AN ASSESSMENT OF THE CONSEQUENCES

OF A SEISMIC EVENT

AT THE BIG ROCK POINT PLANT

CONSUMERS POWER COMPANY

ATTACHMENT

Consumers Power Company
Big Rock Point Plant
Docket 50-155

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I. INTRODUCTION

Review of the seismic design of the Big Rock Point Plant began in 1979 as a part of Systematic Evaluation Program (SEP) Topic III-6. Since that time, relatively significant analyses and modification of the Big Rock Point Plant have been completed. Evaluations performed to date have included 15 major structures and systems; anchorage of 56 equipment items which could have an impact on safety-related electrical equipment during a seismic event; analysis of various major mechanical equipment important to plant response following an earthquake; and, seismic qualification of cable tray and conduit raceway systems. Capital expenditures to date have totaled more than $2 million excluding work performed in house or the cost of modifications. Reevaluation of Big Rock Point seismic resistance therefore, has been the single most resource intensive of the more than 100 original SEP topics. It is also clear from the Nuclear Regulatory Staff's (Staff) draft Safety Evaluation Report (SER) dated October 19, 1982, that seismic reevaluation of Big Rock Point is also the single SEP topic with which the Staff had the most difficulty in coming to a clear conclusion.

In December, 1982 the Staff requested Consumers Power Company to investigate alternative methods of resolving the differences of opinion as to the seismic design adequacy of the plant. These alternative methods could consider the comparison of seismic risks at the Big Rock Point site with those considered acceptable at other more typical nuclear power facilities and an evaluation of the consequences of failures as a result of an earthquake or combinations thereof. In June 1983 the original weak-link analysis was submitted to the staff without any recommendations or conclusions. In November of 1985 Consumers Power Company responded to the staff's request with what is effectively an assessment of the consequences of a seismic event at the Big Rock Point Plant.

In that report the weakest links at Big Rock Point were identified. The identified pieces of equipment were assumed to fail at 0.0G ground acceleration since their true seismic capacity was unknown. Consumers Power Company committed to upgrade or analyze these pieces of equipment during the 1988 refueling outage to improve their seismic capacity.

Since this time, these items have been delayed per letter to the NRC dated May 6, 1988 due to required completion of higher plant priority projects and manpower availability.

In this report the next set of weak-links whose seismic capacities are greater than 0.0G have been identified and proposed fixes developed. The weak-links identified are listed in Table VIII-5. The conclusions from the Seismic Weak-link analysis have been presented to the Big Rock Point Technical Review Group (TRG). These proposed fixes will be ranked by TRG when cost estimates for implementation are available.

In evaluating the effect of an earthquake at Big Rock Point, consequences can be viewed in two different ways. As indicated by the Staff's request there are the consequences of the failure of various plant components and their effect on the transient response of the plant to these seismically-induced failures; there are also the consequences of the earthquake with respect to the health and safety of the public should sufficient systems and equipment in the plant fail with a resultant significant radiological release from the site. Seismic hazards levels are specified for each nuclear site by the NRC to minimize the potential for such a radiological release. The seismic-hazards curve for the Big Rock Point site is presented in NUREG-CR-1582, "Seismic Hazards Analysis" and is graphically displayed along with the hazards curve from the Big Rock Point PRA in Figure I-1 of this report. The Staff has defined the design basis ground acceleration for a given nuclear facility to be that which occurs less frequently than once every thousand years. If Big Rock Point were a new facility being constructed today, the design basis ground acceleration would be less than 0.11g (referring to the Staff's hazards curve). Given that Big Rock Point is significantly different than a typical or average facility being built or in operation today in that it is much smaller, consideration should be given to applying design basis ground acceleration criteria such that the risk or level of protection afforded the public, is commensurate with that which is considered acceptable at a newer typical or average facility.

The public dose following a significant event can be viewed in one of two ways: that dose to the public as a whole; or, that dose an individual may receive as a result of the release. The dose to the public as a whole per curie released has been published for a variety of plants in NUREG-CR-1497 "Radioactive Materials Release from Nuclear Power Plants" and NUREG-CR-1498 "Population Dose Commitment Due to Radioactive Releases from Nuclear Power Plant Sites in 1977." A summary of the information presented in these documents is shown in Table I-1. If the average dose per curie released for these 33 plants represents that dose which could be expected from an average or typical plant*, then it can be seen that a curie released at the Big Rock Point site results in a dose to the public of a factor of 14 lower than a dose from an average or typical site. This is due to the lower population density and its distribution in the Big Rock Point area. Factoring in that the Big Rock Point Plant design included only 10% of the radionuclide inventory of the typical plant, one can see that the risk to the public as a whole is a factor of 140 less severe for a given earthquake at the Big Rock Point site than at a typical site. Risk to individuals in the Big Rock Point vicinity is a different matter, however. The potential dose to an individual is independent of the population density or its distribution. Therefore, the individual risk for a given earthquake is reduced only by the fractional fission product inventory found at Big Rock Point as compared to another site, 10%. The risk to an individual in the Big Rock Point Plant area is therefore a factor of 10 less than that for an individual near a typical facility.

*Assumes meteorology, ratio of dose to inhalation, and terrain similar for these plants.

Using only the individual risk perspective, one can conclude that requiring the design basis ground motion for the Big Rock Point Plant to be equivalent to that which would be applied to a newer typical facility results in the mandatory implementation of a level of protection to the health and safety of the public at least a factor of ten more restrictive than at the newer facility. Such a requirement is not necessary, given the public risk posed by Big Rock. CPCo and the staff concluded that less quantitative, rigorous approaches to determining the seismic risks were appropriate. (CPCo to NRC, June 1, 1983).

Qualitatively, an assessment of a power plant's resistance to an earthquake can be made by using Table I-2 of this section, which gives a brief description of the effects on objects at various Modified Mercalli Intensity (MMI) categories. Using the original plant design seismic loading factor, low end Category VI effects may be expected. If the Staffs hazard curve value for Big Rock Point (if Big Rock were considered as a new facility), high end Category VI - low end Category VII effects may be expected. Since Big Rock Point is constructed of structural steel and reinforced concrete (with masonary catagorized as masonary 'C') the expected damage would include some cracks to the masonary 'C' walls and broken glassware, but no major structural damage is expected to occur. This, therefore, supports the conclusion that qualitative assessments of the need for seismic upgrading of the Big Rock Point Plant are appropriate.

An alternate approach in performing a seismic consequence analysis is to forego the use of hazards curves altogether and attempt to determine the maximum size earthquake the facility can withstand based on knowledge of plant transient response and relative structural capacity of plant systems and components. This is the approach taken in generating this report. A brief description of this approach follows and is presented in greater detail in later sections of the report.

Identifying the design features of the plant most susceptible to the earthquake, those transients which were felt most important or most likely to occur following a seismic event were selected. Three transients were chosen for study: loss of offsite power; a medium steam line break inside containment; and an Anticipated Transient Without Scram (ATWS). Loss of offsite power was chosen because of its effect on essentially every mitigating system at Big Rock Point. Other transients affect only portions of the systems available to shut down and cool the reactor. The medium steam line break inside containment was chosen for the Loss of Coolant Accident (LOCA), because adequate core cooling is dependent on most systems. Steam line breaks inside containment also require the use of enclosure sprays to maintain containment atmosphere within the environmental qualification envelope. A medium break was selected because it requires the satisfactory functioning of the Reactor Depressurization System (RDS) to prevent core damage.

The logic behind each of these transients had been developed previously in the PRA. The loss of offsite power and medium steam line break event trees were extracted from the PRA and modified to reflect the use of only those systems which could potentially be shown to survive the earthquake. The logic (by which failure of the systems were noted) in the event trees was also extracted from the PRA. Fault trees for each of these systems were reviewed and substantially simplified. As an example, the RDS tree was revised to include only a single train of power supplies, sensors, actuation cabinets, and depressurization valves because all four trains are essentially identical to each other in terms of their function, location and structural features. In other words, if one train fails as a result of the seismic event, this study assumed the likelihood of a similar failure in the other trains was quite high. Components which may have dissimilar seismic resistance (such as the two diesel generators) or have functional dissimilarities in the way they operate (such as the fire pumps and their power supplies) were not combined in this manner. Passive components and structures normally unimportant during these transients but whose failures may be made important as a result of ground motion were added to the trees (such as masonry walls).

By combining the fault trees for each core damage sequence, dependencies between the important systems were identified and a complete list of cut sets* for each sequence was produced. All the combinations of all the failures which must occur following an earthquake at Big Rock Point were thus tabulated for each core damage sequence identified in the event trees.

A conservative estimated ground acceleration which would result in the failure of each component in the fault trees was determined. Applying these accelerations to the cut set members, the acceleration at which all members of a cut set will fail was determined. This acceleration was the acceleration at which the strongest component in the cut set failed and represented the seismic resistance of that cut set. The seismic resistances for all the cut sets were ranked with respect to the size of the earthquake necessary to satisfy the cut set. Those cut sets which were satisfied at the smallest ground accelerations represent the seismic "weak-links" at the Big Rock Point Plant. These weak-links are the components and structures at which proposed modifications for upgrading the plant should be aimed. Modifications addressing members of the more seismically resistant cut sets are of little benefit unless the weaker links are also addressed.

It was recognized that the best estimate ground accelerations for many of the components identified in the attachments were not available.

*A minimal cut set is a smallest combination of component failures which, if they occur, will cause the top event to occur, VII-15 NUREG-0492 - Fault Tree Handbook - January 1981.

However, a great deal of information was extracted from the Big Rock Point design and review analyses performed to date as a part of the SEP.

Where this information was not available, a conservative approximation of the failure acceleration was used, in some cases even a value of zero. The more components with which this approach was taken the more artificially important seismic events would become. The weakest of the weak-links combined with the acceleration at which the transient occurs represented an estimate of that size earthquake which would result in a significant release. This estimate obviously does not represent a rigorous deterministic analysis. Nor does it represent a quality PRA evaluation. Risk is presented in units of ground acceleration and no mention of probability is made. The acceleration at which the weakest link fails cannot be translated to a core damage probability by use of a hazards curve because the effect of random failures on top of seismically induced failures has not been incorporated in this analysis and no effort has been made to address the effects of the uncertainties associated with the assumed component capacities.

Although this approach can be used to rank the relative importance of various combinations of failures against one another, it is, also very useful in focusing any future efforts in assessing or upgrading the seismic capacity of the Big Rock Point Plant.

The report is broken up into several sections besides the introduction. These include the following:

Section	Description
II	Selection of Initiating Events Identifies the appropriateness and completeness of selection of the loss of offsite power, medium steam line break and failure to SCRAM (ATWS) in evaluating seismic risks at Big Rock Point.
III	Identification of Important Systems and Equipment Introduction to Big Rock Point systems, structures and components important in the mitigation of the loss of offsite power and medium steam line break events.

TABLE I-1

Plant	Type	Power (Mwe)	Population 2-80 km	Person-Rem (Whole body)	Curies (Noble gas)	10^{-5} Person-rem / Curie
Arkansas	PWR	836	1.6+5	0.09	1,39+4	0.65
Big Rock	BWR	63	1.3+5	0.28	1.34+4	2.09
Browns Ferry	BWR	1067	6.7+5	2.7	1.66+5	1.62
Brunswick	BWR	790	1.9+5	6.3	2.46+5	2.56
Calvert Cliffs	PWR	850	2.4+6	0.7	2.23+4	3.14
Cook	PWR	1054	1.1+6	0.063	3.8+3	1.66
Cooper	BWR	778	1.8+5	0.017	1.27+3	1.34
Crystal River	PWR	825	2.2+5	0.014	3.35+3	0.42
Davis Besse	PWR	906	1.8+6	0.006	1.27+3	0.63
Dresden	BWR	1003	6.4+6	180.0	8.3+5	461
Duane Arnold	BWR	545	5.7+5	0.3	3.87+3	7.7
Fitzpatrick	BWR	821	8.3+5	0.54	2.33+4	2.3
Fort Calhoun	PWR	478	7.5+5	0.044	3.81+3	1.15
Ginna	PWR	490	1.2+6	0.056	3.2+3	1.75
Conn Yankee	PWR	582	3.4+6	2.2	3.12+3	70.5
Hatch	BWR	786	2.8+5	0.1	1.9+3	5.26
Indian Point	PWR	873	1.6+7	12.0	1.6+4	75.0
Kewaunee	PWR	535	6.0+5	.021	2.4+3	0.87
LaCrosse	BWR	50	3.3+5	1.6	4.25+4	3.76
Maine Yankee	PWR	825	5.7+5	.01	2.86+2	3.5
Millstone 1 & 2	PWR	1530	2.5+6	220.0	6.2+5	35.5
Monticello	PWR	536	2.1+6	0.2	6.87+3	2.9
Nine Mile Pt.	BWR	610	8.3+5	.098	3.5+3	2.8
Oconee	PWR	860	7.4+5	0.69	3.56+4	1.94
Palisades	PWR	740	1.0+6	1.5-3	59.9	2.5
Peach Bottom	BWR	1065	4.1+6	5.0	7.11+4	7.0
Pilgrim	PWR	670	4.4+6	52.0	4.13+5	12.6
Prairie Island	PWR	520	2.1+6	0.32	673.0	47.5
Quad Cities	BWR	789	6.7+5	1.3	2.56+4	5.1
Robinson	PWR	665	6.4+5	0.024	4.76+2	5.0
St Lucie	PWR	777	2.9+5	0.58	2.54+4	2.28
Salem	PWR	1090	4.9+6	0.038	19.6	194.0
Zion	PWR	1040	7.0+6	9.5	3.22+4	29.5
means		760 avg	8.1+6			30.3

TABLE I-2

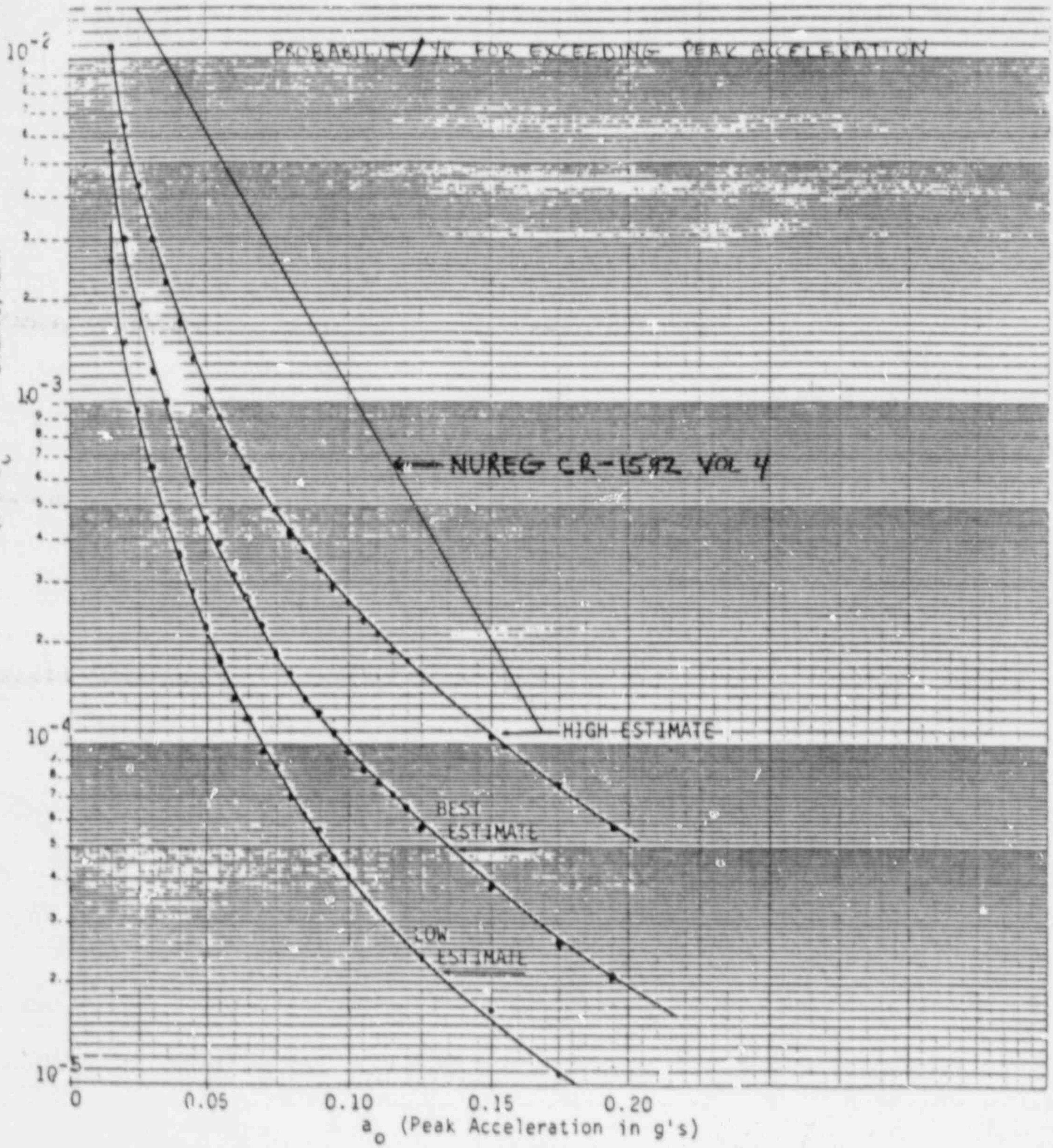
NATLSCO
National Loss Control Service Corporation

Approximate Relationships Between Intensity, Acceleration, Magnitude, and Energy Release

Modified Mercalli Intensity Scale	Description of Effects (Masonry A, B, C, and D Are Defined Below;*)	Maximum Acceleration (g)	Richter Magnitude	Energy Release (ergs)
I	Not felt; marginal and long-period effects of large earthquakes evident		M2	$\approx 10^{14}$
II	Felt by persons at rest, on upper floors, or favorably placed		M3	10^{15}
III	Felt indoors; hanging objects swing; vibration like passing of light trucks occurs; duration estimated; might not be recognized as an earthquake	0.003 to 0.007		10^{16}
IV	Hanging objects swing; vibration occurs that is like passing of heavy trucks, or there is a sensation of a jolt like a heavy ball striking the walls; standing motor cars rock; windows, dishes, and doors rattle; glasses clink; crockery clashes; in the upper range of IV, wooden walls and frame creak	0.007 to 0.015	M4	10^{17}
V	Felt outdoors; duration estimated; sleepers waken; liquids become disturbed, some spill; small unstable objects are displaced or upset; doors swing, close, and open; shutters and pictures move; pendulum clocks stop, start, and change rate	0.015 to 0.03		10^{18}
VI	Felt by all; many are frightened and run outdoors; persons walk unsteadily; windows, dishes, glassware break; knickknacks, books, etc., fall off shelves; pictures fall off walls; furniture moves or overturns; weak plaster and masonry D crack; small bells ring (church, school); trees, bushes shake	0.03 to 0.09	M5	10^{19}
VII	Difficult to stand; noticed by drivers of motor cars; hanging objects quiver; furniture breaks; damage occurs to masonry D, including cracks; weak chimneys break at roof line; plaster, loose bricks, stones, tiles, cornices fall; some cracks appear in masonry C; waves appear on ponds, water turbid with mud; small slides and caveins occur along sand or gravel banks; large bells ring	0.07 to 0.22	M6	10^{20}
VIII	Steering of motor cars affected; damage occurs to masonry C, with partial collapse; some damage occurs to masonry B, but none to masonry A; stucco and some masonry walls fall; twisting, fall of chimneys, factory stacks, monuments, towers, and elevated tanks occur; frame houses move on foundations if not bolted down; loose panel walls are thrown out; changes occur in flow or temperature of springs and wells; cracks appear in wet ground and on steep slopes	0.15 to 0.3		10^{21}
IX	General panic; masonry D is destroyed; masonry C is heavily damaged, sometimes with complete collapse; masonry B is seriously damaged; general damage occurs to foundations; frame structures shift off foundations, if not bolted; frames crack; serious damage occurs to reservoirs; underground pipes break; conspicuous cracks appear in ground; sand an mud ejected in alluviated areas; earthquake fountains and sand craters occur	0.3 to 0.7	M7	10^{22}
X	Most masonry and frame structures are destroyed, with their foundations; some well-built wooden structures and bridges are destroyed; serious damage occurs to dams, dikes, and embankments; large landslides occur; water is thrown on banks of canals, rivers, lakes, etc.; sand and mud shift horizontally on beaches and flat land; rails are bent slightly	0.45 to 1.5		10^{23}
XI	Rails are bent greatly; underground pipelines are completely out of service	0.5 to 3	M8	
XII	Damage nearly total; large rock masses are displaced; lines of sight and level are distorted; objects are thrown into air	0.5 to 7	M9	10^{24}

*Masonry A. Good workmanship, mortar, and design; reinforced, especially laterally, and bound together by using steel, concrete, etc.; designed to resist lateral forces.
Masonry B. Good workmanship and mortar; reinforced, but not designed in detail to resist lateral forces.
Masonry C. Ordinary workmanship and mortar; no extreme weaknesses like failing to tie in at corners, but neither reinforced nor designed against horizontal forces.
Masonry D. Weak materials, such as adobe; poor mortar; low standards of workmanship; weak horizontally.

Figure I-1


PROBABILITY/YR FOR EXCEEDING PEAK ACCELERATION
NUREG CR-1592 VOL 4
HIGH ESTIMATE
BEST ESTIMATE
LOW ESTIMATE
EVENTS/YEAR
(a > a$_o$)
10^{-2}
10^{-3}
10^{-4}
10^{-5}
0
0.05
0.10
0.15
0.20
a$_o$ (Peak Acceleration in g's)

II. SELECTION OF INITIATING EVENTS

In this section, a discussion of the possible initiating transients which could occur at the Big Rock Point Plant as a result of a seismic event will be presented. An initiating event is a failure which requires active response of systems and/or the operator in order to prevent an accident sequence. A number of initiating events were identified as a part of the Big Rock Point PRA (Appendix XII). The development of this list of events was an iterative process utilizing a variety of industry sources (see Table II-1), Big Rock Point Plant specific incidents and the event trees and fault trees of Appendixes I and II of the PRA. This list is considered sufficiently complete to use as a basis for studying the response to a seismic event of the Big Rock Point Plant.

Unless the fragility of the equipment required to fail in order to initiate a transient is assessed, the assumption must be made that any of the initiating transients identified in Appendix XII of the PRA is possible. It is desirable to limit the number of systems required for study following a seismic event due to the complexity and cost of evaluating the seismic resistance of structures and components associated with those systems. For this reason an attempt will be made to categorize the events in PRA Table XII under limited number of transient and accident headings and choose a most limiting event for each category that requires sufficient systems and equipment to completely characterize the Big Rock Point Plant operating response to an earthquake regardless of the transient that occurs.

Four major category headings will be chosen under which the transients and accidents will be compiled: Anticipated Transients; LOCA; reactivity transients; and, external events. The last category, external events, is the category in which earthquakes reside. Other external events include wind, tornado, fire, toxic chemical accidents, airplane crashes, flooding, etc, all believed to be extremely remote simultaneous with an earthquake. This category will not be considered further.

Tables II-2, 3 and 4 contain a tabulation of transients under each of the three remaining categories. The lists begin with what is considered the most limiting transient representative of all the other transients in that category. The three most limiting transients are: Loss of Offsite Power; Medium Steam Line Break Inside Containment; and, ATWS for the anticipated transient, LOCA and reactivity transient categories, respectively. A statement as to why each event listed is believed to be conservatively bounded by the most limiting transient is presented following the transient description.

Only one transient, simultaneous closure of the recirc loop valves was not categorized. Based on both the availability of ac power in order to initiate this transient and the relatively significant amount of time the operator has to terminate the transient by reopening the valves or providing a makeup water source, this transient is probably less limiting than a loss of offsite power. However, as the power to the control circuitry for these valves is normally disabled through the hand switch, a closure of these valves resulting from the earthquake was not considered possible and the transient was not categorized.

TABLE II-1
INDUSTRY AND BRP TRANSIENT INITIATOR REFERENCES

GE Standard SAR

GE Generic ATWS Report NEDO-10349

WASH - 1400

EPRI Report on ATWS EPRI NP-801

Standard Review Plan Chapter 15

Systematic Evaluation Program Topics - Item XV

GE Service Information Letters

EPRI LER Data Base

BRP Control Room Logbooks and SCRAM Reports

BRP LERS

TABLE II-2
ANTICIPATED TRANSIENTS POTENTIALLY RESULTING FROM AN EARTHQUAKE

Description	Discussion
Loss of Offsite Power (LOSP)	Most Limiting Transient
Turbine Trip	Considered as a Part of LOSP
Load Rejection	Considered as a Part of LOSP
Loss of Main Condenser	Considered as a Part of LOSP
MSIV Closure	Considered as a Part of LOSP
IPR Fails Closed	Considered as a Part of LOSP
Miscellaneous Plant Occurrences	No Out-of-Tolerance Conditions Necessarily Exist
Manual SCRAM	No Out-of-Tolerance Conditions Necessarily Exist
Spurious Nuclear Instrumentation	No Out-of-Tolerance Conditions Necessarily Exist
Loss of Feedwater	Considered as a Part of LOSP
Loss of One Feed Pump	Considered as a Part of LOSP
Feedwater Controller Failed Closed	Considered as a Part of LOSP
Recirc Pump Trip	Considered as a Part of LOSP
Recirc Pump Shaft Seizure	Similar to Recirc Pump Trip Only With One Pump
Loss of Auxiliary Power	Considered as a Part of LOSP
Loss of DC Power	DC Power Dependencies Considered as a part of LOSP
Loss of Misc Power Panels	Considered as a Part of LOSP
Service Water Failure	Considered as a Part of LOSP
Instrument Air Failure	Considered as a Part of LOSP

TABLE II-3
LOSS OF COOLANT ACCIDENTS POTENTIALLY RESULTING FROM AN EARTHQUAKE

Description	Discussion
Medium Steam Line Break Inside Cont (MSLB)	Most Limiting Transient - Requires Enclosure Spray, Core Spray, RDS and Post-Incident Cooling
Feedwater Maximum Demand	Assumed to Hydro Primary System and Results in Safety Relief Valve Sticking Open (Same as MSLB)
IPR Fails Open	Blowdown through Turbine Requires Same Systems as MSLB Except Enclosure Spray
Inadvertent Opening of Safety Valve	Requires Same Systems as MSLB
Spurious Opening of Bypass Valve	Same as IPR Failure
Spurious RDS	Requires Same Systems as MSLB
HELB in Recirc Pump Room	Requires Same Systems as MSLB
HELB in Pipe Tunnel	Requires Same Systems as MSLB Except Enclosure Spray
Interfacing LOCA (Inside Cont)	Requires Same Systems as MSLB Except Enclosure Spray
Large LOCA (Inside Cont)	Requires Same Systems as MSLB Except Enclosure Spray
Medium LOCA (Inside Cont)	Requires Same Systems as MSLB Except Enclosure Spray
Large SLB (Inside Cont)	Requires Same Systems as MSLB Except Enclosure Spray
Small SLB (Inside Cont)	Requires Same Systems as MSLB
Large SLB (Outside Cont)	Requires Same Systems as MSLB Except Enclosure Spray
Medium SLB (Outside Cont)	Requires Same Systems as MSLB Except Enclosure Spray
Small SLB (Outside Cont)	Requires Same Systems as MSLB Except Enclosure Spray

TABLE II-4
POTENTIAL REACTIVITY TRANSIENTS RESULTING FROM AN EARTHQUAKE

Description	Discussion
ATWS	Most Limiting Transient
Loss of Feedwater Heater	Momentary Power Transient
Core Spray Injection at Start-Up	Momentary Power Transient
Rod Withdrawal at Start-Up	Momentary Power Transient
Rod Withdrawal at Power	Momentary Power Transient
Rod Drop	Momentary Power Transient
Idle Recirc Loop Start-Up	Momentary Power Transient

III. IDENTIFICATION OF IMPORTANT SYSTEMS AND EQUIPMENT FOLLOWING A SEISMIC EVENT

Three transients were identified in Section II as being those incidents which would require sufficient systems and equipment to characterize completely Big Rock Point Plant operational response following a site seismic event. These transients were: a medium steam line break inside containment; a loss of offsite power; and an ATWS. In this section, a description will be developed of each system important for providing adequate core cooling following any of these three transients. This description will include identification of major components making up each system, location of the equipment with respect to major plant structures and development of success or failure criteria associated with each system function. A list of major active components described in this section and supporting instrumentation and equipment is provided in Table III-1.

Fire Protection System

To provide core cooling following a major LOCA or the actuation of the RDS, the Big Rock Point Plant is equipped with a low-pressure core-spray system referred to as the Fire Protection System (FPS). The FPS is located in four distinct areas of the plant as shown on the system line diagram presented in Figure III-1. These four areas include the screenhouse, turbine building, reactor containment and core-spray heat-exchanger room located within the fuel cask loading dock structure. The FPS consists of two fire pumps (one diesel-powered and one ac-powered). Actuation of the FPS is automatic on low-steam drum level (≤17" below steam drum center line) or low-fire-header pressure (<70 psi electric pump or <60 psi diesel pump). Manual actuation of the FPS can be accomplished from either the RDS Panel in the control room or from local control panels in the screenhouse. The FPS draws water from beneath the screenhouse and discharges to the yard piping through carbon steel and cast-iron piping components within the screenhouse.

The yard piping is buried cast iron. It encircles the turbine and reactor buildings and provides three paths into the plant through which core spray water can flow.

Two of the paths enter the plant through the turbine building. Turbine building fire piping is carbon steel and includes threaded joints and mechanical couplings. The core spray water passes through basket strainer filters prior to entering the containment through sections of welded piping in the turbine building pipe tunnel.

FPS inside the containment is primarily welded carbon steel piping. Water flow to the reactor vessel following reactor depressurization is through two core spray piping paths. Each path contains two normally closed motor-operated valves which automatically open on combined low-reactor-water level (≤2'9" above the core) and low-reactor pressure (≤200 psig). One pair of valves is dc-powered (MO7051 and MO7061) while the other pair is ac-powered (MO7070 and MO7071). Operation of the dc-powered valves permits water flow to a ring sparger through a penetration in the side of the reactor vessel. The ac-powered valves feed a nozzle located in the reactor vessel head. Spray flow distribution from either of these sources is sufficient to provide adequate cooling of all the fuel assemblies.

A third path for FPS is actuated through the fuel-cask loading-dock area to the containment by opening a dc-powered motor-operated valve (MO7072). Manual operation of the valve also can be accomplished locally.

Successful operation of the FPS during a LOCA requires the automatic starting of at least one fire pump, the opening of either the ac or dc motor-operated valves in the core spray lines and maintaining the integrity of the FPS from the fire pumps to the containment building through the screenhouse, yard loop and turbine building. If time permits, the fire pumps can be started within minutes from the control room by operator action. If a significant amount of time is available, the pumps may be started locally in the screenhouse and any FPS failures which might have occurred in the turbine building can be isolated from the yard and bypassed by opening the dc-powered valve (MO7072). Success of this alternate path is considered effective only if the fire piping leading to and in the containment are isolated from ruptured piping so as to prevent significant diversion of core spray water from the containment.

If coincident with a loss of offsite power, success of electrical components (electric fire pump, ac core-spray valves, and fire pump manual-start circuitry in the control room) requires the operation of the emergency diesel generator, the emergency electrical bus and associated electrical switchgear. Again, if significant time is available, the standby diesel generator can be started manually and pick up important loads after connection to the emergency bus.

Reactor Depressurization System

In order to permit the functioning of the low-pressure core spray system during the course of a transient, a method for lowering the pressure of the primary coolant system is necessary. At Big Rock Point, this function is provided by the RDS which allows a means of rapid depressurization of the primary coolant system should a loss of coolant inventory occur. The major components of the RDS are located in the containment, service building, turbine building and screenhouse.

The RDS is shown on a system line diagram of the primary system in Figure III-2. This system consists of four 6-inch diameter pipes each containing two-in-series normally closed valves. The first valve (CV4180 through 4183) is an air-operated fail-open isolation valve. Air pressure is provided to the valve operator through a normally de-energized solenoid valve (SV 4980 through 4983). The second valve, referred to as the depressurization valve, is a pilot-operated solenoid valve which is closed when de-energized. Depressurization of the primary coolant system through these lines requires energization of each solenoid valve resulting in the opening of the isolation valve and depressurization valve in each line. Three of the four paths must be opened in this manner to result in a sufficiently rapid depressurization of the reactor to permit core spray.

Actuation signals are provided through solid-state circuitry located in actuation and sensor cabinets in the service building. Sensors for satisfying RDS actuation logic include steam-drum level instrumentation, reactor level instrumentation (each located in the containment), fire-pump discharge pressure instrumentation (located in the screenhouse) and the two-minute timers (located in the control room). Power sources for energizing the solenoid valves are derived from a set of UPS (uninterruptible power supplies) batteries located in the turbine building. System operation is accomplished as follows. A loss of primary coolant system inventory will result in a lowering of the level in the steam drum. At 17" below the center line of the drum, an actuation signal is sent to each of the two fire pumps in the screenhouse and a two-minute timer starts. The low-steam-drum level signal, fire pump discharge pressure (≥100 psig) and the timing out of the two-minute timers are three of the four signals required to provide an actuation signal to the RDS valves. When the primary coolant level reaches 2'9" above the top of the core, the fourth signal is provided by the reactor level instruments completing the actuation logic and opening RDS valves depressurizing the reactor.

Success of this system requires proper functioning of two of four sets of steam-drum-level instruments, reactor level instruments, fire pump pressure switches and two-minute timers. The UPS power sources for each of the preceding sets of RDS instrumentation must be available and at least one fire pump must start. Proper functioning of each pair of valves in three of the four blowdown paths must also occur.

Time permitting, the operator can manually start a fire pump and actuate the RDS from the control room. Use of this manually initiated method of blowdown in conjunction with a loss of offsite power requires the functioning of the emergency or standby diesel generators, the emergency bus and associated switchgear.

Enclosure Spray System

In the event that the earthquake results in the rupture of a steam line inside containment, the sticking open of a safety relief valve or the spurious actuation of an RDS train, it is possible that fluid escaping the primary coolant system could super heat resulting in an escalation of the containment atmospheric temperature above the 235°F environmental qualification temperature for important RDS and core spray system components. The purpose of the enclosure (ie, containment) spray system is to spray the containment atmosphere with water from the FPS, quenching the superheated steam and thereby lowering the containment atmosphere below the qualification temperature. The enclosure spray system is included as a part of the line diagram in Figure III-1.

The enclosure spray system consists of two spray headers located at the top of the reactor building internal structure (ie, containment) with each header controlled by a normally closed motor-operated valve. One valve (MO7064) is automatically actuated by containment pressure switches located on the outside cable penetration area (~ 2.2 psig). This motor-operated valve requires a dc power supply. An ac-powered motor-operated valve (MO7068) controls the backup containment spray header and is manually actuated from the control room should the dc valve fail to operate.

Success of the enclosure spray system requires that either of the two motor-operated valves open. The enclosure spray system additionally relies on the same FPS inside and outside containment that the core spray system requires for successful operation and includes the operation of either of the two fire pumps. Operation of the ac motor-operated valve requires that the emergency diesel generator, the emergency bus and the associated switchgear are operational when normal ac power is unavailable.

This system is not required when the primary coolant loss results from rupture of piping normally containing saturated liquid or a full blowdown of the RDS.

Post-Incident System

Following RDS and core spray actuation, the containment will begin to fill with water coming from the primary coolant system, the core sprays and the enclosure sprays. On reaching an elevation in the containment between 587 ft and 590 ft (approximately 260,000 gallons), the operator is required to terminate water addition to the containment and initiate decay heat removal by means of the post-incident system. Switchover to post-incident recycle occurs on the order of 4 to 24 hours depending on the nature and location of the source of primary coolant inventory loss.

The major components associated with post-incident recycle are located in the fuel-cask loading-dock area and shown in the line diagram presented in Figure III-1.

Post-incident system initiation consists of starting one of the two core spray pumps, drawing water from the bottom of containment and pumping it through the core spray heat exchanger and core spray system piping back to the core spray and containment spray headers. Either core spray pump can be started remotely by a hand switch from the control room. To remove heat from the containment water, FPS water is supplied to the shell of the core-spray heat-exchanger through motor-operated Valve MO7066 (an ac-powered valve remotely actuated from the control room). Fire water addition to the containment is terminated by closing hand-operated Valves VFP-29 and VFP-30 located in the turbine building.

Successful operation of this system requires the availability of a single fire pump, screenhouse fire piping (the yard loop), core spray piping in the fuel-cask loading-dock area, core spray piping inside containment, the functioning of MO7066 and the operation of a single core-spray pump. Operation of the ac-powered equipment (core spray pump, MO7066, and the electric fire pump) will depend on the emergency or standby diesel generators, emergency bus and miscellaneous 480 V ac distribution equipment if coincident with offsite power failure. Failure of remote operation of MO7066 can be overcome by remote operation of MO7080, which is parallel to MO7066. Local manual operation of either valve is an option. Rupture of underground yard piping also can be bypassed by isolating the yard piping from the screenhouse and connecting fire hose from the hose manifold on the side of the screenhouse to Valve VPI-10 in the shell side of the heat exchanger.

Operation of the system is assumed to be required for a minimum of one month following its initiation at which time natural heat losses from the primary coolant system and containment are sufficient to ensure removal of decay heat.

Primary Coolant System Isolation

Should the transient which occurs following the seismic event not result in a significant loss of primary coolant inventory within the containment, it is beneficial to ensure that no such loss is occurring outside the containment by isolating the main steam line. This function is accomplished by closing the MSIV (MO7050). The MSIV is a dc-powered motor-operated gate valve located just inside the containment shell (see Figure III-2). Automatic operation of this valve requires that electrical equipment associated with the reactor protection system (RPS) be functional (in the event of a loss of offsite power, voltage to the RPS will be lost, automatically sending a signal closing the MSIV) and that dc power is available. (The dc power source is located in the alternate shutdown building). If time permits, the valve can be actuated manually from the control room, or from the alternate shutdown building.

Emergency Condenser System

Assuming a LOCA inside containment is not occurring and successful primary system isolation is accomplished, the emergency condenser system is used for removal of decay heat and ultimately the cooldown of the primary coolant system. The emergency condenser is shown in both line diagrams found in Figures III-1 and III-2. The emergency condenser system consists of a large tank (ie, shell) of water in which two tube bundles are immersed and through which steam from the primary system flows by natural circulation, condenses and returns to the steam drum.

Each emergency condenser tube bundle has a normally open ac-powered inlet valve (MO7052 and MO7062) and a normally closed dc-powered outlet valve (MO7053 and MO7063). The dc-powered motor-operated valves receive a signal to open when a high reactor pressure is experienced

(≥1450 psia) or a loss of station power occurs. Either emergency condenser tube bundle is capable of removing sufficient heat from the primary coolant system to accommodate decay-heat generation and prevent the actuation of steam drum safety relief valves (lowest valve set point is 1550 psi). The emergency condenser shell contains a stored water supply capable of removing the equivalent of four hours of decay heat. To prevent a reactor pressure rise to the safety relief valve set point due to depletion of the emergency condenser shell inventory, a source of makeup water are available through a manually actuated dc-powered valve (SV4947) which can be operated from the control room or the alternate shutdown building. The emergency condenser makeup line draws its water supply from the enclosure spray headers upstream of the dc power-operated Valve MO7064.

Success of the emergency condenser requires automatic or manual operation of either of the two dc-operated outlet valves, manual actuation of the makeup valve from the enclosure spray headers, the dc power source is available from the alternate shutdown building, and maintaining the integrity of the makeup pipe, core spray piping inside and outside containment, yard piping, screenhouse fire piping and at least one fire pump. As was assumed during operation of the core spray system, if turbine building piping failures occur and fire water is supplied to the containment through MO7072, makeup to the emergency condenser shell is successful only if ruptured turbine building piping is isolated from the FPS to prevent diversion of fire water.

If coincident with a loss of offsite power, emergency condenser operation has a dependency on emergency ac power sources (either diesel generator, emergency bus and switchgear) only for operation of the electric fire pump.

Control Rod Drive Makeup

The purpose of the control rod drive system following an earthquake is to supply high-pressure low-volume makeup to the primary coolant system in the long term to overcome shrinkage due to cooldown and normal primary coolant system leakage. Shrinkage of the primary coolant by itself will not result in uncovery of the core but when combined with sufficient primary coolant leakage (<1 gpm unidentified and <10 gpm identified at 1350 psi reactor pressure) could result in extremely low reactor water levels. These levels are not expected to occur for a day or more after shutdown of the reactor at these leakage rates. The core spray system can be manually initiated following cooldown of the reactor to less than 150 psi, but the control rod drive system is the preferred source of makeup as it can be initiated at any time during the cooldown, even with the reactor at elevated pressures.*

Major components of control rod drive makeup are located in the reactor and turbine buildings. The system includes a 25,000 gallon

*It is emphasized that control rod drive makeup is a backup to core spray.

capacity condensate storage tank (which is normally more than half full), welded carbon steel piping located underground, the condensate pump room and the pipe tunnel in the turbine building, and the recirc pump room in the reactor building. Water flows through the underground piping to an air-operated valve (CV4090) which opens on a loss of instrument air or a loss of the condensate pumps to the suction of the control rod drive pumps (25 gpm positive displacement pumps) inside containment. From there the water is pumped to the reactor through the control rod drive mechanisms and the reactor cleanup system.

Success of this system requires periodic operation of at least one control rod drive pump and the opening of the normally closed Valve CV4090 if the condensate pumps are not running or instrument air is not available. Control rod drive pump suction and discharge piping must remain intact as must the condensate storage tank.

Reactor Trip

The electrical equipment and hardware required to ensure shutdown of the reactor by automatic rapid control rod insertion following an earthquake includes the RPS, control rod drive SCRAM piping, air-operated SCRAM valves and the CRD mechanism. Both channels of RPS circuitry are located in the control room and include the logic and relays for terminating power to the SCRAM valve solenoids. Reactor protection sensors most likely to trip the reactor following an earthquake include any or all of the following: reactor pressure (≧1400 psia); low steam drum level (≦8" below drum center line); low reactor level (≦2'9" above the core); high condenser pressure (≧8" Hga); high enclosure pressure (≦1 psig); loss of voltage to reactor protection system (≦52 V); high neutron flux (≧120); or manual SCRAM depending on the nature of the transient which is initiated by the earthquake.

Control rod drive hydraulic equipment which must function following the earthquake includes the piping between the control rod drive mechanism and the SCRAM dump tank and the air-operated SCRAM discharge valves (CVNC10) including the solenoid valves from the instrument air header (SVNC27). The hydraulic piping need not remain intact following the earthquake but must not fail such that flow from the control rod drive mechanism to the SCRAM dump tank is prohibited (such as by crimping of the piping). With the reactor at pressure (>450 psig), the reactor pressure alone is sufficient to permit control rod insertion without the aid of the SCRAM piping to the control rod drive mechanisms or the nitrogen-filled accumulators.

Success of this system requires the tripping or loss of power to the RPS terminating power to the SCRAM solenoid valves causing them to close venting the air from the SCRAM valves. The SCRAM valves fail open on loss of air venting the top of the control rod drive piston to the SCRAM dump tank forcing the control rod into the core due to the resulting large differential pressure across the control rod drive piston. Normal core geometry must be maintained by reactor internals to permit insertion of the control rods.

Success of this system has no dependency on the availability of any ac or dc power supplies.

TABLE ?II-1

BIG ROCK POINT PLANT

IMPORTANT ACTIVE COMPONENTS FOLLOWING A SEISMIC EVENT

Major Active Components	Instrumentation and Equipment Required to Activate Component	Additional Instrumentation Useful in Manual Activation of Components
Electric Fire Pump	PS615 (Fire Pump Discharge Pressure) LT3184 A-3187 (RDS Drum Level) RDS Actuation Cabinet 6.1 Fire Pump Control Panel C17 UPS Batteries	LIRE19A & B (Drum Level Indication) LT-3188 (ASDB Drum Level Indication) /
Diesel Fire Pump	PS612 (Fire Pump Discharge Pressure) LT3184-3187 (RDS Drum Level) RDS Actuation Cabinet 5.1 Fire Pump Control Panel UPS Batteries	LIRE19A & B (Drum Level Indication)
Core Spray Pumps	1A and 2A Buses 2B Bus	LT3171 & 3175 (Reactor Building Level)
Control Rod Drive Pumps	1A and 2A Buses 2B Bus or ASDB Disc-1441	LT3180-3187 & LIRE19A & B (Drum Level Instrumentation) /
Emergency Diesel Generator	Control Panel C18 EDG Batteries UPS A Batteries Undervoltage Relay	
Standby Diesel Generator	Standby EDG Batteries Standby EDG Transformers	

TABLE III-1 (Continued)

Major Active Components	Instrumentation and Equipment Required to Activate Component	Additional Instrumentation Useful in Manual Activation of Components
SV4984-4987 (Depressurization Valves)	RDS Actuation & Sensor Cabinets, UPS Batteries LT3184-3183 (Drum Level) LT3180-3187 (Reactor Level) PS789-796 (Fire Pump Pressure) 2 Minute Timers	LSRE09 A-D (Reactor Level)
CV4180-4183 (Isolation Valves	SV4980-4983 Actuation & Sensor Cabinets UPS Batteries LT3184-3183 (Drum Level) LT3180-3187 (Reactor Level) PS789-796 (Fire Pump Pressure) 2 Minute Timers	
CVNC10 (SCRAM Valves)	SVNC27A & B Reactor Protection Channels A & B Reactor Protection Sensors: PS664-667 (Enclosure Pressure) LSRE09A-D (Reactor Level) LSRE06A-D (Drum Level) PSRE07A-D (Reactor Pressure) PS654-657 (Condenser Pressure) Flux Monitors: RH01A-B (UV Contacts)	
MO7070 and 7071 (Core Spray)	LSRE09B, D, F&H (Reactor Level) PSIG11B, D, F&H (Reactor Pressure) Contact Relay for MO7070 and 7071 Bus 2B	
MO7051 and 7061 (Core Spray)	LSRE09A, C, E&G (Reactor Level) PSIG11A, C, E&G (Reactor Level) dc Bus D01 Station Batteries	

TABLE III-1 (Continued)

Major Active Components	Instrumentation and Equipment Required to Actuate Components	Additional Instrumentation Useful in Manual Actuation of Components
MO7064 (Enclosure Spray	PS636A & B } PS7064A & B } (Enclosure Pressure) dc Buses D10, D02 and D01 Station Batteries	
MO7068 (Enclosure Spray)	Conte Relay for MO7068	PI367 (Enclosure Pressure)
MO7066 (Core Spray Htx)	Co act Relay for MO7066 2A Bus 2B Bus	LT3171 and 3175 (Reactor Building Level)
MO7080 (MO7066 Backup)	Contact Relay for MO7080 2A Bus 2B Bus	LT3171 and 3175 (Reactor Building Level)
MO7072 (Backup Core Spray Supply)	dc Buses D10, D02, and D01 Station Batteries	
MO7053 and 7063 (Emergency Condenser)	PSRE07A-D (Reactor Pressure) Bus D12 Alternate Shutdown System Batteries	PSID28E PIA49 (Reactor Pressure)
SV4947 (Emergency Condenser Makeup)	dc Buses D12 Alternate Shutdown System Batteries	LS3550 (Emergency Condenser Shell Level) PIA49 (Reactor Pressure) PT 1.3 (ASD Reactor Pressure)

FIGURE III-1

REF. 0740640123

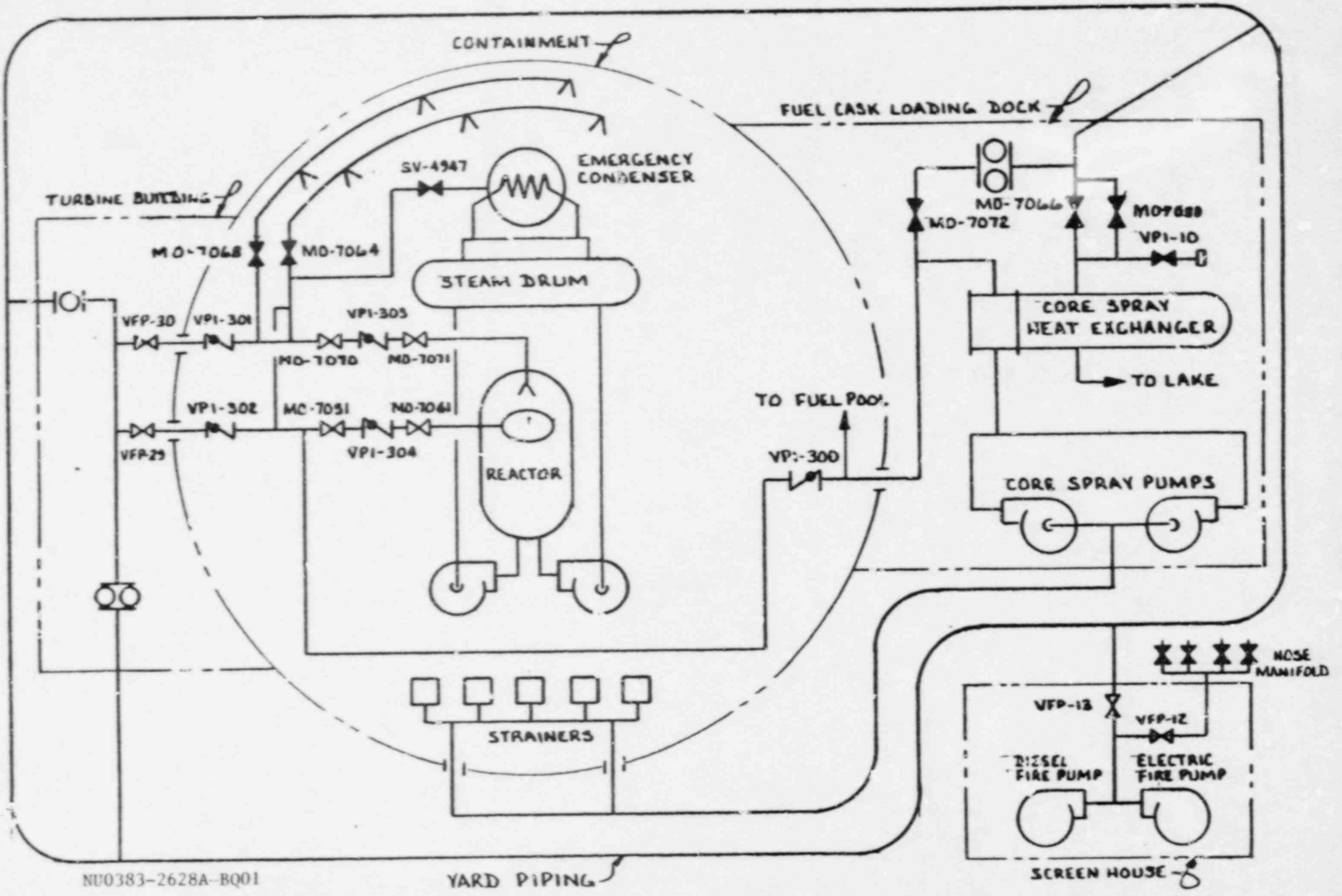


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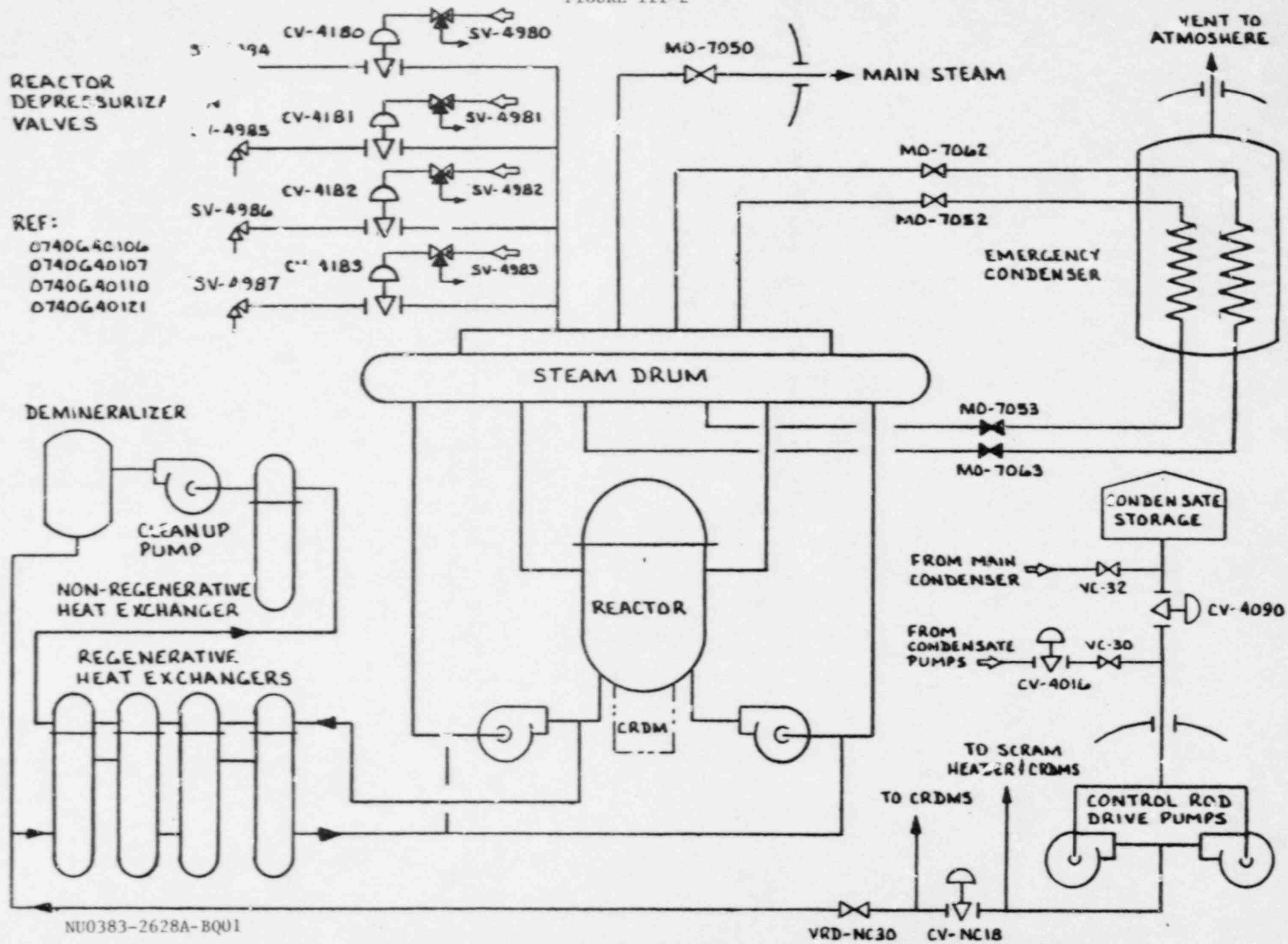
FIGURE III-2



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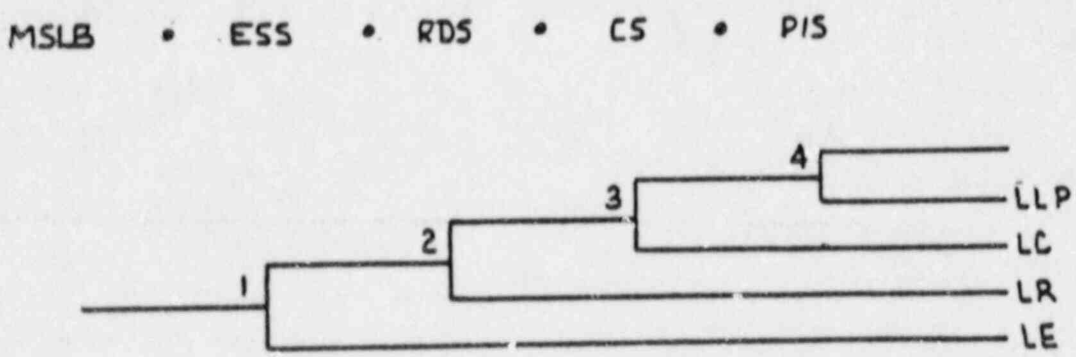
IV. BIG ROCK POINT PLANT RESPONSE TO A SEISMIC EVENT (EVENT TREE DESCRIPTION)

In this section, plant response to two of the three initiating events identified in Section II is described in detail. The loss of offsite power and medium steam line break inside containment transient descriptions take the form of event trees. Logic for these trees was presented initially in Appendix I of the PRA. This logic was modified for the purpose of this study to reflect only those systems which were felt to be easiest to show to be resistant to the forces of the earthquake, those systems are described in Section III. More detail was added to the "loss of offsite power tree" than exists in the PRA for ease of identifying dependencies between systems within the tree and simplifying the process of identifying the seismic weak links where they exist in the plant design. For loss of offsite power sequences leading to the use of long-term cooling methods involving the emergency condenser, a detailed long-term cooling event tree was developed. This tree reflects the need for long-term cooling makeup from the control rod drive system and reflects the use of the RDS, the core spray system and the post incident cooling system as redundant systems to this makeup source. The following transient descriptions begin with an event tree diagram, definitions of event tree headings and are followed by a discussion of each branch point of the event tree.

The third transient, ATWS, is not developed in this section. The potential for this transient to occur will be presented in Section V by identifying those equipment failures which must occur to result in SCRAM failure. Discussion of additional evaluations which will ensure that a failure to SCRAM will not occur will then be presented in Section VIII.

Seismically Induced
Medium Steam Line Break
Inside Containment



MSLB - MEDIUM STEAM LINE BREAK
ESS - ENCLOSURE SPRAY SYSTEM
RDS - REACTOR DEPRESSURIZATION SYSTEM
CS - CORE SPRAY SYSTEM
PIS - POST INCIDENT SYSTEM

FIGURE IV - 1

BP 1 Enclosure Spray

At Branch Point 1 (Figure IV-1) a break in a steam line has occurred as a result of an earthquake. Offsite power is assumed to have been lost also as a result of the earthquake. Because the steam superheats as it leaves the primary system the containment temperature begins to rise toward the environmental qualification temperature of 235°F. On containment pressure attaining 1.7 psig a signal is sent via PS636A & B and PS7064A & B to open MO7064 actuating enclosure sprays. Success at this branch point implies the availability of MO7064, the containment pressure switches, the valve's dc power supply, the integrity of fire piping in the containment, turbine building, yard and screenhouse, and the operation of either of the two fire pumps. The ac fire pump will require the operation of the emergency diesel and 2B bus. Automatic starting of the fire pumps on low drum level or low fire header pressure due to enclosure spray actuation will be required. Should MO7064 fail to open the operator may manually actuate the Backup Enclosure Spray Valve MO7068 which is an ac-powered valve with dependencies on emergency power.

Failure to deliver water to the enclosure spray header is assumed to result in the exceeding of the environmental qualification temperature resulting in the failure of important RDS or core spray equipment by Sequence LE. Successful enclosure spray leads to RDS operation at Branch Point 2.

BP 2 RDS

As the blowdown progresses a low drum level signal at 17" below drum center line will occur starting the fire pumps and a two minute timer in the RDS logic. Drum level will continue to fall to the low reactor water set point of 2'9" above the core. On reaching this level, coincident fire system pressure >100 psi, low drum level and two-minute timer closure a signal will be sent to the solenoid valves on the RDS isolation and depressurization valves. Opening the valves rapidly depressurizing the reactor allowing low-pressure core spray. Successful system operation requires three of the four RDS valve trains to operate, two of four sets of drum level, reactor level, fire pressure and timer instrumentation to operate, and one of the two fire pumps to run. Operation of the sensors and valves depends on the UPS, operation of the electric fire pump depends on emergency ac power. Failure of this system is assumed to result in inadequate depressurization of the reactor and at least limited core damage before a pressure is low enough to allow core spray (sequence LR). Success leads to the need for core spray actuation.

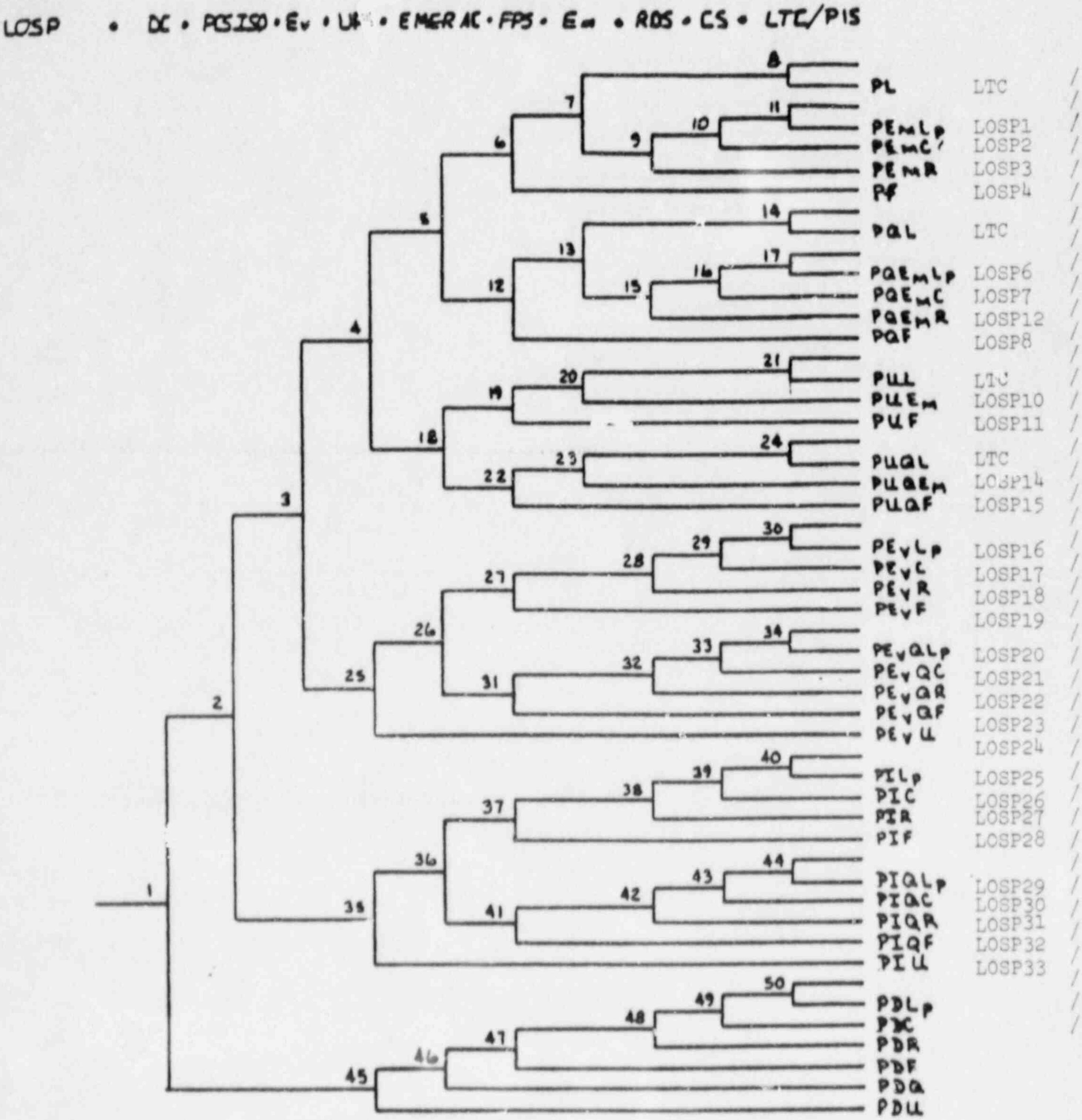
BP 3 Core Spray

A low reactor level coincident with reactor depressurization to <200 psig leads to actuation of ac and dc-powered core spray valves permitting fire system flow through a spray nozzle in the vessel head or a ring sparger around the vessel perimeter. Flow >290 gpm through either of these lines is sufficient to cool an uncovered core. Success of this system depends on opening either the ac or dc-powered valves, the integrity of fire piping in the containment, turbine building, yard and screenhouse and the operation of at least one fire pump. The ac valves have a dependency on the operation of the diesel generator and the 2B bus. Failure of the core spray system leads to an inadequately cooled core by sequence LC, success ultimately leads to the need for post-incident long-term cooling.

BP 4 Post-Incident System

Water will enter the containment by way of primary coolant loss, core spray and enclosure spray. On the addition of ~ 260,000 gallons of water to the containment (587 ft elevation) the operator will place the post-incident system into service and isolate the fire water system water addition to the enclosure by closing hand-operated Valves VFP-29 and 30. Success of this system implies the operation of one core spray pump, the integrity of pump suction piping from the containment sump, discharge piping to the heat exchanger, removal of heat by way of the core spray heat exchanger and the integrity of post-incident system piping back to the core spray system inside containment. Heat removal through the heat exchanger implies the addition of fire water to the heat exchanger shell through MO7066 or parallel valve MO7080 integrity of the yard loop or bypass with a fire hose, integrity of screenhouse fire piping and the operation of at least one fire pump. MO7066, the core spray pumps and the electric fire pump are assumed to require emergency ac power by way of the 2B bus and the emergency or standby diesel generators. Operation of the post-incident system is assumed to be required between 4 and 24 hours following the LOCA. Failure of this system leads to Sequence LLp.

FIGURE IV - 2
Earthquake Generated LOSP Event Tree



EARTHQUAKE GENERATED LOSS OF OFFSITE POWER

BP 1 DC Power

At Branch Point 1 (Figure IV-2), an earthquake has occurred in which significant ground motion resulted at the Big Rock Point Plant Site. A loss of offsite power has been the consequence of this ground motion. Loss of offsite power may result at Big Rock Point following a seismic event due to a variety of failures which are considered to be "weak-links" with respect to the availability of offsite power during such an event. These "weak-links" include the 2400 V voltage regulator in the switchyard, ceramic insulators in the switchyard or at offsite substations, or various masonry walls to which the 2400 V cable to the Station Power Distribution System is mounted. At Branch Point 1 the success or failure of the 125 V dc Distribution System is determined. Failure at this branch point implies that either the station batteries or the 125 V dc MCC D01 has failed as a result of the earthquake, leaving the plant without a station dc power source at Branch Point 45. The functioning of station dc power leads to Branch Point 2. There was not a DC power fault tree developed, rather the DC power dependencies were modeled into the individual trees.

BP 2 MSIV

The first automatic action required will be isolation of the Primary System by closure of the MSIV. Successful MSIV actuation implies success of the Reactor Protection System in generating a containment isolation signal. Failure to close the MSIV is conservatively assumed to result in a blowdown outside the containment either due to a failure of the backup valves to the MSIV to close or due to the failure of steam piping in the pipe tunnel. The MSIV has it's dc power source housed in the alternate shutdown building (ASDB). Successful MSIV closure requires the availablity of the dc batteries in the ASDB. These failures lead to Branch Point 35. Successful MSIV closure leads to Branch Point 3.

BP 3 Emergency Condenser Valves

A loss of offsite power has occurred as a result of an earthquake and the Primary System has isolated by automatic closure of the MSIV. Also generated as a part of station power loss was a signal to open the outlet valves to the emergency condenser (MO7053 and MO7063). This signal was generated by PSRE07 A-D which are ac-powered switches that fail closed on loss of voltage. Had station power been available, high reactor pressure (1450 psia) would have resulted in the closure of these switches. A manual demand to open the dc emergency condenser valves can also occur if the operator is aware of the rising reactor pressure due to decay heat generation. Indication of reactor pressure is available to the operator in the Control Room via PIIA07 which is dependent on emergency ac power operation. The operator has six minutes from the time of isolation of the Primary System until the first safety relief valve lifts (1550 psia) in which to manually actuate the emergency condenser should automatic actuation fail. The emergency condenser valves also have their power source housed in the ASDB. Successful valve operation requires the dc batteries available. Successful automatic or manual initiation of either of the emergency condenser outlet valves leads to Branch Point 4. Failure of both of the valves to open leads to Branch Point 25.

BP 4 UPS

Following isolation of the Primary System, this event tree contains a heading for the UPS power supplies and actuation/sensor cabinets. The UPS will have an impact on several of the systems which follow in this tree, such as operation of the RDS, automatic loading of the emergency generator onto the 2B bus and the functioning of various instrumentation which the operator may require for manual initiation of these or other functions. Success of the UPS in surviving this earthquake leads to the sequences following Branch Point 5. Failure of the UPS leads to Branch Point 18.

BP 5 Emergency AC

An earthquake has occurred, offsite power is unavailable, the Primary System is isolated and the emergency condenser is in service removing decay heat and cooling down the Primary System. Little, if any, dependency on ac power has been required up to this point in the tree. For the next four hours the plant will remain isolated with little loss of primary coolant inventory. By the end of this four-hour period, depletion of the water in the shell of the emergency condenser will have occurred, reactor pressure will have risen back to above normal operating conditions and a safety relief valve will begin to lift (1550 psia) limiting Primary System pressure unless action by the operator is taken to provide a source of makeup to the emergency condenser. Several of the systems following this branch point do have a significant dependence on ac power for success. At this branch point, therefore, the functioning of an emergency ac power source is required. Successful operation of the MSIV and emergency condenser valves, a significant amount of time to ensure that emergency ac power operation is available and successful actuation of either the emergency generator or the standby generator in loading the 2B bus will lead to the Branch Point 6. Failure of the 2B bus or both of the diesel generators will lead to a relatively lengthy station blackout described at Branch Point 12.

BP 6 Fire Pumps

The success or failure of the fire pumps is determined at this branch point. The plant is cooling down on the emergency condenser and emergency ac power is available. Success at this branch point assumes that the Fire System piping in the screenhouse is intact and that either of the two fire pumps, diesel or electric-powered, is running. A loss of the piping integrity or failure of both fire pumps within the first four hours of the transient results in lifting of a steam drum safety relief valve due to the inability to makeup to the emergency condenser shell from the Fire System. Gradual depletion of the Primary Coolant System inventory occurs for the next two hours at which time sufficient water has left the Primary System through relief valve actuation to begin to uncover the core. If the fire pumps are not recovered within this additional two-hour period it is assumed a core damage situation ensues via Sequence Pf due to the inability to provide adequate core spray. Maintenance of the screenhouse piping integrity and successful operation of either fire pump will lead to Branch Point 7.

BP 7 Emergency Condenser Makeup

Successful isolation of the reactor, operation of the emergency condenser and availability of a fire pump will allow use of the emergency condenser to cool the reactor to near shutdown conditions and maintain those conditions in the long term at Branch Point 8. Success of emergency condenser makeup implies the manual operation of dc powered SV4947 allowing fire water flow from the Core Spray System to the emergency condenser shell. As this valve is operated manually, indication that makeup to the shell is required should be available to the operator. This indication is assumed to be provided by an annunciation of shell low-level via LS3550 or by an indication of rising primary system pressure on PIIA07 both being in the Control Room. The valve is powered from the dc batteries housed in the ASDB. Successful valve operation implies that the power source is available.

Emergency condenser makeup also relies on the integrity of core spray piping inside the containment, the yard piping and piping between the yard and containment through the Turbine Building or the Post-Incident System. Should Turbine Building piping be unavailable as a result of the earthquake, it must be isolated from the yard loop and containment piping to prevent diversion of fire water and a flow path to containment must be established through dc-operated MO7072. Credit for this operator action is taken in this tree given the amount of time available to perform these operations (4 hours).

Failure of the emergency condenser makeup by way of failure of the makeup valve, Control Room shell level instrumentation, containment, yard or unisolatable Turbine Building piping failures, or failure to establish a flow path to containment through the Post-Incident System given a failure of Turbine Building piping is assumed to lead to Branch Point 9.

BP 8 Long-Term Cooling

Cooldown of the isolated reactor on the emergency condenser has been successful including makeup to the shell. The reactor is most likely at or near atmospheric conditions. Reactor cooling can continue indefinitely in this manner with the aid of makeup systems such as the control rod drive pumps or core sprays to accommodate any minor leakage which naturally occurs from the Primary Coolant System. Success or failure to cool the core at this branch point using these systems is developed in detail in the long-term cooling event tree which follows this discussion.

BP 9 RDS

Primary System isolation and emergency condenser actuation have been successful and fire pumps are available, but emergency condenser makeup has failed resulting in the repressurization of the reactor on decay heat to the safety relief valve set point. Gradual depletion of the reactor inventory is occurring due to periodic relief valve operation. No credit for makeup to the reactor by control rod drive pump operation is taken due to uncertainty in its ability to operate in the steam environment created by relief valve actuation.

Within six hours of the transient initiation, reactor inventory has reached the RDS actuation set point (2'9" above the core). Successful operation of the RDS at this stage of the transient leads to the need for core spray at Branch Point 10. Failure of RDS implies failure of isolation (CV4180-CV4183) or depressurization (SV4980-SV4983) valves to open or failure of an automatic or manual actuation signal. Automatic actuation of the RDS is dependent on drum (LT3184-LT3187) and reactor level (LT3180-LT3183) instrumentation, fire pump pressure sensors (PS789-PS793) and RDS timers. Manual actuation is assumed to depend on the availability of reactor level indication to the operator. In addition to the RDS level, instrumentation control room annunciation of low reactor level is provided via Reactor Protection Switches LSRE09 A-D. Manual operation of the RDS from the Control Room also has dependency on ac power distribution equipment via Panel 1Y.

Failure of RDS at this branch point is assumed to lead to a gradual uncovering of the core at pressure by Sequence PEmR.

BP 10 Core Spray

Depletion of primary coolant inventory to the point of RDS actuation requires core spray initiation. Successful core spray implies actuation of either of the sets of core spray valves (dc-actuated MO7051, MO7061 or ac-actuated MO7070 or MO7071). Sensors required to actuate these valves include reactor level and Pressure Switches LSRE09 A-H and PSIG11 A-H. Credit is taken for manual initiation of important core spray functions by the starting of a fire pump but only if appropriate drum level instrumentation is available. Manual actuation of the core spray valves can also be accomplished from the control room although credit for this action is not included as a part of the logic of these trees.

As was the emergency condenser makeup, core spray is dependent on the integrity of core spray piping inside containment, yard piping and a path from the yard loop to the containment either through the Turbine Building or the Post-Incident System. The Turbine Building path is that path normally valved in for operation, but should rupture of this piping occur as a result of the earthquake, sufficient time is available to establish the path through the Post-Incident System by opening MO7072 (more than six hours). Again, as with the emergency condenser makeup supply, core spray flow is considered adequate only if any ruptured fire piping which does exist in the Turbine Building is isolated from the yard loop and containment.

Success of this equipment leads to the need for a long-term cooling heat sink at Branch Point 11. Failure is assumed to lead to temporarily uncovered inadequately cooled core by way of Sequence PEmC.

BP 11 Post-Incident System

RDS and core spray actuation having been successful, the containment will begin to fill with water coming from the Primary System, core and enclosure sprays. Reactor and steam drum level will recover quickly following core spray actuation and water will flow from the drum to containment through the

RDS valves. The operator may control flow to the Primary System from outside the containment by regulating flow with Valves VFP-29 and VFP-30 or MO7072, thereby limiting the amount of water which enters containment. This tree assumes, however, that a containment water elevation between 587 feet and 590 feet is reached ultimately and the need for post-incident recycle and cooling is required.

Success of the Post-Incident System requires that at least one of two core spray pumps be started pumping water from the bottom of containment, through the core spray heat exchanger to core spray piping inside the containment. Failure of post-incident piping, core spray piping inside the containment, both of the core spray pumps or their station power buses or Reactor Building level instrumentation implies failure of the system. As assumed for the Core Spray System, diversion of water to rupture of piping in the Turbine Building will also fail the system unless this piping is isolated from the containment.

Heat removal by way of the core spray heat exchanger is required for success of the system. This heat removal is established by the opening of MO7066 remotely from the Control Room or locally by hand. An ac motor operated valve, MO7080, in parallel with MO7066 can also be opened to supply Fire System water to the shell of the core spray heat exchanger. Success of this system implies the integrity of Post-Incident System piping to and from the heat exchanger, the heat exchanger shell itself and yard piping. Yard piping failures which occur may be bypassed with the installation of a fire hose from the hose manifold in the screenhouse to Hand-Operated Valve VPI-10 also in parallel to MO7066.

Failure of this system to supply sufficient flow to the reactor to makeup for decay heat losses implies eventual core uncovery and inadequate core cooling by way of Sequence PEmLp.

BP 12 Fire Pumps Without AC Power

A seismic-generated loss of offsite power has occurred, the Primary System has isolated, an emergency condenser cooling path has been established; station dc power and the UPS are available but emergency ac power is unavailable due to loss of the 2B bus or both of the diesel generators. The Fire System is required to be functional to provide makeup to the emergency condenser and core spray if necessary. The success and failure of this system are identical to that described for the Fire System in Branch Point 6 except that only the diesel fire pump is available due to the failure of onsite ac power. In addition, the ability to start a fire pump manually from the RDS panel in the Control Room is no longer available due to this circuit's ac dependencies. Failure of the diesel pump or screenhouse fire piping under these conditions is assumed to lead to core damage Sequence PQF. Time frame for reaching this state of inadequate core cooling is the same whether or not ac power is available (approximately six hours). Successful fire pump operation allows emergency condenser makeup at Branch Point 13.

BP 13 Emergency Condenser Makeup Without AC Power

Primary power dependency of emergency condenser makeup is on the ASDB dc power. This dependency is a result of DC-Powered Makeup Valve SV4947. There is a minor ac power dependency at this branch point in that Control Room Primary System Pressure Indicator PIIA07 requires an ac power supply.

The description of this branch point is therefore identical to the discussion presented in Branch Point 7 except that continuous reactor pressure instrumentation used as backup verification that emergency condenser makeup is occurring is not available. Failure of makeup leads to the need for RDS and core spray at Branch Point 15; successful operation leads to Long-Term Cooling Systems at Branch Point 14.

BP 14 Long-Term Cooling

As indicated at Branch Point 8, detailed long-term cooling event tree has been developed and is presented following discussion of this loss of offsite power tree. This tree includes a discussion of long-term ac power dependencies.

BP 15 RDS Without AC Power

The success or failure of RDS given primary coolant level has reached 2'9" above the core is identical to that discussed in Branch Point 9 except that the ability to manually actuate a fire pump or manually depressurize the reactor from the Control Room is no longer available due to these circuits' dependencies on ac power. (RDS panel control switch and annunciation power is from Panel 1Y.) Failure of automatic RDS actuation leads to eventual inadequate cooling of the core at elevated pressure in a time frame similar to that which would occur with ac power available (> six hours) by way of sequence PQR. Successful RDS operation leads to the need for core spray at Branch Point 16.

BP 16 Core Spray Without AC Power

As a result of the unavailability of any ac power source, the ac-dependent motor-operated valves (MO7070 and MO7071) of the Core Spray System will not be available to provide core cooling after RDS actuation. The dc-powered valves (MO7051 and MO7061) will be available however as station dc power is assumed to be unaffected. Therefore, with the exception of the ac valves and the ac-powered Control Room instrumentation and actuation circuitry for manually starting a fire pump from the Control Room, the Core Spray System will function or fail as described in Branch Point 10. Failure of the Core Spray System will lead to an uncovered core at Sequence PQC, successful core spray will lead to Branch Point 17 and the need for long-term cooling.

BP 17 Post-Incident System

The success and failure of the Post-Incident System in long-term cooling at this branch point are identical to those described for Branch Point 11 even though there are ac power dependencies in the operation of the core spray pumps and MO7066. Recall from the discussion of this branch point that the path by which water fills the containment is through the RDS valves (well above the reactor core). The operator has the ability to monitor containment level (LT3171 and 3175) and reactor and drum level (LT3180-LT3183 and LT3184-LT3187) and regulate the flow to the Reactor Building from outside containment without ac power availability. It is assumed at this branch point that rather than terminating core spray makeup to the reactor entirely, allowing the core to become uncovered, the operator will regulate core spray flow allowing makeup to cover losses due to decay heat generation until an emergency ac power source for the core spray pumps can be recovered or offsite power is restored. This method of core spray flow regulation can occur for days which is sufficient time to allow restoration or repair of ac power.

Failure of post-incident cooling equipment described at Branch Point 11 after emergency ac power failure then leads to an inadequately cooled core by Sequence PQLp.

BP 18 Emergency AC Without UPS

An earthquake has occurred which has resulted in a loss of offsite power. DC power remained available, the MSIV closed and the emergency condenser outlet valves opened providing a heat sink for the Primary Coolant System.

It is assumed that the UPS have not survived, however, which has implications with respect to the manner in which emergency power is provided to the 2B bus, RDS actuation and Control Room indication of reactor and drum level.

At this branch point the success and failure of emergency power is discussed. Automatic isolation of the 2B bus and loading of the diesel generator cannot be accomplished because of the switchgear dependency on UPS and dc power. However, given that the Primary System is isolated and the emergency condenser is in service, four hours minimum exists before safety relief valve actuation can occur and six hours minimum until the RDS actuation set points are satisfied. Given these time frames, manual operation of the 2B bus electrical switchgear is extremely likely and the importance of the emergency power dependencies on the UPS is very small.

Success at this branch point implies the operation of either the standby or emergency diesel generators and the loading of this power to important equipment on 2B bus. Successful emergency ac power leads to Branch Point 19. Failure of emergency ac power leads to Branch Point 22.

BP 19 Fire Pumps Without UPS

There is no dependency by the fire pumps on the UPS with the exception of the automatic starting of the pumps on low drum level as measured by RDS drum level instrumentation (LT3184-LT3187). Similar to the emergency generator switchgear this dependency is not important at this branch point given the time frame in which makeup to the emergency condenser (four hours following transient initiation) or core spray (six hours) is required. The success or failure of the fire pumps at this branch point is the same as that presented in the discussion for Branch Point 6. Successful fire pump actuation leads to Branch Point 20 emergency condenser makeup. Failure of the fire pumps at this branch point leads to eventual core damage Sequence PUF due to the failure of the makeup source to the emergency condenser and ultimately core spray failure.

BP 20 Emergency Condenser Makeup Without UPS

The description of success or failure at this branch point is identical to that presented for Branch Point 7 as there is no dependency on UPS by the fire water makeup to the emergency condenser. Success of this system leads to long-term cooling Branch Point 21.

However, unlike Branch Point 7, failure of makeup to the emergency condenser by itself leads to inadequate core cooling by way of Sequence PUEm. After having cooled the Primary System adequately for four hours, the emergency condenser has ceased removal of heat from the reactor due to depletion of shell side water. The reactor has repressurized and a safety relief valve is limiting pressure at 1550 psia. Inventory depletion of the Primary Coolant System is occurring because no makeup water is available (like Branch Point 9, no control rod drive makeup is assumed because of the steam environment inside containment). Approximately six hours following the earthquake and loss of power, the reactor water level reaches the RDS actuation set point (2'9" above the core). As no power supply is available for energizing the solenoid-operated RDS isolation and depressurization valves, the reactor remains at pressure and the core slowly becomes uncovered without the ability to provide low-pressure core spray injection.

BP 21 Long-Term Cooling

Similar to Branch Points 8 and 14, long-term cooling given the success of the emergency condenser is developed in a subsequent event tree.

BP 22 Fire Pumps With UPS and Emergency AC Failure

For the same reasons stated in Branch Point 19, the success or failure of the fire pumps has limited dependence on the UPS. However, ac power availability does have a significant influence on this system. Therefore, the description of this branch point is essentially identical to that presented in Branch Point 12 with the failure of the diesel fire pump leading to core damage Sequence PUQF. Successful diesel pump operation allows emergency condenser makeup at Branch Point 23.

BP 23 Emergency Condenser Makeup Without UPS and Emergency AC

There is no dependency of emergency condenser makeup on UPS and only a minor dependency on emergency ac. This dependency was described in Branch Point 13 and involves the inability of the operator to use PIIA07 (primary coolant) as backup verification of the need for adding makeup to the emergency condenser shell. LS3550 (shell level) is assumed to be available, however, as dc power is still functional. Successful makeup addition to the emergency condenser shell at this branch point is still possible, therefore, and leads to the need for long-term cooling systems at Branch Point 24. Failure of emergency condenser makeup piping, Makeup Valve SV4947 or shell level indication is assumed to lead to core damage Sequence PUQEm for the same reasons presented in the description of Branch Point 20.

BP 24 Long-Term Cooling

Development of a long-term cooling event tree follows the description of this event tree.

BP 25 Through 34 Emergency Condenser Valve Failure

At this branch point a loss of offsite power resulting from a seismic event has occurred, dc power is available and the MSIV has closed isolating the Primary System. The assumption is made at this branch point that neither of the emergency condenser outlet valves (MO7053 and MO7063) opened leaving the Primary Coolant System without a heat sink. Reactor pressure has risen to the set point of the first safety relief valve (1550 psia) and Primary System pressure is being limited in this manner. Assuming the reactor was operating at full power at the time of the transient, this reactor state will be reached within six minutes from initiation of the transient.

There are no dependencies on emergency condenser valve operation within the UPS, emergency ac power, fire pumps, RDS, core spray or Post-Incident System with the exception of dc power (which is assumed to be available during these sequences). Emergency condenser makeup obviously will not be required following failure of the emergency condenser valves, hence no branches between Branch Points 27 and 28 or 31 and 32. There is a minor dependency of emergency condenser valve actuation on ac power in that PIIA07 can be an unambiguous indication that the emergency condenser valves need to be opened given that they have failed automatically. This pressure indicator relies on automatic starting of the emergency diesel generator and operation of switchgear connecting it to the 2B bus. Given the limited time frame in which this instrument is being used no credit for manual actuation of the emergency diesel or standby diesel generators is taken in manual operation of these valves.

Given these limited dependencies, it can be seen that the reactor states and sequence descriptions resulting from transient sequences leading to inadequate core cooling as a result of emergency condenser makeup failure are identical to those which occur following emergency condenser valve failure, the only difference being the time required to reach the final reactor state.

Emergency Condenser Makeup Failure Sequences				Corresponding Emergency Condenser Valve Failure Sequences			
Sequence	Time of SRV Actuation	Time of Core Uncovery	BP	Sequence	Time of SRV Actuation	Time of Core Uncovery	BP
PEmLp	4 h	Days	11	PEvLp	6 m	Days	30
PQEmLp	4 h	Days	17	PEvQLp	6 m	Days	34
PEmC	4 h	6 h	10	PEvC	6 m	2 h	29
PQEmC	4 h	6 h	16	PEvQC	6 m	2 h	33
PEmR	4 h	6 h	9	PEvR	6 m	2 h	28
PQEmR	4 h	6 h	15	PEvQR	6 m	2 h	32
PF	4 h	6 h	6	PEvF	6 m	2 h	27
PQF	4 h	6 h	12	PEvQF	6 m	2 h	31
PUF	4 h	6 h	19	No Corresponding	---	---	--
PUQF	4 h	6 h	22	Sequences	---	---	--
PUEm	4 h	6 h	20	PEvU	6 m	2 h	25
PUQEm	4 h	6 h	23				

BP 35 Primary Coolant System Isolation Failure

On loss of offsite power a signal is generated by the Reactor Protection System to close all automatically-actuated containment isolation valves. At this point on the event tree the ASDB dc power supply is assumed to have survived the seismic event but the MSIV has failed to close. No assumptions are being made with respect to the ability of backup valves to the MSIV to close, or if they do close, to prevent leakage from the unisolated main steam line to the main condenser, or of the integrity of the main steam line piping or its branch connections to remain intact following ground motion. It is conservatively assumed, therefore, that failure to close the MSIV following a loss of offsite power leads to a blowdown of the reactor to the Turbine Building or pipe tunnel. If large enough, such a blowdown could occur over the course of several minutes. The energy loss of such a blowdown is more than that added to the Primary System as a result of decay heat eliminating a demand on the emergency condenser valves to open or the need to makeup to the emergency condenser shell.

At this branch point then the success or failure of UPS is discussed. Failure of UPS implies the failure of the batteries, power supply or actuation/sensor cabinets. Failure of more than one UPS is sufficient to disable the RDS leading to an uncovered core with an insufficient blowdown to allow timely core spray initiation. If the blowdown continues outside containment or one or more RDS trains actuates properly, core spray injection can occur eventually although not before limited core damage has occurred. Failure of UPS leads to inadequate core cooling via Sequence PIU. Successful operation of at least three of the four UPS leads to Branch Point 36.

BP 36 Emergency AC Power Following MSIV Failure

As the blowdown occurs due to failure of the MSIV to close, the emergency diesel generator will attempt to start and energize the 2B bus to provide power for important equipment such as the electric fire pump and ac core spray valves. Failure of the diesel generator to start or load the 2B bus leads to Branch Point 41; successful energization of the emergency bus to Branch Point 37. Due to the potentially limited duration of this transient, no credit is taken for manual operation of the emergency or standby diesel generators.

BP 37 Fire Pumps After MSIV Failure

On attaining a low drum level (<17" below center line) a signal will be sent to start both fire pumps. Success at this branch point implies the starting of at least one of the two fire pumps and the maintaining of screenhouse fire piping integrity. This will allow fire pressure permissive (>100 psig) to the RDS actuation logic and provides a source of fire water for core spray. Success leads to Branch Point 38 while failure leads to a lack of adequate core cooling after RDS and core spray actuation fails, Sequence PIF. No credit for operator action to start the fire pumps is implied, again due to the potentially limited time frame over which this transient occurs.

BP 38 RDS With MSIV Failure

As the blowdown outside containment continues, low drum level will initiate a two-minute timer and a low drum level signal to the RDS actuation logic. Further blowdown will result in reaching the low reactor level set point (2'9" above the core). When all four signals occur simultaneously in two of four RDS sensor trains, low drum level, low reactor level, high Fire System pressure and time out of the two-minute timer, the RDS solenoid valves will be energized opening the isolation valve and depressurization valve in each train. Failure of blowdown in more than one train is assumed to lead to an unsatisfactory depressurization of the reactor and limited core damage by Sequence PIR. Successful blowdown through three of the four RDS trains leads to the need for core spray at Branch Point 39. Again, no credit for operator action is taken due to the short duration of the transient, failure of automatically-operated components is assumed to lead to system failure.

