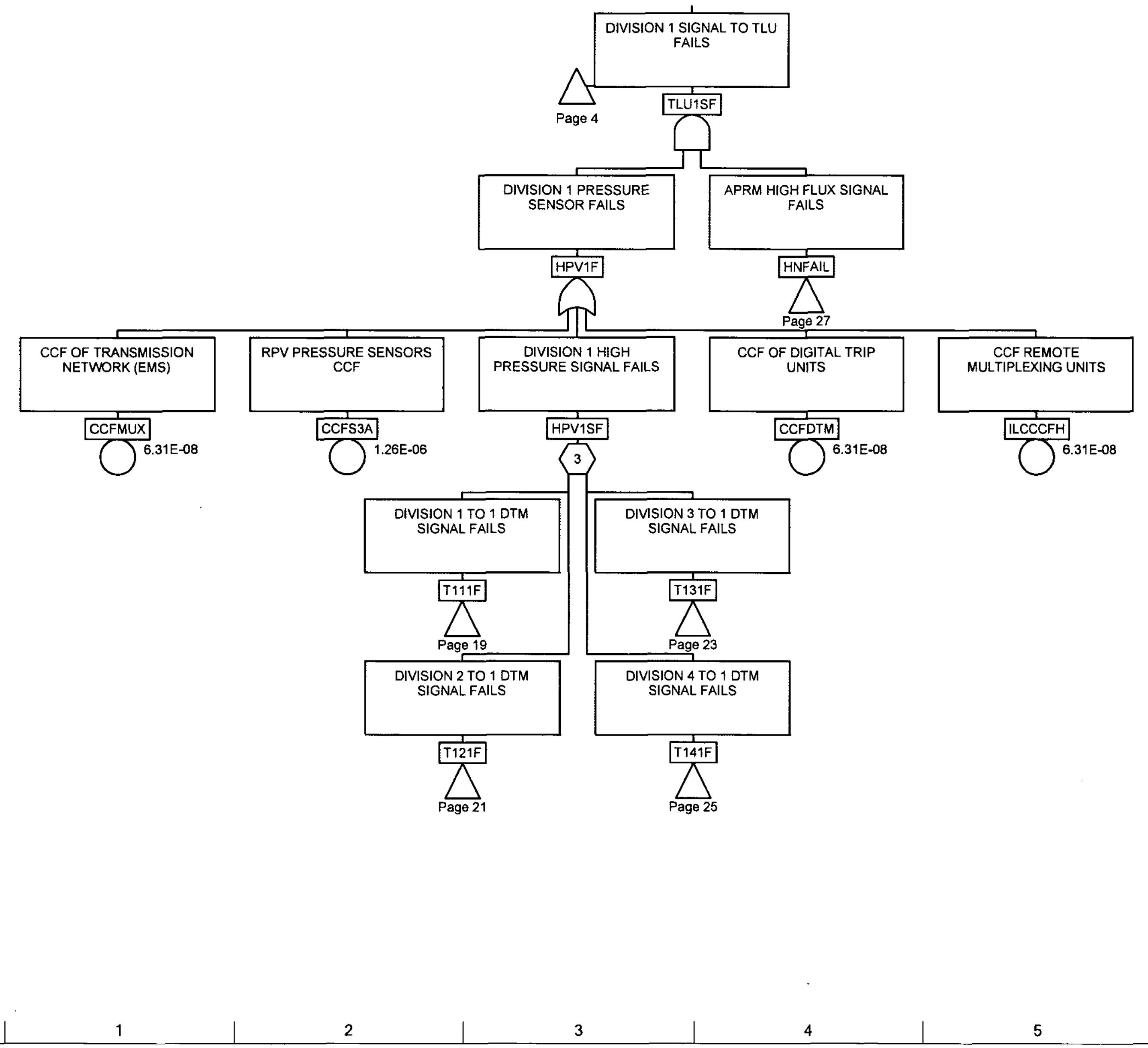
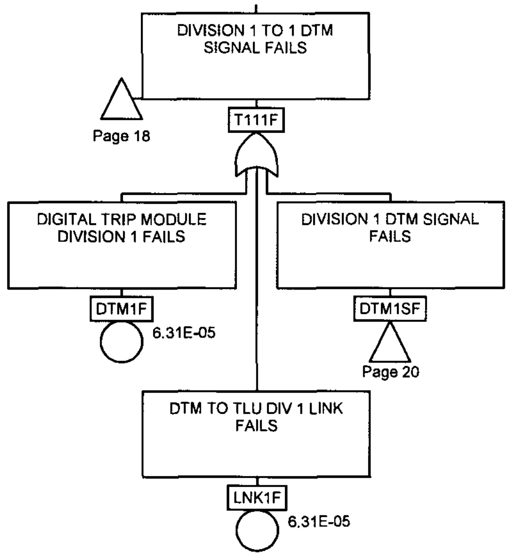


Figure 19D.6-24 Reactor Protection System Fault Tree



1 2

Figure 19D.6-24 Reactor Protection System Fault Tree

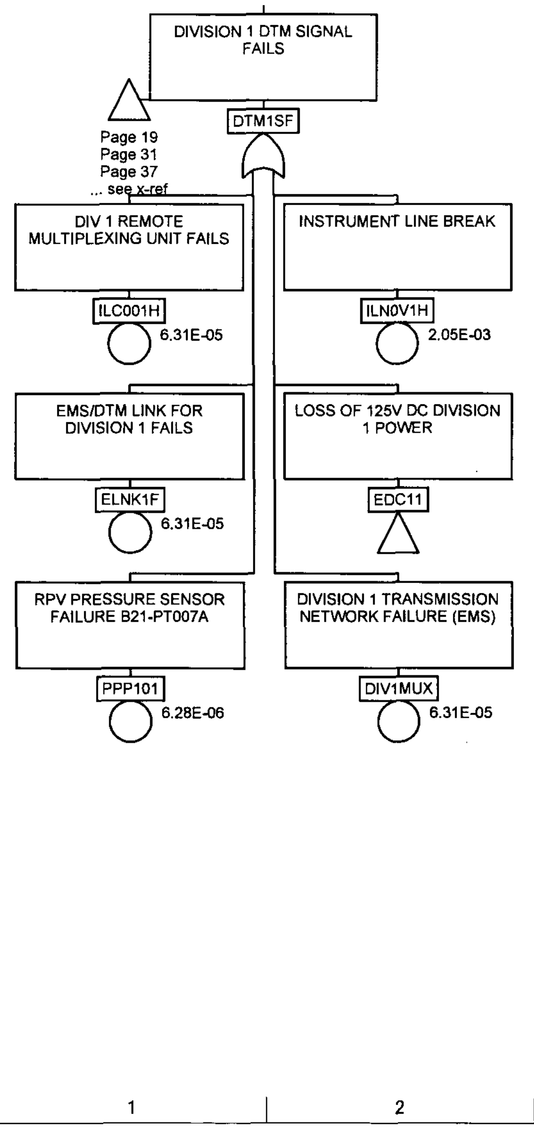


Figure 19D.6-24 Reactor Protection System Fault Tree

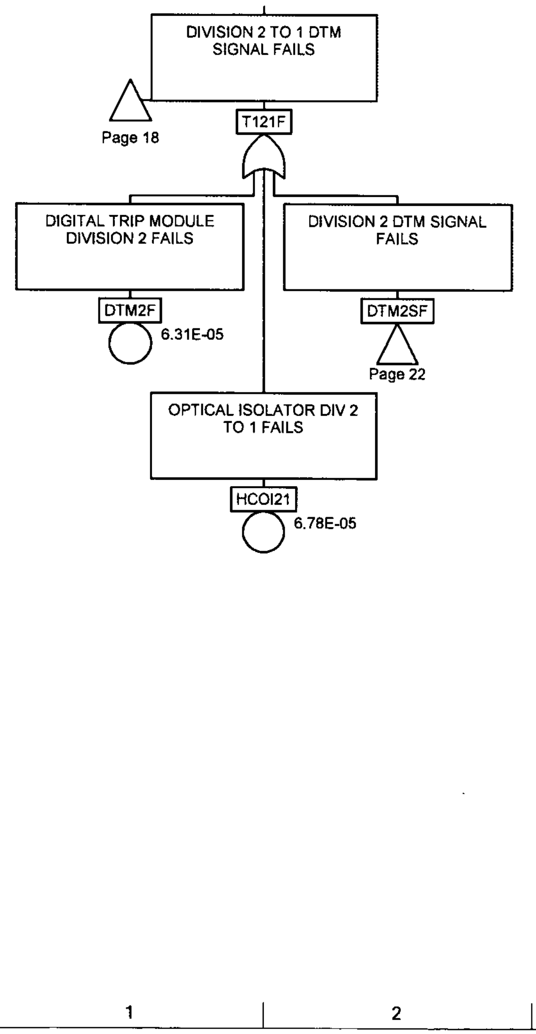


Figure 19D.6-24 Reactor Protection System Fault Tree

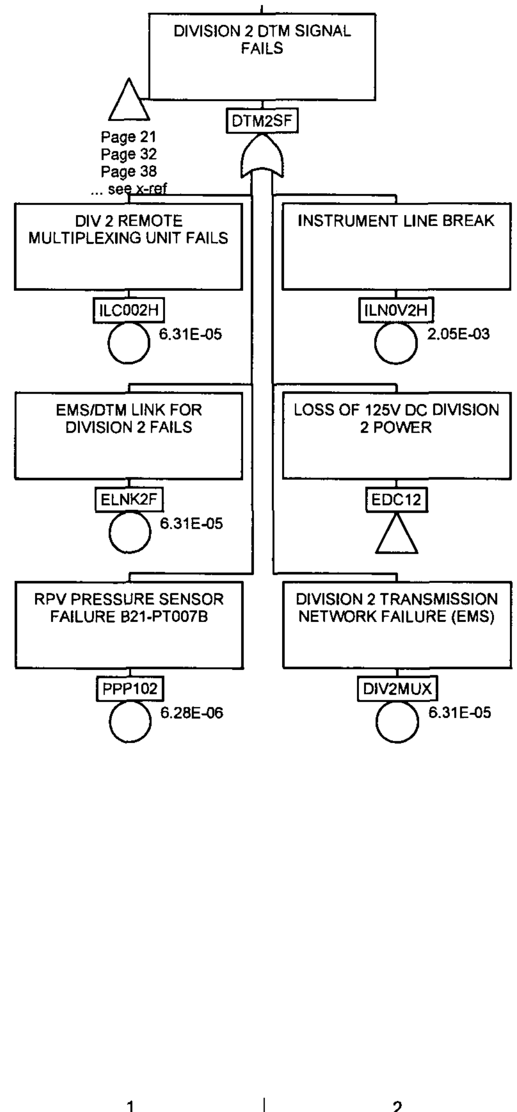


Figure 19D.6-24 Reactor Protection System Fault Tree

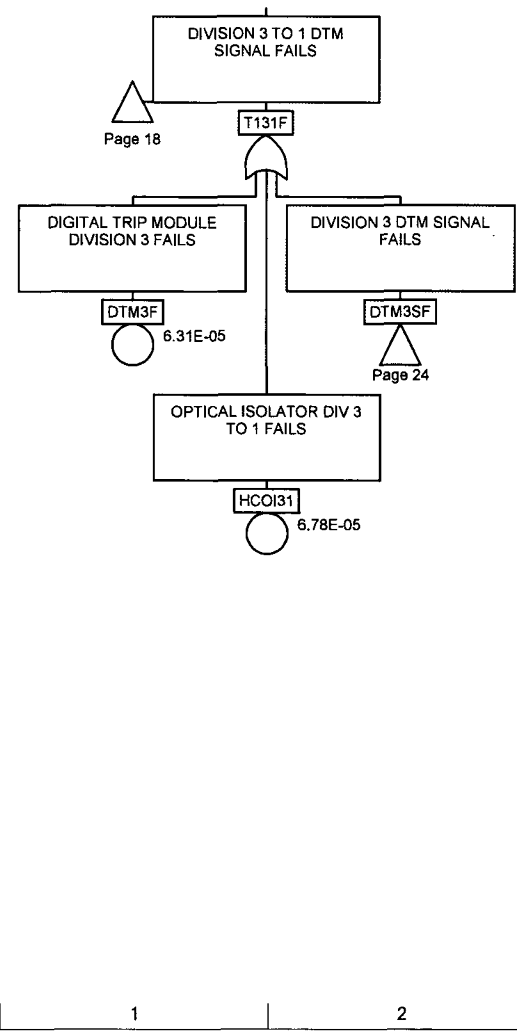
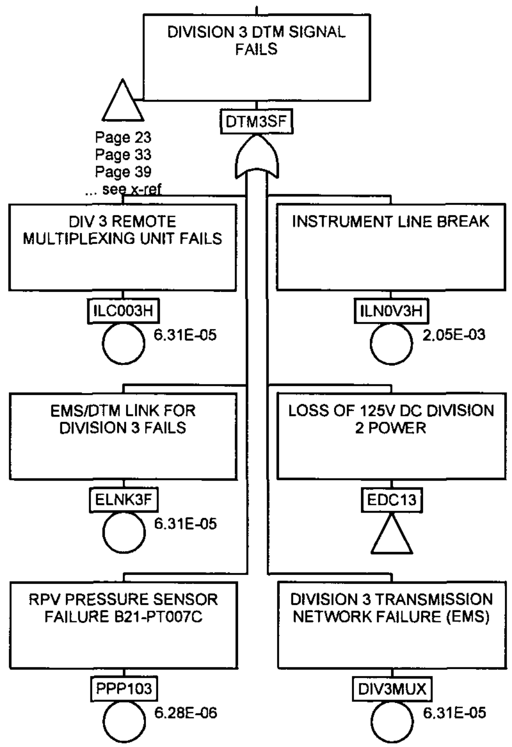


