CRBRP-4

PROBABILISTIC RISK ASSESSMENT

VOLUME 2

Clinch River Breeder Reactor Plant

TEC TECHNOLOGY for ENERGY CORPORATION
Knoxville, Tennessee

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LIST OF ACRONYMS AND ABBREVIATIONS

ABHX	Air blast heat exchanger
AC	Alternating current
ACRS	Advisory Committee on Reactor Safeguards
ACS	Annulus cooling system
AF	Annulus filter
AFS	Annulus filtration system
AFW	Auxiliary feedwater
AHU	Air handling unit
ALMS	Auxiliary liquid metal system
AOV	Air operated valve
APM	Annulus pressure maintenance
ASME	American Society of Mechanical Engineers
B&R	Burns and Roe
BOC	Beginning of cycle
BOP	Balance of plant
BSC	Backup sodium cooler
BWR	Boiling water reactor
C/B	Circuit breaker
CAPS	Cell atmosphere processing system
CB	Control building
CCDF	Complementary cumulative distribution function
CCF	Common cause failure
CCS	Containment cleanup system
CES	Containment exhaust system
CFE	Containment fails early
CIS	Containment isolation system
CLCV	Cold leg check valve
CPS	Containment purge system
CRBRP	Clinch River Breeder Reactor Plant
CRD	Control rod drive
CRDM	Control rod drive mechanism
CST	Condensate storage tank
CWPH	Circulating water pumphouse
DBA	Design basis accident
DBT	Design basis tornado
DC	Direct current
DGB	Diesel generator building
DHRS	Direct heat removal service
DP	Distribution panel
DST	Deaerating storage tank
EBR	Experimental Breeder Reactor
ECP	Engineering change procedure
ECT	Emergency cooling towers
ECW	Emergency chilled water
EDG	Emergency diesel generator
EEB	Electrical equipment building
EH	Electrohydraulic
EM	Electromagnetic
EOC	End of cycle

EOI	Emergency operating instructions
EPZ	Emergency Planning Zone
EPRI	Electric Power Research Institute
EPSW	Emergency plant service water
EVS	Ex-vessel storage
EVSPS	Ex-vessel storage and processing system
EVST	Ex-vessel storage tank
EVTM	Ex-vessel transfer machine
FAI	Fails as is
FA	Fauske and Associates
FAA	Federal Aviation Administration
FC	Fails closed
FCI	Fuel coolant interactions
FFTF	Fast Flux Test Facility
FHC	Fuel handling cell
FM	Flow meter
FMEA	Failure modes and effects analysis
FO	Fails open
FPD	Full power days
GE	General Electric
GPM	Gallons per minute
GV	Guard vessel
HAA	Head access area
HCDA	Hypothetical core disruptive accident
HEDL	Hanford Engineering Development Laboratory
HEP	Human event probability
HPH	High pressure heater
HRA	Human reliability assessment
HRS	Heat rejection system
HTS	Heat transport system
HVAC	Heating, ventilation, and air conditioning
HX	Heat exchanger
I&C	Instrumentation and control
IA	Instrument air
IE	Initiating event
IGRP	Inert gas receiving and processing
IHTS FC	Intermediate heat transport system forced circulation
IHTS NC	Intermediate heat transport system natural circulation
IHTS	Intermediate heat transport system
IHX	Intermediate heat exchanger
ILRT	Integrated leak rate test
INEL	Idaho National Engineering Laboratory
IREP	Interim Reliability Evaluation Program
IRP	Intermediate rotating plug
ISI	Inservice inspection
IVTM	In-vessel transfer machine
LAB	Lower axial blanket
LC	Locked closed
LCO	Limiting conditions for operations
LDS	Leak detection subsystem
LER	Licensee event report
LHE	Latent human error

LHSE	Loss of heat sink early
LHSL	Loss of heat sink late
LMFBR	Liquid metal fast breeder reactor
LO	Locked open
LOCA	Loss of cooling accident
LOF-d-TOP	Loss of flow driven transient overpower
LOP	Loss of offsite power
LOS	Loss of sodium
LP	Low pressure
LPH	Low pressure heaters
LRP	Large rotating plug
LSP	Large scale pool
LTHS	Long term heat sink
LWR	Light water reactor
MCC	Motor control center
MCR	Main control room
MDAFWP	Motor driven auxiliary feedwater pump
MFW	Main feedwater
MG	Motor generator
MM	Modified Mercalli
MOP	Maintenance outline procedure
MOV	Motor operated valve
MS&W	Maintenance shop and warehouse
MTBF	Mean time between failure
MTBM	Mean time between maintenance
MTBT	Mean time between test
MTTR	Mean time to repair
Na	Sodium
NaK	Sodium potassium
NC	Normally closed
NCW	Normal chilled water
NDHX	Natural draft heat exchanger
NI	Nuclear island
NO	Normally open
NPSH	Net positive suction head
NPSW	Normal plant service water
NRC	Nuclear Regulatory Commission
NREP	National Reliability Evaluation Program
NSSFC	National Severe Storms Forecast Center
NSSS	Nuclear steam supply system
NWL	Normal water level
OBE	Operating basis earthquake
OHX	Overflow heat exchanger
OPRA	Oconee Probabilistic Risk Assessment
ORNL	Oak Ridge National Laboratory
OSIS	Outlet steam isolation subsystem
P&ID	Piping and instrumentation drawings
PACC	Protected air cooled condenser
PCRDM	Primary control rod drive mechanism
PHTS FC	Primary heat transport system forced circulation
PHTS NC	Primary heat transport system natural circulation

PHTS	Primary heat transport system
PIC	Pressure indicator controller
PLHS	Protected loss of heat sink
PLOF	Protected loss of flow
PM	Permanent magnet
PO	CRBRP Project Office
PPS	Plant protection system
PRA	Probabilistic Risk Assessment
PRSS	Primary reactor shutdown system
PSAR	Preliminary Safety Analysis Report
PSB	Plant service building
PSIG	Pounds per square inch gauge
PSP	Primary sodium pump
PSSPS	Primary sodium storage and processing system
PSW	Plant service water
PWR	Pressurized water reactor
PWS	Protected water supply
PWST	Protected water storage tank
RAPS	Radioactive argon processing system
RAT	Reserve auxiliary transformer
RCB	Reactor containment building
RC	Reactor cavity
RCV	Reactor cavity vent
RGC	Recirculating gas cooling
RHTICS	Reactor heat transport instrumentation and control system
RI	Rockwell International
RPS	Reactor protection system
RPT	Reactor pump trip
RSB	Reactor service building
RSS	Reactor shutdown system
RTD	Resistance temperature detector
RV	Reactor vessel
SBTF	Sodium boiling test facility
SCFM	Standard cubic feet per minute
SCRS	Secondary control rod system
SDD	System design description
SDS	Sodium dump subsystem
SERC	Southeastern Electric Reliability Council
SGAB	Steam generator auxiliary building
SGAHRS	Steam generator auxiliary heat removal system
SGB	Steam generator building
SGIB	Steam generator intermediate bay
SGR	Switchgear
SGS	Steam generator system
SGTR	Steam generator tube rupture
SHRS	Shutdown heat removal systems
SIA	Systems interactive analysis
SLI	Success Likelihood Index
SMBDB	Structural margin beyond design basis
SOIV	Superheater outlet isolation valve

SRSS	Secondary reactor shutdown system
SRU	System resource unit
SRV	Safety relief valve
SSCCW	Secondary services closed cooling water
SSE	Safe shutdown earthquake
SSPLS	Solid state programmable logic system
SWR	Sodium water reaction
SWRPRS	Sodium water reaction pressure relief subsystem
TBD	To be determined
TBV	Turbine bypass valve(s)
TDAFWP	Turbine driven auxiliary feedwater pump
TGB	Turbine generator building
TIC	Temperature indicator controller
TLS	Top logic structure
TMBDB	Thermal margin beyond design basis
TOP	Transient overpower
TVA	Tennessee Valley Authority
UAT	Unit auxiliary transformer
UIS	Upper internal structure
ULHS	Unprotected loss of heat sink
ULOF	Unprotected loss of flow
ULOS	Unprotected loss of sodium
USS	Unit substation
UTOP	Unprotected transient overpower
VAC	Vital instrument AC
VPS	Vent and purge system
WARD	Westinghouse Advanced Reactor Division
WDS	Water dump subsystem

Section 10

FREQUENCIES OF CORE-DAMAGE SEQUENCES

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Section 10

FREQUENCIES OF CORE-DAMAGE SEQUENCES

This section describes the sequences that have been determined to contribute to the frequency of core damage for the Clinch River Breeder Reactor Plant (CRBRP). These sequences and their frequencies comprise the results of the plant systems and sequence analyses. This section represents a compilation of the core-damage results for input to the integration with the core- and containment-response results, as described in Section 13. A summary of the conclusions drawn from the core damage results presented in this section (Section 10) are provided in Section 14.

The total core damage frequency is calculated to be 3.6×10^{-5}/yr of which 3.7×10^{-6}/yr comes from internal initiators, 3.2×10^{-5}/yr from external events, and 2.0×10^{-7}/yr from common-cause initiators. These are described further in Sections 10.1, 10.2 and 10.3, respectively.

10.1 RESULTS FOR INTERNAL INITIATORS

This section presents the results of the quantification of the frequencies of the core-damage sequences for internal initiators as described in Section 7. This quantification involved combining the system fault-tree models described in Appendix A according to the logic presented for the event-sequence development in Section 3 to obtain sequence-level minimal cut sets.* These cut sets are then quantified by applying initiating-event frequencies and component failure probabilities.

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*A minimal cut set for a sequence is the combination of events comprised of an initiator and the basic faults that lead to the failure of all systems able to provide core cooling under the conditions of the sequence.

The final step in the quantification process is to estimate the probabilities of operator actions occurring during the accident sequence, including actions taken to restore failed functions by repairing equipment or realigning systems (these are referred to as recovery events). This step is taken last to permit consideration of the effects on the reliability of the operator associated with the events in the cut set (e.g., timing of system failures, effects of failures on control room indicators, etc.). A detailed description of the approach taken in modeling and quantifying the human interactions is provided in Section 6.

The frequencies for the core-damage sequences are summarized in Table 10.1-1. The core-damage event tree for internal initiators is presented in Figure 10.1-1.

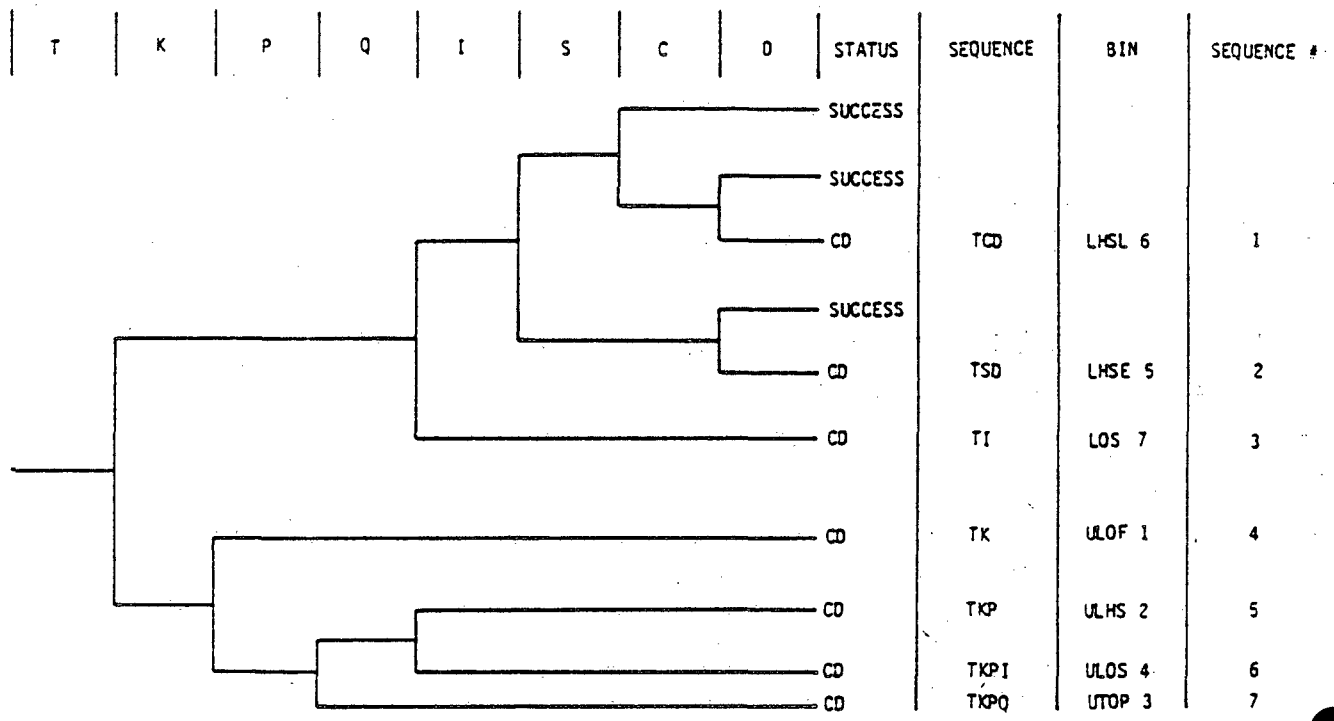
Table 10.1-1

SUMMARY OF CORE-DAMAGE FREQUENCIES
FOR INTERNAL INITIATORS

Bin	Sequence	Description	Frequency
1	TK	Unprotected loss of flow	ε[a]
2	TKP	Unprotected loss of heat sink	2.5-7/yr[b]
3	TKPQ	Unprotected transient overpower	ε
4	TKPI	Unprotected loss of sodium	ε
5	TSD	Loss of heat sink (early)	3.4-6/yr
6	TCD	Loss of heat sink (late)	ε
7	TI	Loss of sodium	ε
		Total	3.7-6/yr

[a]The symbol "ε" indicates that the sequence has been assessed to have a very low frequency ($<10^{-8}$/yr).

[b]2.5-7 = 2.5×10^{-7}



EVENT NAMES

T - Initiator

K - Reactor Shutdown

P - Main PHTS Pump Motors Tripped

Q - Initial Power Normal

I - Sodium Inventory Maintained

S - Short-term Cooling via SGS Maintained

C - Long-term Cooling via SGS Maintained

D - Core Cooling via DHRS Initiated and Maintained

Figure 10.1-1. Core-Damage Event Tree for Internal Initiators

For each of the seven core-damage sequences, the cut sets dominating their frequencies are described. Because of the very large number of cut sets for some sequences, those containing the same initiator and similar subsequent failures have been sorted into groups; these groups are presented in decreasing order of their contribution to the frequency of their respective sequence. The cut sets are identified according to the coded event names used for input to the SETS code for model solution. The events in the cut sets, their probabilities or frequencies, and their sources, are tabulated in Section 10.1.8.

In order to simplify presentation, some cut sets have been factored according to the rules of Boolean algebra, as shown in the example below.

Two cut sets for sequence TSD (prior to the consideration of recovery actions) are as follows:

8.4-8 T4(9.2-2) * S903BE(1.0-3) * EMEDGAS(3.0-2) * EMEDGBS(3.0-2)

7.0-8 T4(9.2-2) * S903BE(1.0-3) * EMEDGAS(3.0-2) * EMEDGBL(2.5-2)

These are equivalent to the following:

1.5-7 T4(9.2-2) * S903BE(1.0-3) EMEDGAS(3.0-2) * [EMEDGBS(3.0-2) + EMEDGBL(2.5-2)]

The first number is the annual frequency of the cut set or group of cut sets (e.g., 8.4-8 = 8.4 x 10^{-8}/yr). The first event name in each cut set always begins with the letter "T" and designates the initiator. The remaining names refer to component or system faults or failures related to operator actions. The numbers in parentheses represent the

probabilities of failure for the remaining events. The symbols "*" and "+" denote the Boolean operators "AND" and "OR," respectively.

Following the description of each cut set is an indication of any potential dependence between events in the cut set and the availability of the containment safety features (CSF, i.e., the annulus cooling, annulus filtration, containment cleanup, and containment isolation systems). This is the input to the integration step described in Section 13.

10.1.1 Bin 1: Sequence TK

Sequence TK indicates a demand for the plant protection system (PPS) to operate, with the main primary heat transport system (PHTS) pumps tripping but the reactor failing to be shut down. This is referred to as an unprotected loss of flow event.

Because there are two redundant and diverse shutdown systems, the likelihood of failure to scram is very low. Since the trip signal for the PHTS pumps comes from the PPS, the probability of successful pump trip combined with failure to scram is even lower. Thus, this sequence was calculated to have a frequency below 10^{-8}/yr, and is an insignificant contributor to core-damage frequency.

10.1.2 Bin 2: Sequence TKP

Sequence TKP is an interruption of the normal heat removal system (e.g., loss of main feedwater, turbine trip, loss of condenser vacuum) during power operation, followed by failure of the PPS function. In this case, the PHTS pumps continue to operate, but the mismatch between the heat

generated and that removed causes a fairly rapid heatup of the sodium. One type of cut set results which occurred for this sequence:

2.5-7 TX(15.4) * PPSFAILS(8.0-8) * RTRIPH(0.2)

Event TX can be any initiator except loss of offsite power (Event T4), since that would directly result in the interruption of power to the main PHTS motors; a reactivity insertion (Event T5), which would lead to an unprotected transient overpower, Sequence TKPQ (see Section 10.1.3); or any of the PHTS sodium leaks, since, with failure of the main PHTS pump motors to trip, an unprotected loss of sodium would result (Sequence TKPI, described in Section 10.1.4). The PPS fails to function (PPSFAILS), and manual scram fails (RTRIPH), either because the operators fail to actuate the shutdown systems in time, or because the failure mode of the shutdown systems is one that cannot be readily recovered from the control room.

CSF state: no direct effects.

The total frequency for sequence TKP (and, consequently, for Bin 2) is estimated to be 2.5×10^{-7}/yr.

10.1.3 Bin 3: Sequence TKPQ

Sequence TKPQ is initiated by an inadvertent positive reactivity insertion, event T5, followed by a failure of the PPS to scram the reactor and to trip the main PHTS pump motors. This constitutes an unprotected transient overpower event. Because the frequency of event T5 is low (0.019/yr) and the unavailability of the shutdown systems is very small, the frequency for sequence TKPQ is calculated to be less than 10^{-8}/yr, and constitutes an insignificant contributor to the frequency of core damage.

10.1.4 Bin 4: Sequence TKPI

Sequence TKPI is a sodium leak in the PHTS followed by failure of the PPS such that the reactor fails to be shut down and the main PHTS pump

motors fail to trip (i.e., an unprotected loss of sodium). Like sequence TKPQ, the low frequencies of the leak initiators (totaling about 0.014/yr) combined with the low probability of failure of the shutdown systems make sequence TKPI an insignificant contributor to core-damage frequency, at less than 10^{-8}/yr.

10.1.5 Bin 5: Sequence TSD

Sequence TSD represents a failure to provide for decay heat removal upon shutdown; decay heat removal can be accomplished either via the heat sinks provided by the steam generating system (event S), or by using the direct heat removal service (DHRS, event D).

Because of the very long time it takes for the primary sodium to heat to saturation, a significant aspect of all cut sets for sequence TSD is the opportunity to recover core cooling if it is unavailable or is interrupted. In the case of a loss of cooling via the steam generating systems, the time for recovery is limited by the time it takes to dry out the steam generator modules, which is about 10 minutes from the start of the transient in the case of a total loss of feedwater supply. However, up to about 10-12 hr is available for recovery of core cooling by successful initiation of the DHRS.

Because of the large number of cut sets comprising Sequence TSD, they have been divided into several groups. Six groups dominate the frequency of sequence TSD, and are described in detail. The total frequency for sequence TSD is estimated to be 3.4×10^{-6}/yr.

TSD Group I

The cut sets for sequence TSD in Group I are initiated by a leak in a PHTS loop that makes that loop unavailable for heat transport and that disables the DHRS (Event T11A, B, or C for PHTS loops A, B, or C). The unaffected loops are unavailable due to a variety of faults. The total frequency for Group I was estimated to be 2.0 x 10^{-7}/yr.

2.4-8 T11A(6.0-5) * Q53RV106BC(2.0-2) * Q53RV106CC(2.0-2)

2.4-8 T11B(6.0-5) * Q53RV106AC(2.0-2) * Q53RV106CC(2.0-2)

2.4-8 T11C(6.0-5) * Q53RV106AC(2.0-2) * Q53RV106BC(2.0-2)

A leak in a PHTS loop makes that loop and DHRS unavailable. The superheater safety relief valve (SRV) at the lowest set-point in each of the remaining loops opens and fails to reclose (Q53RV106AC for loop A, Q53RV106BC for loop B, and Q53RV106CC for loop C). The resultant depressurization causes the steam drums to be isolated from main and auxiliary feedwater.

CSF state: no direct effects.

1.8-8 T11A(6.0-5) * S52CDEFCOM(3.0-4)#

1.8-8 T11B(6.0-5) * S52ABEFCOM(3.0-4)#

1.8-8 T11C(6.0-5) * S52ABCDCOM(3.0-4)#

Like the cut sets above, these three cut sets are initiated by a PHTS leak that makes that loop and DHRS unavailable. In this case, the remaining two loops are unavailable due to common-cause failure of a protected air-cooled condenser (PACC) vent valve in each loop to reclose (S52CDEFCOM for loops B and C, S52ABEFCOM for loops A and C, and S52ABCDCOM for loops A and B).

CSF state: no direct effects.

#These cut sets were identified by searching for occurrence of coupled failures of identical components, and did not appear explicitly in the cut-set printout.

```
1.3-8  T11A(6.0-5) * Q53RV106BC(2.0-2) * [QSPWDC(1.0-3) +
                                          QSWRPRSIC(1.0-3) +
                                          T53SDSRCCL(1.0-3) +
                                          S52EV133EC(1.0-3) +
                                          S52EV133FC(1.0-3) +
                                          S52EV116CC(1.0-3) +
                                          S52EV117CC(1.0-3) +
                                          T53ESRCCL(1.0-3) +
                                          T53SHSRCCL(1.0-3) +
                                          P51BRVCT(1.0-3) ]
```

A leak in PHTS loop A makes that loop and DHRS unavailable, and a stuck-open superheater SRV makes loop B unavailable. Loop C fails due to one of the following: a spurious signal actuates the water dump (QSPWDC) or SWRPRS (QSWRPRSIC); a latent human error results in an open steam drum SRV (T53SDSRCCL); one of the protected air-cooled condenser (PACC) vent valves sticks open (S52EV133EC or S52EV133FC); a steam generator auxiliary heat removal system (SGAHRS) vent valve opens and fails to reclose (S52EV116CC or S52EV117CC); a latent human error causes a SGAHRS vent valve (TS52VENTCL), an evaporator SRV (T53ESRCCL), or a superheater SRV (T53SHSRCCL) to stick open; or the intermediate heat transport system (IHTS) sodium dump valve transfers open (P51BRVCT).

CSF state: no direct effects.

```
1.3-8  T11A(6.0-5) * Q53RV106CC(2.0-2) * [QSPWDB(1.0-3) +
                                          QSWRPRSIB(1.0-3) +
                                          T53SDSRBCL(1.0-3) +
                                          S52EV133CC(1.0-3) +
                                          S52EV133DC(1.0-3) +
                                          S52EV116BC(1.0-3) +
                                          S52EV117BC(1.0-3) +
                                          T53ESRBCL(1.0-3) +
                                          T53SHSRBCL(1.0-3) +
                                          P51BRVBT(1.0-3) ]
```

A leak in PHTS loop A makes that loop and DHRS unavailable, and a stuck-open superheater SRV makes loop C unavailable. Loop B fails as a result of faults analogous to those described above for loop C.

CSF state: no direct effects.

```
1.3-8  T11B(6.0-5) * Q53RV106AC(2.0-2) * [QSPWDC(1.0-3) +
                                          QSWRPRSIC(1.0-3) +
                                          T53SDSRCCL(1.0-3) +
                                          S52EV129EC(1.0-3) +
                                          S52EV129FC(1.0-3) +
                                          S52EV116CC(1.0-3) +
                                          S52EV117CC(1.0-3) +
                                          TS52VENTCL(1.0-3) +
                                          T53ESRCCL(1.0-3) +
                                          T53SHSRCCL(1.0-3) +
                                          P51BRVCT(1.0-3)]
```

A leak in PHTS loop B makes that loop and DHRS unavailable, and a stuck-open superheater SRV makes loop A unavailable. Loop C fails due to the causes described for earlier cut sets.

CSF state: no direct effects.

```
1.3-8  T11B(6.0-5) * Q53RV106CC(2.0-2) * [QSPWDA(1.0-3) +
                                          QSWRPRSIA(1.0-3) +
                                          T53SDSRACL(1.0-3) +
                                          S52EV129AC(1.0-3) +
                                          S52EV129BC(1.0-3) +
                                          S52EV116AC(1.0-3) +
                                          S52EV117AC(1.0-3) +
                                          TS52VENTAL(1.0-3) +
                                          T53ESRACL(1.0-3) +
                                          T53SHSRACL(1.0-3) +
                                          P51BRVAT(1.0-3)]
```

A leak in PHTS loop B makes that loop and DHRS unavailable, and a stuck-open SRV fails loop C. Loop A fails due to causes described above for loop C.

CSF state: no direct effects.

```
1.3-8  T11C(6.0-5) * Q53RV106BC(2.0-2) * [QSPWDA(1.0-3) +
                                          QSWRPRSIA(1.0-3) +
                                          T53SDSRACL(1.0-3) +
                                          S52EV129AC(1.0-3) +
                                          S52EV129BC(1.0-3) +
                                          S52EV116AC(1.0-3) +
                                          S52EV117AC(1.0-3) +
```

TS52VENTAL(1.0-3) +
T53ESRACL(1.0-3) +
T53SHSRACL(1.0-3) +
P51BRVAT(1.0-3)]

A leak in PHTS loop C makes that loop and DHRS unavailable, and a stuck-open superheater SRV fails loop B. Loop A is unavailable due to other causes.

CSF state: no direct effects.

1.3-8 T11C(6.0-5) * Q53RV106AC(2.0-2) * [QSPWDB(1.0-3) +
QSWRPRSIB(1.0-3) +
T53SDSRBCL(1.0-3) +
S52EV129CC(1.0-3) +
S52EV129DC(1.0-3) +
S52EV116BC(1.0-3) +
S52EV117BC(1.0-3) +
TS52VENTBL(1.0-3) +
T53ESRBCL(1.0-3) +
T53SHSRBCL(1.0-3) +
P51BRVBT(1.0-3)]

A leak in PHTS loop C makes that loop and DHRS unavailable, and a stuck-open superheater SRV fails loop A. Loop B fails due to miscellaneous faults.

CSF state: no direct effects.

TSD Group II

Cut sets in Group II are initiated by any event that does not directly result in the unavailability of a heat transport loop, followed by isolation of all three loops due to stuck-open superheater safety relief valves or PACC vent valves. With DHRS unavailable due to a variety of causes, core damage results. The total frequency for Group II is approximately 1.7×10^{-6}/yr.

1.5-6 TXHTF(15.1) * [S52ACECOM(6.1-5) +
S52ACFCOM(6.1-5) +
S52ADECOM(6.1-5) +

S52ADFCOM(6.1-5) +
S52BCECOM(6.1-5) +
S52BCFCOM(6.1-5) +
S52BDECOM(6.1-5) +
S52BDFCOM(6.1-5)] * [R81DHRSH(1.0-4) +
RMISOLATE(2.0-3) * RDHRSH1(5.0-2)]

These cut sets are initiated by any event that does not directly lead to unavailability of a heat transport loop (TXHTF), followed by a common-cause failure of at least one PACC vent valve in each of the three loops to close, leading to isolation of the steam drums (events S52ACECOM, etc.). DHRS fails either because the operators fail to initiate it properly (R81DHRSH), or because the bypass valves for the overflow heat exchanger fail to close (RMISOLATE). In either case the operators fail to close the bypass valves locally (RDHRSH1).

CSF state: no direct effect.

1.9-7 TIOR(3.8-4) * [S52ACECOM(6.1-5) + S52ACFCOM(6.1-5) +
S52ADECOM(6.1-5) + S52ADFCOM(6.1-5) +
S52BCECOM(6.1-5) + S52BCFCOM(6.1-5) +
S52BDECOM(6.1-5) + S52BDFCOM(6.1-5)]

A leak in the reactor vessel or in the PHTS piping within the reactor-vessel guard vessel precludes the use of DHRS. A stuck-open vent valve in each loop, as in the cut sets above, leads to isolation of the steam drums and a total loss of core cooling.

CSF state: no direct effect.

2.4-8 TXHTF(15.1) * Q53RV106AC(2.0-2) * Q53RV106BC(2.0-2) *
Q53RV106CC(2.0-2) * [R81DHRSH(1.0-4) +
RMISOLATE(2.0-3) * RDHRSH1(5.0-2)]

These cut sets are equivalent to those discussed above for TXHTF, with the steam drums isolated due to stuck-open superheater relief valves.#

CSF state: no direct effect.

#A literature search was made to locate data indicative of common-cause failures of relief valves to reclose, but no appropriate data were found.

TSD Group III

Cut sets in this group are initiated by a loss of offsite power, which results in the loss of main feedwater. The train A and B diesel generators fail such that both motor-driven auxiliary feedwater (AFW) pumps and the DHRS makeup pumps are unavailable. Failure of the turbine-driven auxiliary feedwater pump then leads to a loss of core cooling. The cut sets in this group sum to a total frequency of approximately 5.6×10^{-7}/yr.

4.7-7 T4(9.2-2) * EMEDGCOM(8.3-3) * [SMTDPCH53(5.1-3) + SMTDPRC53(1.1-3)] * RECOVERT4(0.1)

A loss of offsite power (T4) and common-cause failure of the A and B diesel generators (EMEDGCOM) results in loss of main feedwater, the unavailability of both motor-driven AFW pumps and both makeup pumps. The turbine-driven AFW pump is unavailable, either because of pump faults (SMTDPCH53), or because of failure of the pump recirculation line (SMTDPRC53). Failure to recover power in time to initiate DHRS successfully (RECOVERT4) leads to core damage.

CSF state: trains A and B of electric power unavailable.

4.3-8 T4(9.2-2) * EMEDGAL(2.5-2) * EMEDGBS(3.0-2) * [SMTDPCH53(5.1-3) + SMTDPRC53(1.1-3)] * RECOVERT4(0.1)

4.3-8 T4(9.2-2) * EMEDGAS(3.0-2) * EMEDGBL(2.5-2) * [SMTDPCH53(5.1-3) + SMTDPRC53(1.1-3)] * RECOVERT4(0.1)

Loss of offsite power (T4) occurs with one diesel failing to start (EMEDGAS, EMEDGBS) and the other unavailable due to latent human errors (EMEDGBL, EMEDGAL) resulting in loss of main feedwater, both motor-driven AFW pumps, and both makeup pumps. The turbine-driven AFW pump is unavailable due to pump faults (SMTDPCH53), or because of failure of the pump recirculation line (SMTDPRC53). Recovery (RECOVERT4) is done by restoring power to DHRS prior to core damage.

CSF state: trains A and B of electric power unavailable.

A number of the lower frequency cut sets in Group III sum to about 4×10^{-9}/yr. Included in this sum are cut sets involving failure of one of the diesels, with failure of the other due to various faults in the plant service water system, or failure of one diesel generator and failure of the motor-driven AFW pump in the other train.

TSD Group IV

Cut sets in this group result in loss of DHRS because of support system failures: e.g. failure of diesel generators or loss of off-site power, or due to loss of cover gas to PHTS pump seals failing all forced circulation. The steam generator system fails because of SGAHRS logic failures which can be recovered by the operator. The total frequency for cut sets of this group is approximately 6.4×10^{-7}/yr.

5.1-7 T4(9.2-2) * [EMEDGAS(3.0-2) * EMEDGBL(2.5-2) + EMEDGAL(2.5-2) * EMEDGBS(3.0-2)] * S903BE(1.0-3) * SGAHRSH1(5.0-2)

Loss of off-site power (T4) occurs, and two emergency diesels A and B are lost due to miscellaneous faults (EMEDGAS and EMEDGBL, EMEDGAL and EMEDGBS). Automatic SGAHRS actuation fails (S903BE) requiring manual actuation by operator (SGAHRSH1).

CSF state: trains A and B of electric power unavailable.

5.0-8 T16(1.0-3) * S903BE(1.0-3) * SGAHRSH1(5.0-2)

SGAHRS fails to actuate (S903BE) on loss of service water flow (T16) requiring operator action to actuate AFW system (SGAHRSH1).

CSF state: no direct effect.

3.8-8 T4(9.2-2) * EMEDGCOM(8.3-3) * S903BE(1.0-3) * SGAHRSH1(5.0-2)

Diesel generators A and B fail because of common causes (EMEDGCOM) when loss of offsite power (T4) occurs, thereby failing DHRS. SGAHRS actuation logic fails (S903BE) failing all decay heat removal. Recovery occurs by manual actuation of SGAHRS by push button operation (SGAHRSH1).

CSF state: trains A and B of electric power unavailable.

2.5-8 T18(0.25) * [G82ACCH(1.0-3) + G82ACCL(1.0-3)] * S903BE(1.0-3) * SGAHRSH1(5.0-2)

Loss of instrument air (T18) fails cover gas supply to PHTS pump seals. Failure to align the standby argon accumulators to the pump seals results in failure of all forced circulation rendering DHRS useless. Secondary side heat removal fails because of failure to actuate SGAHRS (S903BE) which can be recovered by a pushbutton operation by the operator (SGAHRSH1).

CSF state: no direct effects.

Several low frequency cut sets of this group sum to about 1.2×10^{-8}/yr and include combination of support system failures for DHRS, with SGAHRS actuation failures (S903BE).

TSD Group V

The cut sets in Group V are initiated by an event that causes one heat transport loop to be unavailable. The remaining loops are lost due to isolation of their steam drums, and DHRS fails. The cut sets sum to a frequency of approximately 6.3×10^{-8}/yr.

2.1-8 [T7A(4.8-2) + T8A (4.5-2) + T9A(3.0-2) + T14A(2.6-2)] * [Q53RV106BC(2.0-2) * Q53RV106CC(2.0-2) + S52CDEFCOM(3.0-4)] * [R81DHRSH(1.0-4) + RMISOLATE(2.0-3) * RDHRSH1(5.0-2)]

Loop A is unavailable for heat transport due to a steam generator tube rupture (T7A), inadvertent actuation of the loop A SWRPRS (T8A), a leak in IHTS loop A (T9A), or a steam line break inside the loop A superheater outlet isolation valve

(T14A). The remaining two loops are unavailable due to stuck-open superheater SRVs or PACC vent valves, and DHRS is unavailable due to inappropriate actuation.

CSF state: no direct effects.

2.1-8 [T7B(4.8-2) + T8B(4.5-2) + T9B(3.0-2) + T14B(2.6-2)] * [Q53RV106AC(2.0-2) * Q53RV106CC(2.0-2) + S52ABEFCOM(3.0-4] * [R81DHRSH(1.0-4) + RMISOLATE(2.0-2) * RDHRSH1(5.0-2)]

These cut sets are analogous to those above, with the initiator affecting loop B, and the valve failures in loops A and C.

CSF state: no direct effects.

2.1-8 [T7C(4.8-2) + T8C(4.5-2) + T9C(3.0-2) + T14C(2.6-2)] * [Q53RV106AC(2.0-2) * Q53RV106BC(2.0-2) + S52ABCDCOM(3.0-4] * [R81DHRSH(1.0-4) + RMISOLATE(2.0-3) * RDHRSH1(5.0-2)]

These cut sets are analogous to those above, with the initiator affecting loop C, and the valve failures in loops A and B.

CSF state: no direct effects.

TSD Group VI

The cut sets in Group VI are sequences in which main feedwater (MFW) is lost, and SGAHRS fails due to failure to actuate, or PWST failures. Core damage occurs due to failure of the DHRS to be properly initiated. The total frequency for Group VI is about 1.9×10^{-7}/yr.

8.7-8 [T3(4.0) + T4(9.2-2) + T18(0.25)] * S52TKPWST (1.0-4) * [R81DHRSH1(1.0-4) + RMISOLATE(2.0-3) * RDHRSH1(5.0-2)]

Main feedwater is lost due to one of three transients (T3, T4, or T18), auxiliary feedwater fails because of the unavailability of PWST which results in core damage when DHRS is not initiated properly.

CSF state: no direct effects.

5.0-8 T18(0.25) * [G82ACCH(1.0-3) + G82ACCL(1.0-3)] * S52TKPWST(1.0-4)

Loss of instrument air (T18) fails the cover gas supply to the PHTS pump seals requiring use of alternate argon accumulators which fail either because of improper operator action (G82ACCH), or because of latent human errors (G82ACCL). Failure of forced circulation renders DHRS unavailable, and the PWST is unavailable in the absence of main feedwater flow, failing the steam generators.

CSF: no direct effects.

4.3-8 [T3(4.0) + T4(9.2-2) + T18(0.25)] * S903BE(1.0-3) * SGAHRSH1(5.0-2) * [R81DHRSH(1.0-4) + RMISOLATE(2.0-3) * RDHRSH1(5.0-2)]

MFW flow is interrupted due to one of three initiators: loss of MFW (T3), loss of offsite power (T4), or loss of instrument air (T18). SGAHRS fails to actuate automatically (S903BE), and fails to be initiated manually prior to steam-drum dryout (SGAHRSH1). DHRS is unavailable due to the causes discussed previously.

CSF state: no direct effects.

Additional cut sets in Group VI sum to a frequency of about 7 x 10^{-9}/yr. These cut sets include failures of DHRS due to failure of individual electrical buses; failure of the crossover valves in ex-vessel sodium processing system (EVSPS) circuit 1 to open coupled with failure of the dampers for the air-blast heat exchanger (ABHX) B to open; or failure of support services for the makeup pumps, including failure of the chilled water or HVAC.

Additional TSD Cut Sets

Additional lower frequency cut sets for sequence TSD sum to a total of about 1 x 10^{-8}/yr. This includes a series of cut sets initiated by an event that causes interruption of MFW flow, with AFW flow to two of the three loops failing due to failure of a SGAHRS amplifier, and the third

loop isolated due to a stuck-open superheater SRV, followed by failure of DHRS. Similar cut sets arise when failure of the protected-water storage tank causes AFW to be unavailable following a loss of MFW flow. A number of cut sets were attributed to a loss of offsite power, followed by failure of the A or B diesel generator (either directly or due to support-system faults). With failures in the steam generating systems for two loops, core cooling is lost if the operators fail to trip the pump motors in these two loops, since they would backseat the cold leg check valve in the third loop (which has steam generator cooling available). DHRS is lost because of improper actuation. These cut sets have a small total frequency due to the various opportunities for recovery by the operators.

The final group involves failures that lead to unavailability of one heat transport loop with failure of the main PHTS pump motor in that loop to trip, such that the other two loops have no flow. Again, recovery can be accomplished by tripping the main motor, and a long period is available for taking this action.

The cut sets for sequence TSD sum to a total of approximately 3.4×10^{-6}/yr.

10.1.6 Bin 6: Sequence TCD

Sequence TCD represents an event involving successful short-term core cooling, but with failure of long-term cooling to be established and maintained either via the MFW and turbine-bypass systems, or using the PACCs.

When considered in the light of potential recovery actions, the cut sets for Sequence TCD all had very low frequencies. This is because the successful removal of sensible and decay heat in the short-term significantly lengthens the time it would take the primary sodium to heat to saturation if cooling were lost subsequently. (More than 20 hours are available prior to sodium reaching saturation temperature.) In addition, after two hours of successful steam-generator cooling, the PACCs can remove decay heat in essentially a closed-loop cycle, with some makeup to the steam drums on a periodic basis. The likelihood that no feedwater is available, coupled with long-term failures affecting DHRS, make this sequence very unlikely.

10.1.7 Bin 7: Sequence TI

Sequence TI represents a leak in the PHTS sufficient to interrupt all core cooling, either because a main PHTS pump motor continues to operate, supplying sodium to the leak location, or because the leak is within a guard vessel, but the guard vessel also leaks. A leak which can interrupt all core cooling is defined to be a large-size leak in Section 5.5.2.3. Such leaks have very low frequencies (8.0×10^{-8}/yr, overall) as shown in Tables 5.5-5, 5.6-3, and 5.6-4. The low frequency of a leakage coupled with the small probability of guard vessel failures ($<10^{-5}$) such that primary sodium is lost to the reactor cavity make sequences involving loss of sodium insignificant. The cut sets for Sequence TI therefore have no contribution to the overall plant damage frequency.

Also, cut sets which involve the failure of a main PHTS pump motor to trip following a leak are insignificant contributors to core-damage frequency due to the reliability of the pump trip and the ability of the operators to take manual action to terminate pump operation.

10.1.8 Summary of Data in Dominant Cut Sets

In order to facilitate understanding of the events in the cut-set listing, those events listed in Sections 10.1.1 through 10.1.7 are tabulated in the following tables, along with a reference to the report section in which the probability of the event is developed. Table 10.1-2 summarizes the internal initiators, Table 10.1-3 lists the basic event and module unavailabilities appearing in the cut sets, and Table 10.1-4 summarizes the recovery events.

10.2 RESULTS FOR EXTERNAL EVENTS

Section 8 describes the methods, input data and results of the analysis of external-event initiators. The external events treated in this study are earthquakes, tornadoes, and aircraft impact. The results are summarized here to provide in Section 10 a single source for all core-damage frequency results.

Table 10.2-1 summarizes the results for the external events. The tornado and aircraft impact sequences were estimated to have a negligible frequency for core-damage sequences (less than 10^{-8}/yr). The remaining sequences are those caused by an earthquake initiating event.

Table 10.1-2

INITIATOR DESCRIPTION REFERENCE LIST

Event[a]	Frequency (yr^{-1})	Description
T1	8.9	General transient
T2	0.96	Loss of condenser
T3	4.0	Loss of main feedwater
T4	9.2-2[b]	Loss of offsite power
T5	1.9-2	Reactivity insertion
T6ALL	5.7-2	Total loss of PHTS flow
T6A(B)(C)	0.25	Loss of flow in PHTS loop A (B)(C)
T7A(B)(C)	4.8-2	Steam generator tube rupture in loop A (B)(C)
T8A(B)(C)	4.5-2	Inadvertent actuation of SWRPRS in loop A (B)(C)
T9A(B)(C)	3.0-2	Leak in IHTS loop A (B)(C)
T10A(B)(C)[c]	9.5-5	Leak in PHTS loop A (B)(C) within guard vessel
T10R[c]	3.8-4	Leak in reactor vessel or in PHTS piping within its guard vessel
T11A(B)(C)[c]	6.0-5	Leak in PHTS loop A (B)(C) that disables the loop for heat transport and makes DHRS unavailable
T12A(B)(C)[c]	3.8-5	Leak in PHTS loop A (B)(C) that disables the. loop but allows DHRS operation
T13	7.1-3	Steam-line break outside the superheater outlet isolation valves (SOIVs)
T14A(B)(C)	2.6-2	Steam-line break inside the SOIV for loop A (B)(C)
T15	2.0-3	Loss of chilled water
T16	1.0-3	Loss of service water
T17	8.8-5	Loss of essential service ac buses
T18	0.25	Loss of instrument air
T19	--	Loss of instrumentation and control power
T20	--	Loss of dc power
T21A(B)(C)[c]	8.6-5	Leak in PHTS loop A (B)(C) that disables DHRS but allows heat transport in that loop
T22A(B)(C)[c]	9.5-5	Leak in PHTS loop A (B)(C) that allows DHRS and heat transport operation
TX	15.4	Any initiator except T4, T5, T10A (B)(C), T10R, T11A (B)(C), T12A (B)(C), T21A (B)(C), or T22A (B)(C)
TXHTF	15.1	Any initiator that does not lead directly to unavailability of a heat transport loop, T7A (B)(C), T8A (B)(C), T9A (B)(C), T10A (B)(C), T11A (B)(C), T12A (B)(C), or T14 A (B)(C)

[a]The frequencies for these events are derived in Section 5.
[b]9.2-2 = 9.2×10^{-2}.
[c]Small leak frequency. For large leak frequency see Table 5.6-3.

Table 10.1-3

SUMMARY DESCRIPTION OF BASIC EVENTS AND MODULES IN CUT-SET LISTING

Event[a]	Probability	Reference[a]	Description
EMEDGAS	3.0-2[b]	A12	Failure to start of diesel generator A
EMEDGAL	2.5-2	A12	Latent human error on diesel generator A
EMEDGBS	3.0-2	A12	Failure to start of diesel generator B
EMEDGBL	2.5-2	A12	Latent human error on diesel generator B
EMEDGCOM	8.3-3[b]	5.7	Common-cause failure of diesel generators A and B
G82ACCH	1.0-3	A24	Operator fails to align standby argon accumulators to provide cover gas to PHTS pump seals
G82ACCL	1.0-3	A24	Latent human error on standby argon accumulators
P51BRVAT	1.0-3	A14	Loop A IHTS sodium dump valve transfers open
P51BRVBT	1.0-3	A14	Loop B IHTS sodium dump valve transfers open
P51BRVCT	1.0-3	A14	Loop C IHTS sodium dump valve transfers open
PPSFAILS	8.0-8	A18	Failure of both the primary and secondary shutdown systems to function
Q53RV106AC	2.0-2	A4	Failure of the loop A superheater safety relief valve to close
Q53RV106BC	2.0-2	A4	Failure of the loop B superheater safety relief valve to close
Q53RV106CC	2.0-2	A4	Failure of the loop C superheater safety relief valve to close
QSPWDA	1.0-3	A4	Spurious actuation of the water dump for loop A
QSPWDB	1.0-3	A4	Spurious actuation of the water dump for loop B
QSPWDC	1.0-3	A4	Spurious actuation of the water dump for loop C

Table 10.1-3 (Continued)

Event	Probability	Reference	Description
QSWRPRSIA	1.0-3	A4	Inadvertent actuation of SWRPRS for loop A
QSWRPRSIB	1.0-3	A4	Inadvertent actuation of SWRPRS for loop B
QSWRPRSIC	1.0-3	A4	Inadvertent actuation of SWRPRS for loop C
RMISOLATE	2.0-3	A3	Failure of bypass valves for overflow heat exchanger to close upon initiation of the DHRS
S52ABCDCOM	3.0-4	5.8	Common-cause failure of a PACC vent valve in loop A and one in loop B to close
S52ABEFCOM	3.0-4	5.8	Common-cause failure of a PACC vent valve in loop A and one in loop C to close
S52CDEFCOM	3.0-4	5.8	Common-cause failure of a PACC vent valve in loop B and one in loop C to close
S52ACECOM	6.1-5	5.8	Common-cause failure to close of a PACC vent valve in each of the three loops
S52ACFCOM	6.1-5	5.8	Common-cause failure to close of a PACC vent valve in each of the three loops
S52ADECOM	6.1-5	5.8	Common-cause failure to close of a PACC vent valve in each of the three loops
S52ADFCOM	6.1-5	5.8	Common-cause failure to close of a PACC vent valve in each of the three loops
S52BCECOM	6.1-5	5.8	Common-cause failure to close of a PACC vent valve in each of the three loops
S52BCFCOM	6.1-5	5.8	Common-cause failure to close of a PACC vent valve in each of the three loops
S52BDECOM	6.1-5	5.8	Common-cause failure to close of a PACC vent valve in each of the three loops

Table 10.1-3 (Continued)

Event	Probability	Reference	Description
S52BDFCOM	6.1-5	5.8	Common-cause failure to close of a PACC vent valve in each of the three loops
S52EV116AC	1.0-3	3.4	SGAHRS vent valve in loop A fails to reclose
S52EV117AC	1.0-3	3.4	SGAHRS vent valve in loop A fails to reclose
S52EV116BC	1.0-3	3.4	SGAHRS vent valve in loop B fails to reclose
S52EV117BC	1.0-3	3.4	SGAHRS vent valve in loop B fails to reclose
S52EV116CC	1.0-3	3.4	SGAHRS vent valve in loop C fails to reclose
S52EV117CC	1.0-3	3.4	SGAHRS vent valve in loop C fails to reclose
S52EV129AC	1.0-3	3.4	PACC vent valve in loop A fails open
S52EV129BC	1.0-3	3.4	PACC vent valve in loop A fails open
S52EV129CC	1.0-3	3.4	PACC vent valve in loop B fails open
S52EV129DC	1.0-3	3.4	PACC vent valve in loop B fails open
S52EV129EC	1.0-3	3.4	PACC vent valve in loop C fails open
S52EV129FC	1.0-3	3.4	PACC vent valve in loop C fails open
S52TKPWST	1.0-4	A2	PWST vent valve plugs
S903BE	1.0-3	A25	Failure of the SGAHRS system to actuate automatically
SMTDPCH53	5.1-3	A2	Failure of the turbine-driven AFW pump due to faults local to the pump
SMTDPRC53	1.1-3	A2	Failure of the turbine-driven AFW pump due to failure of its recirculation line

Table 10.1-3 (Continued)

Event	Probability	Reference	Description
T53ESRACL	1.0-3	3.4	Loop A evaporator SRV sticks open due to latent human error
T53ESRBCL	1.0-3	3.4	Loop B evaporator SRV sticks open due to latent human error
T53ESRCCL	1.0-3	3.4	Loop C evaporator SRV sticks open due to latent human error
T53SHSRACL	1.0-3	3.4	Loop A superheater SRV sticks open due to latent human error
T53SHSRBCL	1.0-3	3.4	Loop B superheater SRV sticks open due to latent human error
T53SHSRCCL	1.0-3	3.4	Loop C superheater SRV sticks open due to latent human error
T53SDSRACL	1.0-3	3.4	Steam drum SRV for loop A open due to latent human error
T53SDSRBCL	1.0-3	3.4	Steam drum SRV for loop B open due to latent human error
T53SDSRCCL	1.0-3	3.4	Steam drum SRV for loop C open due to latent human error
TS52VENTAL	1.0-3	3.4	SGAHRS vent valve on loop A fails open due to latent human error
TS52VENTBL	1.0-3	3.4	SGAHRS vent valve on loop B fails open due to latent human error
TS52VENTCL	1.0-3	3.4	SGAHRS vent valve on loop C fails open due to latent human error

[a]Reference is for report section number in which event is developed and probability derived.

[b]3.0-2 = 3.0×10^{-2}.

Table 10.1-4

DESCRIPTION OF RECOVERY EVENTS

Event[a]	Probability	Description
RDHRSH1	5.0-2[b]	Failure to recover DHRS by locally isolating overflow heat exchanger bypass
RECOVERT4	0.1	Failure to recover offsite power within about 10 hours
RTRIPH	0.2	Failure to back up scram system manually within about 5 minutes
R81DHRSH	1.0-4	Failure to recover core cooling by properly initiating DHRS within 10-12 hours
SGAHRSH1	5.0-2	Failure to actuate SGAHRS manually when automatic actuation fails

[a]All recovery events are described in Section 6.
[b]5.0-2 = 5.0×10^{-2}.

Table 10.2-1

SUMMARY OF CORE-DAMAGE FREQUENCIES FOR EXTERNAL EVENTS

Bin	Sequence	Description	Frequency
1	EK	Seismic-Induced Unprotected Loss of Flow	1.3-6/yr
2	EKP	Seismic-Induced Unprotected Loss of Heat Sink	ε
3	$EKI\overline{P}$	Seismic-Induced Transient Overpower	ε
4	EKI	Seismic-Induced Unprotected Loss of Sodium	1.5-6/yr
5	ESD	Seismic-Induced Loss of Heat Sink (early)	1.5-5/yr
6	ECD	Seismic-Induced Loss of Heat Sink (late)	ε
7	EI	Seismic-Induced Loss of Sodium	1.4-5/yr
All	All	Tornado Initiators	ε
All	All	Aircraft Impact	ε
		TOTAL	3.2-5/yr

The event tree in Section 8.1 (Figure 8.1-4) was evaluated using the abbreviated fault tree logic also presented in Section 8. The earthquake sequences were solved and grouped by sequence and bin as indicated in Table 10.2-1. Three sequences did not have significant frequencies: EKP, EKPQ, and ECD. The first two are insignificant because event P requires that offsite power remain available, but offsite power is one of the most fragile functions during an earthquake. The last sequence ECD is insignificant for two reasons: (1) the sequence requires a successful short-term heat removal and unsuccessful long-term heat removal--unlikely because the seismically-induced failures of equipment are generally immediate and therefore result in early core-damage sequences, and (2) the long-term sequences all have very long times (greater than 10 hours) for potential recovery actions.

The sequences that are significant sum to a total of 3.2×10^{-5}/yr. The significant cut sets for these sequences are listed in Tables 8.1-18 through 8.1-21 and are not repeated here. The results of the seismic sequences indicate that the following features are important:

- The PHTS piping and components leaks into guard vessels coupled with failures of the guard vessels are the dominant contributors sodium leak sequences (EKI and EI).
- The SWRPRS rupture disk failures are significant contributors to the loss of heat sink sequences.
- Structural failure of the steam generator building is assumed to lead to a core-damage sequence.
- Loss of offsite power is an important seismic event because of its low fragility and is also significant in combination with other electrical switchgear failures.

- The seismically-induced failure of the shutdown systems is more significant than for non-seismic sequences because both the primary and secondary systems could be subject to a common-cause effect--mechanical binding.

10.3 RESULTS FOR COMMON-CAUSE INITIATORS

Core-damage sequences resulting from fires, liquid-metal fires and interactions, turbine missiles, and flooding from internal plant sources are summarized in this section. These events--termed common-cause initiators in this study--are more fully addressed in Section 9.

Table 10.3-1 summarizes the results for common-cause initiators. As the table illustrates, only one sequence is significant--the non-sodium fire initiating event followed by loss of the heat sink in the short term. The significant sequence is the core-damage sequence assumed to result from a large control room fire. Using available data for initiating events and postulating the scenario for propagation of the fire, a 2.0×10^{-7}/yr frequency has been estimated.

No other significant sequences were found. The unprotected sequences are insignificant because the common-cause initiating-event frequencies are low enough (on the order of 10^{-2} or less) that, when coupled with the failure of both scram systems, no dominant sequences result. The physical separation of important equipment that is an integral part of the CRBRP design limits the impact of the location-dependent common-cause events considered.

Table 10.3-1

SUMMARY OF CORE-DAMAGE FREQUENCIES FOR COMMON-CAUSE INITIATORS

Bin	Sequence	Description	Frequency
1-4	All K sequences	Unprotected Accidents for Common-Cause Initiators	ε
5	Fire*SD	Fire-Induced Loss of Heat Sink (early) - FLHSE	2.0-7/yr
5	All other*SD	Other Causes for Loss of Heat Sink (early)	ε
6	Initiator*CD	Late Loss of Heat Sink for all Common-Cause Initiators	ε
7	Initiator*I	Loss of Sodium for all Common-Cause Initiators	ε
		TOTAL	2.0-7/yr

Section 11

ANALYSIS OF CORE-DAMAGE PHENOMENOLOGY

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Section 11

ANALYSIS OF CORE-DAMAGE PHENOMENOLOGY

As part of the determination of the risk potential associated with core-damage accident sequences for the CRBRP, a review of the core-damage phenomenology is necessary. How core damage proceeds, its effects on the primary system boundary, and the timing and energetic potential associated with core damage are important to determining the challenge to containment and the ultimate release of fission products to the environment. This section addresses the phenomenology related to the core-damage processes and describes the use of a core-response event tree to define and estimate the probabilities of core-response scenarios.

11.1 INTRODUCTION

The following subsections contain discussions related to the development of the core-response event tree model and its quantification. Section 11.2 addresses the relationships of this task to other tasks in the CRBRP PRA. Section 11.3 contains a brief description of the reactor enclosure so as to provide terminology necessary for understanding the core-damage process. Section 11.4 describes the progression of core-damage accidents first in general terms, and then more specifically related to each type of core-damage event. Section 11.5 addresses the development of the core-response event tree model for analyzing the core response during an accident. Finally, Section 11.6 addresses the probabilities for each core-response scenario given a specific core-damage initiator.

11.2 INTERRELATIONSHIPS WITH OTHER PRA ELEMENTS

The core-response event tree provides a mechanism for investigating the various physical processes associated with different core-damage sequences in order to assess the potential impact on the ability of the containment to maintain its integrity as the final barrier to fission-product release. The analysis of core-damage phenomenology must therefore be closely coordinated with the definition of core-damage sequences (see Section 3), so that these sequences are defined in a suf ficient level of detail to enable investigation of the various credible physical responses. Core-damage phenomenology must also be coordinated with the containment-phenomenology analysis (see Section 12), to assure that the different core-damage processes that can produce different containment responses are accounted for.

The correlation between the plant-systems analyses and consequence analysis is accomplished through the selection of a set of core-damage bins. Each core-damage bin represents a category of core-damage sequences all of which would be expected to exhibit substantially the same core and containment responses. The set of core-damage bins is intended to comprise a discrete representation of the full spectrum of possible core-damage sequences producing different phenomenological responses.

The definition of core-damage bins and their impact on the core-damage event tree are elements in the iterative process of developing the accident sequences. The core-damage event tree was constructed initially to provide the most convenient means for quantifying the frequency of core damage. A set of core-damage bins was then developed, indicating

the need for further breakdown of some of the event-tree top events. As the core- and containment-phenomenological analyses are completed, some redefinition of the bins may be necessary. As a final check on the adequacy of the bin definitions, the sequence-level minimal cut sets dominating the frequency of each bin were examined to verify that the core and containment responses attributed to the bins appropriately reflect expected response for those cut sets.

The set of core-damage bins has been developed based on the parameters important to the consequence analysis and a review of previous phenomenological analyses of the CRBRP and other PRAs. In addition to the conditional probability of a release from containment, the following parameters have been found to have important effects on the offsite consequences:[1]

- magnitude and isotopic content of release (source term)
- energy of release
- evacuation time
- duration of release

These parameters are discussed in greater detail in Section 12.

The mechanisms by which radionuclides are released from the fuel depend on the core-damage progression in the reactor vessel. The core-damage progression depends, in turn, on the timing associated with core disruption and energetics (if applicable), and on core debris coolability while still in the reactor vessel. Previous phenomenological investigations have indicated that core-damage progression paths can be characterized

by four principal accident types: (1) unprotected (i.e., failure of the plant protection system to function) loss of flow; (2) unprotected loss of heat sink; (3) unprotected transient overpower; and (4) protected loss of heat sink.[2,3,4] In terms of relevant system features, these accident types can be described as follows:

1. Unprotected loss of flow: an initiating event (e.g., a loss of offsite power) causes a flow coastdown and the reactor fails to scram. Another sequence would be an event that results in a scram signal that successfully trips the pumps but fails to insert the rods.

2. Unprotected loss of heat sink: an initiating event causes the loss of the normal heat sink and the reactor fails to scram.

3. Unprotected transient overpower: the initiating event involves a positive reactivity insertion, but the reactor fails to scram.

4. Protected loss of heat sink: an initiating event results in a reactor shutdown, but the shutdown heat removal systems fail to provide adequate core cooling.

Releases from the reactor to the containment could occur due to failure of the reactor vessel closure head or due to failure of the bottom head. Rupture disks between the reactor cavity and the upper reactor containment building are designed to rupture when the sodium from the reactor vessel spills onto the reactor cavity floor following bottom head failure. Thus, in both cases a direct release path to containment is established immediately following vessel failure. The probability and degree of reactor vessel closure head failure are dependent on the energetic potential of core disruption. The timing of bottom head failure depends on whether the core damage proceeds rapidly; as in the case of an unprotected accident or over a longer period, as in the case of a protected accident.

The paths for release of radionuclides from containment could include breaches of containment resulting from missiles or from overpressurization, failure to isolate paths connecting the containment atmosphere to the external atmosphere (i.e., failure to isolate containment), venting via the containment cleanup system, and penetration of the containment basemat. The likelihood of a missile failing containment would depend on the energetic potential of the core-damage accident. The potential from overpressure failure of the containment could depend, in part, on the time history of energy addition to containment and the production and behavior of combustible gases (i.e., hydrogen). The time associated with core damage can be conveniently broken into four categories: (1) very rapid damage due to an unprotected accident; (2) early damage due to loss of sodium inventory; (3) relatively early core damage due to immediate failure of the shutdown heat removal system; and (4) late core damage due to long-term failure of the shutdown heat removal system. The latter two categories suggest a further breakdown of the protected loss of heat sink accident. The breakdown indicated is a logical choice since the success of short-term core cooling results in the removal of substantial sensible heat and time for a reduction in decay power. These combine to extend significantly the time it would take to boil off sufficient sodium to lead to core damage.

Based on the use of the principal accident types, accounting for two protected loss of heat removal categories, and treating the loss of sodium inventory category under both protected and unprotected cases, a

logical breakdown of core-damage sequences to provide an adequate definition of core-damage bins can be defined as follows:

Bin	Description
1	Unprotected loss of flow (ULOF)
2	Unprotected loss of heat sink (ULHS)
3	Unprotected transient overpower (UTOP)
4	Unprotected loss of sodium (ULOS)
5	Protected early loss of heat sink (LHSE)
6	Protected late loss of heat sink (LHSL)
7	Protected loss of sodium inventory (LOS)

Further discussions related to these core-damage bin assignments are provided in Section 11.4, while the interrelationships between the core-response event tree and the containment-response event tree is discussed in Section 12.2.

11.3 DESCRIPTION OF THE REACTOR ENCLOSURE*

The CRBRP is a mixed plutonium-uranium oxide (PuO_2/UO_2) fueled, sodium-cooled, fast reactor having a thermal output of 975 Mwt. A schematic of the reactor is shown in Figure 11.3-1. The schematic depicts the reactor vessel, a closure head which allows for through-the-top refueling, nozzle connections to the primary heat transport system (PHTS); and associated internal structures. The general arrangement of the reactor core on the core support plate within the vessel is also shown in relation to the inner and outer plenum zones.

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*Much of the material in this section is taken directly from Reference 5.

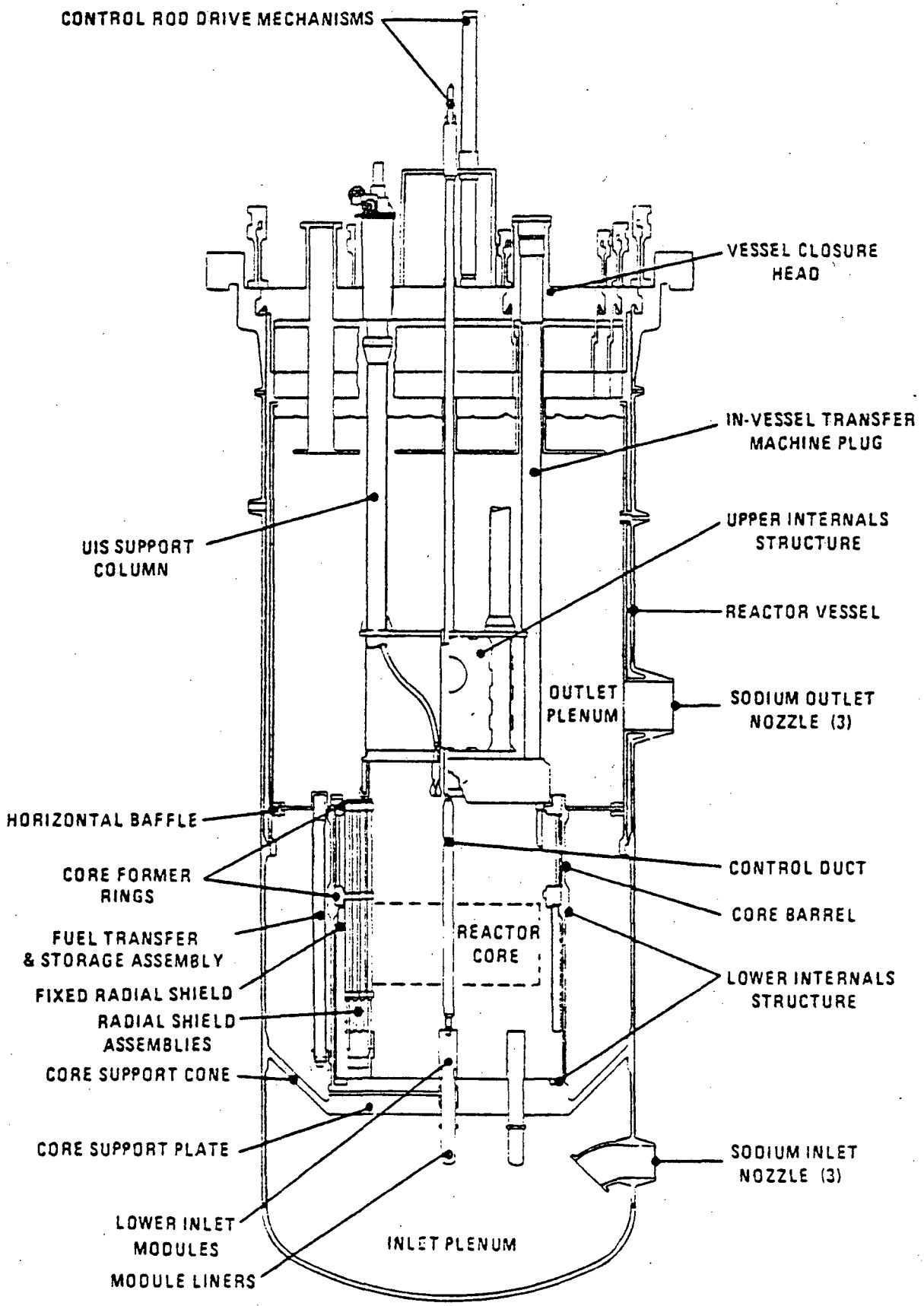

Figure 11.3-1. Reactor and Closure Head Schematic.
(From Reference 5)

The first loading, referred to as beginning of cycle 1 (BOC-1), consists of 156 fuel assemblies, 82 inner blanket assemblies, 126 radial blanket assemblies, and 15 control assemblies (9 primary and 6 secondary) as depicted in Figure 11.3-2.

At the end of the first cycle, three inner blanket assemblies from the sixth row are replaced with fresh fuel assemblies. At the end of cycle two, the entire core (159 fuel assemblies and 79 inner blanket assemblies) is replaced with a BOC-1 core configuration (156 fuel assemblies and 82 inner blanket assemblies). At the end of the third cycle, 6 inner blanket assemblies from row six are replaced with fresh fuel assemblies. The odd numbered cycles therefore have core configurations identical to BOC-1 and BOC-3, whereas the even numbered cycles after BOC-2, have core configurations identical to BOC-4. The inner and outer radial blankets are first replaced at the end of cycles four and five, respectively, and at equivalent time intervals thereafter. The primary control assemblies used for control purposes, must be replaced about every two cycles.

Each fuel assembly consists of 217 fuel rods, an outer duct, and inlet and outlet nozzle areas as shown in Figure 11.3-3. The outer duct is a hexagonal tube that forms a discrete coolant flow path for each assembly. The inlet and outlet nozzles associated with each assembly allow the sodium coolant to pass by the fuel rods as the coolant is pumped upward through the core.

Figure 11.3-4 shows that each fuel rod contains a stack of PuO_2/UO_2 core pellets housed in a stainless steel cladding. These pellets form the

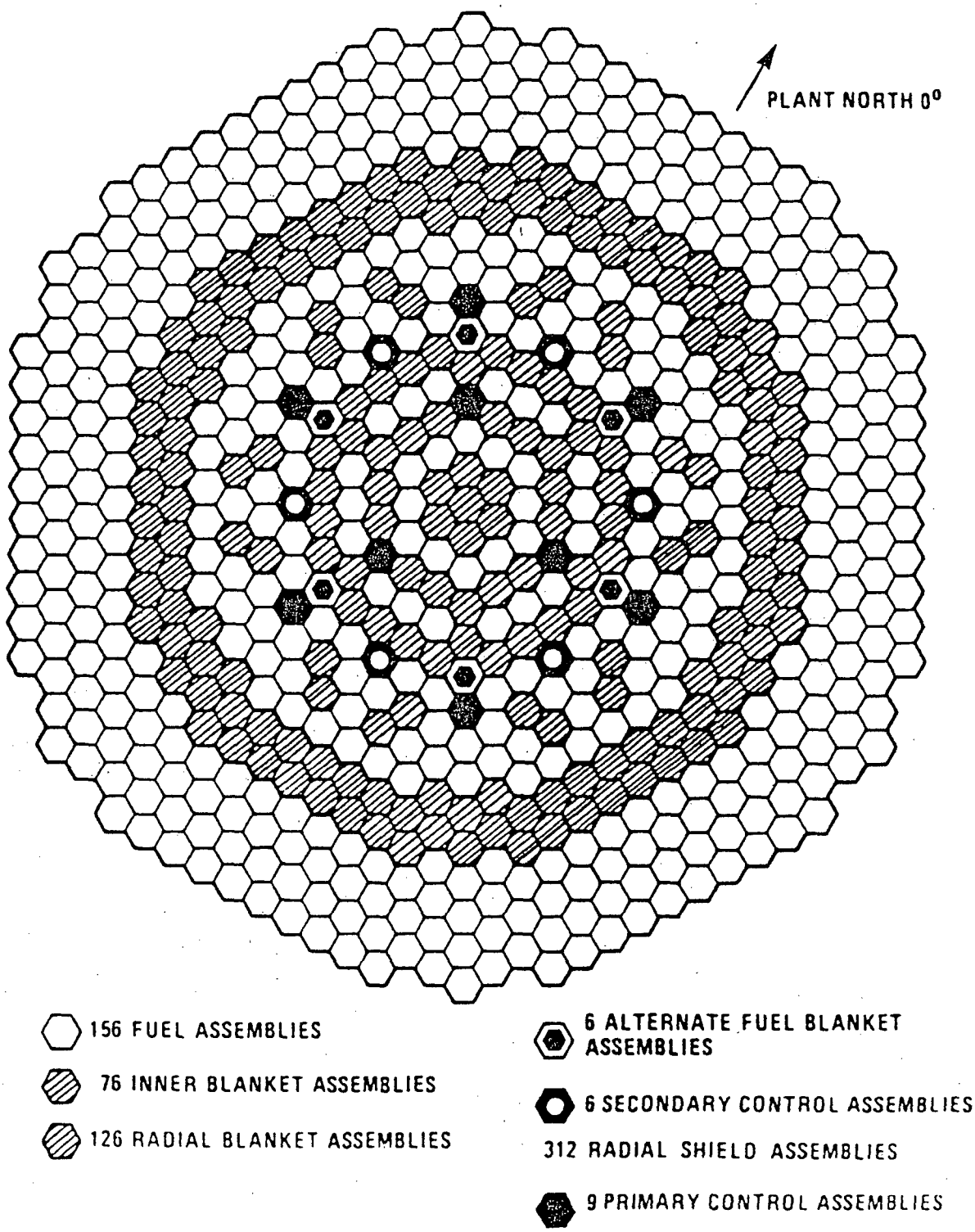

Figure 11.3-2. Clinch River Breeder Reactor Core Layout for BOC-1. (From Reference 5)

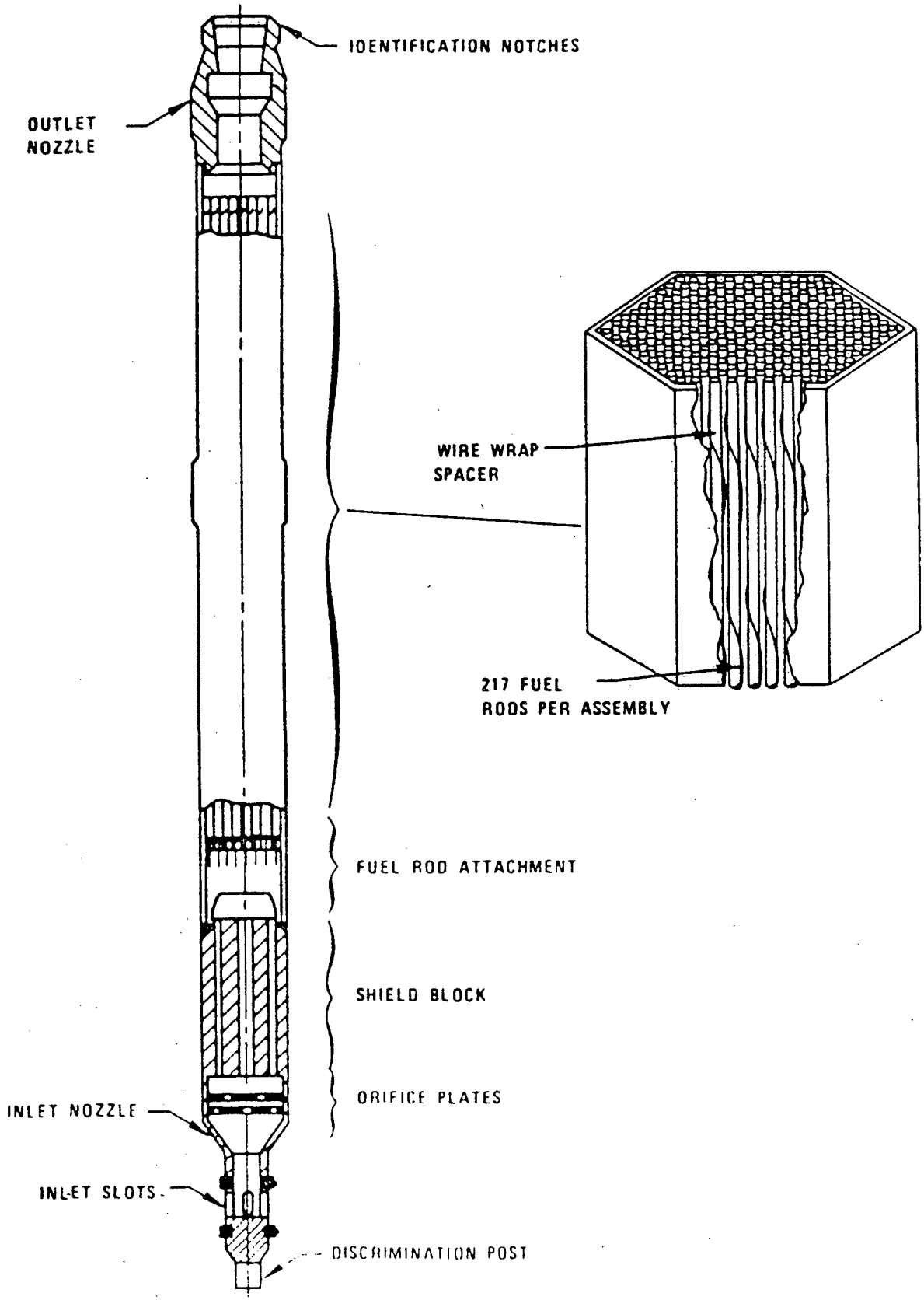

Figure 11.3-3. Fuel Assembly Schematic.
(From Reference 5)

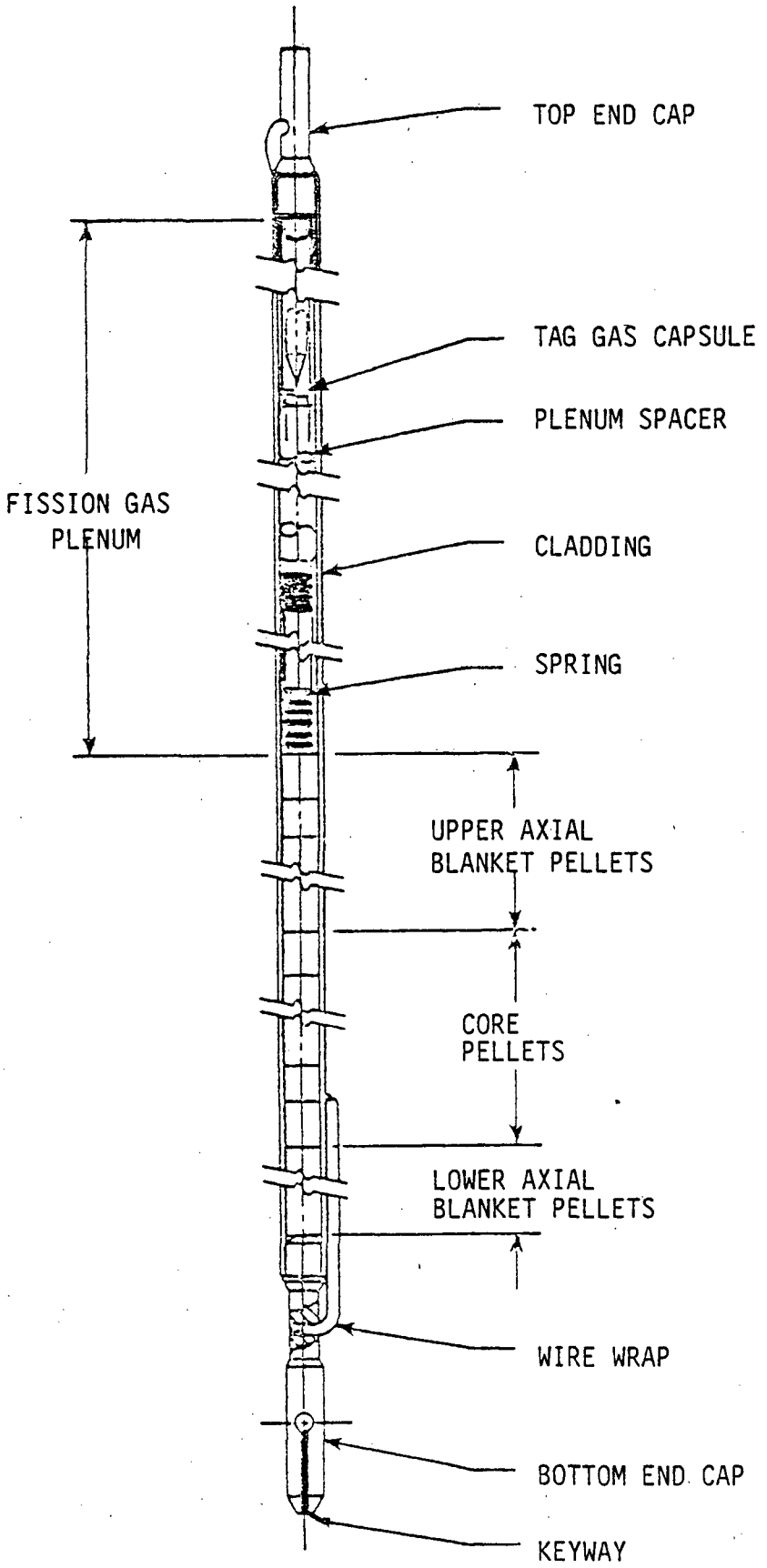

Figure 11.3-4. Fuel Rod Assembly.
(From Reference 5)

36 inch active core region. Fourteen inch stacks of depleted UO_2 pellets form the upper and lower axial blanket regions. A plenum is provided in each fuel rod to collect fission gases released during operation of the core. A small capsule of tag gas that is unique to each assembly is included in the rods for detection and location of leaking fuel rods.

The inner and radial blanket assemblies provide most of the fertile material for breeding fissile plutonium. The blanket assemblies are arranged in a configuration similar to the fuel assemblies, although the number and diameter of the rods are different. Each blanket assembly contains 61 rods with an outside diameter of 0.506 inches. This compares with an outside diameter of 0.230 inches for each fuel pin.

The control assemblies, using boron carbide (B_4C) pellets clad in stainless steel rods, in conjunction with the control rod drives, comprise two redundant, independent reactor shutdown systems. The primary system is the only one of the two that is used to control the net reactivity and power level of the reactor core. However, each system is capable of independently shutting down the reactor with one stuck control assembly.

The sodium coolant removes heat generated by the reactor core to the intermediate heat exchangers. The core inlet and outlet sodium temperatures during power operation are 730°F and 995°F, respectively. Under these conditions, the expected maximum fuel cladding temperatures during power operation are approximately 1300°F.

The general arrangement of the reactor vessel, closure head, guard vessel, and associated structures is shown in Figure 11.3-5. The reactor vessel is supported by the reactor vessel support ledge just below the head access area. The reactor is 59 feet long with a diameter of 20 feet. The sodium containing portion is primarily stainless steel designed for pressures on the order of 15 psig (plus sodium head) and temperatures of 1100°F in the outlet plenum region and 200 psig and 775°F in the inlet plenum region.

The reactor vessel closure head consists of three rotating plugs constructed of SA508 Class 2 steel. Each plug is provided with mechanical and electrical interlocks which prevent plug rotation during reactor operation. A heating and cooling system is provided to maintain the closure head at a nominal 400°F.

A guard vessel surrounds the reactor vessel to assure that the sodium level in the reactor vessel always remains sufficiently above the outlet nozzle to ensure adequate loop flow in the event of a leak in the reactor vessel piping, or connections. It is constructed of 304 stainless steel and extends approximately 6 feet above the minimum safe sodium level. The reactor guard vessel is supported by the guard vessel skirt imbedded into the reactor cavity floor.

11.4 OVERVIEW OF HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS

Design features and safety margins are incorporated into the CRBRP design to assure that the risks of operating the plant are maintained at a low level. These features and margins are provided to prevent accidents

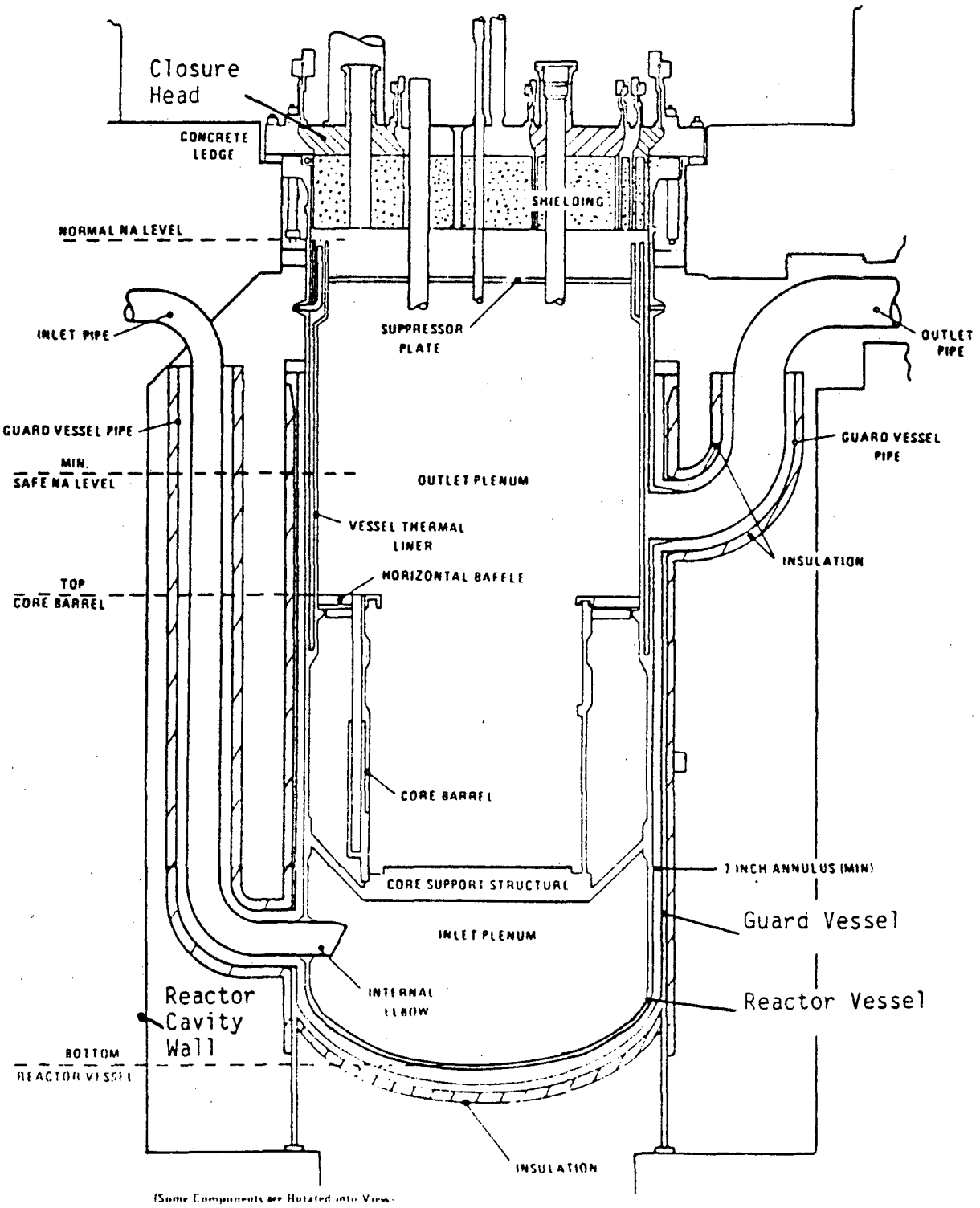

Figure 11.3-5. Reactor Enclosure.

as well as to mitigate the effects of design-basis accidents. In addition, CRBRP incorporates design features strictly for the purpose of mitigating beyond-the-design-basis events. Although excluded from the design basis, core-damage accidents in liquid metal fast breeder reactors (LMFBRs) have received a great deal of attention. Meltdown can possibly yield core configurations of higher reactivity than in the original core geometry. The resulting high temperatures and pressures within the primary boundary, and potential effects on that boundary as well as on containment barriers, has given rise to the study of "energetics" for the LMFBR.

In the early considerations of LMFBR energetics, the term hypothetical core disruptive accidents (HCDAs) was in common use. Theofanous and Bell[3] suggest that this phrase was used to denote the low probability of such accidents as well as the tentative nature of our understanding of such accidents and the hypothetical situations which could lead to them. Reactivity changes associated with core material relocation in these accidents could yield high power transients. In such cases, core disruption and dispersal away from the original geometric configuration may be necessary before permanent subcriticality is achieved.

HCDAs have the potential to release significant amounts of radionuclides from the primary coolant boundary and threaten the integrity of the containment boundary. Such events could occur as a result of failure of the safety systems used to terminate anticipated plant events. As discussed in Section 3.2.1.1, functions of the safety systems include shutting down the reactor, maintaining sodium inventory, and transporting

decay heat from the reactor to the ultimate heat sink (the environment). In short, the safety systems are designed to prevent conditions where significantly more heat is produced by the core fuel than is being removed. Such an imbalance is the necessary condition for a HCDA to occur. Two general categories of events causing this condition have been identified.[2] These are:

- insufficient heat removal, and
- heat generation in excess of removal capability.

Sequences of events leading to either or both of these two general categories are the subject of Section 3. Sequences involving a sustained imbalance in the heat production/heat removal rates will initiate fuel and/or clad damage and may eventually lead to core-wide melting and gross material (i.e., fuel, steel, B_4C, sodium) motion. This progression is important, from an energetics perspective, provided a sufficient fraction of the fuel (typically more than 60% for the CRBRP) remains within the core region.[2] Neutronically active states are then possible through a variety of rearrangements of fuel, blanket, structural, control, and coolant materials. When such states are obtained by fuel compaction in supercritical configurations following highly but temporarily dispersed subcritical fuel states, they are called "recriticalities." Permanent subcriticality, or "termination" (termination of energetic concerns), on the other hand, may occur at any point along the continuum of core-disruption states. When the relocation of the appropriate quantity of fuel occurs in a forceful manner, this condition is referred to as "energetic termination" or "hydrodynamic disassembly," or simply "disassembly." When this relocation is benign

it is referred to as "mild termination" or simply "dispersal." The remainder of Section 11 deals with an analysis of the possible core-response progressions and resulting states which could occur given a HCDA.

11.4.1 Phases of a HCDA

The progression of a HCDA can be divided into four phases.[2,6] These are the initiating phase, meltout phase, large-scale pool phase, and hydrodynamic disassembly. The accident may terminate during any phase, and hydrodynamic disassembly may occur during any of the other three phases.

The initiating phase covers the early phase of the accident. Possible events leading to an imbalance between heat generation and heat removal have been discussed in Section 3 and will be further addressed in Section 11.4.2. During such an imbalance, the core temperature increases, the fuel and/or cladding starts to melt, and relocation begins.

Depending on the particular accident sequence, a number of important physical processes are relevant to determine the core response during the initiating phase as well as the later phases of a HCDA. These processes have been summarized in a number of reports.[2,3,4,6] Examples of these important physical processes include the heatup rate of the fuel, the amount and rate of sodium voiding, Doppler and material worths, burnup, clad-relocation effects, potential for blockage of coolant flow, autocatalytic behavior due to fuel compaction, location of cladding failure, fuel-coolant interaction potential, and the degree of fuel particulate sweepout from the core by sodium flow. Certain processes are

more important for some types of HCDAs than for other types. Each general class of HCDA and the more important physical processes associated with each HCDA are addressed in Section 11.4.2. Depending on the HCDA and subsequent core response, the initiating phase could theoretically end with a partially damaged but coolable core at shutdown or stable power conditions, or progress to the meltout phase, or result in a hydrodynamic disassembly with possible reactor vessel head damage and a subsequent release of fission products to the containment.

The meltout phase is initiated with the melting away of the fuel and blanket assembly hexcans and the subsequent flow of fuel into spaces between assemblies. A major concern during this phase is that a recriticality event could occur resulting in a possible energetic disassembly. If an energetic disassembly does not occur the core will meltdown in a subcritical configuration and eventually cause failure of the reactor and guard vessels. However, if an energetic disassembly does occur the primary coolant boundary integrity will be challenged, possibly causing an immediate challenge to the containment boundary.

Although investigations of core-damage response indicate that it is not expected to occur, the large-scale pool phase would sequentially follow the meltout phase. The large-scale pool phase may involve a "bottled-up" fuel pool which, upon collapsing, could result in a relatively large energetic event.

The term "hydrodynamic disassembly" is used to describe the core response to a sustained, superprompt critical excursion that might be

predicted during any of the three phases discussed above. The primary assumption is that fuel vaporization results in pressures which exceed the mechanical restraint of the core internal structures.

Following are discussions related to each type of HCDA investigated in this study. Important processes and core-response sequences are addressed leading to the development of the core-response event tree in Section 11.5.

11.4.2 Potential Initiators for HCDAs

As discussed above, the basic cause of an HCDA is: heat generation greater than heat removal. The events leading to (1) insufficient heat removal, and (2) greater heat generation (than design basis), are discussed in general terms in order to classify these initiating events into categories to bound the problem as well as the discussion.

11.4.2.1 HCDAs caused by insufficient heat removal. The condition for insufficient heat removal is dependent upon the core power which is to be removed from the reactor core. Two distinct levels of core power used to study HCDAs are:

1. full power - as during normal plant operation, and
2. decay heat - as during the shutdown mode.

For this analysis it is assumed that the reactor is in full power operation when normal operation of the plant is challenged (e.g., loss of offsite power, turbine trip, etc.) thus generating transient conditions in the reactor system. At this point the reactor is signaled

to scram. Depending upon whether or not the scram occurs, the core will initially be in one of the two power levels noted above.

For the protected case (successful scram) the heat removal requirements are far less than for the unprotected case; also the available time before core disruption may vary (by orders of magnitude) for the two cases. Therefore, the "insufficient heat removal" case can be further divided into the protected and unprotected categories, each of which can have four possible initiator scenarios:

1. loss of flow,
2. loss of heat sink,
3. loss of sodium, and
4. local core disruptions.

These scenarios are discussed below in general terms.

11.4.2.1.1 Unprotected HCDA's. An unprotected HCDA occurs due to failure to scram the reactor following a transient condition requiring shutdown. Failure to scram can be the result of scram signal failures, control rod drive mechanism malfunctions, failure of the core support structure, external events (e.g., seismic), etc.

Each of the four potential initiators for unprotected HCDAs can affect the outcome of an HCDA and possibly result in different states of a damaged core. These transient conditions are discussed below including the individual scenarios and core behavior. Those that follow a similar pattern due to the timing and nature of the core disruption progression, are considered as one "class" of accidents and categorized into a core-damage bin for purposes of the analysis.

Unprotected Loss of Flow (ULOF) Accident

This accident has been assessed to bound the consequences of all HCDA's.[2,3,6] ULOF events would occur following an initiating event that causes loss of flow (e.g., loss of offsite power, failure of a trip signal, etc.) without reactor scram.

The core-damage progression for the ULOF event is summarized in References 2, 3, and 6. During the initiating phase, the primary flow coastdown results in a heatup of the core which causes rapid sodium boiling in the fuel channels introducing positive reactivity and a power increase. Fuel temperatures will then increase, resulting in melting of the fuel and cladding. Higher fuel temperatures and sodium voiding induce fuel disruption.

Theofanous and Bell did not rule out the possibility of fission gas fuel compaction generating an energetic response during this phase.[3] Sensitivity studies performed to evaluate this mechanism indicate that the range of reactivity feedback is rather narrow. At low feedback levels the gas may disperse before fuel column integrity is lost and thus little compaction of the fuel can occur. At the high end of reactivity feedback, the disruptive force of vaporizing fuel disperses the fuel in a transient overpower (TOP) event before the fission gas pressure can accelerate the fuel. Theofanous and Bell found that the reactivity insertion levels that could be generated following fission gas-driven fuel compaction are conservatively limited to approximately 50$/second as compared to over 100 $/second required to cause damaging energetics.[2,3,6]

This phenomena is discussed in greater detail in Section 11.6.4.1. Several different accident paths, including variations on rate of fuel collapse, ranging from free-gravity to slow drainage (0.25 g) were considered by the CRBRP Project in Reference 2. It was concluded that all of the analyzed core failures result in a power burst (because of positive reactivity from fuel collapse), but at a level short of super-prompt criticality due to fuel expansion and Doppler effects. These dispersal mechanisms act to shutdown the neutronic activity.

As the ULOF event continues into the meltout phase, the molten fuel behavior is critical to the course of the sequence. From an overview standpoint, energetic levels are possible only if a large fraction of the active fuel can be brought together in a homogeneous liquid form while excluding control material, sodium, cladding, or depleted fuel (blanket assemblies), all of which reduce the reactivity insertion. This would maximize the reactivity inserted and could cause damaging energetic levels. To achieve such a state in the heterogeneous breeder design would require a very precisely-timed sequence of events, some of which may not be physically possible.

As more assemblies undergo melting and surging, further collapse takes place. Each surge, however, will tend to disperse fuel. The longer this process occurs, the more opportunity exists for melting and mixing of blanket assembly materials into the molten fuel. In this mode the surges may become sufficient to drive a large amount of fuel into the blanket assemblies and other interstitial sites shutting down neutronic activity. Blanket and control material may also be mixed with the

molten fuel permanently shutting down the neutronic activity. Another more remote possibility is a final contraction that is sufficient to cause enough reactivity insertion to reach an energetic (though not necessarily structurally damaging) disassembly of the core.

Significant energetic behavior through oscillations of the molten core materials has been concluded to be physically unreasonable by Theofanous and Bell.[3] Their analysis of the requirements of an overpower transient driven by a loss of flow (LOF-d-TOP) having substantial energetics required unlikely assumptions which would maximize reactivity (i.e., high estimate of sodium worth) while ignoring phenomena which would reduce if not eliminate power ramping (i.e., axial fuel expansion, no fuel dispersion, no control material mixing, incoherency effects absent, etc.).