BP 39 Core Spray After MSIV Failure

The reactor has depressurized by RDS actuation after blowdown through the main steam line. The fire pumps and emergency power are successfully operating. At low reactor water level a signal was sent to the core spray valves, and on reaching 200 psig reactor pressure the ac- and dc-operated valves receive a signal to open. Success of the Core Spray System at this branch point implies opening of either of the pairs of ac or dc valves in the core spray lines maintaining the integrity of the core spray piping in the containment, in the Turbine Building and the underground yard piping. No credit is given for isolation of the Turbine Building piping should it rupture as a result of the earthquake or establishing a backup fire water path to containment through the Post-Incident System, again due to the limited time frame over which the transient occurs. Successful core spray leads to long-term cooling at Branch Point 40, failure leads to degraded core cooling Sequence PIC.

BP 40 Post-Incident System With MSIV Failure

Although a result of water loss was effectively a loss of coolant accident outside containment, the Long-Term Cooling System description at this branch point is identical to that at Branch Point 10. Following a transient of much longer duration, one might argue that the main steam line is open outside containment and water overflow from the Primary System preferentially will pass through this line rather than through the RDS valves which are located at a higher elevation. This argument could lead to the potentially unconservative assumption that the Post-Incident System will not be needed for long-term cooling following a reactor blowdown outside containment. The assumption will be made, therefore, that repairs to main steam line components are made following the transient isolating the Primary System and ultimately requiring use of the Post-Incident System. Failure of this system is assumed to gradually lead to inadequate cooling by Sequence PILp.

BP 41 Through 43

A failure of the MSIV to close has resulted in a blowdown of the reactor to the Turbine Building. At this branch point, dc and UPS power are assumed to available but emergency ac power is not available due either to a failure the emergency diesel generator or within the 2B bus. The description of anch Points 41 through 43 is identical to Branch Points 37 through 39 with e exception that the electric fire pump and ac-powered core spray valves are t functional due to the loss of emergency power. Failure of the diesel fire mp or one of the dc core spray valves by themselves will lead to Sequences QF and PIQC. Sequence PIQR occurs just as described in Branch Point 38. ccessful diesel fire pump, RDS and core spray actuation lead to long-term oling Branch Point 44.

on

44 Post-Incident System

e description presented in Branch Point 11 is applicable to Post-Incident stem operation at this branch. Again the assumption is made that the st-Incident System is required, perhaps, as a result of a repair of the main eam line. The standby diesel generator is now considered available for tential use as backup to the emergency generator. Even if both diesels iled as a result of the earthquake, operator action limiting the amount of ter added to containment will postpone the use of the Post-Incident System r as much as several days until repair or restoration of an ac power source r the core spray pumps is made available.

45 UPS With DC Power Failure

e earthquake has resulted in a loss of offsite power. The assumption is de at this branch point that the station dc power supply (station batteries ASDB batteries) also has failed.

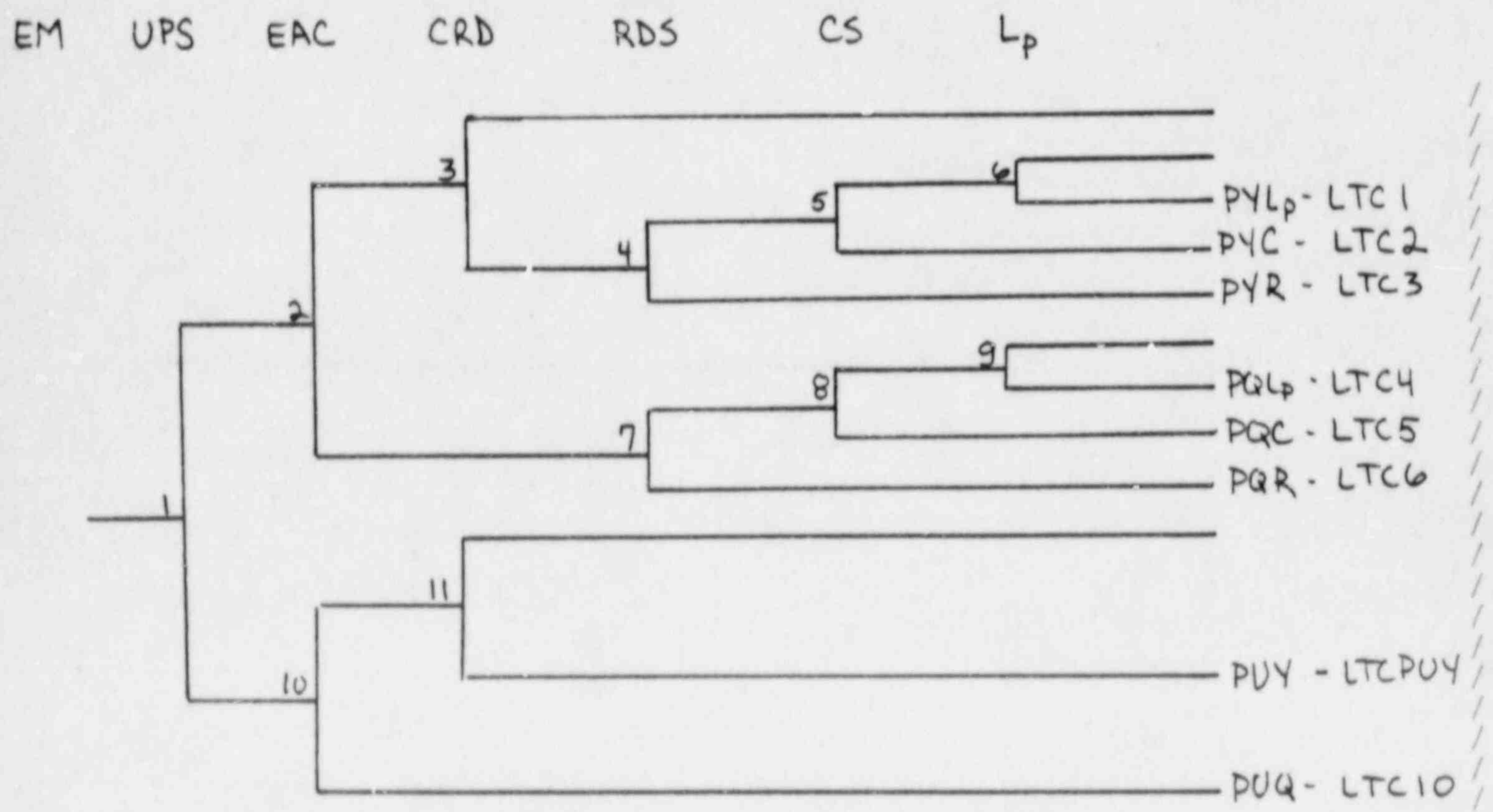
a

power failure automatically results in the inability to isolate the Primary stem by closure of the MSIV. As was assumed at Branch Point 35, a blowdown the reactor to the Turbine Building results. A failure of the UPS has two portant consequences at this branch point. Failure of UPS A disables the ility of automatic loading of the diesel generator onto the 2B bus leaving power source to the ac-powered core spray valves should they be needed. A pid blowdown outside containment will result in core uncovery without core ray as the power supply to the dc core spray is also assumed to be disabled. ilure of more than one of the UPS will result in inadequate reactor pressurization perhaps leading to limited core damage as discussed in Branch int 38. Success of three of the four UPS, at least one being UPS A, leads Branch Point 46. Failure of UPS A or any two UPS is assumed to lead to graded core cooling by Sequence PDU.

46 Emergency AC With DC Power Failure

ilure of the emergency diesel to start or failure of the 2B bus to energize sables the power supply to the ac core spray valves. As stated previously, ould a rapid reactor depressurization occur, no credit for manual action in arting the standby diesel can be taken and core spray will be disabled by y of Sequence PDQ. Successful energization of the 2B bus leads to Branch int 47.

LONG TERM COOLING EVENT TREE



EM	-EMERGENCY CONDENSER MAKEUP
UPS	-UNINTERUPTABLE POWER SUPPLY
EMER AC	-EMERGENCY AC POWER
CRD	-CONTROL ROD DRIVE MAKEUP
RDS	-REACTOR DEPRESSURIZATION SYSTEM
CS	-CORE SPRAY SYSTEM
PIS	-POST INCIDENT SYSTEM

FIGURE IV - 3

Long-Term Cooling Given Emergency Condenser Operation Following a Loss of Offsite Power

The previous tree described a loss of offsite power transient generated by a seismic event. Four branches of that tree lead to a need for long-term cooling assuming the Primary Coolant System was isolated, the emergency condenser was in service and emergency condenser makeup from the Fire System is successful. An assumption is made that offsite power may not be available for several days leading to the need to makeup to the Primary System allowing a recovery of normal level overcoming shrinkage due to the cooldown and normal Primary Coolant System leakage.

Shrinkage by itself is not sufficient to result in uncovering the core. Combined with normal Primary System leakage from packing, seals, etc, over several days there exists the potential for attaining very low Primary Coolant System levels, however. For this reason an additional system, Control Rod Drive Makeup, is required for use during long-term cooling in addition to the emergency condenser. Failure of the CRD makeup, like the emergency condenser, will be assumed to lead to water levels below the low reactor level set point and a need for RDS, core spray and post-incident recycle. Failure of this makeup supply is not expected to result in these low levels for over a day allowing a relatively long time for the operator to establish this makeup supply. In fact, if the Primary Coolant System pressure is maintained below 150 psi by the emergency condenser, the Core Spray System can be used to provide the required makeup. As the CRD System is the preferred source of makeup and as it can be actuated even with the reactor at elevated pressures, the tree which follows was developed conservatively ignoring the potential Core Spray System option.* The core spray can be added if needed by inserting an additional heading following CRD makeup in the long-term cooling tree.

BP 1 UPS

Long-term cooling has been established for the Primary System through use of the emergency condenser. This tree (Figure IV-3) contains a branch point for UPS, similar to the loss of offsite power tree to account for its effect on various important system functions such as providing Control Room indication for reactor drum level and actuation of the RDS if necessary. UPS failure leads to Branch Point 10 whereas success results in Branch Point 2.

BP 2 Emergency AC

Success of emergency ac implies operation of the emergency diesel or standby diesel generator and energization of the 2B bus. This system is necessary to provide a power source for the CRD pumps and is also important in providing power for the electric fire pump, ac core spray valves and various Control Room indications. Emergency ac failure leads to Branch Point 7; successful energization of the 2B bus leads to the operation of the CRD pumps at Branch Point 3.

*It is emphasized that control rod drive makeup is a backup to core spray.

BP 3 CRD Pumps

Successful operation of this system implies that one of the two CRD pumps be operating from the emergency power source, that the condensate storage tank be intact, CV4090 be open, CRD suction and discharge piping be intact and a path to the reactor (most likely through the scram header) be available. Failure of this system is assumed to lead to the need for RDS, core spray and post-incident cooling around a day following the loss of offsite power, Branch Point 4. If UPS is unavailable as in Branch point 10, loss of CRD assumes core damage as UPS is not available for RDS. Periodic operation of the system implies successful long-term cooling.

BP 4 Through 6

RDS, core spray and post-incident cooling are required in these branches. Description of these branch points is identical to that presented for Branch Points 9 through 11 in the loss of offsite power tree except that rather than emergency condenser makeup, CRD makeup failure has occurred. Failure of these systems leads to inadequate core cooling via Sequences PYR, PYC and PYLp.

BP 7 Through 9

Loss of emergency power source automatically implies the inability to makeup with the CRD pumps and the need for RDS, core spray and post-incident cooling. The description of these branches is identical to Branch Points 15 through 17 of the loss of offsite power tree except rather than emergency condenser makeup, CRD makeup failure has occurred. Failure of these systems is assumed to lead to inadequate core cooling by Sequences PQR, PQC and PQLp.

BP 10 Emergency AC With UPS Failure

Given the length of time associated with this branch of the transient (on the order of a day), there is little dependency of ac power on the UPS. This branch point is therefore identical to Branch Point 2. Success of ac power leads to CRD makeup at Branch Point 11, failure is assumed to lead to core damage. Sequence PUQ fails because of the inability to makeup to the reactor or actuate the RDS.

BP 11 CRD Pumps

This branch point is identical to Branch Point 3 except that reactor level instrumentation powered from the UPS is not available. Failure of CRD makeup is assumed to lead to inadequate cooling by way of Sequence PUY because of the inability to makeup to the reactor with the CRD System or actuate the RDS. The RDS dependancy upon UPS (BP-12, sequence PUYR) is redundant to sequence PYR, as the RDS logic model contains all the dependencies on UPS.

V. SEISMIC CAPACITY, FAILURE MODES, EFFECTS, REPAIR AND RECOVERY

This section contains a table identifying each plant component important to the systems described in Sections III and IV. In addition to the components included as a part of the systems, descriptions, plant structures and equipment not normally considered important to the functioning of the system but which may be made important as a result of ground motion are identified, such as major plant structures and masonry walls. The effects of the failure of each component are assessed as well as the expected failure mode. As the purpose of this study was to rank the importance of these plant components in some manner, an assumed capacity for each of these components was also determined. A compilation of this information is presented in Table V-1.

The components chosen were those identified as being important to the systems used to perform important plant shutdown functions in the event tree headings of Section IV. The effects of the failure of each of these components were based on the system-success-and-failure-criteria presented in the system descriptions and event tree discussions. Important passive components whose structural failure could lead to loss of any of these system components were identified previously in plant walkdowns such as those conducted for IE Bulletins (electrical equipment, anchorage, masonry walls).

In Table V-1, equipment dependencies on passive components were identified for major plant equipment and their power sources. To have a more complete system failure analysis; equipment dependencies on cable routings were included in the models. For the equipment identified, all the cables listed in the schemes for the various pieces of electrical equipment were identified. The raceway routing of these cables was then listed and a general walkdown of the raceways was conducted. Of primary concern during the walkdown evaluations were items that may fail the cables by falling onto the cable trays, conduit, or equipment. For instance, the walkway grating by the screenhouse entrance door (SCHDW) was assumed to fail the diesel and electrical fire pumps, due to cables which run underneath this walkway. The seismic capacity of these items was assumed to be that of the structure to which it is attached. For the example above, the walkway was assigned a ground acceleration of 0.500g which is the value of the screenhouse structure.

The capacity of the raceways has been evaluated by the Seismic Qualification Utilities Group (SQUG). The group concluded:

> "...that the existing raceway systems in SEP plants possess substantial inherent seismic resistance and that the seismic qualification of raceway systems is not a significant safety issue."

Based upon this conclusion, the seismic capacity of the cable trays was set equal to the capacity of the structure it is mounted to. Pending final NRC approval of the SQUG conclusions, no further analysis of the cable trays will be undertaken.

Therefore, the bases for the capacities for the components listed in Table V-1 are discussed in this section as well as assumptions made with respect to potential operator response in recovery and repair of the component failures should they occur. The procedure for using the information presented in Table V-1 is discussed in Section VI.

A. Assumed Capacity

In order to evaluate the seismic capacity of Big Rock Point structures, equipment and components on a consistent basis, information from many sources was reviewed. Information sources included analyses, tests, specifications, historical equipment seismic reports and judgments made by experienced engineers. The seismic capacities determined for Big Rock Point structures, equipment, and components are based upon the following:

1. Detailed structural analyses of as-builts,

2. Detailed structural design analyses,

3. Comparison based upon historical performance with respect to items of similar physical appearance and functional requirements, and

4. Visual inspection and evaluation based upon experienced judgments.

Assessments with respect to the adequacy of structures, equipment and components at some level of seismic input were reliably estimated by the above four means. However, the specification of a lower-bound free-field seismic excitation at which an item fails was more difficult. Therefore, a significant amount of judgment was required to normalize all sources of information and to unify all data on a consistent basis.

The results of detailed analysis of as-builts and seismic designs provided margins with respect to code (ASME, ANSI, AISI, AISC, ACI, etc) requirements. An understanding of the margins with respect to the codes themselves provided a basis for establishing mean capacities and variances associated with structural integrity or reliability. A mean capacity minus one standard deviation provides what is referred to in Table V-1 as an "assumed capacity."

An evaluation of items on the basis of historical performance may well be directly amenable to similar potential calculations as to those employed for items analyzed. For some items, such as welded pipe, very few failures due to seismic excitation have occurred Thus, statistics are not suitable. Therefore, the "assumed capacity" is the lowest free field seismic excitation at which a failure has been known to occur. Such an "assumed capacity" will be much more conservative (lower) than that derived statistically from distribution of failures.

Evaluation of items by inspection was done by comparison with items which had a historical performance data base or which had been analyzed at Big Rock Point. An assignment of an "assumed capacity" was based on statistics or lower bounds of historical performance data.

Examples of items evaluated by the general four groupings above are contained in the following table:

BIG ROCK POINT SELECTIVE ASSUMED CAPACITIES

Item	Basis for Evaluation
Masonry Walls	Detailed Analysis to As-Builts
Building Structures	Detailed Analysis to As-Builts
Station Power Battery Racks	Detailed Analysis Design
Fire Piping - Screenhouse	Detailed Analysis Design
Electrical Equipment	Historical Performance
Welded Piping	Historical Performance
Diesel Generator Unit	Inspection - Historical Performance
Turbine Building Fire Piping	Inspection - Comparison with Analyzed Pipe

All assumed capacities reflected an understanding of item location, overall building and local response, and the site specific spectral shape. Scaling of capacities from Regulatory Guide 1.60 (Design Response Spectra for Seismic Design of Nuclear Power Plant) to those of the site specific spectra was based on an overall scale factor of 1.5 where applicable. For plant structures, the assumed capacity was associated with a potential lack of serviceability. The capacities were elevated from the results obtained by D'Appolonia (Report 78-435, August 1981 - Volumes I - X, Seismic Safety Margin Evaluation - Big Rock Point Nuclear Power Plant Facilities). Serviceability is generally defined as code limits except for secondary steel members in highly redundant structures where code limits were handled more liberally. A scale factor of 1.5 for spectral shape between Reg Guide 1.60 and the Big Rock Point site specific spectra was used to relate D'Appolonia results to the site specific spectra. The increase in allowables of 1.6 for the safe shutdown earthquake (NRC-Standard Review Plan 3.8.4) was used for some steel member allowables in evaluating capacities. For concrete, no ductility factor or Standard Review Plan allowable increase was employed. Thus, concrete allowables were those determined strictly from the ultimate strength design method per ACI-349-76.

The assumed capacities for masonry and for some other equipment items in Table V-1 were determined from fragility analysis similar to that of Kennedy et al (Probabilistic Seismic Safety Study of an Existing Nuclear Power Plant, Nuclear Engineering and Design 59 (1980) 315-338). The assumed capacities for equipment not limited by building capacity are based on a minus one standard deviation from the estimated mean capacity. The standard deviations are composite values based upon both random variabilities and variabilities associated with the uncertainty of the mean value.

B. Operator Initiated Recovery and Repair

It was not assumed that because a particular structure or component listed in Table V-1 did not survive an earthquake that the function provided by the failed component was unavailable for the duration of the transient. Numerous options for repair or recovery of systems are available to the Big Rock Point operator depending on the nature of the failures and the time frame in which the failed system functions are required. In this section, the ground rules for assumed operator response in overcoming component failures identified in Table V-1 are presented. These ground rules were used as a basis for identifying operator response following an earthquake for which credit was taken in the event tree branch point descriptions of Section IV.

Recovery of failed systems was assumed to be possible if sufficient major components of a system which provided a given function survive the earthquake but were not operating because of secondary failure modes. An example of this situation would be the failure of the diesel generator to start because undervoltage relays did not generate any actuation signal but, given sufficient time, manual operation of the generator or standby generator could be accomplished providing power to the emergency bus. The following assumptions were made in recovering a system which was not adequately performing its function due to seismically induced failures:

1. No operator action was assumed to occur while ground motion was occurring. This assumption was applied to any recovery actions required to occur over the first few minutes of the transient. As examples, no credit for manual RDS or core spray actuation was taken if a rapid blowdown of the reactor was occurring due to LOCA or MSIV isolation failures.

2. Limited manual recovery of systems which have survived the earthquake is assumed during the first few minutes following the earthquake. Examples of these operator actions include manual operator actuation of the emergency condenser outlet valves or enclosure spray valves. These actions are assumed to be possible only for those systems that do not have to function while ground motion is in progress, whose actuation can occur from the control room; and, are considered successful only if instrumentation is available to the operator indicating the need to actuate the system and has also survived the earthquake.

3. Manual recovery of systems from outside the control room was considered possible only if a relatively significant time frame was available to the operator to perform these actions (eg, hours). Such actions included operation of the standby diesel generator or isolation of ruptured fire system piping in order to establish a flow path to the reactor building through the post incident system. These types of operator action were considered possible only if the systems were currently designed

to be operated in this manner (ie, no credit was taken for "modifying" a system for unusual operation within a time frame of several hours). Again, indication that these types of actions were necessary needs to be available to the operator.

4. Repair or modification of a system was not considered possible in a time frame on the order of days. An example of this type of operator action was demonstrated in the dependency of the core spray pumps on ac power. If the emergency bus or 1A and 2A buses failed to survive the earthquake due to the collapse of a nearby block wall, the only way power could be supplied to these pumps, if required, would be to provide temporary jumpers from one of the diesel generators directly to the pump motor, bypassing normal station power distribution equipment. Credit for this type of repair or modification was taken only when significant time frames were available for repairs or mods.

TABLE V-1
COMPONENT FAILURE MODES, EFFECTS AND CAPACITIES

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
Containment Building Internal Structure	Loss of components anchored to reactor building concrete structures (ie, piping, instrumentation, electrical cabinets, electrical and pneumatic valves)	.4g - interpretation of D'Appolonia Report 78-435, Volume II Shear Failure of Reinforced Concrete Wall
Fuel Cask Loading Dock	Failure of Post Incident System components located in core spray	.4g - interpretation of D'Appolonia Project Report 78-435, Volume VI Column Interaction
Turbine Building a. Pedestal	Failure of components anchored to pedestal and pipe tunnel structures (ie, core spray and CRD pump suction piping)	.25g - interpretation of D'Appolonia Project Report 78-435, Volume III Sliding of Turbine Pedestal
b. Steel Superstructure	Failure of components anchored to steel members of turbine building (ie, piping, turbine building crane, electrical cabinets)	.32g - interpretation of D'Appolonia Project Report 78-435, Volume III Failure of Overall Steel Superstructure
c. Foundation	Failure of components anchored to concrete structures in the turbine building other than turbine pedestal or pipe tunnel (ie, electrical cabinets, motor control centers, electrical instrumentation batteries	.35g - interpretation of D'Appolonia Project Report 78-435, Volume III Failure of Foundation Below Steel Superstructure
Service Building	Loss of equipment anchored to service building structures (ie, control room electrical panels, and electrical instrumentation, RDS actuation/sensor cabinets)	.35g - interpretation of D'Appolonia Project Report 78-435, Volume III Failure of Steel Structures
Screenhouse	Loss of components anchored to screenhouse and diesel generator room structures (ie, diesel generator electrical panels, electrical instrumentation batteries, crane)	.5g - interpretation of D'Appolonia Report 78-435, Volume VIII

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
Stack	Los of turbine building, service building and electrical penetration components (ie, control room, RDS and station power room equipment)	.25g - interpretation of D'Appolonia Project Report 78-433, Volume IV Tensile Stress in Vertical Rebar
Block Walls[1]		
M100.01	Loss of CRD pump suction piping in condensate pump room	.33g - interpretation of SMA Report 13703.01-R003 Loss of Blocks or Collapse
M100.02	Loss of condensate piping between hot well and condensate storage tank (CRD pump suction piping)	1.4g
M100.03	Loss of UPS A (RDS power supply and power supply for emergency diesel generator circuit breaker)	.63g
M100.04	Loss of UPS B	.53g
M100.05	Loss of threaded fire piping in turbine building	.53g
M100.06	Loss of UPS A and B	.33g
M100.07	Loss of UPS C and D	.37g
M100.08	Loss of UPS A and C	.31g
M100.09	Loss of UPS B and D	.31g
M100.10	Loss of UPS A	.33g
M100.11	Loss of UPS C	.37g
M100.12	Loss of UPS C	.16g

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
M100.13	Loss of UPS D and threaded fire piping in turbine building	.16g
M100.14	Loss of dc power source (station batteries, and 125 V dc Bus D01, D02 and D10)	.53g
M100.15	Loss of threaded fire piping in turbine building	.53g
M100.16(2)	Loss of dc power source (station batteries, 125 V dc Bus D01, D02 and D10, core spray piping and 2400 V cable (offsite power))	.53g
M100.17(2)	Loss of 2400 V cable and contactors for motor operated valves MO 7072 and MO 7064	>>1.0g
M100.18(2)	Loss of station ac distribution equipment (2400 V Bus, 1A, 2A and 2B Buses)	.13g
M100.19(2)	Loss of 2400 V cable and core spray piping in turbine building and dc power sources adjacent to black wall M100.16	.53g
M100.20	Loss of core spray piping in turbine building	.12g
M100.21(2)	Loss of 2400 V Cable (offsite power) and threaded fire piping in turbine building)	.11g
M100.22	Loss of PIS-core spray HX cooling water	.300g - (Small wall near ground level, assign medium value of 0.3g) / /
M100.23	Fails cables to control room from alternate shutdown building. ie., Emergency condenser make-up MSIV and Emergency condenser valves.	.110g - (Similar to M100.21 given smallest value of block walls)

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode	
K200.1IS	Loss of electrical power to MO 7066	.4g - (same as fuel cask loading dock due to size of wall and encased all four sides by loading dock concrete structure)	
2400 V Voltage Regulator(2)	Loss of offsite power	.25g Historical capacity of unanchored switch yard equipment Falls from supports	
Emergency Condenser(2) Shell	LOCA, RDS valve and fire piping damage	.12g - interpretation of SMA Report 13702.01-R003 Support failure	
Nonregenerative(2) Heat Exchanger	LOCA	.13g - interpretation of SMA Report 13702.01-R003 Support failure	
Reactor Cooling Water Heat Exchanger	Fails CRD pump #1	.40 (Used containment building internal structure).	/
Cleanup(2) Demineralizer	LOCA	.11g - interpretation of SMA Report 13702.01-R003 Support failure	
Reactor(2) Internals	ATWS	0.2g - interpretation of SMA Report 13702.01 Support Plate Alignment pins shearing.	/
CRDM Discharge(2) Piping	ATWS	> .3g - interpretation of SMA Report 13702.01 Crimping	/
	LOCA	> .3g - interpretation of SMA Report 13702.01 Rupture	/

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
Ceramic Insulators	Loss of offsite power	.2g Historical capacity
Fire Piping in Screenhouse (Cast iron components)	Inability to supply water to yard piping and failure of automatic RDS actuation	.[illegible]5g - interpretation of Catalytic Calculation Book 2076, October, 1981 Rupture/leakage
Yard Piping	Inability to supply water to turbine building or containment through post incident system	.2g - Interpretation of D'Appolonia Report 78-435, Volume IX Rupture/leakage
Fire Piping in Turbine Building Threaded and Victolic Couplings	Inability to supply water to containment through turbine building and potential diversion of water from fire piping in containment if coincident with failure of Valves VPI 301 or 302	.15g - Inspection Failure due to internal load of basket strainers
Welded Piping		.25g (Same as turbine pedestal as penetrates pipe tunnel wall)
Fire Piping in Core Spray Heat Exchanger Room	Inability to supply water to containment through Post incident system or shell side of core spray heat exchanger	.4g (Same as fuel cask loading dock due to historical performance of welded carbon steel piping) Rupture/leakage
Fire Piping Inside Containment	Inability to provide makeup to emergency condenser shell, core spray, or containment spray	.4g (Same as reactor building internal structure due to historical performance of carbon steel piping) Rupture/leakage

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
Electric and Diesel Fire Pumps (Pumps 6 and 7)	Inability to supply water to fire piping in screenhouse and failure of automatic RDS actuation	.25 - interpretation of Catalytic Engineering Evaluation Book 2076 October 1981 Rupture/leakage of cast iron casing
Core Spray Pumps (Pumps 2A and B)	Inability to recycle water from the containment sump to core sprays for long-term heat removal after a LOCA or RDS actuation	.4g (securely anchored to fuel cask loading dock foundation) Loss of function or rupture/leakage
Control Rod Drive Pumps and Piping Inside Containment (Pumps 4A and B)	Inability to makeup to reactor in long term to accommodate normal primary coolant system leakage	.4g (securely anchored to reactor building internal structure) Loss of function or rupture/leakage
Containment	Piping outside	.25g (anchored to pipe tunnel walls) Rupture leakage
Emergency Diesel Generator	Failure to supply automatic emergency ac power to 2B Bus	.5g (securely anchored to diesel generator room floor) Loss of function
Standby Emergency Diesel Generator	Failure to supply backup emergency ac power to 2B Bus (manually actuated)	Not characterized Loss of function
Emergency Diesel Fuel Supply	Failure of emergency diesel to run	>1.0g (Buried tank) Leakage/rupture
Diesel Fire Pump Fuel Supply	Failure of diesel fire pump to run	>1.0g (buried tank) Leakage/rupture

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
Standby Diesel Fuel Supply	Failure of standby diesel generator to run	Not characterized Leakage/rupture
Core Spray Heat Exchanger	Failure to remove decay heat or diversion of post incident system water during long-term cooling after LOCA or RDS actuation	.4g (securely anchored to fuel cask loading dock foundation) Leakage/rupture
Station Batteries	Loss of power to major dc powered electrical equipment (125 V dc Bus D01)	.35g (securely anchored to turbine building foundation) Loss of function
UPS Batteries	Loss of power to RDS sensors and solenoid valves (UPS A also supplies power to emergency diesel generator switchgear)	.35g (Securely anchored to) turbine building foundation Loss of function
Emergency Diesel Generator Batteries	Loss of power to emergency generator starting circuit and motor starter (EDG failure to start)	.5g (securely anchored to diesel generator room floor) Loss of function
Standby Diesel Generator Batteries	Loss of power to standby diesel generator motor starter (standby EDG failure to start)	Not characterized Loss of function
Diesel Fire Pump Batteries	Loss of power to diesel fire pump starting circuit and motor starter (diesel fire pump failure to start)	.5g (securely anchored to screenhouse floor) Loss of function
125 V dc Bus	Loss of power to major dc powered electrical equipment (Bus D02, dc core spray valves)	.35g (securely anchored to turbine building foundation) Loss of function/overturning
125 V dc Bus (D02)	Loss of power to major dc powered equipment (Bus D10, station annunciators, misc station power switchgear)	.35g (securely anchored to turbine building foundation) Loss of function

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
125 V dc Bus (2D)	Loss of power to major dc powered equipment (emergency condenser outlet valves, MSIV, emergency condenser make-up valve)	.60 (mounted rigidly to ASDB wall)
125 V dc Bus (D10)	Loss of power to dc powered enclosure spray valve and backup fire water supply valve to core spray through PIS	.35g (securely anchored to turbine building foundation) Loss of function
480 V Bus 1A and 2A	Loss of power to ac powered core spray pumps core spray heat exchanger valve, CRD pumps Loss of function/overturning	.35g (securely anchored to turbine building foundation)
480 V Bus 2B	Loss of power to ac powered core spray valves, electric fire pump Bus 1A and 2A, auto throwover panel	.35g (securely anchored to turbine building foundation) Loss of function/overturning
Bus 1 and 2 and tation Power	Causes secondary failure of auto throwover panel and contactors for core spray valves and ac powered containment spray valve	.35g (securely anchored to turbine building foundation) Overturning
Auto Throwover Panel (C05)	Loss of power to panels 1Y and 3Y	.35g (securely fastened to turbine building north wall, turbine building foundation) Loss of function
1Y Panel (BUS 1Y)	Loss of ac power to control room instrumentation (RDS manual actuation switches, reactor pressure indication, RPS drum level indication, emergency condenser shell level indication)	.32g (securely anchored to turbine building structural steel) Loss of function
RDS Sensor and Actuation Cabinets	Loss of power to RDS valves, sensors and control room indication (reactor and drum level) automatic and manual actuation failure of RDS and fire pump start circuitry (auto only)	.35g (securely anchored to computer room floor, service building) Loss of function/overturning

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
Panels C01, C02 and C40	Loss of indication and control for main control room and RDS panels	.35g (securely anchored to control room floor, service building) Loss of function/overturning
Panel C30	Causes loss of instrumentation on south wall of steam drum enclosure reactor pressure switches for emergency condenser valves and core spray valves, reactor level switches for core spray valves and RDS control room reactor pressure and drum level indication	.4g (securely anchored to reactor building internal structure) Overturning
Electric and Diesel Fire Pump Control Panels (C17 and C09)	Failure of diesel and electric fire pumps to start	.5g (securely anchored to screen-house structure) Loss of function
Emergency Diesel Generator Control Panel (C18)	Failure of emergency diesel generator to start	.5g (securely anchored to diesel generator room wall) Loss of function
Standby Diesel Generator Setup and Step m Transformers	Failure of standby diesel generator to supply emergency power to 2B Bus	.15g (historical performance of unanchored transformers-SSMRP) Loss of function
UPS Battery Chargers	Failure of UPS power supplies to RDS	.35g (securely anchored to turbine building foundation) Overturning onto UPS batteries
Station Battery Chargers	Failure of DC power supplies	.35g (securely anchored to turbine building foundation) Overturning onto station batteries or DC distribution panels

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
MSIV (MO 705J)	Failure to isolate primary system potential blowdown outside containment	.40g (okay as pressure boundary) MSIV is sufficiently restricted that it will not impact surrounding structures.
Emergency Condenser Outlet Valves (MO 7053, MO 7063)	Loss of primary system decay heat sink	Not characterized (okay as pressure boundary). Fails to open because of operator impact on surrounding structure due to seismic motion
Emergency Condenser Makeup Valve (SV 4947)	Loss of primary system decay heat sink	.4g (securely anchored to reactor building internal structure) Fails to open
Ac Core Spray Valves (MO 7C70, MO 7071)	Loss of redundant core spray nozzle	Not characterized (okay as pressure boundary.) Fails to open because of operator impact on surrounding structure due to seismic motion
DC Core Spray Valves (MO 7051, MO 7061)	Loss of primary core spray ring sparger	Not characterized (okay as pressure boundary). Fails to open because of operator impact on surrounding structure due to seismic motion
Ac and dc Enclosure Spray Valves (MO 7068, MO 7064)	Failure of RDS and core spray equipment following LOCA due to exceeding environmental qualification envelope	.4g (well anchored to reactor building internal structure motion limited by supports with reasonable clearance to surrounding structures >6")

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
Core Spray Heat Exchanger Valves (MO 7066, MO 7080)	Inability to supply remotely actuated cooling water to side of core spray heat exchanger (loss of long term cooling heat sink following LOCA or RDS actuation	.4g (securely anchored to foundation of fuel cask loading dock, motion limited by supports and good clearance from surrounding structures) Fails to open
Backup Fire Supply to Containment (MO 7072)	Inability to supply remotely actuated cooling water to fire system inside containment through post incident system (loss of emergency condenser or core spray makeup if coincident with turbine)	.4g (securely anchored to foundation of fuel cask loading dock, motion limited by supports and reasonable clearance with surrounding structures) Fails to open
Core Spray Check Valves (VPI 301, 302)	Core spray or emergency condenser water makeup diversion to turbine building if coincident with turbine building welded fire piping failure	Not characterized Internal failure causing back flow
RDS Isolation Valves (CV 4180 thru 83)	Loss of ability to depressurize reactor allowing low pressure core spray	.4g (securely anchored to reactor building internal structure catalytic) Failure to open
RDS Isolation Solenoid (SV 4980 thru 83)	Loss of ability to vent air from CV 4180-83	.4g (securely fastened to RDS isolation valves) Failure to open
RDS Depressurization Valves (SV 4984 thru 87)	Loss of ability to depressurize reactor allowing low pressure core spray	.4g (securely fastened to reactor building internal structure)
CV4090	Control rod drive pump suction valve CV4090. Loss of reactor make-up via loss of control rod drive suction line.	0.25g (given valve of turbine building pedestal). Valve is mounted to pipe tunnel wall.

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
SV4894	Solenoid valve for CV4090. Loss of reactor make-up via loss of control rod drive suction line.	0.25 (given valve of turbine building pedestal).
I&C Transformers	Loss of Panel 1Y and contactors for ac core spray and enclosure spray valves	.35g (securely anchored to north wall of station power room-turbine building foundation) Fall from supports
Station Power Room Cooling Unit	Loss of Panel 1Y	.32g (secured by turbine building structural steel) Falls from supports
Tool Crib	Loss of UPS B and D	Not characterized Overturn of tool cabinet
Screen	Loss of UPS B and D	.32 (securely attached to steel superstructure of turbine building)
Lights Near Station Batteries	Shorting of station batteries	.32g (secured to turbine building structural steel) Falls from anchors
Lights Near UPS Batteries	Shorting of UPS batteries	.13g (capacity of most fragile UPS blockwall - secured to steel on ceiling) Falls from anchors
Steel Enclosure for Drum Level Mirror and Emergency Light	Loss of RDS drum level Transmitters C&D	.4g (securely anchored to reactor building internal structure) Falls from anchors

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
Vent Ducts in SFP Heat Exchanger Room	Loss of RDS reactor level Transmitter A	.4g (secured to reactor building internal structure) Falls from support
Hypochlorite Tank	Loss of screenhouse fire piping	.5g (securely anchored to screenhouse structure) Falls from support
Circulating Water Piping in Screenhouse	Loss of fire pumps due to flooding	.2g - Inspection Rupture/leakage
Heater, Lights, Battery Charger in Diesel Generator Room	Loss of diesel generator batteries	.5g (securely anchored to diesel generator room wall) Falls from supports
Diesel Generator Cooling Water Head Tank and Muffler	Failure of diesel generator to run	.5g (securely anchored to diesel generator room structure)
Turbine Building Crane	Failure of turbine building fire piping	.32g - interpretation of D'Appolonia modeled as a part of turbine building structure Fails with turbine building structural steel
Reactor Building Crane	Failure of fire system piping in containment or LOCA	.23g - SMA Report 13703.01R003 gantry legs buckling Rail overturns
Cleanup Demin Hoist	Failure of enclosure spray valves	Not characterized Load/hoist falls from rail

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
Screenhouse Trolley	Failure of fire system piping in screenhouse	.5g captured by screen-house structural steel Falls from rails
RDS Hoist	Failure of isolation Valves CV 3182 and 3183	.4g (securely anchored to reactor building internal structure with large gussets and baseplate Overturns
1T Emergency Condenser Beam	Fails emergency condenser level instrument piping (drains shell)	.4g (securely anchored to reactor building internal structure with large gussets and baseplate) Overturns
Reactor Building Crane	LOCA plus potential failure of core spray piping inside containment	.22g interpretation of Whiting Analysis
LT3180-3183	RDS steam drum level instrumentation leads to inability to automatically actuate RDS, automatically start fire pumps and start RDS drum level indication in control room	.4g (reactor building internal structure) anchored to south wall of steam drum Loss of function
LT3184-3187	RDS reactor level instrumentation leads to inability to automatically actuate RDS. Also results in loss of RDS reactor level indication in control room	.4g (reactor building internal structure) anchored to wall in spent fuel pool heat exchanger room Loss of function
LSRE09A-H	Reactor level switches. Leads to inability to automatically actuate ac or dc core spray valves	.4g (reactor building internal structure) anchored to south wall of steam drum and spent fuel pool heat exchanger room wall

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
LS3550	Emergency condenser shell level switch leads to failure of emergency condenser low level annunciation in control room	.12g (emergency condenser shell) attached to piping on side of emergency condenser shell
LIRE19A and B	Drum level indicators leads to loss of ac powered drum level indication in control room	.35g (turbine building column uplift)
LTRE20A	Steam drum level transmitter. Leads to loss of ac powered drum level indication in control room	4.g (reactor building internal structure) located in Panel C30 anchored to floor near personnel lock Loss of function
LERE08B	Steam drum level element. Leads to loss of ac powered drum level indication in control room	4.g (reactor building internal structure) attached to east end of steam drum Loss of function
PSIG11A-H	Reactor pressure switches. Leads to loss of automatic opening of core spray valves on low reactor pressure	.4g (reactor building internal structure) anchored to south wall of drum enclosure and fuel pool heat exchanger room walls Loss of function
PIIA07	Reactor pressure. Leads to loss of reactor pressure indication in control room	.35g (turbine building column uplift) located in Panel C02 in control room Loss of function
PSID28E	Reactor high pressure switch. Leads to loss of high reactor pressure annunciation in control room	.4g (turbine building column uplift) located on control room Panel C02 Loss of function

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
PSRE07	Reactor pressure switches. Loss of automatic actuation of emergency condenser outlet valves	.4g (reactor building internal structure) located on south wall of steam drum enclosure Loss of function
PS636A and B	Enclosure pressure. Loss of automatic actuation of enclosure spray valve	Not characterized. Mounted to angle iron in cable penetration area Loss of function
PS7064A and B	Enclosure pressure. Loss of automatic actuation of enclosure spray valve	Not characterized. Mounted to angle iron in cable penetration area Loss of function
PS612 and 615	Fire pump discharge pressure. Loss of automatic starting of fire pumps on low discharge pressure	.5g (screenhouse) located in fire pump control panels anchored to screenhouse floor Loss of function
PS789-796	Fire pump discharge pressure. Loss of automatic actuation of RDS	.5g (screenhouse) anchored to screenhouse wall Loss of function
PI367	Enclosure pressure. Loss of control room enclosure pressure indication	.35g (turbine building column uplift) located in control room Panel C02 Loss of function
PT174	Enclosure pressure. Loss of control room enclosure pressure indication	.35g (turbine building column uplift) anchored to wall in cable penetration area Loss of function

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
LT3171 and 3175	Enclosure level. Loss of control room indication of enclosure water level	.4g (reactor building internal structure) anchored to wall in recirculation pump room entrance area Loss of function
HSPM07	Fire pump hand switch. Loss of ability to manually start fire pumps locally	.5g (screenhouse) located in fire pump control panels anchored to screenhouse floor Loss of function
HSVEC1	Hand switch for SV4947. Loss of ability to makeup to emergency condenser shell	.35g (service building) located in control room Panel C01 Loss of function
HSPBRDS	RDS push buttons. Loss of ability to start fire pumps or manually actuate RDS from control room	.35g (service building) located in control room Panel C40 Loss of function
HS7053	Emergency condenser outlet valve hand switch. Loss of ability to manually actuate emergency condenser	.35g (service building) located in control room Panel C01 Loss of function
HS7068	Ac enclosure spray valve hand switch. Loss of ability to manually actuate M07068 from control room	.35g (service building) located in Panel C02 in control room. Fail open

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
HS7066	Core spray heat exchanger valve hand switch. Loss of ability to remotely actuate MO7066.	.35g (service building) located in Panel C02 in control room.
HS7080	Core spray heat exchanger back-up valve hand switch. Loss of ability to remotely actuate MO7080.	.35g (service building) located in Panel C02 in control room.
HSPM04	Hand switch for control rod drive pumps. Loss of CRD pumps.	.35g (service building) located in control room.
HSPM02	Hand switch for core spray pumps. Loss of core spray pumps.	.35g (service building) located in Panel C01.
HSSDL	Reactor depressurization system hand switch (drum and reactor level), loss of RDS	.35g (service building) located in Panel C40.
HSRDS	Reactor depressurization system hand switch. Loss of RDS	.35g (service building) located in Panel C40.
CB3550	LS3550 circuit breaker. Loss of emergency condenser level indication in control room	.32g (turbine building superstructure) located in panel in station power room Fail open
CB1YFI	PI1A49 circuit breaker. Loss of reactor pressure indication in control room	.32g (turbine building superstructure) located in Panel 1Y anchored to wall in station power room Fail open
CB1YLI	Circuit breaker in the 1Y panel for reactor	0.32 (turbine building superstructure) located in 1Y panel, mounted to vertical column in station power room.

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
CB7072	MO7072 circuit breaker. Loss of power to MO7072	.35g (turbine building foundation) located in Panel D10 located in station power room Fail open
DISC-1441	Loss of CRD PP#1 (ASDB connection to pump)	.40g (secured to wall in EDG room)
CB1YPB	Circuit breaker to RDS control panel. Loss of ability to manually actuate fire pumps or RDS from control room	.35g (turbine building foundation) located in Panel 1Y anchored to wall in station power room. Fail open
PNL3Y	PIIA07 circuit breaker (in panel 3Y). Loss of pressure indication in control room.	.35g (turbine building foundation) located in Panel 3Y anchored to wall in station power room Fail open
CB7053	Circuit breakers for emergency condenser outlet valves. Loss of ability to actuate emergency condenser	.35g (turbine building foundation) located in Panel D01 anchored to station power room Fail open

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (Z?GA) and Failure Mode
CB7051	Circuit breakers for dc powered core spray valves. Loss of core spray flow through MO7051 and 61	.35g (turbine building foundation) located in Panel D01 anchored to station power room floor Fail open
CBEDG	Emergency diesel generator circuit breaker. Loss of ability to energize emergency bus via EDG	.35g (turbine building foundation) located in Bus 2B anchored to station power room floor Fail open
CB7070	Circuit breakers for ac powered core spray valves. Loss of core spray flow through MO7070 and 71	.35g (turbine building foundation) located in Bus 2B anchored to station power
CBPM06	Ac fire pump circuit breaker. Loss of ac fire pump	.35g (turbine building foundation) located in Bus 2B anchored to floor in station power room Fail open power room Fail open
CBSDG	Circuit breakers for the stand-by diesel generator Loss of stand-by diesel generator.	0.35g (turbine building foundation) located in Bus 2B in station power room.

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
CB7068	Circuit breaker for ac enclosure spray valve. Loss of ability to actuate MO7068	.35g (turbine building foundation) located in Bus 2B anchored to floor in station power room. Fail open.
CBPM04	Circuit breakers for control rod drive pumps. Loss of CRD pumps.	0.35g (turbine building foundation) located in Bus 1A and Bus 2A in station power room.
CBPM02	Circuit breakers for core spray pumps. Loss of core spray pumps.	0.35g (turbine building foundation) located in Bus 1A and 2A in station power room.
CB7064	Circuit breaker for dc enclosure spray valve Loss of ability to actuate MO7064	.35g (turbine building foundation) located in Bus D01 anchored to floor of station power room. Fail open.

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
CB1A2B	1A-2B, 2A-2B tie breakers. Loss of ability to disconnect 2B bus from auxiliary buses	.35g (turbine building foundation) located in Buses 1A and 2A anchored to floor of station power room Fail closed
CB2A2B	Tie breaker for Bus 2A to 2B. Loss of ability to connect the 2A bus to the EDG.	0.35 (turbine building foundation) located in Bus 2A in station power room.
CB7050	Circuit breaker for MSIV. Loss of ability to isolate primary system	.35g (turbine building foundation) located in Bus D01 anchored to floor of station power room
CB7066	Circuit breaker for core spray heat exchanger valve MO7066. Loss of ability to remotely actuate MO7066.	.35g (turbine building foundation) located in Bus 2A, anchored to floor of station power room.

Footnotes:

(1) See Figure V-1 for location of block walls. (Note: block wall M100.PIS is in the fuel cask loading dock area and, therefore, not shown in Figure V-1.)

(2) Failure of these components leads to one of the three initiating events.

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
CB7080	Circuit breaker for core spray heat exchanger back-up valve MO7080. Loss of ability to remotely actuate MO7080.	.35g (turbine building foundation) located in Bus 2B, anchored to floor of station power room.
Access Hatch	Condenser circulating water pump access hatch in screenhouse. Failure of electric fire pump, RDS switches PS789.	.5g (given value of of screenhouse structure).
Air Duct	Air duct in core spray test tank area. Failure of containment level indication (LT3171 and LT3175).	.4g (given value of containment building internal structure).
Computer Equipment	Computer, desks and printers in computer room. Loss of RDS cabinets in computer room.	.35g (given value of service building)
Computer Room Ceiling	Computer room ceiling tiles and supports. Loss of RDS cabinets in computer room.	.35g (given value of service building)
Computer Room Wall	Plaster wall in computer room. Loss of cables from RDS cabinets in computer room to control room.	.35g (given value of service building)
Telephone Room Wall	Wall between computer room and telephone room. Loss of cables from RDS cabinets in computer room to control room.	.35g (given value of service building)

Footnotes:

(1) See Figure V-1 for location of block walls. (Note: block wall M100.PIS is in the fuel cask loading dock area and, therefore, not shown in Figure V-1.)

(2) Failure of these components leads to one of the three initiating events.

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
Emergency Lights and Eye Wash Stations	Lights and emergency eye wash units near the diesel fire pump and emergency diesel generator batteries. Loss of this equipment.	.5g (given value of screenhouse)
File and Storage Cabinets	File drawer and test cabinet, storage cabinets, behind the RDS cabinets in the computer room. Loss of RDS.	.35g (given value of service building)
Floor Grating	a. Grating over rod drive access area. Loss of containment level switches LS3562, 3564 and 3565.	.4g (given value of containment building).
	b. Grating in electrical penetration room, inside containment. Loss of LSRE09	.4g (given value of containment building).
	c. Grating near personnel lock area. Loss of rod drive pumps and LT3180-3184.	.4g (given value of containment building).
	d. Grating at steam drum enclosure access area. Loss of LS3550 and MO7053.	.4g (given value of containment building).
	e. Grating from upper reactor cooling water heat exchanger room to electrical penetration room. Loss of MO7051 and 61 and LSRE09.	.4g (given value of containment building).
Floor Plate	a. Metal floor plate between ventilation unit and clean-up demin pit, at personnel lock area. Loss of MO7070, MO7064 and MO7068.	.4g (given value of containment building).

Footnotes:

(1)See Figure V-1 for location of block walls. (Note: block wall M100.PIS is in the fuel cask loading dock area and, therefore, not shown in Figure V-1.)

(2)Failure of these components leads to one of the three initiating events.

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
	b. Second level mezzanine floor plate in outer electrical penetration room. Loss of SV4980 and 84, LT3180 and 84 and PI367.	.35g (given value of service building)
Instruments	Instrumentation mounted to wall in rod drive access area. Loss of containment level switches LS3562, 3564 and 3565.	.4g (given value of containment building).
Junction Boxes	a. JB-21 mounted to screenhouse wall. Loss of PS789.	.5g (given value of screenhouse).
	b. JB-UPS-A mounted on top of UPSA battery charger cabinet. Loss of UPSA and LT3180 and 84, CV4180, PS789 and SV4984.	.35g (given value of service building).
	c. JB-97 in sphere ventilation room (air shed). Loss of MO7072, Core Spray pump, MO7066, MO7080 and MO7072.	.35g (given value of service building).
	d. Terminal box mounted to wall above the second level in the outer electrical penetration room. Loss of LS3550, LT3171 and 75, LS3562, 64 and 65, MO7070 and 71, LSRE09, rod drive pumps, MO7050, MO7053 and 63, CV4180, SV4980, MO7051 and 61, MO7064, MO7068.	.35g (given value of service building).

Footnotes:

(1) See Figure V-1 for location of block walls. (Note: block wall M100.PIS is in the fuel cask loading dock area and, therefore, not shown in Figure V-1.)

(2) Failure of these components leads to one of the three initiating events.

TABLE V-1 (Continued)

Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
Vent Line	Lube oil tank vent line in outer electrical penetration room. Loss of LT3184 and 80, SV4984, SV4980.	.35g (given value of service building).
Metal Hatch	Access hatch cover to the regen/non-regen heat exchanger room. Loss of LT31/5.	.4g (given value of containment building).
Recirc Pump Valve	Recirc pump suction valve operator, MO-N003A, located over rod drive access area. Loss of containment level switches LS3562, 64 and 65.	.4g (given value of containment building).
Overhead Light	Light suspended from ceiling in core spray pump room. Loss of MO7066.	.4g (given value of fuel cask loading dock).
Poison Tank	Liquid poison storage tank at emergency condenser level. Loss of LS3550.	.4g (given value of containment building).
Pipe Support	Pipe and conduit support located at the north side of the emergency condenser level. Loss of LS3550.	.4g (given value of containment building).
Screen	Safety screen mounted to floor, south of the diesel fire pumps. Loss of diesel fire pumps.	.5g (given value of screenhouse).
Loose Equipment	a. Tools and equipment in rod drive access area. Loss of containment level switches LS3562, 64 and 65.	.4g (given value of containment building).

Footnotes:

(1) See Figure V-1 for location of block walls. (Note: block wall M100.PIS is in the fuel cask loading dock area and, therefore, not shown in Figure V-1.)

(2) Failure of these components leads to one of the three initiating events.

TABLE V-1 (Continued)

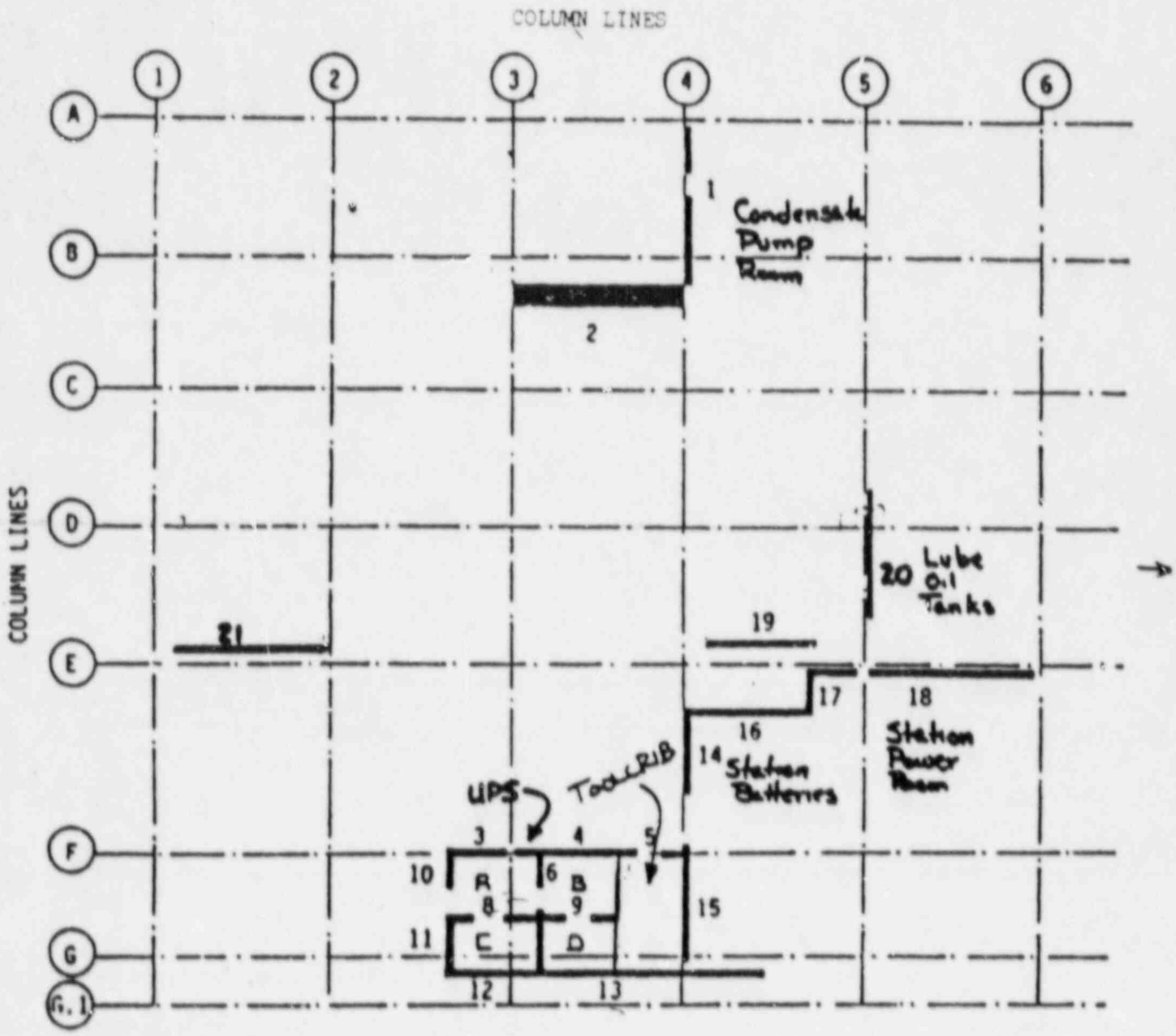
Component	Consequences of Failure	Assumed Capacity (ZPGA) and Failure Mode
	b. Cabinets, desks and equipment in Room 441, Radiation Protection counting room in containment. Loss of LT3175.	0.4g (given value of containment building).
Steel Ceiling	Steel ceiling plate in electrical penetration room, inside containment. Loss of cables coming into containment.	.4g (given value of containment building).
Vent Unit	Ventilation units across from the clean-up demin access area. Loss of LT3184.	.4g (given value of containment building).
Vent Unit 2	Ventilation ducts in sw corner of containment - loss of ECS circuits	.4g (secured to Reactor building internal structure) / /

Footnotes:

(1) See Figure V-1 for location of block walls. (Note: block wall M100.PIS is in the fuel cask loading dock area and, therefore, not shown in Figure V-1.)

(2) Failure of these components leads to one of the three initiating events.

FIGURE V-1



PLAN VIEW OF MASONRY WALLS

VI. METHODOLOGY FOR IDENTIFICATION OF THE SEISMIC "WEAK-LINKS" AT BIG ROCK POINT

Three transients were identified in Section II as being those transients which require sufficient systems and equipment to characterize adequately Big Rock Point operational response to a given seismic event. The list of systems associated with these three transients and the manner in which they interact was sufficiently complete to assure that all important plant system functions were identified regardless of the actual transient which might occur as a result of the earthquake. The three transients chosen were: loss of offsite power; medium steam line break inside containment: and an ATWS. In this section, the procedure by which the most fragile combinations of equipment given any of these transients can be identified will be described. These most fragile combinations of equipment are referred to as the "weakest-links" with respect to attaining safe shutdown following an earthquake at the Big Rock Point site.

The event trees developed for the purpose of this study were presented in Section IV and are similar to the event trees presented in the Big Rock Point PRA. The system functions important to safe shutdown of the reactor are identified in the event tree headings. As stated in Section IV, these headings differ slightly from the headings of the event trees presented in the PRA as, conservatively, they do not include those systems or functions which are not easily shown to be capable of surviving the earthquake. They also contain more detail than do the PRA event trees with respect to those functions which will most likely be required. Existing systems or sets of equipment were identified which will fulfill each function identified by the event tree headings. The logic by which each of these systems succeeds or fails also was extracted from the PRA in the form of fault trees. For any given sequence, a fault tree was applied to each heading in the sequence for which a system functionally failed. By combining the fault trees for each system failure and performing Boolean logic on this combination of trees, the dependencies between each of the systems in a sequence was identified and a listing of minimum combinations of all the failures which must occur to result in a particular reactor state was developed.

As an example, there exists in the loss-of-offsite-power tree (at the end of Branch Point 10) a Sequence PEmC which defines a given set of system failures required to lead to a plant state in which inadequate core cooling occurs. The particular system functional failures which were assumed to occur in this sequence are makeup to the emergency condenser (Em) and core spray failure (C). Given that emergency condenser makeup depends on the FPS for its water source (just as does the core spray) there are some obvious dependencies between these two systems (core spray piping being a specific example).

The fault tree logic for emergency condenser makeup and core spray are presented in Apendix A of this report. The definitions of the bottom events in these trees are presented at the end of that section. It can be seen from the fault trees that failure of either yard piping (PPYARD) or fire piping inside the containment (PP02) was sufficient to satisfy the logic of the trees and fail both systems. These two trees in Appendix A were combined under an AND gate and Boolean logic was performed by use of the SETS code. The results of this exercise are presented in Table VI-1 of this section. This table contains a partial list of the combinations of all the failures (cut sets) which must occur to attain the plant state PEmC. It can be seen that the yard piping and fire piping inside containment do in fact show up as single events leading to the failure of both of these systems, confirming the dependency. Effectively, identification of this dependency in this manner indicates that the occurrence of either of these single events following an earthquake by themselves are sufficient to lead to a plant state in which inadequate core cooling occurs. This exercise has been completed for all the system failures which occur in each sequence of the event trees presented in Section IV. There exists a table of cut sets similar to that presented in Table VI-1 for each sequence. Tens of thousands cut sets exist for the three-event trees as a whole with the size of the cut sets ranging from one to seven members.

In examining the event trees in Section VII, it may be noted that the detail of the trees has been substantially simplified over what exists in the PRA. As an example, the RDS tree was revised to include only a single train of power supplies, sensors, actuation cabinets and depressurization valves because all four trains are essentially identical to each other in terms of their function, location and structural features. In other words, if one train fails as a result of a seismic event this study assumes the likelihood of a similar failure in the other trains is quite high. Table VI-2 contains a list of components which were modularized in this manner. Components which may have a dissimilar seismic resistance (such as the two diesel generators) or have functional dissimilarities in the way they operate (such as the fire pumps and their power supplies) were not combined. Passive components and structures normally unimportant during these transients but whose failures may be made important as a result of ground motion were added to the trees (such as masonry walls).

Additional modularization of bottom events was performed as needed as the fault trees were combined and run through SETS. This modularization consisted of combining a set of independent bottom events beneath an OR gate into a single bottom event. Bottom events simplified in this manner were compressed everywhere that specific combination of events occurred in the trees being run. No bottom event was included in a compressed event identifier that occurred by itself elsewhere in the tree so as not to lose the ability to identify all dependencies between systems. The definitions of these compressed events were saved for later use in identifying the weakest-links in the seismic response of the plant.

Determination of the weakest-links after an earthquake requires knowledge of the seismic strength of the component and the response of the structure to which the component is mounted to ground motion. Given these, a best estimate ground acceleration which will result in the failure of a given component located at a specific location in the plant can be determined. Applying this acceleration to each component within a cut set, one can then determine the acceleration at which all members of a cut set will fail. This acceleration is the acceleration at which the strongest component in the cut set fails and represents the seismic resistance of that cut set. Those cut sets which are satisfied at the lowest ground acceleration are the seismic weak-links at the Big Rock Point Plant.

Section V presented a table of failure modes and effects on all components for which a bottom event exists in the fault trees of Section VII. For each component a conservative ground acceleration was presented above which this study assumes the component fails. Some components have not been characterized in sufficient detail to estimate a ground acceleration at which they will fail. These components have been assigned an arbitrary capacity of zero g. This approach artificially raises the importance of these components for seismic events and allows a relative determination of the value of evaluating these components further.

The acceleration at which the most fragile of the weak-links is satisfied is representative of that size earthquake the plant can be expected to survive without sufficient seismically-induced failures to result in core damage. It is these weak-links at which further evaluations or plant modifications should be aimed if any are necessary. Evaluation and modification of components in the more seismically resistant cut sets produces little measurable benefit unless the weaker-links are also addressed.

The fault trees for each of the system headings in the loss of offsite power, long-term cooling and medium steam-line break-event trees are represented in Section VII. The evaluation of the Big Rock Point Plant as it exists today using this methodology is presented in Section VIII. Evaluation of potential modifications of the weak-links identified by this method is also presented in Section VIII.

TABLE VI-1

LOSP2 =

PPYARD +

CRANE75T +

PP02 +

SPEPRM +

M10018 +

TURBLDG +

RXBLDG +

PPHTC * IST-1 +

TBERM * IST-1 +

IST-8 * IST-9 +

PP04W * IST-9 +

PP03 * IST-9 +

M10019 * IST-9 +

M10014 * IST-9 +

M10017 * IST-8 +

M10017 * PP04W +

M10017 * PP03 +

M10019 * M10017 +

M10014 * M10017 +

PP04W * VPI301 +

PP03 * VPI301 +

M10019 * VPI301 +

TBERM * IST-3 * IST-4 +

TBERM * UPSCHG * IST-3 +

TABLE VI-1 (Continued)

TBERM * JBUPSA * IST-3 +

TBERM * M10010 * IST-3 +

TBERM * M10006 * IST-3 +

TBERM * M10003 * IST-3 +

TBERM * PPHTC * IST-3 +

PPHTC * IST-2 * IST-3 +

M10019 * PPHTC * IST-3 +

M10014 * PPHTC * IST-3 +

RE092ND * IST-1 * IST-13 +

LSRE09 * IST-1 * IST-13 +

IST-1 * IST-10 * IST-11 +

WALKWAY * IST-1 * IST-10 +

RE092ND * IST-1 * IST-10 +

PSIG11 * IST-1 * IST-10 +

LSRE09 * IST-1 * IST-10 +

VPI301 * M10020 * IST-8 +

M10014 * VPI301 * M10020 +

TBERM * IST-3 * IST-5 * IST-6 +

TBERM * M10017 * IST-3 * IST-6 +

TBERM * SDGBAT * IST-3 * IST-7 +

TBERM * SDGFUE * IST-3 * IST-7 +

TBERM * CBSDG * IST-3 * IST-7 +

TBERM * SDGTR2 * IST-3 * IST-7 +

TBERM * SDGTR1 * IST-3 IST-7 +

TBERM * SDG * IST-3 * IST-7 +

WALKWAY * IST-1 * IST-2 * IST-13 +

TABLE VI-1 (Continued)

PSIG11 * IST-1 * IST-12 * IST-13 +

M10019 * RE092ND * IST-3 * IST-4 * IST-13 +

M10019 * UPSCHG * RE092ND * IST-3 * IST-13 +

M10019 * JUBUPSA * RE092ND * IST-3 * IST-13 +

M10019 * M10010 * RE092ND * IST-3 * IST-13 +

M10019 * M10008 * RE092ND * IST-3 * IST-13 +

M10019 * M10006 * RE092ND * IST-3 * IST-13 +

M10019 * M10003 * RE092ND * IST-3 * IST-13 +

M10014 * RE092ND * IST-3 * IST-4 * IST-13 +

M10014 * UPSCHG * RE092ND * IST-3 * IST-13 +

M10014 * JBUPSA * RE092ND IST-3 * IST-13 +

M10014 * M10010 * RE092ND * IST-3 * IST-13 +

M10014 * M10008 * RE092ND * IST-3 * IST-13 +

M10014 * M10006 * RE092ND * IST-3 * IST-13 +

M10014 * M10003 * RE092ND * IST-3 * IST-13 +

RE092ND * IST-2 * IST-3 * IST-4 * IST-13 +

UPSCHG * RE092ND * IST-2 * IST-3 * IST-13 +

JBUPSA * RE092ND * IST-2 * IST-3 * IST-13 +

M10010 * RE092ND * IST-2 * IST-3 * IST-13 +

M10008 * RE092ND * IST-2 * IST-3 * IST-13 +

M10006 * RE092ND * IST-2 * IST-3 * IST-13 +

M10003 * RE092ND * IST-2 * IST-3 * IST-13 +

M10019 * LSRE09 * IST-3 IST-4 * IST-13 +

M10019 * UPSCHG * LSRE09 * IST-3 * IST-13 +

M10019 * JBUPSA * LSRE09 * IST-3 * IST-13 +

M10019 * M10010 * LSRE09 * IST-3 * IST-13 +

TABLE VI-1 (Continued)

M10019 * M10008 * LSRE09 * IST-3 * IST-13 +

M10019 * M10006 * LSRE09 * IST-3 * IST-13 +

M10019 * M10003 * LSRE09 * IST-3 * IST-13 +

M10014 * LSRE09 * IST-3 * IST-4 * IST-13 +

M10014 * UPSCHG * LSRE09 * IST-3 * IST-13 +

M10014 * JBUPSA * LSRE09 * IST-3 * IST-13 +

M10014 * M10010 * LSRE09 * IST-3 * IST-13 +

M10014 * M10008 * LSRE09 * IST-3 * IST-13 +

M10014 * M10006 * LSRE09 * IST-3 * IST-13 +

M10014 * M10013 * LSRE09 * IST-3 * IST-13 +

LSRE09 * IST-2 * IST-3 * IST-4 * IST-13 +

UPSCHG * LSRE09 * IST-2 * IST-3 * IST-13 +

JBUPSA * LSRE09 * IST-2 * IST-3 * IST-13 +

M10010 * LSRE09 * IST-2 * IST-3 * IST-13 +

M10008 * LSRE09 * IST-2 * IST-3 * IST-13 +

M10006 * LSRE09 * IST-2 * IST-3 * IST-13 +

M10003 * LSRE09 * IST-2 * IST-3 * IST-13 +

IST-2 * IST-3 * IST-4 * IST-10 * IST-11 +

WALKWAY * IST-2 * IST-3 * IST-4 * IST-10 +

RE092ND * IST-2 * IST-3 * IST-4 * IST-10 +

PSIG11 * IST-2 * IST-3 * IST-4 * IST-10 +

LSRE09 * IST-2 * IST-3 * IST-4 * IST-10 +

UPSCHG * IST-2 * IST-3 * IST-10 * IST-11 +

UPSCHG * WALKWAY * IST-2 * IST-3 * IST10 +

UPSCHG * RE092ND * IST-2 * IST-3 * IST10 +

UPSCHG * PSIG11 * IST-2 * IST-3 * IST10 +

TABLE VI-1 (Continued)

UPSCHG * LSRE09 * IST-2 * IST-3 * IST-10 +

JBUPSA * IST-2 * IST-3 * IST-10 * IST-11 +

JBUPSA * WALKWAY * IST-2 * IST-3 * IST-10 +

JBUPSA * RE092ND * IST-2 * IST-3 * IST-10 +

JBUPSA * PSIG11 * IST-2 * IST-3 * IST-10 +

JBUPSA * LSRE09 * IST-2 * IST-3 * IST-10 +

M10010 * IST-2 * IST-3 * IST-10 * IST-11 +

M10010 * WALKWAY * IST-2 * IST-3 * IST-10 +

M10010 * RE092ND * IST-2 * IST-3 * IST-10 +

M10010 * PSIG11 * IST-2 * IST-3 * IST-10 +

M10010 * LSRE09 * IST-2 * IST-3 * IST-10 +

M10008 * IST-2 * IST-3 * IST-10 * IST-11 +

M10008 * WALKWAY * IST-2 * IST-3 * IST-10 +

M10008 * RE092ND * IST-2 * IST-3 * IST-10 +

M10008 * PSIG11 * IST-2 * IST-3 * IST-10 +

M10008 * LSRE09 * IST-2 * IST-3 * IST-10 +

M10006 * IST-2 * IST-3 * IST-10 * IST-11 +

M10006 * WALKWAY * IST-2 * IST-3 * IST-10 +

M10006 * RE092ND * IST-2 * IST-3 * IST-10 +

M10006 * PSIG11 * IST-2 * IST-3 * IST-10 +

M10006 * LSRE09 * IST-2 * IST-3 * IST-10 +

M10003 * IST-2 * IST-3 * IST-10 * IST-11 +

M10003 * WALKWAY * IST-2 * IST-3 * IST-10 +

M10003 * RE092ND * IST-2 * IST-3 * IST-10 +

M10003 * PSIG11 * IST-2 * IST-3 * IST-10 +

M10003 * LSRE09 * IST-2 * IST-3 * IST-10 +

TABLE VI-1 (Continued)

M10019 * IST-4 * IST-3 * IST-10 * IST-11 +

M10019 * WALKWAY * IST-4 * IST-3 * IST-10 +

M10019 * RE092ND * IST-4 * IST-3 * IST-10 +

M10019 * PSIG11 * IST-4 * IST-3 * IST-10 +

M10019 * LSRE09 * IST-4 * IST-3 * IST-10 +

M10019 * UPSCHG * IST-3 * IST-10 * IST-11 +

M10019 * WALKWAY * UPSCHG * IST-3 * IST-10 +

M10019 * RE092ND * UPSCHG * IST-3 * IST-10 +

M10019 * PSIG11 * UPSCHG * IST-3 * IST-10 +

M10019 * LSRE09 * UPSCHG * IST-3 * IST-10 +

M10019 * JBUPSA * IST-3 * IST-10 * IST-11 +

TABLE VI-2

System	Major Components Modularized Due to Locational, Functional and Structural Similarities	Major Components Not Modularized	Discussion
Fire Protection System	MO 7070, MO 7071 (AC Core Spray Valves) MO 7051, MO 7061 (DC Core Spray Valves) PSIG 11 A through H (Reactor Pressure) LSRE09 A through H (Reactor Level)	Electric Fire Pump Diesel Fire Pump	AC and DC core spray valves were not modularized due to dependencies on different power sources. Fire pumps not modularized due to dependencies on different power sources. UPS A separated from other UPS to account for EDG starting circuit dependency.
RDS	UPS A-D Sensor Actuation Cabinets LT 3180 through 3183 (Drum Level) LT 3184 through 3187 (Reactor Level) RDS Timers PS 789 through 796 (Fire Pump Pressure) CV 4180 through 4183 (Isolation Valves) SV 4980 through 4983 (Isolation Valve Air Supply) SV 4984 through 4987 (Depressurization Valves)		
Enclosure Spray	PS 636 A and B (Containment Pressure) PS 7064 A and B (Containment Pressure)	MO 7064 (DC Encl Spray) MO 7068 (AC Encl Spray)	Enclosure spray valves not modularized due to dependencies on different power sources.
Emergency Condenser	MO 7053 and MO 7063 (DC Outlet Valves) PSRE07 A through D		

TABLE VI-2 (Continued)

System	Major Components Modularized Due to Locational, Functional and Structural Similarities	Major Components Not Modularized	Discussion
Emergency Power		Emergency Diesel Generator Standby Diesel Generator	Diesel generators not modularized due to dissimilarities in starting circuitry (one automatic, one manual) and structures in which they are housed (one in the screenhouse, one on a truck bed).
Post Incident System	Core Spray Pumps 1A and 2A Bus		
CRD Makeup	Control Rod Drive Pumps 1A and 2A Bus		

VII. FAULT TREE LOGIC SUMMARY

The fault tree logic used to identify the structural weak-links following a seismic event at the Big Rock Point Plant is contained in Appendix A. The trees are drawn for those systems identified as being important during an earthquake in Section III of this report. A fault tree is available for each system heading located in the event trees of Section IV. The bottom events are described in the following table and include all components and dependencies defined in the failure modes and effects table of Section V.

The first four fault trees were used for determining cut sets leading to core damage during LOCAs and Primary System Isolation Failures (with MSIV fault tree). These four trees include Post-Incident System, Core Spray, Reactor Depressurization System and Enclosure Spray and included all dependencies on ac and dc power sources, UPS and all shared system components. The remaining trees were used to evaluate the loss of offsite power and long-term cooling event trees and have had the ac and UPS power dependencies removed as these power sources are included in independent event tree system headings. There are two emergency condenser makeup, emergency condenser valve, core spray, RDS and fire pump trees to account for the availability or absence of emergency power.

VIII. RESULTS AND CONCLUSIONS

A. Results

Table VIII-1 of this section contains the results of the event tree/fault tree evaluations performed using the methodology described in Section VI. This list of cutsets represent the seismic weak-links at Big Rock Point. The cutsets presented in Table VIII-1 were limited to cutsets whose seismic capacities were determined to be less than 0.30G. This cut off point was arbitrarily chosen as a point to limit printing output.

In Table VIII-1, the cutsets are ranked from the most fragile to those with the greatest seismic capacity. The first column in this table list the sequence designator used throughout the analysis to represent the sequence PEvR (Loss of station power with Emergency Condenser Valve failure and failure of RDS). These designations are listed in Table VIII-2. The second column in Table VIII-1 lists the cutset capacity, or ground acceleration at which core damage is assumed to occur due to the sequence of events represented in the cutset. The cutsets are grouped by initiating events, ie., Loss of Station Power (LOSP), LOCA and Long Term Cooling (LTC). Cutset component identifier definitions are listed in Table VIII-3.

As stated in Section VI, the most fragile cutsets are those of which backfits or further analysis will be directed. To determine which component(s) of the cutsets our upgrading effort will be directed at, a list was compiled of all of the basic events whose ground acceleration was less than 0.30G, that were present in the weak-links cutsets (Table VIII-4). It is from among these basic events that the components for upgrading will be chosen. Cutset by cutset, the basic events were reviewed and a determination was made as to which component of the cutset would be easiest and most cost effective to fix, that would raise the capacity of the cutset. Since the upgrading of the seismic capacity of a particular basic event often raised the capacity of several cutsets, there are not as many weak-links to upgrade as there are weak-link cutsets. Table VIII-5 groups the basic events chosen for upgrading by ground acceleration. We can see by the summarization of failures why these basic events have such a pervasive effect on safety.

Several methods are available for raising the assumed capacity of each of the cutsets in which the above mentioned weak-links are a part. Note that an upgrade of any one of the weak-link basic events in a cutset can raise the capacity of a cutset. This is because the capacity of the cutset is equivalent to the strongest component of the cutset, therefore it is not necessary to consider each weak-link in a given cutset. Backfits may take a variety of forms ranging from simple procedure revisions to major structural upgrading of plant structures and equipment.

Proposed fixes for the identified weak-links are listed in Table VIII-6. These are merely recommendations to the plant staff and are not the only alternatives.

ATWS was not handled in the same manner as the other two initiating events, LOSP and LOCA because of the lack of fault tree and event tree logic. Instead, a reactor internal evaluation was completed by Structural Mechanics Associates (SMA 13703.01 October, 1985) to assess the seismic resistance of equipment important to shutting down the plant. The analysis addressed two technical issues, 1) the ability of the plant to SCRAM and 2) the integrity of the reactor vessel supports. The results of the SMA reactor internal evaluation indicated that the core assembly would shift and preclude scram at 0.20G. The reactor vessel support evaluation indicated the supports would fail at 0.21G. Since these items are greater than the 0.12G ground response spectrum, it is not mandatory that we perform seismic upgrades. However, the Technical Review Group will consider any modifications useful in raising the seismic capacity of the weak-links of Big Rock Point.

B. Conclusions

Numerous backfits and evaluations are suggested in the previous section as potential dispositions for raising the cutset capacities to levels at which they are no longer limiting. These dispositions vary in complexity from reanalyzing to major structural reinforcement of large items such as masonry walls and emergency condenser supports. In selecting the backfits to be upgraded, it is desirable to select those which are most pervasive in their effect on upgrading the seismic capacity of the plant and the least costly in terms of capital expenditures and other resources. Cost/benefit analyses are useful in the weighing of one backfit against another during analyses such as these. Unfortunately, the methodology used in this report does not easily lend itself to performing detailed cost benefit evaluations.

The Safe Shutdown Earthquake (SSE) was determined to be 0.12G by Reg Guide 1.6C. In order to establish an adequate margin of safety, the analysis was bound to 0.18G. The weak-links less than or equal to 0.12G (the SSE) were presented to the Technical Review Group (TRG) in the January 1988 meeting. These weak-links are the Emergency Condenser supports and block walls M10018 and M10021. The TRG requested an investigation of the cost involved in upgrading these two weak-links. Proposed fixes for these weak-links will be implemented. Weak-links greater than 0.12G will be reviewed by the TRG and considered for upgrading provided they are not cost prohibitive after the items in the 0.12G category have been resolved. All cost beneficial weak-links greater than 0.12g will be upgraded. They will be reviewed by TRG in order of the weakest-links first.

By upgrading the weak-links identified as those that fail equal to or below the SSE, and proceeding beyond the SSE by 50% or until it becomes cost prohibitive, the conclusion is reached that Big Rock Point is seismically resistant, with an adequate margin of safety, to earthquakes that are likely to occur in this general local.

TABLE VIII-1

	LOSP18	.00E+00	MO7053 * PS6362ND +
	LSOP27	.00E+00	PS6362ND * ISOSCRAM +
	LOSP25	.00E+00	MO7070 * MO7051 * ISOSCRAM +
	LOSP17	.00E+00	CRANE3T * MO7051 * MO7053 +
	LOSP17	.00E+00	MO7070 * MO7051 * MO7053 +
	LOSP26	.00E+00	CRANE3T * MO7051 * ISOSCRAM +
#	LOSP18	1.20E-01	ECSHELL * RDSCV2ND +
	LOSP17	1.20E-01	CRANE3T * MO7051 * ECSHELL +
	LOSP17	1.20E-01	M10021 * VPI301 * M10020 * ECSHELL +
	LOSP17	1.20E-01	M10021 * VPI301 * M10020 * MO7053 +
	LOSP17	1.20E-01	MO7070 * MO7051 * ECSHELL +
#	LOSP3	1.20E-01	VPI301 * M10020 * IST-7 * IST-9 +
#	LOSP3	1.20E-01	VPI301 * M10020 * PS6362ND * IST-7 +
	LOSP18	1.20E-01	ECSHELL * PS6362ND +
	LOSP18	1.20E-01	MO7053 * RDSCV2ND +
	LOSP26	1.20E-01	M10021 * VPI301 * M10020 * ISOSCRAM +
	LOSP27	1.20E-01	RDSCV2ND * ISOSCRAM +
	LOSP25	1.20E-01	ISOSCRAM * VPI301 * M10020 +
	LOSP16	1.20E-01	VPI301 * IST-1 * IST-6 +
#	LOSP1	1.20E-01	VPI301 * M10020 * IST-8 +
#	LOSP2	1.20E-01	VPI301 * M10020 * IST-8 +
	LOSP2	1.30E-01	M10018 +
	LOSP1	1.30E-01	M10018 +
	LOSP16	1.30E-01	M10018 +
	LOSP14	1.30E-01	M10018 +
	LOSP19	1.30E-01	M10018 +
	LOSP15	1.30E-01	M10018 +

#Denotes sequences new to this analysis.

TABLE VIII-1 (Continued)

	LOSP12	1.30E-01	M10018 +
	LOSP6	1.30E-01	M10018 +
#	LOSP28	1.30E-01	ISOSCRAM * M10018 +
#	LOSP26	1.30E-01	ISOSCRAM * M10018 +
#	LOSP25	1.30E-01	ISOSCRAM * M10018 +
#	LOSP27	1.30E-01	ISOSCRAM * M10018 +
#	LOSP20	1.30E-01	ECSHELL * M10018 +
#	LOSP23	1.30E-01	ECSHELL * M10018 +
#	LOSP23	1.30E-01	MO7053 * M10018 +
#	LOSP20	1.30E-01	MO7053 * M10018 +
#	LOSP31	1.30E-01	ISOSCRAM * M10018 +
#	LOSP6	1.30E-01	VPI301 * M10020 * IST-2 * IST-4 +
#	LOSP26	1.30E-01	MO7070 * DC * ISOSCRAM +
#	LOSP21	1.30E-01	ECSHELL * M10018 +
#	LOSP21	1.30E-01	MO7053 * M10018 +
#	LOSP26	1.30E-01	CRANE3T * DC * ISOSCRAM +
	LOSP4	1.30E-01	M10018 +
	LOSP3	1.30E-01	M10018 +
	LOSP7	1.30E-01	M10018 +
	LOSP33	1.30E-01	ISOSCRAM * UPS2ND +
#	LOSP33	1.30E-01	ISOSCRAM * M10018 +
#	LOSP29	1.30E-01	ISOSCRAM * M10018 +
#	LOSP30	1.30E-01	ISOSCRAM * M10018 +
	LOSP11	1.30E-01	M10018 +
	LOSP17	1.30E-01	M10018 +
	LOSP10	1.30E-01	M10018 +

#Denotes sequences new to this analysis.

TABLE VIII-1 (Continued)

#	LOSP17	1.30E-01	CRANE3T * DC * MO7053 +
#	LOSP17	1.30E-01	CRANE3T * DC * ECSHELL +
#	LOSP17	1.30E-01	MO7070 * DC * MO7053 +
#	LOSP17	1.30E-01	MO7070 * DC * ECSHELL +
	LOSP24	1.30E-01	ECSHELL * UPS2ND +
	LOSP24	1.30E-01	M10018 +
	LOSP3	1.30E-01	M10018 +
	LOSP5	1.30E-01	M10018 +
	LOSP24	1.30E-01	MO7053 * UPS2ND +
#	LOSP10	1.30E-01	UPS2ND * M10021 * VPI301 * M10020 +
#	LOSP32	1.30E-01	ISOSCRAM * M10018 +
#	LOSP20	1.30E-01	ECSHELL * M10018 +
#	LOSP20	1.30E-01	MO7053 * M10018 +
#	LOSP10	1.50E-01	UPS2ND * PP06C * VPI301 * M10020 +
#	LOSP10	1.50E-01	UPS2ND * PP05T * VPI301 * M10020 +
#	LOSP26	1.50E-01	PP05T * ISOSCRAM * VPI301 * M10020 +
#	LOSP26	1.50E-01	PP06C * ISOSCRAM * VPI301 * M10020 +
#	LOSP17	1.50E-01	PP06C * M10020 * VPI301 * ECSHELL +
#	LOSP17	1.50E-01	PP06C * M10020 * VPI301 * MO7053 +
#	LOSP17	1.50E-01	PP05T * M10020 * VPI301 * ECSHELL +
#	LOSP17	1.50E-01	PP05T * M10020 * VPI301 * MO7053 +
#	LOSP17	1.60E-01	M10013 * M10020 * VPI301 * MO7053 +
#	LOSP17	1.60E-01	M10013 * M10020 * VPI301 * ECSHELL +
#	LOSP26	1.60E-01	M10013 * VPI301 * M10020 * ISOSCRAM +
	LOSP24	1.60E-01	ECSHELL * M10012 +
	LOSP24	1.60E-01	MO7053 * M10012 +

#Denotes sequences new to this analysis.

TABLE VIII-1 (Continued)

#	LOSP10	1.60E-01	M10012 * M10013 * VPI301 * M10020 +
#	LOSP10	1.60E-01	UPS2ND * M10013 * VPI301 * M10020 +
#	LOSP10	1.60E-01	M10012 * PP06C * VPI301 * M10020 +
#	LOSP10	1.60E-01	M10012 * PP05T * VPI301 * M10020 +
#	LOSP14	1.60E-01	M10012 * UPS2ND * ECSHELL +
#	LOSP14	1.60E-01	M10012 * UPS2ND * LS3550 +
#	LOSP14	1.60E-01	M10012 * UPS2ND * PP06C * VPI301 * M10020 +
#	LOSP14	1.60E-01	M10012 * UPS2ND * M10021 * VPI301 * M10020 +
#	LOSP14	1.60E-01	M10012 * UPS2ND * PP05T * VPI301 * M10020 +
#	LOSP14	1.60E-01	M10012 * UPS2ND * M10013 * VPI301 * M10020 +
#	LOSP10	1.60E-01	M10012 * M10021 * M10020 * VPI301 +
#	LOSP10	2.00E-01	UPS2ND * PPYARD +
	LOSP3	2.00E-01	PS6362ND * IST-1
#	LOSP15	2.00E-01	M10012 * UPS2ND * PPSWCW +
	LOSP19	2.00E-01	PPSWCW * ECSHELL +
	LOSP19	2.00E-01	PPSWCW * MO7053 +
	LOSP3	2.00E-01	IST-1 * IST-9 +
	LOSP26	2.00E-01	PPYARD * ISOSCRAM +
	LOSP17	2.00E-01	PPYARD * MO7053 +
	LOSP17	2.00E-01	PPYARD * ECSHELL +
	LOSP28	2.00E-01	ISOSCRAM * PPSWCW + * LOSP10 2.00E-01 M10012 * PPYARD +
	LOSP11	2.00E-01	UPS2ND * PPSWCW +
	LOSP11	2.00E-01	M10012 * PPSWCW +
	LOSP4	2.00E-01	PPSWCW +
	LOSP6	2.00E-01	IST-1 * IST-4 * IST-5 +
	LOSP6	2.00E-01	PPYARD * IST-4 * IST-5 +

#Denotes sequences new to this analysis.

TABLE VIII-1 (Continued)

	LOSP6	2.00E-01	VPI301 * M10020 * IST-1 * IST-4 +
	LOSP6	2.00E-01	PPYARD * VPI301 * M10020 * IST-4 +
#	LOSP14	2.00E-01	M10012 * UPS2ND * PPYARD +
#	LOSP1	2.00E-01	M10020 * VPI301 * PPYARD +
	LOSP2	2.00E-01	PPYARD +
	LOSP2	2.20E-01	CRANE75T +
	LOSP1	2.20E-01	CRANE75T +
	LOSP14	2.20E-01	M10012 * UPS2ND * CRANE75T +
#	LOSP6	2.20E-01	CRANE75T * IST-4 +
	LOSP17	2.20E-01	CRANE75T * ECSHELL +
	LOSP16	2.20E-01	IST-1 * IST-5 +
	LOSP17	2.20E-01	CRANE75T * MO7053 +
	LOSP25	2.20E-01	ISOSCRAM * CRANE75T +
	LOSP10	2.20E-01	M10012 * CRANE75T +
	LOSP10	2.20E-01	UPS2ND * CRANE75T +
	LOSP26	2.20E-01	ISOSCRAM * CRANE75T +
#	LOSP17	2.50E-01	PP04W * VPI301 * ECSHELL +
#	LOSP17	2.50E-01	PP03 * VPI301 * ECSHELL +
#	LOSP17	2.50E-01	PP04W * VPI301 * MO7053 +
#	LOSP17	2.50E-01	PP03 * VPI301 * MO7053 +
	LOSP19	2.50E-01	PPSCNH * MO7053 +
#	LOSP15	2.50E-01	M10012 * UPS2ND * PPSCNH +
#	LOSP15	2.50E-01	M10012 * UPS2ND * PM07 +
#	LOSP19	2.50E-01	PPSCH * ECSHELL * PM06 +
#	LOSP19	2.50E-01	PM07 * ECSHELL * PM06 +
#	LOSP19	2.50E-01	PPSCNH * ECSHELL +

#Denotes sequences new to this analysis.