Figure 19D.6-24 Reactor Protection System Fault Tree



1	2

Figure 19D.6-24 Reactor Protection System Fault Tree

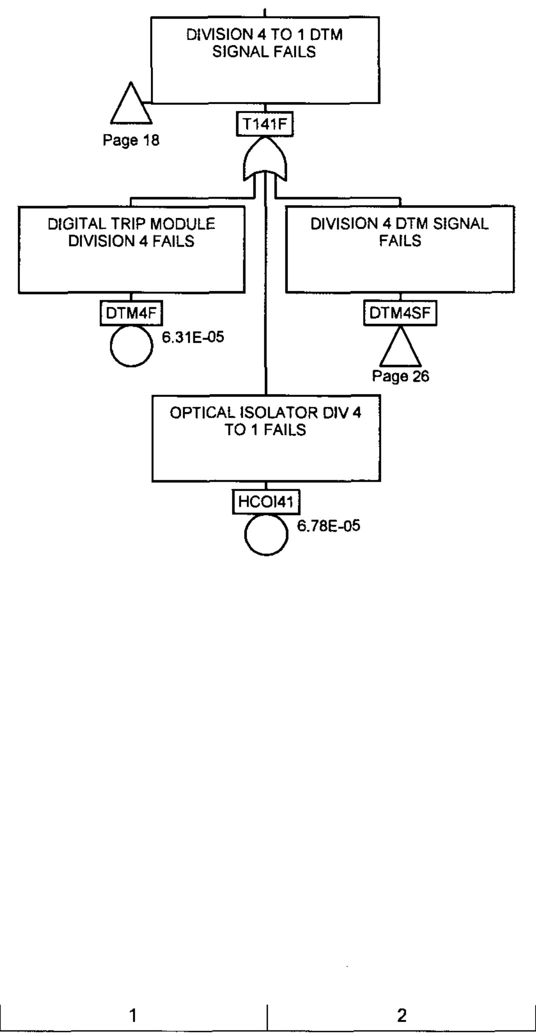


Figure 19D.6-24 Reactor Protection System Fault Tree

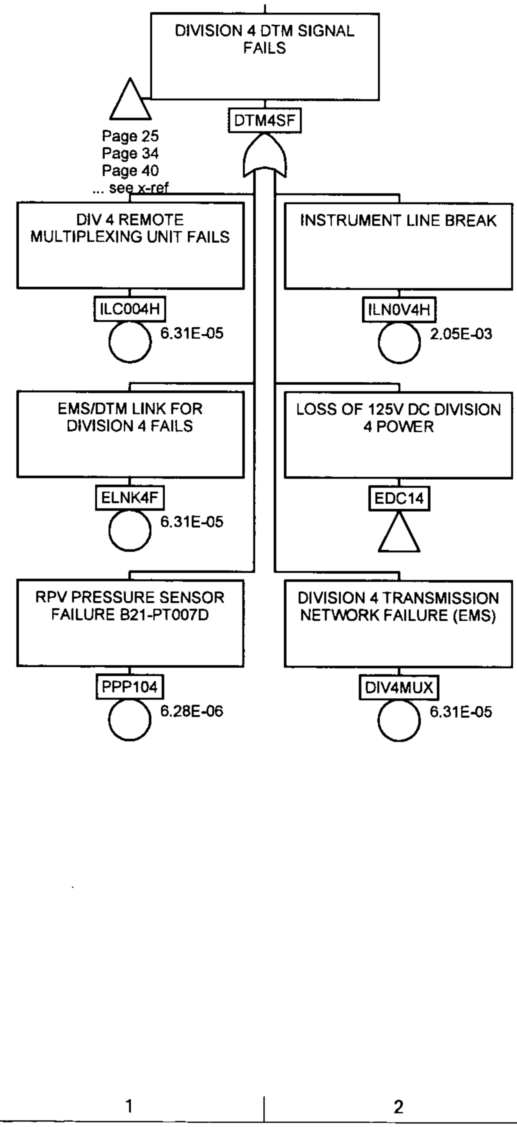


Figure 19D.6-24 Reactor Protection System Fault Tree

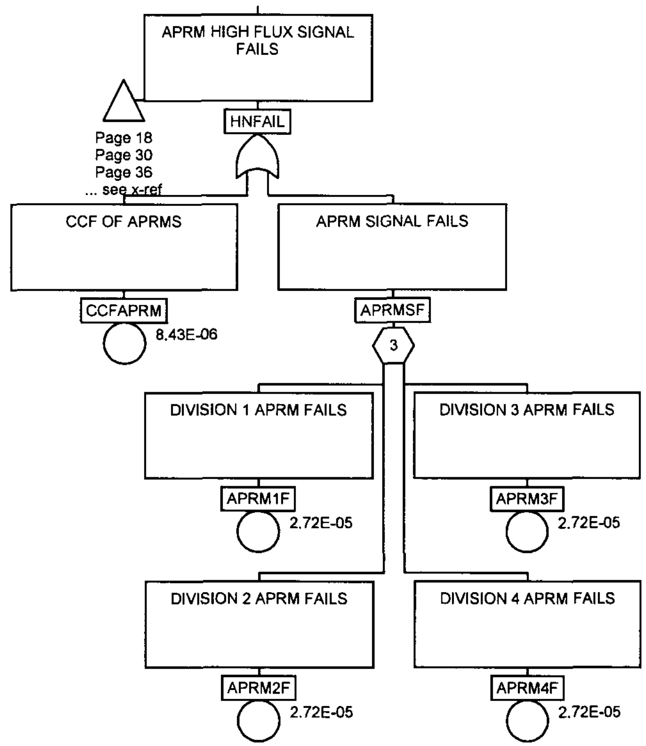


Figure 19D.6-24 Reactor Protection System Fault Tree

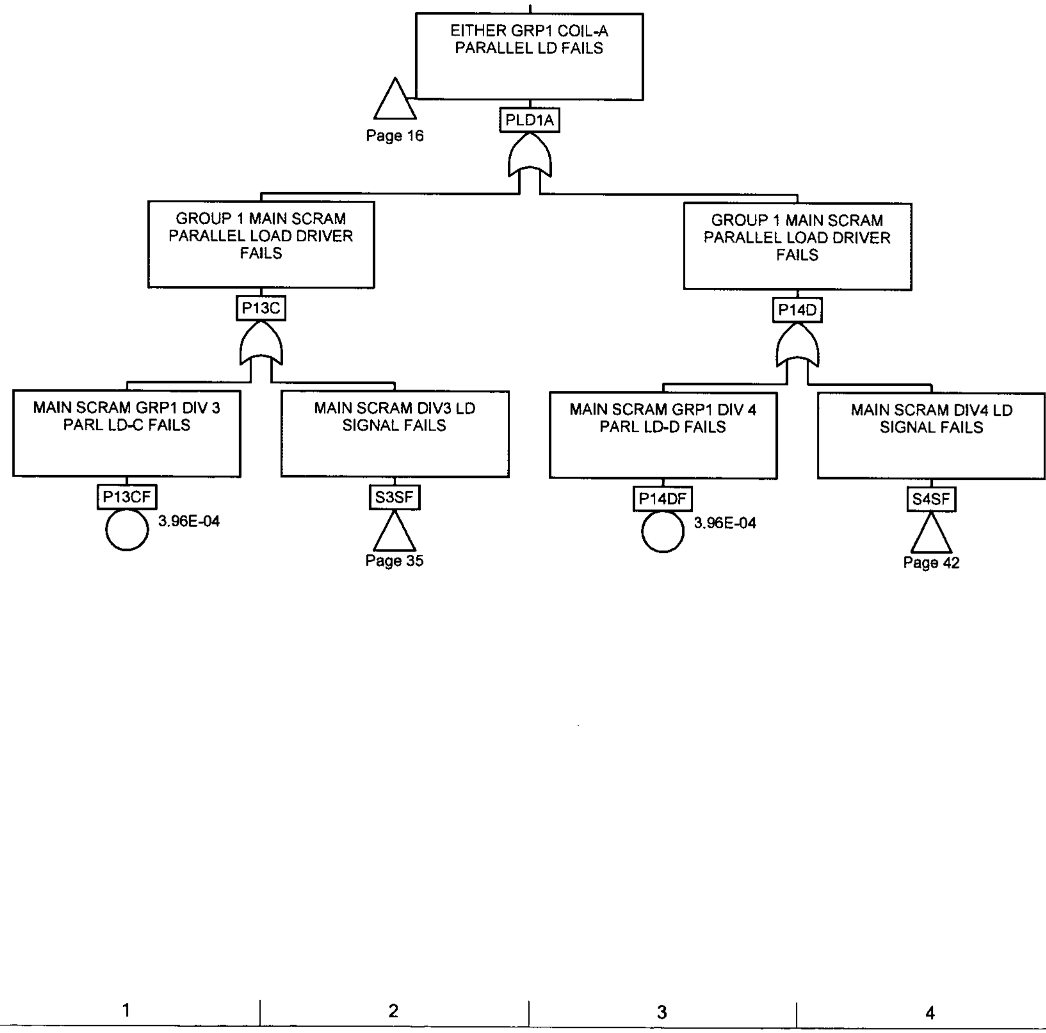


Figure 19D.6-24 Reactor Protection System Fault Tree

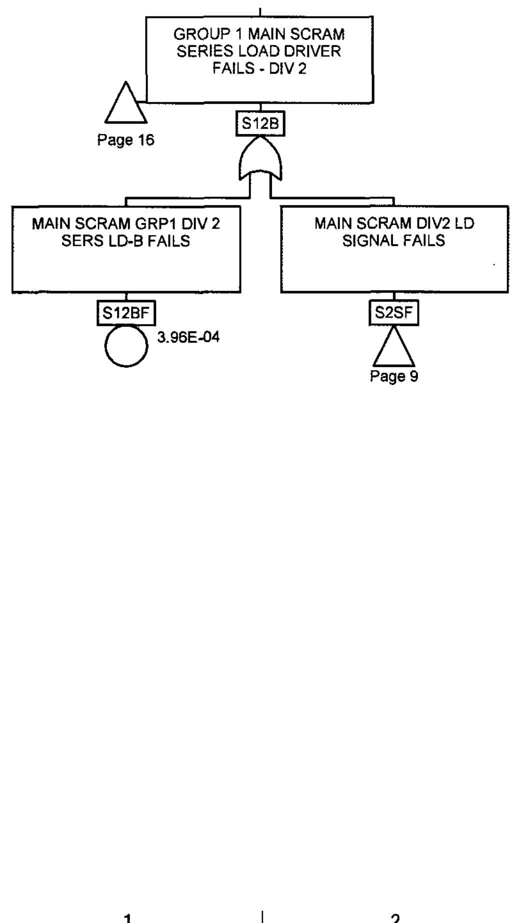


Figure 19D.6-24 Reactor Protection System Fault Tree

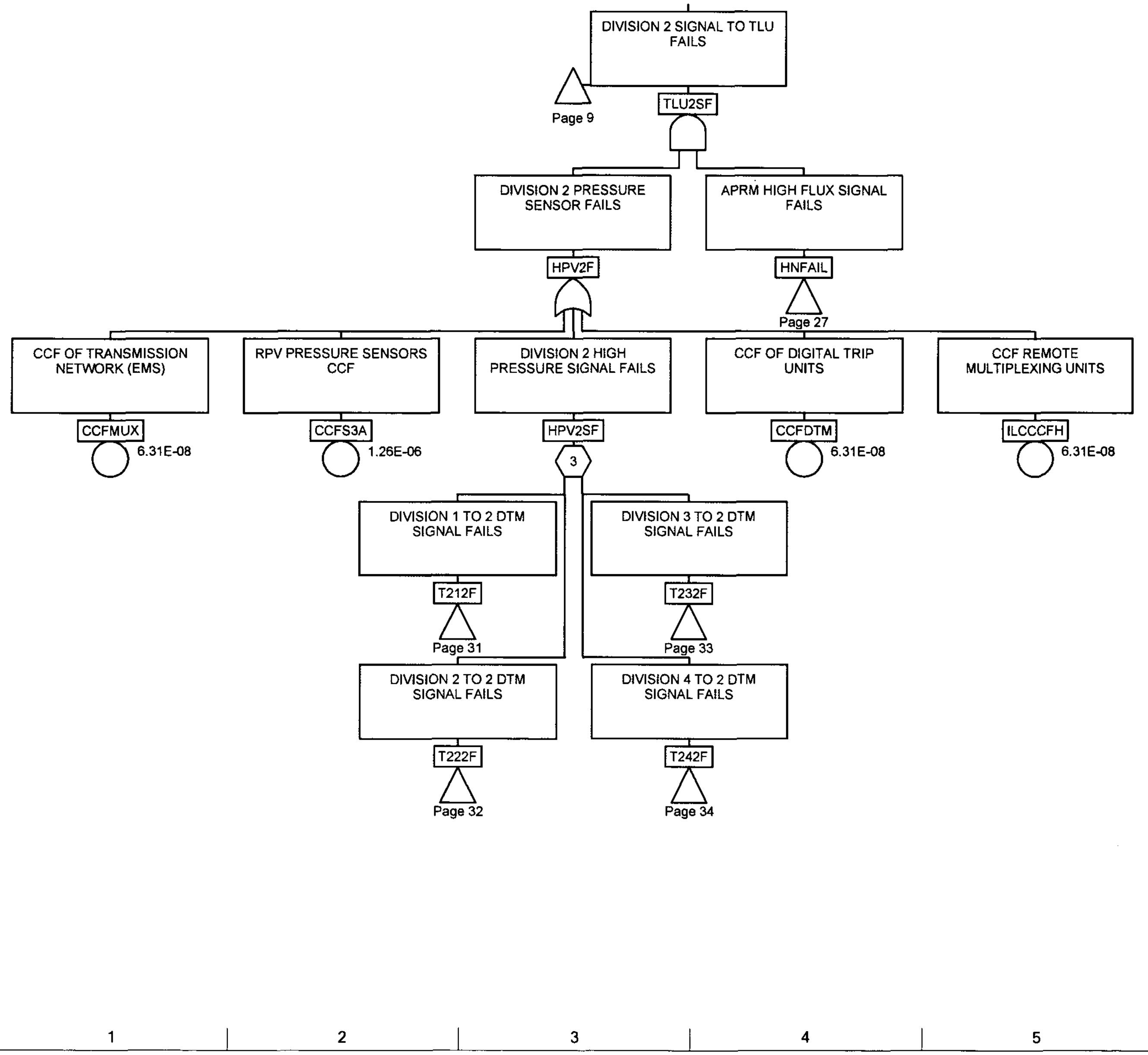


Figure 19D.6-24 Reactor Protection System Fault Tree

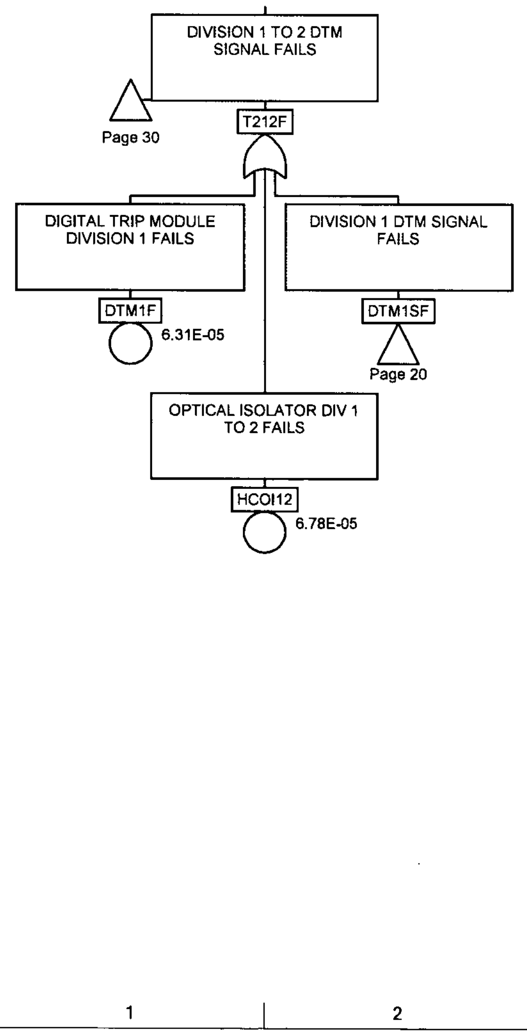
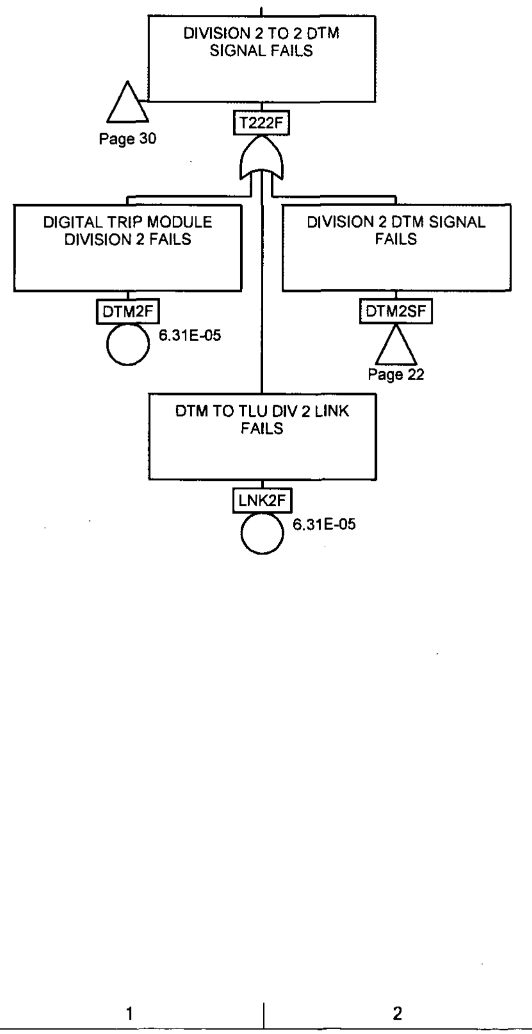


Figure 19D.6-24 Reactor Protection System Fault Tree


DIVISION 2 TO 2 DTM SIGNAL FAILS
Page 30
T222F
DIGITAL TRIP MODULE DIVISION 2 FAILS
DIVISION 2 DTM SIGNAL FAILS
DTM2F
6.31E-05
DTM2SF
Page 22
DTM TO TLU DIV 2 LINK FAILS
LNK2F
6.31E-05
1
2