The distribution of core debris following termination of an HCDA depends on the prior progression path and on the specific HCDA initiator. If the ULOF event is non-energetically terminated in the meltout phase, as predicted in Reference 2, the fuel debris is expected to be roughly equally distributed above and below the active core region. However, if the HCDA is terminated energetically, a greater fraction of the fuel could be ejected upward. Based on the results of analytical models and experimental tests, reported in Section 3 of Reference 2, the CRBRP Project concluded that more than half of the fuel debris ejected upward will settle on the upper internals structure and primarily the horizontal baffle (assuming it remains horizontal). The rest of the fuel debris ejected upward was estimated to enter the PHTS piping and settle at various locations upstream of the IHX.[2]

The downward moving fuel debris which, depending on the progression path, is estimated to comprise between 50% to 10% of the active core; is predicted to either ablate the steel plug below the molten fuel front or to flow through the shield orifice blocks into the inlet flow modules. The fuel that accumulates in the inlet flow modules may initially form a particulate bed. However, as this bed grows in size, the inlet module geometry will no longer be able to accommodate the fuel debris. In addition, the bed depths within the modules may exceed coolable limits causing the previously particulate fuel to become molten.

Fuel debris upon leaving the inlet modules, will likely settle on the bottom head and form another particulate bed. Any molten fuel that enters the lower plenum will probably fragment and particulate when in contact with the bulk sodium.[2] The heat rejection capability of the debris-bed in the bottom head will depend, to a great extent, on the debris-bed depth and composition. For shallow beds the heat will be removed by conduction and convection.

Sodium boiling will occur within deeper beds creating vapor channels that vent from the interior of the bed to the overlying sodium. If the heat generation rate (associated with fission products within the fuel) or bed depth are such that bed dryout is not precluded, the temperature of the bed will rise until the steel, and potentially the fuel within the bed becomes molten.[2] Bed dryout implies that sodium cannot penetrate into the bed to provide an upward heat removal path and that sodium film boiling is occurring throughout the bed. This continuing process leads to eventual failure of the vessel bottom.

If the vessel bottom does not fail due to meltthrough of a non-coolable debris bed it will most likely fail as a result of creep rupture. In either case, some combination of fuel, steel (molten or particulate), and sodium will eventually penetrate through the reactor vessel and guard vessel and enter the reactor cavity. The phenomena from this point on is the subject of Section 12.

In short, the ULOF accident includes the following characteristics and has been assigned core-damage bin 1:

1. Fuel failure with sodium around the core,
2. Very short time to core melt (tens of seconds),
3. Small possibility of initiating phase energetics with a high likelihood of entering the meltout phase,
4. Debris removal most likely through the reactor vessel bottom.

Unprotected Loss of Heat Sink (ULHS) Accident

A postulated ULHS would be characterized by failure to trip the reactor on loss of main feedwater so that the heat sink is lost without an adequate reduction in reactor power.

The ULHS core-damage sequence has not been analyzed in great detail. Reference 7 points out a need for such analysis and mentions that the entire PHTS heats up in a matter of minutes.

Except for the very early stages of the accident, the core-damage progression for the ULHS event is expected to be similar to the ULOF accident as discussed above. In fact, the ULHS event can be easily

transformed into a ULOF event by tripping the primary pumps or by their subsequent failure. However, it is during these early stages of the accident when fuel melting begins, that the continued forced flow of sodium due to operation of the primary pumps may allow for early fuel sweepout. This characteristic provides less of a chance of early energetics than for a ULOF event. Therefore, the ULHS event has been assigned core-damage bin 2 due to this potentially significant factor as well as its unique initiating characteristics.

Unprotected Loss of Sodium (ULOS) Accident

Failure to scram in conjunction with a loss of sufficient sodium (below the minimum safe level) due to a leak will result in a ULOS accident. A ULOS accident can occur following reactor vessel/PHTS pipe rupture or leak with failure of the guard vessel or following a PHTS pipe rupture or leak with failure to trip the primary sodium pumps.

Like the ULHS event, this event has not been assessed in detail due to the assumption that it is bounded in frequency and consequences by other events. It is assumed here that ULOS would be similar to ULOF during the core disruption period if the primary sodium pumps are tripped or otherwise fail. However, in the ULOS accident (given continued flow), overheating of the reactor core will not occur until the sodium level in the reactor vessel falls below the minimum safe level and siphon is broken in all three PHTS loops. Thus at the time of fuel disruption the volume of sodium above the core can be substantially different for the ULOS event relative to the ULOF event. As discussed in Section 11.6, this smaller volume of sodium may have an impact on the energy transmitted

to the reactor vessel head following energetics. In addition, sodium may be present in the reactor cavity by the time the core debris melts through the reactor vessel depending on the initial cause for the loss of sodium. This could affect the coolability of the debris bed in the reactor cavity such that the resulting containment failure probabilities are different from those involving other types of unprotected events. The above characteristics make it desirable to categorize the ULOS event as core-damage bin 4 for analysis using the core-response event tree.

Unprotected Local Core Events

Local core failures might also result in transients leading to a HCDA.[2] Positive reactivity can be induced by the occurrence of a large void in the core. The introduction of a void could be the result of fuel pin cladding failures which discharge the built-up fission gas into the coolant flow or by a coolant system leak near the coolant pump intake which allows aspiration of cover gas into the system. In the first case the void would be very localized and the resulting power ramp would be insufficient to fail the fuel bundles directly involved. Even if melt were to occur, the slow rate of fuel heat up would allow some clad heating to occur. Failure by clad ballooning, channel blocking, and melting would follow. The fuel would be expected to exit the failure location slowly and be sweptout, drain or disperse. Extensive failure in adjacent rods would not be expected. Inert gas entering the system via a leak is insufficient to cause significant perturbation. If large amounts of the core could be blanketed, the timing of void formation is likely to resemble a ULOF. The possibilities of local fuel blockage

have also been discussed in Reference 2, but such scenarios have been judged to be incredible.

The failed fuel monitoring system will play an important role here, especially in the initial stages of the accident, to warn the operator of the situation and initiate a plant trip manually. In any event, the consequences of any postulated sequence of events is bounded by the other unprotected events and therefore, no further analysis is considered.

11.4.2.1.2 Protected HCDA's. In the protected core HCDAs, the reactor trips at the start of accident conditions or a transient initiator. Thus, the heat removal requirements are based upon the decay heat generated by the fission products, and sensible heat from the PHTS and reactor internals.

Protected Loss of Flow (PLOF) Accident

A PLOF event can happen because of shutdown of primary system pony motors following a reactor trip. This will result in loss of all forced circulation through the primary system, but density differences in the sodium surrounding the core will set up flow by natural circulation. Detailed tests have been performed[6] and analysis conducted which has demonstrated that natural circulation will support heat fluxes corresponding to 8% of full power without any damage to the core. Therefore unlike ULOF, a protected loss of flow will not cause any core damage by itself. Some unpublished tests demonstrate natural circulation can provide sufficient heat removal up to 25% of full power.

In view of the above information it seems unlikely that core damage can occur only because of loss of flow on reactor trip. That is to say, other functions would also need to fail prior to core disruption. Two of these functions are the availability of the heat removal medium (i.e., sodium) and of the heat sinks, and PLOF does not necessarily imply failure of either. PLOF is therefore not treated as an independent core damage bin since it is dominated by the loss of heat sink (PLHS) and the loss of sodium (LOS) events, which are discussed in the following paragraphs.

Protected Loss of Heat Sink (PLHS) Accident

During the shutdown state, insufficient heat removal can be caused by a loss of heat sink resulting in a core disruptive accident. There are a number of heat sinks available for this purpose, e.g., SGAHRS, PACCS, and DHRS. A detailed discussion of these and their behavior (with success criteria) can be found in Section 3 of this report. One important core response identified in the event tree discussion is that the loss of heat sink can happen in two distinct time frames: early and late.

Since time to core melt is an important factor in determining the consequences of a HCDA, the loss of heat sink category is therefore divided into:

a. Loss of heat sink early, LHSE, and

b. Loss of heat sink late, LHSL.

The protected loss of heat sink (PLHS) event is briefly described below.

Upon losing all heat sinks and a successful reactor trip, the reactor core begins to raise the temperature of the surrounding sodium

slowly. Once the sodium surrounding the core reaches saturated conditions it starts evaporating rapidly. The core will not likely be damaged unless all the sodium above the core has boiled off[3,8,9] or unless the heat flux in any given coolant channel exceeds the dryout heat flux. Clad melting will commence soon after the core is uncovered of sodium. Downward cladding relocation will then take place[9] possibly forming plugs at the lower blanket assembly. Theofanous and Bell[3] predict that the sodium vapor velocities would be inadequate to carry the molten metal to the upper blanket assembly, and therefore, exit blockage does not occur.

What happens to the fuel pellets after the cladding has melted is an issue of great controversy. In a paper on core coolability following loss of heat sink accidents[10] the authors indicate that "fuel pin toppling, slumping, and meltdown processes will follow core uncovery...." Fuel slumping is discussed in Reference 8 which gives six cases of fuel assembly failures and discusses the status of recriticality for each. Reference 9 also gives four cases of fuel failure modes with discussions on possible recriticality for each.

A comprehensive study of fuel behavior under different HCDA conditions was performed by Theofanous and Bell[3] who assume that gross fuel motion occurs (resulting in rearrangement of fuel pellets into a more compact geometry), with failure of the subassembly walls, before the fuel starts to melt. Theofanous and Bell also state that the remaining above-core steel structure would fall on top of the fuel pellets and blanket rubble. They estimate that this pile will reach mild criticality

because of gradual melting and draining mechanisms. At this point the neutronic activity will accelerate, increasing the rate of fuel melting but will then collapse due to mixing of the molten upper axial blanket.

It should be noted that without sodium above the core as a medium for transferring energy, damage to the reactor vessel head can be ruled out following a protected loss of heat sink (PLHS) recriticality event.[3,8]

The vessel failure mechanism is also in question since there is the possibility of interference between the reactor vessel and the guard vessel due to thermal expansion and creep at high temperatures. The CRBRP PRA (Phase I) report[4] also discusses failure modes of the reactor vessel due to thermal expansion.

The vessel has four potential failure modes. These are at the ring seals on the reactor head assembly, the inlet and outlet nozzles through interference with the guard vessel and bending over-stress failure at the coolant piping support locations. The ring seals are located on risers where failure due to temperature must be examined. The failure of the seal would provide a path to the upper containment. Boil-off of the sodium will take a very long time if it is lost solely through this path.

Regardless of the seal ring failure, the vessel will continue to creep until contact of the inlet nozzles with the vessel guard pipe. Contact of the outlet pipe has been calculated to occur almost simultaneously with the inlet pipe/guard vessel contact assuming that the reactor vessel and guard vessel are at the same temperature.

This assumption is reasonable because of the small separation of the reactor and guard vessel and the insulation on the outside of the guard vessel; however, the temperature of the guard vessel should actually be slightly lower. The growth of the guard vessel at the outlet nozzle would therefore be reduced favoring contact of the inlet nozzle first. Since in reality it would seem that construction tolerances will result in off center placement of the primary system piping within the guard vessel, contact of the inlet or outlet nozzles does seem approximately equally likely.

As the vessel undergoes thermal growth, the primary system piping will also be undergoing expansion. This expansion could lead to bending stresses which may fail the pipe(s) at any of several locations, pipe bends or restraint locations.

Failure of the seal ring appears likely because of the relatively low design temperature of the seals. Failure of the inlet or outlet nozzle, or both, appears to be inevitable. Failure of the piping due to thermal and bending stresses is less clear.

In brief the PLHS accident includes the following characteristics:

1. Slow heatup of primary system (~1 -2°F per minute) allows greater time for recovery.

2. The core is likely coolable as long as it's surrounded by sodium.

3. Two principal times of interest exist: early and late loss of heat sinks.

4. Steel melting prior to fuel melting (i.e., enters meltout phase directly).

5. Reactor vessel structural failure possible prior to approaching coolant saturation temperature.

With these characteristics the early loss of heat sink (LHSE) and late loss of heat sink (LHSL) events are treated as separate core-damage accidents and assigned to core-damage bins 5 and 6, respectively.

Protected Loss of Sodium (LOS) Accident

The LOS event can occur if the primary guard vessel fails to hold the primary sodium after a break in the PHTS (below the core level). This scenario will result in uncovering the core, thereby removing the heat transport medium. This will cause overheating of the core fuel resulting in a fuel melt and cladding failure. The core disruptive process would be very similar to the PLHS condition from this point on. In other words, LOS will follow the same accident process path as the PLHS case.

The containment failure probabilities and radionuclide release paths may be significantly different when comparing PLHS and LOS events. Therefore, the LOS event is assigned to core-damage bin 7 and will be examined as a separate case.

Protected Local Core Events

Local core events in a protected state are unlikely to cause severe accidents. They could potentially occur due to blockages within coolant channels such as the presence of foreign material in the primary coolant, a broken wrapper wire, excessive pin bowing, or excessive clad swelling.

Blockage of flow to an assembly is deemed less likely than blockage within an assembly (see Figure 11.4-1). The CRBRP is designed to provide

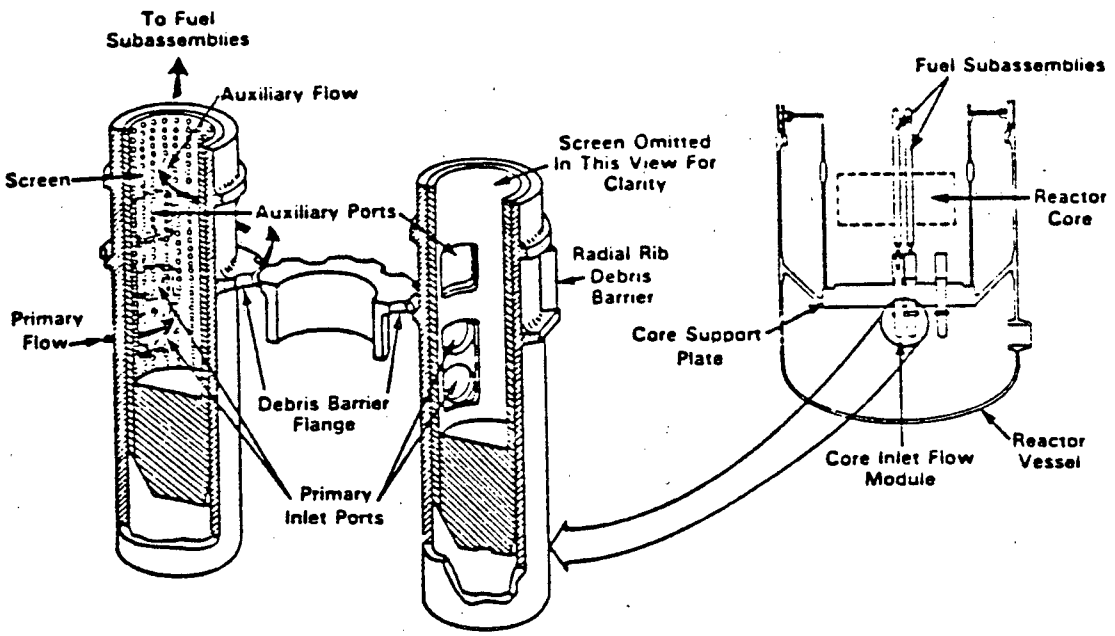

Figure 11.4-1. CRBRP Lower Inlet Flow Module. (Reference 21)

redundant flow paths to the subassemblies to preclude any object from starving flow to an entire subassembly. Only very small particles less than 0.25 inches in diameter could enter the subassembly inlet. Particles less than 0.25 inches in diameter would be trapped at the bottom of the fuel rod bundles. Based on analyses reported in Reference 21, a major buildup of particles that causes more than 80% of the flow area to be blocked, would not increase the coolant temperature by more than 200°F. This would not cause boiling even in the hottest fuel coolant channel. In addition, the CRBRP Project has outlined procedures for filtering out particles larger than 100 microns prior to startup. During operation the sodium purity will be maintained through the operation of cold traps.

Although non-mechanistic, total blockage of a fuel assembly would result in about a 10¢ reactivity insertion (maximum). In addition, the fuel-melt process would tend to be dispersive causing reduced reactivity.[2] Blockage growth due to ballooning and swelling of cladding resulting in release of fission gases and fuel particles would be very slow and would be detected by the failed fuel monitoring system in its early stages. Protected local core events therefore do not add significantly to the plant risk, and in the worst case it appears that they would cause failure of only a few fuel channels. Therefore, they will not be considered further in this analysis.

11.4.2.2 <u>HCDAs Caused by excessive power generation</u>. This type of accident can occur because of positive reactivity insertion in the reactor core, which can happen because of inadvertent control rod withdrawal,

insertion of fuel elements at improper locations, core voiding by coolant boiling or fission gap entrapment, etc. All of these are classified as the UTOP event described below.

Unprotected Transient Overpower (UTOP) Accident

This event is indicative of a rapid positive reactivity insertion and may occur in several different ways. It may be initiated by a sudden change in core geometry, by the introduction of voids in the core, or by inadvertent control rod withdrawal.

The most probable UTOP event, as an initiating transient, would occur through a failure of the electrical circuitry causing a withdrawal of one or more control rods. This event is mechanistically possible although with low probability of occurrence. Initiation of the event through external forces (such as seismic events) is difficult to describe in a mechanistic manner and would be expected to have a very low probability of occurrence.

The ability to generate energetic disassembly, recriticalities or nearly coherent fuel melting have been assessed by many researchers. The following description of a UTOP event is based on information from a number of references.[2,3,4,6]

The insertion of reactivity at power and at full flow results in rapidly increasing power. The fuel elements heat rapidly and expand. This occurs very quickly and before any significant heat transfer to the cladding. As the fuel begins melting it will pressurize the cladding causing it to rupture. Molten fuel will be forced into the coolant.

The cladding may fail in such a way as to trap sodium in fuel assembly coolant paths, and thus set up the potential for small scale fuel-coolant interactions. Other possibilities exist; the molten fuel may be sweptout of the channel by the flowing coolant, or it could be temporarily held up in the core region until nearly all fuel channels have also melted. Under some circumstances, the combination of fuel motion, clad motion, and sodium motion could result in positive reactivity feedback that leads to fuel vaporization and the potential for an energetic disassembly.

If a UTOP event is initiated by an external event which causes a change in the core geometry, the structural damage has to be such that control rods stick in place with a net relocation of fuel, increasing the average fuel density. In such an event, the fuel pins may be bent bringing fuel in closer proximity; however, the possibility of the fuel assemblies failing in such a manner as to reduce the core volume is remote. It is plausible that gaps will be formed next to pins or even assemblies which have been bent in another direction (i.e., the core may have a pattern of dense fuel areas and alternating void areas). It is also very likely that the condition will be created over a period of a few seconds rather than on the order of tens of milliseconds. Fuel disruption in the higher power channels would occur like the scenario described previously with the exception that some channels may be blocked by damaged fuel pins providing an opportunity for fuel-coolant interactions. In other core assemblies the availability of large escape

paths may allow sweepout of the fuel with minimum fuel-coolant interaction. The timing of individual failures will be random because of the stochastic damage process which would occur during a seismic event. Thus, the highly incoherent nature of the response is expected to limit the ability to transfer damaging levels of energy to structures.

As described above, the higher power assemblies may be involved in a mild to moderate energetic event. Any molten fuel not removed from the core during the energetic excursion may be partially sweptout of the core by the reestablishment of coolant flow and may partially drain into the lower blanket assemblies and lower plenum. Adjacent pins and assemblies will disrupt shortly following the initial failures. The residence time of the molten fuel from the initial failure will establish the power ramp in the adjacent fuel. Less energetics would be expected from these secondary failures, although core damage from the initial energetic response may affect the secondary failures. The staggered nature of the failures appear to limit the size of any energetic response.

The ability to cool the remaining debris and any undamaged assemblies after the initial disruption is uncertain. Flow to the core will still be in progress (due to natural circulation) even if protection circuitry has shutoff the PHTS pumps. The disarray and damage to the coolant channels may eliminate or substantially reduce flow to a large portion of the central core area. Fuel remaining in these areas after the initial fuel discharge would consist mainly of fuel rod extremities and

fuel pins which sustained structural damage but which did not participate in the initial power ramp. These could melt such that most of the molten fuel would be expected to drain into the lower blanket and plenum where quenching and freezing would occur. The cladding would likely freeze in the upper portion of the lower blanket, thereby trapping some of the fuel.

If PHTS pump flow is not lost during the disruption process, fuel may be coolable in the relatively intact portions of the core. Since substantial amounts of core material would be lost in the initial disruption, it is reasonable to expect the core to become subcritical. Following subcriticality the decay heat could be removed by natural circulation and the PHTS pumps would no long be required. In Reference 2 it is reported that the SAS-3A code predicted termination of the UTOP event by sweepout with less than 4% of the core ejected. The expected status of the core at the end of the disruption/dispersal phase would have some fragmented fuel scattered in the region above the core, some amount of pins relatively undamaged and coolable, a small amount of fuel material frozen in the lower blanket assembly, and large amounts of fragmented frozen fuel material in the lower vessel head. Depending on the amount of material, the size of the fragments and the porosity of the debris bed, a coolable geometry may be maintained. Otherwise, the debris will heat up, coalesce, melt and cause a breach of the lower vessel. The melt will be collected in the guard vessel where it is possible that the core material will eventually fall into the reactor cavity before a large quantity of sodium collects in the cavity.

Loss of vessel head integrity during the initiating phase due to energetics would not have any effect on the remaining core-damage sequence except that extensive core structural damage would be expected. Blockage of a large portion of the remaining core cannot be ruled out. If enough material to cause shutdown is not removed, the sequence would progress similar to the unprotected loss of flow (ULOF) case or more slowly if the core is shutdown by the disassembly.

While some of the characteristics of the UTOP are very similar to a ULOF event, probabilistic differences may be evident in the core response. In particular, chances for early termination of the event during the initiating phase appear greater and so the UTOP initiator is assigned core damage bin 3 and separately analyzed using the core-response event tree.

11.5 THE CORE-RESPONSE EVENT TREE

The core-response event tree used to address core-disruption scenarios is presented in this section. The event tree is used as an interfacing model between the core-damage event tree described in Section 3 and the containment-response models addressed in Section 12. Inputs for the core-response event tree are the core-damage sequences from the plant-system analyses. The outcomes of the core-response event tree are end states which summarize the physical status of the core and reactor vessel.

Events affecting the potential for liner failure and subsequent sodium-concrete interactions such as whether sodium or core debris are likely

to fall into the reactor cavity first, are also described as part of the core-response event tree end states.

11.5.1 Identification of Top Events

During the progression of a HCDA, the same functions of interest in the core-damage event tree are also of interest in defining the sequence of events during the core-melt phase of the accident. These are basically neutronic shutdown, the presence of sodium coolant, and the ability to remove heat. One or more of these functions may not exist for a particular core-damage sequence being examined. All three can affect whether the core-melt process is subsequently halted, or if continuation of the melt process causes an energetic or relatively benign failure of the reactor vessel. If the potential exists for prompt critical conditions, the potential for energetic disruption, the degree of the energetic disruption, and the resulting effects on the reactor vessel are of particular interest. Finally, whether or not the reactor cavity cell liner fails affects the potential for sodium-concrete interactions, the production of hydrogen and other non-condensibles, and the resulting containment failure potential.

The core-response event tree is illustrated in Figure 11.5-1 using these physical events as the bases for selecting the events depicted on the tree. Each event is defined below:

- Permanent Non-Energetic Termination - Success for this event implies that a state of early, permanent neutronic shutdown has been achieved in a non-energetic manner, and maintained with no subsequent recriticalities. For protected accidents this translates into the requirement that recriticalities be ruled out following core melt. Success for the unprotected accident implies that the reactor has achieved a state of permanent subcriticality through fuel removal. For initiators where the

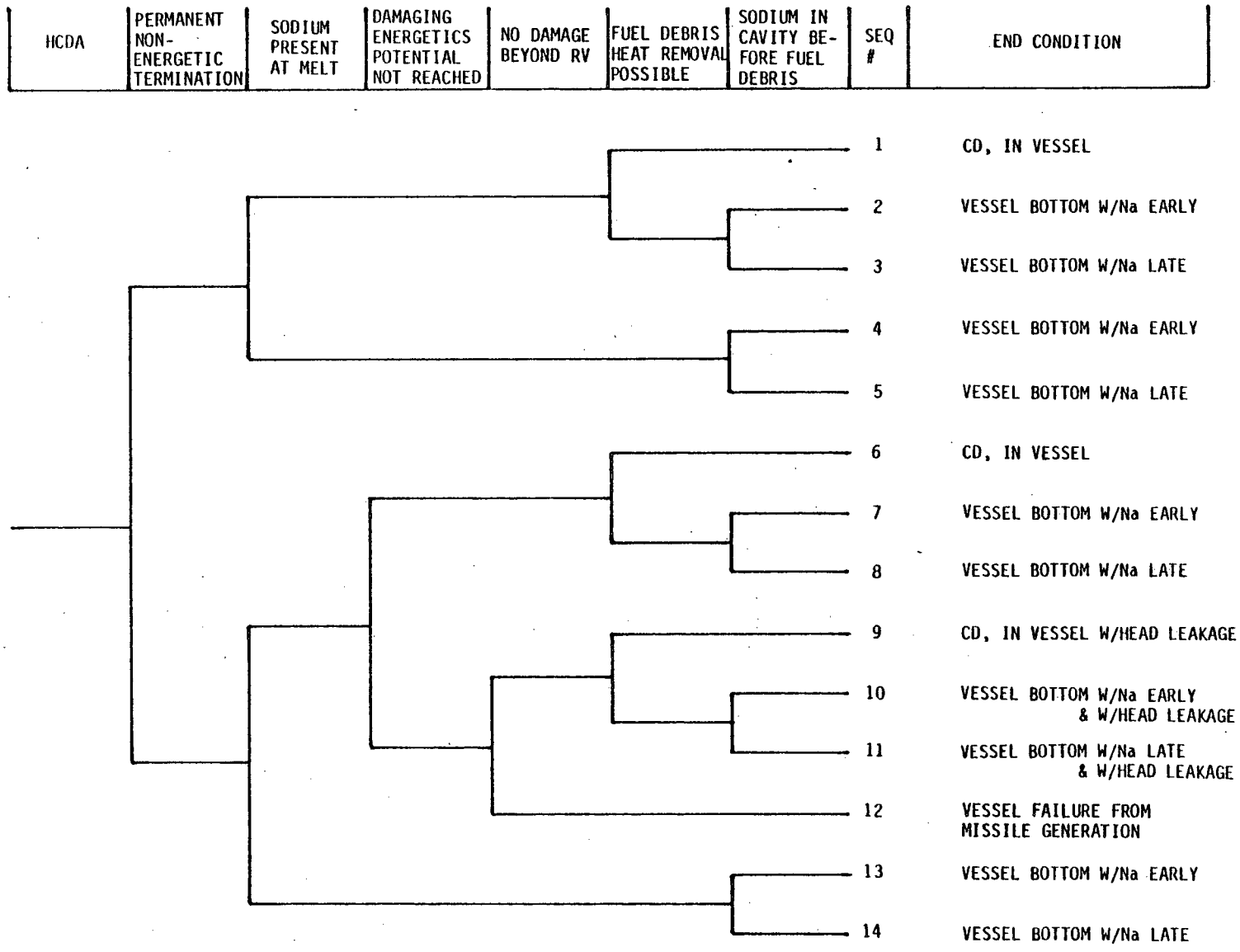

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Figure 11.5-1. Core-Response Event Tree.

main PHTS pumps are running, such as the UTOP, the dominant mode of fuel removal is sweepout. For initiators that involve pump trip and subsequent flow coastdown, fuel could potentially be removed by dispersion. However, the success path requires that the dispersion process is not driven by prompt critical core conditions; otherwise the potential for reactor vessel head damage must be considered.

- Sodium Present at Melt - Success for this event implies that sodium is above and around the core at the time of fuel melt or fuel disruption.

  For a protected accident, severe core damage is not expected as long as sodium is present above the core. This implies that sodium has to be lost either by evaporation, or drained out of the vessel by some kind of reactor vessel failure (due to vessel interactions, creep, etc.) before core melt can occur. Energetics is only a concern for sequences where recriticalities cannot be ruled out ("permanent non-energetic termination" criteria are not met) and sodium remains above the core as a medium for transferring energy to the reactor vessel head. Thus, for an unprotected HCDA, the chances of vessel damage are increased since sodium will very likely be above the core at the time of fuel disruption.

- Damaging Energetics Potential Not Reached - Success for this event implies that energetic fuel disruption either does not occur or if it does, potentially damaging levels are not attained so as to pose a threat to the integrity of the reactor vessel or primary system.

  This event addresses those energetics that can potentially challenge the reactor vessel integrity. It is estimated that no damaging energetic release is possible in the absence of sodium above the reactor core.

- No Damage Beyond RV - The success path for this event implies that a potentially damaging engergetic event has resulted in damage that is limited to the reactor vessel head. For example, head seal damage may occur resulting in a significant release of sodium to the upper containment.

  The failure path for this event implies that head failure produces a missile that is capable of breaching the reactor containment boundary.

- Fuel Debris Heat Removal Possible - Success for this event implies that the core has a coolable geometry even after some degree of core melt and that the heat removal required to prevent the core from melting any further is available.

If the success criteria are met the debris within the reactor vessel will remain coolable for an indefinite period of time. It is assumed here (where applicable) that there are enough heat sinks (e.g., DHRS, SGAHRS) available to cool the core following termination of the accident (because the heat sink is recovered or it was never lost). In the absence of sodium, this event is bypassed, since no cooling medium is available to transport heat to the available heat sinks.

- Sodium in Cavity Before Core Debris - Success for this event implies that sodium is present in the reactor cavity before significant amounts of core debris can come in contact with the cavity floor.

  Presence of sodium in the reactor cavity can affect the potential for a sodium-concrete reaction by cooling the debris bed, therefore lessening the generation of non-condensible gases. It also may affect the possibility of containment liner failure because an ablative attack, in the absence of sodium, will increase the failure likelihood of the liner.

11.5.2 Sequence Description and Outcomes

Based on the understanding of core disruption phenomena developed by other information sources and summarized in Section 11.4, the core-response scenarios are depicted on the core-response event tree using the event names identified in the previous subsection. The event tree is constructed as a general model for examining all HCDAs. The progression of each HCDA will be examined and discussed in Section 11.6. That discussion will include the probabilities associated with each sequence on the core-response event tree for each class of HCDA examined.

The scenarios depicted on the core-response event tree represent possible sequences of events during the progression of core damage ending in seven categories of end states. These end states range from a partially damaged core with no primary system failure (i.e., a mitigated sequence) to a failure of the reactor vessel head with subsequent potential for direct containment failure due to missile generation.

Each HCDA initiator is examined for its potential to achieve a state of non-energetic permanent neutronic shutdown and whether sodium is present at the time of core disruption. The energetics potential and degree of energetics are considered only for those sequences in which permanent non-energetic neutronic shutdown is not achieved and sufficient sodium is present as a medium for transferring energy to the reactor vessel head above the core. This is somewhat in contrast to the Phase I report[4] which considered a mechanical challenge to the reactor vessel in connection with core disruptive accidents to be an incredible event. This is not to say that the probability of reaching a damaging energetics potential is high, but Reference 3 does consider potentially damaging energetics, particularly for unprotected loss of flow events. The possibility of damaging energetics has therefore been retained in the core-response event tree for the conditions noted above to provide a means for more closely examining the probabilities of vessel damage.

Whether or not sufficient fuel debris heat removal is possible (analogous to the debris bed coolability event in the Phase I report) is examined for any sequence in which the sodium coolant is present and where damaging energetics have not occurred which would severely damage the reactor vessel (thus precluding the possibility of heat removal). Of course, certain HCDA initiators will also preclude heat removal and thus the "down" branch for this event would be followed.

Finally the event associated with either sodium or core debris first in the reactor cavity is analogous to the primary system failure and reactor cavity cooling discussions in the Phase I report. For those scenarios

where reactor vessel bottom failure is considered the dominant mode of vessel failure, this event is examined since it can have an impact on the potential for containment failure and subsequent release to the environment.

There are seven end states for the core-response event tree. These end states represent possible different challenges to the containment and fission product release potentials since the location and energetic potential of the release can differ with each end state. The end states and their applicable sequences are summarized below (see Figure 11.5-1).

End State	Sequences	Comments
Core Damage in Vessel	1,6	Partial core damage only, primary system intact.
Vessel Bottom with Na Early	2,4,7,13	Core melt proceeds to vessel bottom failure. Sodium is likely to enter the reactor cavity first, possibly affecting the potential for reactor cavity liner failure.
Vessel Bottom with Na Late	3,5,8,14	Same as above except core debris enters the reactor cavity before sodium, allowing for a greater chance of liner failure.
Core Damage with Head Leakage	9	Partial core damage only, with vessel head leakage.
Vessel Bottom with Na Early and Head Leakage	10	Core melt proceeds to vessel bottom failure with the addition of head leakage to the upper containment. Sodium is available in the reactor cavity at the time the core debris falls into the cavity.
Vessel Bottom with Na Late and Head Leakage	11	As above except the core debris falls first allowing for a greater chance of liner failure.
Catastrophic Vessel Failure	12	Severe damage to the vessel head results in potential threat to containment integrity from missile generation.

11.6 CORE-RESPONSE EVENT TREE QUANTIFICATION

11.6.1 Introduction to Method of Quantifying the Event Tree

The quantification of the core-response event tree was performed subjectively following review of analyses and literature that were deemed helpful in delineating the phemonena involved. Of particular help were the CRBRP PRA Phase I core phenomenology analysis (Reference 4) and NUREG CR-3224, "An Assessment of CRBR Core Disruptive Accident Energetics" by T. G. Theofanous and C. R. Bell (Reference 3). The details of a particular phenomena were addressed only as necessary to assign a best-estimate, qualitative probability (e.g., very unlikely) to each branch point of the event tree. Qualitative probabilities were then translated into quantitative probabilities as follows:

Unlikely	= 0.1	Likely	= 0.9
Very Unlikely	= 0.01	Very Likely	= 0.99
Extremely Unlikely	= 0.001	Extremely Likely	= 0.999
Non-mechanistic	= 1×10^{-4} and lower		

It must be recognized that there is considerable subjectivity involved in the assignment of these quantitative probabilities. In each case we have attempted to substantiate each assignment recognizing the present understanding of core-melt phenomena. As stated previously, the assignments were intended to be best estimates, but often little information was available for quantifying qualitative judgements. In these latter cases, an attempt was made to lean toward conservative probabilistic assignments. This was done so that potentially important core responses were not prematurely rejected before the total sequence probabilities, including chance of release to the environment, are calculated and then re-examined as necessary.

11.6.2 Permanent Non-Energetic Termination

This event was defined in Section 12.5.1 for both unprotected and protected HCDA initiators. Recall that for event success, the reactor must achieve a permanent subcritical state without first going prompt critical and the potential for recriticalities must be ruled out. Following is a more detailed discussion of how this event was applied to specific HCDA initiators and the rationale used in the quantification process.

11.6.2.1 Unprotected HCDA's

ULOF. As discussed in Section 11.4.2.1.1 the initiation of ULOF is characterized by a failure to scram in conjunction with a loss of primary flow and subsequent heatup of the core. In a little more than 10 seconds after initiation of pump coastdown, sodium boiling is initiated throughout the core, especially in the high power-to-flow channels. The fast Doppler and slower fuel axial expansion negative reactivity feedback effects are overcome by the positive reactivity feedback effects of sodium voiding in the central high power channels. (Driver assemblies around the perimeter of the core have negative sodium void coefficients due to neutron leakage.)

Clad melting and downward relocation begins about 5 seconds after sodium voiding and introduces even more positive reactivity into the core. Steel blockages are expected to form at the axial ends of the core before any fuel disruption occurs. In the Reference 6 best-estimate case of BOC-1 (low burnup, relatively low sodium void coefficient) fuel slumping and drainage will result in a subprompt power burst. Fuel (now

a vapor) dispersal in conjunction with the previously mentioned negative reactivity feedback effects are expected to render the core subcritical.

However, in the case of the end of the fourth fuel cycle (EOC-4), which has higher burnup and a relatively higher sodium void coefficient, a moderate superprompt power burst followed by rapid fuel dispersal is predicted. Again, the negative reactivity associated with fuel dispersal is expected to render the core subcritical as it enters the meltout phase.

The conclusions reached by Theofanous and Bell[3] regarding the core response of CRBRP to the ULOF initiating phase are in general agreement with the conclusions reached by the CRBRP Project.[6] The trends identified in their independent assessment were very similar, and in some cases identical, to those identified by the CRBRP Project. The trends include substantial neutronic activity dominated by gravity-driven oscillatory fuel motions that lead to power bursts and a reduced time interval between clad melting and fuel disruption. This latter trend, called "co-disruption" has the effect of minimizing the extent of fuel/steel separation and core exit blockage formation and thus enhancing termination due to fuel dispersion.[3]

In summary, it appears that neutronic shutdown during the initiating phase of an ULOF will occur only following sufficient fuel vapor dispersal. The co-disruption trend favors fuel dispersion due to the fact that clad blockages will have less time to form prior to fuel disruption. However, it appears that for many cases, the fuel dispersion

requires superprompt critical or near superprompt critical conditions. Further, based on the available literature, recriticalities are difficult to rule out in the meltout and potential large scale pool phases of the ULOF accident.[3,4,6]

On this basis, permanent, non-energetic termination of the ULOF in the initiating phase appears unlikely since superprompt critical conditions and/or recriticalities in subsequent phases are difficult to rule out. With some potential conservatism, the probabilities for this event are therefore assigned as shown below.

Assigned probabilities:	Yes	0.1
	No	0.9

UTOP. The permanent termination potential in the initiating phase of a UTOP event is affected by several different factors. To a great extent, the ramp rate of the initiator is a major factor affecting the potential for autocatalytic behavior. For small ramp rates, (i.e., <20¢/sec) autocatalytic behavior, and hence energetics, can be ruled out all together.[3] A failure modes and effects analysis performed on the reactor control system concluded that it would require a total of seven subsystem failures to result in a control rod bank withdrawal of up to 9 inches per minute (maximum mechanical speed) and six subsystem failures to result in a single rod withdrawal of up to 9 inches per minute. The respective reactivity ramp rates associated with these withdrawals are 12.6¢/sec and 2.1¢/sec, respectively. It was further concluded that ramp rates of 10-12¢/sec are more than three orders of magnitude more

likely than those of 15-20¢/sec.[12] Therefore, ramp rates that are large enough to cause autocatalytic behavior can reasonably be ruled out.

As previously discussed in Section 11.5, early fuel removal via sweepout, is a very important phenomenon during the UTOP initiating phase that tends toward neutronic shutdown. Unlike the ULOF, the positive reactivity additions from sodium voiding, clad relocation and fuel compaction are not of major importance in the UTOP event. The Reference 6 best-estimate analysis for the UTOP indicated that the reactor would not progress into the meltout phase but instead achieve a state of permanent subcriticality. This conclusion is also consistent with the conclusions reached by Theofanous and Bell.[3]

In summary, success of the permanent termination event seems likely for the UTOP due to expectations of early fuel sweepout and absence of early autocatalytic behavior. However due to uncertainties associated with the amount of fuel sweepout, a high probability of success cannot be assigned for this initiator. On this basis the following probabilities were assigned.

Assigned probabilities:	Yes	0.9
	No	0.1

<u>ULHS</u>. Due to its perceived low frequency, the ULHS event has not been addressed to any great extent by the LMFBR technical community.[3] As explained in Section 11.4.2.1.1, the unprotected loss of heat sink accident is characterized by failure to trip the reactor on loss of main feedwater so that the heat sink is lost without an adequate reduction in

reactor power. Although the primary system will heat up rapidly following a loss of heat sink, the primary pump flow (the main primary pumps are assumed to be running for this event) will provide a means for fuel sweepout upon core disruption. Thus, the core response to the ULHS event would exhibit some characteristics of both the ULOF (rapid coolant boiling relative to the UTOP) and the UTOP (potential for fuel sweepout). The potential for meeting the success criteria of the permanent termination event is related to the relative strength of positive reactivity effects, as seen in the ULOF (e.g., sodium voiding, cladding melt and relocation) versus the relative strength of the negative reactivity effects, as seen in the UTOP (fuel sweepout, axial fuel expansion). Without conclusive evidence as to which effects dominate in the case of ULHS events, the selected probabilities for success/failure of permanent neutronic termination are between those already discussed.

Assigned probabilities:	Yes	0.5
	No	0.5

ULOS. The ULOS branch-point probabilities for neutronic shutdown are assumed to be identical to the probabilities for the ULOF. This is due to the fact that, in the long term, the events are very similar. In the ULOS event the core begins to overheat when the primary system heat sink is lost following a loss of siphon in the PHTS loops. As in the case of the ULOF, the primary pumps are not assumed to run upon loss of siphon in the PHTS loops. Thus, from the perspective of neutronic shutdown, the events are treated as if they were identical.

Assigned probabilities:	Yes	0.1
	No	0.9

11.6.2.2 Protected HCDA's

LHSE. The phenemena associated with protected loss of heat sink accidents was discussed in considerable detail in Section 11.4.2.1. As described in that section, the time frame of even an early protected loss of heat sink accident is long relative to that of an unprotected HCDA. The cladding and hexcan wall material is expected to melt and flow downward into the lower axial blanket region, since the steel melting temperature (~1425°C) is much lower than the melting temperature of the control rod material (~2350°C) and the fuel (~2825°C). The liquid steel, once inside the available gaps, will freeze and form a steel plug. This plug is expected to extend upward a minor distance (0.2m) into the lower portion of the core.[3] Many of the columns of fuel rod and control rod pellets are expected to remain temporarily intact due to sintering during power operation. However, toppling or gradual melting of the columns will eventually occur, possibly resulting in mild criticalities. The lower melting point of the control material (B_4C) relative to the fuel may cause the control material to disrupt, melt and drain earlier than the fuel. Control rod material escape from the core region seems unlikely, however, because of the presence of the lower steel plug and the large margin to B_4C sublimation (boiling point > 3500°C). The recriticality potential following meltdown of the fuel material may be offset somewhat by the ingress of radial and axial blanket material into the pool. However, separation or stratification of the control rod material from the fuel may occur due to differences in specific weight (2.5 for B_4C versus ~9 for the fuel).[3] As discussed

in Section 11.4.2.1, Theofanous and Bell predict that a mild recriticality will be achieved sooner or later; if not by toppling of the unclad fuel and control pellets, then by gradual melting and draining.[3] They further predict that upon reaching criticality the rate of fuel collapse and melt would be accelerated.

In summary, the reactor core is subcritical by definition upon initiation of an early protected loss of heat sink accident. However, due to the slow heatup of the core and the subsequent material motions, the reactor is predicted to become recritical. On this basis it is deemed unlikely that the neutronic termination success criteria can be met.

Assigned probabilities:	Yes	0.1
	No	0.9

LHSL. Although this event would occur on a greater time scale than the previous one, it would not likely have a substantial effect on the long-term potential for permanent neutronic termination.

On this basis the LHSL event cannot justifiably be assigned a success probability different than that for the LHSE event.

Assigned probabilities:	Yes	0.1
	No	0.9

LOS. The melting sequence and the probability of attaining permanent neutronic termination for the LOS event is considered similar to the PLHS events. The primary reason the events are considered to have similar melting sequences is because the LOS event becomes a PLHS event when

the sodium level in the reactor vessel falls below the overflow nozzles and siphon is broken in the PHTS loops. The extent to which the LOS resembles the LHSE or the LHSL event is a function of the leak rate. In either case the probability of meeting the permanent termination success criteria is the same.

Assigned probabilities:	Yes	0.1
	No	0.9

11.6.3 Sodium Present at Melt

11.6.3.1 Unprotected HCDA's

ULOF. As previously explained in Section 11.6.2.1, sodium boiling is expected to occur about 10 seconds following initiation of the ULOF event. Since, for the ULOF event, the reactor would be at near nominal power level, the boiling would be unstable and clad dryout would occur soon after boiling inception. As a result, the subassembly coolant channels will undergo extensive voiding (liquid sodium will be expelled out of both ends of the coolant channel) and dryout while the core remains surrounded by liquid sodium. However, it is very unlikely that the entire region of sodium above the core could undergo extensive voiding prior to core disruption. Thus, the liquid sodium above the core will remain available as a medium for transferring energy to the reactor vessel head in the event of an energetic disassembly of the core. The values assigned to the probability of sodium presence at the time of fuel melt are shown below.

Assigned probabilities:	Yes	0.99
	No	0.01

UTOP. In the UTOP event, sodium boiling near the coolant channel exits begins several tens of seconds later than it does in the ULOF event.[6] The boiling begins in the highest power to flow channels and eventually progresses into other fuel coolant channels. Although the increased pressure drop associated with sodium boiling in the channel exits reduces sodium flow, single-phase liquid sodium will remain in the core region. As the power in the core rises, the boiling interface will slowly move upstream. Clad failure due to high fuel vapor pressure is expected soon after boiling inception (<5 seconds).[6] Again, sodium surrounding the core will very likely remain available as a medium for transferring energy to the primary system boundary in the event of an energetic disassembly of the core.

Assigned probabilities:	Yes	0.99
	No	0.01

ULHS. The sodium boiling phenomena for the ULHS event would be similar to the boiling phenomena for UTOP, if no pump trip is assumed, and similar to the ULOF boiling phenomena if the pumps are assumed to be tripped. As in the case of the ULOF and the UTOP, it is very unlikely that the large mass of sodium above or around the core could be vaporized in the small time frame associated with the core disruptive accident.

Assigned probabilities:	Yes	0.99
	No:	0.01

ULOS. In the ULOS event, core overheating will not occur until the sodium level within the reactor vessel falls below the minimum safe

level and siphon is broken in all three PHTS loops. In the event of an energetic disassembly, the volume of sodium above the core that could act as a medium for transferring energy to the reactor vessel head could range from about one third* to none of the original above-core volume, depending on the timing of energetics and the leak rate of the sodium. Based on available literature, it appears that smaller leak rates have a considerably greater frequency than larger leak rates.[17,18] This fact, in conjunction with the fact that upon a loss of siphon in the PHTS loops core disruption would proceed on a time scale similar to a ULOF (tens of seconds), suggests that in most cases the volume of sodium above the core at the time of fuel disruption would correspond to about one third of the original above-core volume rather than zero. On this basis, the probability that sodium will be above the core at the time of core disruption is deemed only somewhat less likely for the ULOS relative to other unprotected HCDA's.

Assigned probabilities:	Yes	0.9
	No	0.1

11.6.3.2 Protected HCDA's

LHSE. Following initiation of an early protected loss of heat sink accident the primary system temperature will slowly begin to rise. Although natural circulation through the PHTS loops is assumed to be lost, since all active heat sinks are not available, density differences within the reactor vessel will allow for some convective flow patterns

*The one third above core volume was obtained by estimating the volume of sodium below the outlet nozzle centerline and above the active core.

within the reactor core. If adiabatic conditions are assumed (zero heat transfer from the primary coolant boundary) the decay heat would go entirely into heating of the primary sodium and primary system structures.

The response of the SNR-300 [a 300 MW(e) LMFBR] to a postulated loss of all active heat sinks was assessed by Voseebrecker and Kellner.[13] The assessment, which included treatment of temperature-dependent insulation heat losses from the primary system and considered the effect of natural convection within the reactor core, concluded that boiling would never occur. The peak sodium temperature in the core, that was reached at about 70 hours following initiation of the accident, corresponded to about 1400°F (more than 200° subcooled). The eventual drop in temperature was attributed primarily to the increased radiation heat transfer through the insulation material at higher temperatures. On this basis the assumption of adiabatic heat transfer from the primary system boundary appears conservative.

However, Bari performed an equivalent study for CRBRP and concluded that the heat losses through the insulation would be negligible.[13] It was further concluded that bulk sodium boiling would occur approximately 3 to 4 hours into the accident and that sustained dryout would follow, leading to meltdown of the fuel. Thus, in Bari's assessment, core melt was assumed to occur while the core was still surrounded by sodium.

Fauske, in the CRBRP Phase I report concluded that extended channel dryout could not occur as long as sodium remained above the core.[4] It

was predicted, based on analytical and experimental evidence, that sodium would re-enter a coolant channel almost immediately following channel dryout.

In order to prevent the re-entry of sodium into the coolant channels upon reaching a channel dryout condition, the gravitational pressure drop gradient associated with the liquid sodium must not exceed the vapor velocity and its associated frictional pressure drop gradient. This requirement can be translated into a minimum heat flux that must be exceeded to prevent sodium re-entry.[4] Fauske has calculated this minimum heat flux using approximation methods for conditions of low flow convection with varying degrees of inlet subcooling. As shown in Table 11.6-1, the values obtained using various techniques are in excellent agreement.

Also included in the table are heat flux limits calculated by Khatib-Rahbar and Cazzoli for a CRBRP full assembly.[15] Khatib-Rahbar and Cazzoli used equations similar to the Kutateladze re-entry criterion proposed by Dunn and modified by Fauske and Ishii. Although the dryout heat flux values are lower than those predicted by Fauske, they are still above the expected heat flux immediately following shutdown and well above the heat flux expected when the sodium saturation temperature is reached (by nearly an order of magnitude).[3] In addition to calculating the dryout heat fluxes, Khatib-Rahbar and Cazzoli have calculated the excursion heat fluxes. The excursion heat flux corresponds to the heat flux, for a given condition, where boiling becomes unstable leading to film dryout.

Table 11.6-1

COMPARISON OF DRYOUT HEAT FLUX CALCULATIONS FOR A CRBRP FUEL ASSEMBLY

Tin (°K)	Fauske CHF[13]	Dryout Heat Flux Fauske Re-Entry[13,4]	Khatib-Rahbar[15] Numerical	Excursion Heat Flux Khatib-Rahbar[15] Excursion
673	~ 9 W/cm^2	~ 10 W/cm^2	7.30 W/cm^2	25.7 W/cm^2
873	Not Calculated	Not Calculated	6.15 W/cm^2	15.3 W/cm^2
1073	~ 7.8 W/cm^2	~ 8.7 W/cm^2	~ 5.35 W/cm^2	5.8 W/cm^2

Notice from the table that as the inlet temperature increases, as would be expected in a loss of heat sink accident, the dryout heat flux approaches the excursion heat flux. The impact of inlet temperature (degree of sub-cooling) is further delineated in Figures 11.6-1 through 11.6-3. As the inlet temperature is increased, the interval between dryout heat flux (re-entry) and excursion heat flux is reduced. On this basis, flow excursions that lead to dryout at or near saturated inlet conditions cannot be ruled out.

When boiling within the coolant channels becomes unstable (from a quasi-stable state) the voids within the coolant channels expand under high pressure, driving liquid sodium out of the coolant channel. Although the coolant expulsion will result in a large reactivity increase, it is not expected to result in a reactor recriticality because the maximum void worth (~ 4$ at EOC-4) is sufficiently below the shutdown worths of the primary (~ 6$) and the secondary (~ 3.4$ with one stuck rod)* reactor shutdown systems (at EOC-4).

Any vapor bubble, that has expanded from within a coolant channel to outside of the core, will begin to condense and collapse. Once the vapor-generated pressures begin to decay, sodium may re-enter the coolant channels. However, this re-entry is expected to result in "chugging" - a phenomena where the vapor bubble continues to grow and collapse.[3] There is still some question as to whether or not the time intervals between re-entry and bubble growth would be insufficent to prevent core coolability.

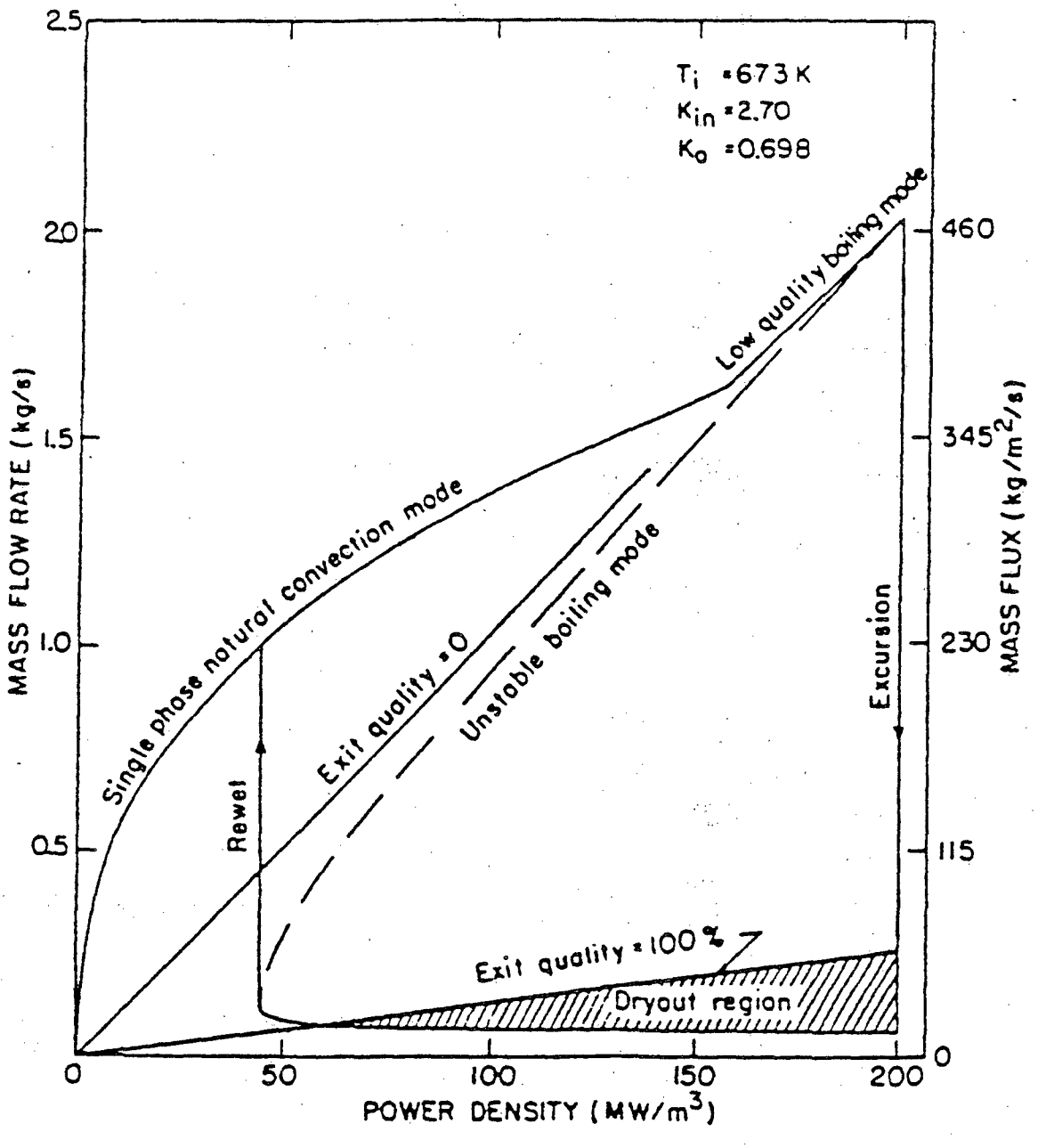


Figure 11.6-1. Boiling Map for CRBR Fuel Assembly at Inlet Temperature of 673K. (Reference 15)

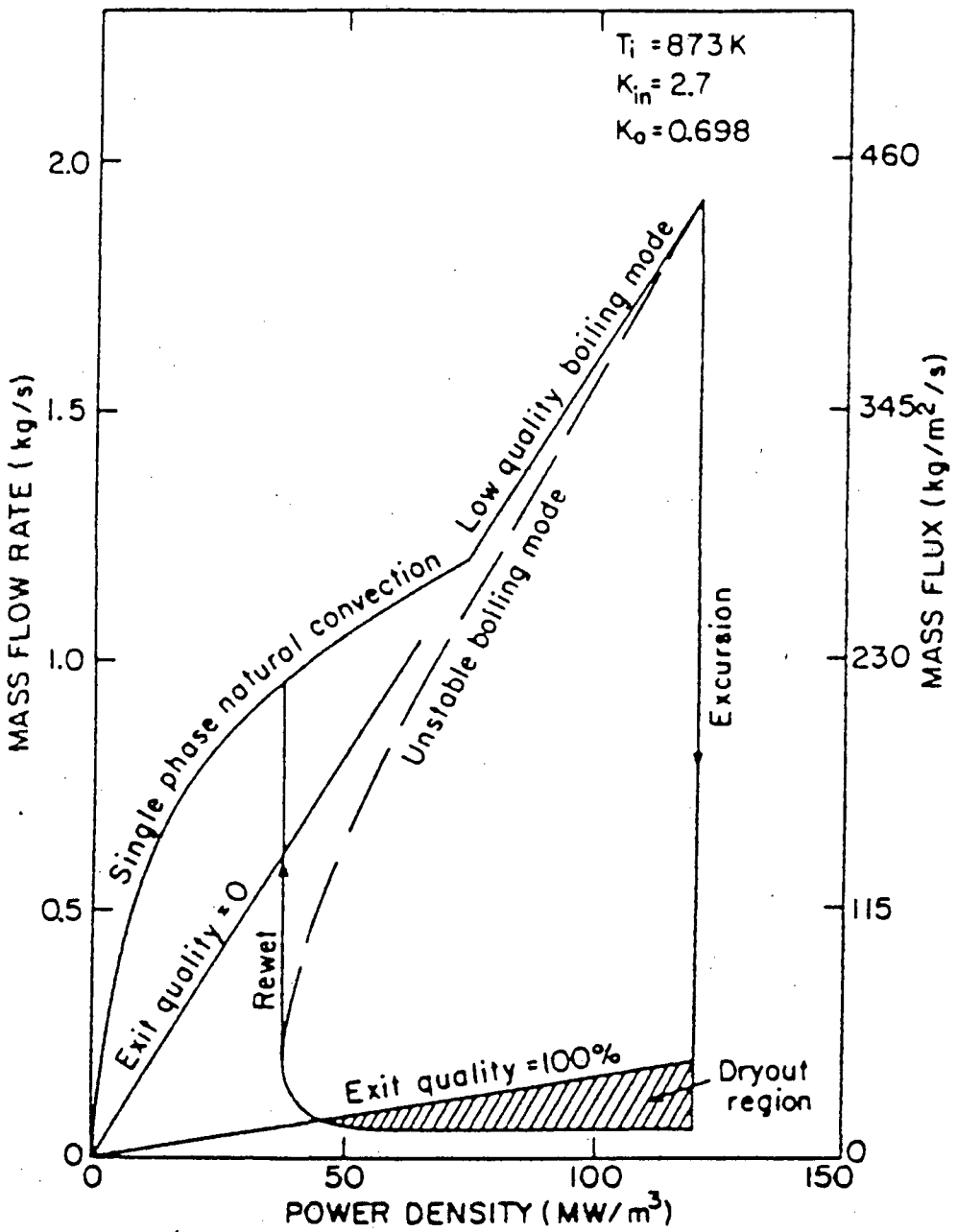


Figure 11.6-2. Boiling Map for CRBR Fuel Assembly at Inlet Temperature of 873K. (Reference 15)

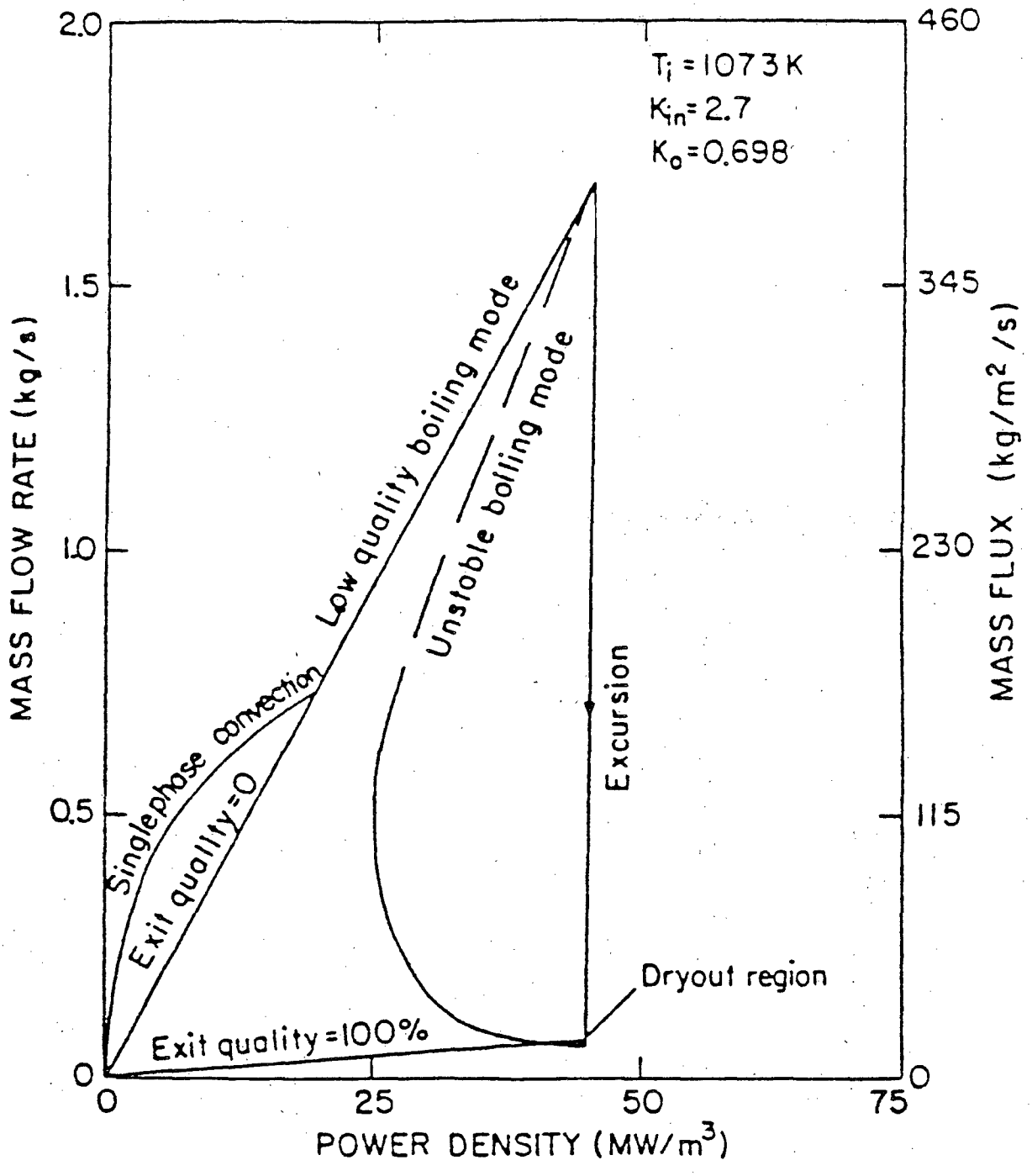


Figure 11.6-3. Boiling Map for CRBR Fuel Assembly at Inlet Temperature of 1073K. (Reference 15)

Indirect experimental evidence tends to support the re-entry phenomena discussed above. Perhaps the most pertinent experimental support comes from the Sodium Boiling Test Facility (SBTF) experiments performed at Oak Ridge National Laboratory (ORNL). The SBTF experiments indicate that the coolability limits are not highly dependent on the degree of subcooling.[3] However, due to limitations in the test apparatus, the SBTF experiments have not addressed conditions of inlet saturation. Thus, due to lack of experimental evidence of sodium re-entry at zero inlet subcooling, some degree of uncertainty must be attributed to the re-entry phenomena discussed above.

Based on the above discussion, it was deemed unlikely that core melt could occur while sodium was still present above the core. However, in light of the uncertainties associated with the re-entry phenomena, especially those related to saturated inlet conditions, a greater probability cannot be justifiably assigned to the downward path of the "Sodium Present at Melt" mode.

Assigned probabilities:	Yes	0.01
	No	0.99

LHSL. The primary difference between this event and the previous event is the decreased heat flux levels. Given a much lower heat flux level, the probability of exceeding the dryout or excursion heat flux becomes much lower (see Section 11.6.4.5).

Assigned probabilities:	Yes	0.001
	No	0.999

LOS. In order for the LOS event to cause core damage, the sodium level in the reactor vessel must fall below the minimum safe level causing a loss of siphon in all three PHTS loops. Until that time, decay heat would be removed from the core at a near-normal rate. Following a loss of siphon in the loops, the LOS event would resemble a PLHS event with lower sodium temperatures. Whether the decay heat power level associated with the LOS would be more characteristic of the LHSE or the LHSL event would depend on the leak rate and on any action the operator might take to makeup sodium to the reactor vessel. However, it is deemed extremely unlikely that the duration of the loss of heat sink (whether early or late) prior to the time that the core is uncovered would be long enough to lead to fuel disruption conditions (re-entry phenomena aside).

Assigned probabilities:	Yes	0.001
	No	0.999

11.6.4 Damaging Energetics Potential Not Reached

In order to determine the relative probability of damaging energetics for a given initiator, the HCDA progression path was divided into the three phases; the initiating phase, the meltout/annular pool phase and the large-scale pool phase. A best estimate assessment of the energetics potential within each of these phases was made for each of the HCDA initiators. The probability of damaging energetics (energy levels greater than the structural margin of the primary coolant boundary) was then determined for each phase of the event by multiplying the probability of damage to the primary coolant boundary, given energetics in a particular phase, by the probability that the phase is entered. By

summing the probabilities of damaging energetics over all three phases of a given event, a total probability of damaging energetics was determined. The notation used is shown below.

$P(E_I)$ = The probability of energetics in the initiating phase

where $P(D/E_I)$ = The probability of energetics beyond the primary coolant boundary structural margin (damaging energetics) given energetics in the initiating phase

so that $P(E_I')$ = The probability of damaging energetics in the initiating phase = $P(E_I) \cdot P(D/E_I)$

and $P(E_T')$ = The total probability of damaging energetics for the event

$= P(E_I') \cdot P(_I) + P(E_M') \cdot P(_M) + P(E_L') \cdot P(_L)$

where $P(_I)$ = The probability of entering the initiating phase

$P(_M)$ = The probability of entering the meltout phase

$P(_L)$ = The probability of entering the large-scale pool phase

$P(E_M')$ = The probability of damaging energetics in the meltout phase

$P(E_L')$ = The probability of damaging energetics in the large-scale pool phase

The above approach was taken because the probability for damaging energetics is very dependent on HCDA phase, which in turn depends on the initiator. Due to the complexities involved in HCDA sequences, assigning a single value for the damaging energetics potential, without considering the contribution from each applicable phase, would be inappropriate. The approach used here not only provides a convenient framework for estimating the total damaging energetics potential of a

given initiator, but also allows a relative comparison of damaging energetics potential.

11.6.4.1 ULOF

Initiating phase. In Reference 6, an extensive analysis was performed on the ULOF event using the SAS-3D code. Variations of the base case were analyzed including very pessimistic assumptions relative to the best estimate. The variations also included degree of fuel collapse and reduced fuel vapor pressure with and without fuel axial expansion and nominal clad worth. Based on the analysis, no hydrodynamic disassembly was predicted for the initiating phase of the ULOF event.

In an independent assessment, Theofanous and Bell identified two major areas of concern that could potentially lead to ULOF initiating phase energetics in LMFBR's.[3] The first concern was the LOF-driven-TOP (LOF-d-TOP). As explained in Section 11.4.2.1.1, the LOF-d-TOP occurs when the positive sodium void reactivity during the ULOF event results in a high overpower condition much like the TOP. The LOF-d-TOP can cause pin failures in unvoided assemblies while surrounding higher power-to-flow channels are voided. If the failures were to occur at the core midplane, rapid motion of the fuel toward the failure location could result in autocatalytic conditions. Theofanous concluded that because of the considerably lower sodium void coefficient associated with the heterogeneous reactor core, (as opposed to the earlier homogeneous core design) the LOF-d-TOP condition is avoided in the CRBRP.

The second concern was plenum fission gas-induced fuel compaction. This phenomena was previously discussed in Section 11.4.2.1.1. After the reactor has been operating continuously for a relatively long period of time, the fission gas buildup within the plenum causes considerable plenum pressurization. Upon fuel disruption within the core region the plenum fission gas pressure, if unrestricted, could drive blanket and undisrupted driver assemblies downward toward the failure location.

Under this type of condition, energetic behavior would be difficult to rule out. The CRBRP Project analyzed this mechanism and concluded that there would be adequate time for fission gas to escape prior to fuel disruption.[5,14] However, Theofanous and Bell did not agree with these conclusions[3] and recommended that steps be taken to limit the action of fission gas pressure during the initiating phase of the ULOF. As a result, the project committed to a minor design modification of the plenum spacer that would substantially reduce fission gas blowdown following fuel disruption.[16]

A realistic upper bound on the ramp rate associated with fuel compaction has been placed at $50/sec. Since it was concluded that ramp rates near $200/sec were required to challenge the CRBRP vessel head integrity,[3] it seems unlikely that the previously discussed fuel compaction phenomena could pose a challenge to the primary coolant boundary integrity.

In summary, since the two phenomena Theofanous and Bell identified that could lead to autocatalytic behavior have been avoided by virture of the

CRBRP design, it is deemed very unlikely that an energetic event could occur in the initiating phase of the ULOF. Furthermore, if either of the two phenomena did occur, they would not likely challenge the reactor vessel head integrity.

The probability of energetics beyond the structural margin of the reactor vessel (and in particular the reactor vessel head) during the initiating phase of the ULOF accident can be described as follows:

$$P(E_I') = P(E_I) \cdot P(D/E_I)$$

where $P(E_I')$ = The probability of damaging energetics in the initiating phase beyond the primary coolant boundary structural margin.

$P(E_I)$ = Probability of energetics in the initiating phase.

$P(D/E_I)$ = Probability of damage beyond the primary coolant boundary structural margin given energetics in the initiating phase.

Based on the previous discussion, the following values are assigned:

$P(E_I)$ = 0.01 (very unlikely)

$P(D/E_I)$ = 0.1 (unlikely)

Thus $P(E_I')$ = 0.001 for the initiating phase of the ULOF event.

<u>Meltout/Annular Pool Phase</u>. Upon termination of the ULOF initiating phase, the reactor core may be subcritical by several dollars due to upward (above the core) fuel dispersal. The ejected fuel will begin to ablate the cladding and the hexcans will begin to rupture and melt, allowing liquid fuel to boil and flow into gaps between hexcans. Fuel penetration into empty control assembly ducts following melt-through is

also likely. The Reference 6 best-estimate assessment of the meltout phase concluded that "escape paths" for fuel removal would develop within 10-20 seconds and that permanent subcriticality conditions could be achieved. In addition, the potential for fuel penetration (removal) during the meltout phase of the ULOF event has been supported by a number of experiments.[19]

Although fuel removal during the meltout is predicted to render the core subcritical, the potential for recriticalities does exist. Fauske (Phase I of the CRBRP PRA)[4] concluded that the potential for recriticality occurrence due to gravity-driven fuel compaction is difficult to rule out in the meltout phase. Fauske points out that the fuel compaction is limited by vapor separation and cannot introduce reactivity ramp rates exceeding 20¢/s/assembly. Ramp rates on this order would be mitigated by core-wide incoherencies in the early (meltout) phase.

Theofanous and Bell also recognize that core-wide coherence is unlikely during the meltout phase due to the fact that the time interval prior to subassembly wall disintegration permits only a few oscillations on the axial fuel mass distribution. These oscillations, caused by the high heat loss (and unstable) environment of the subassembly pool, will result in local power pulses. When the dispersive fuel vapor pressure subsides, the fuel will relocate (or collapse) under the influence of gravity. This fuel collapsing phenomena is depicted in Figure 11.6.4 using the classical Bethe-Tait Model. However, the Bethe-Tait Model is highly conservative and inaccurate for the situation described. As shown in Figure 11.6.-5, breakup of the accelerating fuel slug is

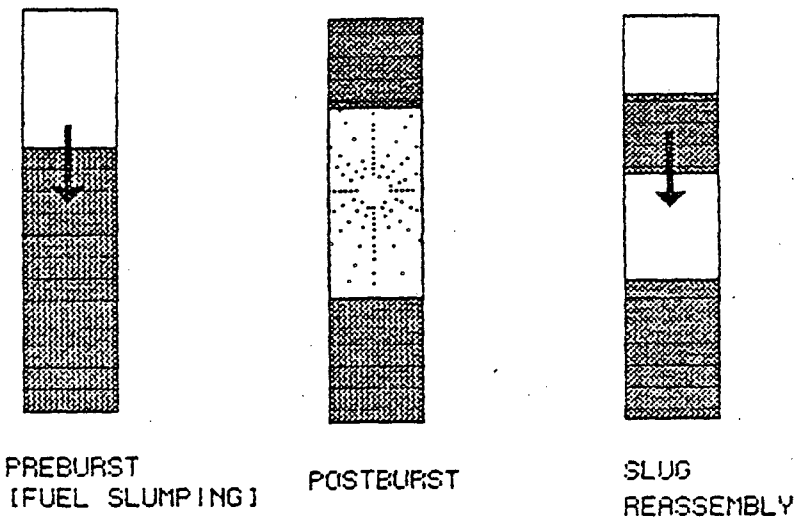


Figure 11.6-4. Bethe-Tait Model of Fuel Reassembly.

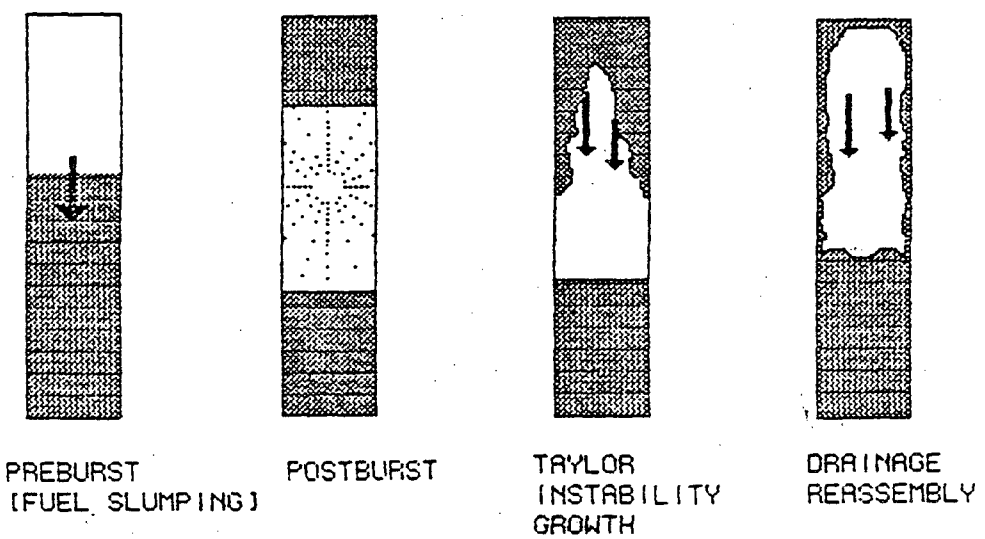


Figure 11.6-5. Model of Fuel Reassembly with Taylor Instability Induced Break-up. (Reference 3)

expected to occur as the result of a physical process known as the Rayleigh-Taylor instability. Rayleigh-Taylor instabilities occur at the interface of a lighter fluid (fuel vapor) and a heavier fluid (molten fuel) when the fluid acceleration is in the direction of the heavier fluid. Based on the above breakup process, Theofanous and Bell conclude that the ramp rates associated with gravity-driven recriticalities are small.[3]

On this basis it can be stated that a recriticality event is self-dispersive and that a reccurrence or amplification does not appear likely because of vapor/liquid-breakup processes resulting from the growth of Rayleigh-Taylor instabilities.

It has been estimated that if core-wide incoherences are ignored, ramp rates associated with gravity-driven recriticality would be limited to 30-100$/sec.[3]

The potential for fuel compaction driven by fuel-coolant thermal interactions (FCI) is ruled out on the basis of the inherent inability of fuel to trap sodium and the unattainable requirement for fine scale fragmentation (of fuel) and intermixing (of fuel and sodium).[3]

In summary, although fuel removal early in the meltout phase may lead to permanent subcriticality, gravity-driven recriticalities cannot be ruled out. If such recriticalities occurred, they would be limited to small ramp rates due primarily to incoherencies and vapor/liquid-breakup processes. Even if the incoherency effect is removed, the ramp rates would not be large enough to challenge the reactor vessel head integrity. Fuel compaction due to FCI can reasonably be ruled out.

The probability of energetics beyond the structural margin of the reactor vessel during the meltout/annular pool phase of the ULOF accident is delineated in the same manner that it was for the initiating phase of the ULOF. The relevant probabilities for the meltout phase are as follows:

$$P(E_M) = 0.01 \text{ (very unlikely)}$$

$$P(D/E_M) = 0.1 \text{ (unlikely)}$$

Thus $P(E_M) = 0.001$ for the meltout phase of the ULOF event

Large-Scale Pool Phase. The large-scale pool (LSP) phase would be entered in the event that permanent fuel removal in earlier phases were not to occur. In the CRBRP PRA Phase I report,[4] Fauske concludes that, "no credible mechanisms have been discovered that can lead to prevention of sufficient fuel removal in connection with the subassembly disruption (meltout) phase to necessitate further consideration into the whole-core pool phase." Although this conclusion is consistent with the Reference 6 analysis, the energetic potential of the LSP phase was analyzed. The open system (no upper blockages) was expected to remain subcritical due to the dispersive nature of the boiling pool in conjunction with earlier losses in the meltout and initiating phases. In the closed, or bottled-up configuration, system fuel crust formation at the boundaries would limit heat losses causing the pool to vigorously boil. Blowdown of the pool upon a pool opening (caused by blockage melting) would result in substantial fuel removal. Further melting of steel and blanket material would have the effect of diluting the pool, which would result in a permanent neutronic shutdown.

High reactivity ramp rates (~100$/sec) were observed by Theofanous and Bell in their assessment of the energetics potential of the LSP phase. The energetic regime however, was only associated with idealized perfectly symmetric (geometry and power distribution) and entirely homogeneous pools.[3] However, the energetics level did not exceed the structural margin of the primary system boundary.

In summary the formation of an entirely homogeneous large-scale pool in the ULOF event is deemed very unlikely. The energetics potential associated with the large-scale pool is likely below that which would lead to a loss of primary system boundary integrity.

The probabilities assigned to LSP energetics and loss of primary system integrity are:

$P(E_L) = 0.1$ (Unlikely)

$P(D/E_L) = 0.1$ (Unlikely)

$P(E_L') = 0.01$

<u>Total ULOF Damaging Energetics Potential</u>. The probability of energetics beyond the primary system structural margin has been estimated for the initiating phase, the meltout phase and the large-scale pool phase of the ULOF event. Based on previous discussion, the probability that the ULOF progresses into a particular phase can be reasonably estimated as follows:

P(I) = Probability that the ULOF enters into the initiating phase
= 1.0

P(M) = Probability that the ULOF enters into the meltout phase
= 0.99

P(L) = Probability that the ULOF enters into the large-scale pool
= 0.01

Thus, the total probability that the ULOF event will result in damaging energetics is:

$$P(E_T') = P(E_I') \cdot P(I) + P(E_M') \cdot P(M) + P(E_L') \cdot P(L)$$

$$P(E_T') = (0.001)(1) + (0.001)(0.99) + (0.01)(0.01) = 2 \times 10^{-3}$$

Assigned probabilities:	Yes	0.998
	No	2×10^{-3}

11.6.4.2 UTOP

Initiating Phase. The phenomena related to the neutronic shutdown in the initiating phase of the UTOP event was briefly described in Section 11.6.2.1. As explained in that section, non-energetic, permanent neutronic shutdown in the initiating phase is the expected path of the UTOP event. The CRBRP Project performed an analysis of the UTOP energetics potential during the initiating phase using the SAS-3D code.[6] Variations from the base case (4.1¢/sec insertion rate) that were analyzed included pessimistic assumptions such as; complete flow blockage, arbitrary midplane failure with pessimistic Doppler and material worth at 50¢/s, extreme blockage and limited fuel removal, etc. Only one case, (arbitrary midplane failure at 10¢/s with SAS/FCI calculations) produced an energetic event. The energetics associated with this event was well below the structural margin of the primary coolant boundary. This same case, when analyzed using a more mechanistic method of calculation (PLUTO2 instead of SAS/FCI) produced zero energetics. The assessment of the UTOP by Theofanous and Bell focused on defining the margins for autocatalytic behavior for assumed midplane failures.[3]

Autocatalytic behavior depends to a large extent on the relative timing of fuel motions (pellet movement inside pins upon midplane failure) versus external fuel motions (sweepout).

The timing depends on the degree of core-wide coherence. The core-wide coherence in turn, depends on the core configuration and the reactivity ramp rate of the initiating event. As the ramp rate increases, coherence also increases.

Theofanous and Bell performed an analysis of the UTOP event for the most coherent core configuration (EOC-3) using the SAS4A computer code with PLUTO2.[3] The analysis indicated that the time between failures was on the order of 300 ms for the first six groups of assemblies. The PLUTO2 sweepout calculation, which was adjusted to the TREAT L8 experimental data, showed that negative reactivity effects associated with sweepout, cancelled out the positive reactivity effects associated primarily with internal fuel motion. Shutdown was predicted within 100 ms following sweepout.

As mentioned in Section 11.6.2.1 in their analysis of the UTOP, Theofanous and Bell assumed initiating ramp rates on the order of 10-12¢/sec. Ramp rates in excess of 10-12¢/sec., although extremely unlikely, will increase the extent of core-wide coherence and thus increase the likelihood of energetics.

Based on the above discussion, the following probabilities have been assigned for the initiating phase of the UTOP.

$P(E_I) = 0.001$ (extremely unlikely)

$P(D/E_I) = 0.1$ (unlikely)

$P(E_I') = 1 \times 10^{-4}$

Meltout/Annular Pool Phase. Although the UTOP event is not expected to progress into the meltout phase, such a progression can be postulated if particular pessimistic assumptions are made. For example, it could be postulated that instead of achieving a state of permanent subcriticality, the UTOP may exit the initiating phase at a stable power level beyond the heat removal capability of the heat transport system. This UTOP event may resemble a ULOF event upon a loss of flow following primary system boundary failure. However, because of the greater propensity for further fuel dispersal (fewer steel blockages) the consequences would be bounded by the traditional ULOF event.

There is a general consensus within the LMFBR community that the potential for energetics for unprotected initiators is bounded by the ULOF event.[3,4,5,6,13] In addition, once an unprotected HCDA progresses into the meltout or large scale pool phase, there appears to be diminished sensitivity to the initiator. Thus, it seems reasonable to consider that the probabilities assigned to the ULOF energetics potential, for the meltout and large-scale pool phases, would represent an upper bound for other initiators.

Therefore, the probabilities assigned to the damaging energetics potential of the UTOP event during the meltout phase are the same value applied to the ULOF event.

$P(E_M') = 0.001$ for the meltout phase of the UTOP event

Large-Scale Pool Phase. Based on the discussion above, the probability assigned to the damaging energetics potential of the large-scale pool phase is the same value applied to the ULOF event.

$P(E_L') = 0.01$

Total UTOP Damaging Energetics Potential. The probability that the UTOP progresses into a particular phase is estimated below based on previous discussion.

P(I) = Probability that the UTOP enters into the initiating phase = 1.0

P(M) = Probability that the UTOP enters into the meltout phase = 0.1

P(L) = Probability that the UTOP enters into the large-scale pool phase = 0.01

Thus, the total probability that the UTOP event will result in damaging energetics is:

$$P(E_T') = P(E_I') \cdot P(I) + P(E_M') \cdot P(M) + P(E_L') \cdot P(L)$$

$$= (1 \times 10^{-4})(1.0) + (0.001)(0.1) + (0.01)(0.01)$$

$$= 3 \times 10^{-4}$$

Assigned probabilities:	Yes	0.9997
	No	3×10^{-4}

11.6.4.3 ULHS.

Initiating phase. Detailed investigations into the energetics potential of the ULHS event have not been performed, primarily because of the perceived low probability associated with the event. However, the energetics potential of the ULOF event is expected to bound the energetics potential of the ULHS event due to the fact that there is a much greater

propensity for fuel removal via sweepout for the ULHS event. It is conservatively assumed that, despite the favorable propensity for fuel sweepout, the probability for energetics is identical to that of the ULOF event.

Since there would likely be sodium above the core in the event of a hydrodynamic disassembly (see Section 11.6.3.1), the probability of reactor vessel damage beyond the structural margin would be no greater, and probably less, than that of the ULOF event.

$P(E_I) = 0.01$ (very unlikely)

$P(D/E_I) = 0.1$ (unlikely)

$P(E_I') = 0.001$ for the initiating phase of the ULHS event

Meltout/Annular Pool Phase. Based on the discussion for the UTOP meltout phase, the probabilities assigned to the damaging energetics potential of the ULHS event during the meltout phase are the same values applied to the ULOF event.

$P(E_M') = 0.001$ for the meltout phase of the ULHS event.

Large-Scale Pool Phase. Based on the discussion for the UTOP large-scale pool phase the following probability was assigned.

$P(E_L') = 0.01$ for the large-scale pool phase of the ULHS event

Total ULHS Damaging Energetics Potential. The only significant difference between the ULOF event and the ULHS event that is expected to have an effect on the potential for damaging energetics, is the probability of entering the meltout phase. While the ULHS exhibits fuel removal characteristics similar to the UTOP, it also exhibits characteristics

similar to the ULOF such as rapid coolant boiling, early clad failure (relative to fuel), potential blockages, etc. Rather than equating the probability of entering the meltout phase for the ULHS event to the corresponding probability for the UTOP or the ULOF, a median value of 0.5 is used.

$$P(E_T') = (0.001)(1) + (0.001)(0.5) + (0.01)(0.01)$$
$$= 0.0016$$

Assigned probabilities

Yes: 0.9984

No: 0.0016

11.6.4.4 ULOS.

Initiating Phase. In Section 11.6.3.1 it was explained that core overheating would not occur unless the sodium level in the reactor vessel fell below the PHTS outlet nozzles causing a loss of siphon in all three PHTS loops. In the event that siphon was lost in all three loops, the core would begin to heat up and the scenario would closely resemble the ULHS (pumps assumed to be tripped) or more generally the ULOF. For this reason, it was deemed appropriate to set the probability of energetics for the ULOS event equal to that of the ULOF.

In the event of an energetic disassembly of the reactor, the volume of sodium above the core may range from zero to approximately one third of the original above-core volume (See Section 11.6.3.1). The effect of above-core sodium volume on the probability of damage to the reactor vessel head given energetics P(D/E) was investigated.

No Sodium Above the Core. A major assumption in the core-response model is that sodium must be present above the core as a medium for transferring energy in order for reactor vessel head damage to occur. For this case the P(D/E) is considered to be essentially zero.

One-Third of Above Core Volume of Sodium Above the Core. When sodium is present above the core upon a hydrodynamic disassembly, the expanding high temperature and pressure fuel vapor bubble will drive the "sodium slug" upward until it impacts with the reactor vessel head. The partition of energy within the reactor vessel depends on the relative strengths of the energy release, core barrel, reactor vessel walls, vessel head, etc. If the core barrel and reactor vessel walls are assumed to be non-rigid, they will undergo considerable plastic deformation and absorb a significant amount of energy before the sodium slug is accelerated. However, if the reactor vessel walls and core barrel are rigid, they will not be capable of absorbing as much energy and more energy will be imparted on the reactor vessel head upon impact with the sodium slug (gun barrel effect). Based on CRBRP-3, Volume 1, Hypothetical Core Disruptive Accident Considerations in CRBRP,[2] only about 22% of the total work energy goes into strain energy of the reactor vessel walls and core barrel. More than 75% of the total work energy goes into upward kinetic energy of the sodium slug. Therefore, for purposes of this study, the reactor vessel was considered rigid and elastic energy and cover gas compression was neglected. The latter is a conservative assumption because gas compression would tend to "cushion" the impact of a sodium slug. Because the upper internal structure (UIS)

was not modeled in this study, mitigating effects associated with the UIS were ignored.

The UIS was also not included in the REXCO-HEP model used by the CRBRP Project to assess the reactor vessel structural response to an HCDA. However, the effect of the UIS in attenuating the sodium slug kinetic energy has been experimentally investigated in a series of CRBRP scale model tests performed at SRI International. The UIS was found to reduce the slug kinetic energy at impact by about a factor of two. In addition, Lacko[20] developed a simplified model for determining the reduction in the total work energy of an expanding gas bubble for a large (1000 MW) LMFBR. Since the simplified model does not account for the kinetic energy lost to turbulence in the upper plenum produced by the jetting action of the coolant through the UIS, all of the work energy released by the expanding bubble goes into accelerating the sodium slug. The energy reduction due to the presence of the UIS was calculated with the simplified model using input parameters comparable to the third CRBRP scale model test (SM-3) input parameters. Relatively close agreement was shown to exist between the energy reductions predicted by the simplified model, the SRI scale model tests and the more detailed SIMMER-II analytical model. In general, the UIS is expected to reduce the slug kinetic energy at impact by about 50%.[22]

The details of a simplified calculation to calculate coolant slug kinetic energy are shown in Appendix B1. The results indicate that the kinetic energy of the smaller slug (~ 300 MJ) would be significantly greater than the kinetic energy of the larger slug (~ 100 MJ).

The percentage of the sodium slug kinetic energy that is expended in performing work on the RV head is determined in Appendix B2. The results show that only slightly more kinetic energy must be absorbed by the reactor vessel head for the small sodium slug ($\sim$ 15 MJ) than for the larger sodium slug ($\sim$ 13 MJ). The partioning of the remaining slug kinetic energy is illustrated in Figure 11.6-6. Based on this simplified assessment, the probability of damage to the reactor vessel head given energetics in the ULOS event is deemed identical to that for the ULOF event.

Thus, for the initiating phase of the ULOS, the following probabilities have been assigned:

$$P(E_I) = 0.01$$

$$P(D/E_I) = 0.1$$

$$P(E_I) = 1 \times 10^{-3}$$

Meltout/Annular Pool Phase. Since the ULOS phenomena upon core disruption is very similar to the ULOF phenomena, the probability assigned to energetics during the ULOS meltout phase is the same as the probability assigned to energetics in the meltout phase of the ULOF.

$$P(E_M) = 0.01$$

$$P(D/E_M) = 0.1$$

$$P(E_M) = 1 \times 10^{-3}$$

Large-Scale Pool Phase. The arguments made in the last section concerning the resemblance of the ULOS to the ULOF still remain valid.

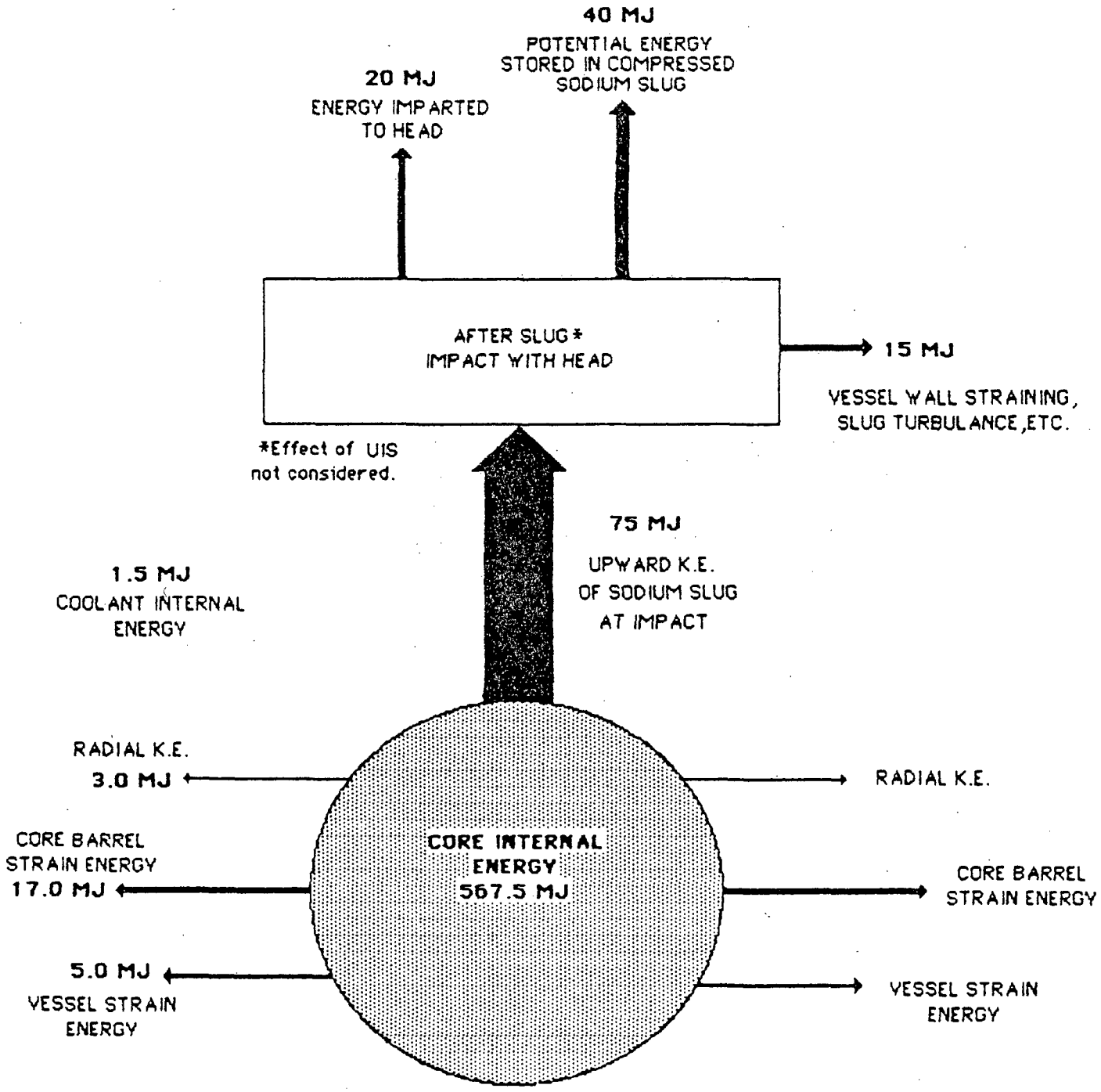


Figure 11.6-6

Energy Partitions Following a Hydrodynamic Disassembly of the Core.

(Reference 2 - Structural Margin Case)

Also, in Section 11.6.4.2 it was suggested that upon entering the meltout or large-scale pool phases the core phenomena is insensitive to the "type" of HCDA initiator.

Although it is recognized that there would be less of a tendency to have some fraction of sodium above the core pool in the LSP phase (due to longer time scale for leak and sodium boil off), it will be conservatively assumed that some sodium will remain above the core as a medium for transmitting energy to the reactor vessel head.

$P(E_L) = 0.1$

$P(D/E_L) = 0.1$

$P(E_L') = 0.01$

<u>Total ULOS Damaging Energetics Potential</u>. The probability that the ULOS progresses into a particular phase is estimated to be similar to the corresponding probabilities associated with the ULOF event

P(I) = Probability that the ULOS enters the initiating phase = 1.0

P(M) = Probability that the ULOS enters the meltout phase = 0.99

P(L) = Probability that the ULOS enters the large-scale pool = 0.01

The total probability that the ULOS event will result in damaging energetics is:

$$P(E_T') = P(E_I') \cdot P(I) + P(E_M') \cdot P(M) + P(E_L') \cdot P(L)$$

$$= (0.001)\ (1.0) + (0.001)\ (0.99) + (0.01)\ (0.01)$$

Assigned probabilities:	Yes	0.998
	No	2×10^{-3}

11.6.4.5 LHSE. A major assumption made in this study is that there is no potential for energetics for a protected HCDA without sodium above the core as a medium for transferring energy. Thus, in order for the "damaging energetics potential not reached" event to apply to the LHSE event, the assumption is made that sodium is still present above the core at the time of fuel disruption. This translates into a requirement that sodium re-entry (discussed in greater detail in Section 11.6.3.2) either does not exist or is insufficient to prevent channel dryout.