TABLE VIII-1 (Continued)

#	LOSP19	2.50E-01	PPSCNH * MO7053 * PM06 +
	LOSP19	2.50E-01	PM07 * MO7053 * PM06 +
	LOSP4	2.50E-01	PPSCNH +
	LOSP29	2.50E-01	ISOSCRAM * PPSCNH +
	LOSP28	2.50E-01	ISOSCRAM * PM07 * PM06 +
	LOSP4	2.50E-01	PM07 * PM06 +
	LOSP10	2.50E-01	UPS2ND * PP04W * VPI301 +
	LOSP10	2.50E-01	M10012 * PP04W * VPI301 +
	LOSP10	2.50E-01	M10012 * PP03 * VPI301 +
	LOSP10	2.50E-01	UPS2ND * PP03 * VPI301 +
	LOSP11	2.50E-01	M10012 * PPSCNH +
	LOSP25	2.50E-01	ISOSCRAM * VPI301 * PP04W +
	LOSP25	2.50E-01	ISOSCRAM * VPI301 * PP03 +
#	LOSP6	2.50E-01	PP04W * VPI301 * IST-4 +
#	LOSP6	2.50E-01	PP03 * VPI301 * IST-4 +
#	LOSP11	2.50E-01	UPS2ND * PM07 * PM06 +
	LOSP17	2.50E-01	CRANE75T * MO7053 +
	LOSP26	2.50E-01	PP04W * VPI301 * ISOSCRAM +
	LOSP26	2.[illegible]0E-01	PP03 * VPI301 * ISOSCRAM +
	LOSP11	2.50E-01	UPS2ND * PPSCNH +
	LOSP11	2.50E-01	M10012 * PM07 * PM06 +
#	LOSP11	2.50E-01	UPS2ND * PPSCH * PM06 +
	LOSP3	2.50E-01	PP03 * VPI301 * IST-9 +
	LOSP3	2.50E-01	PP03 * VPI301 * PS6362ND +
	LOSP3	2.50E-01	PP04W * VPI301 * IST-9 +
	LOSP3	2.50E-01	PP04W * VPI301 * PS6362ND +

#Denotes sequences new to this analysis.

TABLE VIII-1 (Continued)

	LOSP19	2.50E-01	PPSHCN * MO7053 +
	LOSP19	2.50E-01	PM07 * ECSHELL * PM06 +
	LOSP19	2.50E-01	PPSCNH * ECSHELL +
	LOSP19	2.50E-01	PPSCNH * MO7053 * PM06 +
	LOSP1	2.50E-01	PP04W * VPI301 +
	LOSP1	2.50E-01	PP03 * VPI301 +
	LOSP2	2.50E-01	PP03 * VPI301 +
	LOSP2	2.50E-01	PP04W * VPI301 +
#	LOSP3	3.00E-01	PP03 * IST-8 * IST-9 +
#	LOSP3	3.00E-01	PP03 * PS6362ND * IST-8 +
#	LOSP3	3.00E-01	IST-7 * IST-8 * IST-9 +
#	LOSP3	3.00E-01	PS6362ND * IST-7 * IST-8 +
#	LOSP3	3.00E-01	PP04W * IST-9 * IST-8 +
#	LOSP3	3.00E-01	PP04W * PS6362ND * IST-8 +

LOCA

CSLOCA	.00E+00	MO7070 * MO7051 +
CSLOCA	.00E+00	CRANE3T * MO7051 +
RDSLOCA	.00E+00	PS6362ND +
ESCLCA	.00E+00	PS6362ND +
ESCLCA	.00E+00	PS636 * PBSHIELD +
ESCLCA	.00E+00	PS7064 * PBSHIELD +
ESCLCA	.00E+00	CRANE3T * PBSHIELD +
ESCLCA	1.10E+00	M10021 +
CSCLCA	1.10E+00	M10021 +
RDSLOCA	1.20E+00	ECSHELL +
CSLOCA	1.20E+00	ECSHELL +

#Denotes sequences new to this analysis.

TABLE VIII-1 (Continued)

PISLOCA	1.20E+00	ECSHELL +
PISLOCA	1.20E+00	VPI301 * M10020 +
ESLOCA	1.20E+00	ECSHELL +
ESLOCA	1.30E+00	UPS2ND +
ESLOCA	1.30E+00	M10018 +
PISLOCA	1.30E+00	M10018 +
CSLOCA	1.30E+00	M10018 +
RDSLOCA	1.30E+00	UPS2ND +
RDSLOCA	1.30E+00	M10018 +
CSLOCA	1.50E+00	PP06C +
CSLOCA	1.50E+00	PP05T +
ESLOCA	1.50E+00	PP05T +
ESLOCA	1.50E+00	PP06C +
ESLOCA	1.60E+00	M10013 +
CSLOCA	1.60E+00	M10013 +
RDSLOCA	1.60E+00	M100UPS +
RDSLOCA	2.00E+00	PPSWCW +
ESLOCA	2.00E+00	PPSWCW +
CSLOCA	2.00E+00	PPSWCW +
CSLOCA	2.00E+00	PPYARD +
PISLOCA	2.00E+00	PPSWCW +
ESLOCA	2.00E+00	PPYARD +
ESLOCA	2.20E+00	CRANE75T +
PISLOCA	2.20E+00	CRANE75T +
CSLOCA	2.20E+00	CRANE75T +
CSLOCA	2.50E+00	PP04W +

#Denotes sequences new to this analysis.

TABLE VIII-1 (Continued)

CSLOCA	2.50E+00	PPSCNH +
CSLOCA	2.50E+00	PP03 +
CSLOCA	2.50E+00	PPSCNH * PM06 +
CSLOCA	2.50E+00	PM07 * PM06 +
RDSLOCA	2.50E+00	PPSCHN +
RDSLCOA	2.50E+00	PM07 * PM06 +
PISLOCA	2.50E+00	PPSCHN +
PISLOCA	2.50E+00	VPI301 * PP03 +
PISLOCA	2.50E+00	VPI301 * PP04W +
PISLOCA	2.50E+00	PM07 * PM06 +
ESLOCA	2.50E+00	PPSCNH +
ESLOCA	2.50E+00	PP03 +
ESLOCA	2.50E+00	PP04W +
ESLOCA	2.50E+00	PM07 * PM06 +
ESLOCA	2.50E+00	PPSCN * PM06 +
		LONG TERM COOLING
LTC10	1.30E-01	M10018
LTC1	1.30E-01	M10018
LTC6	1.30E-01	M10018
LTC2	1.30E-01	M10018
LTC3	1.30E-01	M10018
LTCPUY	1.30E-01	M10018
LTC5	1.30E-01	M10018
LTC10	1.60E-01	UPS2ND * M10012 +
LTC6	1.60E-01	UPS2ND * M10012 +
LTC5	1.60E-01	UPS2ND * M10012 +

#Denotes sequences new to this analysis.

TABLE VIII-1 (Continued)

LTC2	2.50E-01	PPCRD4 * PPYARD +
LTC2	2.50E-01	PPCRD4 * CRANE75T +
LTC2	2.50E-01	CV4090 * CRANE3T * DC +
LTC2	2.50E-01	PPCRD3 * CRANE3T * DC +
LTC2	2.50E-01	PPCRD4 * MO7070 * DC +
LTC2	2.50E-01	CV4090 * CRANE3T * MO7051 +
LTC2	2.50E-01	PPCRD3 * CRANE3T * MO7051 +
LTC2	2.50E-01	PPCRD3 * MO7070 * MO7051 +
LTC2	2.50E-01	CV4090 * PPO4W * VPI301 +
LTC2	2.50E-01	PPCRD3 * PPO4W * VPI301 +
LTC2	2.50E-01	CV4090 * PPO3 * VPI301 +
LTC2	2.50E-01	PPCRD3 * PPO3 * VPI301 +
LTC2	2.50E-01	PPCRD4 * PPO6C * VPI301 * M10020 +
LTC2	2.50E-01	CV4090 * PPO5T * VPI301 * M10020 +
LTC2	2.50E-01	PPCRD3 * PPO5T * VPI301 * M10020 +
LTC2	2.50E-01	PPCRD4 * M10021 * VPI301 * M10020 +
LTC2	2.50E-01	CV4090 * M10013 * VPI301 * M10020 +
LTC2	2.50E-01	PPCRD3 * M10013 * VPI301 * M10020 +
LTC2	2.50E-01	CV4090 * CRANE75T +
LTC2	2.50E-01	CV4090 * PPYARD +
LTC2	2.50E-01	PPCRD3 * CRANE75T +
LTC2	2.50E-01	PPCRD3 * PPYARD +
LTC2	2.50E-01	PPCRD4 * PPO3 * VPI301 +
LTC2	2.50E-01	PPCRD3 * MO7070 * DC +
LTC2	2.50E-01	CV4090 * MO7070 * DC +
LTC2	2.50E-01	PPCRD4 * CRANE75T * MO7051 +

#Denotes sequences new to this analysis.

TABLE VIII-1 (Continued)

LTC2	2.50E-01	PPCRD4 * PP04W * VPI301 +
LTC2	2.50E-01	CV4090 * MO7070 * MO7051 +
LTC2	2.50E-01	PPCRD3 * MO7070 * MO7051 +
LTC2	2.50E-01	PPCRD4 * CRANE3T * DC +
LTC2	2.50E-01	CV4090 * M10021 * VPI301 * M10020 +
LTC2	2.50E-01	CV4090 * PP06C * VPI301 * M10020 +
LTC2	2.50E-01	PPCRD3 * M10021 * VPI301 * M10020 +
LTC2	2.50E-01	PPCRD4 * PP05T * VPI301 * M10020 +
LTC2	2.50E-01	PPCRD4 * M10012 * VPI301 * M10020 +
LTC2	2.50E-01	PPCRD3 * PP06C * VPI301 * M10020 +
LTC1	2.50E-01	CRANE75T * PPCRD4 +
LTC1	2.50E-01	VPI301 * M10020 * PPCRD4 +
LTC1	2.50E-01	PP04W * VPI301 * CV4090 +
LTC1	2.50E-01	PP04W * VPI301 * PPCRD3 +
LTC1	2.50E-01	PP03 * VPI301 * PPCRD4 +
LTC1	2.50E-01	CRANE75T * PPCRD3 +
LTC1	2.50E-01	CRANE75T * CV4090 +
LTC1	2.50E-01	PP04W * VPI301 * PPCRD4 +
LTC1	2.50E-01	M10020 * VPI301 * CV4090 +
LTC1	2.50E-01	PP03 * VPI301 * CV4090 +
LTC1	2.50E-01	M10020 * VPI301 * PPCRD3 +
LTC1	2.50E-01	PP03 * VPI301 * PPCRD3 +
LTC3	2.50E-01	CV4090 * PS6362ND +
LTC3	2.50E-01	PPCRD4 * RDSCV2ND +
LTC3	2.50E-01	PPCRD3 * RDSCV2ND +
LTC3	2.50E-01	CV4090 * RDSCV2ND +

#Denotes sequences new to this analysis.

TABLE VIII-1 (Continued)

LTC3	2.50E-01	PPCRD3 * PS6362ND +
LTC3	2.50E-01	PPCRD4 * PS6362ND +
LTCPUY	2.50E-01	CV4090 * UPS2ND +
LTCPUY	2.50E-01	PPCRD3 * UPS2ND +
LTCPUY	2.50E-01	CV4090 * M10012 +
LTCPUY	2.50E-01	PPCRD3 * M10012 +
LTCPUY	2.50E-01	PPCRD4 * M10012 +
LTCPUY	2.50E-01	PPCRD4 * UPS2ND +

#Denotes sequences new to this analysis.

TABLE VIII-2
SEQUENCE DESCRIPTOR DESIGNATORS

DESIGNATOR	SEQUENCE DESCRIPTOR		
LOCA			
PISLOCA	LLp		
CSLOCA	LC		
RDSLOCA	LR		
ESLOCA	LE		
LOSP			
(LTC)	PL		
LOSP1	PEmLp		
LOSP2	PEmC		
LOSP3	PEmR		
LOSP4	PF		
LOSP5 (LTC)	PQL		
LOSP6	PQEmLp		
LOSP7	PQEmC		
LOSP8	PQF		
LOSP9	PUL		
LOSP10	PUEm		
LOSP11	PUF		
LOSP12	PQEmR		
LOSP13 (LTC)	PUQL		
LOSP14	PUQEm		
LOSP15	PUQF		
LOSP16	PEvLp		
LOSP17	PEvC		
LOSP18	PEvR		
LOSP19	PEvF		
LOSP20	PEvQLp		
LOSP21	PEvQC		
LOSP22	PEvQR	LTC	
LOSP23	PEvQF	LTC1	PYLp
LOSP24	PEvU	LTC2	PYC
LOSP25	PILp	LTC3	PYR
LOSP26	PIC	LTC4	PQLp
LOSP27	PIR	LTC5	PQC
LOSP28	PIF	LTC6	PQR
LOSP29	PIQLp	LTCPUY	PUY(3)
LOSP30	PIQC	LTC10	PQUY
LOSP31	PIQR		
LOSP32	PIQF		
LOSP33	PIU		

TABLE VIII-3

BOTTOM EVENT DEFINTION LIST

NAME	GROUND ACCEL	DESCRIPTION
31802ND	0.40	VENT DUCT AND RCW PIPING
31842ND	0.40	SD MIRROR & EMERG LIGHT
ACSHTCH	0.50	CCW PUMP ACCESS HATCH IN SCREENHOUSE ROOF
AIRDUCT	0.40	AIRDUCT IN CORE SPRAY TEST TANK AREA
AIRDUCT1	0.40	AIRDUCT IN CORE SPRAY TEST TANK AREA
ASDBAT	0.60	ASDB BATTERIES
ASDBBE	0.60	ASD BUILDING FAILURE
ASDBLDG	0.60	ASDB STRUCTURE
ASDCHG	0.60	ASDB BATTERY CHARGER
BATCHG	0.35	DC BATTERY CHARGER
BEAM1T	0.40	1 TON BEAM ON ECS LEVEL
BUS1A	0.35	AC BUS 1A
BUS1Y	0.32	PANEL 1Y
BUS2A	0.35	AC BUS 2A
BUS2B	0.35	BUS 2B
C05	0.35	AUTO THROWOVER PANEL
CAC61	0.35	RDS ACTUATION CAB
CB1A2A	0.35	CIRCUIT BREAKER 1A-2A
CB1A2B	0.35	CIRCUIT BREAKER 1A-2B
CB1Y	0.32	CIRCUIT BREAKER 1Y
CB1YLI	0.32	CIRCUIT BREAKER RX LEVEL INDICATOR
CB1YPB	0.32	CIRCUIT BREAKER RDS PUSH BUTTON
CB2A2B	0.35	2A-2B TIE BREAKER
CB2B	0.35	CIRCUIT BREAKER 2B
CB3550	0.35	CIRCUIT BREAKER LS-3550
CB7050	0.35	MSIV CIRCUIT BREAKER
CB7051	0.35	CIRCUIT BREAKER MO7051 & 61
CB7053	0.35	CIRCUIT BREAKER MO-7053 & 63
CB7054	0.35	CIRCUIT BREAKER MO-7054
CB7064	0.35	CIRCUIT BREAKER MO7064
CB7066	0.35	CIRCUIT BREAKER MO-7066
CB7068	0.35	CIRCUIT BREAKER MO-7068
CB7070	0.35	CIRCUIT BREAKER MO-7070
CB7072	0.35	CIRCUIT BREAKER MO-7072
CB7080	0.35	CIRCUIT BREAKER MO-7080
CB721D33	0.35	CIRCUIT BREAKER LS-3550
CBEDG	0.35	EDG CIRCUIT BREAKER
CBPM02	0.35	CIRCUIT BREAKER CORE SPRAY PUMPS
CBPM04	0.35	CIRCUIT BREAKERS CRD PUMPS
CBPM04A	0.35	CIRCUIT BREAKERS CRD PUMPS

TABLE VIII-3 (Continued)

CBPM05	0.35	EFP CIRCUIT BREAKER
CBPM06	0.35	CIRCUIT BREAKERS EFP
CBSDG	0.35	CIRCUIT BREAKERS SDG
CDST	0.95	CONDENSATE STORAGE TANK
CO5	0.35	AUTO THROWOVER PANEL
COMPEQP	0.35	EQUIP IN COMPUTER RM
COMPRMC	0.35	COMPUTER RM CEILING
COMPWL	0.35	COMPUTER ROOM WALLS
COOLUNIT	0.35	HEATING/COOLING UNIT NEAR 1Y
CRANE25T	0.32	25T CRANE
CRANE2T	0.50	2T CRANE
CRANE3T	0.00	3T CRANE
CRANE75T	0.22	75 T CRANE FAILS
CRDFIL	0.40	CRD FILTERS
CRDPD	0.40	CRD PULSE DAMPENER
CRTRWL	0.35	COMP RM/TELEPHONE WALL
CV4090	0.25	CRD PUMP SUCTION VLV
CV4180	0.40	RDS ISOLATION VLV CV-4180-83
DC	0.13	DC POWER GATE
DCBAT	0.35	DC BATTERIES
DCBUS	0.35	DC BUS D01
DCLTS	0.32	LIGHTS ABOVE STATION BATTERIES
ECSHELL	0.12	EMERG. CONDENSER SHELL FAILS
EDG	0.50	EDG
EDGBAT	0.50	EDG BATTERIES
EDGBE	0.50	EDG FAILURE
EDGBLDG	0.50	EDG BUILDING STRUCTURE
EDGFUE	0.99	EDG FUEL
EDGFUSES	0.99	EDG UNERGROUND FUEL TNK
EDGTRN	0.99	OPERATOR EROOR TO TRANSFER EDG TO ASDB
EDGTRNSFR	0.40	TRS-1442 AND TRS-1401
EMLTEYS	0.50	EMERG LIGHT & EYE WASH STATION
EQUIPLCK	0.40	EQUIPMENT LOCK STRUCTURE
FDSCCOMP	0.35	FILE DRAWER & STORAGE IN COMPUTER RM
FLRGRAT	0.40	FLOOR GRATING OVER ROD DRIVE ACCESS
FLRGRAT1	0.40	FLOOR GRATING OVER ROD DRIVF ACESS
FLRPLT	0.40	FLOOR GRATES, DEMIN AREA
FPPINH	0.35	FIRE PUMP INHIBITOR HAND SWITCH
GRASA	0.40	WALKWAT GRATING AT SD ACCESS AREA
GRAT2	0.40	FLOOR GRATING IN CONT. ELECTRICAL PENETRATION RM
GRATPER	0.40	WALKWAT GRATING NEAR PERSONNEL LOCK
HS7053	0.35	ECS OUTLET VLV HAND SWITCH
HS7066	0.35	HAND SWITCH FOR EC OUTLET VALVES
HS7068	0.35	HAND SWITCH MO-7068
HS7080	0.35	HAND SWITCH FOR MO-7080
HS7902	0.35	ECS MAKE-UP VLV
HSPM02	0.35	CORE SPRAY PUMP CONTROL SWITCH
HSPM04	0.35	CRD PUMPS HAND SWITCH
HSPM07	0.50	HAND SWITCH FOR FIRE PUMPS

TABLE VIII-3 (Continued)

HSRDS	0.35	RDS CONTROL SWITCHES
HSSDL	0.35	RDS HAND SWITCH DRUM & RX LEVEL
HSVEC1	0.40	HAND SWITCH FOR ECS MAKEUP VLV
HX01	0.40	CORE SPRAY HX
HYPOTNK	0.50	HYPOCHLORITE TNK
INST1	0.40	INSTURMENTS IN ROD DRIVE ACCESS
ISOSCRAM	0.00	MSIV CONTROL SIWTCH
ISOVLV3	0.99	RDS ISOLATION VALVES
JB21	0.50	JUNCTION BOX 21 IN SCREENHOUSE
JBUPSA	0.35	JUNCTION BOX UPSA CABINET
LEQUIP	0.40	LOOSE EQUIPMENT IN ROD DRIVE ACCESS
LERE08	0.40	STEAM DRUM LEVEL ELEMENT
LI3380	0.35	SD LEVEL INDICATION
LIIA40	0.35	RDS RX WATER LEVEL INDICATOR C40
LIRE19	0.35	SD LEVEL INDICATION
LOCAHOSE	0.99	FIRE HOSE TO CORE SPRAY HEAT EXCHANGER
LOTVL	0.35	LUBE OIL TANK VENT LINE
LPNLAS	0.35	LOCAL PANEL AND SPHERE VENT ISOLATION VLV IN AIR
LS3550	0.12	ECS SHELL LEVEL SWITCH
LS3562	0.00	LEVEL SWITCH
LS3564	0.00	LEVEL SWITCH
LSRE09	0.40	RX LEVEL SWITCHES RE09
LT3171	0.40	CONTAINMENT LEVEL TRANSMITTER
LT3175	0.40	CONTAINMENT LEVEL TRANSMITTER
LT3180	0.40	RDS RX LEVEL TRANSMITTER
LT3184	0.40	SD LEVEL TRANSMITTER
LTIA39	0.40	RVG RX WATER LEVEL TRANSMITTER
LTRE20	0.40	SD LEVEL TRANSMITTER
M10001	0.33	BLOCK WALL
M10003	0.63	BLOCK WALL
M10004	0.53	BLOCK WALL
M10005	0.53	BLOCK WALL
M10006	0.33	BLOCK WALL
M10008	0.31	BLOCK WALL
M10009	0.31	BLOCK WALL
M10010	0.33	BLOCK WALL
M10012	0.37	BLOCK WALL
M10013	0.16	BLOCK WALL
M10014	0.53	BLOCK WALL
M10016	0.53	BLOCK WALL
M10017	0.99	BLOCK WALL
M10018	0.13	BLOCK WALL
M10019	0.53	BLOCK WALL
M10020	0.12	BLOCK WALL
M10021	0.11	BLOCK WALL
M10022	0.30	BLOCK WALL
M100PIS	0.40	BLOCK WALL MODULE
M100UPS	0.16	COMBINATION BLOCK WALLS
METHAT	0.40	ACCESS HATCH TO REGEN-NON-REGEN RM

TABLE VIII-3 (Continued)

MO7050	0.40	MSIV
MO7050BE	0.40	MSIV
MO7051	0.00	CORE SPRAY VLV MO7051 & 61
MO7053	0.00	ECS VLV MO7053 &63
MO7054BE	0.40	MO-7054
MO7064	0.40	ENCLOSURE SPRAY VLV
MO7064BE	0.40	ENCLOSURE SPRAY VLV
MO7066	0.40	FIRE WATER TO CORE SPRAY HX VLV
MO7066BE	0.40	FIRE WATER TO CORE SPRAY HX VLV
MO7068	0.40	ENCLOSURE SPRAY VLV
MO7068BE	0.40	ENCLOSURE SPRAY VLV
MO7070	0.00	CORE SPRAY VLV MO7070 &71
MO7070BE	0.00	CORE SPRAY VLV MO7070 & 71
MO7072	0.40	FIRE WATER VLV AROUND PIS
MO7072BE	0.40	FIRE WATER VLV AROUND PIS
MO7080	0.40	BYPASS VLV FOR MO7066
MON003A	0.40	RECIRC PP SUCTION VLV MANUAL OPERATOR
OVLEYW	0.50	OVERHEAD LIGHT&EYE WASH NEAR FIRE BAT
OVRHDLT1	0.40	OVERHEAD LIGHT IN CORE SPRAY HX RM
P7BAT	0.50	DFP BATTERIES
P7FUE	0.99	DFP UNDERGROUND FUEL
P7FUEL	0.99	DFP UNDERGROUND FUEL
PBRDS	0.35	RDS PUSH BUTTONS
PBSCRH	0.25	FIRE PMP PUSH BUUTON, SCREEN HOUSE
PBSHIELD	0.00	LEAD SHIELD PT-174
PI367	0.35	PRESSURE IND
PIIA07	0.40	RX PRESSURE INDICATION
PIIA07BE	0.40	RX PRESSURE INDICATION
PM02	0.40	CORE SPRAY PUMPS
PM04	0.40	CONTROL ROD DRIVE PUMPS
PM04A	0.40	CONTROL ROD DRIVE PUMPS
PM05	0.25	EFP
PM06	0.25	EFP
PM07	0.25	DFP
PM7BAT	0.50	DFP BATTERIES
PNC09	0.50	DFP CONTROL PANEL
PNL20	0.40	PANEL C20
PNL30	0.40	PANEL C30
PNL3Y	0.35	PANEL 3Y
PNLC09	0.50	DFP CONTROL PANEL
PNLC17	0.50	EFP CONTROL PANEL
PNLC18	0.50	PANEL C18
PNLC30	0.40	PANEL C30
PNLD01	0.35	PANEL D01
PNLD02	0.35	DC DISTR. BUS D02
PNLD10	0.35	DC PANEL D10
PNLD2D	0.60	ASDB DC DISTR. PANEL 2D
POSTNK	0.40	POISON TANK
PP01	0.40	FIRE PIPING IN CORE SPRAY PUMP RM

TABLE VIII-3 (Continued)

PP02	0.40	CORE SPRAY PIPING INSIDE CONTAINMENT FAILS
PP03	0.25	WELDED FIRE PIPING FAILS
PP04W	0.25	WELDED CORE SPRAY PIPING
PP05T	0.15	THREADED FIRE PIPING TURB. BLDG
PP06C	0.15	VICTROLIC COUPLING-FIRE PIPING
PPCRD1	0.40	CRD PIPE FROM PUMPS TO NC-18
PPCRD2	0.40	CRD PIPE FROM CONTAINMENT TO PUMP SUCTION
PPCRD3	0.25	CRD PIPE IN PIPE TUNNEL
PPCRD4	0.25	CRD PIPE IN CONDENSATE PUMP ROOM
PPCRD5	0.99	CRD PIPE UNDERGROUND FROM CDST
PPHTC	0.32	HEATING PIPING, STATION POWER RM
PPSCN	0.25	SCREENHOUSE PIPING
PPSCNH	0.25	SCREENHOUSE FIRE PIPING
PPSUP1	0.40	PIPE SUPPORTS, N SIDE OF ECS LEVEL
PPSWCW	0.20	CIRC WATER PIPING
PPYARD	0.20	CAST IRON UNDERGROUND FIRE PIPING
PS612	0.50	PRESSURE SWITCH
PS615	0.50	PS-615 EFP DISCHARGE PRESSURE
PS632ND	0.00	VENT DUCT&PIPING MONITOR NEAR PRESS. SWITCH
PS636	0.00	ENCLOSURE SPRAY VLV PRESSURE SWITCH
PS6362ND	0.00	VENT DECT&ILRT PIPING NEAR PRESS.SWITCH
PS7054	0.00	PRESSURE SWITCHES
PS7064	0.00	ENCLOSURE SPRAY VLV PRESSURE SWITCHES
PS789	0.50	FIRE PUMP DISCHARGE PRESS SWITCH
PSID28	0.40	RX HIGH PRESSURE SWITCH
PSIG11	0.40	RX HIGH PRESSURE SWITCH
PSRE07	0.40	ECS OUTLEY VLV CONTROL SWITCH
PT174	0.35	PRESSURE TRANSMITTER
RAC11	0.35	RDS ACTUATION AND SENSOR CABINET
RDSCV2ND	0.12	ECS SHELL AND 2 TON WINCH
RDSHOIST	0.40	RDS HOIST ON ECS LEVEL
RDSLTLS	0.40	LT3180 AND LT3184
RDSPIPE	0.40	RDS PIPING OUTSIDE SD
RE092ND	0.40	RCW ASDB SPF PIPING BY RE09 SWITCH
RE092ND	0.40	RCW ABD SPF PIPING IN RE09 AREA
RO0702ND	0.35	I & C TRANSFORMER AND AC BUS 1 &2
RO702ND	0.35	I&C TRANSFORMER AC BUS 1&2
RO7064	0.99	RELAY CONTACTS MO-7054
RO7070	0.35	RELAY CONTACTS FOR MO7070 & 71
ROOM441	0.40	ROOM 441, MISC. EQUIPMENT
RXBLDG	0.40	RX BUILDING FAILS
RXCLCHX	0.40	RX COOLING WATER HX
SCHDW	0.50	SCREENHOUSE WALKWAY
SCHTLY	0.50	2T TROLLEY IN SCREENHOUSE
SCNHBLG	0.50	SCREENHOUSE STRUCTURE
SCNHSBLD	0.50	SCREENHOUSE STRUCTURE
SCNHSBLG	0.50	SCREENHOUSE STRUCTURE
SCREEN	0.32	SCREEN IN FRONT OF TOOLCRIB
SCRN	0.50	SCRENN BEHIND DFP

TABLE VIII-3 (Continued)

SDG	0.00	STANDBY DIESEL GENERATOR
SDGBAT	0.00	STANDBY DIESEL GEN BATTERIES
SDGFUE	0.00	STANDBY DIESEL GEN FUEL TANK
SDGTR1	0.15	SDG TRANFORMER #1
SDGTR2	0.15	SDG TRANSORMER #2
SDLVL	0.35	LIRE19 AND LTRE20 AND LERE08
SLMEPRM	0.35	2ND LEVEL MAZZANINE, OUTER CABLE
SPEPRM	0.40	STEEL CEILING-INNER ELECTR. PENETRATION
SV4894	0.40	SOLENOID FOR CV-4090
SV4947	0.40	ECS MAKEUP VLV
SV498	0.40	RDS ISOLATION VLV 4980-83
SV4984	0.40	RDS DEPRESS VLV
TBERM	0.35	TERMINAL BOX ELECTRICAL PEN. RM
TESTCAB	0.35	TEST CABINET IN COMPUTER RM
TOOLCRIB	0.00	TOOL CABINETS IN TOOL CRIB
TRANSTCH	0.50	CRD PP #1 DISC. SWITCH 1441
TURBLDG	0.32	TURBINE BUILDING FAILS
UPS	0.35	UPS BATTERIES
UPS2ND	0.13	VENT, DUCTS, LIGHTS, UPS
UPSA	0.35	UPSA BATTERIES
UPSABE	0.35	UPSA FAILURE
UPSBAT	0.35	UPS BATTERIES
UPSBE	0.35	UPS FAILURE
UPSCHG	0.35	UPS BATTERY CHARGER
UVRE2B	0.50	UNDERVOLTAGE RELAY ON 2B
VENT2	0.40	SHPERE HEATING &COOLING UNIT DUCTS
VENTUNT	0.40	VENTILATION CABINETS W OF PERSONNEL LOCK
VPI301	0.00	CORE SPRAY CHECK VLV 301 & 302
WALKWAY	0.40	WALKWAY GRATING TO ELECT PENETRATION RM
XLEVEL	0.40	CONTAINMENT LEVEL TRANSMITTERS

TABLE VIII-4

COMPONENTS OF CUTSETS THAT ARE LESS THAN 0.30G

PPYARD	0.20
PPSWCW	0.20
CRANE75T	0.22
PP04W	0.25
PP03	0.25
PPSCNH	0.25
PM07	0.25
PM06	0.25
PPCRD4	0.25
CV4090	0.25
PPCRD3	0.25
ECSHELL	0.12
RDSCV2ND	0.12
CRANE3T	0.00
MO7051	0.00
M10021	0.11
VP1301	0.00
M10020	0.12
MO7053	0.00
MO7070	0.00
PS636ND	0.00
ISOSCRAM	0.00
M10018	0.13
DC	0.13
UPS2ND	0.13
PP06C	0.15
PP05C	0.15
M10012	0.16
M10013	0.16
PPSWCW	0.20
PBSHIELD	0.00

IST-6 LOSP16	
PP03	0.25
PP04W	0.25
M10020	0.12

IST-1 LOSP6	
LS3550	0.12

IST-7 LOSP3	
M10013	0.16
M10021	0.11
PP05T	0.15
PP06C	0.15

IST-9 LOSP3	
RDSCV2ND	0.12

IST-5 LOSP16	
TRNYPIPL	0.20
CRANE75T	0.22

IST-1 LOSP16	
MO7053	0.00
ECSHELL	0.12

IST-2 LOSP6	
PP06C	0.15
M10013	0.16
M10021	0.11
PP05T	0.15

IST-4 LOSP6	
UPS2ND	0.13
M10012	0.16

IST-1 LOSP3	
CRANE75T	0.22
PPYARD	0.20

IST-8 LOSP1	
PP06C	0.15
M10013	0.16
M10021	0.11
PP05T	0.15

TABLE VIII-5

WEAK-LINKS

WEAK-LINKS AT 0.11G

M10021 This block wall is located between the machine shop and the laydown area in the turbine building. The 2400 V switchgear cable and the fire piping penetrate this block wall. Failure of this wall causes failure of the Core Spray and Enclosure Spray systems during a LOCA only because of lack of time to isolate the turbine building fire piping and circumvent the fire system pipe break.
(See Figure V-1 for location of this block wall)

WEAK-LINKS AT 0.12G

ECSHELL The emergency condenser shell support failure causes the failure of the emergency condenser. This failure not only causes loss of the primary heat removal source during a loss of station power, but it also could cause a loss of the EDS valves, and loss of the fire water supply to the core spray system either through fire water make-up line to EC rupture or the EC shell falling off the ECS level and rupturing the surrounding enclosure spray and core spray piping.

M10018 This block wall is the west wall of the station power room. Failure of this wall takes out the 2400V bus, Bus 1A, 2A and 2B. Because of the large amount of dependence in all systems on these power supplies, failure of this block wall can fail multiple safety systems simultaneously.
(See Figure V-1 for location of this block wall).

WEAK-LINKS AT 0.13G

M10020 This block wall is the south wall of the lube oil tank room. Failure of this wall will damage or fail the fire water supply to containment. This particular segment of piping can be isolated from the fire system by manual closure of VFP-29 and 30. Closure of these valves eliminates flow diversion of the fire system and makes other paths of fire water available, given time is available as in loss of station power sequences where there is a need for

TABLE VIII-5 (Continued)

	core spray, enclosure spray or emergency condenser make-up. (See Figure V-1 for location of this block wall).
OR VPI301	In order for fire water supply to core spray and enclosure spray to fail from a pipe rupture in the lube oil tank room, VPI-301 and VPI-302, the core spray check valves, must fail to hold, thereby preventing an alternate path of fire water supply to the containment from flowing out of these paths. Therefore M10020 will only cause a failure if VPI-301 and 302 fail simultaneously with M10020. Therefore we can fix either M10020 or VPI-301 and VPI-302. A deterministic analyses of these check valves was never performed, therefore we assumed that they would fail a 0.0G.
UPS2ND	Failure of the vent ducts and lights in the UPS room to stay secured are assumed to fail fail the UPS batteries, and therefore the RDS system.
	WEAK-LINKS AT 0.15G
PP06C	Failure of the Victrolic coupled fire piping in the Turbine Building will cause an inability to supply water to containment through the Turbine Building. This piping failure cauaes a problem in a LOCA situation in as not enough time is available to isolate the turbine building fire water supply and circumvent the break to supply water to containment.
PP05T	Failure of the Threaded fire piping in the Turbine Building will cause an inability to supply water to containment through the Turbine Building. This piping failure cauaes a problem in a LOCA situation in as not enough time is available to isolate the turbine building fire water supply and circumvent the break to supply water to containment.
	WEAK-LINKS AT 0.16G
M10013	This block wall is located along the east wall of the stock room. Failure of this block wall may damage or fail the UPS batteries or the fire piping. This appears only in the LOCA event trees as there is

TABLE VIII-5 (Continued)

	not time to isolate a turbine building fire piping break during a LOCA. (See Figure V-1 for location of this block wall).
M100UPS	M100UPS is a combination of block walls, the most limiting of which is M10013. It's effects due to failure, is explained above. All of the other walls that makeup M100UPS have ground accelerations greater than 0.3g.

WEAK-LINKS AT 0.20G

PPYARD	Failure of the fire piping in the yard can inhibit water supply to the turbine building or the containment for core spray, enclosure spray or use for emergency condenser makeup. No credit is taken for bypassing the yard loop via the LOCA hose in the case of a LOCA in as there is not enough time available.
PPSWCW	Loss of the circulationg water flow due to the expansion joint rupturing causes loss of the main condenser heat removal and loss of the fire pumps due to flooding. This catastrophic loss of fire water leads to core damage in any event tree.
ATWS	Shearing of the alignment pins in the core support plate causes lateral movement of the support plate, preventing rod insertion.

TABLE VIII-6

OPTIONAL FIXES

WEAK-LINKS AT 0.11G

M10021 1. Structurally reinforce the block wall.

2. Reassess the seismic fragility of the block walls using nonlinear dynamic analysis. (Computech Engineering Services proposal, February 10,1986).

OR 3. Relocate the affected equipment and cables to a more suitable location.

WEAK-LINKS AT 0.12G

ECSHELL 1. Improve emergency condenser supports possibly by attaching the other 3 gussets to the floor pads.

M10018 1. Secure the wall to the building structure.

2. Structurally reinforce the wall.

3. Reassess the seismic fragility of the block wall using nonlinear dynamic analysis. (Computech Engineering Services proposal, February 10, 1986).

OR 4. Relocate the affected equipment and cables to a more suitable location.

WEAK-LINKS AT 0.13G

M10020 1. Secure wall to building structure.

2. Structurally reinforce the block wall).

3. Reassess the seismic fragility of the block walls using nonlinear dynamic analysis. (Computech Engineering Services proposal, February 10, 1986).

OR 4. Relocate the affected equipment and cables to a more suitable location.

TABLE VIII-6 (Continued)

VPI301	1. Analyze valves (VPI-301 and 302).
UPS2ND	1. Place cage like structure over batteries to prevent damage.
	2. Place cage like structure over vent ducts and lights to prevent damage to the batteries.

WEAK-LINKS AT 0.15G

PP06C & PP05T	1. Replace threaded joints and victrolic couplings with welded joints.
	2. Current analysis techniques has shown that piping is much more resilient than previously thought. Reanalyze.

WEAK-LINKS AT 0.16G

M10013	1. Secure wall to the building structure.
	2. Structurally reinforce the block wall.
	3. Reassess the seismic fragility of the block walls using nonlinear dynamic analysis. (Computech Engineering Services proposal, February 10, 1986).
	4. Relocate the affected equipment and cable to a more suitable locations.
M100UPS	Fixing M10013 will take care of M100UPS.

WEAK-LINKS AT 0.20G

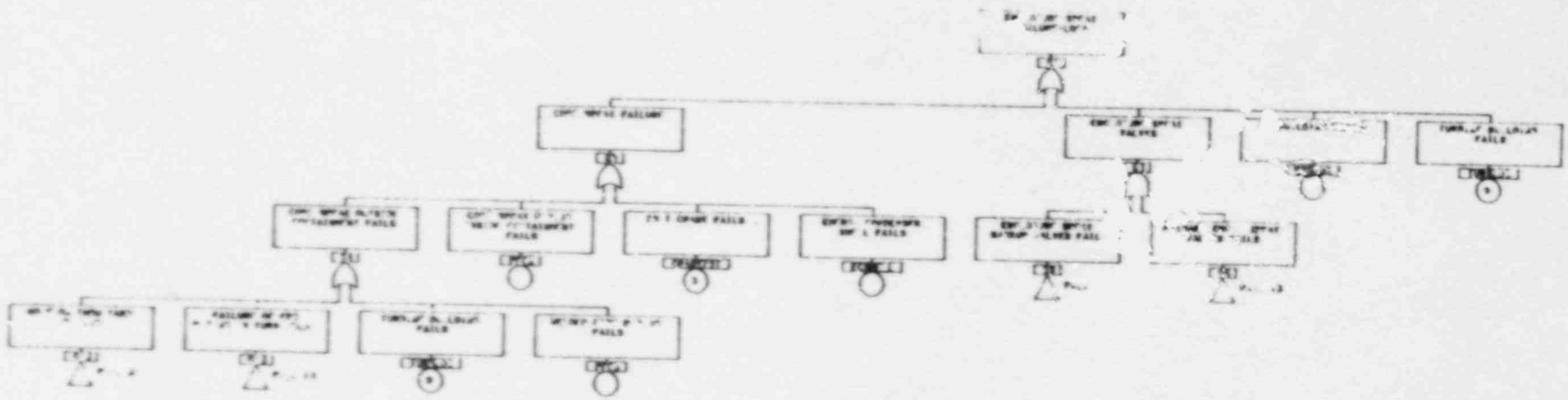
PPYARD	1. Replace yard piping with non-cast iron piping.
	2. Install a redundant fire system for the yard pipe section.
PPSWCW	1. Provide analysis.
ATWS	1. Install larger diameter core support plate alignment pins and bolts.

APPENDIX A

FAULT TREES

ENCLOSURE SPRAY FAILURE - LOCA
FAULT TREES

17 Pages



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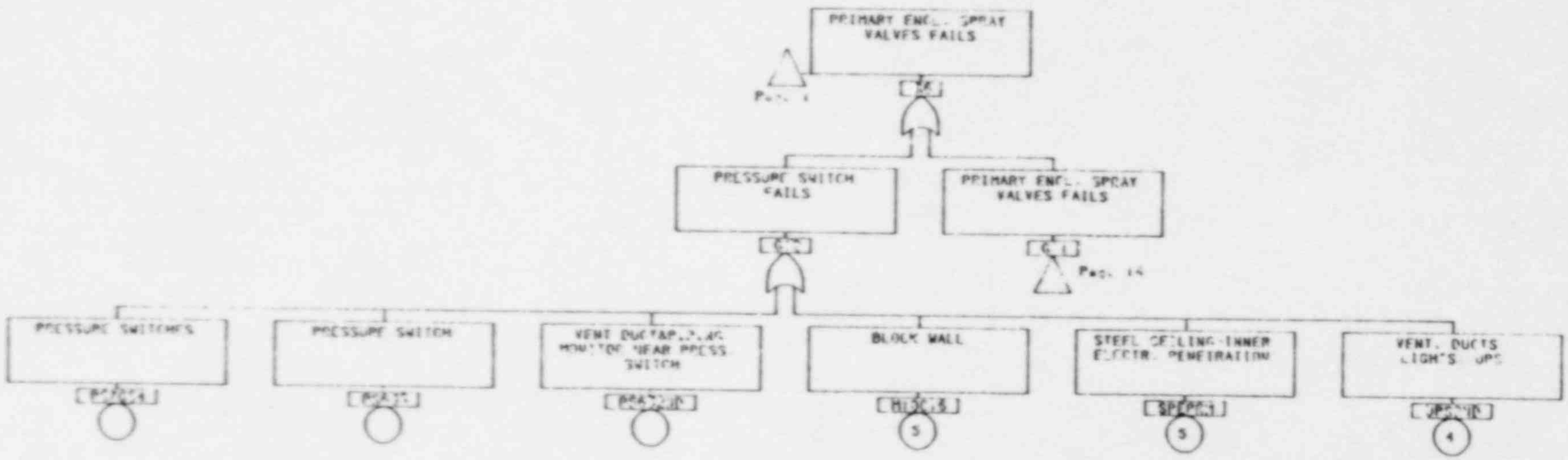

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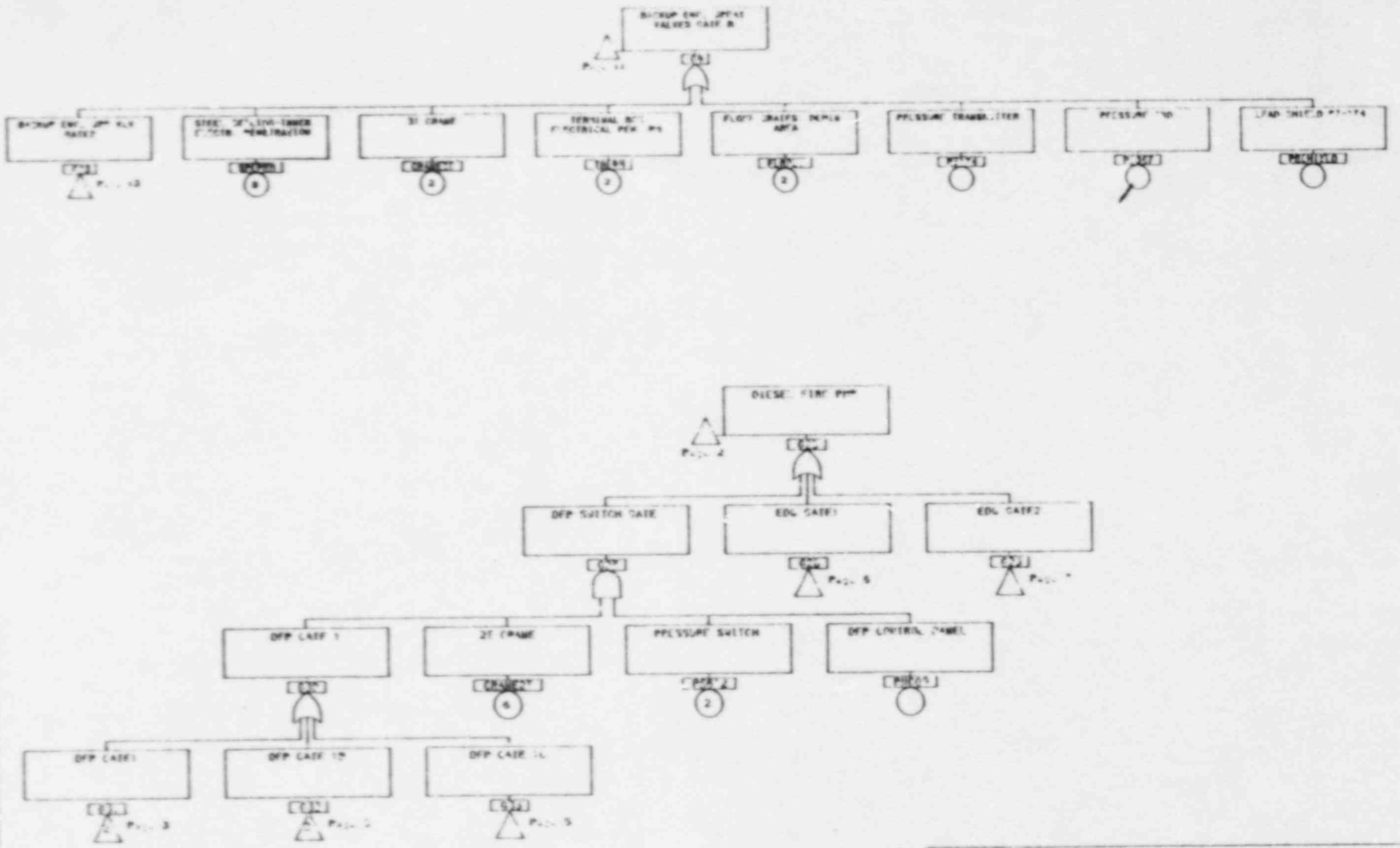

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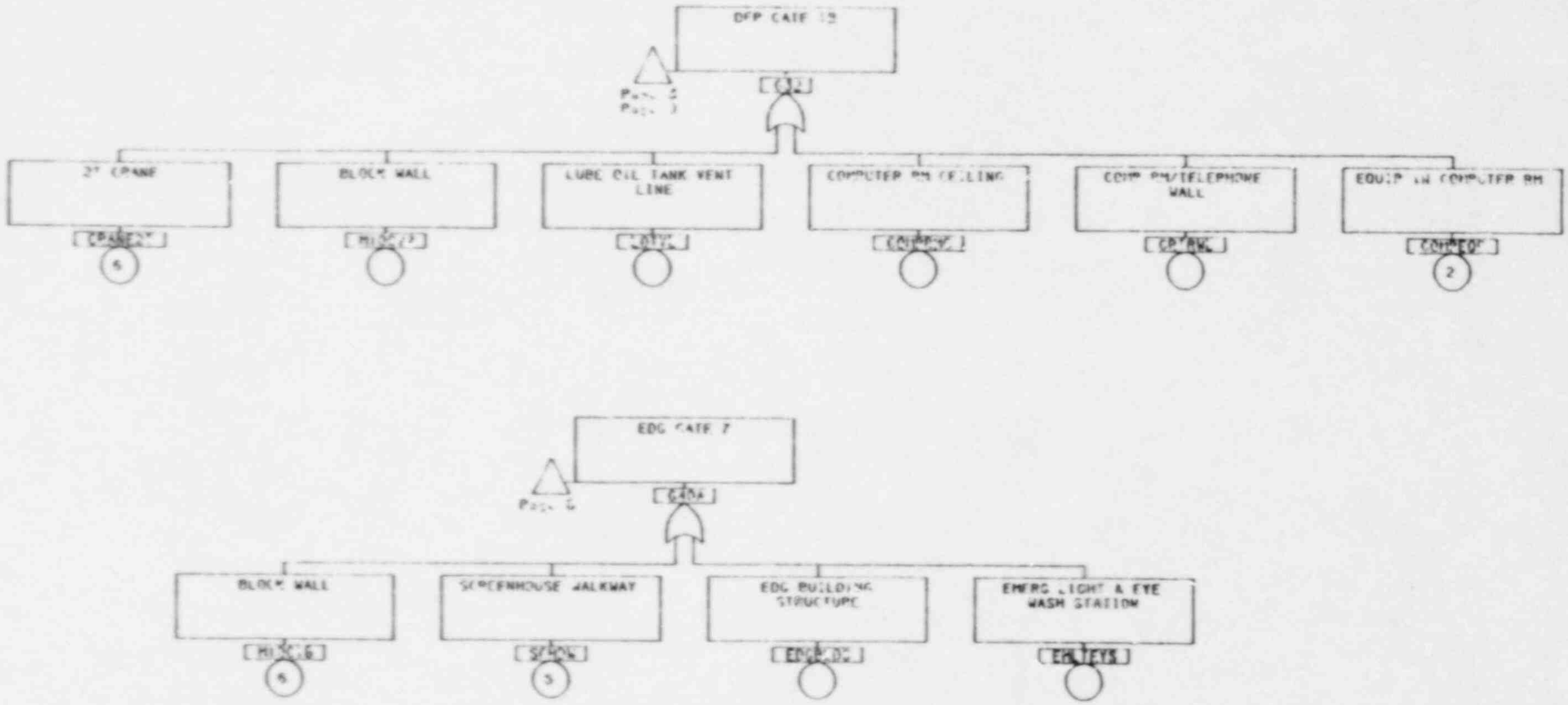

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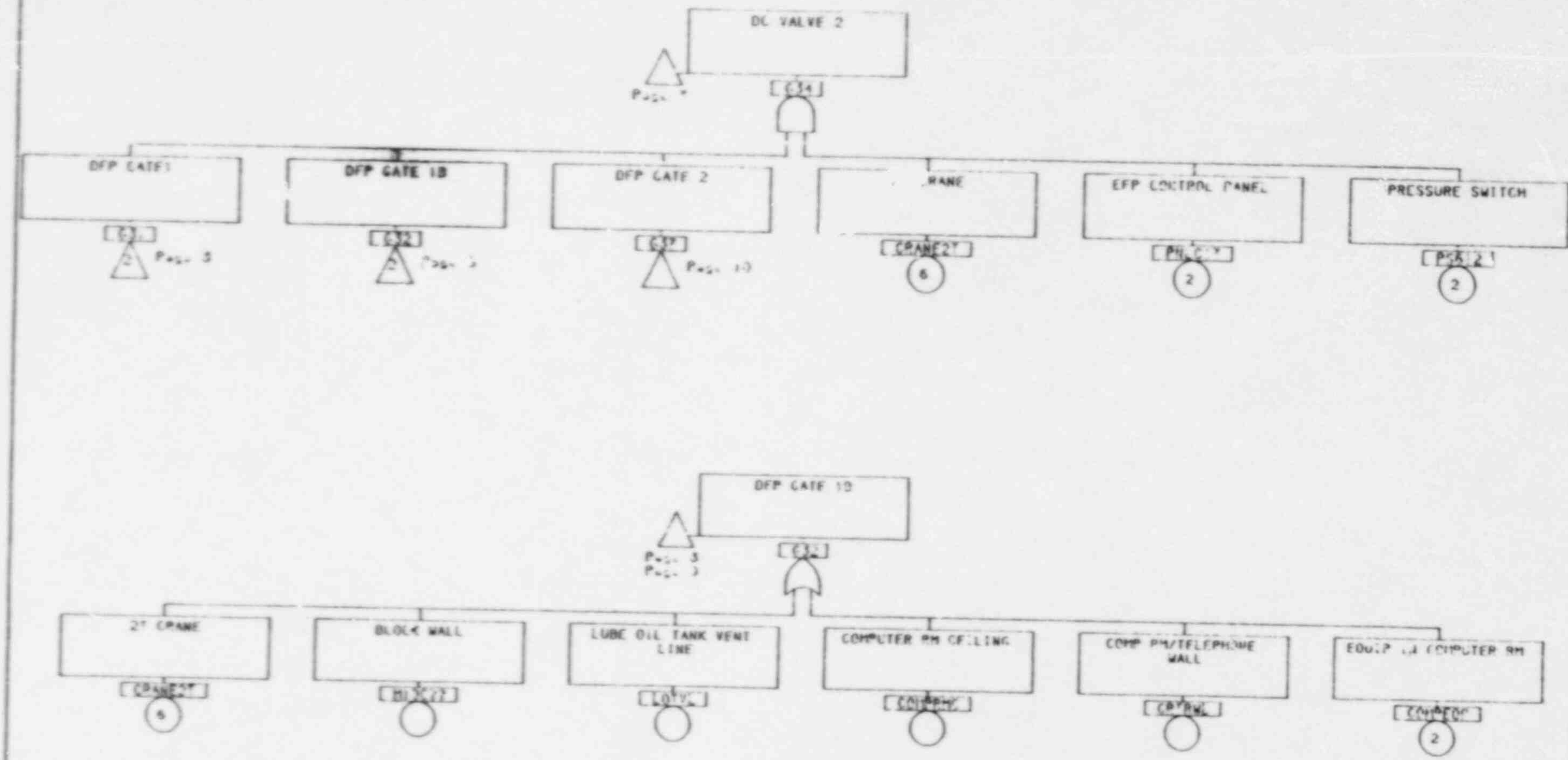

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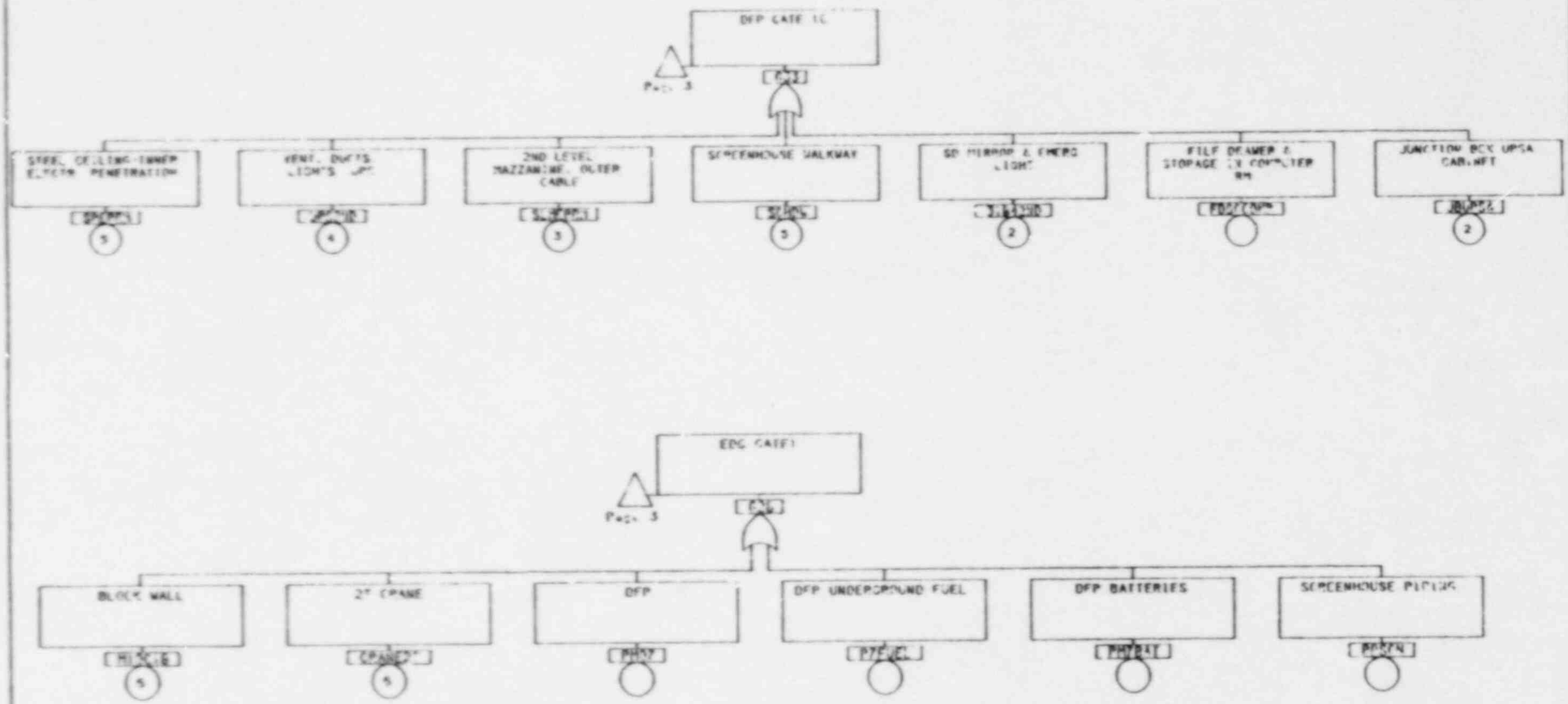

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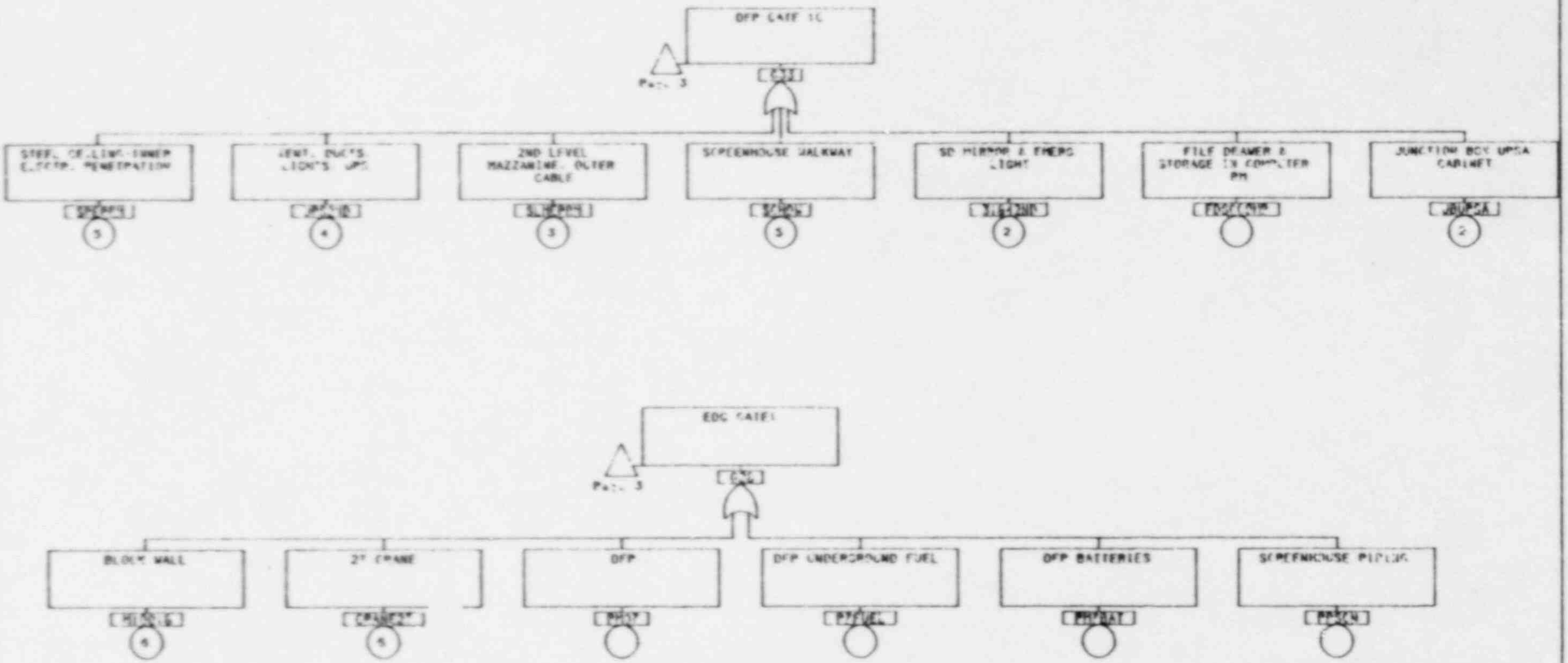

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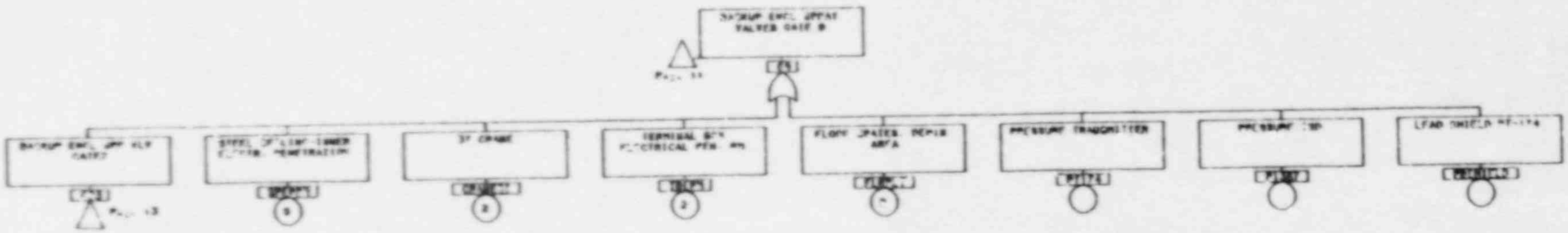

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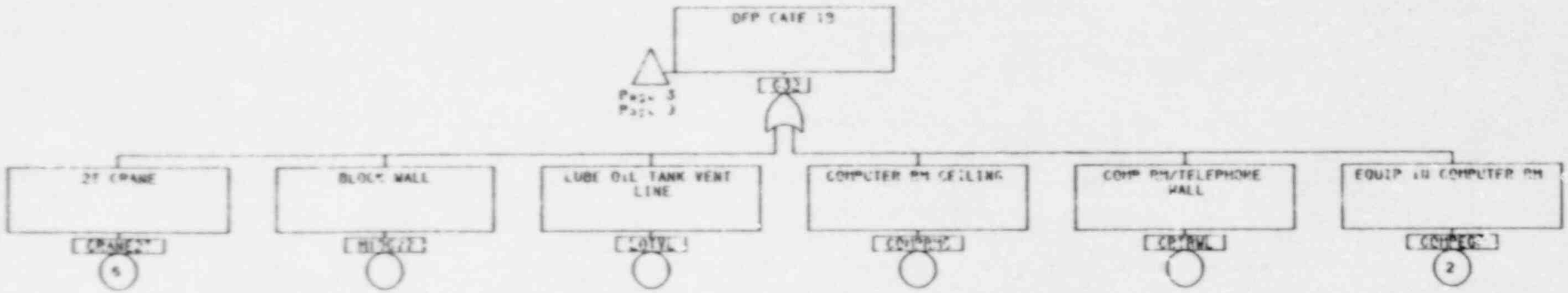

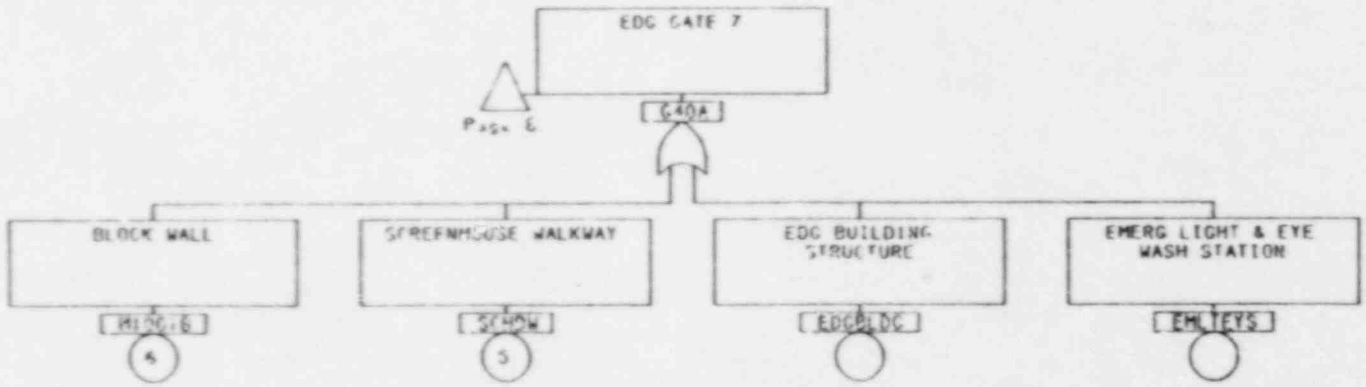

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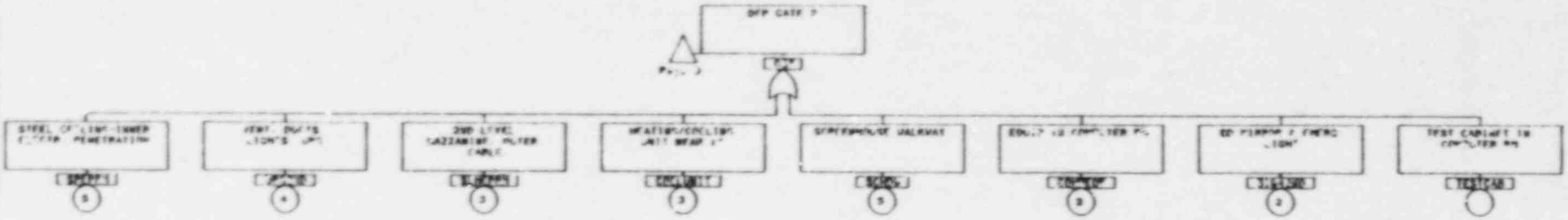

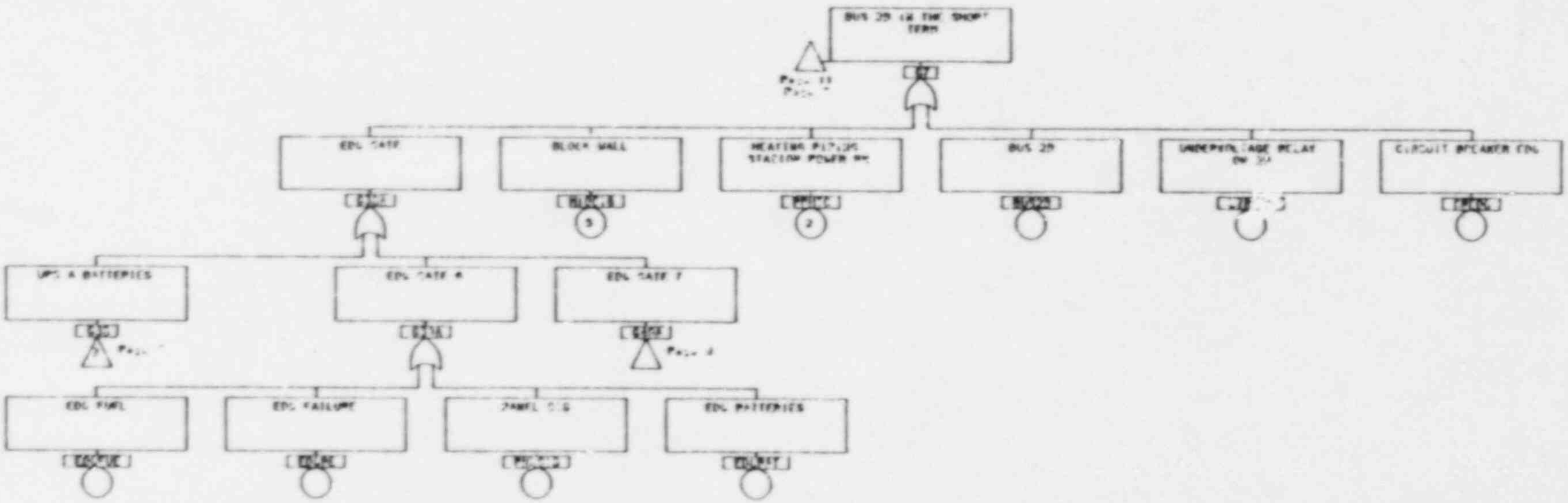

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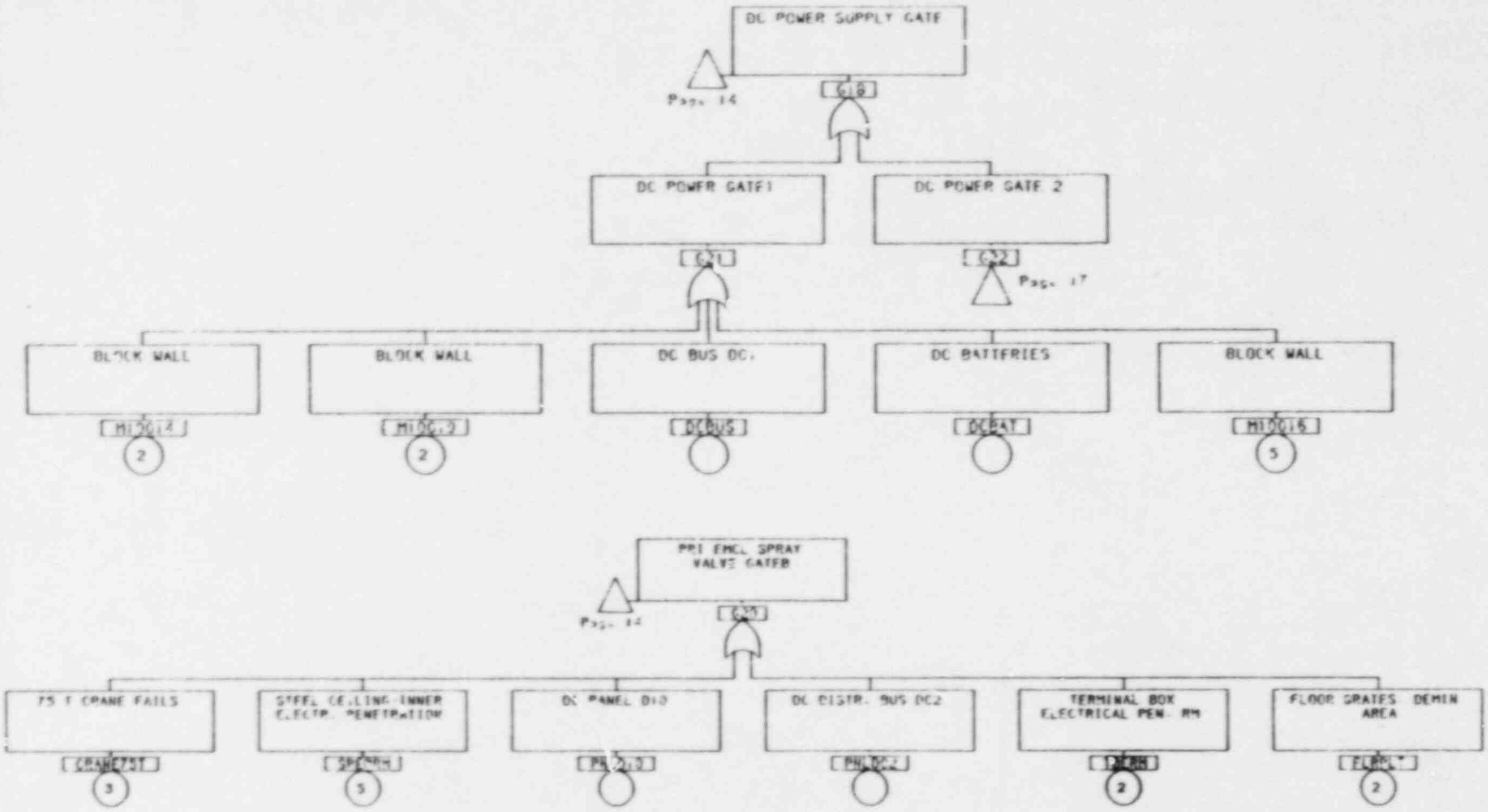

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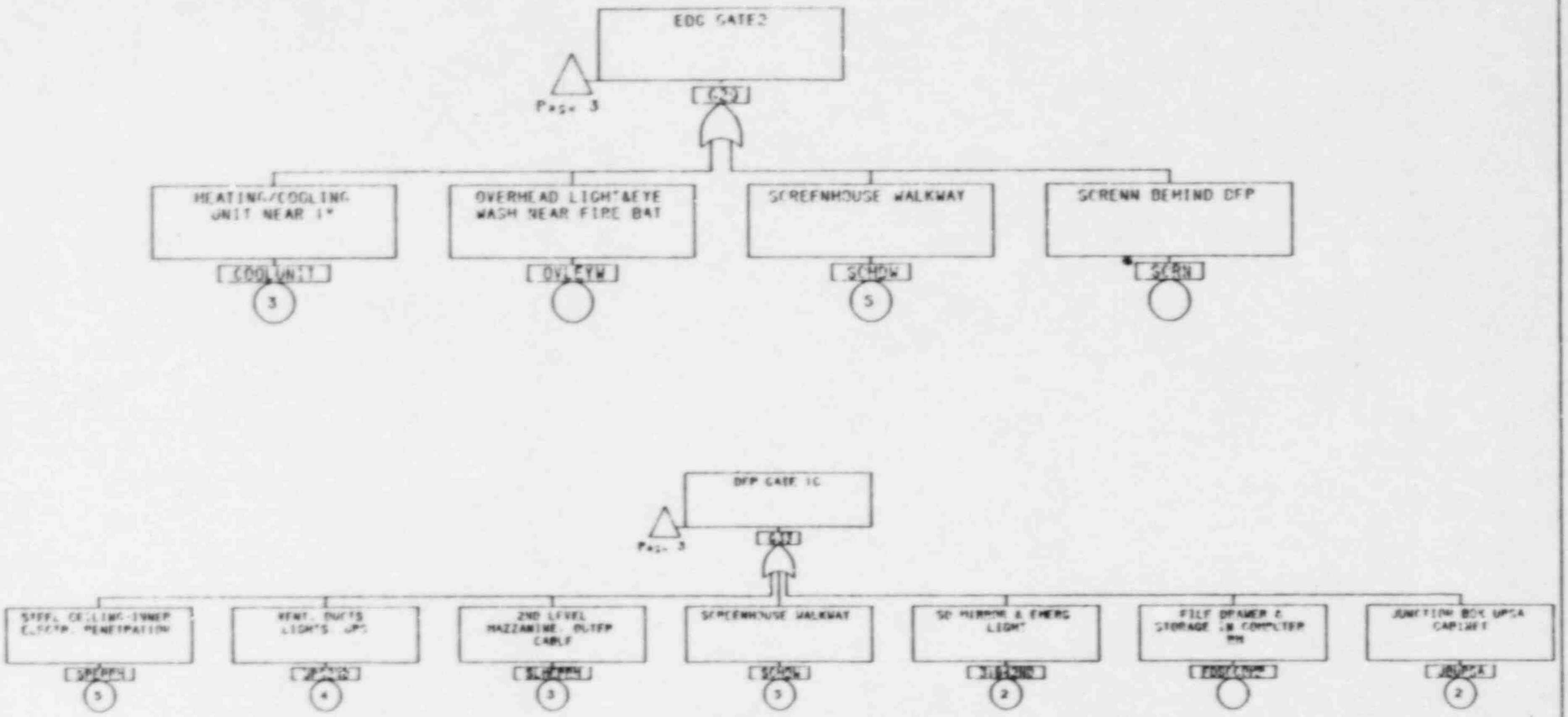

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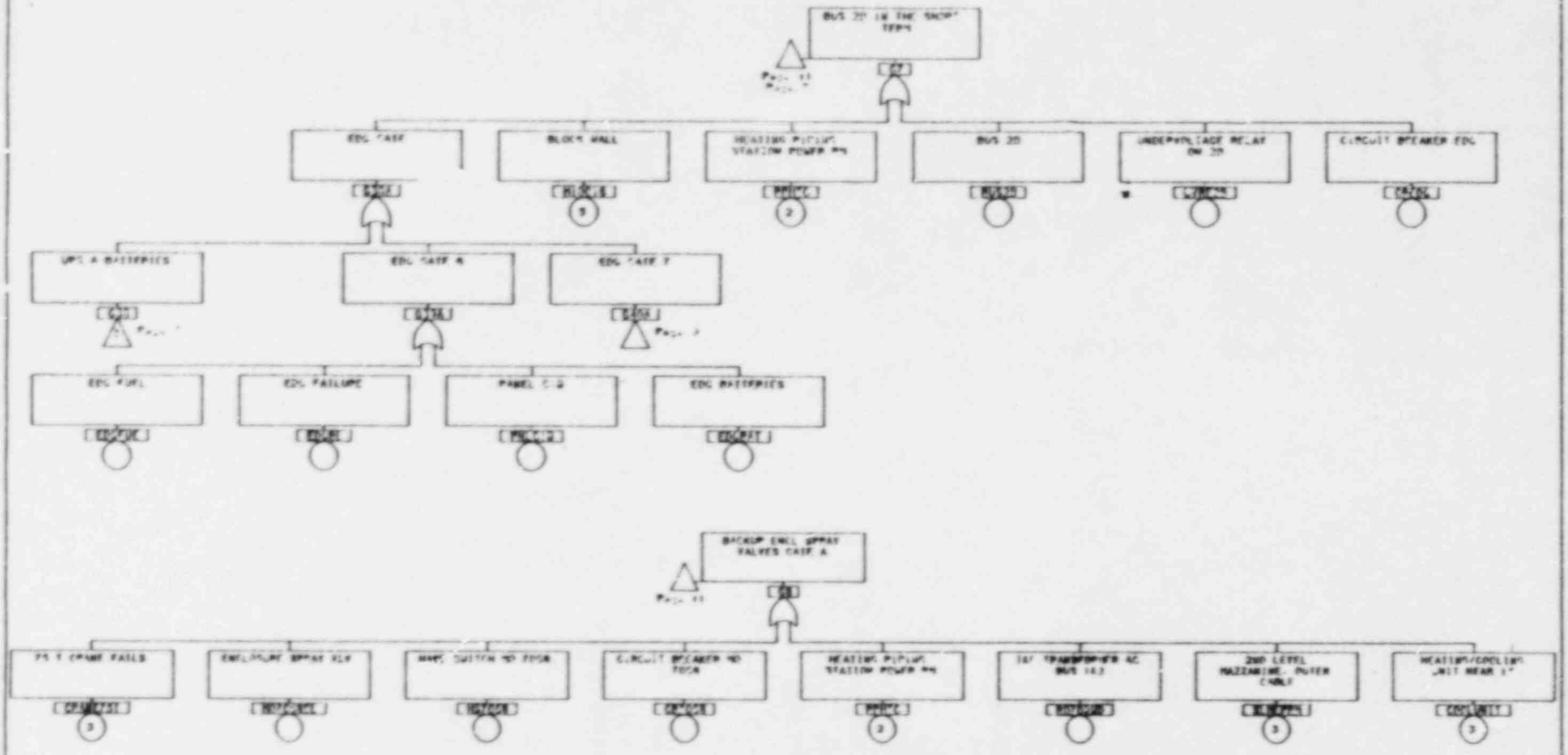

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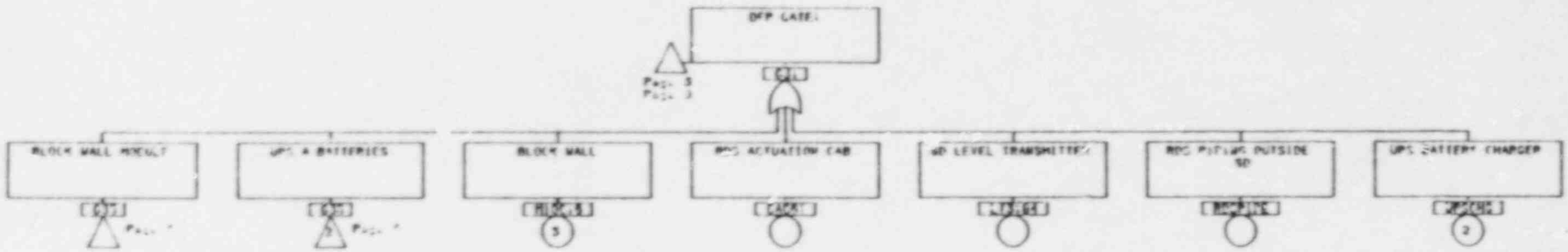

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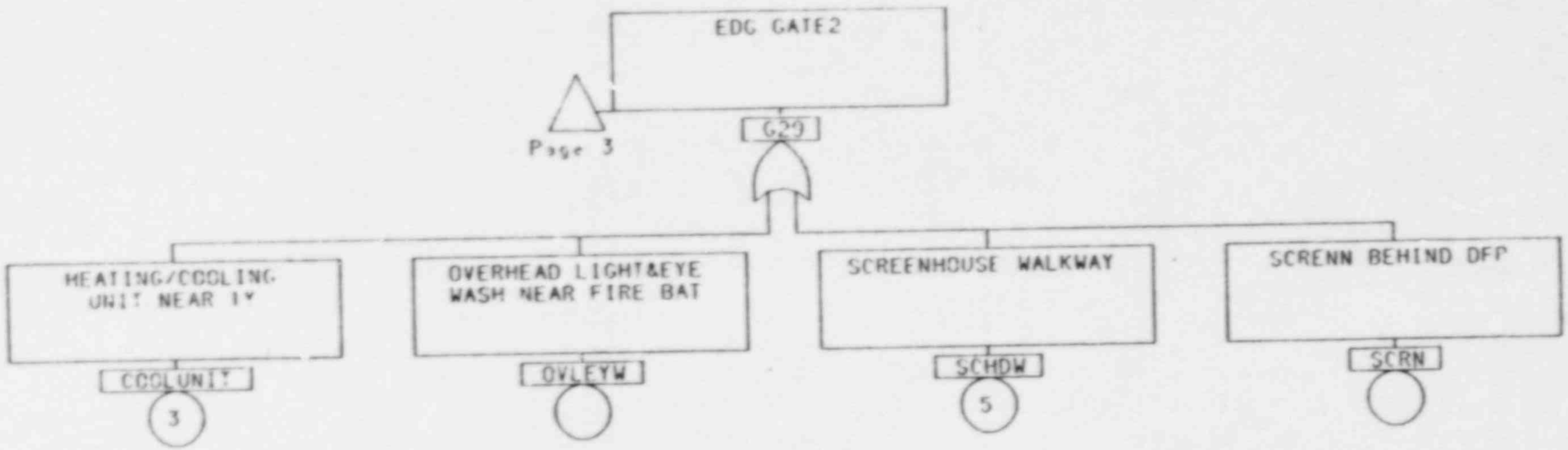

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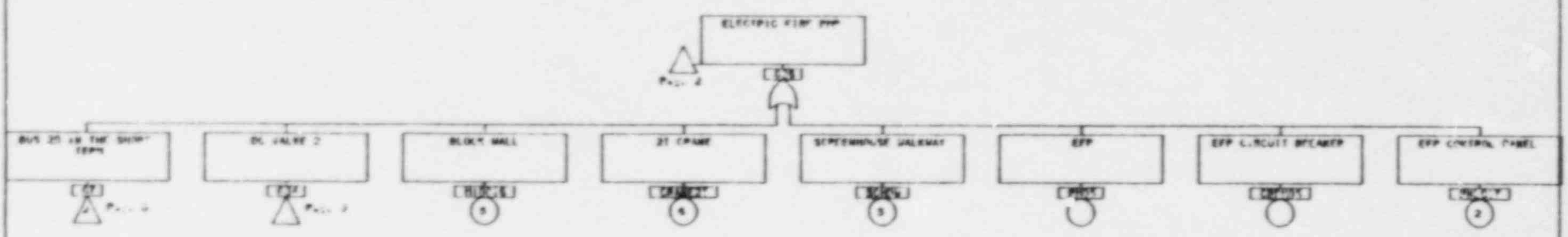

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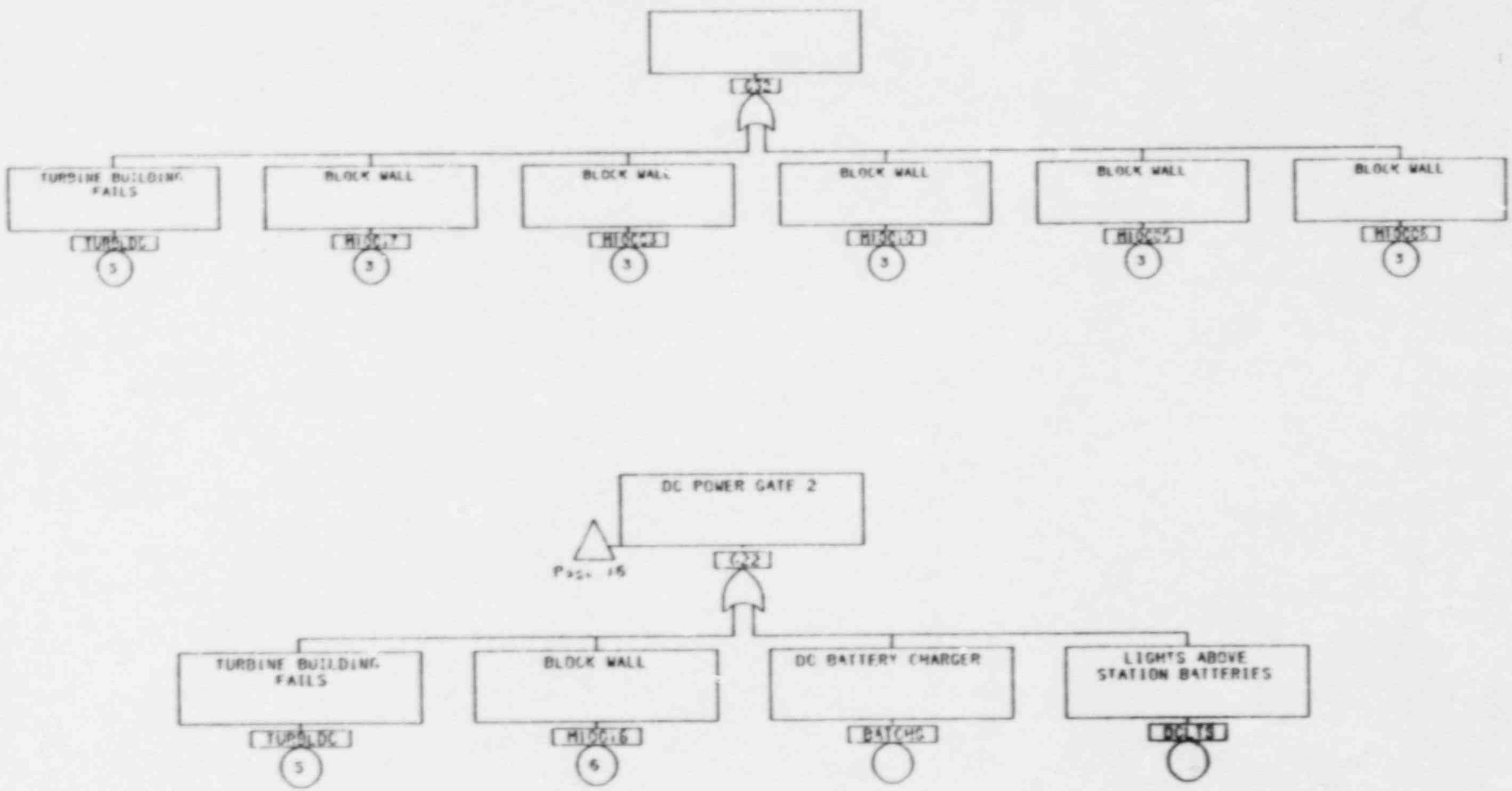

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FAULT TREES

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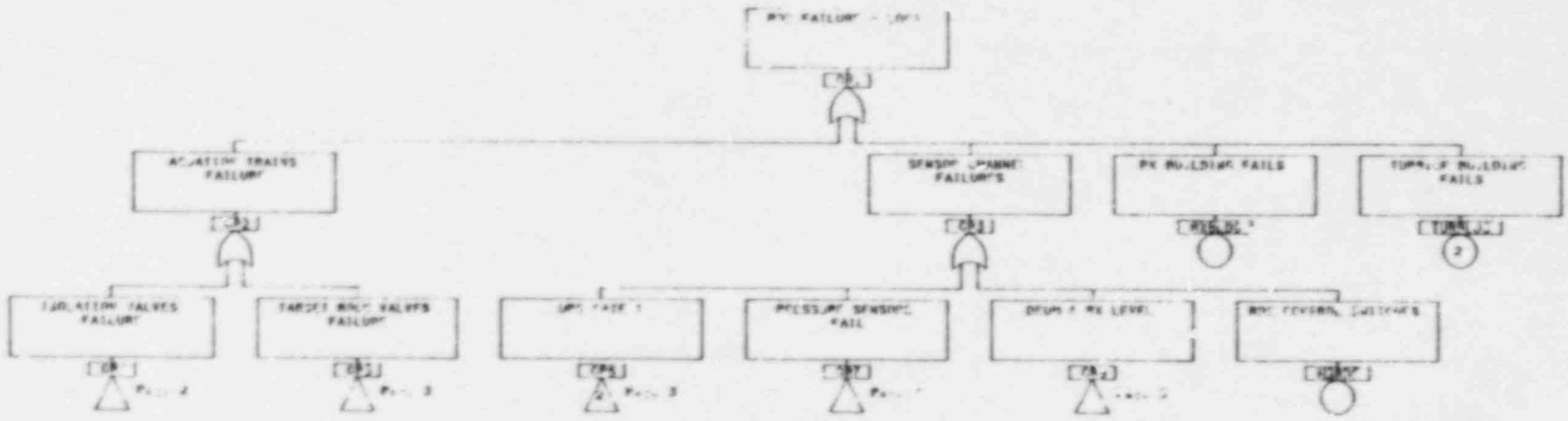