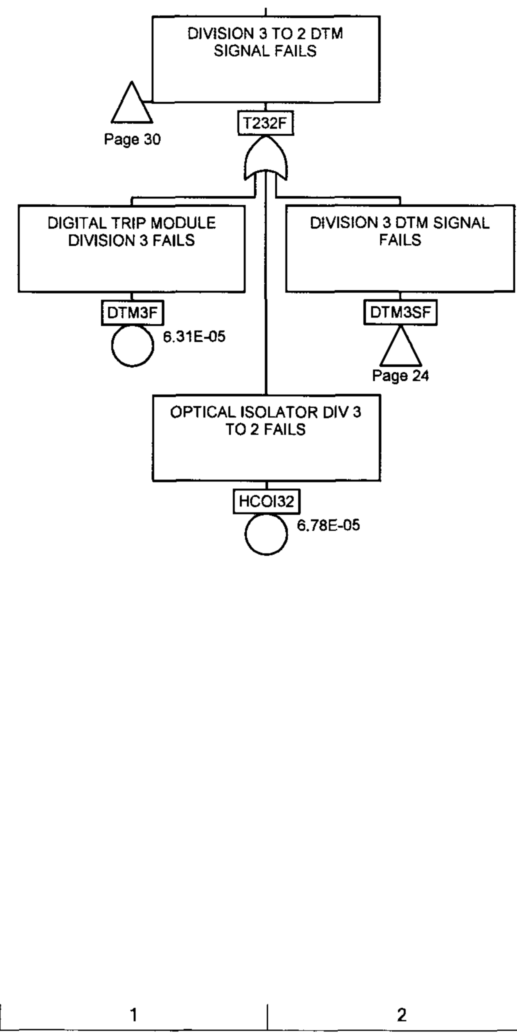


Figure 19D.6-24 Reactor Protection System Fault Tree

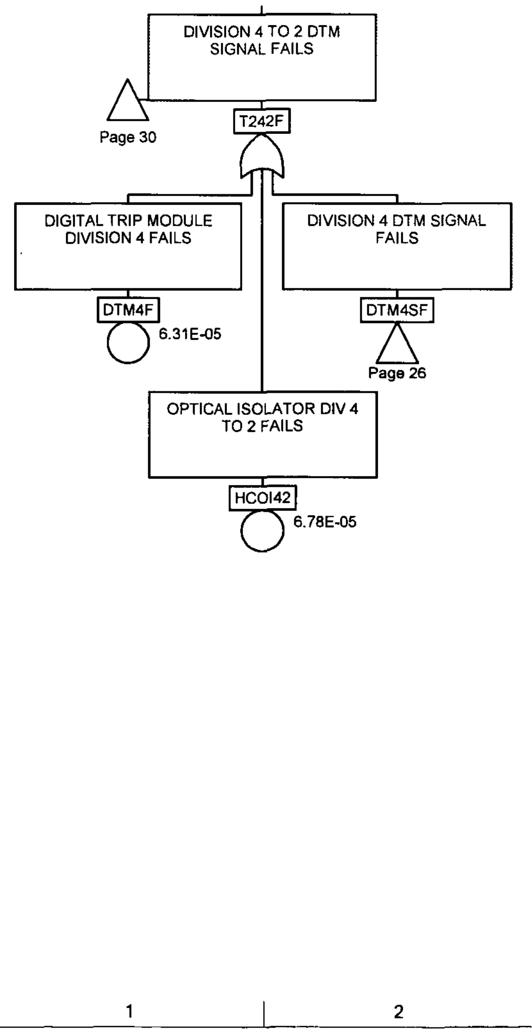


Figure 19D.6-24 Reactor Protection System Fault Tree

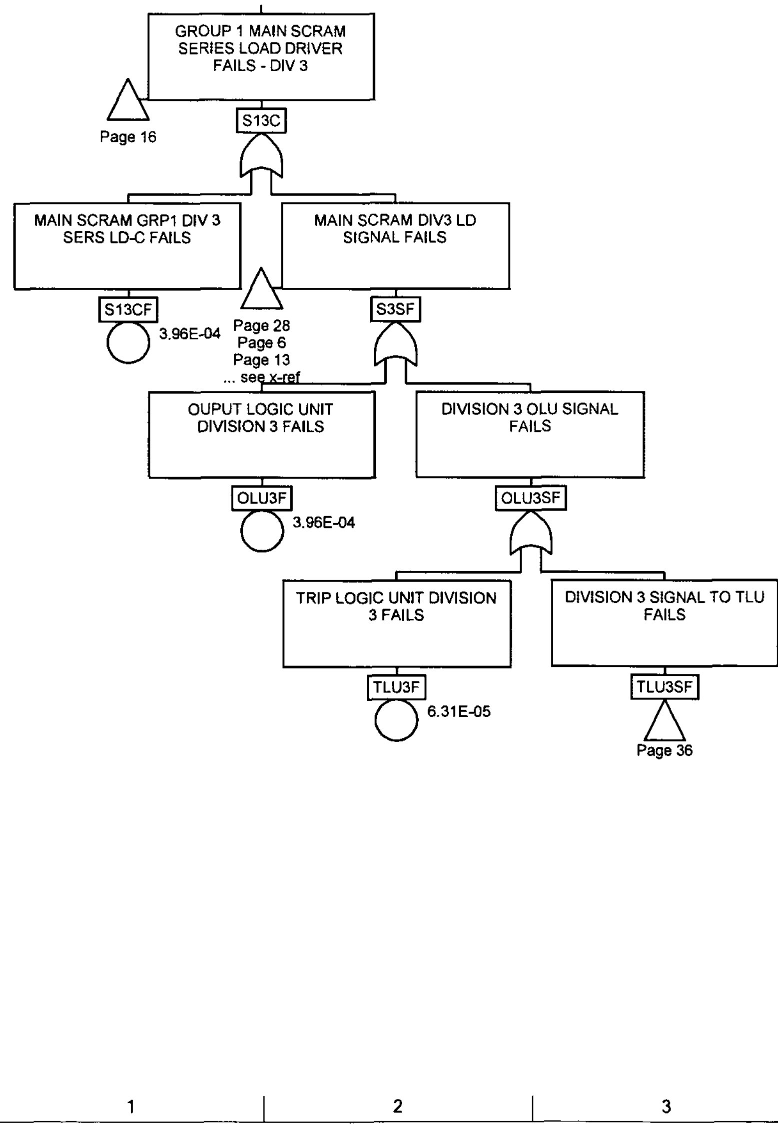


Figure 19D.6-24 Reactor Protection System Fault Tree

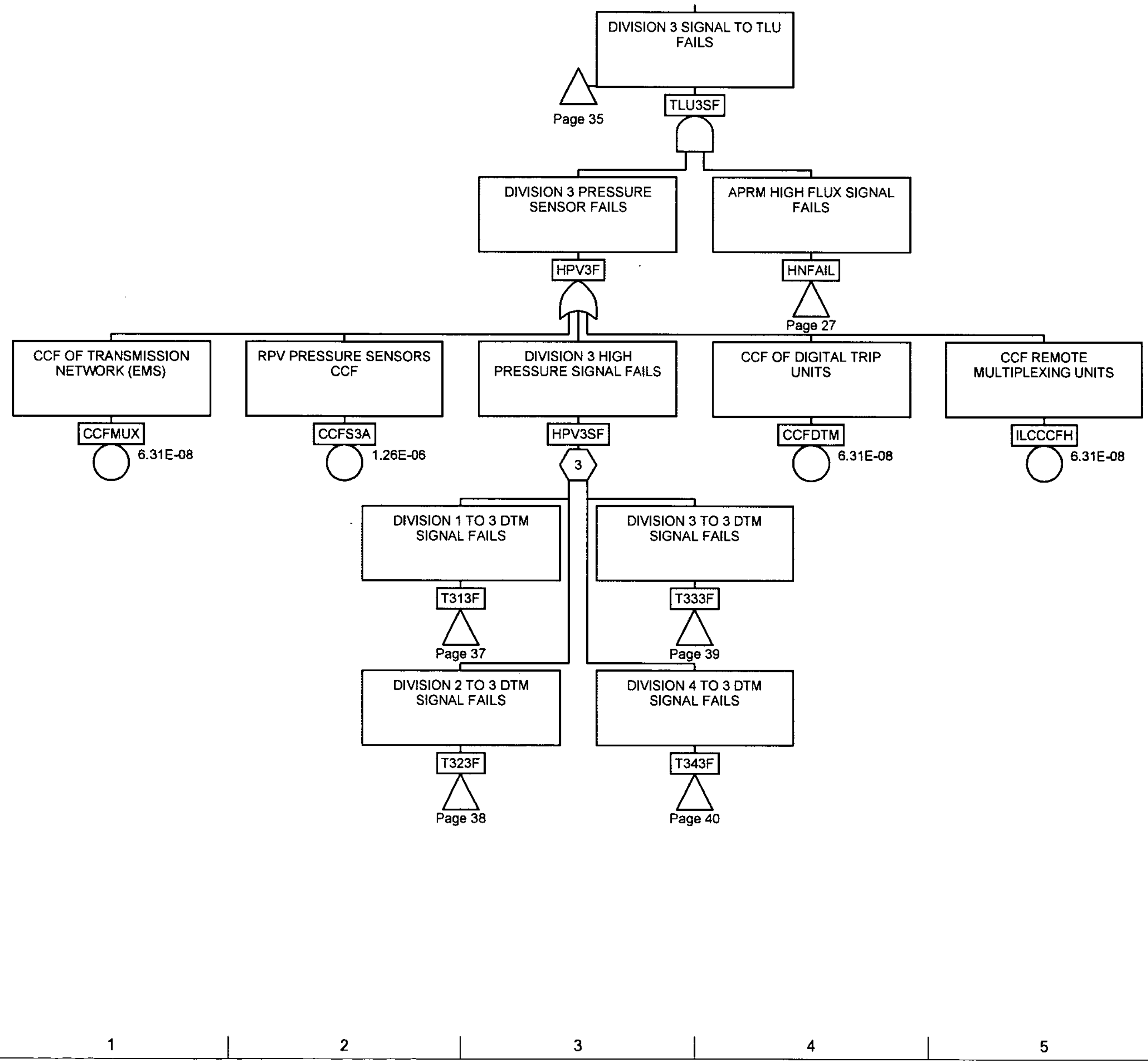


Figure 19D.6-24 Reactor Protection System Fault Tree

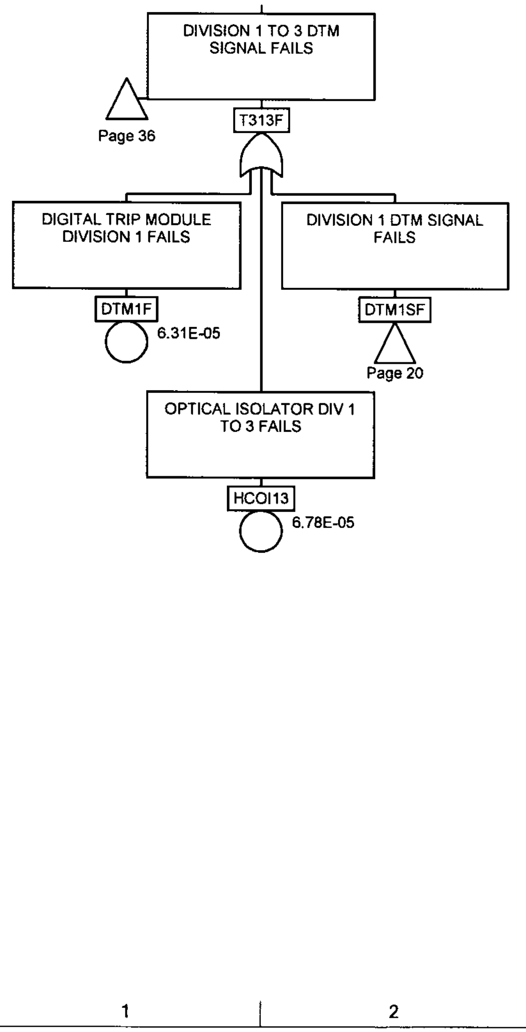


Figure 19D.6-24 Reactor Protection System Fault Tree

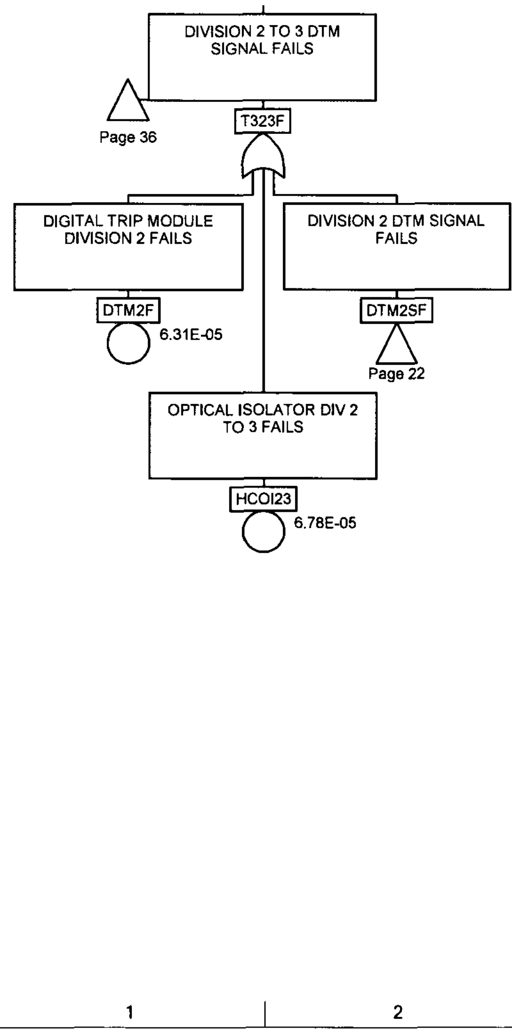


Figure 19D.6-24 Reactor Protection System Fault Tree

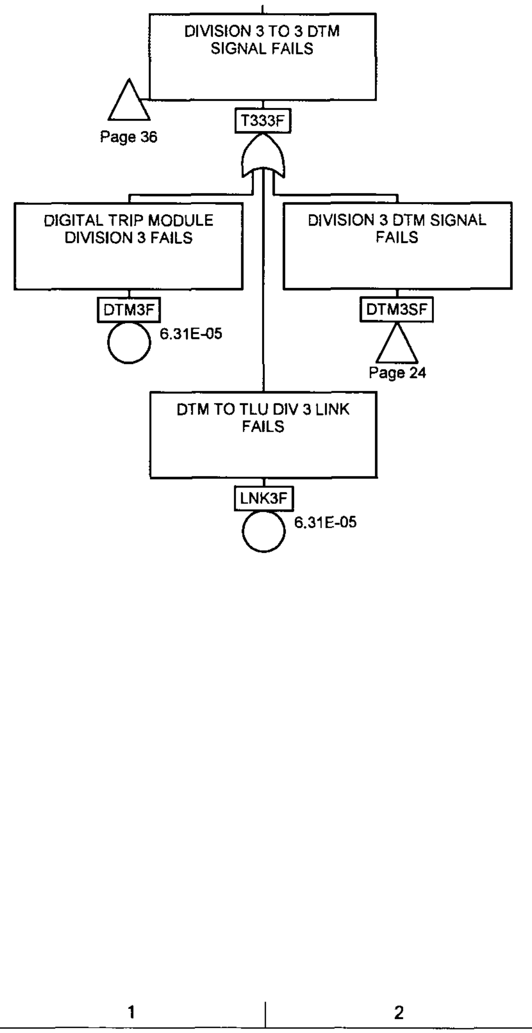


Figure 19D.6-24 Reactor Protection System Fault Tree

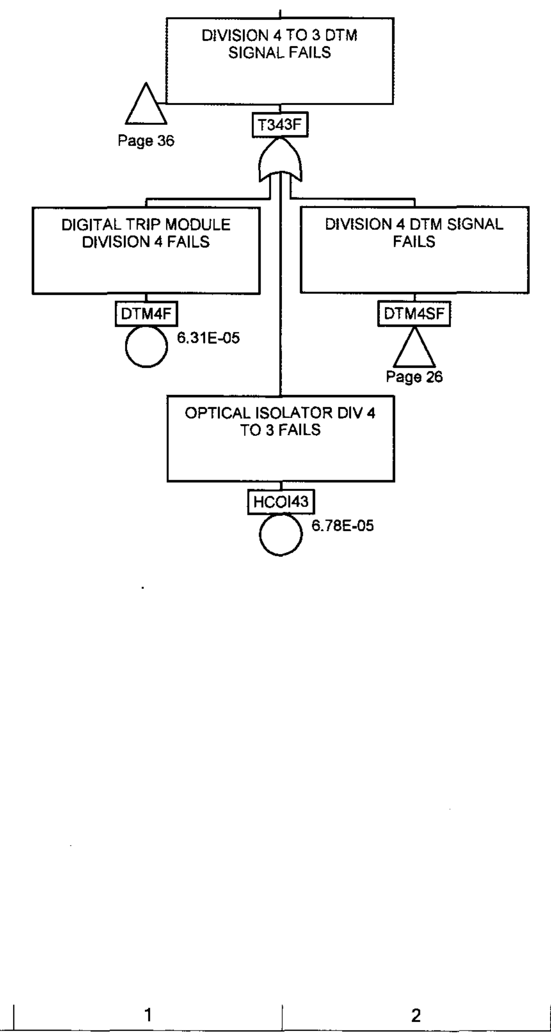


Figure 19D.6-24 Reactor Protection System Fault Tree

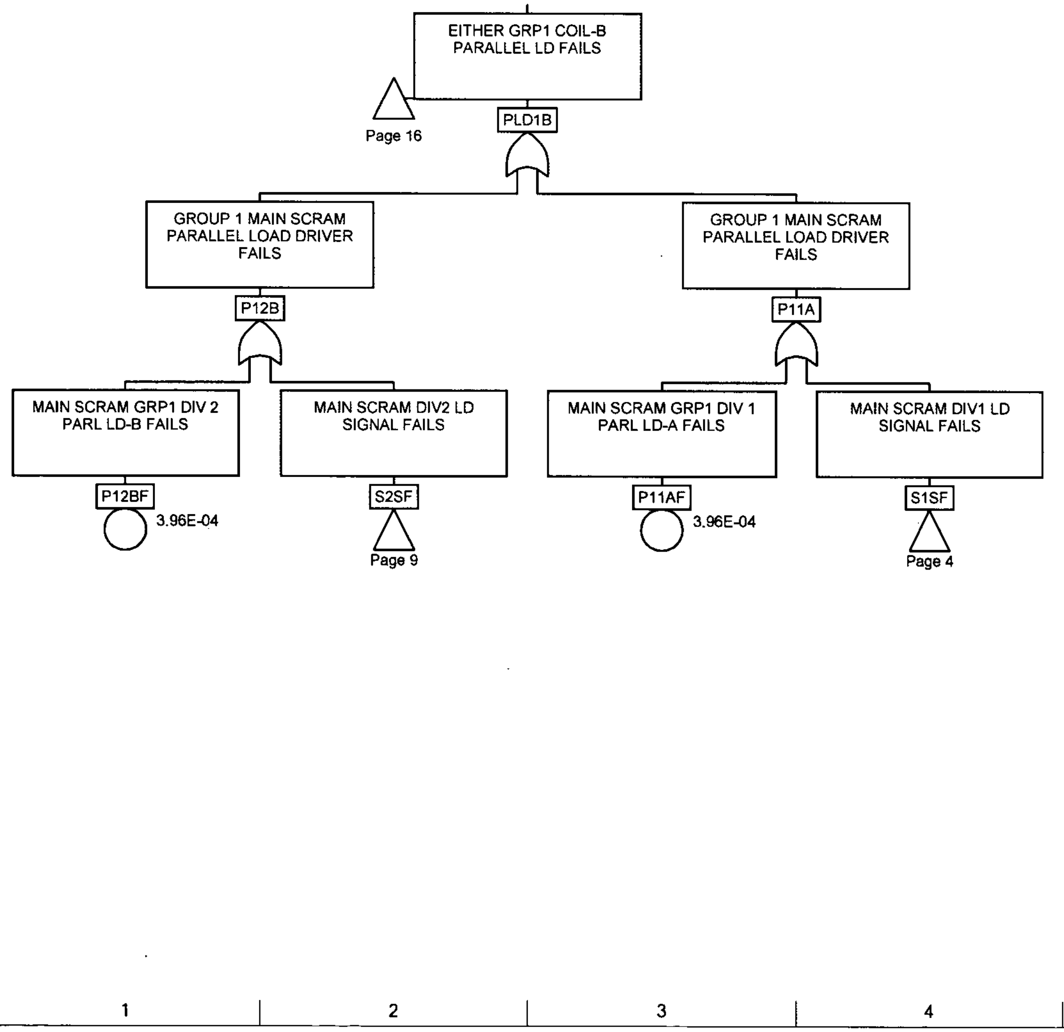


Figure 19D.6-24 Reactor Protection System Fault Tree

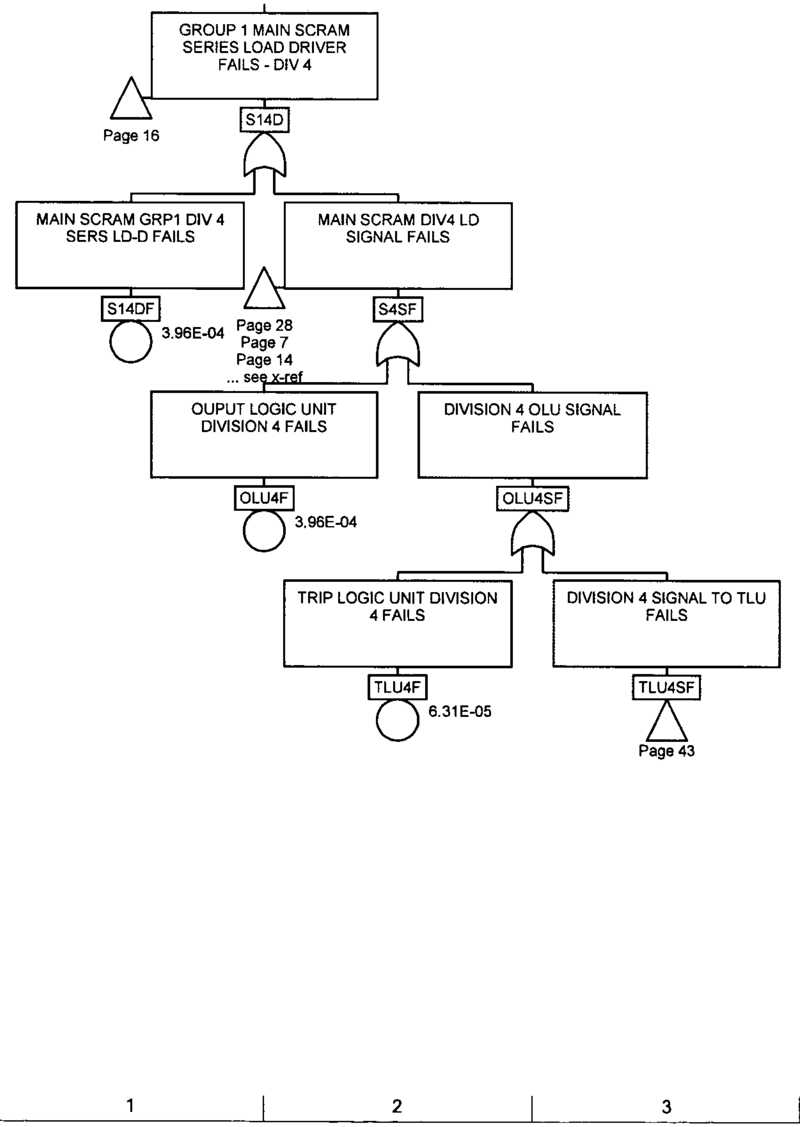


Figure 19D.6-24 Reactor Protection System Fault Tree

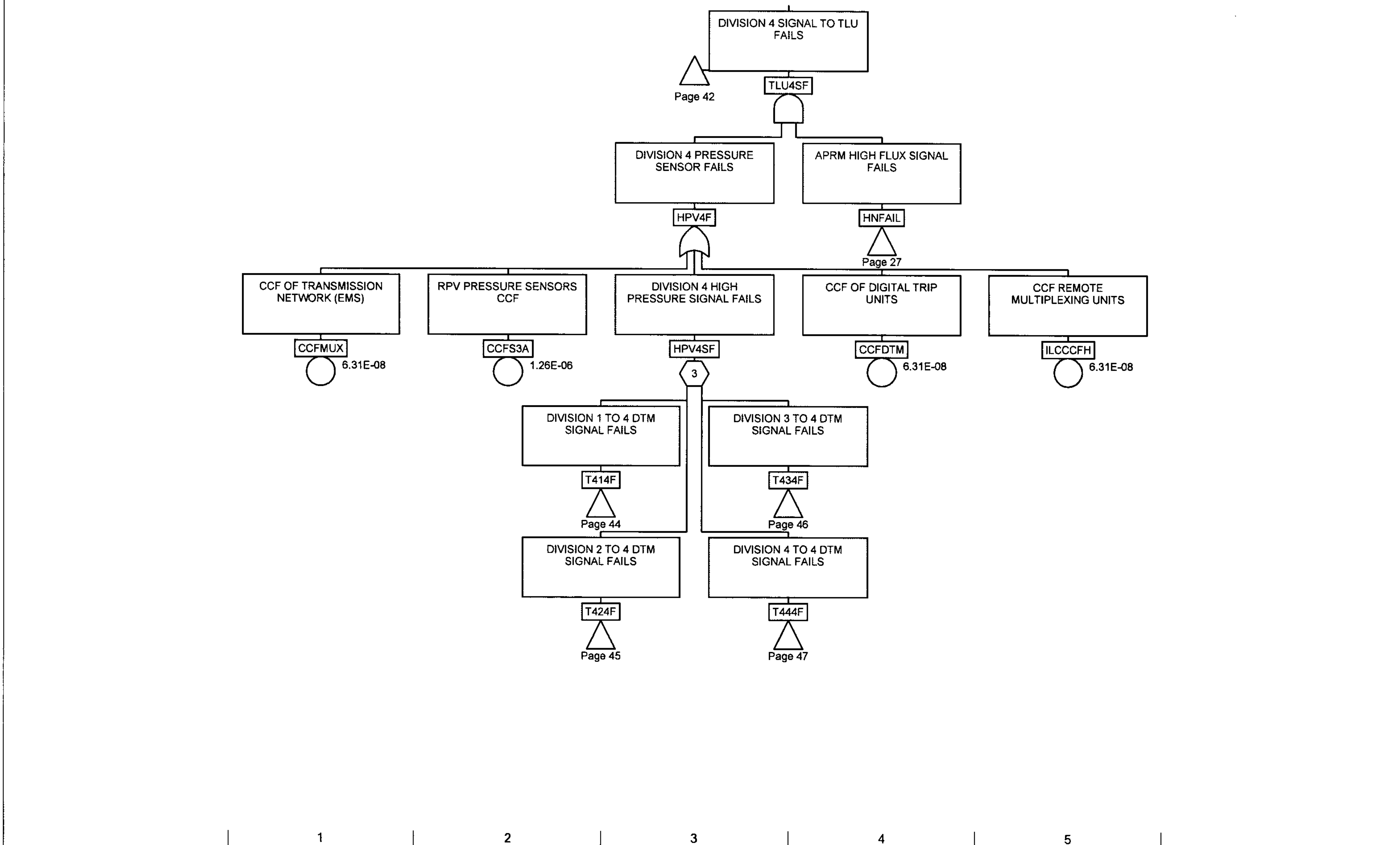


Figure 19D.6-24 Reactor Protection System Fault Tree

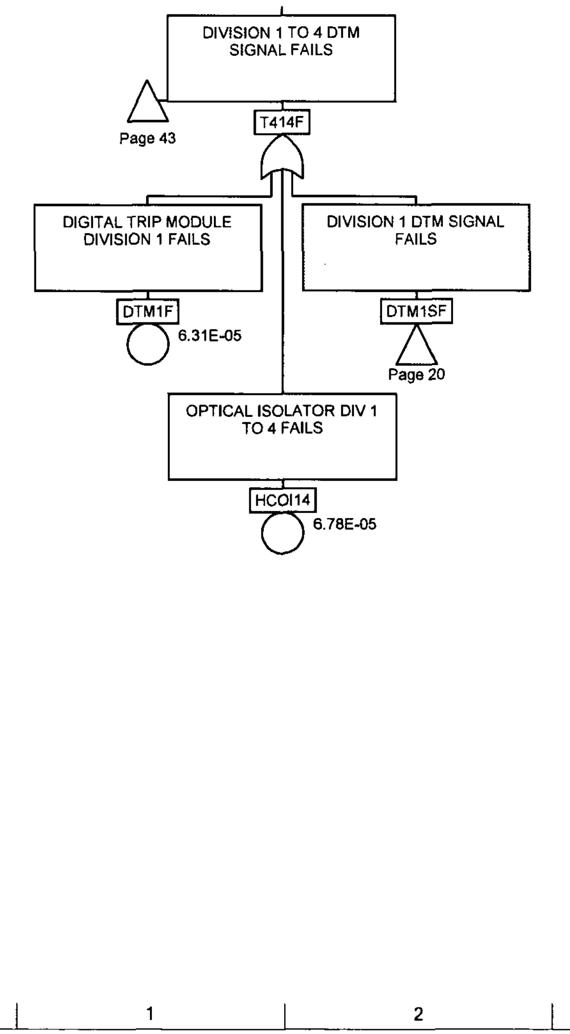


Figure 19D.6-24 Reactor Protection System Fault Tree

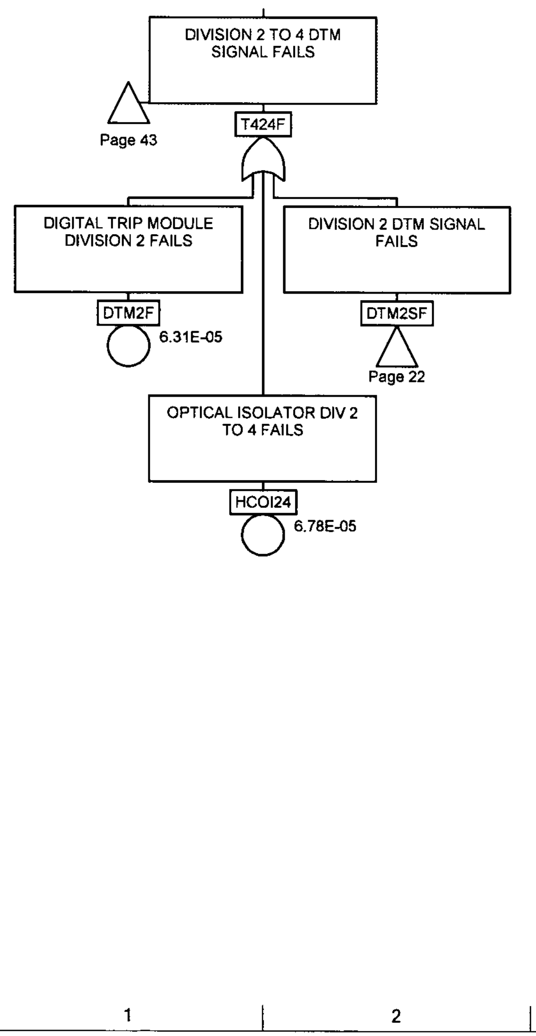
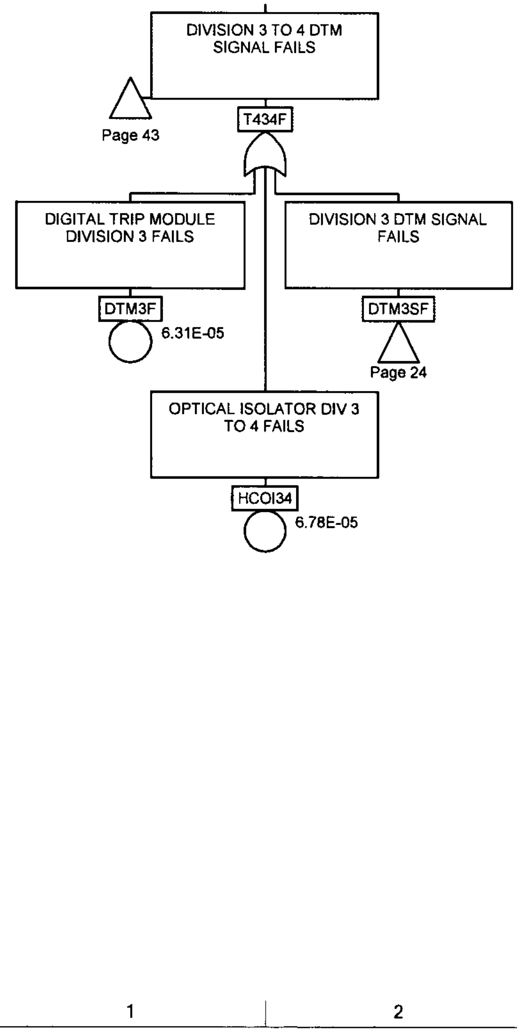