Due to the characteristics of protected loss of heat sink accidents, (i.e., initiation with melting of steel and fuel) the meltout phase is entered directly, bypassing an initiation phase.

Meltout/Annular Pool Phase. As explained in Section 11.6.2.2, recriticalitics cannot be ruled out in the meltout phase of the LHSE accident. Recriticalitics would occur due to fuel settling following melting or toppling of the fuel pellets. If a critical condition is obtained, the fuel will melt faster and could potentially lead to ramp rates on the order of 60$/sec.[3] Large ramp rates (i.e., $\geq \sim$ \$30/sec) will lead to energetic fuel dispersal followed by permanent neutronic shutdown. Recent studies predict that ramp rates $\geq$ \$30/sec would occur about 50% of the time.[23] Smaller ramp rates will probably not lead to permanent fuel dispersal and the probability of enerting the large scale pool phase will be increased.

The ramp rates that are postulated do not appear to be sufficient to challenge the primary system integrity. However, the fact that (by

definition) sodium remains above the core will increase the likelihood that the primary system integrity will be compromised. Thus, based on the above discussion the following probabilities have been assigned.

$P(E_M) = 0.1$ (Unlikely)

$P(D/E_M) = 0.1$ (Unlikely)

$P(E_M) = 1 \times 10^{-2}$

Large-Scale Pool Phase. The large-scale pool phase of the LHSE event is exemplified by the formation of a stable, boiled-up and subcritical pool configuration. The formation of the pool would take a considerable amount of time given the low heat flux conditions. Since the entire core, including the blanket region, will have heated-up at a relatively even rate, freezing and plugging of fuel is deemed unlikely. Thus, a bottled-up pool is considered unlikely. Nonetheless, the energetics potential resulting from a large-scale, bottled-up pool is considered below.

A bottled-up pool could potentially result in rapid fuel re-entry and pool collapse. Fauske concluded in his assessment of the LHSE, that the special conditions required for a sufficiently energetic fuel-coolant interaction suggest that a pressure-driven fuel compaction event is unlikely even with sodium in the upper plenum. His conclusions were based on numerous prototypic experiments and well established physical principles.[13]

The large-scale pool is expected to achieve a state of permanent fuel removal by eventual fuel dispersal (via boiling) or permanent subcriticality by dilution with blanket material.

Since the LHSE large-scale pool is not expected to be bottled-up, the probability of energetics is expected to be low.

$P(E_L) = 0.01$ (Very Unlikely)

$P(D/E_L) = 0.1$ (Unlikely)

$P(E_L) = 1 \times 10^{-3}$

Total LHSE Damaging Energetics Potential. The probability that the LHSE event progresses into a particular phase was briefly discussed in this section. Due to the extended time frame associated with the LHSE event, the potential for entering the LSP phase is considered to be much greater than it is for unprotected accidents resulting in the following assessment.

$P(I) = N/A$

$P(M) = 1.0$

$P(L) = 0.5$

$$P(E_T) = P(M) \cdot P(E_M) + P(L) \cdot P(E_L)$$
$$= (1.0)(1 \times 10^{-2}) + (0.5)(1 \times 10^{-3})$$
$$= \sim 1 \times 10^{-2}$$

Assigned probabilities: Yes 0.99

No 0.01

11.6.4.6 LHSL. The LHSL event is expected to reach the initial conditions of the LHSE event at a much later point in time (on the order of tens of hours). However, from the perspective of energetics potential, the difference between the two events is insignificant.

Assigned probabilities: Yes 0.99

No 0.01

11.6.4.7 LOS. As explained in earlier sections, the LOS event becomes a LHSE or a LHSL event, depending on the leak rate, upon a loss of siphon in all three PHTS loops. Thus, the same probabilities are used.

Assigned probabilities: Yes 0.99

No 0.01

11.6.5 No Damage Beyond Reactor Vessel

The above node was defined in Section 11.5.1. Instead of evaluating the probability of the above event for each HCDA initiator, a general evaluation was performed to determine whether or not the failure path (failure of the RV head with subsequent potential for direct containment failure due to missile generation) was mechanistic.

Based on the simplified analysis performed in Appendix B3, catastrophic vessel failure due to HCDA energetics-generated missile is non-mechanistic. It should be noted that the simplified analysis was extremely conservative, especially in neglecting the substantial structural margin provided by the CRBRP reactor vessel head to accommodate energetic events on the order of 200$/sec.[3]

It should be mentioned that whether the CRBRP structural margin requirements for the reactor vessel head (to be capable of accommodating a sodium slug with 75MJ of upward axial kinetic energy) are met, depends on the elimination of a premature failure mode. Following a series of

scale model hydrostatic and dynamic tests (SM Series) it was discovered that failure of the reactor vessel head was occurring due to kinematic disengagement of the plugs rather than material strain.[14] The disengagement was caused following closure of the gap between the intermediate rotating slug (IRP) and the large rotating plug (LRP). Closure of the gap caused the plugs to pry apart laterally, leading to disengagement. Since the CRBRP Project had committed to modify the RV head design to eliminate this kinematic failure mode, it will be assumed that structural margin to accommodate energetics is achievable.

In summary, it was concluded that catastrophic containment vessel failure resulting from a HCDA-generated missile is non-mechanistic.

Assigned probabilities:	Yes	~ 1
	No	$<10^{-4}$

11.6.6 Fuel Debris Heat Removal Possible

The fuel debris heat removal potential is dependent on the in-vessel conditions following HCDA termination, the distribution of core debris within the primary system, and the ability of structures to contain core debris.

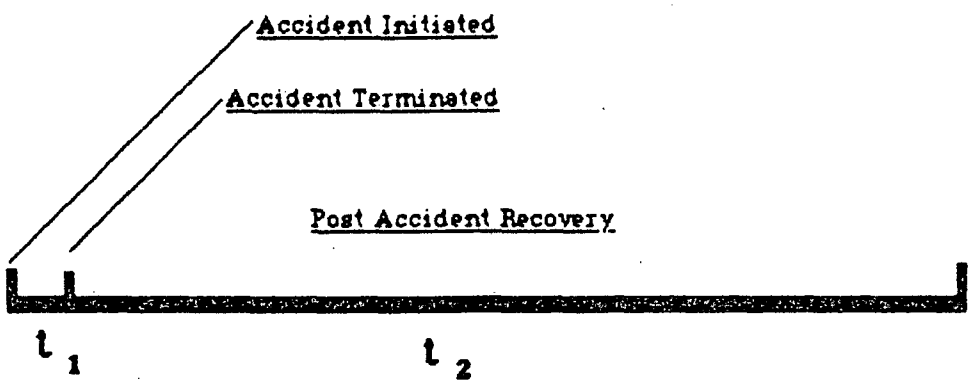
Attempts to re-establish decay heat removal paths (i.e., DHRS, SGAHRS) following termination of the accident were considered in assigning probabilities of success to the "Fuel Debris Heat Removal Possible" event. The chance that the decay heat removal systems will be available was considered to be very good for unprotected HCDA's that meet the "Permanent Non-Energetic Termination" success criteria.

As shown in Figure 11.6-7, upon initiation of an unprotected HCDA there would be only tens of seconds for any recovery actions to be taken. However, if the primary coolant boundary integrity is maintained following termination of the HCDA and PHTS loop flow is possible, then the decay heat removal systems could be utilized to remove decay heat from the core debris. The potential for fuel debris coolability would then depend on the core configuration (i.e., potential for large debris beds, etc.).

Protected HCDAs on the other hand occur on a much longer time scale. (See Figure 11.6-7.) Upon initiation of a protected HCDA there would be many hours to days to recover the decay heat removal systems prior to fuel disruption. If attempts at recovering the decay heat removal systems had been unsuccessful up to the time of fuel disruption, the chances of recovering those systems shortly thereafter is estimated to be very small. Thus, for all practical purposes, the probability that a heat removal path is established following a protected HCDA is considered to be small. Although, this assumption is somewhat conservative, it is considered to be reasonable.

11.6.6.1 Unprotected HCDAs

ULOF. For permanent neutronic termination success sequences, blockages would be expected both above and below the core. This is because the process of achieving permanent neutronic termination involves fuel and steel removal which leads to blockages above and below the core. In Reference 2 it was estimated that fuel would be distributed in roughly equal quantities above and below the active core region.



Timescale for Unprotected Accidents

Timescale for Protected Accidents

- For Unprotected Accidents: t_1 = Tens of Seconds

  t_2 = Hours

- For Protected Accidents: t_1 = Hours to Days

  t_2 = Hours

Figure 11.6-7. HCDA Progression Timescale.

Some fuel is expected to be retained by the shield/orifice blocks and inlet modules possibly causing them to melt. Debris beds that form primarily on horizontal surfaces could be cooled by internal sodium convection within the bed. If the debris bed is too deep or closely distributed or if the heat generation rate is too high, the bed will experience dryout, melt, and form a fuel/steel pool. Under these circumstances vessel meltthrough would be inevitable. Some in-pile experimental data suggest that bed dryout may not be a sufficient condition for melting.[2] On this basis it is entirely possible that the bed depth at which fuel melting would occur would be greater than bed depths associated with dryout. In Reference 2 debris beds up to 2-6 inches were considered to be stable and coolable.[2]

The potential for re-establishing a heat removal path following non-energetic accident termination is considered good for the ULOF event. Upon establishing pony motor flow, the degree of fuel particle settling and thus the propensity for forming a debris bed would be be significantly diminished. Under these conditions the fuel would be more likely to remain particulate and the fuel debris could remain coolable as long as the heat removal path(s) remain available.

In summary, for the non-energetic termination sequences, long-term coolability of the fuel debris would be possible if a heat removal path can be re-established. If such a heat removal path can not be established, a debris bed will likely form on horizontal surfaces. It is uncertain if the debris bed would remain coolable or not. The debris

bed coolability would depend on the debris bed depth which would depend on the amount of fuel removal necessary to terminate the accident.

In light of the above uncertainties and lacking any detailed investigation of the ability to provide long-term coolability if a heat removal path is re-established, the probability of event success was assigned a value of 0.5. This value not only reflects the uncertainties but also suggests that there is no dominant physical phenomena associated with this sequence that would tend to favor the success path over the failure path.

The potential for fuel debris heat removal is considered to be diminished for sequences that do not meet the permanent neutronic termination success criteria but do meet the "Damaging Energetics Potential Not Reached" criteria. This is not only due to the fact that "mild" energetics (less than that which could damage the primary system integrity) may occur within this sequence but also because the core conditions at termination will most likely correspond to a much less coolable configuration.

On this basis the probability of meeting the "Fuel Debris Heat Removal Possible" success criteria is unlikely.

For sequences involving damaging energetics the fuel would be expected to remain primarily in the upper plenum area. This is due to the difference in the strength of the above and below core structures. Since the lower-core structures are considerably stronger than the above-core structures, more damage would be sustained by the upper structure,

allowing for a greater path for removal of core materials. In Reference 2 it was estimated that as much as 90% of the fuel could be ejected upward. Fuel would eventually move downward through the shield/orifice blocks and inlet modules to the bottom head. The ability to re-establish a heat removal path is deemed much less likely for sequences that involve energetics due to the effects of energetics on the primary system boundary (i.e., potential pump failures, check valve closures, etc.) The large fraction of fuel in the upper plenum region may settle out initially as a debris bed. However, due to a large amount of fuel involved, the debris bed would not remain coolable. Any subsequent debris beds that may form as the core material moves downward would eventually become non-coolable. It is expected that bottom head failure will eventually occur.

In summary, the "fuel debris coolable" success sequences for "damaging energetic" events are extremely unlikely due to the probable inability to re-establish a heat removal path. Furthermore, even if a heat removal path is re-established, there is a large degree of uncertainty associated with the condition of the core and its ability to be cooled under the circumstances.

On this basis it was considered extremely unlikely that the fuel debris could be coolable for sequences that involve damaging energetics.

Assigned probabilities:

Permanent Neutronic Termination Success Sequences:	Yes	0.5
	No	0.5

Permanent Neutronic Termination Failure Sequences/No Damaging Energetics:

Yes	0.1
No	0.9

Permanent Neutronic Termination Failure Sequences/Damaging Energetics:

Yes	0.001
No	0.999

UTOP. For permanent neutronic termination success sequences, where the HCDA is expected to terminate following fuel sweepout, the fuel debris is expected to be retained or frozen on the above-core structures and throughout the PHTS piping. If fuel sweepout and subsequent neutronic termination occurs early, the coolability potential of the fuel is estimated to be good. This is especially true in light of the fact that chances of re-establishing a heat removal path are very good.

Due to the presence of continuous hydraulic forces, the settling (by gravity) of fuel debris into a bed is considered unlikely. Debris accumulation in elbows is considered credible. However, such accumulations are estimated to be coolable. Even if debris beds form, the amount of fuel removed from the core would be much less than for the ULOF event, and thus debris bed depths would be less likely to exceed coolable limits.

Particle levitation for the permanent neutronic termination failure UTOP sequences would be limited to the sequences where pony motors continue to run. It will be assumed here that damaging energetics sequences (sequences where the "Damaging Energetics Potential Not Reached" criteria is not met) will preclude forced convection of sodium. In the

absence of forced convection, debris beds will form. As in the case of the ULOF, debris beds that form following permanent neutronic termination failure sequences will not likely remain coolable. The propensity for fuel debris to remain coolable on horizontal surfaces within the upper internal structure will be reduced relative to the non-subcritical termination sequences.

Assigned probabilities:

Permanent Neutronic Termination Success Sequences:	Yes	0.9
	No	0.1
Permanent Neutronic Termination Failure Sequences/No Damaging Energetics (Assumed Flow):	Yes	0.1
	No	0.9
Permanent Neutronic Termination Failure Sequences/Damaging Energetics (Assumed No Flow):	Yes	0.001
	No	0.999

ULHS. For the forced convection conditions assumed, the probability of core coolability is set equal to that of the UTOP.

ULOS. For this event the debris coolability heavily depends on the presence of sodium which in turn depends on leak location. Fuel debris retention in the upper internal structure will be impossible due to the absence of sodium.

Fuel debris will eventually migrate downward into a sodium-filled pool. Debris beds are very credible for the ULOS due to the absence of flow

and a limited sodium environment. Since siphon is broken in all PHTS loops, there is little chance of establishing heat removal paths. In general the probability of meeting the success criteria was estimated to be extremely unlikely.

All sequences: Yes 0.001

No 0.999

11.6.6.2 Protected HCDAs

LHSE. Due to the large amount of sodium vaporization that occurs in the latter stages of the LHSE event it seems very unlikely that fuel debris can be kept coolable following disruption. If the mass of sodium above the core is boiled away (e.g., through the head seals) then significant heat can be removed from the fuel debris via radiation. The steel plug that is expected to form in the lower axial blanket region will temporarily impede the progression of a molten debris bed. The core support structure may fail under the severe thermal and mechanical loading conditions. Fuel debris may also be temporarily held up in the shield/orifice blocks and the lower inlet modules. Since the entire system is gradually heating up, saturated or near saturated sodium conditions in the lower plenum cannot be ruled out. If such conditions do exist, the debris in the lower plenum will experience dryout and begin to penetrate the reactor vessel bottom head.

In general, since a heat removal path will not be available, it was considered very unlikely that the success criteria could be met. However, if the permanent non-energetic termination success criteria is met, the

heat removal requirements would be much less than those of an unprotected accident.

Assigned probabilities:

Permanent Neutronic Termination Success Sequences:	Yes	0.01
	No	0.99
All Other Sequences:	Yes	0.001
	No	0.999

LHSL. The heat removal requirements of the LHSL event are estimated to be much less than the heat removal requirements of the LHSE event because of the significant amount of sensible heat following shutdown that has already been removed and because the decay heat power levels are very small. This significant reduction in decay heat removal requirements is estimated to increase the likelihood of coolability for the LHSL event. In some cases, just the increased radiation heat transfer from the system at higher temperatures could provide an adequate means of cooling. Heat removal through the head seals is also a means of cooling the system. However, the above only holds true for permanent neutronics termination success sequences. Otherwise, the phenomenological outcome of the LHSL event is similar to the LHSE phenomenological outcome.

Assigned probabilities:

Permanent Neutronic Termination Success Sequences:	Yes	0.1
	No	0.9

Permanent Neutronic Termination Failure Sequences: Yes 0.001

No 0.999

LOS. The coolability of the core debris depends on the presence of sodium which depends on the leak rate and location. However, any sodium that might be retained in the vessel will probably be at saturated or near saturated conditions and is not expected to provide cooling.

Assigned probabilities:

All sequences: Yes 0.001

No 0.999

11.6.7 Sodium In Cavity Before Fuel Debris

The logic used in quantifying this node is straightforward. For sequences where sodium is present at melt, there is no substantial evidence that supports whether fuel debris or sodium enters the reactor cavity first. It is expected that once the fuel debris (sodium) enters the reactor cavity, the sodium (fuel) will be "right behind". For the ULOS and LOS sequences where sodium was not present at melt, it was deemed extremely likely that sodium entered the reactor cavity first. Based on the previous discussion of the PLHS events, there is a likely chance that sodium will enter the reactor cavity first. The mechanism for removing sodium from above the core in the UTOP, ULOF AND ULHS sequences is not considered to involve a loss of primary sodium into the reactor cavity. Therefore, the UTOP, ULOF and ULHS sequences without sodium present at melt were treated the same way as the UTOP, ULOF and ULHS sequences with sodium present at melt.

Assigned probabilities:

For Sodium Present at Melt Sequences:	Yes	0.5	(for all HCDAs except the LOS and ULOS)
	No	0.5	
	Yes	0.999	(for LOS and ULOS)
	No	0.001	

For Sodium Not Present at Melt Sequences:

- LOS AND ULOS:	Yes	0.999
	No	0.001
- LHSE and LHSL:	Yes	0.9
	No	0.1
- UTOP, ULOF and ULHS:	Yes	0.5
	No	0.5

11.7 SUMMARY AND CONCLUSIONS

As explained in the beginning of this section, review of the core-damage phenomenology is an essential part of the determination of risk associated with the Clinch River Breeder Reactor. The interrelationships between the core-damage phenomenology and other PRA elements was discussed in Section 11.2. The correlation between the plant systems analyses and the consequence analysis is accomplished through the selection of a set of core-damage bins. The set of core-damage bins comprise a discrete representation of the full spectrum of possible core-damage sequences producing different phenomenological responses. Core-damage sequences in LMFBR's have traditionally been referred to as Hypothetical

Core Disruptive Accidents (HCDAs). Unlike core-damage accidents in light water reactors, LMFBR core-damage accidents or HCDAs can potentially yield core configurations of higher reactivity than the original core geometry.

HCDAs can be caused by events that lead to insufficient heat removal or substantially high heat generation. HCDAs can be further categorized based on whether or not the reactor shutdown system functions as required.

Initiators where the reactor is shutdown, but core heat removal is absent for an extended period of time are referred to as "protected" HCDAs. If core heat removal is absent due to an unavailability of the systems that provide the heat sink, the accident is referred to as a protected loss of heat sink (PLHS), with an early loss of heat sink described as a loss of heat sink early (LHSE) event. Likewise if core heat removal is absent due to a late unavailability of the systems that provide the heat sink, then the accident is referred to as a loss of heat sink late (LHSL) event. If core heat removal is absent due to a loss of sodium from the primary system the accident is referred to as a loss of sodium (LOS) event.

HCDA initiators where the reactor is not shutdown are referred to as "unprotected". The unprotected HCDAs that involve insufficient heat removal include the unprotected loss of flow (ULOF), the unprotected loss of heat sink (ULHS) and the unprotected loss of sodium (ULOS). The ULOF accident involves failure of the reactor shutdown system with

subsequent tripping of the main primary heat transport system pumps. The ULHS and ULOS correspond to the PLHS and the LOS events without shutdown.

An unprotected HCDA initiator that involves substantially high heat generation is the unprotected transient overpower (UTOP) accident. The UTOP accident involves a reactivity insertion in addition to a failure to shutdown the reactor. Due to their phenomenological responses, each of the above-mentioned HCDA initiators was placed in a separate core-damage bin.

A core-response event tree was developed in Section 11.5. The event tree was constructed to accomodate each HCDA initiator (or core-damage bin). In Section 11.6 the event tree modes were quantified for each HCDA initiator, based on applicable and available studies and on engineering judgement. For some HCDA initiators, literature was either non-applicable or unavailable. This is due, in part, to the fact that some of the HCDA initiators, namely the ULOS, the LOS, and the ULHS, have received limited analysis by the LMFBR technical community. The LHSE and LHSL events have been analyzed to some extent by individuals such as Fauske and Bari. The ULOF and UTOP accidents, on the other hand, have undergone a considerable amount of scrutiny for more than two decades. Perhaps one of the reasons that more attention has been given to unprotected accidents than to protected accidents is due to the fact that unprotected accidents occupy a time scale on the order of tens of seconds whereas protected accidents occupy a time scale ranging from tens of hours to days.

The quantified core-response event trees for each initiator are shown in Figures 11.7-1 through 11.7-7. It should be noted that the uncertainties associated with the non-traditional HCDA initiators (i.e., the LOS, ULOS and ULHS) are, in general, somewhat greater than for the traditional HCDA initiators (i.e. the ULOF and UTOP). In all instances the uncertainty is inclusive in the quantification of the top events. In many instances the core response to a non-traditional HCDA initiator was deemed similar to the core response of a traditional HCDA initiator. For example, in the presence of flow, a ULHS event will experience sweepout similar to the UTOP event. If the pumps are tripped it can reasonably be assumed that the core response to the ULHS is closer to that of the ULOF event.

In addition to a varying degree of uncertainty among initiators there exists a significant degree of uncertainty among event tree top events for a particular initiator.

For example, it is known with a fairly high degree of certainty that for the UTOP event, sodium will be present at the time of fuel disruption. However, the uncertainties associated with whether heat removal is possible for a particular ULOF sequence are much greater.

The results are summarized in Table 11.7-1. The table shows the probability that each type of core-damage accident (denoted by core-damage bins) ends in a certain core-response end state. The values are obtained by summing up the accident sequence probabilities for "like" end states from each event tree using two significant figures of precision. The probabilities are conditional on the probability of each

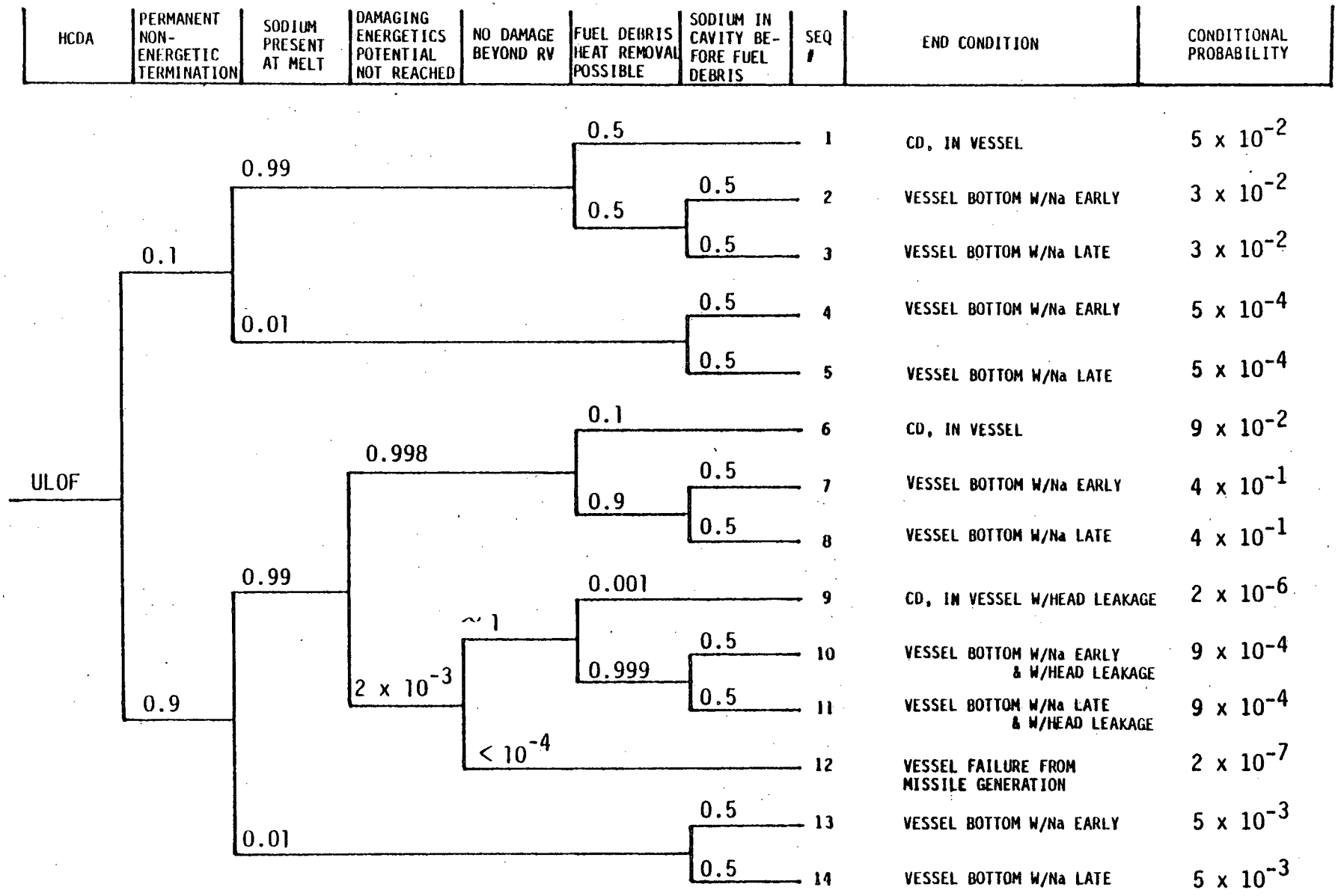


Figure 11.7-1. Core-Response Event Tree for ULOF.

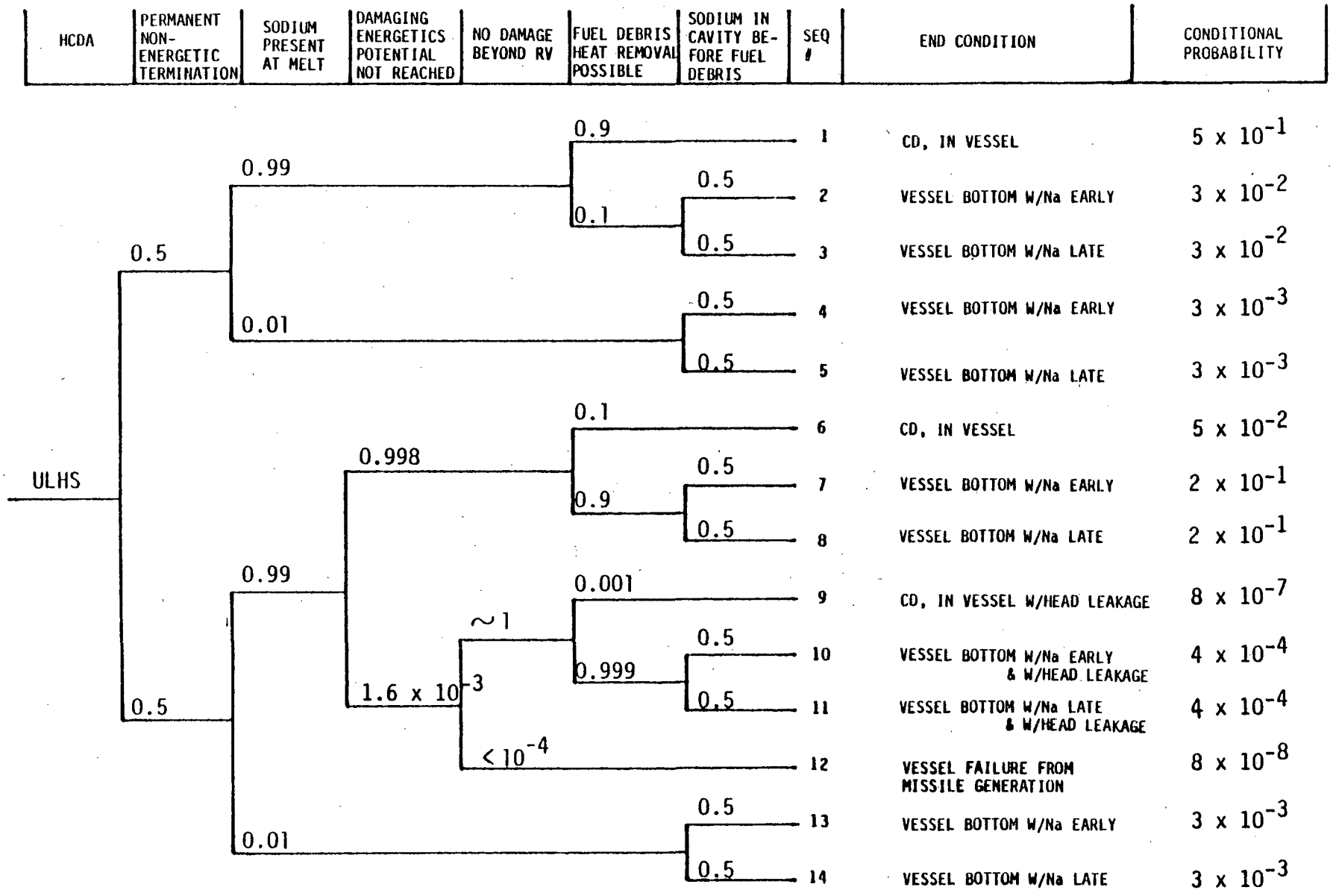


Figure 11.7-2. Core-Response Event Tree for ULHS.

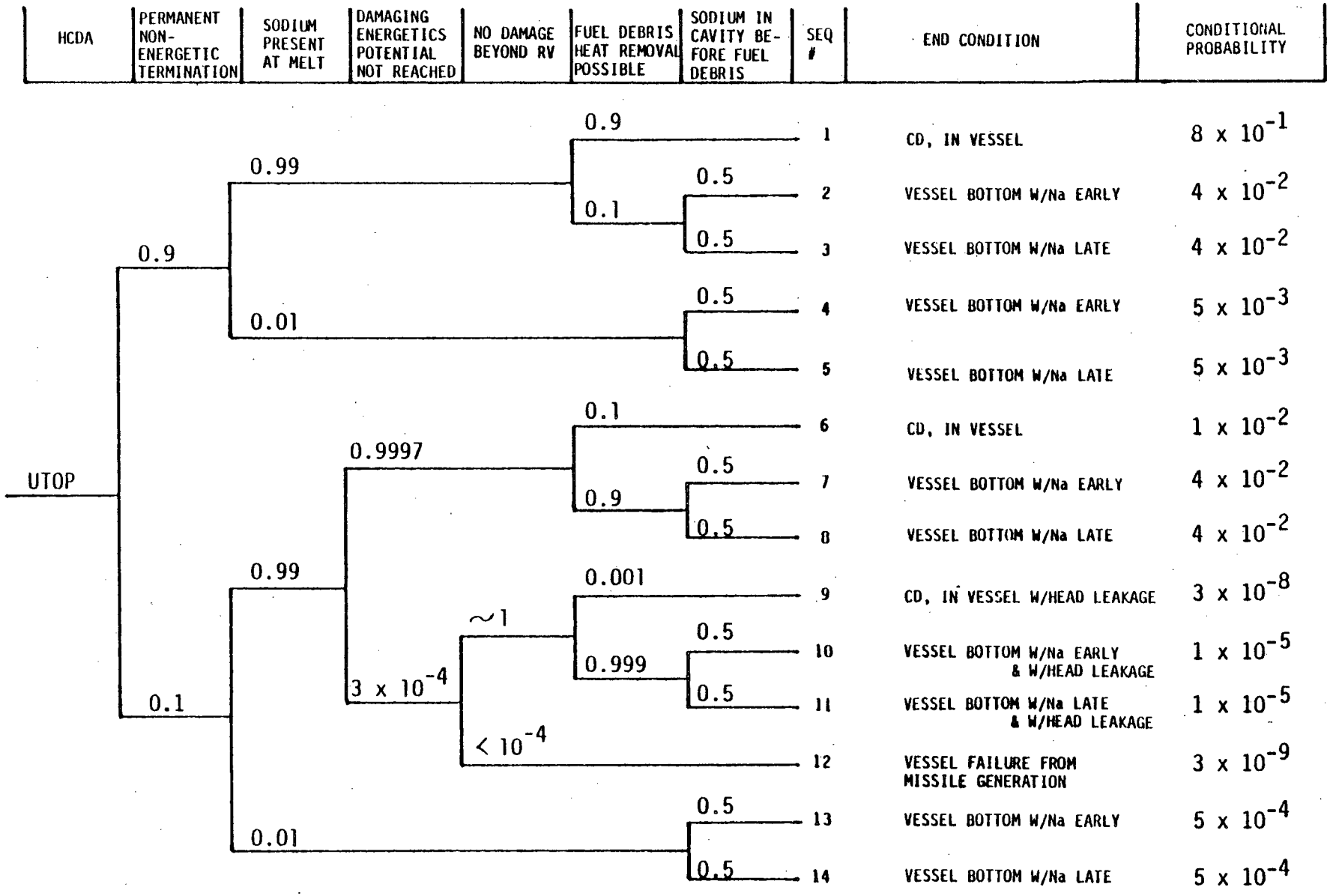


Figure 11.7-3. Core-Response Event Tree for UTOP.

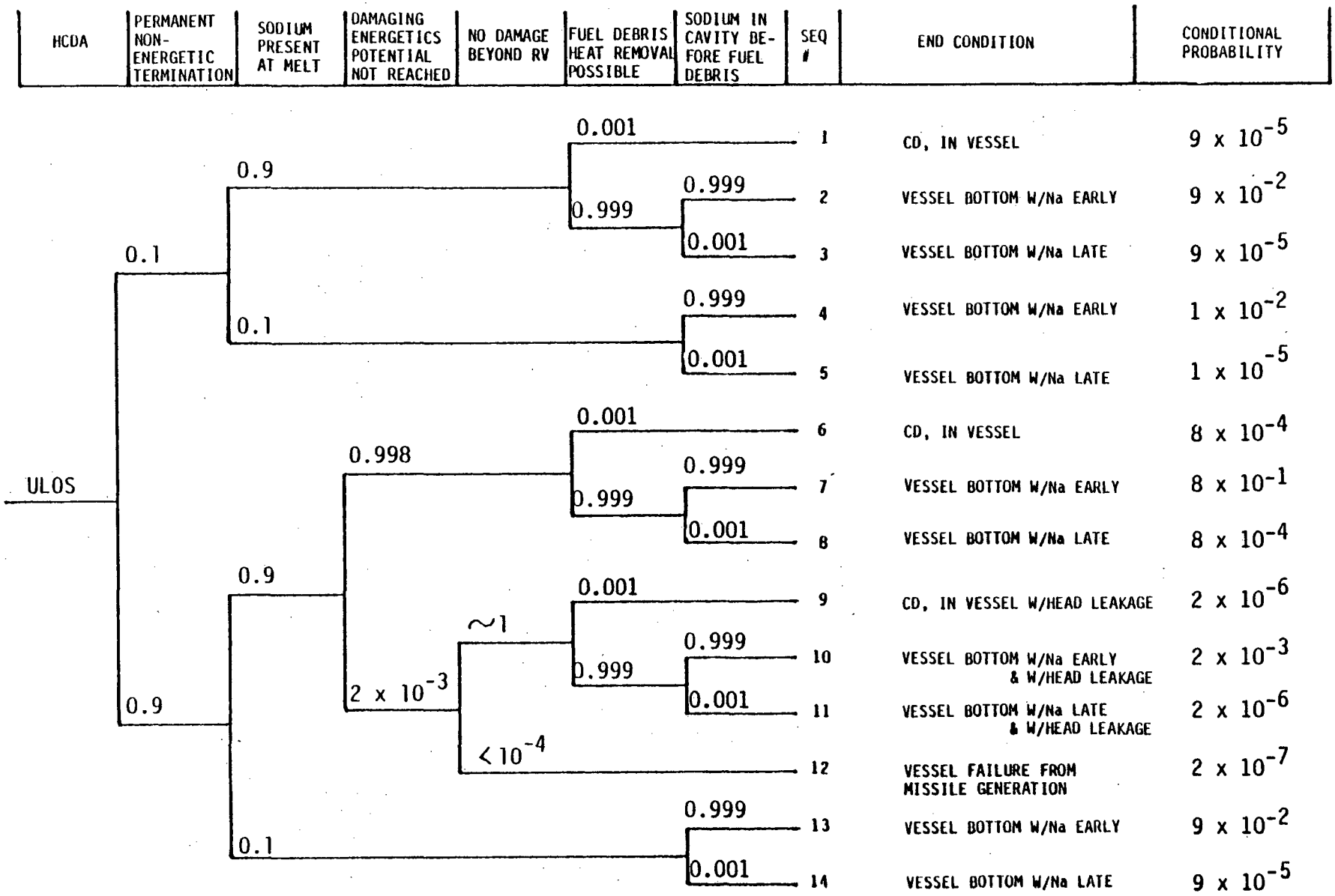


Figure 11.7-4. Core-Response Event Tree for ULOS.

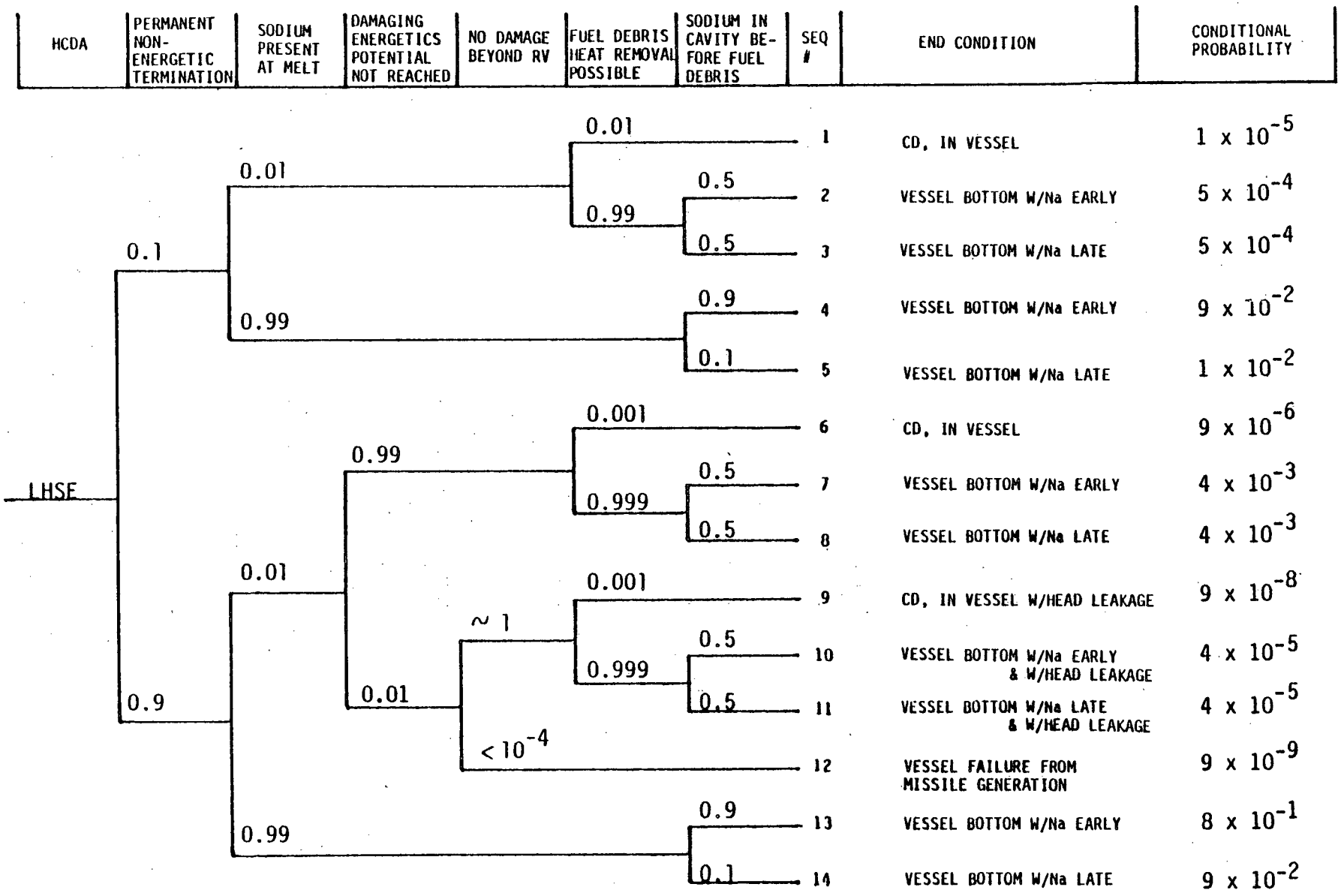


Figure 11.7-5. Core-Response Event Tree for LHSE.

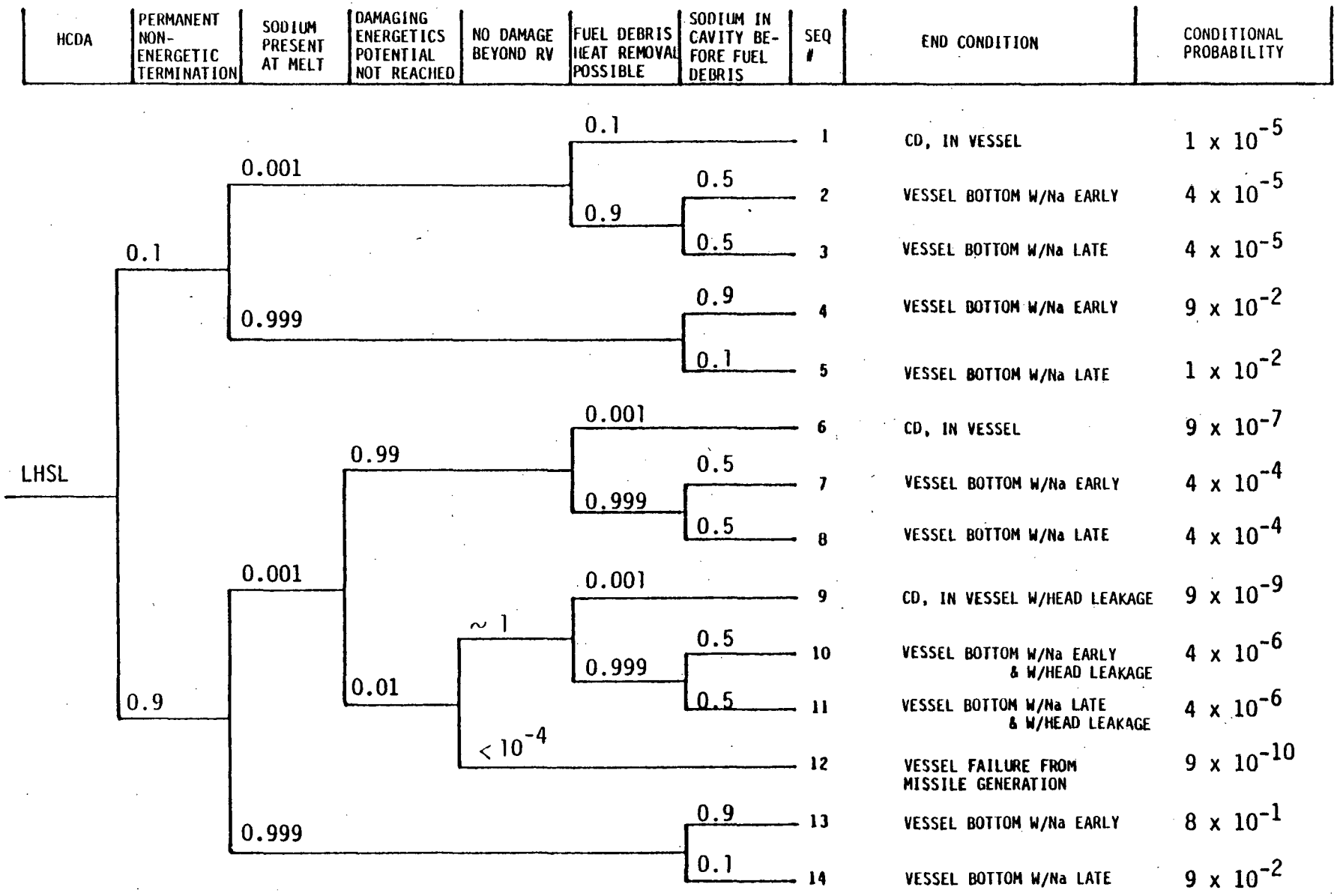


Figure 11.7-6. Core-Response Event Tree for LHSL.

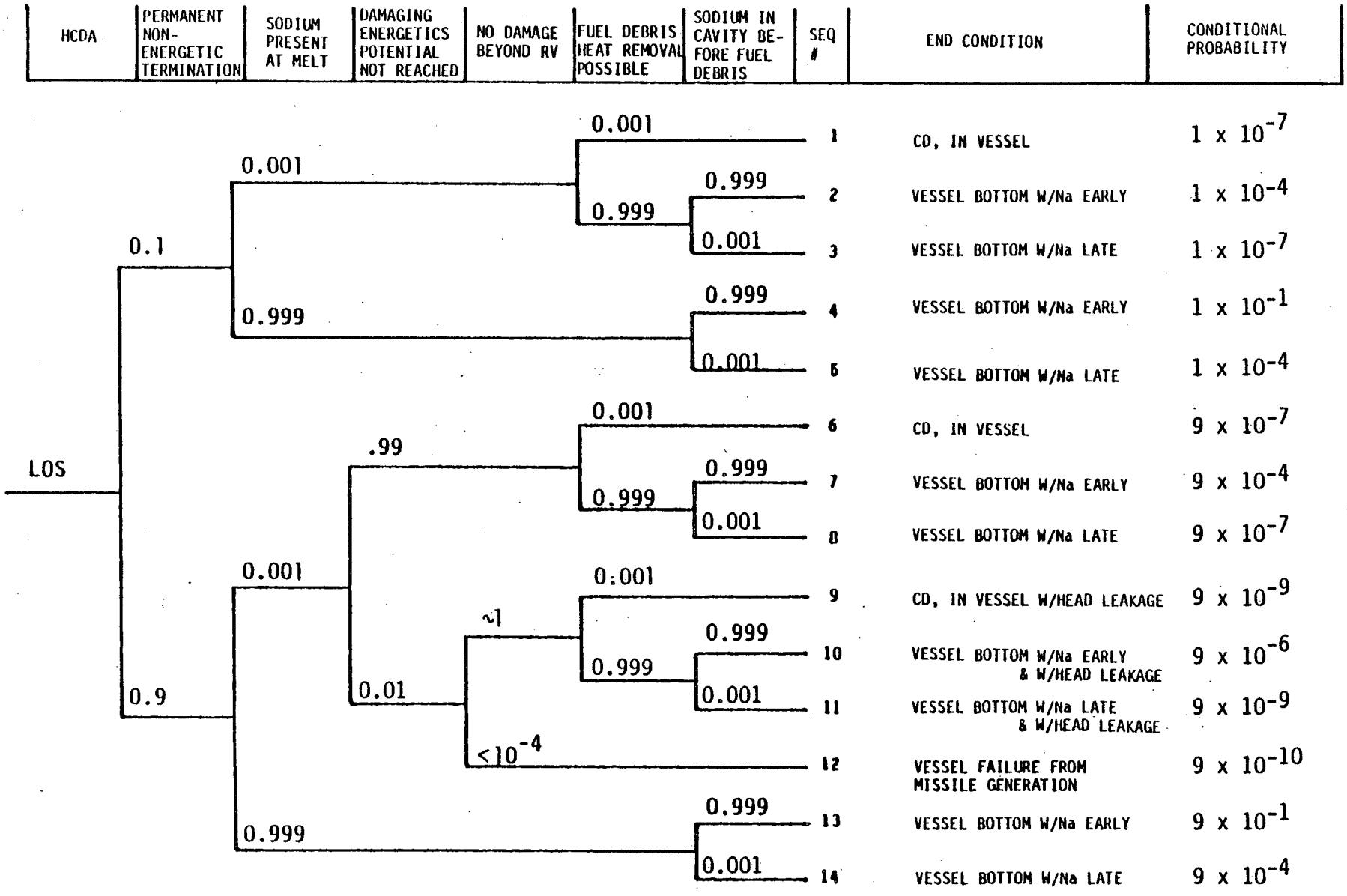


Figure 11.7-7. Core-Response Event Tree for LOS.

Table 11.7-1

CONDITIONAL PROBABILITIES FOR CORE-RESPONSE END STATES

State	1	2	3	4	5	6	7
Core-Damage Bin \ Core-Response End States	Core Damage, In-Vessel	Vessel Bottom w/Na Early	Vessel Bottom w/Na Late	Core Damage In-Vessel w/Head Leakage	Vessel Bottom w/Na Early & w/Head Leakage	Vessel Bottom w/Na Late & w/Head Leakage	Vessel Failure From Missile Generation
ULOF	1.4-1[a]	4.3-1	4.3-1	1.8-6	8.9-4	8.9-4	2.0-7
ULHS	4.9-1	2.5-1	2.5-1	7.9-7	4.0-4	4.0-4	7.9-8
UTOP	8.1-1	9.4-2	9.4-2	3.0-8	1.5-5	1.5-5	3.0-9
ULOS	9.0-4	9.9-1	1.0-3	1.6-6	1.6-3	1.6-6	1.6-7
LHSE	1.9-5	8.9-1	1.0-1	9.0-8	4.5-5	4.5-5	9.0-9
LHSL	1.0-5	9.0-1	1.0-1	9.0-9	4.5-6	4.5-6	9.0-10
LOS	9.9-7	~1	1.0-3	9.0-9	9.0-6	9.0-9	9.0-10

[a]1.4-1 = 1.4×10^{-1}

initiator's occurrence considering other system failures which lead to the core-damage event. The total probability of an accident outcome is therefore the probability of the core-damage event times the core-response probability (shown here) times the containment-response probability (discussed in Section 12).

For the ULOF event, the accident is most likely to end up with reactor vessel bottom failure (86%) although another likely end state is partial core damage with no loss of primary system integrity (~14%). All other end states are at least two orders of magnitude less probable than the above two core-response end states.

The ULHS event is also most likely to end up in the same two core-response end states as for the ULOF event. In this case, there is nearly a 50-50 chance of proceeding to reactor vessel bottom failure or just partial core-damage for cases where the accident is mitigated before primary system failure.

The UTOP event, due to the prediction of early fuel sweepout during the accident, has an even higher chance of ending in a partial core-damage state (~81%). This is because the fuel sweepout, following core disruption, tends to shutdown the reactor and allow for a coolable core configuration. The end states that do end in reactor vessel failure are predicted to do so by vessel bottom failure (~18%).

Vessel bottom failure with sodium entering the reactor cavity first, is nearly always predicted as the end state for ULOS events (~99%). This outcome is driven by the fact that the loss of sodium precludes a likely

chance for mitigation of the event with the sodium coolant most likely entering the reactor cavity when melting of the core begins.

The protected accidents; LHSE, LHSL, and LOS are all predicted to end in reactor vessel bottom failure with sodium most likely in the reactor cavity first. The probability of this end-state ranges from 89% to nearly 100% for the three protected accidents. This prediction is dominated by the relatively slow progression of these events and the fact that the sodium level has to drop significantly (and hence may enter the reactor cavity first) before fuel damage begins.

For all accidents, other possible end states have significantly less chance of occurring than those mentioned above. The added failures of reactor vessel head leakage or an energetic reactor vessel rupture are considered to be remote possibilities. The chances of such occurrences range from a .0016 probability for a ULOS event to end up with reactor vessel bottom failure and head leakage, to negligible chances for energetic reactor vessel failure following any of the accident sequence types.

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SECTION 12

CONTAINMENT-RESPONSE ANALYSIS

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Section 12

CONTAINMENT-RESPONSE ANALYSIS

12.1 INTRODUCTION

In the last section, the risk potential associated with core-damage accidents was assessed only to the extent that it considered the response of the reactor core to specific HCDA initiators. As such, this assessment did not consider the effects of core damage outside of the primary coolant boundary. The end states of the core-response event tree represent the transition points between the core response to an HCDA (within the primary coolant boundary) and containment response to an HCDA (outside of the primary coolant boundary). The relationships between the containment-response analysis and the core-response analysis end states are discussed in Section 12.2. A description of the design features used in mitigating the consequences of a core-melt accident is provided in Section 12.3. Section 12.4 provides an overview of some of the more important phenomena associated with the containment response to a core-melt accident. These phenomena include: initial conditions in the reactor cavity, fuel debris coolability, sodium pool boiling, chemical reactions in the reactor containment building, sodium-concrete reactions, hydrogen burning and containment failure modes. Based on these phenomena a containment-response event tree is developed in Section 12.5. In Section 12.6 the containment-response event tree is quantified for each core-response event tree end state and for each HCDA initiator. Section 12.7 contains a summary of results and conclusions based on the containment-response analysis.

12.2 INTERRELATIONSHIPS WITH OTHER PRA ELEMENTS

The containment-response event tree includes branch points that represent both system failures and phenomena. The system success/failure branch point probabilities reflect basic event failures that may or may not be common with a particular HCDA initiator sequence. For example, assume that a protected loss of heat sink accident (PLHS) was initiated, in part, due to a loss of 1E power. The probability of each branch point in the containment-response event tree that reflects the status of a system must then account for the fact that 1E power was unavailable at the time the accident was initiated. The probability that the systems directly dependent on 1E power would be available on demand to mitigate the consequences of a core-melt accident would then depend on the probability of regaining 1E power between t_i and t_D, where t_i is the time of accident initiation and t_D is the time the systems are called upon to perform their designated function.

The branch point probabilities for containment-response phenomena depend to a great extent on the core-response event tree end conditions, covered in Section 11, which in turn depend upon the specific HCDA initiator. For example, the probability that the reactor cavity liner fails early could depend on whether fuel debris or sodium entered the cavity first. The probability that fuel debris (or sodium) enters the reactor cavity first would depend on the HCDA initiator. For example, as discussed in Section 11, fuel debris would be more likely to enter the reactor cavity first for an unprotected loss of flow (ULOF) event than for a protected loss of sodium (LOS) event.

As will be shown later in this section, the end result of the containment-response event tree is a radiological release matrix. Each end state of the containment-response event tree will correspond to one or more of the radiological release matrix categories. This matrix represents the end result of the PRA as defined within its final scope.

In Section 13, the HCDA initiator sequences, the core-response sequences and the containment-response sequences will be integrated, and the dominant sequences (in terms of frequency) will be identified.

12.3 DESCRIPTION OF DESIGN FEATURES USED IN MITIGATING THE CONSEQUENCES OF A CORE-MELT ACCIDENT

12.3.1 Containment/Confinement

The CRBRP containment/confinement safety feature is designed to mitigate the consequences of in-containment design basis accidents by providing a final barrier to the release of radionuclides. The function of the containment/confinement safety feature is to limit the leakage of radionuclides from within the containment boundary to the environment.* The containment/confinement safety feature, shown in Figure 12.3-1, consists of a steel shell containment, surrounded by a reinforced concrete confinement, and the containment isolation system (not shown).

The steel shell is a free-standing structure with a flat circular base that is embedded in concrete below the operating floor of the reactor containment building (RCB). Above the operating floor the steel shell

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*The confinement also serves the function of limiting the release of radionuclides from outside of the containment (i.e., from within the reactor service building) to the environment.

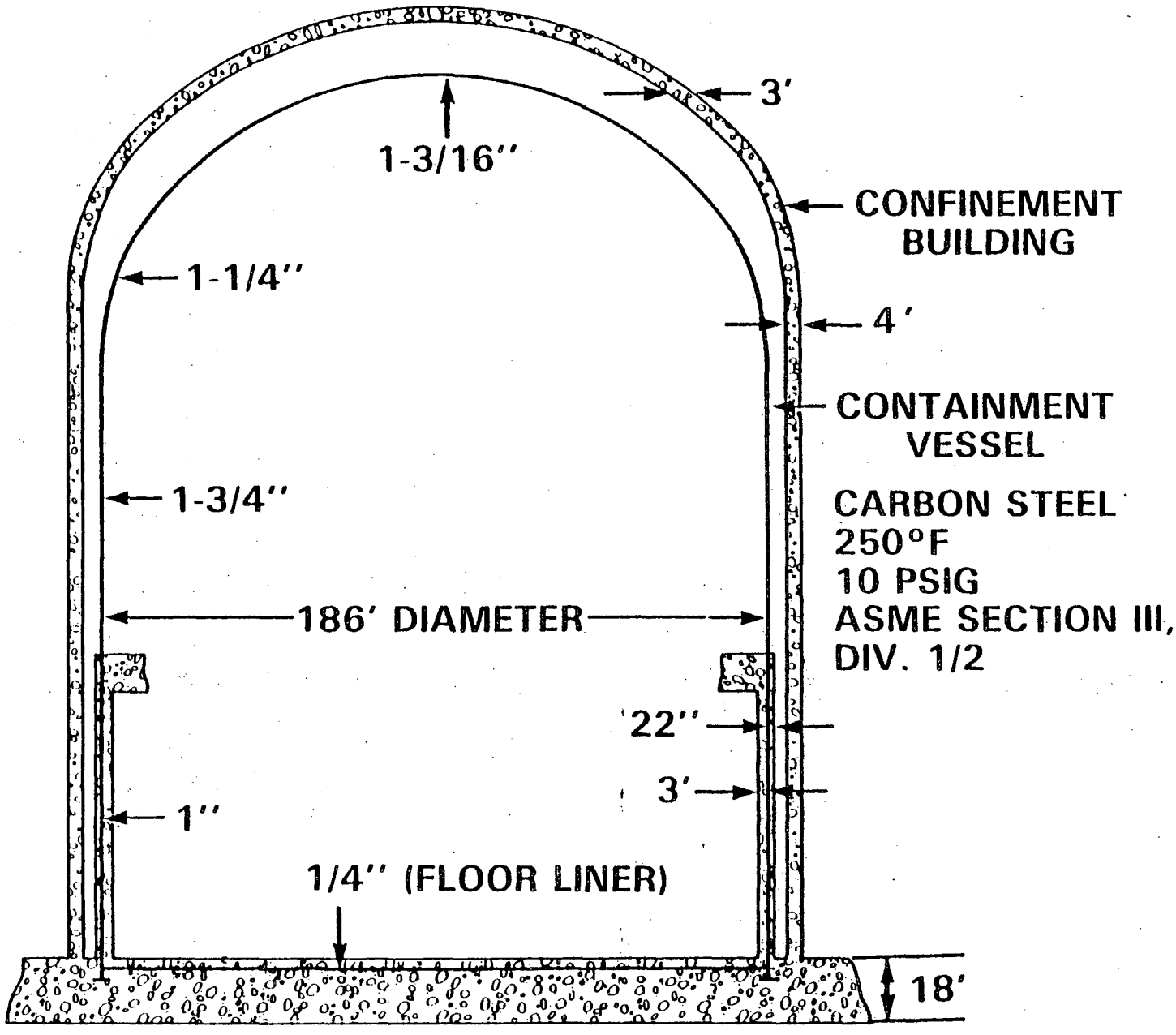

Figure 12.3-1. Containment/Confinement Structure. (Reference 2)

ranges in thickness from 1-3/16" to 1-3/4". The steel shell is 1" thick below the operating floor where cells are located that contain the primary heat transport system (PHTS), the primary sodium storage tank (PSST), and various other support systems (see Figure 12.3-2).

The atmosphere within cells containing primary sodium is inerted with nitrogen ($< 2\%$ O_2) to prevent extended sodium burning in the event of a sodium spill. The cells are lined with 3/8" carbon steel to prevent the contact of sodium and concrete (see Figure 12.3-3). The cell liner design consists of a modular prefabricated wall and ceiling panels composed of welded carbon steel liner plate backed by lightweight insulating concrete. The cell liner panels are anchored with structural embedments at the corners of the cell to provide stability. The carbon steel cell liner plate on the floor is attached to steel embedded into the concrete. A precast layer of lightweight insulating concrete is located between the floor liner plate and the underlying structural concrete. A cell liner vent system is provided to limit the pressure buildup behind the cell liners to 5 psig following sodium spills. Steam that is produced due to heating of the concrete is released into the gap between the liner plate and the insulating concrete. The steam generated behind the reactor cavity walls is vented to a non-inerted cell (cell 105) whereas the steam generated behind the reactor cavity floor liner is vented to the operating floor.

The atmosphere within the inerted cells is maintained and cooled by the recirculating gas cooling system (RGCS). The atmosphere is continually

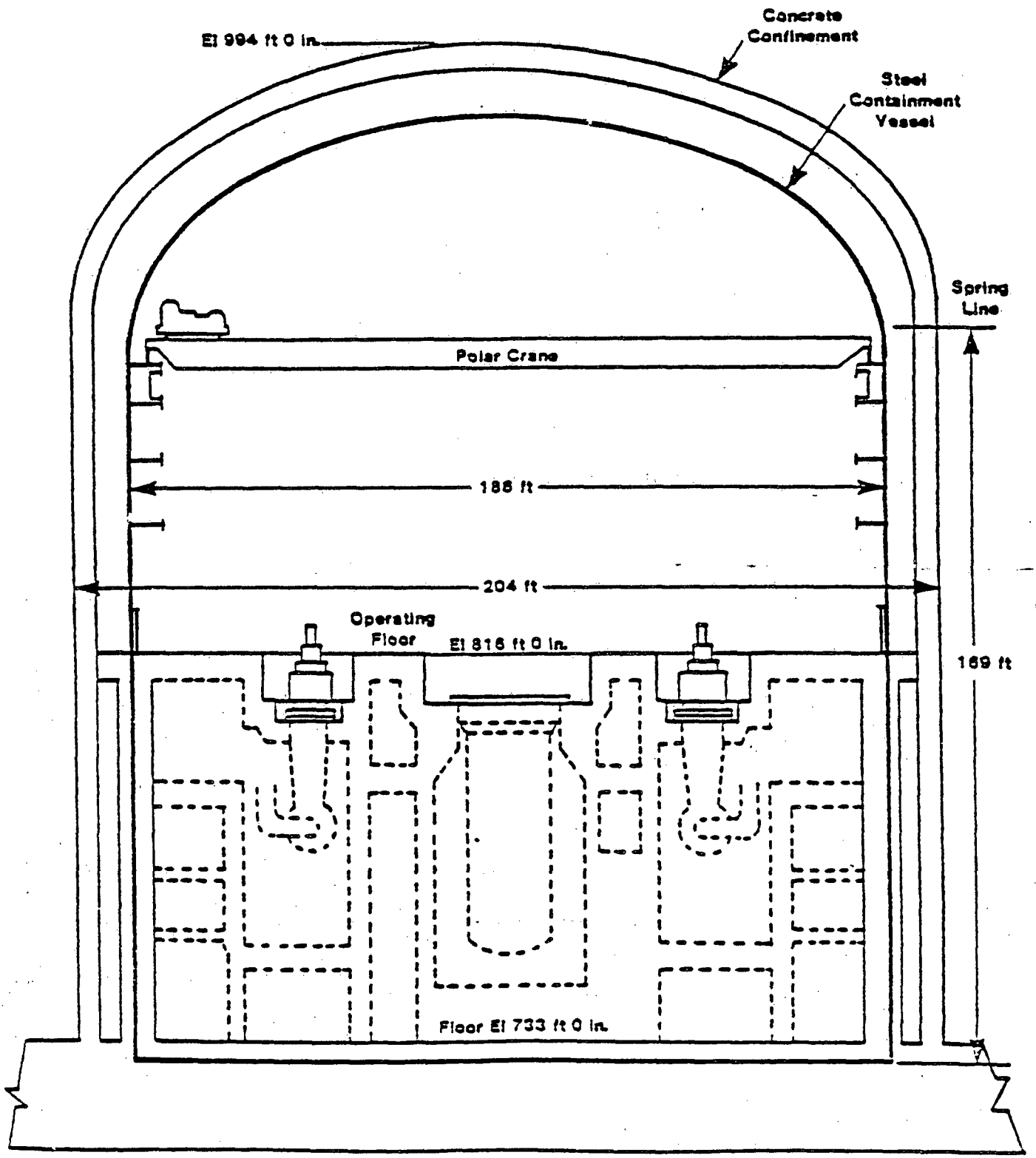

Figure 12.3-2. Containment/Confinement Cross Section Showing PHTS Cells.

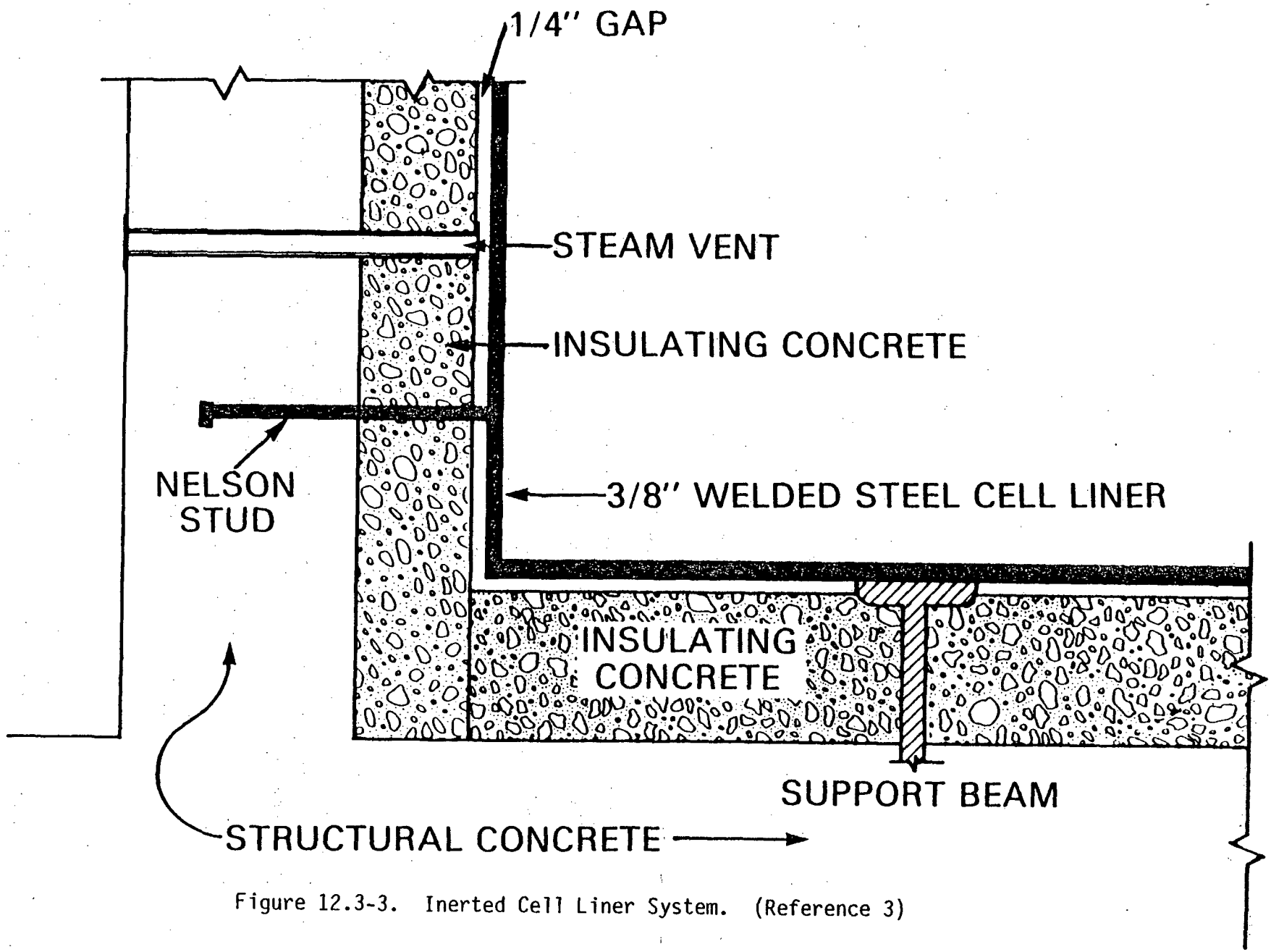

Figure 12.3-3. Inerted Cell Liner System. (Reference 3)

monitored for leakage into and out of the cells. Technical specification limits are placed on the leakage rate and on oxygen ingress into the cells.

The free volume within the containment is approximately 5,000,000 ft^3, of which 3,600,000 ft^3 is above the operating floor and 1,400,000 ft^3 is below.

The annulus between containment and the confinement building is about 5 ft wide above the operating floor. The free volume of the annulus is approximately 1,000,000 ft^3. The concrete confinement building is 3 ft thick at the dome and 4 ft thick at the cylindrical walls.

The design temperature of the containment steel shell is 250°F. The internal design pressure for the containment is 10 psig with an allowable leak rate of 0.1 volume percent per day into the annulus.* The ultimate capacity of the containment is delineated in Table 12.3-1.

The containment is maintained at a negative pressure with respect to the atmosphere (-1/8 inch w.g.). The pressure within the annulus (-1/4 inch w.g.) is lower than the containment pressure to ensure that any potential leak from the containment is filtered by a filtration sytem before being vented to the atmosphere. The containment is protected from large external pressures by two independent vacuum relief systems, each of which is capable of fully relieving postulated external pressures in the annulus. The vacuum relief system consists of two self-actuated

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*This leak rate assumes successful function of the containment isolation system.

Table 12.3-1

CONTAINMENT STRUCTURAL CAPABILITY[1]**

Steel Shell Temperature (°F)	Capacity Based on Allowable Stress Pressure (psig)	Capacity Based on Ultimate Stress* Pressure (psig)
100	46.8	51.1
150	43.9	53.5
200	42.7	55.2
250	41.9	56.5
300	41.4	57.3
350	41.0	57.8
400	40.3	57.8
450	39.2	57.5
500	37.7	56.7
550	36.0	55.6
600	34.7	54.1
650	34.0	52.2
700	33.6	49.9

*This criterion states that the primary stress shall not exceed 85% of the ASME Code, Section III, Appendix F limits (0.7 x ultimate stress)

valves connected in series that open if the pressure in the annulus becomes greater than the pressure in containment. The valves are set to close before the pressure in containment approaches the pressure of the outside atmosphere to preclude the possibility of leakage out of the containment.

The negative pressure within the annulus is maintained by means of the annulus filtration system (AFS) (see Figure 12.3-4 and Section A19). The AFS uses two redundant filter-fan units to filter 14,000 CFM of air. Of this air, 3000 CFM is exhausted, and the remaining 11,000 CFM is recirculated. Thus, any leakage from the containment into the annulus would be filtered by the annulus filtration system prior to being released to the environment.

The containment isolation system provides a means for automatic and manual closure of valves in lines that penetrate containment to ensure that the containment functions as the final barrier to radionuclide release. (The IHTS piping is considered to be an extension of the containment boundary and is, therefore, not isolated.)

There are four types of containment isolation valves. Automatic containment isolation valves, which close upon receiving a signal from the containment isolation system (CIS) logic train, comprise the first type. The CIS logic train receives its input from two radiation monitors, one in the head access area (HAA) and the other in the containment HVAC exhaust duct. Containment penetrations that are isolated by automatic containment isolation valves include the containment ventilation air

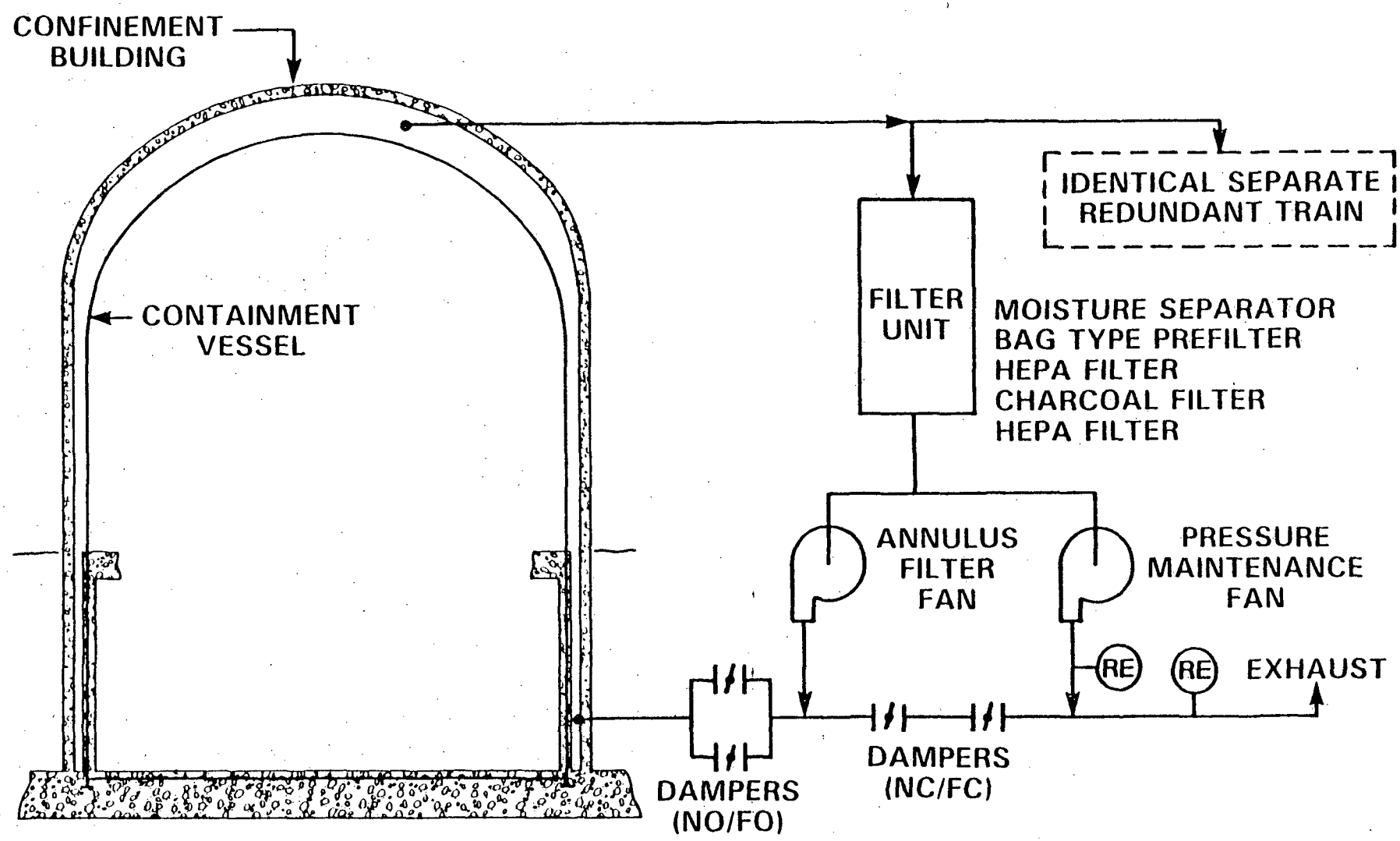

Figure 12.3-4. Confinement Annulus Filtration System. (Reference 4)

exhaust, instrument air and the cell atmosphere processing system (CAPS) inlet header. The second type of containment isolation valves includes those that must be operated remote manually, usually from the main control room. The emergency chilled water supply and return line are examples of penetrations that have remote manual isolation valves. The third type of containment isolation valves includes those that are back pressure regulated such that if the supply side pressure drops below a specified limit, the valves automatically close. The argon supply line is an example of a line isolated by back pressure regulated valves. This type of valve would help ensure that radioactive argon or sodium could not be released from containment following argon supply depressurization or a substantial increase in the cover gas pressure. Last are the containment isolation valves that must be manually closed to isolate containment. All valves that fall in this category are closed during normal reactor operation.

12.3.2 Reactor Cavity Vent To RCB

The reactor cavity-to-RCB vent is shown in Figure 12.3-5. This system is specifically designed for accidents beyond the design basis. The purpose of the reactor cavity vent is to prevent overpressurization of the reactor cavity after penetration of the reactor vessel and guard vessel and to promote maximum exchange of heat between the vented cavity gases and the pipeway cell structures before releasing the gas above the operating floor.[1]

The reactor cavity vent path is estabished by the failure of rupture disks following the entry of sodium into the reactor cavity. The

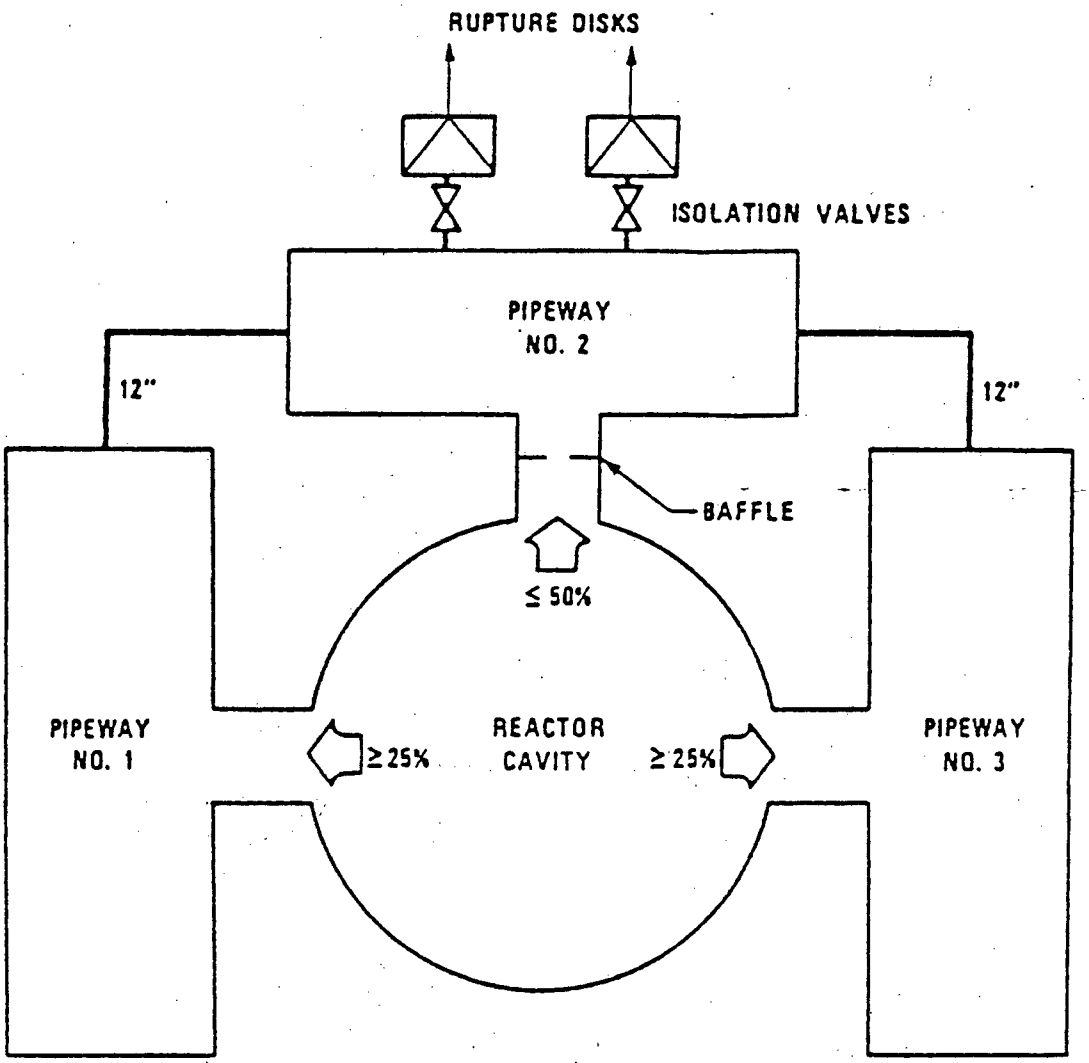

Figure 12.3-5. Reactor Cavity Vent System. (Reference 1)

rupture disks open when the pressure in the reactor cavity exceeds the rupture disk setpoint as a result of air volume displacement by sodium. (The rupture disks are designed not to rupture for design basis leaks.)

Approximately 50% of the gases generated in the reactor cavity are vented directly to pipeway cell No. 2, where the vents to the RCB are located. Pipeway cell No. 1 and No. 3 each receive about 25% of the reactor cavity gases. The purpose for diverting flow to pipeway cells other than No. 2 is to take advantage of the heat sink provided by the pipeway structure surfaces. If rupture disks were provided in every pipeway cell, the rupture of one disk would cause reactor cavity depressurization and likely prevent the utilization of the other pipeway cell structures as heat sinks. The pipeway cells are slanted toward the reactor cavity to promote reflux of condensed sodium.

The reactor cavity gas that flows into pipeway cell No. 2 is vented up to the operating floor through two 12" diameter pipes.

The rupture disks can be isolated by remote manually operated gate valves, shown in Figure 12.3-5.

12.3.3 Annulus Cooling System

The annulus cooling system is specifically designed for accidents beyond the design basis. The purpose of the annulus cooling system is to remove heat via forced convection from the containment and confinement structures in order to ensure their structural integrity following a core-melt accident.

The annulus cooling system draws outside air into the annulus through an opening in the confinement structure. By means of redundant vane axial fans, air is blown upward in a spiral path between the containment and confinement building. The spiral air flow path is provided by partitions in the annulus. The annulus filtration system cannot be in operation at the same time as the annulus cooling system, since all bypass leakage into the annulus after initiation of the annulus cooling system would be directly released to the environment. However, bypass leakage into the annulus prior to operation of the annulus cooling system would be filtered by the annulus filtration system. A more detailed description of the annulus cooling system is provided in Section A20.

12.3.4 Vent And Purge Systems

Like the annulus cooling system, the vent and purge systems are also specifically designed for accidents beyond the design basis. The purposes of the vent and purge systems are to reduce the pressures and the hydrogen concentration in containment and to allow for the controlled burnoff of sodium and hydrogen with oxygen. Although physically different, the vent and purge systems must operate in conjunction to serve this function. Upon reaching the criteria for venting following a core-melt accident, a vent path from the containment to the containment cleanup system (CCS) is established by opening remote manual containment isolation valves shown in Figure 12.3-6. (There are two 24" vent lines from the RCB to the CCS.) When the containment pressure is reduced to atmospheric, the CCS exhaust blowers are turned on and the purge line

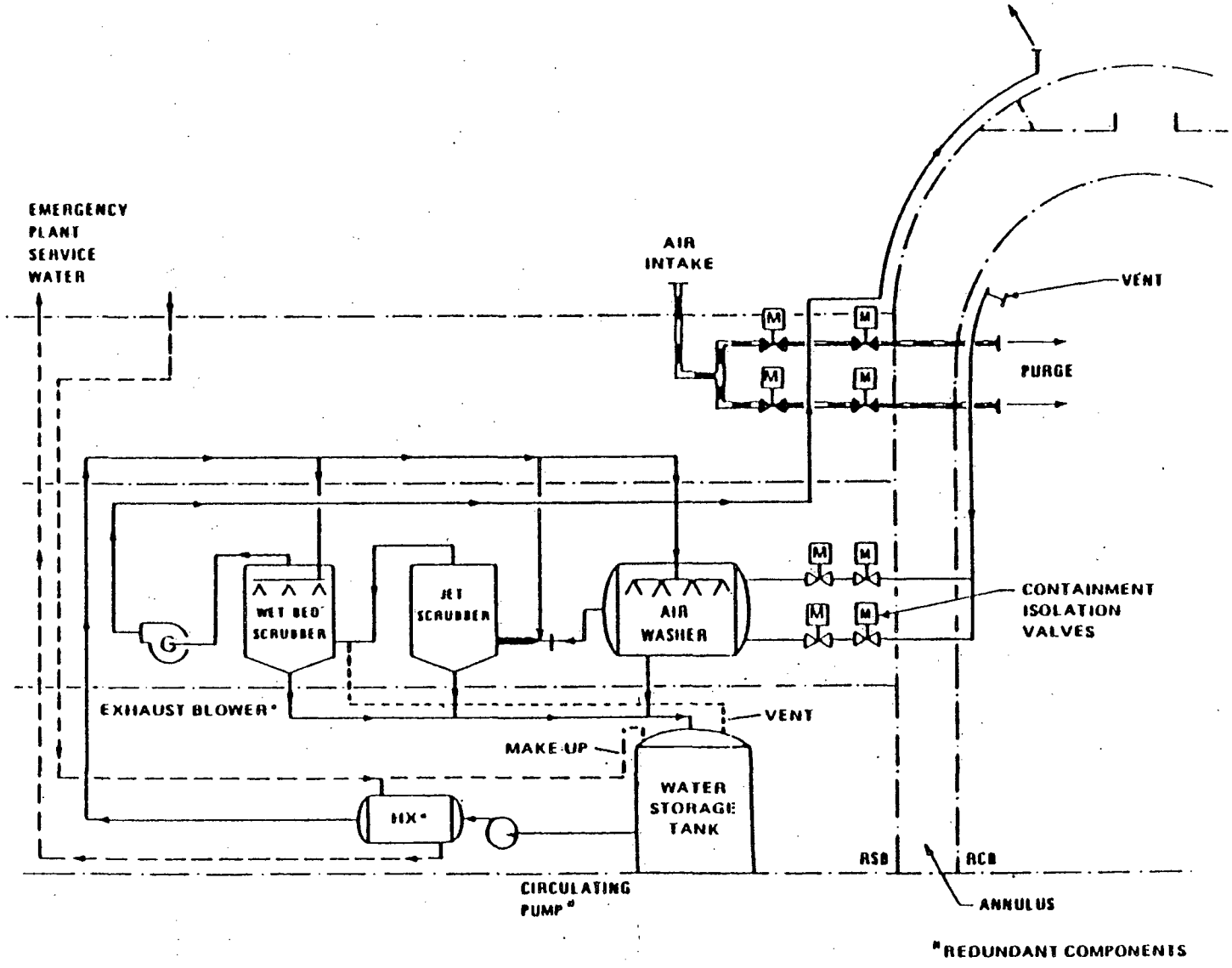

Figure 12.3-6. Containment Vent, Purge and Cleanup Systems. (Reference 1)

containment isolation valves are opened remote manually by the operator. The cleanup system exhaust blowers produce a suction to purge air into the containment. Purging of air into the containment will reduce the hydrogen concentration and increase the oxygen concentration. The increase in the oxygen concentration allows controlled burning of sodium and hydrogen. A more detailed description of the vent and purge systems is provided in Section A21.

12.3.5 Containment Cleanup System

The containment cleanup system, shown in Figure 12.3-6, is specifically designed for accidents beyond the design basis. The purpose of the CCS is to filter the RCB atmosphere that is exhausted during venting operations prior to releasing it to the environment. The filtering is done by a series of scrubbers that are physically located in the reactor service building.

The scrubbers include a jet venturi scrubber in series with a high efficiency wetted fiber bed scrubber unit. An air washer is upstream of the scrubbers to ensure that all of the sodium oxide that has entered the CCS train is reacted to form sodium hydroxide.

The CCS filter train is designed to provide an overall filtration system efficiency of 99% for vented solids and liquids and 97% for all vented vapors (excluding noble gases). Standby redundancy is provided on all active CCS components. A more detailed discussion of the CCS is provided in Section A21.

12.3.6 Instrumentation for Beyond-Design-Basis Accidents

Containment atmospheric pressure and temperature, steel shell temperature and hydrogen concentration can all be measured following a core-melt accident, with instrumentation that is specifically designed for beyond-the-design basis accidents. The containment pressure can be measured at temperatures up to 1100°F with a capillary line that is connected to a pressure element and transmitter located outside of the containment. Containment atmospheric temperature is measured with sensors at the top of the RCB. The signal conditioning equipment for the RCB atmospheric temperature sensors is located in the steam generator building (SGB). Similar instrumentation is used to measure the steel shell temperature. Continuous hydrogen analyzers located in the SGB and connected to the RCB through redundant and independent sampling lines would be used to sample hydrogen following a core-melt accident. The inlet to the sampling lines is located at the top of the RCB to ensure that an accurate indication of hydrogen concentration would be obtained if hydrogen stratification were to occur.

All of the instrumentation described above is redundant and designed to remain functional following a safe shutdown earthquake, and is also qualified to remain functional under the severe environmental conditions imposed by a core-melt accident.

12.4 DESCRIPTION OF CONTAINMENT RESPONSE TO A CORE-MELT ACCIDENT

12.4.1 Overview of the End States of the Core-Response Event Tree

The majority of the end states of the core-response event tree in Section 11 corresponded to conditions in which the core debris eventually penetrates the reactor vessel and guard vessel and relocates in the

reactor cavity. Only one sequence (assessed to be non-mechanistic) corresponded to an early failure of the containment as a result of an HCDA-generated missile. Another sequence that proved to be non-mechanistic (i.e., 1×10^{-4} and lower for all sequences) was the sequence designated "CD, In-Vessel with Head Leakage." The remaining sequences that pose a challenge to the containment integrity all involve penetration of the reactor vessel bottom head. Thus, the most likely challenge to containment integrity will correspond to scenarios in which the core is in the reactor cavity with or without sodium leakage through the reactor vessel head. In Reference 1 the CRBRP Project performed a sensitivity study on the amount of sodium leakage through the reactor vessel head. It was concluded that head leakage prior to 24 hours would not result in conditions that challenge the containment integrity without venting, purging, or annulus cooling system operation. From this it was concluded that scenarios involving head leakage do not pose an extraordinatry threat to containment integrity. Since from a consequence perspective the containment response to a core in the reactor cavity with sodium leakage through the reactor vessel head is similar to the containment response to a core in the reactor cavity without head leakage, only the latter will be explicitly addressed in this section. On this basis the containment response will be described for only the scenario, "Reactor Vessel Bottom Fails with no Head Leakage."

12.4.2 Reactor Vessel Bottom Failure Sequences

Initial Conditions in the Reactor Cavity

As explained in Section 11, vessel bottom failure (most likely due to creep rupture) would be followed by relocation of sodium and core

materials from the reactor vessel into the reactor cavity. In Reference 1 it was estimated that following vessel penetration, approximately one million gallons of sodium would drain and siphon from the reactor vessel and PHTS loops. On this basis, the sodium pool that would be formed within the reactor cavity would be approximately 17 feet in height (see Figure 12.4-1). The sodium pool could theoretically reach 26 feet in height in the unlikely event that the entire sodium volume in the reactor vessel was displaced due to excessive reactor vessel pressurization relative to the reactor cavity.

Reference 1 assumed that the complete fuel and blanket inventory was released to the reactor cavity following vessel penetration. This assumption is conservative since an equivalent height of sodium within the reactor vessel would be capable of preventing further melting of much of the fuel and blanket material retained above the bottom head at the time of penetration. In addition, depending on the size of the vessel rupture, only a fraction of the fuel and blanket material in the lower plenum area would likely be released to the reactor cavity immediately.

In Section 11.6.7 it was concluded that for HCDA sequences where sodium was present at the time of fuel disruption, there was no substantial evidence that supported whether fuel debris or sodium enters the reactor cavity first. It was also concluded in Section 11 that, if fuel debris entered the reactor cavity first, there would be a greater chance of liner failure. However, even if fuel enters the reactor cavity first it would be immediately followed by sodium and, if molten, become quenched and fragment. It is uncertain as to whether the initial contact of

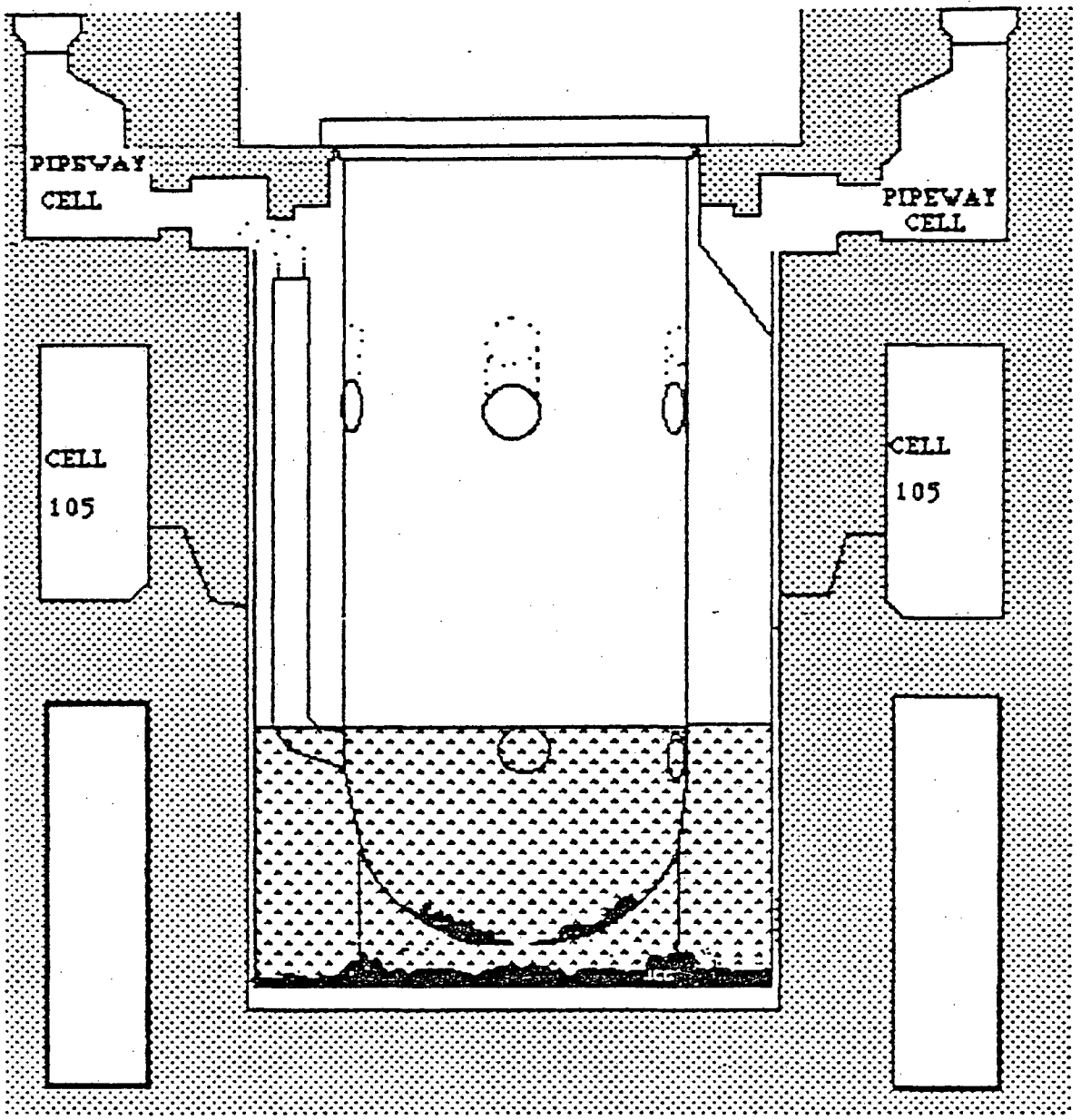

Figure 12.4-1. Reactor Cavity Cross Section Showing Sodium Pool Depth.

molten fuel debris with the reactor cavity floor liner would be of sufficient duration to result in liner failure. Even if immediate liner failure did occur it would probably be limited to local areas of molten fuel-liner contact.