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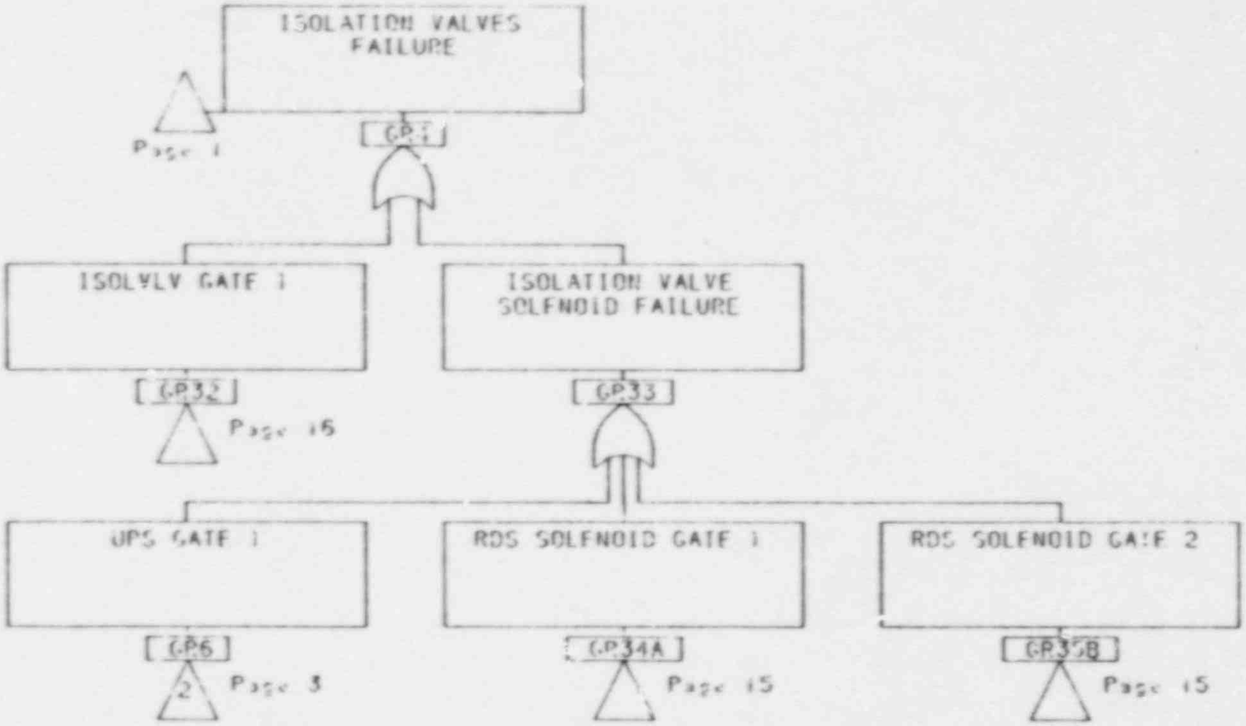

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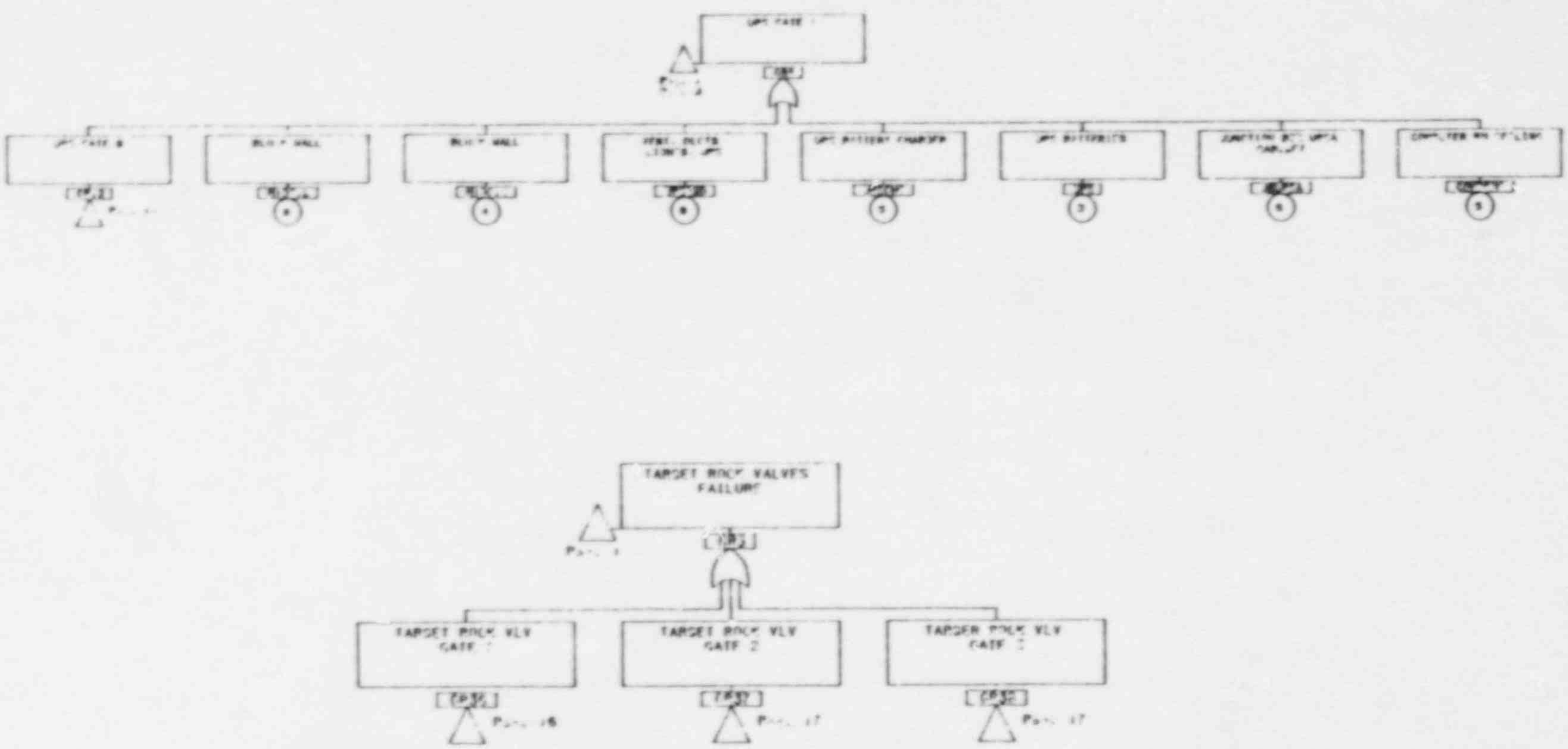

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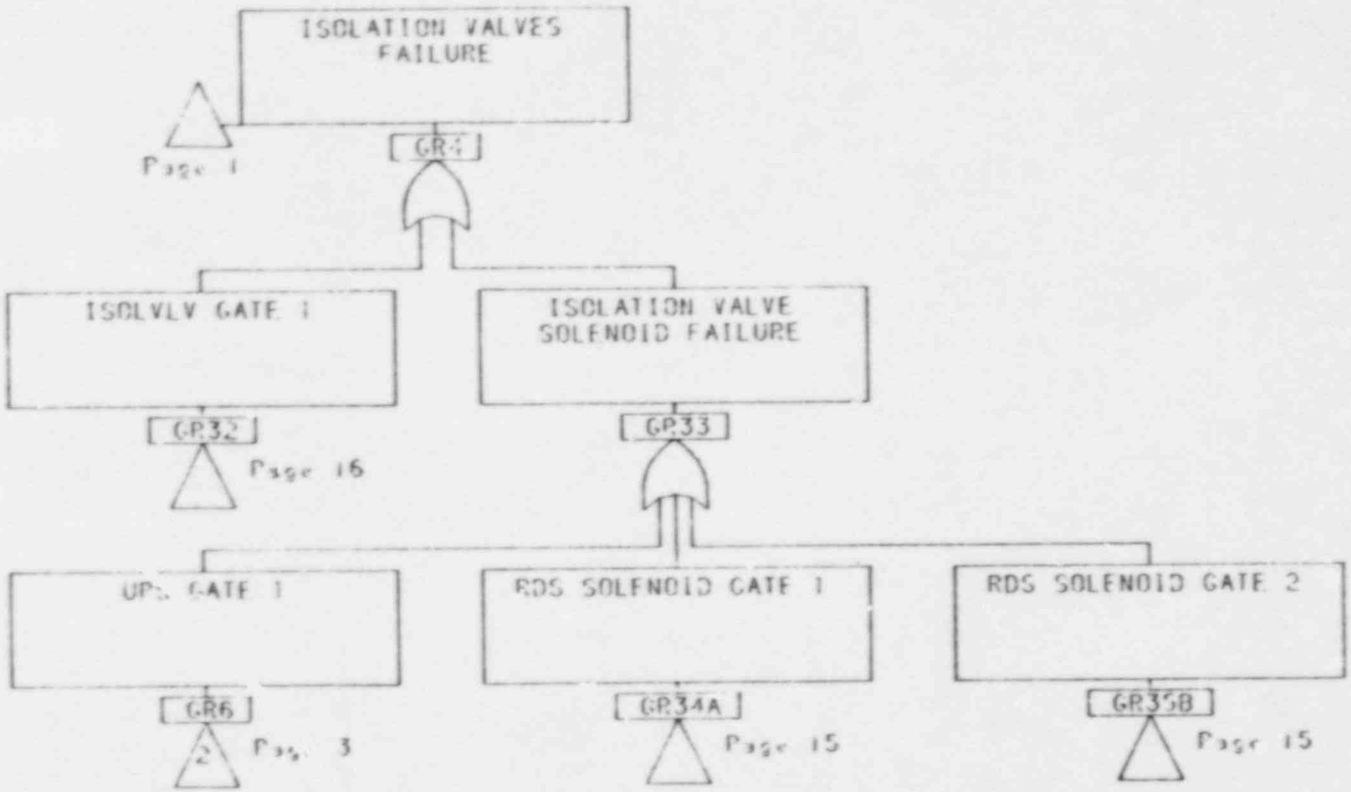

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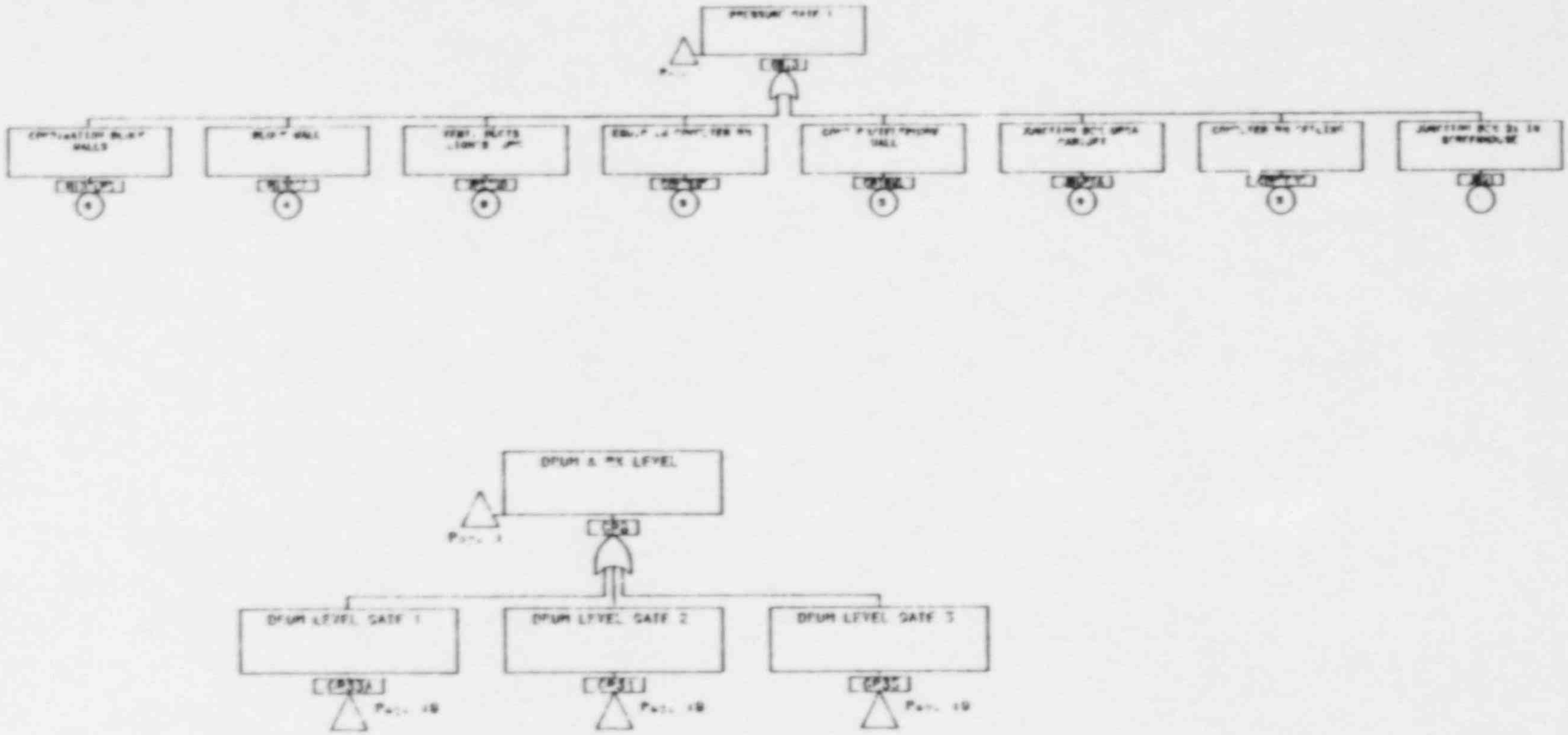

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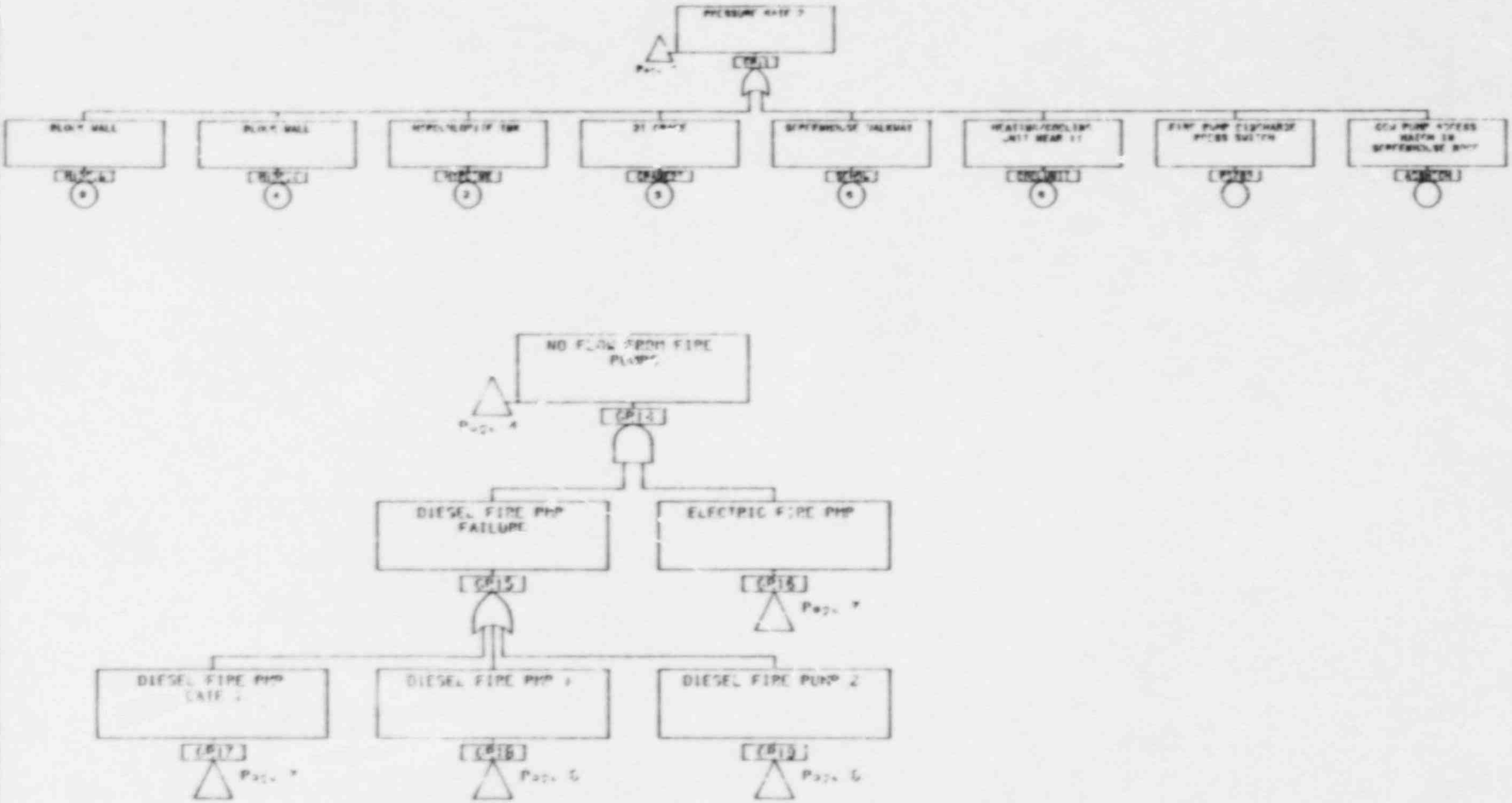

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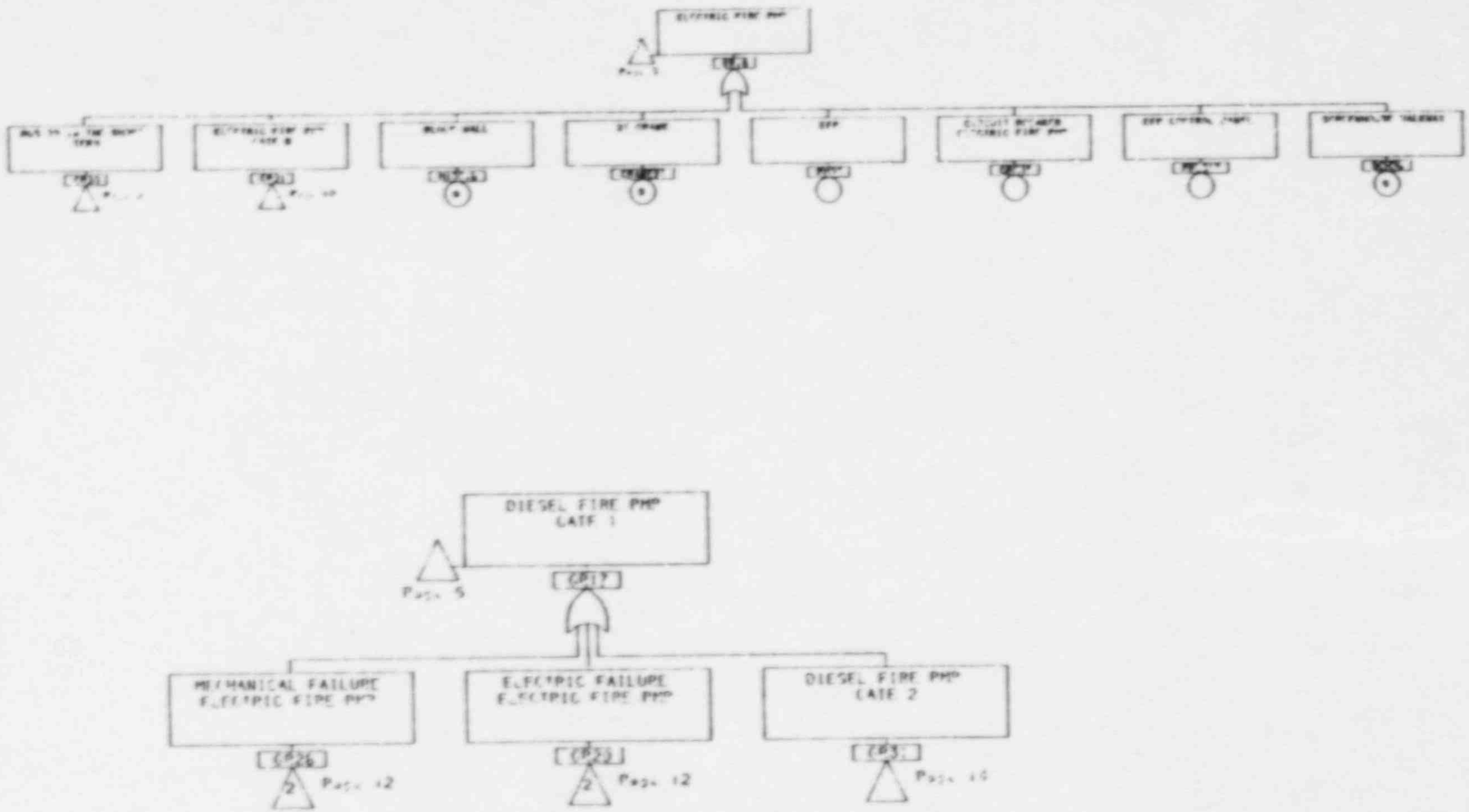

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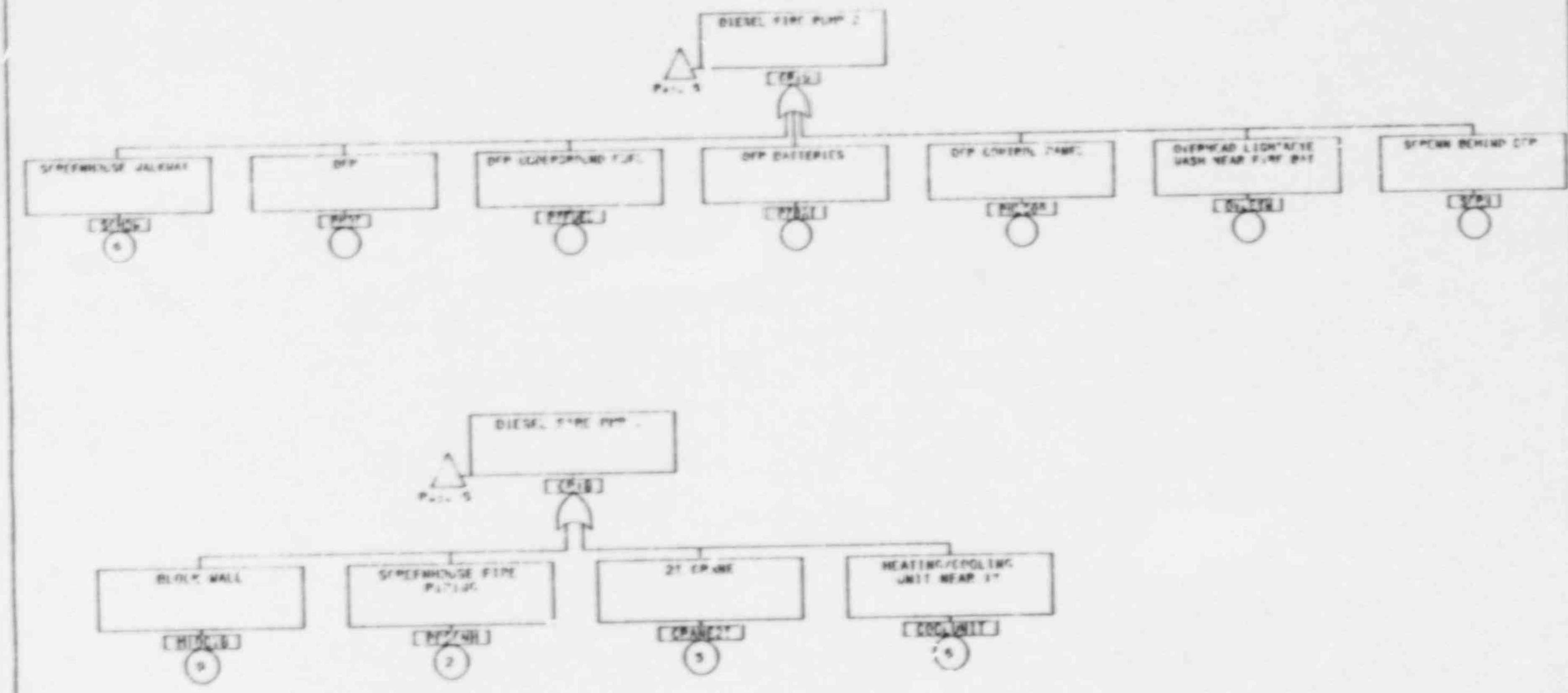

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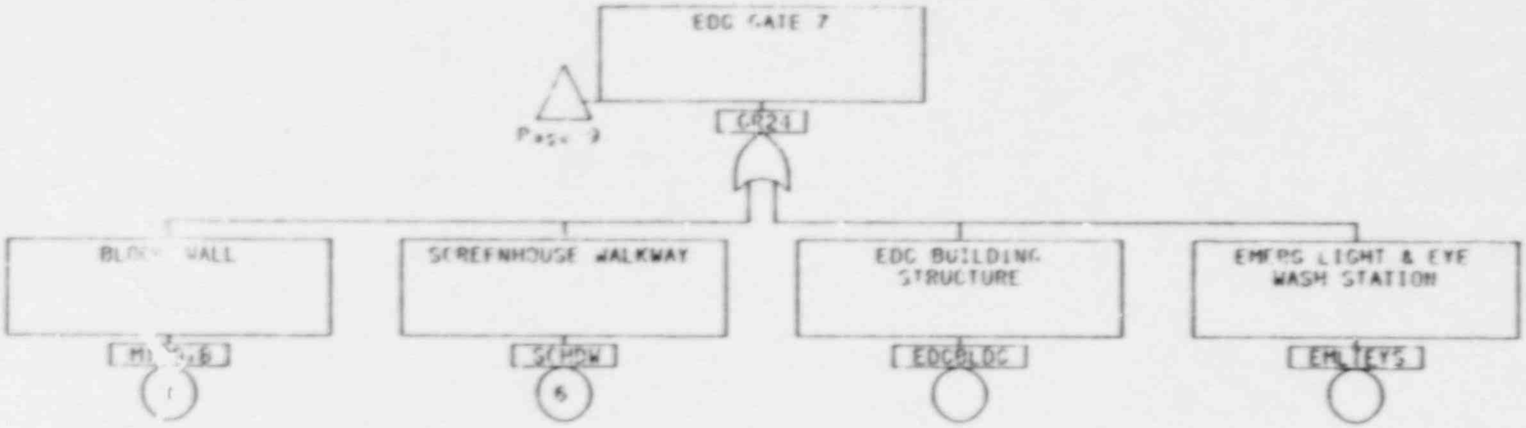

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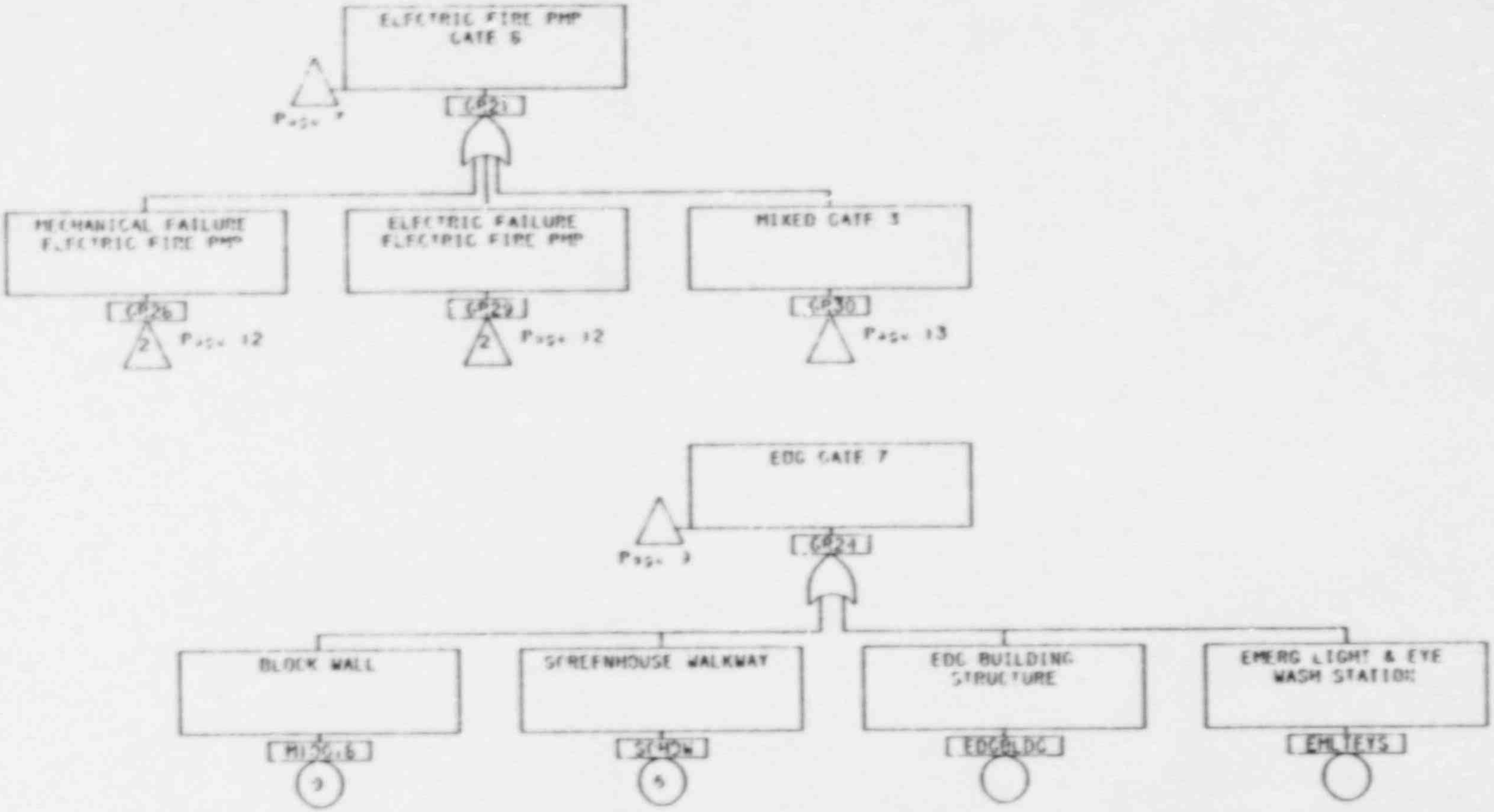

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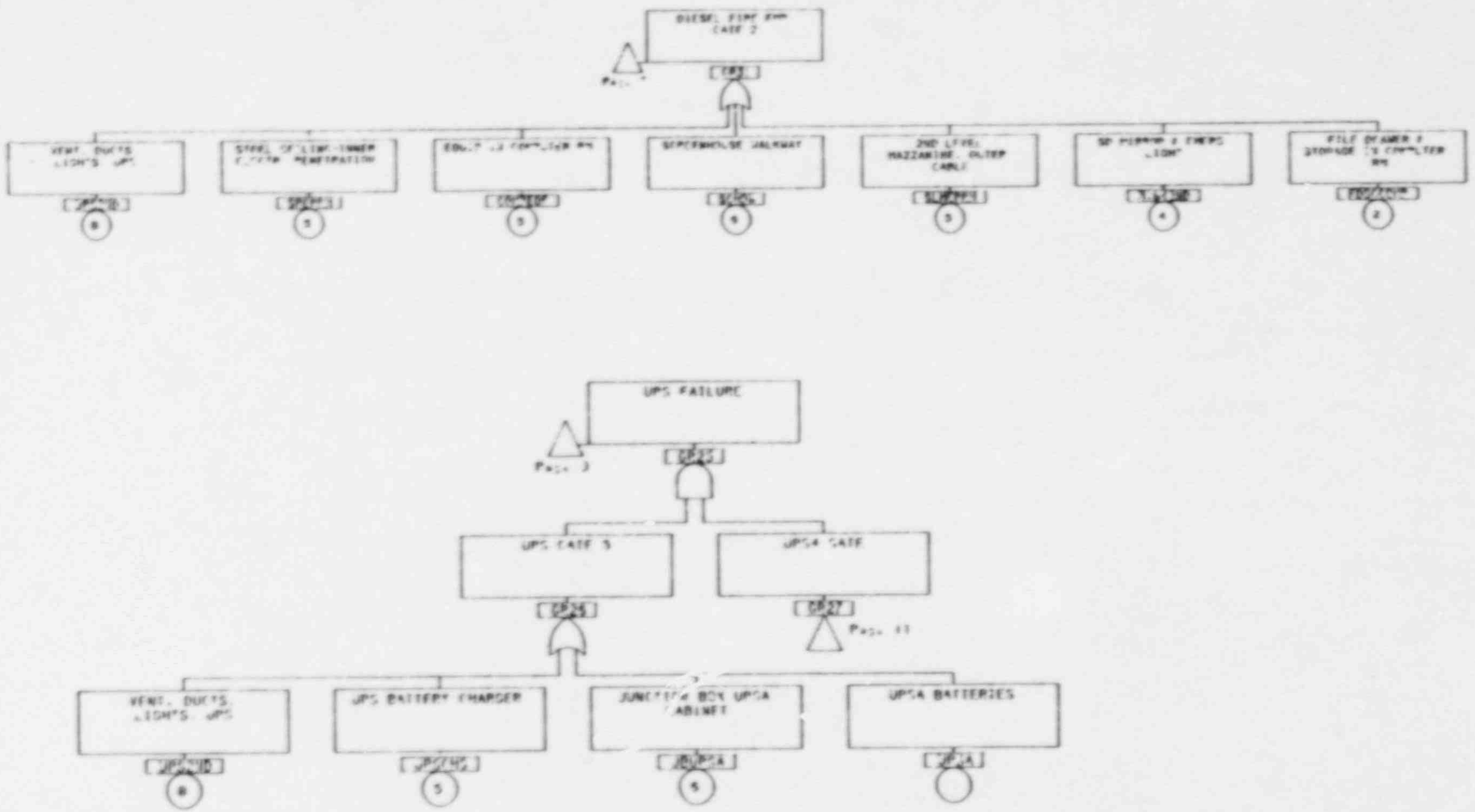

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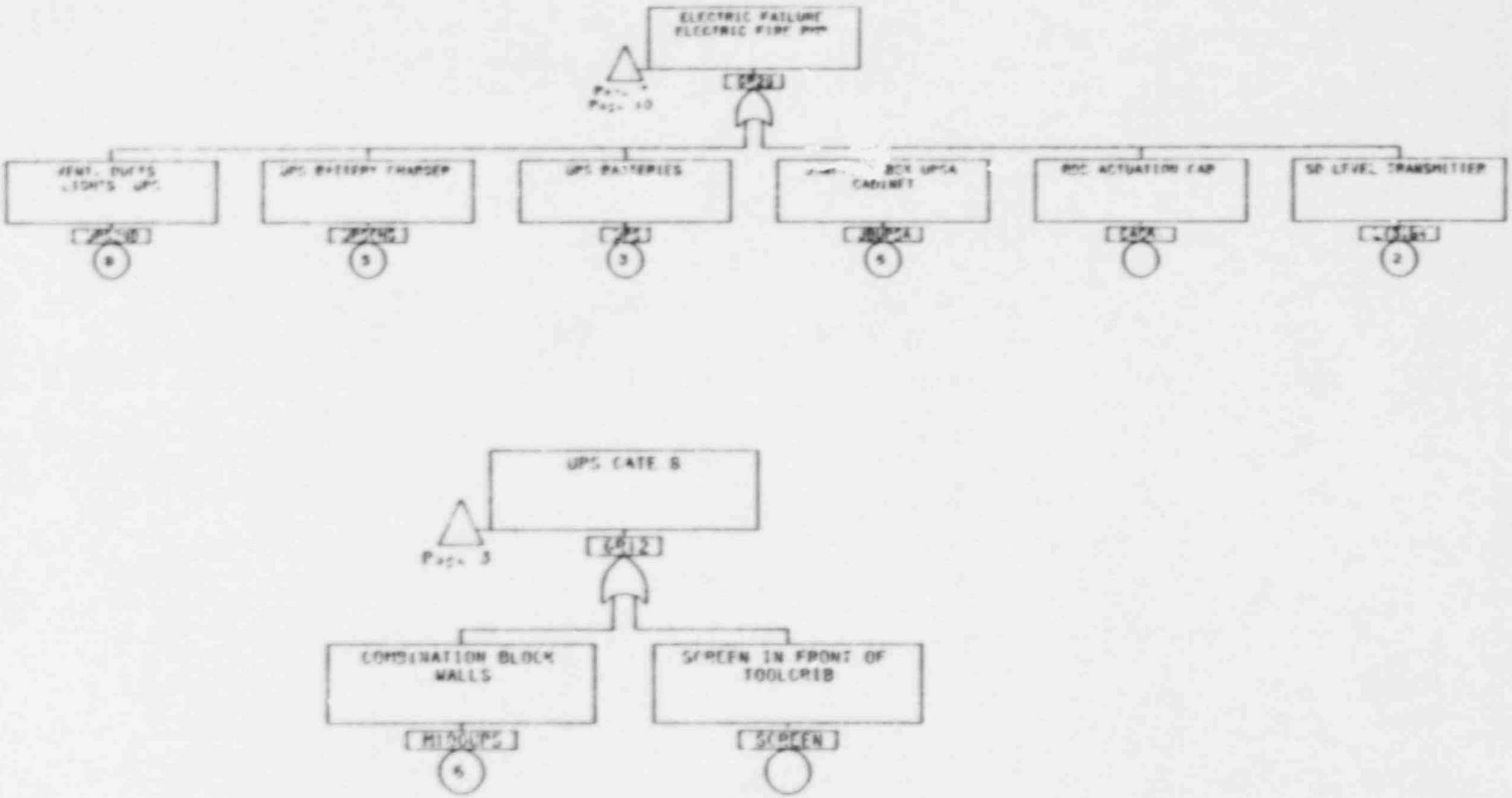

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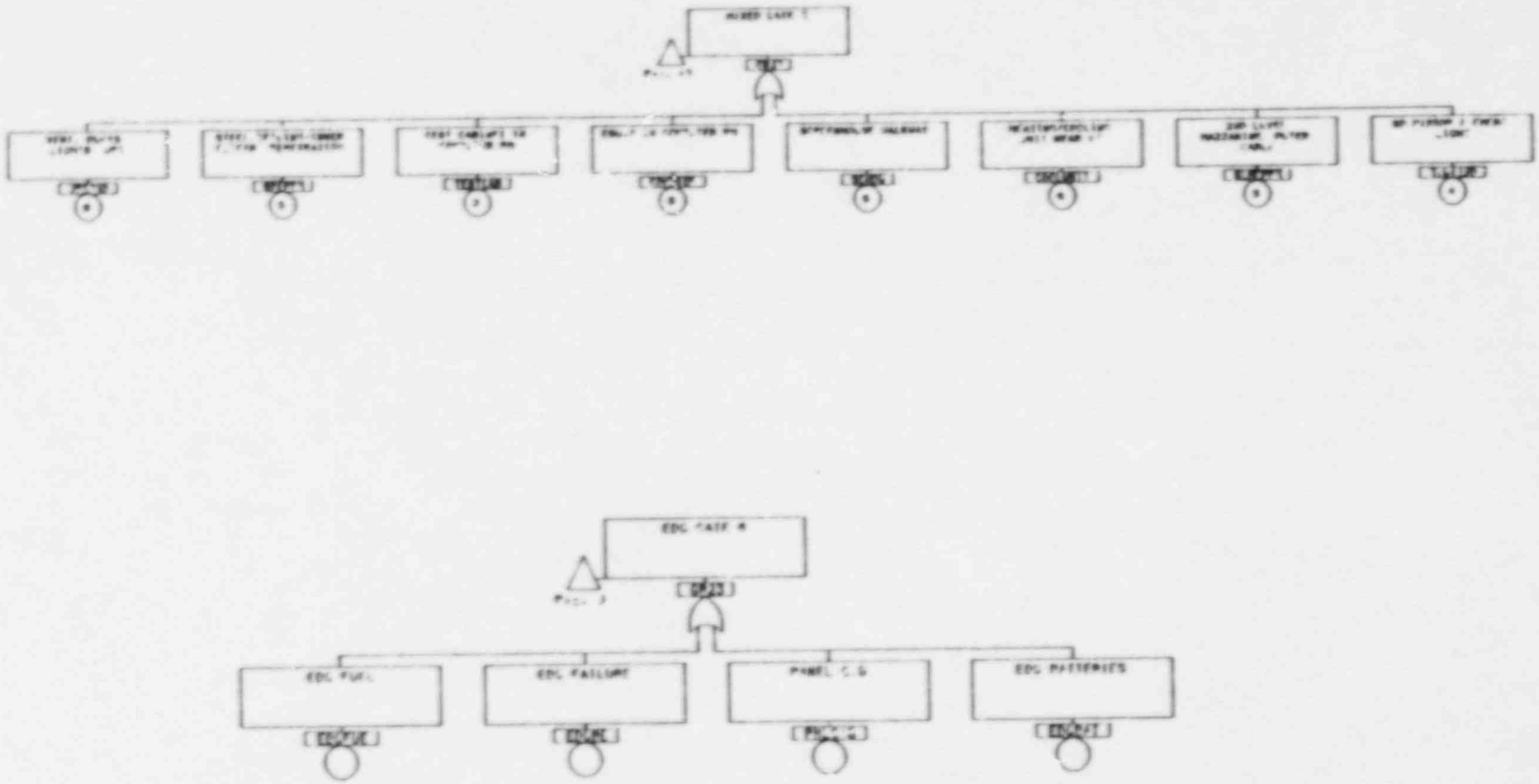

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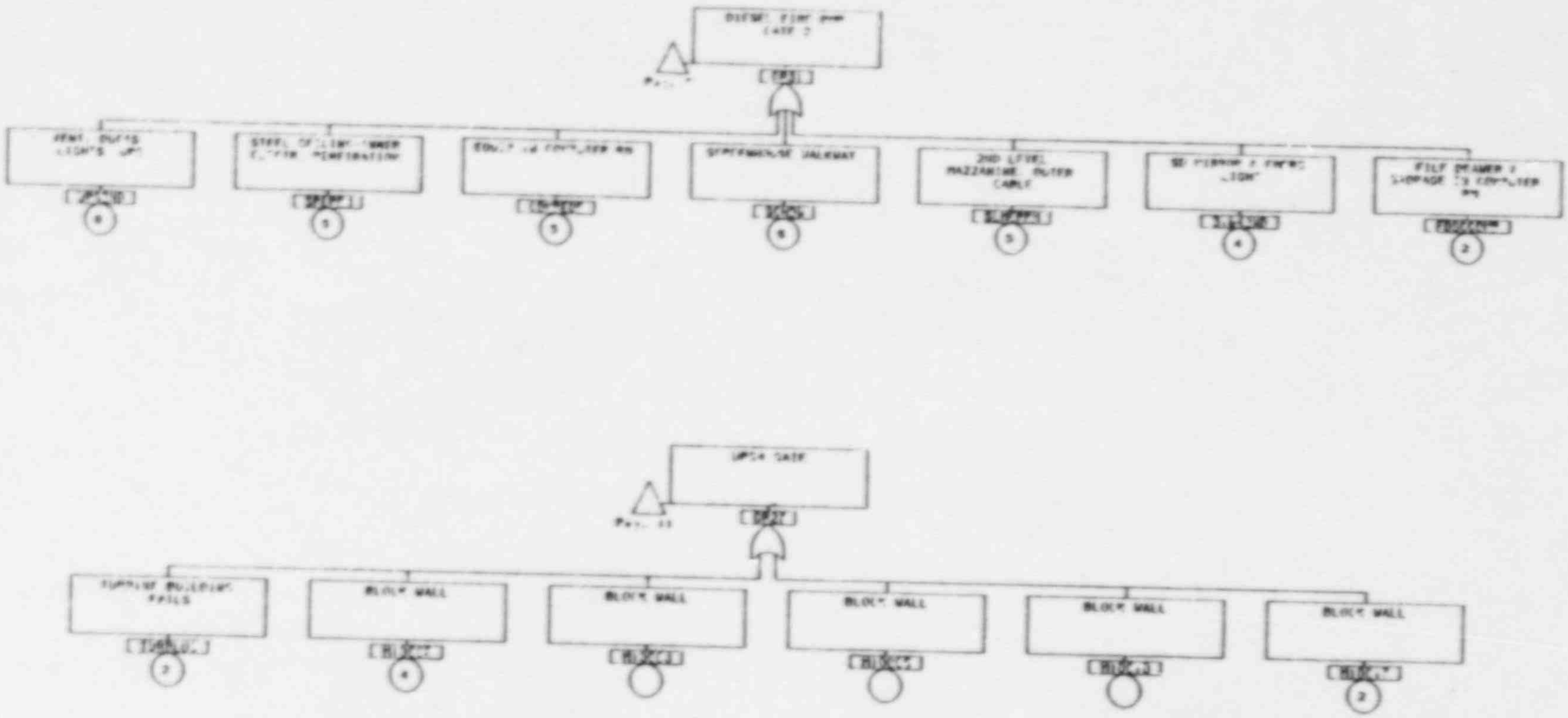

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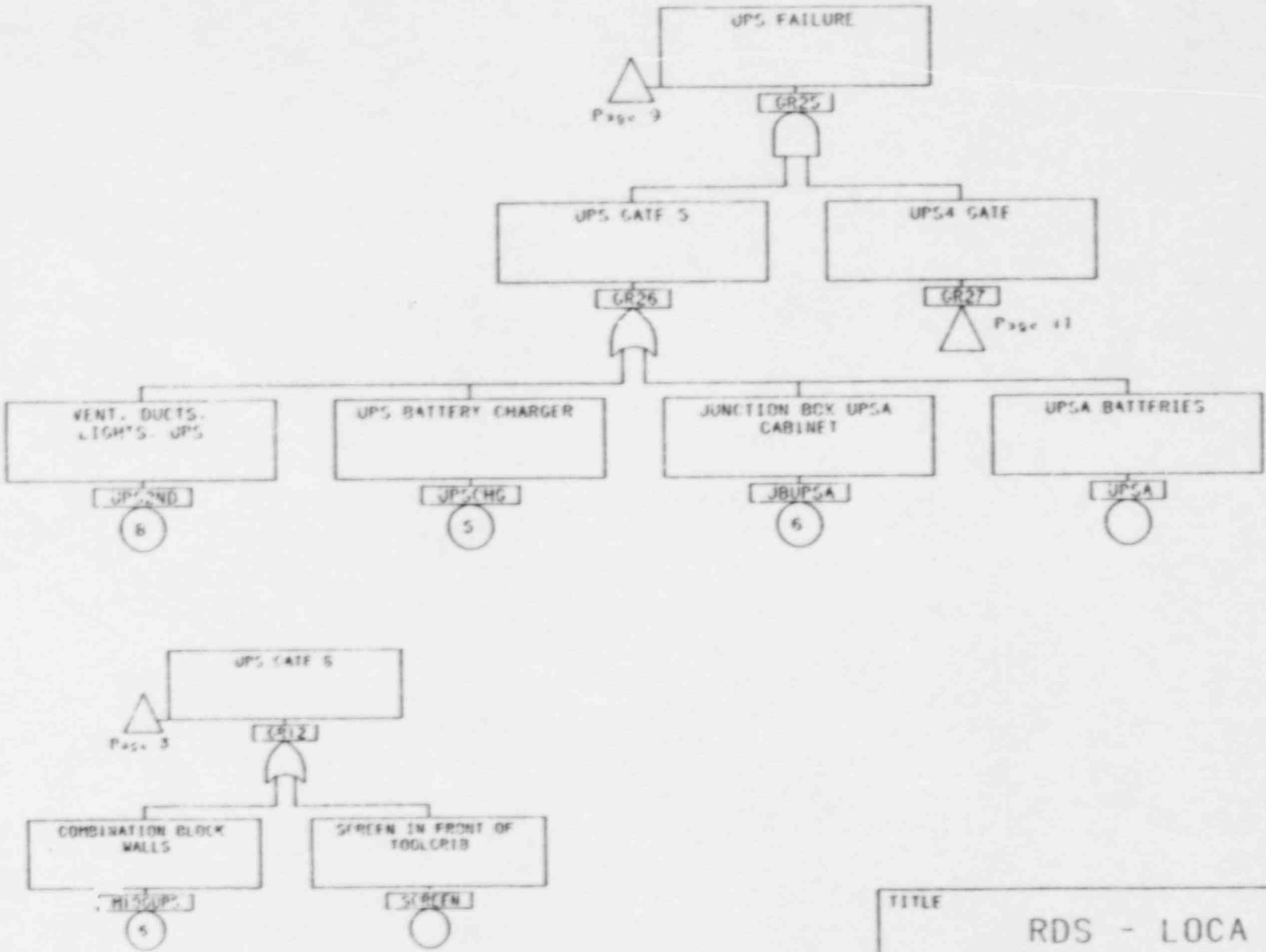

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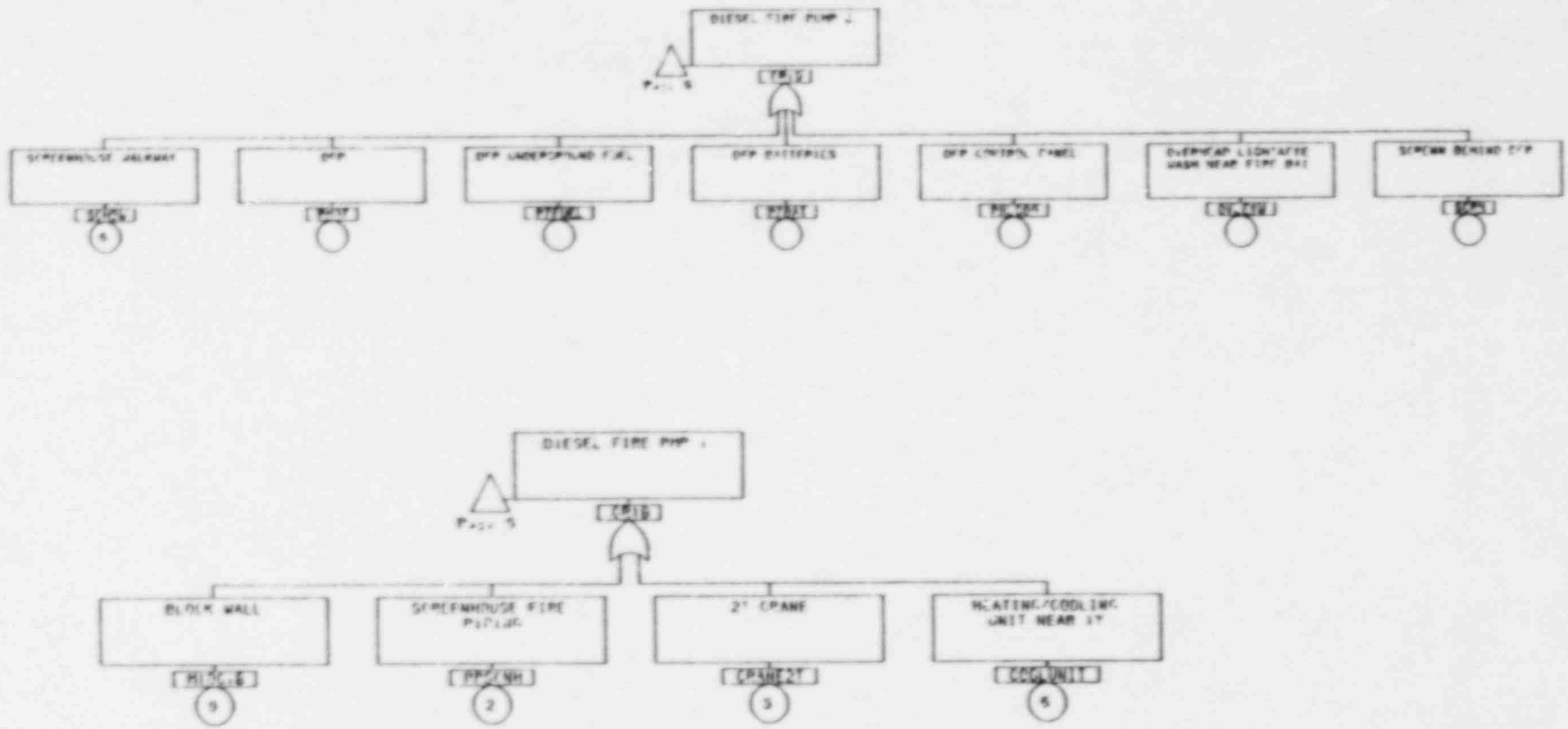

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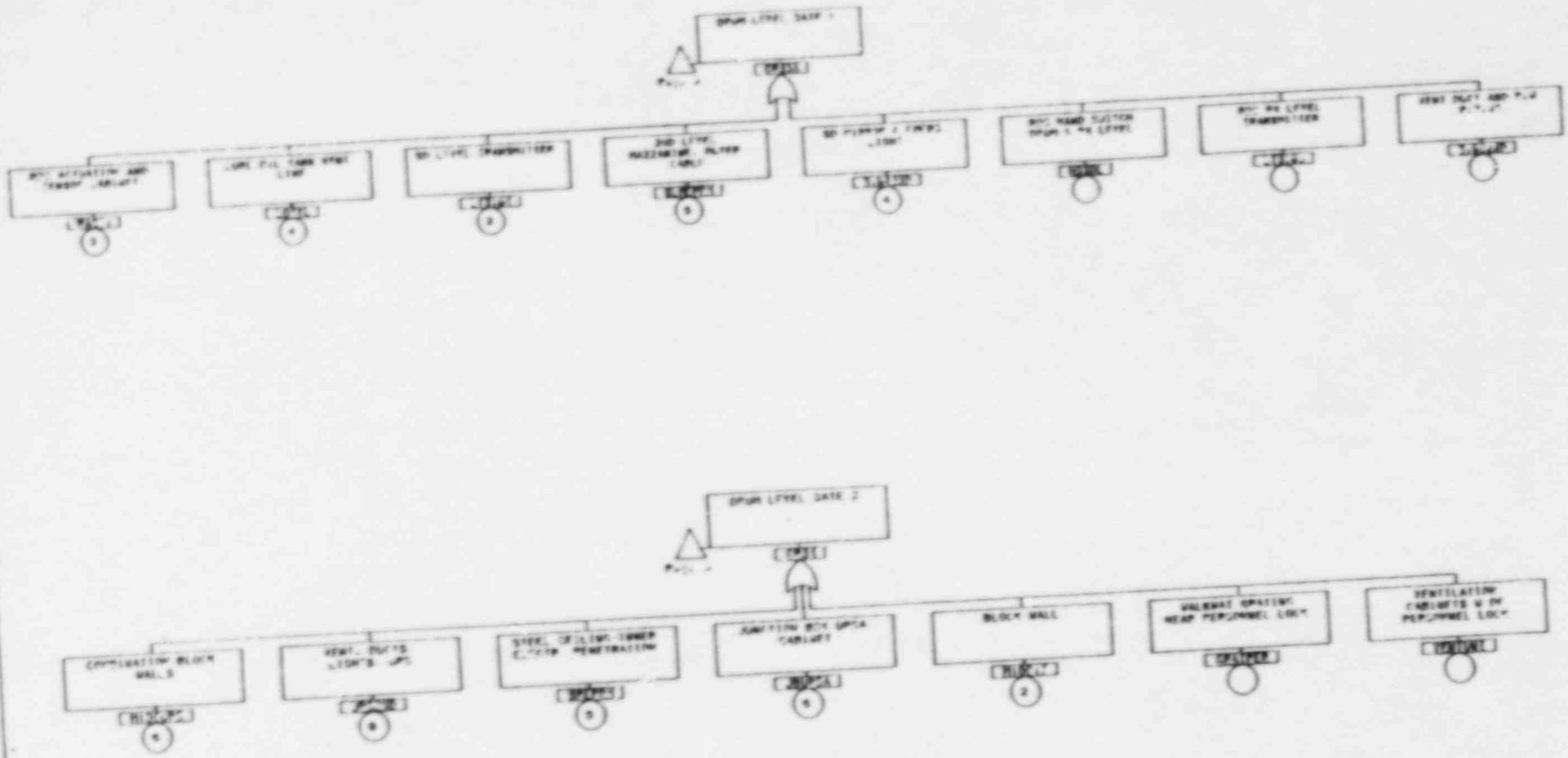

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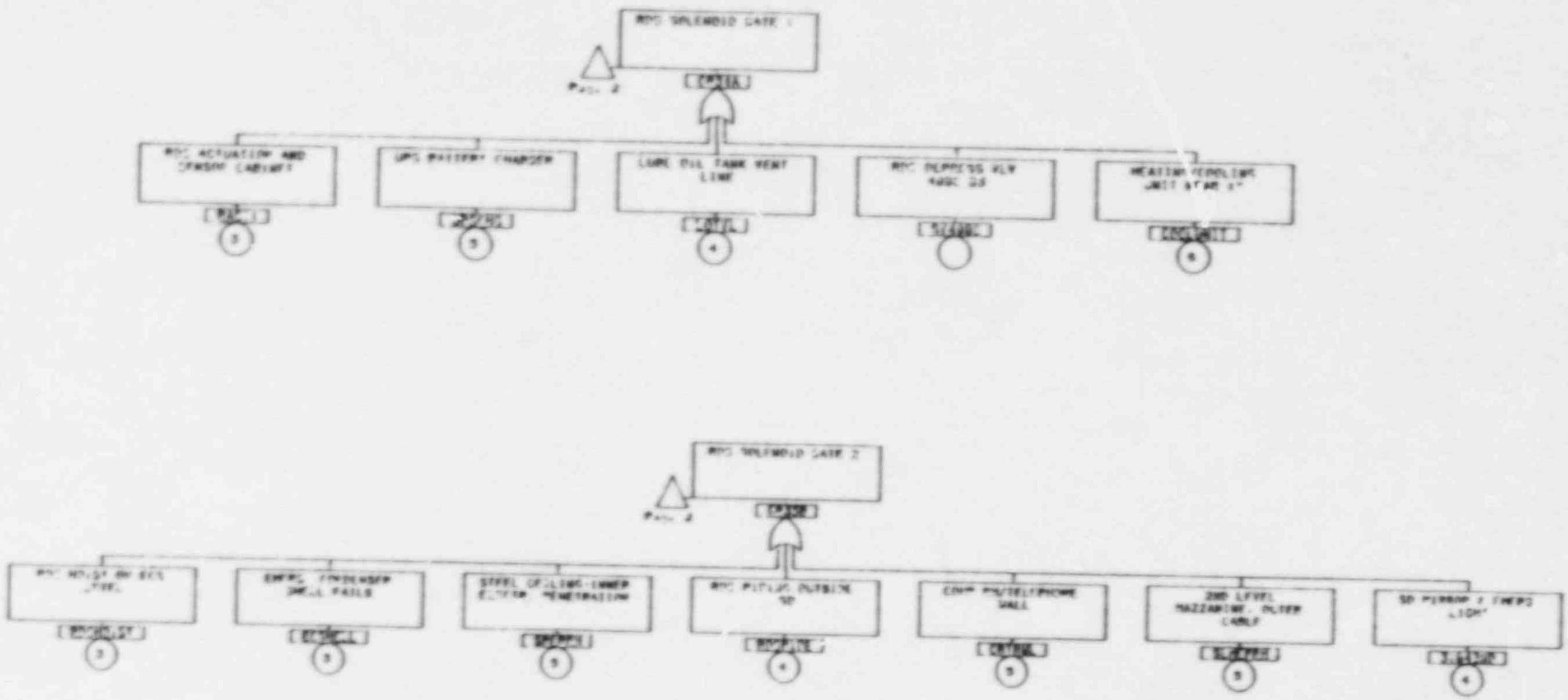

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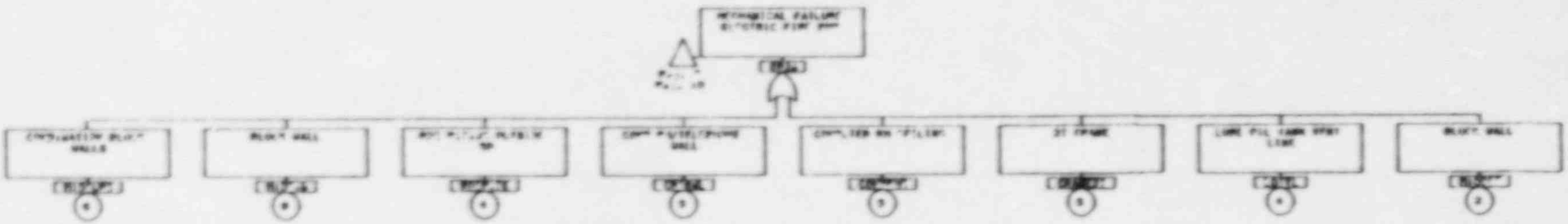

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CORE SPRAY - LOCA
FAULT TREES

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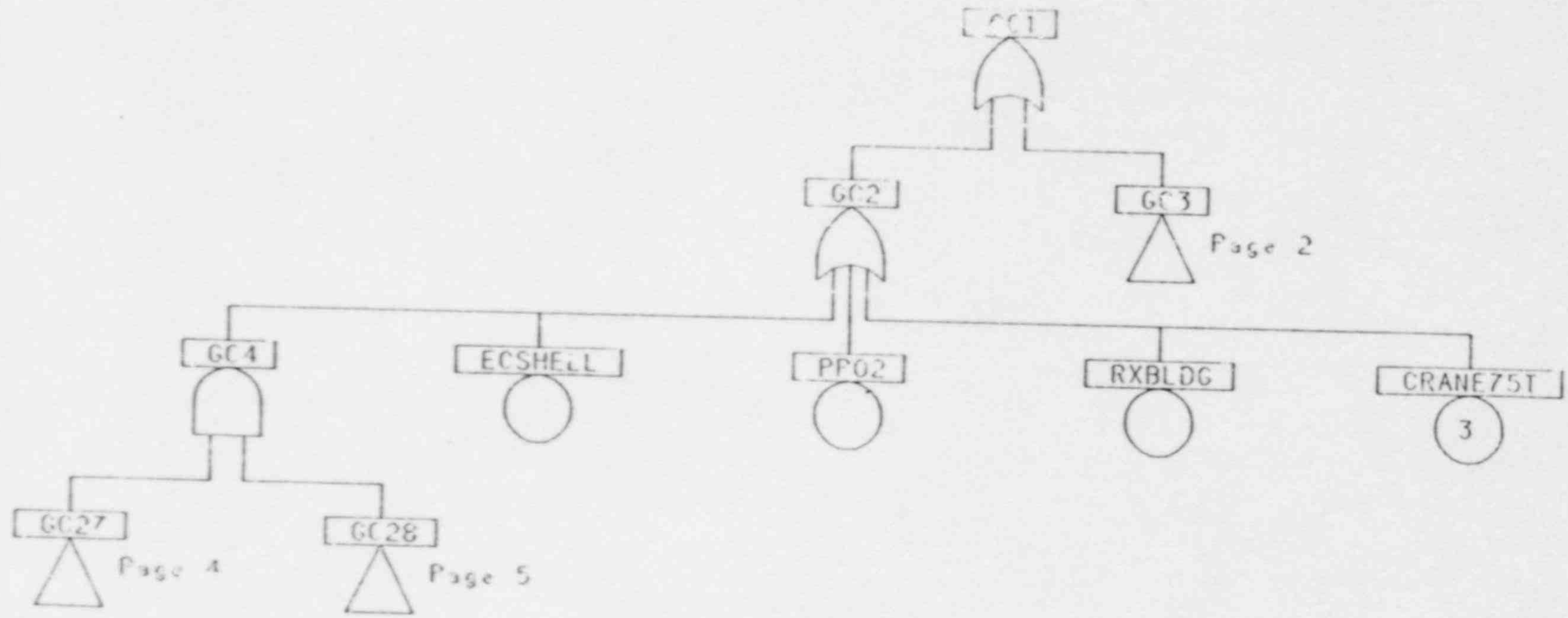

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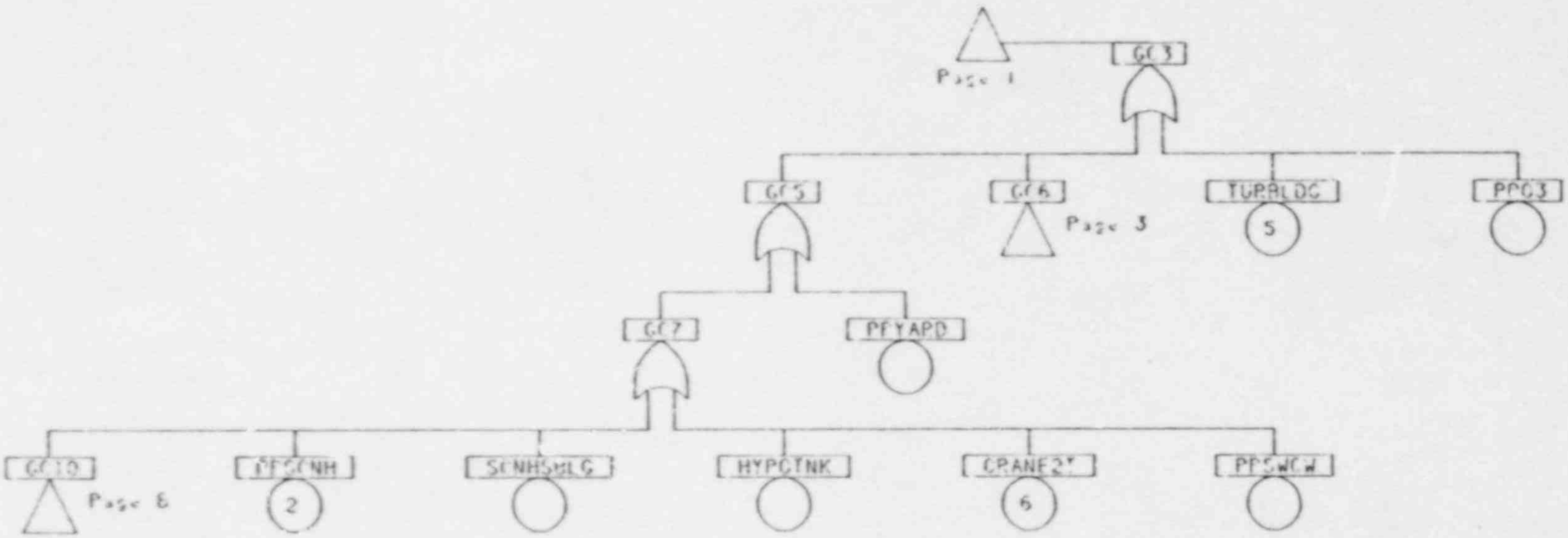

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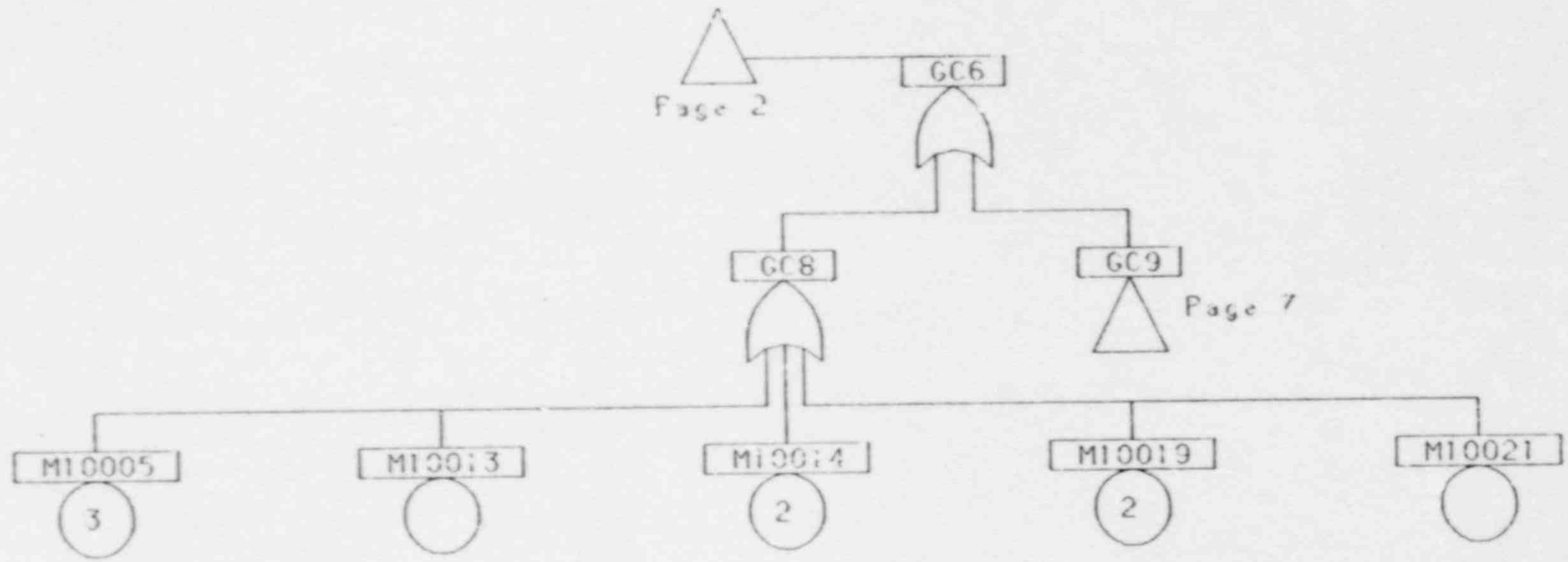

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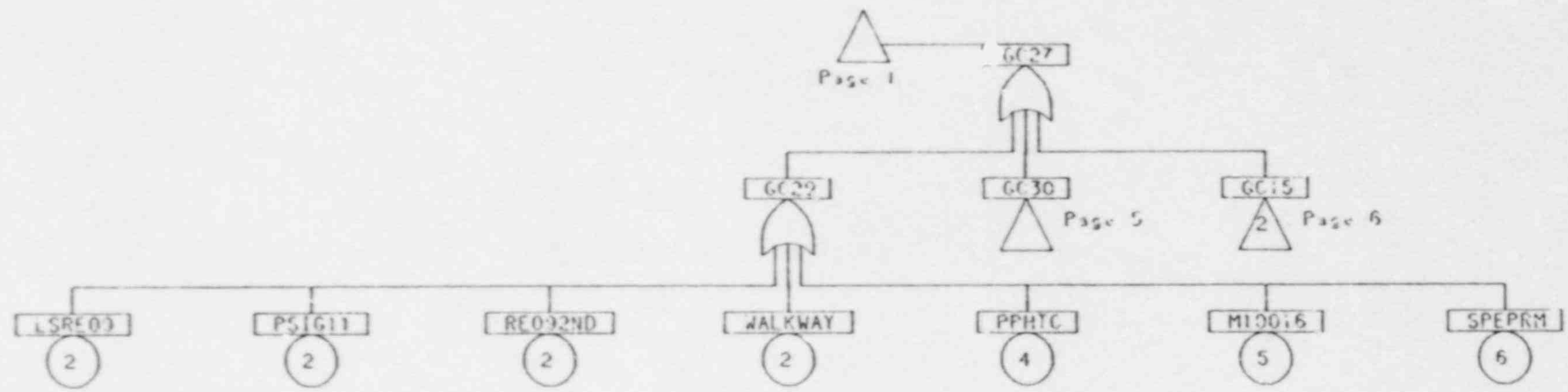

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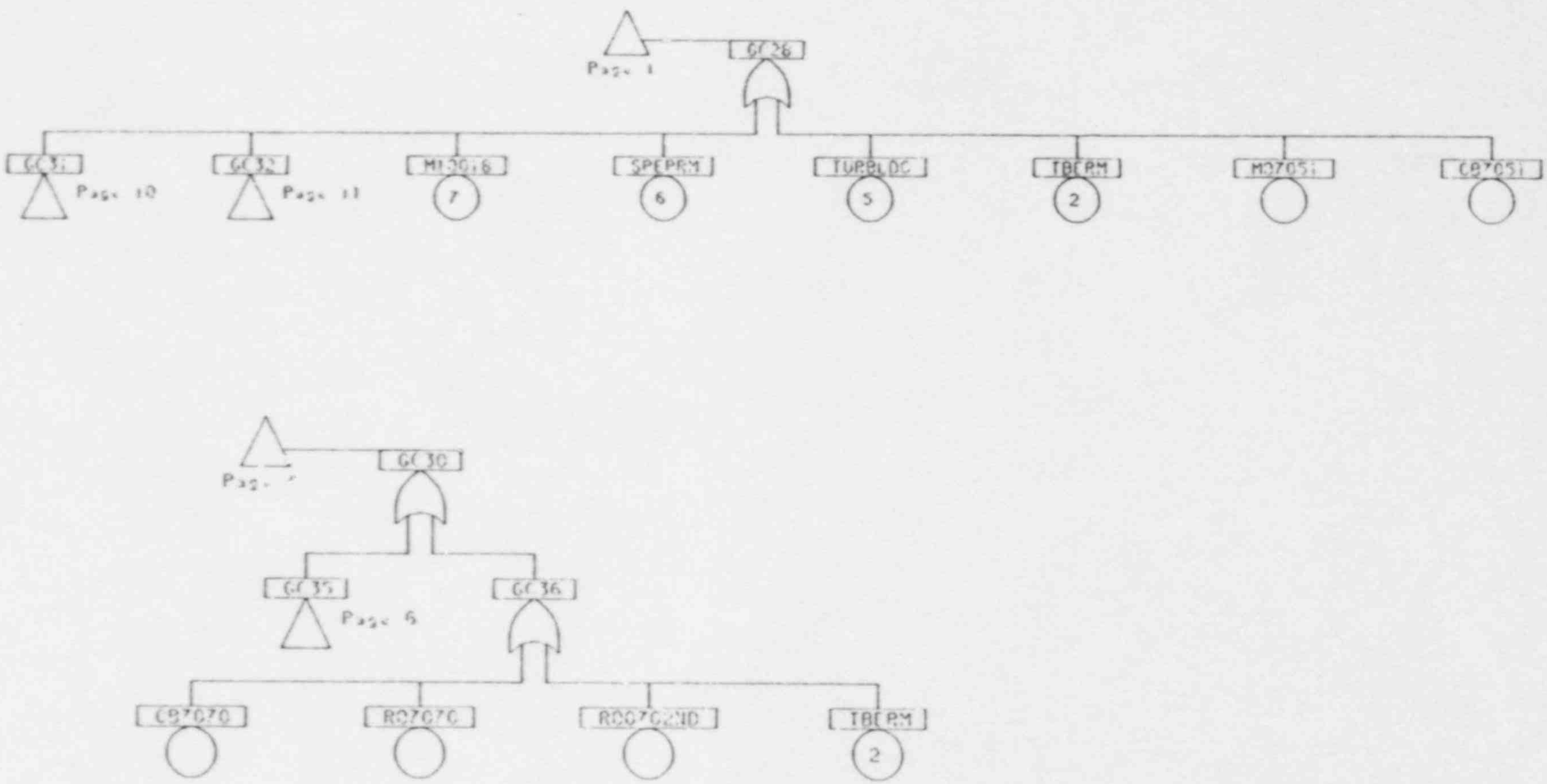

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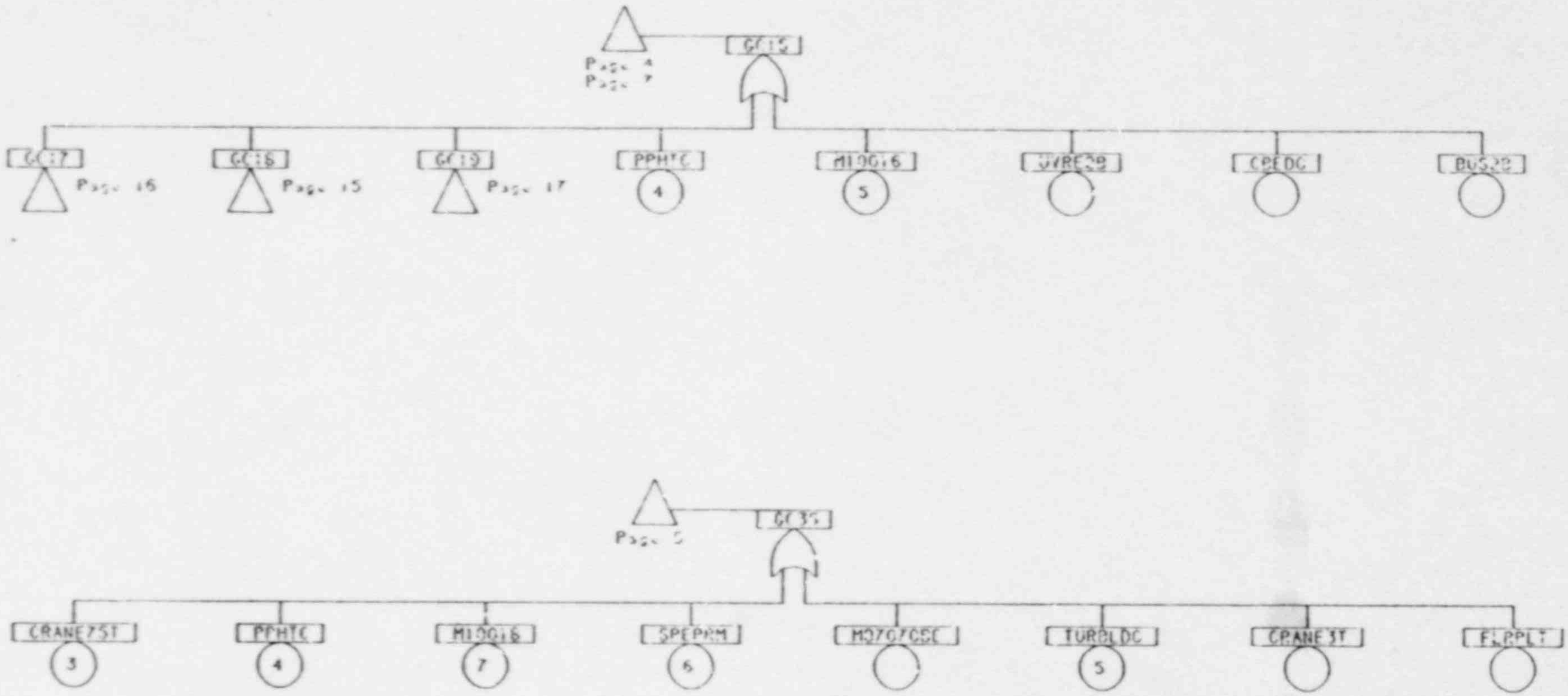

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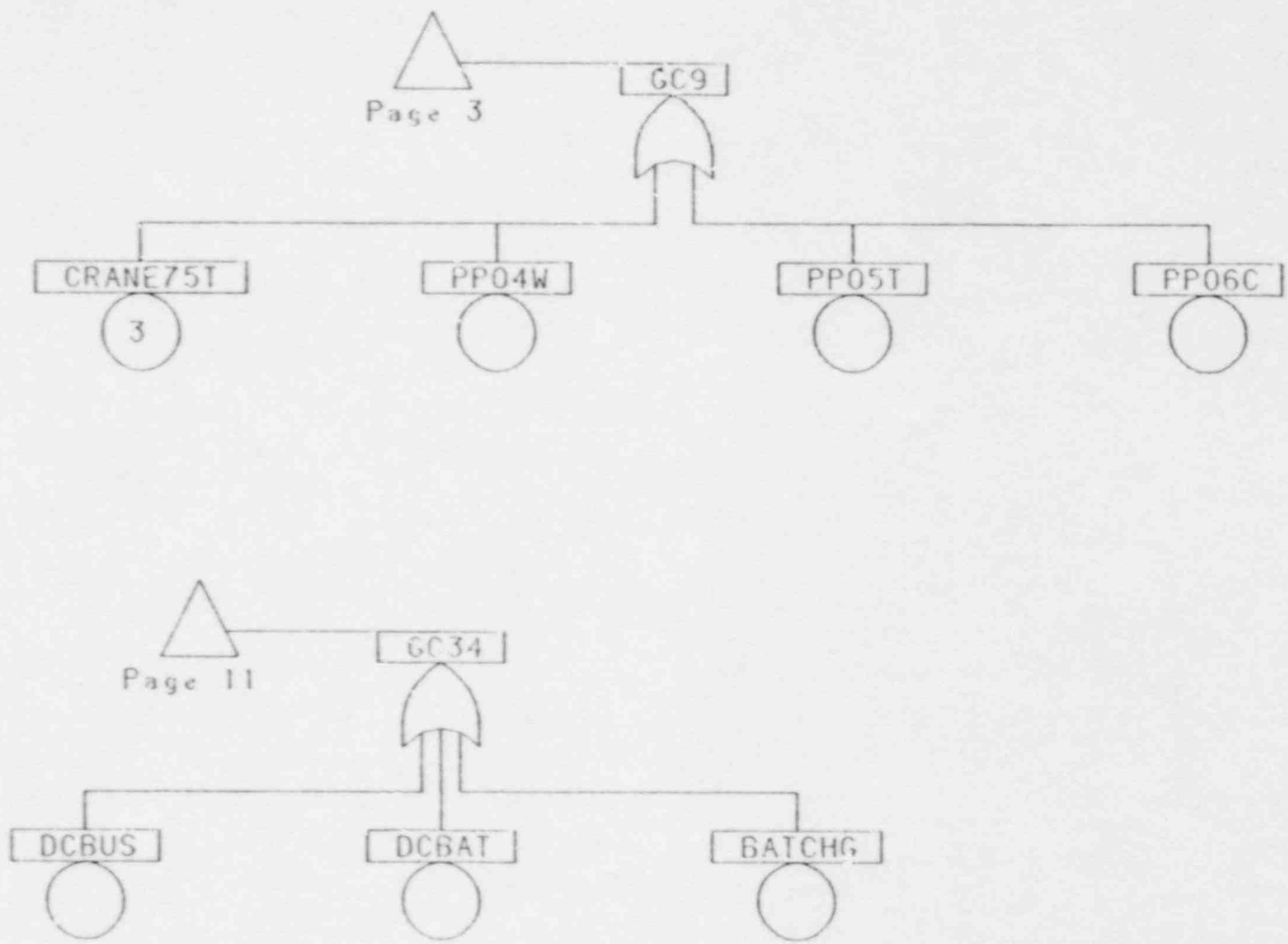

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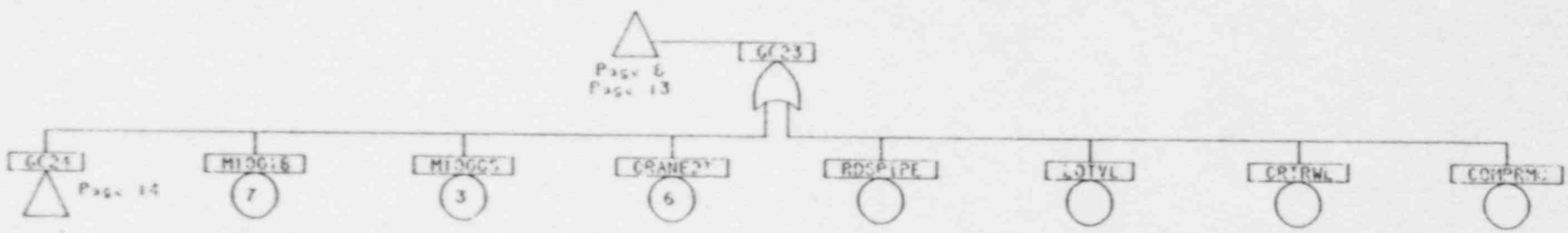

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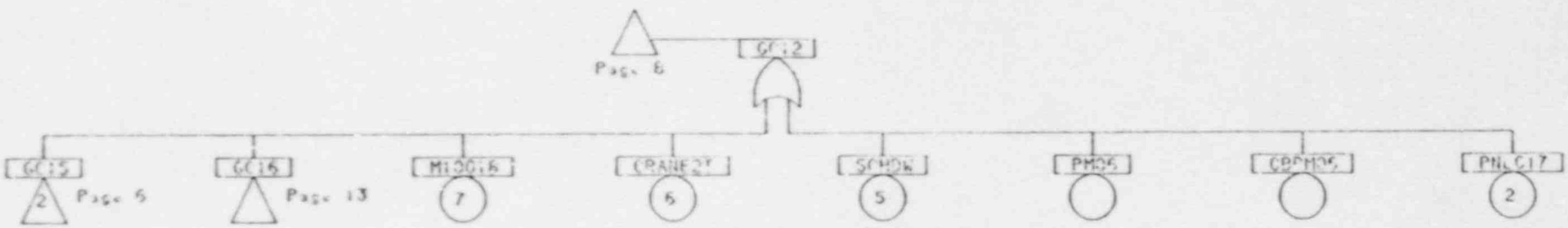

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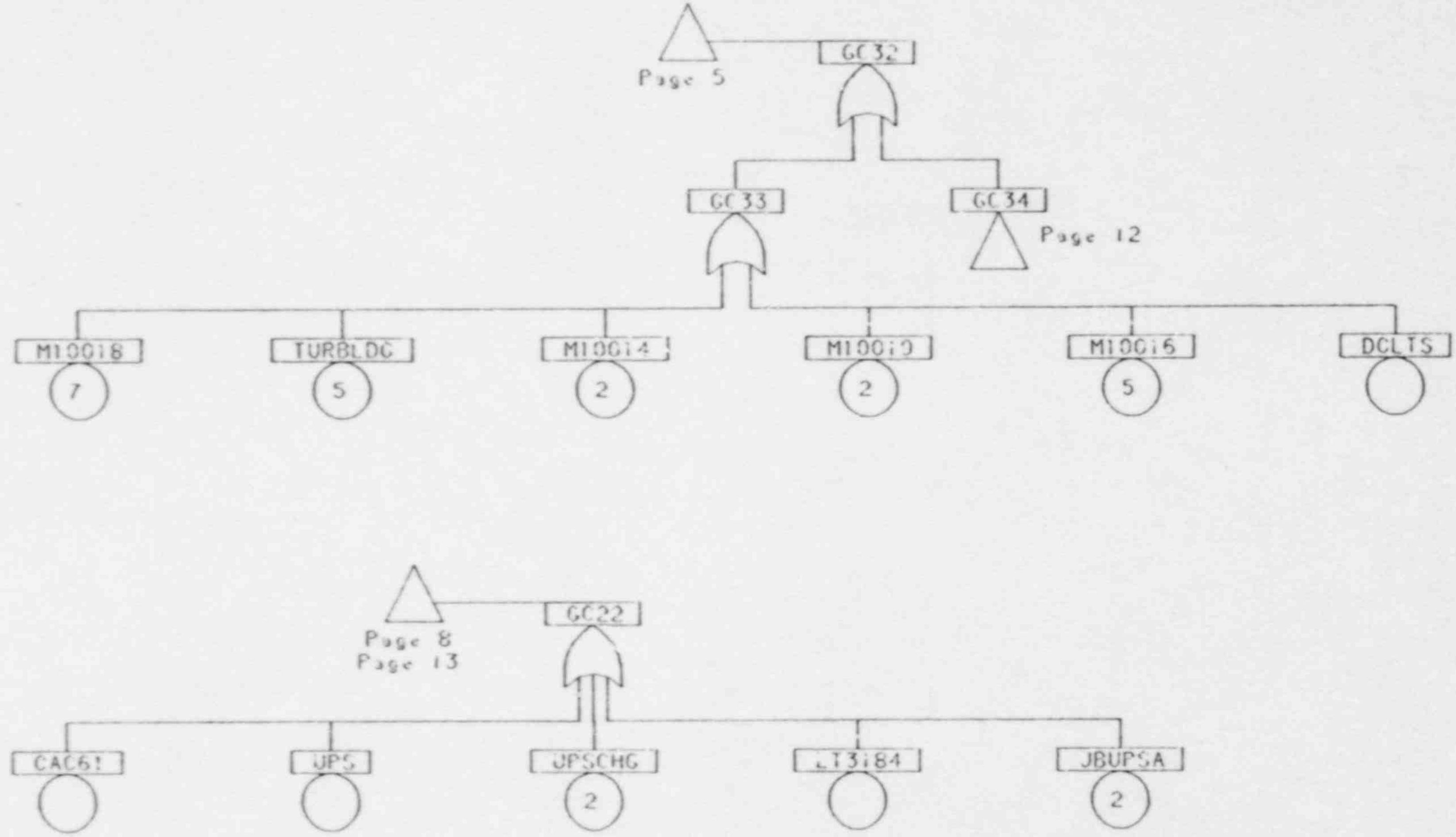

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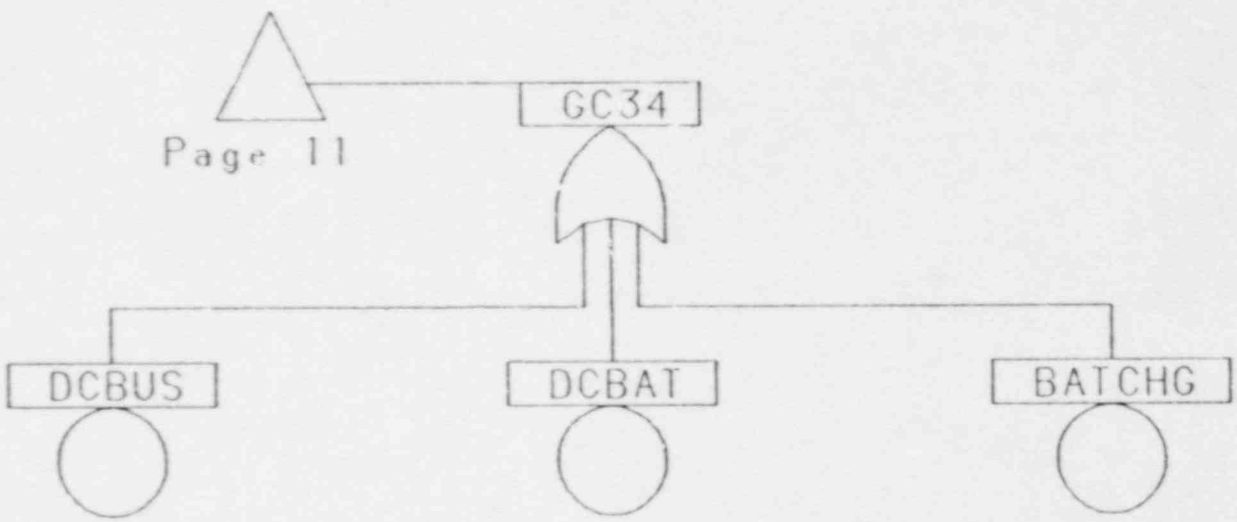