Figure 19D.6-24 Reactor Protection System Fault Tree


DIVISION 3 TO 4 DTM SIGNAL FAILS
Page 43
T434F
DIGITAL TRIP MODULE DIVISION 3 FAILS
DIVISION 3 DTM SIGNAL FAILS
DTM3F
6.31E-05
DTM3SF
Page 24
OPTICAL ISOLATOR DIV 3 TO 4 FAILS
HCOI34
6.78E-05
1
2

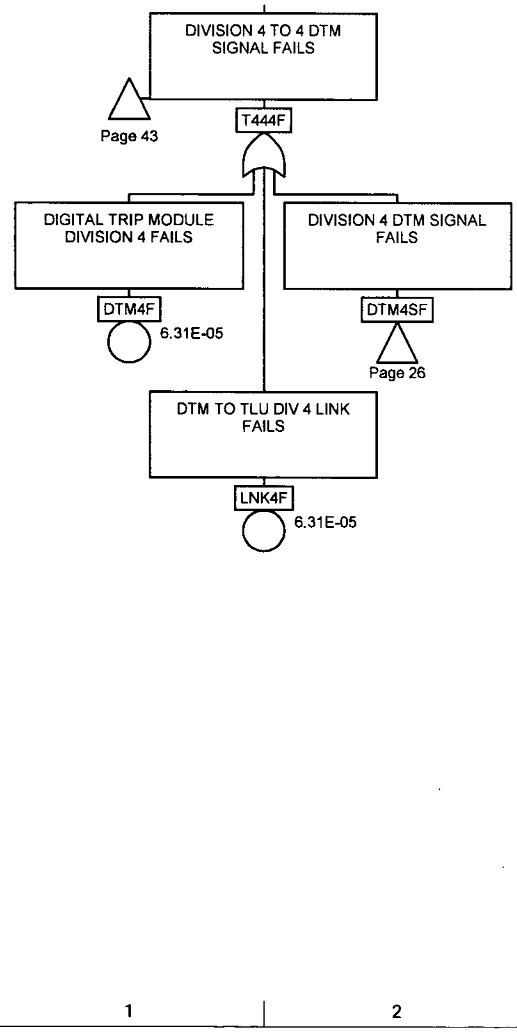


Figure 19D.6-24 Reactor Protection System Fault Tree

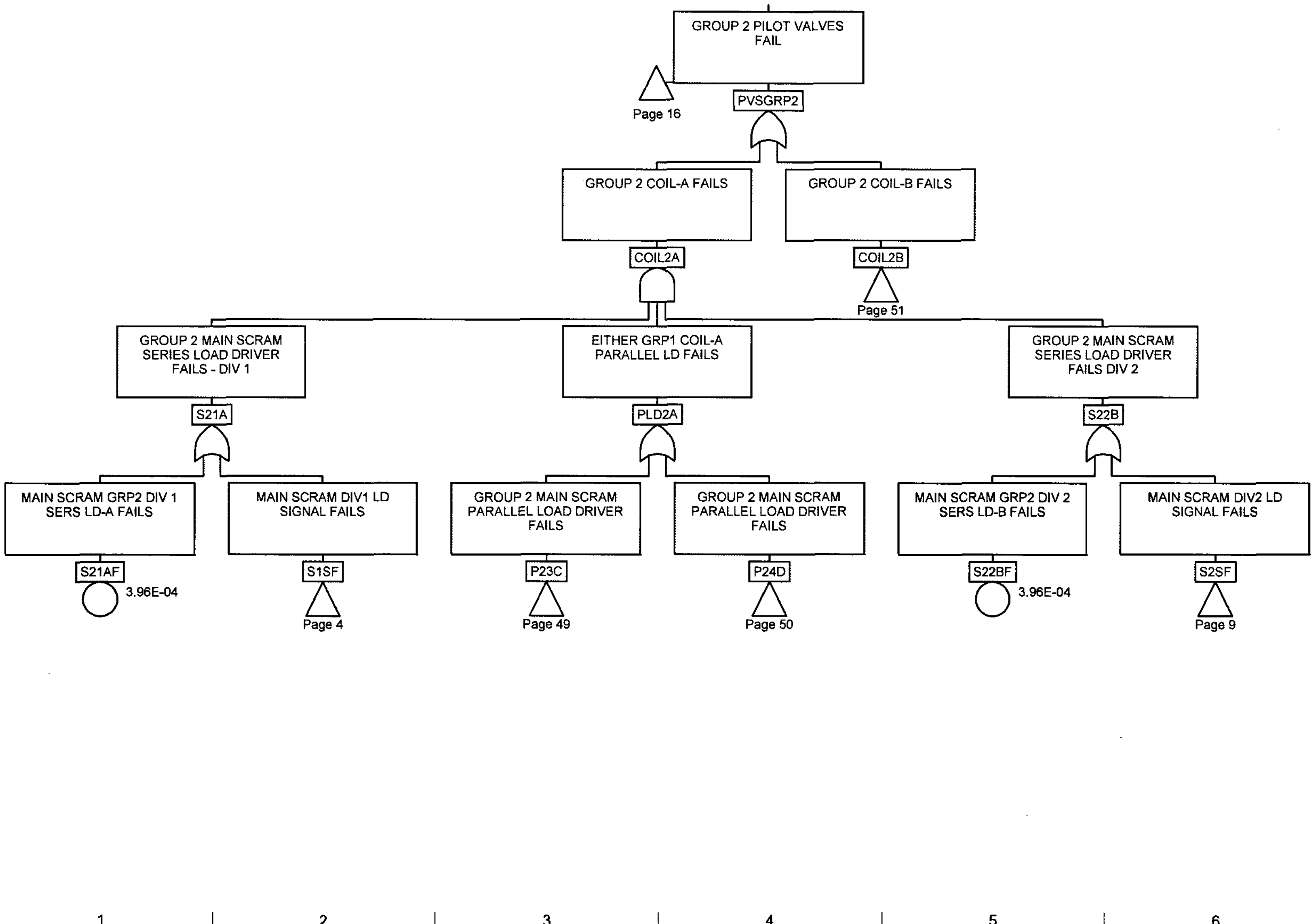


Figure 19D.6-24 Reactor Protection System Fault Tree

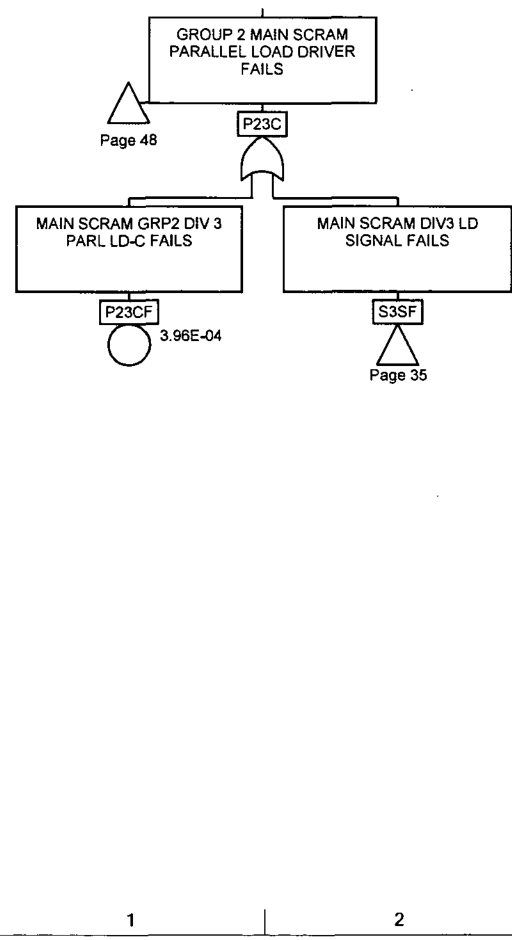
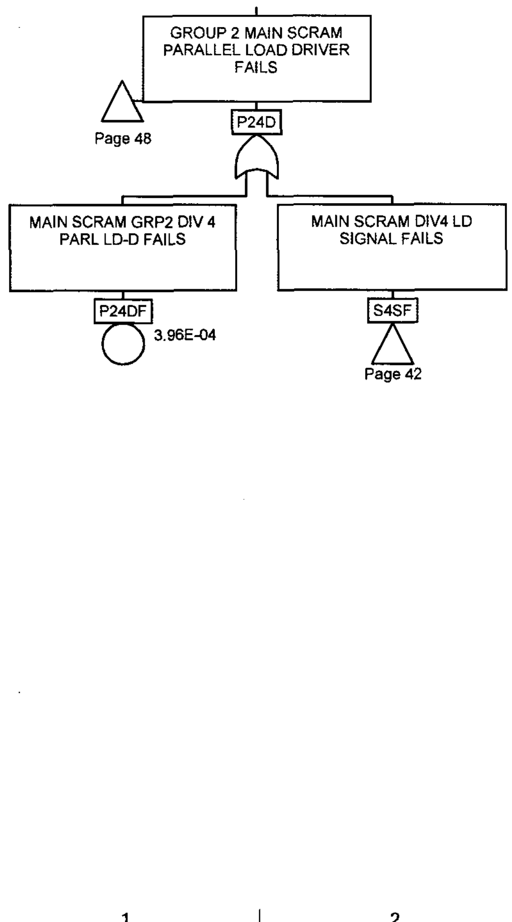


Figure 19D.6-24 Reactor Protection System Fault Tree



GROUP 2 MAIN SCRAM PARALLEL LOAD DRIVER FAILS
Page 48
P24D
MAIN SCRAM GRP2 DIV 4 PARL LD-D FAILS
MAIN SCRAM DIV4 LD SIGNAL FAILS
P24DF
3.96E-04
S4SF
Page 42
1
2