The potential for liner failure is dependent on the initial temperature of the sodium entering the reactor cavity. Liner failure could occur earlier in time for those sequences in which very hot sodium (~1600°F) enters the reactor cavity first, or later in time if the sodium were relatively cool (~800°F). It is expected that the liner failure potential would tend to increase with an increase in sodium temperature. The liner failure mode in this instance would still likely involve only local ruptures due to excessive thermal strains. Fragmentation of fuel debris entering the sodium pool would result in rapid heat transfer to the sodium pool, but is not expected to result in substantial fuel-coolant interactions. The consequences of liner failure will be discussed later in this section.

Reactor Cavity to Reactor Containment Building Vent

The pressures generated in the reactor cavity due to volume displacement of air by sodium will cause the reactor cavity to reactor containment building (RC-RCB) rupture disks to burst. Once the rupture disks have burst, a direct path from the reactor cavity to the upper reactor containment building would be established, reducing the differential pressure between the reactor cavity and the reactor containment building. Various failures of the RC-RCB vent system were analyzed in Reference 1. These failures include partial RC-RCB vent line plugging,

complete plugging of one of the vent lines that interconnect the pipeway cells and delayed opening of the rupture disks. The consequences of RC-RCB vent malfunction are discussed in greater detail in Reference 1.

Assuming that the rupture disks do open immediately following vessel penetration, a "puff" of noble gases would enter the upper containment building. Containment isolation is then initiated upon receiving a signal from the radiation monitors located in the head access area (HAA) and the 100% outside air system exhaust duct. Unlike LWR's, the CRBRP containment is isolated based on one parameter only. The annulus filtration system is designed to filter any radionuclides (except for noble gases) that leak into the annulus from the containment.

Fuel Debris Coolability

The factors governing the potential for fuel debris coolability within the reactor cavity are the same factors that applied to fuel debris coolability within the reactor vessel discussed in Section 11 of this report. These factors include debris bed depth (relative to the bed depth leading to dryout), the presence of sodium, and the availability of heat sinks. In Reference 1, the CRBRP Project predicted that debris bed "self-leveling" will occur within the reactor cavity due to the fluid agitation induced by sodium boiling. The debris bed was assumed to be of uniform depth and to cover the entire floor area of the reactor cavity. Debris bed uniformity is made possible, in part, by the design of flow ports in the guard vessel support skirt that allow core debris to be sweptout onto the reactor cavity floor. Since the bed depth calculated was less than the bed depth corresponding to dryout, the fuel

was not predicted to melt until all of the sodium above the debris bed boiled away. In addition, a sensitivity analysis concluded that a factor of 2 to 4 is available between the bed dryout depth and the depth corresponding to a uniform distribution of fuel debris within the reactor cavity. In general, the containment response to core melt was deemed to be insensitive to the degree of bed leveling.

Reference 1 also concludes, based on experiments performed at Argonne National Laboratory, that the "effect of subcooled sodium may increase the margin between the bed dryout thickness and the depth of a debris bed uniformly distributed over the reactor cavity floor."

Sodium Pool Boiling

The time to sodium pool boiling would depend to a great extent on the core-response event tree end conditions. Sodium from the reactor vessel may be present in the reactor cavity many hours prior to reactor vessel/guard vessel penetration for the protected loss of sodium (LOS) accident. The sodium pool in the reactor cavity would lose heat to the surrounding reactor cavity walls and to the upper reactor cavity. The reactor cavity liners are designed to accommodate large sodium spills and would be less likely to fail for the LOS event since the sodium temperatures would be relatively low (750-1000°F). If the liners do not fail, the sodium in the reactor cavity would not come into contact with concrete and the reactor cavity liner vent would allow the steam released from the concrete to be vented to the upper containment (RC floor liner) or to cell 105 (RC wall liners). As heat is removed from the sodium pool, the pool temperature will decrease possibly even to the

point where the sodium freezes (~210°F). If the fuel debris that penetrates the reactor/guard vessel and falls into the sodium pool in the reactor cavity is in a molten form (e.g., due to a previous absence of sodium in the reactor vessel), the fuel will fragment upon entering the sodium pool. The finer particles of fuel will transfer heat to the sodium pool more efficiently and the sodium temperatures will begin to rise.

It is very conceivable that the fuel debris decay heat level would be so low, and the margin to sodium boiling so large, that the sodium saturation temperature would not be exceeded for at least ten hours or more.

The sodium in the reactor cavity following a protected loss of heat sink (PLHS) accident, on the other hand, would likely exhibit a margin to boiling much smaller than that for the LOS accident. Thus, sodium boiling would be expected sooner for the PLHS event than for the LOS event.

Chemical Reactions in the RCB

As sodium boiling commences, volatile fission products and fuel entrained in the sodium will be transported up into the reactor containment building (RCB). The free sodium that is transported up into the RCB could theoretically react with the concrete of the operating floor, burn with oxygen present in the air and react with the water vapor vented from the reactor cavity floor liner. Since the sodium entering the upper containment building would be a vapor, it seems unlikely that it could condense and react with concrete before being consumed in reactions with oxygen. On this basis, sodium-concrete interactions are not

considered to be important early-on in the core-melt scenario. The CACECO* model used by the CRBRP Project in Reference 1, assumes that sodium vapor reacts instantly with oxygen to form sodium oxide (Na_2O) aerosols. The Na_2O was then assumed to react with water vapor to form sodium hydroxide (NaOH).

The assumption that the sodium vapor reacts instantly with oxygen was based on approximately 45 different experiments performed at Hanford Engineering Development Laboratory (HEDL) in which a jet of sodium vapor was introduced into a chamber containing varying quantities of water vapor.[4] The experiments indicated that the formation of hydrogen (a by-product of sodium-water reactions) was negligible for water vapor concentrations less than the oxygen concentration. When the water vapor concentration exceeded the oxygen concentration, hydrogen was produced in accordance with the molar fraction of sodium and water vapors present.** However, as reported in Reference 5, NRC interpreted these experimental results a different way. They postulated that the sodium vapor reacted with water vapor in proportion to its concentration but that the temperatures in the "reaction zone" (of the experimental test section) were high enough to allow the hydrogen to burn with oxygen and,

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*The CACECO computer code model, originally developed for the Fast Flux Test Facility (FFTF) is used primarily to assess the thermal and pressure loadings on structures within containment following an LMFBR core-melt accident.

**Some test results actually showed that H_2 was produced in excess of what would be expected if the sodium vapor and water vapor reacted in accordance with their molar fractions. This phenomena was attributed to hydrogen contamination from previous experiments.

therefore, not be observed in the experiment exhaust gas measurements. NRC's interpretation of the experimental test results would not seem to explain the production of hydrogen in the tests in which the water vapor concentration exceeded that of the oxygen. On this basis, it is assumed that H_2 production will be negligible when the water vapor concentration is below that of the oxygen.

Consequences of Liner Failure

Up until this time in the accident progression it has been assumed that reactor cavity liner failure has not occurred. As previously mentioned in this section, it is deemed likely that the liner failure would initially be limited to local ruptures of the floor liner caused by excessive thermal strains.

Once the liner fails, sodium will come into contact with the water and carbon dioxide (CO_2) generated from the heated concrete. The applicable reactions in the sodium pool are as follows:

Reaction	Heat of Reaction
$2Na + H_2O = Na_2O + H_2$	1,600 Btu/lbm Na
$2Na + 2H_2O = 2NaOH + H_2$	4,514 Btu/lbm Na
$4Na + 3CO_2 = 2Na_2CO_3 + C$	4,326 Btu/lbm Na
Na + Concrete	331 Btu/lbm concrete

The ratio of sodium hydroxide (NaOH) to sodium oxide (Na_2O) produced is dependent on the hydrogen partial pressure and the system temperature. Any aerosols generated below the sodium pool surface would likely be

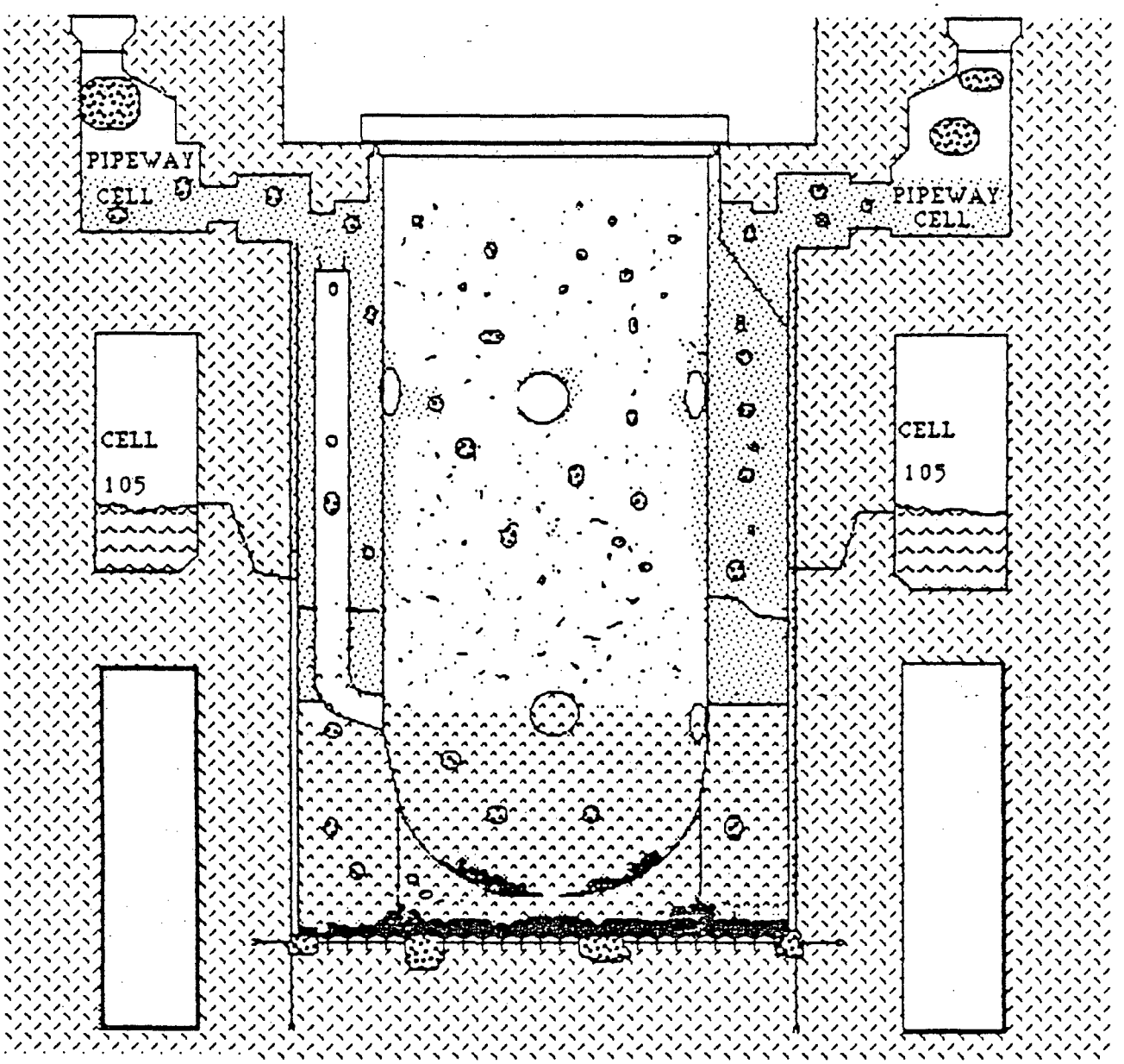
retained within the pool.[1] Hydrogen produced from sodium-water reactions would, however, leave the pool and accumulate in the upper containment building until hydrogen ignition criteria are met (see Figure 12.4-2). The hydrogen ignition criteria will be discussed later in this section.

The heat generated from the reactions would be added to the sodium pool, increasing the rate of boiling in saturated sodium pools and decreasing the margin of subcooling if sodium pool boiling has not begun. For the base case scenario presented in Reference 1, the following energy inputs to the reactor cavity were calculated for the first 36 hours following reactor vessel/guard vessel penetration.

Source	Energy (MW-sec)	% of Total
Fission Product Decay (14.28 MW at 1000 seconds to < 5 MW at 36 hours)	1.0×10^6	89
Sodium - H_2O Reaction	9.0×10^4	8
Sodium - CO_2 Reaction	2.2×10^4	2
Sodium - Concrete Penetration	1.1×10^4	1

Sodium-Concrete Reaction Rates

The rate at which non-condensibles, including hydrogen, would be produced is a function of the sodium-concrete reaction rate. In the CRBRP Project (Reference 1) base case analysis, the concrete-sodium reaction rate was estimated for both horizontal and vertical submerged surfaces based on available experimental evidence. For horizontal surfaces, the reaction was estimated to occur at a rate of 1/2 inch per hour for four hours resulting in a total penetration of two inches. For vertical



KEY

- LIQUID SODIUM
- HYDROGEN BUBBLES
- WATER
- SODIUM VAPOR

Figure 12.4-2. Reactor Cavity Cross Section Showing Hydrogen Evolution and Transport to the Pipeway Cells.

surfaces, the reaction was estimated to occur at a rate of one inch/hour for four hours resulting in a total penetration of four inches. However, at the time the above estimates of sodium-concrete reaction rates had been made, a very limited number of supporting experiments had been completed. For this reason, the CRBRP Project later developed a Sodium-Concrete Interactions Development Program to further understand the sodium-concrete interaction phenomena. The experimental data base for sodium-concrete interactions to date includes variations of concrete test article size, sodium temperature, concrete test article configuration (horizontal slabs, cylinder, crucibles, etc.) and type of concrete (prehydrated limestone, dolomitic limestone, calcitic limestone, etc.). The test results generally indicated that sodium-concrete reactions are "self limiting" due to a build-up of a reaction product layer.[6-18] This reaction product layer, which has been characterized as having a sandstone-like texture, builds up as the sodium-concrete reaction progresses and eventually becomes so thick that significant sodium-concrete interactions are prohibited. This is because sodium from the pool must trickle down through the reaction product layer to get to the concrete. This "trickle down" phenomena is illustrated in Figures 12.4-3a-d. The total penetration depths from the test ranged from about zero to 7.5 inches. Penetration rates were observed to be as high as 10 in/hr. However, high penetration rates were also short lived (no greater than 20 minutes).*

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*In several LS-series tests performed at Sandia, several sodium-concrete reactions that exhibited high penetration rates (> 7 in/hr) were short lived due to the fact that the sodium in the test article had been consumed.

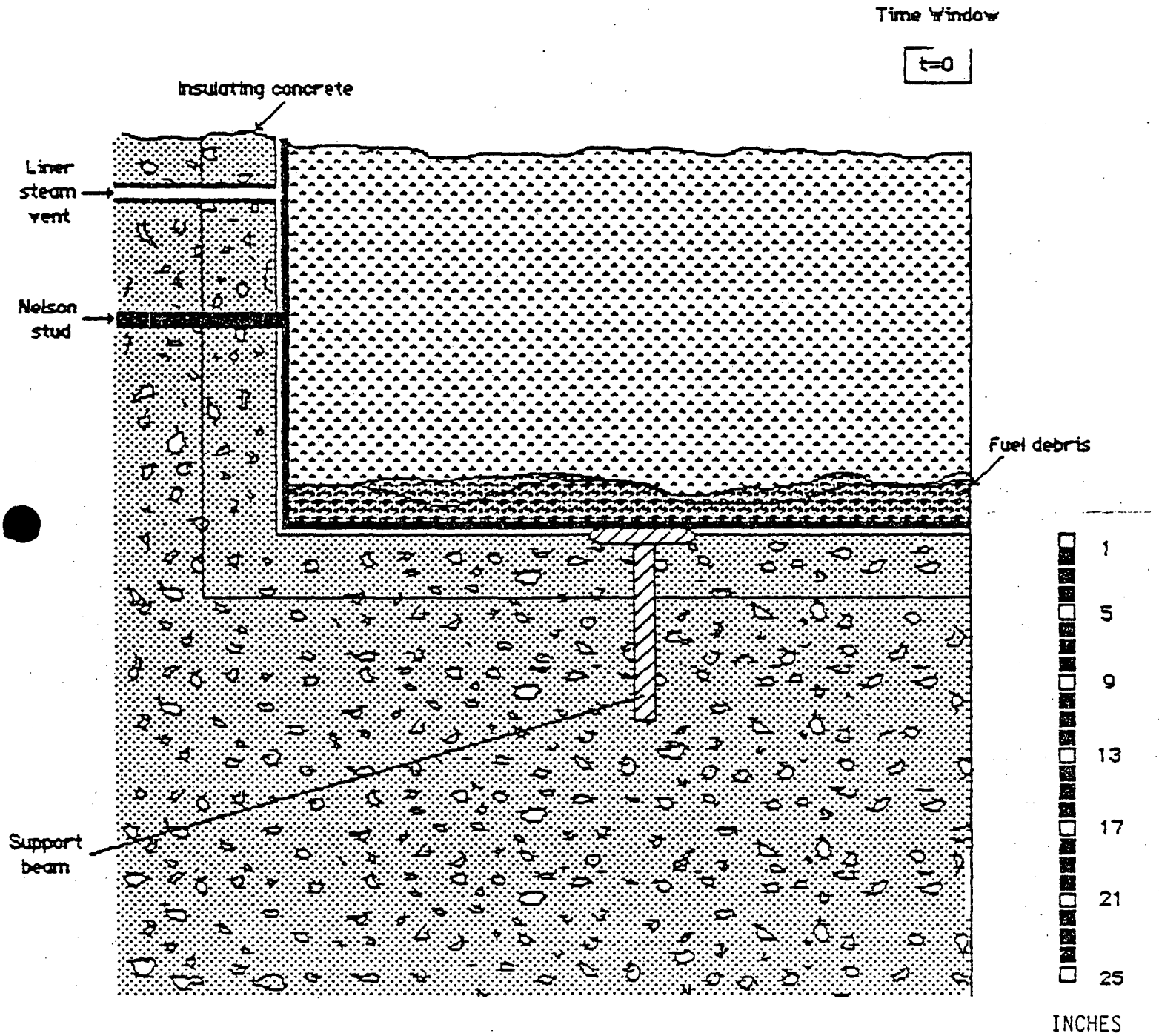

Figure 12.4-3a. Sodium-Concrete Interaction at T = 0 Seconds.

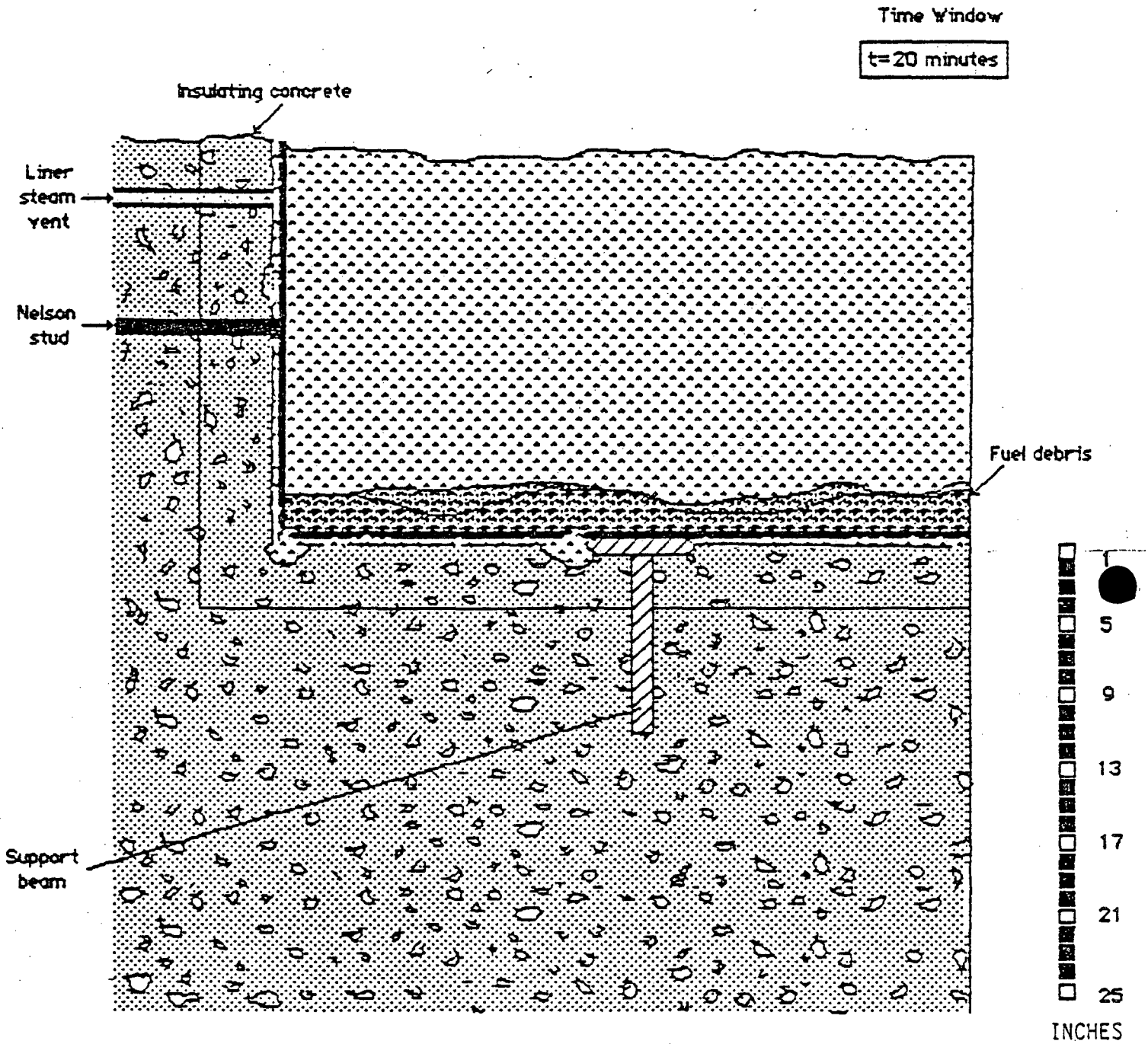

Figure 12.4-3b. Sodium-Concrete Interaction at T = 20 Minutes.

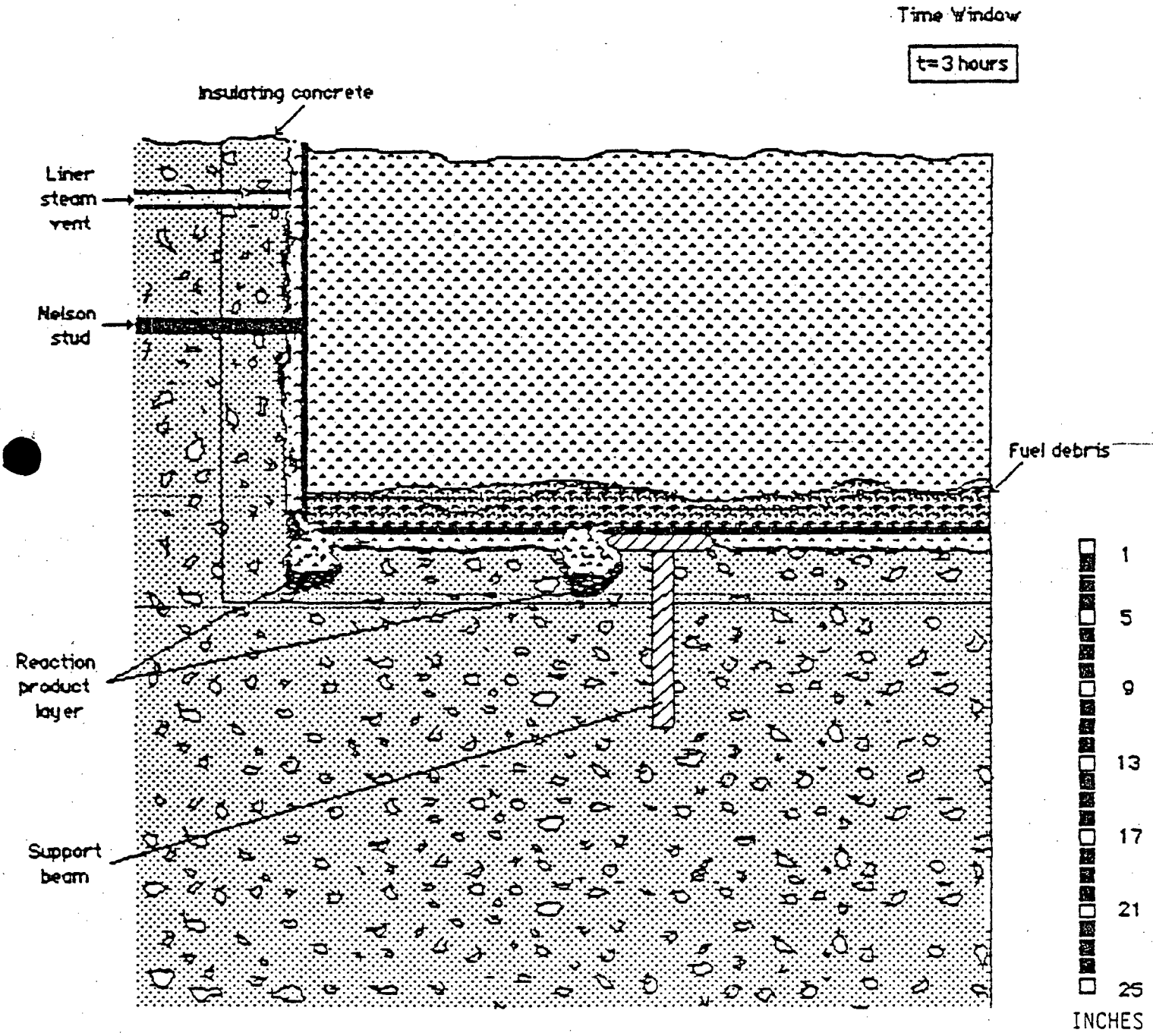

Figure 12.4-3c. Sodium-Concrete Interaction at T = 3 Hours.

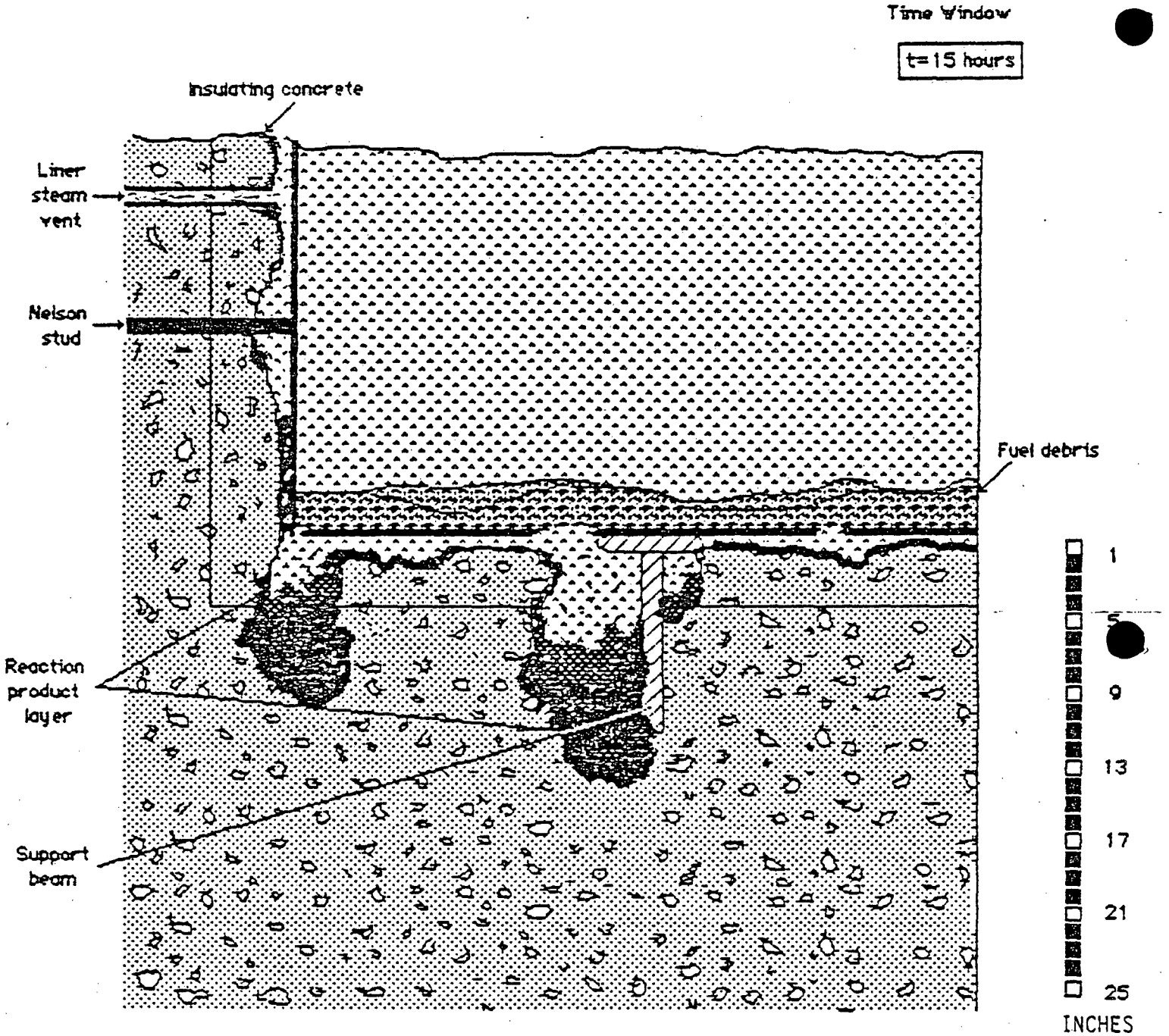

Figure 12.4-3d. Sodium-Concrete Interaction at T = 15 Hours.

The CRBRP Project, upon examining the results of recent sodium-concrete reaction experiments have selected a "reasonable upper bound" reaction rate of seven inches per hour for 20 min followed by one inch per hour thereafter (until the sodium has boiled-off). Since this rate is supported by experimental evidence, it will be accepted as a best estimate of the upper bound penetration rate.

Hydrogen Burning Phenomena

In order for the hydrogen to burn with the oxygen, certain ignition criteria must be satisfied. The ignition criteria are satisfied when the containment oxygen concentration is above 5% and the hydrogen concentration is above 4% and an ignition source is present. Based on Reference 1, this ignition source requirement is met if the hydrogen-nitrogen mixture entering containment is above 1450°F or contains at least 6 g/m^3 of sodium at temperatures above 500°F. The 6 g/m^3 sodium aerosol concentration is believed to be a conservative upper bound. Experimental evidence suggests that the ignition criteria could be met with sodium aerosol concentrations on the order of 1 g/m^3. However, as shown in Figure 12.4-4, the lower criteria for an ignition source would have an insignificant effect on the timing of hydrogen burning.[39]

If the oxygen concentration is above 5% but below 8%, the oxygen-hydrogen reaction would be incomplete and only hydrogen in excess of 4% would be expected to burn. However, for oxygen concentrations above 8%, the hydrogen would be burned completely.

If the ignition criteria are met, hydrogen flowing upward through the pipeway cells would produce a flame upon entering the non-inerted upper

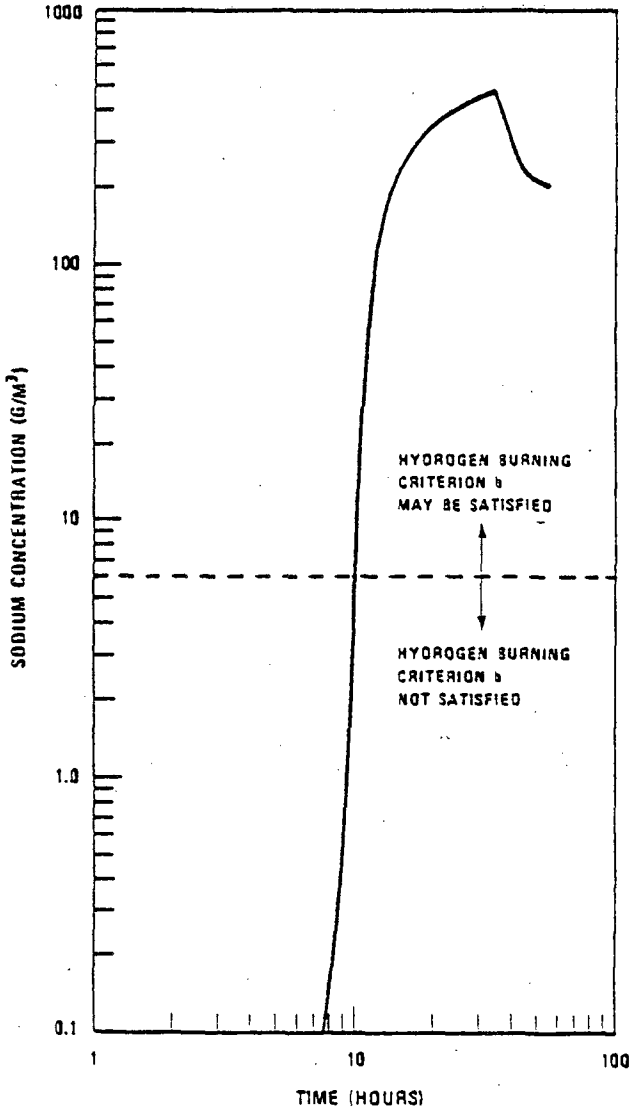

Figure 12.4-4. Sodium Concentration Entering Containment

containment building (the oxygen concentration in air is about 21 volume percent). Initially, the main constituent of the flame would be hydrogen (assuming that most of the nitrogen in the reactor cavity has already been expelled). Because hydrogen has a low molecular weight and consequently a high volumetric flow rate per unit mass, the nearly pure hydrogen flame was predicted to exhibit laminar flow.[1] However due to inherent instabilities and turbulence from upstream geometry effects, laminar flow would not be expected to last for very long. The flame would likely go from a laminar regime into a transitional turbulent regime. Turbulent flow will result in a greater entrainment of surrounding atmosphere which will tend to dilute the concentration of hydrogen in the flame. In addition, a laminar flame that leaves the nozzle will likely become turbulent at some height above the nozzle, depending on the Reynolds number.

In Reference 1 it was determined that the total height of a laminar hydrogen flame is primarily a function of the hydrogen volumetric flow rate whereas the height of a turbulent hydrogen flame was more a function of the nozzle diameter. On this basis, a turbulent hydrogen flame from a single nozzle with a diameter of 12 inches was predicted to be about 80 pipe diameters high or 80 feet. A flame height of 89 pipe diameters was observed in the auto-ignition test performed at the Hanford Engineering Development Laboratory. However the diameter of the nozzle used in this test was only 0.18 inches. In a 1970 study by the U.S. Bureau of Mines, a hydrogen diffusion flame for a 30 inch diameter vent was observed to be only about 25 pipe diameters high or 63 feet for subsonic flow.[19]

If it is assumed that flame height is a function of pipe diameter, as in Reference 1, and that the flame height (in pipe diameters) linearly decreases for increasing pipe diameters, then a relationship like that shown in Figure 12.4-5 could be postulated for the two data points presented above. For such a relationship, the flame height for a 12" diameter nozzle used in CRBRP would be 65 feet. Since only two data points are being considered here, it is deemed inappropriate to draw any definitive conclusions regarding the relationship between flame height and pipe diameter. Although there is some disagreement as to whether hydrogen diffusion flame heights are more a function of pipe diameter than hydrogen concentration or flow rate, observations of hydrogen diffusion flame in the chemical industry generally support flame heights on the order of 80 pipe diameters for subsonic flow and 100 pipe diameters for supersonic flow.[37] On this basis it will be assumed in this report that hydrogen diffusion flame heights on the order of 65-80 pipe diameters are not unreasonable for pipe diameters of one foot.

It should be noted, however, that the above discussion applies only to a nearly pure hydrogen diffusion flame. In the CRBRP core-melt scenario, sodium will become entrained in the hydrogen flowing from the reactor cavity to the RCB. Eventually, the flow will be dominated by sodium rather than hydrogen. The flame height predicted in Reference 1 for the sodium dominated flame is shown by the late time period predictions in Figure 12.4-6. Note also that the early 80 foot hydrogen flame was predicted to last for about 5 hours.

Because of these flame conditions, the effect of hydrogen burning on the structural integrity of the containment has also been considered.

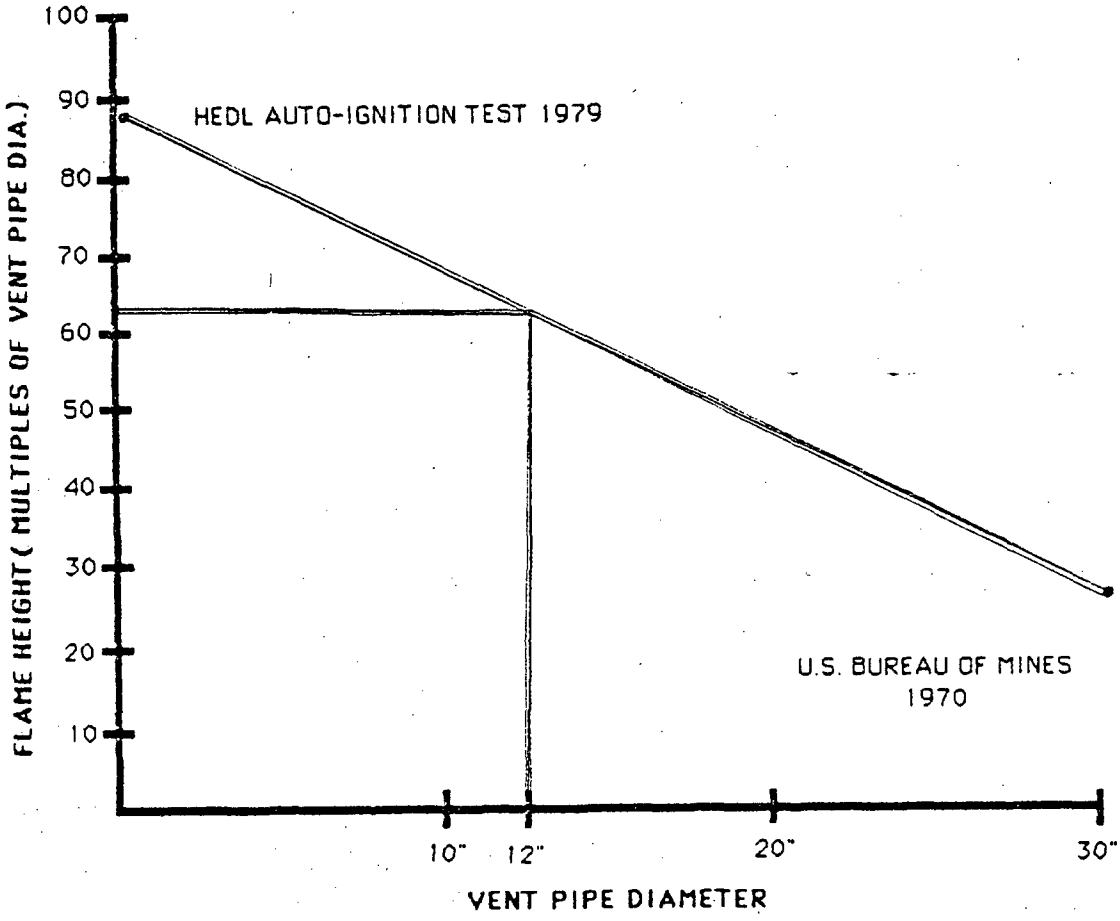

Figure 12.4-5. Flame Length: HEDL Auto-Ignition Test and U.S. Bureau of Mines Study.

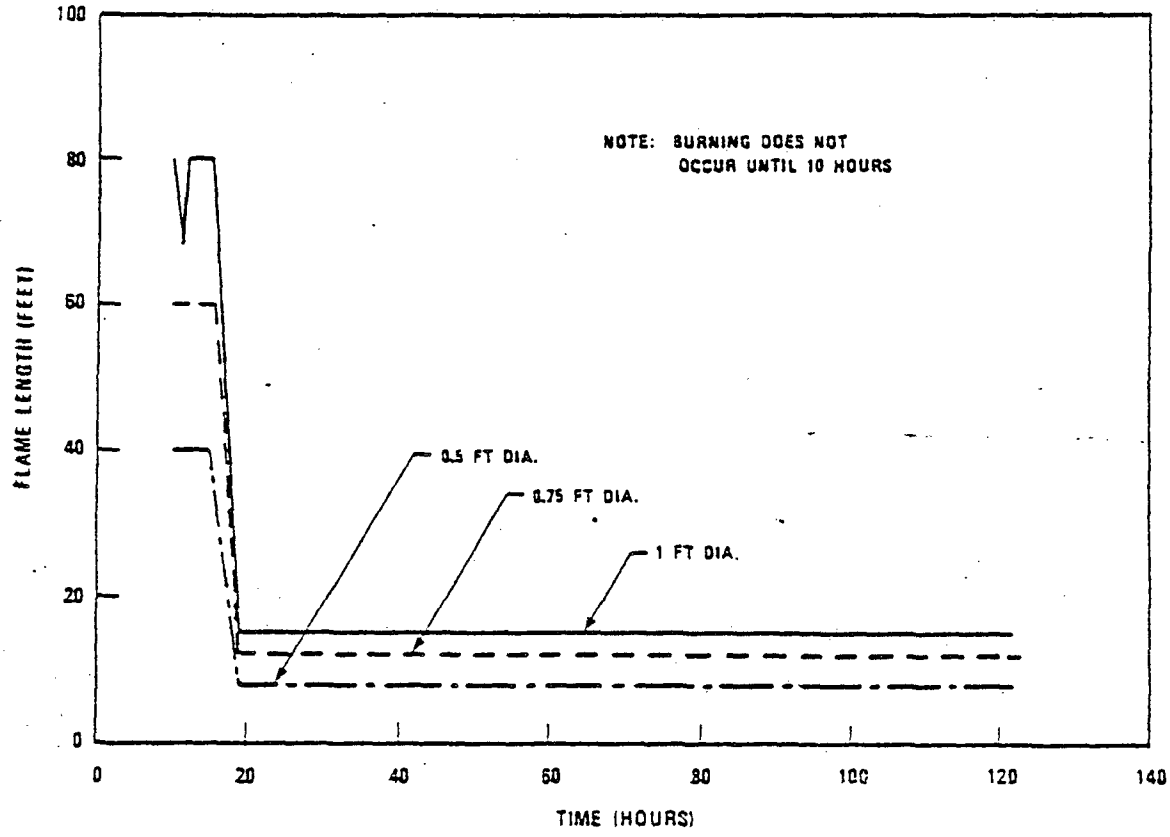

Figure 12.4-6. Flame Length as a Function of Nozzle Diameter. (Reference 1)

As shown in Figure 12.4-7, a flame 80 ft high would not come near the containment steel shell. The temperature of the gases produced from H_2-O_2 reactions in the vicinity of the flame was estimated to be about 3500-4000°F if it is assumed that there is no cooling of these gases and that only a stoichiometric volume of air was mixed with the hydrogen.[1] Based on experimental evidence, the temperature of the flame would be greatest about halfway up the length and decrease with flame height.[19]

In Reference 1, it was determined that the gases rising from the flame would be mixed with the surrounding atmosphere and would not have an adverse effect on the containment steel shell integrity. However, Figure 12.4-7 also shows that the top of the turbulent hydrogen flame would be at the same elevation as the polar crane. A study of the damage caused by the hydrogen flame produced during the Three Mile Island (TMI) accident indicated that the thermal damage in the polar crane region of the TMI containment was limited to charring of an electrical cable.[29] However, it is believed that the TMI hydrogen flame lasted for less than a minute,[38] whereas the CRBRP turbulent hydrogen dominated flame was predicted in Reference 1 to remain at a height of 80 feet for about 5 hours. The effect of preferentially heating the polar crane for a period of about five hours (if the crane were to be postioned over the vent nozzle) is addressed in Appendix B, Section 5. Of particular concern is the effect of the differential thermal expansion of the polar crane girder and the subsequent potential for slumping of the polar crane.

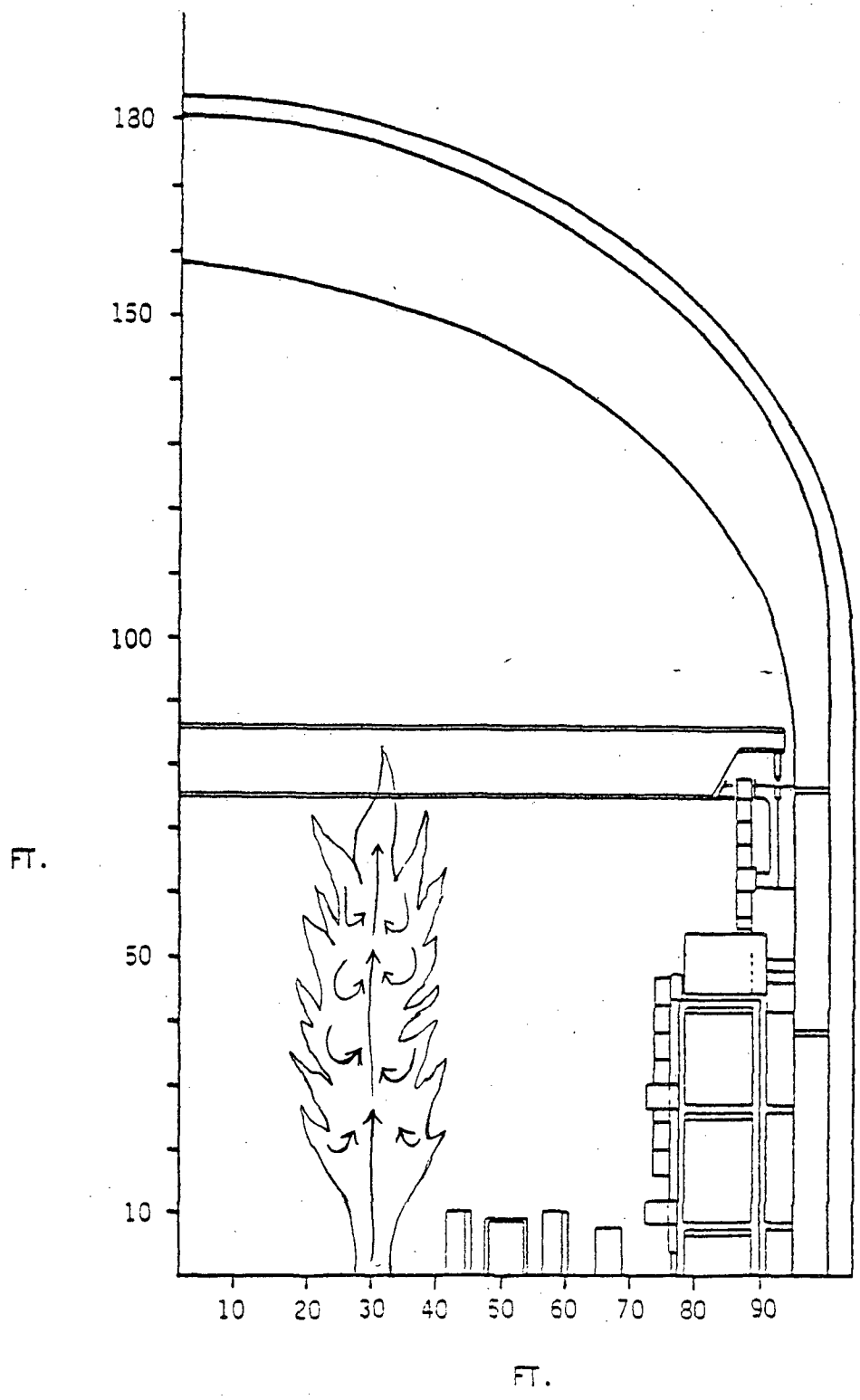

Figure 12.4-7. Containment Cross Section Showing Distance of Hydrogen Flame from Containment Boundary.

In reviewing this potential failure, it was necessary to consider whether or not the polar crane will be directly over one of the reactor cavity vents. The discussion of the event in Section 12.4 assumes that the polar crane girder is, in fact, directly over the vents. However as shown in Figure 12.4-8 the anticipated parked position of the crane is outside the boundaries of the hydrogen flame. Although, review of procedures did not indicate that there was a required position for parking the polar crane, examination of RCB layout drawings did indicate that the ladders to the polar crane were located on top of the electrical, instrumentation, and control cubicles. Thus, the ends of the anticipated parking position represent the location of the ladder to the polar crane. Even if it is assumed that there is an equally likely chance that the polar crane would occupy any given position, the probability that any end of the polar crane comes within the boundary of the hydrogen flame would be less than two-thirds.

Although, containment failure as a result of the mechanisms described in Appendix B, Section 5 cannot be ruled out at this time, it is deemed unlikely that the conditions are sufficient to cause containment failure. Even if indications were that a problem might exist due to preferential heating of the polar crane, Reference 1 points out that the hydrogen flame height could be easily reduced by decreasing the effective diameter of the vent line(s). One way of doing this is to increase the number of nozzles downstream of each rupture disk such that the combined area is equivalent to the original nozzle area (i.e., simply adding more nozzles without decreasing the nozzle diameter would not decrease the height of a turbulent hydrogen flame). The CRBRP design utilizes two 12-inch diameter reactor

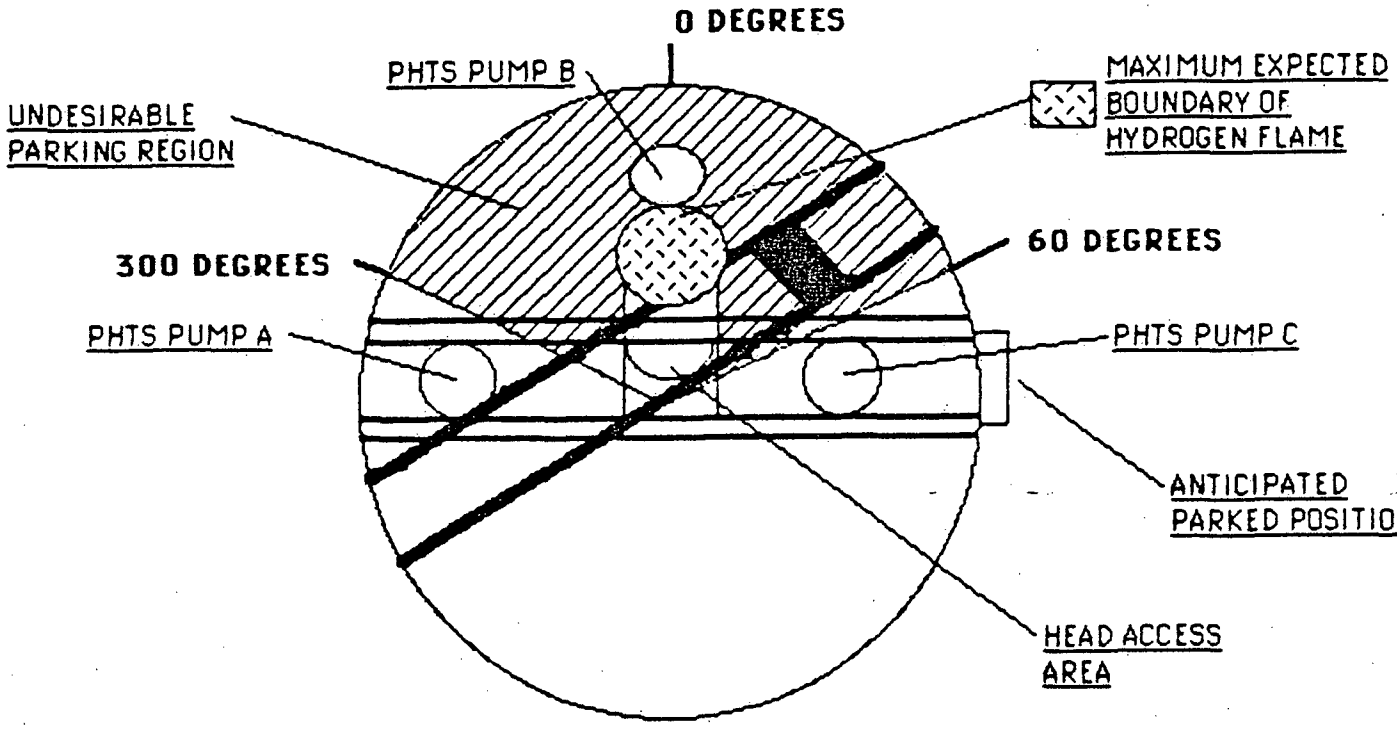

Figure 12.4-8. Map of RCB Showing Hydrogen Flame Boundary.

cavity to RCB vent lines (nozzles); each nozzle located downstream of a rupture disk and normally open block valve. For each nozzle, the maximum height for a turbulent hydrogen flame based on the Reference 1 assessment, would be about 80 feet. The effect of adding more nozzles can be best explained by first considering the two 12-inch diameter RCB vent lines each having an area of 0.785 ft^2 as a single vent line with an area of 1.57 ft^2. It can be observed from Figure 12.4-6 that the flame height is assumed to increase approximately twenty feet for every 0.25 ft increase in nozzle diameter based on the assumption that the flame height is a constant 80 pipe diameters. A single nozzle with an area of 1.57 ft^2 would have a diameter of 1.41 ft, and thus a potential flame height of 113 feet. The effect of increasing the number of nozzles downstream of each rupture disk is illustrated in Figure 12.4-9. If the total area of the single vent line is distributed between two nozzles the flame height would be reduced to 80 feet. However, if the total area of the single vent line is distributed between three nozzles, the flame height would drop to about 65 pipe diameters. Four nozzles with a total area of 1.57 ft^2 would reduce the flame height by about 50% to 57 feet.

Hydrogen Detonation Potential

When the hydrogen ignition criteria discussed previously are no longer satisfied, hydrogen would gradually accumulate in the reactor containment building.

The detonation limits of hydrogen in air for ambient conditions has been estimated to be between 18.2% and 58.9% by Breton. These values were later quoted by Lewis and Van Elbe and have been generally accepted.

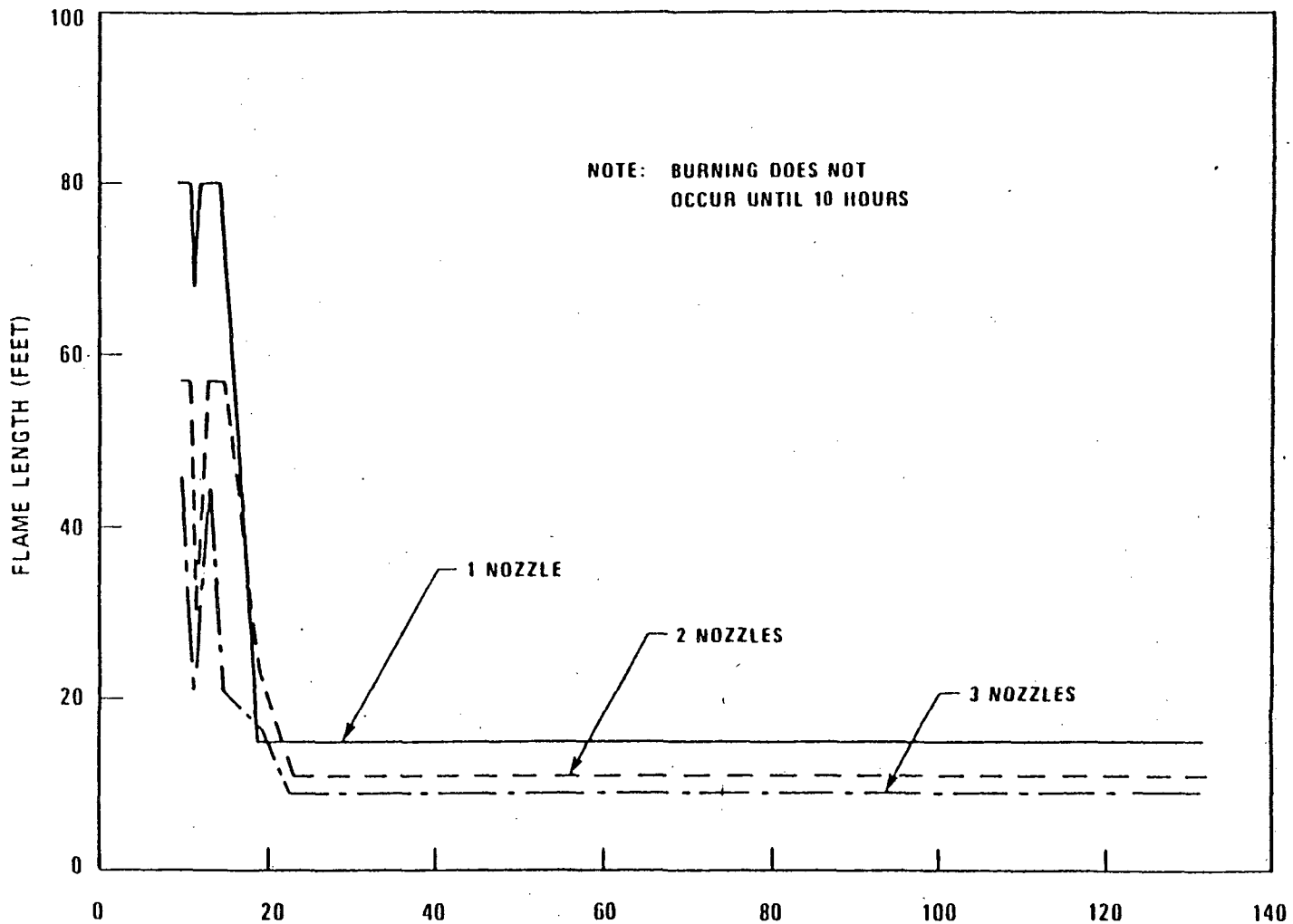

Figure 12.4-9. Dependence of Flame Length on Number of Nozzles. ($0.7853\ Ft^2$ Total Area) (Reference 1)

However, a 15% hydrogen mixture in air with moisture content below 0.005% has been found to support a stable detonation.[23] In addition, detonation has been observed to occur in air at hydrogen concentrations as low as 13.8%.[28] Although there appears to be little data on the effect that temperature and pressure have on detonation limits, the data that is available seems to suggest a very small widening of the detonation limits with increasing temperature and pressure.[26] Thus, based an available data, it appears that the hydrogen detonation limits, under CRBRP core-melt conditions could be below the generally accepted 18.2% lower limit.

Hydrogen meters, located in the upper containment dome, are provided in the CRBRP design to measure the hydrogen concentration in containment following a core-melt accident. According to Operator Procedure BDB-10-1,[34] once the operator has verified a core disruptive accident he/she must monitor several key parameters, including containment hydrogen concentration, every 15 minutes unless the rate of increase indicates a more frequent period.[29] If the hydrogen concentration at any time exceeds 6%, the operator would be instructed to initiate vent and purge. As the containment is vented the hydrogen flow rate from the reactor cavity to the upper reactor containment will increase due to the depressurization of the upper reactor containment building. This will cause an initial increase in the reactor containment building hydrogen concentration. However, this hydrogen concentration would be below the detonable limit as long as venting is initiated when the hydrogen concentration is not significantly above 6%.[1] The hydrogen concentration in containment will decrease as the hydrogen reacts with the oxygen brought into the upper RCB during containment purge.

If venting is initiated when the hydrogen concentration is significantly above 6%, the hydrogen concentration could potentially reach detonable limits as the hydrogen flow from the reactor cavity is increased. The oxygen introduced into the RCB from purging could then allow a detonation to occur.

Another potential mechanism for detonation would exist if the hydrogen concentration in containment approached detonable limits before the ignition criteria were satisfied. This could theoretically occur if the liner failed very early in the scenario and the sodium pool remained substantially subcooled. (The 6 g/m^3 of sodium required as an ignition source does not necessarily require that the sodium pool reach saturation temperatures). However, several phenomena would tend to offset this effect. First, the heat added to the sodium pool by the sodium-concrete reactions following liner failure would drive pool boiling and thus, the early presence of an ignition source. Second, since liner failure is expected to be localized the hydrogen generation rate may be less than that assumed in Reference 1. Third, the duration of hydrogen generation would be relatively short due to the self limiting nature of sodium-concrete reactions discussed previously. Finally, when the hydrogen concentration exceeded 6%, the operator would be required to initiate vent and purge operations, albeit earlier than desirable.

A traditional concern related to hydrogen detonation is that of stratification. Hydrogen stratification has been postulated when containment conditions allow the formation of local "pockets" that contain hydrogen concentrations well in excess of those in the global containment environment.

In Reference 1, hydrogen stratification was not predicted to occur early in the core-melt scenario for CRBRP due to mixing of the containment atmosphere as a result of hydrogen flame turbulence. Rapid mixing would be more likely to cause the hydrogen concentration to rise uniformly throughout the entire upper containment volume, whereas slow mixing could cause high concentrations of hydrogen to develop locally. Several studies that have recently been performed for light water reactors indicate that "the rate of mixing depends strongly on the presence of forced convection and the rate and location of hydrogen release."[20,21]

Since the CRBRP upper RCB containment is not compartmentalized, as in many LWRs, it seems unlikely that local hydrogen accumulation could occur. Thus, location does not seem to be of importance for the CRBRP containment. In addition, the forced convection turbulence of the hydrogen flame seems adequate to provide mixing of the containment atmosphere such that hydrogen stratification is precluded. If the sodium has boiled off, hydrogen will be generated at a much slower rate.* However in the absence of sodium as an ignition source, the hydrogen concentration could rise above 4% without being consumed in reactions with oxygen. If for this scenario, or any other scenario, the hydrogen monitors in the upper containment dome detected hydrogen concentrations in excess of 6% the operators would initiate vent and purge in accordance

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*Hydrogen is generated after sodium boildry from the following reactions: $3Fe + 4H_2O = Fe_3O_4 + 4H_2$ and $2Cr + 3H_2O = Cr_2O_3 + 3H_2$

with Operating Procedure BDB 10-1.[34]* The mixing action of the vent and purge system is expected to preclude hydrogen stratification later in the scenario.

In summary, hydrogen detonation due to stratification (either local or global) is not expected because the generation of hydrogen would be accompanied by a hydrogen flame which would mix the containment atmosphere. In the event that hydrogen stratification were to occur, it could be sensed by the H_2 monitors and, if necessary, venting and purging actions could be taken.

Criticality Potential in the Reactor Cavity

The results of three assessments of the criticality potential in the reactor cavity are reported in Reference 1. The three assessments reviewed stages during the accident including a boiling sodium pool, a dry debris bed, and fuel distribution after a few months. In all cases the $k_{effective}$ is approximately 0.51 or lower. Even with uncertainties, the chance of attaining critical conditions after vessel penetration appears negligible. On this basis ex-vessel criticalities in the reactor cavity are ruled out.

Fission Product Release Mechanisms

The initial release of fission products includes those materials which are not expected to be retained to an appreciable extent. These include the noble gases and more volatile elements such as cesium.

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*Hydrogen concentration is only one of several criteria that would require vent and purge system operation.

During the sodium boil-up phase in the reactor cavity, radioactivity trapped in the sodium pool enters the RCB atmosphere as the sodium pool boils. At this time, the halogens and remaining volatile elements are released. It is reported in CRBRP-3[1] that the fission products would probably be released as a gas from the sodium pool and later condensed in the RCB atmosphere allowing time to agglomerate before being vented. The non-volatile fission products will form particulates in the sodium with a significant partitioning of the fuel and sodium.

After evaporation of the sodium pool in the cavity, any further release could occur from surface vaporization, particle convective levitation, and gas sparging.[1] The first two mechanisms should release little radioactivity. The latter mechanism, involving steam and CO_2 bubble-up through the sodium pool, could release about 26 g of plutonium over a several month period to the RCB. Accounting for 99% filter efficiency and aerosol fallout and plateout, the plutonium released to the atmosphere over a several month period could be on the order of one tenth of a gram. This assumes, of course, operation of the containment cleanup system and venting and purging of the RCB to maintain the hydrogen concentration to acceptable levels.

Extended Liner Integrity Scenario

In the Reference 1 base case analysis it was assumed that the reactor cavity floor liner failed at the instant the fuel debris entered the reactor cavity. However, as explained previously in this section, liner failure is not necessarily expected to be immediate, and the liner could possibly stay intact for an extended period of time after the fuel

debris has entered the reactor cavity. If the liner integrity is maintained for an extended period of time, hydrogen production, and thus venting and purging, would likewise be delayed. Since the heat from the sodium-concrete reactions would not be added to the sodium pool, the time to boildry would be greater. In Reference 32 it was estimated that the time to sodium boildry would increase from 130 hours for the Reference 1 base case to about 700 hours for the extended liner integrity case.

Although the conditions in the upper RCB are less severe for the extended liner integrity case than for the Reference 1 base case, the integrated heat load to the reactor cavity structures is much greater. A study performed by the CRBRP Project on extended liner integrity indicated that because of the extended heat loads to the reactor cavity structures, partial wall failure between cell 105 and the reactor cavity could occur.[32] (See Figure 12.4-10)

Since cell 105 contains the water vented from behind the reactor cavity, a path between cell 105 and the reactor cavity could result in extensive sodium-water reactions. If this were to occur, the hydrogen in containment could rapidly exceed 6% and the addition of non-condensibles could exceed the capability of the vent/purge system to maintain a negative pressure in containment.[32]

Perhaps a more mechanistic way for sodium to come into contact with cell 105 would be as a result of pipeway floor structural failure.[32] Failure of the pipeway floor could occur suddenly, causing all of the sodium in the pipeway cell floor to come into contact with cell 105 instantly.

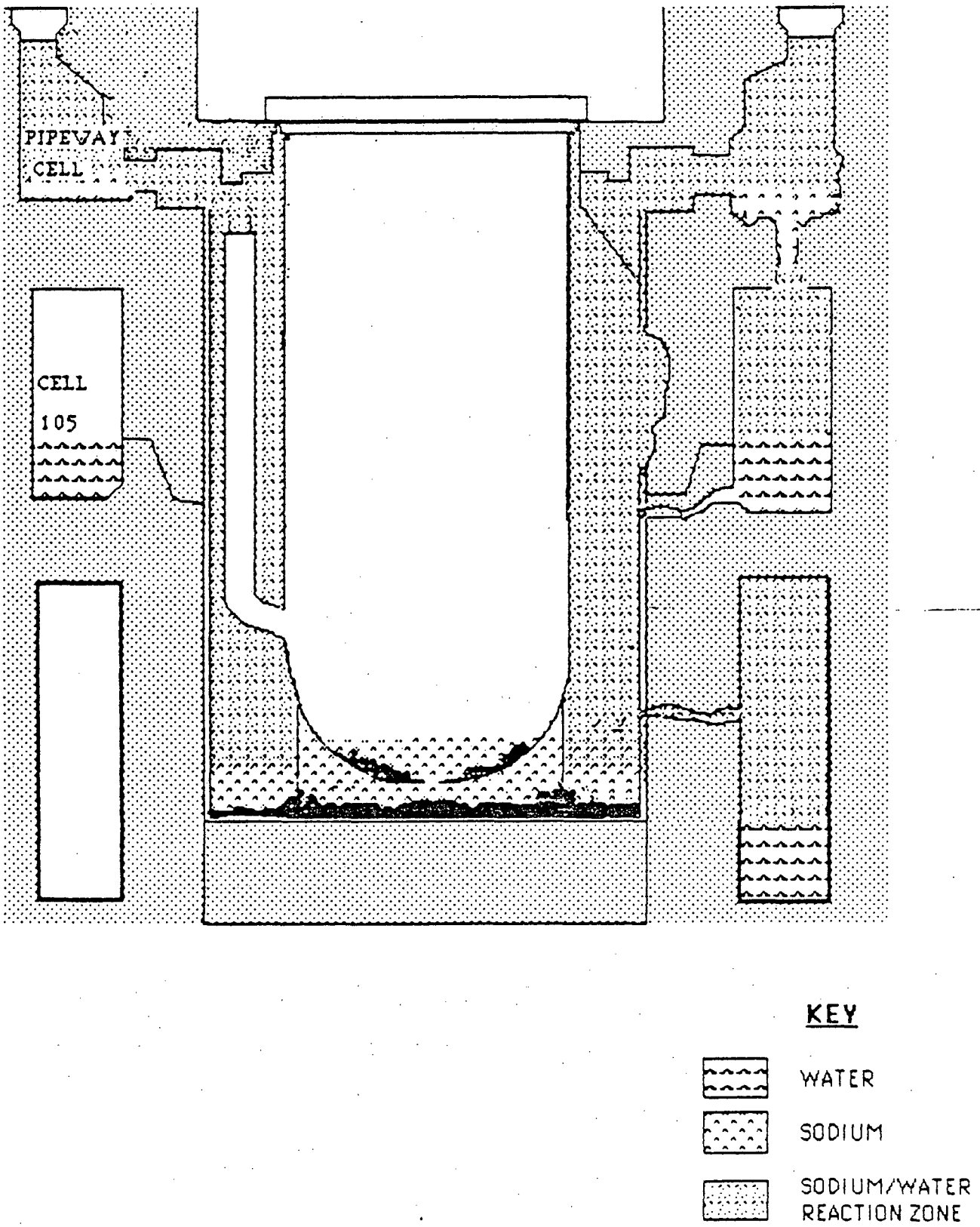


Figure 12.4-10. RCB Cross Section Showing Potential Structural Failure Modes for the Extended Liner Integrity Scenario.

However, since the pipeway cells are designed to drain condensed sodium back into the pipeway cell, the sodium pool would only be about 2 in. deep for a total mass of about 3000 lbs per cell.

Another study looked at the potential for degraded concrete rubble on the sides of the reactor cavity walls to fall into the reactor cavity sodium pool.[32] The study made varying assumptions about the amount of rubble falling within the pool over a given time period. In general it was concluded, that the existing beyond-the-design-basis features could accommodate any one of the above scenarios for the conditions:

- Sodium leakage to cell 105 from the reactor cavity no greater than 10,000 lbm/hour (27 gpm),
- Sequential massive failures of all three pipeway cell floors all within a time period of no less than 10 hours, and
- Concrete rubble from the reactor cavity wall up to the entire four inches of light weight insulating concrete falling into the reactor cavity sodium pool in a short time period (3 hours).

In summary, in the event that the reactor cavity liner remained intact for an extended period of time, it would be possible for conditions more severe than those discussed in Reference 1 to occur. However, a scenario involving loss of containment integrity would require that very pessimistic assumptions be placed on the extent of structural damage induced by extended liner integrity.

Overview of the Upper Reactor Containment Building Response

Figure 12.4-11 summarizes the response of the upper containment building for the Reference 1 base case. In the base case scenario the reactor cavity floor liner is assumed to "vanish" allowing for early production

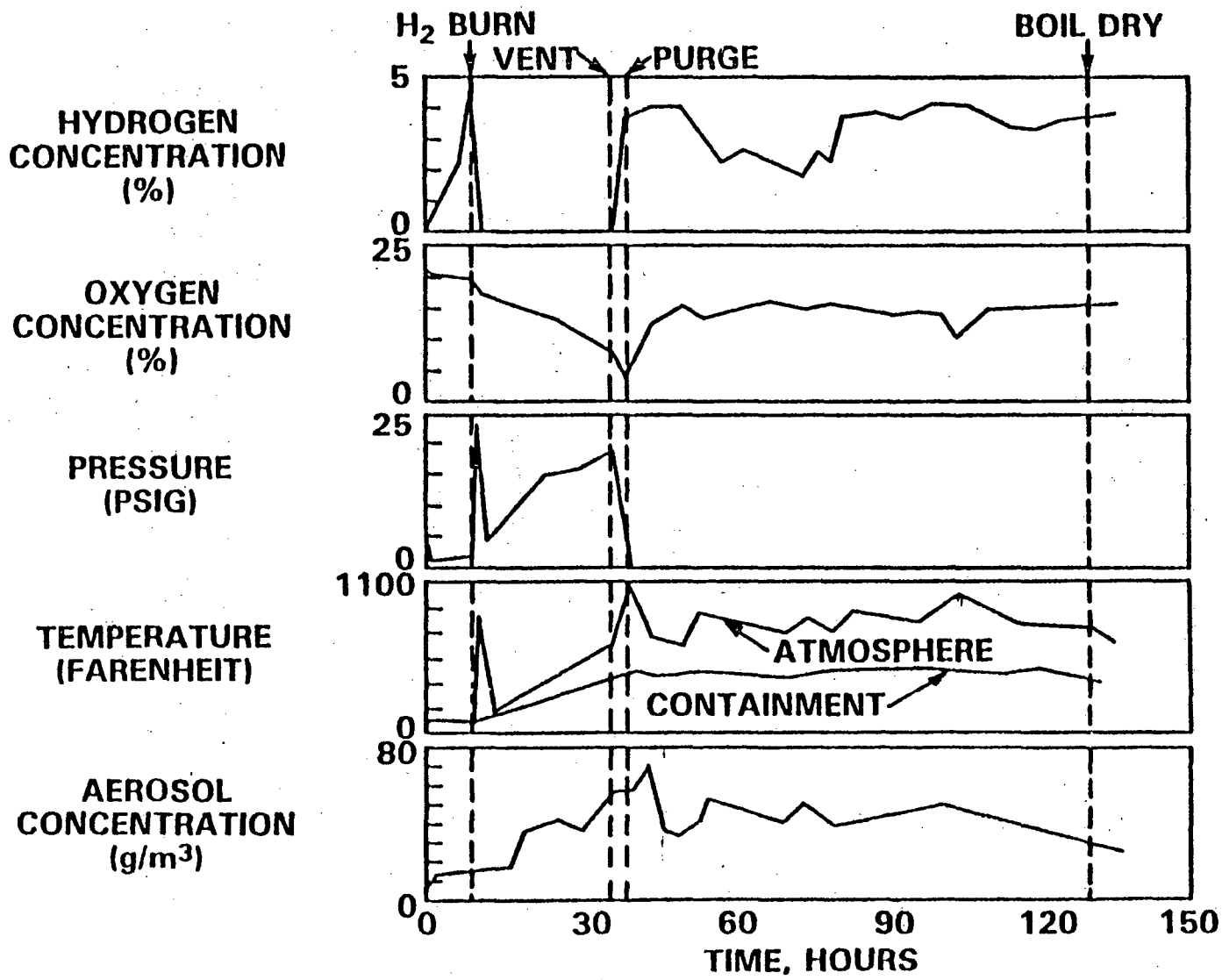


Figure 12.4-11. Upper RCB Response to a Core-Melt Accident.

of hydrogen. However, as previously discussed the reactor cavity floor liner would not be expected to vanish, and the hydrogen curve shown in Figure 12.4-11 could shift to the right. The oxygen concentration curve would then shift accordingly since, when the hydrogen concentration exceeds 4%, the hydrogen would begin to burn (assuming the ignition criteria are met). The pressure spike of about 22 psig is attributed to the assumption that the hydrogen burns instantly. Since the failure pressure of the containment is in excess of 30 psig, the containment integrity is maintained (see Table 12.3-1). The burning of the hydrogen also results in a sharp rise in the bulk containment atmosphere temperature and a slower rise in the containment steel shell temperature.

At 36 hours the annulus cooling system is started to lower the containment and confinement temperatures. Since the oxygen concentration has gone below 10%, venting and purging are initiated. The hydrogen concentration is low due to the fact that it is being burned by oxygen above 8% (above 5% for incomplete burning). Once the oxygen is depleted, the hydrogen concentration will rise unless a new source of oxygen is introduced through purging. During venting the hydrogen concentration in the RCB increases as the hydrogen flow from the reactor cavity increases.

Initially, the aerosols generated (primarily NaOH) would remain airborne within the upper RCB. Gravitational settling and plateout will occur and eventually the RCB operating floor will have several feet of aerosols on top of it. When venting is initiated some of the aerosols will be absorbed by the containment cleanup system.

12.5 THE CONTAINMENT-RESPONSE EVENT TREE

The containment-response event tree, shown in Figure 12.5-1, is used to delineate the response of the containment to the conditions represented by the core-response event tree end conditions in Chapter 11. The end conditions of the containment-response event tree will be placed into categories corresponding to the relative magnitude of radiological consequences associated with each sequence.

12.5.1 Definition and Description of Top Events

IVC: In Vessel Cooling

Definition

Event IVC success criteria is met if the necessary conditions to keep the damaged core coolable and within the confines of the reactor vessel have been satisfied. The purpose of including this top event as part of the containment-response event tree is to illustrate the fact that core-response event tree end states designated "CD, in-vessel" bypass the containment-response event tree altogether. As explained earlier in Section 12.4.1, core-response event tree end states designated "CD, in-vessel with head leakage" are not being considered since the probability of occurrence is 2×10^{-6} or lower for all HCDA initiators and the consequences of this event would be less than an event with reactor vessel bottom failure.

Dependencies

Although there are no dependencies between the IVC event and the other containment-response top events, there is a dependency between the IVC event and the type of HCDA initiator. Some unprotected HCDA initiator sequences could potentially lead to long term, stable decay heat removal from the damaged core within the reactor vessel. However, long term core coolability is unlikely for protected HCDA sequences since they will very likely involve an absence of sodium around the core at the time of fuel disruption.

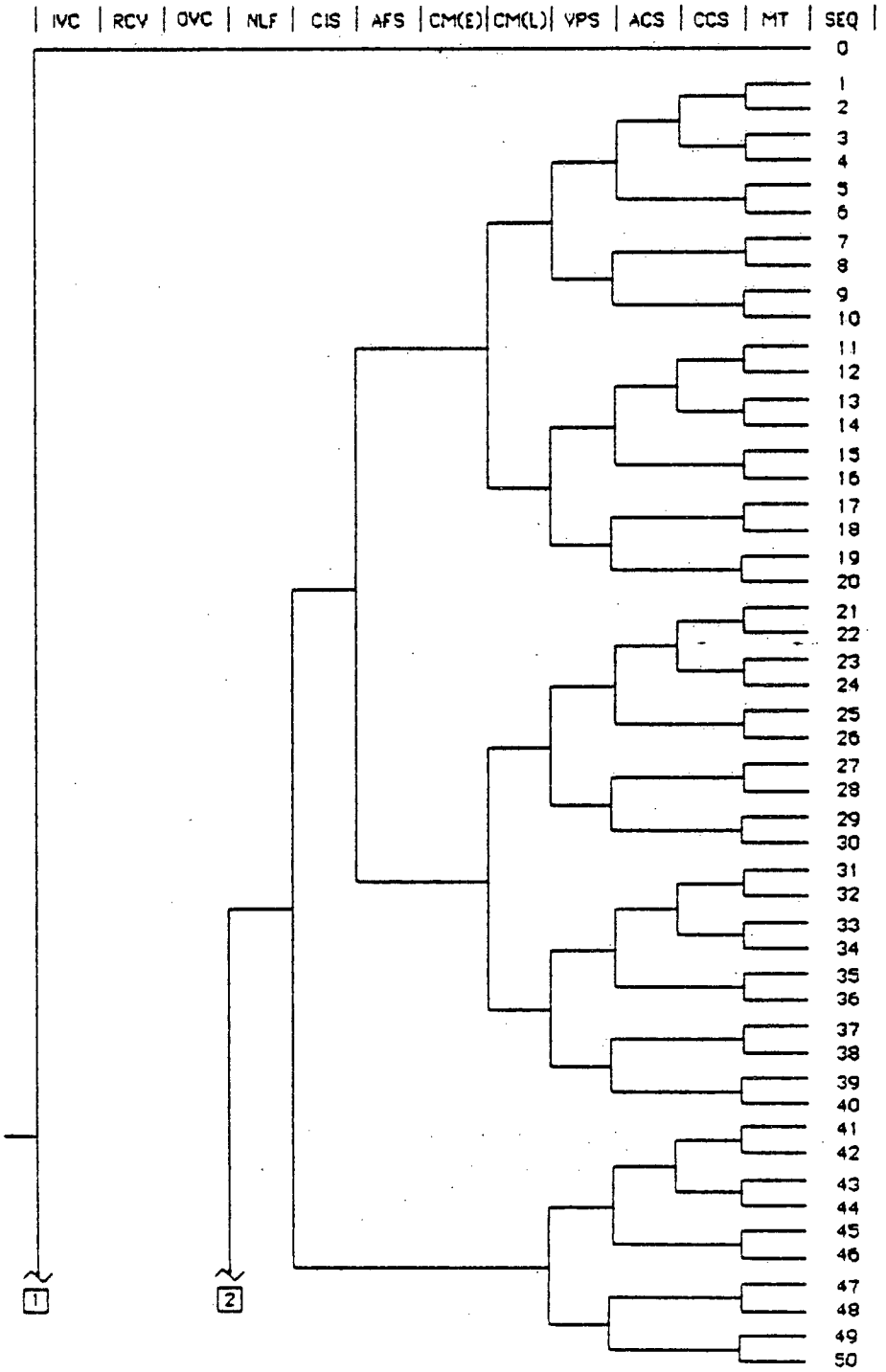


Figure 12.5-1. Containment-Response Event Tree.

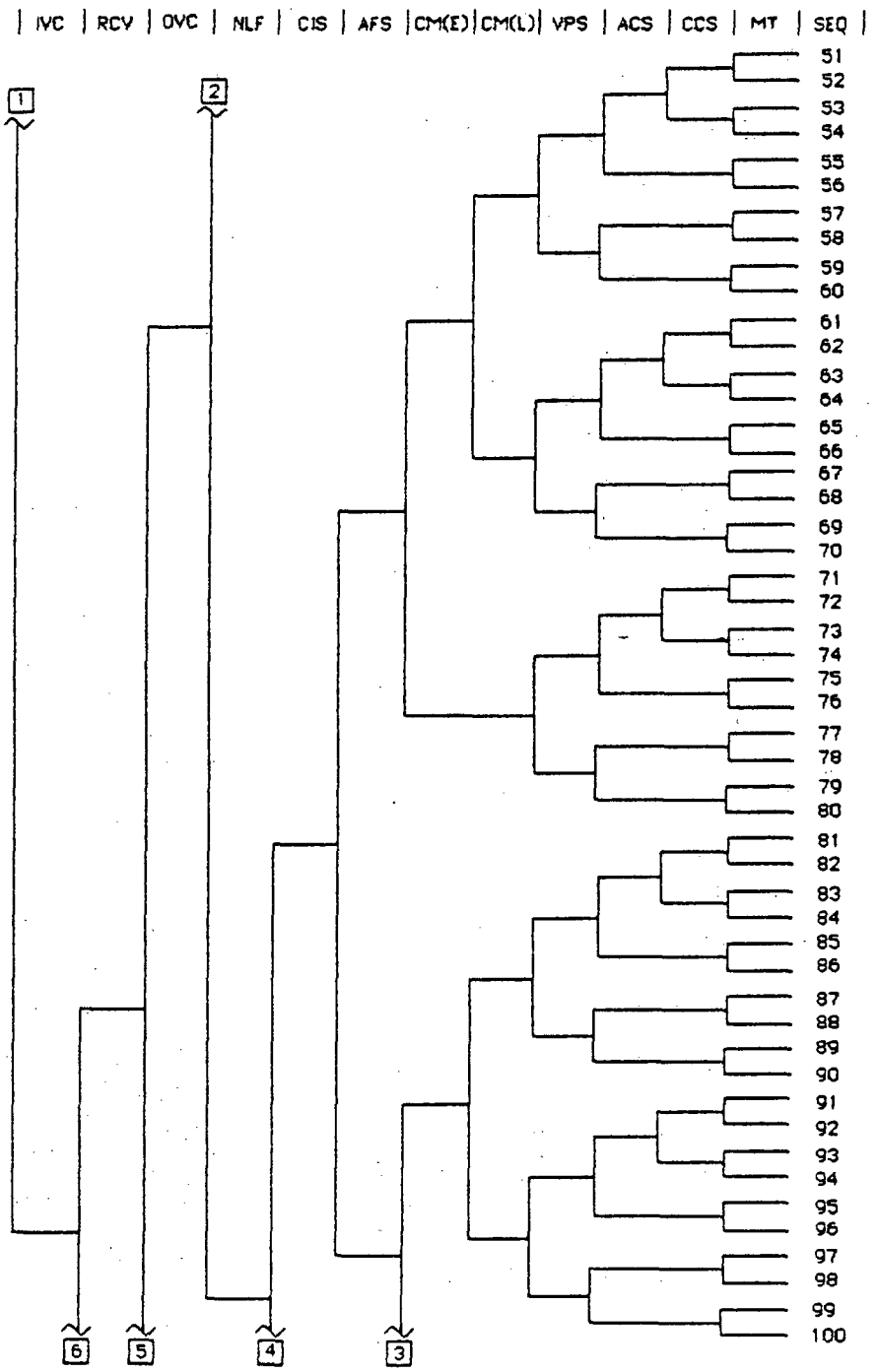


Figure 12.5-1 (Continued)

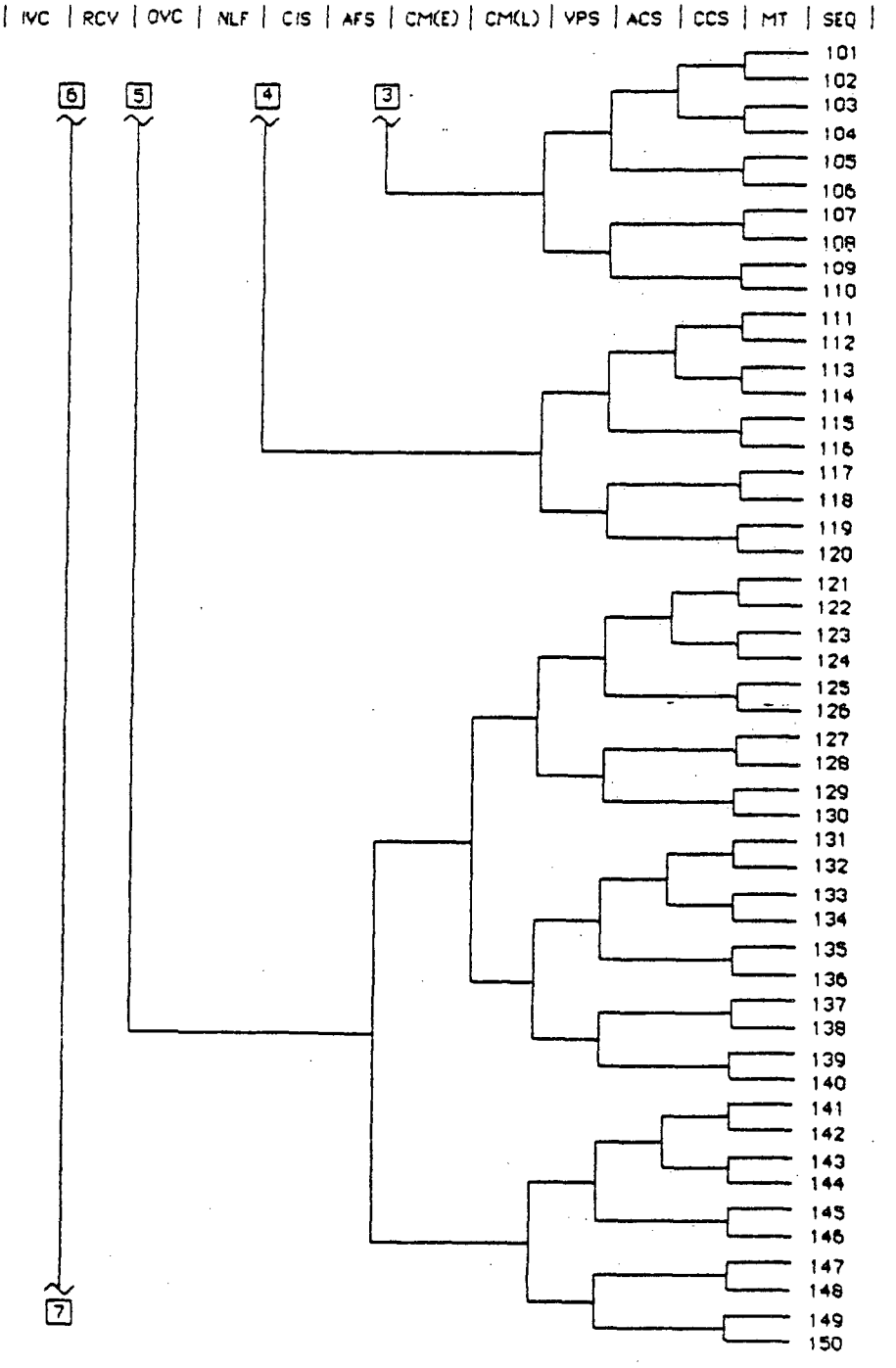


Figure 12.5-1 (Continued)

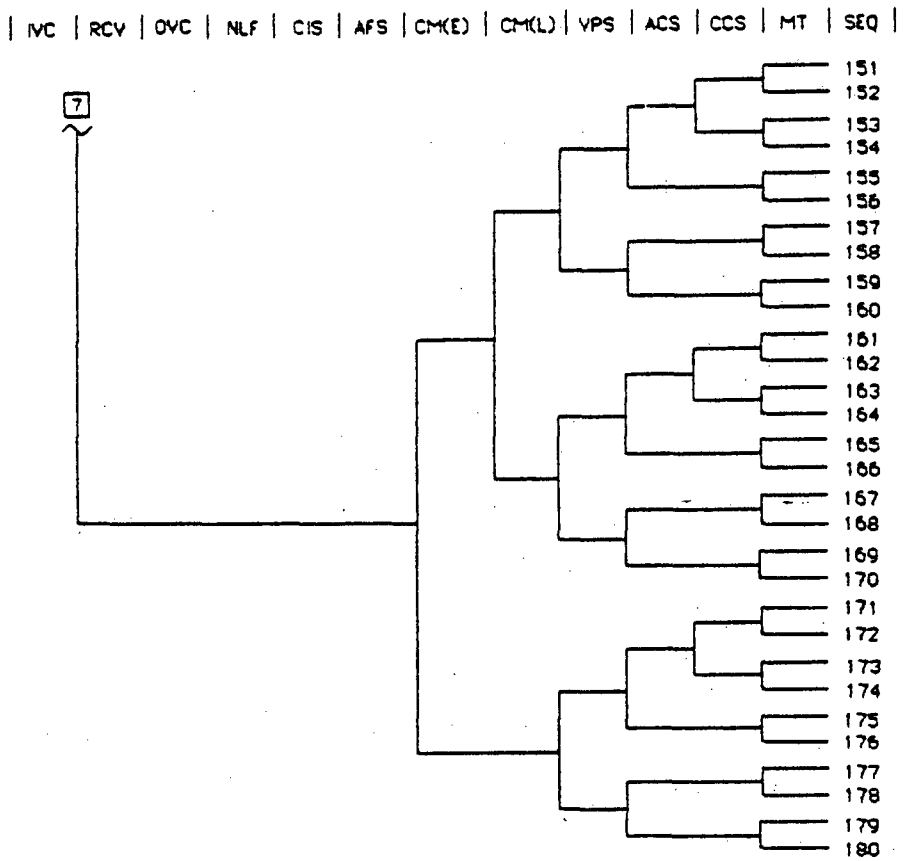


Figure 12.5-1 (Continued)

There are three core-response event tree sequences that fall within the IVC category. The end conditions for all three sequences represent a state of long term coolability of the damaged core. The assumption is made here that for core-response sequences 1 and 6 of the core-response event tree in Chapter 11, there is no radionuclide release outside of the primary system envelope.

RCV: Reactor Cavity Vent

Definition

Success for this event is attained if at least one of the two vent paths between the reactor cavity and the upper containment open within twenty hours of the time when core debris and sodium have spilled into the reactor cavity. The reactor cavity vent provides a release path to the upper containment for the gases generated in the reactor cavity. Success for this event is not attained (e.g., due to inadvertent closure of the RCV isolation valves) if buildup of gases in the reactor cavity and pipeway cells is allowed to occur for longer than 20 hours. As discussed in Section 12.6.8, a buildup of gases for a period of twenty hours or more is assumed to result in structural failure of the reactor cavity and/or pipeway cells due to overpressurization. Additionally, if the isolation valves are suddenly opened after a buildup of hydrogen in the inerted lower containment atmosphere, a potentially detonable hydrogen concentration could be introduced into the non-inerted upper containment.

Dependencies

Discussed under dependencies of other events.

OVC: Out-of-Vessel Cooling

Definition

This event is defined as the presence of the conditions necessary to keep the fuel debris that has relocated into the reactor cavity in a coolable state.

Success for this event implies that a coolable debris bed is permanently established within the reactor cavity until sodium boildry. In order for the above criterion to be met, fuel debris bed heat flux levels must not exceed dryout limits. Failure implies that a coolable debris bed cannot be established and that reactor cavity liner failure and subsequent concrete penetration will occur.

Dependencies

The OVC event is only applicable to the core-response sequences that either do not meet or are not applicable to the IVC event success criterion. If the OVC criteria are met, reactor cavity liner failure might be precluded for several hours. However, if the success criteria of the OVC are not met early reactor cavity floor liner failure (see event NLF) is assured. The OVC event is bypassed for the RCV failure sequence since the implied failure to establish a reactor cavity to RCB vent path will result in substantially diminished heat transfer from the sodium-fuel pool. Therefore, in all likelihood, a non-coolable debris bed would be expected.

NLF: No Liner Failure

Definition

Success for this event is defined as no failure of the reactor cavity floor liner or pipeway cell liners prior to the evacuation of the Emergency Planning Zone (EPZ). If the success criteria of this event are met, catastrophic containment failures would be precluded prior to the time required to evacuate the emergency planning zone (EPZ). This is true because all of the mechanisms that could potentially cause containment failure prior to evacuation of the EPZ require hydrogen production and, therefore, failure of the reactor cavity liner. Furthermore, if early hydrogen production is precluded, venting and purging would be unnecessary until after the EPZ had been evacuated.

If the success criteria are not met, containment failure prior to evacuation of the emergency planning zone cannot be ruled out.

Dependencies

Since liner failure is assured if the core debris in the reactor cavity is not coolable, a branch point for NLF success was not included for sequences that did not meet the OVC criteria. Similarly, early liner failure is implied for RCV failure sequences due to the diminished heat transfer conditions between the debris bed and its surroundings. Therefore, the RCV failure sequences exclude a branch point for event NLF.

CIS: Containment Isolation System

Definition

Success for this event implies that a leakage rate sufficient to cause a substantial increase in radionuclide releases to the atmosphere does not occur. The CIS event success criteria are not met if the leakage from a structurally intact containment exceeds 230 volume percent per day. As shown in Appendix B, a leak rate of 230 volume percent per day would preclude buildup of hydrogen beyond four percent and would preclude overpressurization of the containment.

Dependencies

If the CIS success criteria are met, the containment will be in a "bottled-up" state. This bottled up state will prevent leakage beyond that specified above. However, the bottled up state can be assumed to allow hydrogen gas and other non-condensibles to build-up within the confines of the reactor containment. If the CIS success criteria are not met, hydrogen build-up would be precluded, but at the cost of an early release of radionuclides to the environment. Since it is assumed that for the RCV failure sequence the fission products would be bottled-up in the reactor cavity, the CIS event is bypassed for that sequence.

Additionally, it was concluded that the CIS event in conjunction with the OVC failure sequence was unlikely and uninteresting from a phenomenological perspective since the consequences would be no different than other more likely sequences already represented by the tree. As a result the OVC failure sequence does not include a node for event CIS.

AFS: Annulus Filtration System

Definition

The success criteria for the AFS event are met if the AFS system maintains a 1/4" w.g. negative pressure in the annulus and filters the annulus space during containment isolation. Failure of this event implies that containment leakage into the annulus is released directly to the environment without being filtered.