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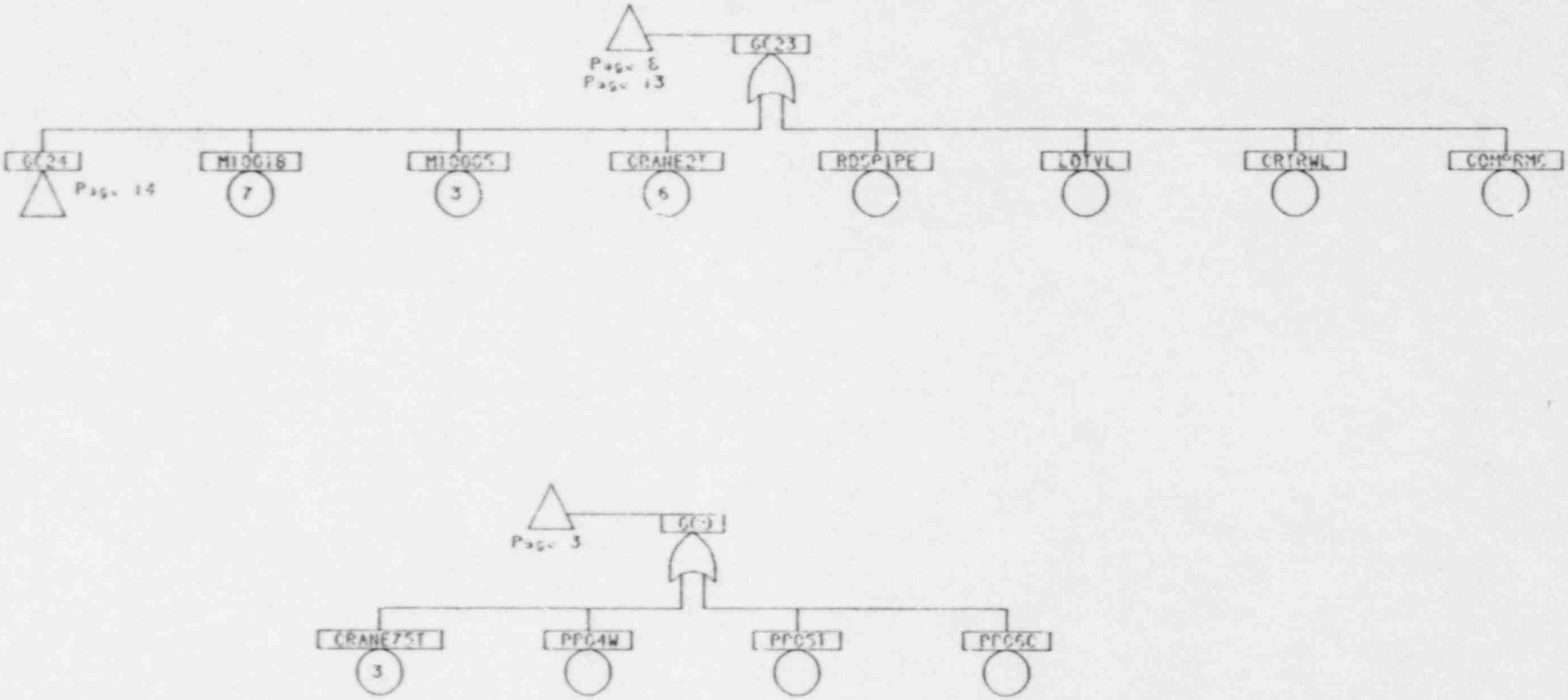

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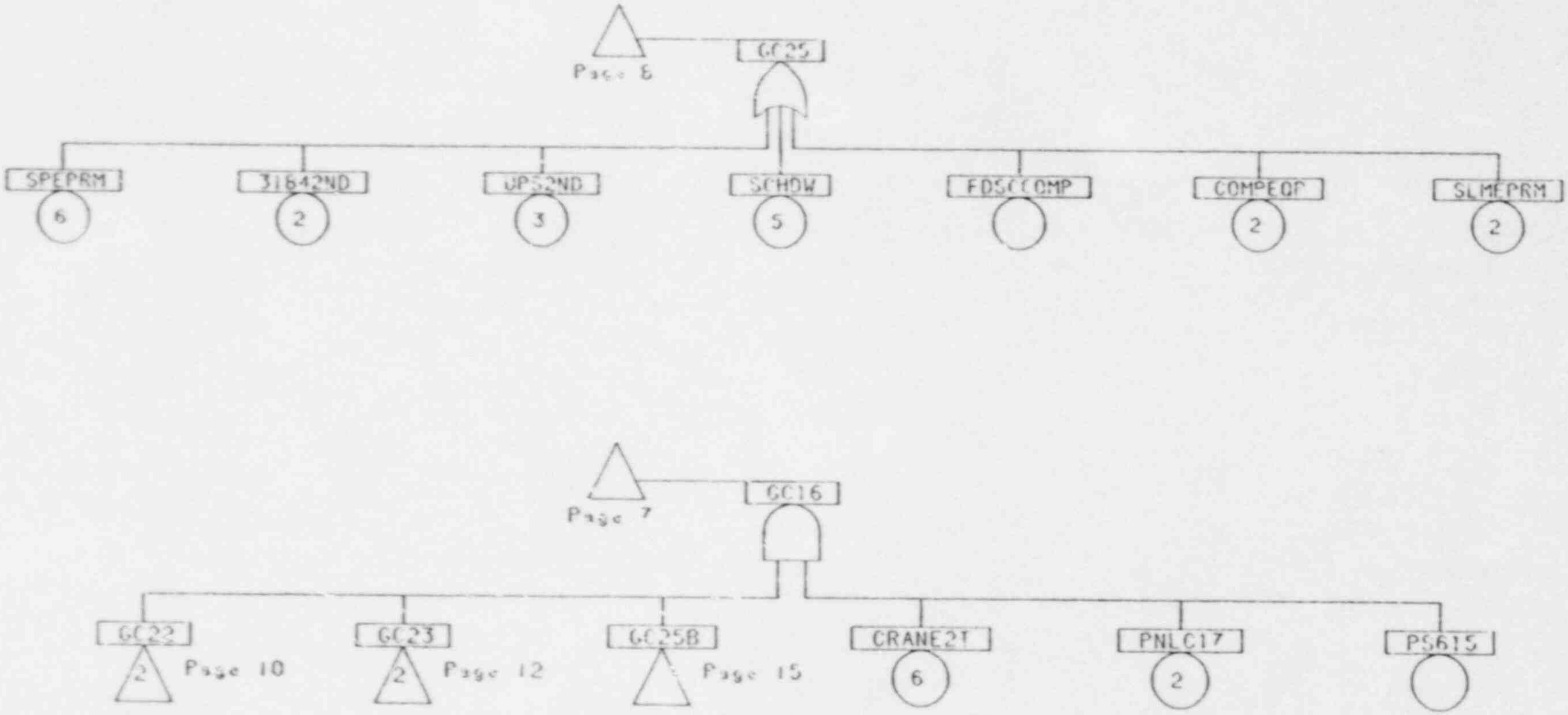

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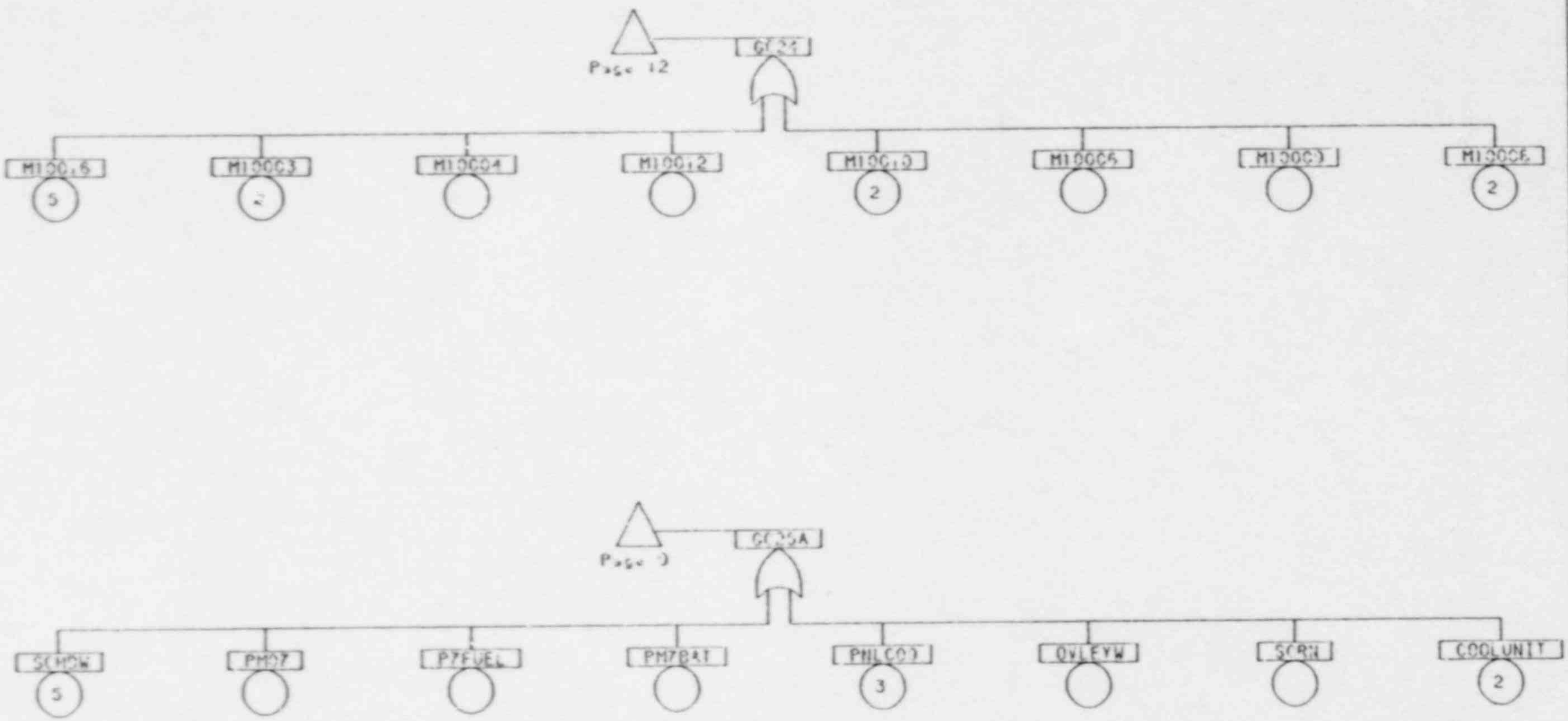

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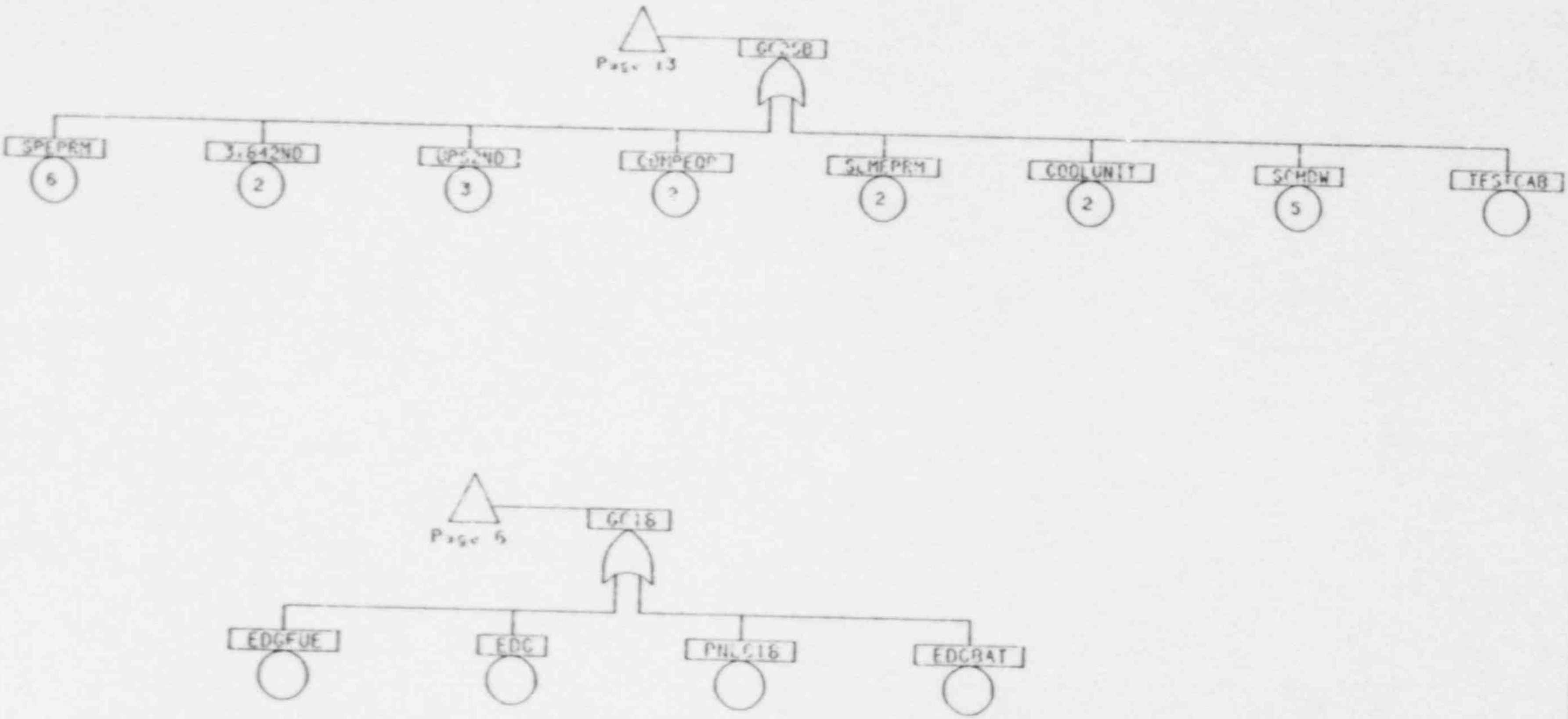
CORE SPRAY - LOCA

DRAWING NUMBER	DATE
Page 14	4/27/88



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CORE SPRAY - LOCA

DRAWING NUMBER	DATE
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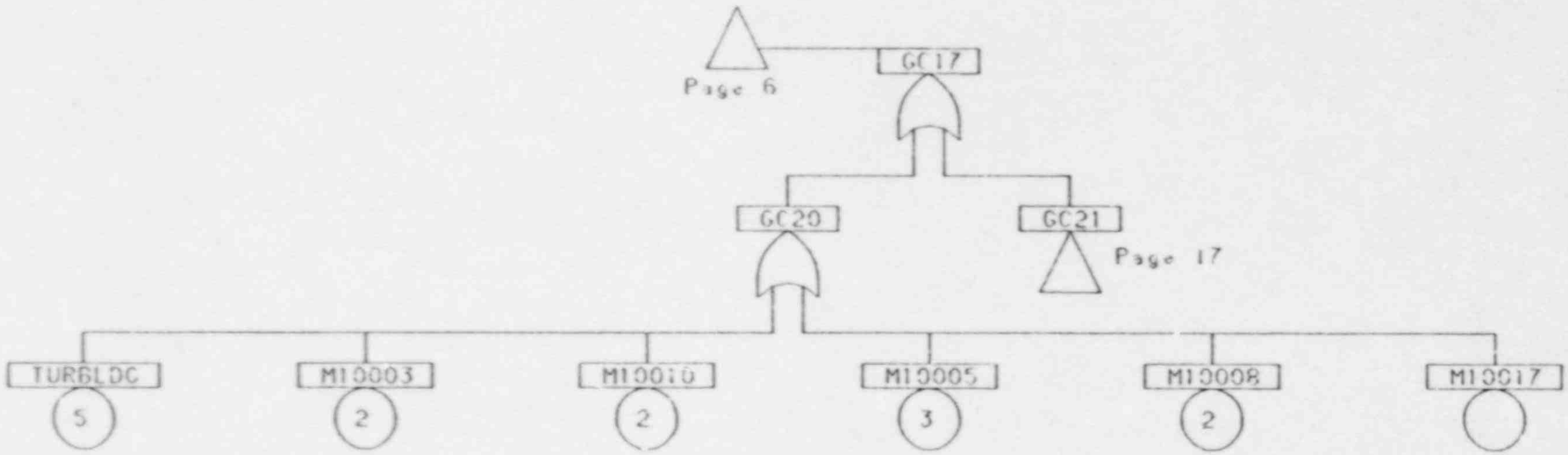

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CORE SPRAY - LOCA	
DRAWING NUMBER	DATE
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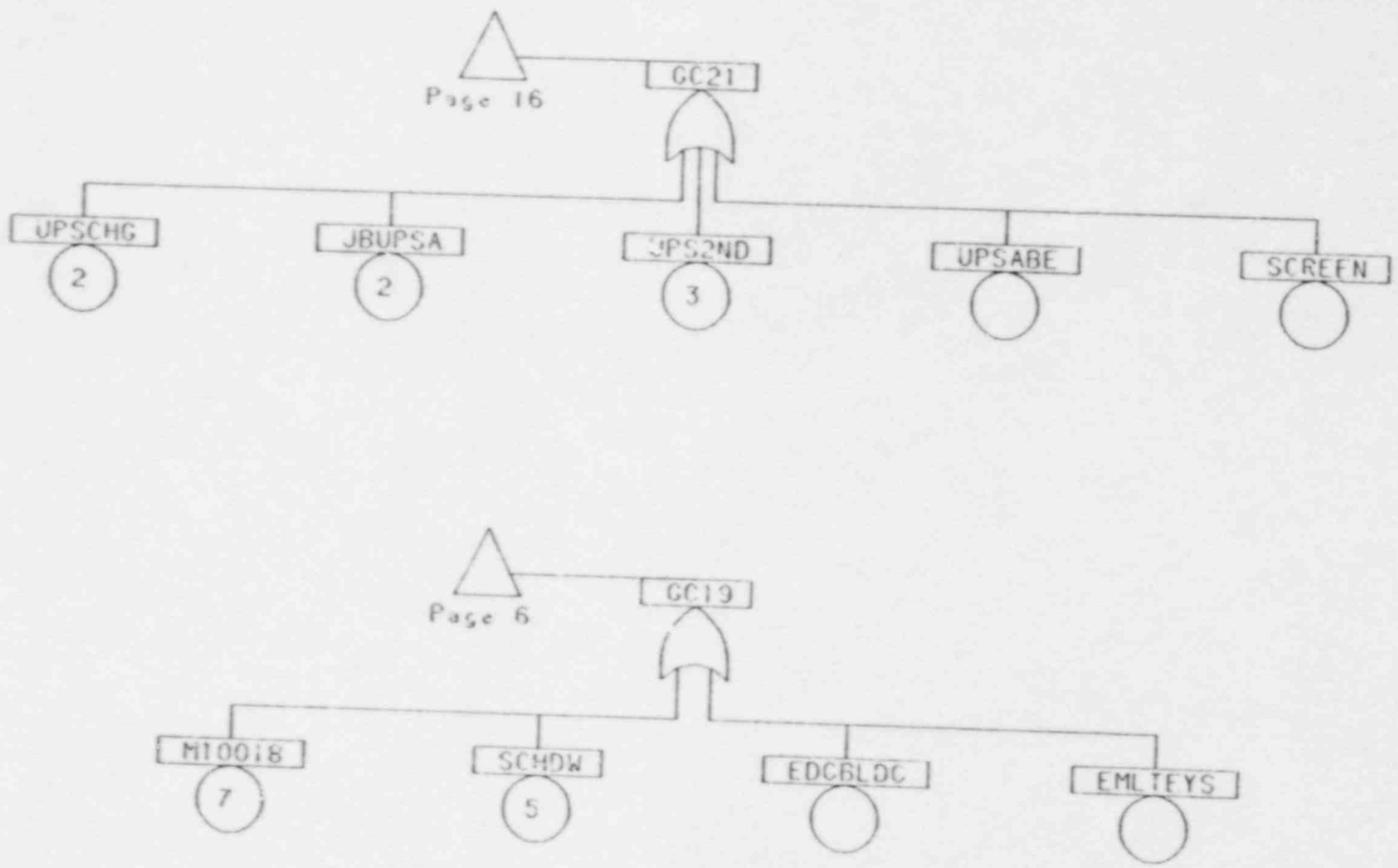

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DRAWING NUMBER Page 17	DATE 4/27/88

POST INCIDENT SYSTEM - LOCA
FAULT TREES

15 Pages

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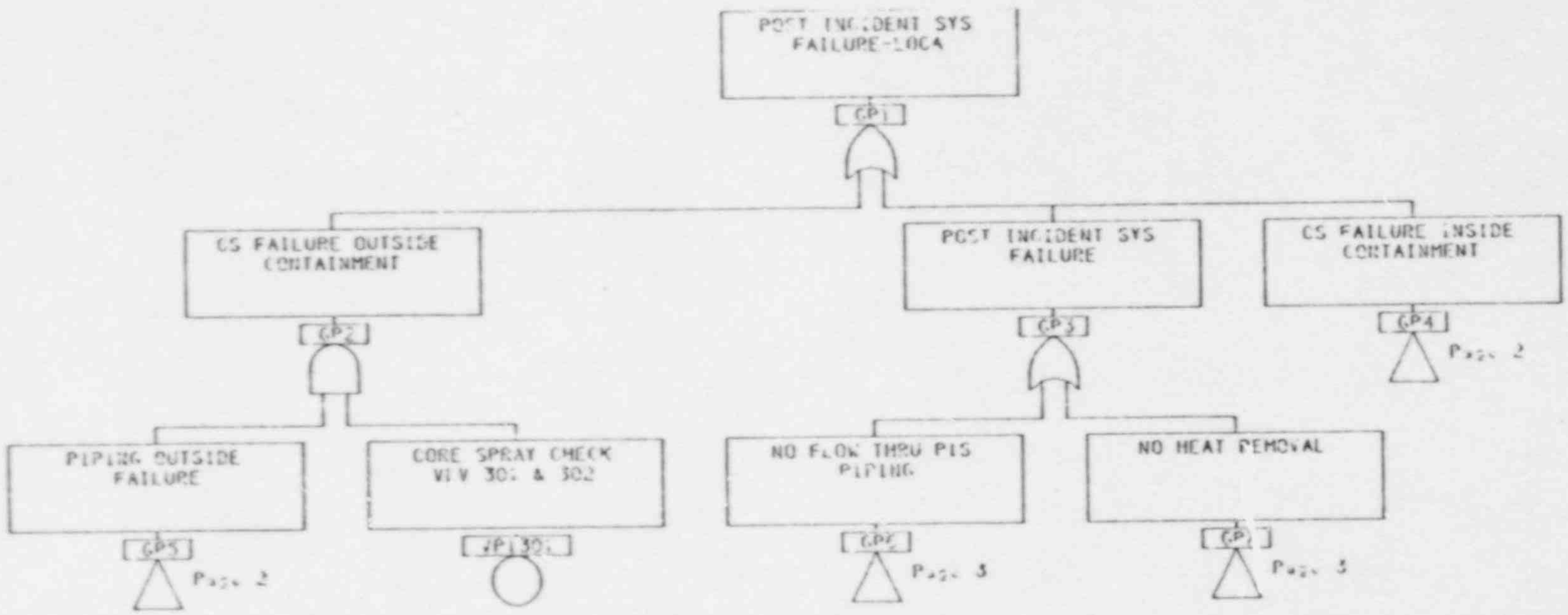

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DRAWING NUMBER
Page 1

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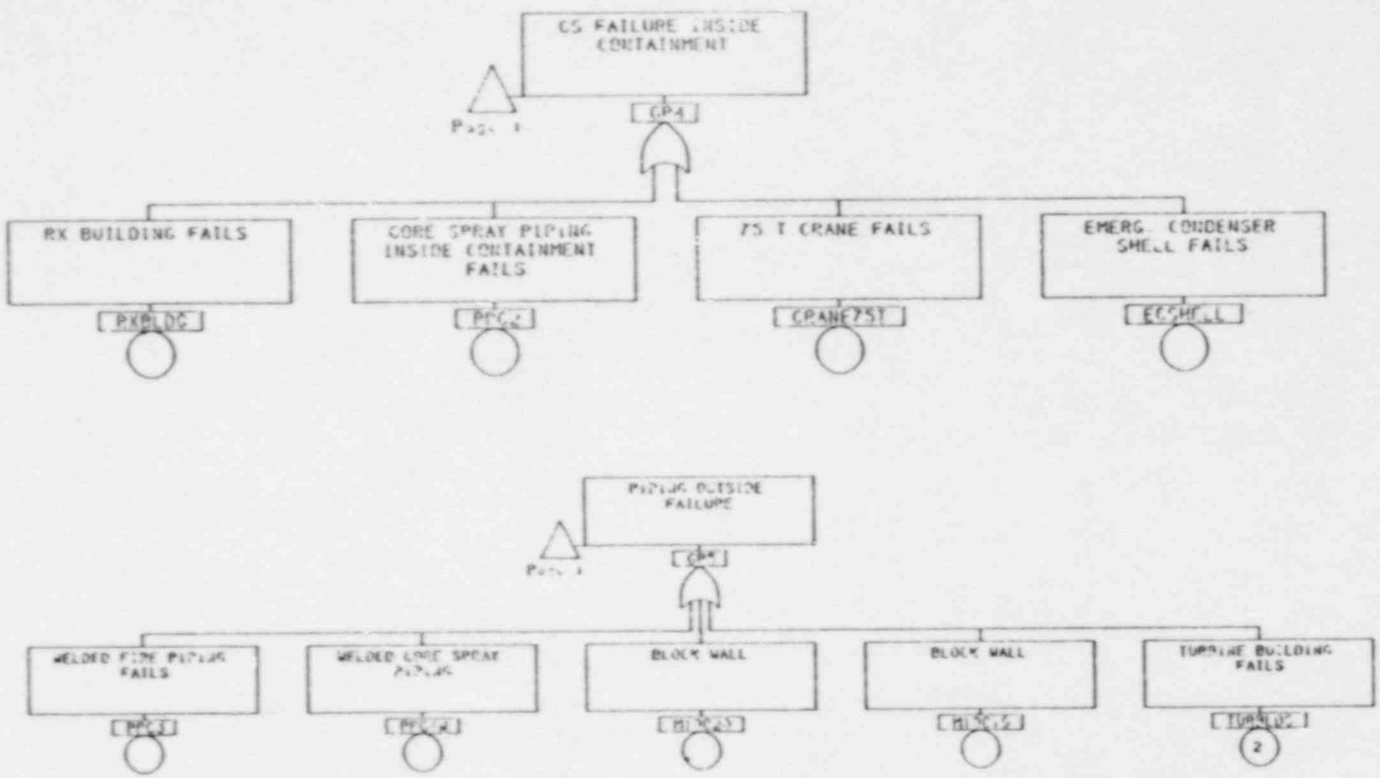

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Page 2

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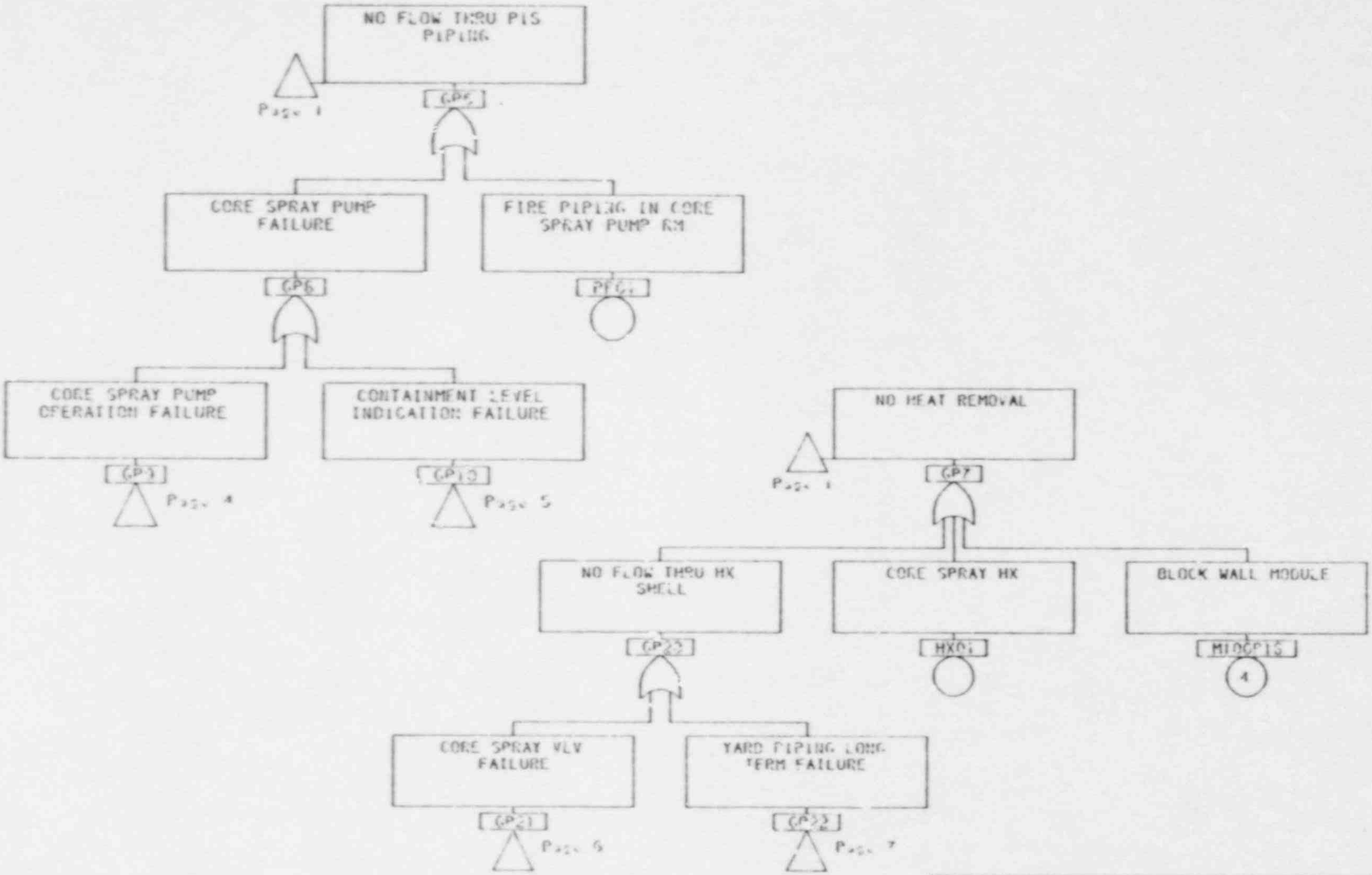

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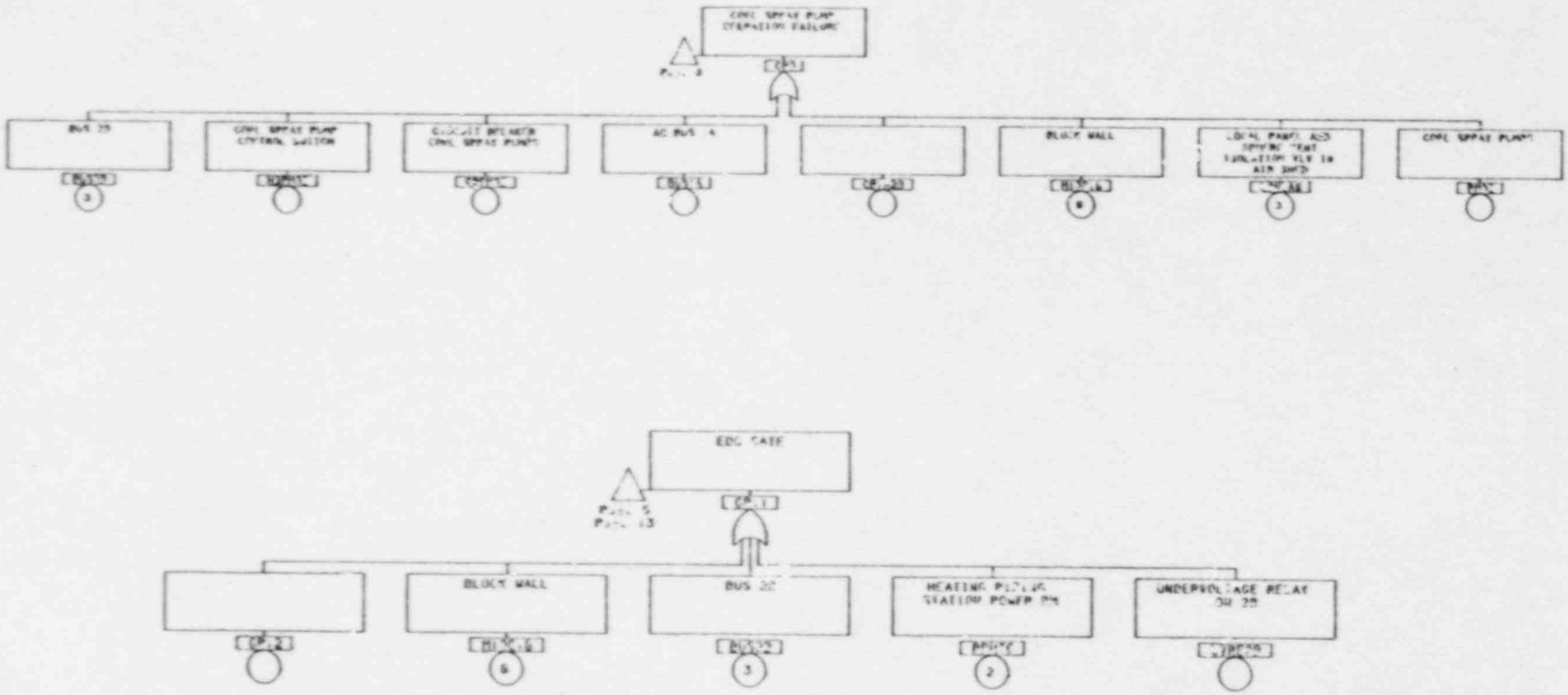
DRAWING NUMBER Page 3	DATE 4/27/88



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DRAWING NUMBER
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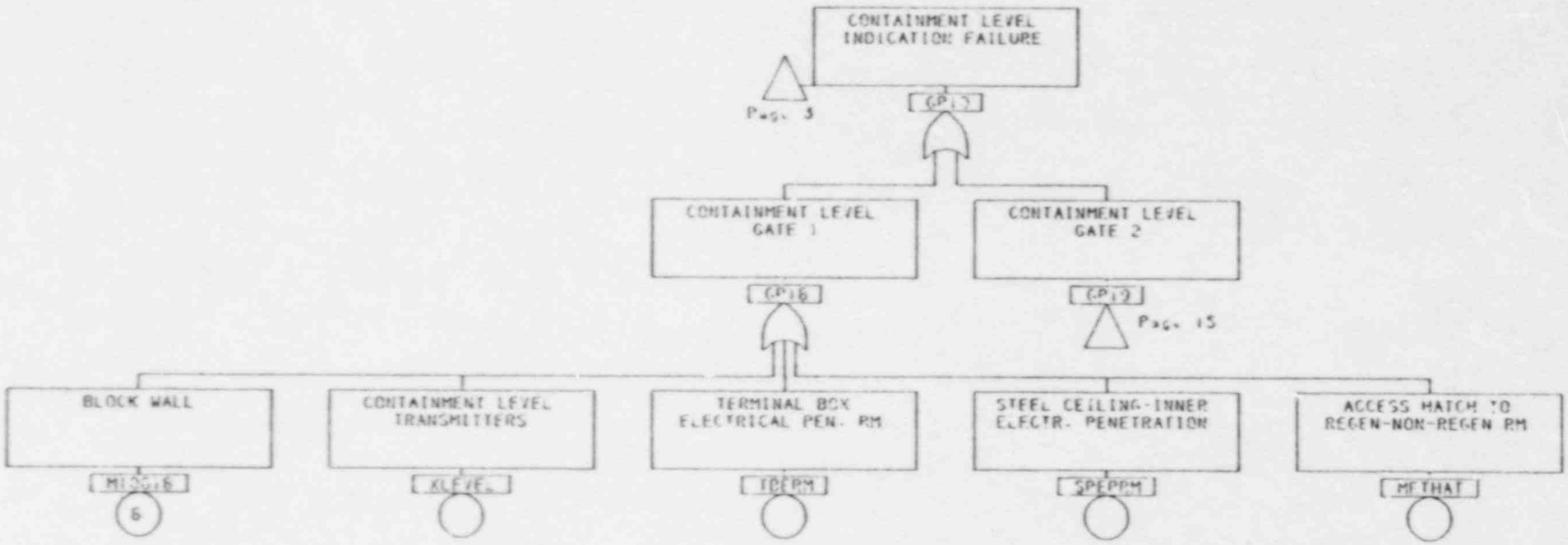

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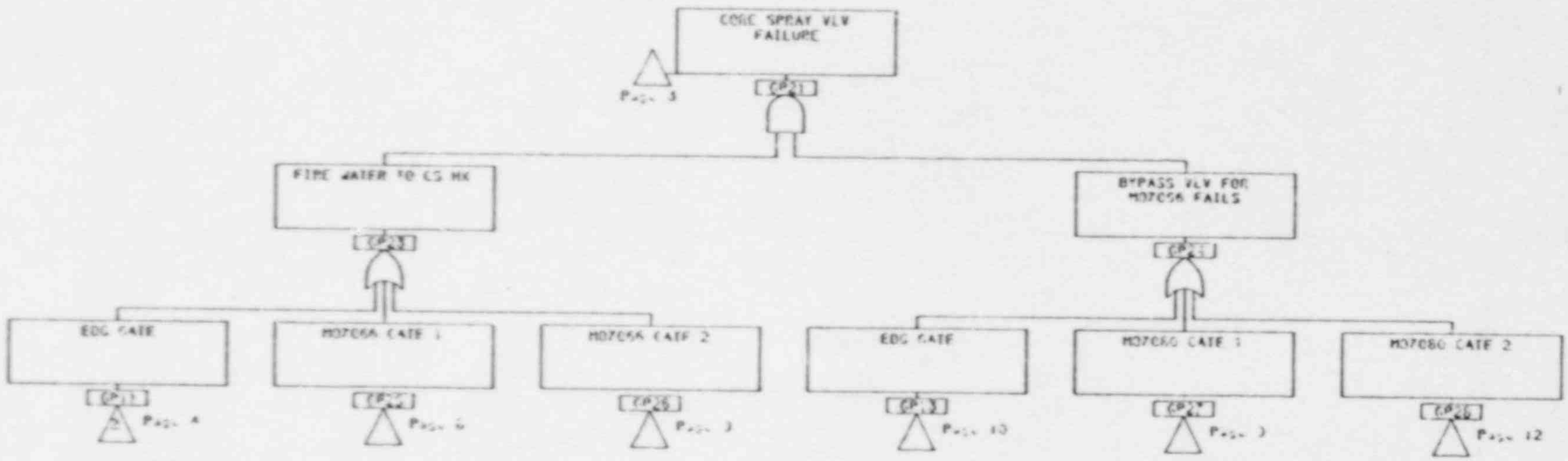
DRAWING NUMBER Page 5	DATE 4/27/88



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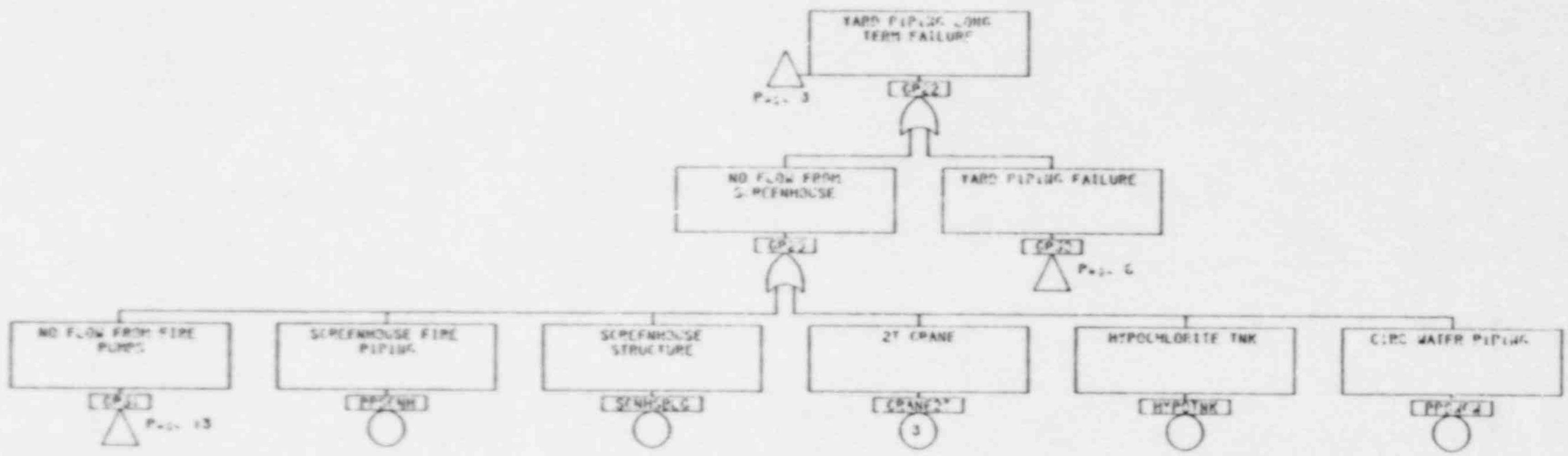
POST INCIDENT SYSTEM - LOCA

DRAWING NUMBER	DATE
Page 6	4/27/88



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POS' INCIDENT
SYSTEM - LOCA

DRAWING NUMBER	DATE
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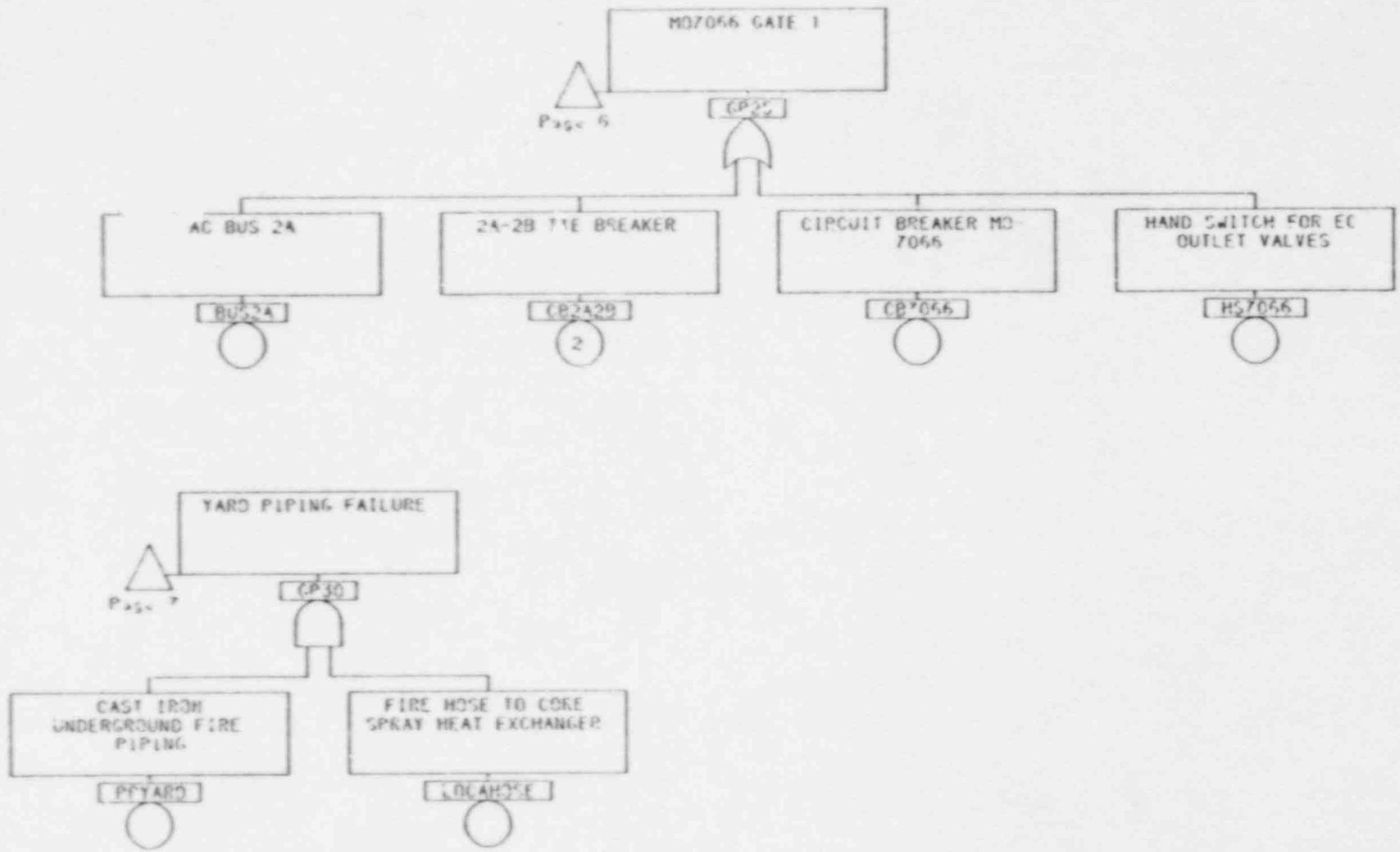

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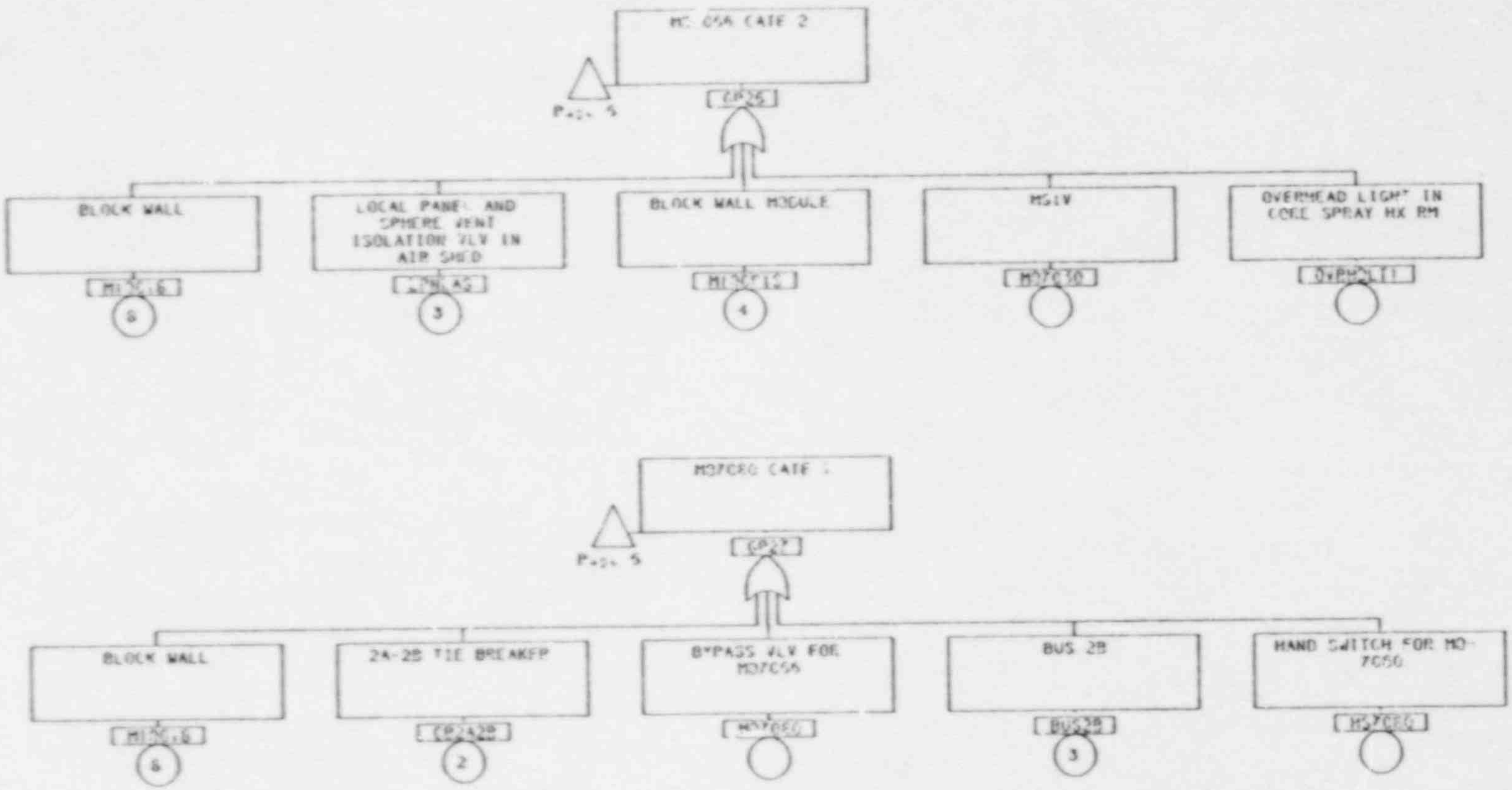
DRAWING NUMBER: Page 8	DATE: 4/27/88



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SYSTEM - LOCA

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Page 9

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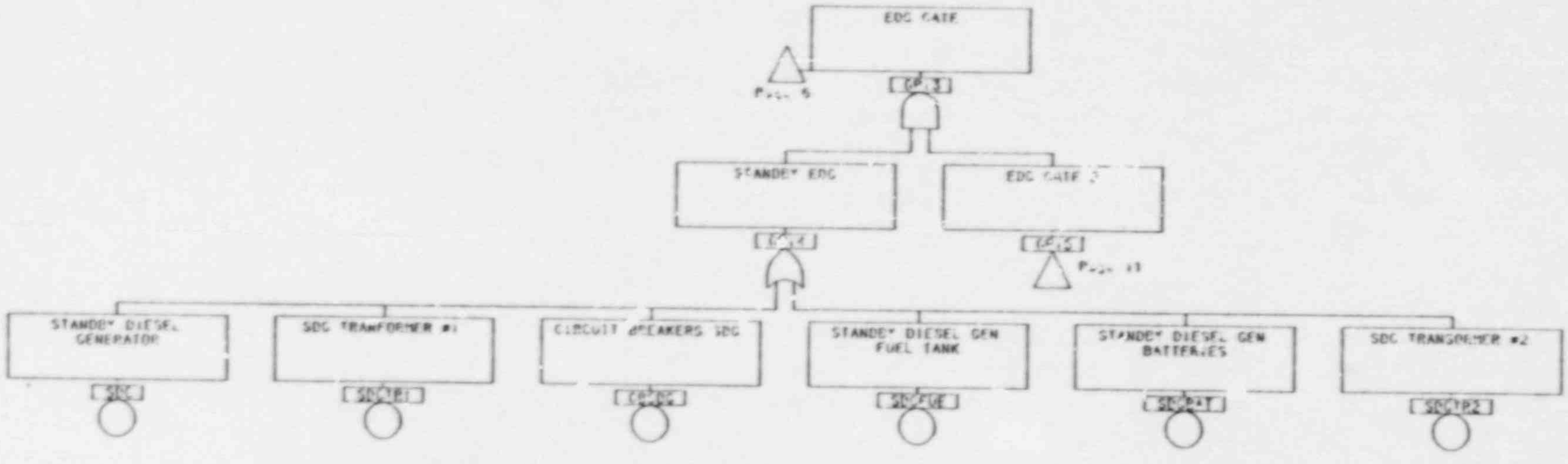
4/27/88



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POST INCIDENT SYSTEM - LOCA

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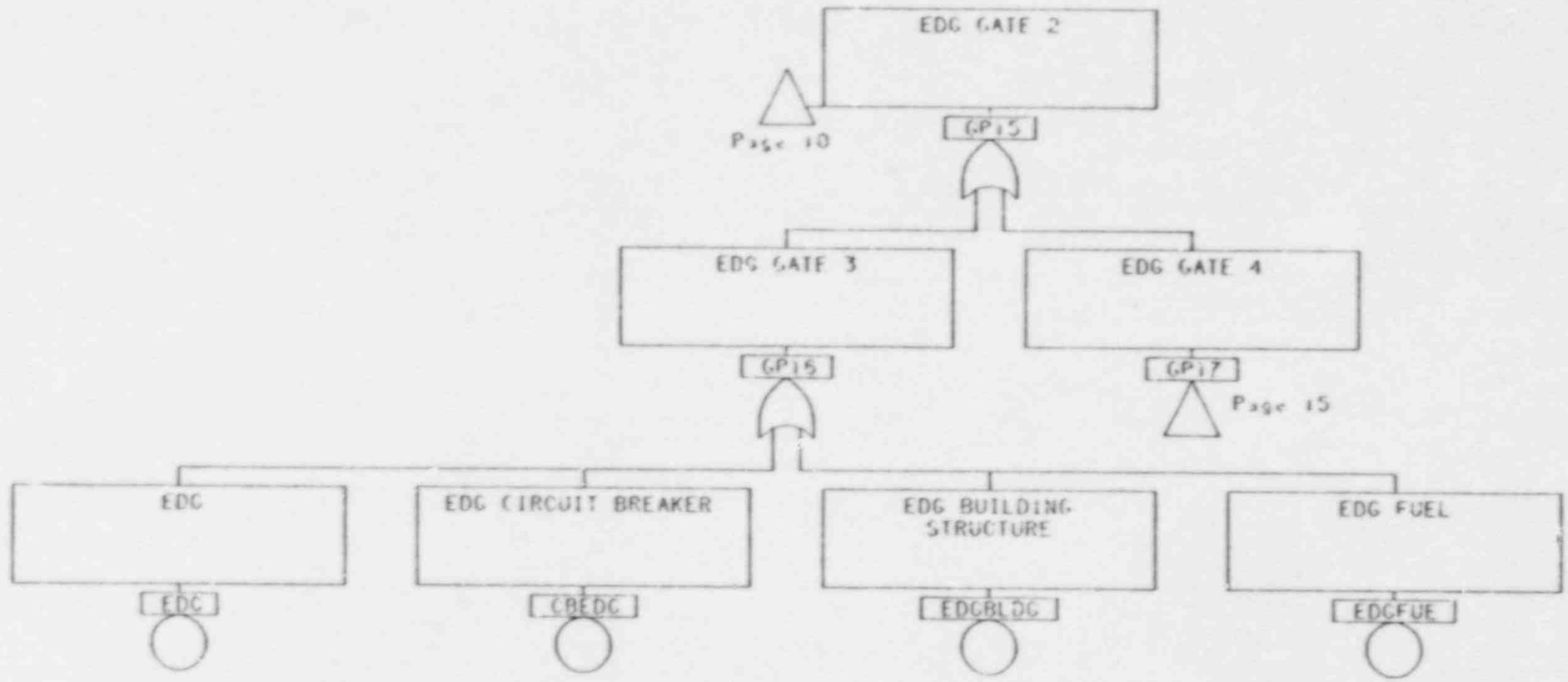
DATE: 4/27/88



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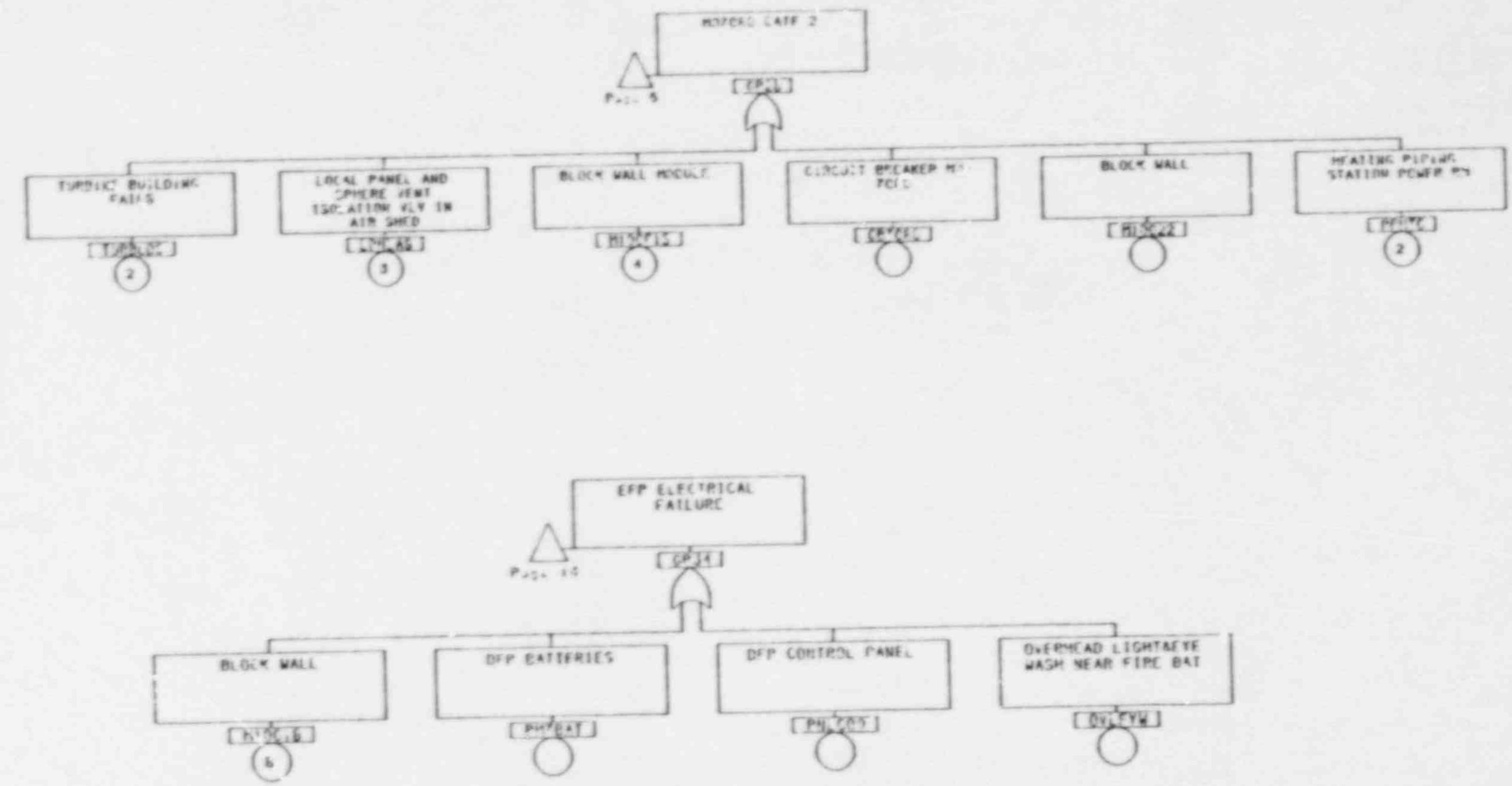
DRAWING NUMBER Page 11	DATE 4/27/88



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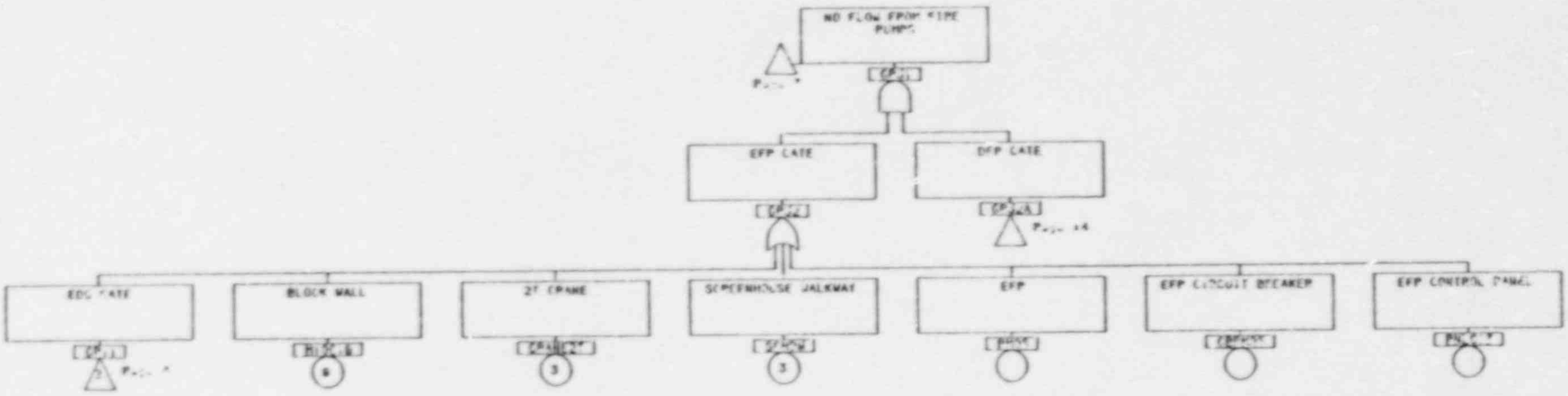

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Page 13

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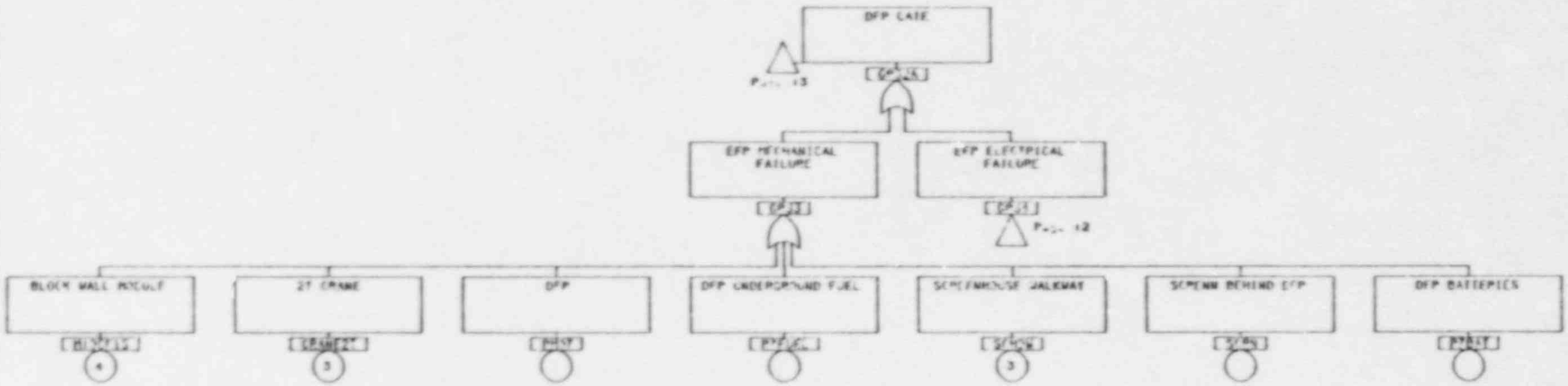
4/27/88



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POST INCIDENT SYSTEM - LOCA

DRAWING NUMBER	DATE
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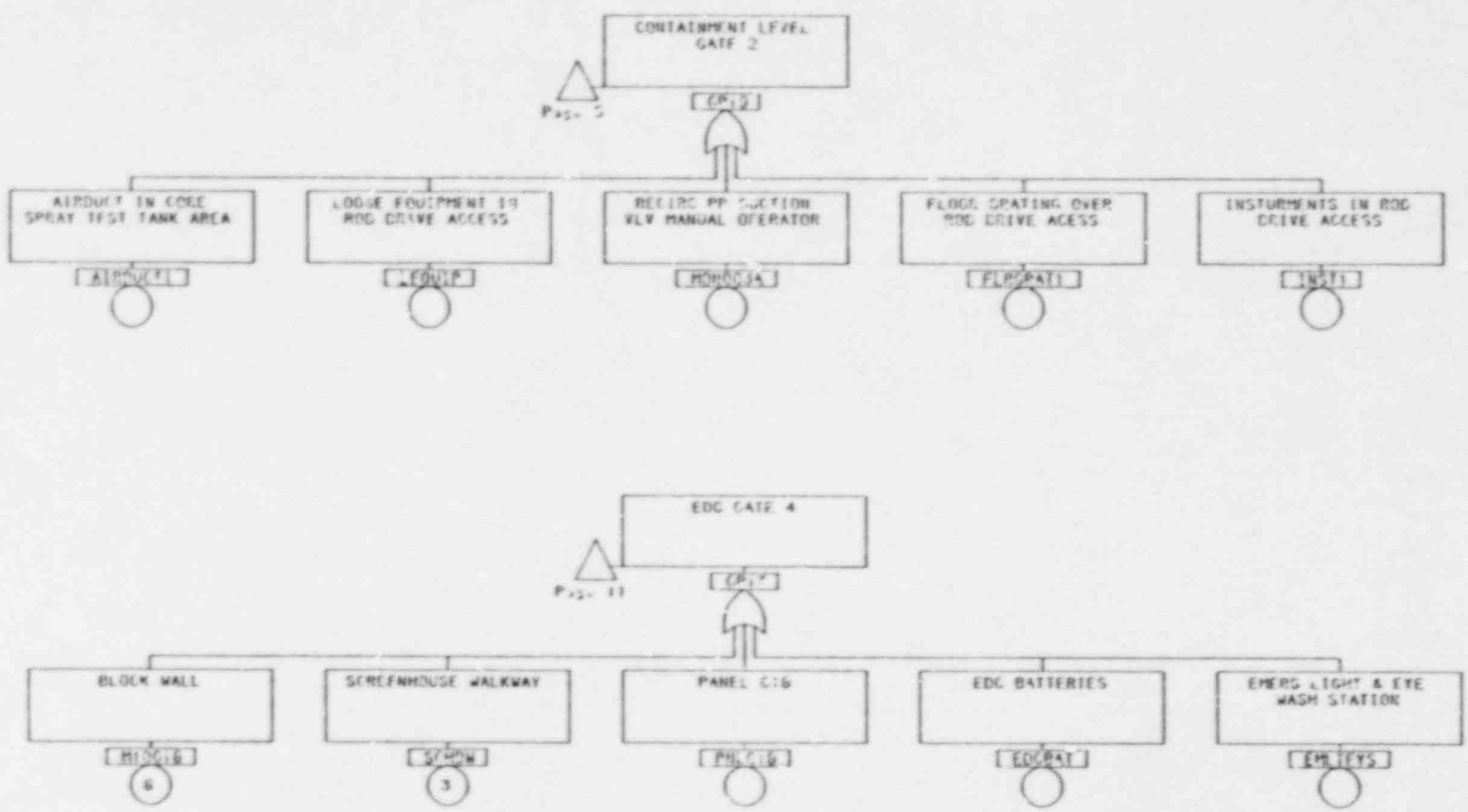

TITLE
POST INCIDENT SYSTEM - LOCA

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CONTROL ROD DRIVE MAKE-UP
FAULT TREES

8 Pages

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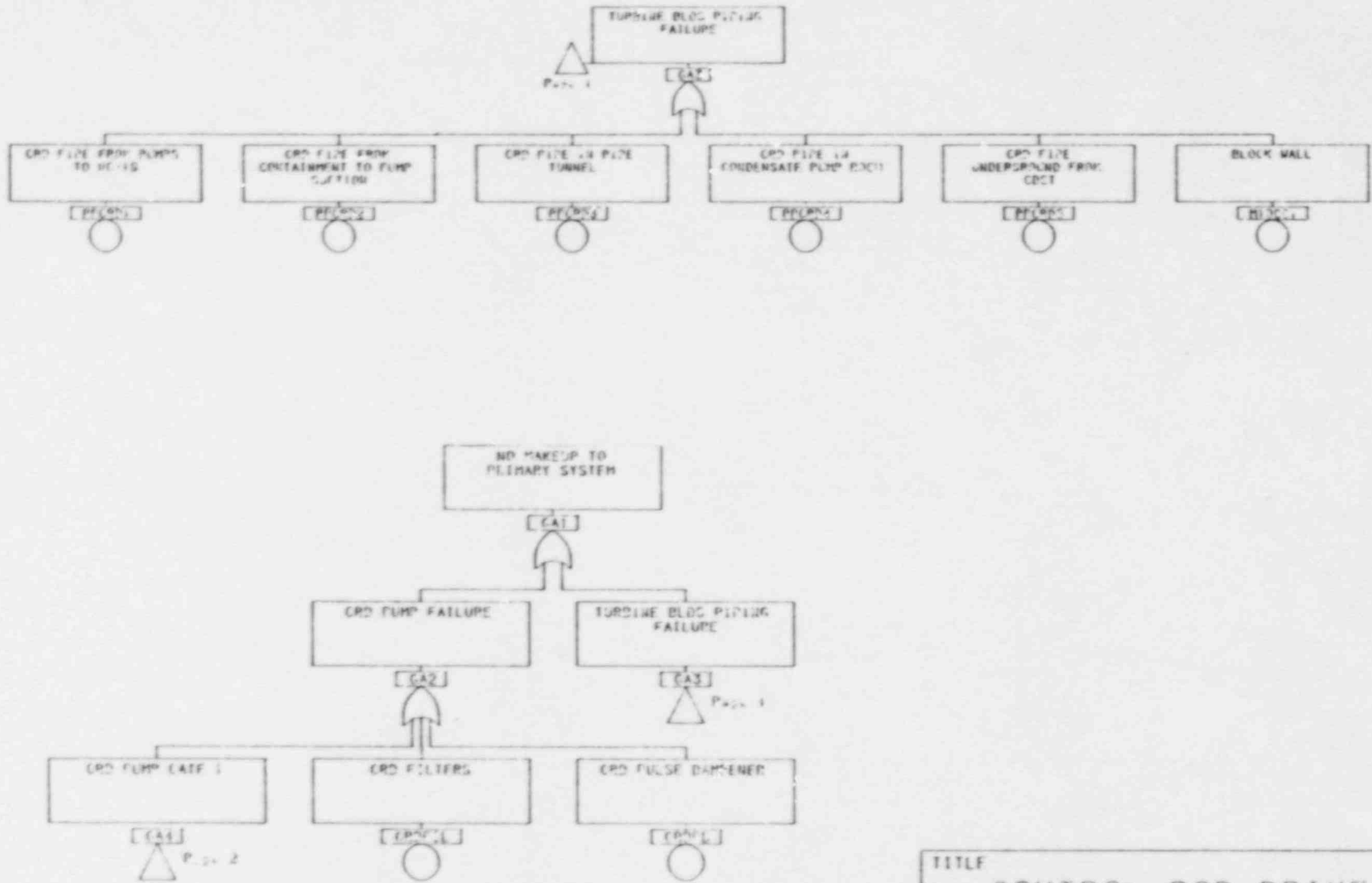

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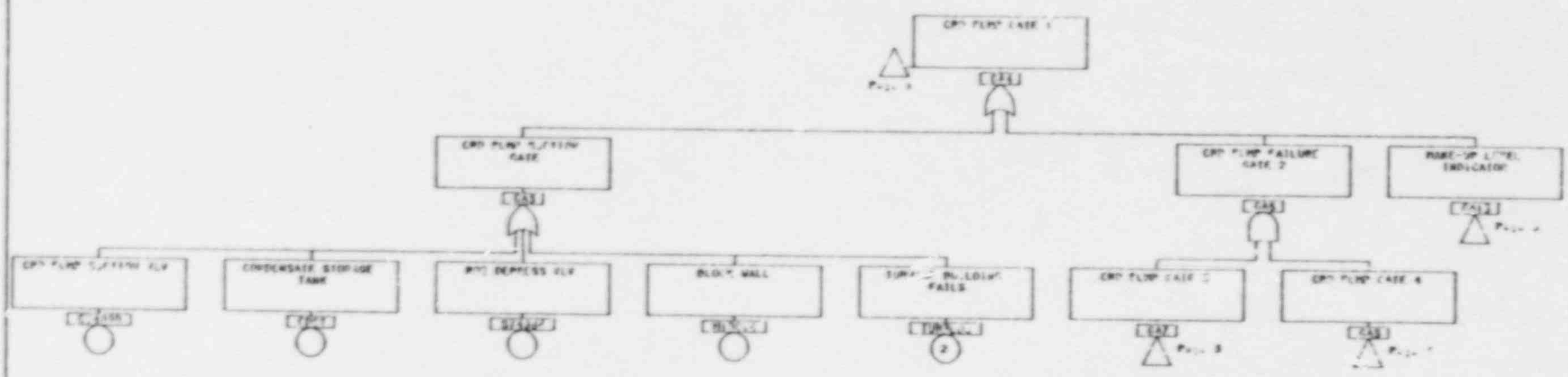
CONTROL ROD DRIVE
MAKE-UP

DRAWING NUMBER	DATE
Page 1	4/27/88



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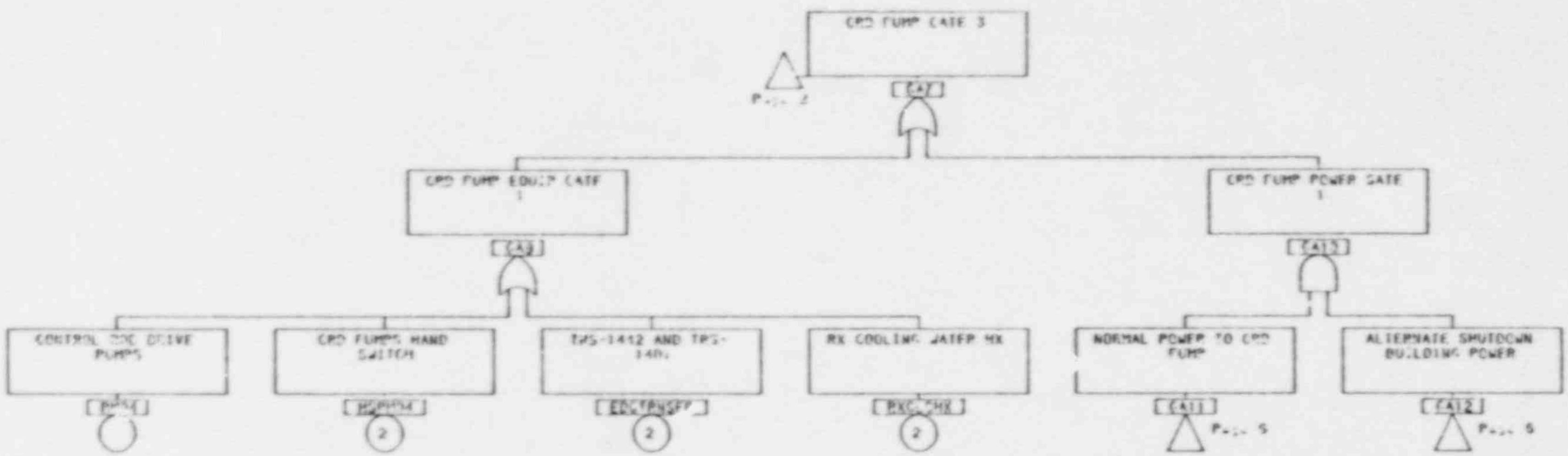
CONTROL ROD DRIVE
MAKE-UP

DRAWING NUMBER	DATE
Page 2	4/27/88



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CONTROL ROD DRIVE MAKE-UP

DRAWING NUMBER Page 3

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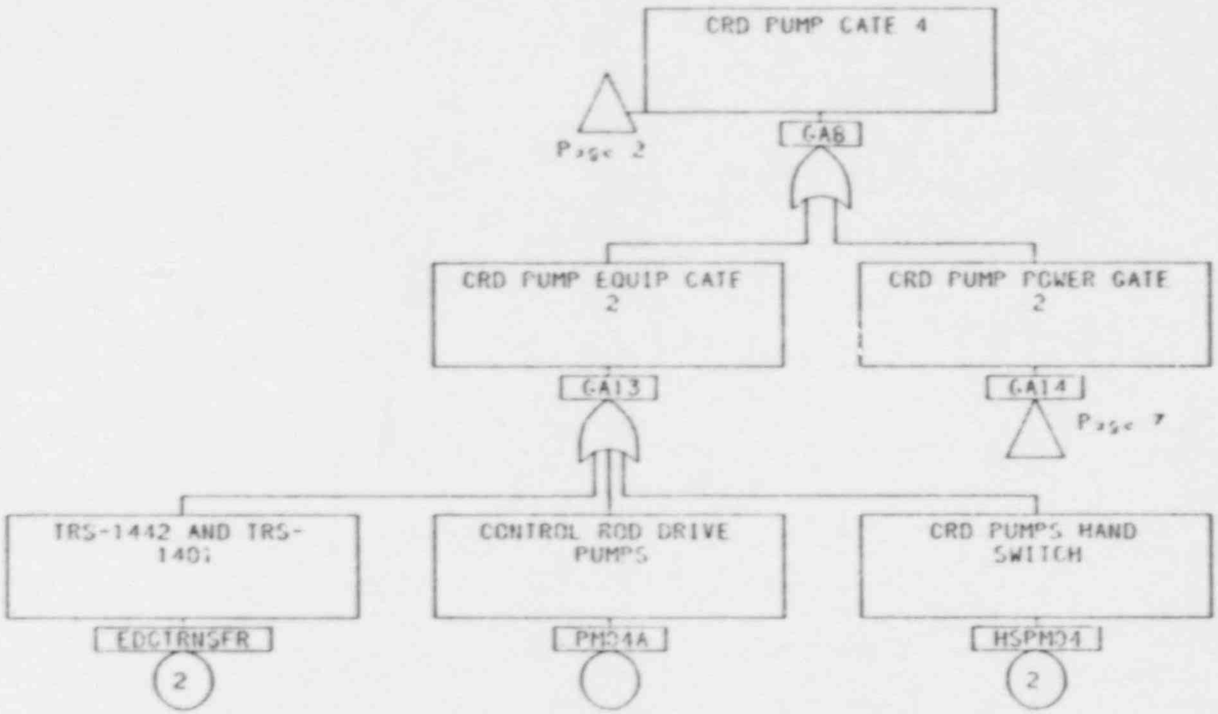

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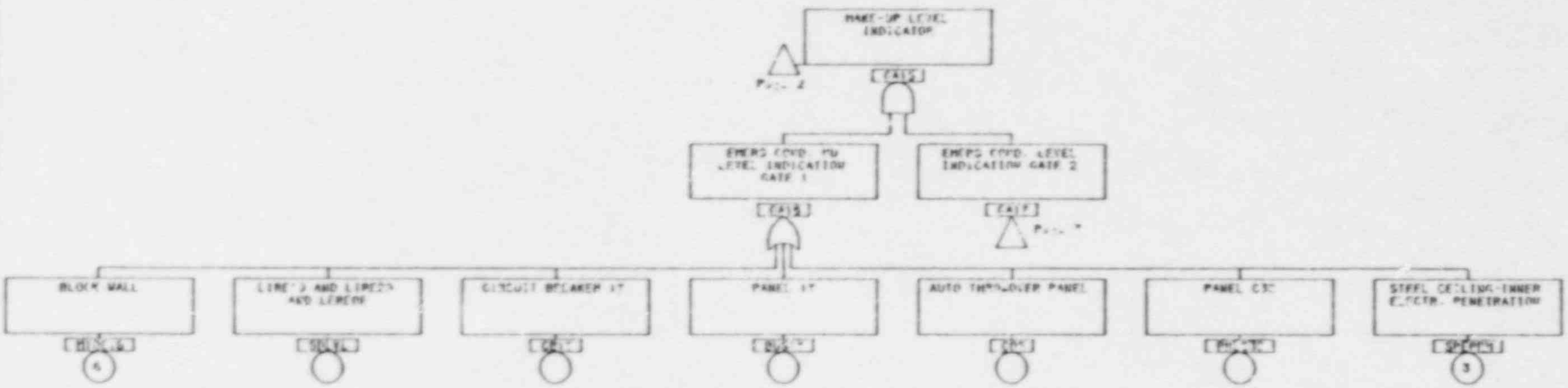
CONTROL ROD DRIVE
MAKE-UP

DRAWING NUMBER	DATE
Page 4	4/27/88



TITLE
CONTROL ROD DRIVE
MAKE-UP

DRAWING NUMBER	DATE
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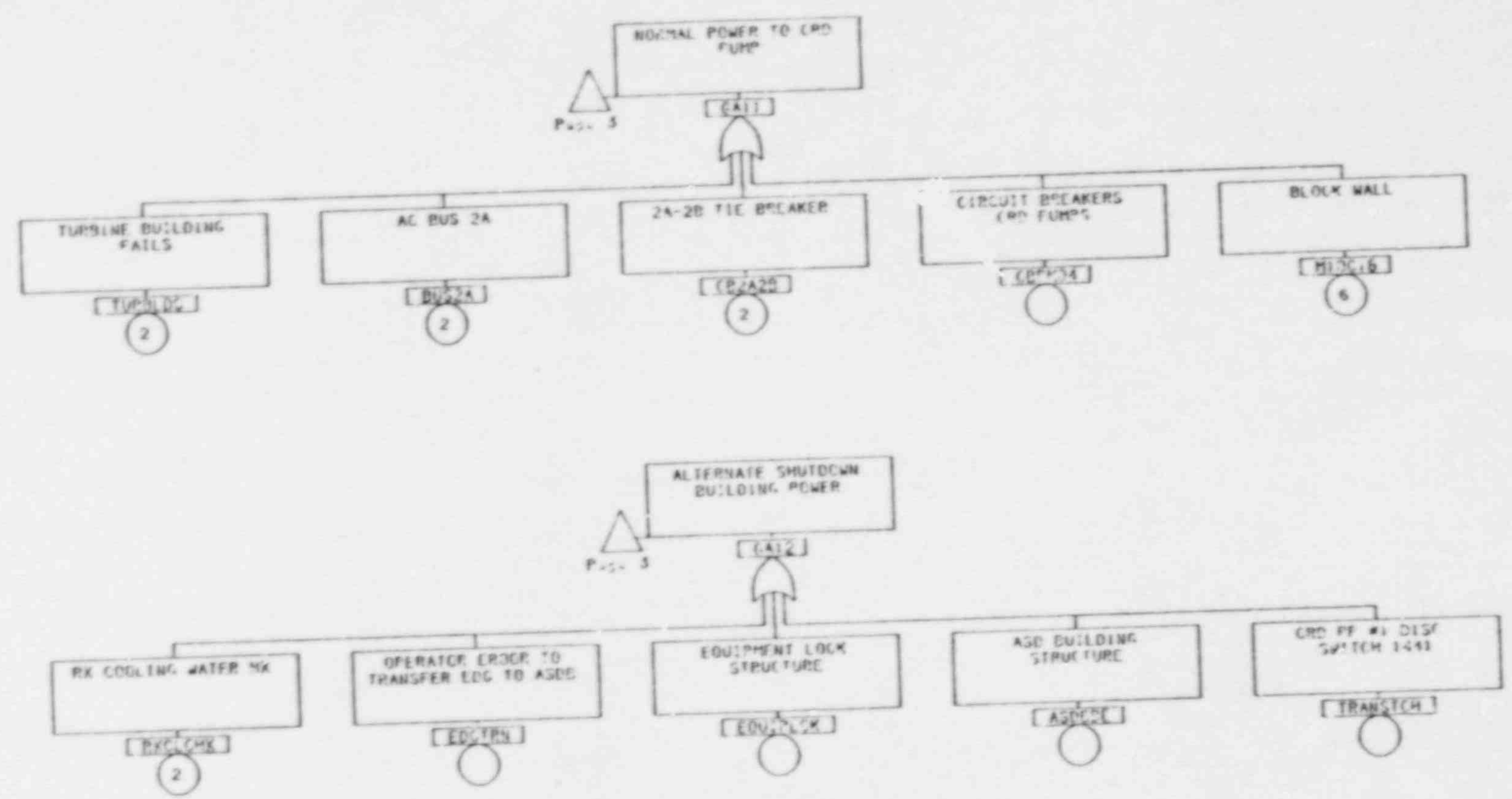
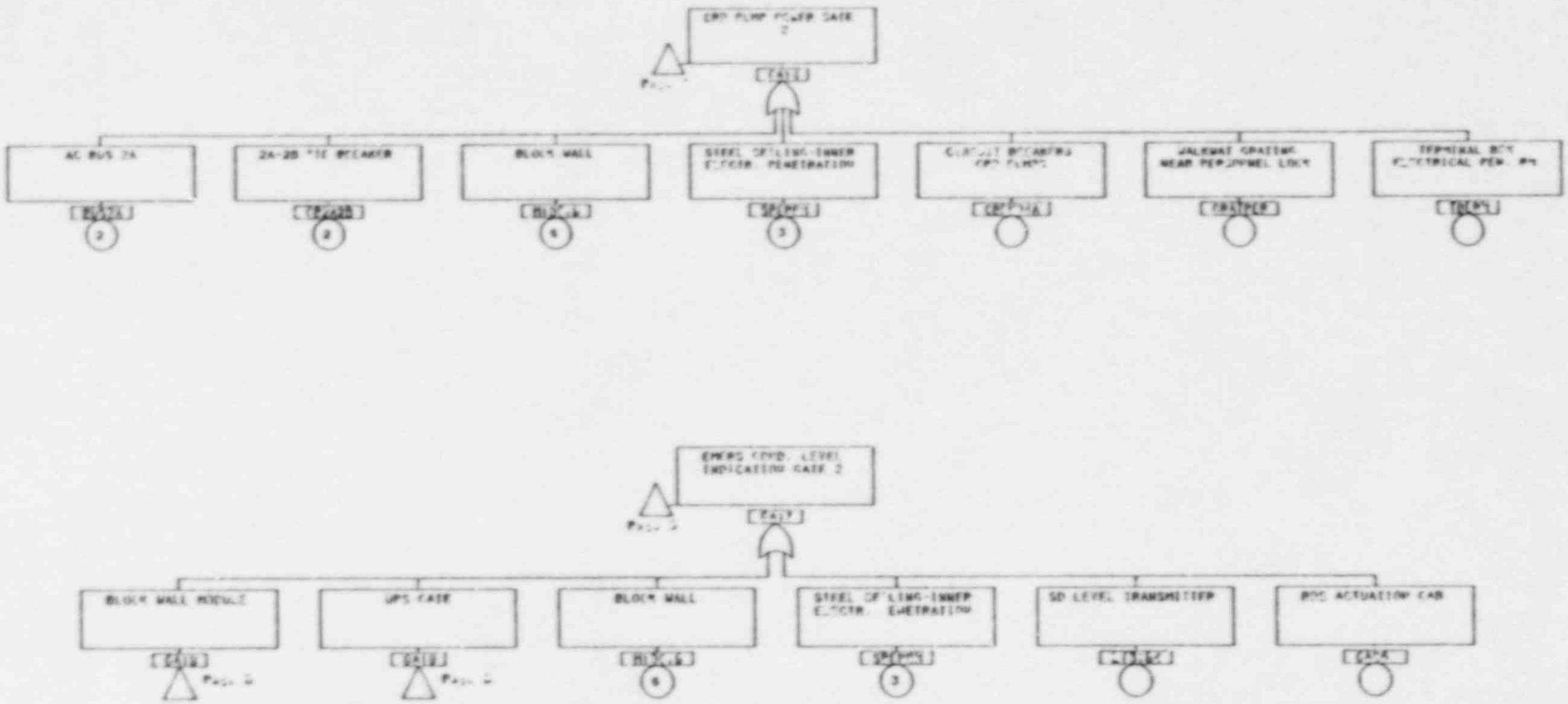


TITLE
CONTROL ROD DRIVE
MAKE-UP

DRAWING NUMBER	DATE
Page 6	4/27/88



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CONTROL ROD DRIVE
MAKE-UP

DRAWING NUMBER
Page 7

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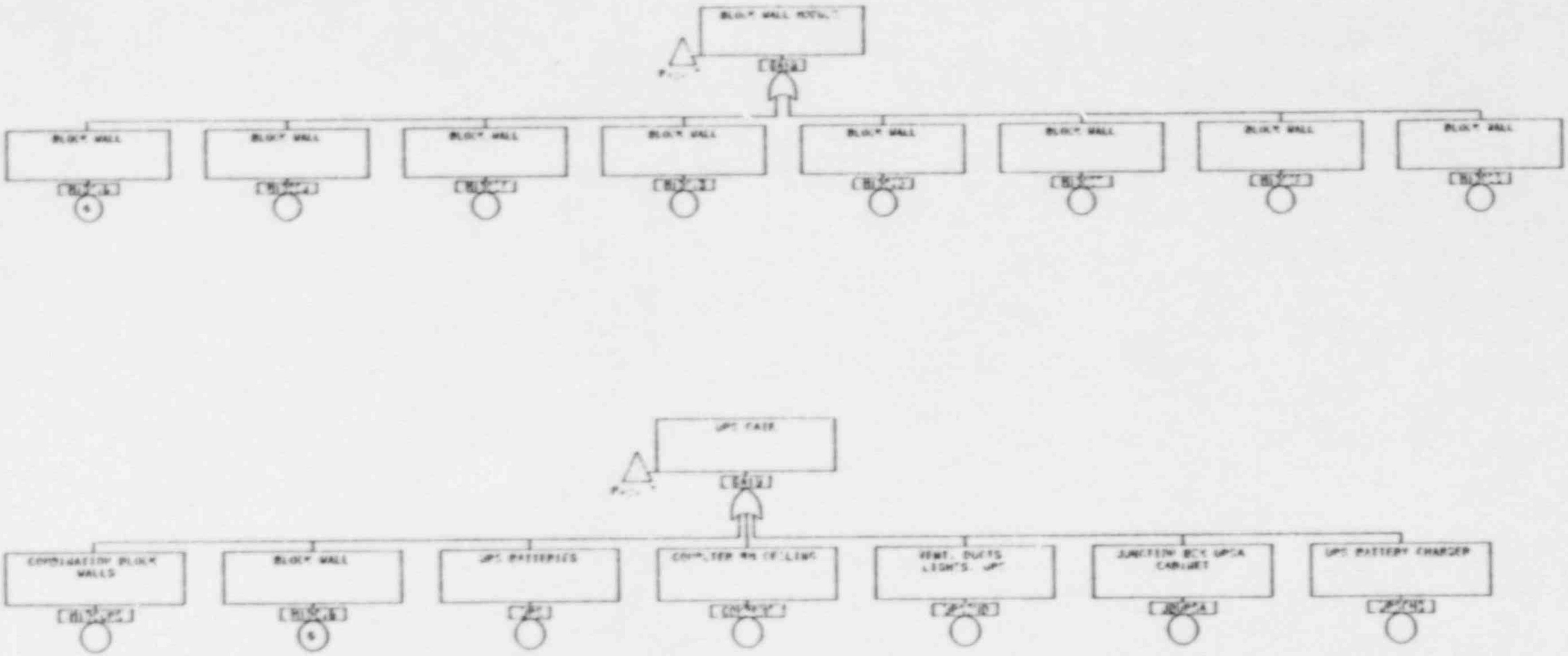