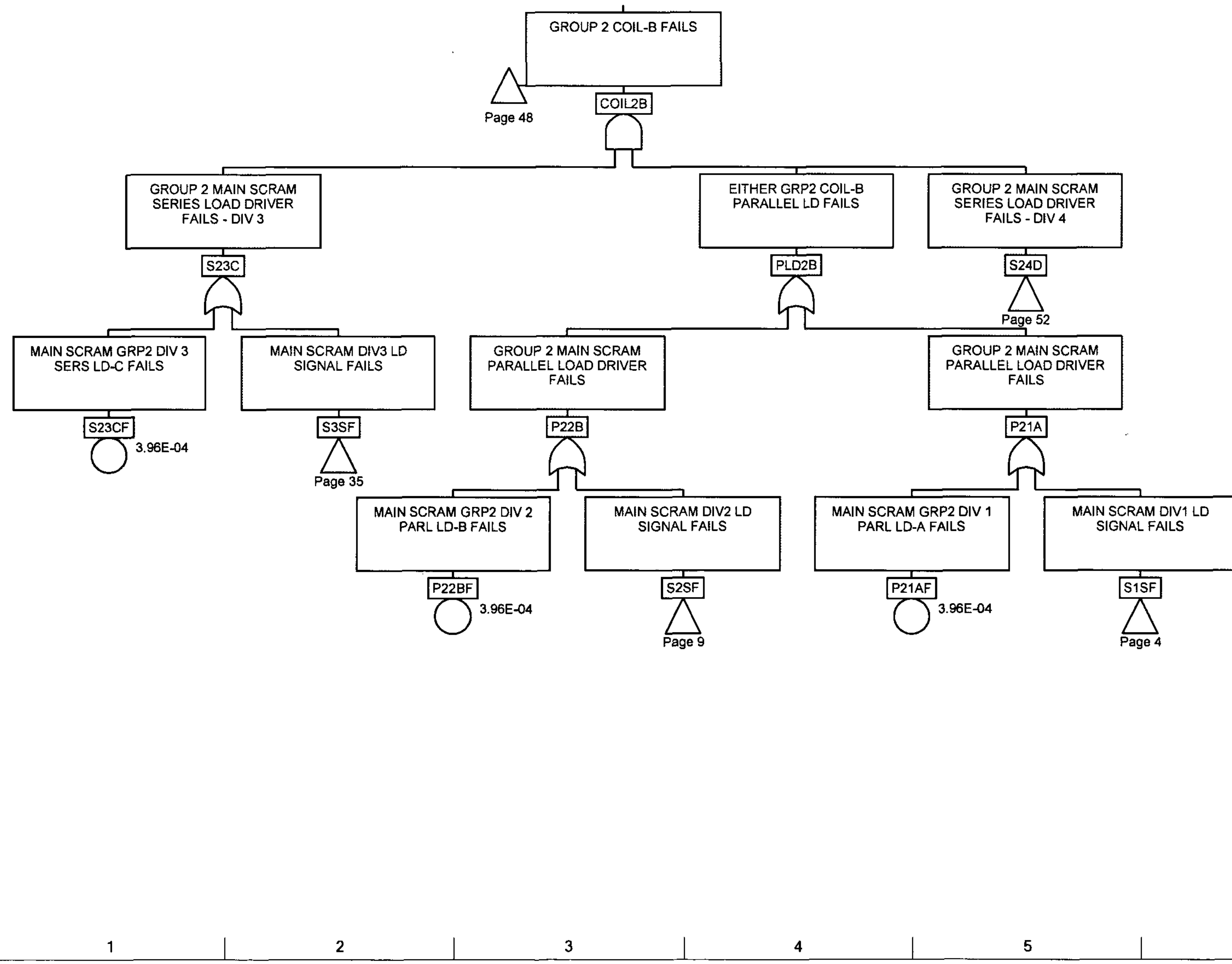


Figure 19D.6-24 Reactor Protection System Fault Tree

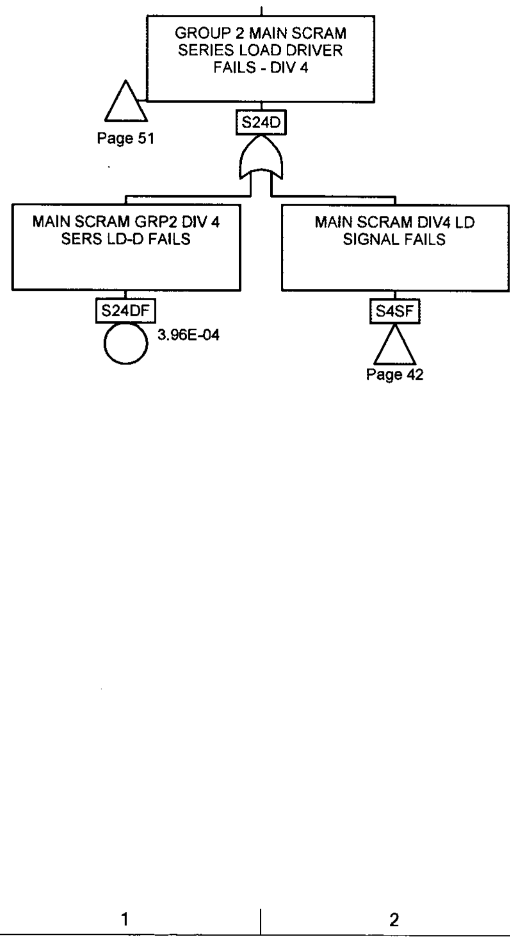


Figure 19D.6-24 Reactor Protection System Fault Tree

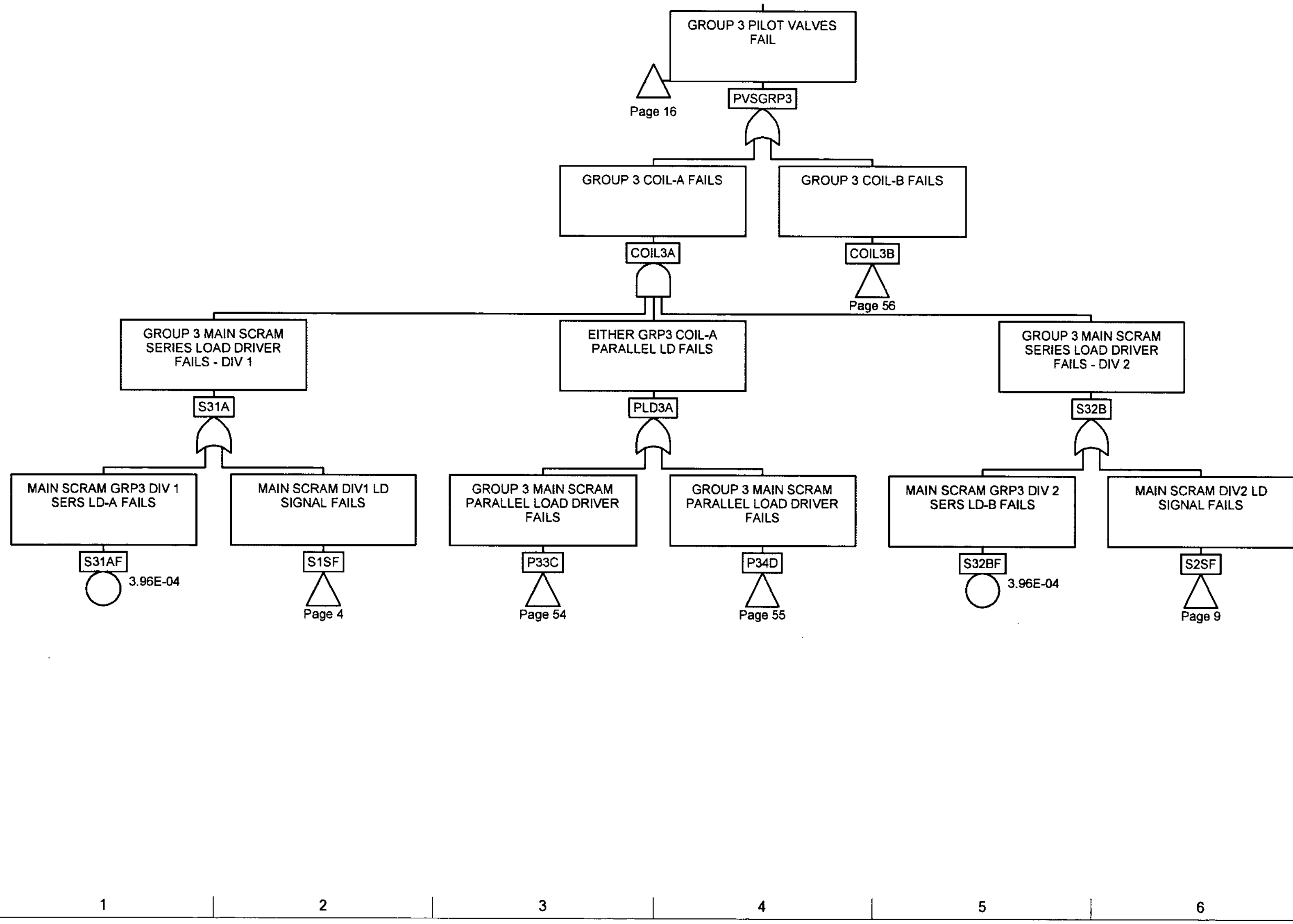


Figure 19D.6-24 Reactor Protection System Fault Tree

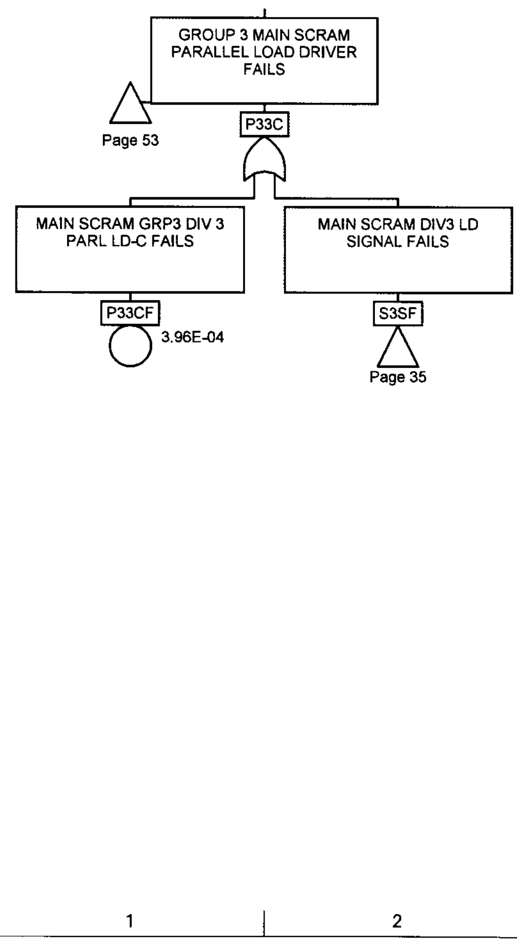


Figure 19D.6-24 Reactor Protection System Fault Tree

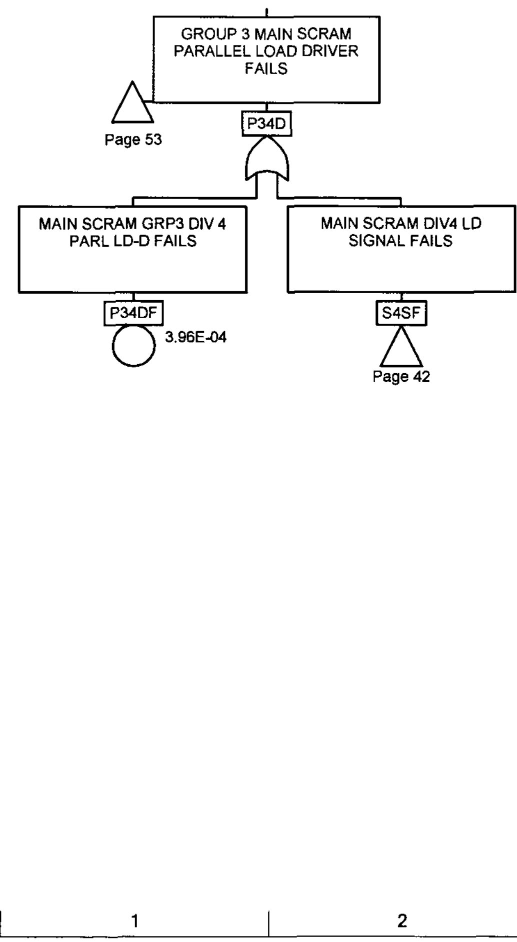
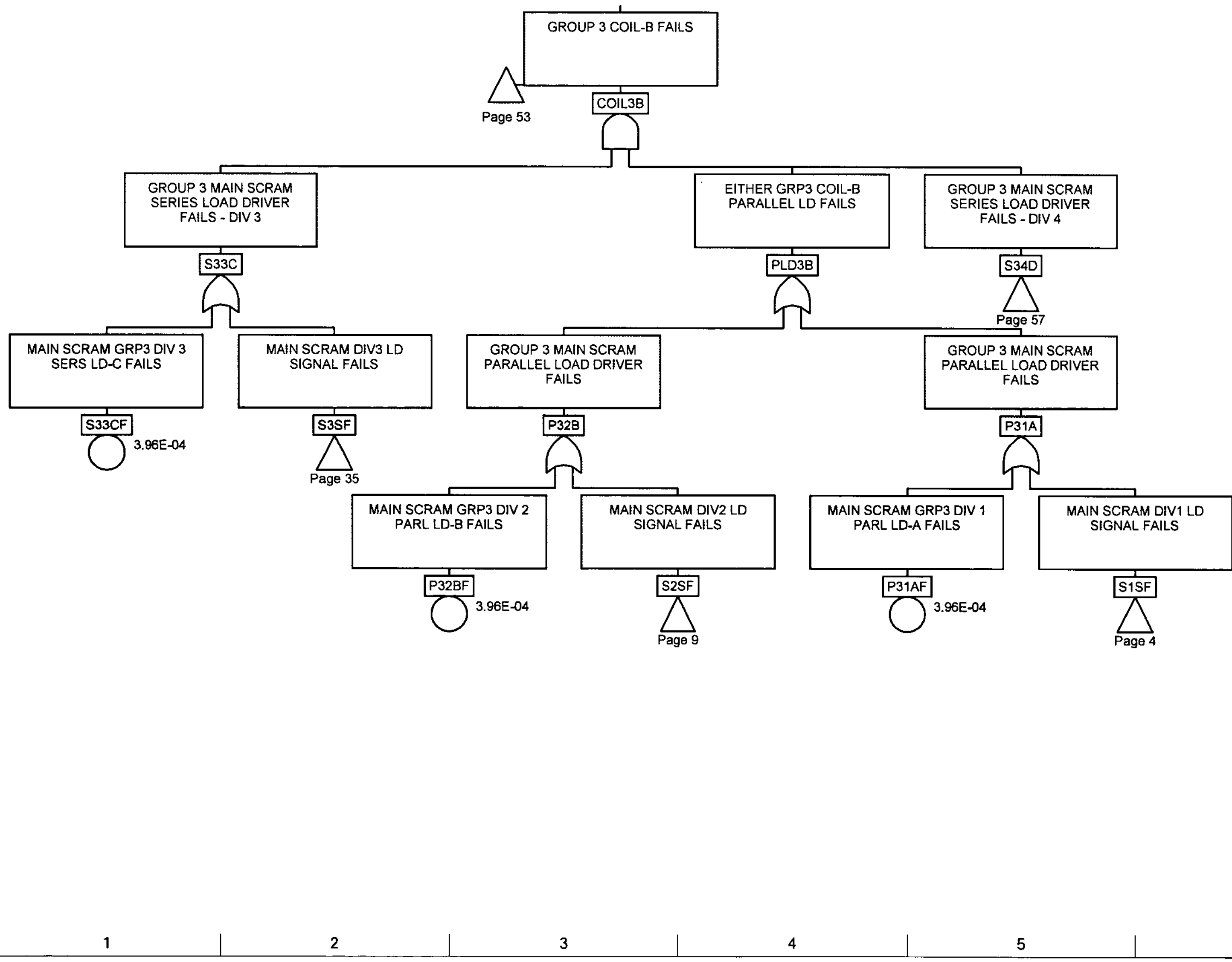


Figure 19D.6-24 Reactor Protection System Fault Tree


GROUP 3 COIL-B FAILS
Page 53
COIL3B
GROUP 3 MAIN SCRAM SERIES LOAD DRIVER FAILS - DIV 3
S33C
EITHER GRP3 COIL-B PARALLEL LD FAILS
PLD3B
GROUP 3 MAIN SCRAM SERIES LOAD DRIVER FAILS - DIV 4
S34D
Page 57
MAIN SCRAM GRP3 DIV 3 SERS LD-C FAILS
S33CF
3.96E-04
MAIN SCRAM DIV3 LD SIGNAL FAILS
S3SF
Page 35
GROUP 3 MAIN SCRAM PARALLEL LOAD DRIVER FAILS
P32B
GROUP 3 MAIN SCRAM PARALLEL LOAD DRIVER FAILS
P31A
MAIN SCRAM GRP3 DIV 2 PARL LD-B FAILS
P32BF
3.96E-04
MAIN SCRAM DIV2 LD SIGNAL FAILS
S2SF
Page 9
MAIN SCRAM GRP3 DIV 1 PARL LD-A FAILS
P31AF
3.96E-04
MAIN SCRAM DIV1 LD SIGNAL FAILS
S1SF
Page 4
1
2
3
4
5

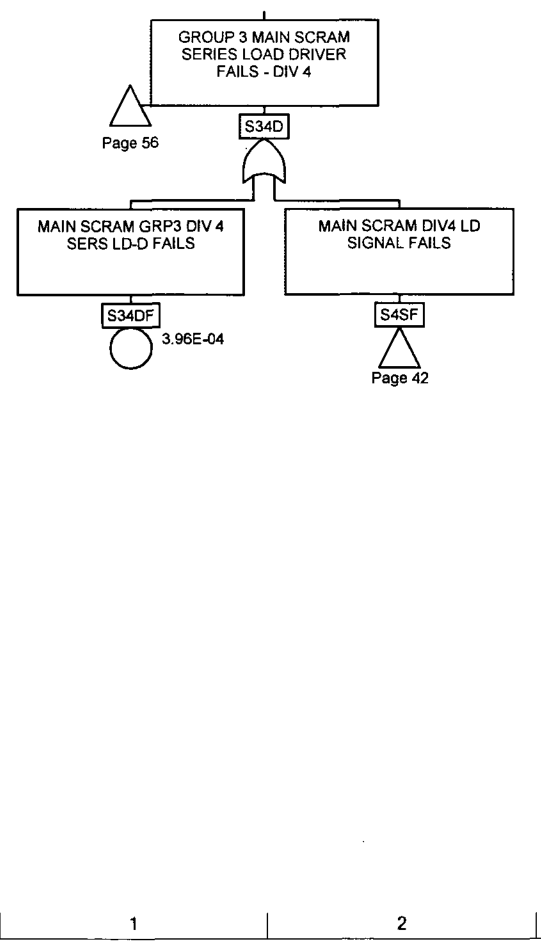


Figure 19D.6-24 Reactor Protection System Fault Tree

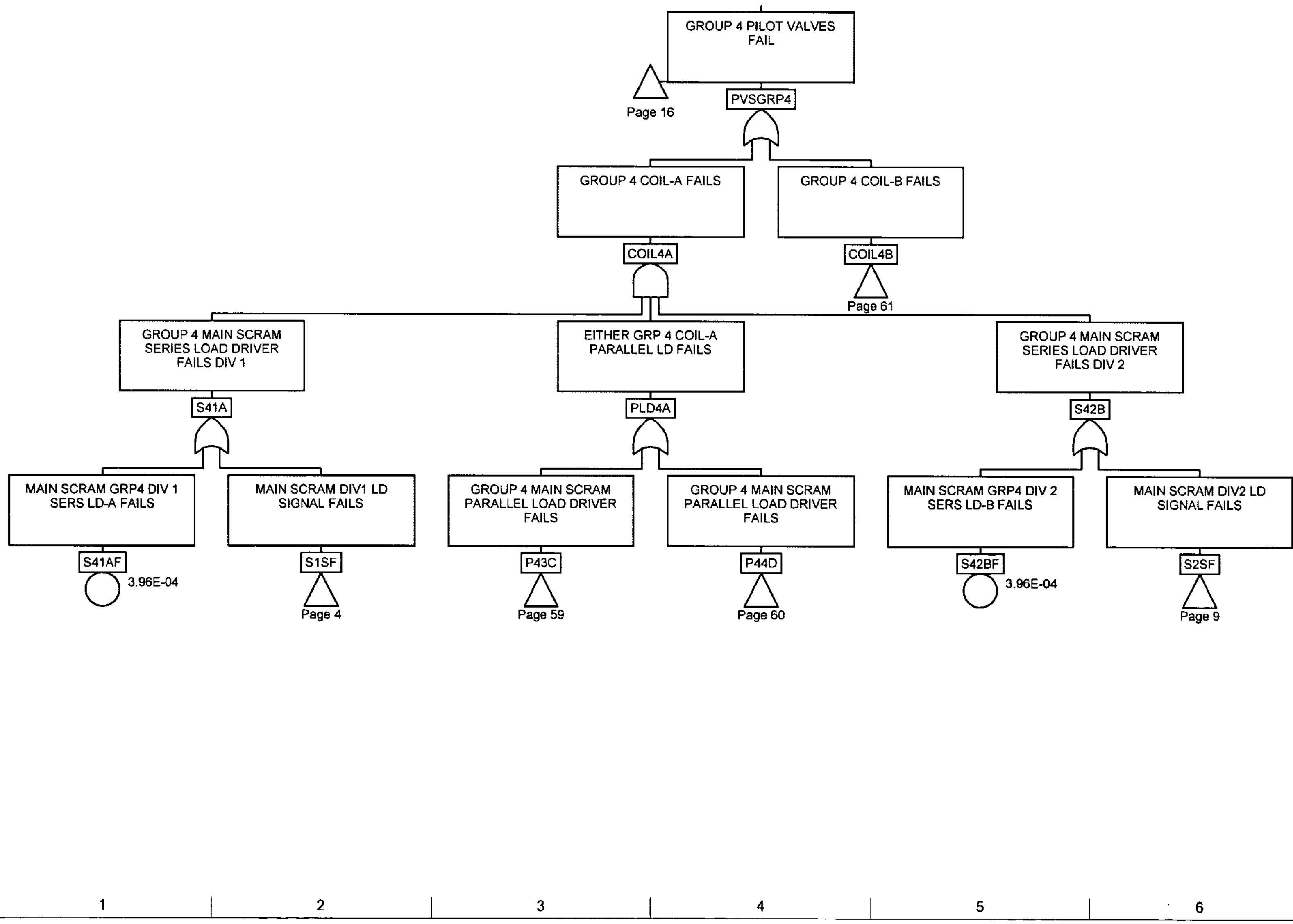


Figure 19D.6-24 Reactor Protection System Fault Tree

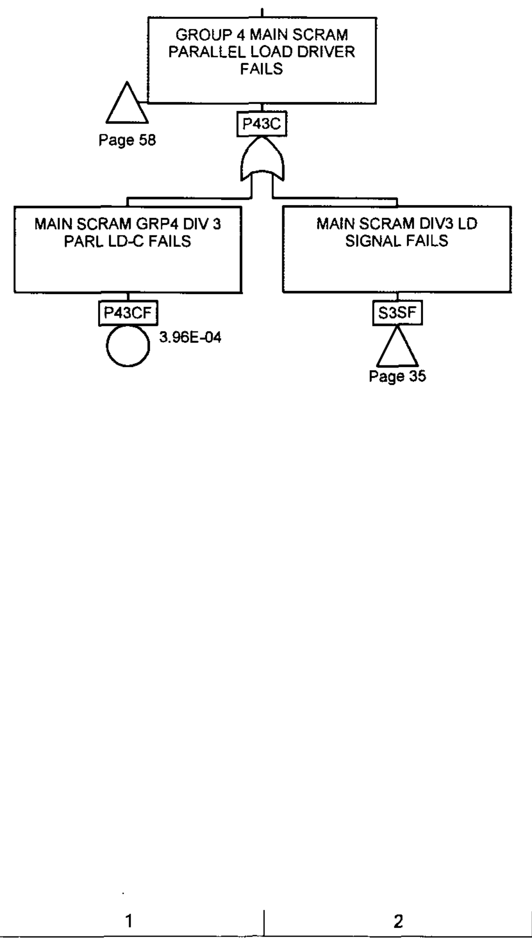


Figure 19D.6-24 Reactor Protection System Fault Tree

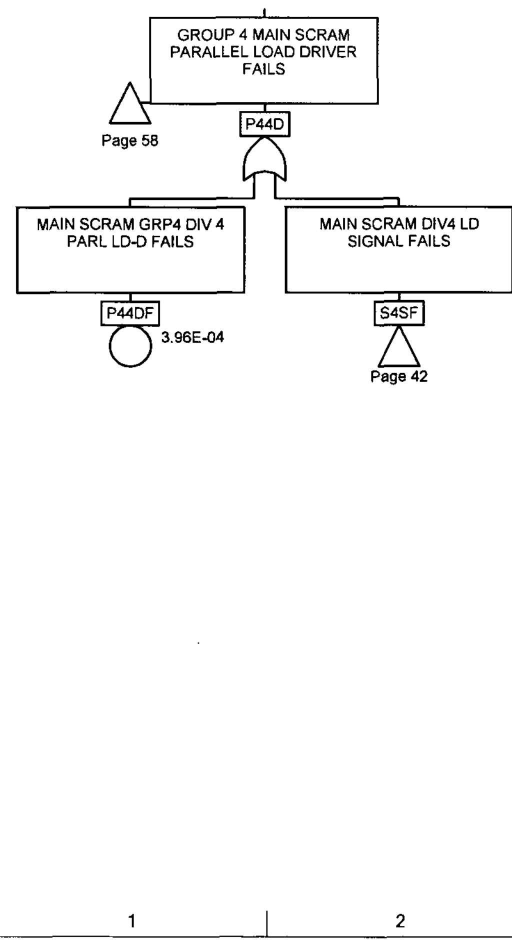


Figure 19D.6-24 Reactor Protection System Fault Tree

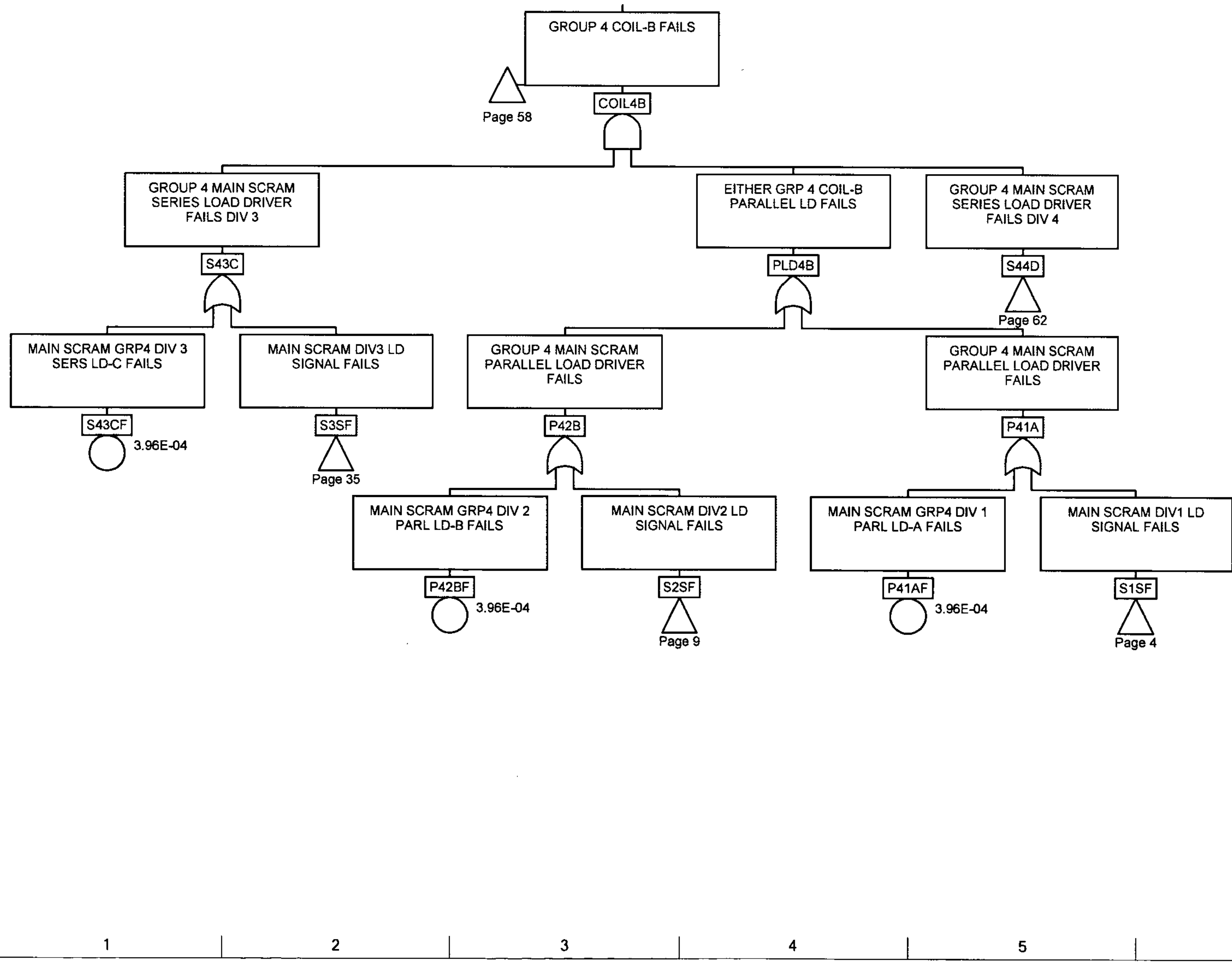
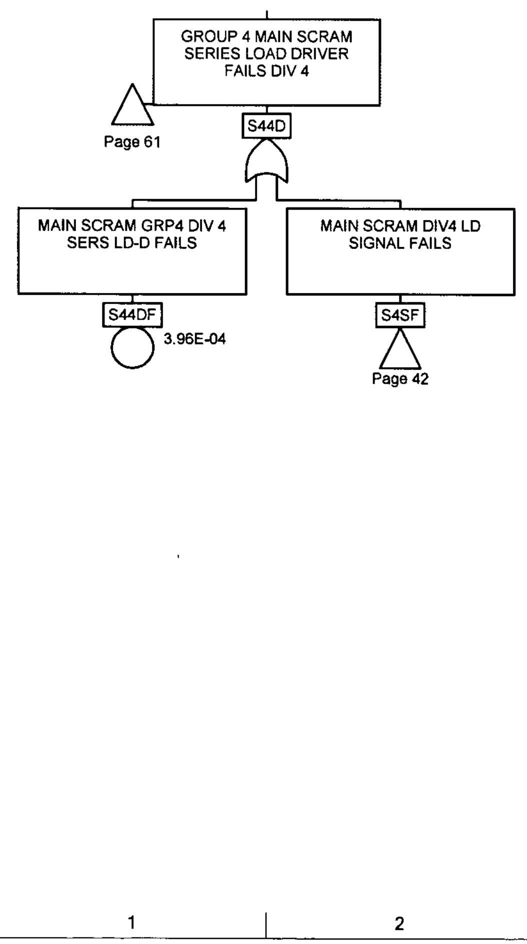