This event was included in the containment-response event tree because it can significantly affect the radiological consequences attributed to containment leakage into the annulus. It should be noted that CRBRP-3, Volume II (Reference 1) assumes that the annulus filtration system is operating at an efficiency of 99% up until the time of venting and purging or operation of the annulus cooling system. The 1% unfiltered leakage from the annulus is the primary non-noble contributor to the 2 hour exclusion boundary doses reported in CRBRP-3, Volume II. (Noble gases are not filtered out by the annulus filtration system.) Some AFS failure sequences would allow 100% of the annulus air to be vented directly to the environment without being filtered. Presumably, this could increase the non-noble fission product releases by nearly two orders of magnitude. This could potentially have the effect of raising halogen-sensitive organ doses (e.g., thyroid doses) correspondingly.

Dependencies

The AFS event is bypassed for RCV failure sequences since these sequences would not involve a fission product release to the upper RCB for at least 20 hours and since the AFS would be primarily effective in mitigating consequences early in the scenario (i.e. 0 to 36 hours). CIS failure sequences do not consider the AFS event because the consequences

of a containment isolation failure (containment leakage of 230 volume percent per day or greater) would not be substantially exacerbated by the AFS event failure.

CM(E): Early Containment Integrity Maintained

Definition

Success for event CM(E) is attained if the integrity of the containment vessel (excluding isolation failures) survives at least up to evacuation of the emergency planning zone (EPZ). Mechanisms that were investigated as potential challenges to the containment prior to the time required to evacuate the EPZ include the following:

- Detonation of hydrogen within containment, and
- Differential thermal expansion of the polar crane under the influence of a hydrogen flame.

NOTE: HCDA-generated missiles were considered non-mechanistic in Section 11 and will not be considered here.

Dependencies

This event only applies to sequences that do not meet the NLF or OVC success criteria since these sequences are the only ones that are capable of causing containment failure prior to the EPZ evacuation time. Furthermore, CIS failure sequences bypass the CM(E) event since for these sequences hydrogen buildup beyond 4% is precluded. Although the criteria for event CM(E) may not be met, operation of the vent and purge system, containment cleanup, and annulus cooling systems is considered here as potential steps to mitigate the consequences of containment failure.

CM(L): Late Containment Integrity Maintained

Definition

Success for event CM(L) is attained if failure of the containment, not related to beyond-design-basis system malfunction, does not occur after the emergency planning zone has been evacuated. The mechanisms that

could potentially fail containment include those that were listed for CM(E) plus additional failure potential associated with extended liner integrity for sequences where the NLF success criteria is met.

Dependencies

The CM(L) event is not applicable to sequences where the CM(E) success criteria are not met. CIS failure sequences bypass the CM(L) event since hydrogen buildup beyond 4% and over-pressurization are precluded.

Although the criteria for CM(L) may not be met, the operation of vent and purge systems, as well as containment cleanup and annulus cooling systems, is considered here for potentially mitigating the consequences of containment failure.

VPS: Vent and Purge Operation Successful

Definition

Success for this event is attained if at least one of two vent paths and at least one of two purge paths are established in the necessary order to relieve pressure within the containment and to reduce hydrogen concentrations (if applicable). Event VPS would be unsuccessful if, for example, the operator failed to open up the purge lines upon venting, when the hydrogen concentration was approaching the detonable limit. Another failure of the vent and purge system could result from failure to open the vent line isolation valves or a plugging of the vent line from aerosols.

Dependencies

Since vent and purge operations would very likely take place after evacuation of the EPZ, failure to vent and purge is not expected to have an effect on early radiological consequences.

ACS: Annulus Cooling System Operates as Required

Definition

Success for this event is attained if the annulus cooling system cools the reactor containment steel shell and the confinement dome as

necessary to prevent their failure. The operator would actuate the annulus cooling system when the containment steel shell approaches a prescribed value (~400 to 500F). When the steel shell temperature drops to 200F, the ACS is turned off.

Successful operation of the ACS requires the following:

- 2 of 4 containment building dome exhaust dampers, and
- 3 of 6 annulus cooling fans, with dampers, which are configured in two trains of three fans

If the ACS success criteria are not met, the containment steel shell, or more likely the confinement dome, could fail.

Dependencies

Operation of the annulus cooling system requires bypassing the annulus filtration system. Failure of the annulus cooling system could cause the containment or confinement to fail even if the CM(E) and CM(L) success criteria have been met. Operation of the ACS is considered following containment failure (i.e., CM(E) and CM(L) failure sequences) as a measure to prevent further failures. For simplicity, failure of the annulus cooling system implies failure of both the containment and the confinement.

CCS: Containment Cleanup System

Definition

The success criteria for the CCS event require that the containment cleanup system operate as designed during venting operations. This in turn requires that the passive components of the CCS (e.g., quench tank, jet venturi scrubber, high efficiency fibrous scrubber) and at least one of two active components of the CCS (e.g., containment cleanup circulation pump) remain functional. If the containment cleanup system does not function as required, the dose consequences will be significantly higher than they would be if the CCS were operating as required.

Dependencies

Since the CCS is on the vent line train, the VPS event failure sequences bypass the CCS event as well. The CCS is not assumed to be effective for ACS failure sequences since failure of ACS would likely fail both containment and confinement.

MT: Meltthrough Does Not Occur

Definition

Success for this event is achieved if the core debris has not penetrated the basemat and the radionuclides of the melt front do not come into contact with groundwater. This event was included to assess the potential for a longer term radionuclide release to the groundwater and ultimately to the Clinch River, following core melt.

Dependencies

None

12.5.2 Sequence Description and Outcomes

The containment-response event tree sequences begin with an initial condition that corresponds to a core-response event tree end condition. The core-response event tree end conditions have been grouped into the following five categories:

1. Vessel bottom failure without RV head leakage
2. Vessel bottom failure with RV head leakage
3. In-vessel retention of core debris
4. In-vessel retention of core debris with RV head leakage
5. HCDA-generated, missile-induced containment failure

The first category of core-response event tree end conditions, "vessel bottom failure (without RV head leakage)," was the most probable one.

The estimated combined probability of achieving this end condition ranged from about 15% for the UTOP to greater than 99% for the ULOS and the protected HCDA initiators. The resulting containment-response sequences and their outcomes given vessel bottom failure are briefly described in Table 12.5-1.

The combined probability of achieving end conditions corresponding to the second category, "vessel bottom failure with RV head leakage" was less than 1% for all of the HCDA initiator sequences. As mentioned in Section 12.4.1, the containment response to the second category would be very similar to the containment response to the first category of containment-response end conditions. On this basis, only the first category of containment-response sequences is delineated here.

It should be noted here that the core-response event tree was modeled in a manner that suggests that RV head leakage requires energetics. However, under the prolonged high-temperature conditions of the protected HCDA the RV head seals may fail prior to vessel penetration. This phenomenon could increase the probability of achieving this second category of end conditions somewhat for protected loss of heat sink accidents. Unprotected accidents and loss of sodium accidents will occur on a time scale too short to allow for non-energetic RV head seal damage prior to bottom head penetration. Head seal thermal damage may occur tens of hours later, however.

The third category of core-response end conditions is not applicable to the containment-response event tree since the core debris is retained within the primary coolant boundary and no sodium is released from the

Table 12.5-1

SEQUENCE DESCRIPTION FOR
REACTOR VESSEL BOTTOM HEAD FAILURE WITHOUT HEAD LEAKAGE

Sequence Number(s)	Description
1-10	These sequences correspond to a situation in which the core debris is coolable in the reactor cavity and reactor cavity liner integrity is maintained prior to evacuation of the emergency planning zone. Sequences 1 and 2 correspond to operation of all beyond-design basis features as required. Sequences 3-10 involve failure of one or more beyond-design-basis features. Sequence 1 does not result in basemat penetration as does Sequence 2. These sequences involve containment failures related to system failures only.
11-20	Sequences 11-20 involve failure of the containment after the EPZ has been evacuated. Depending on the size of the containment breach, the beyond-design basis features would likely be able to mitigate the consequences of such a failure. If a breach of containment fails the steel shell but not the confinement, the containment atmosphere will flow into the annulus and could be filtered by the annulus filtration system. Once venting is initiated, the flow will reverse. Air will be drawn into the containment through the failure location and (assuming that the pressure differential between the containment and the atmosphere is greater than the pressure differential between the containment and the annulus) the containment atmosphere could be vented through the containment cleanup system. Failure of the annulus cooling system would prevent the above recovery action since it is assumed that both the confinement and the containment fail. Sequence 11 corresponds to successful operation of all the beyond-design-basis features.

Table 12.5-1 (Continued)

Sequence Number(s)	Description
21-40	Sequences 21-40 are similar to sequences 1-20 except that the annulus filtration system does not operate as required. This will result in an early dose rate larger than that predicted in the CRBRP-3, Volume II base case for the exclusion boundary and low population zone because the non-nobles that leak past the containment into the annulus would not be filtered before being released to the environment.
41-50	These sequences correspond to a failure to isolate containment. Since failure to isolate containment as defined in Section 12.5 will preclude hydrogen buildup beyond 4% and prevent overpressurization, the CM(E) and CM(L) events are bypassed. Sequences 41 thru 46 correspond to operator initiation of vent and purge to reduce the consequences of failure to isolate containment. Like the sequences that involved failure of the annulus filtration system, the CIS failure paths result in an early release of radionuclides greater than the CRBRP-3, Volume II base case.
51-60	Sequences 51-60 correspond to a situation in which the RC liners have failed prior to the time required to evacuate the emergency planning zone but the hydrogen produced following liner failure does not pose an immediate threat to containment. Later, hydrogen production can be controlled by burning with oxygen and if necessary through operation of the vent and purge system. These sequences correspond most closely with the Reference 1 base case.
61-70	Sequences 61-70 correspond to late (after the EPZ has been evacuated) failure of the containment as similarly discussed for sequences 11-20. For these sequences the reactor cavity floor liner failed prior to the EPZ evacuation time.

Table 12.5-1 (Continued)

Sequence Number(s)	Description
71-80	Sequences 71-80 correspond to early (before the EPZ has been evacuated) failure of the containment. Although the failure mechanism would be similar to that described in sequences 61-70 the consequences would likely be greater due to the earlier potential for release of fission products.
81-110	These sequences correspond to a failure of the annulus filtration system which is successful for sequences 51-80. Thus, air drawn into the annulus through the failure location could not be filtered by the annulus filtration system.
111-120	Sequences 111-120 involve a failure to isolate containment. This failure is defined to be large enough such that hydrogen burning in containment is precluded.
121-130	These sequences correspond to a situation in which a non-coolable fuel debris bed in the reactor cavity causes immediate liner failure. However, for these sequences failure of the upper containment is averted.
131-140	These sequences correspond to a situation in which late failure of the containment vessel has occured. The scenario is very similar to an early reactor cavity floor liner failure scenario in which the fuel debris was coolable. The consequences of sequences 131-140 would be similar to the consequences of sequences 61-70.
141-150	These sequences correspond to a situation in which early failure of the containment vessel has occurred with failure of the AFS. From a consequence perspective, sequences 141-150 are similar to sequences 101-110.
151-180	These sequences all correspond to an inadvertant closure of the reactor cavity vent line isolation valves. Sequences 151-160 correspond to no containment failure, sequences 161-170 correspond to a late containment failure and sequences 171-180 correspond to an early containment failure.

RV head to pose a challenge to containment. The combined probability for achieving this third end condition ranged from less than 1% for the protected HCDA initiators to about 80% for the UTOP. If sodium is released to the upper containment through the RV head and the core debris remains coolable within the RV, as in the fourth category of end conditions, the threat to containment is limited only to the upper containment. As explained in Section 12.4.1, the containment response to the fourth category of end conditions was not specifically considered due to its low probability of occurrence and resulting similar consequences as those already covered by the more likely vessel bottom failure sequences.*

The containment response to the fifth, and last, category of core-response end conditions was also not specifically considered due to its extremely low probability of occurrence.

12.5.3 Description of Radionuclide Release Categories

Radionuclide releases were categorized according to the estimated magnitude and timing of the release. The magnitude of the release is described by three broad categories; low, medium, and high. If the release was estimated to occur prior to the time required to evacuate the 10 mile emergency planning zone (EPZ), it was categorized as an early release. Based on Reference 2, the estimated evacuation time of the 10-mile EPZ is 8 hours 45 minutes or about 9 hours.** For purposes of

* A complete assessment of public risk could require modeling of these low probability events.

**This estimate contains a 1-hour 50-minute preparation time factor.

this PRA, the evacuation will be assumed to have been initiated the instant that a "core on the floor" has been verified by the operators (i.e., the instant that fuel debris has entered the reactor cavity). This is a conservative assumption for the protected loss of heat sink accident, since many hours would have passed between initiation of the HCDA and a "core on the floor" condition. However for the remaining initiators the assumption is deemed reasonable since the estimated time between initiation of the HCDA and a core on the floor is of the order of 1000 sec or about 16 minutes. A map illustrating the boundaries of the EPZ is shown in Figure 12.5-2. If the release occurred after this time it was categorized as a late release. Releases that occurred many months after the accident were categorized as long-term. Releases due to cleanup efforts were not addressed in this study. The release categories are shown in Table 12.5-2.

12.6 QUANTIFICATION OF THE CONTAINMENT-RESPONSE EVENT TREE

12.6.1 Introduction to Method of Quantifying the Tree

The events that comprise the containment-response event tree can be divided into several categories. The first category of top events are phenomenological in nature and essentially independent of the system failures leading to HCDAs. For example, these events would not be dependent on loss of 1E power. However, these events are, in some cases, sensitive to core-response event tree end states. Included in this first category are events IVC (in-vessel coolable), OVC (out-of-vessel coolable), NLF (no liner failure), CM(E) (early containment failure), CM(L) (late containment failure), and MT (meltthrough). The approach used in quantifying this first category of events is similar to the

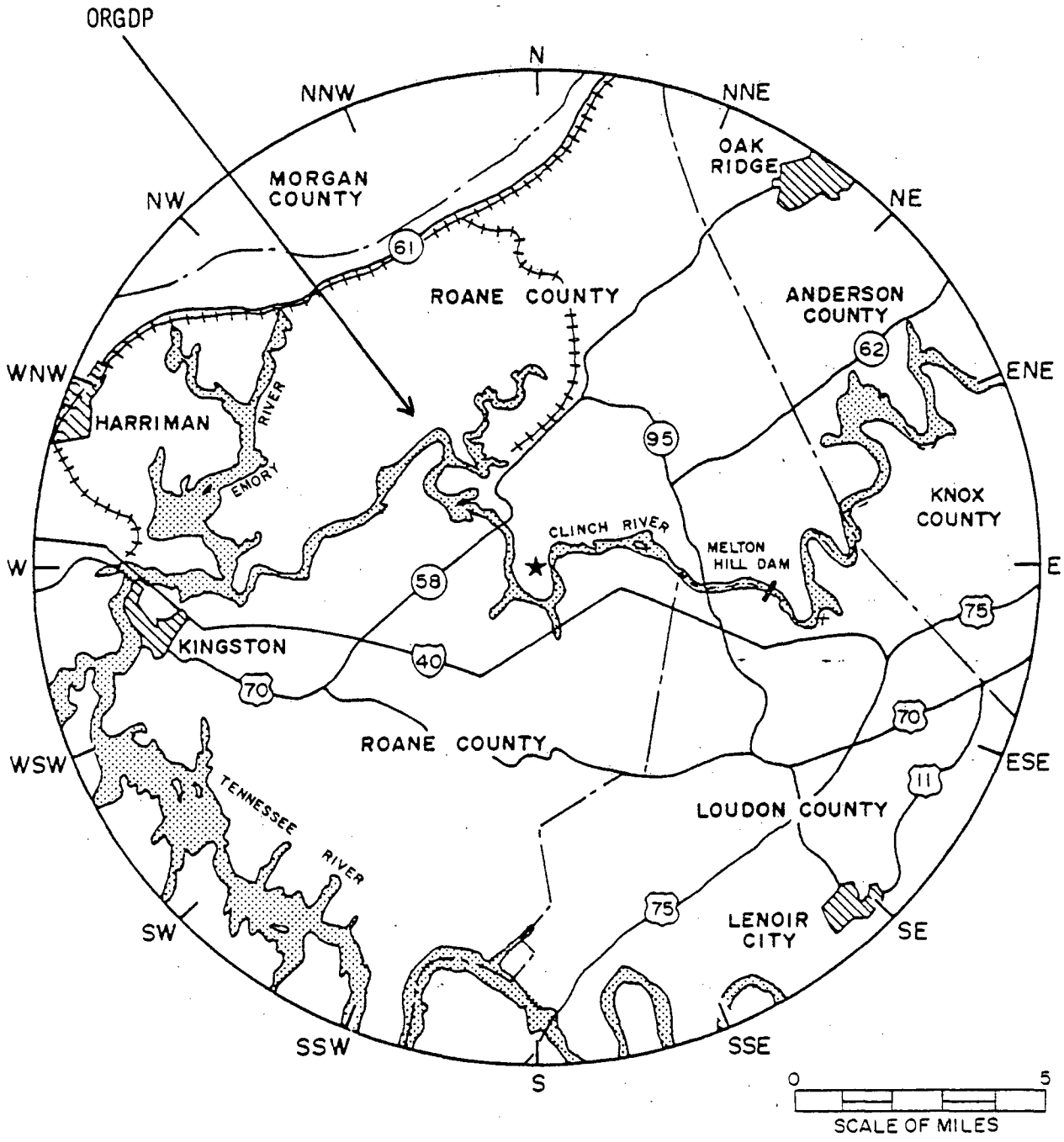


Figure 12.5-2. Map Showing Boundaries of the 10-Mile Emergency Planning Zone.

Table 12.5-2

RADIOLOGICAL RELEASE MATRIX

Time / Magnitude	Early 0 < t < 9 hrs	Late t > 9 hrs	Long-Term > 6 months
LOW	1	2	3 All even numbered sequences (i.e., all MT sequences)
MODERATE	4 41,71*,101*,111,141,171*	5 1,11*,21,51,61*,81,121,131*,151,161*	6 7,57,87,127,157
HIGH	7 43,47,49,73,75,77,79,103,105,107,109,113,117,119,143,147,149,173,177,179	8 3,5,9,13,15,17,19,23,25,27,29,31,33,35,37,39,45,53,55,59,63,65,67,69,83,85,89,91,93,95,97,99,115,123,125,129,133,135,137,139,145,153,155,159,163,165,167,169,175	9

Note 1: Moderate release roughly corresponds to doses in Reference 1 base case analysis. Doses higher than base case are designated "high" and doses lower as "low."

Note 2: Only odd numbered sequences (n) are listed. Even numbered sequences (n+1) are identical to odd numbered sequences (i.e. - appear in same category) except that foundation mat meltthrough adds an additional long term release contributor.

Note 3: Early releases associated with bypass leakage are not included here.

*Assumes the confinement does not fail along with the containment.

approach used in quantifying the core-response phenomena in Section 11 of this report. The probability assignments were made subjectively following review of analyses, test results, and literature dealing with various aspects of LMFBR containment response to a core-melt accident.

The second category of events are system-related events that are directly dependent on the system failures that lead to HCDAs. Included in this category are events CIS (containment isolation system), AFS (annulus filtration system), VPS (vent purge system), ACS (annulus cooling system), and the CCS (containment cleanup system). A typical system failure dependency that would affect the category two events would be "loss of 1E power." The top events in this second category are quantified in Section 13 based upon fault tree analyses reported in Appendix A.

The third and last category of events comprises system-related events that are not directly dependent on the system failures that lead to HCDAs. Only the RCV (reactor cavity vent) event corresponds to this category.

Since the category two events are directly dependent on system failures leading to HCDAs, they will not be quantified in this section. Instead two separate states corresponding to total event success or total event failure will be propagated through the containment-response event tree along with the category one and three branch point probabilities. By doing this, dominant sequences can be selected based on phenomenology and sequence-independent system failures alone. System-dependent sequence failures will then be coupled, in Section 13, with dominant phenomenological sequences to obtain dominant containment-response sequences.

12.6.2 In-Vessel Cooling

As explained in Section 12.5.1, the IVC event is only applicable to core-response event-tree end states designated as "CD, In Vessel."

CD, In-Vessel" end states:

Yes: 1.0

No: 0.0

All other end states:

Yes: 0.0

No: 1.0

12.6.3 Reactor Cavity Vent

Since the RCV event success/failure probability is not directly dependent on the system failures leading to initiation of HCDAs, it is quantified in this section. Based on the fault tree analysis performed in Appendix A26 the following probabilities have been assigned to the RCV event:

Yes: ~1

No: 1×10^{-4}

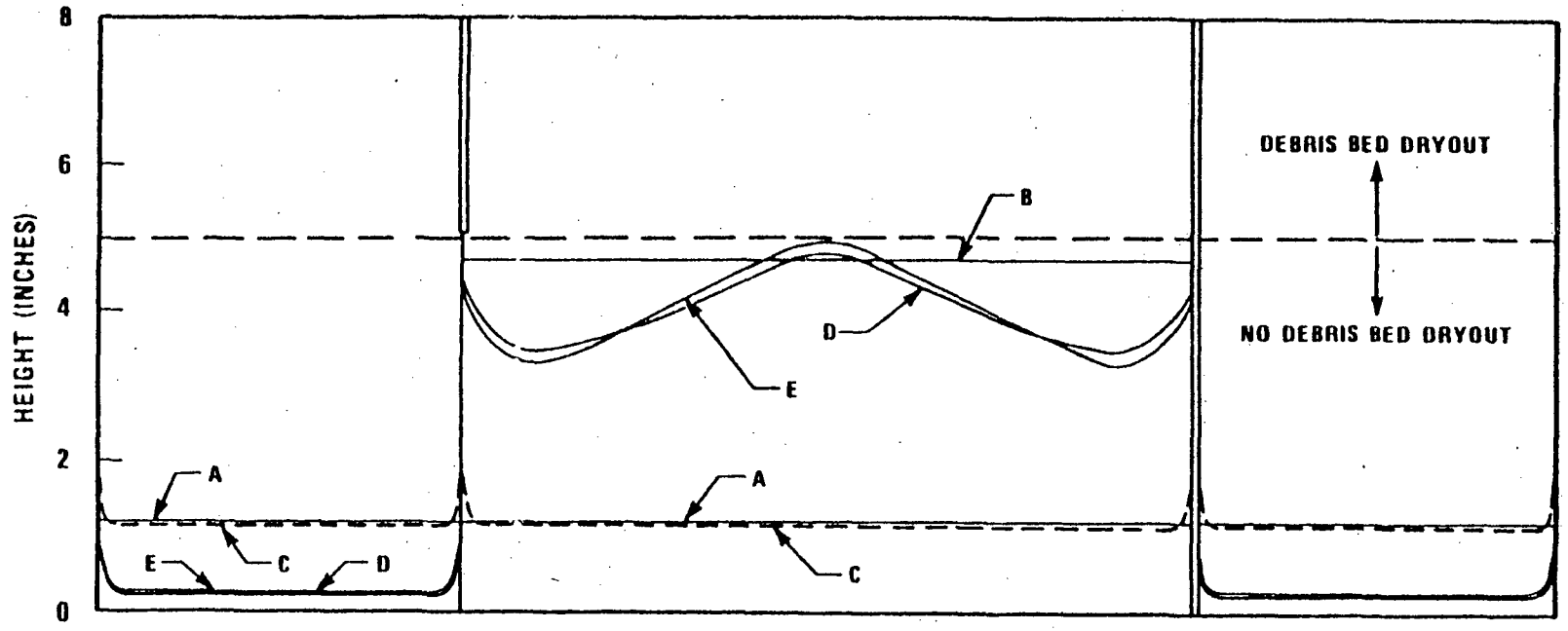
12.6.4 Out of Vessel Coolability

In quantifying the OVC containment-response event, no distinction will be made between core-response sequences designated, "vessel bottom failure without RV head leakage" and sequences designated, "vessel bottom failure with RV head leakage." This is justified because there is no conclusive evidence that indicates that sodium leakage from the head (if any) following energetics, would be significant relative to the volume of sodium that would fill the reactor cavity following vessel breach.

The phenomena associated with fuel debris coolability were briefly discussed in section 12.4. As explained in that section, the factors governing the potential for fuel debris coolability within the reactor cavity are the same factors that applied to in-vessel fuel debris coolability; namely, (1) debris bed depth, (2) the presence of sodium, and (3) the availability of heat sinks.

The debris bed depth expected would depend on the amount of fuel debris swept out by sodium which, to a certain extent, would depend on the size of the rupture. As explained in Section 12.4, some of the core debris would be expected to be temporarily held up in the reactor vessel, thus effectively reducing the debris bed depth in the reactor cavity. In Reference 1 the effect of debris bed uniformity on margin to debris bed dryout was examined (see Figure 12.6-1). Debris bed dryout was not predicted even for the extreme non-uniform spreading case where nearly all of the fuel debris is retained within the guard vessel skirt. Since this study was performed for the ULOF case after 1000 seconds, the debris bed depths corresponding to dryout would be somewhat higher for fuel debris distributions with lower decay heat levels. Although it is expected that HCDA sequences (e.g., the UTOP event) may initially have varying in-vessel fuel debris distributions, the debris bed depth in the lower RV head at the time of failure is expected to be roughly equivalent for purposes of containment-response assessment.

Sodium will also be expected in the reactor cavity any time fuel has penetrated the bottom head of the reactor vessel. As discussed in Section 12.4, the temperature of the sodium, and thus the margin to boiling, will depend on the specific HCDA initiator.



A UNIFORM SPREADING (40' DIA)
B UNIFORM SPREADING (20' DIA)
C UNIFORM SPREADING (40' DIA) EXCEPT AT VERTICAL SURFACES
D NON-UNIFORM SPREADING
E EXTREME NON-UNIFORM SPREADING

*INCLUDES STEEL CLADDING, WIRE WRAP, DUCTS AND PART OF THE REACTOR VESSEL AND GUARD VESSEL LOWER HEADS.

Figure 12.6-1. Debris Bed Configurations that Would Not Result in Dryout. (Reference 1)

The early availability of heat sinks is not expected to be sensitive to specific HCDA initators. The additional heat removal path associated with reactor vessel head failure (i.e., through the head seals) is neglected because of its greater resistance relative to the RC to RCB vents. Heat removal through the head seals might be important for the RCV failure sequences however.

As previously alluded to, another important factor affecting fuel debris coolability is the decay heat level of the fuel debris at the time it enters the reactor cavity. The lower the fuel debris decay heat level, the lower the probability that the dryout heat flux will be exceeded. The decay heat level will vary considerably for various initiators. For example, if it is assumed that 100% of the fuel debris enters the reactor cavity 1000 seconds after the ULOF has been initiated, the decay power level would correspond to 14.28 MW(th). However, if following a LHSL accident the fuel debris entered the reactor cavity at 12 hours the decay heat level would have dropped down to less than 6 MW(th). Thus decay heat levels may play an important role for fuel debris distributions that are marginally coolable.

ULOF, UTOP, and ULHS

The above HCDA initiators are grouped together because they exhibit similar initiator-sensitive characteristics that most influence whether or not the fuel debris in the reactor cavity will remain coolable. The decay heat levels, the time prior to vessel failure, and the initial sodium temperature upon entering the reactor cavity are similar for all three events. The temperature of the sodium that enters the reactor

cavity would be expected to be about 1000°F (990°F was the value used in the Reference 1 analysis). Perhaps the most important factor that ties the above three HCDA initiators together is the fact that the assigned probability that sodium or fuel debris enters the reactor cavity first (see Section 11) is identical for all three. In Section 11 it was stated that if the fuel entered the reactor cavity first, the sodium would be right behind (and visa versa). The turbulence produced when the sodium and fuel debris enter the reactor cavity and the subsequent fragmentation of the fuel is expected to allow the majority of the fuel to be suspended long enough for it to be swept out from under the guard vessel skirt (i.e., the fuel sweepout time is less than the time associated with fuel particle settling).[1] It is on this basis that a uniform fuel distribution is predicted in the reactor cavity.

In light of the fact that a uniform debris bed is deemed likely, and using the Reference 1 analysis, including Figure 12.6-1 as guidance, the following probabilities were assigned to OVC success for the ULOF, UTOP, and ULHS end states:

Assigned Probabilities:	Yes:	0.99	(very likely)
	No:	0.01	

LHSE and LHSL

In the absence of an analysis, such as that performed in Reference 1 for the ULOF, it is difficult to estimate, with a high degree of confidence, the times to boiling and boildry for the LHSE and LSHL. However, based on the assessment performed in Section 11 and on the Phase I PRA report, it can be safely assumed that the sodium temperatures upon entering the

reactor cavity would be higher than for the ULOF event. Thus, the margin of subcooling would be decreased for the PLHS events (LHSE, LHSL) relative to the ULOF event. Further, the slow heat up process of these events will heat up the reactor cavity structure early and diminish its ability as a heat sink upon sodium entry into the reactor cavity. An important factor relative to the chances of forming a coolable debris bed is whether sodium or fuel debris enters the reactor cavity first. Unlike the ULOF event, the PLHS events may involve a failure of the vessel at elevated temperature which may allow sodium to get into the reactor cavity long before the fuel penetrates the bottom head. Depending on the failure location (e.g., outlet nozzle region vs inlet nozzle region) a relatively deep sodium pool will be present in the reactor cavity at the time of bottom head penetration. As the fuel enters the reactor cavity it will fragment and would be somewhat dispersive. However because of a significantly reduced sodium head above the melt front at the time of penetration, the extent of sweepout would be expected to be less than that for the ULOF event. Using Figure 12.6-1 as a guide, the expected debris bed depth would be less than that corresponding to dryout. However, the margin to dryout depth would be substantially diminished relative to the ULOF case. On the other hand, the decay heat level at the time of vessel penetration would be significantly lower for the "PLHS with Sodium Early" end state relative to the ULOF and state. It is anticipated that, for protected accidents, the lower decay heat level would tend to offset the diminished dryout margin. On this basis, the probability that the OVC criteria are met

for the PLHS with sodium early is deemed approximately equivalent to the probability that the OVC criteria are met for the UTOP, ULOF, and ULHS events.

PLHS with Sodium Early.

Assigned Probabilities:

Yes: 0.99
No: 0.01

PLHS with Sodium Late.

The PLHS with sodium late endstate would correspond to a situation where the fuel debris bed would be relatively level and, in addition, decay heat levels would be relatively low. Like the ULOF event, the PLHS with sodium late would result in fuel suspension and sweepout from under the guard skirt, but unlike the ULOF event, the decay heat level of the fuel at the time of vessel penetration would be relatively low (by about a factor of three). On this basis, the PLHS with sodium late event is expected to be even more likely to meet the OVC success criteria than the ULOF, ULOF and ULHS events.

Assigned Probabilities:

Yes: 0.999
No: 0.001

LOS

As explained in Section 12.4, there are many factors associated with the LOS event that favor the OVC success criteria. The sodium would be expected to enter the reactor cavity early relative to the fuel. The time period between sodium entry and fuel debris entry would be

expected to be greater than the corresponding time period for the ULOF because of the "sluggishness" attributed to protected accidents. However, for some LOS sequences that involve PHTS leaks, the sodium may be delayed getting into the reactor cavity. As the sodium does flow into the reactor cavity it will transfer heat to the pipeway cell structures which will effectively lower the temperature of the sodium that eventually does enter the reactor cavity.

When the fuel debris penetrates the reactor vessel bottom head and guard vessel and enters the reactor cavity, it will fragment into finer particulates. The smaller particles of fuel will rapidly transfer heat to the surrounding sodium pool which will result in pool turbulence. The pool turbulence will have the effect of temporarily suspending the fuel particles and possibly allowing some fraction of the fuel debris to relocate outside of the guard vessel support skirt. However, like the PLHS case with sodium early, the fraction of fuel swept out from under the guard vessel skirt would be much smaller than the best estimate ULOF case. With lack of any evidence that would discriminate the LOS* from the PLHS with sodium early, the following values have been assigned to OVC success for the LOS event.

Assigned probabilities:

Yes: 0.99
No: 0.01

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*It is implied that for the LOS end state the sodium enters the reactor cavity first.

ULOS

The primary difference between the ULOS and the LOS is in the timing of the penetration. The ULOS decay heat level will be higher than the LOS decay heat level at the time of bottom head failure. These differences are deemed significant enough to assign a value to the ULOS OVC event success lower than that assigned for the LOS end state.

Assigned probabilities:

Yes: 0.9
No: 0.1

12.6.5 No Liner Failure

In the Reference 1 analysis, the reactor cavity floor liner was assumed to fail completely and instantaneously, immediately following fuel debris penetration. The calculated time to failure for the other liners, based on the analysis performed with the ANSYS computer code for the Reference 1 base case, is shown below.

Cell Liner Region	Liner Failure Time (Hours)*
Pipeway Cell Floor and Roof	30
2'-6" Thick Pipeway Wall	35
4' Thick Pipeway Wall	40
Lower Submerged Reactor Cavity Wall	50
Head Access Area Pipeway Wall	55
Upper Submerged Reactor Cavity Wall	70
Non-Submerged Reactor Cavity Wall	80
Double-Heated Pipeway Wall	90

The failure criteria that were utilized in determining liner failure times were based upon considerations of the steel tensile properties and the effects of creep and thermal expansion. The tensile properties of

*Based on Table 3-12 of Reference 1.

liner steel (SA 516) were explored in a comprehensive test program as part of the CRBRP development program.[1] The observed stress-strain relationships as a function of temperature were utilized in determining the strain allowables. Test results also indicated that at high temperature and stress levels considerable creep could occur in just a few hours.**

However, Reference 1 goes on to state that, "in the case of structural elements which are mainly subjected to thermal strains (such as the cell liners), the effect of creep will generally be to relieve or relax compressive strains by exchanging mechanical strains (caused by thermal expansion) with creep strains." Since the Reference 1 failure criteria are deemed conservative, the expected liner failure times are considered to be bounded by those listed above.

In summary, based on the analysis performed in Reference 1, it appears that the reactor cavity liners, other than the floor liner, will survive for tens of hours following entry of sodium and/or fuel debris in the reactor cavity.

Although it appears that the reactor cavity wall liners will not fail for tens of hours, the same cannot be said of the reactor cavity floor liner. However, there is some experimental evidence from the LT-1 test that indicates the CRBRP cell liner design can accommodate sodium spills at temperatures of 1100°F.[30] The LT-1 test consisted of spilling 1100°F

**Creep is defined here as the time dependent strain in materials that undergo continuous stress.

sodium at a rate of 100 gal/min into a test article that had a liner configuration similar to CRBRP on the floor and on an adjacent wall and an FFTF cell liner configuration on the opposite wall. After the sodium cooled down, electric heaters were turned on to heat-up the sodium pool. The sodium temperature was maintained between 1460°F and 1580°F for approximately six days. After a week of cooling down, the sodium was drained and the cell liners were inspected. Although some nelson studs had broken, there were no failures of the liner itself. With this single data point, it will be assumed in this report that the reactor cavity floor liner will remain intact as long as the sodium temperature is less than 1100°F. It should be noted, however, that results of the LT-1 test should not be applied directly to a situation in which core debris, as well as sodium, is falling into the reactor cavity. It has been estimated, based on expert consultation,[35] that floor liner failure will be likely prior to the EPZ evacuation time if (1) fuel debris enters the reactor cavity or (2) if the sodium temperature entering the reactor cavity is in excess of about 1300°F. A table was constructed for the purpose of assigning probabilities to NLF event failure (See Table 12.6-1). As shown in the table, the liner failure probability is the same for all events if fuel enters the reactor cavity. The liner is also likely to fail prior to vessel bottom penetration for the PLHS sequences in which very hot sodium (~ 1600°F) enters the reactor cavity. Although there is a chance that the liner could fail prior to vessel bottom penetration for loss of sodium events, this chance is deemed unlikely and is not believed to effect the overall probability of event NLF.

Table 12.6-1

MATRIX OF LINER FAILURE PROBABILITIES FOR CORE-RESPONSE ENDSTATES

HCDA Initiator Endstate	Floor Liner Failure Probability as a Result of			
	Sodium in Cavity Before RV Bottom Head Failure		Sodium and Fuel in Cavity	
	Sodium Temp.	NLF_1	Sodium Temp.	NLF_2*
• ULOF • ULHS • UTOP	N/A	---	~1000°F	0.9
• LHSE and LHSL Sodium in RC late	N/A	---	~1600°F	0.9
• LHSE and LHSL Sodium in RC early	~1600°F	>0.9	~1600°F	>0.9
• LOS • ULOS	750-1000°F	0.1	750-1000°F	0.9

*Since it has been determined that early liner failure is likely if fuel enters the reactor cavity, the value for NLF_2 is the same for all HCDA initiator end states.

In summary, it is estimated that reactor cavity wall liner failure would require tens of hours, whereas, depending on the initial conditions, reactor cavity floor liner failure will be likely within the time required to evacuate the emergency planning zone.

The following probabilities have been assigned for event NLF success.

Core-response end states other than LHSE and LHSL with sodium early:

Yes	0.1
No	0.9

LHSE and LHSL with sodium early:

Yes	0.01
No	0.99

12.6.6 Containment Isolation System

The CIS event falls within the second category of events discussed in Section 12.6.1 and will be quantified in Section 13. The analysis used in quantifying the CIS event is described in Appendix A22. Only total event success or failure will be considered here:

Yes: $\overline{CIS}$

No: CIS

12.6.7 Annulus Filtration System

The AFS event falls within the second category of events discussed in Section 12.6.1 and will be quantified in Section 13. The fault tree analysis used to quantify the AFS event is shown in Appendix A19. Only total event success or failure will be considered here:

Yes: $\overline{AFS}$

No: AFS

12.6.8 Early Containment Integrity Maintained

As explained in Section 12.5, this event examines the probability of a containment failure that is not a result of a system failure. (An example of a system failure that could effectively fail containment under prolonged core-melt conditions is the failure of the annulus cooling system.) After an attempt to identify potential containment threats prior to the 10-mile EPZ evacuation time, the following scenarios were identified.

Failure to Vent and Purge at 6% Hydrogen Concentration

If the reactor cavity floor liner failed soon after fuel debris entered the reactor cavity it would be theoretically possible that hydrogen could accumulate to limits that require venting and purging prior to the evacuation time of the EPZ. Such a case was assessed by the CRBRP Project in Reference 1. The case, referred to as the Margin Assessment Case, was not based on any sodium-concrete reaction penetration rates seen to date. Instead the Margin Assessment case was fabricated to coincide with the 10-mile EPZ evacuation time. Since the sodium-concrete reaction rates are considerably greater than those seen from experiments it is deemed highly unlikely that a hydrogen concentration of 6% could be produced within the 8 hours, 45 minutes required to evacuate the EPZ. In addition, the assumption that the liner vanishes at the time the fuel debris enters the reactor cavity is conservative. As explained in Section 12.4, only local failures would be expected. However, it has been estimated that once a liner penetration area has exceeded about 6 in^2, the entire area of concrete under the failed liner section will be in contact with sodium that flows into the liner gap.

Each liner section corresponds to about 1/8th of the entire reactor cavity floor liner area. In order for the Reference 1 assumption regarding the reactor cavity floor liner to be valid, a failure of area 6 in^2 or more would have to occur in each of the 8 reactor cavity floor liner sections.

Scenarios that would tend to increase the likelihood that 6% hydrogen could be generated prior to the 10-mile EPZ evacuation time include additional reactor cavity wall or pipe way cell liner failures and a liner failure prior to the time that fuel debris enters the reactor cavity. The latter scenario would be more possible for the PLHS events since sodium at 1600°F could enter the reactor cavity and cause liner failure prior to RV bottom head penetration.

In addition to considering the potential for hydrogen buildup, one must consider the failure of the operator to detect a 6% hydrogen concentration and to take appropriate action. (This would be considered in the VPS event.) Another possibility is that the operator recognizes that he must vent but does not, for example, because of a direct order from the state governor or other official.

One concern is that the hydrogen filters on the hydrogen sampling line could become plugged and prevent the operator from obtaining a reliable indication of the hydrogen concentration in containment. The hydrogen sample must be filtered to remove solid or liquid reaction products. Tests performed at Hanford Engineering Development Laboratory on hydrogen sampling line filters similar to those of the CRBRP design showed that free CO_2 reacted with the NaOH and Na_2O on the filter causing it to

plug.[40] The potential for hydrogen filter plugging would then depend on the probability that a significant amount of free CO_2 can reach the filter. An analysis performed by Westinghouse indicated that the free CO_2 load to the filter in the Hanford Test was over three orders of magnitude larger than it was estimated to be for a CRBRP core-melt sequence.[41] Only a small amount of CO_2 was predicted to reach the filter in the CRBRP core-melt sequence because of the tendency for CO_2 to react with the NaOH and Na_2O already in the RCB atmosphere. CO_2 would be vented to the RCB atmosphere from the reactor cavity floor liner vent early in the scenario when the calcium carbonate in the limestone concrete decomposes. (Recall from Section 12.3 that the CO_2 released to the sodium pool will react to form Na_2O and C and will remain below the surface of the pool.)

It should also be noted that the CACECO model used by the CRBRP Project does not consider production of CO_2 since the CACECO model was developed for FFTF which utilizes basalt rather than limestone concrete.

In summary, based on the above analysis performed by Westinghouse, it appears that arguments can reasonably be made as to why the free CO_2 loading on the filter would be minimal. What is uncertain at this point, however, is the amount of CO_2 and NaOH or Na_2O that would be required in order to result in plugging. On this basis, hydrogen sampling line plugging cannot be ruled out.

If hydrogen concentrations well in excess of 6% were generated, the venting process would increase the hydrogen concentration even more. If

the hydrogen concentration was great enough for detonation, the action of purging would allow detonation to occur by allowing oxygen to enter the RCB atmosphere.

It is not a foregone conclusion that hydrogen detonation will fail containment. The failure potential would depend on the reactor containment building's ability to withstand a shock wave, etc. An estimate of the probability of containment failure would require two steps: (1) a calculation of the shock wave pressures as a function of time and (2) a structural analysis of the response of the containment to the impulsive loadings.[28]

Polar Crane Differential Thermal Expansion Fails Containment

This failure mode was discussed in Section 12.4 and in Appendix B, Section 5. Unless an ignition source is present, hydrogen burning would not occur. In Reference 1 ignition was predicted to occur slightly before sodium boiling at 9 hours. Hydrogen burning would most likely occur very near the threshold between CM(E) and CM(L). Given that the time required to meet the ignition criteria is fixed at about 9 hours, it is not deemed likely that the hydrogen flame would occur much earlier than the time predicted in Reference 1. Thus, it is expected that this scenario will be of greater importance to the CM(L) event. It should also be mentioned that although the PLHS events may involve early hydrogen production (i.e., before bottom head penetration) the relatively small margin to boiling expected for the PLHS will cause the ignition criteria to be met earlier. If the ignition criteria are met earlier

the hydrogen will burn sooner. This will reduce the chance of hydrogen buildup beyond 6% but increase the failure probability associated with heatup of the polar crane.

RC to RCB Vent Closure Fails RC

The consequences of an inadvertent closure of the reactor cavity to reactor containment building isolation valves were addressed earlier in Section 12.4. The major concern is that the extent of reactor cavity pressurization will cause the reactor cavity structure to fail. This could lead to interaction of sodium with cell 105 which could potentially lead to containment overpressurization.

Another concern is the accumulation of hydrogen in the reactor cavity that could be suddenly introduced into the RCB upon a later opening of the isolation valves downstream of the rupture disk. Reference 1 considered such a scenario and assumed that the isolation valves were closed for 17 hours, allowing the pressure in the reactor cavity to build-up. Upon opening the isolation valves the pressure in containment after cavity depressurization was found to be 30 psig. Containment failure was not predicted since the steel shell temperature at the time of the spike was 110°F.

Since failure was not predicted after 17 hours, it is deemed unlikely that it would occur prior to the 10-mile EPZ evacuation time. Structural failures of the reactor cavity may be more limiting in this case.

Based on the above discussions the following sequences will be quantified. Note that sequences, rather than core-response event tree end states, are being quantified here since the CM(E) event is most dependent on these sequences.

NLF Failure Sequences, OVC Success Sequences

For all of the core-response event tree end states that follow this sequence, with the exception of the PLHS events, the ignition criteria would not likely be met until very near eight hours and 45 minutes. Thus, insufficient time would be available for a hydrogen flame to cause containment damage, such as by heatup of the polar crane. Also, as previously discussed it is doubtful that the hydrogen concentration could buildup to 6%, let alone detonable limits, in that time frame. Perhaps the greatest challenge would come from the PLHS sequences since the ignition source could be present early. On this basis the following values were assigned:

Core-response end states other than the PLHS:

Yes: 0.99

No: 0.01

PLHS (LHSE and LHSL):
For both Na early and Na late

Yes: 0.9

No: 0.1

OVC Failure Sequences

This sequence involves early liner failure and exhibits a strong potential for the early presence of an ignition source. On this basis the following probabilities were assigned:

All core-response end states:

Yes: 0.9

No: 0.1

RCV Failure Sequences

Based on previous discussions it is deemed that containment failure is very unlikely prior to 8 hours and 45 minutes for the RCV failure sequence.

Yes: 0.99

No: 0.01

12.6.9 Late Containment Integrity Maintained

In addition to the potential containment failure mechanisms identified for CM(E) there were two additional mechanisms identified for the CM(L) event. These additional failure mechanisms are briefly described below.

Large Time Interval Between H_2 Generation and Ignition Allows Buildup of H_2

In the CRBRP Reference 1 base case analysis the liners are assumed to vanish the instant that fuel debris enters the reactor cavity. Sodium-concrete reactions commence, producing hydrogen which, in the absence of an ignition source, begins to accumulate. When the ignition criteria are met at 9 hours, the hydrogen is assumed to burn instantaneously, causing

a rapid pressure rise in containment of about 25 psig. It is theoretically possible that, because of larger margins of subcooling than those predicted in Reference 1, the ignition criteria may not be met for tens of hours. This could allow hydrogen concentrations to build up to limits that would require venting. If venting were not initiated, or if venting were delayed, a potential for hydrogen detonation would exist.

It is very unlikely, however, that the operator would fail to recognize the need to vent, unless the hydrogen filters become plugged as discussed previously in Section 12.6.8.

Interaction of Cell 105 with Sodium

For extended liner integrity cases there is a possibility of reactor cavity wall structural failures that lead to interaction of cell 105 water with the reactor cavity sodium. This scenario was previously discussed in Section 12.4 and will not be elaborated here.

The potential containment failure modes discussed in Section 12.6.8 are also applicable here. In fact, because these events are not limited in time for the CM(L) mode as they were for the CM(E) mode, the potential threat to containment from these events appears to be greater.

Of special concern is the effect of the hydrogen flame on the polar crane. As in Section 12.6.8, the quantification of the CM(L) mode will require consideration of sequences rather than core-response event tree end states.

Both NLF Failure and Success Sequences and OVC Failure Sequences

Considering all of the threats to containment integrity discussed in Section 12.4 and in Appendix B, Section 5, the following values have been subjectively assigned for CM(L):

Assigned Probabilities:

Yes: 0.9

No: 0.1

RCV Failure Sequences

Based on discussions in Section 12.6.8 it would appear that containment failure cannot be ruled out if the reactor cavity vent isolation valves are closed for more than about 20 hours. Given that the valves are still closed at about 9 hours, it is unlikely that they would be opened prior to 20 hours. If the valves are opened after 20 hours, the pressure in containment may exceed the ultimate capacity. Even if the valves are not later opened, structural failure of the reactor cavity leading to interaction of cell 105 and sodium could cause containment to fail. On this basis the following values were assigned:

Assigned Probabilities:

Yes: 0.1

No: 0.9

12.6.10 Vent/Purge System

The VPS event falls within the second category of events discussed in Section 12.6.1 and will be quantified in Section 13. The fault tree

analysis used to quantify the VPS event is shown in Appendix A21. Only total event success or failure will be considered here:

Yes: $\overline{VPS}$

No: VPS

12.6.11 Annulus Cooling System

The ACS event falls within the second category of events discussed in Section 12.6.1 and will be quantified in Section 13. The fault tree analysis used to quantify the ACS event is shown in Appendix A20. Only total event success or failure will be considered here.

Yes: $\overline{ACS}$

No: ACS

12.6.12 Containment Cleanup System

The CCS event falls within the second category of events discussed in Section 12.6.1 and will be quantified in Section 13. The analysis used in quantifying the CCS event is described in Appendix A21. Only total event success or failure will be considered here.

Yes: $\overline{CCS}$

No: CCS

12.6.13 Meltthrough Does Not Occur

The extent of fuel penetration into the basemat following sodium boildry is an area of uncertainty. In Reference 1, the CRBRP Project performed a parametric study of post boildry phenomena. In the study, the fuel was assumed to penetrate through the concrete by melting. Sufficient experimental evidence is available to suggest that fuel oxides and molten steel would not react with concrete oxides.[30] However, the CO_2

and H_2 gases released from the heated concrete would act to oxidize the molten steel. The oxidized steel (iron) would lower the melting point of the oxide mixture allowing melting to occur.[1]

The parametric study performed in Reference 1 considered the heat transfered from a molten pool of fuel and concrete to the surrounding environment. In the study, the thermal resistance of the crust overlying the molten pool was varied from 0.1 hr-ft^2-°F/Btu (low resistance) to 1.0 hr-ft^2-°F/Btu (intermediate resistance) and then to 10 hr-ft^2-°F/Btu (high resistance). The percentage of heat transfered upward ranged from as high as 90% for the low resistance case to as low as 5% for the high resistance case. The heat not transfered upward is distributed radially and downward to the surrounding-concrete structures. In both the low and intermediate resistance cases the reactor vessel head and the reactor cavity wall failed prior to 600 hours after the HCDA. (This failure time could be less if the ACS failed to operate.)

No structural failures were predicted for the high resistance case as long as the annulus cooling system was operating. However, the high resistance case resulted in the greatest penetration of the basemat. This penetration was about 20 feet below the reactor cavity floor liner, or only about six feet above the bottom of the foundation mat (See Figure 12.6-2). Although the fuel would not penetrate the remaining six feet, cracking and general concrete degradation that would allow contact of radionuclides with ground water cannot be ruled out at this time.

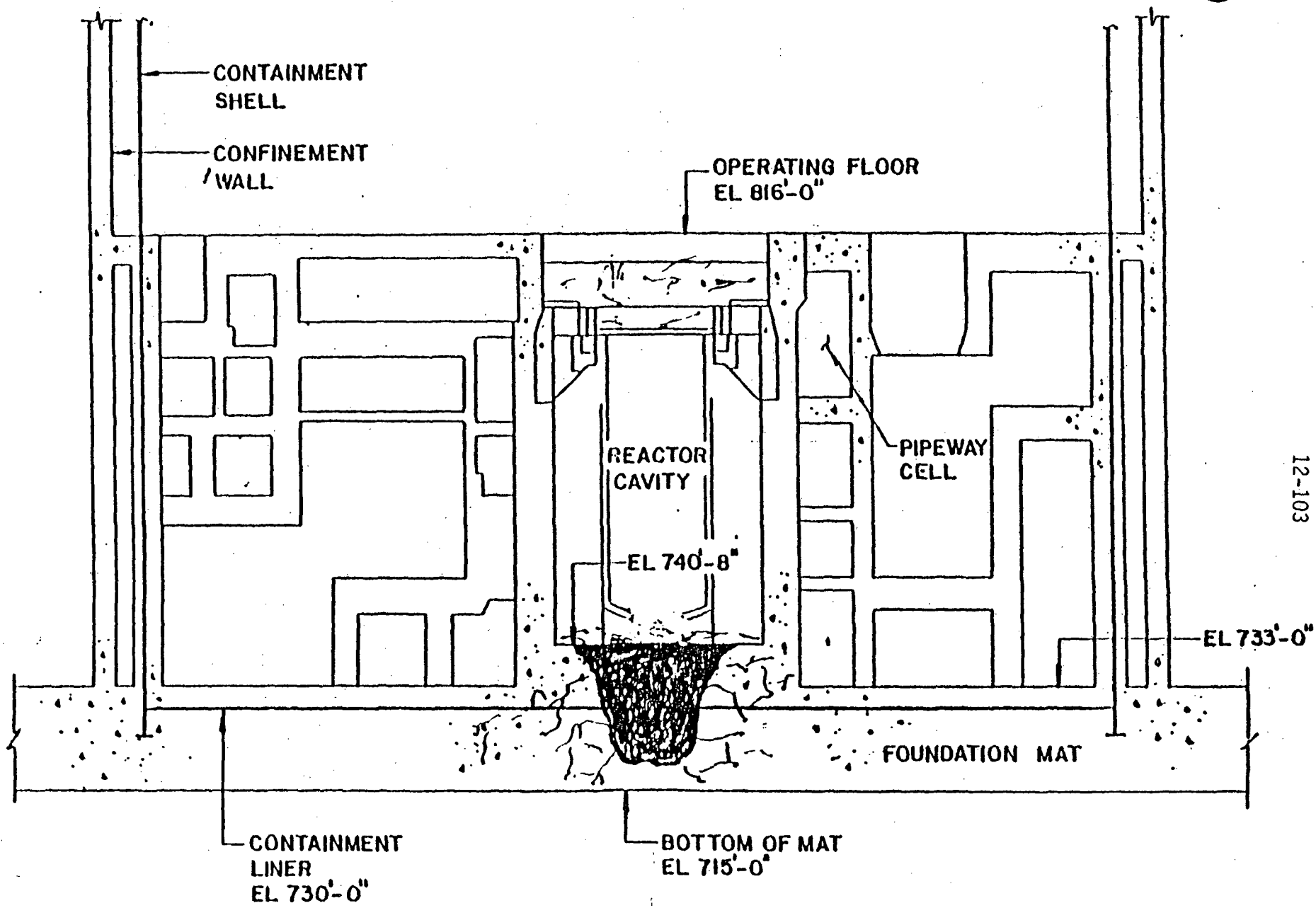


Figure 12.6-2. RCB Cross Section Showing Basemat Penetration

The consequences of a non-coolable debris bed early in the core-melt scenario were assessed by the CRBRP Project in a letter supplied to NRC in November, 1982.[36] As in the CRBRP-3 Volume II (Reference 1) base case analysis, the reactor cavity floor liner was assumed to vanish at the time of failure. However, unlike the base case analysis, the fuel debris is assumed to sink below the reaction products layer. The reaction products layer would then prevent sodium from coming into contact with the fuel debris allowing the debris to melt. This scenario is unlikely since the reactor cavity liner is not expected to "vanish" and since the reaction product layer would provide some support of the fuel debris in the regions of liner failure. Nonetheless, the penetration distance into the basemat prior to sodium boildry was predicted to be only five feet.

In summary, based on available studies, it does not appear that basemat penetration is inevitable either before or after sodium boildry. Furthermore, the case assessed in Reference 1 was for a ULOF HCDA. Since the decay heat levels of the fuel debris would be greatest for accidents within this category, the maximum penetration depth calculated in the parametric study is believed to be bounding. The actual penetration depth would be expected to be less than the 20 feet presented for the high resistance case. Contact of radionuclides and groundwater cannot be ruled out for penetration depths as great as 20 feet. However, since total penetration would not be expected to be this great, it is deemed unlikely that basemat penetration and/or significant radionuclide-ground water contact would occur following core melt.

All sequences:

Yes: 0.9

No: 0.1

12.7 SUMMARY AND CONCLUSIONS

A detailed summary of the containment-response analysis as well as the core-response analysis will be provided in Section 13. In this section the containment-response event tree was developed based on an understanding of the phenomena associated with an LMFBR core-melt accident. Although only the phenomenological branch point probabilities were assigned in this section, a best estimate outcome can be predicted for the case where the beyond-the-design basis features operate as required. The quantified containment-response event trees are shown in Figures 12.7-1 through 12.7-5.

Following the entry of sodium and fuel debris into the reactor cavity, the reactor cavity vent rupture disks will open, allowing the reactor cavity to depressurize and the reactor cavity gases to flow into the upper RCB. The concrete behind the cell liners will begin to heat up and eventually release steam and CO_2 to the upper RCB and to cell 105 by means of the behind-the-liner vent system. The reactor cavity floor liner will likely fail early (within the time required to evacuate the Emergency Planning Zone) if fuel debris enters the reactor cavity or if the sodium that enters the reactor cavity is at relatively high temperatures (above 1300°F). However the failures would be local and would propogate very slowly. The remaining reactor cavity liners (e.g., wall liners) would be expected to remain intact for several tens of hours.

Based on the results of a number of experimental programs, sodium-concrete reaction rates would be limited to about 7 inches per hour for 20 minutes followed by one inch per hour until termination. In addition, sodium-concrete reactions were deemed to be "self terminating" due to build-up of a reaction product layer that prevents the sodium from coming into contact with the concrete. When sodium reacts with the steam released from the heated concrete, hydrogen will be produced and released to the upper RCB through the reactor cavity vent system. Hydrogen will not burn, however, until an ignition source is present and the hydrogen concentration is above 4%. Sodium vapor entering the upper RCB is the most likely ignition source in the CRBRP core-melt scenario. Coincidentally, both criteria for hydrogen burning would be met at about 10 hours if the sodium entering the reactor cavity is at a temperature of about 1000°F (~10 hours is required to raise the sodium temperature from 1000°F to saturation). Hydrogen burning could occur sooner after reactor vessel guard vessel penetration for the protected loss of heat sink event since hot sodium (~1600°F) could spill into the reactor cavity prior to vessel failure.

Once the burning criteria are met, the hydrogen in the containment will react with the oxygen in containment. This burning will cause an initial pressure spike in containment that should not exceed the containment pressure capability. The visible flame produced from the hydrogen oxygen reactions could theoretically remain at a height of about 80 feet for five hours. However, certain phenomena, including sodium entrainment and variations of hydrogen concentration with flame height,[40] would have the effect of lowering the flame height, and should be explored in

greater detail before a definitive statement can be made regarding the anticipated flame height. Potential threats to containment integrity as a result of the hydrogen burning scenario were identified. These threats include the differential thermal expansion of the polar crane girder as it is preferentially heated by the hydrogen flame and as it is indirectly heated over the longer term along with the containment atmosphere. Although the total differential thermal expansion of the polar crane was estimated to be greater than a foot in both cases, it was judged that containment failure would not occur, in part due to the greater ductility of the containment steel shell at higher temperatures. The containment steel shell temperatures, although relatively high, are maintained at an acceptable level by operation of the annulus cooling system. Another scenario identified as a potential threat to containment involves the slumping of one of the polar crane girders as it is preferentially heated by the hydrogen flame. Again this scenario was not estimated to fail containment. Although containment failure due to thermal buckling was a concern (particularly in the polar crane region and at the 816 level) it was not specifically addressed in this analysis. In any event, it is anticipated that adequate safety margins exist in the design to accommodate the increased containment shell temperatures.

Threats to containment, not necessarily associated with hydrogen burning, include significant interaction between cell 105 water and sodium and failure of the reactor cavity vent path to open. The only other identified means for significantly pressurizing containment is hydrogen detonation. One possible way in which hydrogen detonation could occur involves failure of the operators to initiate vent and purge

operations when the hydrogen concentration in containment exceeds 6%. This in turn could be caused by a plugging of the filter on the hydrogen sampling line.

In summary, considering all of the potential containment failure modes, loss of containment integrity following evacuation of the emergency planning zone was deemed unlikely. Furthermore, loss of containment integrity prior to this time was deemed even less likely. Although mention was made of containment and beyond-the-design basis systems and their relationship to containment phenomena, no attempt was made in this section to assess the likelihood of these features successfully performing their designated function. The failure probabilities associated with the beyond-the-design basis features for mitigating the consequences of a core-melt accident will be considered in Section 13.

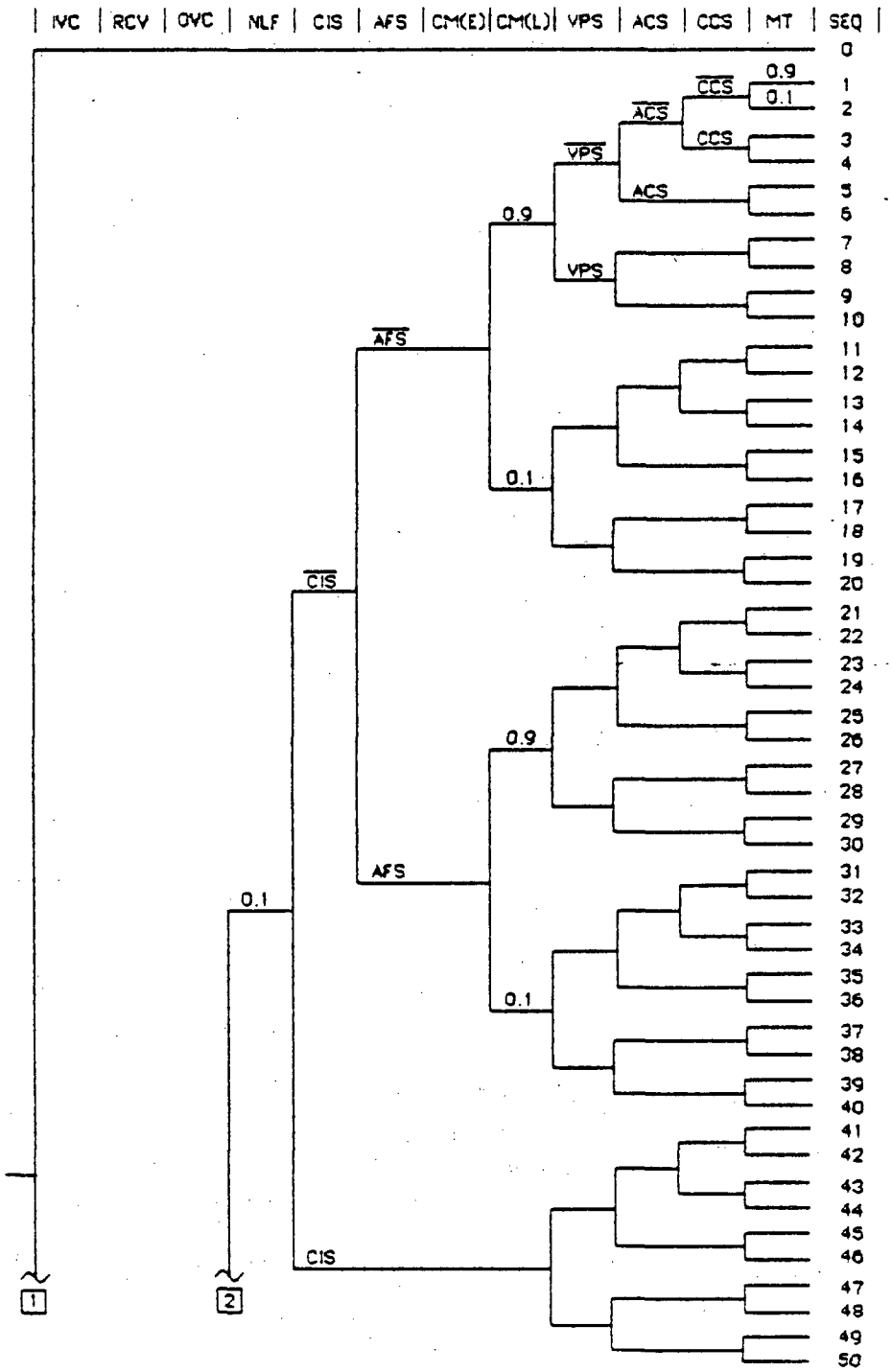
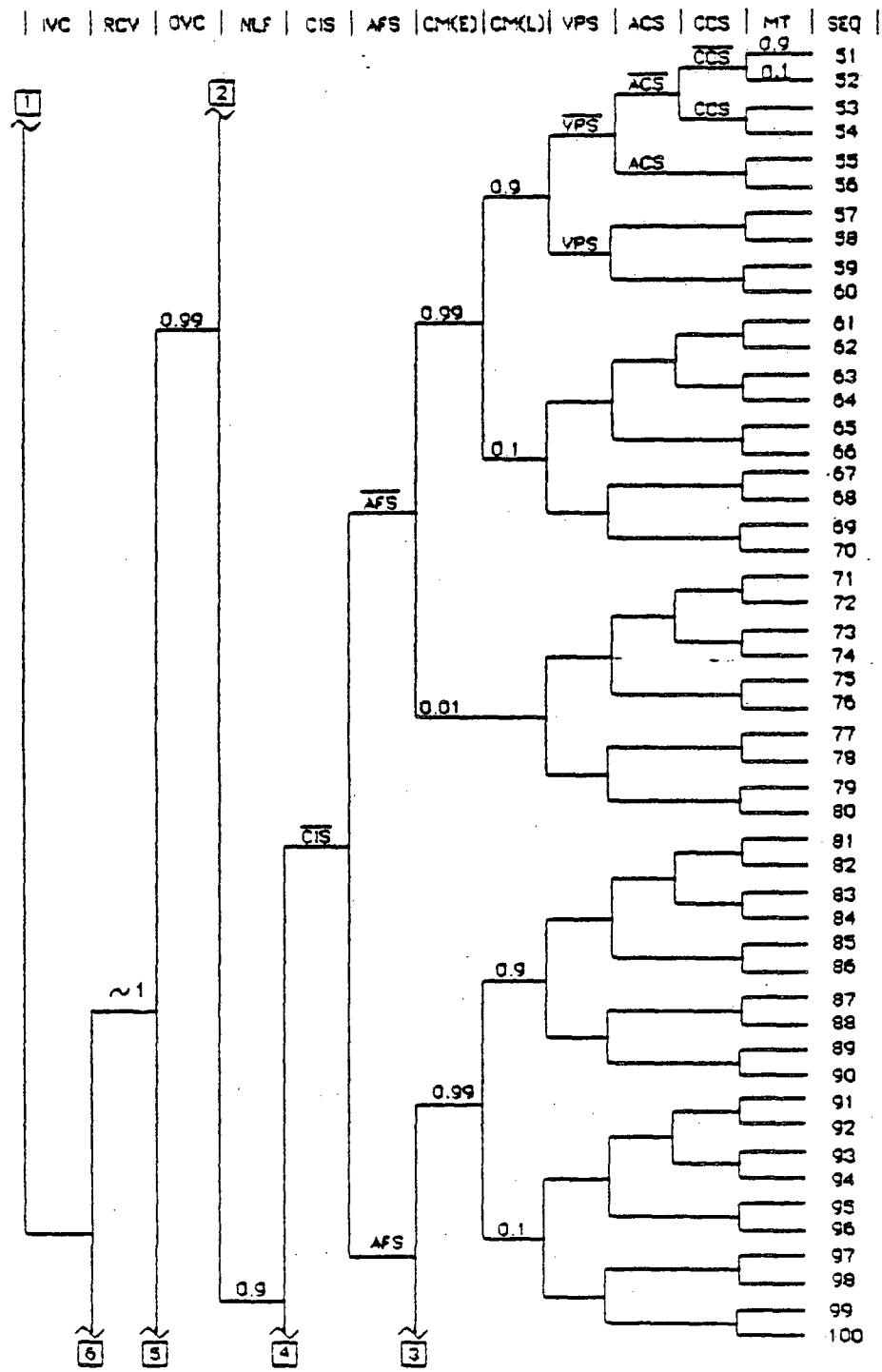


Figure 12.7-1. Containment-Response Event Tree (ULOF, UTOP, and ULHS).



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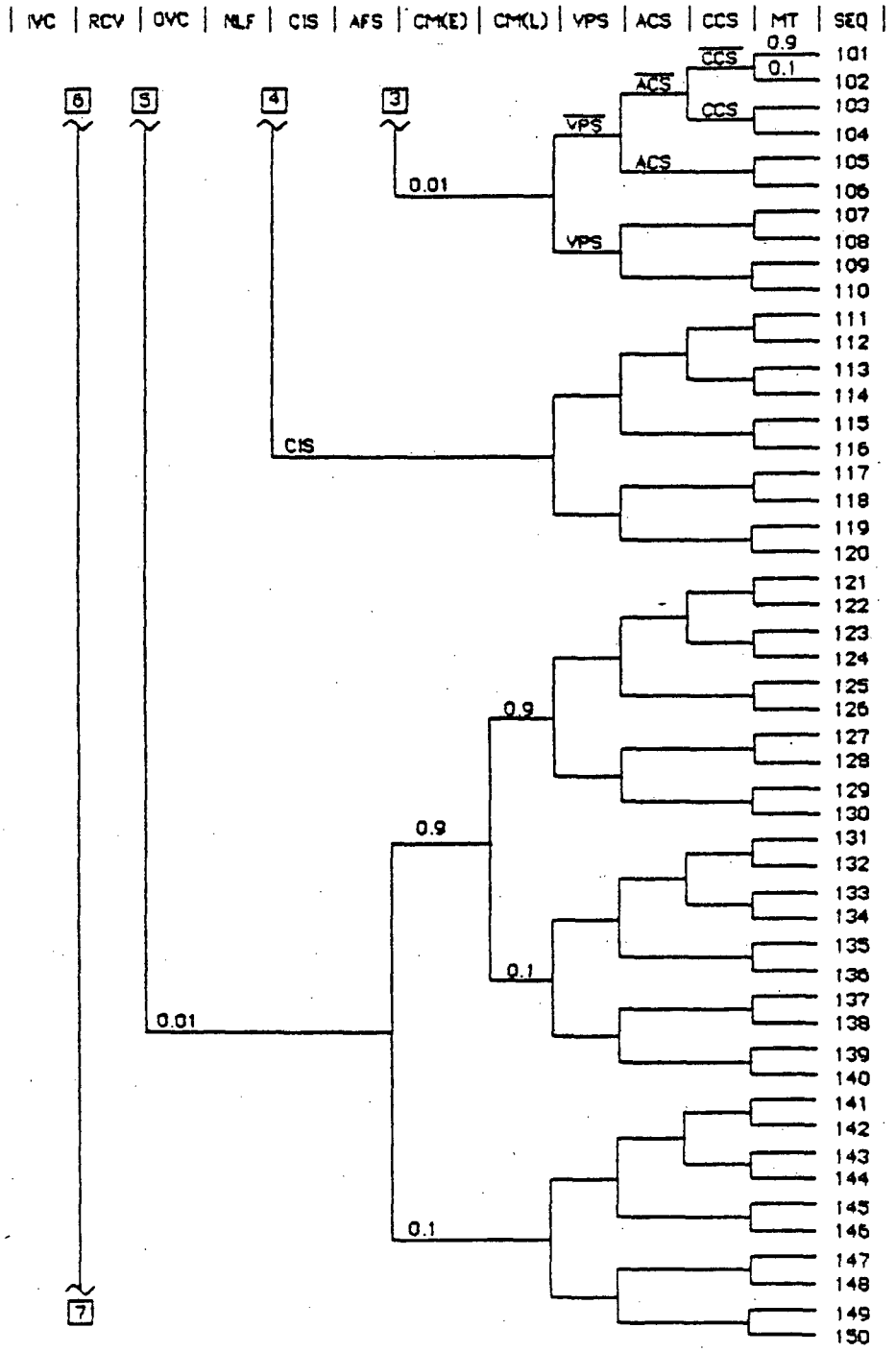

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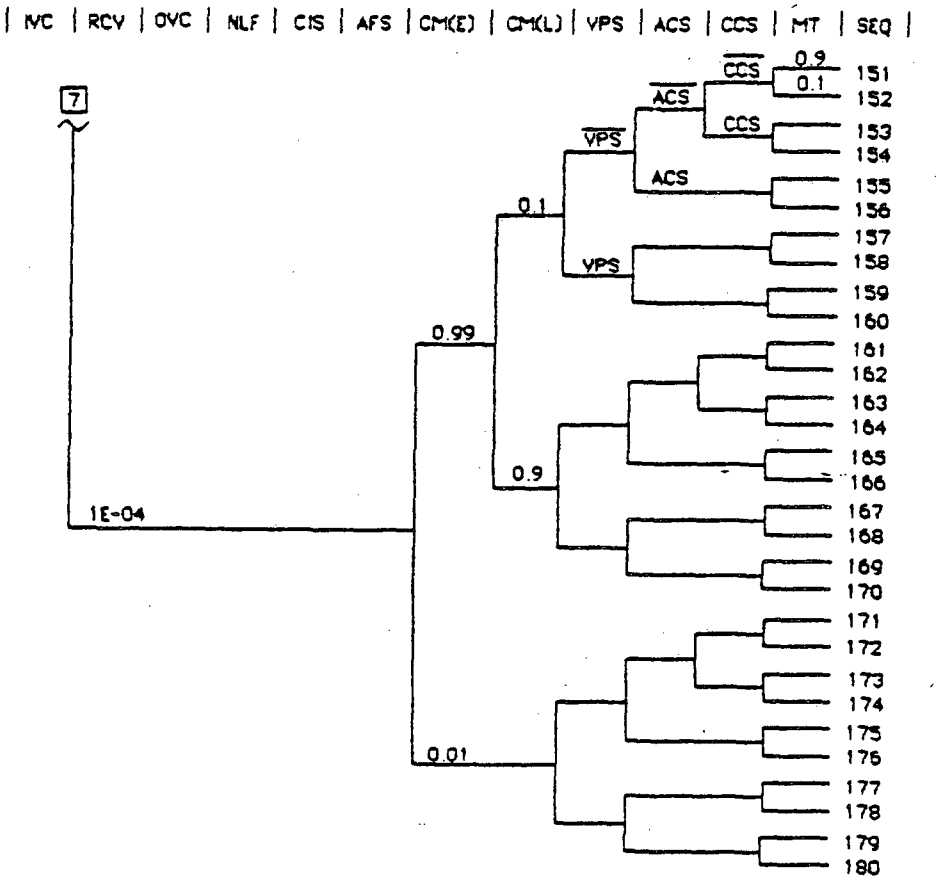

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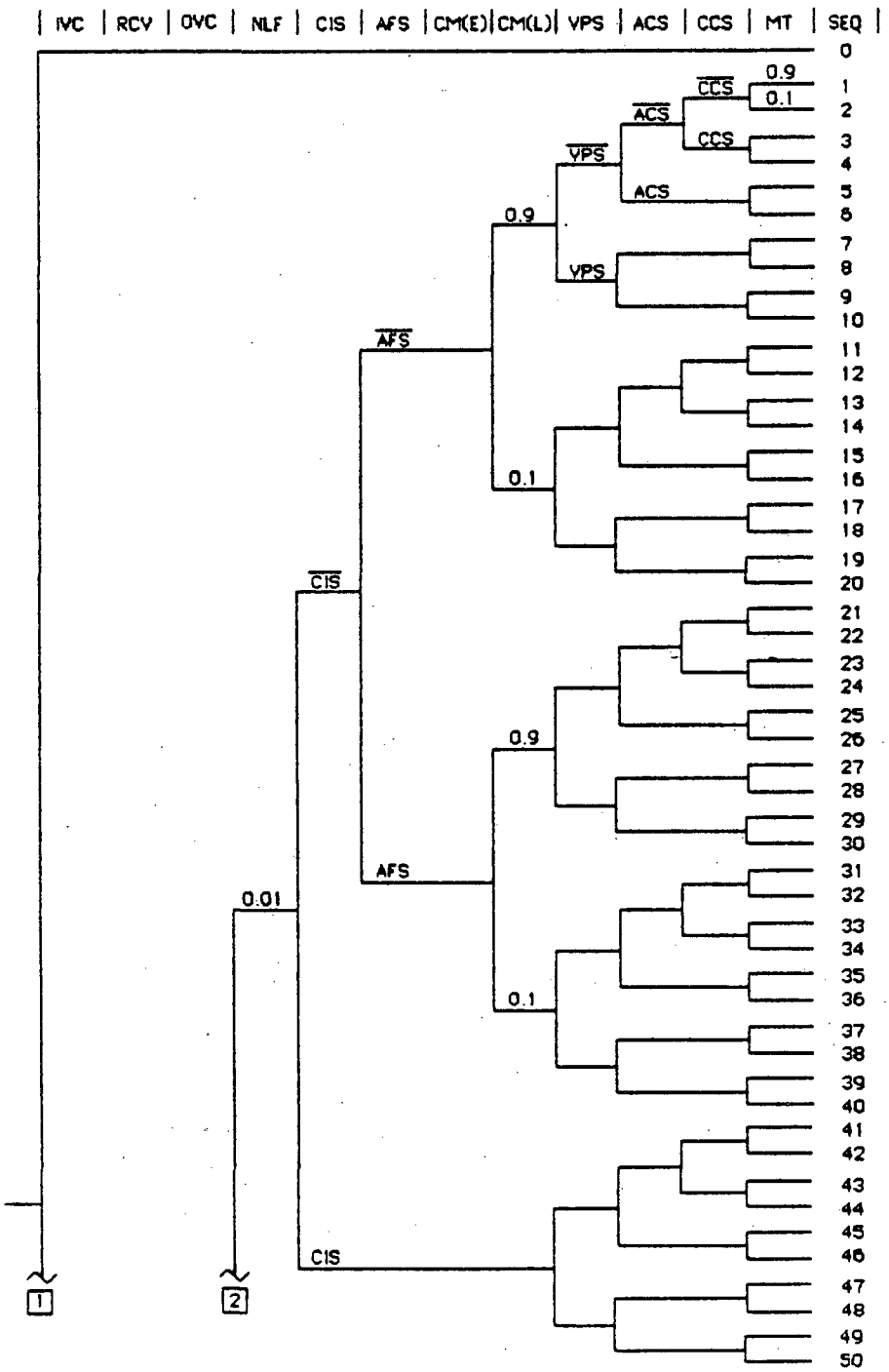
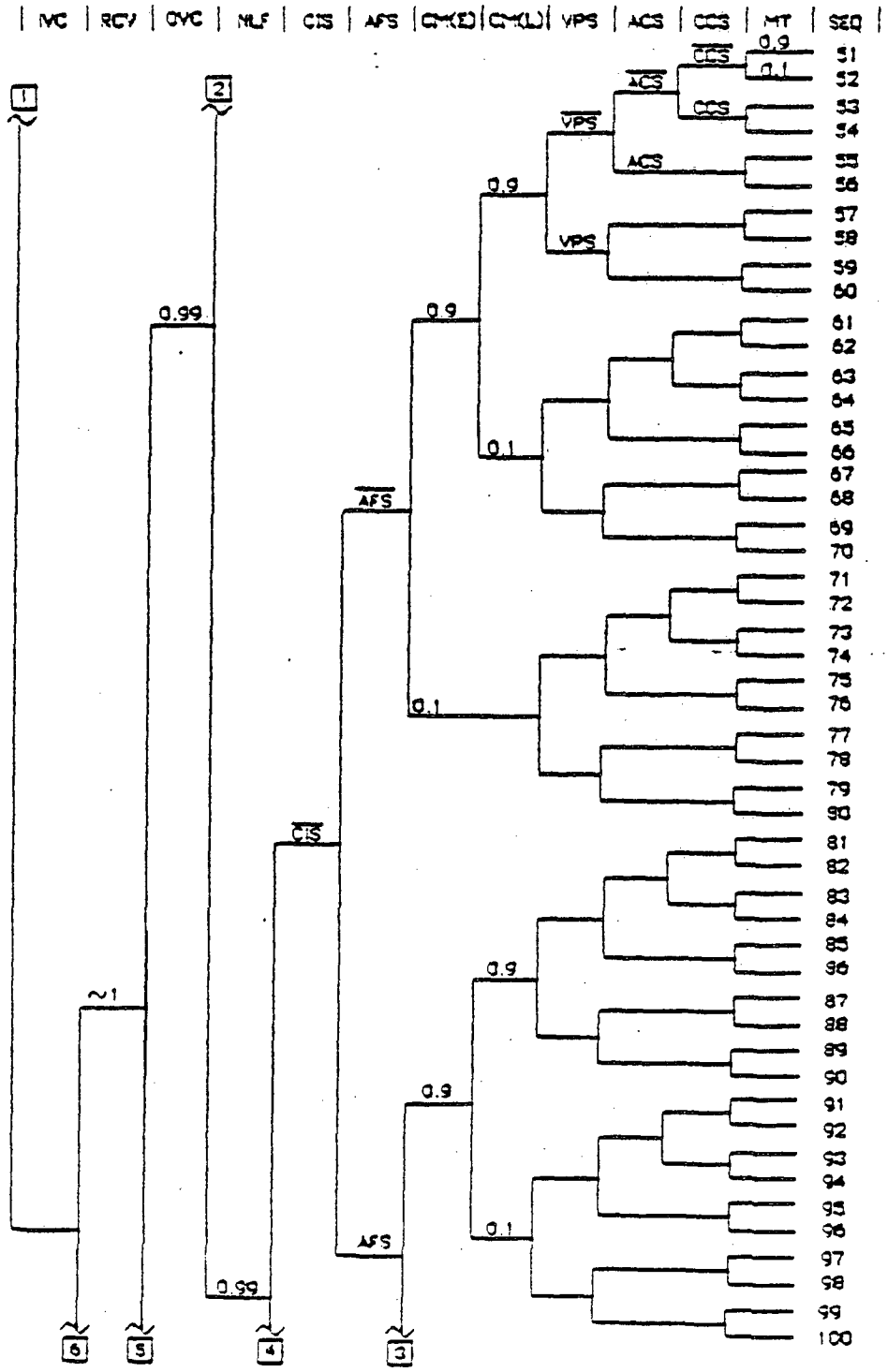


Figure 12.7-2. Containment-Response Event Tree for PLHS (Early and Late) with Na in the Reactor Cavity Before Core Debris.



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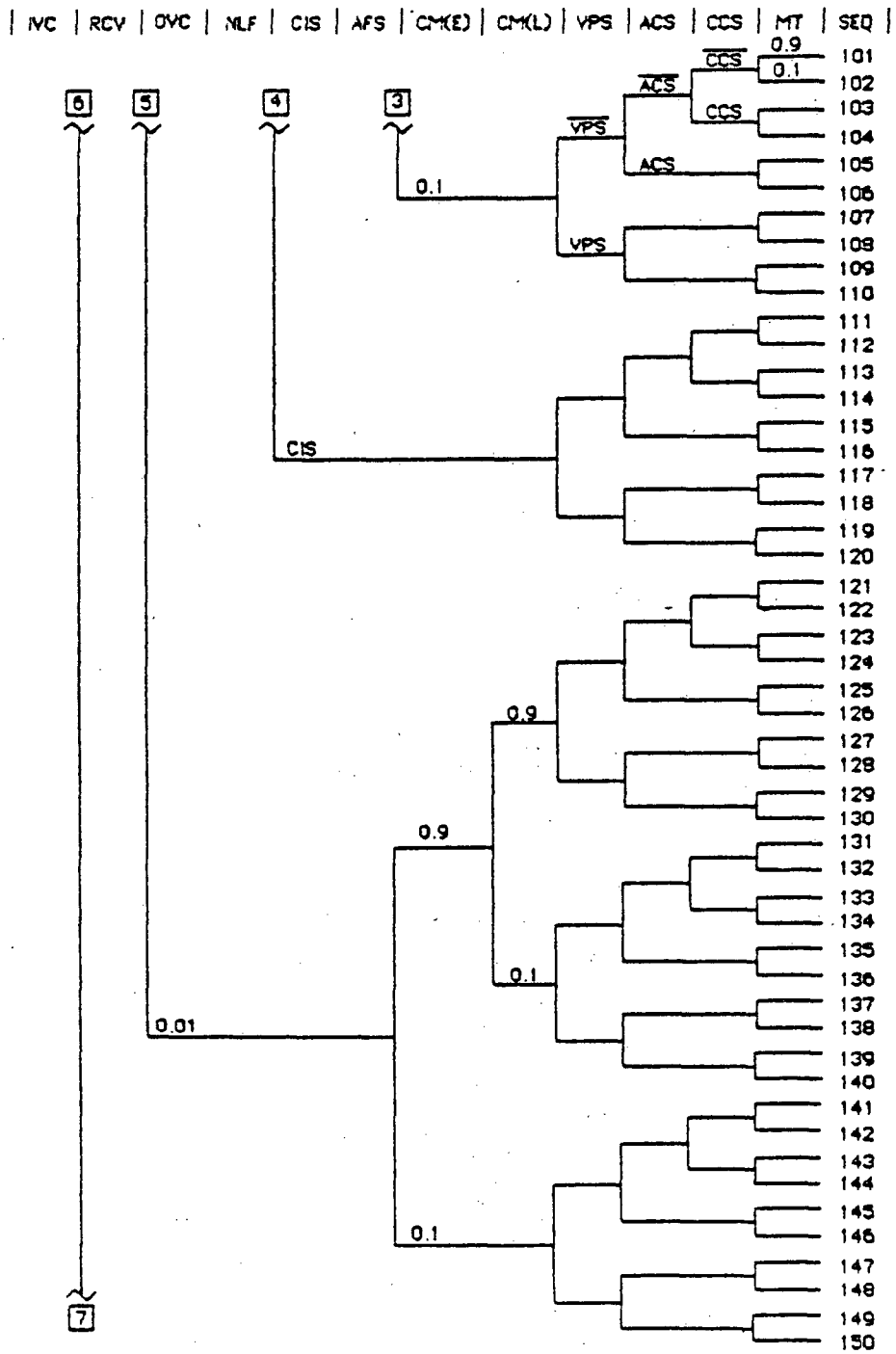

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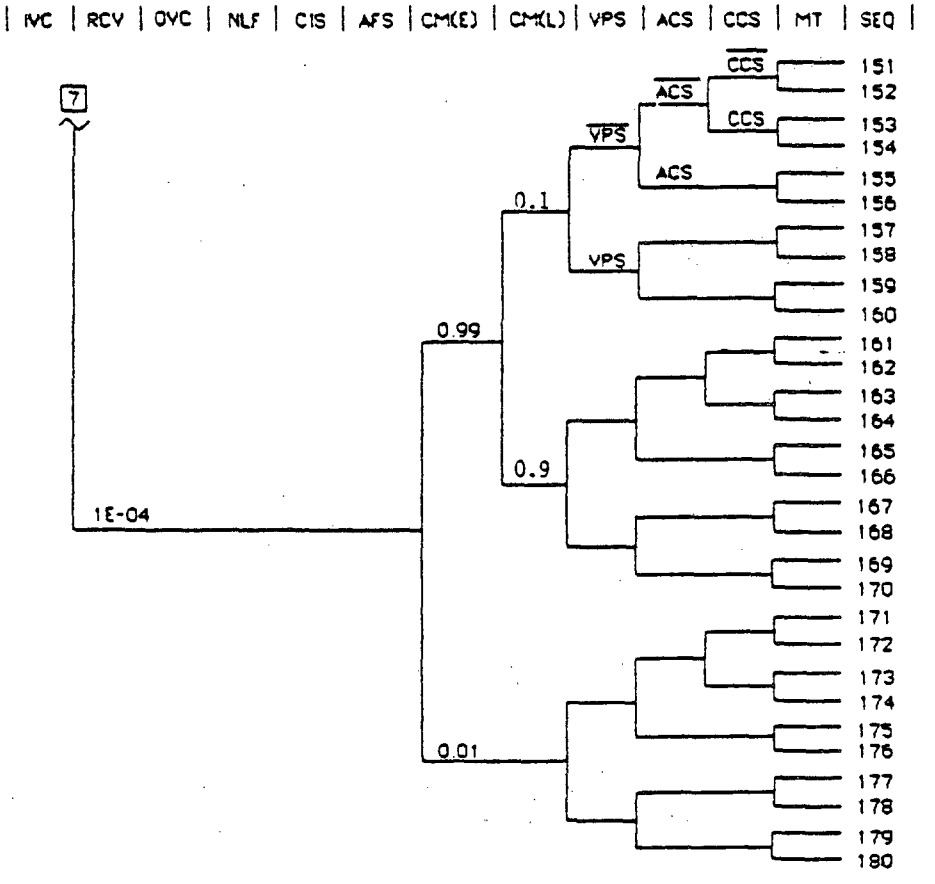

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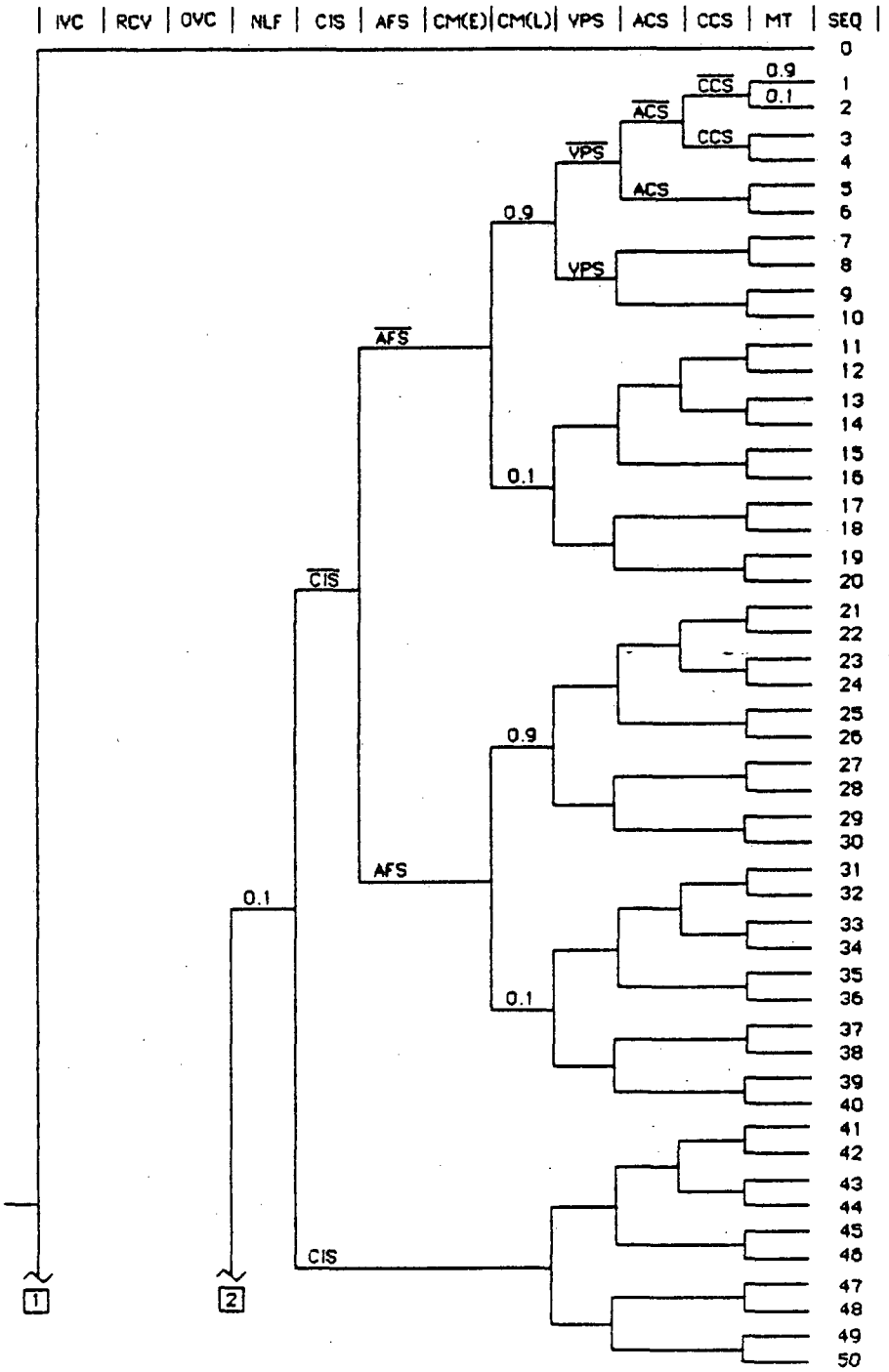
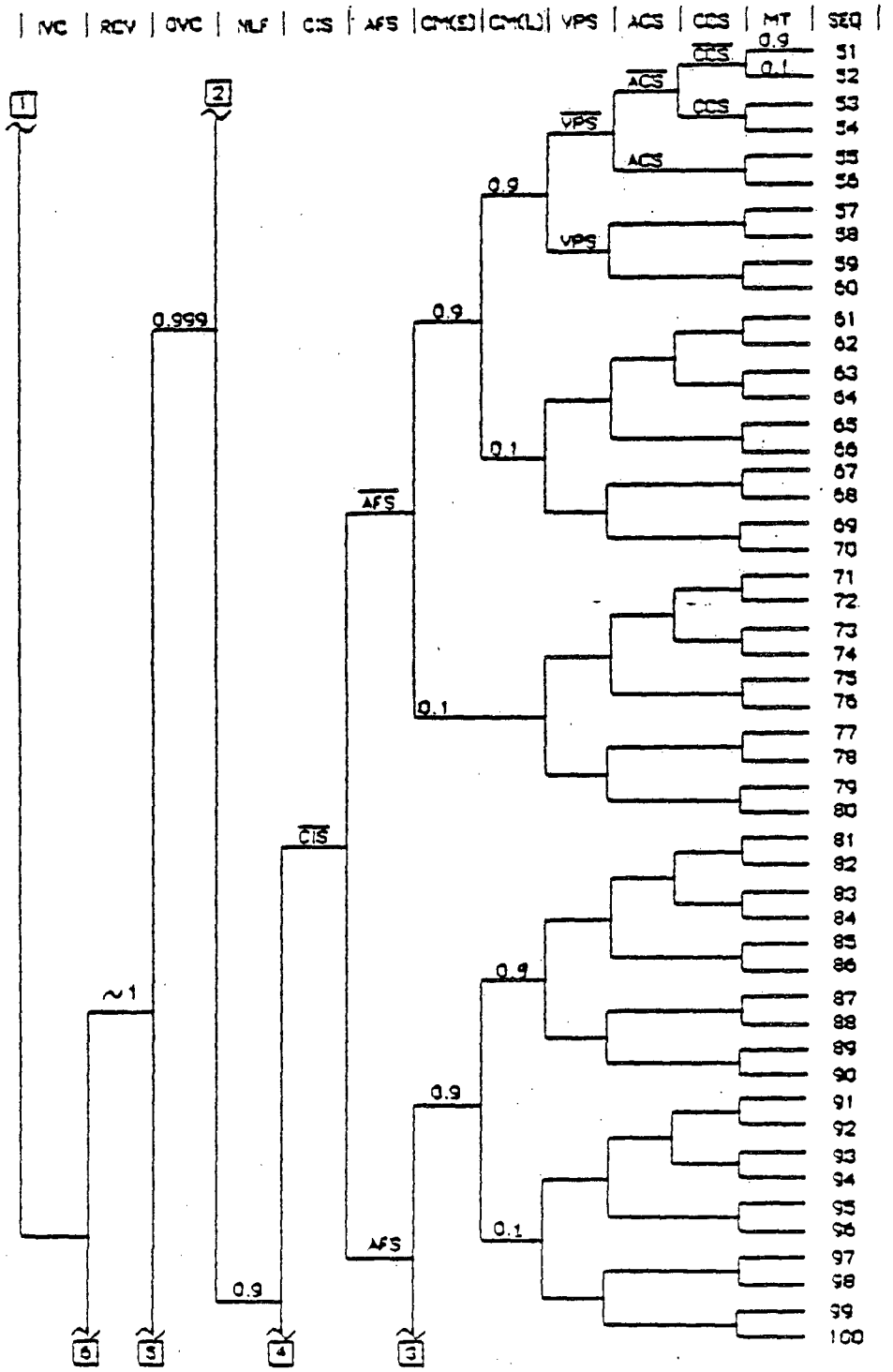


Figure 12.7-3. Containment-Response Event Tree for PLHS (Early and Late) with Fuel in Reactor Cavity Before Na.