TITLE CONTROL ROD DRIVE MAKE-UP	
DRAWING NUMBER Page 8	DATE 4/27/88

MAIN STEAM LINE ISOLATION VALVE
FAULT TREES

1 Page

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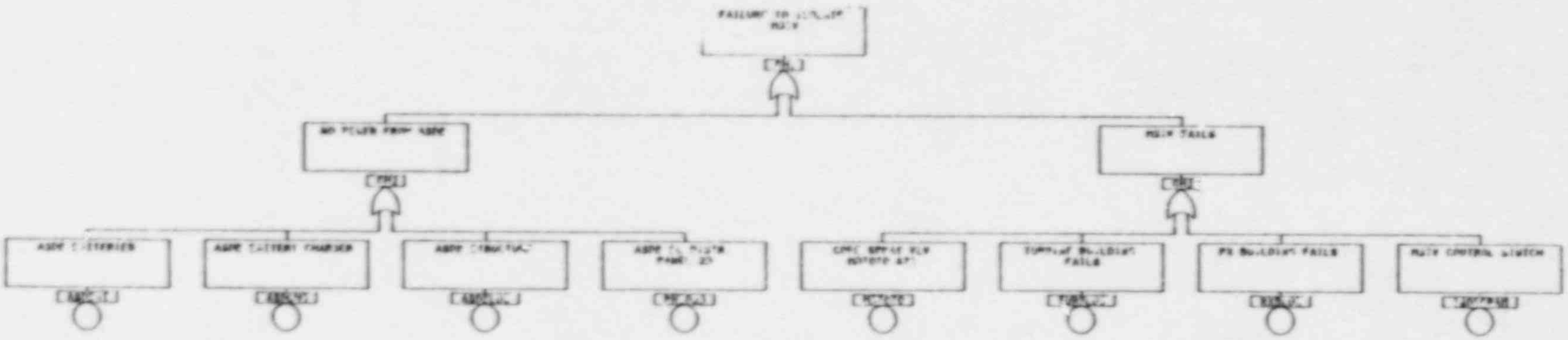

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MAIN STEAM LINE
ISOLATION VALVE

DRAWING NUMBER: Page 1

DATE: 4/27/88

EMERGENCY CONDENSER VALVES
FAULT TREES

8 Pages

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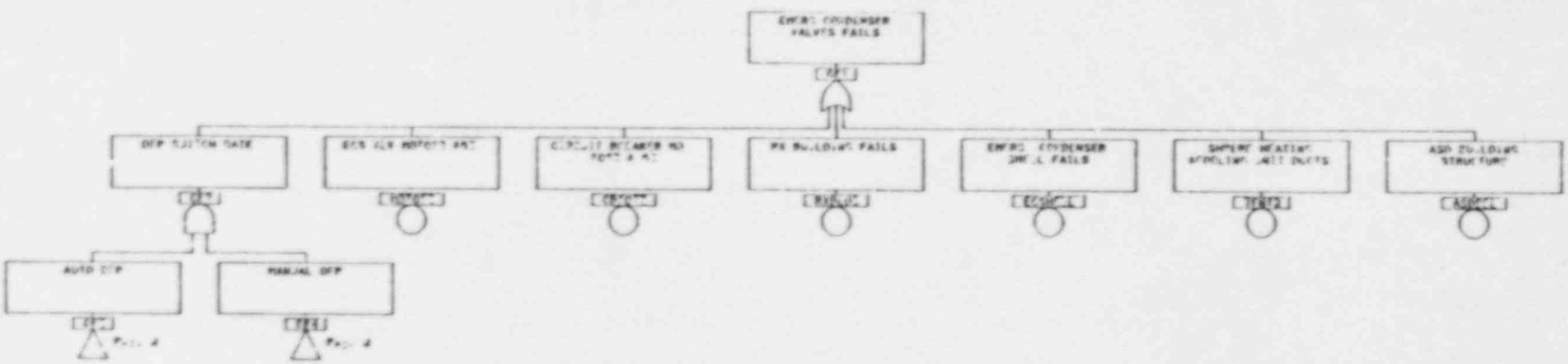

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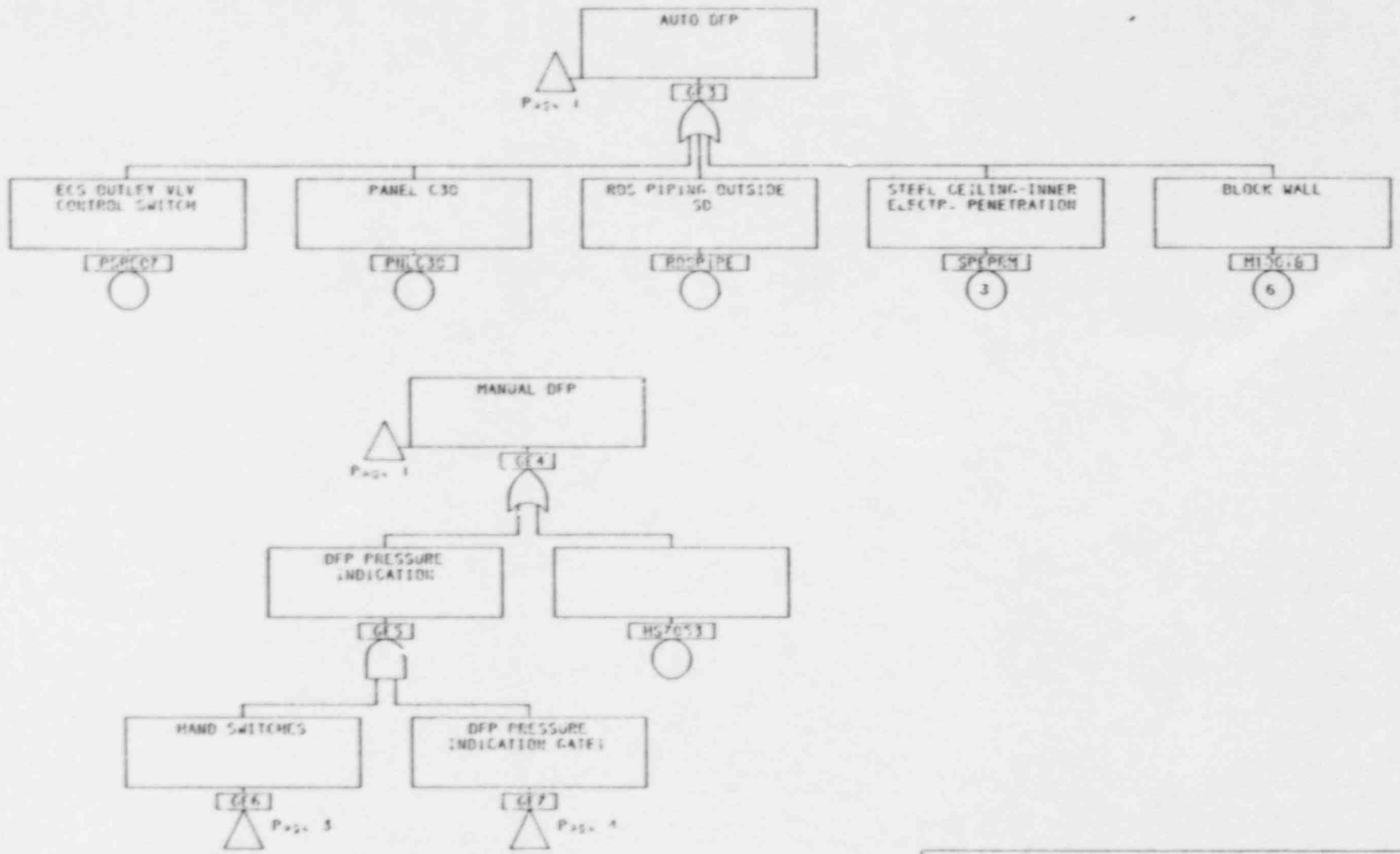
DRAWING NUMBER	DATE
Page 1	4/27/88



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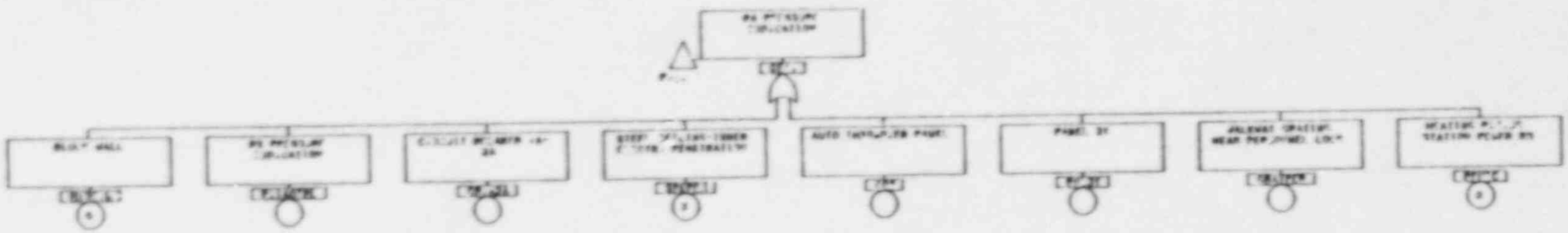
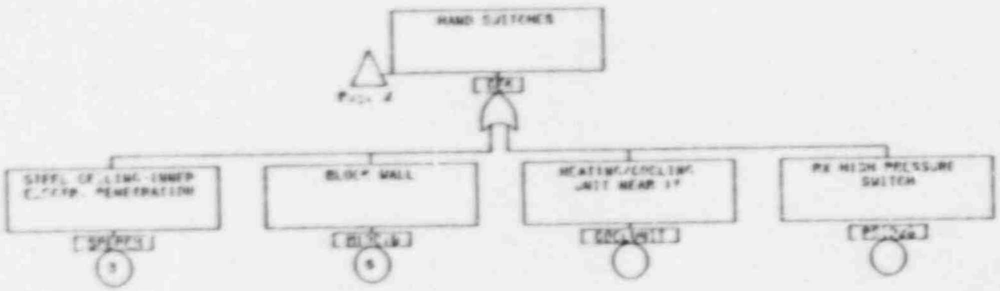
DRAWING NUMBER	DATE
Page 2	4/27/88





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EMERGENCY CONDENSER VALVES

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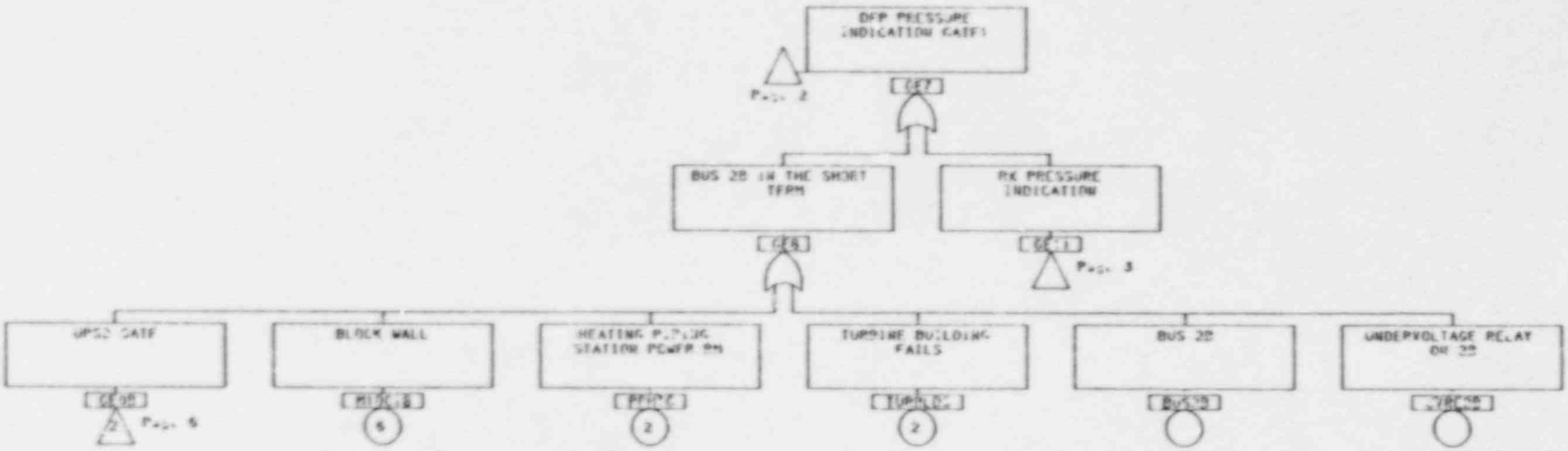
Page 3	4/27/88



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EMERGENCY CONDENSER VALVES

DRAWING NUMBER	DATE
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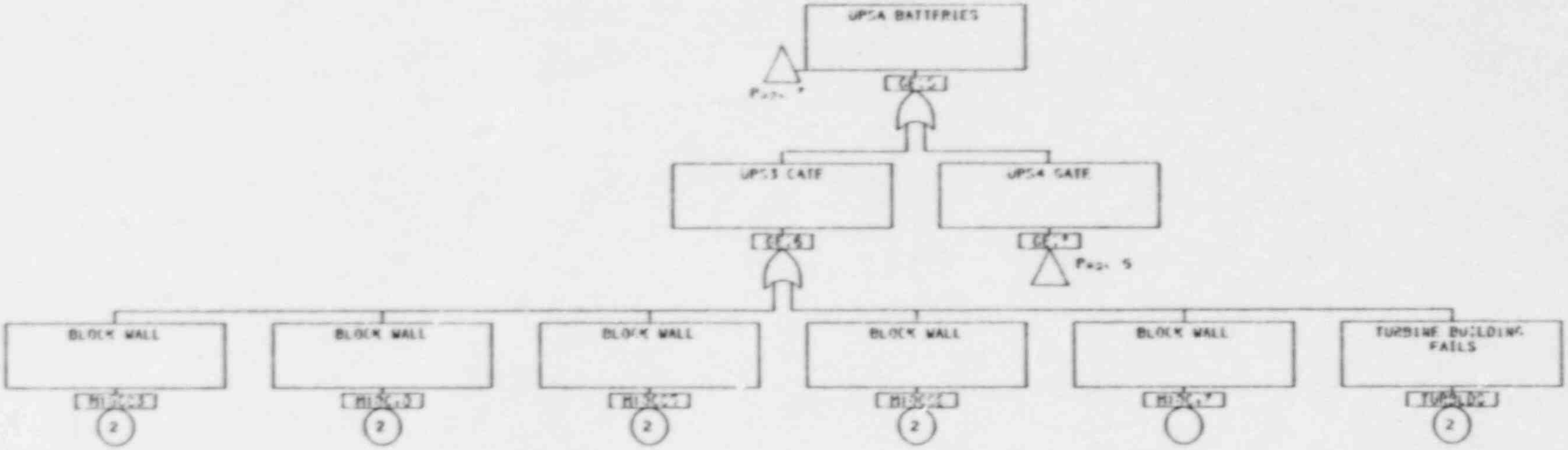

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EMERGENCY CONDENSER VALVES

DRAWING NUMBER	DATE
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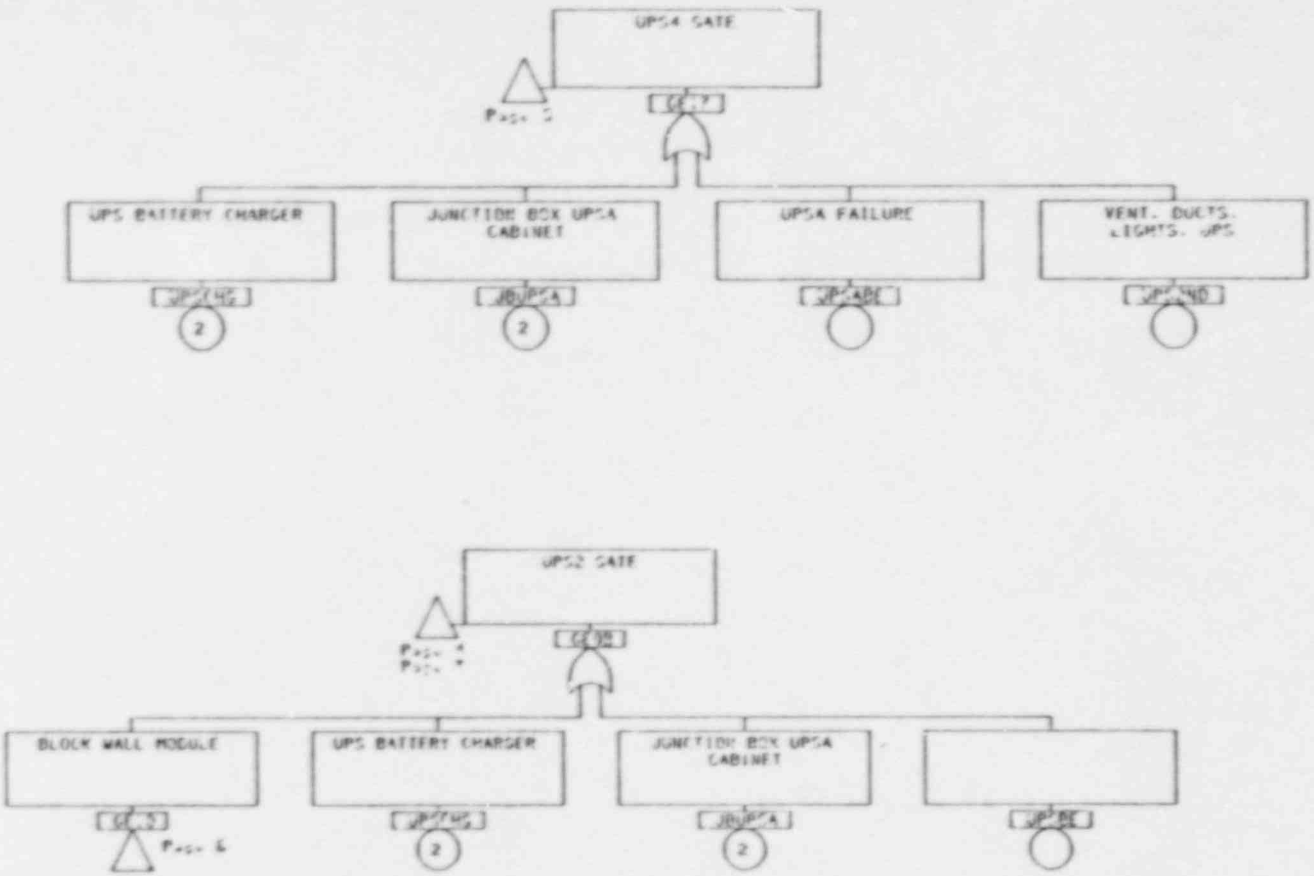

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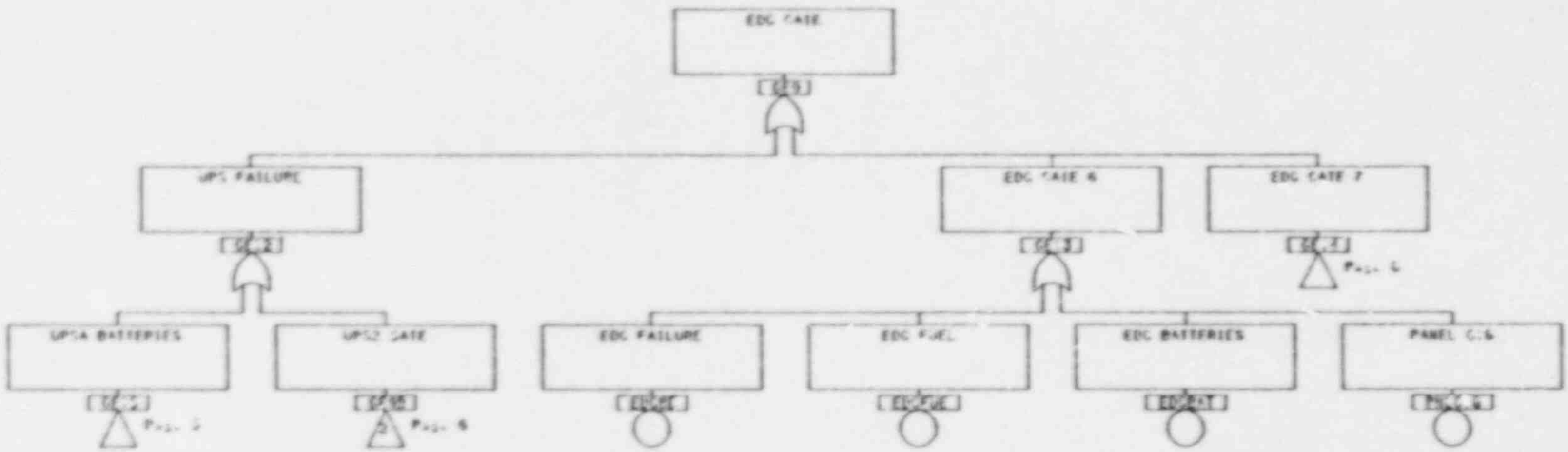

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EMERGENCY CONDENSER VALVES

DRAWING NUMBER	DATE
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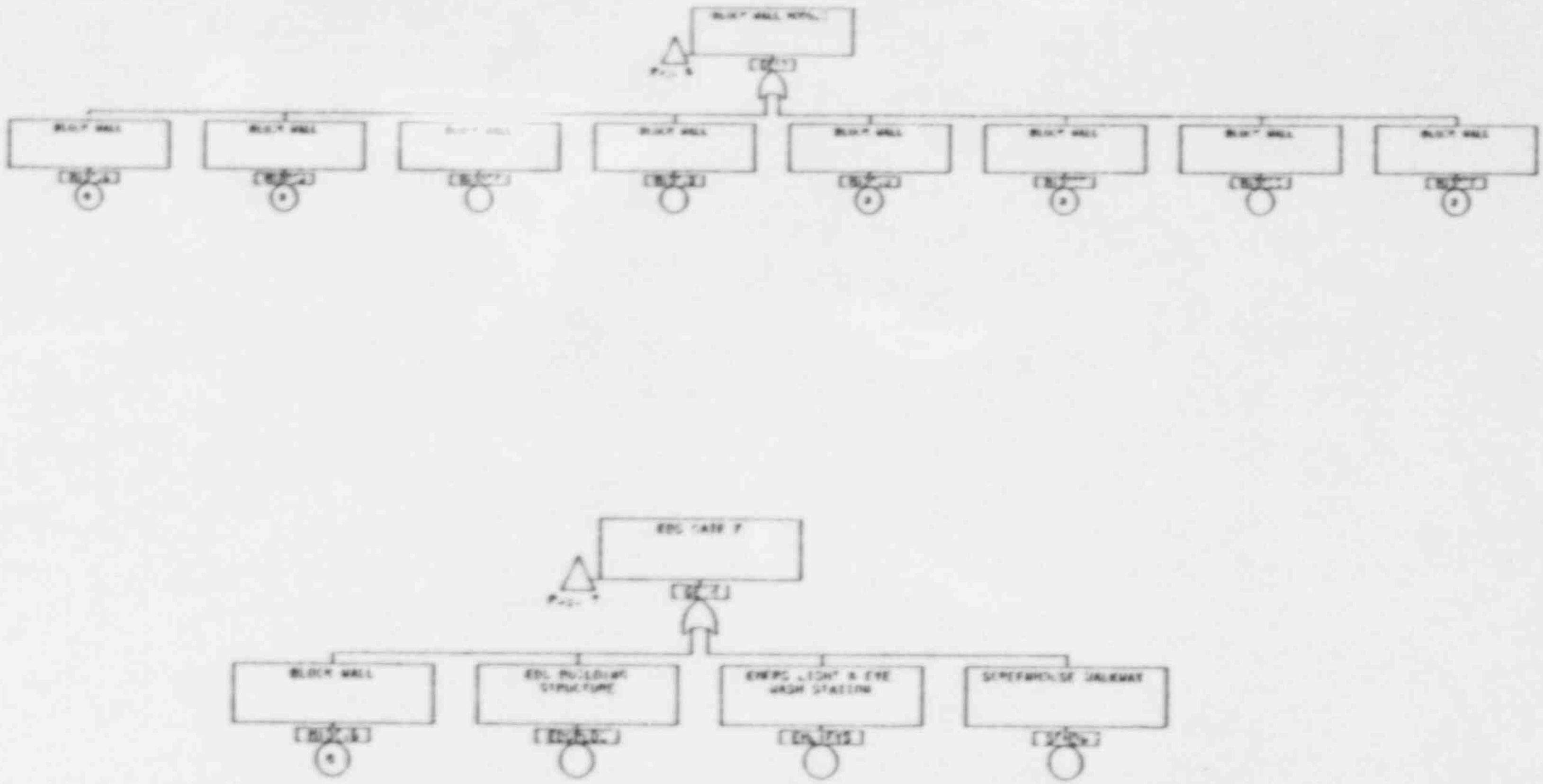

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EMERGENCY CONDENSER
VALVES

DRAWING NUMBER
Page 8

DATE
4/27/88

CORE SPRAY W/O AC POWER
FAULT TREES

8 Pages

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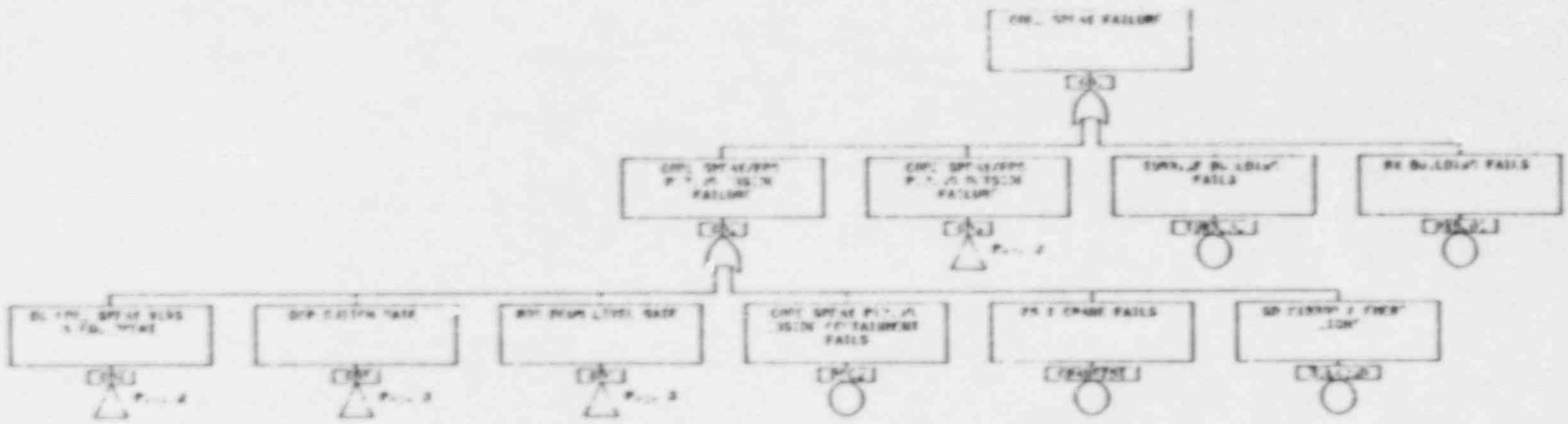

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CORE SPRAY W/O AC POWER

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Page 1

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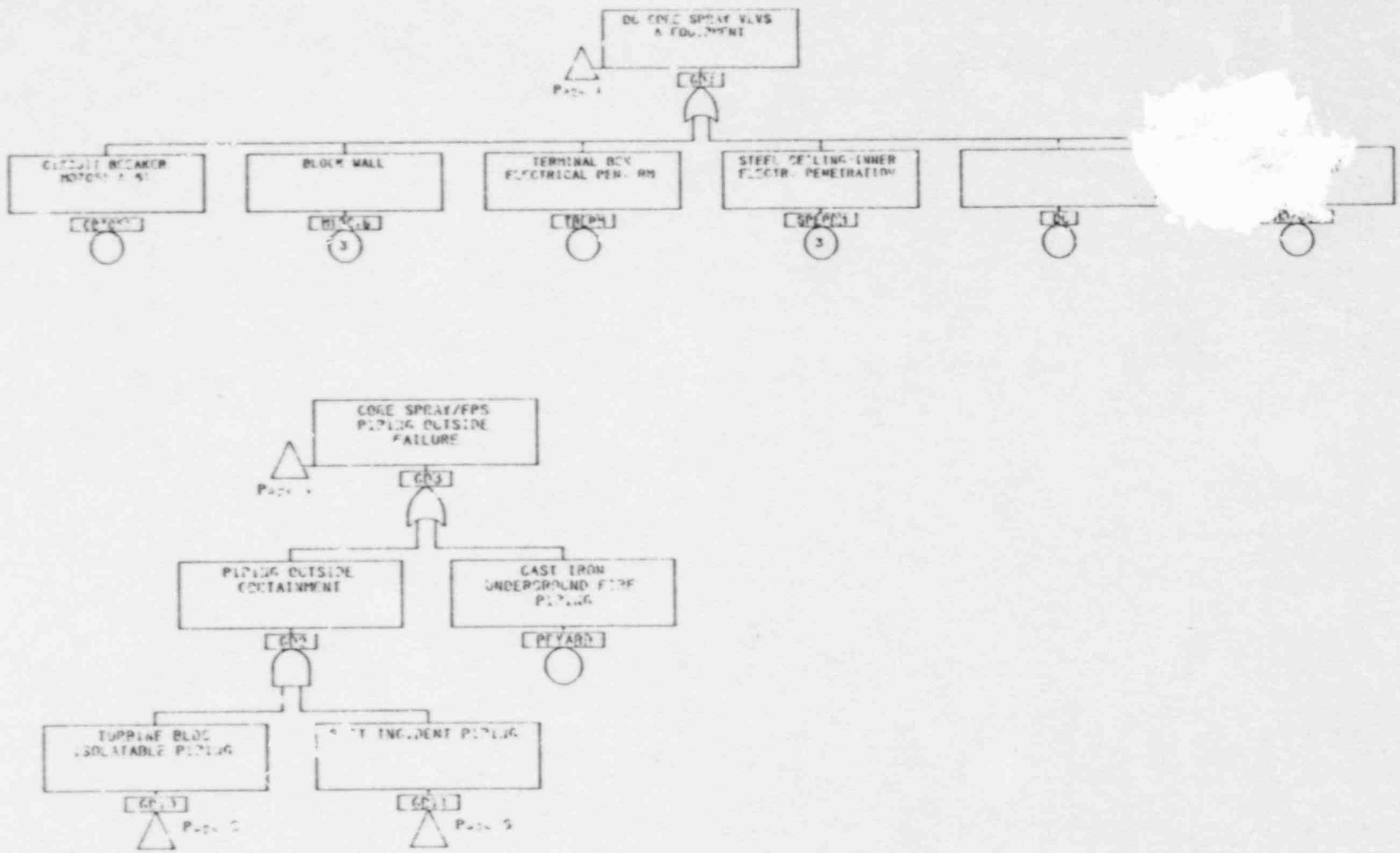

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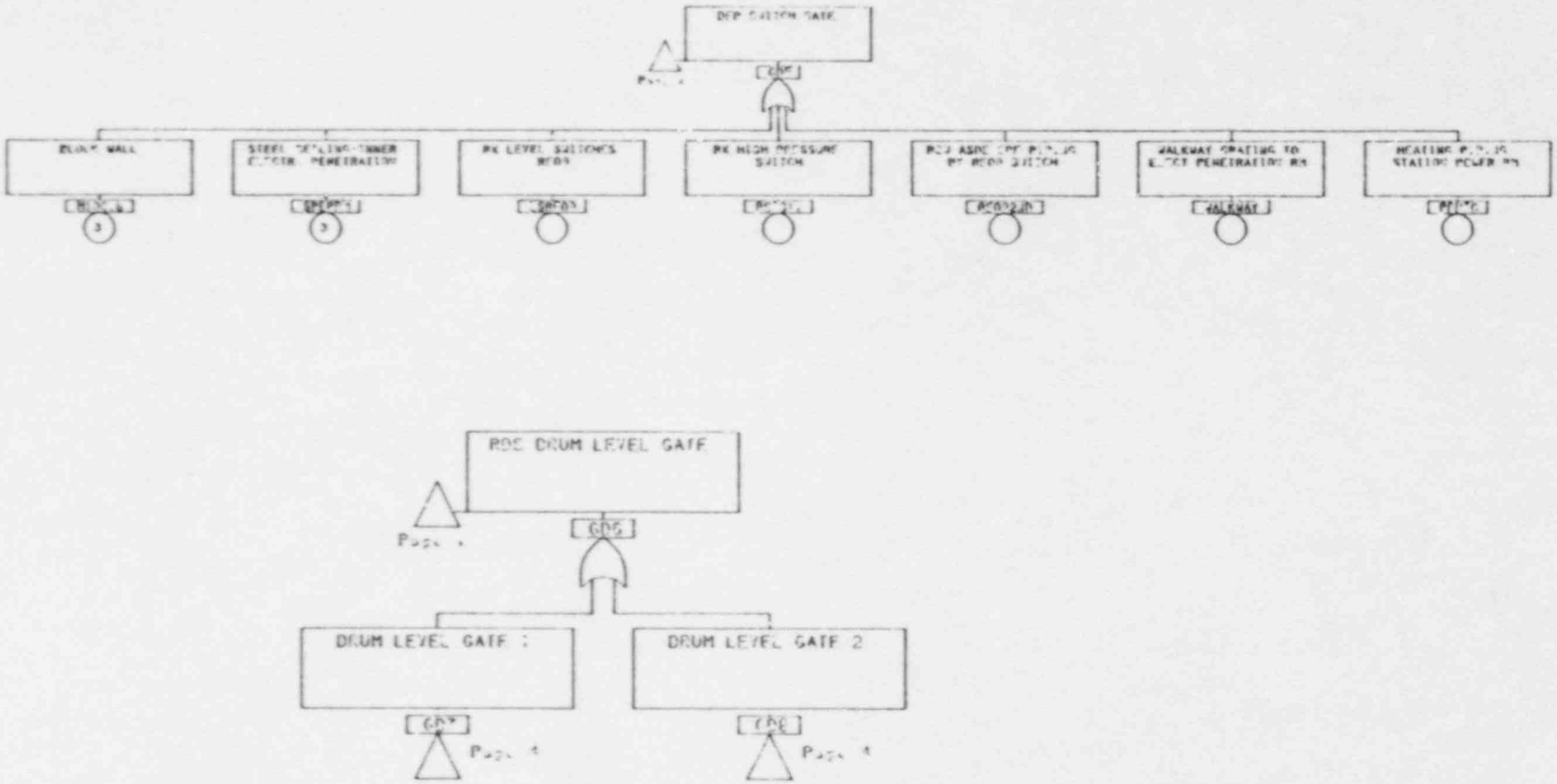
DRAWING NUMBER Page 2	DATE 4/27/88



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CORE SPRAY W/O AC POWER

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Page 3

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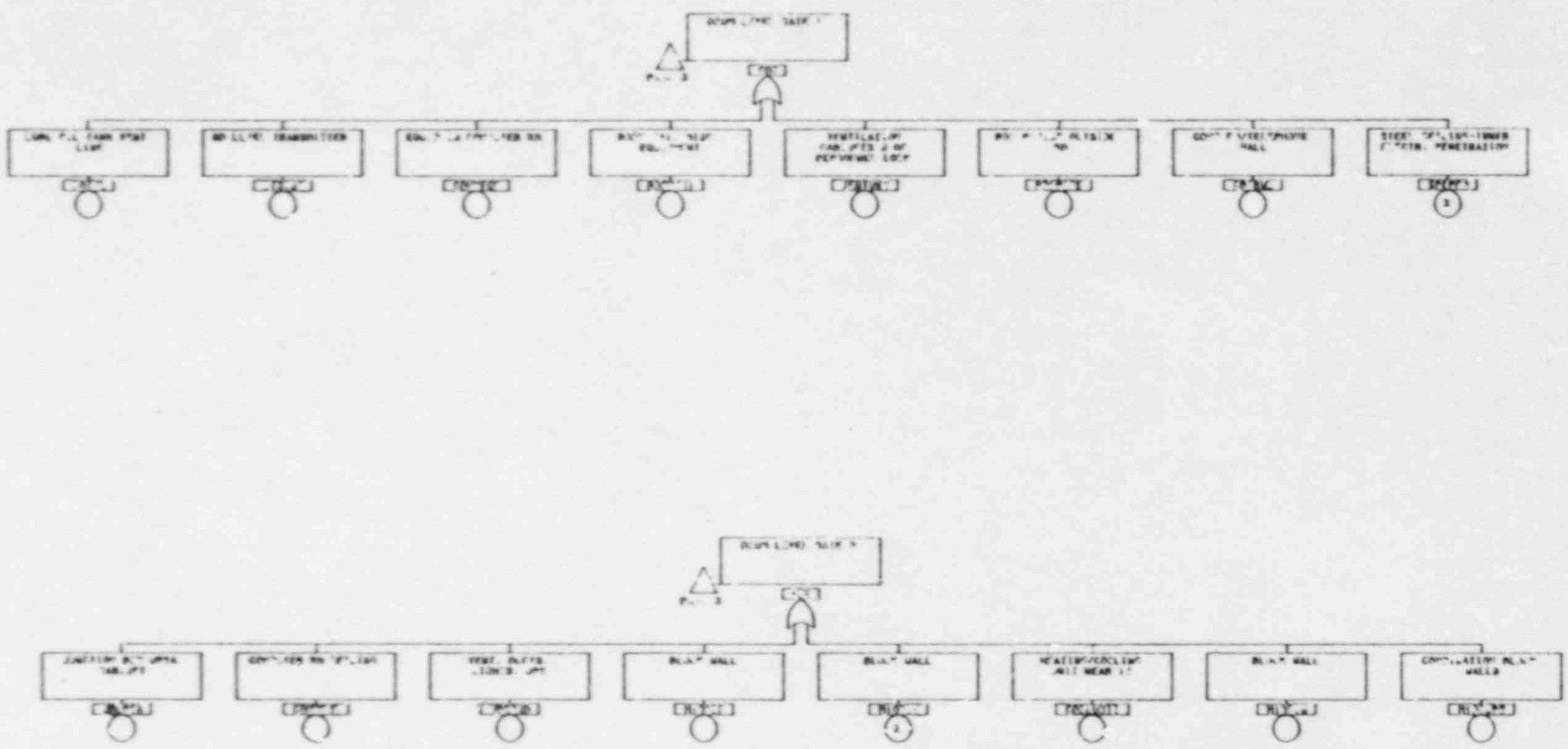

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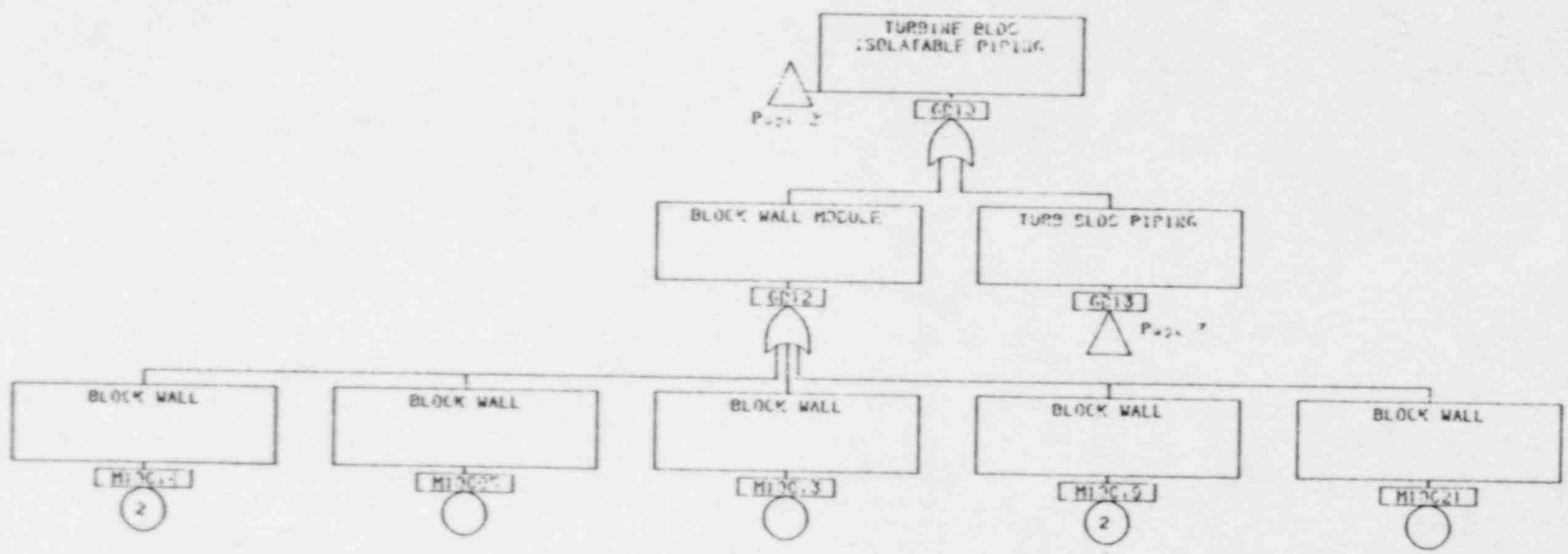
DRAWING NUMBER Page 4	DATE 4/27/88



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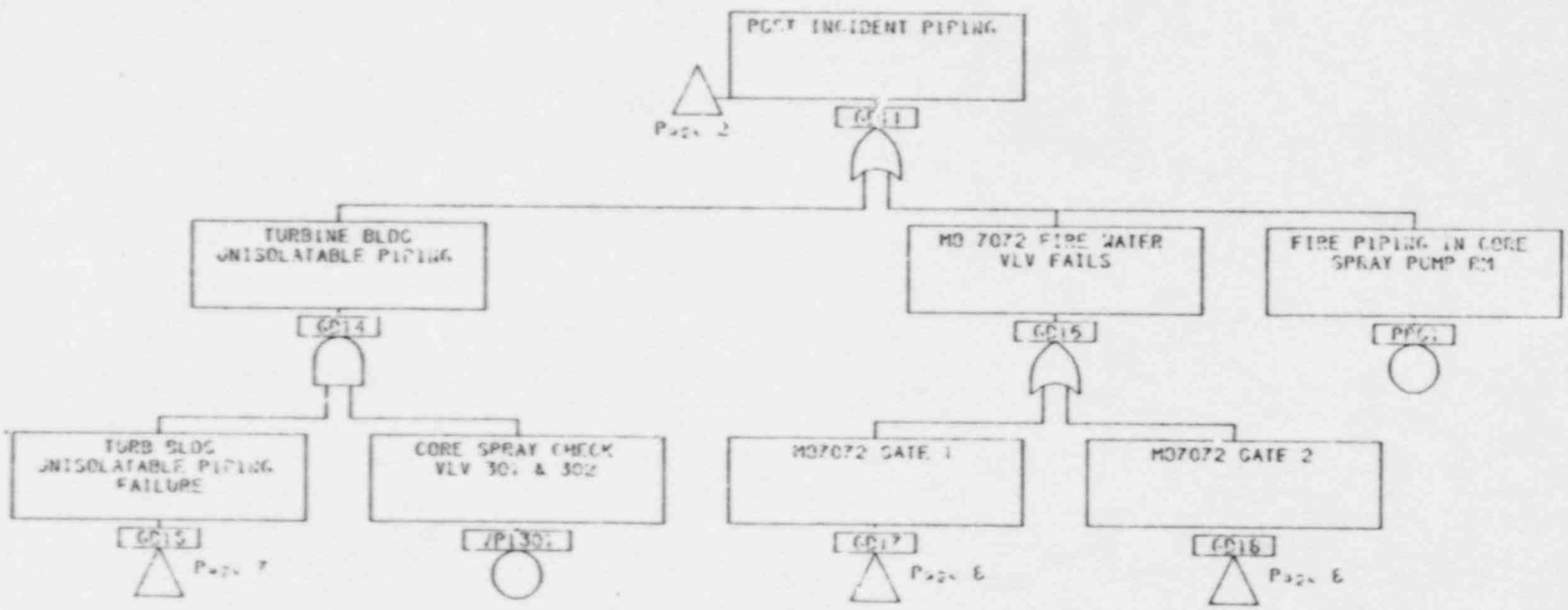
CORE SPRAY W/O AC POWER

DRAWING NUMBER	DATE
Page 5	4/27/88



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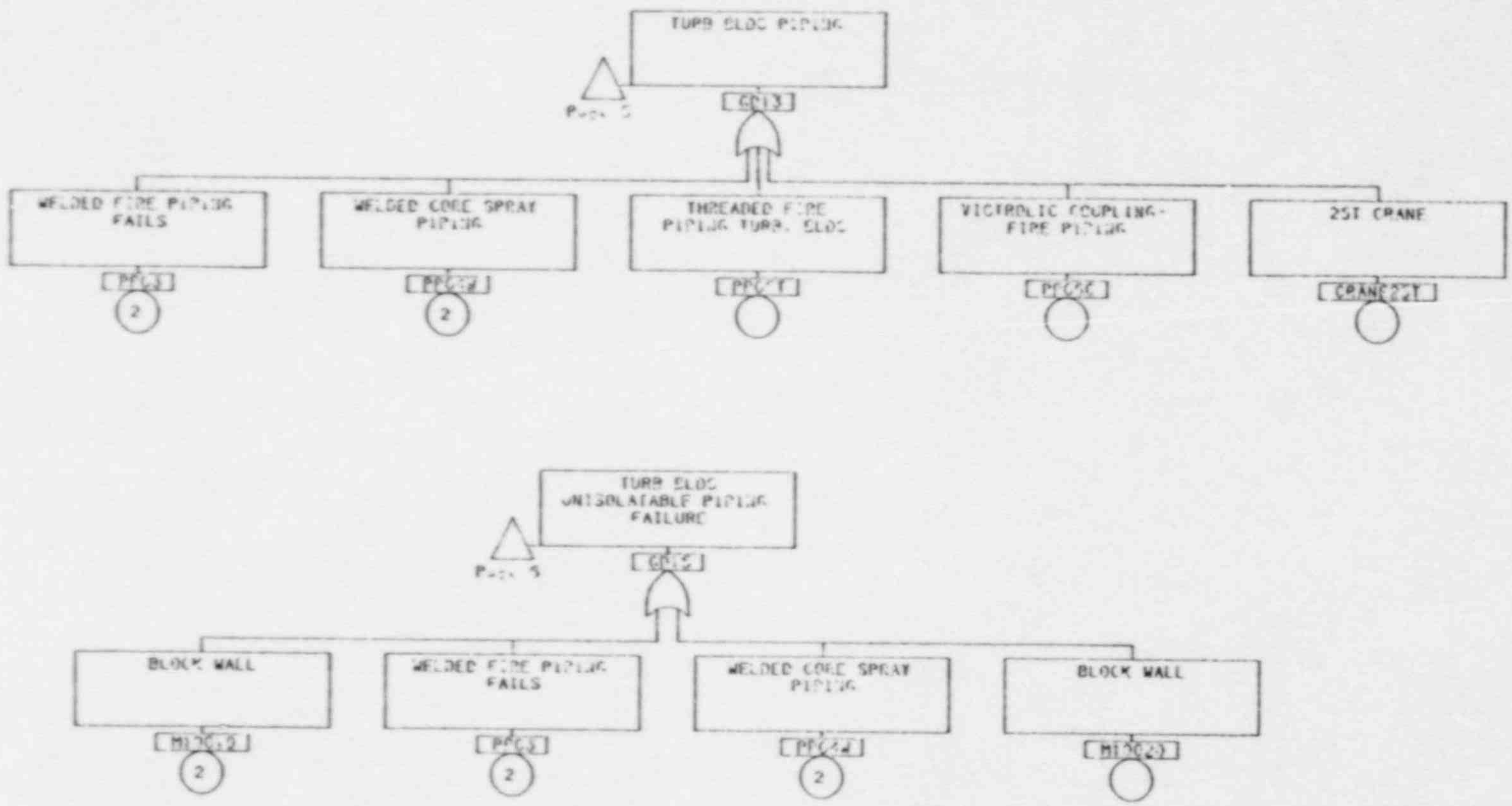
DRAWING NUMBER: Page 6	DATE: 4/27/88



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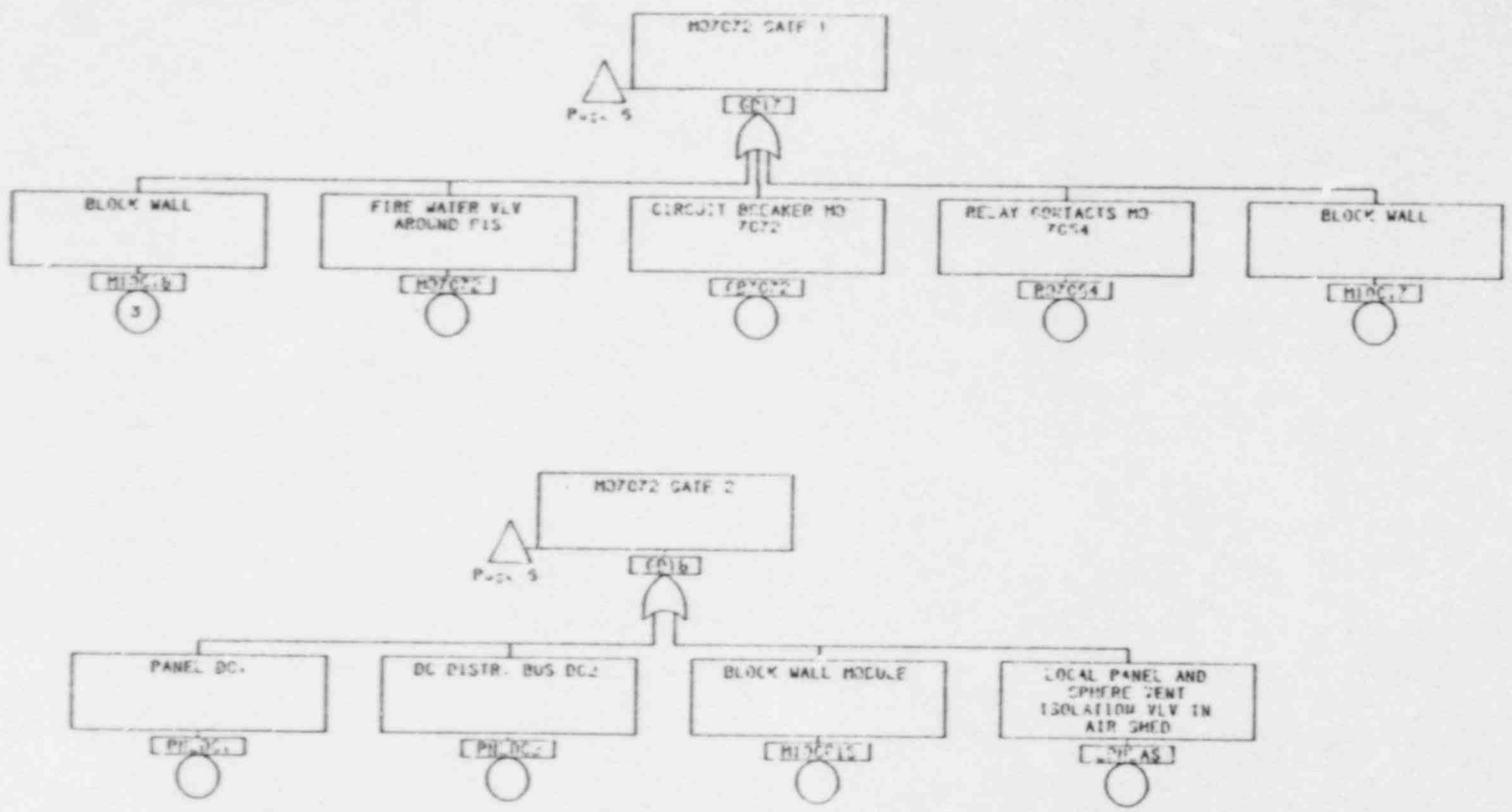
CORE SPRAY W/O AC POWER

DRAWING NUMBER	DATE
Page 7	4/27/88



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CORE SPRAY W/O AC POWER	
DRAWING NUMBER: Page 8	DATE: 4/27/88

POST INCIDENT FAILURE - LOSP
FAULT TREES

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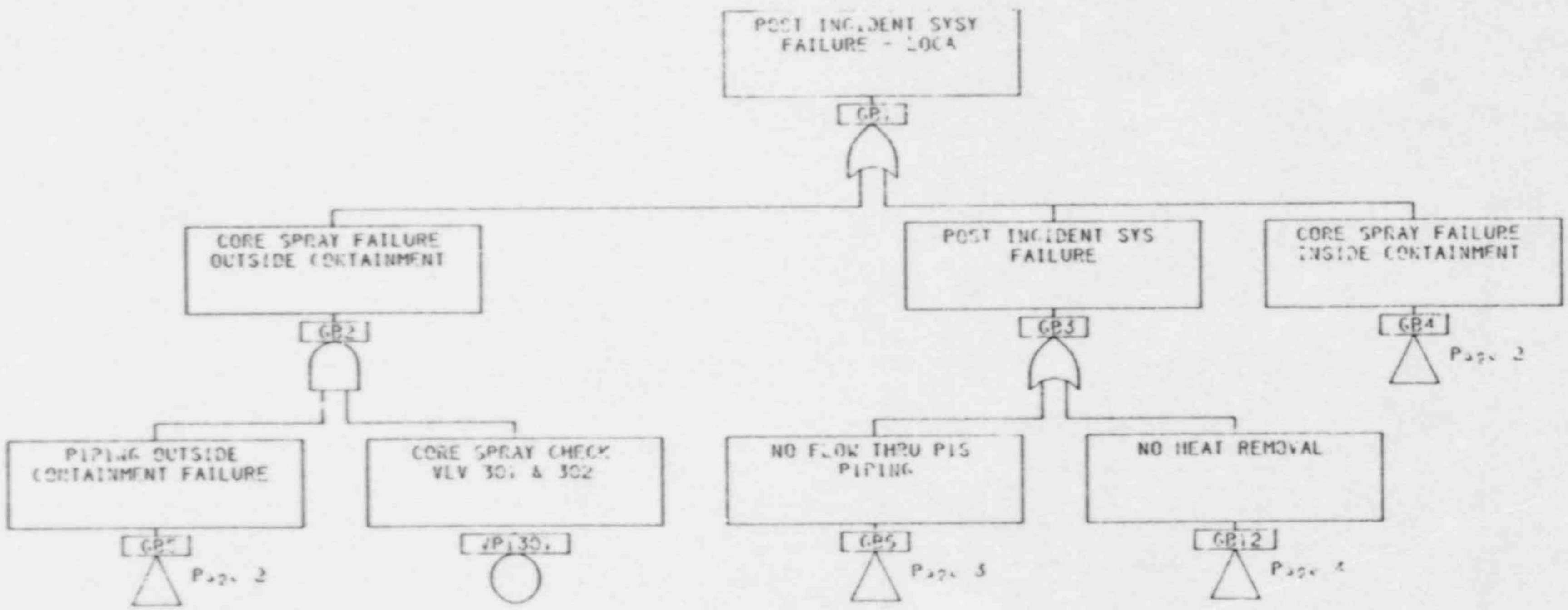

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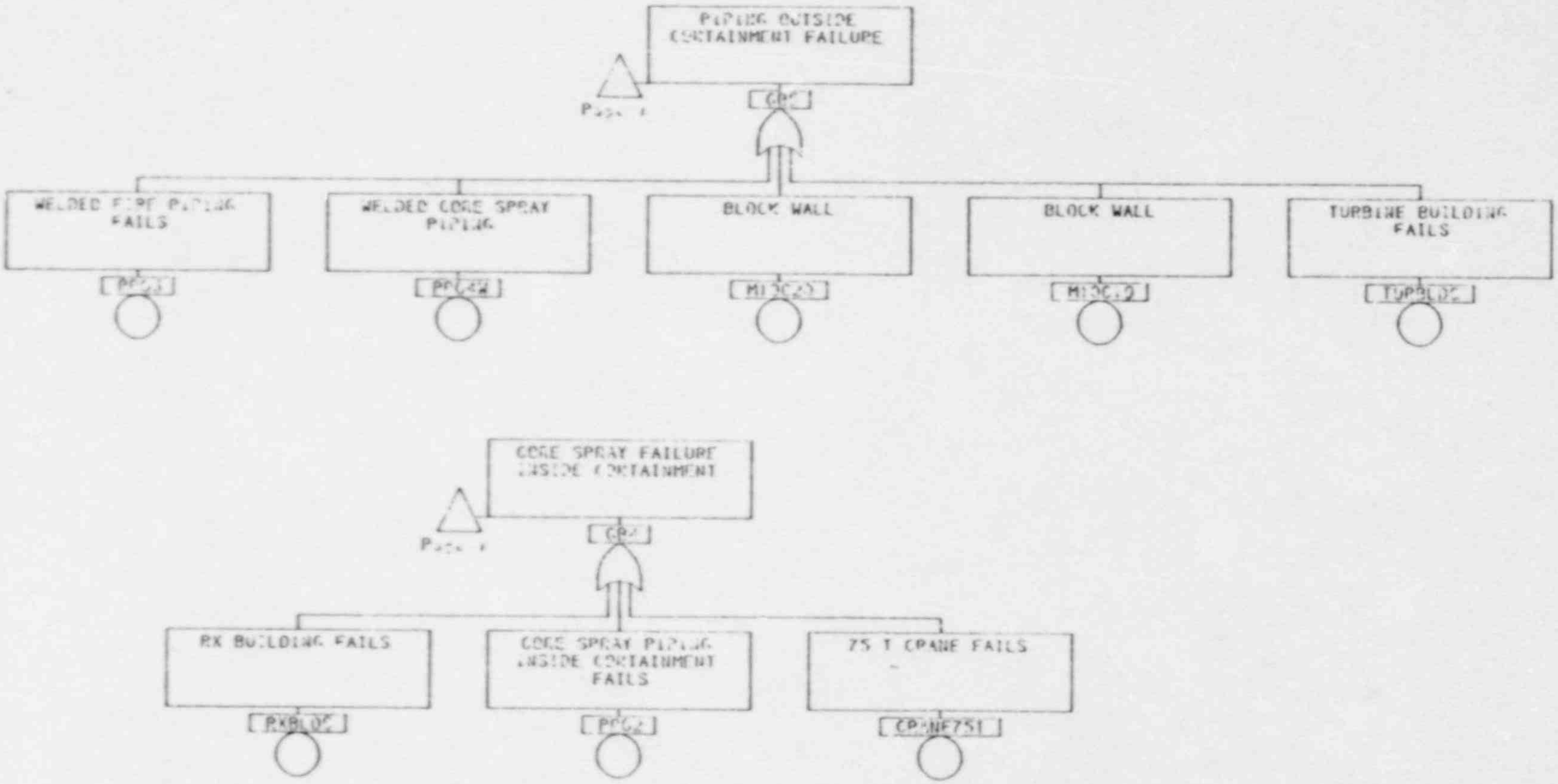
POST INCIDENT FAILURE-LOSP

DRAWING NUMBER	DATE
Page 1	4/28/88



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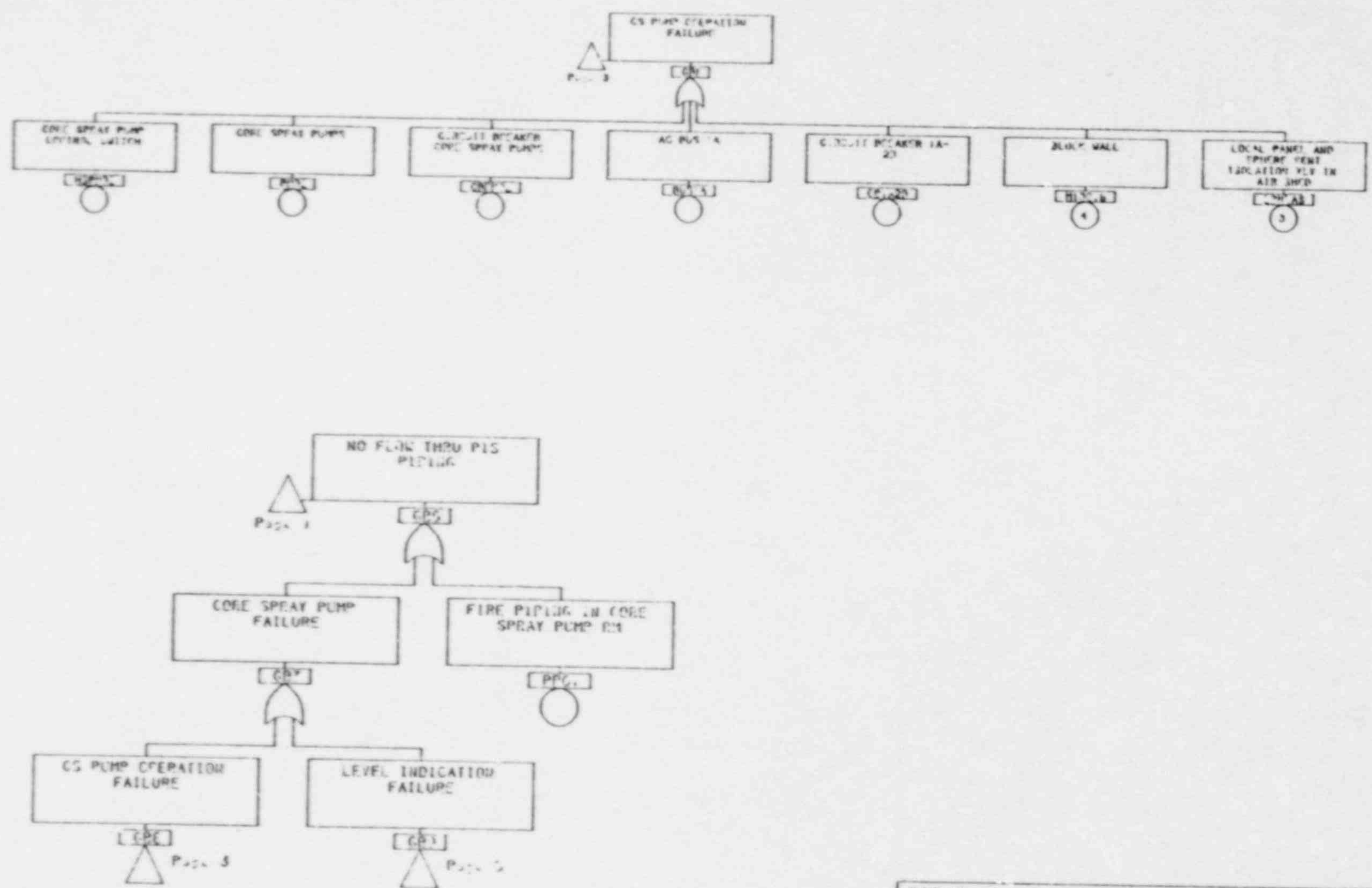
DRAWING NUMBER	DATE
Page 2	4/28/88



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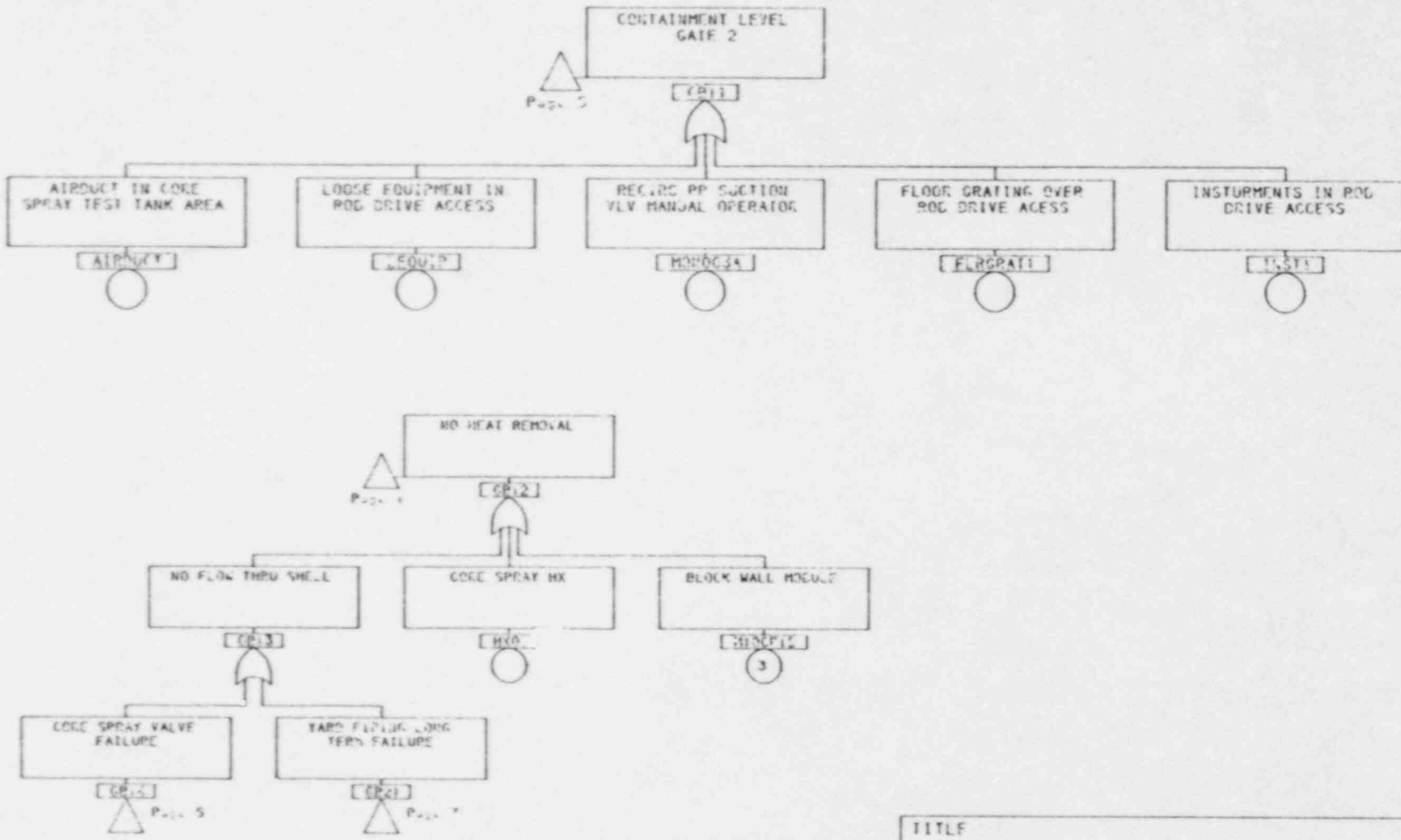
POST INCIDENT FAILURE-LOSP	
DRAWING NUMBER	DATE
Page 3	4/28/88



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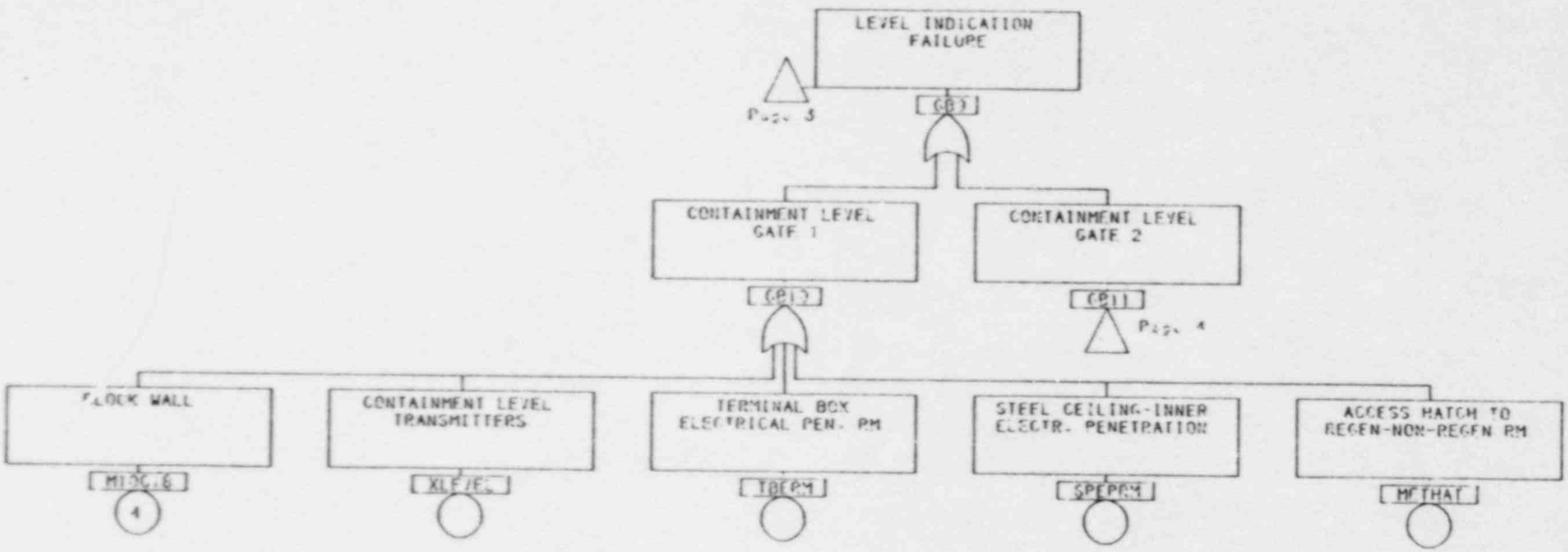
DRAWING NUMBER: Page 4	DATE: 4/28/88



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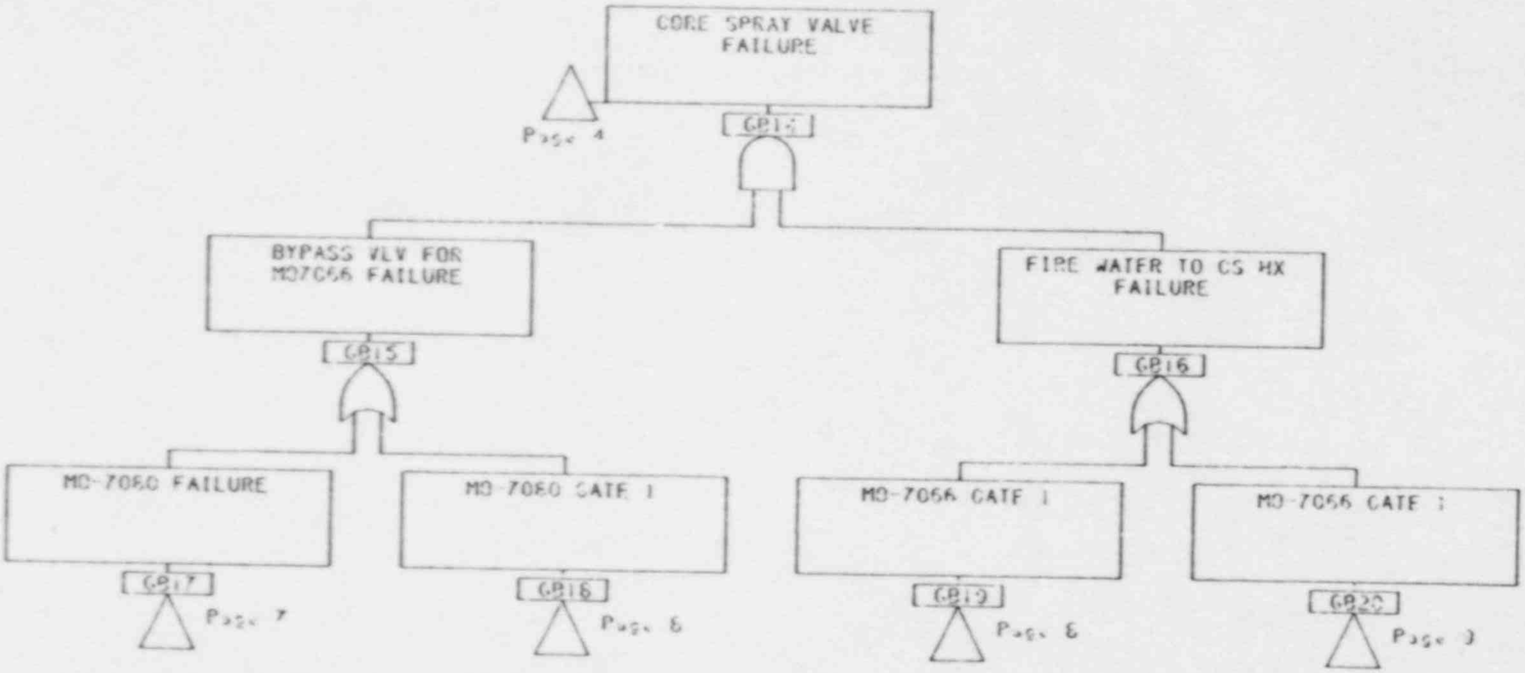

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POST INCIDENT
FAILURE-LOSP

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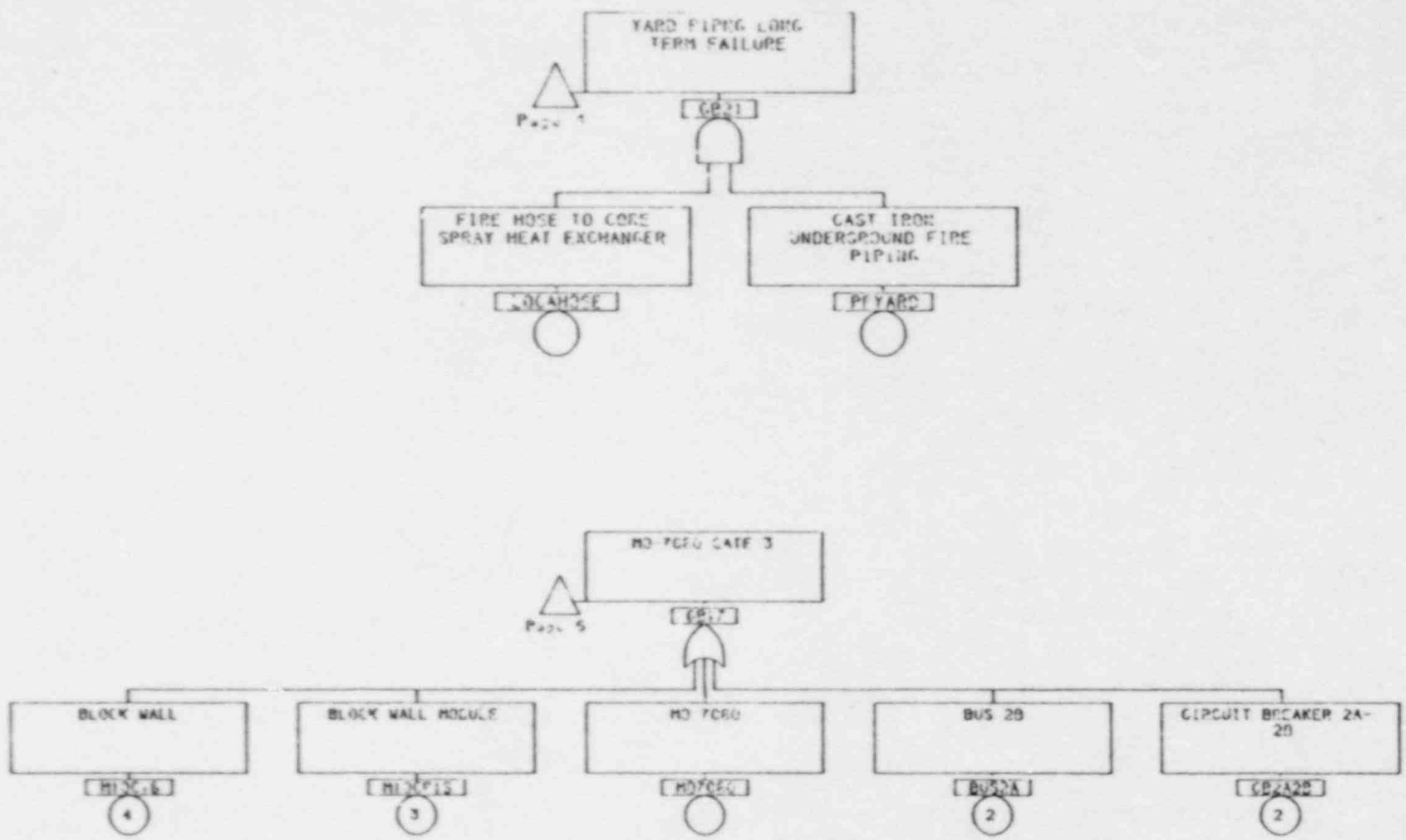

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POST INCIDENT FAILURE-LOSP

DRAWING NUMBER	DATE
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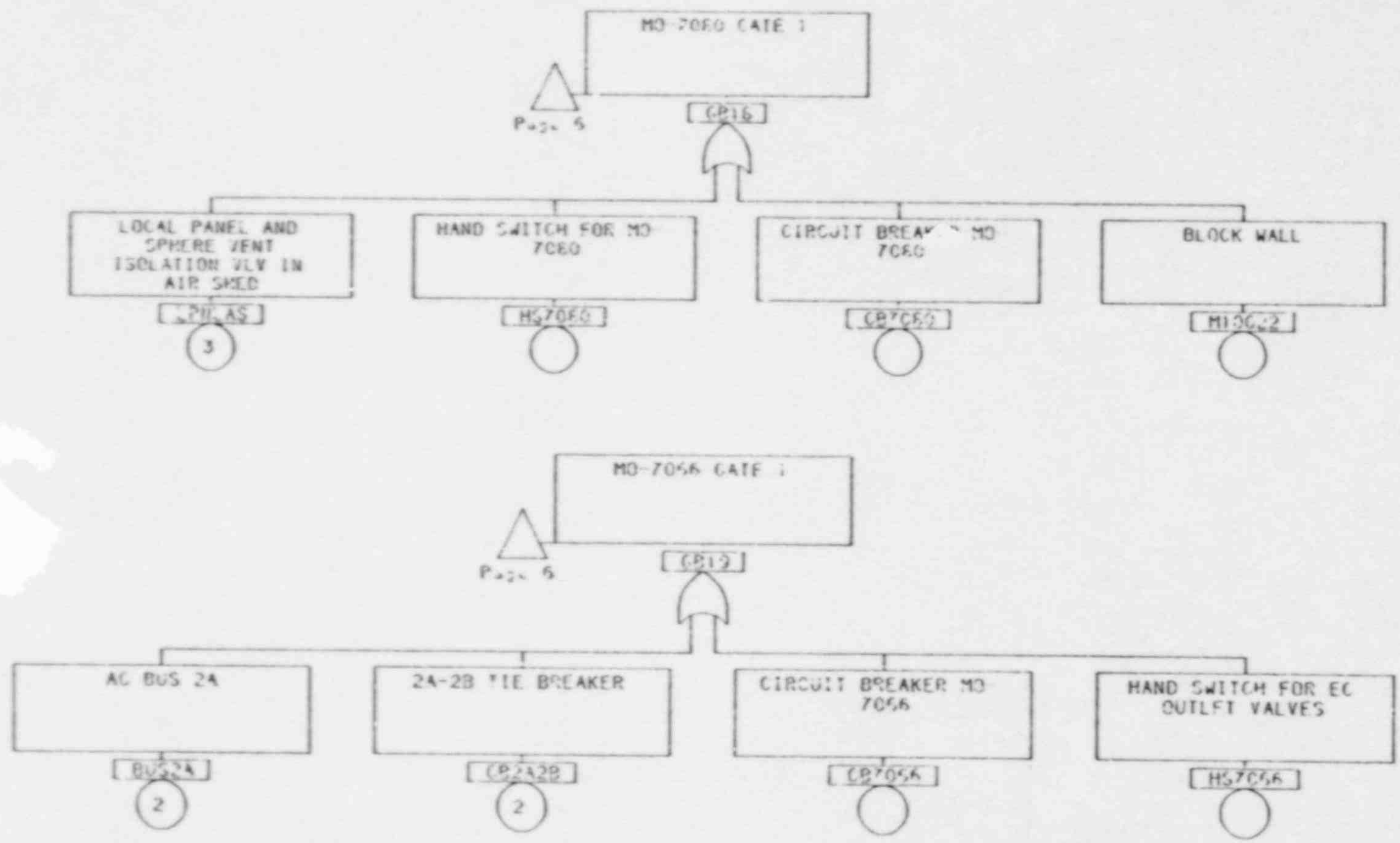

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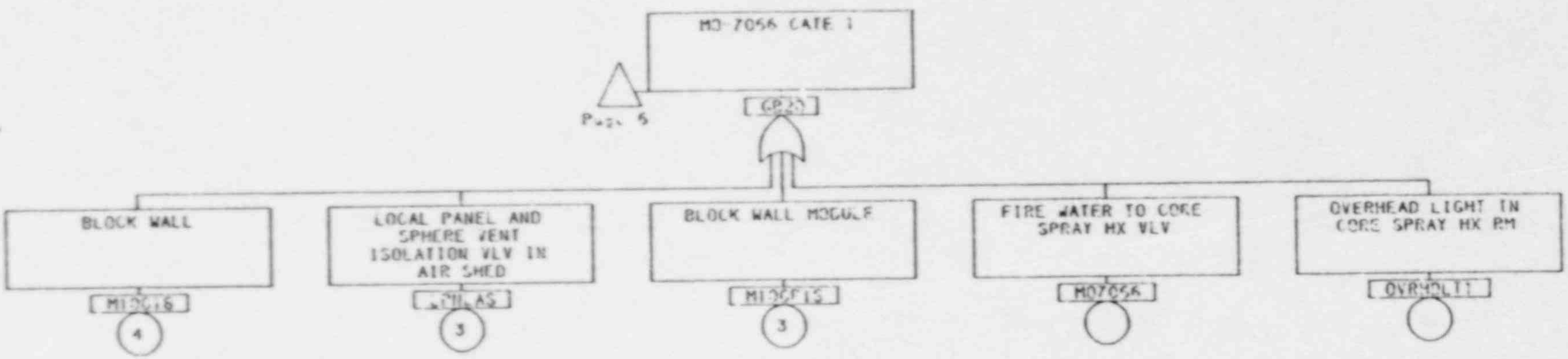
DRAWING NUMBER Page 8	DATE 4/28/88



TITLE	
POST INCIDENT FAILURE-LOSP	
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EMERGENCY CONDENSER MAKEUP W/O AC
FAULT TREES

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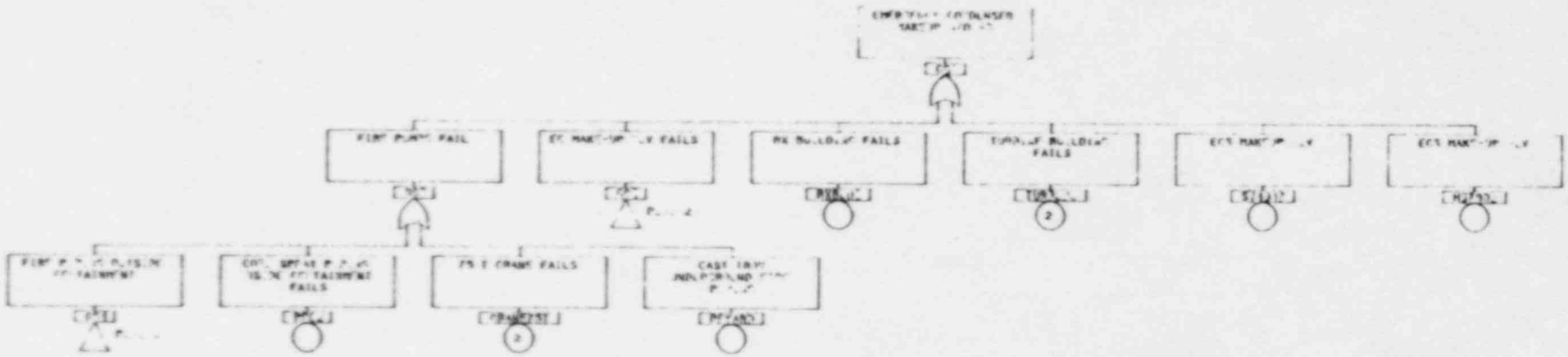
NU0383-2628A-BQ01-NL04



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EMERGENCY CONDENSER
MAKEUP W/O AC

DRAWING NUMBER	DATE
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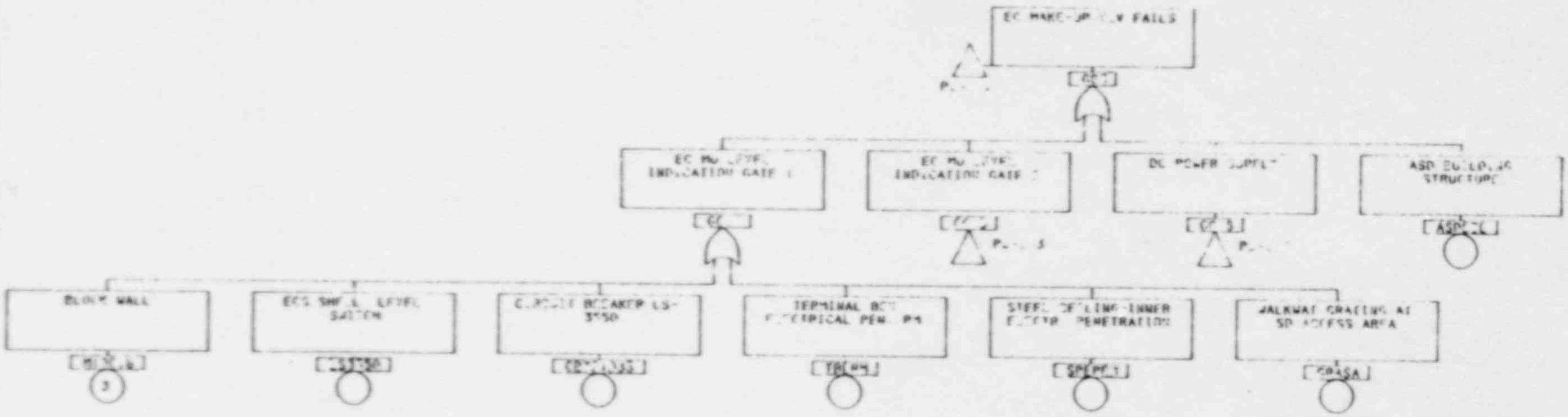

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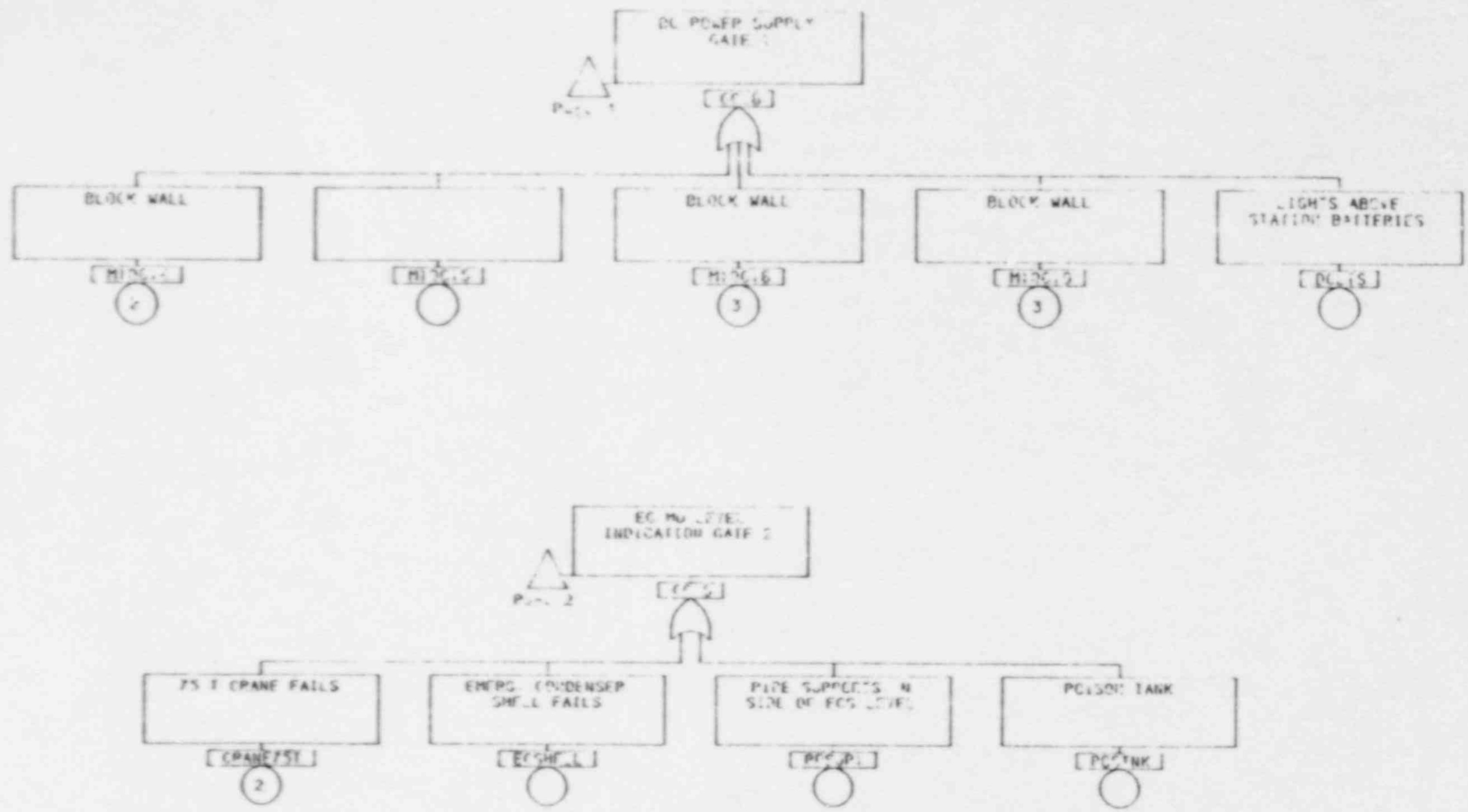
EMERGENCY CONDENSER
MAKEUP W/O AC