Figure 19D.6-24 Reactor Protection System Fault Tree


GROUP 4 MAIN SCRAM SERIES LOAD DRIVER FAILS DIV 4
Page 61
S44D
MAIN SCRAM GRP4 DIV 4 SERS LD-D FAILS
MAIN SCRAM DIV4 LD SIGNAL FAILS
S44DF
3.96E-04
S4SF
Page 42
1
2

Name	Page	Zone	Name	Page	Zone
APRM1F	27	2	BS14D	5	2
APRM2F	27	2	BS14D	7	2
APRM3F	27	3	BS14DF	7	1
APRM4F	27	3	BS14F	11	5
APRMSF	27	2	BS14F	14	2
BCOIL1AB	2	4	BS14F	15	3
BCOIL1AT	2	2	BS14FF	14	1
BCOIL1BB	11	5	CCFAPRM	27	1
BCOIL1BB	15	2	CCFDTM	18	4
BCOIL1BT	11	3	CCFDTM	30	4
BFUSEA	2	5	CCFDTM	36	4
BFUSEB	11	6	CCFDTM	43	4
BKUPLG	1	2	CCFMUX	18	1
BKUPLG	2	4	CCFMUX	30	1
BKUPS	1	2	CCFMUX	36	1
BKUPVA	2	3	CCFMUX	43	1
BKUPVB	2	4	CCFOLU	1	1
BKUPVB	11	3	CCFRLY	1	1
BPLDA	2	4	CCFS3A	18	2
BPLDA	10	2	CCFS3A	30	2
BPLDB	11	4	CCFS3A	36	2
BPLDC	2	2	CCFS3A	43	2
BPLDC	5	2	CCFTLU	1	3
BPLDD	15	2	COIL1A	16	2
BS11A	2	2	COIL1B	16	4
BS11A	3	2	COIL2A	48	3
BS11A	10	1	COIL2B	48	4
BS11AF	3	1	COIL2B	51	3
BS11G	11	2	COIL3A	53	3
BS11G	15	2	COIL3B	53	4
BS11GF	11	1	COIL3B	56	3
BS12B	2	3	COIL4A	58	3
BS12B	8	2	COIL4B	58	4
BS12B	10	2	COIL4B	61	3
BS12BF	8	1	DIV1MUX	20	2
BS12E	11	3	DIV2MUX	22	2
BS12E	12	2	DIV3MUX	24	2
BS12E	15	1	DIV4MUX	26	2
BS12EF	12	1	DTM1F	19	1
BS13C	2	4	DTM1F	31	1
BS13C	5	1	DTM1F	37	1
BS13C	6	2	DTM1F	44	1
BS13CF	6	1	DTM1SF	19	2
BS13H	11	4	DTM1SF	20	2
BS13H	13	2	DTM1SF	31	2
BS13H	15	3	DTM1SF	37	2
BS13HF	13	1	DTM1SF	44	2
BS14D	2	5	DTM2F	21	1

Name	Page	Zone	Name	Page	Zone
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DTM2F	38	1	HNFAIL	27	2
DTM2F	45	1	HNFAIL	30	4
DTM2SF	21	2	HNFAIL	36	4
DTM2SF	22	2	HNFAIL	43	4
DTM2SF	32	2	HPV1F	18	3
DTM2SF	38	2	HPV1SF	18	3
DTM2SF	45	2	HPV2F	30	3
DTM3F	23	1	HPV2SF	30	3
DTM3F	33	1	HPV3F	36	3
DTM3F	39	1	HPV3SF	36	3
DTM3F	46	1	HPV4F	43	3
DTM3SF	23	2	HPV4SF	43	3
DTM3SF	24	2	ILC001H	20	1
DTM3SF	33	2	ILC002H	22	1
DTM3SF	39	2	ILC003H	24	1
DTM3SF	46	2	ILC004H	26	1
DTM4F	25	1	ILCCCFH	18	5
DTM4F	34	1	ILCCCFH	30	5
DTM4F	40	1	ILCCCFH	36	5
DTM4F	47	1	ILCCCFH	43	5
DTM4SF	25	2	ILN0V1H	20	2
DTM4SF	26	2	ILN0V2H	22	2
DTM4SF	34	2	ILN0V3H	24	2
DTM4SF	40	2	ILN0V4H	26	2
DTM4SF	47	2	LNK1F	19	2
EDC11	20	2	LNK2F	32	2
EDC12	2	3	LNK3F	39	2
EDC12	22	2	LNK4F	47	2
EDC13	11	4	MAINLD	1	3
EDC13	24	2	MAINLG	1	4
EDC14	26	2	MAINLG	16	4
ELNK1F	20	1	MAINS	1	4
ELNK2F	22	1	OLU1F	4	1
ELNK3F	24	1	OLU1SF	4	2
ELNK4F	26	1	OLU2F	9	1
HCOI12	31	2	OLU2SF	9	2
HCOI13	37	2	OLU3F	35	2
HCOI14	44	2	OLU3SF	35	3
HCOI21	21	2	OLU4F	42	2
HCOI23	38	2	OLU4SF	42	3
HCOI24	45	2	P11A	41	4
HCOI31	23	2	P11AF	41	3
HCOI32	33	2	P12B	41	2
HCOI34	46	2	P12BF	41	1
HCOI41	25	2	P13C	28	2
HCOI42	34	2	P13CF	28	1
HCOI43	40	2	P14D	28	4

Name	Page	Zone	Name	Page	Zone
P14DF	28	3	PVSGRP3	16	4
P21A	51	5	PVSGRP3	53	4
P21AF	51	5	PVSGRP4	16	5
P22B	51	3	PVSGRP4	58	4
P22BF	51	3	RPS	1	2
P23C	48	3	RPSSLG	1	2
P23C	49	2	S11A	16	1
P23CF	49	1	S11A	17	2
P24D	48	4	S11AF	17	1
P24D	50	2	S12B	16	2
P24DF	50	1	S12B	29	2
P31A	56	5	S12BF	29	1
P31AF	56	5	S13C	16	3
P32B	56	3	S13C	35	2
P32BF	56	3	S13CF	35	1
P33C	53	3	S14D	16	4
P33C	54	2	S14D	42	2
P33CF	54	1	S14DF	42	1
P34D	53	4	S1SF	3	2
P34D	55	2	S1SF	4	2
P34DF	55	1	S1SF	11	2
P41A	61	5	S1SF	17	2
P41AF	61	5	S1SF	41	4
P42B	61	3	S1SF	48	2
P42BF	61	3	S1SF	51	6
P43C	58	3	S1SF	53	2
P43C	59	2	S1SF	56	6
P43CF	59	1	S1SF	58	2
P44D	58	4	S1SF	61	6
P44D	60	2	S21A	48	2
P44DF	60	1	S21AF	48	1
PLD1A	16	2	S22B	48	6
PLD1A	28	2	S22BF	48	5
PLD1B	16	4	S23C	51	2
PLD1B	41	2	S23CF	51	1
PLD2A	48	4	S24D	51	5
PLD2B	51	4	S24D	52	2
PLD3A	53	4	S24DF	52	1
PLD3B	56	4	S2SF	8	2
PLD4A	58	4	S2SF	9	2
PLD4B	61	4	S2SF	12	2
PPP101	20	1	S2SF	29	2
PPP102	22	1	S2SF	41	2
PPP103	24	1	S2SF	48	6
PPP104	26	1	S2SF	51	4
PVSGRP1	16	2	S2SF	53	6
PVSGRP2	16	3	S2SF	56	4
PVSGRP2	48	4	S2SF	58	6

Name	Page	Zone	Name	Page	Zone
S2SF	61	4	T212F	31	2
S31A	53	2	T222F	30	3
S31AF	53	1	T222F	32	2
S32B	53	6	T232F	30	4
S32BF	53	5	T232F	33	2
S33C	56	2	T242F	30	4
S33CF	56	1	T242F	34	2
S34D	56	5	T313F	36	3
S34D	57	2	T313F	37	2
S34DF	57	1	T323F	36	3
S3SF	6	2	T323F	38	2
S3SF	13	2	T333F	36	4
S3SF	28	2	T333F	39	2
S3SF	35	2	T343F	36	4
S3SF	49	2	T343F	40	2
S3SF	51	2	T414F	43	3
S3SF	54	2	T414F	44	2
S3SF	56	2	T424F	43	3
S3SF	59	2	T424F	45	2
S3SF	61	2	T434F	43	4
S41A	58	2	T434F	46	2
S41AF	58	1	T444F	43	4
S42B	58	6	T444F	47	2
S42BF	58	5	TFUSEA	2	1
S43C	61	2	TFUSEB	11	1
S43CF	61	1	TLU1F	4	2
S44D	61	5	TLU1SF	4	3
S44D	62	2	TLU1SF	18	4
S44DF	62	1	TLU2F	9	2
S4SF	7	2	TLU2SF	9	3
S4SF	14	2	TLU2SF	30	4
S4SF	28	4	TLU3F	35	2
S4SF	42	2	TLU3SF	35	3
S4SF	50	2	TLU3SF	36	4
S4SF	52	2	TLU4F	42	2
S4SF	55	2	TLU4SF	42	3
S4SF	57	2	TLU4SF	43	4
S4SF	60	2			
S4SF	62	2			
T111F	18	3			
T111F	19	2			
T121F	18	3			
T121F	21	2			
T131F	18	4			
T131F	23	2			
T141F	18	4			
T141F	25	2			
T212F	30	3			

Figure 19D.6-24 Reactor Protection System Fault Tree

19D.7 HUMAN ERROR PREDICTION

Refer to Reference 19D.7-1

19D.7.1 REFERENCES

19D.7-1 Shehane, M., Update of DCDRA Chapter 19D.7, RSC Engineers, Inc., RSC 10-16, August 2010.

19D.8 DEPENDENT FAILURE TREATMENT

19D.8.1 SUMMARY

Dependent failures have been included as integral parts of the overall PRA system and functional analyses. Dependencies of frontline systems on support systems and interdependencies between frontline or support systems have been modeled explicitly in the fault tree and event tree analyses. Common-cause failures (CCFs) have also been explicitly included from the standpoint of multiple component failures within systems, and as a result of human error.

19D.8.2 GENERAL CONSIDERATIONS

Dependent failures, including those frequently categorized as common-mode or common-cause, must be realistically and adequately addressed in any comprehensive evaluation of risk. The term dependent failure refers to two or more elements failing as a result of the same cause or failure mode. A number of considerations are important in the analysis of common-cause failures, including the following:

(1) Common-cause effects generally are of greatest significance in redundant systems. The analysis was structured to insure that no important common cause effects were overlooked, particularly for backup and support systems.

(2) Common-cause effects are limited by design, manufacture, and procedural diversity. Such factors were taken into consideration in the analysis.

(3) Isolation barriers and physical separation of redundant systems and components can reduce or eliminate common causes of failure, and these factors were considered and taken into account in the analysis as appropriate.

19D.8.3 MULTIPLE EQUIPMENT FAILURES FROM A COMMON CAUSE

Causes of multiple equipment failures include common manufacturing errors, design errors, and extreme environmental conditions. Failures of this type generally have a low frequency of occurrence and would not significantly influence system failure probabilities, except in those cases where design redundancy is incorporated to achieve very low overall system or function failure probabilities.

Consequently, systems for which common-cause effects of the above type are potentially most significant—and, therefore, addressed explicitly—included the following:

(1) emergency power supply system,

(2) automatic depressurization system,

(3) control rod drive system,

(4) reactor protection system, and

(5) instrumentation and control.

Design-related CCFs from the above systems are included in the analysis for the following special components:

ESF logic,
transmission network (MUX),
sensor & transmitter miscalibration,
output logic units,
digital trip units,
trip logic units,
main scram load drivers,
backup scram relays,
pressure sensors,
APRMs,
diesel generators,
batteries,
offsite power sources,
safety relief valves.

Multiple equipment failures generally do not occur simultaneously. Usually there will be a noticeable time period between the first and any subsequent failures, thus, providing advance information on a potentially developing problem. If the first failure is detected and its cause determined before subsequent failures occur, loss of system functions can be avoided and corrective action can be taken.

19D.8.4 MULTIPLE FAILURES DUE TO HUMAN ERROR

Multiple failures can also occur due to human error, if an operator or maintenance technician repeatedly makes the same mistake. Multiple instrument miscalibration is judged the most likely event of this type and is explicitly included in the analysis. Human error as a common-cause failure mechanism is discussed further in Subsection 19D.7.

19D.8.5 FUNCTIONAL INTERDEPENDENCIES

Interdependencies within and between systems are treated rigorously in the fault tree and event tree analyses. Examples of interdependencies that are accounted for are electric power, service water, room cooling, and control instrumentation.

Interdependency factors that could affect automatic or manual initiation of systems were evaluated. In some cases, the same sensors, signal transmitters, reference fluid pressure columns or logic units are used in the initiation of more than one system. This type of dependency between systems was represented in detail to assess the probability of safety system operability. In the ABWR PRA, many sensors provide input to more than one system. Each of these sensors and their logic are designated by unique acronyms throughout the analysis to insure that dependencies among systems are rigorously accounted for in the functional fault tree evaluations.

19D.8.6 GENERIC COMPONENT CCFS

In response to a request from the Nuclear Regulatory Commission, additional common cause failures were added to the basic analysis and the effects on system availability and core damage frequency were evaluated. Component CCFs for the following systems were identified, evaluated and included:

HPCF	2/2 trains	14 components
RHR Core Flooding Mode	3/3 trains	24 components
RHR SP Cooling Mode	3/3 trains	25 components
RBCW/RSW	Internal to each division	4 components
RBCW/RSW	Between divisions A & B	6 components
RBCW/RSW	Between divisions A & C	6 components
RBCW/RSW	Between divisions B & C	6 components
RBCW/RSW	Between divisions A, B & C	6 components

These are the systems where generic component common-cause failures might have a significant effect. Generic component CCFs that are included in the system analysis are for pumps, pump auxiliary equipment, manual valves, motor-operated valves, check valves, room air conditioners, spargers, strainers, circuit breakers, flow transmitters, heat exchangers, and temperature elements.

For the RBCW/RSW CCFs internal within each division of RBCW/RSW, the component CCFs were included at appropriate places within the fault tree structures. For all other cases (interdivisional or between trains), the individual component CCFs were summed and added-in at the top as a CCF module. The RBCW/RSW interdivisional CCFs were added-in at the top of the fault trees for the safety systems that use RBCW/RSW. For updating of the values the pump failure probability was considered the dominating factor in the CCF and was used to determine a factor from which a new probability could be calculated. This is documented in the data update (Reference 19D.8-2).

Component CCFs were identified wherever redundancy occurs in the fault trees of the above systems (generally, for every "and" gate). The component CCFs were quantified using the "multiple Greek letter" method and using the CCF factors given in RSC 08-06 (Reference 19D.8-1). Where common-cause factors were not given for specific component types, the recommended "generic" factors were used. For those cases, the results should be considered as "bounding" and are probably conservative.

For the update to the data, values for independent failure probabilities were calculated and then the probability of a common cause failure was used to compute the common cause factor. This was then used with the updated independent failure probability to update the common cause probability. Thus, the method of CCF analysis has not changed, only the failure probabilities have changed due to the data update. This is documented in RSC-CALKNX-2010-0501 (Reference 19D.8-2).

The numerical results of the component CCF analysis in terms of system CCF probabilities are given below:

HPCF CCFs	2/2 loops	1.85E-3
RHR Core Flooding CCFs	3/3 loops	8.85E-4
RHR SP Cooling CCFs	3/3 loops	1.39E-3
RBCW/RSW CCFs within a division	1/2 pumps	1.16E-5
RBCW/RSW CCFs between Div. A & B	2/2	4.94E-6
RBCW/RSW CCFs between Div. A & C	2/2	4.94E-6
RBCW/RSW CCFs between Div. B & C	2/2	4.94E-6
RBCW/RSW CCFs between Div. A, B, & C	3/3	4.55E-6

The numerical results of this analysis also can be viewed from two additional perspectives: the effect on system unavailability (Table 19D.8-1), and the effect on core damage frequency (Table 19D.8-2).

The effects of the added generic component CCFs on system unavailability are significant. The most significant effect is on the core flooding mode of RHR, where the system unavailability with component CCFs is over 22 times the system unavailability without component CCFs.

For the HPCF system, the most significant CCF contributors are common-cause failure of the pumps. For the updating of the value after new failure rates were added the common cause failure probability calculated was a function of the overall pump failure value. It was determined that would be the dominating factor in CCF of this system.

The individual divisions of the RBCW/RSW systems were not significantly affected by component CCFs. However, the interdivisional CCFs have a measurable effect on core damage frequency, as shown in Table 19D.8-2 and discussed below.

The most significant effect on CDF is due to the CCFs between all three divisions of RBCW/RSW. This is primarily due to the failure of both HPCF and RHR Core Flooding, given loss of all RBCW/RSW divisions. All other CCFs have relatively little effect on CDF.

The updated common cause failure of all three divisions of RBCW/RSW is based on common-cause failure of pumps as they were considered the dominating common cause components.

The total effect on CDF of the addition of these CCFs (~8% increase) is not insignificant, partially due to the low absolute value of the ABWR CDF.

19D.8.7 REFERENCES

19D.8-1 Eddy, C., Establishment of Model to Evaluate Plant Specific Changes, RSC Engineers, Inc. RSC 08-06, April 2010.

19D.8-2 Lee, A.M., Update of the Toshiba DCD PRA MOR Component Failure and Initiating Event Data to Support the Toshiba DCDRA, RSC Engineers, Inc., RSC CALKNX-2010-0501, Revision 1, July 2010.

Table 19D.8-1
Effect on System Unavailability

System	Base Unavailability	Unavailability with CCFs*	% Increase
HPCF	1.60E-3	3.45E-3	116
RHR (flood)	3.97E-5	9.25E-4	2230
RHR (cool)	1.73E-4	1.56E-3	802
RBCW/RSW Div. A	4.44E-4	4.70E-4	5.86
RBCW/RSW Div. B	4.44E-4	4.70E-4	5.86
RBCW/RSW Div. C	4.44E-4	4.70E-4	5.86

*CCFs within that system

Table 19D.8 2
Effect on Core Damage Frequency

System	CDF Increase/yr	% Increase
HPCF	2E-10	<1
RHR (flood)	1E-10	<1
RHR (cool)	8E-10	<1
RBCW/RSW A, B, & C	5.6E-9	6.2

19D.9 CDF SENSITIVITY TO OUTAGE TIMES AND SURVEILLANCE INTERVALS

19D.9.1 SUMMARY

As a consequence of 1992 GE-NRC discussions of ABWR DSER questions regarding applicability of GESSAR test and maintenance (T&M) unavailabilities to the ABWR PRA, it was agreed that ABWR T&M unavailabilities would be increased over those of GESSAR to provide utility operational flexibility. Consequently, T&M values for RCIC, HPCFB, HPCFC, RHRA, RHRB and RHRC were each raised to two percent in the PRA model, and the calculated core damage frequency of 9.80E-8 reflects inclusion of these values.

19D.9.2 SENSITIVITY TO TEST AND MAINTENANCE OUTAGE TIMES

CDF sensitivity to T&M outage times was assessed by varying system values individually as well as in combination. Results presented in attached Tables 19D.9-1 and 19D.9-2 illustrate the impact of increasing system T&M unavailabilities by a factor of five from two to ten percent. Ten percent was judged to be a reasonable upper bound for T&M unavailability for a single system. As can be seen, calculated CDF is most sensitive to the RCIC system T&M unavailability. This is due in large part to the fact that station blackout sequences dominate CDF, and in these sequences RCIC is essential for successful core cooling. Since no credit was taken in the Level 1 PRA for fire water injection, this calculated CDF sensitivity to RCIC T&M unavailability is actually somewhat conservative. In addition, ample time is available for maintenance of RCIC during refueling outages without CDF risk implications, since during shutdown the system is unable to perform its ECCS function.

Second in importance is the T&M unavailability of HPCFB. HPCFB includes a hardwired manual initiation backup in the control room. This provides a diverse means of manually initiating HPCFB in the event of essential multiplexing system common mode failure, a feature which increases the importance of HPCFB relative to other ECCS systems.

CDF is very insensitive to the T&M unavailability of systems other than RCIC and HPCFB, either individually or in combination. Table 19D.9-3 provides a summary of bounding scenarios in which individual systems are removed completely from service.

These results support the conclusions drawn from Tables 19D.9-1 and 19D.9-2. Further, even with RCIC the most sensitive system completely out of service, the core damage frequency goal of 1.0E-5 is still satisfied.

19D.9.3 SENSITIVITY TO SURVEILLANCE INTERVALS

Since no changes were made to established BWR surveillance intervals (GESSAR II values were applied in the ABWR PRA), sensitivity to changes in surveillance intervals was not investigated.