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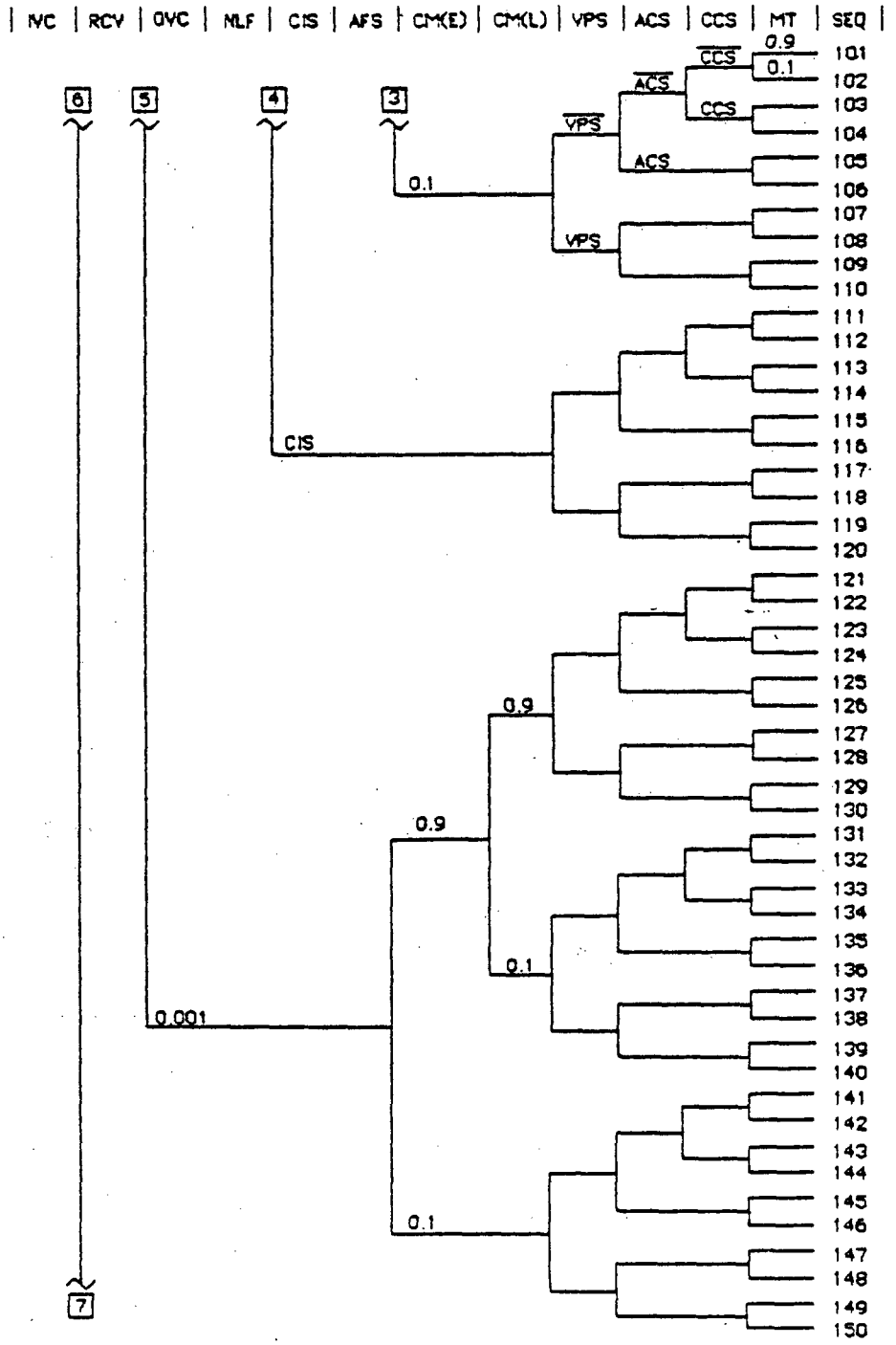

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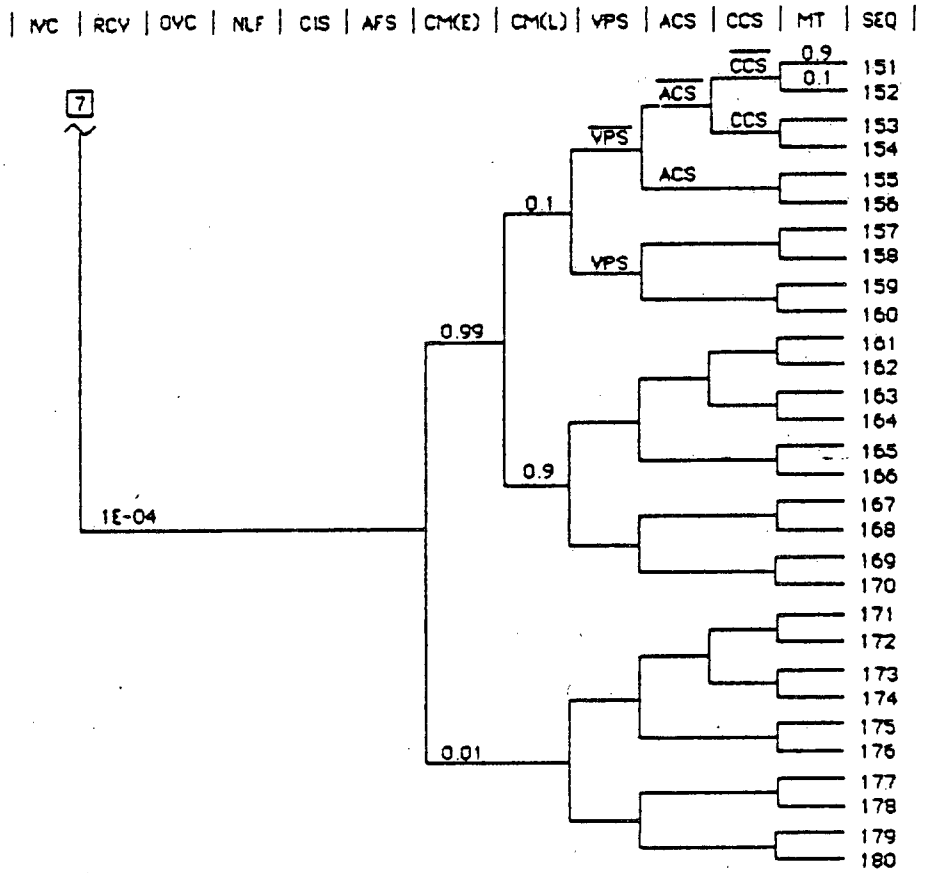

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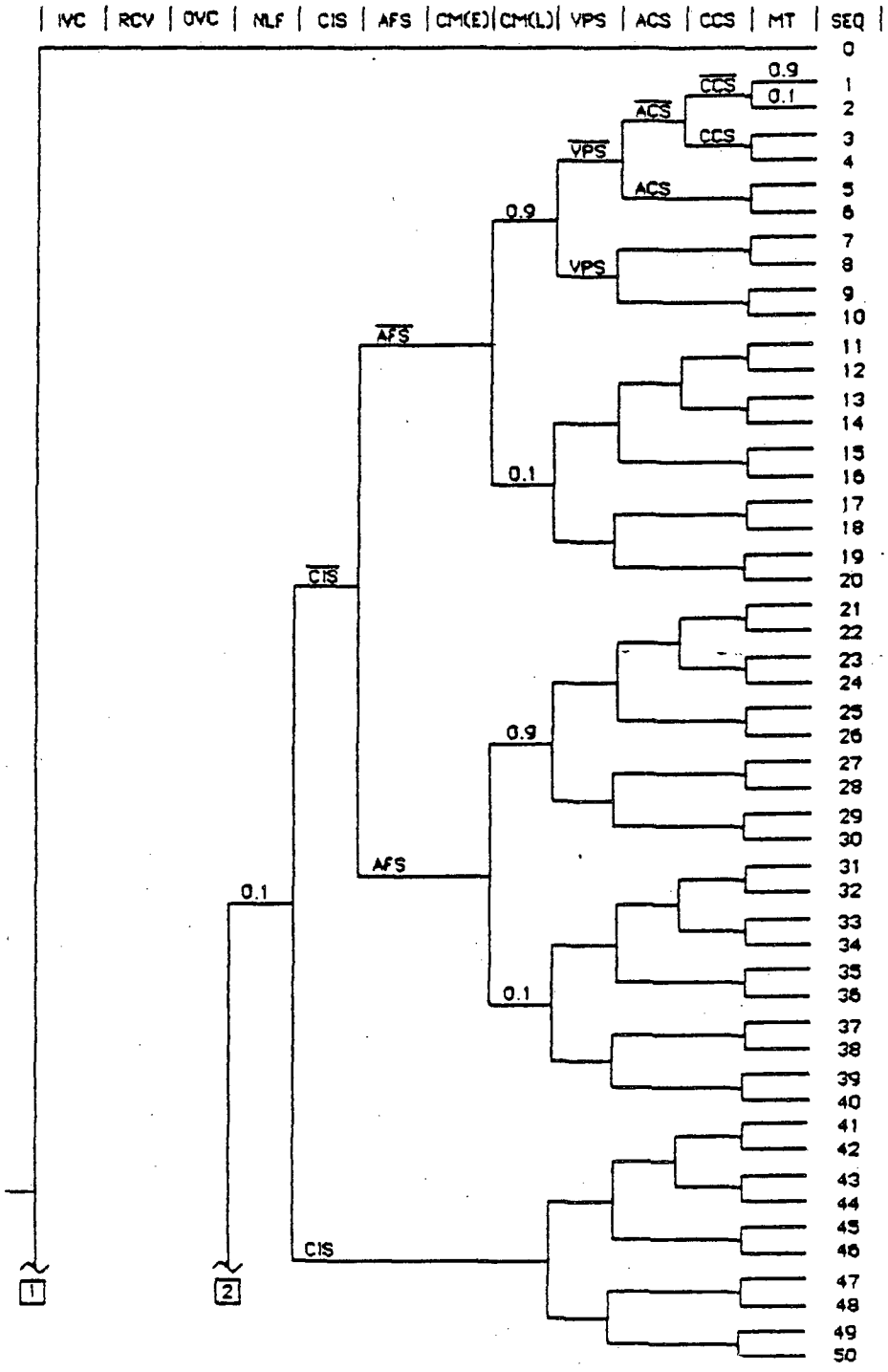
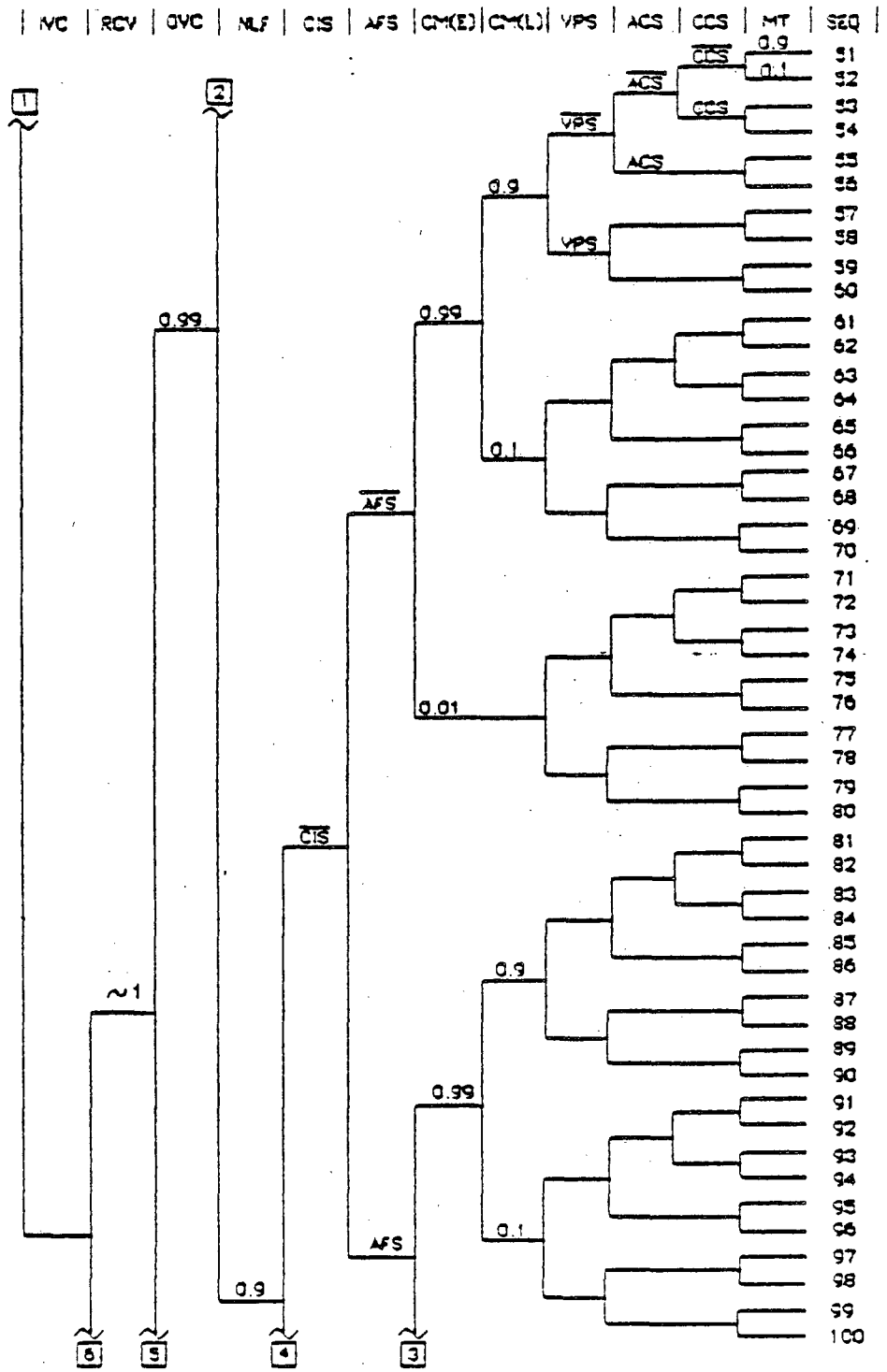


Figure 12.7-4. LOS Containment-Response Event Tree.



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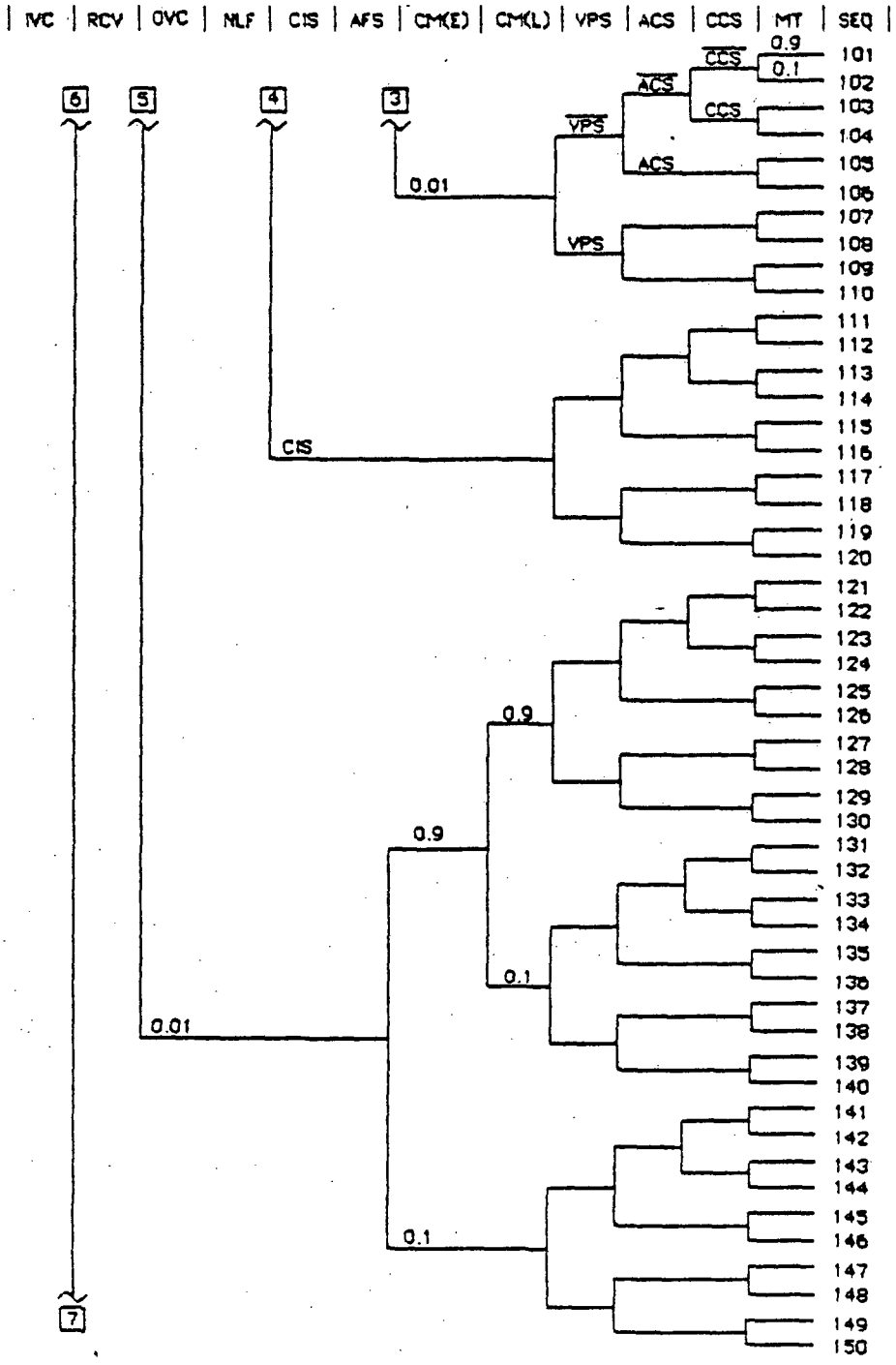

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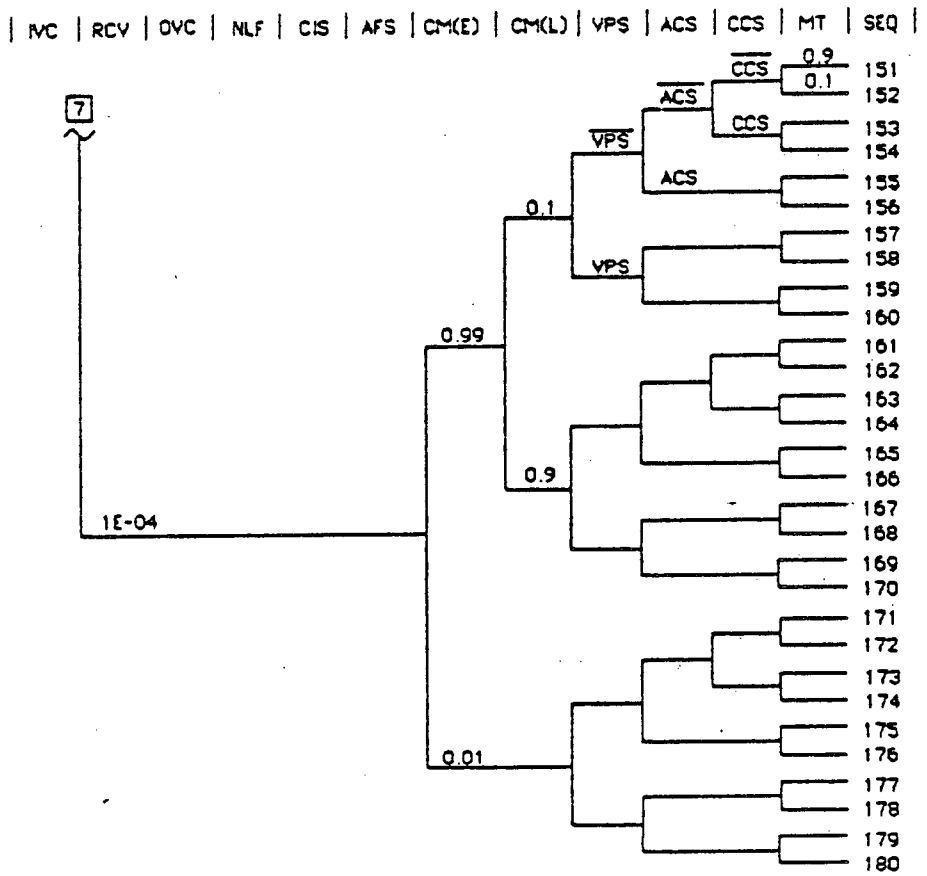

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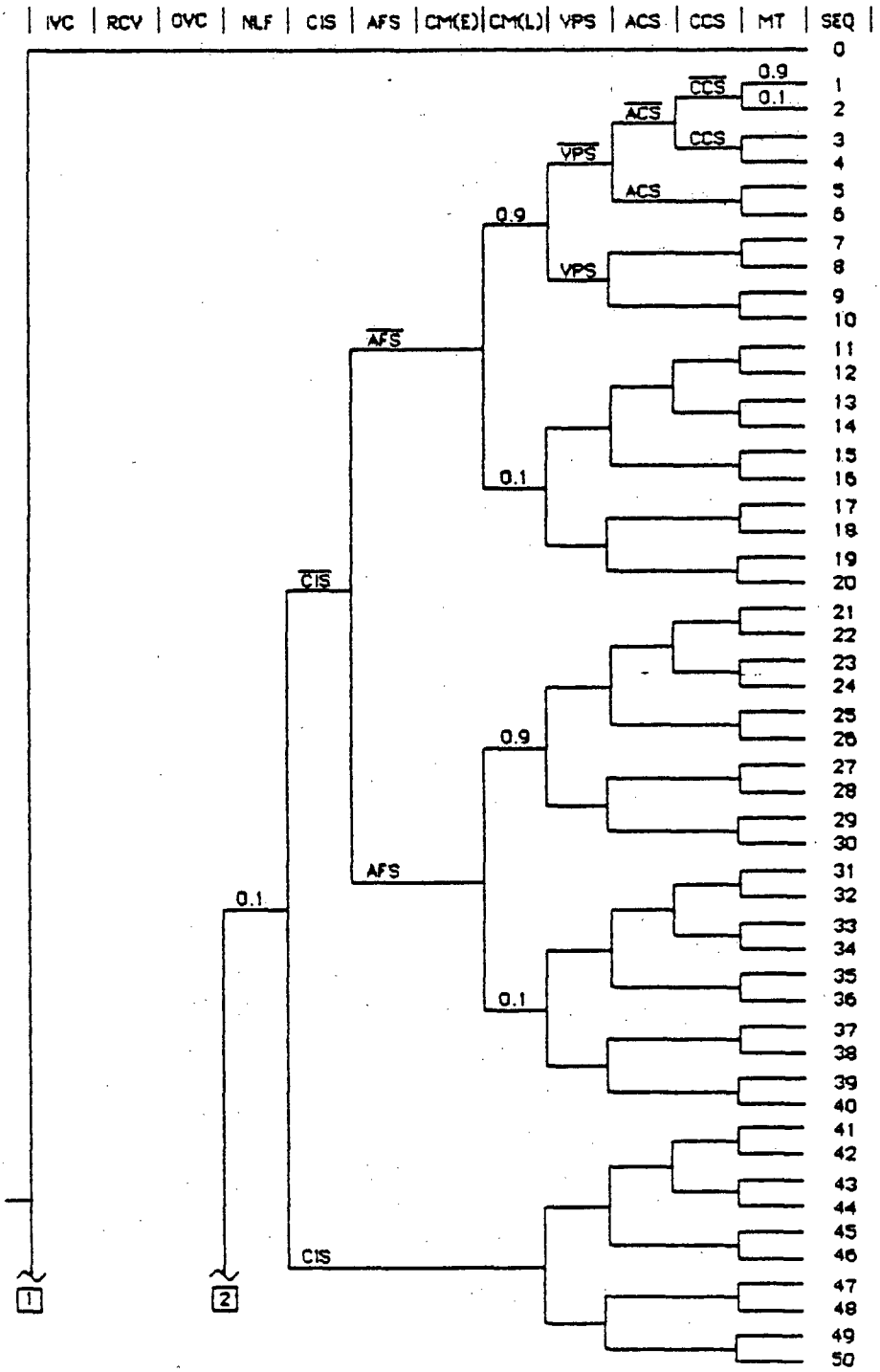
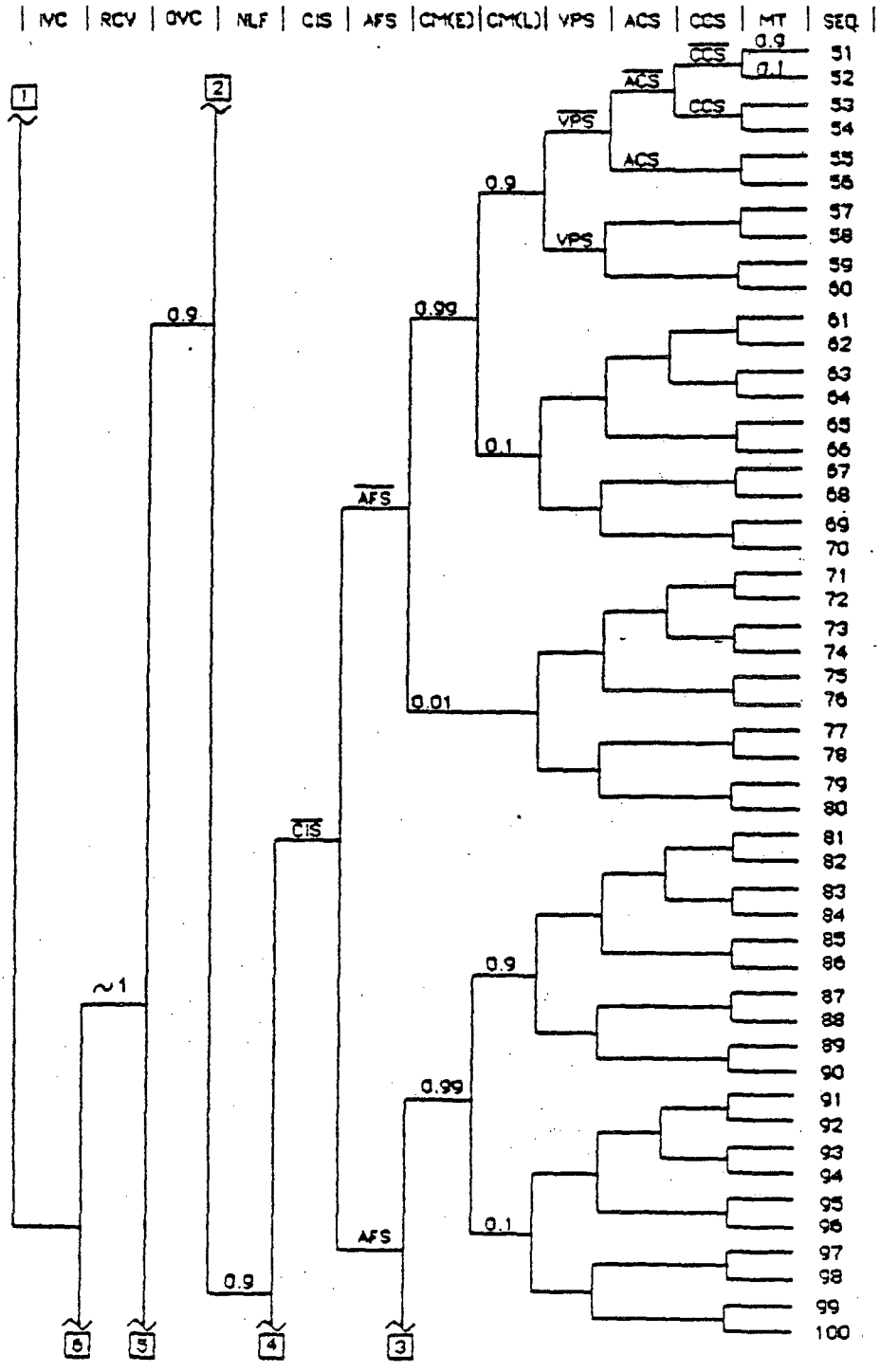


Figure 12.7-5. ULOS Containment-Response Event Tree.



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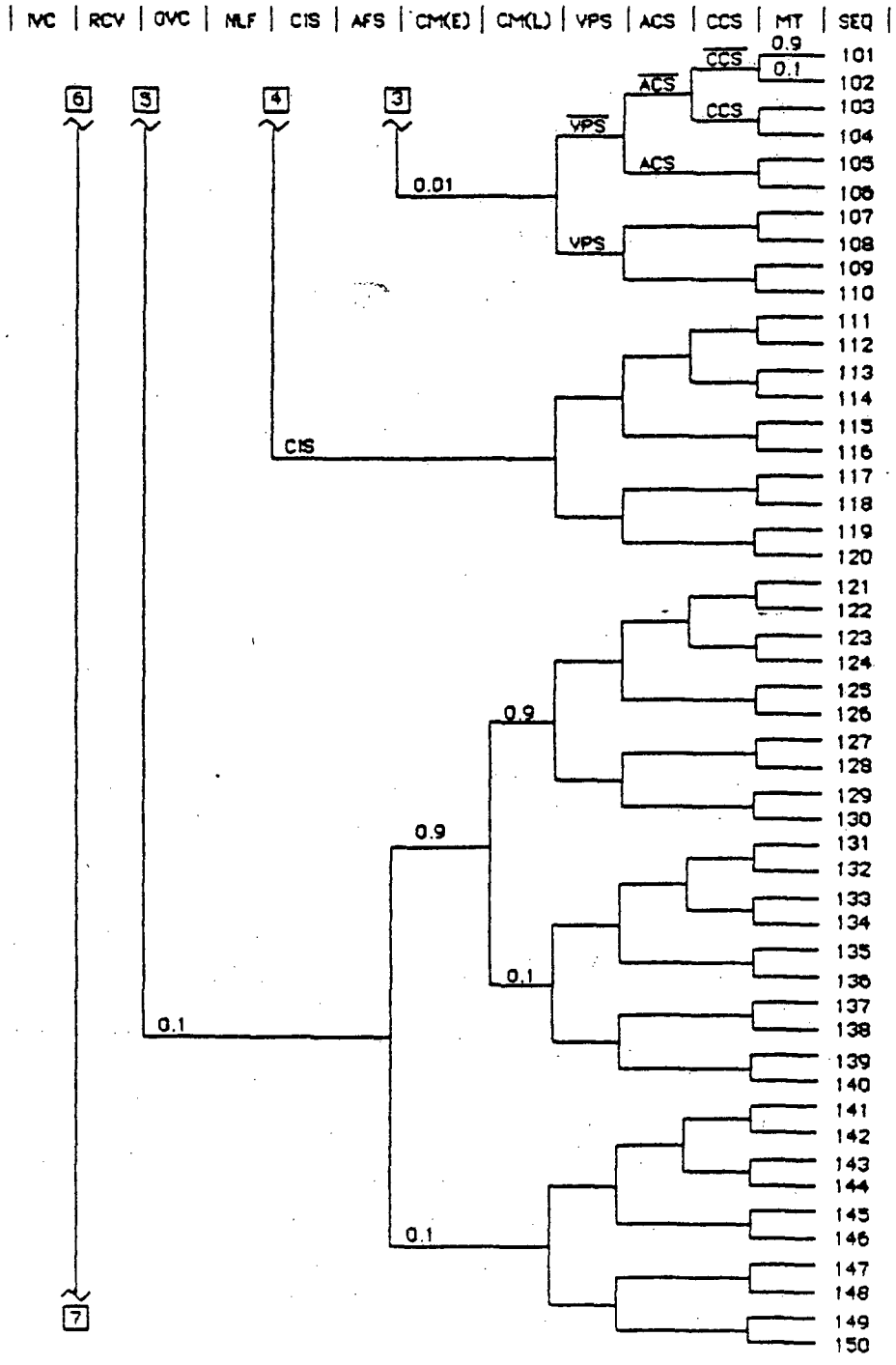

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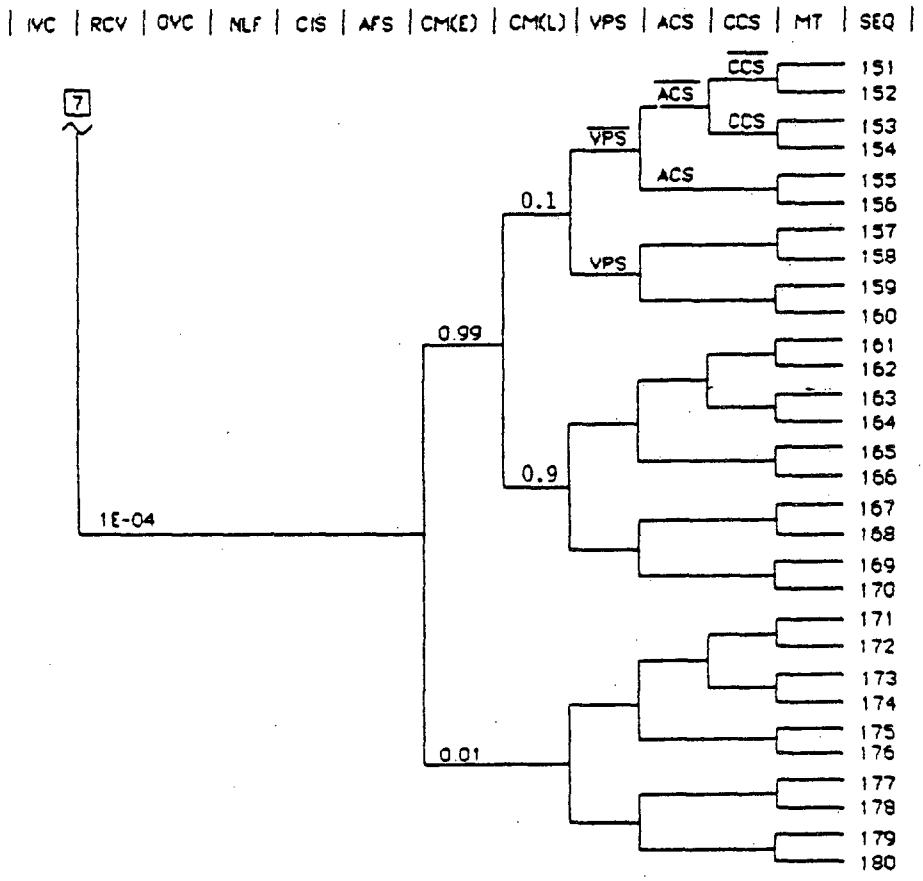

Figure 12.7-5 (Continued)

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Section 13

RESULTS

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Section 13

RESULTS

The analysis and results of each major task of this PRA have been presented in the preceeding sections of this report. Only a summary of the results of each of these tasks is presented here. More detailed discussions of the results may be found in the appropriate sections of the report: Section 10 for core-damage sequences and their frequencies; and Sections 11 and 12 for core and containment phenomenology, respectively. The integration of the core-damage sequences with the core and containment phenomenology is discussed in Section 13.3, while the engineering insights derived from this study are provided in Section 14.

13.1 DOMINANT SEQUENCES CONTRIBUTING TO CORE-DAMAGE FREQUENCY

The frequencies of each type of core-damage sequence and for each initiator group are listed in Table 13.1-1. The total core-damage frequency is estimated to be 3.6×10^{-5} per year. The table indicates that one sequence type, early loss of heat sink, dominates the frequency. Within this category the external events, specifically seismic, dominate. The other important sequence in terms of frequency is the loss of sodium, with seismic events contributing all of the frequency estimate for this accident sequence.

The earthquake-initiated sequences have been evaluated to have the largest frequency, 3.2×10^{-5}/yr; this comprises essentially all of the

Table 13.1-1

SUMMARY OF ANNUAL CORE-DAMAGE FREQUENCIES

Bin	Sequence	Internal Initiators	External Events	Common-Cause Initiators	Total Frequency (yr^{-1})
ULOF	TK[a]	ε[b]	1.3-6	ε	1.3-6
ULHS	TKP	2.5-7[c]	ε	ε	2.5-7
UTOP	TKPQ	ε	ε	ε	ε
ULOS	TKPI	ε	1.5-6	ε	1.5-6
LHSE	TSD	3.4-6	1.5-5	2.0-7	1.9-5
LHSL	TCD	ε	ε	ε	ε
LOS	TI	ε	1.4-5	ε	1.4-5
Total Frequency (yr^{-1})		3.7-6	3.2-5	2.0-7	3.6-5

Sequence Descriptions

ULOF - Unprotected loss of flow
ULHS - Unprotected loss of heat sink
UTOP - Unprotected transient overpower
ULOS - Unprotected loss of sodium
LHSE - Loss of heat sink - early
LHSL - Loss of heat sink - late
LOS - Loss of sodium

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[a]The sequences are defined here with the non-specific internal initiator (T), but the same description applies for other initiators.

[b]The symbol "ε" indicates that the core-damage sequence has been assessed to have a very low total frequency, less than 10^{-8}/yr.

[c]$2.5\text{-}7 = 2.5 \times 10^{-7}$

external event contribution to core-damage. The other external events--tornado and aircraft impact--were found not to be important. The seismic sequences contributing most to the core-damage frequency are the LHSE and LOS sequences and include failures of primary piping and guard vessels, failures of sodium-water reaction pressure-reduction system (SWRPRS) rupture disks, loss of offsite power coupled with other equipment failures, and structural failure of important buildings (see Table 13.1-2).

The sequences initiated by internal events contribute approximately one-tenth of the total core-damage frequency. The failure of short-term heat removal (LHSE, Sequence TSD) is the most likely sequence. Unprotected loss of heat sink accidents (ULHS, Sequence TKP) involving a failure to scram and a loss of heat sink, contribute about 7% to the total internal core-damage frequency. The dominant contributors to the core-damage frequency from internal initiating events are summarized in Table 13.1-3.

The common-cause initiators (fire, liquid-metal interactions, floods and turbine missiles) were found to be small contributors to the core-damage frequency. The only contribution specifically identified is due to a large control room fire which causes loss of available heat sinks in the early time frame (fire-induced LHSE).

13.2 PHENOMENOLOGICAL SEQUENCE ANALYSIS RESULTS

The physical process of core damage and melt were analyzed up to the point of containment failure (or maintenance of containment integrity).

Table 13.1-2

DOMINANT CONTRIBUTORS TO CORE-DAMAGE SEQUENCES INITIATED BY EXTERNAL EVENTS

Bin	Initiator	Description	Frequency
LHSE (Sequence ESD)*	Seismic event	Cooling is lost as a result of seismic-induced failures of SGS cooling, due to evaporator/superheater rupture disk failure, and of DHRS due to loss of air-blast heat exchangers, loss of power or service water, or missile impacts. In addition, PHTS leaks contribute to DHRS failure for some scenarios. Control building and steam generator building damage also contribute to loss of cooling scenarios.	1.5-5/yr
LOS (Sequence EI)*	Seismic event	A seismic-induced PHTS leak in loop A or B or C with a loss of integrity of the corresponding loop's guard vessel causes a loss of sodium and eventual core damage. A reactor vessel-guard vessel leak is included in this scenario.	1.4-5/yr

*See Section 8.1 for external event nomenclature; these sequences correspond to the TSD and TI sequences for internal initiators.

Table 13.1-3

DOMINANT CONTRIBUTORS TO CORE-DAMAGE SEQUENCES
INITIATED BY INTERNAL EVENTS

Bin/Initiator	Description	Frequency
LHSE (Sequence TSD)		
Any initiator except those that directly cause failures of a heat transport loop	The heat transport loops fail due to steam leaks through safety/relief and vent valves, followed by DHRS failure due to hardware faults or human error.	1.7-6/yr
Loss of offsite power	Diesel generators A and B fail due to either common causes, failure to start, or latent human errors. The turbine-driven AFW pump is unavailable due to either pump or recirculation line faults.	5.6-7/yr
Initiators that fail MFW system	Coupled with failure of automatic actuation of SGAHRS and its recovery by operator.	6.4-7/yr

The process was divided into two phases: core and containment phenomenology. The scenarios of core melt, vessel penetration, core entry into the reactor cavity, and the response of the containment were outlined using event trees. The results of the analyses are reported in terms of accident sequences (which include physical events) and their associated frequencies for each type of core-damage sequence. The end state of the core-damage sequence analysis--the core-damage bins--are the entry point for the core-response event tree, and the significant end states of the core-response tree are analyzed using the containment-response event tree.

The results of the core- and containment-response analyses are presented in the two sections that follow.

13.2.1 Core-Response Analysis

The core-response analysis results are summarized in Table 13.2-1. The core-damage sequence end states, or bins, (e.g., ULOF, LHSE, etc.) are the entry for the core-response event tree. The table presents the conditional probabilities of the seven core-response end states for each of the core-damage bins. The results for each end state are discussed briefly below.

For the unprotected loss of flow (ULOF) sequence, the dominant core-response end states are 2 and 3: reactor vessel bottom penetrations with either sodium (Na) or fuel entering the reactor cavity first. These two end states together have about a 0.86 conditional probability.

Table 13.2-1

CONDITIONAL PROBABILITIES OF CORE-RESPONSE END STATES

State	1	2	3	4	5	6	7
Core-Response End States / Core-Damage Bin	Core Damage, In-Vessel	Vessel Bottom w/Na Early	Vessel Bottom w/Na Late	Core Damage In-Vessel w/Head Leakage	Vessel Bottom w/Na Early & w/Head Leakage	Vessel Bottom w/Na Late & w/Head Leakage	Vessel Failure From Missile Generation
ULOF	1.4-1[a]	4.3-1	4.3-1	1.8-6	8.9-4	8.9-4	<2.0-7
ULHS	4.9-1	2.5-1	2.5-1	7.9-7	4.0-4	4.0-4	<7.9-8
UTOP	8.1-1	9.4-2	9.4-2	3.0-8	1.5-5	1.5-5	<3.0-9
ULOS	9.0-4	9.9-1	1.0-3	1.6-6	1.6-3	1.6-6	<1.6-7
LHSE	1.9-5	8.9-1	1.0-1	9.0-8	4.5-5	4.5-5	<9.0-9
LHSL	1.0-5	9.0-1	1.0-1	9.0-9	4.5-6	4.5-6	<9.0-10
LOS	9.9-7	~1	1.0-3	9.0-9	9.0-6	9.0-9	<9.0-10

[a]1.4-1 = 1.4×10^{-1}

Core-response end state 1, core damage retained in the reactor vessel, is predicted to occur about 14% of the time. The conditional probabilities associated with head leakage due to energetics is small and more severe failures involving missile generation are considered non-mechanistic.

The unprotected loss of heat sink (ULHS) accidents have core responses similar to ULOF as discussed above, except that the primary pump flow leads to greater core sweepout and therefore a greater chance (49%) for in-vessel coolability.

The unprotected transient overpower (UTOP) sequences were calculated to have an 81% probability of core material being retained in the reactor vessel, or at least within the primary system. The early fuel sweepout dominates this sequence phenomenology making severe energetics very unlikely. The sequences that do end up in a breach of the reactor vessel do so by vessel bottom penetration (states 2 and 3). Damaging energetics are extremely unlikely for this accident. It should be noted that the calculations for this sequence are dependent on the conclusion that there are no large ramp rates (greater than 20¢/sec) for the transient overpower event.

The unprotected loss of sodium (ULOS) event was estimated to nearly always end in one core-response state, namely state 2--vessel bottom penetration with sodium entering the reactor cavity before fuel. The possibility of retention of core material in the vessel for this case is minimal because there is no inventory of sodium for core heat removal. Once again, damage to the head from energetics is very unlikely and severe head failure is non-mechanistic.

The protected loss of heat sink (PLHS) scenario, both early and late (LHSE and LHSL), behave similarly. The probability of retention of the core in-vessel is small and the probabilities of head leakage or severe damage are small or negligible. The dominant core-response states are 2 and 3, vessel bottom penetration with sodium entering the reactor cavity before or after core debris.

The final sequence, protected loss of sodium (LOS), has a conditional probability of near unity of resulting in a state 2 core-response end state. The likely failure mode is vessel bottom penetration with sodium entering the reactor cavity before core debris.

13.2.2 Containment-Response Analysis

The containment response to HCDAs was analyzed using a containment-response event tree to delineate the potential sequences which could occur following the core debris entry into containment. The tree includes both physical events (e.g., out-of-vessel debris bed coolability) and the containment/confinement safeguard system failures that could affect the progression of the sequence. Due to the inclusion of the system failures, the results of the quantification of this event tree are dependent on the actual core-damage sequence. For example, the annulus cooling system designed to remove heat from the containment and confinement building, would be inoperable for a station blackout (total loss of all ac power) core-damage sequence.

The results of the containment-response analysis are summarized in Table 13.2-2 for the dominant internal initiator sequences only. This is

Table 13.2-2

SUMMARY OF MOST LIKELY CONTAINMENT-RESPONSE SEQUENCES FOR DOMINANT INTERNAL INITIATED HCDAS

HCDA	Sequence[a]	Conditional Probability
ULHS	1	0.08
	2	0.01
	11	0.01
	51	0.70
	52	0.08
	61	0.08
	62	0.01
	71	0.01
	81	0.01
	121	0.01
PLHS-early (LHSE) with Sodium in Cavity Early (Electric power initially failed)	21	0.01
	81	0.61
	82	0.07
	89	0.11
	90	0.01
	91	0.07
	92	0.01
	99	0.01
	101	0.07
	102	0.01
	109	0.01
PLHS-early (LHSE) with Sodium in Cavity Early (Electric power available)	1	0.01
	51	0.70
	52	0.08
	61	0.08
	62	0.01
	71	0.09
	72	0.01
	81	0.01
	121	0.01

Table 13.2-2 (Continued)

HCDA	Sequence[a]	Conditional Probability
PLHS-early (LHSE) with Fuel in Cavity Early (Electric power initially failed)	21	0.07
	22	0.01
	29	0.01
	31	0.01
	81	0.56
	82	0.06
	89	0.10
	90	0.01
	91	0.06
	92	0.01
	99	0.01
	101	0.07
	102	0.01
	109	0.01
PLHS-early (LHSE) with Fuel in Cavity Early (Electric power available)	1	0.08
	2	0.01
	11	0.01
	51	0.65
	52	0.07
	61	0.07
	62	0.01
	71	0.08
	72	0.01
	81	0.01

[a]Corresponds to sequence number on containment-response event trees; see Section 12.7.

because the dominant external (seismic) and common-cause (fire) induced sequences affect success/failure probabilities for both systems in the core-damage and containment-response event trees at the same time. It is therefore easier to discuss the integrated results of these sequences (i.e., after combining the probabilities of core damage, core-response end states, and containment-response end states) rather than to present the results of just the containment analysis portion of the PRA for the external-event and common-cause sequences.

Even in the case of internal initiators, one significant common mode potential was found which could affect the successful operation of both the core cooling systems and the containment safeguard systems at the same time. This common mode potential is failure of electric power. Such a failure affects the probabilities of success or failure of the containment safeguard systems as well as the core cooling systems and hence two sub-categories of results (with and without power) are presented for the LHSE event, the only HCDA where this failure significantly influenced the results of the overall analysis of internal initiators. In this case, recovery of electric power was considered between the time of core damage and the time when the containment systems are needed to mitigate the event.*

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*The annulus filtration system is needed early in the event while the vent/purge system, the annulus cooling system, and the containment cleanup system are not needed until ~10 hours or more after reactor vessel failure.

For the ULHS event, the most likely containment-response scenario following reactor vessel bottom failure involves successful operation of the reactor cavity vent system and all containment safeguard systems (sequence 51). Early cavity liner failure is expected but with coolability of the debris bed in the reactor cavity until sodium boildry. Basemat meltthrough is not expected and containment integrity is expected to be maintained. Other less likely sequences, each with probabilities of 0.08, involve basemat failure, loss of containment integrity after evacuation of the emergency planning zone (EPZ), or success of all containment features including no cavity liner failure.

The LHSE event results in different containment responses depending primarily on the availability of electric power, although whether sodium or fuel enters the reactor cavity first does introduce additional minor effects. For the case of a loss of electric power, the most likely containment-response scenario is the same as that for ULHS, except AFS fails due to the loss of power (sequence 81). Power is restored after core damage and vessel bottom failure so that the other containment safeguard systems can mitigate the consequences of the accident and prevent a loss of containment integrity. For the case in which electric power is available, the most likely containment response is the same as for the ULHS event.

13.3 INTEGRATION OF CORE-DAMAGE SEQUENCES AND PHENOMENOLOGY

This section integrates the results of the core-damage sequences with the core-and containment-response event trees. The basic information used in

this section has been described in detail in Sections 10, 11 and 12; Sections 8 and 9 contain details on external events analysis and common-cause failures.

This section has been divided into four subsections. Sections 13.3.1, 13.3.2, and 13.3.3 discuss integration of core-damage sequences due to internal initiators, external events, and common-cause initiators, respectively. Section 13.3.4 gives the summary of results and frequencies of release in the radiological release matrix. The radiological release matrix is discussed in Section 12 in detail. Since the integration process follows the same principal steps, detailed description of the procedure is described only once in Section 13.3.1. The rest of the bins/sequences are not discussed in detail; only features unique to each type of sequence are identified in their respective sections.

The overall integration effort is represented in Figure 13.3-1. The sequences from the core-damage event trees (Sections 3, 8, and 9) are applied as inputs to the core-response event tree (Section 11). Dominant sequences resulting from the core-response event tree analysis are then propagated through the containment-response event tree (Section 12) conditional on the success and failure of the containment safeguard systems (e.g., annulus filtration and annulus cooling systems, containment isolation system). Frequencies of the most likely containment-response end states are then placed in the appropriate segments of the radiological release matrix, as shown in various tables throughout this section.

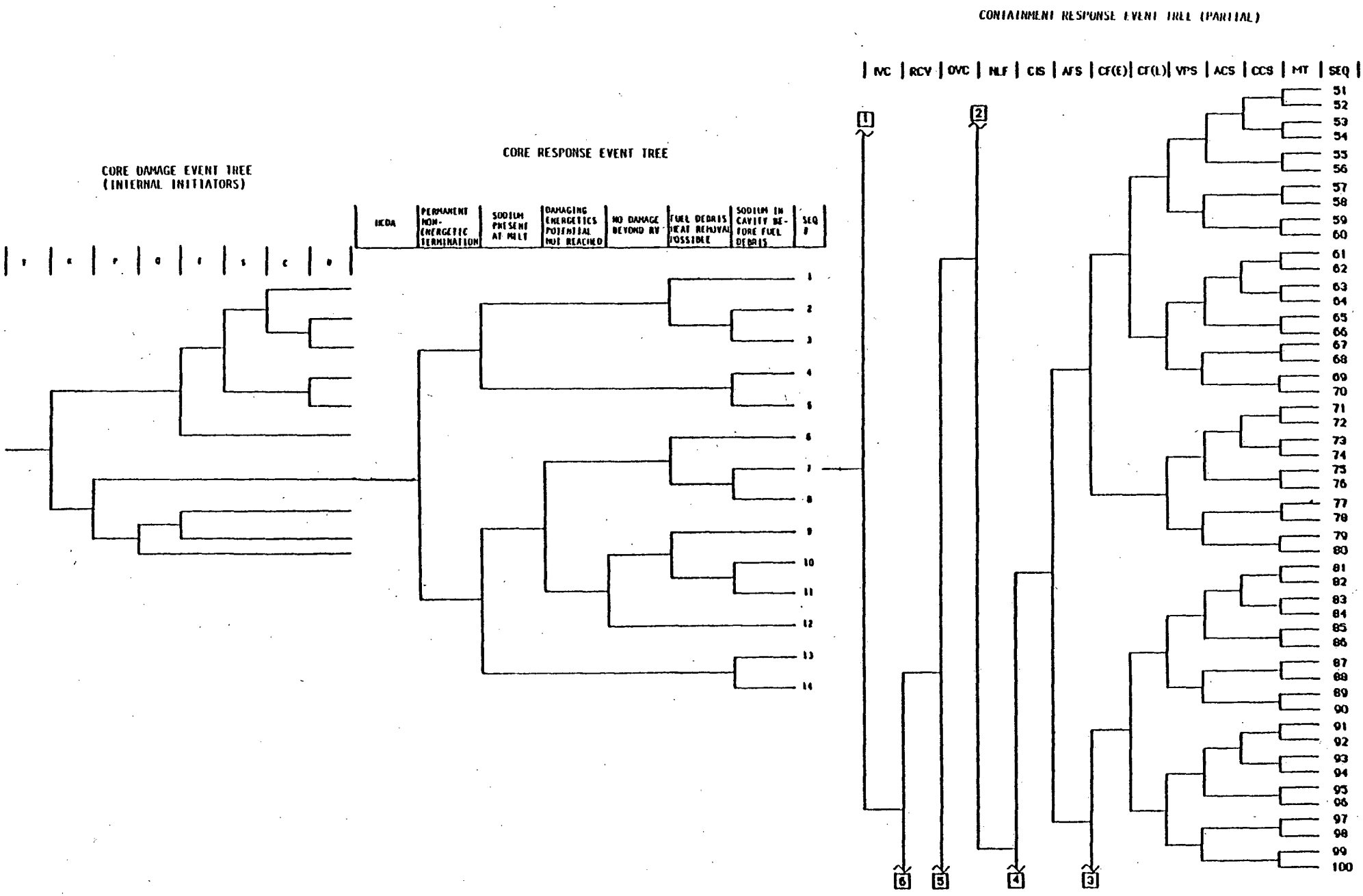


Figure 13.3-1. Principle of Integration of Core-Damage Sequences and Phenomenology.

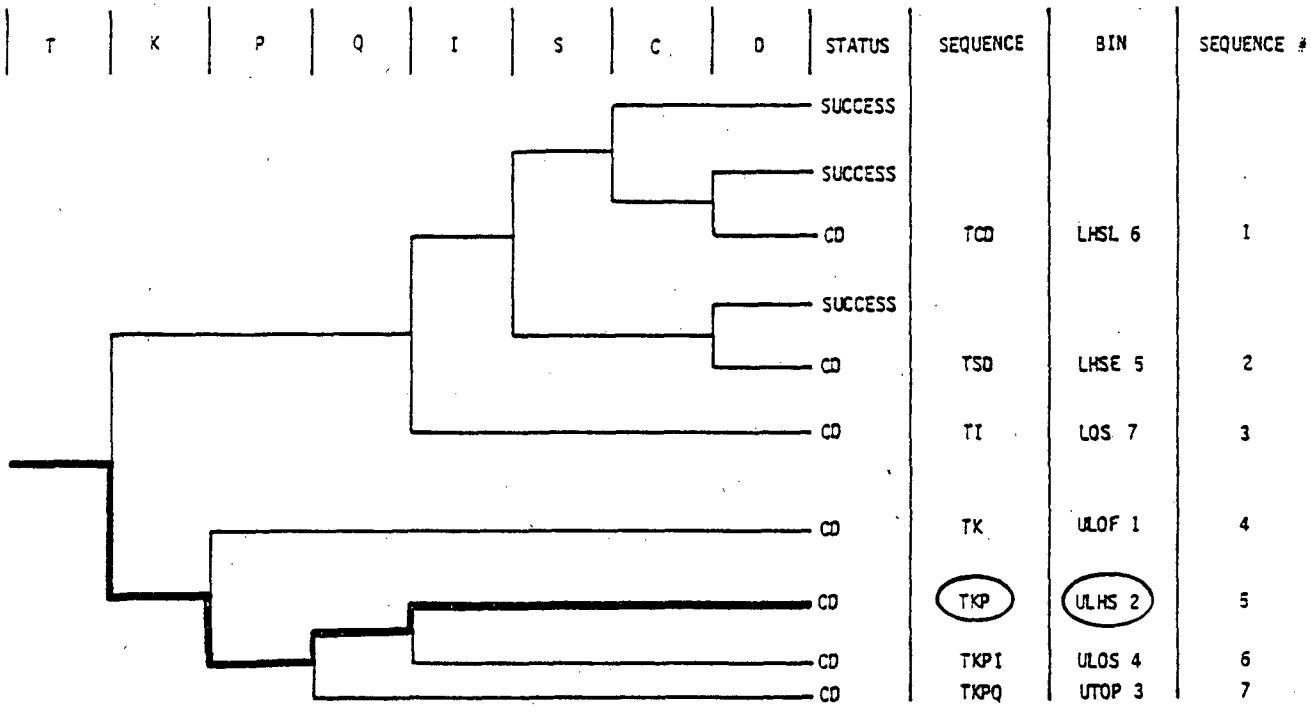
13.3.1 Integration of Internally Initiated Sequences

Section 10.1 contains the results of the core-damage analysis for internal initiators. This section extends those results and integrates them with core-and containment-response phenomena to determine their radiological release categories. The discussion that follows uses the same chronological steps as Section 10.1. As an example, only one sequence (Sequence ULHS-TKP) is discussed in detail; the remaining sequences are described only briefly.

ULOF (Sequence TK). The unprotected loss of flow (ULOF) accidents (Sequence TK) caused by internal initiators were assessed to have a very low frequency of occurrence ($<10^{-8}$/yr) as shown in Table 13.1-1. This sequence is therefore not discussed any further here.

ULHS (Sequence TKP). Figures 13.3-2, 13.3-3, and 13.3-4 show the core-damage, core-response, and containment-response event trees respectively for the unprotected loss of heat sink (ULHS) transient (TKP or sequence #5 of Figure 13.3-2) which belongs to Bin 2. Core damage caused by ULHS is likely to follow the path shown by broadened lines on these event trees (not accounting for the relatively unimportant "CD, In Vessel" sequences on the core-response event tree).

The discussion begins given a hypothetical core disruptive accident (HCDA) has already occurred. That is to say, all the features required to prevent an ULHS accident have failed and core disruption has started to occur. This is represented as the right-most point of sequence #5 of Figure 13.3-2, or the left-most point of Figure 13.3-3.



EVENT NAMES

T - Initiator

K - Reactor Shutdown

P - Main PHTS Pump Motors Tripped

Q - Initial Power Normal

I - Sodium Inventory Maintained

S - Short-term Cooling via SGS Maintained

C - Long-term Cooling via SGS Maintained

D - Core Cooling via DHRS Initiated and Maintained

Figure 13.3-2. Core-Damage Event Tree Showing ULHS (TKP) HCDA.

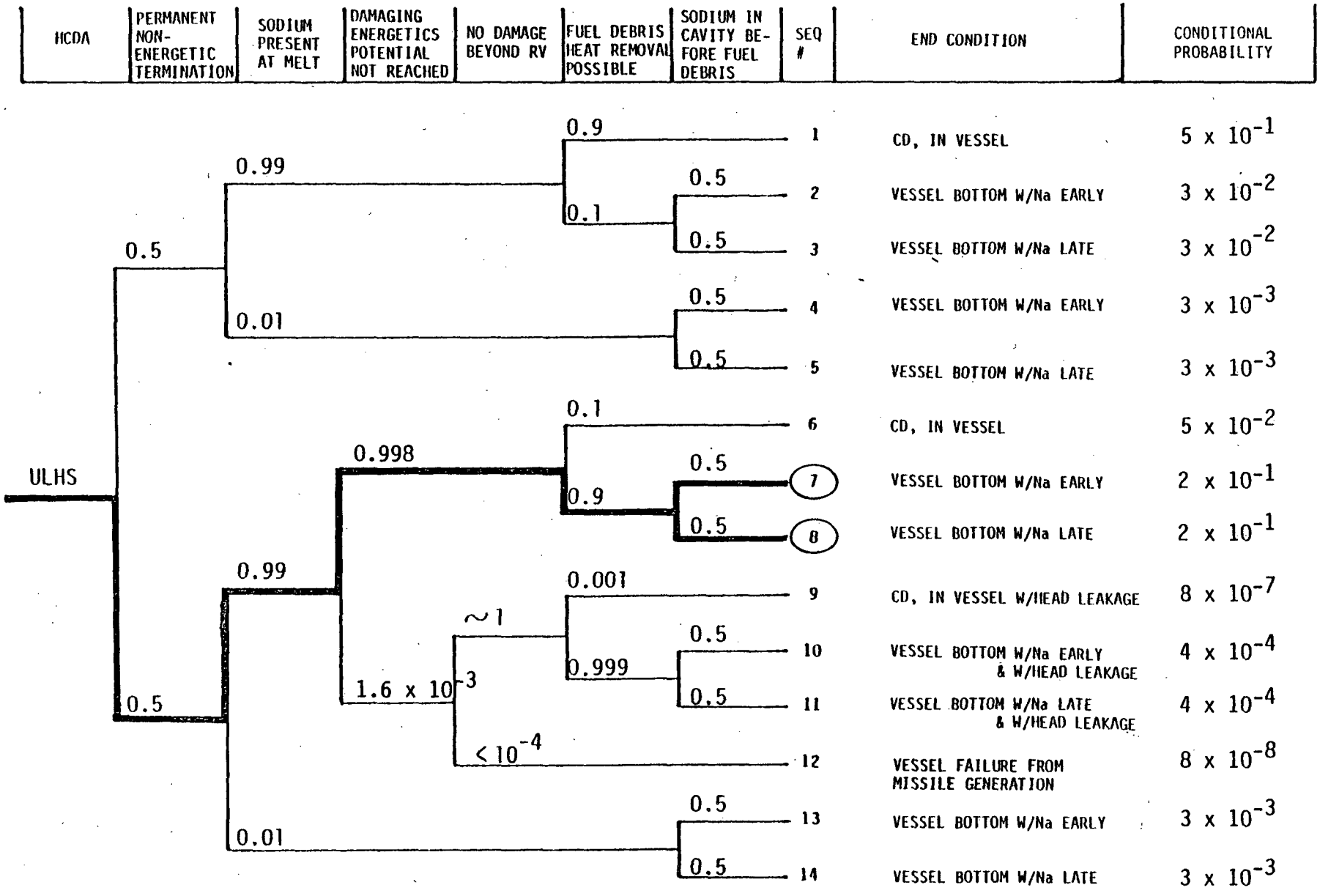


Figure 13.3-3. Core-Response Event Tree for ULHS (TKP).

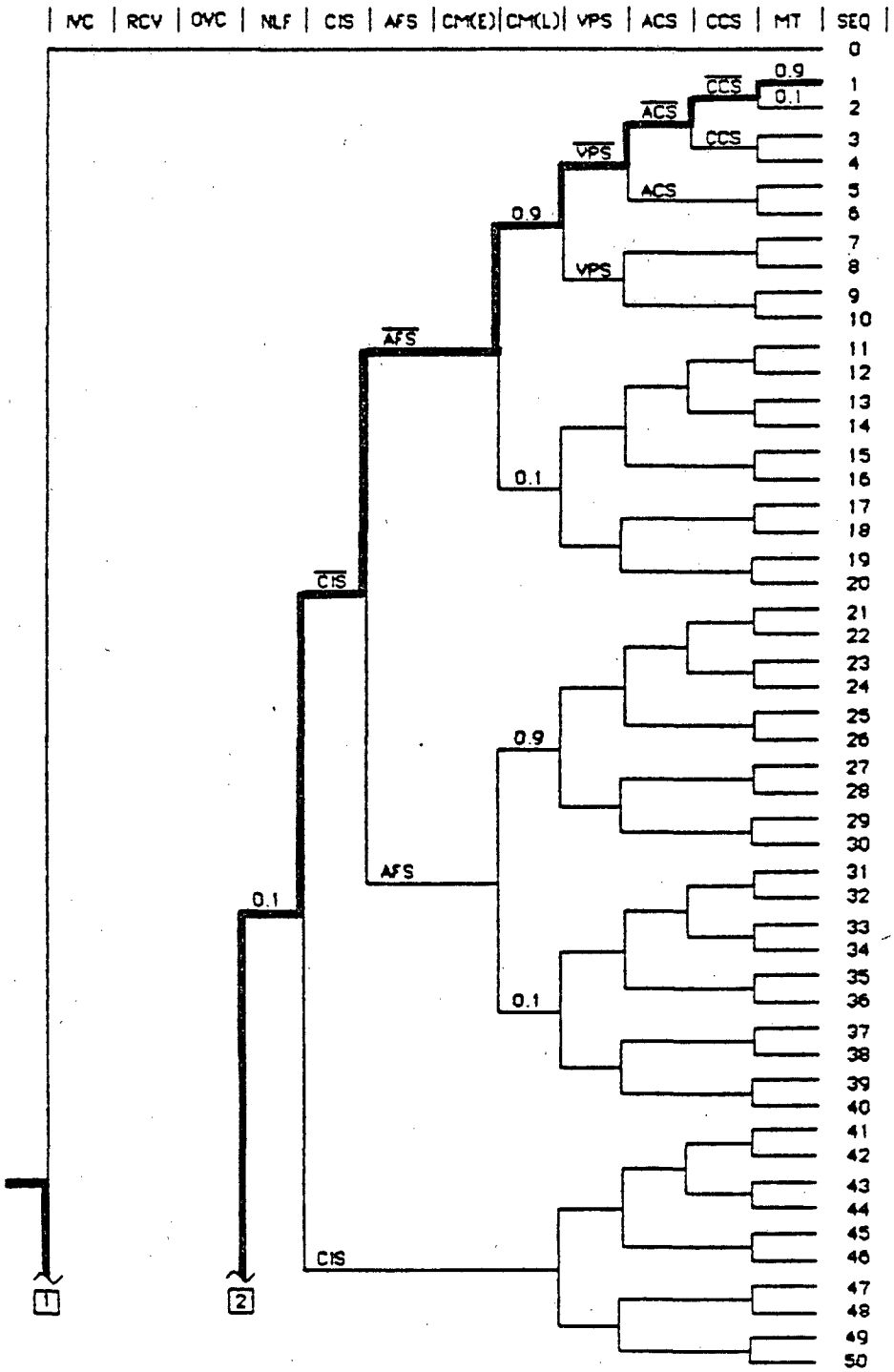


Figure 13.3-4. Containment-Response Event Tree for ULHS (TKP).

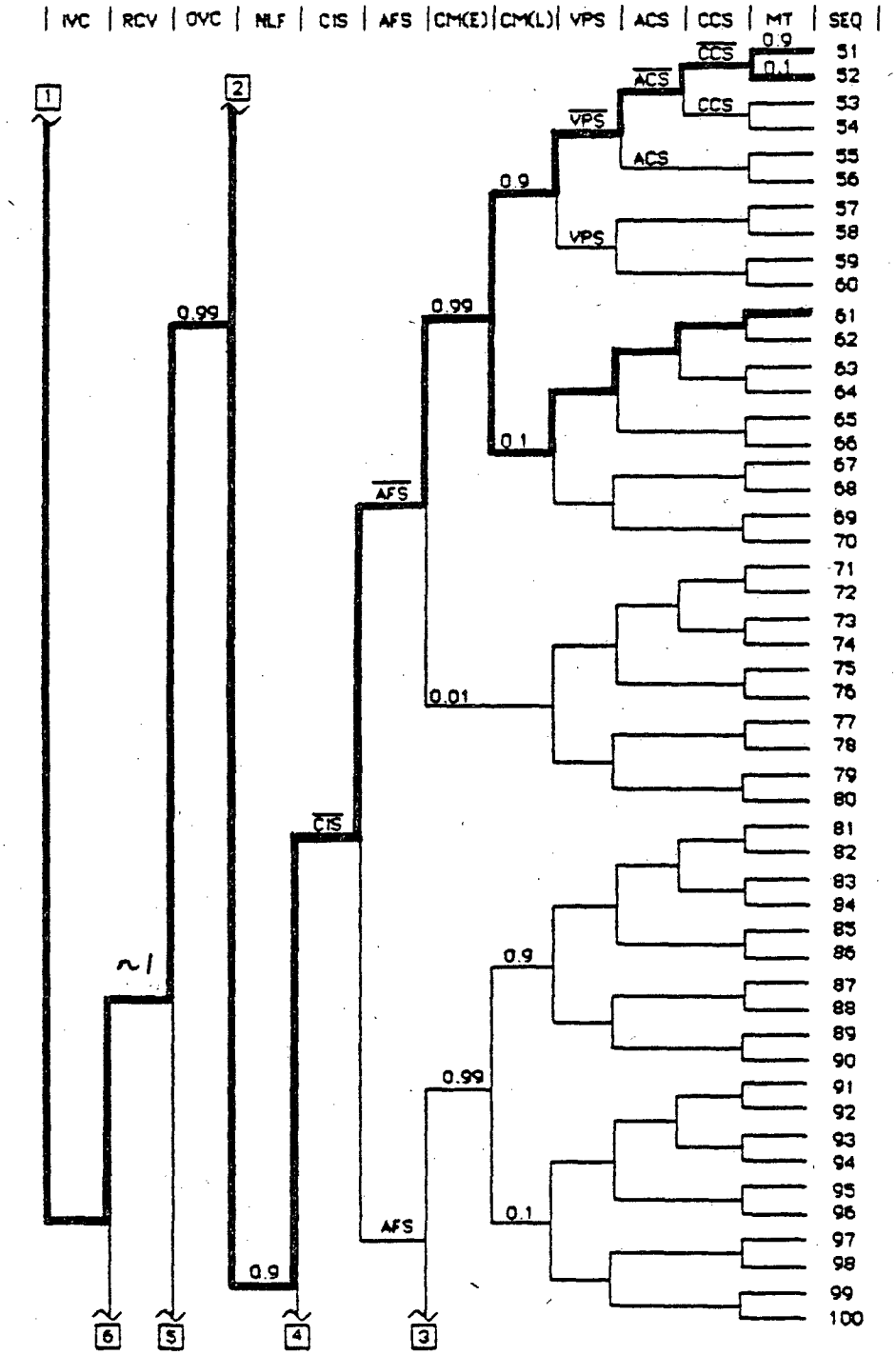


Figure 13.3-4 (Continued)

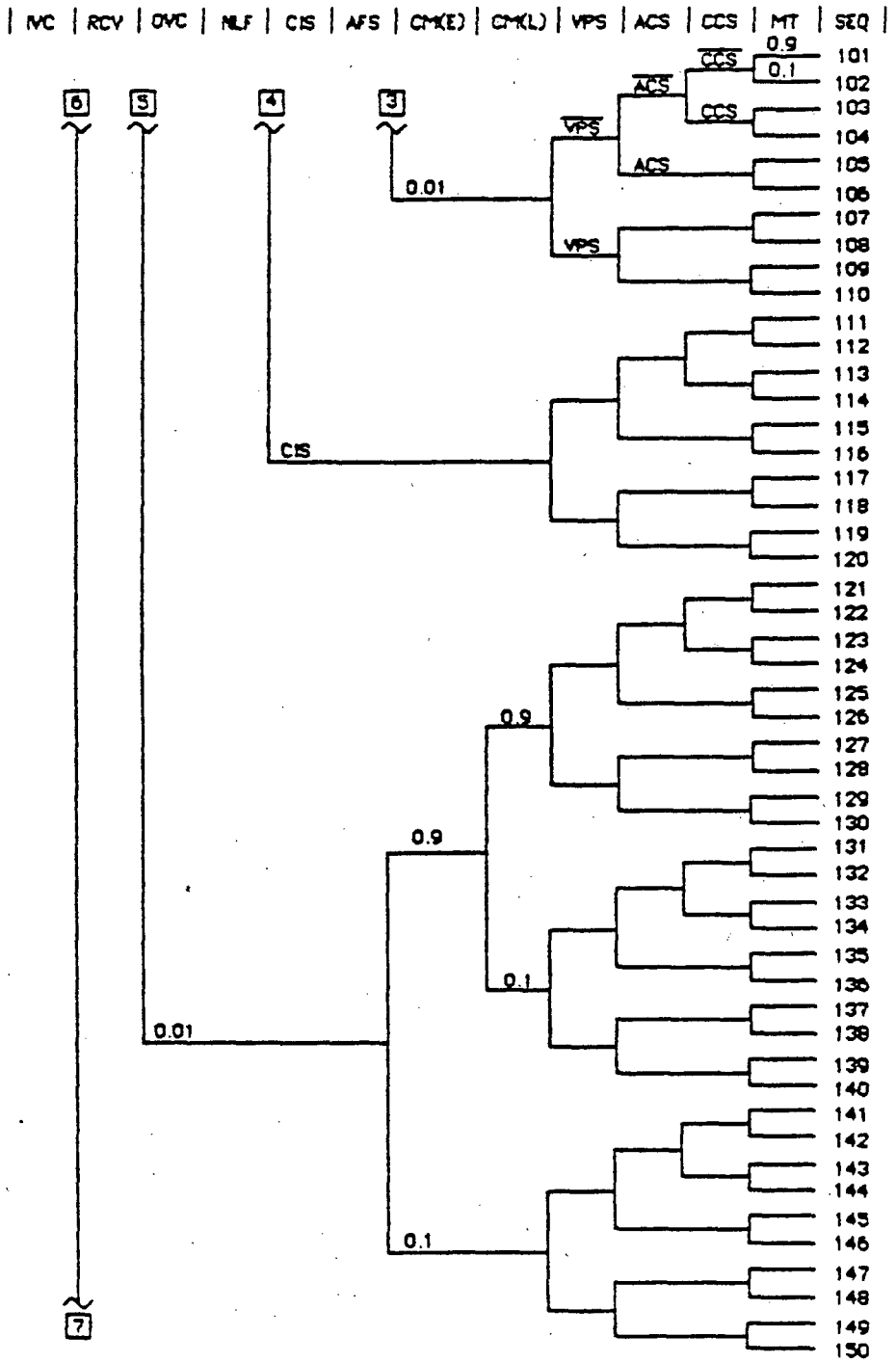


Figure 13.3-4 (Continued)

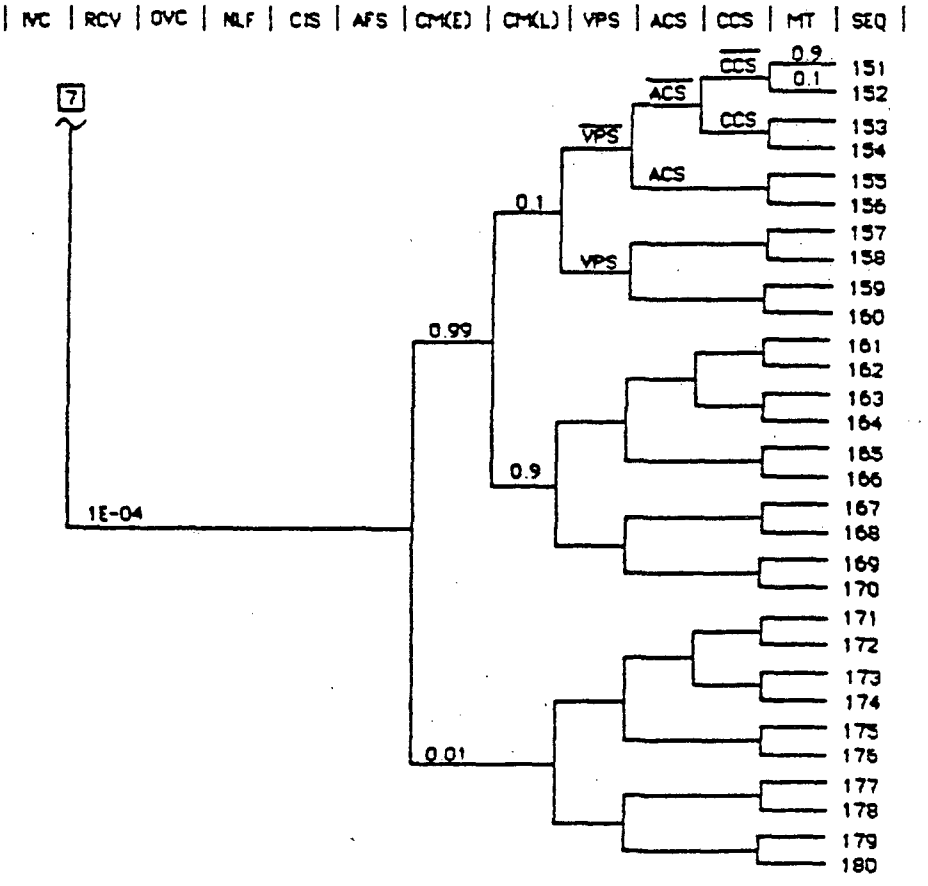


Figure 13.3-4 (Continued)

Once core disruption starts, the events are likely to follow the paths represented by Sequences 1, 6, 7, or 8 in Figure 13.3-3. Sequences 1 and 6, which have approximately a 55% combined probability of occurrence, both lead to a damaged core which remains in the reactor vessel. Therefore, these two sequences do not contribute significantly to the radiological release to the containment or surroundings (Sequence #0 on Figure 13.3-4). Core-damage progression represented by Sequences 1 and 6 (Figure 13.3-3) is therefore not propagated any further.

The next two series of events, represented by Sequences 7 and 8 (in Figure 13.3-3) each have approximately a 20% likelihood of occurrence. Both of these result in failure of the reactor vessel bottom head with the reactor core debris falling into the reactor cavity. The effect of sodium dropping into the reactor cavity earlier or later than the core debris is not of significance in the ULHS case, so they both will be treated as similar entry points for the containment-response event tree (Figure 13.3-4).

The conditional probabilities of the seven core-response end states (that is the end condition of the core-response phase of the accident as shown by the 'End Condition' column of the core-response event tree — Figure 13.3-3) are shown in Table 13.3-1. These probabilities are the sum of the probabilities of each of the core-response sequences which lead to a particular core-response end state. Table 13.3-1 shows these results for all the core-damage bins. Table 13.3-2 contains the

Table 13.3-1

CONDITIONAL PROBABILITIES OF CORE-RESPONSE END STATES

State	1	2	3	4	5	6	7
Core-Response End States / Core-Damage Bin	Core Damage, In-Vessel	Vessel Bottom w/Na Early	Vessel Bottom w/Na Late	Core Damage In-Vessel w/Head Leakage	Vessel Bottom w/Na Early & w/Head Leakage	Vessel Bottom w/Na Late & w/Head Leakage	Vessel Failure From Missile Generation
ULOF	1.4-1[a]	4.3-1	4.3-1	1.8-6	8.9-4	8.9-4	<2.0-7
ULHS	4.9-1	2.5-1	2.5-1	7.9-7	4.0-4	4.0-4	<7.9-8
UTOP	8.1-1	9.4-2	9.4-2	3.0-8	1.5-5	1.5-5	<3.0-9
ULOS	9.0-4	9.9-1	1.0-3	1.6-6	1.6-3	1.6-6	<1.6-7
LHSE	1.9-5	8.9-1	1.0-1	9.0-8	4.5-5	4.5-5	<9.0-9
LHSL	1.0-5	9.0-1	1.0-1	9.0-9	4.5-6	4.5-6	<9.0-10
LOS	9.9-7	~1	1.0-3	9.0-9	9.0-6	9.0-9	<9.0-10

[a]1.4-1 = 1.4×10^{-1}

Table 13.3-2

FREQUENCIES OF CORE-RESPONSE END STATES (INTERNAL INITIATORS)

STATE	1	2	3	4	5	6	7
Core-Response End States / Core-Damage Bin	Core damage In Vessel	Vessel Bottom w/Na Early	Vessel Bottom w/Na Late	Core damage In-Vessel w/Head Leakage	Vessel Bottom w/Na Early & w/Head Leakage	Vessel Bottom w/Na Late & w/Head Leakage	Vessel Failure From Missile Generation
ULOF (TK)	ε[a]	ε	ε	ε	ε	ε	ε
ULHS (TKP)	1.2-7[b]/yr	6.0-8/yr	6.0-8/yr	ε	ε	ε	ε
UTOP (TKPQ)	ε	ε	ε	ε	ε	ε	ε
ULOS (TKPI)	ε	ε	ε	ε	ε	ε	ε
LHSE(TSD) i [c]	ε	1.1-6/yr	1.2-7/yr	ε	ε	ε	ε
ii	ε	2.0-6/yr	2.2-7/yr	ε			ε
LHSL (TCD)	ε	ε	ε	ε	ε	ε	ε
LOS (TI)	ε	ε	ε	ε	ε	ε	ε

NOTES: [a] 'ε' implies that these end states have frequencies $<10^{-8}$/yr.

[b] $1.2\text{-}7 \equiv 1.2 \times 10^{-7}$.

[c] i - Electric power not initially available.
ii - Electric power available.

frequencies of each of these bins for internal initiators (including ULHS-TKP), which were calculated by taking the product of the bin frequencies (of Table 13.1-1) and the conditional probabilities of Table 13.3-1.

Due to the nature of the structure of the containment-response event tree and its insensitivity to the core response-end states, only one generic containment-response event tree was required (see Section 12). The probabilities of occurrence of the different events are adjusted according to the HCDA being investigated and the core-response phenomena expected. Section 12 contains more details on this topic.

The likely containment-response paths to be followed by the ULHS sequences are highlighted in Figure 13.3-4 as delineated in Table 13.2-2. These dominant Sequences are 1, 51, 52, and 61. Other sequences (2, 11, 62, 71, 81, 121) each have a conditional probability of about 0.01 for the containment response occurring as outlined by these sequences. Each containment-response sequence ends in a particular end state as defined by the radiological release matrix discussed in Section 12 and reproduced here as Table 13.3-3. The matrix distinguishes among high, moderate, and low releases to the environment (where moderate releases correspond approximately to the CRBRP-3 base case analysis) as well as the time of those releases where 9 hours corresponds to the approximate time to evacuate the emergency planning zone (EPZ) after a "core-on-the-floor"

Table 13.3-3

RADIOLOGICAL RELEASE MATRIX

Time / Magnitude	Early 0 < t < 9 hrs	Late t > 9 hrs	Long-Term > 6 months
LOW	1	2	3 All even numbered sequences (i.e., all MT sequences)
MODERATE	4 41,71*,101*,111,141,171*	5 1,11*,21,51,61*,81,121,131*,151,161*	6 7,57,87,127,157
HIGH	7 43,47,49,73,75,77,79,103,105,107,109,113,117,119,143,147,149,173,177,179	8 3,5,9,13,15,17,19,23,25,27,29,31,33,35,37,39,45,53,55,59,63,65,67,69,83,85,89,91,93,95,97,99,115,123,125,129,133,135,137,139,145,153,155,159,163,165,167,169,175	9

Note 1: Moderate release roughly corresponds to doses in Reference 1 base case analysis. Doses higher than base case are designated "high" and doses lower as "low."

Note 2: Only odd numbered sequences (n) are listed. Even numbered sequences (n+1) are identical to odd numbered sequences (i.e. - appear in same category) except that foundation mat meltthrough adds an additional long term release contributor.

Note 3: Early releases associated with bypass leakage are not included here.

*Assumes the confinement does not fail along with the containment.

condition is imminent and evacuation begins. The sequence numbers indicated correspond to the sequence numbers on the containment-response event trees which are structurally the same for all HCDAs.

By multiplying the ULHS frequency values shown in Table 13.3-2 (which include the core damage frequency and core-response probabilities taken together) by the containment-response probabilities shown in Table 13.2-2, and applying the radiological release matrix, the overall frequency of a ULHS-TKP event resulting in a certain release end state can be obtained. For example, a ULHS-TKP event with vessel bottom failure and sodium entering the reactor cavity first (6.0×10^{-8}/yr) resulting in containment-response sequence 51 (0.70) yields a probability per year (frequency) of 4.2×10^{-8}/yr for resulting in a moderate release to the environment which would occur after the 9-hour EPZ evacuation time period. Summing up the frequencies of all other dominant containment-response sequences for a given radiological release end state results in the total estimated frequency of that end state for the given HCDA of interest.

The radiological release end state results for the ULHS-TKP event are summarized in Table 13.3-4. The results indicate that releases from the plant due to a ULHS event are estimated to occur at about a frequency of 1.2×10^{-7}/yr or less. A moderate-late release is more likely to occur than a moderate-early or low-long-term release which are 1/100 and 1/10 the above frequency, respectively. All other possible releases are estimated to occur at frequencies less likely than the three mentioned here.

Table 13.3-4

FREQUENCIES OF RADIOLOGICAL RELEASE END STATES FOR ULHS (INTERNALLY INITIATED)

Time / Magnitude	EARLY $0 < t < 9$ hrs	LATE $t > 9$ hrs	LONG-TERM $t \geq 6$ months
LOW	ε	ε	1.2-8/yr[b]
MODERATE	1.2-9/yr	1.2-7/yr	ε
HIGH	ε	ε	ε

[a] 'ε' implies frequency $<10^{-9}$/yr.
[b] 1.2-8 = 1.2×10^{-8}.

UTOP (Sequence TKPQ). The unprotected transient overpower (UTOP) HCDA does not contribute significantly to the overall core-damage frequency as shown in Table 13.1-1. This HCDA is therefore not analyzed further regarding its contribution to the radiological release matrix.

ULOS (Sequence TKPI). The unprotected loss of sodium (ULOS) HCDA also does not contribute significantly and is not analyzed here.

LHSE (Sequence TSD). The protected loss of heat sink early (LHSE) event is the only HCDA that is influenced significantly by the availability or non-availability of electric power, as was discussed earlier. The two cases are therefore analyzed separately. The cases being:

Case (i) - Electric power initially fails contributing to the occurrence of core damage. Power may or may not be recovered within 10 hours after core penetration to mitigate the consequences of the accident, and

Case (ii) - Electric power is available.

The conditional probabilities for core-response end states for LHSE HCDAs are given in Table 13.3-1, which shows that a LHSE event is very unlikely to end up with core debris remaining in the vessel confinement. Table 13.3-2 shows the frequencies of these core-response end states for both LHSE cases (i) and (ii). Reviewing Table 13.2-2, it is noted that there is a marked difference in the containment response depending on the state of electric power availability at the onset of the accident and whether sodium or fuel enters the reactor cavity first. By multiplying the frequencies of the LHSE events found in Table 13.3-2 with the appropriate probabilities in Table 13.2-2, the contributions to each end state in the radiological release matrix are obtained.

The results of this evaluation are shown in Table 13.3-5. Depicted are the conditions for whether or not electric power is initially available at the start of the event and whether sodium or fuel enters the reactor cavity first. Moderate-late releases are most likely among the radiological release end states for all categories of LHSE events. It should be noted that some high-early releases are expected at frequencies of $\sim 10^{-8}$/yr and high-late releases at $\sim 10^{-7}$/yr for conditions where a LHSE event involves a loss of electric power. In these cases, the major contributors to the sequence frequencies involve the failure of all containment safeguard systems since electric power is not restored in time to mitigate the accident.

LHSL (Sequence TCD). The protected loss of heat sink late (LHSL) accidents were calculated to be insignificant contributors to core-damage frequency, as shown in Table 13.1-1. These HCDAs are therefore not analyzed further.

LOS (Sequence TI). The protected loss of sodium (LOS) event also provides insignificant contributions to the overall core damage frequency (see Table 13.1-1). It is therefore not analyzed further.

13.3.2 Integration of External Event Sequences

As discussed in Section 13.3.1, seismically induced core-damage accidents comprise essentially all of the external event contribution to core-damage. Hence, the integration of external event sequences with the core-response and containment-response event trees centered about

Table 13.3-5

FREQUENCIES OF RADIOLOGICAL RELEASE END STATES FOR LHSE (INTERNALLY INITIATED)

Time / Magnitude	EARLY 0 < t ≤ 9 hrs	LATE t > 9 hrs	LONG-TERM t ≥ 6 months
LOW	ε[a]	ε	[b]LHSE(Na early)i = 1.1-7/yr[c] LHSE(Na early)ii = 2.0-7/yr LHSE(Na late)i = 1.2-8/yr LHSE(Na late)ii = 2.2-8/yr Total 3.4-7/yr
MODERATE	LHSE(Na early)i = 8.8-8/yr LHSE(Na early)ii = 2.0-7/yr LHSE(Na late)i = 9.6-9/yr LHSE(Na late)ii = 2.0-8/yr Total 3.2-7/yr	LHSE(Na early)i = 7.6-7/yr LHSE(Na early)ii = 1.8-6/yr LHSE(Na late)i = 8.4-8/yr LHSE(Na late)ii = 2.0-7/yr Total 2.8-6/yr	ε
HIGH	LHSE(Na early)i = 1.1-8/yr LHSE(Na late)i = 1.2-9/yr Total 1.2-8/yr	LHSE(Na early)i = 2.3-7/yr LHSE(Na late)i = 2.5-8/yr Total 2.6-7/yr	ε

[a] 'ε' implies frequency $<10^{-9}$/yr.

[b] Na early or late refers to whether sodium (Na) enters reactor cavity before or after the fuel; i - Electric power not available initially; ii - Electric power available.

[c] 1.1-7 = 1.1×10^{-7}.

the integration of the seismic sequences. The approach used to integrate these sequences differs from that taken to integrate internal initiator sequences since: (1) the common structural shaking caused by earthquakes may result in common-cause failures of containment systems and (2) the failure probabilities for the containment system components are, in general, functions of the earthquake acceleration.

The integration of seismically-induced core-damage accident sequences with the core-response and containment-response event trees was accomplished by completion of the following steps:

1. Determination of the susceptibility of containment safeguard systems to earthquakes (fragility analysis),
2. Determination of containment-response end state prime implicants (qualitative solution),
3. Calculation of each end state's frequency using the methods of Section 8.1.4.2, and
4. Calculation of each release bin's frequency according to the radiological release matrix.

A review of the core-response and containment-response event trees showed that only six events were susceptible to seismic events:

1. <u>No liner failure (NLF)</u>

   The reactor cavity liner may be damaged due to collapse of the floors and walls inside the containment building.

2. <u>Annulus filtration system (AFS)</u>

   The AFS may fail due to ductwork collapse, fan failure, or loss of electric power.

3. Containment fails early (CFE) - i.e., failure of CM(E)

   The steel containment liner may be breached during an earthquake.

4. Vent and purge system (VPS)

   Containment venting or purging may be precluded due to ductwork failures or loss of electric power.

5. Annulus cooling system (ACS)

   The ACS may be failed due to ductwork or fan failure, or loss of electric power.

6. Containment cleanup system (CCS)

   The CCS may be failed due to pipe leaks or loss of electric power.

Additional events were added to the containment systems' fault trees to model these seismic failures; Table 13.3-6 lists the exact modifications made. Table 13.3-7 lists the basic events added along with a brief description of each event.

The Clinch River Breeder Reactor Plant Reserve Seismic Margins Study[1] provided a median capacity ($\check{A}$) for the steel containment liner of 0.88 g. No quantitative estimate of uncertainty for this value was given; the logarithmic standard deviation (β_c) was assumed to be 0.50. Fragility parameters ($\check{A}$ and β_c) for ductwork, fans, and the CCS piping were estimated from data contained in four LWR risk assessments (Zion[2], Indian Point[3], Oconee[4], and McGuire[5]) as discussed in Section 8.1.2. Table 13.3-8 lists the data used to estimate these parameters and the final results.

Table 13.3-6

INCORPORATION OF SEISMIC FAILURES INTO THE CONTAINMENT-RESPONSE EVENT TREE

Event	Gate Definition
NLF	NLF = NLFBE + RCB
AFS	CM100 (seismic) = CM100* (non-seismic) + AFSDUCT + AFSFANS
CFE	CFE = CFEBE + CONTAIN
VPS	H25DTAY (seismic) = H25DTAY* (non-seismic) + VPSDUCTA
	H25DTBY (seismic) = H25DTBY* (non-seismic) + VPSDUCTB
ACS	AM1 (seismic) = AM1* (non-seismic) + ACSDUCT + ACSFANS
CCS	W75TKY (seismic) = W75TKY* (non-seismic) + CCSPIPE

*Correspond to events depicted in the fault trees for these systems as provided in the appropriate sections of Appendix A.

Table 13.3-7

LIST OF BASIC EVENTS ADDED TO THE CONTAINMENT-RESPONSE EVENT TREE FOR SEISMIC ANALYSIS

Event	Description
ACSDUCT	Seismic failure of the ACS ductwork
ACSFANS	Seismic failure of the ACS fans
AFSDUCT	Seismic failure of the AFS ductwork
AFSFANS	Seismic failure of the AFS fans
CCSPIPE	Seismic failure of the CCS piping
CFEBE	Containment failure basic event (non-seismic reasons)
CONTAIN	Seismic failure of containment vessel
NLFBE	Reactor cavity liner fails (non-seismic reasons)
RCB	Seismic failure of the containment building internal walls and floors
VPSDUCTA	Seismic failure of the train A VPS ductwork
VPSDUCTB	Seismic failure of the train B VPS ductwork

Table 13.3-8

DATA USED TO ESTIMATE CONTAINMENT SYSTEM COMPONENT FRAGILITY

Event	Zion (.17)[a]			Indian Point (.15)			Oconee (.1)			McGuire (.15)			CRBRP (0.25g)		
	$\hat{A}$	RSM	β_c	$\hat{A}$	RSM	β_c	$\hat{A}$	RSM	β_c	$\hat{A}$	RSM	β_c	$\hat{A}$	RSM	β_c
CCS Piping	.73[b]	4.29	.41	.70[b]	4.67	.36	.35[c]	3.50	.60	.85[b]	5.67	.52	1.13	4.53	.47
Ductwork	.97[d]	5.71	.65	1.12[d]	7.47	.60	-	-	-	1.50[e]	10.00	.47	1.93	7.73	.57
Fans	1.74[f]	10.24	.49	1.16[f]	7.73	.43	>2.0[g]	-	-	1.61[h]	10.73	.63	2.39	9.57	.52

[a]Values shown in parenthesis are the SSE accelerations.

[b]Based on refueling water storage tank.

[c]Based on high pressure service water storage tank.

[d]Based on containment ventilation system ductwork.

[e]Based on ductwork in reactor building installed after November 1, 1980.

[f]Based on containment ventilation system fans.

[g]Based on reactor building cooling fans.

[h]Based on containment air return system fans.

Prime implicants for the end-state sequences were found using the SETS[6] computer program. Due to the complexity of these sequences, only prime implicants whose frequencies exceeded 1.0×10^{-9}/yr were found. The site seismicity data (Section 8.1.1), component fragility data (given above for the containment systems and presented in Section 8.1.2 for all other components), and the end-state prime implicants were combined using the ESA[7] computer program to determine the end-state frequencies. (See Section 8.1.4 for details of the computational method.)

Integrating the entire analysis by combining the seismic core-damage sequences, the core-response end state probabilities, and the modified containment-response event tree expressions for all accelerations examined, led to the radiological release end-state frequencies shown in Table 13.3-9. Table 13.3-10 displays the significant containment-response sequences and their frequencies which led to the radiological release matrix results. Based on the results, early releases are expected at frequencies of $\sim 10^{-6}$/yr for LOS events and $\sim 10^{-7}$/yr for ULOS events. These early releases are dominated by loss of containment integrity early due to the earthquake (containment-response sequence 71 dominates for ULOS and LOS events leading to moderate-early releases) or loss of containment integrity with failure of the CCS due to a seismic-induced pipe failure (containment-response sequence 73 dominates for LOS events leading to a high-early release). Late releases dominate the expected release end states for all four accident types contributing to the external event release frequencies. Containment-response sequences 1, 11, 51, 52, 61, and 62 dominate the moderate-late releases. These sequences involve success of all containment systems with or without

Table 13.3-9

FREQUENCIES OF RADIOLOGICAL RELEASE END STATES FOR EXTERNAL EVENTS

Time / Magnitude	EARLY 0 < t < 9 hrs	LATE t > 9 hrs	LONG-TERM t > 6 months
LOW	ε[a]	ε	ULOF = 1.1-7/yr[b] ULOS = 1.5-7/yr LHSE = 1.5-6/yr LOS = 1.4-6/yr
MODERATE	ULOF = ε ULOS = 2.8-7/yr LHSE = ε LOS = 1.5-6/yr	ULOF = 4.1-7/yr ULOS = 9.2-7/yr LHSE = 1.3-5/yr LOS = 1.4-5/yr	ε
HIGH	ULOF = ε ULOS = ε LHSE = ε LOS = 1.3-7/yr	ULOF = 6.6-7/yr ULOS = 2.5-7/yr LHSE = 2.4-6/yr LOS = 5.5-7/yr	ε

[a] 'ε' implies frequency <10-9/yr.