DRAWING NUMBER	DATE
Page 2	4/28/88



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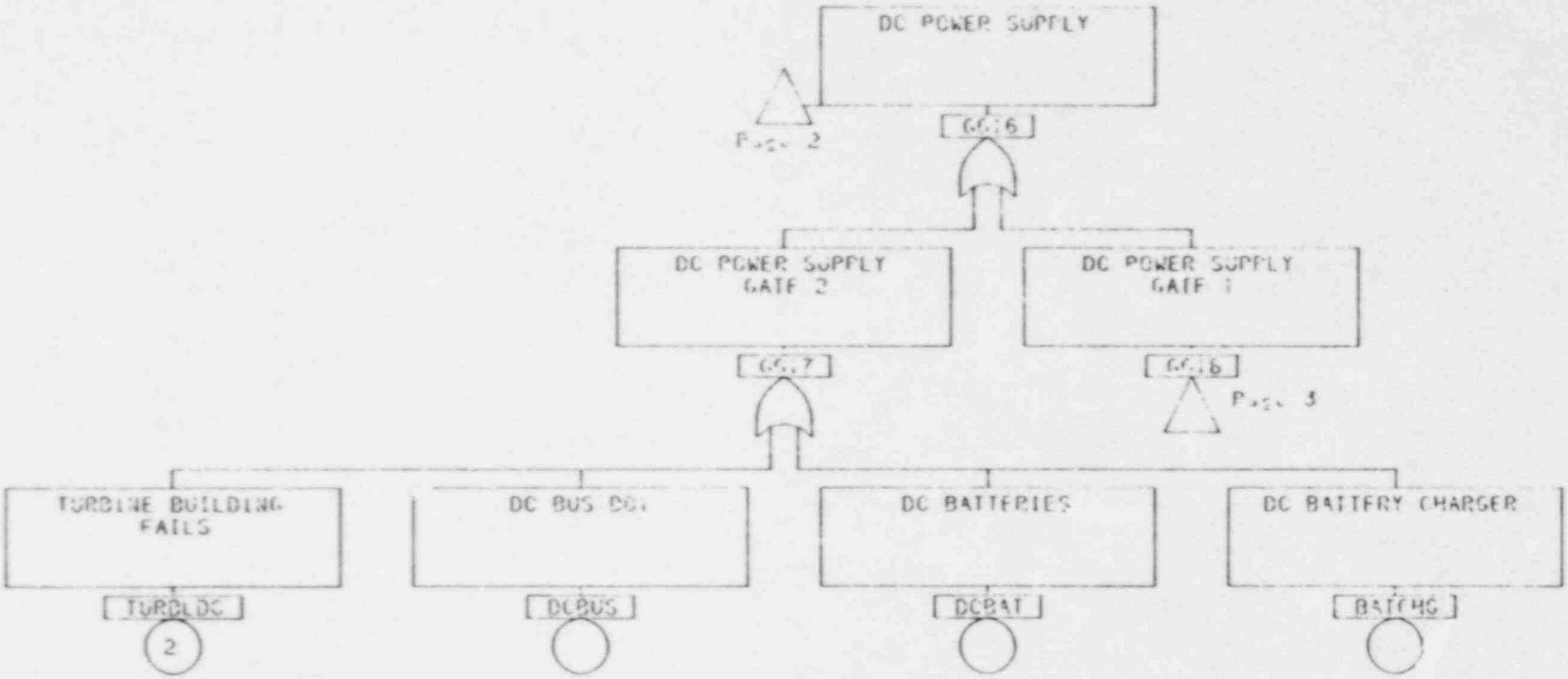
EMERGENCY CONDENSER MAKEUP W/O AC	
DRAWING NUMBER Page 3	DATE 4/28/88



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EMERGENCY CONDENSER MAKEUP W/O AC

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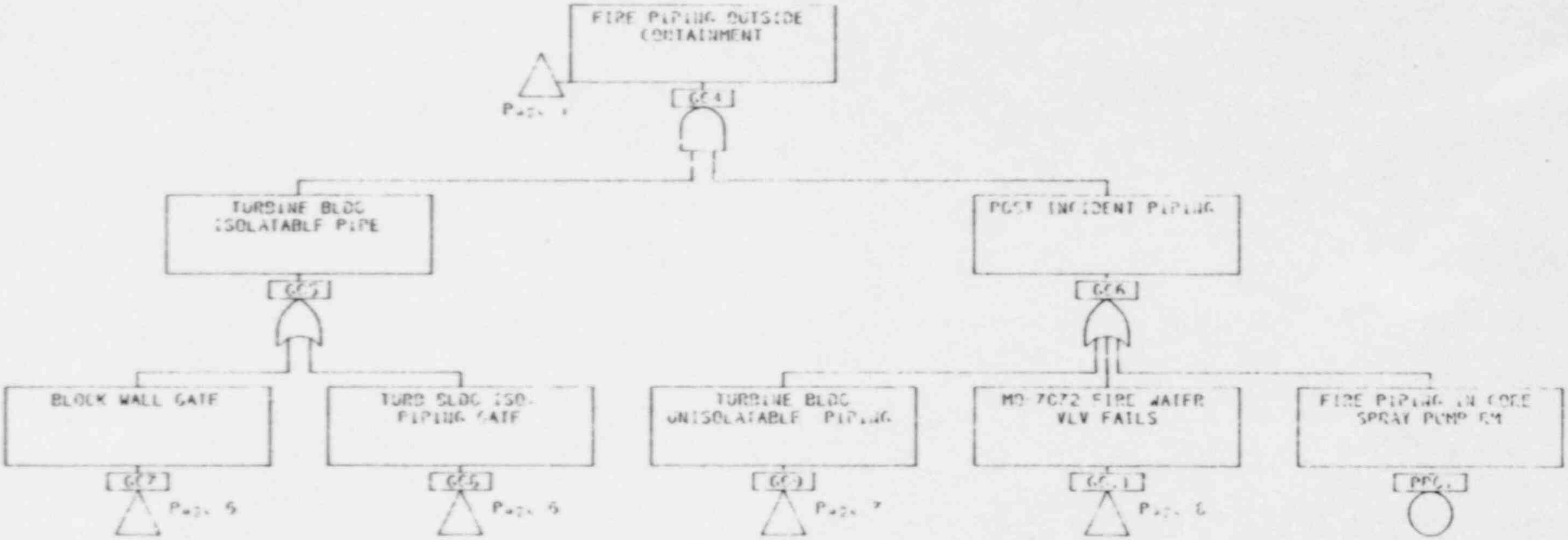

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EMERGENCY CONDENSER MAKEUP W/O AC

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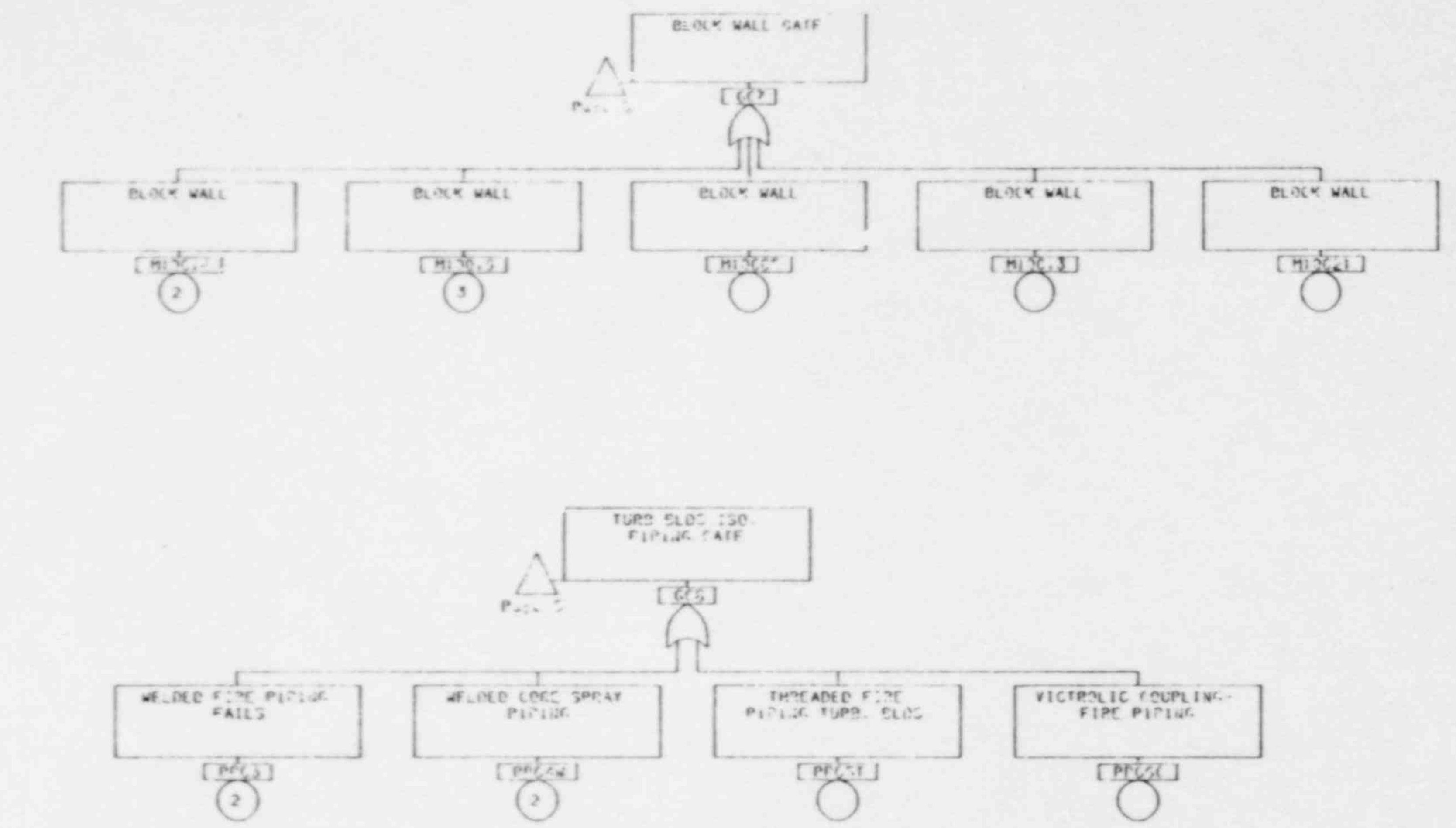

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EMERGENCY CONDENSER MAKEUP W/O AC	
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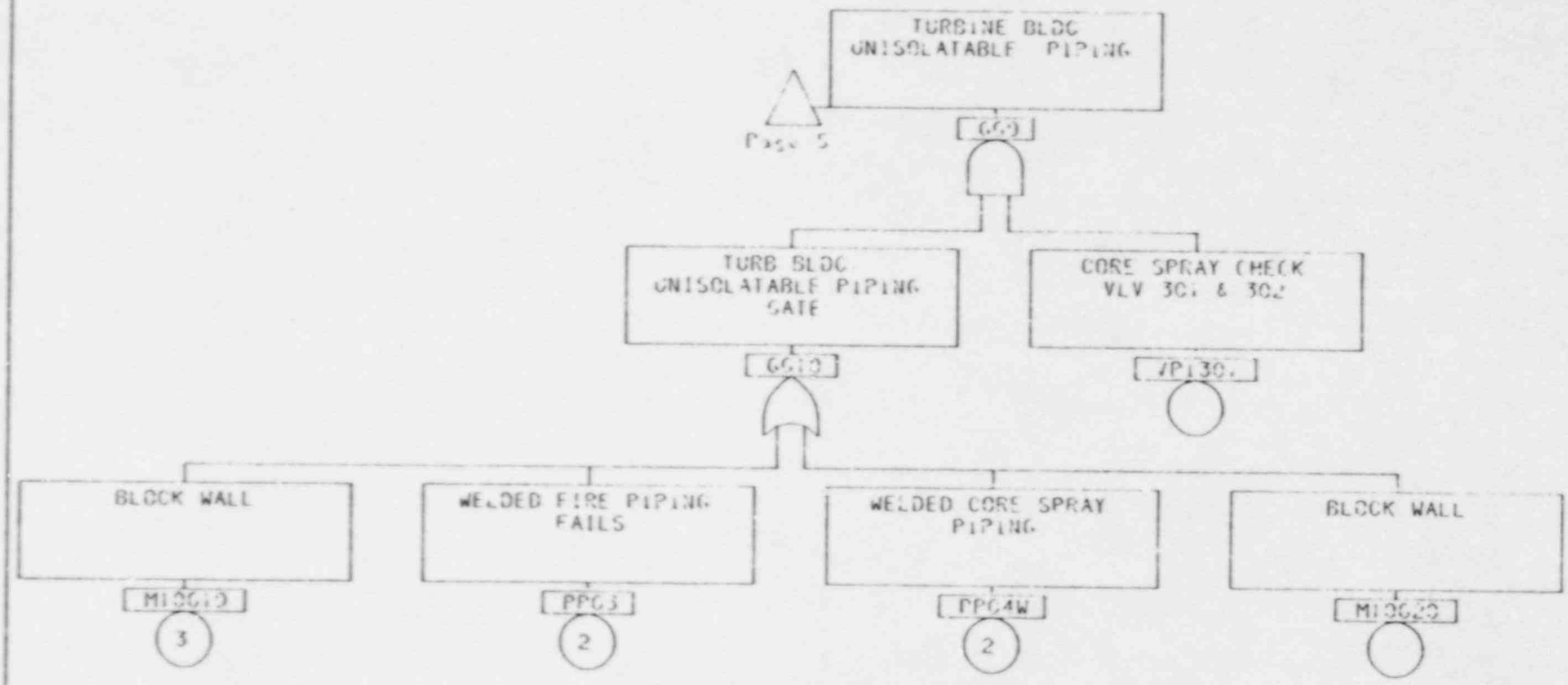

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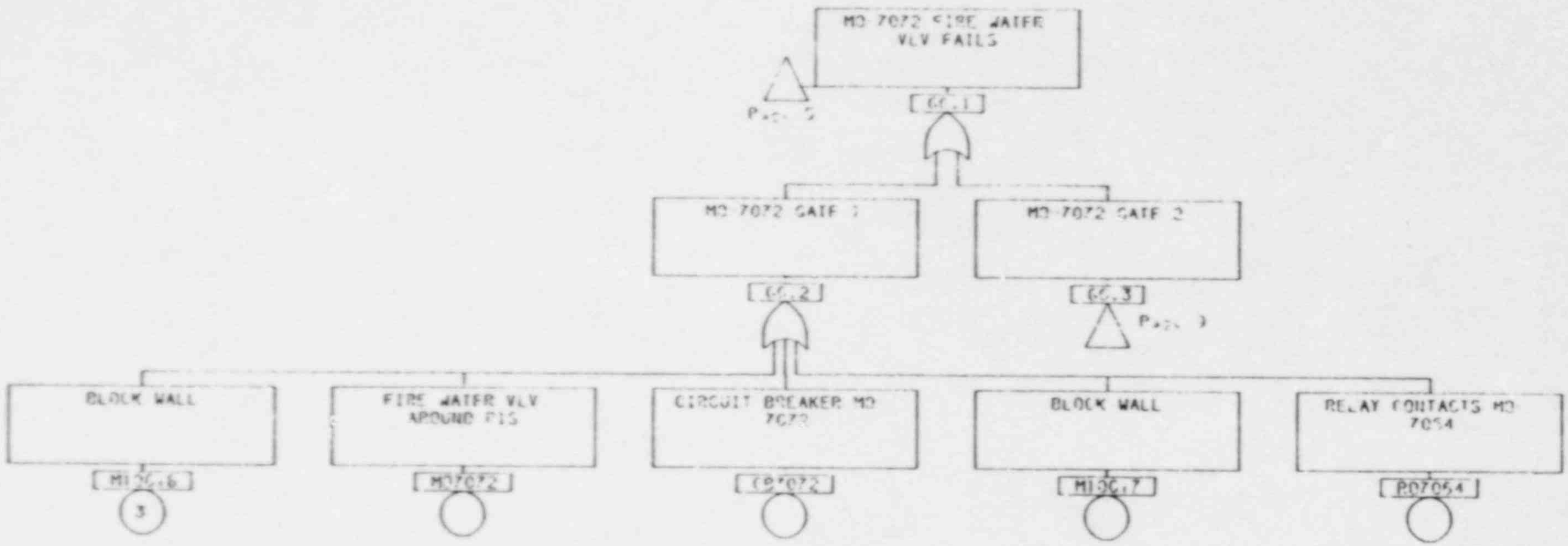

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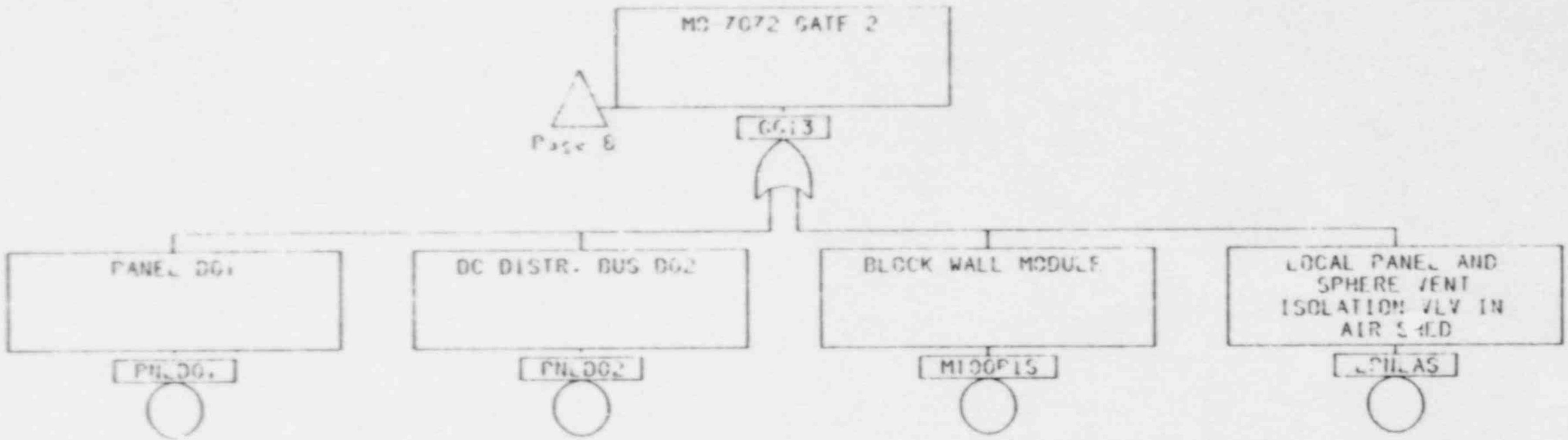

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EMERGENCY CONDENSER MAKEUP W/O AC

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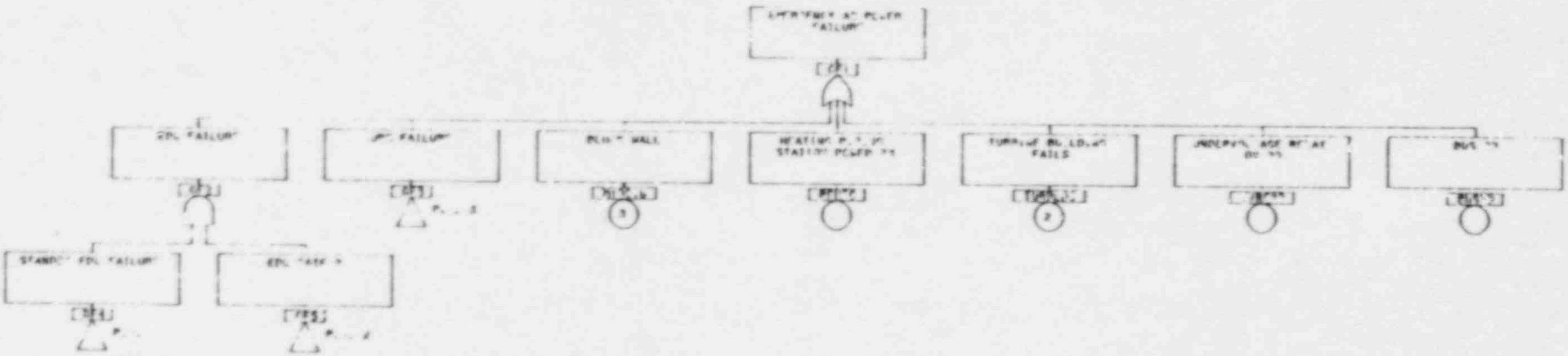
EMERGENCY AC POWER FAILURE
FAULT TREES

6 Pages



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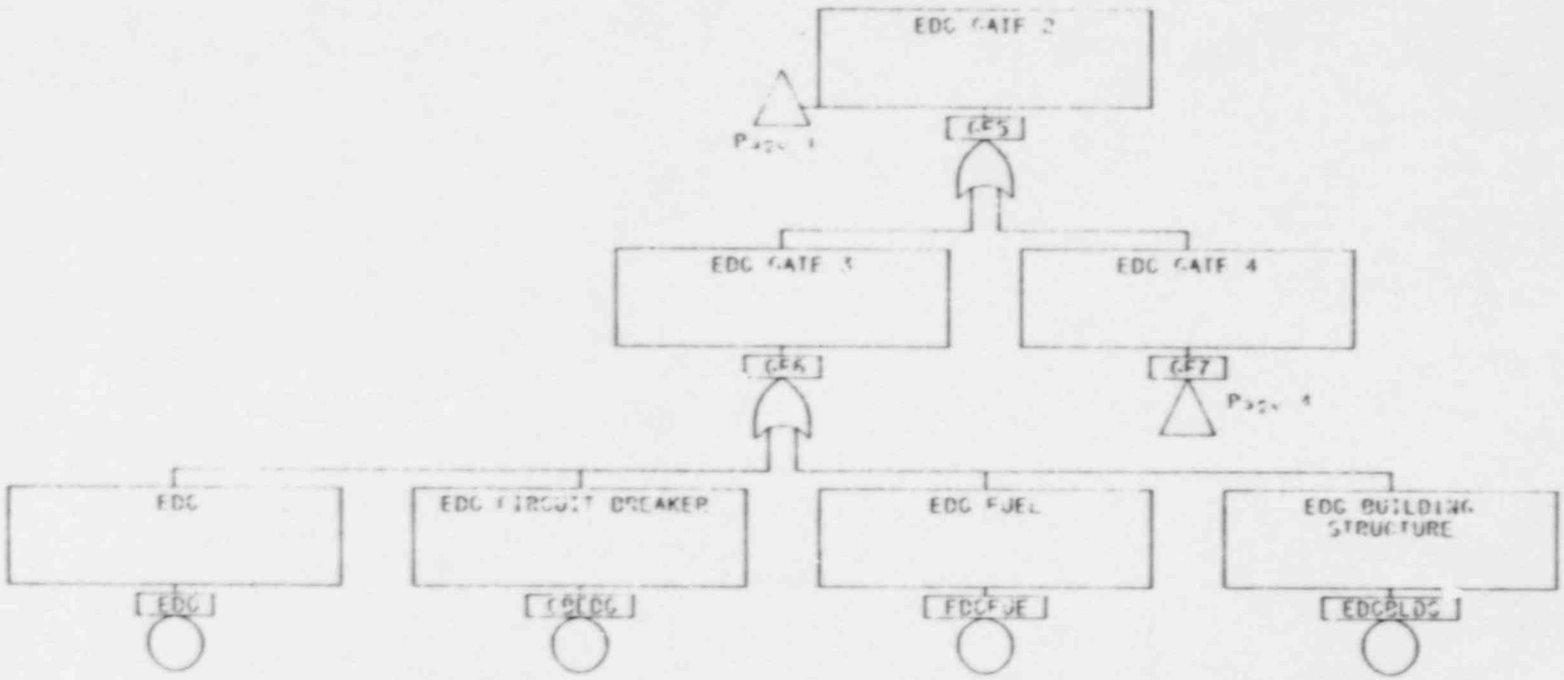

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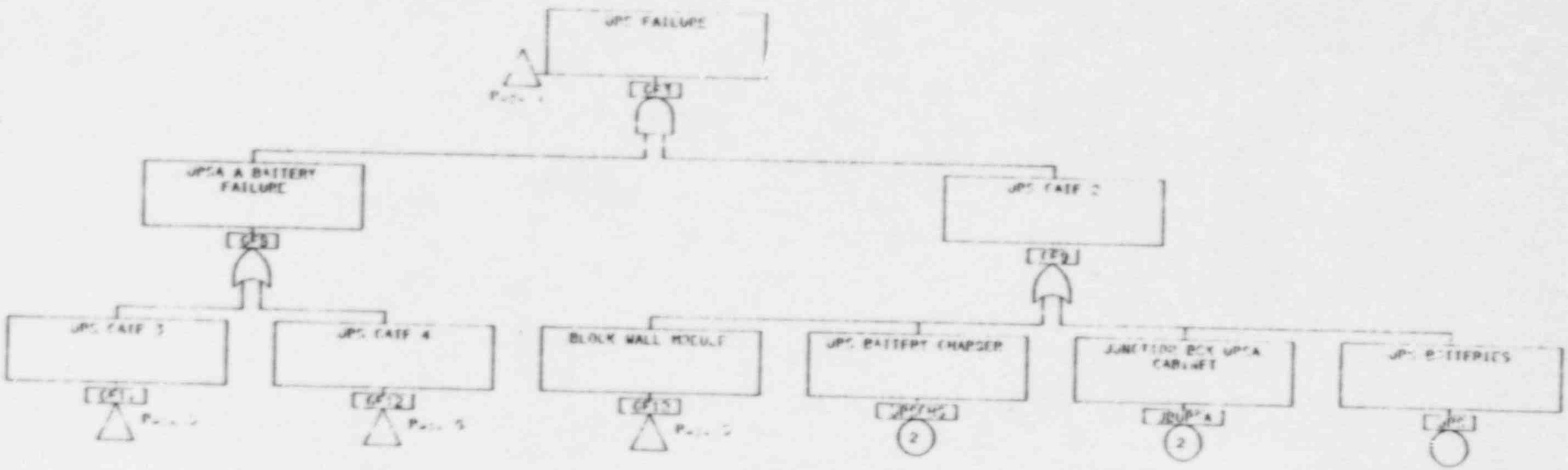
DRAWING NUMBER Page 2	DATE 4/28/88



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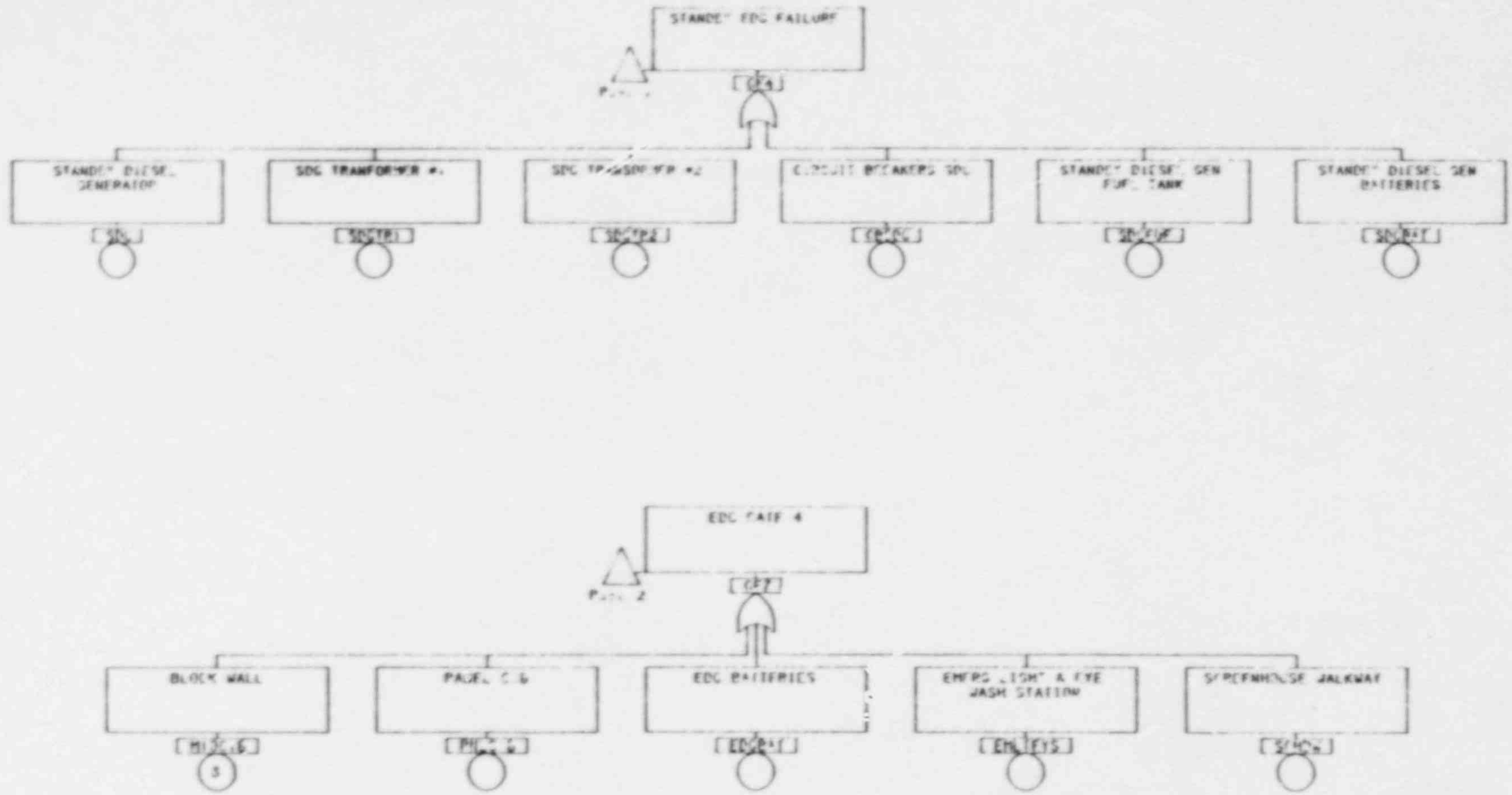

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EMERGENCY AC POWER FAILURE

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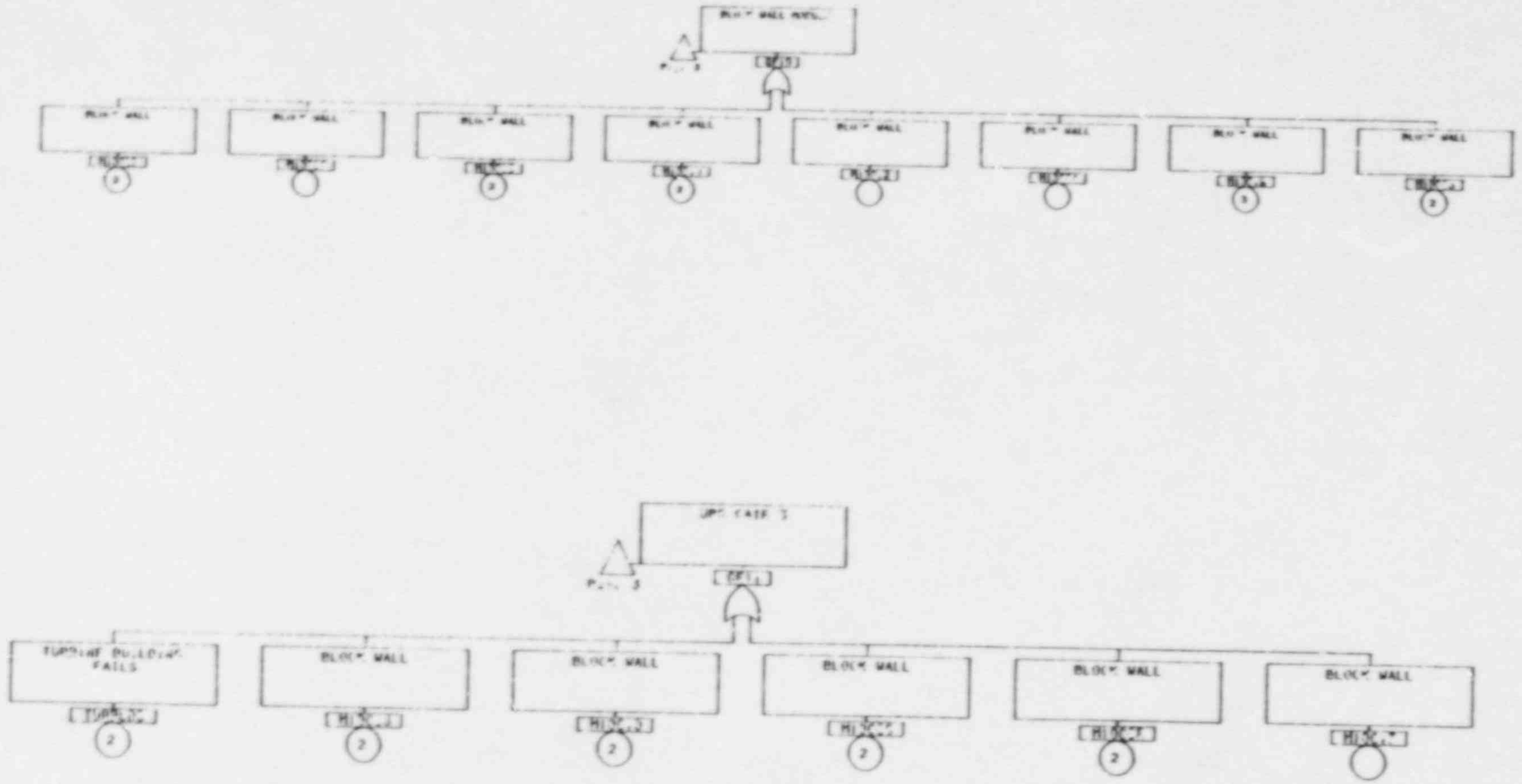

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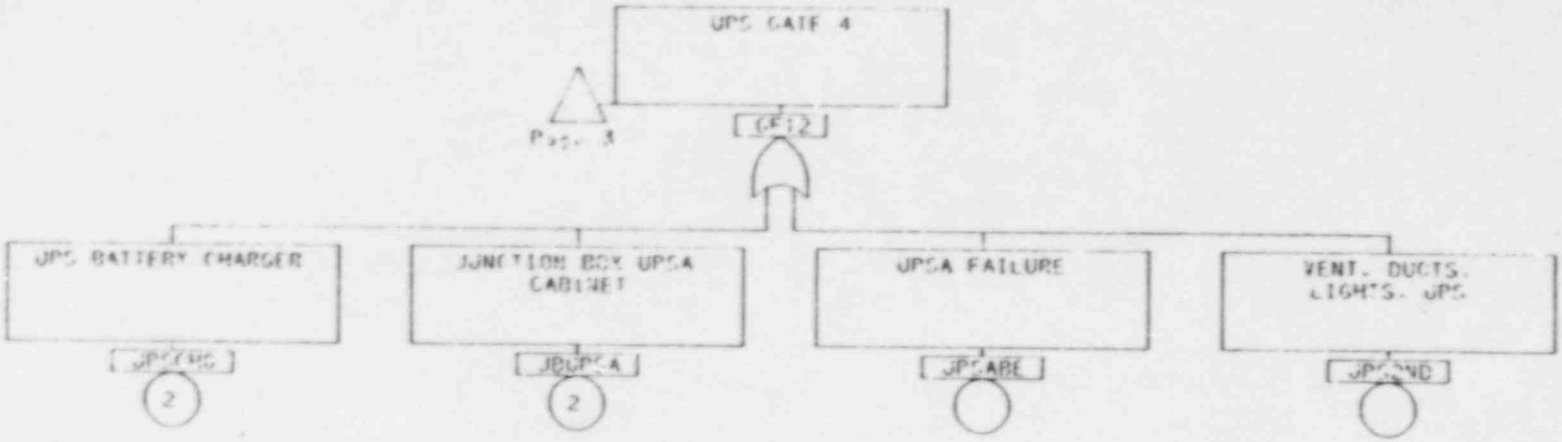

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UPS SYSTEM FAILURE
FAULT TREES

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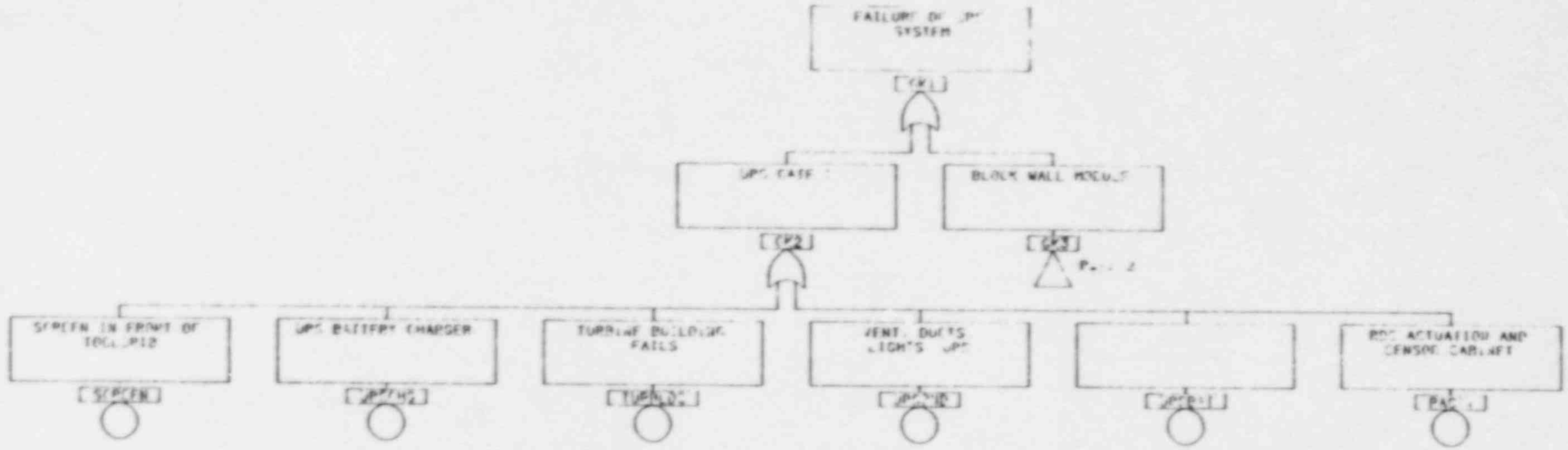

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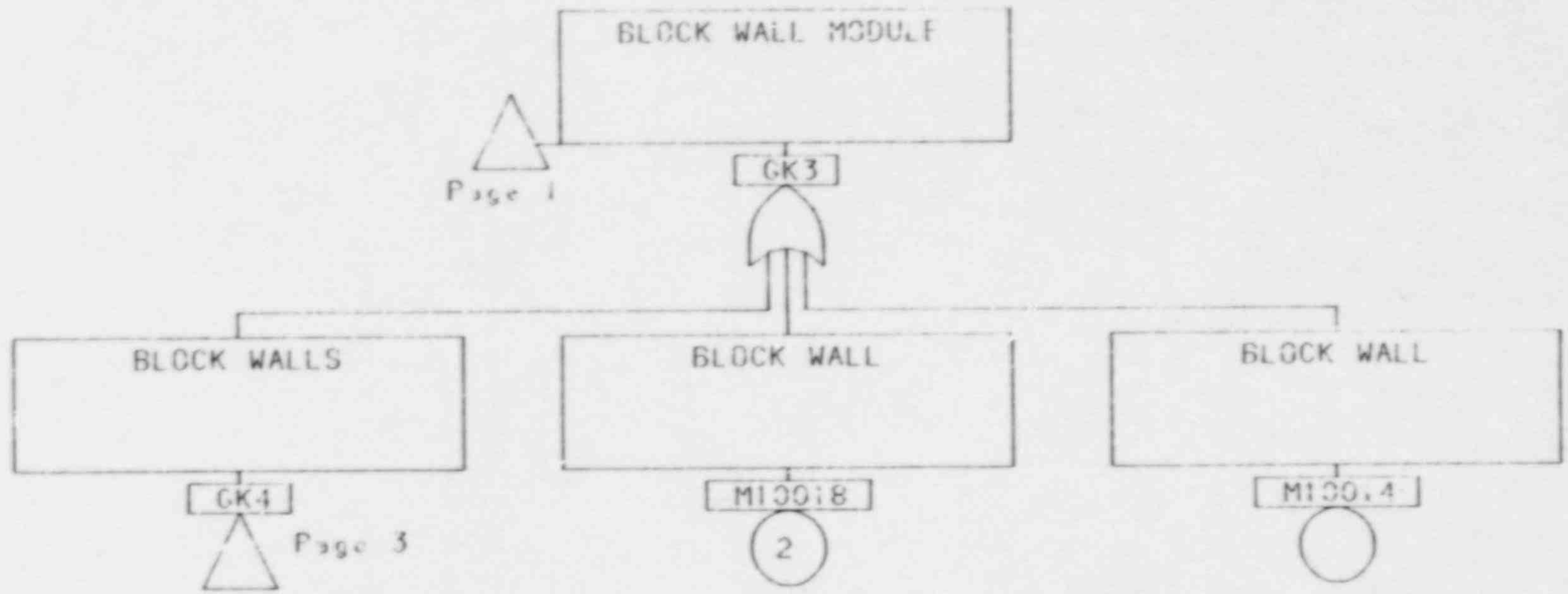

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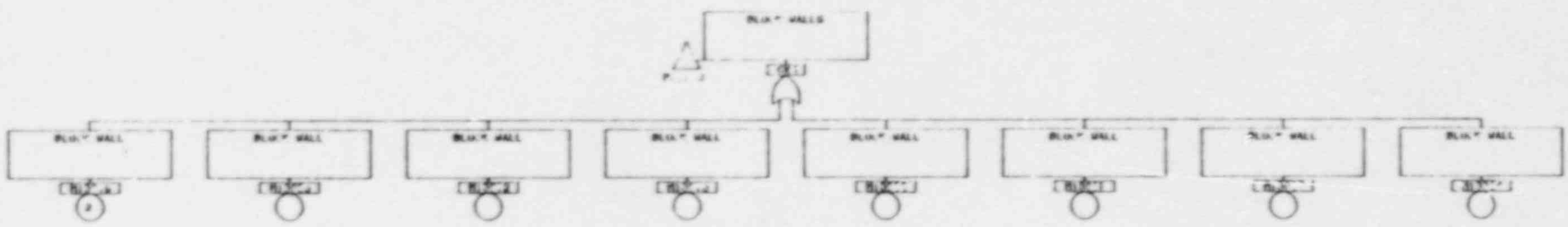

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FIRE PROTECTION SYSTEMS W/O AC
FAULT TREES

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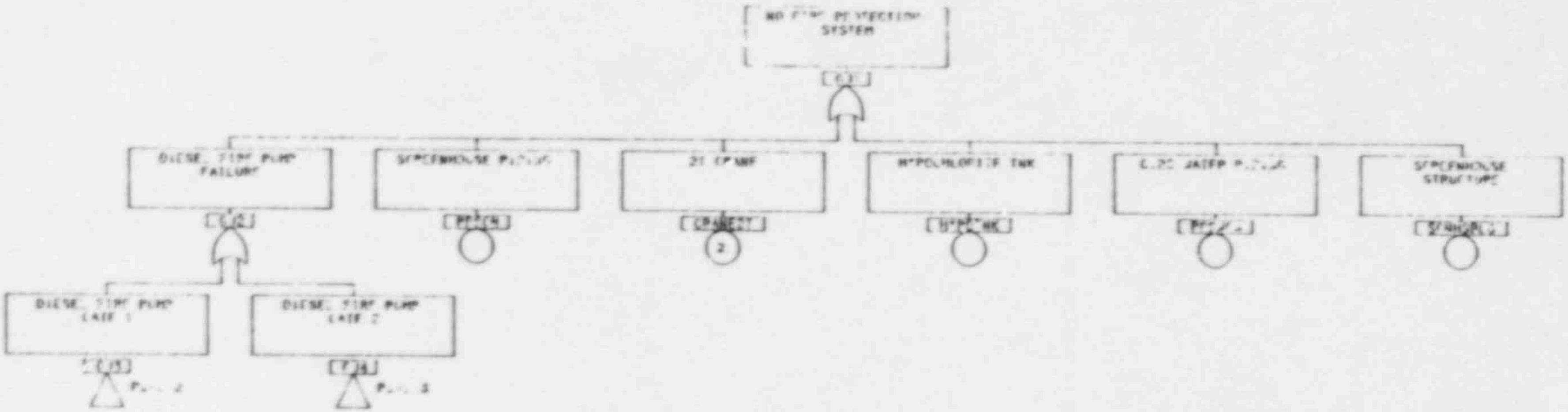

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SYSTEMS W/O AC

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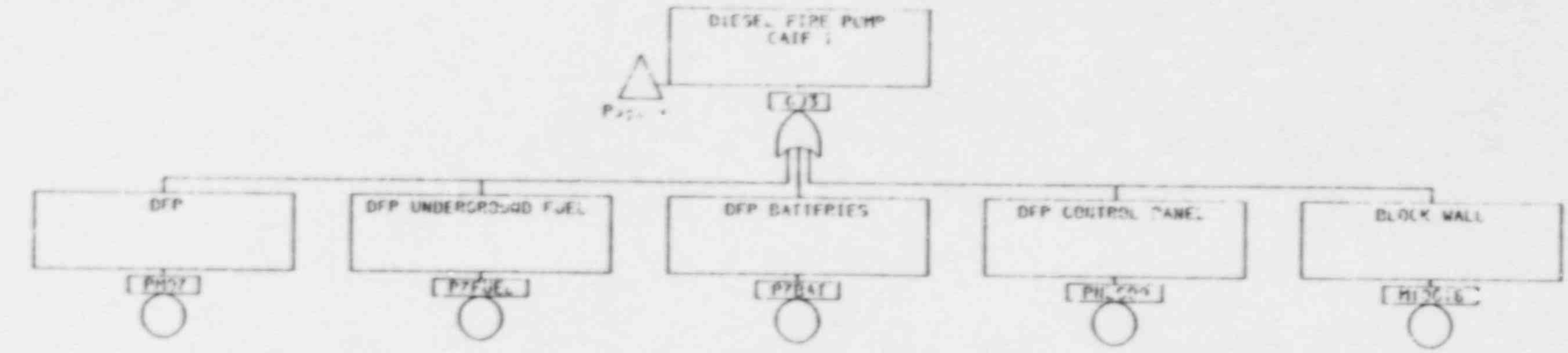

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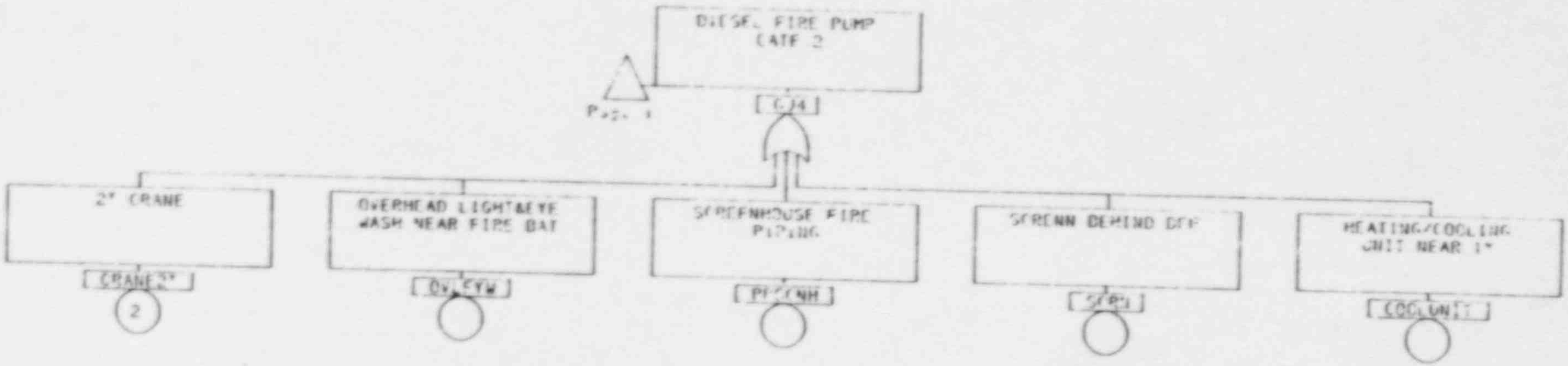

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FIRE PROTECTION
SYSTEMS W/O AC

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FIRE PROTECTION SYSTEM
FAULT TREES

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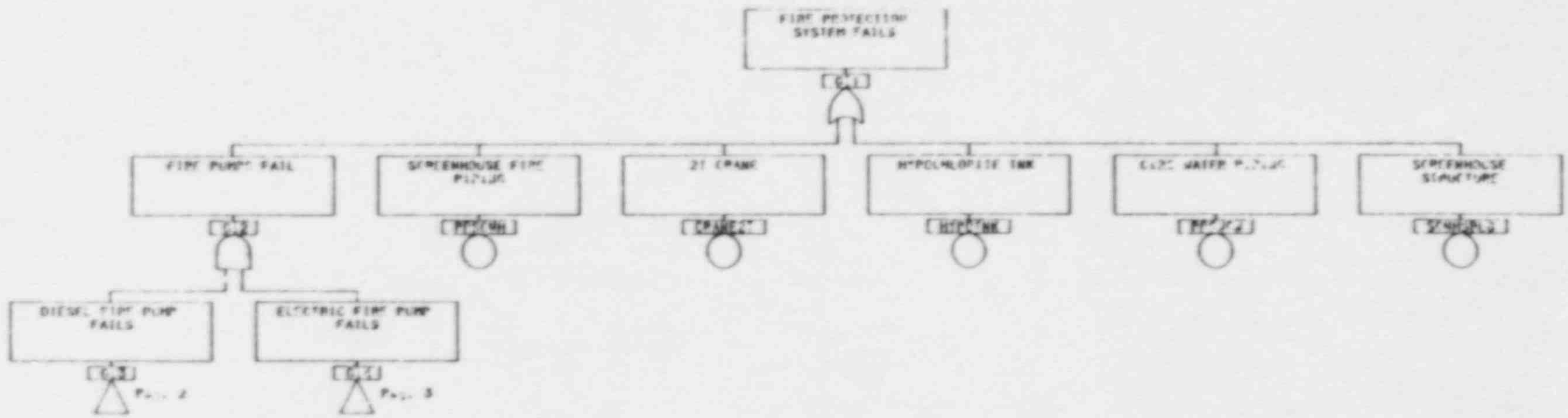

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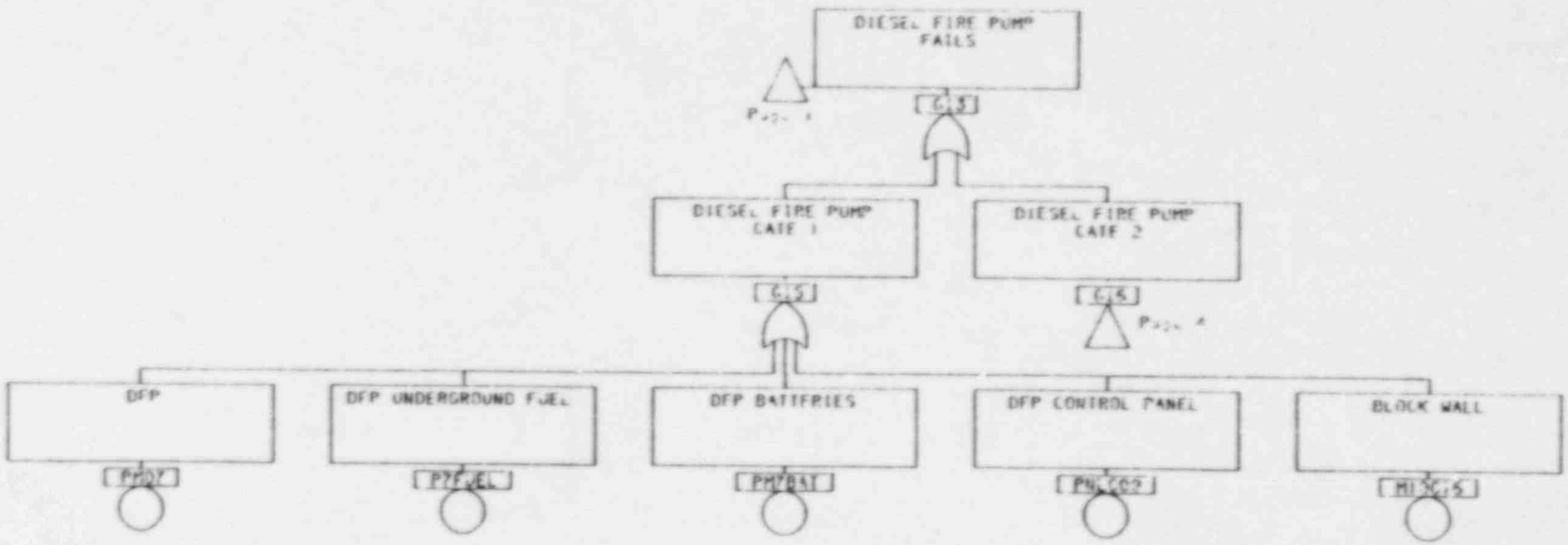

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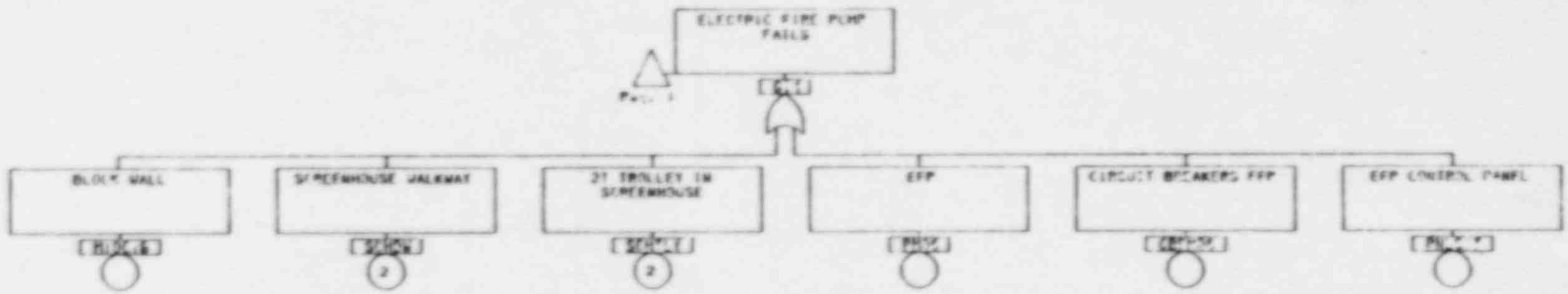

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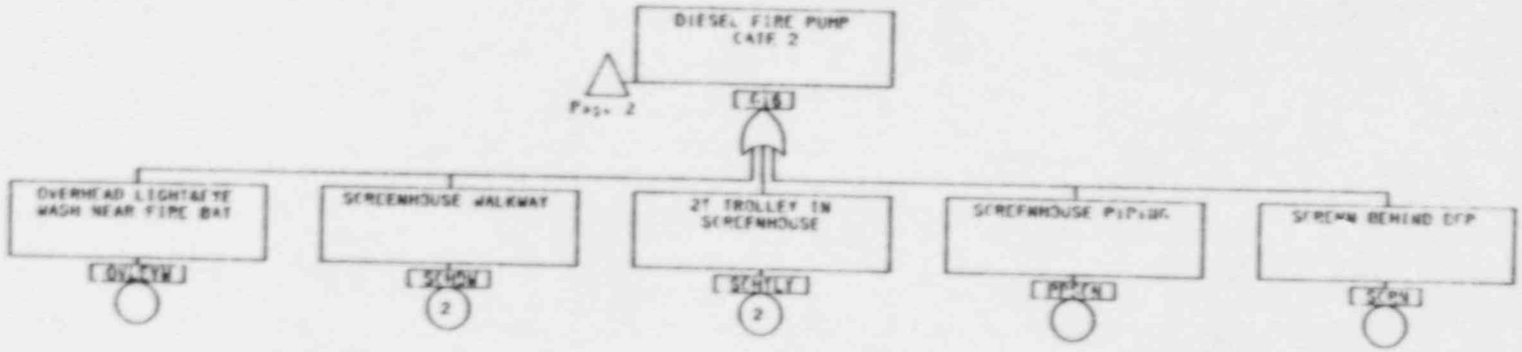

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FIRE PROTECTION SYSTEM

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EMERGENCY CONDENSER MU FAILURE
FAULT TREES

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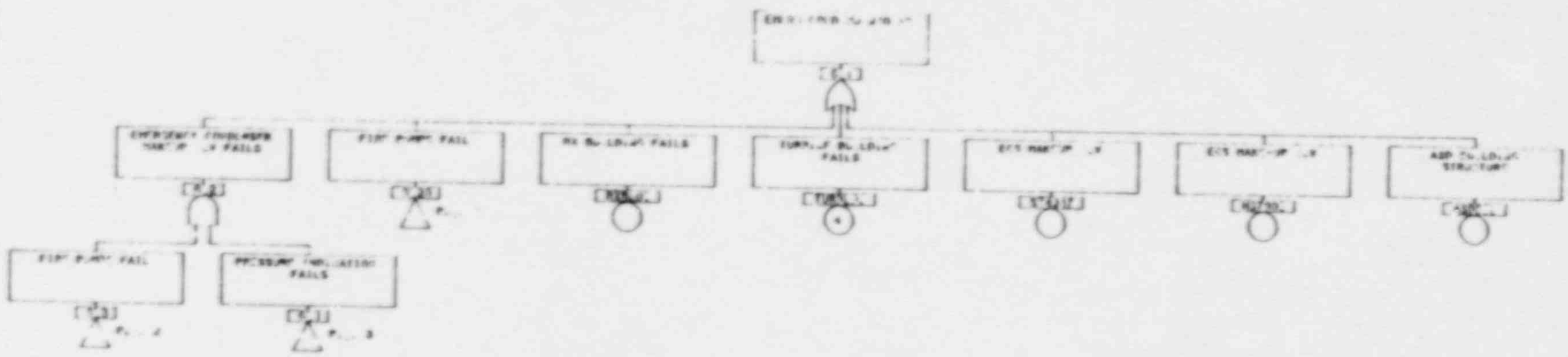

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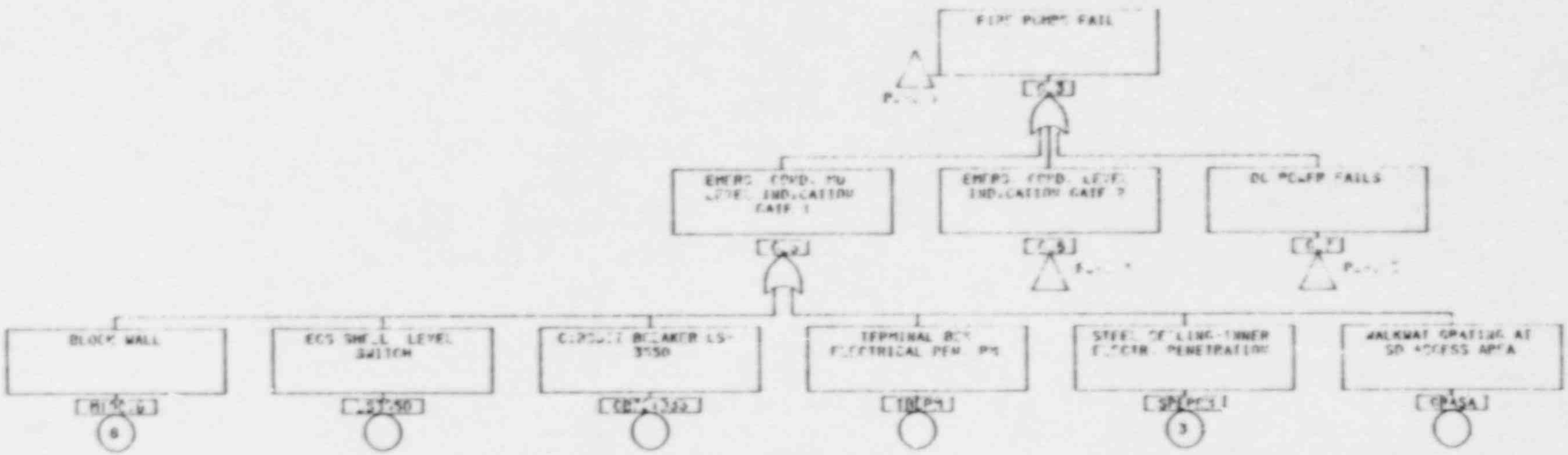

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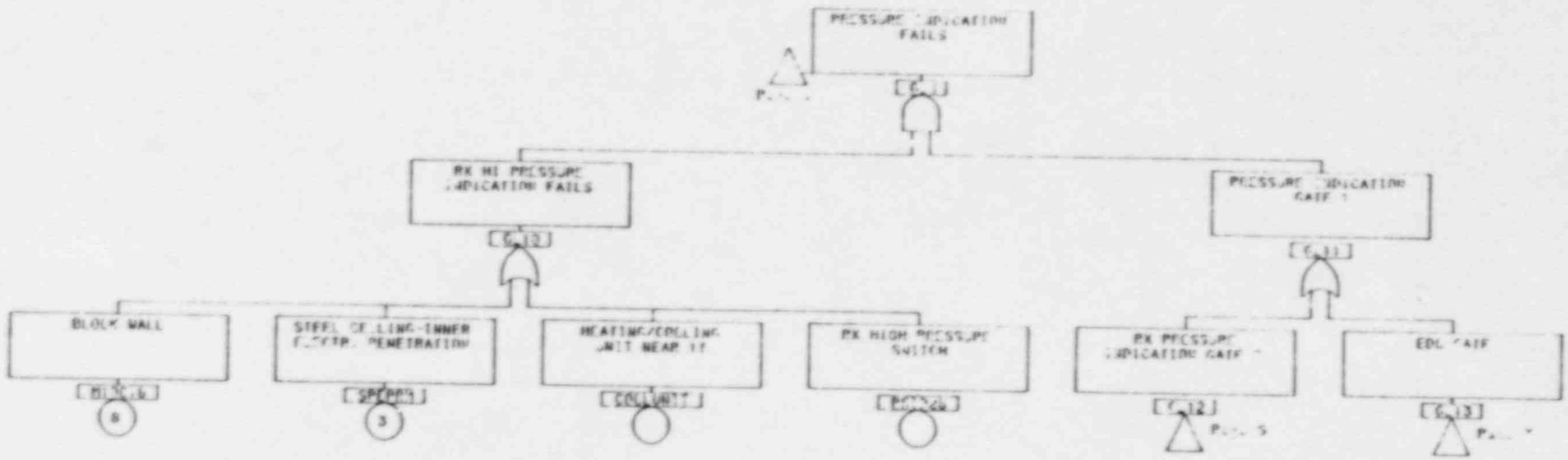

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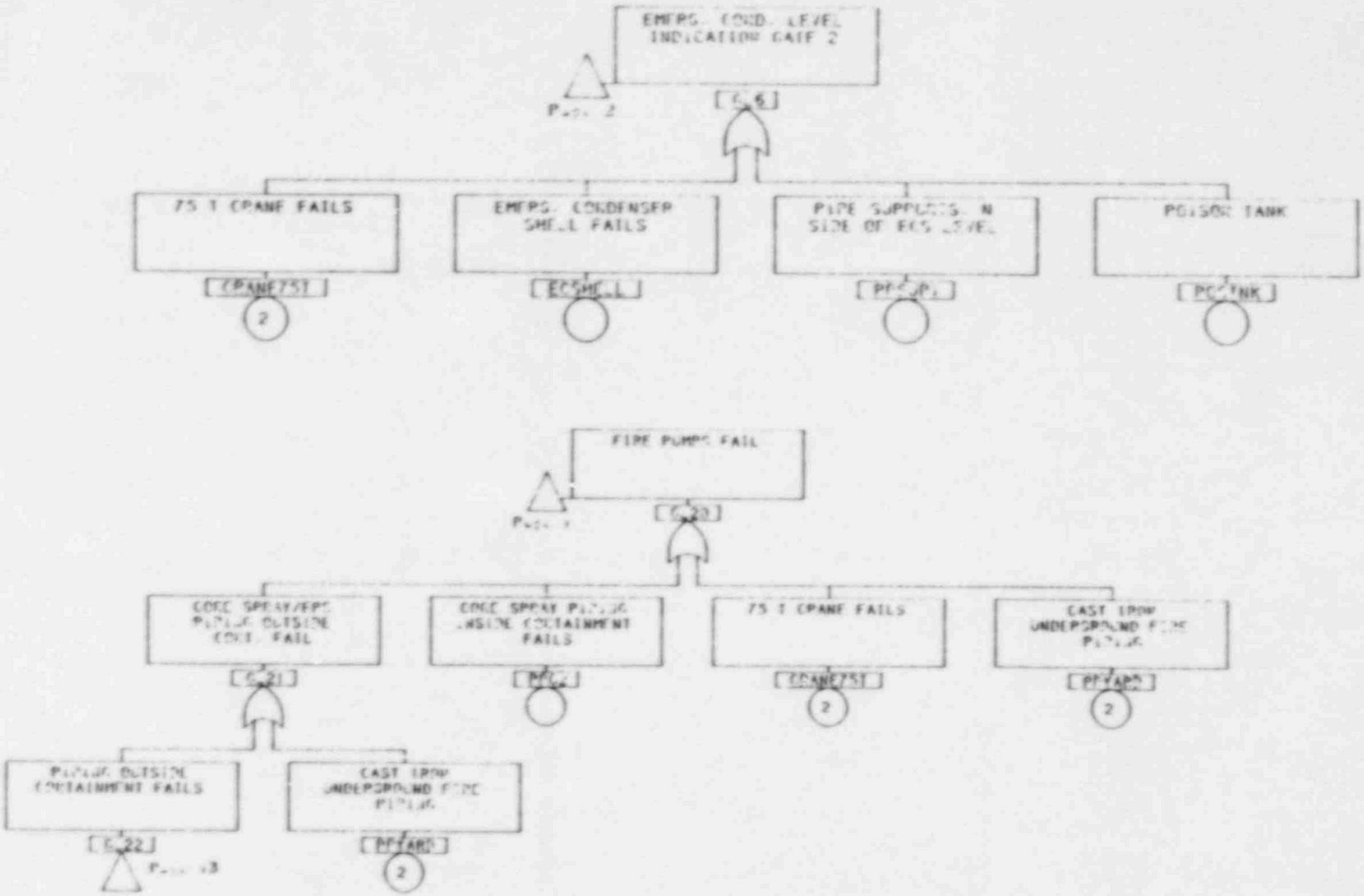

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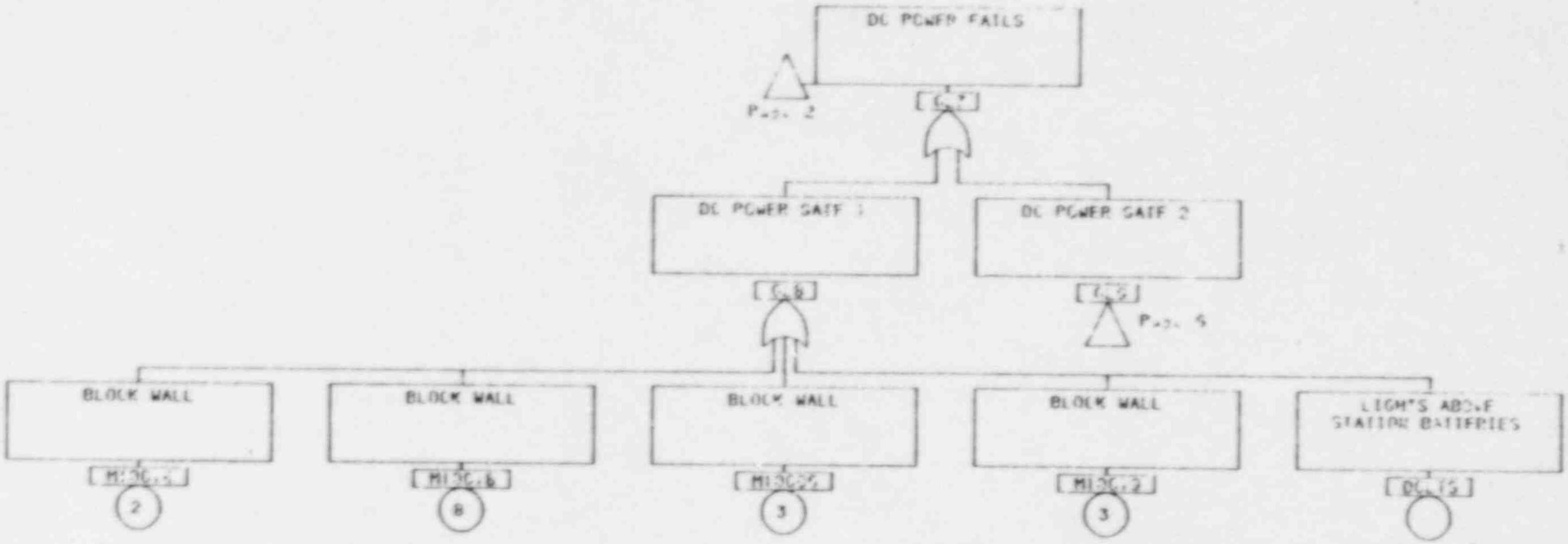

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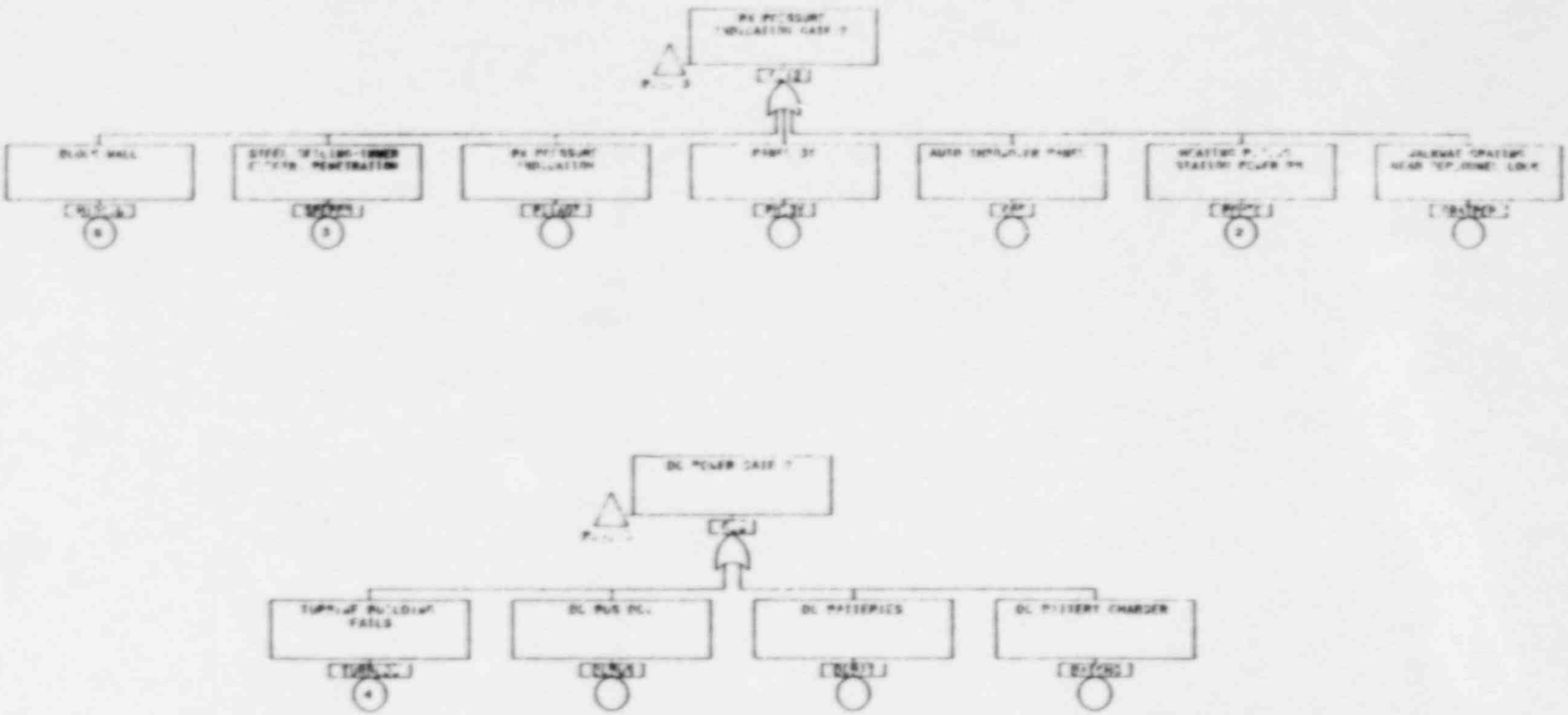

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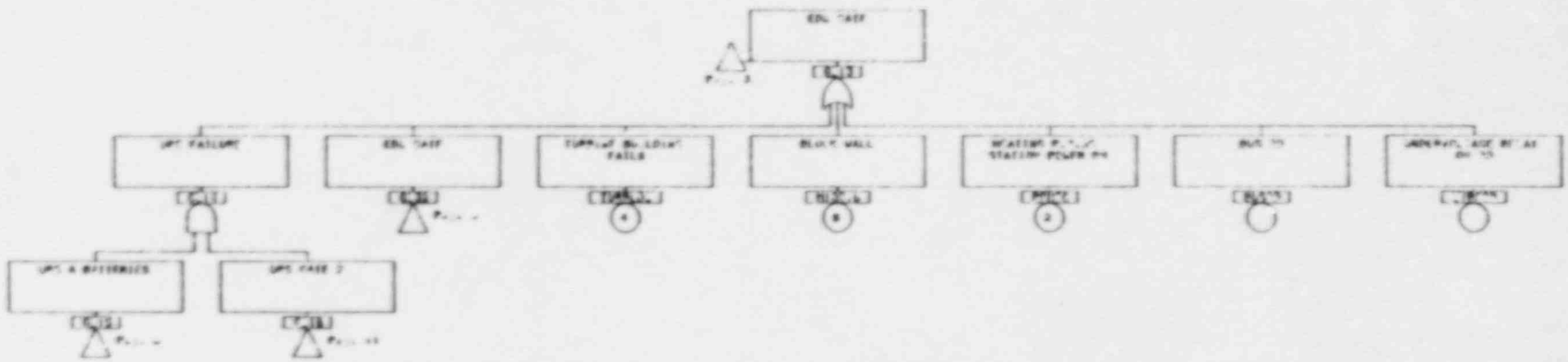

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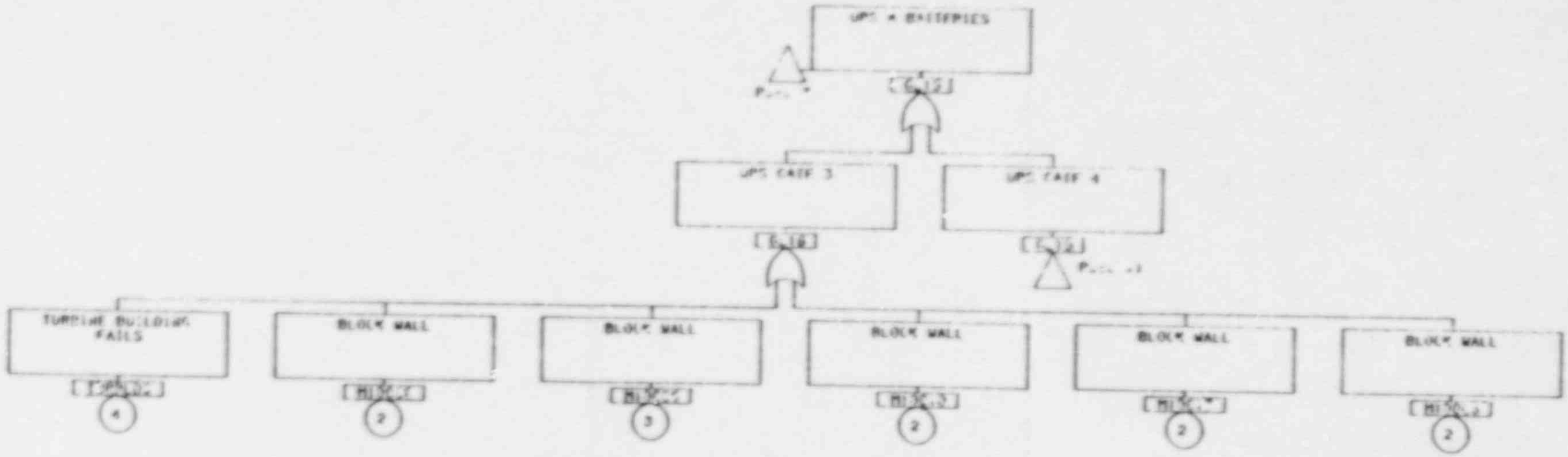

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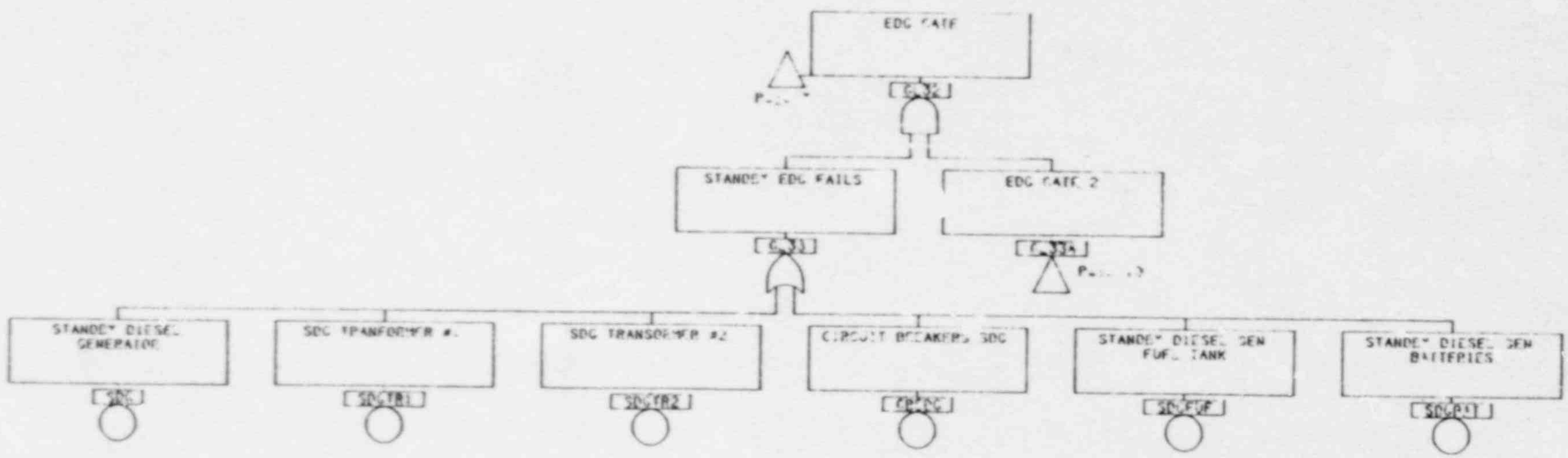

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EMERGENCY CONDENSER
MU FAILURE

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RDS FAILURE W/O AC
FAULT TREES

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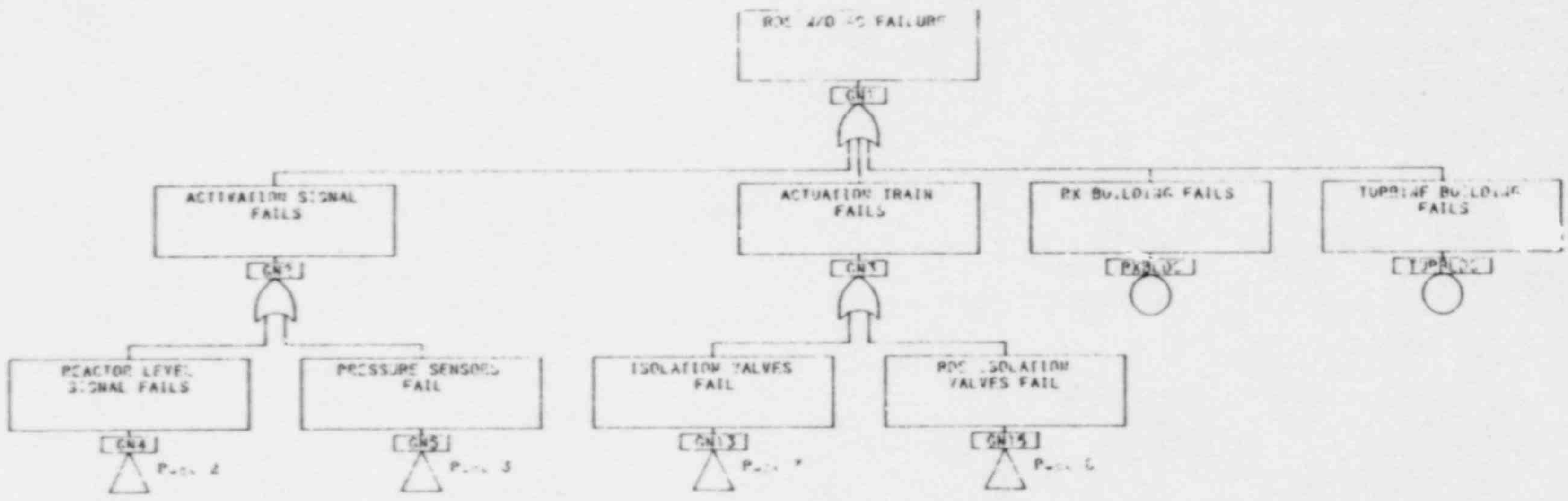

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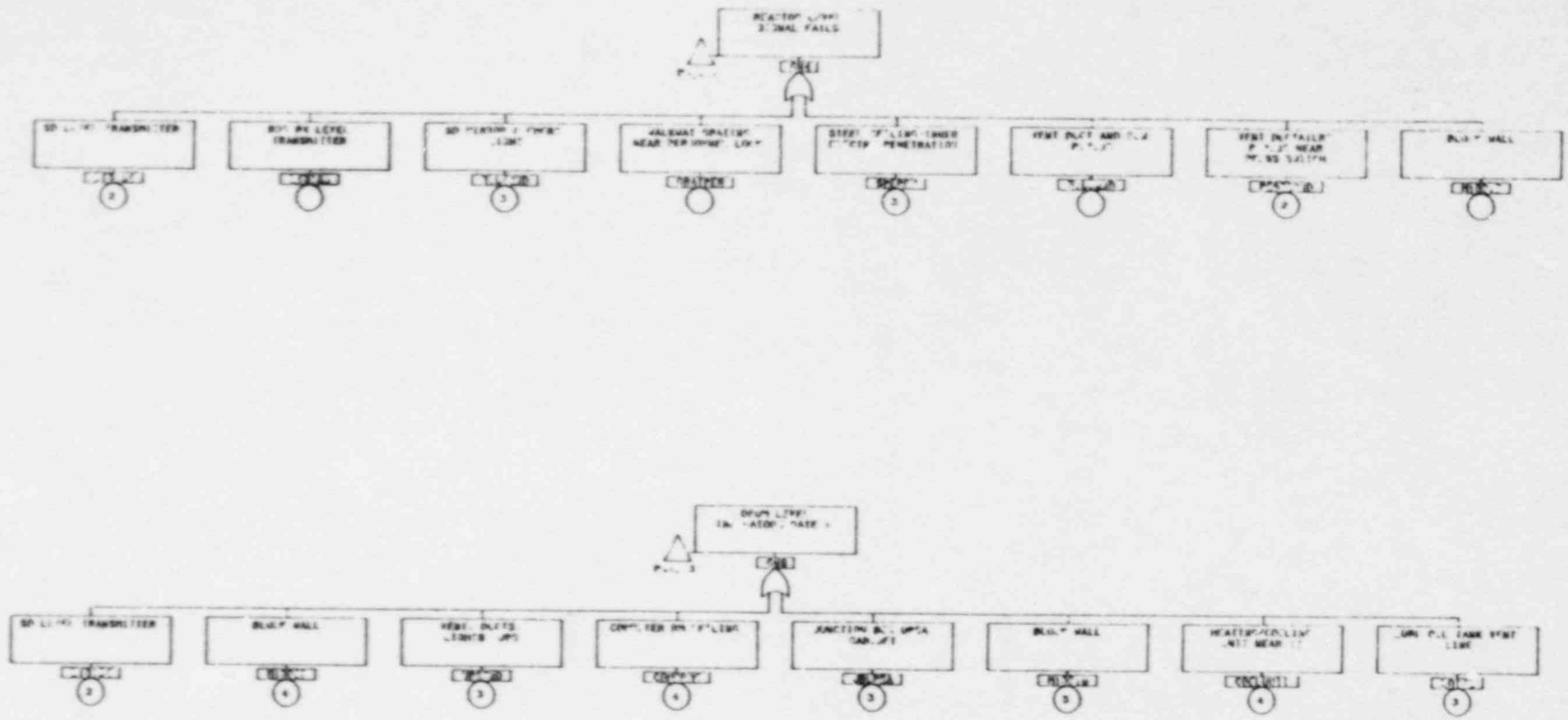

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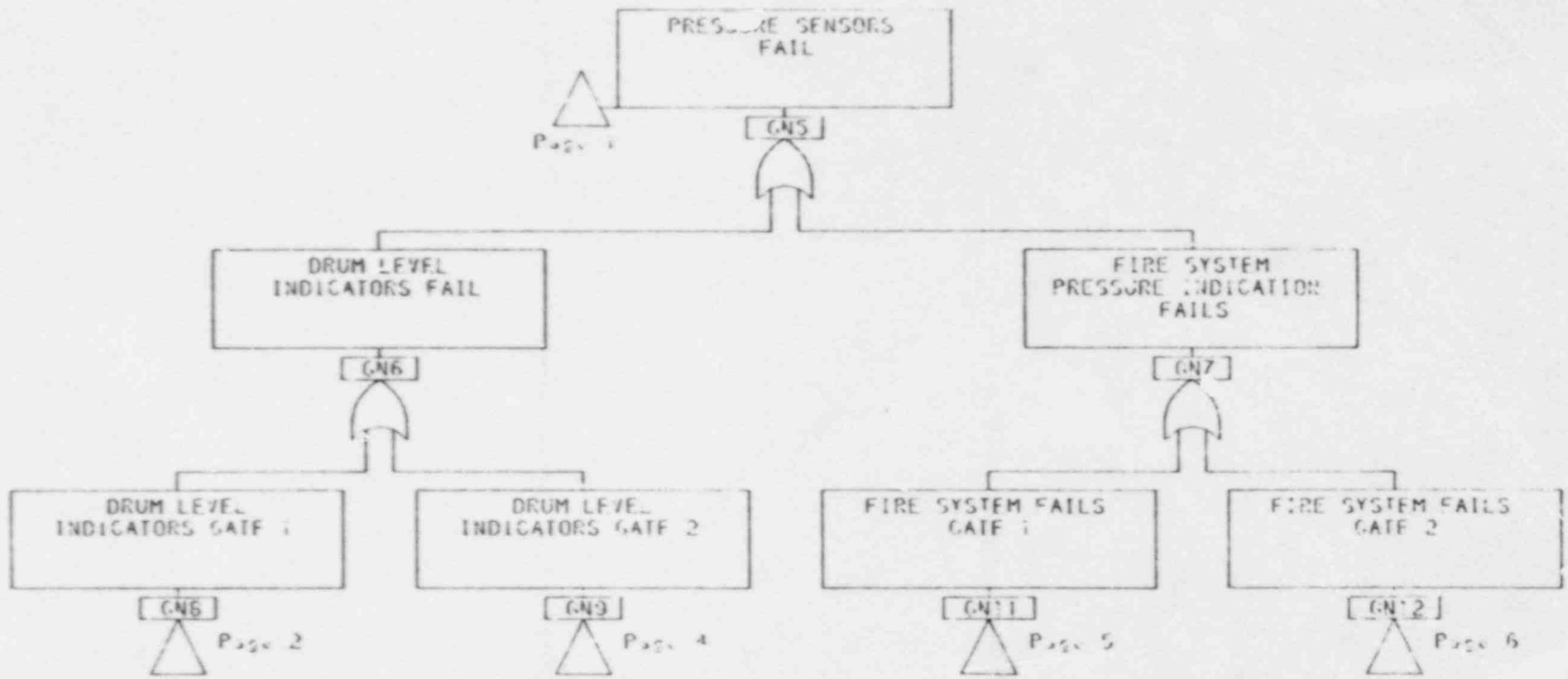