19D.9.4 REFERENCES

19D.9-1 Eddy, C., Establishment of Model to Evaluate Plant Specific Changes, RSC Engineers, Inc. RSC 08-06, April 2010.

Table 19D.9-1
CDF Sensitivity to T&M Outage Unavailabilities

System	Single System Perturbations to Base Case T&M Value of Two Percent						
RCIC	2.00E-2	1.00E-1	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2
HPCFB	2.00E-2	2.00E-2	1.00E-1	2.00E-2	2.00E-2	2.00E-2	2.00E-2
HPCFC	2.00E-2	2.00E-2	2.00E-2	1.00E-1	2.00E-2	2.00E-2	2.00E-2
RHRA	2.00E-2	2.00E-2	2.00E-2	2.00E-2	1.00E-1	2.00E-2	2.00E-2
RHRB	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2	1.00E-1	2.00E-2
RHRC	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2	1.00E-1
CDF	9.80E-8	1.56E-7	9.90E-8	9.84E-8	9.81E-8	9.80E-8	9.80E-8
% INCR	0	59	1	<1	<1	<1	<1

Table 19D.9-2
CDF Sensitivity to T&M Outage Unavailabilities

System	Multiple System Perturbations to Base Case T&M Value of Two Percent						
RCIC	1.00E-1	2.00E-2	2.00E-2	1.00E-1	2.00E-2	1.00E-1	1.00E-1
HPCFB	2.00E-2	1.00E-1	2.00E-2	1.00E-1	1.00E-1	1.00E-1	1.00E-1
HPCFC	2.00E-2	2.00E-2	1.00E-1	2.00E-2	1.00E-1	1.00E-1	1.00E-1
RHRA	1.00E-1	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2	1.00E-1
RHRB	2.00E-2	1.00E-1	2.00E-2	2.00E-2	2.00E-2	2.00E-2	1.00E-1
RHRC	2.00E-2	2.00E-2	1.00E-1	2.00E-2	2.00E-2	2.00E-2	1.00E-1
CDF	1.56E-7	9.91E-8	9.85E-8	1.57E-7	9.94E-8	1.58E-7	1.59E-7
% INCR	59	1	<1	60	1	61	62

Table 19D.9-3
CDF Sensitivity to T&M Outage Unavailabilities

System	Impact of Single Systems Completely Removed from Service					
RCIC	2.00E-2	1.00	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2
HPCFB	2.00E-2	2.00E-2	1.00	2.00E-2	2.00E-2	2.00E-2	2.00E-2
HPCFC	2.00E-2	2.00E-2	2.00E-2	1.00	2.00E-2	2.00E-2	2.00E-2
RHRA	2.00E-2	2.00E-2	2.00E-2	2.00E-2	1.00	2.00E-2	2.00E-2
RHRB	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2	1.00	2.00E-2
RHRC	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2	2.00E-2	1.00
CDF	9.80E-8	8.10E-7	1.12E-7	1.05E-7	1.00E-7	9.90E-8	9.90E-8
% INCR	0	727	14	7	2	1	1

19D.10 DATA UNCERTAINTY FOR ABWR PRA

19D.10.1 INTRODUCTION

This analysis presents the results of a quantitative data uncertainty analysis for the Advanced Boiling Water Reactor (ABWR) Level 1 Probabilistic Risk Assessment (PRA). Completeness uncertainty was not analyzed. Modeling uncertainty is addressed in Subsection 19.3.1.3(1)(a) of the DCD.

19D.10.2 PURPOSE AND SUMMARY OF CONCLUSIONS

The purpose of this study was to determine and propagate data uncertainty in the internal events analysis in the ABWR Level 1 PRA, to provide the probability distribution describing the uncertainty in the calculated core damage frequency (CDF).

The uncertainty analysis results show that the ABWR CDF has the distribution shown in Figure 19D.10-1, having a mean value of 9.80E-8 per reactor-year and an error factor of 2.96, (calculated as the 95th percentile divided by the median). The 95th percentile of the distribution is 2.35 times the mean value or 2.30E-7. The 5th percentile is 2.99E-8 per reactor-year.

The basic event WPMPCCFRCWABC (i.e., Reactor Building Cooling Water CCF for Train A, B, and C) is the highest contributor to uncertainty in the CDF as well as to the mean value of the CDF. The remaining contributors are identified in Subsection 19D.10.6.1.

The results of the uncertainty show that the 95th percentile is only moderately sensitive to the error factors (EFs) of the basic events, and hence that lack of precise EF values has a rather small effect on the outcome. For example, doubling the EF values of each basic event simultaneously increases the 95th percentile of the CDF by only 3.9%. When all EFs are set equal to 15, the 95th percentile increases by only 8.3%. (Note 1 in Subsection 19D.10.8).

Coupling between components was considered as a source of CDF uncertainty. The effect of coupling has been shown to be more and more negligible as sample size increases. Since very large sample sizes are used in the analysis coupling is considered to be negligible.

Possible bias uncertainty was analyzed by multiplying all mean values of basic events by two (which case is referred to as the "X2 case" in this report). It was found that seven of the top ten basic events in the nominal case, ranked by F-V importance, were in the top ten ranking in the X2 case. Similarly, four of the top six accident sequences were the same in both cases. This is an indication that insights gained from the PRA as to the relative importance of the top ten basic events, and top six accident sequences, will be correct, even if the input data is biased.

19D.10.3 APPROACH

The effects of uncertainty in PRA data were analyzed as follows:

(1) The sources of data were identified.

(2) Error factors were assigned to the PRA data.

(3) The uncertainties were propagated across the fault trees and event trees using Monte Carlo simulation.

(4) The accuracy of the computerized mathematical modeling was established.

A sensitivity analysis was also performed on the mean values of the input data, on the truncation limits, and on the EFs.

19D.10.4 DATA ANALYSIS

The types of data analyzed in this report are listed in column 1 of Table 19D.10-1.

Each entry of the second column represents the source reference for the point estimate (mean) data which was used in the Level 1 calculation. The sources for the Error Factor (EF) estimates used in this analysis are shown in column 3. Each point estimate in the analysis is treated as the mean value of a log normal distribution. All EF values used represent the ratio of the 95th percentile over the 50th percentile.

19D.10.4.1 Error Factors for Human Error Probabilities

An EF was assigned for each human error probability (HEP) from Reference 19D.10- based on the estimated magnitude of the HEP, and on the stress level which applies to the action. For this report, only two stress levels were used, corresponding to events prior to and events during the accident. Table 19D.10-2 shows the EFs used in both situations. A basic assumption is that the tasks are performed by experienced personnel only (licensees with at least 6 months' familiarity with the plant). For the updated values an error factor of 3 was used for a HEP >1E-2, EF of 5 was used for 1E-2 to 1E-4, and 10 for <1E-4.

The HEP EFs which are used are considered conservative because Reference 19D.10-1 is based to a large extent on either (a) derived data, or (b) data from older nuclear power plants (pre-1982). Hence, refinements in training, in operations, and in human factors engineering will tend to group the HEP variability between operators closer together and make the EFs smaller than the values used herein. Reference 19D.10-1 gives EFs as step functions of the HEP. This is evidently an approximation to a smooth curve. The variability of these EFs will have some effect on the variability of the top event. The amount of effect this has is described in Subsection 19D.10.6.

19D.10.4.2 Error Factors for Component Failure Rates

Reference 19D.10-2 was used as the main source for EF data on component failure rates (FRs). Reference 19D.10-2 is a data study where existing generic data was updated with the latest component failure data from the NRC using a Bayesian update.

19D.10.4.3 Error Factors for Special Cases

(1) Common-cause failures (CCFs) were assigned an EF of 15.

(2) An EF of 5 was assigned for each accident initiation frequency.

(3) Undeveloped events were given an EF of 15.

(4) Components which were not listed in the EF database were estimated using the same technique as was used for the updated HEP actions.

19D.10.4.4 Error Factor Applicability to PRA Data

An initial hurdle of the uncertainty analysis was the fact that some of the ABWR PRA basic event failure data for components were given in terms of probabilities of failure, whereas the EF data source (Reference 19D.10-2) gives variance values for failure rates which can be made into EFs. This issue was resolved as follows:

When $\lambda t << 1$, then

$q(t) = 1\text{-}e^{-\lambda t} \approx \lambda t$ (Equation 19D.10-1)

where

λ = component failure rate, and

t = mission time

So the 95th and 50th percentiles of q(t) are approximately given by

$q_{0.95}\ (t)\ 1 - \exp\ (-\lambda_{0.95} t) \approx \lambda_{\ 0.95} t$ (Equation 19D.10-2)

$q_{0.5}\ (t)\ 1 - \exp\ (-\lambda_{0.5} t) \approx \lambda_{\ 0.5} t$ (Equation 19D.10-3)

The error factor is defined as

$$EF = \frac{q_{0.95}(t)}{q_{0.5}(t)} = \frac{\lambda_{0.95}}{\lambda_{0.5}}$$ (Equation 19D.10-4)

The percentage error in this approximation was tested and found to be less than 5% whenever the 95th percentile of the failure rate times mission time was less than or equal to 0.1.

19D.10.5 UNCERTAINTY AND SENSITIVITY ANALYSIS

19D.10.5.1 Mathematical Models

19D.10.5.1.1 Applicability of Lognormal Distribution

An issue for the ABWR uncertainty analysis was whether the lognormal distribution was applicable to the probabilities, unavailabilities and unreliabilities (rather than failure rates). This was answered affirmatively by applying a test to doubtful cases. The method is based on the spill-over of probability mass out of the [0,1] interval, (in which all probability values must theoretically be contained). In a few cases the spill-over was not trivial. However, because such cases give worst case values, no adjustment was made to the uncertainty analysis.

19D.10.5.1.2 Sampling Uncertainties

UNCERT is a PC code, developed by Science Applications International Corporation, which is used for generating probability distributions of a top event when probability distributions of the initiating and subsequent events are specified. The fault tree description must be given to UNCERT in the form of cut sets. A Monte Carlo technique is used for calculation of the histogram of the top event probability.

19D.10.5.1.2.1 Sampling of the Tails

Sampling is performed using the method for generating random numbers which is described in Section 19D.10.5.1.2.1 of Reference 19D.10-1. SAIC has checked the accuracy of the sampling for the 95th percentile of two lognormal distributions each with mean E-4, one with EF=3 and the other with EF=15. The results are within the 95th percentile confidence interval for both of these cases.

19D.10.5.1.3 Coupling Uncertainties

Based on experience with the issue of coupling, as the sample size goes up coupling has less of an impact. Due to the sample sizes being 100,000 samples the effect of coupling was considered to be negligible.

19D.10.5.1.3.1 Not Used

19D.10.5.1.3.2 Not used

19D.10.5.1.3.3 Cut Set Truncation Uncertainties

The CAFTA code was used for generating the cut sets in the level 1 calculation.

CAFTA was used in the uncertainty analysis to generate cut sets upon which to do the Monte Carlo simulation. The cut sets obtained are all cut sets having a probability of occurrence greater than a chosen truncation limit. The following sensitivity analyses were performed to investigate the adequacy of the truncations used to obtain the main CDF result.

(1) A computer run was performed with all basic events multiplied by a factor of 2, and the truncation limit set equal to that used during the point estimate run (nominal case).

(2) The total probability of the additional cut sets picked up for the worst case (the X2 case) run, was compared to the probability of the top event in the nominal case. If this is small, than it is clear that the truncated cut sets do not contribute much to the uncertainty of the result. This is so, because multiplying all mean values by a factor of 2 is an extreme worst case which would essentially never be obtained if a Monte Carlo analysis were run with any value of EF applied across all basic events appearing in the cut sets.

19D.10.5.2 Sensitivity Analysis on the Mean Values of the Basic Events

Sensitivity of analysis bias error uncertainty was first investigated by multiplying the probability of each basic event by 2. The effect of multiplying all basic event probabilities by 2.0 (the X2 case), and running a PRA quantification with point estimates is equivalent to choosing a value above the 85th percentile for each basic event (Note 2 in Subsection 19D.10.5.1.3.3). This case is rather conservative and would almost never be obtained by random sampling. The results of the X2 case are discussed in Subsection 19D.10.6.3.

The combined effect of bias error uncertainty and EF was investigated by increasing the probability of each basic event by a factor of two, keeping the same EFs, and using Monte Carlo simulation. The mean value CDF thus obtained is 1.92E-7/year, or 2.0 times the base case CDF. The 95th percentile is 4.54E-7/year. This shows the effect of possible systematic bias error in the PRA.

19D.10.5.3 Sensitivity Analysis on the EFs

To calculate the sensitivity of the 95th percentile to the value of EF, a curve of the 95^{th} percentile divided by the mean value is plotted. This curve is shown in Figure 19D.10-2.

The most sensitive region of interest is for the smaller values of EF, and the least sensitive region is between 5 and 15. The sensitivity is judged excellent between 5 and 15 because the variation in the 95th percentile will be less than 3% in that region whenever EF is changed by one unit.

19D.10.5.4 Sensitivity Analysis on Coupling of Basic Events

Based on experience with the issue of coupling typically as the sample size goes up coupling has less of an impact. Due to the sample sizes being 100,000 samples the effect of coupling is considered to be negligible.

19D.10.5.4.1 Not Used

19D.10.5.4.2 Not Used

19D.10.6 DISCUSSION OF RESULTS

The uncertainty analysis was run several times with sample sizes of 100,000 for each run.

19D.10.6.1 The Top Ten Contributors to Uncertainty in the CDF

The top 20 events (as ranked by the Fussell-Vesely (F-V) importance measure) were chosen for the calculation of their uncertainty importance. This was obtained by calculating the standard deviation of the top event probability (i.e., the CDF) when only the selected event's probability is allowed to vary. (All other basic event probabilities are held constant at their point estimates.) Runs of 100,000 samples each were made for each of the selected events. The results are given in Table 19D.10-5.

Table 19D.10-5 shows the 10 top contributors to uncertainty in the CDF. The basic events listed in Table 19D.10-5 are defined in Subsections 19D.4 and 19D.6. Nine of the contributors are also in the top ten list when ranked by the Fussell-Vesely (F-V) importance measure.

19D.10.6.2 The Effect of Error Factors on the Top Event Distribution

The model was run several times (100,000 samples per run) with a variety of EFs to determine the effect of the adequacy of the EF treatment used in the uncertainty analysis. Table 19D.10-6 summarizes the results.

19D.10.6.3 Uncertainty Due to the Truncation Limits Used in Generating the Cut Sets

The uncertainty from the truncation by the CAFTA software was studied in two ways:

(1) Case 1—Lower Truncation Limits

The truncation limit used in the basic PRA run was E-12. Additional runs were made with all sequences truncated at E-13. The difference in CDF was less than 1%.

(2) Case 2—Multiplying The Point Estimates By Two

This case was considered for several reasons:

(a) It provides an additional indication of the sensitivity of importance measures to point-estimate bias.

(b) It brings up a different group of cut sets than those brought up in case 1 above.

In conclusion, the truncation limits of the PRA are adequate, and that the calculated top event probabilities are representative of the true CDF.

19D.10.6.4 Robustness of the Top Events Cut Sets, and Sequences

Case 2 above (i.e., with all event and sequence frequencies multiplied by two) was used to analyze the robustness of the PRA in several ways discussed below.

19D.10.6.4.1 Robustness of the Fussell-Vesely (F-V) Importance Measure

Table 19D.10-7 shows how the top 10 events with the highest F-V importance are changed in the X2 case. The events are defined in Subsections 19D.4 and 19D.6.

Table 19D.10-7 shows that eight of the same events are in the top ten in both cases. This shows that the F-V importance measure ranking is very robust with respect to bias increase in the point estimates. Table 19D.10-8 shows the top ten events in the X2 case.

19D.10.6.4.2 Robustness of the Six Top Accident Sequences

Table 19D.10-9 shows how the six top accident sequences transpose when all of the basic event unavailabilities are multiplied by two (i.e., in the X2 case). The accident event sequences are illustrated in the event trees of Subsection 19D.4. Four of the same sequences are in the top six in both the base case and in the X2 case. It is very interesting that in the base case, the top six sequences contribute 93% of the probability mass of the CDF.

19D.10.7 CONCLUSION

The ABWR PRA is robust with respect to uncertainty in the basic event probability data.

As a result, the 95th percentile of top event frequency does not change much with reasonable increases in the mean values, and in the EFs of the basic events.

19D.10.8 NOTES

Note 1—Setting All EFs to 15.

The EFs were set equal to 15 in order to determine how sensitive the ABWR uncertainty analysis results were to the EFs used in the ABWR fault trees. The results showed that when all EFs were set to 15, the 95th percentile of the CDF increased by only 8.8%. This result is not surprising when considered in terms of the information presented in Table 19D.10-5 and Figure 19D.10-2. Table 19D.10-5 presents the top ten contributing basic events (BEs) to the CDF uncertainty. Figure 19D.10-2 is a sensitivity curve for $x_{0.95}$/mean (where $x_{0.95}$ is the 95th percentile of the BE probability (BEP) as a function of the uncertainty (EF) of the estimate of that BEP. Since the mean is a constant, Figure 19D.10-2 is also a sensitivity curve for the 95th percentile. Table 19D.10-5 and Figure 19D.10-2 provide the following pertinent information:

(1) Table 19D.10-5 shows that three of the top ten contributors to uncertainty already have EF = 15 in the base case, and the change will not affect the BEP distributions in these cases. Two were raised from EF = 5 to EF = 15, and five were raised from EF = 3 or less to EF= 15 in the EF sensitivity study.

(2) Figure 19D.10-2 shows that the difference in the values of the 95th percentile is less than 4% when a base case EF=10 is raised to EF = 15. (I.e., $x_{0.95}$/mean changes from about 2.44 to about 2.53). Similarly, when a base case EF=5 is raised to EF = 15, the difference is about 9.52%. (I.e., $x_{0.95}$/mean changes from about 2.31 to about 2.53).

Combining this information from Table 19D.10-5 and Figure 19D.10-2, we see that the 95th percentile of the top ten contributors to CDF uncertainty changed a small amount, on the average, when the EFs of the top ten BEs are raised to an EF = 15.

Note 2—Multiplying the Mean Values by Two

The effect of multiplying all basic event probabilities by 2.0 results in all probabilities being above the 85th percentile for all basic events, irrespective of whether the error factor is small or not. It is shown below that the maximum value of the 85th percentile ($x_{0.85}$) divided by the mean (x) can be calculated as follows:

$(x_{0.85}/ x)max = \exp(0.5Z^2_{0.85})$ (Equation 19D.10-5)

where

$Z_{0.85}$ = the 85th percentile of the standard normal distribution (which is equal to 1.04).

The value of this ratio is approximately 1.72. This shows that the 85th percentile of any lognormal distribution is always less than two times its mean value.

Derivation Of The Value For (x0.85/ x)max—First, the EF can be written in terms of the median and the 95th percentile as

$EF = x_{0.95}/x_{0.5}$ (Equation 19D.10-6)

where

$x_{0.95}$ = 95th percentile

$x_{0.5}$ = 50th percentile

From this definition it follows that

$EF = \exp(1.645\sigma)$ (Equation 19D.10-7)

and that

$\sigma = \ln EF/1.645$ (Equation 19D.10-8)

where

σ = the standard deviation of the standard normal distribution.

The 85th percentile, $x_{0.85}$, can also be written as

$x_{0.85} = x_{0.5} \exp(Z_{0.85} \ln EF/1.645)$ (Equation 19D.10-9)

Substituting Equation 19D.10-1 into the equation for the mean, x, of a lognormal distribution gives:

$x = x_{0.5} \exp((\ln EF)^2/2(1.645)^2)$ (Equation 19D.10-10)

Dividing Equation 19D.10-1 by Equation 19D.10-2 gives

$x_{0.85}/ x = \exp(1.04 \ln EF/ 1.645 - 0.5 (\ln EF/1.645)^2)$ (Equation 19D.10-11)

This equation is a maximum when

$\ln EF = (1.04) (1.645) = 1.711$ (Equation 19D.10-12)

This value of EF gives the maximum value of the ratio $x_{0.85}/ x$:

$(x_{0.85}/ x)max = \exp(0.5(1.04)^2) = 1.72$ (Equation 19D.10-13)

19D.10.9 REFERENCES

19D.10-1 Eddy, C., Establishment of Model to Evaluate Plant Specific Changes, RSC Engineers, Inc. RSC 08-06, April 2010.

19D.10-2 Lee, A., Documentation of the RSC Generic Component Failure Database for PSA Studies, RSC Engineers, Inc., RSC 10-02, April 2010.

19D.10-3 Eide, S.A., Wierman, T.E., Gentillon, C.D., et al., Industry Average Performance for Components and Initiating Events at U.S. Commerical Nuclear Powerplants. USNRC, NUREG/CR-6928, February 2007.

19D.10-4 Shehane, M., Update of Selected Human Action to support the Toshiba DCDRA, RSC Engineers, Inc., RSC CALKNX-2010-0506, July 2010.