[b] 1.1-7 = 1.1×10^{-7}

Table 13.3-10

DOMINANT CONTAINMENT-RESPONSE SEQUENCES
FOR SEISMIC EVENTS

Radiological Release End State	Event	Sequence	Frequency (yr^{-1})
Low-long term	ULOF	40	5.9-8[b]
		62	3.5-8
		30	6.0-9
	ULOS	62	9.6-8
		72	3.3-8
	LHSE (Na early)[a]	62	1.0-6
		52	1.1-7
		64	9.5-8
		40	7.6-8
	LHSE (Na late)[a]	52	1.1-7
		2	2.2-8
	LOS	52	1.0-6
		2	1.5-7
		72	1.4-7
		62	1.3-7
Moderate-early	ULOS	71	2.5-7
		72	2.7-8
	LOS	71	1.3-6
		72	1.5-7
		101	7.7-8
Moderate-late	ULOF	61	3.1-7
		11	3.6-8
		51	2.8-8
		62	2.8-8
	ULOS	61	8.1-7
		11	9.8-8
	LHSE (Na early)	61	9.3-6
		51	1.1-6
		62	1.1-6
	LHSE (Na late)	51	9.5-7
		1	2.0-7
		52	1.0-7
		61	1.0-7
	LOS	51	1.0-5
		1	1.5-6
		61	1.2-6
		52	1.0-6

Table 13.3-10 (Continued)

Radiological Release End State	Event	Sequence	Frequency (yr^{-1})
High-early	LOS	73	1.0-7
		75	2.4-8
		74	1.1-8
High-late	ULOF	39	5.4-7
		29	5.8-8
		40	5.8-8
	ULOS	63	2.0-7
		53	2.0-8
		64	2.0-8
	LHSE (Na early)	63	8.5-7
		39	6.8-7
		65	3.0-7
		53	9.5-8
		64	9.5-8
		29	7.6-8
		40	7.6-8
	LHSE (Na late)	29	1.2-7
		53	8.9-8
		55	3.1-8
	LOS	53	3.1-7
		55	8.7-8
		3	4.5-8
		63	3.4-8
		54	3.4-8

[a]Implies sodium (Na) or fuel enters reactor cavity first.

[b]5.9-8 = 5.9×10^{-8}.

cavity liner failure and with or without containment integrity maintained in the late time frame depending on phenomenology effects. Sequences 29, 39, 53, 55, or 63 dominate the high-late releases depending on the type of event (LHSE, ULOF, ...). These sequences involve failure of one or more containment systems with containment integrity maintained at least until the start of the late time frame. The long-term releases are the meltthrough failure sequences dominated by sequences 40, 52, or 62 depending on the event of interest.

13.3.3 Integration of Common-Cause Initiator Sequences

Table 10.3-1 summarizes the core-damage frequencies for common-cause initiator sequences. Only fire-induced early loss of heat sink accidents are shown to have any significant contributions; therefore, only the fire-initiated sequences will be discussed here.

Fire-Induced Early Loss of Heat Sink Accidents

The fire-induced loss of heat sink early [FLHSE] accidents behave very similar to the LHSE accidents. Section 9.2 describes the main contributor to the FLHSE frequency to be a control room fire which propagates to the point of failing all core cooling and containment systems. Such a scenario could occur since all systems have controls in the control room area and therefore could be seriously affected by the control room fire. The most likely failure is due to the unavailability of electric power which would fail the core cooling and containment systems. The containment isolation system, however, is assumed to fail safe (i.e., operate properly) in the case of a control room fire since these valves

close on loss of power conditions. The FLHSE HCDA, therefore, is similar to a LHSE event with power unavailable. The core-response conditional probabilities for the LHSE event (see Table 13.2-1) apply here making the frequency of FLHSE with vessel bottom failure and sodium in the reactor cavity first equal to approximately 1.8×10^{-7}/yr (core-damage frequency of 2.0×10^{-7}/yr times the conditional probability of 0.89 from Table 13.2-1.) A FLHSE event with vessel bottom failure and fuel in the cavity first (sodium late) is estimated at 2.0×10^{-8}/yr.

Applying the containment-response event trees for the PLHS (early) event, i.e. LHSE, and assuming loss of power conditions with no chance for electric power restoration due to fire damage, yields the results shown in Table 13.3-11. Table 13.3-12 shows the dominant containment-response event tree sequences and their frequencies which led to the radiological release matrix results.

These sequences result in the potential for high-early or high-late releases due to the loss of power conditions since the containment systems cannot be operated to mitigate the consequences of the event. The sequences that dominate the high-early release end state involve reactor cavity liner failure, loss of containment integrity early due to phenomenological effects, and failure of all containment safeguard systems. The high-late release end state is dominated by sequences involving similar circumstances as for the high-early case but with containment integrity maintained. The long-term release end state includes the dominant meltthrough sequences.

Table 13.3-11

FREQUENCIES OF RADIOLOGICAL RELEASE END STATES FOR FIRE SEQUENCES

Time / Magnitude	EARLY $0 < t \leq 9$ hrs	LATE $t > 9$ hrs	LONG-TERM $t \geq 6$ months
LOW	ε[a]	ε	[b]FLHSE(Na early) = 1.8-8/yr[c] FLHSE(Na late) = 2.0-9/yr Total 2.0-8/yr
MODERATE	ε	ε	ε
HIGH	FLHSE(Na early) = 1.8-8/yr FLHSE(Na late) = 1.8-9/yr Total 2.0-8/yr	FLHSE(Na early) = 1.6-7/yr FLHSE(Na late) = 1.8-8/yr Total 1.8-7/yr	ε

[a] 'ε' implies frequency $<10^{-9}$/yr.

[b] Na early or late indicates whether sodium or fuel enters the reactor cavity first.

[c] 1.8-8/yr = 1.8×10^{-8}/yr

Table 13.3-12

DOMINANT CONTAINMENT-RESPONSE SEQUENCES
FOR FIRE-INDUCED HCDA

Radiological Release End State	Event	Sequence	Frequency (yr^{-1})
Low-long term	FLHSE(Na early)[a]	90	1.4-8/yr[b]
		110	1.8-9/yr
		100	1.6-9/yr
	FLHSE(Na late)	90	1.5-9/yr
High-early	FLHSE(Na early)	109	1.6-8/yr
		110	1.8-9/yr
	FLHSE(Na late)	109	1.6-9/yr
High-late	FLHSE(Na early)	89	1.3-7/yr
		90	1.4-8/yr
		99	1.4-8/yr
	FLHSE(Na late)	89	1.3-8/yr
		29	1.6-9/yr
		90	1.5-9/yr
		99	1.5-9/yr

[a]Indicates whether sodium (Na) or fuel enters the reactor cavity first.

[b]1.4-8 = 1.4×10^{-8}/yr.

13.3.4 Summary of Results

The results developed in the previous sections are summarized here. Table 13.3-13 shows the frequencies for the radiological release end states for all the significant sequences. The moderate-late release category dominates with frequencies of the most likely seismic sequences in the low 10^{-5}/yr range and the most likely internally initiated sequence in the low 10^{-6}/yr range. High release category sequences are of particular note; dominated by seismic sequences for both early and late releases.

Some HCDAs can occur at lower frequencies but could potentially have more severe consequences (e.g. - ULOF that is internally initiated; containment vessel failure by missile generation which could result in an early, more energetic release of fission products). Effort was expended in these cases to assure that the frequency of a significant release of fission products was considered low enough to be an insignificant contributor to the radiological release matrix. Only a thorough analysis providing feedback from the results of consequences to the public would assure this with more certainty. This was not done due to the termination of the CRBRP project and the elimination of the consequence analysis task from this study.

13.4 LIMITATIONS AND UNCERTAINTIES

Owing to the cancellation of the CRBRP, a detailed uncertainty analysis was not performed for this PRA. After the CRBRP termination, the objectives of the program were refocused on the basic model development and the engineering insights derived from their evaluation. The numbers,

Table 13.3-13

SUMMARY - RADIOLOGICAL RELEASE END STATES BY HCDAS

Time / Magnitude	EARLY 0 < t < 9 hrs	LATE t > 9 hrs	LONG-TERM t > 6 months
LOW	ε[a]	ε	LHSE (seismic) = 1.5-6/yr[b] LOS (seismic) = 1.4-6/yr LHSE (internal) = 3.4-7/yr ULOS (seismic) = 1.5-7/yr ULOF (seismic) = 1.1-7/yr FLHSE (fire) = 2.0-8/yr ULHS (internal) = 1.2-8/yr
MODERATE	LOS (seismic) = 1.5-6/yr LHSE (internal) = 3.2-7/yr ULOS (seismic) = 2.8-7/yr ULHS (internal) = 1.2-9/yr	LOS (seismic) = 1.4-5/yr LHSE (seismic) = 1.3-5/yr LHSE (internal) = 2.8-6/yr ULOS (seismic) = 9.2-7/yr ULOF (seismic) = 4.1-7/yr ULHS (internal) = 1.2-7/yr	ε
HIGH	LOS (seismic) = 1.3-7/yr FLHSE (fire) = 2.0-8/yr LHSE (internal) = 1.2-8/yr	LHSE (seismic) = 2.4-6/yr ULOF (seismic) = 6.6-7/yr LOS (seismic) = 5.5-7/yr ULOS (seismic) = 2.5-7/yr FLHSE (fire) = 1.8-7/yr	ε

a'ε' implies a frequency $<10^{-9}$/yr
b1.5-6 = 1.5×10^{-6}/yr

propagated and reported in this study are mean values. The previous subsections have identified some insights; (see Section 14) the limitations and key uncertainties are summarized qualitatively here.

13.4.1 Core-Damage Sequence Development

There are a large number of uncertainties that enter the evaluation of core-damage sequences. The project team's assessment of these uncertainties and their impact on the results is discussed below for specific elements of the analysis.

Initiating Events

The initiating event development was extensive and included consideration of LWR experience, sodium facility experience and the CRBRP-specific systems that could cause initiating events. The largest uncertainty is associated with events unique to the design of the CRBRP that have not yet been postulated. The functional review for initiators should have reduced the possibility of a major omission. The completeness of the initiating event list is, overall, considered to be a small contributor to uncertainty. It should be noted that some external event initiators were left out of scope (e.g., external floods, or nearby facility accidents) and would require analysis to complete the study.

Most of the data for initiating events was drawn from industry experience and the quantitative uncertainty due to initiating event frequency is also considered small. Of all the initiating events, the initiator frequencies for sodium leaks probably have the most uncertainty associated with them due to lack of data.

Sequence Delineation

The functional approach to sequence development is believed to result in a fairly comprehensive delineation of accident sequences. The development of success criteria for different functions was more difficult--relying on available studies and using engineering judgement to develop realistic criteria. In some cases, only licensing-type calculations were available and there is therefore some uncertainty as to realism in the success criteria. On the other hand, the lack of proven operating experience on some systems of unique design also results in some uncertainty. Overall, the sequence delineation is believed to have a moderate level of uncertainty relative to other parts of the analysis principally because there may be functional dependencies that were not recognized that would add other accident sequences. Also, the success criteria could be conservative, leading to over-estimation of some sequence frequencies.

Systems Analysis

The systems fault-tree analysis is also subject to limitations and uncertainties. Information was not complete for all parts of the design and several limitations are identified throughout Appendix A. If the design of these additional elements were carried out similar to the design of the systems analyzed, it is unlikely that major surprises would develop. These limitations are believed to result in a moderate level of uncertainty because the sequence results indicate that the basic plant design (i.e., the amount of redundancy) would not change as the result of new information. However, subtleties of design could lead to new sequence cut sets.

The system dependencies were researched extensively but the lack of operating experience does introduce a moderate level of uncertainty about the completeness of the system modeling. Known dependencies were treated directly in the fault trees and models were reviewed and iterated many times during the study.

The system level success criteria are believed to be less important than the overall system/functional criteria established in the sequence development.

Data

The largest source of uncertainty in the data base is associated with the components designed to carry or process liquid metals. A particular sensitivity discovered through a review of the core-damage results is the frequency and magnitude of leak initiators for sodium-bearing components. Some experience does exist but extensive data is currently unavailable, particularly for components unique to the CRBRP. For example, LWR experience includes a wide variation in steam generator reliability. The CRBRP system could also be subject to unique characteristics which could make them more or less susceptible to important failure modes.

Available data was reviewed for common-cause failure rates for important equipment, but this review was hampered by a lack of detailed information. Common-cause failure potential does pose a fairly substantial uncertainty--particularly for the evaluation of this plant which is for the most part triply redundant. Since the loops are nearly identical, many cut sets were generated with identical equipment failing in each loop.

Another aspect of data that is important is the potential for burn-in or wear-out failures affecting the plant during different stages of its operation. Again, this is most important for CRBRP-unique components which may not have applicable experience to determine failure characteristics prior to their use. The analysis did use first year initiator data to account for some first-time occurrences.

Uncertainty due to data variability is the element usually calculated in PRAs and reported as the uncertainty. The other sources identified in this section may actually be more important than data uncertainty. Typical results of detailed uncertainty analysis for other PRAs indicate that uncertainty due to this source results in error factors of 10 (ratio of 95th percentile to the median), or roughly two orders of magnitude uncertainty. There is no indication that this study would yield different results.

Human Reliability Assessment

This subtask of the analysis was also subject to limitations associated with lack of information; e.g., procedures for the plant. Several assumptions had to be made concerning operator training and procedural guidance. These limitations were made less significant, however, by the dominating element affecting the human reliability assessment--long times for potential recovery. The net result is that there are very few critical human interactions in the results for this study. Overall the human reliability assessment is seen (with respect to uncertainty) as comparable to other PRAs. The real uncertainty lies in the ability to quantify the events (the methodology) rather than any specific aspect of the CRBRP; and this uncertainty is moderate to large.

External Event Analysis

Additional uncertainties apply to the external event analyses. In particular, the seismic fragilities (failure rate versus acceleration) were developed based on limited information, comparison with other studies, and engineering judgement. The fragilities used in other studies are very uncertain also--but more detail was available to judge the applicability of data. The uncertainty in the seismic results is believed to be moderate to large.

Common-Cause Initiators

Although a great deal of time was expended analyzing the common-cause initiators, no significant sequences were found. Some design details limited the analysis (location of some components, exact locations and flow paths of drains) but overall the analyses were a comprehensive probabilistic treatment. The uncertainties associated with these analyses are large, but it is the project team's assessment that even with consideration of large uncertainties, the relative importance of these events would still hold.

13.4.2 Phenomenological Analyses

Core Phenomena

The uncertainties associated with the specific aspects of core phenomenology are actually inherent to the methodology--the event tree quantification attempts to systematically delineate the potential phenomena as they are currently understood. The probabilities applied are, for some branch points, indications of the uncertainty in the knowledge to be able to predict the actual physical progression of the sequence.

A 0.9 to 0.1 probability split does not necessarily indicate that 10% of the time a sequence would take the alternate path--the event may be deterministic in nature and it would always take one path. The probabilities are used to indicate the best judgement as to which scenario is likely based on the available information.

This being the case, there is a great deal of uncertainty discussion within Section 11. The most important considerations are summarized below.

Only limited new analysis was performed in support of this study. The limitations of the core phenomena analysis are therefore directly linked to the limitations in the body of knowledge which supports the breeder design. The most significant lack of information relates to protected accidents--the past research effort has been on unprotected accidents and the applicability of many of the results to protected accidents is not entirely clear.

A particular issue important to the results for protected accidents is the phenomena associated with the presence of sodium above the core at the time of fuel disruption. Based on analysis performed for Phase I of this study, the assumption was made here that there will not be sodium above the core for protected loss of heat sink events. The analysis of this phenomena is not conclusive, however, and it would be an important element of protected accident phenomenology research.

A number of assumptions were made in the phenomenology analysis which would affect the results. Additional verification of these assumptions

would reduce the uncertainty associated with this part of the analysis.

The key assumptions include the following:

- The reactor vessel head meets the structural design requirements despite observed kinematic failure modes in scale model tests,
- Sodium is very likely to be above the core at the time of fuel disruption for many accidents,
- Leaks that result in protected loss of sodium events would allow the reactor vessel to drain prior to bottom head penetration,
- The primary pumps are assumed to be running for the unprotected loss of heat sink and the transient overpower events, and
- The transient overpower results considered ramp rates below 20¢/sec.

Additional uncertainties are associated with lack of research for particular phenomena.

- There is no experimental evidence for situations including saturated inlet conditions,
- The time between fuel channel dryout and sodium reentry is uncertain but a significant time difference could possibly allow fuel disruption.

These potential effects would alter the assessment of radionuclide release to the public (if this analysis were to be performed as a follow-on to that presented here).

Containment Phenomena

The containment-response analysis is also subject to the same limitations as the core-phenomenology analysis - a lack of information on protected events. This is generally less limiting for the containment analysis due to lower sensitivity to the initiator category. A moderate uncertainty is associated with this lack of information.

Because the containment-response event tree includes systems, the uncertainties associated with systems and data analysis apply to the containment-response analysis. The results (if carried out to radionuclide analysis) would also be very sensitive to two elements of uncertainty associated with systems analysis: (1) the impact of time on the static fault tree models and (2) the success criteria associated with systems being operated beyond their intended capacity. The expected release results are greatly affected by the operation of these containment systems. The uncertainty in the assumptions concerning the operability of these systems following containment failure could greatly affect the results of risk calculations. Further analysis to verify containment failure modes and their impact of all other systems would be required to reduce this uncertainty.

Containment failure associated with hydrogen burning, particularly as it could affect the polar crane, was deemed relatively unimportant but is a source of significant uncertainty in the PRA team's judgement. Additional research on flame height, temperature profiles above the flame and the physical effects of the heatup would provide additional confidence in an accurate treatment of this phenomena. In particular, structural analyses more sophisticated than those performed in Appendix B would be beneficial in confirming that polar crane elongation or its falling on the containment floor will not cause containment failure.

The interaction between containment failure and confinement failure would also require additional analysis to understand the effect of one on the other and the resulting radionuclide release.

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Section 14

Conclusions

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Section 14

CONCLUSIONS

Specific conclusions from each major task of the PRA have been presented in the preceeding sections of this report. The most important of these conclusions are summarized here with particular emphasis on the engineering insights derived from the study. More detailed discussions on the insights may be found in the appropriate sections of the report: Section 10 for core-damage sequences and their frequencies; Sections 11 and 12 for core and containment phenomena, respectively; and Section 13 for the results of integration of the core-damage sequences with the core and containment phenomena.

14.1 INSIGHTS FROM CORE-DAMAGE ANALYSIS

The insights from the analysis of the core-damage sequences are divided into two sub-sections. Section 14.1.1 discusses those plant features that directly affected the core-damage frequency. Section 14.1.2 includes those insights that are also of interest but that did not significantly affect the final results.

14.1.1 Plant Features Important to Core-Damage Frequency

The plant features that are most significant with respect to core-damage frequency are summarized below. In addition to the features resulting in dominant sequences, the design elements responsible for low frequencies of certain sequence types are also outlined.

Timing of Core-Damage Sequences

Clearly the most important feature of the plant that affected nearly all of the results is the timing associated with core heat-up and damage.

The accident sequences can be grouped into three categories relative to the timing of the accident progression:

- Unprotected events (failure to shut down the reactor) which evolve over a period of minutes,
- Loss of heat sink early (within the first two hours before the passive systems can remove all of the heat) in which core damage would not occur for 10-12 hours, and
- Loss of heat sink late (after two hours of successful cooling) for which 20 hours or more is available before the onset of core damage.

The first category is the only one that evolves quickly in time, but the unprotected events have a very low frequency due to the design of the plant protection systems. The second category is the most significant in terms of core-damage frequency, but the results were greatly affected by the availability of recovery actions. These recovery events account for the ability to use alternate means of cooling to correct some of the faults affecting the sequence (e.g., recovery of offsite power) within the 10-12 hours available. The recovery events do not account for "heroic" or highly innovative solutions which may also be possible in a 12 hour time frame, but which are not within the state-of-the-art to assess probabilistically.

The third category of sequences was found to have a negligible frequency for two reasons: (1) the cooling systems--the protected air-cooled condensers--are essentially passive, and (2) the 20 hours for recovery

allows even more combinations of alternative equipment to be used to cool the core. For these sequences, the effect of recovery actions is also cummulative--a cooling method that works for 1 hour may add an additional 5-10 hours to the total time available for action.

In summary, the timing of core heat-up had a large effect on the accident sequences by causing many of the scenarios to have low frequency. The heat capacity of the sodium coolant enables a significant grace period during which a loss of cooling can be investigated and action to restore cooling can be attempted.

Seismic Design

The CRBRP is well designed to meet the criteria associated with the design basis earthquake. The frequency of core damage due to earthquakes with a magnitude below 0.25 g is small compared to higher acceleration earthquakes.

The seismic-induced core-damage sequence frequency dominates the overall core damage frequency. This is reasonable because the redundancy of the three heat removal loops is most compromised by events which could affect the similar equipment in the loops simultaneously. The results indicate that the rupture disks in the loops are the dominant failure mode for the significant seismic sequences. The analysis of the rupture discs included an examination of the coupling between the disks in different loops (based on orientation and location) and a model for the coupling between the two disks which are in series for each rupture disk location. The resultant model is believed to be a realistic representation based on available information. The data for the fragility of

the disks is limited, and this is one of the most important uncertainties in the results.

The failures of the three loops must be accompanied by failures of the direct heat removal service (DHRS) before all means of core cooling is lost. The DHRS failures that appear in the important seismic sequences include failures of the heat exchangers, loss of service water cooling, leaks in the DHRS system or the primary system, and power failures. All of the important DHRS failure modes are seismically induced.

Failures of buildings and support structures also contribute to the seismic core-damage frequency. This analysis was less detailed due to lack of more specific information, but important building failures coupled with other events are included in the final results.

The other type of seismic sequence involves a leak in the primary system followed by failure of the corresponding guard vessel--the failures occurring as a result of the seismic acceleration. About one-half of the total frequency is associated with these types of sequences. The leak sequences are important because they have the potential to fail the heat removal through all three loops and, at the same time, fail the DHRS heat removal mode.

Plant Protection System Design

The results of this study indicate that the frequencies of unprotected* core-damage sequences are one to two orders of magnitude less likely

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*Failure to shut down following an initiating event.

than protected events. The two reactor shutdown systems are reliable, very diverse and nearly entirely independent. The frequency of unprotected accidents for earthquake initiators was estimated to be about an order of magnitude more likely than for other initiators. The large earthquakes were judged to have the potential to result in common-cause failures causing mechanical failure of the rods to insert into the core.

The frequency of these accidents was assessed using available data, which is very limited. Considerable engineering judgement was involved in the assessment of the probability of common-cause failures between the two systems due to causes that could not be specified. The quantification drew on the work that has been done in the light water reactor industry, but accounted for the additional redundancy associated with the CRBRP design. Even considering the uncertainties associated with the frequency of these accidents, their relative order of importance compared to the other sequence types would probably remain the same.

The phenomenology for unprotected events is also dependent of the status of primary heat transport system (PHTS) flow. The unprotected sequences initiated by internal events were found to fall into the category for which flow was available because the same event that would cause the protection systems to fail would most likely also fail the signal that would cause the main PHTS pumps to trip. For the seismic sequences, the opposite is true--nearly all unprotected events are associated with a loss of flow because the pumps fail on loss of offsite power. Offsite power loss is very likely in earthquakes of any significance.

Steam Generating System Depressurization

An important element of many of the internal event sequences is the potential for steam leaks resulting in the depressurization and isolation of steam generating loops. There are a large number of leak paths (13 per loop) through safety, relief and vent valves. The relatively small inventory in a loop coupled with the limited time available for isolating individual valves or modules before the steam drum in the affected loop is isolated make this an important plant feature. In addition, these events could be operationally difficult; i.e., difficult to diagnose, particularly when more than one steam relief is occurring. These events were calculated to be the largest cause of steam generating loop unavailability for non-seismic sequences.

Sodium Leak Initiators

The frequency of core damage due to a loss of primary coolant (not caused by earthquakes) is less than 1% of the total frequency. The leak initiating events were found to have very low frequencies, and for the leaks that do occur, a number of options are available prior to a loss of all core cooling. The frequency of pipe leaks was estimated using an approach that quantified the phenomenology of the pipe break. The result compares favorably with other probabilistic analyses for pipe break frequency for low pressure sodium piping. Only very limited data for actual pipe experience under conditions similar to the breeder exists, and therefore, there is a significant degree of uncertainty associated with the pipe break sequences. In addition to low initiator frequency, the location of potential pipe leaks also affected the overall result. The pipe leaks were grouped into categories that defined

the plant response to the initiators. For many leak locations, the effect on the plant is limited.

Guard Vessels

The design of the guard vessels for the PHTS, reactor vessel, and intermediate heat transport system is also an important feature, as it provides an additional safeguard against sodium leaks. The guard vessels do play a role in the seismic sequences--leaks within guard vessels followed by failure of the guard vessels account for a significant portion of the seismic core-damage frequency. Additional analysis of the seismic response of the guard vessels is suggested by these results.

The internal event leak sequences were fairly low in frequency even without consideration of guard vessels. The frequencies of important sequences from leaks within guard vessels is negligible.

Closed Loop Cooling - PACCs

The protected air cooled condensers (PACCs) offer a method of closed loop and nearly passive cooling which is very reliable. This, coupled with the long time for recovery (greater than 20 hours) after a successful early cooling phase, make long-term loss of heat sink sequences negligible. The only active requirement for the PACCs is the addition of makeup water to handle losses, but this too does not have to be done for a great deal of time and there are numerous methods, both standard and improvised, of supplying this water. No credible scenarios were discovered which would result in core damage due to long-term heat sink failures after a successful short-term response.

Separation of Equipment

Extensive analysis of common-cause initiators did not yield any dominant core-damage sequences. The most important reason for this is that the separation of equipment in the plant greatly limits the effects of any common cause that is dependent on physical proximity. The physical separation is due directly to the design criteria for separation and is indirectly supplemented by separation required for liquid-metal safety considerations. The common-cause analyses focused on potential compromises of this independence, but no significant sequences were found.

Independence of Steam Generating Loops

Another element confirmed principally through the examination of low frequency sequences is that the steam generating loops are largely independent. The most frequent accident sequences did involve failures of all three loops, but overall probability of three-loop failure is fairly small. The major cause of three loop failure was assessed to be common-cause failure of identical equipment.

Direct Heat Removal Service

The DHRS capability is another reason for the relatively low frequency of core damage. It affords an alternative means for shutdown heat removal that can be used for most accident sequences. Changes to the design that would make DHRS more resistant to earthquake-induced failures and less susceptible to PHTS leaks would decrease the core-damage frequency even further. Many parts of DHRS are normally operating, thus the status of important parts of the system is known more readily than for standby systems. DHRS availability during the conditions of failure of

the cover gas system can be improved by making the switchover to standby argon accumulators automatic instead of manual.

14.1.2 Additional Insights

A number of other observations were made during the analysis that, while not significant in terms of core-damage frequency, could be important relative to future design considerations. The insights are summarized here; additional detail may be found in the report section appropriate to the specific topic.

SGAHRS Actuation

The steam generator auxiliary heat removal system (SGAHRS) actuation system was analyzed in detail and found to be fairly reliable with dominant failure modes all being human errors. The dominant failure modes are recoverable from the control room with manual actuation. One of the design objectives of SGAHRS was to limit steam releases through safety/relief valves. This objective is accomplished for most scenarios, but failure of turbine bypass (steam dump) would not directly result in SGAHRS actuation and a steam release could occur.

Steam Drum Dryout

Several of the accident sequences result in loss of makeup to the steam drums and subsequent dryout in 10 minutes. It was assumed in the study that after dryout had occurred, no water would be added. This was due to concerns about the potential damaging effects of relatively cold water entering the hot system. This issue would require a clear resolution to provide direction to operators who could be faced with a

dilemma--no core heat removal, feedwater available, but uncertainty as to whether the introduction of feedwater will help the situation. This issue is also important relative to the financial risk associated with loop damage if the re-introduction of feedwater leads to severe damage.

Operator Action During Unprotected Accidents

Certain failures of the plant protection system could be mitigated by timely operator response. This was credited as a recovery for the appropriate sequences with a five minute time frame being critical. Recent studies[1,2] indicate that feedback effects on reactivity due to the thermal expansion of the fuel and core could increase this time to fifteen or thirty minutes. These considerations could change the analysis of unprotected accidents; however, at the time of this study, the examination of all the potential positive and negative impacts of this feedback were not fully examined. Of particular relevance to the PRA sequences identified are two items: (1) the potential mechanical effects of the thermal expansion and the resultant impact on the shutdown system, and (2) the time dependence and the long-term stable condition following the accident.

Emergency Chilled Water Actuation

The emergency chilled water system actuates automatically on loss of flow in the normal chilled water system. Some normal chilled water failures could lead to continued flow but no cooling, thereby requiring manual actuation of emergency chilled water. Minor design changes or clear procedural guidance could limit this potential problem.

Division III of Electric Power

The third division of electric power was a new plant modification included in the PRA. The results of the sequence analysis indicate that some changes in load distribution could better utilize the three divisions of power. The heating ventilating, and air conditioning (HVAC) loads were found to be less than optimum for reliability of the overall plant during use of the diesels.

Latent Human Errors

Latent human errors, i.e., errors occuring as results of test or maintenance activities, were not found to be important from an overall CRBRP risk perspective. However, LWR experience suggests that latent errors occur most often in events that:

1. breach containment integrity
2. partially disable a safety system
3. affect systems in different units in multi-unit plants, or
4. initiate reactor trip or lead to some form of plant upset.

For the CRBRP, errors that may lead to the breach of containment integrity were quantified based on LWR experience as adapted to CRBRP containment penetrations. This type of latent event was relatively important to the failure of containment safeguards. Since the CRBRP includes triple redundancy in many major safety systems, the partial disabling of safety systems was relatively unimportant. Latent errors that initiate plant upset were included in the data used to estimate initiator frequencies and not generally singled out; their relative importance is not believed to be significant.

The latent error frequency during the first years of the plant operation might also be somewhat of a concern due to the unfamiliarity of some of the equipment. This could be more of a problem for the CRBRP which would involve entirely new types of systems for most of the personnel; e.g., electro-magnetic pumps.

Programs were in place to evaluate all experience at facilities, such as the Fast Flux Test Facility (FFTF), and to determine the applicability to the CRBRP design. That program would probably have helped to limit latent human errors if the insights were disseminated to the proper personnel.

Front-Line Operator Actions

In general, the CRBRP is designed to automatically respond to an accident condition and there are, as a result, few front-line actions that the operating crew would have to take, without assuming multiple system failure. The notable exceptions are: (1) the manual actuation of emergency chilled water under some loss of normal chilled water scenarios and (2) the manual alignment of standby argon accumulators to provide cover gas to PHTS pump seals upon loss of cover gas.

Recovery actions are those that require the operator and substitute his actions for systems that have failed. Notable examples include: (1) response to an unprotected accident; (2) the manual actuation of SGAHRS following failure of the automatic actuation system; and (3) the manual actuation of DHRS as a last resort in most protected accidents.

These actions are necessarily important because they represent one of possibly several levels of defense to mitigating an accident. Assuming average training and procedure support, these actions do not presume an extraordinary workload on the operator, except possibly for an unprotected accident, which would leave little time to diagnose and act, or the loss of cover gas scenario which would require action within containment.

Concerning the operator action associated with multiple system failure, most of the important actions relative to core damage would offer significant time for decision, action, and recovery. The capacity for heat removal of the primary system would allow time for the effects of the multiple failures to be assessed. Aside from the unprotected accidents, there are no scenarios involving fast-acting transients in the primary system. Thus, the primary system is probably less complex operationally than LWRs since there are no effects analogous to blowdown and rapid phase changes.

The secondary systems can involve operational complexities. The number of different modules and the need to assess and correct situations quickly could be important to the overall safety of the plant. The secondary system leaks described previously involve common-cause failures occurring in multiple loops. Action to diagnose and correct the problems could be complicated by the multiple loop involvement and by the number of possible leak paths. Once these events had progressed to the point where loop isolation on low pressure automatically occurred, the action to remedy the situation would be limited by the time to dryout. After dryout, the loop would be unavailable as an

option, even though the ability to provide water might be restored. This would perhaps be the biggest dilemma associated with the accident sequences found to be important in this study--available heat removal paths that could not be used because of the potential for damage on re-introduction of water.

Long-Term Events

As discussed in the previous section, some accident scenarios (in fact, most protected accidents) would evolve over long durations — 10-20 hrs. No method exists to quantify the likelihood of actions that affect the course of such long-term events, either positively or negatively. It was generally assumed that recovery from long-term events was reliable and that the contribution to the CRBRP core damage frequency from these events was negligible.

Component Cooling Systems

Many other PRAs have found the reliance of equipment on common cooling systems to be a dominant contributor to core-damage frequency. The CRBRP design includes many systems (NCW, ECW, NSW, ESW, and SSCCW) which supply cooling to important equipment. Many of these systems are also quite redundant. Dependencies through cooling water systems do not contribute to the overall core-damage frequency.

Combustible Loading

In addition to the separation of important equipment, another reason for the unimportance of fire initiators is the absence of any concentrated combustible loading. No one cell contains more than 3% of the total plant combustible loading.

Liquid-Metal Spill Design Features

As with all common-causes acting through the local environment, the separation of equipment limits the importance of liquid-metal fires. Another reason for their lack of significance is the inclusion of design features to mitigate spills--piping jackets, drains, fire suppression decks, lined cells, and inerting. Some features of liquid-metal spill scenarios were reviewed, but quantification of the effects was difficult. A good example is the effects of aerosols outside the cell in which the incident initiated. Even low concentrations of aerosols could have impacts that have not yet fully been assessed. It is still believed, however, that the frequency of core damage due to liquid-metal spills is low.

Turbine Building Flood Scenarios

Turbine building flood events were thoroughly reviewed to determine potential core-damage risk. Although this risk was found to be negligible, a turbine building flood could lead to significant damage of equipment and plant unavailability. Experience indicates a real potential for this initiating event. In addition, the isolation of circulating water system flow is very difficult due to backflow from the normal plant service water through the circulating water discharge pipe. Some relatively minor changes to the turbine building design, the inclusion of procedures for this event, and some direct indications of flood could limit this potential financial risk.

Compressed Air System Design

The loss of instrument air initiating event was found to be unimportant relative to the core-damage frequency. It does however lead to a loss of

feedwater and other system failures throughout the plant. The initiator frequency is dominated by passive failures of a large number (13) of valves between the compressors and the supply header. Minor design improvements could limit the frequency of this initiator.

14.2 INSIGHTS FROM PHENOMENOLOGICAL ANALYSES

14.2.1 Insights From Core Phenomena Analysis

A review of the core response analyses and results indicate a number of insights relative to research efforts and the state of knowledge concerning this phenomenology. Other insights are included in Section 11.

End States of Core-Damage Phenomenology

The methodology used in this study examined a full range of potential outcomes for the core-damage sequences. Two end states, reactor vessel bottom penetration and core-damage with debris retention in the reactor vessel, were the dominant outcomes probabilistically for all classes of accidents. End states with more severe outcomes were predicted to be orders of magnitude less likely.

Damaging Energetics

The results of this study indicate that damaging energetics are very unlikely due to the design of the overall reactor system. The design strength of the reactor head coupled with the energy absorbing capability of the core barrel and upper internal structure virtually eliminates catastrophic head failure and makes head damage unlikely. Reactivity ramp rates required for head damage were estimated not to occur. The probability of generation of containment-failing missiles

due to energetics was found to be extremely low because the scenario is non-mechanistic.

In-Vessel Coolability

The possibilities for retention and cooling of a disrupted core within the primary system were found to be probabilistically important for several sequences. The operability of the primary pumps resulting in sweepout from the core is the principal phenomenon occurring.

Protected vs. Unprotected Events

One important insight is that there is only limited research available for some sequence types. In particular, the protected events and the unprotected loss of sodium events are significant in this study, but there was very little information to support the quantification of the event trees. The engineering judgement used to assess these sequences had to rely on analogy to available research. Investigation into the appropriateness of these analogies and the examination for phenomena unique to these accidents would be required to provide assurance that the characterization described in this study is appropriate for these accidents.

The protected accidents evolve over a longer time period (hours as compared to seconds for the unprotected events) and some of the phenomena may be less critical, but a full understanding of potential outcomes would benefit from additional research.

The evaluations for unprotected transient overpower and unprotected loss of flow (UTOP and ULOF) are based on over two decades of research on these accidents. Although probabilistic studies have been somewhat

limited, it is the project team's assessment that the analyses of these two accidents are subject to smaller uncertainty than the other sequence types.

The analysis also pointed out an interesting finding relative to energetics, although it was not important to the results due to the head strength considerations discussed above. Based on a simplified analysis, the unprotected loss of sodium event was calculated to impart slightly more energy to the head than the ULOF event--traditionally considered to be the bounding case.

Reactivity Feedback

Another consideration for estimating the likelihood of shutdown for the protected accidents was discovered late in the study. The effect of radial fuel pin bowing and axial control rod expansion on the reactivity of the core were not considered. The negative reactivity feedback associated with the expansion of the core has been shown in recent studies to result in an automatic reduction in core power. Not enough information was available to fully incorporate this as quite a few questions remain as to mechanical effects, time dependence, and dependencies on other phenommena. This consideration is particularly relevant to new designs that could perhaps be self-terminating.

14.2.2 Insights From Containment-Phenomena Analysis

The containment-phenomenology analysis, analogous to the core-damage phenomenology, used available research to assess the probabilities of the full range of end states that could result from a core-damage

accident. The overall result is dependent on the nature of the accident sequences because of the system failure events that are included in the containment tree. However, several insights were derived from the actual assessment of containment phenomenology.

Hydrogen Flame Effects

The area of greatest concern in the assessment of containment phenomenology involved the potential for polar crane interactions with the hydrogen flame that would occur approximately 10 hours after core debris entered the cavity. Although the failure of containment by this mode was assessed as unlikely (0.1), the phenomenon remains a concern relative to assurance of long-term containment integrity. Provisions to park the crane at a location not above the vent which would be the source of the flame were in place but additional precautions could be taken to limit this potential failure mode.

The phenomenology associated with flame height and the effects of local crane heating was studied thoroughly based on available information, but additional analysis that included an integrated review of the entire scenario and that fully examined both subtleties and local effects on the crane or shell would be required to fully address this issue.

Early Containment Failure

The early containment failures were found to be rather unlikely (<0.01) for most sequences. However, several scenarios could be established for early failure and the probabilities are affected by considerations other than purely phenomenological ones. For example, the vent/purge operation is dependent on operator action and is part of the overall accident

management strategy. Incorrect or untimely operations involving vent/purge could possibly lead to concentrations of hydrogen that could detonate upon initiation of purge. Clear direction on when to use vent purge and the use of unambiguous instrumentation detailing containment conditions (H_2 concentrations) would be valuable assets to prevention of this scenario.

The manual isolation valves between the reactor cavity and the reactor containment are another feature that could affect the probability of containment failure. If the valves were inadvertantly left closed, the reactor cavity would overpressurize following the entry of the core into the cavity. Strict control of these isolation valves would further limit the possibility of containment failure via this mode.

Containment Isoation

The unavailability of containment isolation was calculated to be less than for light water reactor containments due to the limited number of penetrations. In addition, the inerting and separation of cells in containment allows for somewhat of a continuous status check for some failure modes, because pressures are continuously monitored.

Out of Vessel Coolability

Out of vessel core coolability debris was found to be probable for all types of sequences--very likely for unprotected sequences not involving a loss of sodium initiator and also for the protected loss of heat sink with sodium in the reactor cavity after or at the same time as core

debris. The sweepout under the vessel is expected to occur more rapidly than particle settling thus resulting in coolable geometrics.

Containment Failure Due to Missiles

Because of the core response result that energetic-caused head failure is probabilistically negligible, containment failure by missile generation was not found to be a concern.

Basemat Penetration

The analysis of concrete interaction indicated that basemat penetration is not inevitable or likely. The sodium-concrete interaction is probably self-terminated prior to the breach of the basemat.

14.2.3 Insights from the Integration of Core-Damage Sequences and Phenomenology

Although the calculation of the magnitude and frequency of the radiological releases was not part of the reduced scope of the study, the results of the core-damage sequence analysis and the core-and containment-phenomenology analyses were integrated as described in Section 13. Because of the reduced scope, the results of that integration are only preliminary. However the insights gained from that analysis could help direct any future efforts in this area.

The results of the integration were presented earlier in Table 13.3-13. The dominant category in terms of frequency was the moderate release in the late time frame (>9 hours after the core debris breached the reactor vessel). For most of these sequences the containment integrity is maintained, or the containment integrity fails but most of the containment

systems work, thus reducing the release. The high-late category also has a significant frequency, for these sequences one or more containment systems is unavailable. For the seismic sequences the containment system failures are due predominantly to seismic-induced system failures or power failures. Power failures and phenomenological containment failures (hydrogen burn) dominate the high-late category for internal initiators.

The early releases for the internal events are dominated by phenomenological events that cause containment failure. For the seismic initiators the dominant consideration is the seismic-induced failure of the containment cleanup system piping. Containment integrity failure directly as a result of the earthquake also contributes.

There are several insights that are apparent from a review of the results of the sequence integration. The dominant system failures that affect the containment results are caused by power failures that affect containment systems. The annulus filtration system was found to be the most affected by power failures because its mission is nearer the start of the accident. Most of the other containment systems are not needed for 10-12 hours after the core debris has entered containment at which time the probability of power being restored is greater. Nevertheless the availability of power still effects these later systems. Due to the recent addition of the third train of power to the design it is possible that more reliable alignments of power for the containment systems could be devised, or that more flexibility in recovering power to these systems could be provided.

The most important seismic effect involves the failure of the containment cleanup system piping due to the earthquake. This failure mode was the dominant cause of high releases. Seismic-induced failure of containment was also important for several of the sequences, particularly as a cause of early releases.

14.3 THE USE OF CRBRP PRA RESULTS IN CONSIDERATION OF FUTURE DESIGNS

Probabilistic risk assessment methodology can be used in concert with more traditional design practices to help improve the designs of reactor systems or to optimize available designs. It was originally a goal of this study to establish a risk management program that would be used in the final design stages of the plant and be used after plant startup in review of operating experience and analysis of modifications and safety concerns. The first step in this process had already been implemented with the development of a design feedback procedure to formally review PRA-generated design changes. After cancellation of the CRBRP, the goals of this study were redirected toward completion of the models with emphasis on lessons-learned that would be useful in the future. The insights described above in Sections 14.1 and 14.2 are directly applicable to future design considerations. They are summarized here again with emphasis on suggested changes or improvements which could perhaps offer even more optimal safety.

It must first be noted that the results of this study do not seem to indicate any significant deficiencies in the design that was analyzed.

There are no particularly dominant design features in terms of negative impact on safety. In comparison to PRAs of similar scope on light water reactors, the results calculated here indicate that the CRBRP core-damage frequency is in the same range as that of other plants for seismic events and is actually somewhat lower than the calculated frequencies at other plants for most other initiators. When these results were obtained for this study considerable time and resources were expended reviewing the models, data and results to ensure that the reported sequences and frequencies are as accurate as the methodology allowed. This review included the examination of sequence types or design subtleties that had been found in other PRAs (e.g. service water system dependencies or pump seal failures). This additional review and quality assurance step did not identify any additional sequences--but it did uncover the reasons for the relatively low core-damage frequency.

The design features most important to the results are described below. Two important aspects must be noted relative to these findings: (1) the uncertainties associated with PRA must be recognized when drawing conclusions from these insights and (2) suggestions on changes that would improve safety may not be required or optimal. The results in the previous two sections have concentrated on engineering insights. Although these were derived using probabilistic analysis, most are also just as significant when considered in a more traditional engineering sense, without probabilities. The second note of caution refers to the fact that design optimization includes consideration of factors other

than safety--costs, reliability, etc. The changes suggested may not be necessary or a more integrated assessment (such as a financial risk assessment) could suggest different combinations of improvements.

Liquid-Metal Coolant

The use of liquid metal as a primary coolant is the reason for the relatively slow evolution of most of the accident sequences. The timing considerations were, in turn, very important to the frequency of core damage for most types of accidents. The analysis of initiating events due to the use of liquid-metal coolant did not identify important sequences. For example, liquid-metal fires or leaks were not dominant sequences.

The overall conclusion is that the liquid-metal coolant when used with design features such as those in the CRBRP design has significant advantages relative to safety as measured by core-damage frequency. Other considerations such as plant reliability, personnel safety and economic risk of non-core damage accidents would need to be considered to make a broader conclusion.

Seismic Design

The results for the PRA indicate that seismic-induced sequences are the dominant contributor to core-damage frequency, and they are probably also the dominant public health risk sequences. Probabilistic seismic analysis is subject to considerable uncertainty but the results do suggest a need for additional seismic design assurance in several areas:

- The rupture disks in the secondary loops needed to protect against sodium-water interactions are the dominant cause of failure for the three loops for seismic events. The data for the fragility of these disks is very limited but even without consideration of probabilities it seems apparent that these disks are the weakest point in the loops relative to earthquakes. The model used to calculate the core-damage frequency included the consideration of each disk in a pair as independent up to the safe shutdown earthquake, and sensitivity to coupling assumptions was also reviewed. The orientation of the disks in the loops suggested a very high degree of coupling, changing this could impact the results somewhat. Test data for different earthquake intensities and under conditions simulating actual loop conditions would be required to more thoroughly assess this risk. Future plants' seismic designs could be developed with additional recognition of the importance of the impact of these rupture discs, although their reliability in performing their intended function should be lessened by seismic hardening.

- The potential for public risk associated with core-damage events is significantly affected by both the integrity of the containment and the operability of containment systems. The results of this study suggest that the operability of some of the containment systems, particularly the containment cleanup system, could be improved for some earthquakes larger than the design basis earthquake.

- Power failures are usually found to be significant contributors to earthquake-induced core-damage sequences, and this was true for this study also. Power failures were important to the failure of the direct heat removal service and also to the containment systems. Additional hardening of power supplies and relocation of some redundant load centers that are at locations of similar seismic susceptibility would change the results for this plant.

- The reactor protection system was found to be highly reliable but additional analysis of the seismic interaction and the potential for common-cause failure could provide additional safety assurance.

- The important seismic sequences include leaks followed by failures of associated guard vessels. The fragility of the guard vessels and the potential for systems interactions between guard vessels and the piping they enclose is also warranted.

Containment Design/Hydrogen Interactions

One of the most important phenomena associated with the integrity of the containment is the overall ability to withstand all of the effects of hydrogen deflagration. This is the one containment failure mode that is most important in the internal sequence results. Probably the most important insight from the containment analysis is that the interaction of any potential hydrogen flame with the containment structure or other systems could play an important role in post-accident consequences. For this design, the potential interaction with the polar crane was found to be deserving of more study. Although not identified as the most likely scenario, new designs that eliminate the interaction of hydrogen with the polar crane would probably be an improvement over the current design. Additional analysis of flame height, temperature profiles and effect on the structures from the present design could reduce the uncertainty associated with this failure mode. In addition, administrative steps to ensure that the crane is parked away from potential flame location and addition of a new vent design downstream of the cavity rupture disks to reduce flame height would probably also be warranted.

Closed-Loop Cooling Systems

The nearly passive protected air cooled condensers (PACCs) that are part of this design are very important to the overall results. The PACCS can assume the decay and sensible heat load of the primary system after 1-2 hours of cooling using more standard systems such as the auxiliary feedwater system. The comparison of these results to those of other PRAs identified this design element as being a very important reason for lower core-damage frequencies. No changes to the current design were

suggested by this study. One area of consideration in future designs, however could be the practicality of extending this concept (at least as a backup) to include heat removal from the onset of a shutdown. This could further improve the safety of the design and offer a clear advantage over other reactor designs.

Direct Heat Removal Service (DHRS)

The inclusion of the DHRS in the design is another reason for the relatively low core-damage frequency--the results for internal events would have increased approximately two orders of magnitude if the system were not included in the design. The DHRS system offers not only redundancy, but diversity and therefore additional independence from the types of common-cause failures affecting all three loops. Although not found to be dominant probabilistically, the systems analysis suggests that future designs for DHRS-type systems that are less sensitive to primary leaks would offer an advantage over the design considered here. This would ensure this additional redundancy for all types of initiators including a larger range of primary system leaks.

Unprotected Accidents

As described previously, the design of the redundant and diverse shutdown systems appears to be very good and the system analysis did not identify any weaknesses. Future designs could probably incorporate this design directly.

Although not incorporated directly in this study, future designs may benefit from consideration of recent research which suggests that by optimizing the reactivity feedback parameters, a self-terminating design

similar to the CRBRP core design could be developed. This would significantly change the physical phenomena associated with the unprotected accidents.

14.4 CONCLUSIONS

The achievement of the program objectives has been documented in the preceeding sections of this report. System models have been completed for over twenty-five plant systems and subsystems. The logic models were integrated and quantified to obtain realistic descriptions and best estimate frequencies for a full range of accident sequences that could affect the plant. The results of the sequence analysis have been used to generate insights concerning important plant features relative to safety--including identification of the elements which contribute positively to safety. Finally, the models were prepared in a top-down functional approach that would allow their use in consideration of other breeder or liquid-metal reactor designs.

The core and containment analyses were performed for the various types of accidents identified in the core-damage sequence analysis. Available research, both deterministic and probabilistic, was integrated into an overall structure that was quantified by systematically translating qualitative obervations into probabilistic results. The end products include sequence descriptions and probabilities for the physical processes associated with the different types of core-damage sequences. These results were used to identify insights about plant features or the understanding of physical phenomena.

The integration of all the models, phenomenological and system, was used to provide additional probabilistic focus on the features or phenomena that could be important to risk. Although the risk equation was not completed (i.e., health effects were not estimated) due to the reduced scope of this program, a preliminary grouping of accidents by potential timing and magnitude of radionuclide release was performed. While not conclusive, this grouping serves to identify the relative importance of accident sequences.

The results of this study indicate that the frequency of core-damage is 3.6×10^{-5}/yr. The frequency of core damage for all causes except seismic is about an order of magnitude lower than has been assessed in most recent LWR PRAs. The seismic core-damage frequency is similar to that assessed for most LWRs. An in-depth search was made for additional accident sequences after the results of the sequence analysis were completed.

No additional sequences were found but, more importantly, a review of the results did highlight the reasons for the relatively low core-damage frequency:

- The thermal capacity of the primary sodium causes all accidents (except those involving failure to shutdown the nuclear reactor) to evolve over relatively long time frames (10 hours or more). The removal of any heat at the beginning of the accident can extend this time out to a day or more.

  The potential for recovery of failed equipment or use of alternate modes of cooling is substantial for these long time frames.

- The protected air cooled condensers (PACCs) offer a nearly passive method of long-term cooling--a unique decay heat removal concept.

- The direct heat removal service (DHRS) offers a diverse cooling method for accidents in which the three heat transport loops are

not available for heat removal. The DHRS function is limited by the sodium leak sequences--a design more responsive to a full range of sodium leaks would make DHRS even more reliable as a redundant and diverse cooling mode. Even though it is a Class 1E system, the DHRS function could fail because of failure of cover gas supply to PHTS pump seals (a non-1E system), which would result in a loss of forced flow in the primary.

- The dominant sequences were initiated by earthquakes. The available methodology and the somewhat limited scope of the seismic analysis contributes additional uncertainty to these results, but it does appear that the lower limit of the core-damage frequency is limited by earthquake initiators. The plant is well designed to the design basis earthquake.

- The redundant and diverse shutdown systems make unprotected accidents very unlikely. The design was thoroughly reviewed but no substantial common-cause failures were postulated. The mechanical effect associated with seismic events is the most important element and is highly uncertain.

These major findings are discussed in more detail in other parts of this report.

The phenomenological analyses had three major conclusions:

- The probability of reactor vessel head damage due to energetics is very low, and the probability of containment failure due to energetics is negligible.

- It is suggested that further analysis be conducted regarding containment failure due to thermal or mechanical effects associated with the heatup of the polar crane by the hydrogen flame.

- The phenomena associated with the protected accidents (successful reactor trip) are not well defined. Past research has focused on the potentially more severe unprotected accidents. The protected accidents are probabilistically more important.

Thus it can be seen that the probabilistic approach to safety analysis provides an additional perspective on safety issues. The integrated plant models for the CRBRP have generated insights and conclusions that are not only specific to the CRBRP design but that can be translated into design considerations for other breeder or liquid-metal reactor designs.

Appendix A
DETAILED SYSTEMS ANALYSIS

TABLE OF CONTENTS

NOTE: Refer to individual sections for detailed tables of contents.

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Appendix A, Section 1

INTRODUCTION

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INTRODUCTION

A1.1 INTRODUCTION

Each section of Appendix A presents the reliability analysis for a system for the CRBRP PRA. For each system, a fault tree and corresponding reliability data are presented, along with detailed supporting documentation. The systems for which analyses were performed are identified in Table A1-1.

Section 4 of the main report provides a description of the objectives of the systems analysis and a summary of the technical methods used. This introduction to Appendix A describes the conventions used in analysis in constructing the fault trees and in naming the gates and events.

A1.2 ORGANIZATION OF APPENDIX A

The systems analysis report for most systems is divided into four subsections (a more detailed outline is provided in Table A1-2):

AX.1 System Description
AX.2 System Model
AX.3 Failure Data
AX.4 System Level Results and Insights

(Letter X denotes section number for each system as listed in Table A1-1, e.g. A2.1 for System Description of Auxiliary Feedwater System).

The system description section (AX.1) contains information about the function, design and operation of the system. It includes simplified piping and instrumentation drawings (P&IDs) as well as the interface requirements for the system.

Table A1-1

Appendix A Outline
Detailed Systems Analysis

A1	Introduction
A2	Auxiliary Feedwater System
A3	Direct Heat Removal Service
A4	Steam Generator System
A5	Protected Air Cooled Condensers and SGAHRS Vents
A6	Normal Plant Service Water System
A7	Emergency Plant Service Water System
A8	Heat Rejection System
A9	Normal Chilled Water System
A10	Emergency Chilled Water System
A11	Secondary Services Closed Cooling Water System
A12	Electric Power System
A13	Recirculating Gas Cooling System
A14	Primary and Intermediate Heat Transport Systems
A15	Main Feedwater and Condensate Systems
A16	Steam Dump System
A17	Heating, Ventilating and Air Conditioning Systems
A18	Plant Protection System
A19	Annulus Filtration System
A20	Thermal Margin Beyond Design Base - Annulus Cooling System

Table A1-1 (Continued)

A21	Thermal Margin Beyond Design Basis - Containment Cleanup System
A22	Containment Isolation System
A23	Compressed Gas
A24	Cover Gas System
A25	Steam Generator Auxiliary Heat Removal System Actuation Logic
A26	Reactor Cavity Vent

Table A1-2

OUTLINE OF SYSTEMS ANALYSIS SECTIONS

AX.1 SYSTEM DESCRIPTION

AX.1.1 Function

AX.1.2 Design - including simplified P&ID

AX.1.3 Interfaces with other systems

--electric power
--instrumentation
--cooling water
--HVAC, etc.

AX.1.4 Operation

--normal
--emergency

AX.1.5 Test and Technical Specifications

AX.1.6 Maintenance Requirements

AX.2 SYSTEM MODEL

AX.2.1 Modeling Assumptions

AX.2.2 Detailed Fault Tree Model

--overview
--success criteria
--top events
--transfers to other systems
--discussion on fault trees

AX.2.3 Modularized Fault Tree Model

--fault tree simplifications

NOTE: X denotes the report section number as listed in Table A1-1, e.g., A24.1 for System Description of Cover Gas System.

Table A1-2 (Continued)

AX.3 FAILURE DATA

AX.3.1 Basic Event Data

--overview and assumptions
--hardware failure data
--maintenance data
--human error data

AX.3.2 Module Data

AX.4 SYSTEM LEVEL RESULTS AND INSIGHTS

AX.5 REFERENCES

NOTE

Fault trees and data tables after Section AX.5.

Section AX.2 describes the development of the fault-tree model for the system. It includes modeling assumptions specific to the system; detailed, as well as modularized fault trees with discussions on success criteria. For a discussion on modularization see Sections 4.2.2 and A1.3. In general, Sections AX.1.1 and AX.1.2 (Function and Design, respectively) discuss the purpose served by the system including the basic design elements and components. The simplified P&IDs show the system configuration modeled. They show the skeleton layout of the systems for more clarity, ignoring equipment not important to the system reliability and do not include (generally) any information pertaining to the actuation logic. Sub-section AX.1.4 discusses system states during normal and off-normal plant operation. Special emphasis is given to train realignment and components actuated during transient conditions.

The section on data (AX.3) includes definitions of modules, their failure data, and the data of other basic events in the form of data tables. The data tables are arranged so as to show the module, its failure data and definition on one line. Components of the module are shown below the module name indented to the right with their failure data. Data on latent and dynamic human errors, as well as other basic events are generally shown at the end of the data table. Since dynamic operator actions can be dependent on the specific accident sequence in which they arise, the data tables in this appendix only show screening values for dynamic human errors. Actual data used for these errors are shown in the section on Core Damage Results (Section 10) and Human

Reliability Analysis (Section 6). The mission time (after reactor trip) used for most systems was 24 hours. The data base used here is discussed in Section 5.

Any findings important to the reliability and performance of individual systems are discussed in Section AX.4 - System Level Results and Insights.

The section on modeling assumptions (Section AX.2.1) is perhaps the most important section relative to the authenticity of the model. Every effort has been made to list the key assumptions that form the bases for the fault-tree models. For example, for the case of a system with two parallel trains, one of which is normally in use, the models assume that the "A" train is in operation and the "B" train is in standby. Such assumptions are made to simplify the fault tree model and to make the analysis more coherent. Any assumptions made regarding system configuration are also listed in Section AX.2.1.

Discussions about system success criteria, the fault tree top events, and transfers to other systems, and comments are included in the section describing the detailed fault tree model (Section AX.2.2). An exception to this is the Electric Power System (Section A12), in which the detailed fault trees are not included in the discussion. Section AX.2.3 includes discussions on the simplified fault trees as the result of modularization.

A1.3 MODULARIZATION

Modularization is the process of creating independent subtrees by combining non-recurring basic events into single events (called modules).

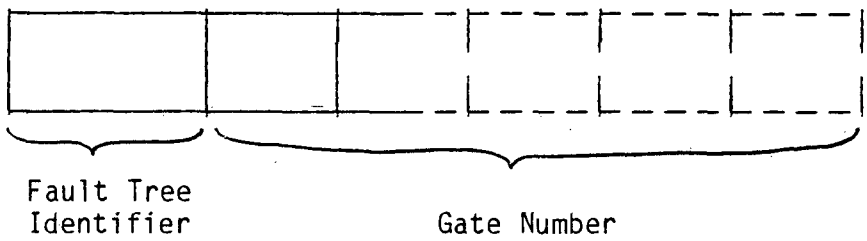
This reduces the size of the system fault trees, therefore reducing the effort needed in quantifying the core-damage cut sets. The modularized fault trees are used for input into SETS (Set Equation Transformation System)[1] to determine the core-damage cut sets. The fault tree naming scheme (see Section A1.4) for the modularized fault trees is similar to the detailed fault trees; at times the gate names have been modified to differentiate between the two fault trees. The respective sections give details on such changes.

A1.4 FAULT TREE NAMING SCHEME

Since the idea behind modeling the systems was to understand plant reliability under accident conditions, all the system level fault trees were coded in a form acceptable by various reliability analysis codes. SETS accept up to 16 alphanumeric characters for the names of gates and basic events, but since other codes used in reliability analysis allow a maximum of 10 characters only, it was decided to use a 10-character naming scheme in the event that another code was used in the future.

In the naming scheme used, all the gate names start with an alphabetic character (to identify the system) followed either by numerals, or another alphabetic character and more numerals. Figure A1-1 describes the gate naming scheme with examples, and Table A1-3 lists the designators used to represent various systems. The gate naming scheme used for the Transient Event Tree Top Logic is described in Section 3.4.1 of the main report (Table 3.4-1).

GATE NAMING SCHEME



Example: i) E100 For Gate #100 in the Electric Power Tree model

ii) S110B Gate #110B in the fault tree model for SGAHRS

iii) SM110B Gate #110B in the modularized SGAHRS fault tree

NOTES: i) The gate names can be up to 10 alphanumeric characters long

ii) Leading zeroes avoided; e.g., instead of D001, D1 was used.

iii) In example ii above a gate name of S110B is shown. It was often necessary to use alphabetic characters to extend an already established gate to create a new gate or modify one in later fault tree revisions.

iv) Some analysts used the letter "M" followed by the system identifer, to differentiate between the detailed and modularized fault-trees. In example iii above gate SM110B implies modularized version of gate S110B.

Figure A1-1. Gate Naming Scheme.

The naming scheme for basic events is more complex than the gate naming scheme, and is described in Figure A1-2. The basic event name starts with the fault tree identifier (Table A1-3) followed by the SDD number of the system. The component type (e.g. manual valve, pump, etc.) in

Table A1-3

SYSTEM CODES FOR FAULT TREES
(Fault Tree Identifiers)

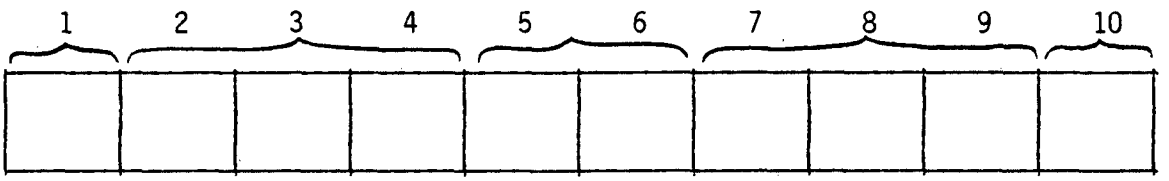
A	:	Annulus Cooling System
B	:	Steam Dump System
C	:	Annulus Filtration System
D	:	Reactor Cavity Vent
E	:	Electric Power System
F	:	Main Feedwater and Condensate Systems
G	:	Cover Gas System
H	:	Heating, Ventilating, and Air Conditioning System
I	:	Intermediate Heat Transport System
J	:	Instrument Air/Compressed Gas System
K	:	Containment Isolation
L	:	Ex-Vessel Storage/Processing System
M	:	Recirculating Gas Cooling System
N	:	Heat Rejection System
P	:	Primary Heat Transport System
Q	:	Steam Generator System (includes SWR Pressure Reduction)
R	:	Direct Heat Removal Serivce (includes ALMS)
S	:	SGAHRS (includes AFW, PACC)

Table A1-3 (Continued)

T	:	Transient Event Tree Top Logic
U	:	Undeveloped System Interfaces
V	:	Containment Cleanup System
W	:	Plant/Emergency Service Water System
X	:	Normal/Emergency Chilled Water System
Z	:	Plant Protection System

abbreviated form, listed in Tables A1-4a and b, is next, followed by the component identifier/number as shown on the P&IDs. Identification of the failure mode is done by using alphabetic characters again, as described in Table A1-5.

BASIC EVENT NAMING SCHEME



1 Fault Tree Identifier (Table A1-3) (e.g., P for PHTS).

2-4 CRBRP System Designator/SDD number (e.g., 51A, 23).

5-6 Component Type/Name (Tables A1-4a/b) (e.g., MV for Motor-Operated Valve).

7-9 Component Identifier/Number (e.g., valve number 25A, pump number 1B).

10 Failure Mode (Table A1-5) (e.g., S for Fails to Start; H for Dynamic Operator/Human Action)

NOTES: 1. Column 1 is the 1st character of basic event name.

2. In case of short system designators (e.g., 23), component type/name starts in column 4. In other words leading blanks and zeros to be avoided. The basic event name can have less than 10 characters, but not more.

Figure A1-2. Basic Event Naming Scheme.

NOTES: (Continued)

3. There were some instances when the component identifier (columns 7-9) had four letters. In such cases, the system designator (SDD number, columns 2-4) was cut short. An example can be "Temperature Element (thermocouple) TE72AA of system 75A (Emergency Plant Service Water System) fails high," can be represented as:

W	75	TC	72AA	A
(i)	(ii)	(iii)	(iv)	(v)

i. Fault Tree Identifier for service water system (Table A1-3).

ii. SDD # for service water system (actual number 75A, but reduced here to 75 to maintain the 10 character naming scheme).

iii. Component type: TC for thermocouple/temperature element (Table A1-4b).

iv. Component number: 72AA (from P&ID).

v. Failure mode: "A" for fails high; i.e., fails with a high output signal (Table A1-5).

Figure A1-2 (Continued). Basic Event Naming Scheme.

Table A1-4a

COMPONENT CODE/ABBREVIATIONS FOR MECHANICAL COMPONENTS

AC	Accumulator
AV	Air-operated valve (Pneumatic)
AH	Air handling unit
CH	Chiller
CM	Compressor
CN	Condenser
CR	Cooler
CV	Check Valve
DA	Damper, air-operated
DD	Damper, check valve type
DG	Diesel generator
DM	Damper, motor-operated
DT	Duct
DV	Diaphragm valve
EJ	Expansion joint
ER	Evaporator
EV	Electro hydraulic valve
FL	Filter/strainer
FN	Fan
FR	Flow restrictor

Table A1-4a (Continued)

GB	Gas bottle
HX	Heat exchanger
MR	Motor cooler, pump
MT	Mixing tee
MV	Motor-operated valve
OR	Orifice
PC	Pipe cap
PE	Pump, electromagnetic
PI	Pipe, venturi
PM	Pump, motor driven
PT	Pump, turbine driven
PV	Pressure vessel
RD	Rupture disc
RV	Relief valve (or safety)
SD	Steam drum
SH	Super heater
SV	Solenoid valve
TK	Tank
TU	Tubing
XV	Manual valve
ZV	Regulating valve

Table A1-4b

COMPONENT CODE/ABBREVIATIONS FOR ELECTRICAL COMPONENTS

AM	Amplifier
AN	Annunciator
AT	Automatic transfer switch
BC	Battery charger
BS	Bus
BY	Battery
CA	Cable
CB	Circuit breaker
CC	Capacitor
CL	Control logic
CO	Connector
CT	Current transformer
DC	DC power supply
DE	Diode/rectifier
DP	Distribution panel
FE	Flow element
FT	Flow transmitter
FU	Fuse
GC	Governor valve controller
GS	Ground switch

Table A1-4b (Continued)

HR	Heater
HT	Heat tracing
IL	Interlock mechanism
IN	Instrumentation
IV	Inverter
KS	Lock-out switch
LA	Lighting arrestor
LR	Level sensor
LS	Limit switch
LT	Light
ME	Meter
MO	Motor
MS	Motor starter
ND	Neutron detector
OT	Potential (or control) transformer
PS	Pressure sensor
RA	Radiation monitor
RE	Relay
RS	Resistor
SC	Speed controller
SW	Manual switch

Table A1-4b (Continued)

TC	Thermocouple, temperature sensor/element
TI	Timer
TR	Transformer, power
TS	Torque switch
TZ	Position transmitter
VR	Voltage regulator
VT	Voltage transformer
WR	Wire

Table A1-5

FAILURE MODE CODES

A	Fails high (electrical output - electrical components only)
B	Fails low (electrical output - electrical components only)
C	Fails to close
D	Fails as is (electrical output - electrical components only)
E	Fails to energize (electrical equipment)
F	Fails to perform intended operation (Note 1)
H	Dynamic operator action (Note 2)
I	Inadvertent self actuation (electrical components only) (Note 3)
L	Latent human error (Note 4)
M	Unavailable due to maintenance
N	No output, open line
O	Fails to open
P	Plugged
R	Fails to run
S	Fails to start
T	Transfers to opposite position (open/closed)
U	Unavailable due to testing
X	Electrical shorts
Y	Leakage/rupture
Z	Calibration shifts to unacceptable range

NOTES:

1. Used only when a more specific failure mode did not apply.

2. Used for any operator action during an accident, that is after the occurrence of a transient. For example, recoveries.

3. Used for electrical equipment which can actuate by itself. For example, a relay can actuate without an input signal, generating a false output condition. Not used for valves transferring (used T for this) to an unsafe position.

4. Any operator/maintenance-crew errors prior to the accident initiator.

The naming scheme used for module events is shown in Figure A1-3. Basically it uses the letter "M" after the fault tree identifier (Table A1-3) with some kind of number or name for the module. The modules are treated much like basic events in the analysis.

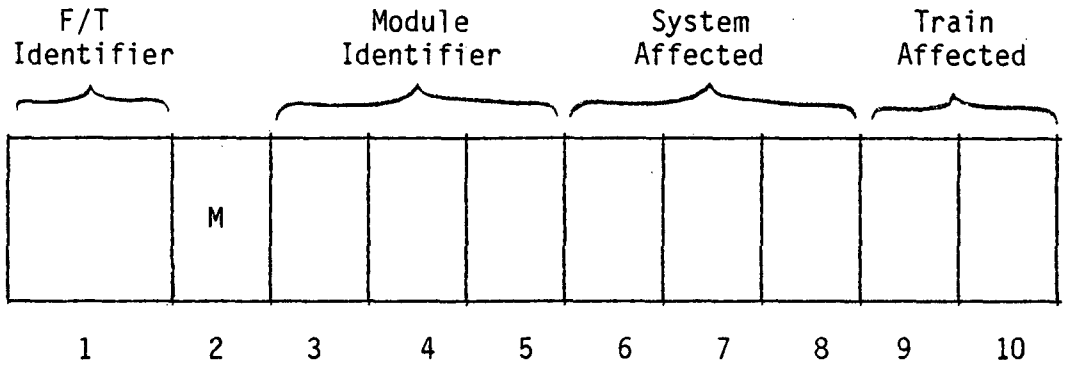
A1.5 FAULT TREE GATE SYMBOLS

The basic elements of a fault tree were used to represent modeling logic for all systems; References 1, 2, and 3 list all such symbols and give description of each. Figure A1-4 includes the fault tree symbols generally used throughout this analysis with short descriptions of each.

The most common deviation from the symbols shown in Figure A1-4 was using the intermediate event symbol, with or without a small circle underneath, to represent a basic event or module. This was done for fault tree drawing purposes, since it was easier to write event and gate descriptions in a rectangle than in a circle. The easiest way of differentiating between an intermediate event and a basic event then is that an intermediate event will always have either a "gate" or a "transfer in" symbol (see Figure A1-4) immediately following it; whereas, a basic event or module will have its name written under the rectangle. Examples of this can be seen throughout Sections AX.2.2 and AX.2.3.

Sometimes use was also made of "x out-of y" type logic gates. For example, a 2 out-of 3 gate (written as 2/3) implies that the event will occur if any two of the three allowable inputs are present. The symbols used for such a gate were either an OR or an AND gate with x/y (2/3 in

MODULE NAMING SCHEME



Column(s)	Contents
1	fault tree identifier (Table A1-3)
2	"M" to represent modularized event
3-5	module number; e.g., 123, 12A, 1 etc.
6-8	system affected, e.g., 73A, 23
9-10	train/group/division affected; e.g., 1A, 1B, etc.

NOTES:

1. The module name can have less than 10 characters.
2. Letter "M" in the 2nd column represents that the basic event is a modularized event.
3. In general, the module names used were more analyst dependent, and did not necessarily contain all the information identified above.

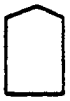
Figure A1-3. Module Naming Scheme.

PRIMARY EVENT SYMBOLS

BASIC EVENT – A basic initiating fault requiring no further development

CONDITIONING EVENT – Specific conditions or restrictions that apply to any logic gate (used primarily with PRIORITY AND and INHIBIT gates)

UNDEVELOPED EVENT – An event which is not further developed either because it is of insufficient consequence or because information is unavailable



EXTERNAL EVENT – An event which is normally expected to occur

INTERMEDIATE EVENT SYMBOLS

INTERMEDIATE EVENT – A fault event that occurs because of one or more antecedent causes acting through logic gates

GATE SYMBOLS

AND – Output fault occurs if all of the input faults occur

OR – Output fault occurs if at least one of the input faults occurs

INHIBIT – Output fault occurs if the (single) input fault occurs in the presence of an enabling condition (the enabling condition is represented by a CONDITIONING EVENT drawn to the right of the gate)

TRANSFER SYMBOLS

TRANSFER IN – Indicates that the tree is developed further at the occurrence of the corresponding TRANSFER OUT (e.g., on another page)

TRANSFER OUT – Indicates that this portion of the tree must be attached at the corresponding TRANSFER IN

Figure A1-4: Fault Tree Symbols (Reference 3).

the case of the above example) written inside the gate symbol. Examples of this can again be found in Sections AX.2.2 and AX.2.3.

A1.6 REFERENCES

1. Worrell, R. B. and Stack, D. W., 1980. A SETS User's Manual for the Fault Tree Analyst, NUREG/CR-0465, SAND77-2051, Albuquerque, New Mexico.

2. McCormick, N. J., 1981. Reliability and Risk Analysis: Methods and Nuclear Power Applications, Academic Press.

3. U.S. Nuclear Regulatory Commission, 1981. Fault Tree Handbook, NUREG-0492, Washington, D.C.

Appendix A, Section 2

AUXILIARY FEEDWATER SYSTEM

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Appendix A, Section 2

AUXILIARY FEEDWATER SYSTEM

A2.1 SYSTEM DESCRIPTION

A2.1.1 Function

The function of the auxiliary feedwater (AFW) system is to provide cooling water to the steam generator system (SGS) for decay heat removal when main feedwater is unavailable, The short term mission of the AFW system is to replace water lost in the SGS due to steam venting. In the long term, the AFW system replaces water lost due to leakage through the protected air-cooled condensers (PACCs) and the SGS.

A2.1.2 System Design

Auxiliary feedwater is a subsystem of the steam generator auxiliary heat removal system (SGAHR). The SGAHRS removes heat from the SGS by steam venting or PACC operation. The AFW system provides makeup water to replace the expended water. The water is supplied to the steam drums by a redundant AFW pump and piping layout (See Figure A2-1). The pumps take suction from 68,160 usable gallons of water in the protected water storage tank (PWST).

There are three pumps: two 716-gpm electric-motor-driven (MD) AFW pumps (52AFP002A,B) in parallel and one 1432-gpm steam turbine-driven (TD) AFW pump (52AFP001). The electric motors are powered by normal or emergency plant ac power; the turbine is driven by steam from the steam drums. Either the combined motor-driven pumps or the turbine-driven pump are capable of supplying the required flow rate to all three loops,

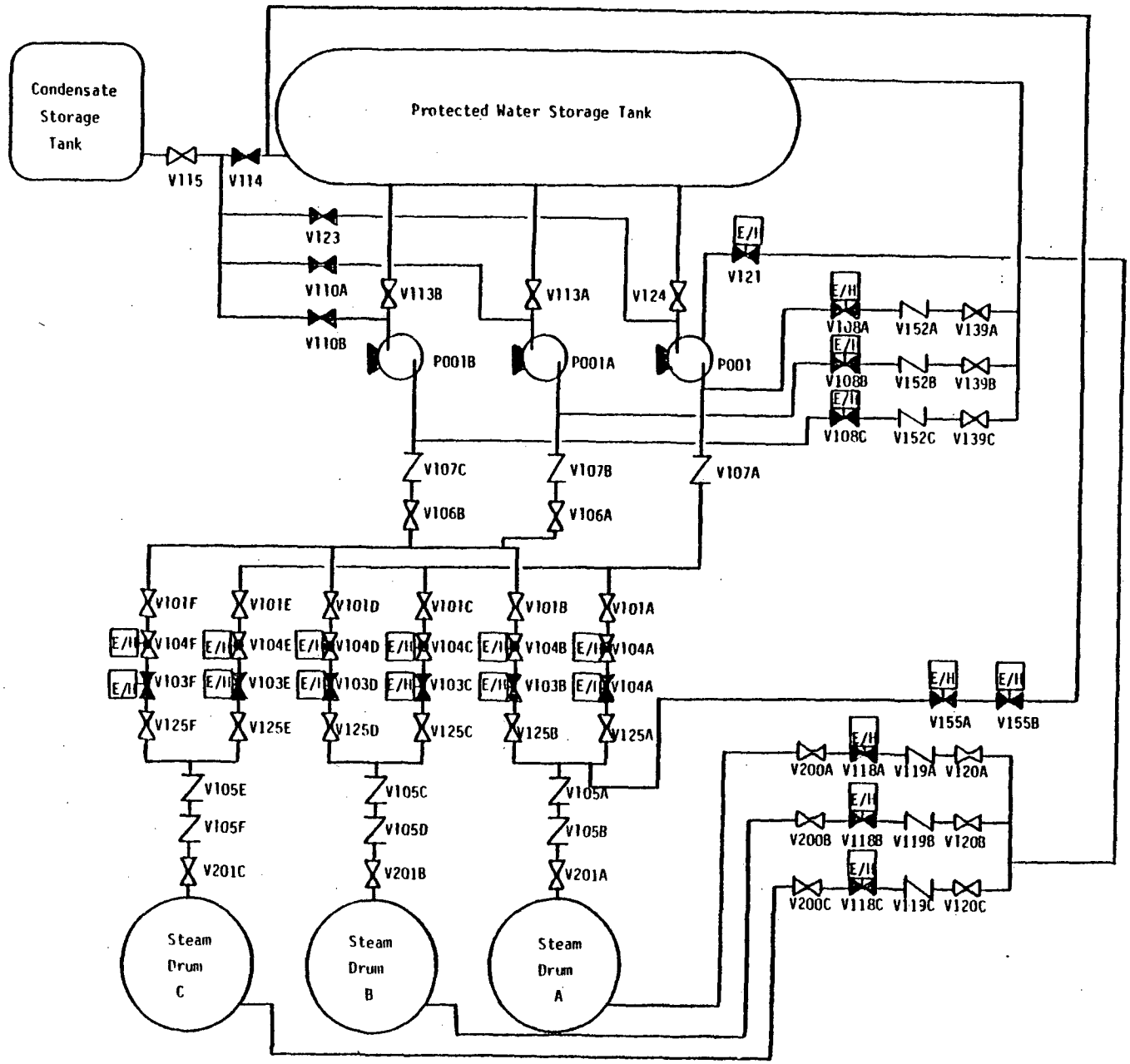

Figure A2-1. Auxiliary Feedwater Simplified P&ID.

including the margin for a flow-limited condition in the event of a pipe break.* The MDAFW pumps are headered together and supply flow-through control valves to each of the three drums. Similarly, the TDAFW pump supplies flow through control valves to each of the three drums. The TDAFW and MDAFW pump supply lines to each drum are joined downstream of the control valve stations for the final run into each drum.

The AFW system is designed to provide enough water to remove reactor decay heat and sensible heat through SGS venting until the PACCs can remove the plant heat load in a closed loop manner. The AFW system is also used to provide the small quantities of makeup required by PACC operational leakage.[5,7]

A2.1.3 Interfaces with Other Systems

The following systems interface with the AFW system:[7]

- Building electrical power system,
- Nuclear island heating, ventilating, and air conditioning (HVAC) system,
- SGAHRS,
- SGS, and
- Reactor heat transport instrumentation and control system.

A2.1.3.1 Building electrical power system. AFW pump motors and valves receive electrical power from the building electrical power system sources listed below.[2] The bus and circuit breaker numbers are not available for valves.

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*The actual success criteria for the fault tree is different as explained in Section A2.2.

Equipment No.	Power	Bus No.	Breaker	Division
52AFK001A	4160V-1E	12NIE003A	YG-3A	I
52AFK002B	4160V-1E	12NIE003B	YG-3B	II
52AFV103A	480V-1E			I
52AFV103B	480V-1E			II
52AFV103C	480V-1E			II
52AFV103D	480V-1E			III
52AFV103E	480V-1E			I
52AFV103F	480V-1E			III
52AFV104A	125V Vital DC			II
52AFV104B	125V Vital DC			I
52AFV104C	125V Vital DC			III
52AFV104D	125V Vital DC			II
52AFV104E	125V Vital DC			III
52AFV104F	125V Vital DC			I
52AFV108A	480V-1E			III
52AFV108B	480V-1E			I
52AFV108C	480V-1E			II
52AFV118A	480V-1E			I
52AFV118B	480V-1E			II
52AFV118C	480V-1E			III
52AFV121	125V Vital DC			III
52AFV155A	480V-1E			I
52AFV155B	480V-1E			II

A2.1.3.2 Nuclear island HVAC. The following table specifies the cells and equipment of the AFW system that require HVAC.[1] The temperatures resulting from a loss of HVAC are shown after 2-hour and 30-hour periods. Information for cells 204, 204A, and 204B is not available.

Cell	Equipment No.	Loss of HVAC After 2 Hours	Temperature After 30 Hours
202	52AFV101A 52AFV101B 52AFV103A 52AFV103B 52AFV104A 52AFV104B 52AFV108A 52AFV155A 52AFV155B	134°F	162°F

Cell	Equipment No.	Loss of HVAC After 2 Hours	Temperature After 30 Hours
202A	52AFN001 52AFP001 52AFV121 52AFV139A	144°F	172°F
202B	52AFV125A 52AFV125B	Assume to be similar to Cell 202	
204	52AFV106A 52AFV106B 52AFV108A 52AFV108B 52AFV139B 52AFV139C 52AFV152B 52AFV152C	N/A	N/A
204A	52AFK001A 52AFP002A 52AFV107B	N/A	N/A
204B	52AFK001B 52AFP002B 52AFV107C	N/A	N/A
206	52AFV101C 52AFV101D 52AFV101E 52AFV101F 52AFV103C 52AFV103D 52AFV103E 52AFV103F 52AFV104C 52AFV104D 52AFV104E 52AFV104F 52AFV125C 52AFV125D 52AFV125E 52AFV125F	106°F	113°F

Cell	Equipment No.	Loss of HVAC After 2 Hours	Temperature After 30 Hours
215	52AFT001 52AFV110A 52AFV110B 52AFV113A 52AFV113B 52AFV124	102°F	119°F
221	52AFV119A 52AFV120A	109°F	110°F
222	52AFV119B 52AFV119C 52AFV120B 52AFV120C	109°F	110°F
241	52AFV105A 52AFV105B 52AFV118A 53SGV200A 53SGV201A	125°F	140°F
242	52AFV105C 52AFV105D 52AFV118B 53SGV200B 53SGV201B	125°F	140°F
243	52AFV105E 52AFV105F 52AFV118C 53SGV200C 53SGV201C	125°F	140°F

A2.1.3.3 Steam generator auxiliary heat removal system. As described in Section A2.1.2, AFW is a subsystem of the SGAHRS.[7]

A2.1.3.4 Steam generator system. The AFW system provides makeup water to the steam drums which are major components of the SGS, when the SGAHRS is activated. The steam drums, in return, provide steam to drive the turbine-driven pump.[8]

A2.1.3.5 Reactor heat transport instrumentation and control system. The reactor heat transport instrumentation and control system (RHTICS) receives signals from its instrumentation that is located throughout the SGS and the AFW system. The RHTICS measures steam drum level, pressure, and flow rates. The signals are processed by the plant protection system and returned to RHTICS for action. RHTICS actuates the SGAHRS, and controls AFW components so that sufficient makeup water is delivered to the steam drums as needed during SGAHRS operation.[6]

A2.1.4 Operation

Auxiliary feedwater is a standby system that is used when the SGAHRS is initiated. It can also be manually operated for testing or other purposes. The SGAHRS is initiated by a high steam-to-feedwater flow ratio or low steam drum water level on any loop. It protects against those events resulting in a loss of steam drum water inventory, or a loss of the normal flow path through the main condenser. Following a reactor shutdown in which the SGAHRS is required, the superheater outlet isolation valves on each loop are closed and the SGS pressure rises, causing the relief valves to open. Also, upon SGAHRS initiation, the SGAHRS vent-control valves at both the superheater exit and the steam drum on each loop are allowed to initiate their pressure-control function, the AFW pumps are started, and the AFW isolation valves are opened.

Upon receipt of the SGAHRS initiation signal, the electrical circuit breakers to the two motor-driven pumps are closed, resulting in both

pumps coming on-line and operating at constant speed. In addition, the drive-turbine steam supply isolation valves (52AFV118A,B,C) are opened automatically upon receipt of the SGAHRS initiation signal. These valves close automatically on a per loop basis upon indication of a sodium/water reaction or steam drum pressure less than 200 psig. A pressure-regulating valve reduces the steam supply pressure to the 1000 psig required at the turbine inlet. The turbine-drive mechanism is equipped with a governor to provide speed regulation.

Manual controls are also provided for starting or stopping each AFW pump at the operator's direction. However, following SGAHRS initiation, special procedures have to be followed to turn off any of the pumps. The operating instructions restrict the conditions under which any of the pumps may be turned off during steam venting. Following steam venting, the pumps are shut down and the initiation signal reset to allow automatic restart when required for makeup to the steam drums.

The AFW isolation valves (52AFV103A through F) open automatically on actuation of SGAHRS initiation signal and close automatically upon indication of sodium/water reaction, high steam-drum level, steam drum pressure less than 200 psig, or an AFW flow greater than 150% rated flow for 5 s. The control channels which isolate the auxiliary feedwater on both the motor-driven and turbine-driven pump legs are completely separate all the way through the sensors so that no single failure can isolate both legs. Remote operator action in the control room using special procedures is required for isolation of pipe breaks. These valves are manually closed at the end of steam venting following a trip

or automatically shut following venting by a high drum level trip. They reopen automatically when the drum level falls to the low drum level trip point.

The AFW flow control valves (52AFV104A through F) control AFW flow as a function of a steam-drum level error. Each valve is positioned by a controller that compares measured steam drum level against a setpoint value and generates a position demand signal to the flow control valve. Manual control of the AFW flow control valve is provided by incorporating a manual/automatic transfer station between the flow controller and the valve positioner. A flow limiter restricts maximum flow through each control valve to 105±5% rated AFW flow. To prevent addition of large amounts of unneeded AFW, the set point for each of the legs supplying a drum is different. The leg from the motor-driven pumps is the prime supply and has a set point at 4 in below the normal water level (NWL). This leg, when operational, will maintain the drum at 4 in below the NWL following the initial transient and level recovery. The second leg from the turbine-driven pump is the backup. It has a set point 18 in below NWL. This backup leg is not required unless additional failures occur following an initiating event. For instance, if one diesel generator providing power to a motor-driven pump should fail, the remaining motor-driven pump could require backup from the turbine-driven pump. Under these conditions, the control valve in the turbine-driven pump leg would open when the drum level falls to 18 in below NWL. That level is then maintained until the plant heat load falls to a point at which the motor-driven pump could raise the level

back to its set point of 4 in below NWL. When the level rises above its set point of 18 in below NWL, the control valve in the turbine-driven pump leg closes. The level set points for both the motor-driven pumps and turbine-driven pump have a dead band of ±2 in.

Water drawn from the PWST by the AFW pump is pumped through parallel loops to each steam drum. The PWST contains 68,160 usable gallons of water, enough for 30 days of design basis AFW operation. In addition, the pumps can draw water from the condensate storage tank (CST), which serves as an alternate water supply. The lines from the PWST and the CST are teed together at each pump inlet. The operator must manually switch from one source to the other. The alternate water supply (the CST) is also used to fill the PWST.

To prevent overheating and vibration of the AFW pumps at reduced flow, each pump is provided with a recirculation line from the pump discharge back to the PWST. The valve in the recirculation line is opened by a pressure switch that is activated by increased AFW pump discharge pressure when the pump is throttled. The valves in the motor-driven pump recirculation lines are opened when the pump discharge pressure rises to 1958 psig and closed when the pressure drops below 1855 psig. The valve in the turbine-driven pump recirculation line opens when the pump discharge pressure rises to 1893 psig and closes when the pressure drops below 1820 psig. These valves are normally open when the system is on standby. When SGAHRS initiation occurs, and prior to the pumps reaching rated speed, the valves will begin to close. When the pumps reach rated speed, the valves will take the appropriate position as a function of pump discharge pressure as noted above.

Downstream from the pump outlets are check valves to prevent reverse flow. The two MDAFW pumps in addition have manual isolation valves downstream from the check valves. The MDAFW pump discharge lines are then headered together and supply flow through valve stations to each steam drum.

Similarly, the TDAFW pump supplies flow through another set of valve stations to each steam drum. The valve stations are identical and contain, in the direction of flow, a manual isolation valve, a venturi flow meter, an electrically operated AFW flow-control valve, an electrically operated AFW isolation valve, and another manual isolation valve.

Downstream from the valve stations, the TDAFW pump supply line to each loop tees into the corresponding MDAFW pump supply line for that loop. Downstream from the tee, the AFW flows through two check valves and a manual isolation valve, and is then fed into the steam drum. The electrically operated AFW isolation valves downstream from the AFW flow control valves are the only normally closed valves in the AFW flow path during normal plant operation.

The AFW pump drive turbine (52AFN001) can be supplied with steam from any combination of steam drums. Each steam supply line contains, in the direction of flow, a locked-open manual isolation valve, a normally closed electrically operated isolation valve (52AFV118A,B,C), a check valve, and another locked-open manual isolation valve. The three lines then tee together into a line containing the normally closed pressure control valve. When the SGAHRS is initiated, the electrically operated

steam isolation valves are opened, and steam is supplied to the pressure-control valve. The drive turbine pressure-control valve (52AFV121) is opened at a slow rate to prevent a pressure spike at the turbine throttling valve. Steam is supplied from the steam drums at pressures up to 1550 psig. The drive turbine pressure control valve reduces this pressure to 1000 psig. At the exit of the turbine exhaust line, there is an exhaust restrictor that throttles the exhaust flow to provide the required turbine backpressure and limit the exhaust velocity.

The MDAFW pumps can be supplied with either offsite power or power from the emergency diesel generators. Each pump is connected to one of the three diesels with no switch-over capability provided.

The AFW pumps are sized to supply the required flow under the worst case design-basis conditions of:

- A pipe break on Loop #1 with flow limited by the control valves
- Steam drum venting on Loop #2
- Superheater venting on Loop #3

The TDAFW pump is a 1432 gpm pump; it can supply the flow required by the above criteria to each loop. The MDAFW pumps are sized (716 gpm) such that combined they can supply the required flow. The requirements for most accidents is any one of the three pumps.

During closed loop operation, the AFW system is employed to supply any makeup water that may be required. When not supplying water, the AFW system will be held in standby mode.[7]

A2.1.5 Test and Technical Specification Requirements

Several tests are performed at various time intervals to check the operability of the system. AFW flow from each motor-driven pump is checked quarterly (MOP 52-034) by testing the following flow paths: (1) pump recirculation flow (89,500 lb/hr or 25% rated flow), (2) AFW flow limiting point via the test line (277,000 lb/hr or 77% rated flow), and (3) full flow test using both the recirculation line and the test line (358,000 lb/hr or 100% rated flow). The turbine-driven pump is also tested (MOP 52-035) in a similar fashion, but at lower relative flow rates.

All AFW active, actuated valves are fail-safe valves. The Actuated Valve Fail-Safe Test (MOP 52-037) is performed annually to test the position of critical valves after loss of actuator and solenoid power. The valves tested include the AFW isolation valves (52AFV103 A through F), the AFW flow-control valves (52AFV104 A through F), and the drive turbine steam supply isolation valves (52AFV118 A through C).

Several AFW valves are tested biannually for leakage rate to meet ASME code requirements. The Valve Leak Rate Test (MOP 52-038) tests the AFW isolation valves and the drive turbine steam supply isolation valves.

All AFW check valves are equipped with mechanical exercisers for testing and movement of the valve disk. The Check Valve Test (MOP 52-039) is performed quarterly by exercising the following valves: the AFW check valves (52AFV105A through F), the AFW pump discharge check valves (52AFV107A,B,C), the drive turbine steam supply check valves

(52AFV119A,B,C), and the AFW pump recirculation check valves (52AFV152A,B,C).

The System Functional Test (MOP 52-040) is designed to demonstrate the operability of the entire AFW Subsystem. This quarterly-performed test is designed to take place under conditions close to design, and it will approximate the complete sequence that brings the AFW Subsystem into operation for a reactor shutdown following postulated accidents. The test is performed when the reactor is at or above 40% rated power.

The technical specifications that are applicable to the AFW are presented below.

A. The PWST must be on standby and available to the operational AFW pumps at all times during normal plant operation. The water level in the PWST shall be greater than 142 in (total tank volume less margin) at all times during power operation, or an orderly transition to hot shutdown conditions shall be initiated. This water level is equivalent to a water volume of 9111 ft^3 (total tank volume less margin).

B. The turbine-driven AFW pump (52AFP001) may be taken out of service for maintenance provided the two motor-driven AFW pumps (52AFP002A&B) are tested and shown to be operable immediately prior (within 4 hours) to removal of the turbine-driven AFW pump. This operability test (MOP 52-034 in Section 6)[7] shall be performed in turn on both the normal and emergency diesel generator power supplies.

C. One of the motor-driven AFW pumps may be taken out of service for maintenance provided the turbine-driven AFW pump is tested (MOP 52-035 in Section 6)[7] and shown to be operable, and the remaining motor-driven AFW pump is tested and shown to be operable. This testing shall be accomplished immediately prior (within 4 hours) to removal of the pump from service. The motor-driven pump shall be tested on both normal and emergency diesel generator power. The diesel generator corresponding to the available motor-driven pump must be operable.

D. An AFW pump shall not be taken out of service for maintenance for longer than 8 hours when core power is at or above 5% rated, or for longer than 7 days for other normal operations (OPDD-10, Section 7.8.1.1.2.11).[3]

E. Only one AFW control valve station (between manual valves 52AFV-101 and -125) can be removed from service for maintenance at a time for a period not to exceed 8 hours if core power is above 5% or 7 days for other normal operations (OPDD-10, Section 7.8.1.1.2.11).[3] The control and isolation valves in the parallel AFW control valve station of the affected loop shall be tested and shown operable within 4 hours of isolation of the aforementioned control valve station. Maintenance can be performed on only one control valve station in one loop at any given time.[7]

F. During normal reactor power operation (three loops) the SGAHRS short-term and two SGAHRS long-term heat removal subsystems shall be operable. The operable short-term heat removal subsystems must correspond to operable long-term subsystems.

G. During two loop reactor power operation the SGAHRS short-term and two SGAHRS long-term heat removal subsystems shall be operable. The operable long-term heat removal subsystem must correspond to two operating heat transport loops.

H. During two-loop and three-loop reactor power operation, the turbine-driven auxiliary feedwater pump may be taken out of service for maintenance, provided the emergency diesels are activated for the two electrically driven auxiliary feedwater pumps. The two feedwater pumps shall be tested for operability on both the normal and emergency power supplies.

I. During two-loop and three-loop reactor power operation, one of the electrically driven auxiliary feedwater pumps may be taken out of service for maintenance, provided the turbine-driven auxiliary feedwater pump has been tested and shown to be operable, and the other electrically-driven auxiliary feedwater pump has been tested and shown to be operable with normal and emergency power supplies. The emergency diesel generator corresponding to the standby pump must be operable.

J. During periods of hot shutdown or refueling shutdown, PHTS and IHTS temperatures shall not subsequently rise above the steady state design temperature limit (see Sections 16.3.2 and 16.3.3, respectively).[10] This is assured by the following considerations:

1. At all times during plant shutdown periods, a minimum of 2 decay heat removal loops individually capable of removing the full decay heat load shall be available. The two loops may consist of two main HTS loops and their associated heat dumps or 1 HTS loop and/or Direct Heat Removal Service (DHRS).