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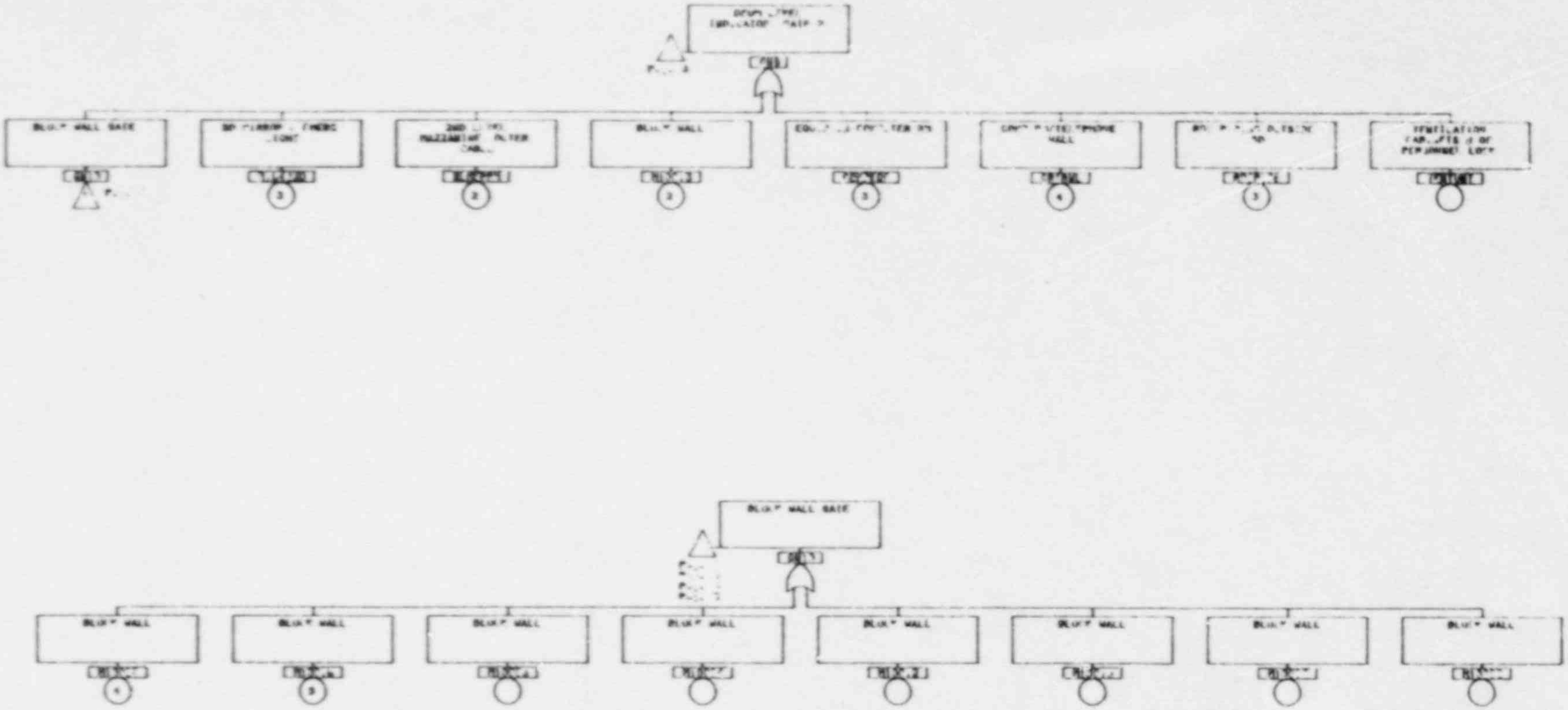

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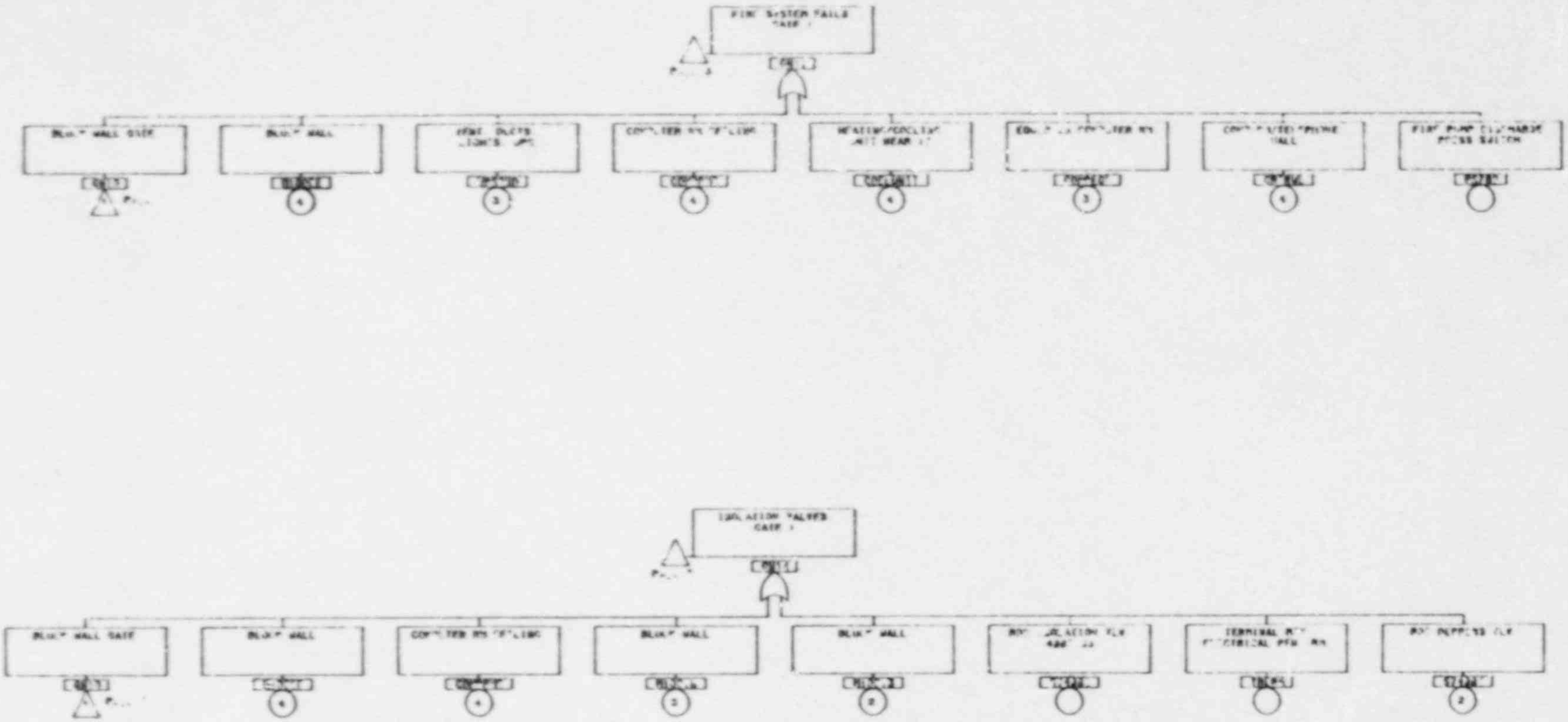

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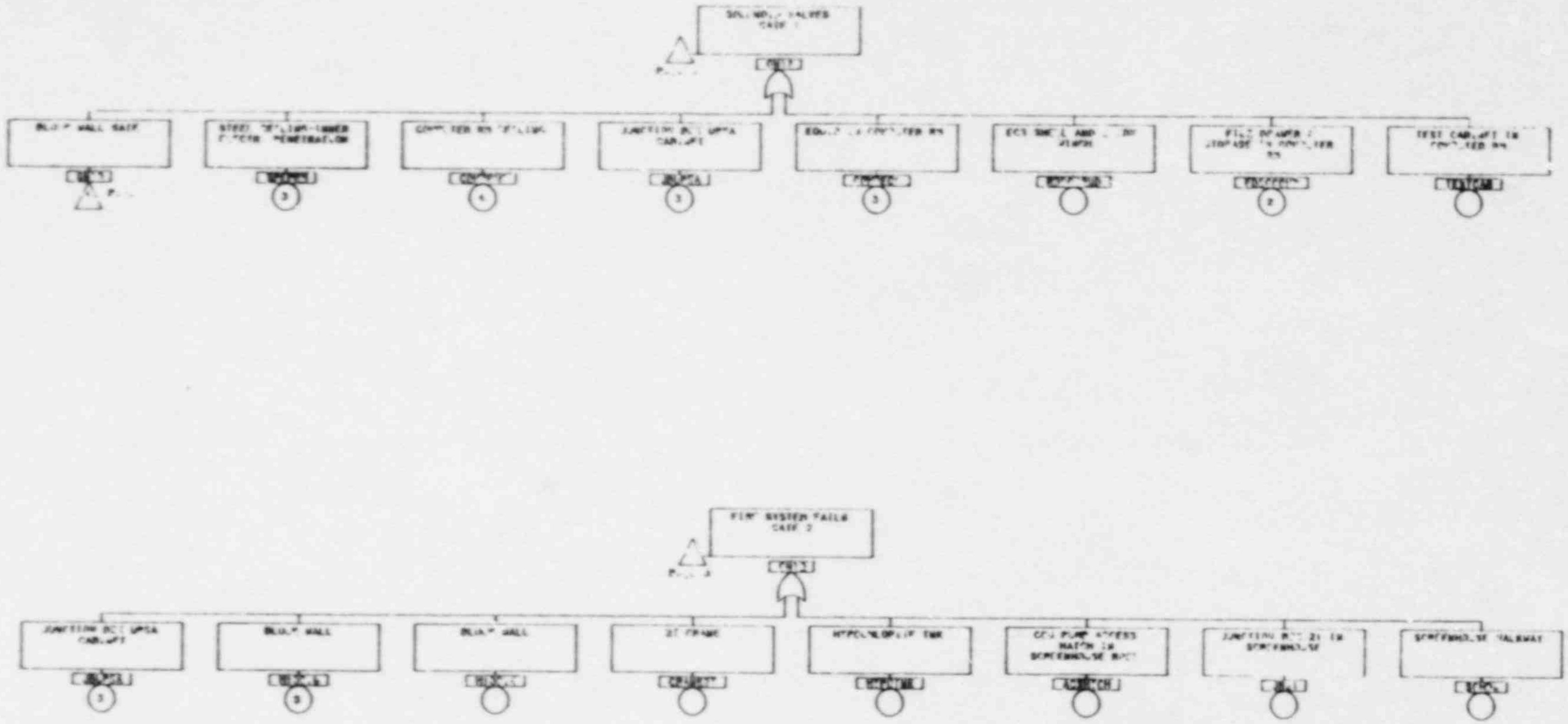

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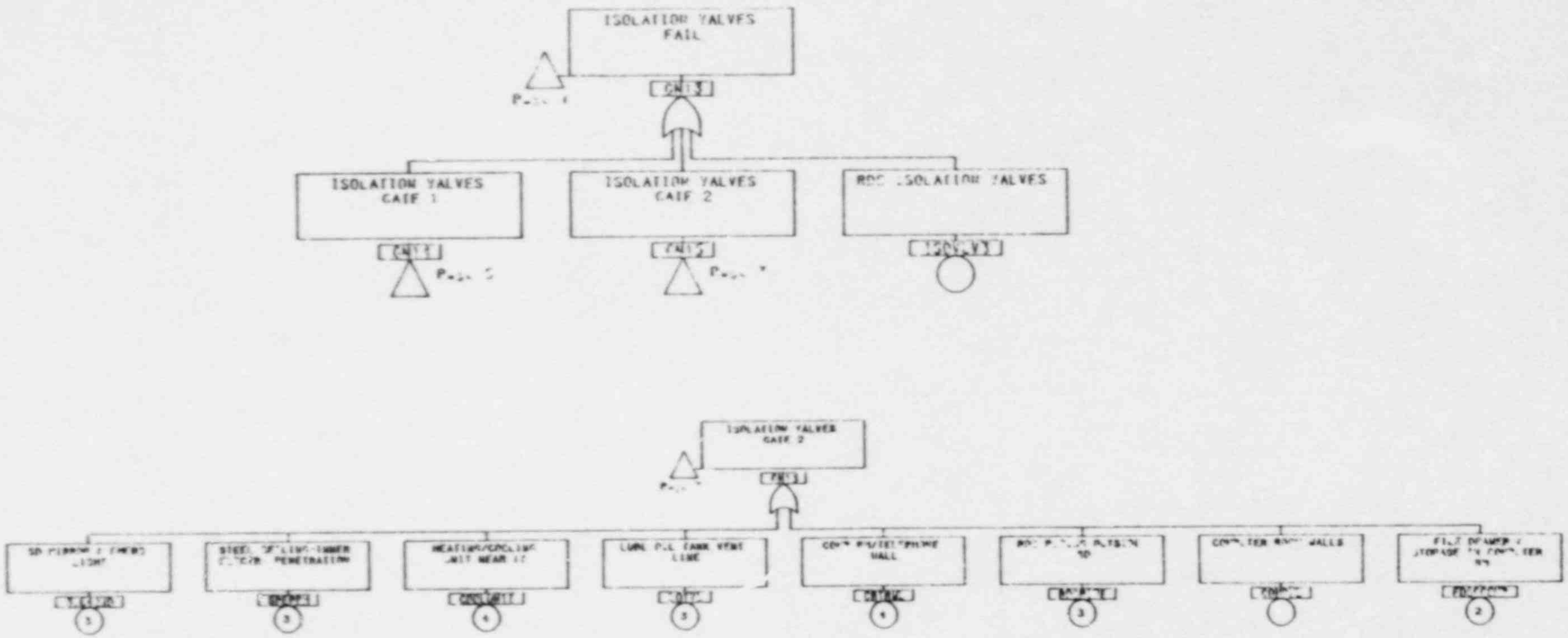

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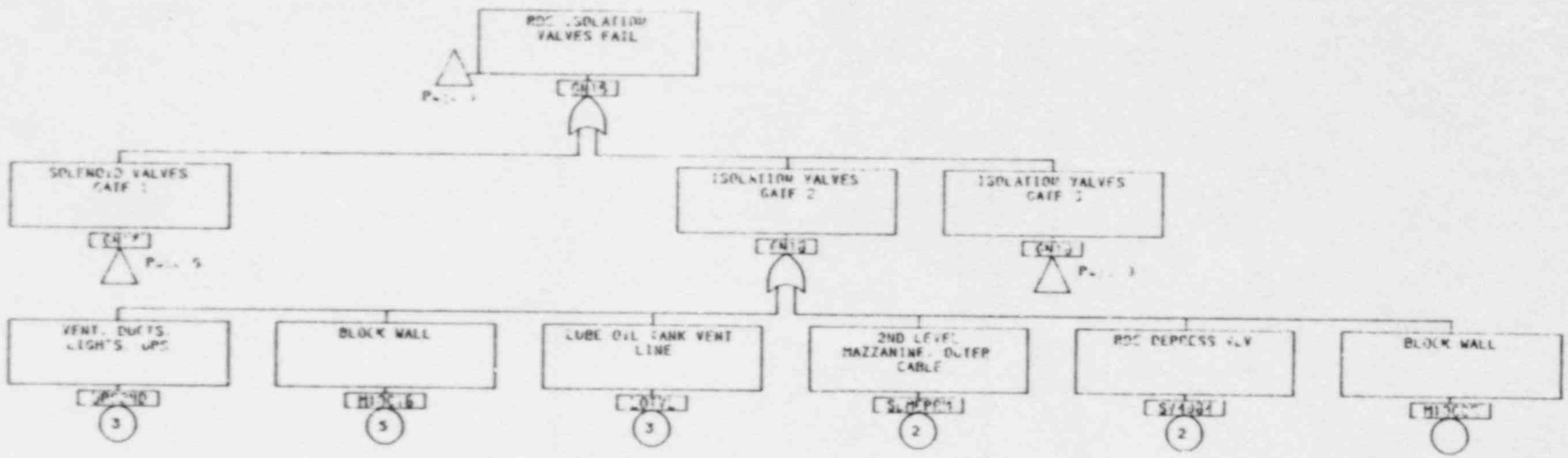

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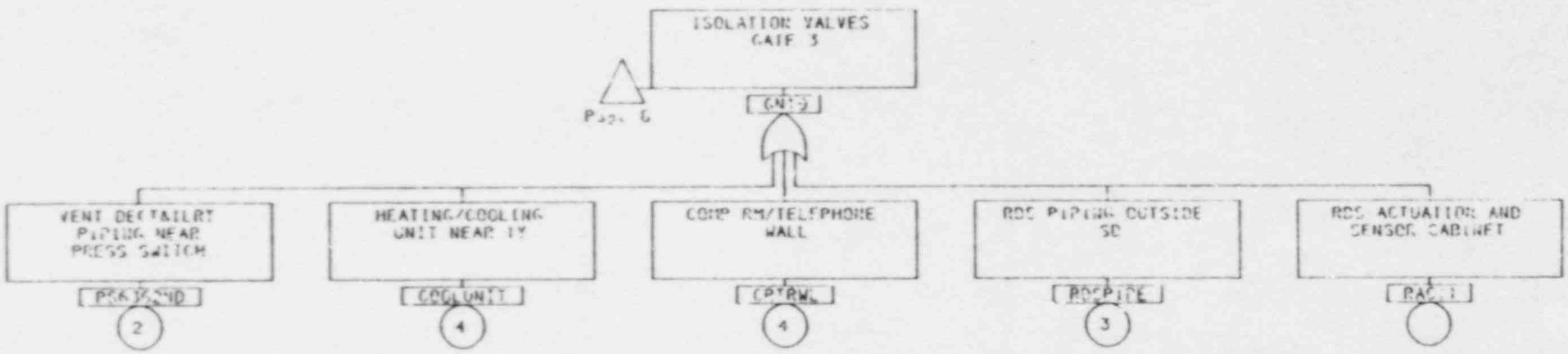

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EMERGENCY CONDENSER VALVE FAILURE W/O AC
FAULT TREES

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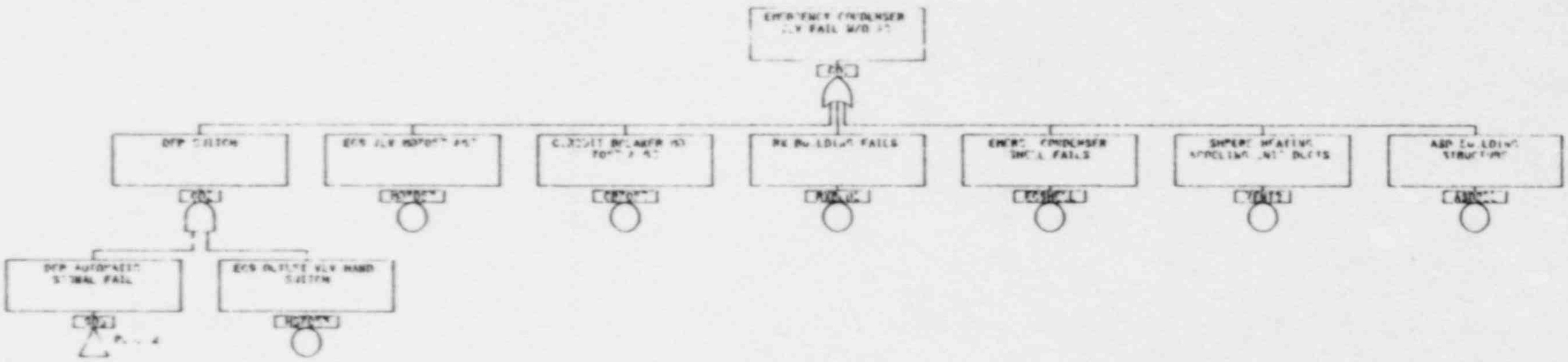

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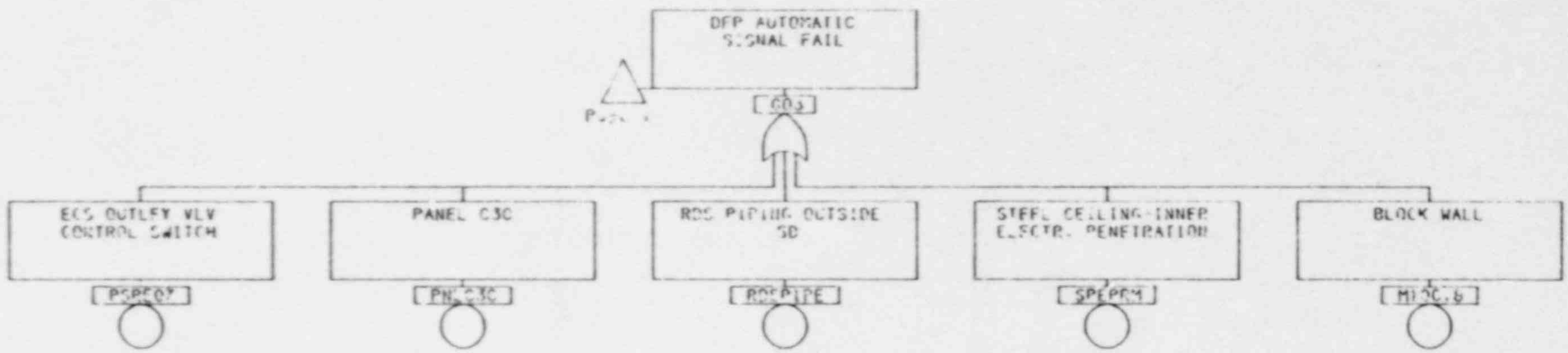

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VLV FAILURE W/O AC

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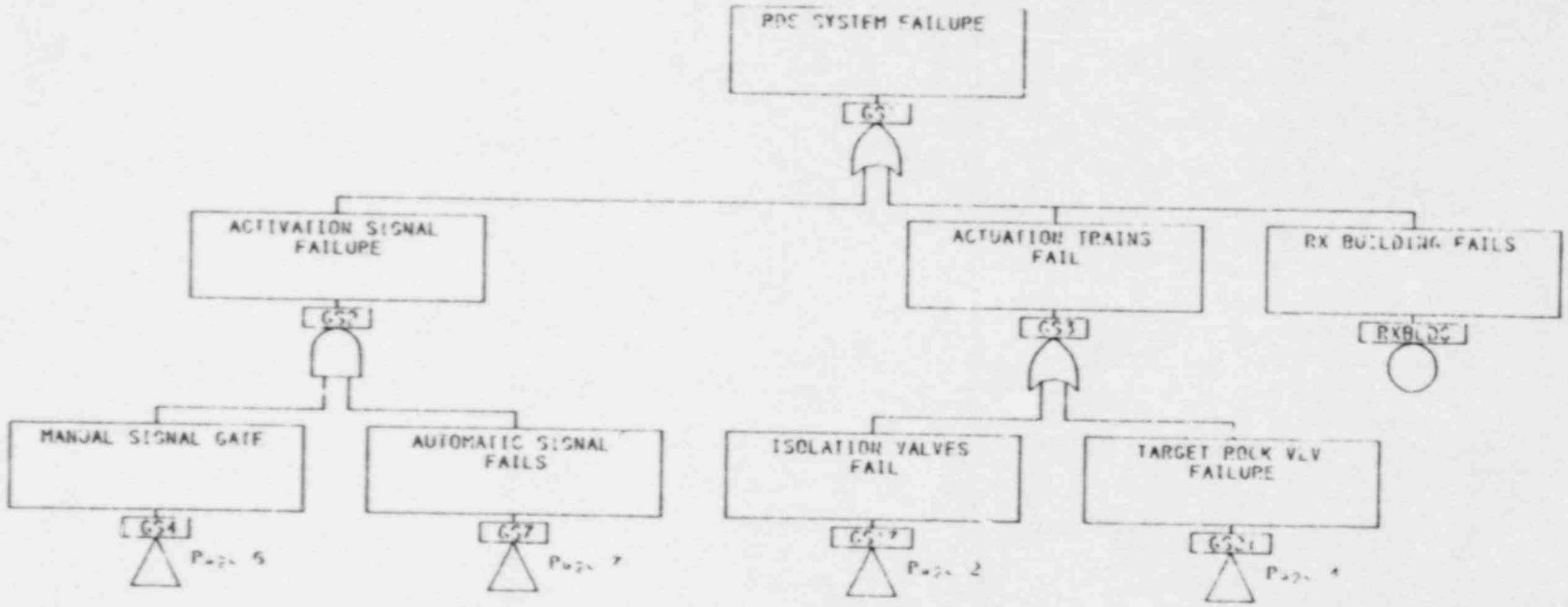

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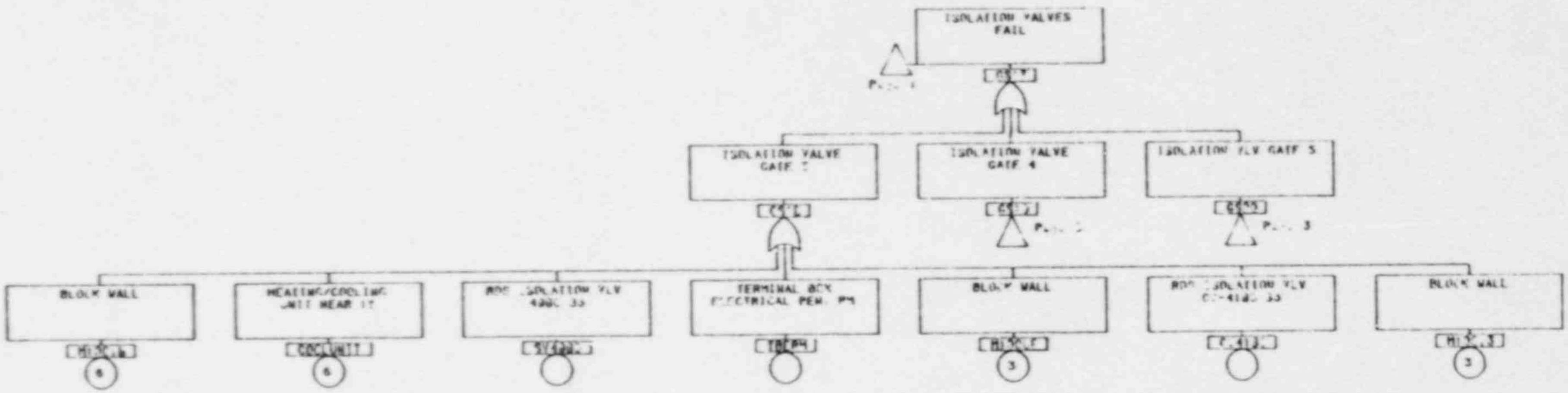

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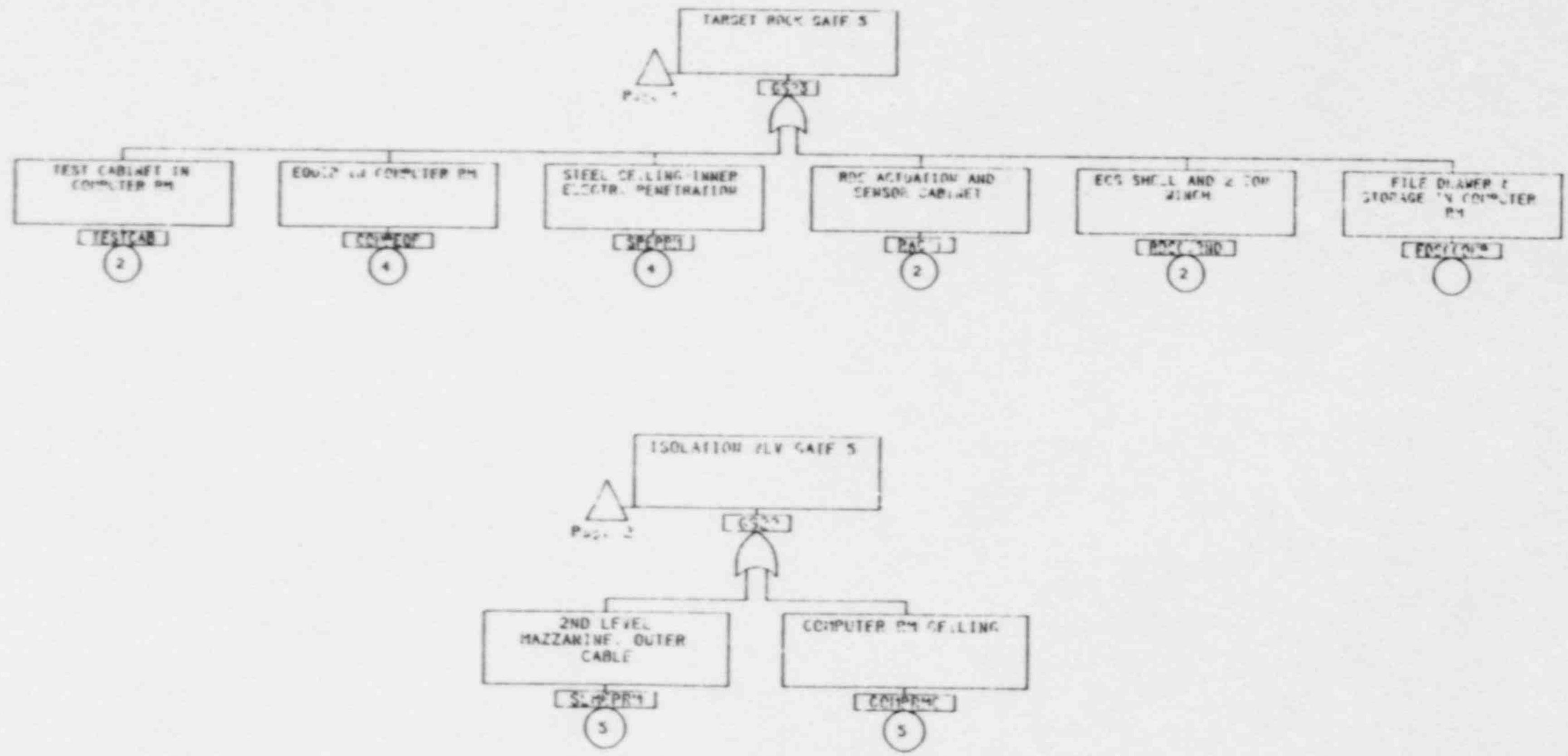

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RDS FAILURE

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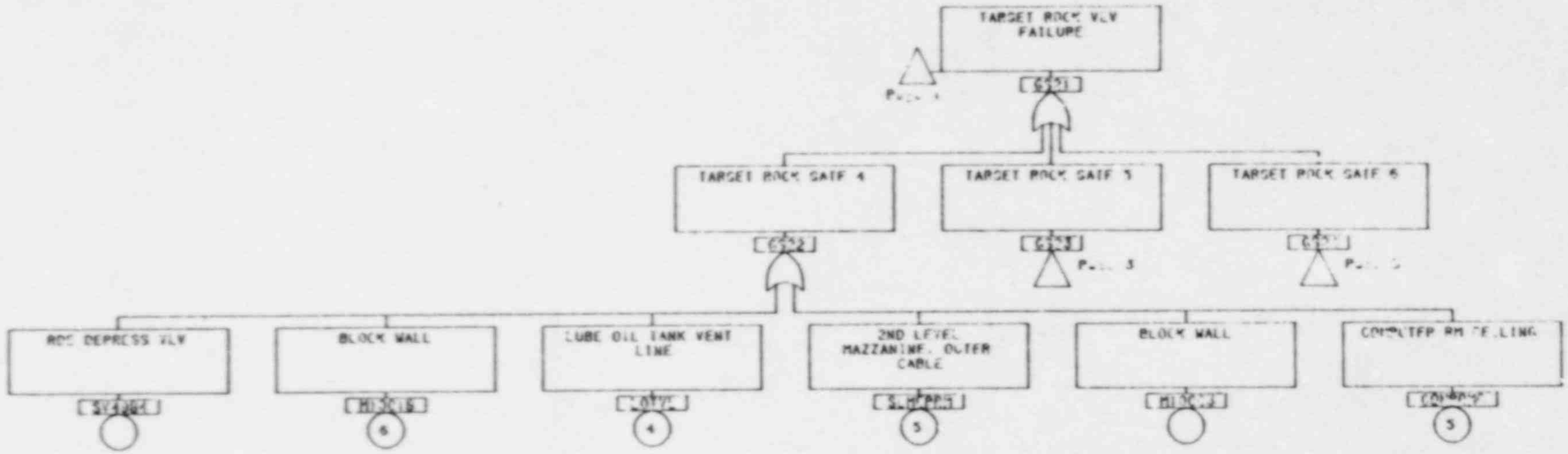

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RDS FAILURE

DRAWING NUMBER	DATE
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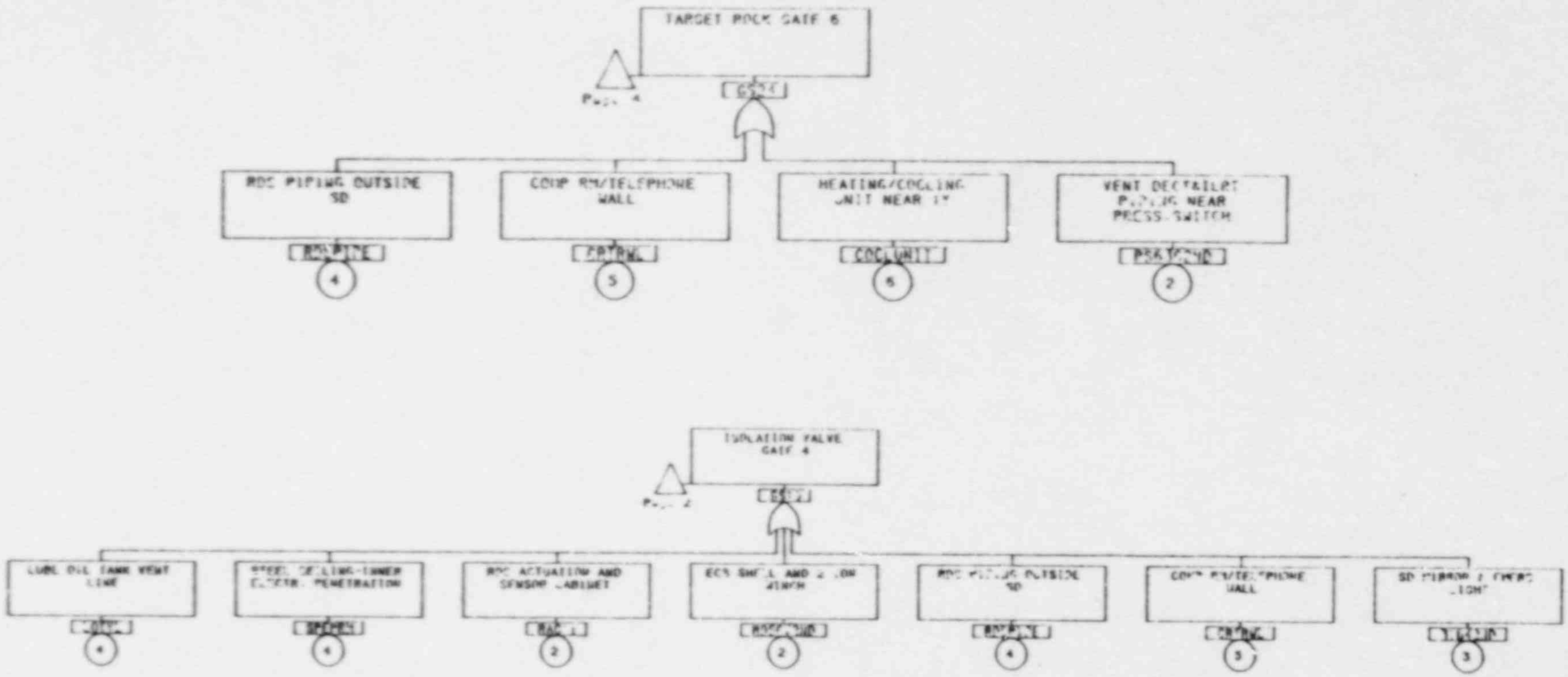

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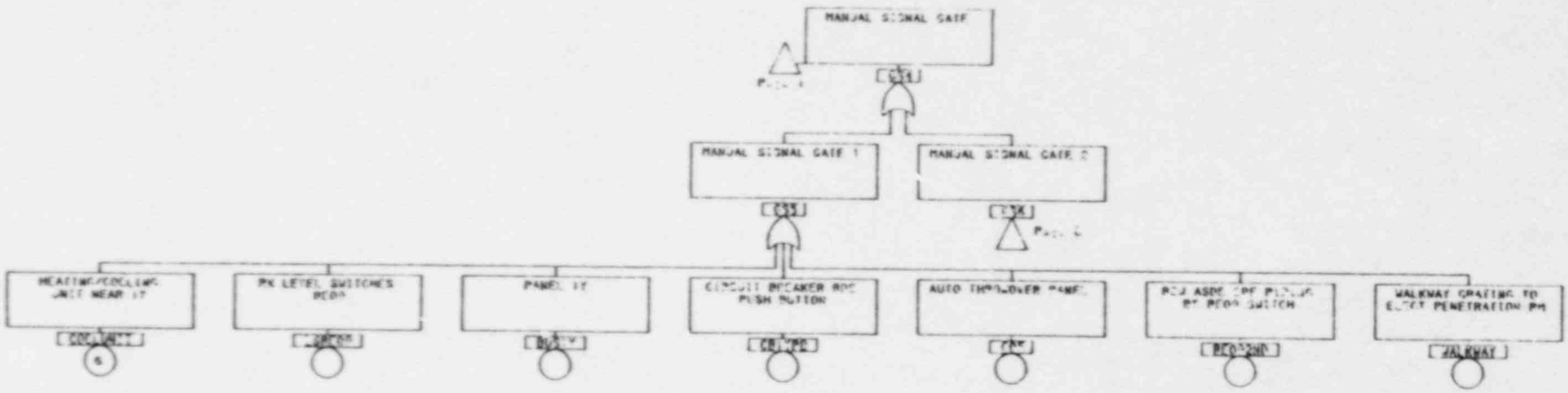

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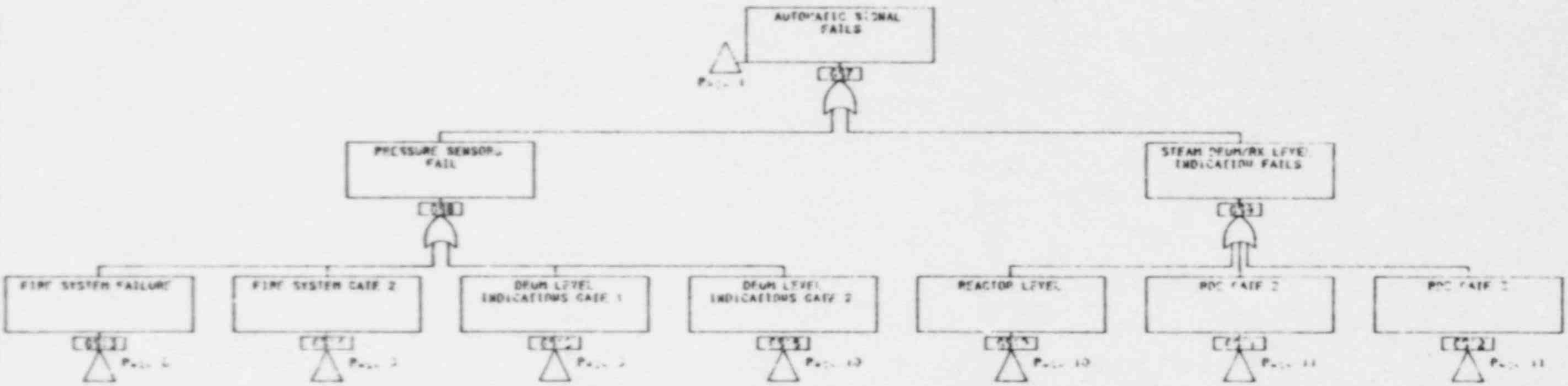

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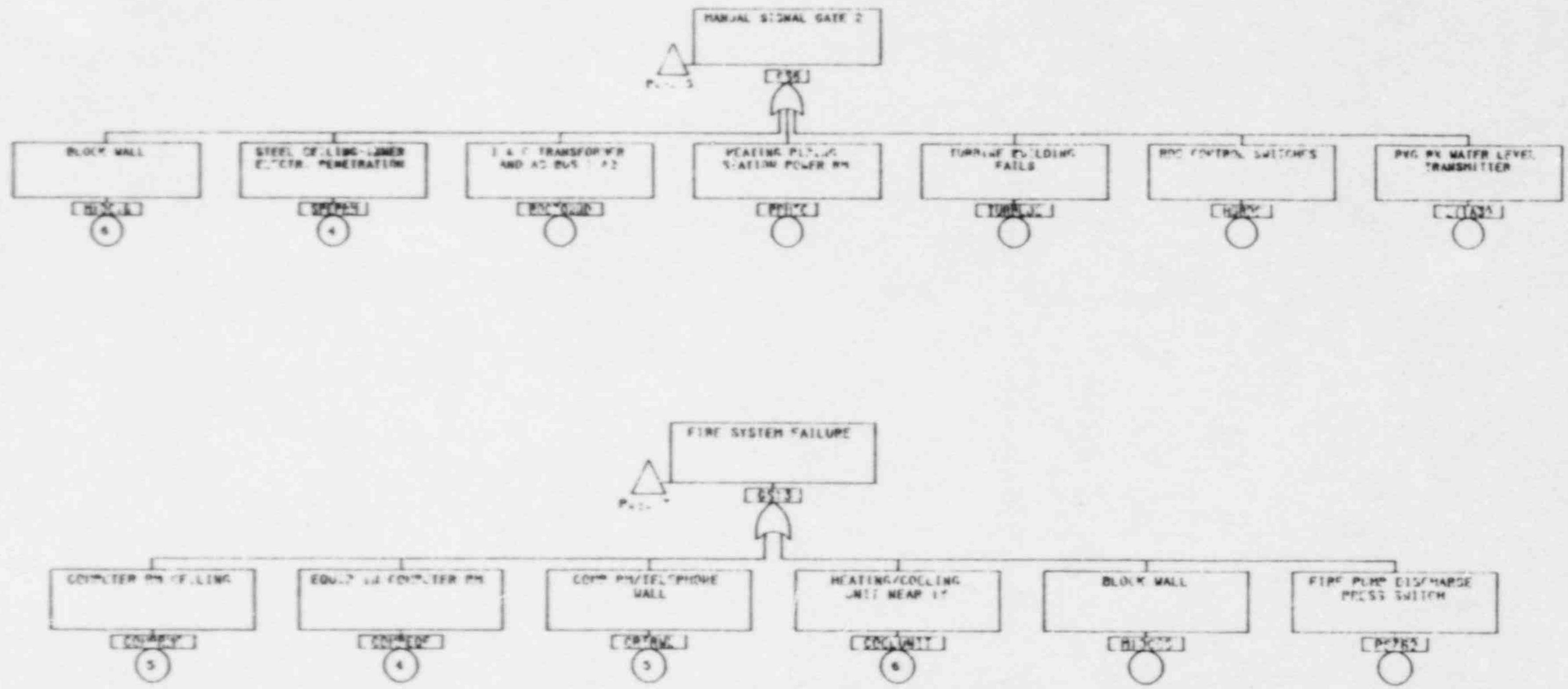

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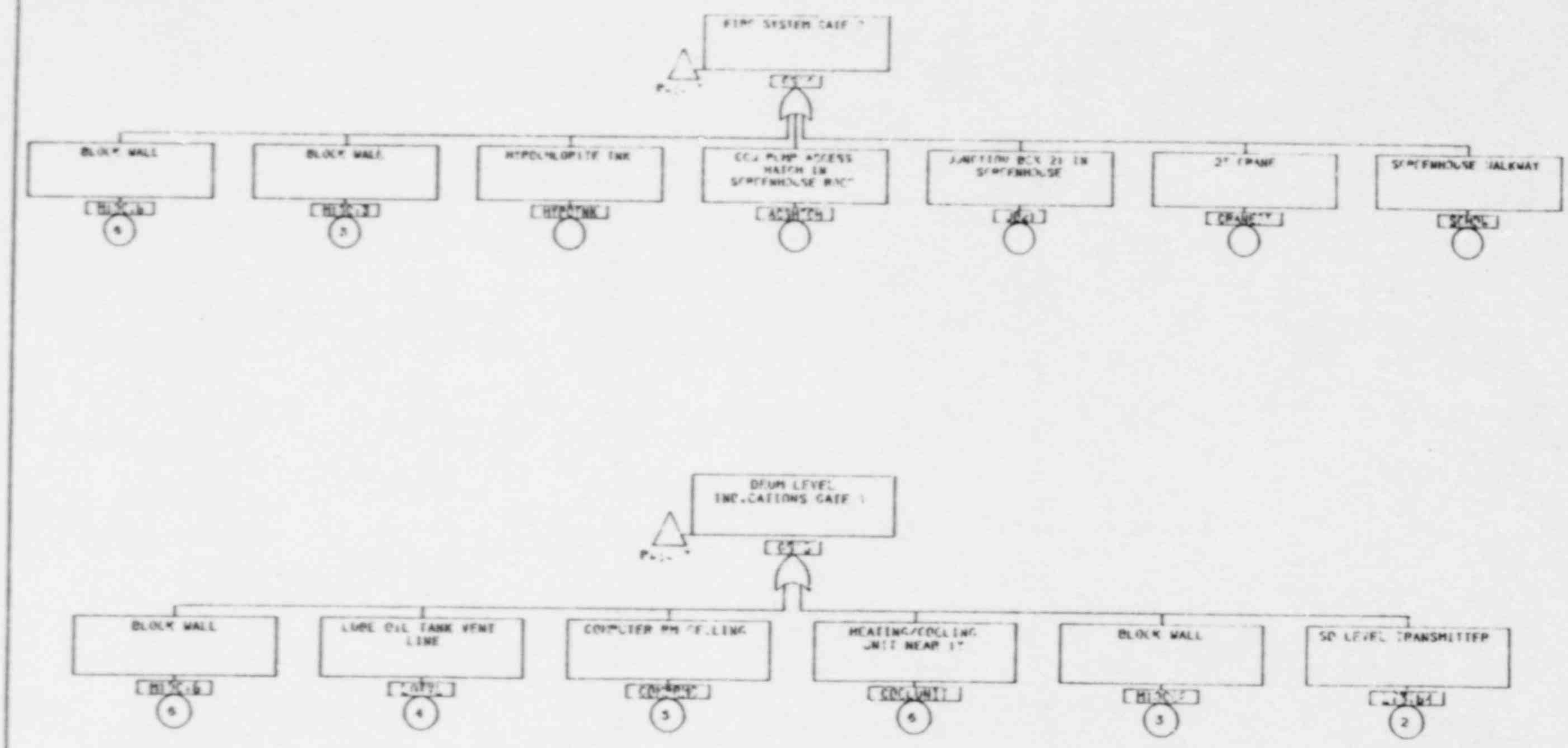

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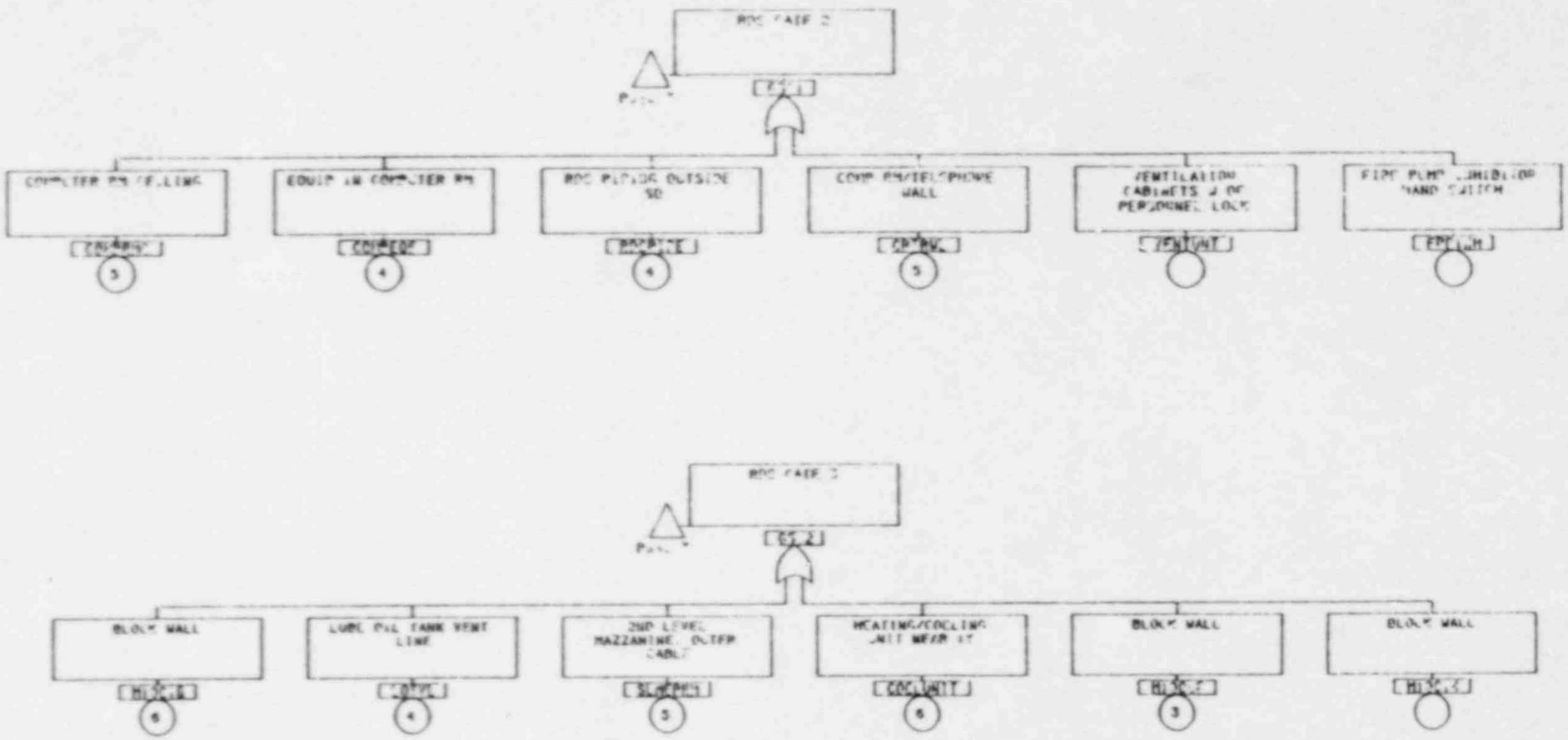

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RDS FAILURE

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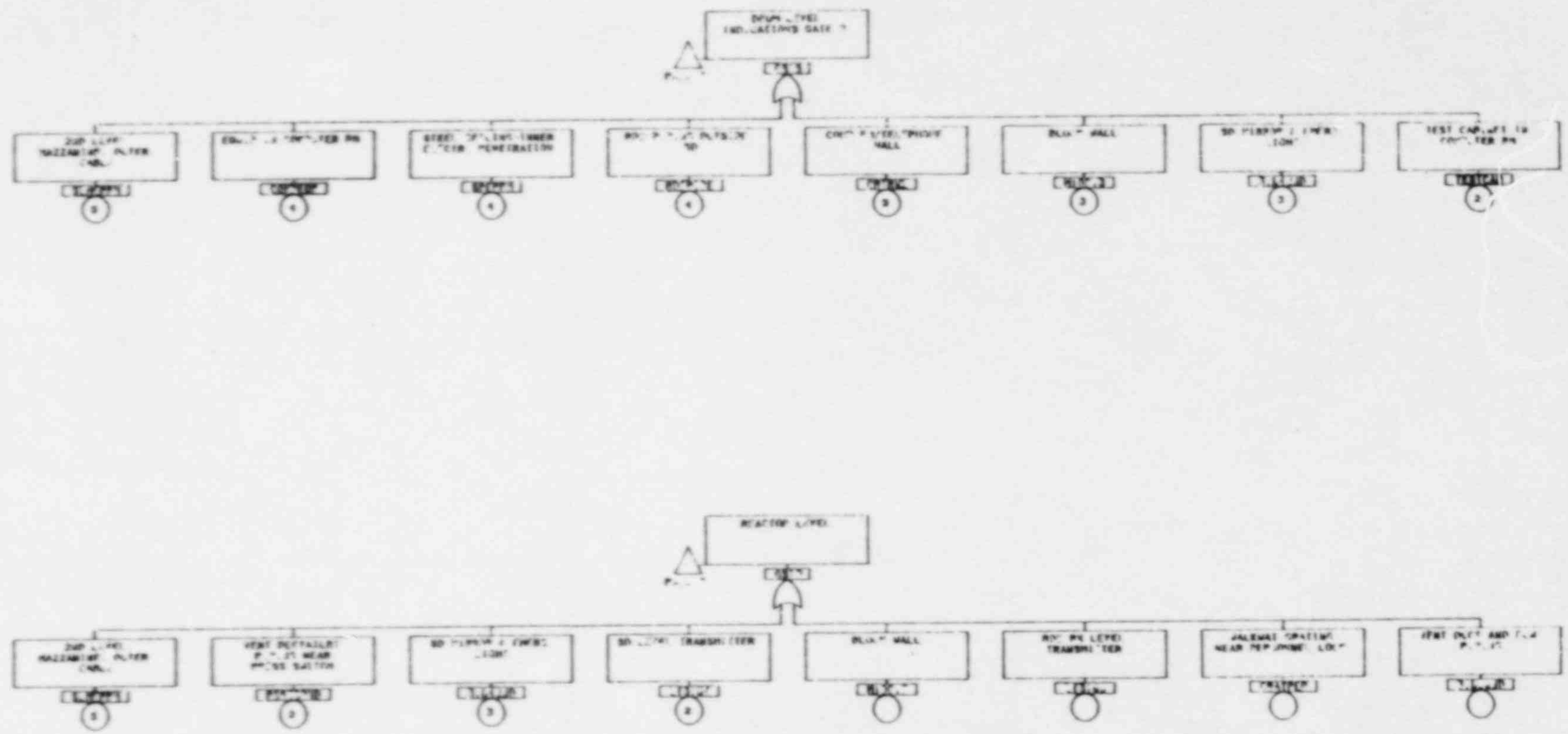

TITLE	
RDS FAILURE	
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CORE SPRAY SYSTEM FAILURE
FAULT TREES

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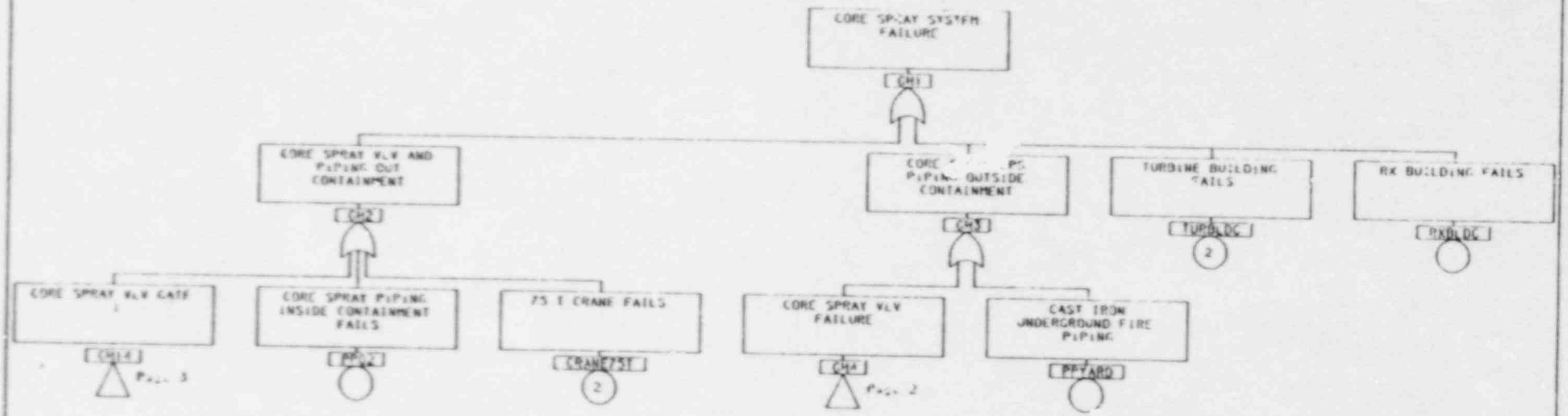

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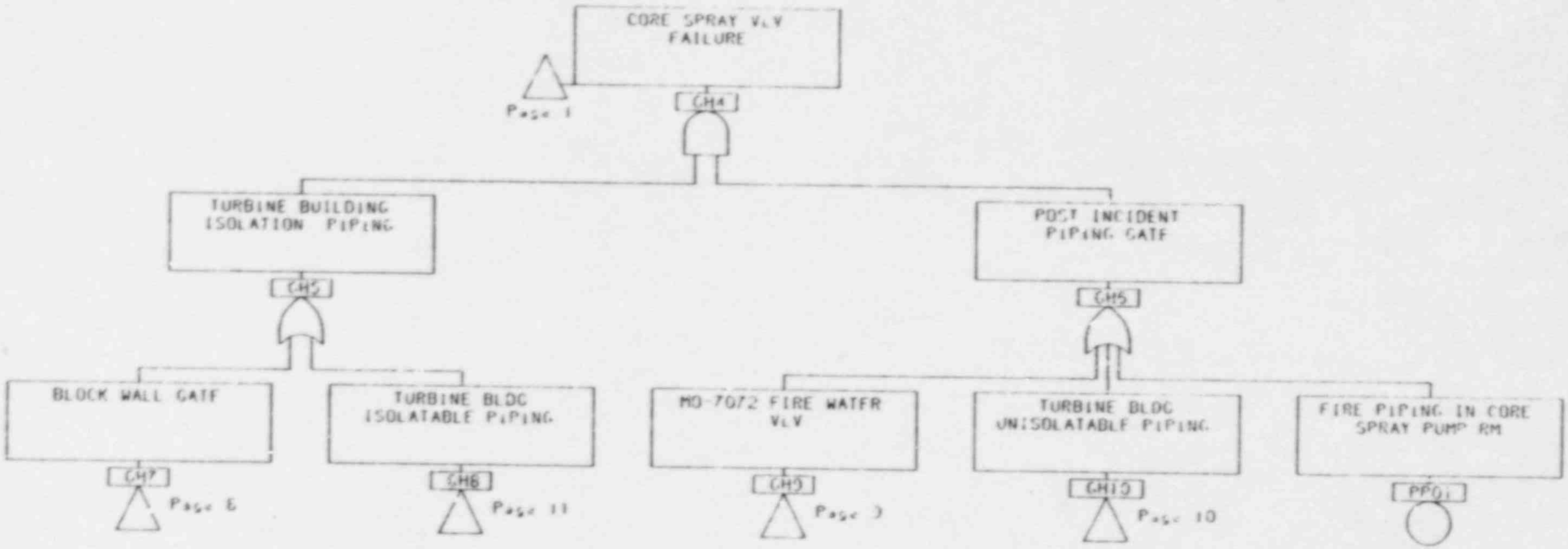
DRAWING NUMBER Page 1	DATE 4/29/88



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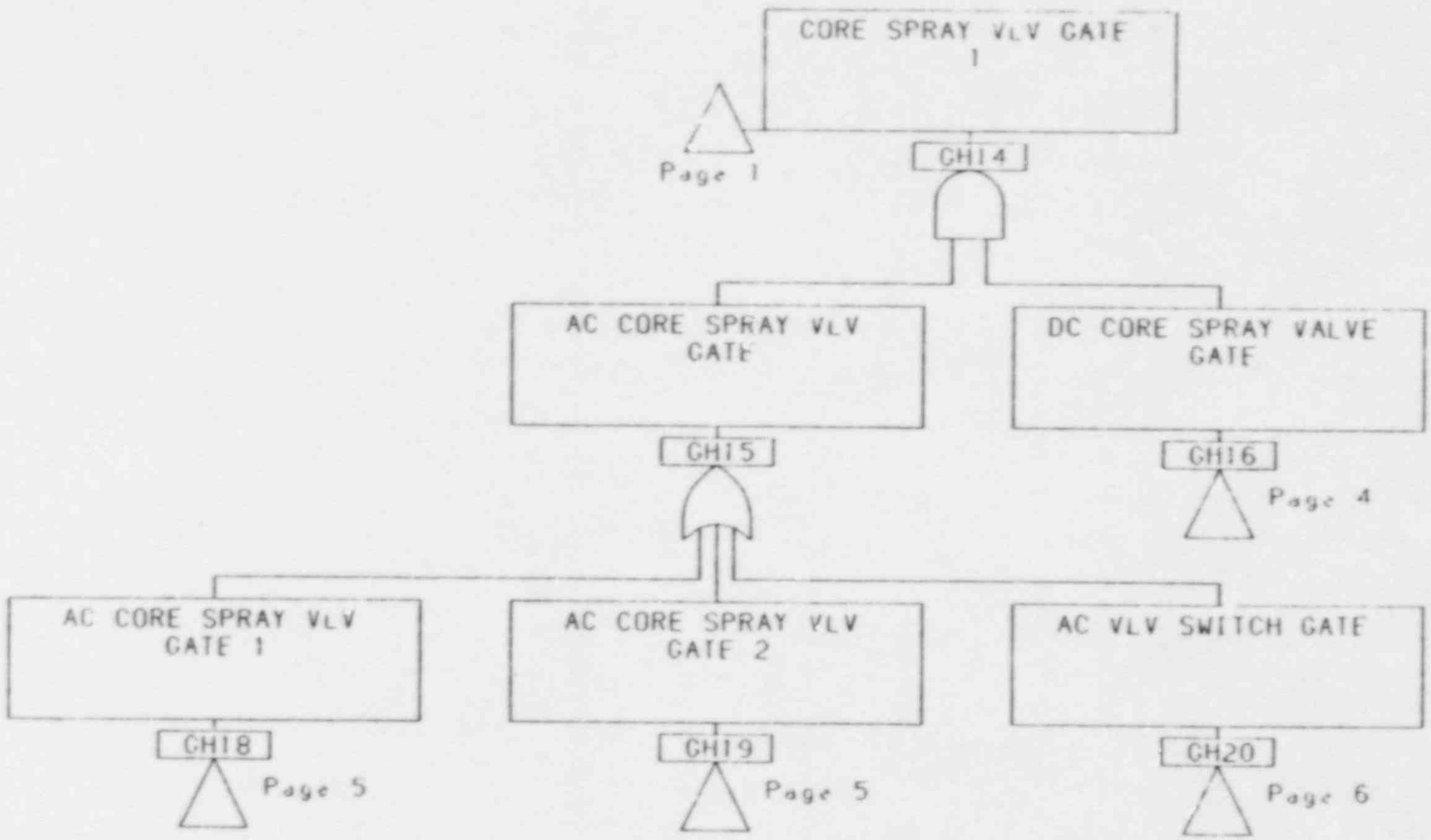

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CORE SPRAY SYSTEM FAILURE

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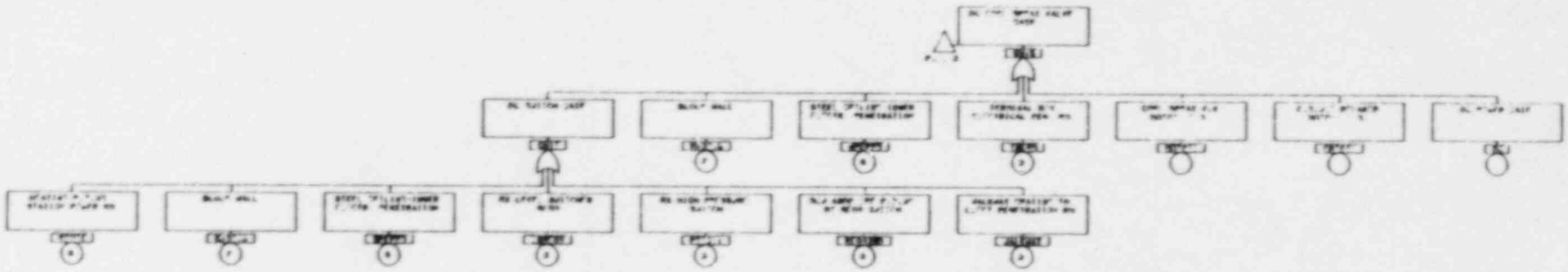

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CORE SPRAY SYSTEM
FAILURE

DRAWING NUMBER	DATE
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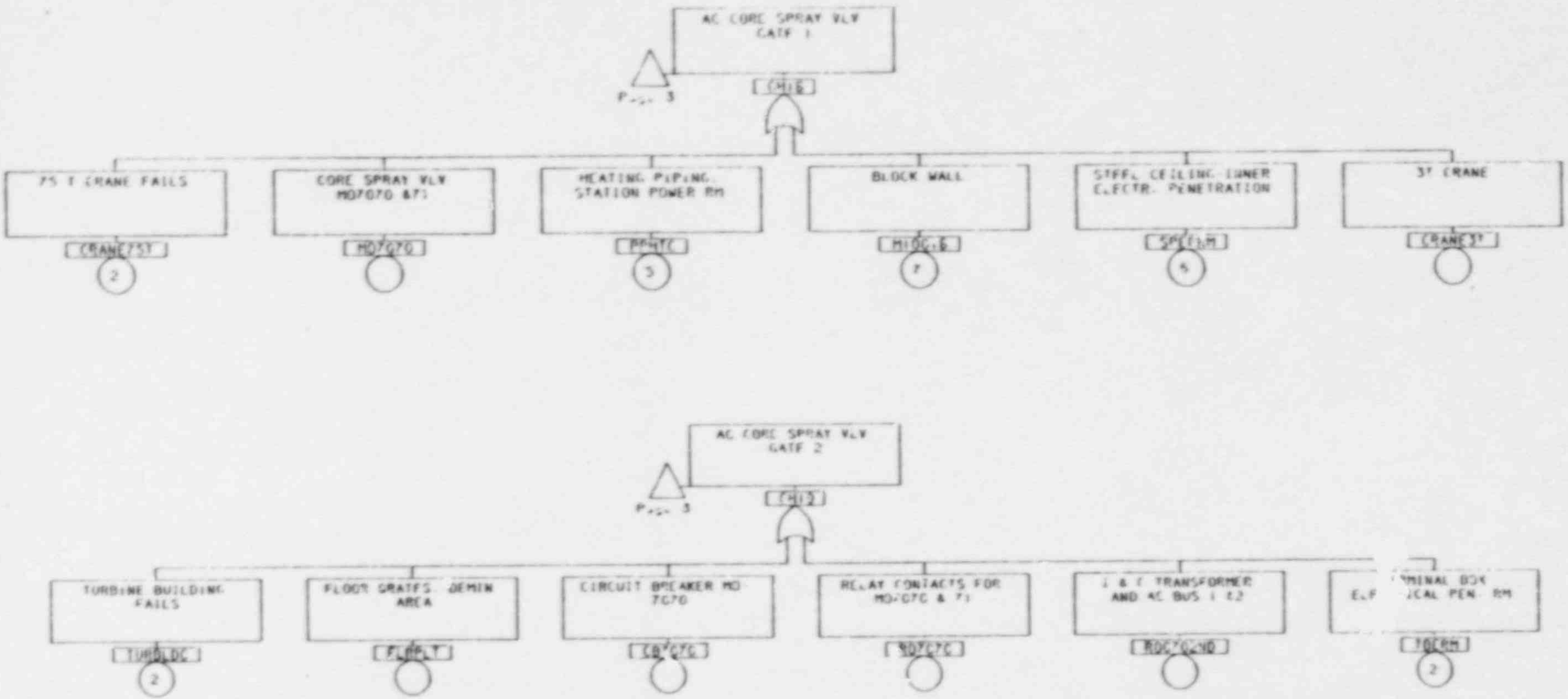

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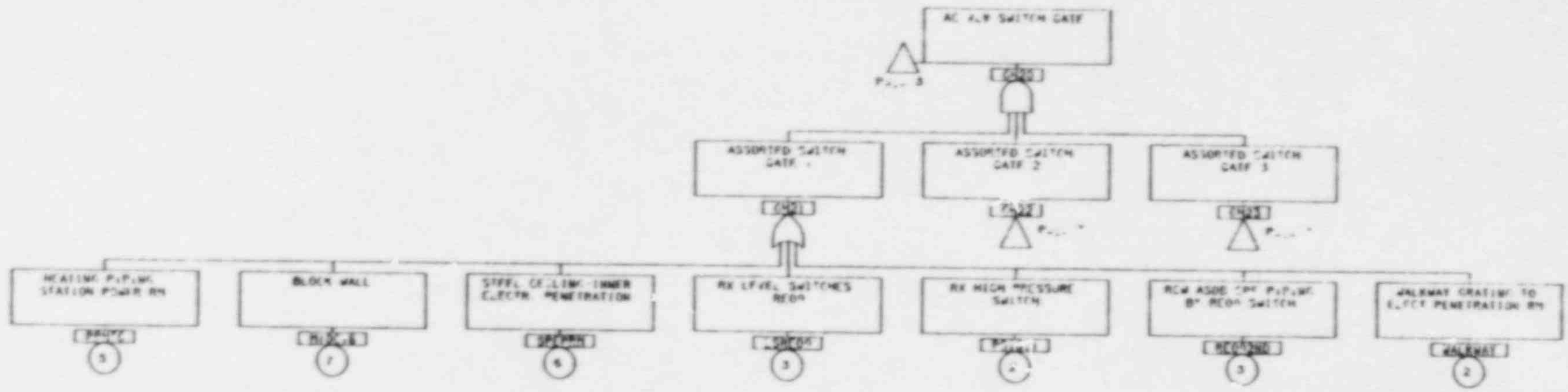
DRAWING NUMBER	DATE
Page 5	4/29/88



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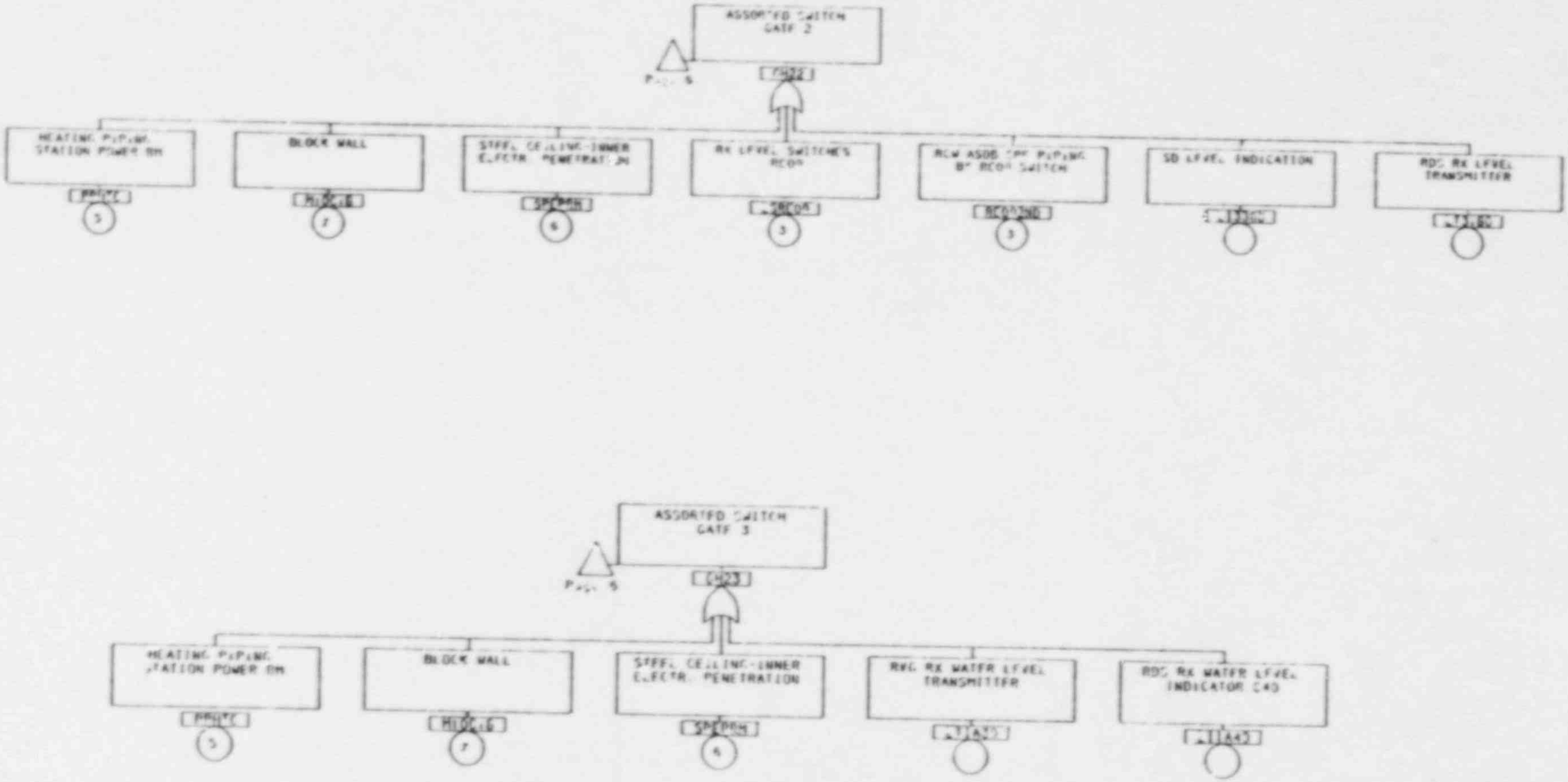
CORE SPRAY SYSTEM FAILURE

DRAWING NUMBER	DATE
Page 6	4/29/88



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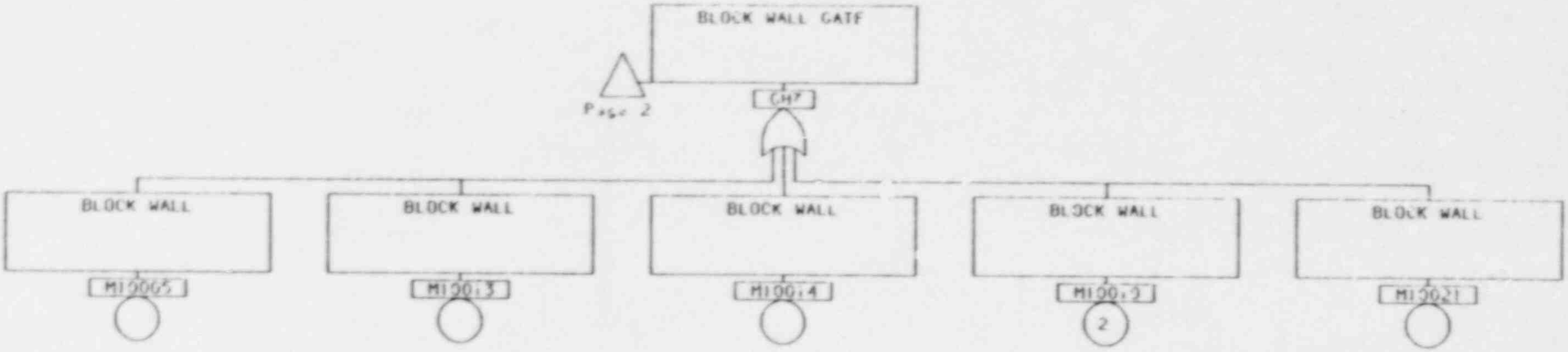
CORE SPRAY SYSTEM FAILURE

DRAWING NUMBER	DATE
Page 7	4/29/88



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CORE SPRAY SYSTEM FAILURE

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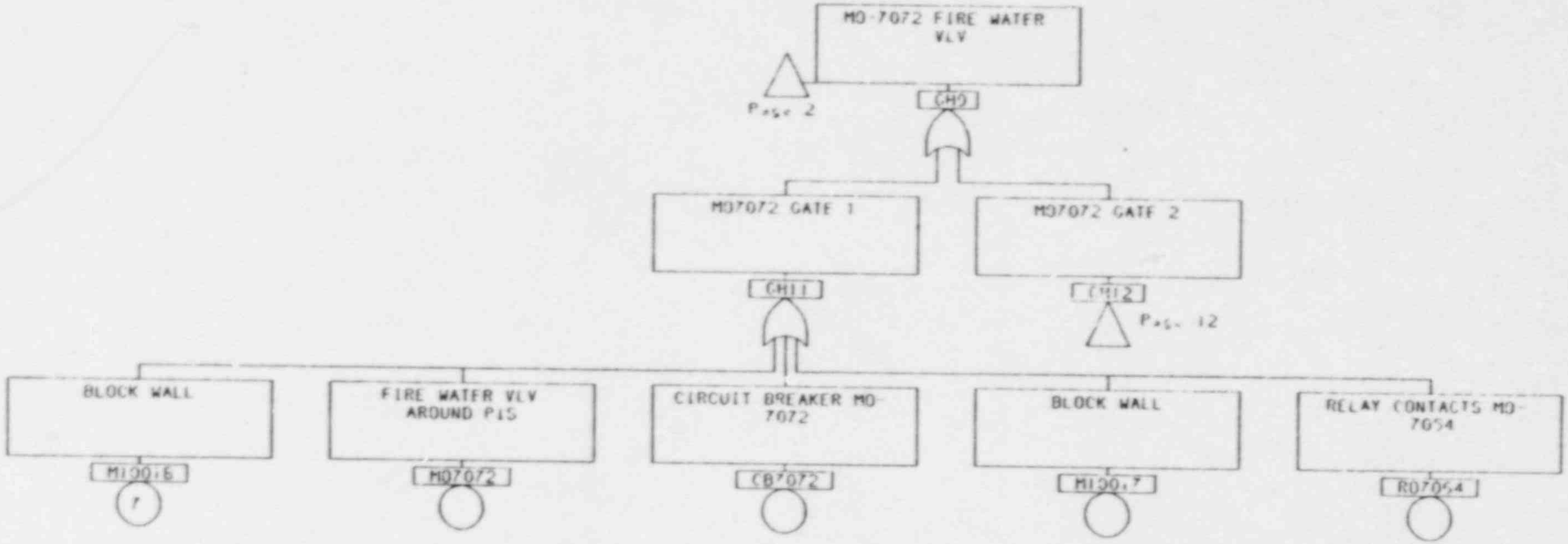
DATE

4/29/88



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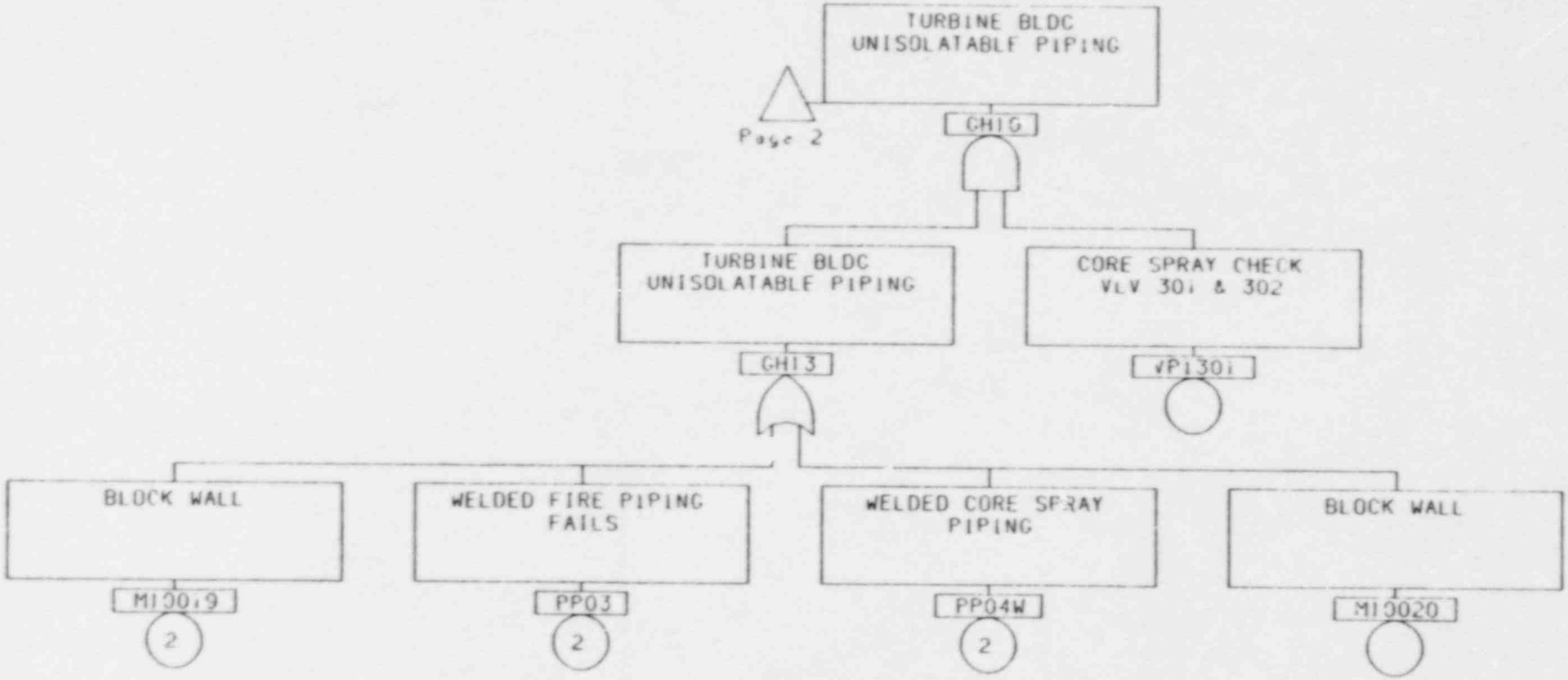
DRAWING NUMBER Page 9	DATE 4/29/88



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CORE SPRAY SYSTEM FAILURE

DRAWING NUMBER	DATE
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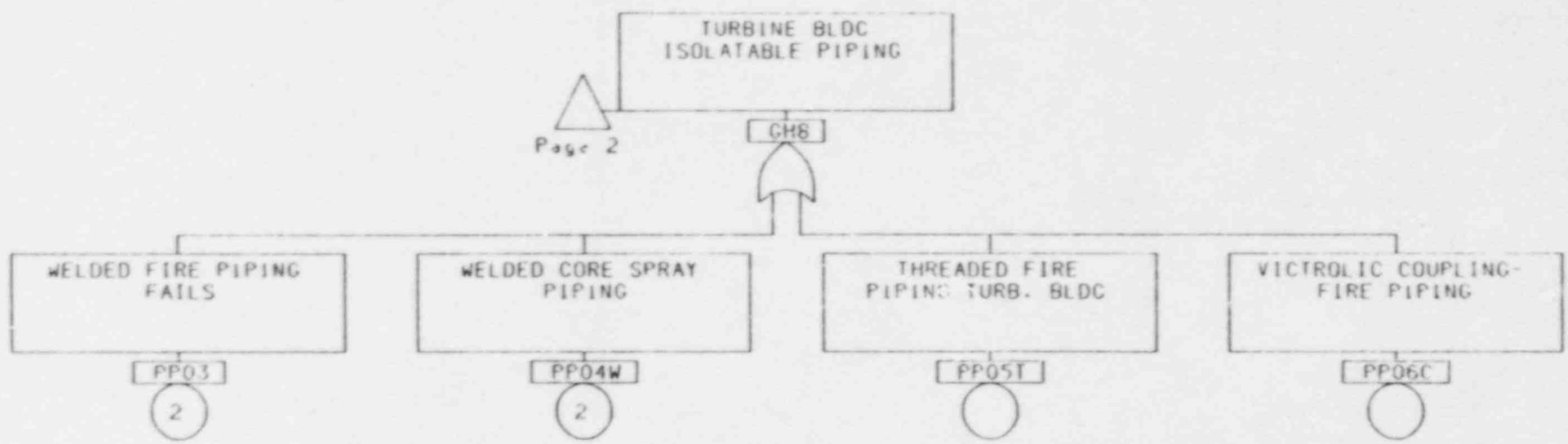

TITLE
CORE SPRAY SYSTEM FAILURE

DRAWING NUMBER	DATE
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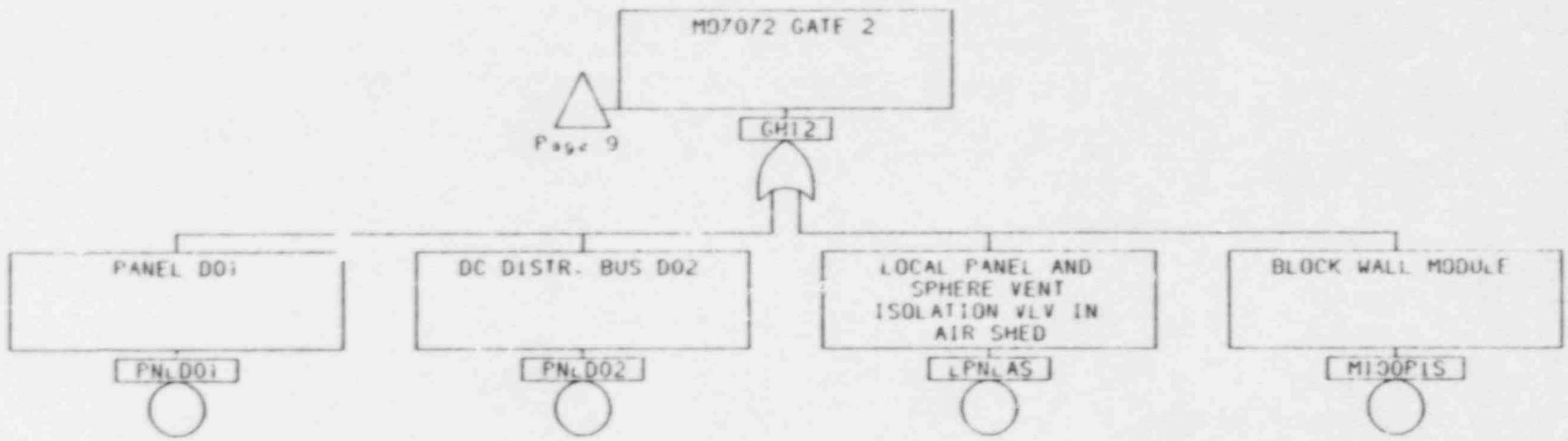

TITLE
CORE SPRAY SYSTEM FAILURE

DRAWING NUMBER	DATE
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APPENDIX B

SEISMIC FRAGILITY OF BIG ROCK POINT CORE ASSEMBLY AND REACTOR VESSEL SUPPORTS

NU0383-2628A-BQ01-NL04

ADDENDUM

EVALUATION OF ALTERNATE SHUTDOWN
BUILDING MODIFICATION

EVALUATION OF ALTERNATE SHUTDOWN BUILDING MODIFICATION OMISSION

During the review process, it was determined that certain pieces of equipment associated with the Alternate Shutdown Building Modification had not been included in the analysis. This omission has been evaluated and it has been determined that there is no need to reanalyze the fault trees and the event trees.

The Equipment that was not included in the analyses is an follows:

1. Alternate Shutdown Building and Equipment Located Within

   This building was designed and constructed to be seismically resistant. The limiting ground acceleration has been determined to be 0.6G, a great deal higher than the SSE of 0.12G, therefore, this omission has no impact on seismic resistance.

2. Underground Conduit from the Alternate Shutdown Building to the Equipment Lock

   This conduit is buried 6 feet underground and was designed and installed to meet seismic requirements, therefore, this omission has no impact on seismic resistance.

3. Conduit from Equipment Lock Area to Equipment

   The conduit runs around the perimeter of containment and is in no danger of degradation from falling objects, therefore, this omission has no impact on seismic resistance.

4. Block Wall M10023

   Conduits from the Alternate Shutdown Building to the control room penetrate this wall. No seismic resistance analysis has been performed on this wall so the most limiting block wall ground acceleration has been assigned (0.12G).

Of the above omissions, only block wall M10023 is considered a threat to the structural integrity of the equipment in question.

In all fault trees associated with the omission (MSIV, EC-VALVES, EC-MAKE-UP), block wall M10023 is a single failure. That is, it alone can fail the above mentioned equipment. Therefore, replacement of the above systems' contribution in the existing cutsets with M10023 produces a list of cutsets that were omitted because of the error.

Below is a list of new cutsets developed by replacing the above failures.

LOSP18	1.20E-01	M10023 * RDSCV2ND
LOSP17	1.10E-01	M10023 * CRANE3T * M07051
LOSP17,26	1.20E-01	M10023 * M10021 * VP1301 * M10020
LOSP17,26	1.10E-01	M10023 * M07070 * M07051
LOSP3	1.10E-01	M10023 * IST-7 * IST-9
LOSP3	1.10E-01	M10023 * PS6362ND * IST-7
LOSP18	1.10E-01	M10023 * PS6362ND
LOSP18,27	1.10E-01	M10023 * RDSCV2ND
LOSP26	1.20E-01	M10023 * VP1301 * M10020 * M10021
LOSP25	1.20E-01	M10023 * VP1301 * M10020
LOSP16	1.20E-01	M10023 * VP1031 * IST-6
LOSP1	1.10E-01	M10023 * IST-8
LOSP17	1.10E-01	M10023 * CRANE3T * M07051
LOSP2	1.10E-01	M10023 * IST-8

M10023 is the limiting component in a few of the above cutsets. Therefore, it should be considered as a weak-link in the seismic issue and should be addressed. An analysis was never performed on this block wall because important equipment was never affected by its failure until the Alternate Shutdown Modification was installed. By conservative assumption, the lowest ground acceleration determined by analysis for all other block walls at Big Rock Point was assigned to this wall (0.11G). Since this analysis shows the block wall's importance, M10023 will be included in the block wall reanalysis, associated with the proposed fixes for block walls M10018 and M10021.

Below is a list of new cutsets developed by replacing the above failures.

LOSP18	1.20E-01	M10023 * RDSCV2ND
LOSP17	1.10E-01	M10023 * CRANE3T * M07051
LOSP17,26	1.20E-01	M10023 * M10021 * VP1301 * M10020
LOSP17,26	1.10E-01	M10023 * M07070 * M07051
LOSP3	1.10E-01	M10023 * IST-7 * IST-9
LOSP3	1.10E-01	M10023 * PS6362ND * IST-7
LOSP18	1.10E-01	M10023 * PS6362ND
LOSP18,27	1.10E-01	M10023 * RDSCV2ND
LOSP26	1.20E-01	M10023 * VP1301 * M10020 * M10021
LOSP25	1.20E-01	M10023 * VP1301 * M10020
LOSP16	1.20E-01	M10023 * VP1031 * IST-6
LOSP1	1.10E-01	M10023 * IST-8
LOSP17	1.10E-01	M10023 * CRANE3T * M07051
LOSP2	1.10E-01	M10023 * IST-8

M10023 is the limiting component in a few of the above cutsets. Therefore, it should be considered as a weak-link in the seismic issue and should be addressed. An analysis was never performed on this block wall because important equipment was never affected by its failure until the Alternate Shutdown Modification was installed. By conservative assumption, the lowest ground acceleration determined by analysis for all other block walls at Big Rock Point was assigned to this wall (0.11G). Since this analysis shows the block wall's importance, M10023 will be included in the block wall reanalysis, associated with the proposed fixes for block walls M10018 and M10021.

SMA 13703.01

SEISMIC FRAGILITY OF BIG ROCK POINT CORE ASSEMBLY AND REACTOR VESSEL SUPPORTS

prepared for

CONSUMERS POWER COMPANY
Jackson, Michigan

October, 1985

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