Table 19D.10-1
Data Sources

Type of Data	Point Source Estimates	EF Estimates
Initiating Event Data	19D.10-10	19D.10-10 Judgment
Component Failure Data	19D.10-9 (Updated) 19D.10-3 19D.10-4	19D.10-9 Judgment
Human Error Prediction Data	19D.10-1 19D.10-3 19D.10-4 19D.10-11 (Updated)	19D.10-1 19D.10-8 Judgment

Table 19D.10-2
EF Values for HEPs

Circumstances Under Which Error Occurs	Mean HEP Estimate	EF
Before Initiator (Type A)	<0.001	10
	0.001 to 0.01	3
	>0.01	5
After Initiator (Type B)	<0.001	10
	>0.001	5

Table 19D.10-3 Not Used

Table 19D.10-4 Not Used

Table 19D.10-5
Top Ten Contributors to Uncertainty in the CDF

Basic Event Name	EF	F-V Rank	Standard Deviation (xE08)
WPMPCCFRCWABC	15	6	1.98
WPMPCCFRSWABC	15	7	1.9
RCIMAINT	3	1	1.08
EBY1CCF	15	12	0.962
COND	3	2	0.772
CRD	3	3	0.478
RLU001DW	5	10	0.384
RPM001DW	2.83	4	0.349
RTU001DH	5	11	0.326
RLS039HW	2.38	8	0.272

Table 19D.10-6
Sensitivity of 95^{th} Percentile with Respect to EF Values

EF Modification	Mean Value (xE08/year)	95^{th} Percentile (xE08/year)
All EFs = Half Base Case	9.80	22.2
Base Case	9.80	23.0
All EFs = Twice Base Case	9.76	23.7
All EFs = 5	9.77	22.6
All EFs = 15	9.75	24.7

Table 19D.10-7
F-V Importance Comparison 1

F-V Rank in Base Case	Component	F-V Importance Value	Rank in x2 Case
1	RCIMAINT	0.148	1
2	COND	0.105	4
3	CRD	0.065	3
4	RPM001DW	0.0626	6
5	WPMPCCFRCWABC	0.0568	7
6	WPMPCCFRSWABC	0.0568	8
7	RLS039HW	0.0491	9
8	RLS045HW	0.0491	10
9	RLU001DW	0.0309	18
10	RTU001DH	0.0272	11

Table 19D.10-8
F-V Importance Comparison 2

F-V Rank in x2 Case	Component	F-V Important Value	Rank in Base Case
1	RCIMAINT	0.157	1
2	CCFTLU	0.15	37
3	CRD	0.118	3
4	COND	0.115	2
5	HPBMAINT	0.0715	34
6	RPM001DW	0.0664	4
7	WPMPCCFRCWABC	0.0579	5
8	WPMPCCFRSWABC	0.0579	6
9	RLS039HW	0.0518	7
10	RLS045HW	0.0518	8

Table 19D.10-9
Top Six Sequences Comparison and Frequency

Base Case		X2 Case	
Sequence	CDF Contribution (/yr)	Sequence	CDF Contribution (/yr)
BE2SEQ2	3.47E-8	BE2SEQ2	6.71E-8
BE0SEQ1	3.32E-8	BE0SEQ1	3.32E-8
BE8SEQ5	1.32E-8	BE8SEQ5	2.56E-8
TISSEQ5	4.70E-9	TE2SEQ6	1.34E-8
TE8SEQ5	3.60E-9	TISSEQ6	1.07E-8
S1SEQ3	1.27E-9	TISSEQ5	1.02E-8

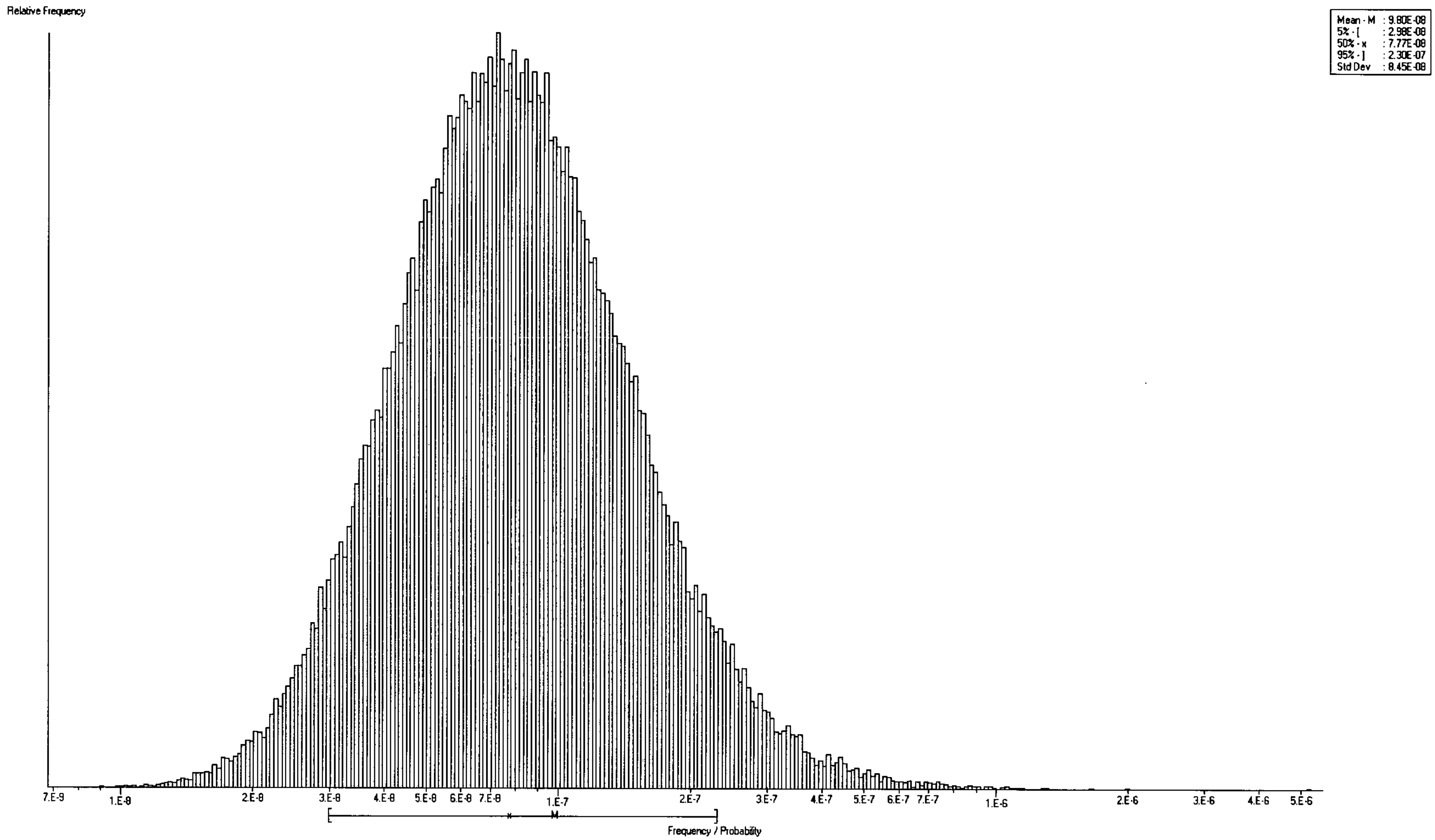

Figure 19D.10-1
Core Damage Frequency Distribution

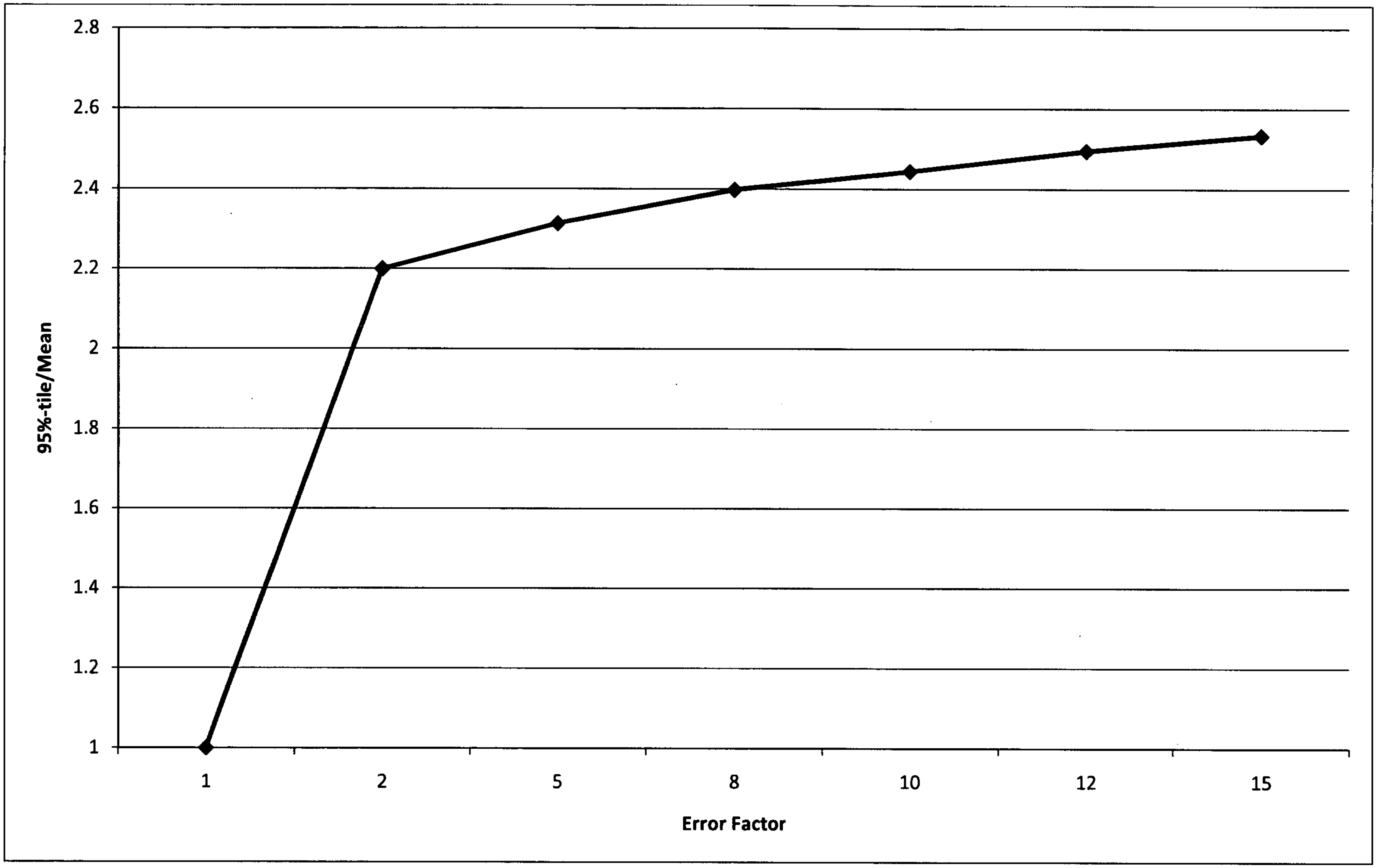

Figure 19D.10-2
Values of 95th Percentile Divided by the Mean Versus Error Factor

19D.11 ABWR COMPARISON TO GRAND GULF CDF SEQUENCES

19D.11.1 INTRODUCTION

A comparison was made of dominant ABWR PRA core damage sequences with those published in Reference 19D.11-1 for Grand Gulf Nuclear Station. This comparison was oriented toward highlighting differences in results on the basis of accident sequence initiators and specific features which provide defense against core damage.

19D.11.2 SUMMARY OF RESULTS

Table 19D.11-1 provides a comparison of ABWR and Grand Gulf PRA mean core damage frequency results grouped by accident sequence type. As can be seen, station blackout sequences dominate total CDF for each plant. The only other significant contributors to core damage for either plant involve sequences resulting in ATWS or loss of the injection function. Reasons for significant differences between the ABWR and Grand Gulf CDF sequence results are presented in the subsections which follow.

19D.11.2.1 Station Blackout

Observations regarding differences in CDF between ABWR and Grand Gulf which result from station blackout sequences (ABWR is lower by factor of 47.7) are as follows:

(1) The lower CDF for station blackout sequences in the ABWR is largely attributable to incorporation of the combustion gas turbine in the ABWR design. This design enhancement provides a diverse defense against station blackout events.

(2) To a lesser extent, incorporation of three complete divisions of high and low pressure ECCS and heat removal capability in ABWR improve the probability of safe recovery from a station blackout.

(3) The Grand Gulf PRA took credit for fire water injection as a long term follow-on to RCIC success. This lowers Grand Gulf CDF relative to ABWR for those sequences in which station blackout duration is greater than station battery life.

(4) Assumed loss of offsite power frequency is different between the two PRAs due to new data being used for the ABWR, 0.0359 for ABWR versus 0.11 for Grand Gulf, and such a difference would also contribute to the factor of 47.7 difference in calculated CDF.

19D.11.2.2 ATWS

Calculated CDF due to ATWS sequences is a factor of approximately 133 lower in the ABWR than that published in the Grand Gulf PRA. This difference in CDF between ABWR and Grand Gulf sequences is due primarily to the following:

(1) ABWR incorporates a diverse means of automatically inserting control rods in response to a demand, i.e., automatic run in with the FMCRD electric motors, which backs up the hydraulic scram feature of Grand Gulf. This feature directly reduces the frequency of ATWS sequences.

(2) A major contributor to mechanical CRD unavailability in Grand Gulf is eliminated in the ABWR FMCRD design, i.e., the scram discharge volume.

(3) In ABWR, the RPS and ARI systems are solid state logic designs with integral automatic self testing features. This decreases system electrical unavailability relative to Grand Gulf.

(4) In ABWR, SLCS system initiation is automatic with manual backup, while in Grand Gulf it is manual only. This feature further reduces ATWS frequency in ABWR relative to Grand Gulf.

(5) An additional design feature of ABWR is that ADS is automatically inhibited in the event of ATWS. This feature further reduces the contribution of ATWS sequences to CDF relative to the Grand Gulf design.

19D.11.2.3 Loss of Injection

As can be seen from Table 19D.11-1, calculated CDF due to loss of the injection function is higher in the Grand Gulf PRA by a factor of 2.3 than that calculated for ABWR. A number of differences in both system design and PRA modeling contribute to this, including the following:

(1) The total frequency of accident sequence initiators in the Grand Gulf PRA which can lead to loss of injection is approximately 7 per year, a factor of about 3.1 greater than that used in the ABWR PRA. This difference results from applying new generic initiating event data, and including an additional transient of one unplanned reactor shutdown per year in the ABWR analysis. This difference directly reduces the ABWR CDF relative to the Grand Gulf for these sequences.

(2) Credit is taken for the plant fire water system as a backup source of low pressure coolant injection in the Grand Gulf PRA. Credit was not taken for the firewater system in the ABWR Level 1 PRA. Grand Gulf CDF due to loss of injection is lowered by the inclusion of this system.

(3) Grand Gulf also credits the operator for manually cross-tying Train B of the Standby Service Water System (SSW) to the injection line of Train B of LPCI as a success path for coolant makeup injection into the reactor vessel. This further lowers the Grand Gulf CDF relative to ABWR.

(4) Compared to Grand Gulf, ABWR has one additional emergency High Pressure Core Cooling System, and thus both high and low pressure injection capability in each of the three ECCS divisions. This feature reduces loss of injection sequence CDF of ABWR relative to Grand Gulf.

(5) ABWR feedwater and condensate pumps are motor driven. This feature improves the availability of feedwater injection relative to Grand Gulf.

It is reasonable that the composite of the differences in design features, PRA modeling, and assumptions outlined above account for the lower calculated CDF value due to loss of the injection function in the Toshiba ABWR PRA.

19D.11.2.4 All Other

Sequences other than those described above such as LOCAs and transients with loss of long term heat removal were not found to be substantial contributors to calculated CDF in either PRA. Consequently, results were not examined in detail to assess the bases for differences.

19D.11.3 DISCUSSION

In order to put a comparison of ABWR and Grand Gulf PRA results into proper perspective, two separate areas need to be addressed:

(1) Design differences between the two plants which will impact both the magnitude and relative importance of the various accident sequences,

(2) Differences in the modeling, methods, and many assumptions made in each of these risk assessments.

19D.11.3.1 Design Differences

The first of these is relatively straight-forward. Compared to Grand Gulf, ABWR has an additional high pressure injection system, and thus both high and low pressure injection capability in each of the three ECCS divisions. Also, motor-driven feedwater and condensate pumps enhance reliability of the core cooling and heat removal functions in the ABWR. In addition, ABWR has decay heat removal capability in each of the three RHR loops while this capability exists in only two of the three low pressure loops in Grand Gulf. The CUW System can also be used to remove decay heat with reactor system at high pressure.

The ABWR design also incorporates a combustion turbine generator to provide a diverse source of emergency power. This system can supply power to any of the three ECCS divisions and serves as a defense against station blackout. ABWR also has an added means of initiating control rod insertion (automatic run in with the FMCRD electric motors) to provide further defense against ATWS events. This additional defense reduces the risk from ATWS events as does another design feature which automatically inhibits ADS in the event of ATWS.

Solid state logic with integral automatic self testing features improves ABWR safety system reliability. Overall, ABWR also has greater automation and less dependence on operator action than Grand Gulf.

19D.11.3.2 Modeling, Methods, and Assumptions

Differences in modeling, methods and assumptions are less easily assessed. The initiating events considered are comparable and assumed loss of offsite power frequency is somewhat lower due to new data being incporporated—0.0359 for ABWR versus 0.11 for Grand Gulf. Frequencies of other transient initiators are substantially lower in ABWR than for Grand Gulf (7.2 per year) however, since the ABWR PRA uses newer data available which concludes there are 2.24 per year resulting in reactor scram. This difference in assumptions also leads directly to the ABWR ATWS sequence initiating frequency being lower by an equivalent amount. LOCA initiator frequencies in each of the PRAs are comparable.

The overall approach to modeling, i.e., fault trees, event tree, and accident sequence development and evaluation are essentially similar. There are a number of differences in application, however, and many uncertainties in details of the Grand Gulf analyses. Examples of these differences include use of the fire water system as a backup source of low pressure coolant injection in the Grand Gulf PRA. Credit was not taken for the firewater system in the ABWR Level 1 PRA. Other differences having uncertain impact include human actions modeled and values assigned these actions, fault tree and event tree model differences, data base differences, and the details of recovery of diesel generators and offsite power.

19D.11.3.3 Results in Perspective

Table 19D.11-1 provides a comparison of ABWR and Grand Gulf PRA mean core damage frequency results. As can be seen, station blackout sequences dominate the total CDF in each case. The lower CDF for these sequences in the ABWR is largely attributed to incorporation of the combustion gas turbine, which provides a diverse defense against this eventuality.

The ABWR PRA core damage frequency of 9.80E-8 includes twenty sequences greater than 1.0E-9. These twenty sequences contribute 82.6% of the total CDF, with sixteen station blackout sequences alone contributing approximately 74%.

The sequence with the highest frequency is a long-term station blackout sequence (greater than eight hours) involving loss of all three divisions of AC power (including the combustion turbine generator). This sequence represents 34% of the total core damage frequency. The second most dominant sequence is a short-term station blackout (less than two hours), and the RCIC is in maintenance. This sequence accounts for 10.6% of total core damage frequency. The third most dominant sequence is an short term station blackout (less than two hours) followed by failure of RCIC to start and run. It represents 4.5% of total CDF. The remaining sequences are primarily transients followed by failure to provide core cooling. Contribution of ATWS sequences is less than 1% and the contribution of Class II heat removal sequences is negligible.

The overall comparison in Table 19D.11-1 attests to the importance of the diversity afforded by the combustion turbine generator and the added redundancy provided by three complete divisions of emergency core cooling (both high and low pressure) and heat removal trains.

19D.11.4 REFERENCES

19D.11-1 Eddy, C., Establishment of Model to Evaluate Plant Specific Changes, RSC Engineers, Inc. RSC 08-06, April 2010.

Table 19D.11-1
Comparison of ABWR vs. Grand Gulf PRA Core Damage Frequency Results

	Grand Gulf*		ABWR	
Initiator	CDF	%	CDF	%
Station Blackout	3.9E-6	97	8.17E-8	83
ATWS	1.1E-7	3	8.24E-10	1
Loss of Injection	1.3E-8	<1	5.73E-9	6
All other	<1.0E-8	<<1	9.79E-9	10
Total	4.00E-6	100	9.80E-8	100

*From Section 1.4.1 from Reference 19D.11-1

Table 19D.11-2 Not Used

19D.12 NOT USED

19D.13 NOT USED

Review and Quality Page

Reviewer Directions:

Provide detailed technical or global editorial comments here. Individual editorial or illustrative comments may be electronically provided (tracking) or attached to this review sheet.

Resolution Process:

Originator must provide resolutions for all comments.

Reviewer is to approve all proposed resolutions prior to completing the review process. No review is complete until this step is accomplished.

Reviewer Comment	Originator Resolution of Comment	Reviewer Approval
1. Editorial comments indicated in document	Fixed everything found.	CLE
2. Check table formatting for all tables.	Checked.	CLE
3. Equations should be numbered.	Equations numbered.	CLE
4. 19D.10-3 needs to have the data points connected with a line.	Done.	CLE
5. Remove the zero from exponents (ie 1E-04 should be reported as 1E-4) for consistency.	Done.	CLE
6. Figure numbers should not have a colon after them.	Fixed.	CLE
7. Check references, should be referencing the ABWR SSAR.	Done.	CLE