2. At least one auxiliary feedwater pump in the short-term heat removal subsystem must be operable. The short-term heat removal subsystem must be available to supply makeup water to the operable long-term heat removal subsystem as required.

If the specification of Item K is not met, the components subjected to high temperatures shall be examined and evaluated for suitability for return to power operation.

During plant operation at least two of the three SGAHRS long-term cooling circuits shall be operable to assure heat removal capability if one of these loops suffers a fault that would interfere with its heat removal function. The remaining loop(s) will be capable of removing the required heat load without exceeding plant safety limits.

The PACC and piping connecting them to the steam drum constitute the long-term heat removal circuit. Each circuit is uniquely associated with a steam drum. To assure redundant long-term cooling, the two circuits associated with the operating heat transport loops must be available during two loop operation. Then, if one of the heat transfer loops loses its ability to remove heat, the plant can be shut down and the decay heat removed by the remaining heat transfer loop with redundancy being provided by the DHRS.

The AFW system, in combination with the SGAHRS vent valves in the SGS, constitutes the short-term heat removal circuit. During plant operation, this system shall be available to supply feedwater to the three steam drums to assure adequate removal of decay heat and to avoid steam drum dryout if the normal feedwater supply is interrupted.

During plant operation on two loops, only two short-term cooling circuits are available since the third heat transport loop is not operating. To assure redundant short-term cooling, the two short-term circuits associated with the operating heat transport loops must be available.

During periods of reactor shutdown, three independent decay heat removal means must be available to provide redundancy in the unlikely event one of them becomes inoperable. Therefore, when the DHRS is not available, three main heat transport loops must be available for long-term decay heat removal. If DHRS is available, then only two main heat transport loops must be available. When decay heat is sufficiently low that the long-term heat removal circuit(s) and/or DHRS can maintain plant temperatures steady or decreasing, then it is not necessary to maintain active short-term circuit(s). This is done to assure the availability of cooling water to keep the long-term circuit(s) operating as long as required.

During plant operation the protected source of water shall be available to the AFW pumps to guard against a loss of normal feedwater accident. The protected water volume will be 9111 ft^3, which is sufficient for the short-term heat removal circuit venting and long-term heat removal.

A2.1.6 Maintenance Requirements

The AFW system is designed so that all maintenance and in-service inspection procedures can be performed during normal plant operation or during reactor outages. However, scheduled maintenance for AFW is

planned to be performed during reactor outages only, while unscheduled maintenance will be performed, when required, during both normal plant operation and reactor outages. Equipment such as motors, valve actuators, sensors, and other components are designed to permit inspections and adjustments.

Scheduled maintenance will be performed on all components in compliance with the requirements as defined by the component supplier, or as required by the component to maintain its functional capability.

All components of AFW are located in the cells of the auxiliary bay of the SGB. In-place maintenance is provided for the PWST, AFW pumps, drive motors and turbine, water/steam valves, and piping. The AFW pump seals and certain valve internals are repaired by removal and repair or replacement. Scheduled maintenance operations will be performed both during normal plant operation and during reactor outages.

In-service inspection will be performed on all components in compliance with the ASME Section XI, Division 1, requirements so as to insure that components continue to meet Code requirements.[5,7]

Precautions must be followed when removing any AFW equipment from service. The SGAHRS short-term subsystem (AFW) must be available (on standby) at all times during normal plant operation as defined by Section 7.8.1.1.2.11 of OPDD-10.[3] Short-term subsystem availability is defined as the capability of supplying auxiliary feedwater to all three steam drums and venting steam from the vent control valves of all three loops.

A2.2 SYSTEM MODEL

A2.2.1 Modeling Assumptions

The following assumptions were made when modeling the AFW System.

A. PWST is assumed to be full (68,160 usable gallons) at the time of the accident, and will provide a sufficient amount of water for the AFW mission.

B. Because the alternate water supply (from the condensate storage tank) must be accessed by opening a manual valve, the use of the alternate water supply will not be modeled in the AFW fault tree, but will be treated as a possible recovery action.

C. Loss of HVAC in a cell causing temperatures to exceed 135°F in that cell will fail all electrical components in that cell unless otherwise environmentally qualified.

D. Failure data sources do not distinguish between pumps and their drive motors or turbines. Therefore, a failure of any critical part of the pump or its drive mechanism will fail the entire unit in this fault tree.

E. The mission time for AFW is 4 hours.

F. Loss of HVAC to cells containing AFW motor and turbine driven pumps was assumed to not fail those pumps.

A2.2.2 Detailed Fault Tree Model

A2.2.2.1 Top event. There are three top events in the AFW model. They are titled:

A. "Insufficient AFW flow to steam drum A,"

B. "Insufficient AFW flow to steam drum B," and

C. "Insufficient AFW flow to steam drum C.

The logic development below these top events is similar except that the path to steam drum A has a test line that can fail open, and thus cause the top event. The fault tree models the AFW system from the AFW input to the steam drums, to the PWST, and to the steam supply from the steam drums.

The three fault trees have identical logic through the control headers, and share the two inputs from the common headers. These are titled:

A. "Loss of flow from the TDAFW pump" and

B. "Loss of Flow From the MDAFW pump".

The TDAFW logic includes loss of flow through the pump, loss of steam supply to the drive turbine, and failures of the turbine and pump. The MDAFW pump logic splits into identical sides for the two pumps. This logic includes loss of flow through the MDAFW pumps, and failures of the motors and pumps.

A2.2.2.2 Transfers to other systems. There are several transfers to other fault trees from the AFW model. A transfer entitled "Failure to actuate components controlled by SGAHRS division (x) amplifier" to gate (S900A,B,C) reflects failure of the SGAHRS actuation system to provide the appropriate signals to initiate function of the AFW components.

A transfer to the building electric power system fault trees is found when loss of power can result in failure of a valve or pump. A transfer entitled "fails to open due to 125V vital dc, division III power failure" is found in the logic for the drive turbine pressure control valve. Another transfer is found under the logic for loss of power to MDAFW pump A, entitled "loss of power from bus 12NIE003A." A similar transfer is found for MDAFW pump B, entitled "loss of power from bus 12NIE003B."

A2.2.2.3 Success criteria. In order for the AFW system to operate successfully, either of the following conditions must be satisified:

A. One of two MDAFW pumps (52AFP002A, B) operate as designed, or

B. The turbine driven pump (52AFP001) operates as designed.

A2.2.3 Modularized Fault Tree Model

The detailed fault-tree model was simplified by combining some of the basic events of Figure A2-2 into modules. The resulting modularized fault tree is depicted by Figure A2-3. The basic events and the logic used to combine them can be found in Table A2-1. Latent human errors affecting individual components were combined on a major pipe segment or train basis, as were events reflecting unavailability due to maintenance.

A2.3 FAILURE DATA

A2.3.1 Basic Event Data

Data was applied according to the rules and tables found in Section 5, for hardware, and Section 6, for human-reliability events. The events in the detailed fault tree are listed in Table A2-1, with their probabilities.

A2.3.2 Module Data

The probabilities for the modules are listed in Table A2-1, and are calculated from the probabilities for the primary events in the detailed fault tree, according to the module definitions.

A2.4 SYSTEM LEVEL INSIGHTS AND RESULTS

The modularized fault tree for the AFW system was solved using the Set Equation Transformation System (SETS) algorithm to determine the minimal cut sets of the system model. The cut sets with the highest probability

of failure, as indicated from the SETS output, were those involving electric power failures. Included in these cut sets were essential bus and transformer failures, as well as diesel generator faults. In addition to these electrical faults, failures in the SGAHRS actuation logic appeared in the dominant cut sets.

A2.5 REFERENCES

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2. Clinch River Breeder Reactor Plant, ECP G1135, "SGAHRS: Update Electrical Requirements", Revision 5.

3. Clinch River Breeder Reactor Plant Overall Plant Design Description, OPDD-10, Revision 146.

4. Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report. Section 16.2.4.2.

5. Clinch River Breeder Reactor Plant Probabilistic Risk Assessment - Phase I, Main Report and Appendices, EGG-EA-6162, January 1983.

6. Clinch River Breeder Reactor Plant System Design Description, Reactor Heat Transport Instrumentation and Control System, SDD-56, Revision 99.

7. Clinch River Breeder Reactor Plant System Design Description, Steam Generator Auxiliary Heat Removal System, SDD-52, Rev. 83.

8. Clinch River Breeder Reactor Plant System Design Description, Steam Generator System, SDD-53, Rev. 100.

9. General Electric Company Drawing 261R121, Rev. 39, Steam Generator P&ID.

10. General Electric Company Drawing 852E355, Rev. 12, SGAHRS Logic.

11. General Electric Company Drawing 852E661, Rev. 31, Steam Generator Auxiliary Heat Removal System P&ID.

12. General Electric Company Drawing 852E886, Rev. 2 SGAHRS Initiation Logic.

13. General Electric Company Drawing 908E803, Rev. 13, SGAHRS Instrumentation Diagram.

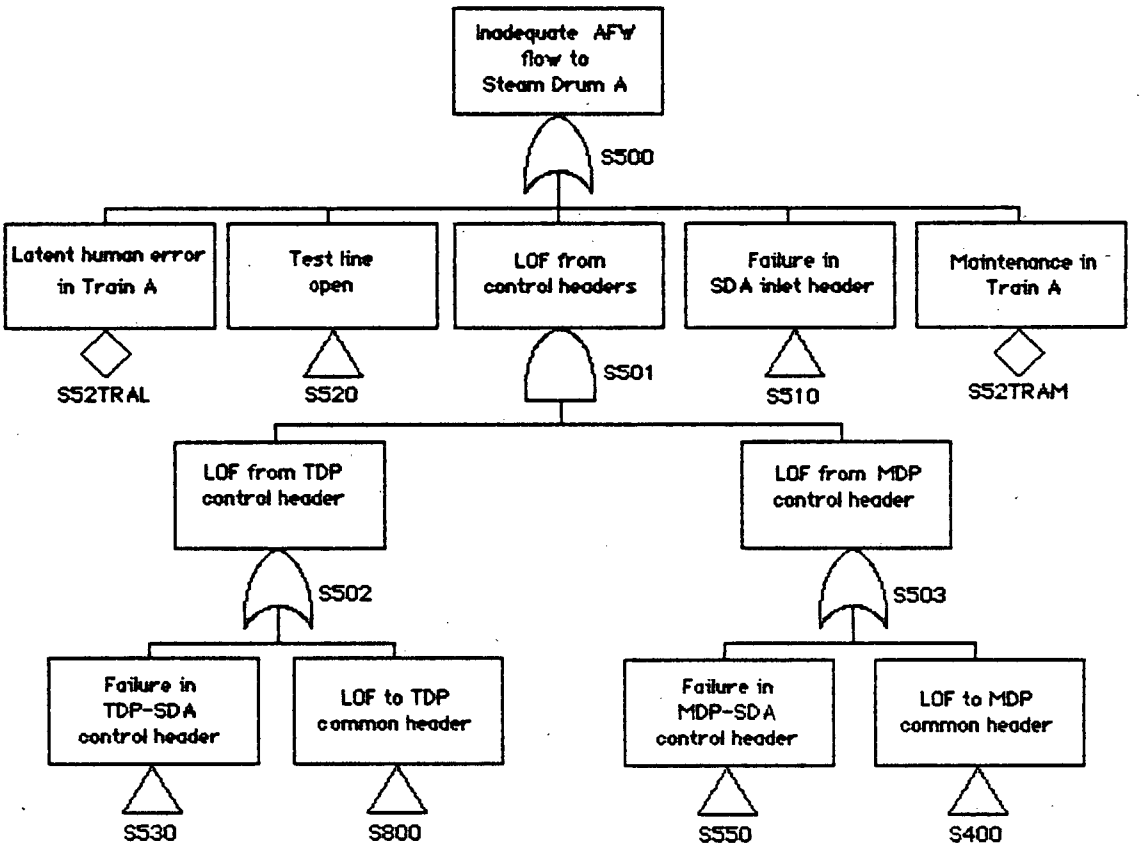
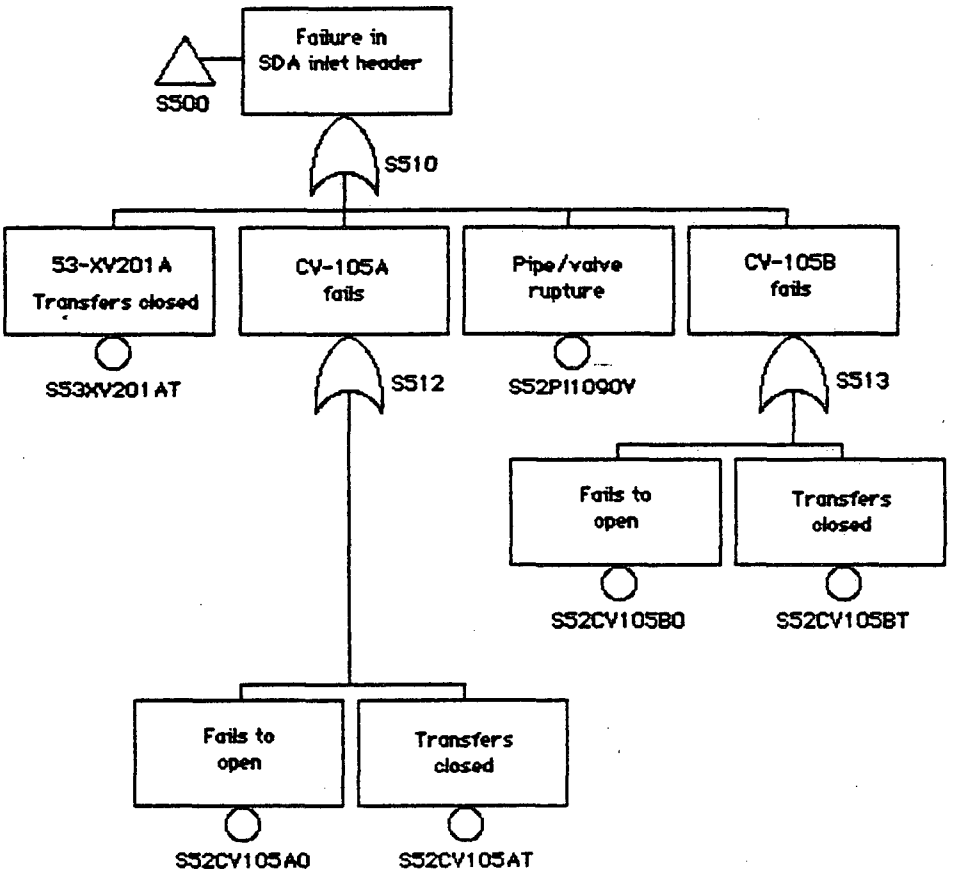
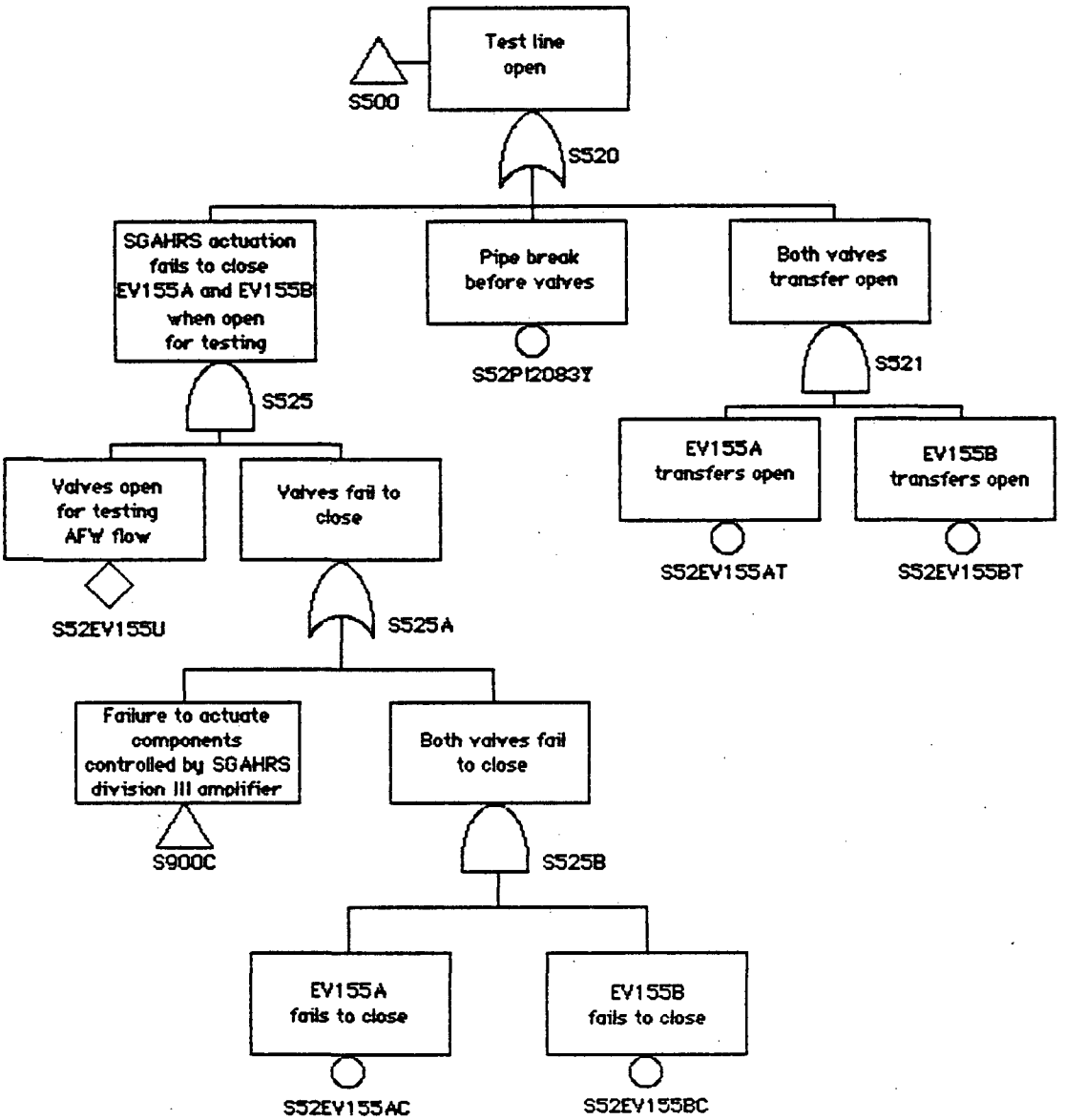
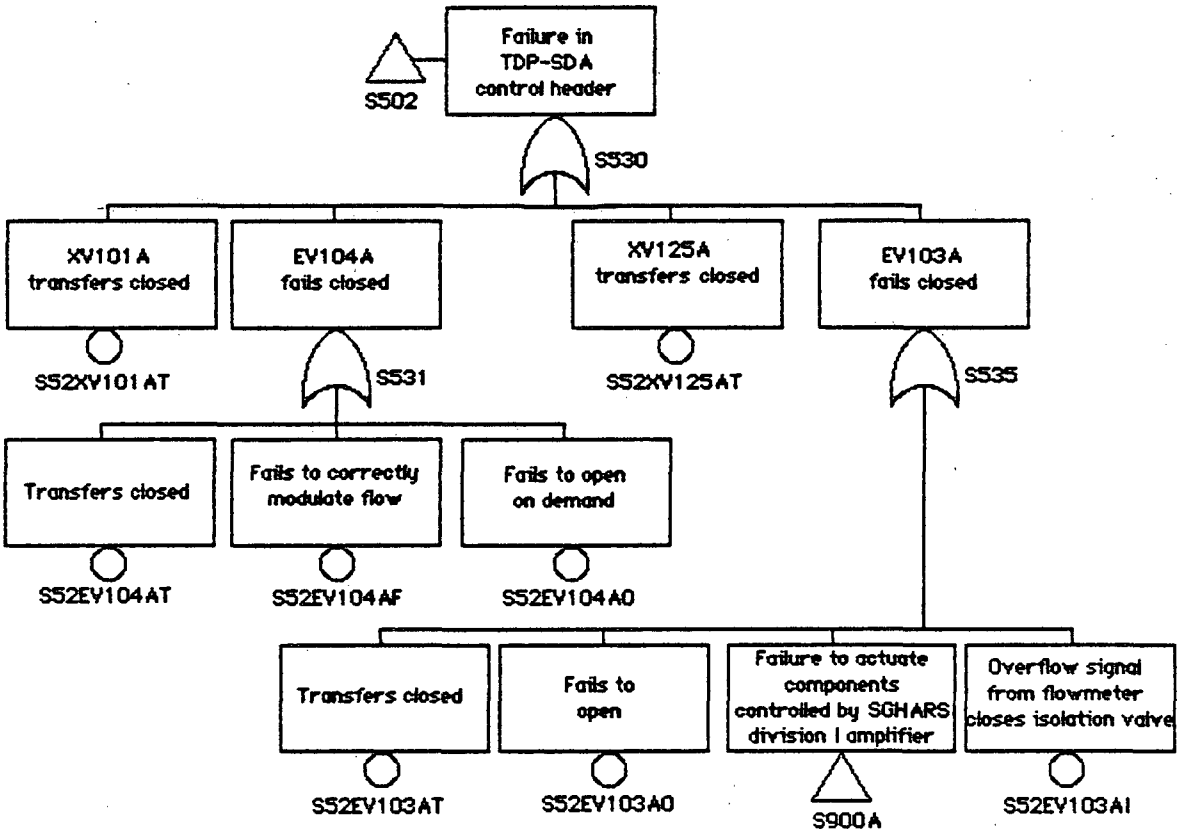
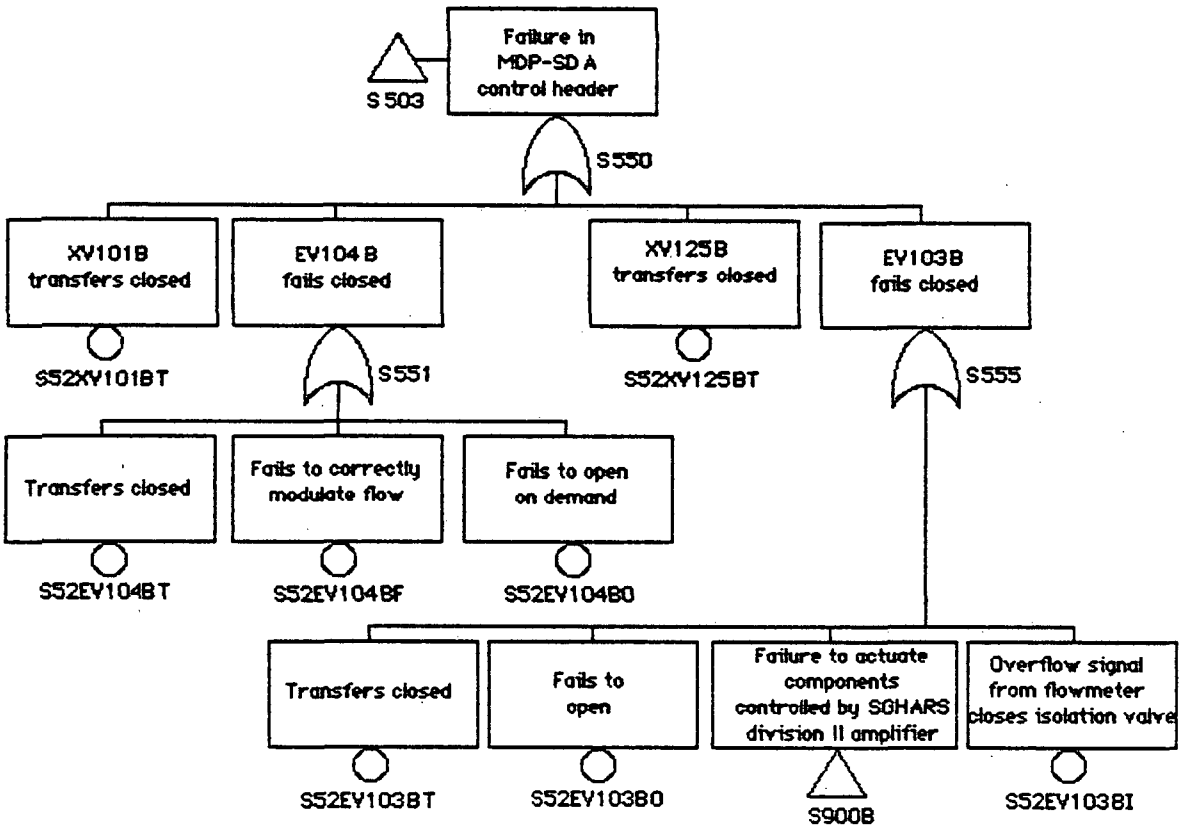


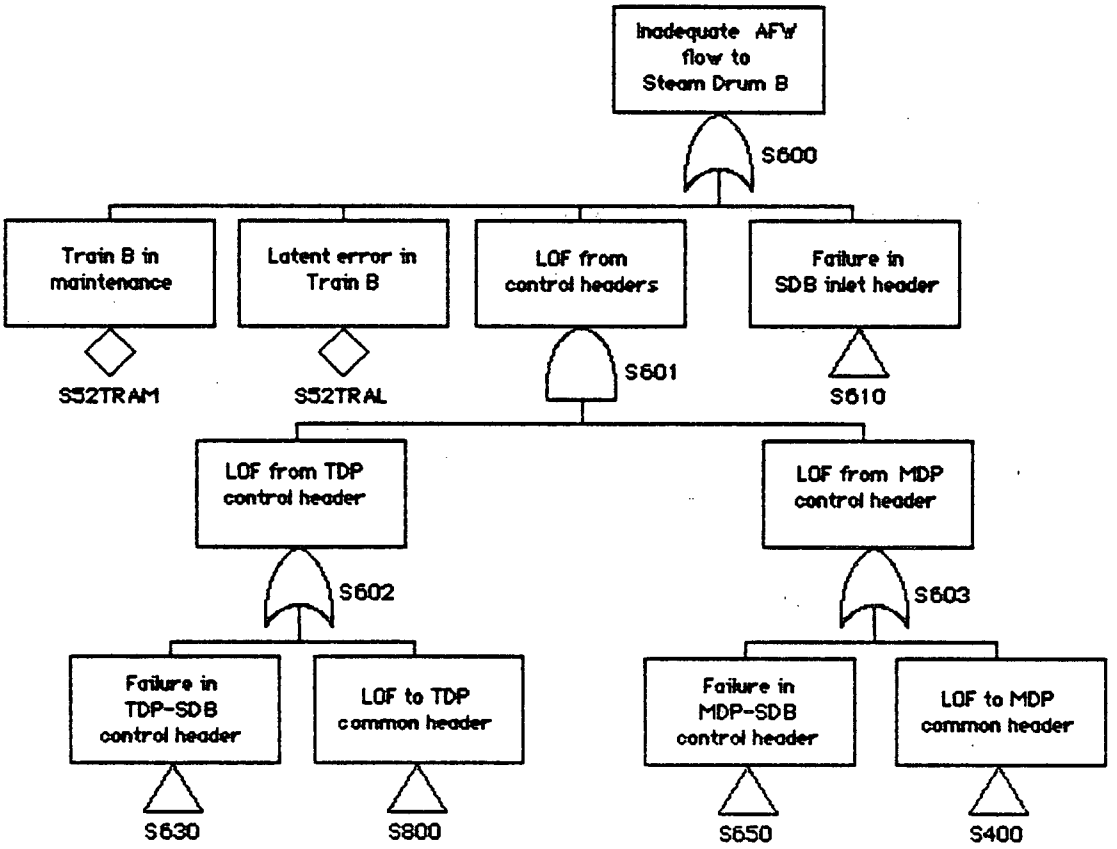
Figure A2-2. Auxiliary Feedwater System Detailed Fault Tree.

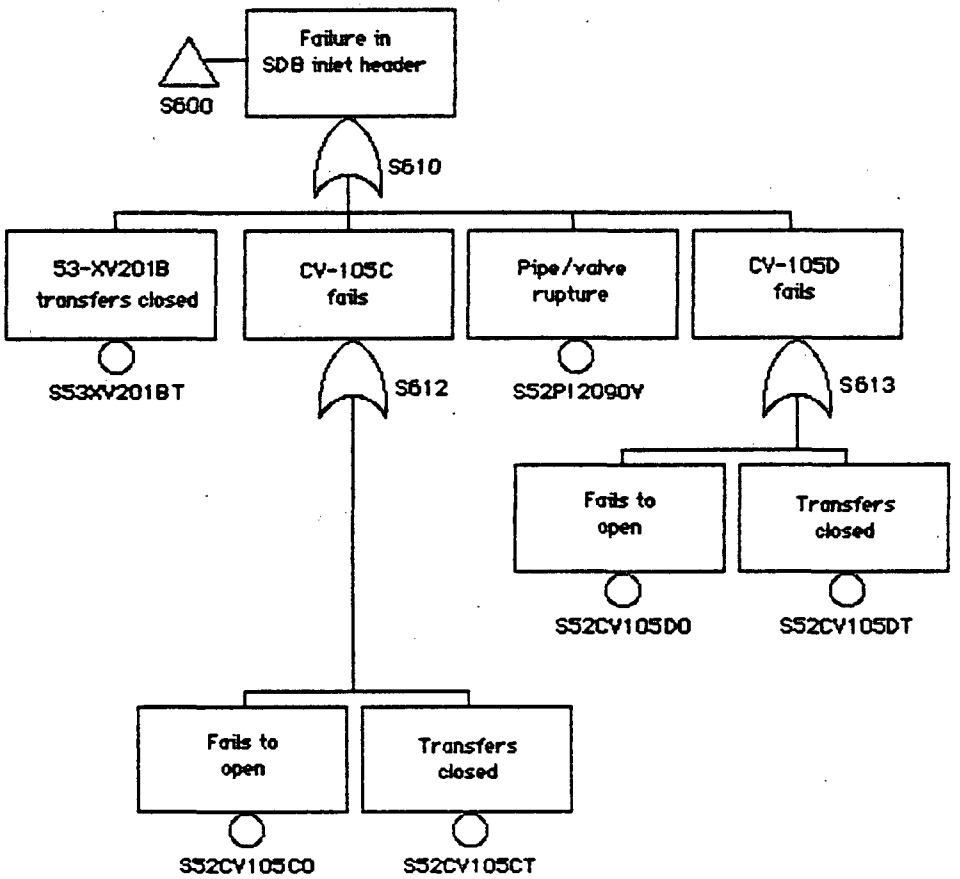

Failure in
SDA inlet header
S500
S510
53-XV201A
Transfers closed
CV-105A
fails
Pipe/valve
rupture
CV-105B
fails
S53XV201AT
S512
S52PI1090V
S513
Fails to
open
Transfers
closed
S52CV105B0
S52CV105BT
Fails to
open
Transfers
closed
S52CV105A0
S52CV105AT

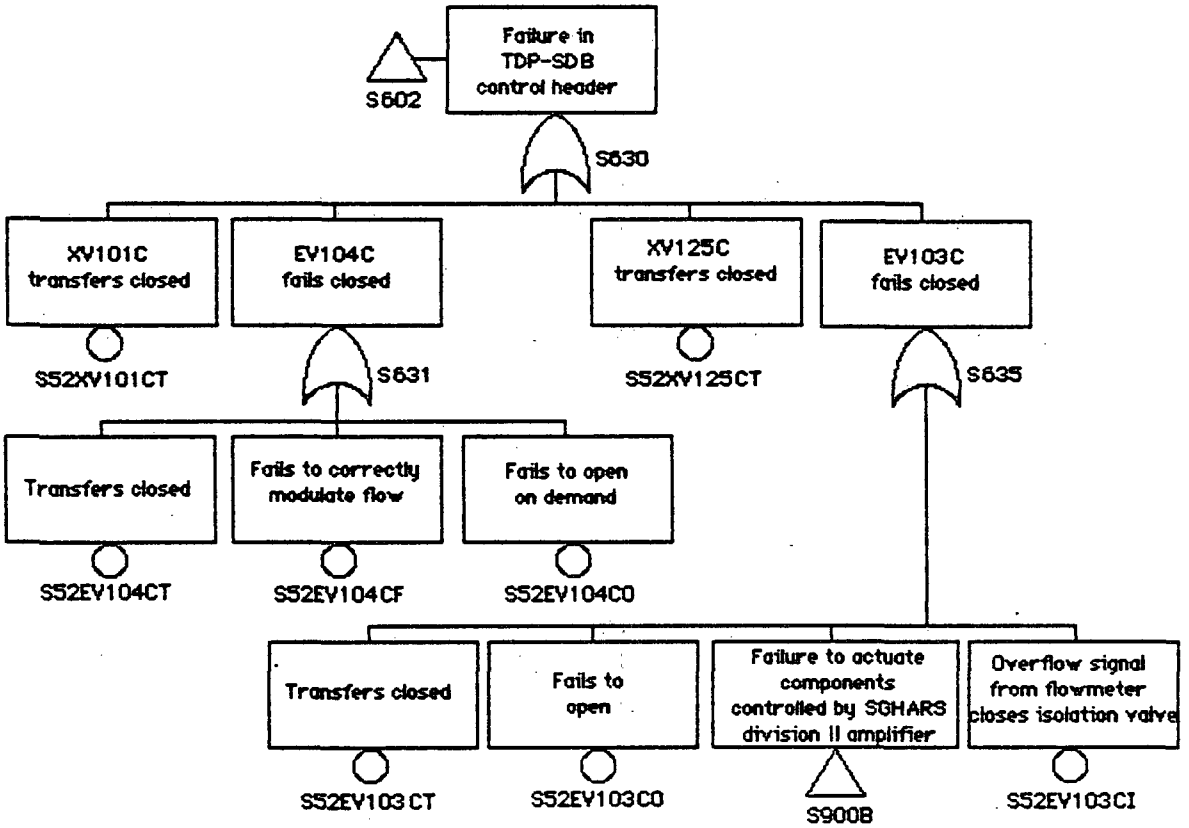

Test line open
S500
S520
SGAHRS actuation fails to close EV155A and EV155B when open for testing
Pipe break before valves
S52P12083Y
Both valves transfer open
S521
S525
EV155A transfers open
EV155B transfers open
S52EV155AT
S52EV155BT
Valves open for testing AFW flow
Valves fail to close
S52EV155U
S525A
Failure to actuate components controlled by SGAHRS division III amplifier
Both valves fail to close
S900C
S525B
EV155A fails to close
EV155B fails to close
S52EV155AC
S52EV155BC

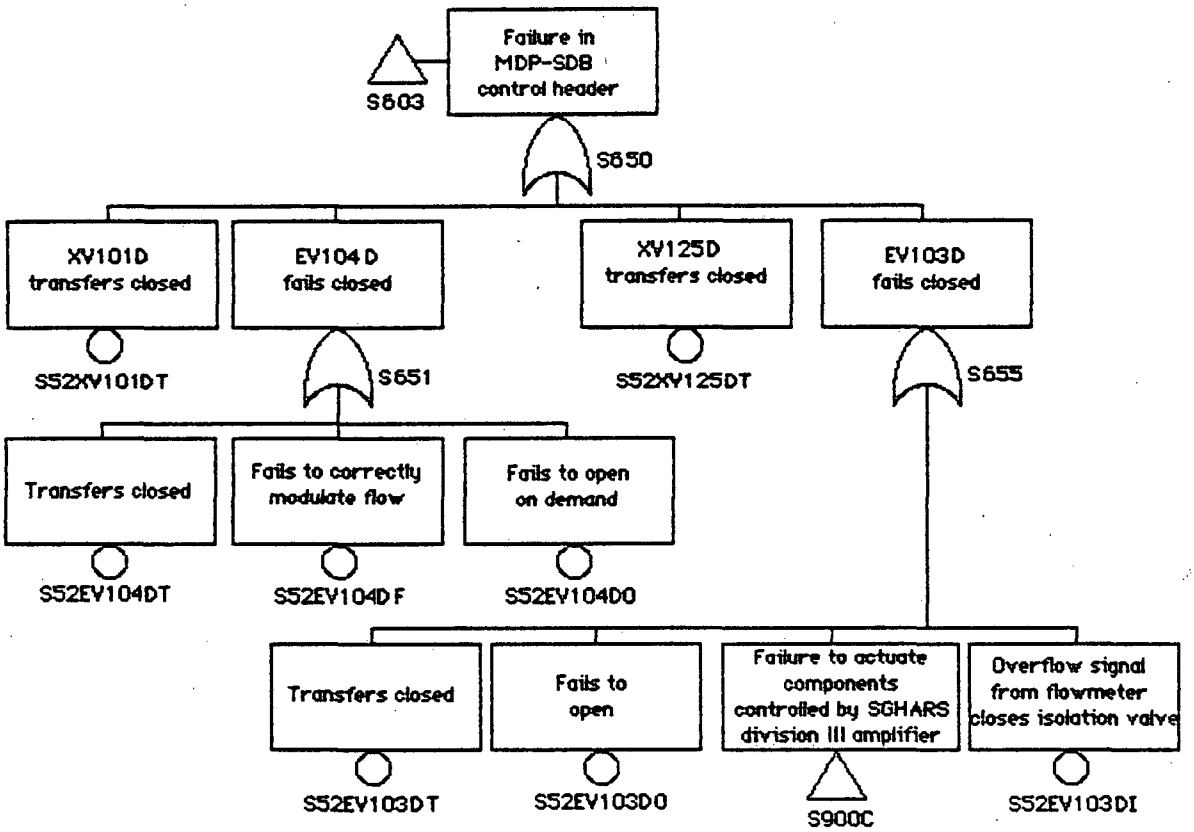

Failure in TDP-SDA control header
S502
S530
XV101A transfers closed
EV104A fails closed
XV125A transfers closed
EV103A fails closed
S52XV101AT
S531
S52XV125AT
S535
Transfers closed
Fails to correctly modulate flow
Fails to open on demand
S52EV104AT
S52EV104AF
S52EV104AO
Transfers closed
Fails to open
Failure to actuate components controlled by SGHARS division I amplifier
Overflow signal from flowmeter closes isolation valve
S52EV103AT
S52EV103AO
S900A
S52EV103AI

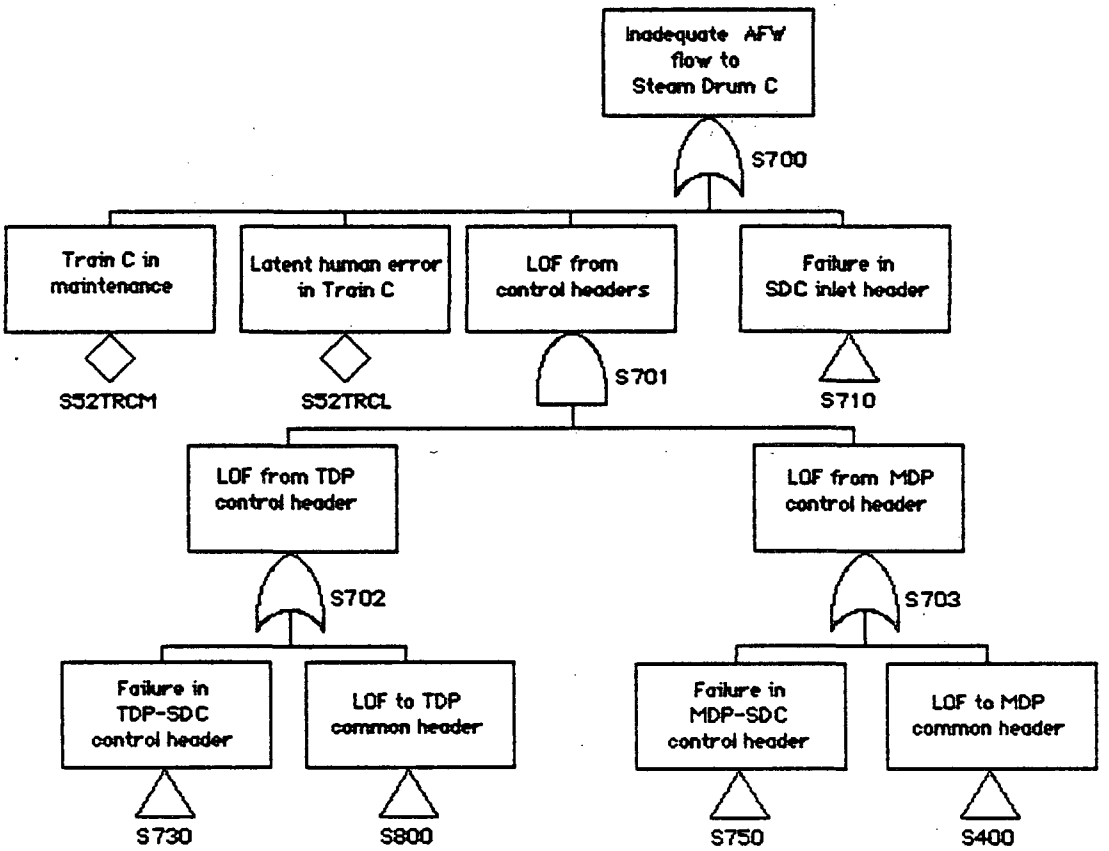

Failure in MDP-SD A control header
S 503
S550
XV101B transfers closed
EV104 B fails closed
XV125B transfers closed
EV103B fails closed
S52XV101BT
S551
S52XV125BT
S555
Transfers closed
Fails to correctly modulate flow
Fails to open on demand
S52EV104BT
S52EV104BF
S52EV104BO
Transfers closed
Fails to open
Failure to actuate components controlled by SGHARS division II amplifier
Overflow signal from flowmeter closes isolation valve
S52EV103BT
S52EV103BO
S900B
S52EV103BI

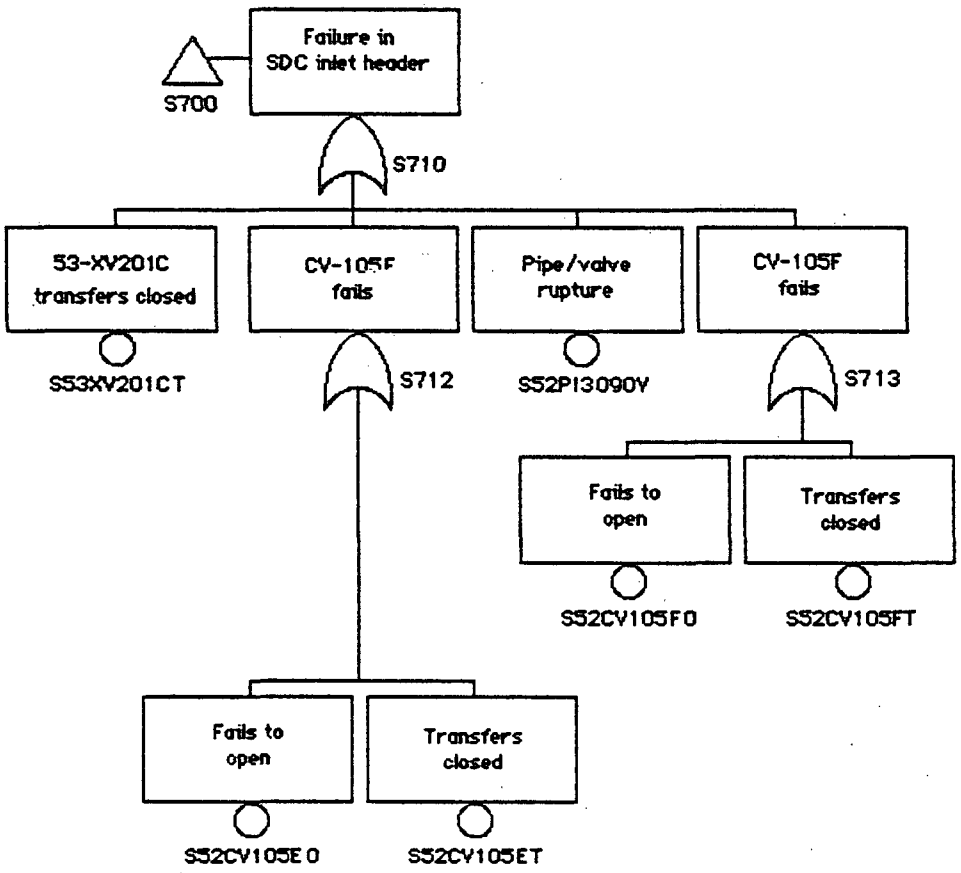

Inadequate AFW flow to Steam Drum B
S600
Train B in maintenance
Latent error in Train B
LOF from control headers
Failure in SDB inlet header
S52TRAM
S52TRAL
S601
S610
LOF from TDP control header
LOF from MDP control header
S602
S603
Failure in TDP-SDB control header
LOF to TDP common header
Failure in MDP-SDB control header
LOF to MDP common header
S630
S800
S650
S400

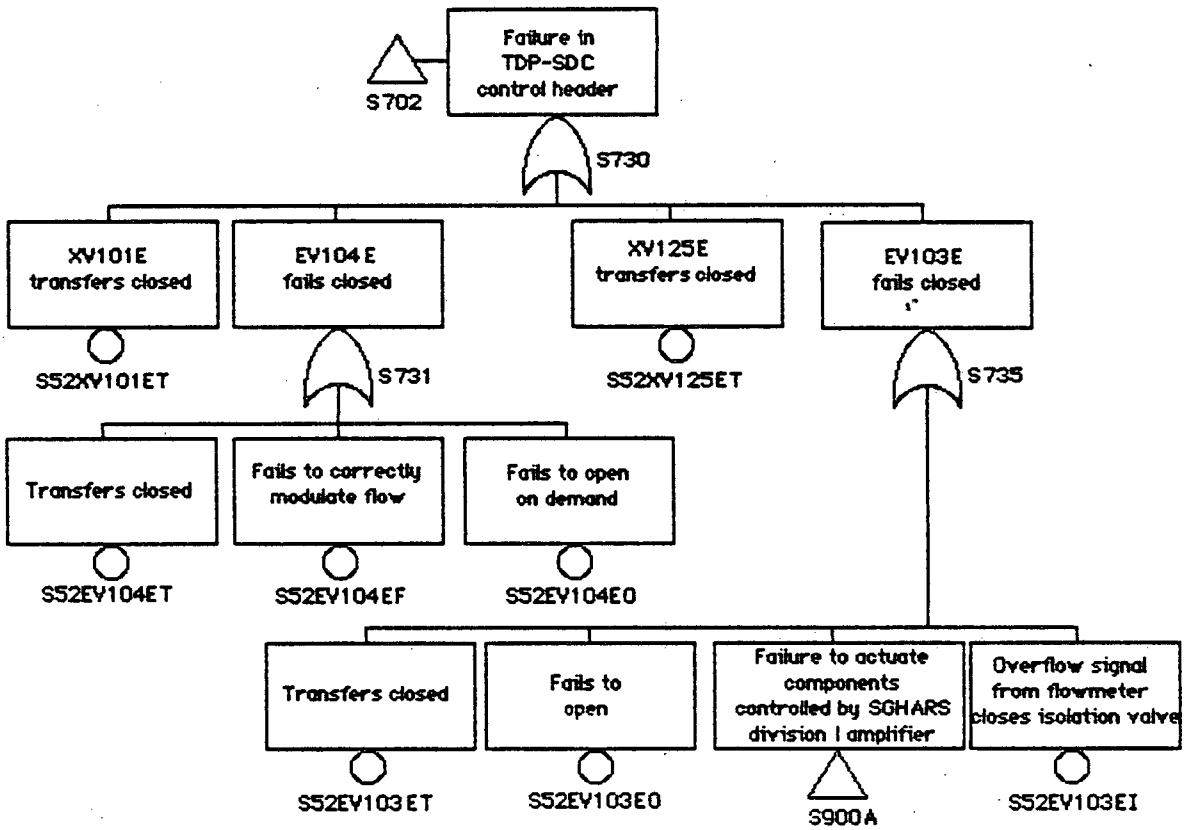

Failure in
SDB inlet header
S600
S610
53-XV201B
transfers closed
CV-105C
fails
Pipe/valve
rupture
CV-105D
fails
S53XV201BT
S612
S52PI2090Y
S613
Fails to
open
Transfers
closed
S52CV105DO
S52CV105DT
Fails to
open
Transfers
closed
S52CV105CO
S52CV105CT

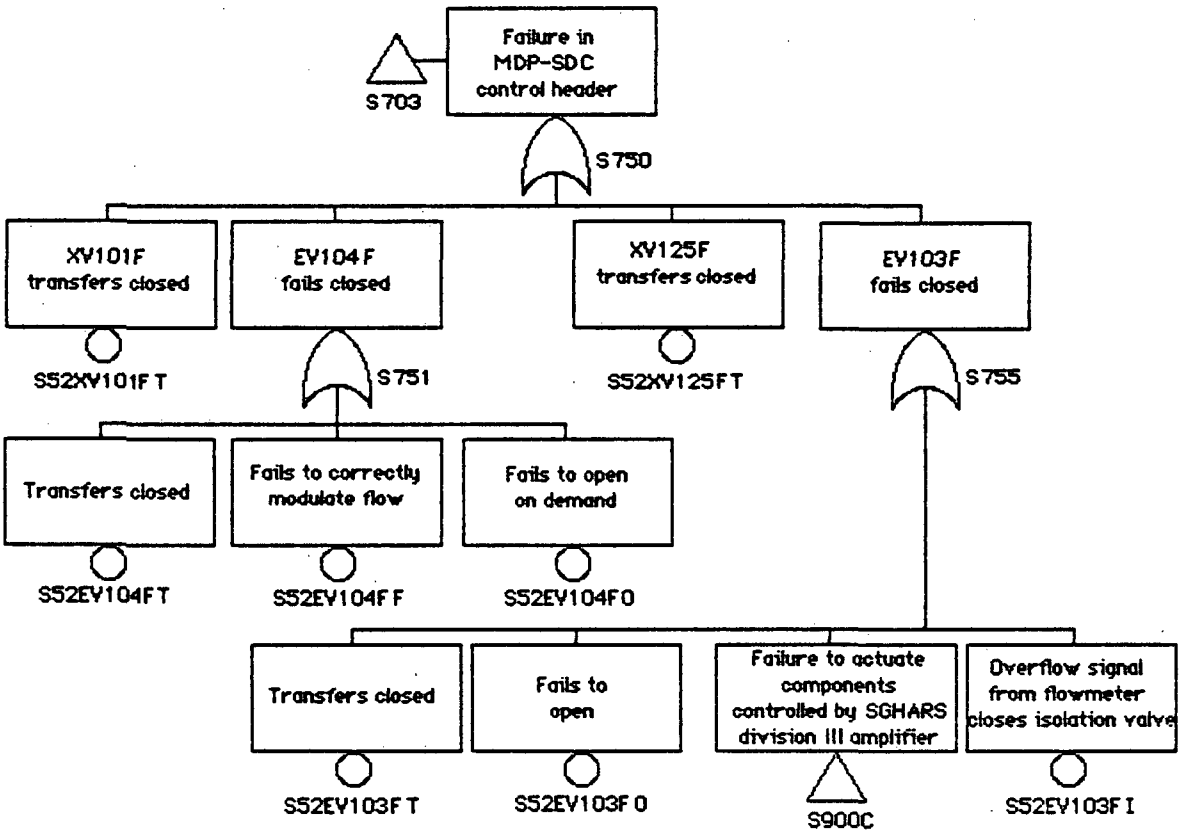

Failure in TDP-SDB control header
S602
S630
XV101C transfers closed
EV104C fails closed
XV125C transfers closed
EV103C fails closed
S52XV101CT
S631
S52XV125CT
S635
Transfers closed
Fails to correctly modulate flow
Fails to open on demand
S52EV104CT
S52EV104CF
S52EV104CO
Transfers closed
Fails to open
Failure to actuate components controlled by SGHARS division II amplifier
Overflow signal from flowmeter closes isolation valve
S52EV103CT
S52EV103CO
S900B
S52EV103CI

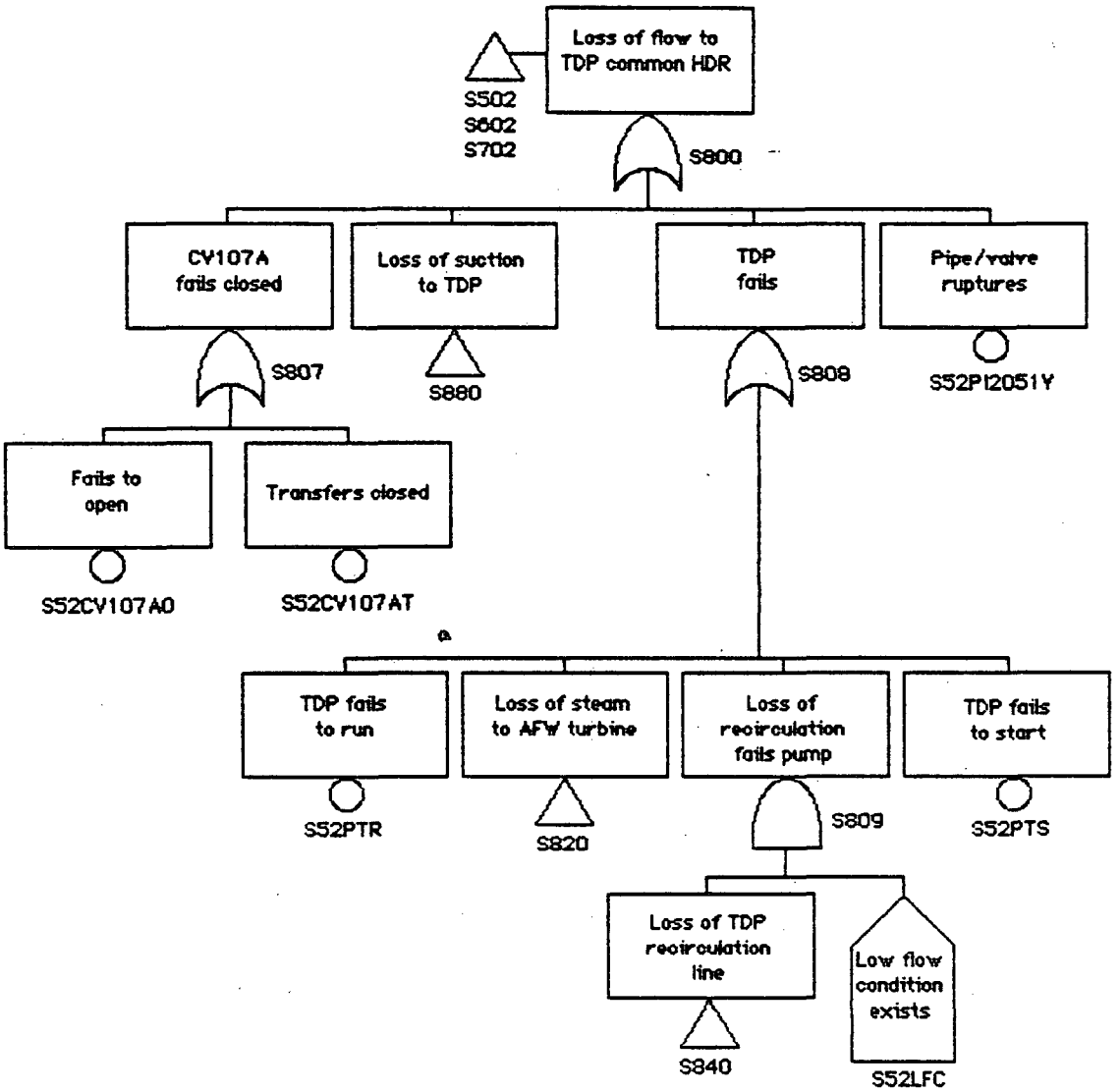

Failure in MDP-SDB control header
S603
S650
XV101D transfers closed
EV104D fails closed
XV125D transfers closed
EV103D fails closed
S52XV101DT
S651
S52XV125DT
S655
Transfers closed
Fails to correctly modulate flow
Fails to open on demand
S52EV104DT
S52EV104DF
S52EV104DO
Transfers closed
Fails to open
Failure to actuate components controlled by SGHARS division III amplifier
Overflow signal from flowmeter closes isolation valve
S52EV103DT
S52EV103DO
S900C
S52EV103DI

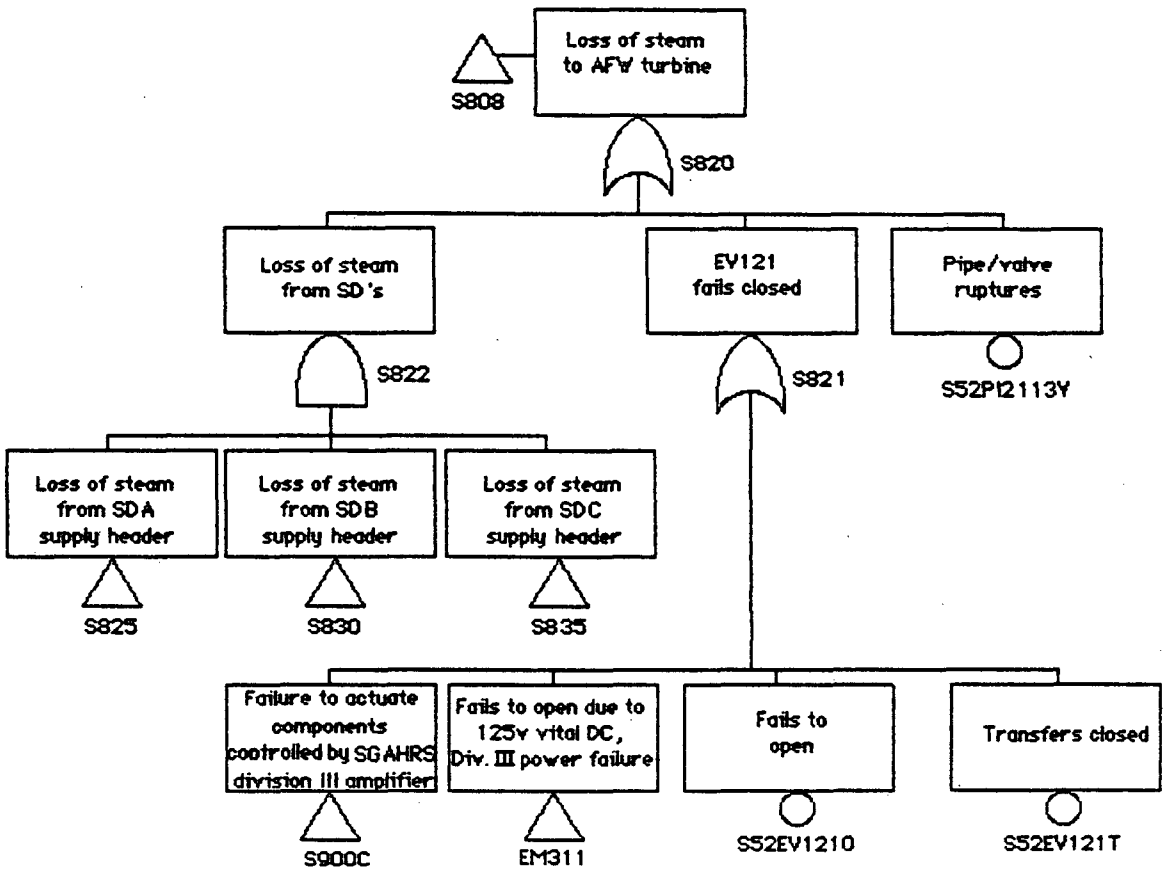

Inadequate AFW flow to Steam Drum C
S700
Train C in maintenance
Latent human error in Train C
LOF from control headers
Failure in SDC inlet header
S52TRCM
S52TRCL
S701
S710
LOF from TDP control header
LOF from MDP control header
S702
S703
Failure in TDP-SDC control header
LOF to TDP common header
Failure in MDP-SDC control header
LOF to MDP common header
S730
S800
S750
S400

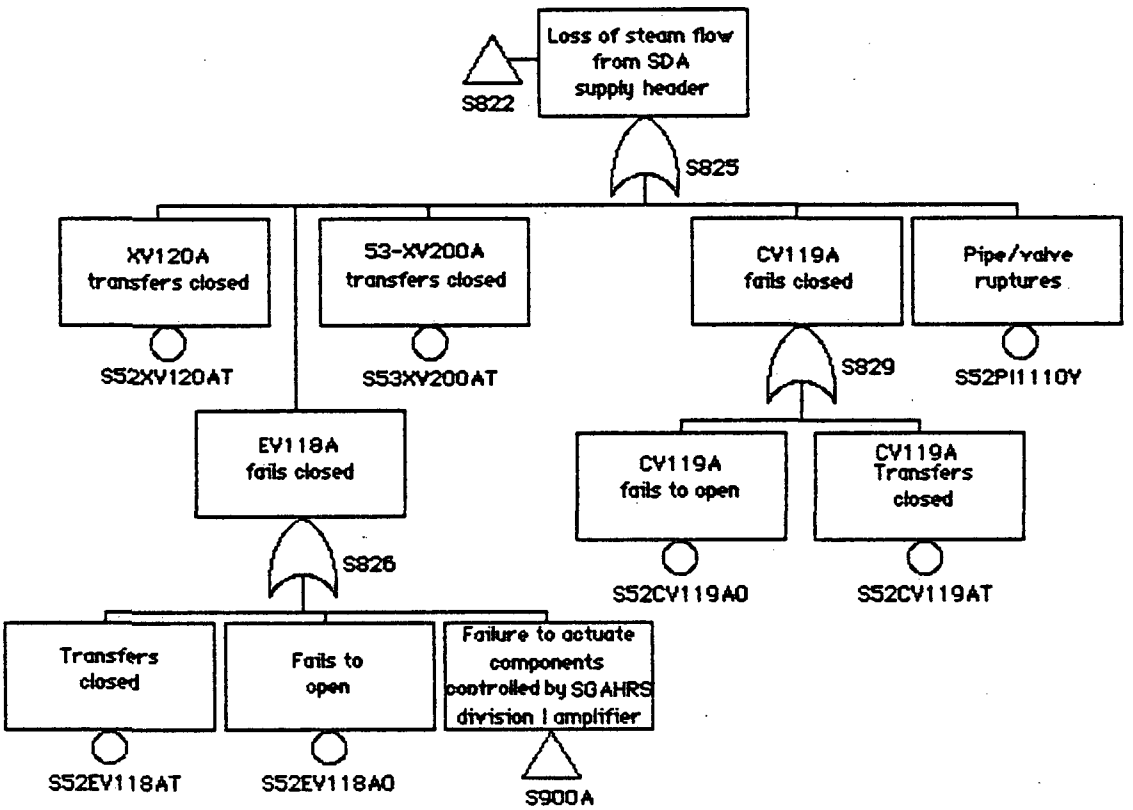

Failure in
SDC inlet header
S700
S710
53-XV201C
transfers closed
CV-105F
fails
Pipe/valve
rupture
CV-105F
fails
S53XV201CT
S712
S52PI3090Y
S713
Fails to
open
Transfers
closed
S52CV105FO
S52CV105FT
Fails to
open
Transfers
closed
S52CV105EO
S52CV105ET

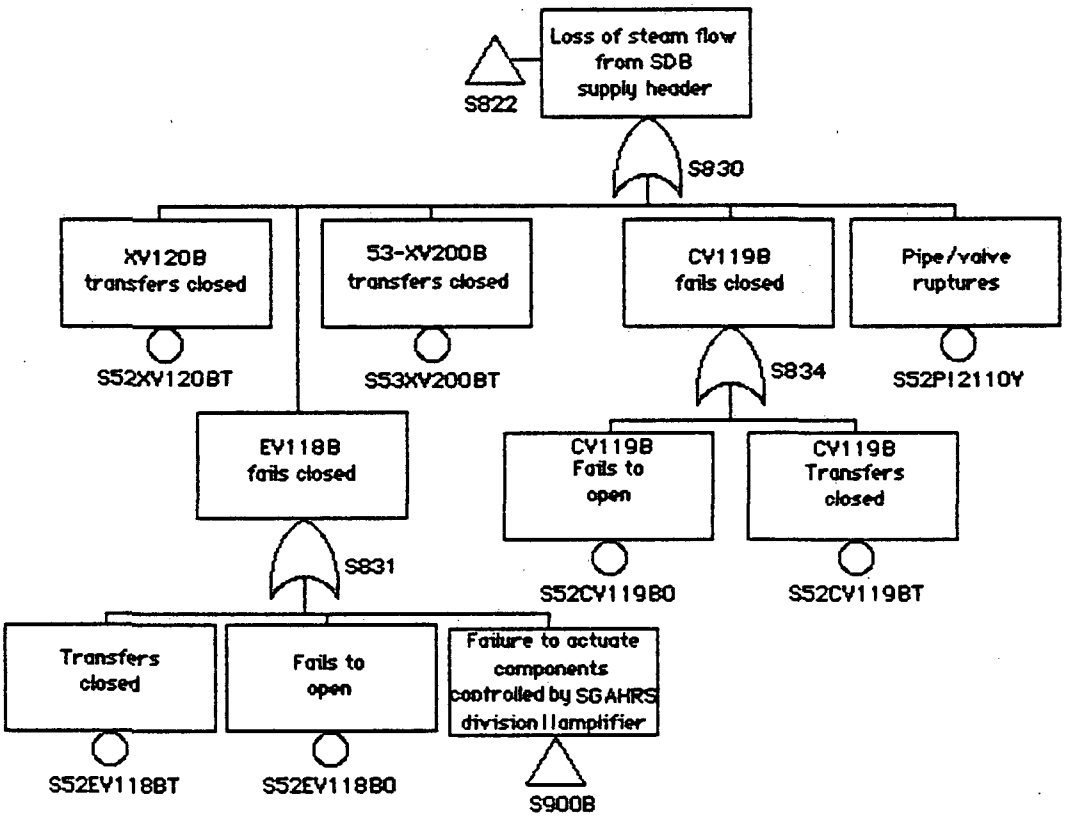

Failure in TDP-SDC control header
S702
S730
XV101E transfers closed
EV104E fails closed
XV125E transfers closed
EV103E fails closed
S52XV101ET
S731
S52XV125ET
S735
Transfers closed
Fails to correctly modulate flow
Fails to open on demand
S52EV104ET
S52EV104EF
S52EV104EO
Transfers closed
Fails to open
Failure to actuate components controlled by SGHARS division I amplifier
Overflow signal from flowmeter closes isolation valve
S52EV103ET
S52EV103EO
S900A
S52EV103EI

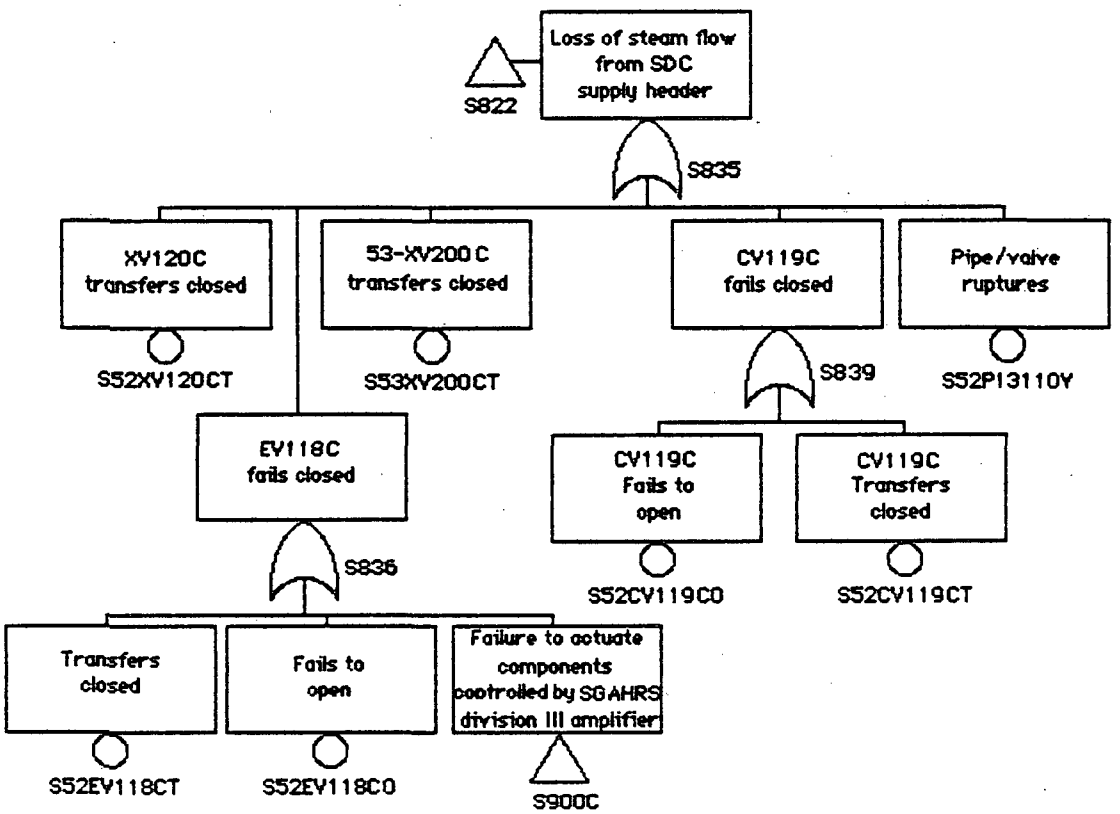

Failure in MDP-SDC control header
S703
S750
XV101F transfers closed
EV104F fails closed
XV125F transfers closed
EV103F fails closed
S52XV101FT
S751
S52XV125FT
S755
Transfers closed
Fails to correctly modulate flow
Fails to open on demand
S52EV104FT
S52EV104FF
S52EV104FO
Transfers closed
Fails to open
Failure to actuate components controlled by SGHARS division III amplifier
Overflow signal from flowmeter closes isolation valve
S52EV103FT
S52EV103FO
S900C
S52EV103FI

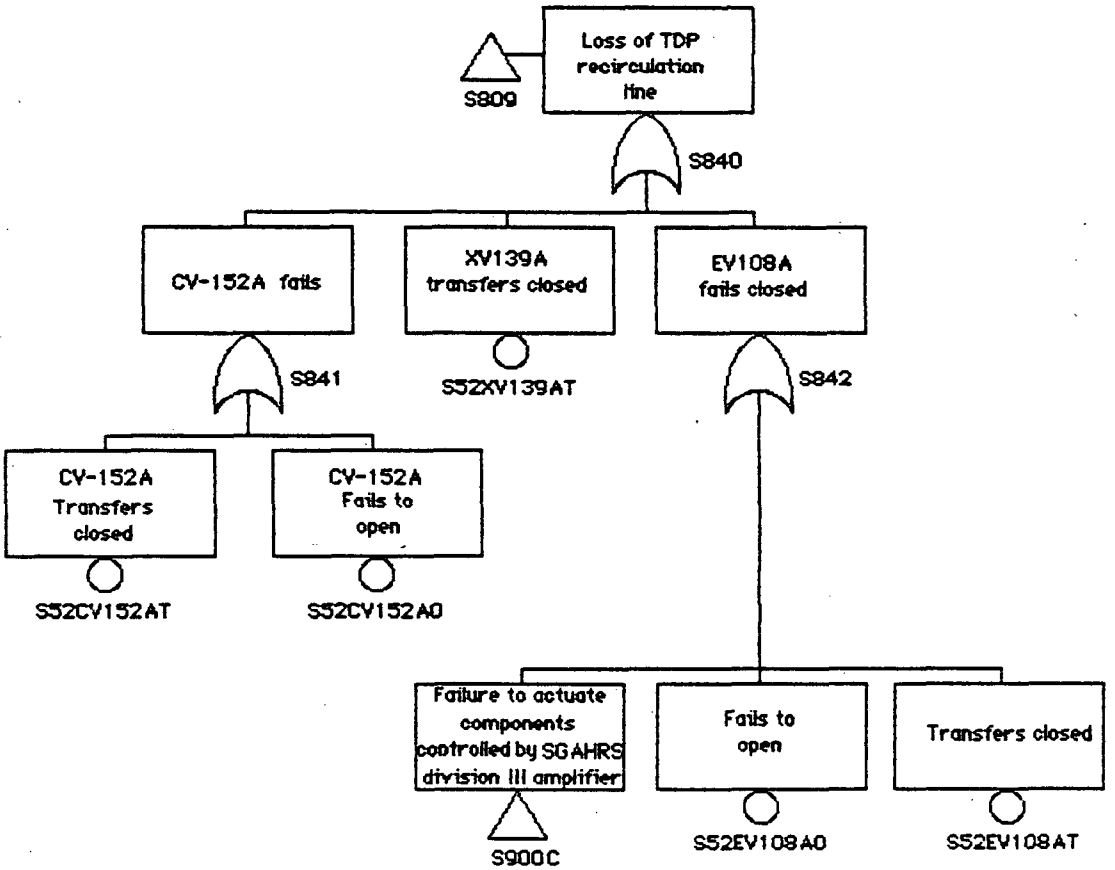

Loss of flow to TDP common HDR
S502
S602
S702
S800
CV107A fails closed
Loss of suction to TDP
TDP fails
Pipe/valve ruptures
S807
S880
S808
S52PI2051Y
Fails to open
Transfers closed
S52CV107AO
S52CV107AT
TDP fails to run
Loss of steam to AFW turbine
Loss of recirculation fails pump
TDP fails to start
S52PTR
S820
S809
S52PTS
Loss of TDP recirculation line
Low flow condition exists
S840
S52LFC

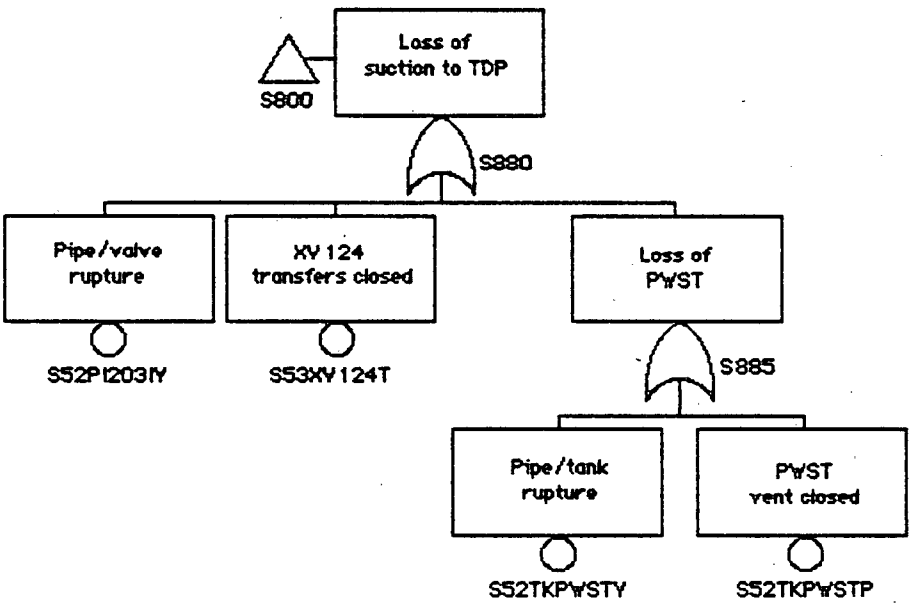

Loss of steam to AFW turbine
S808
S820
Loss of steam from SD's
EV121 fails closed
Pipe/valve ruptures
S822
S821
S52PI2113Y
Loss of steam from SDA supply header
Loss of steam from SDB supply header
Loss of steam from SDC supply header
S825
S830
S835
Failure to actuate components controlled by SG AHRS division III amplifier
Fails to open due to 125v vital DC, Div. III power failure
Fails to open
Transfers closed
S900C
EM311
S52EV1210
S52EV121T

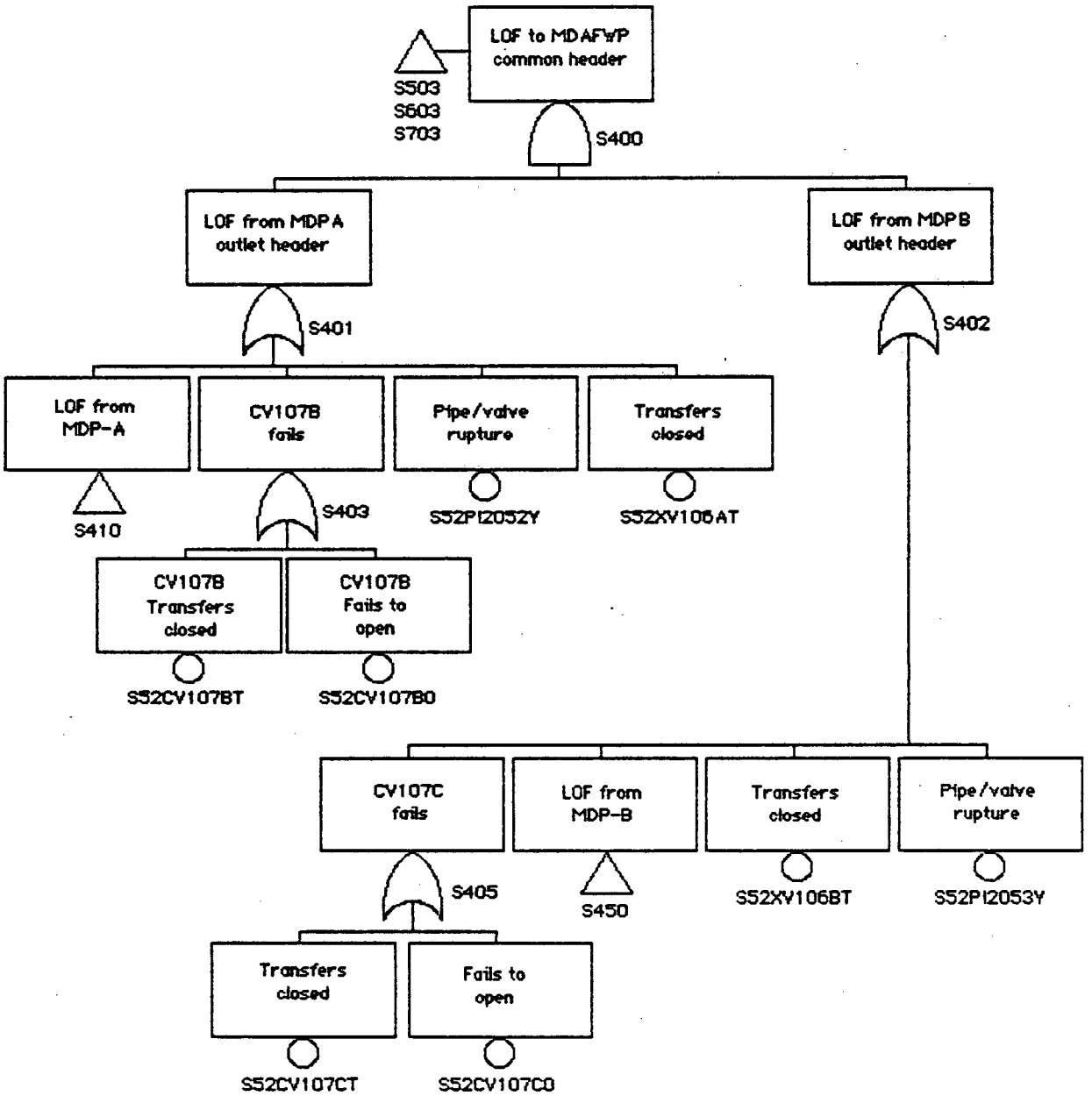

Loss of steam flow from SDA supply header
S822
S825
XV120A transfers closed
S52XV120AT
S3-XV200A transfers closed
S53XV200AT
CV119A fails closed
S829
Pipe/valve ruptures
S52PI1110Y
EV118A fails closed
S826
CV119A fails to open
S52CV119A0
CV119A Transfers closed
S52CV119AT
Transfers closed
S52EV118AT
Fails to open
S52EV118A0
Failure to actuate components controlled by SGAHRS division I amplifier
S900A

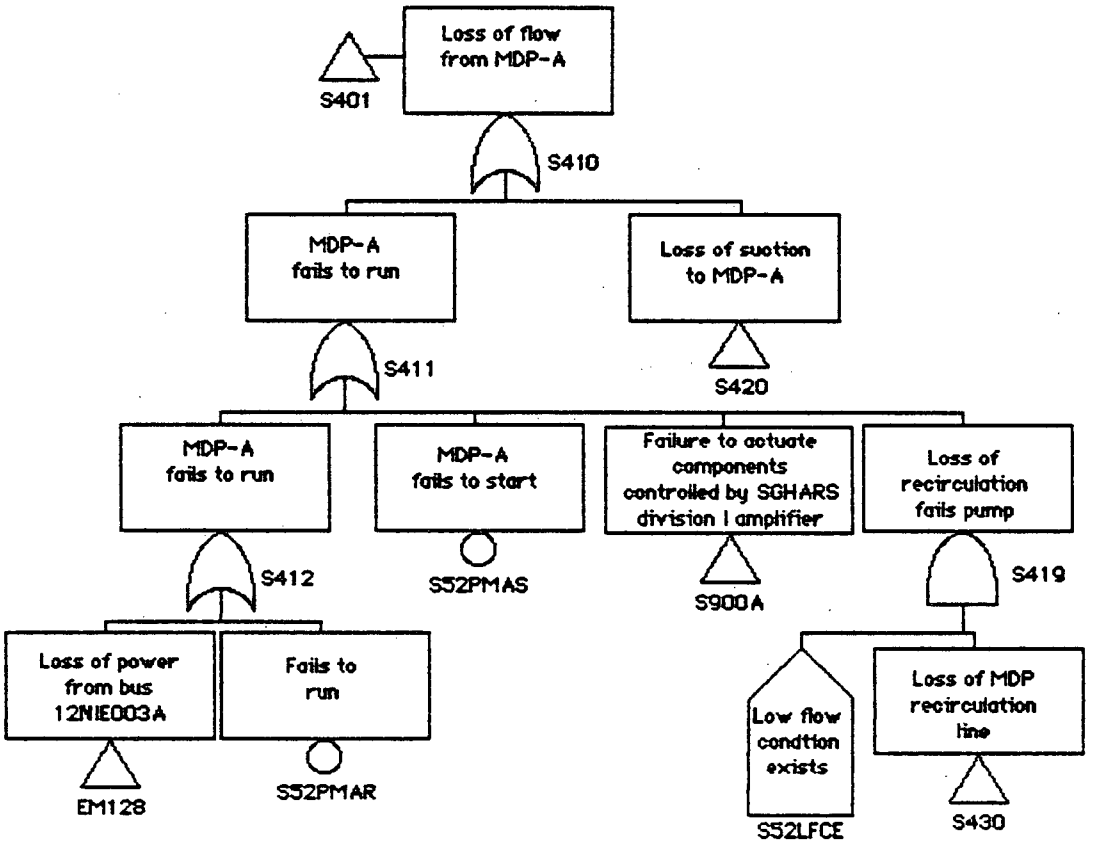

Loss of steam flow from SDB supply header
S822
S830
XV120B transfers closed
53-XV200B transfers closed
CV119B fails closed
Pipe/valve ruptures
S52XV120BT
S53XV200BT
S834
S52PI2110Y
EV118B fails closed
CV119B Fails to open
CV119B Transfers closed
S831
S52CV119BO
S52CV119BT
Transfers closed
Fails to open
Failure to actuate components controlled by SGAHRS division II amplifier
S52EV118BT
S52EV118BO
S900B

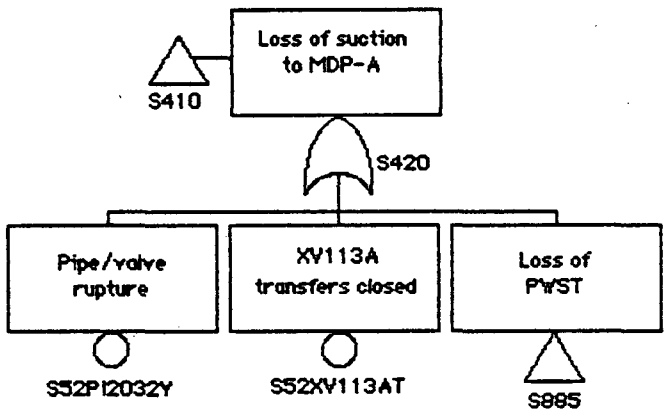

Loss of steam flow from SDC supply header
S822
S835
XV120C transfers closed
53-XV200C transfers closed
CV119C fails closed
Pipe/valve ruptures
S52XV120CT
S53XV200CT
S839
S52P13110Y
EV118C fails closed
CV119C Fails to open
CV119C Transfers closed
S836
S52CV119CO
S52CV119CT
Transfers closed
Fails to open
Failure to actuate components controlled by SG AHRS division III amplifier
S52EV118CT
S52EV118CO
S900C

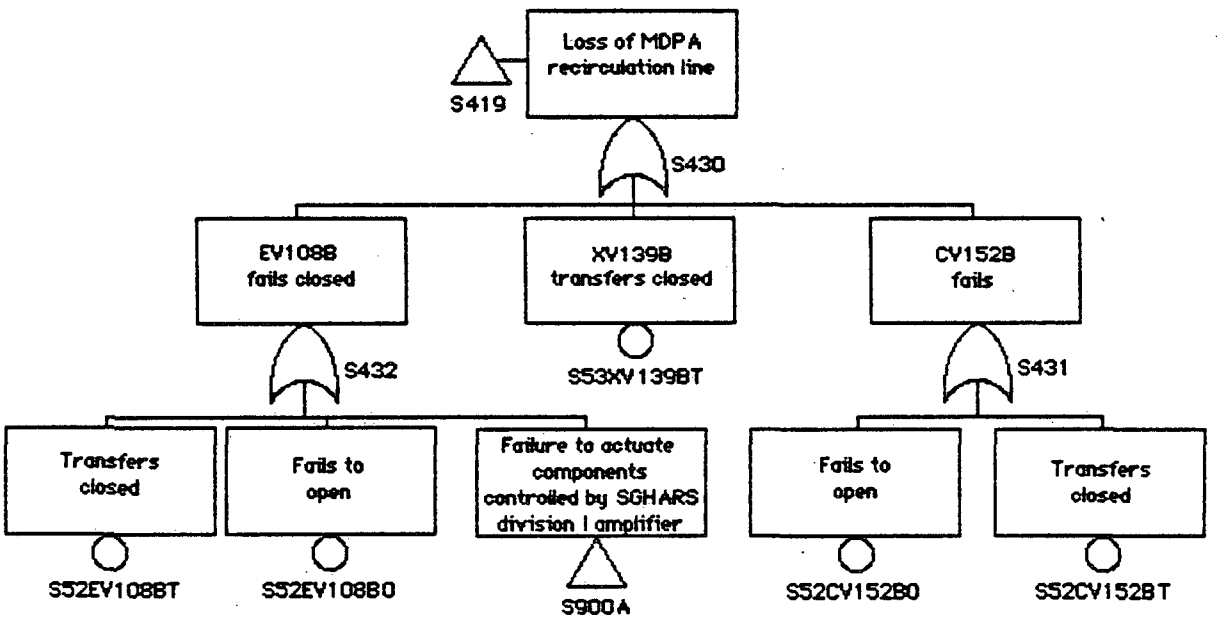

Loss of TDP recirculation line
S809
S840
CV-152A fails
XV139A transfers closed
EV108A fails closed
S841
S52XV139AT
S842
CV-152A Transfers closed
CV-152A Fails to open
S52CV152AT
S52CV152AO
Failure to actuate components controlled by SG AHRS division III amplifier
Fails to open
Transfers closed
S900C
S52EV108AO
S52EV108AT

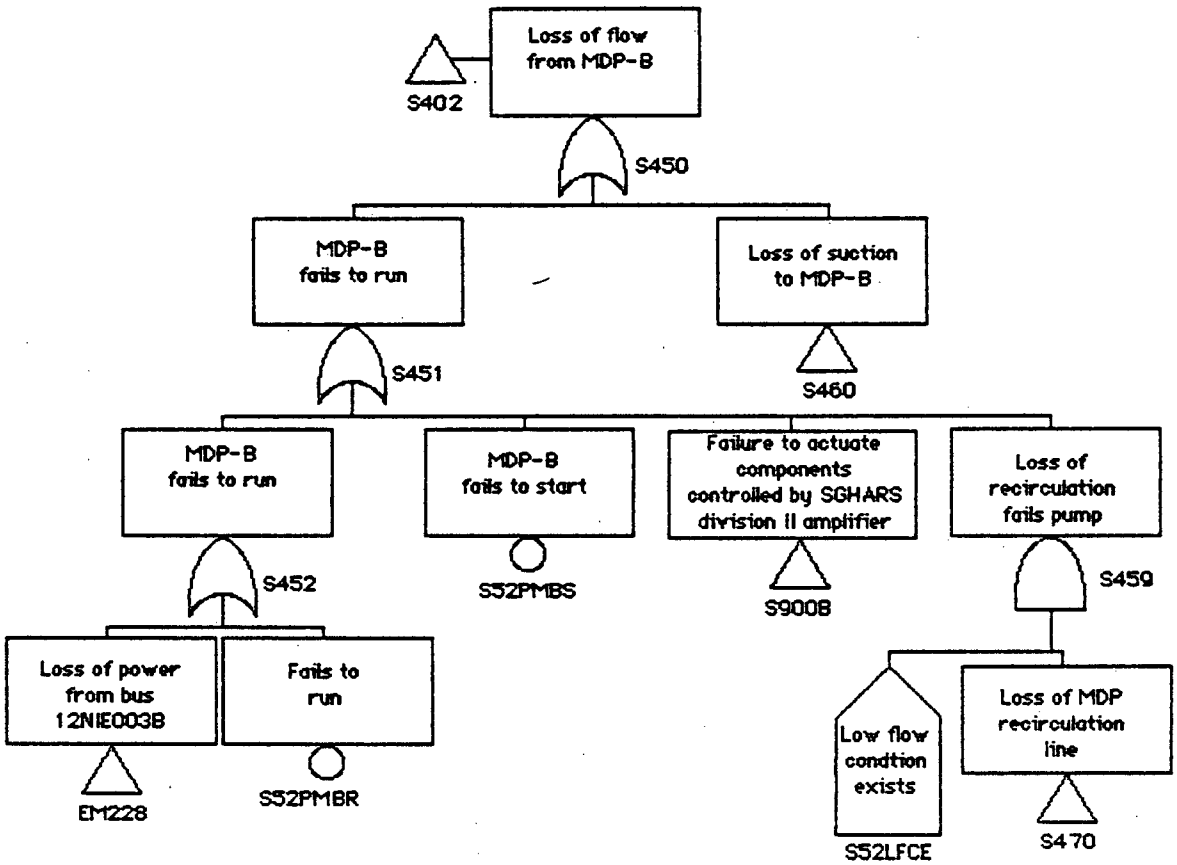

Loss of suction to TDP
S800
S880
Pipe/valve rupture
XV 124 transfers closed
Loss of PWST
S52P1203IY
S53XV124T
S885
Pipe/tank rupture
PWST vent closed
S52TKPWSTY
S52TKPWSTP

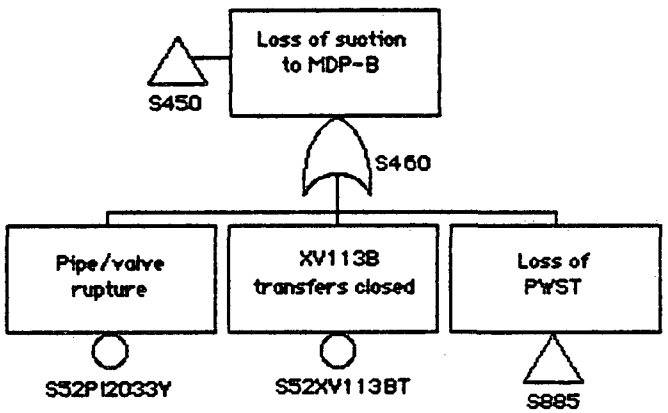

LOF to MDAFWP common header
S503
S603
S703
S400
LOF from MDPA outlet header
LOF from MDPB outlet header
S401
S402
LOF from MDP-A
CV107B fails
Pipe/valve rupture
Transfers closed
S410
S403
S52PI2052Y
S52XV106AT
CV107B Transfers closed
CV107B Fails to open
S52CV107BT
S52CV107BO
CV107C fails
LOF from MDP-B
Transfers closed
Pipe/valve rupture
S405
S450
S52XV106BT
S52PI2053Y
Transfers closed
Fails to open
S52CV107CT
S52CV107CO

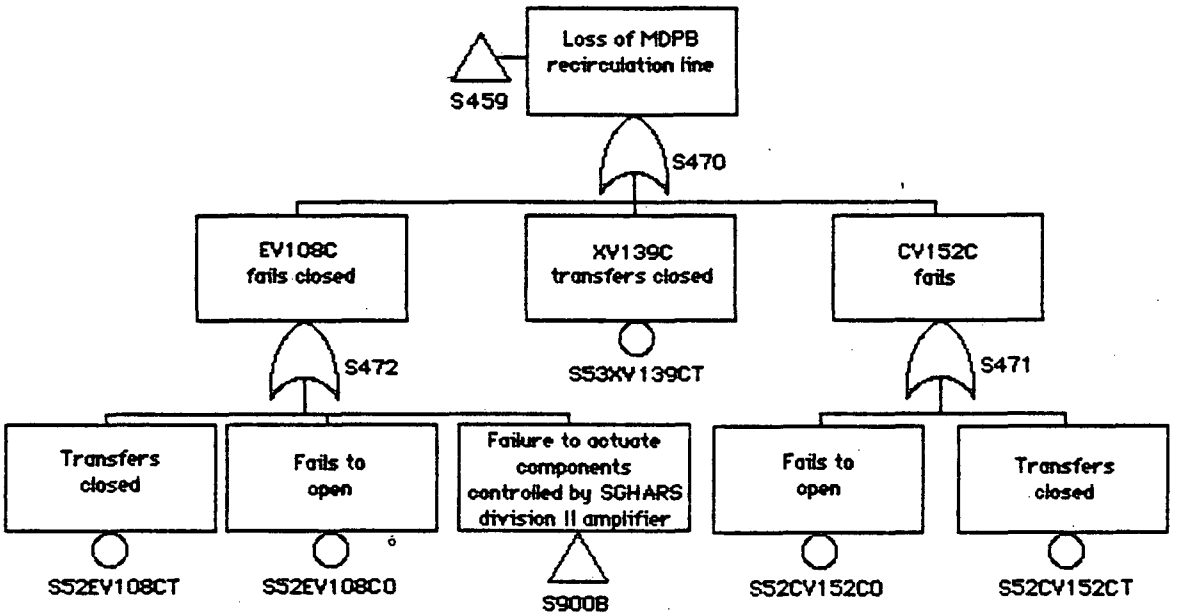

Loss of flow from MDP-A
S401
S410
MDP-A fails to run
Loss of suction to MDP-A
S411
S420
MDP-A fails to run
MDP-A fails to start
Failure to actuate components controlled by SGHARS division I amplifier
Loss of recirculation fails pump
S412
S52PMAS
S900A
S419
Loss of power from bus 12NIE003A
Fails to run
Low flow condtion exists
Loss of MDP recirculation line
EM128
S52PMAR
S52LFCE
S430


Loss of suction to MDP-A
S410
S420
Pipe/valve rupture
XV113A transfers closed
Loss of PWST
S52P12032Y
S52XV113AT
S885


Loss of MDPA recirculation line
S419
S430
EV108B fails closed
XV139B transfers closed
CV152B fails
S432
S53XV139BT
S431
Transfers closed
Fails to open
Failure to actuate components controlled by SGHARS division I amplifier
Fails to open
Transfers closed
S52EV108BT
S52EV108BO
S900A
S52CV152BO
S52CV152BT


Loss of flow from MDP-B
S402
S450
MDP-B fails to run
Loss of suction to MDP-B
S451
S460
MDP-B fails to run
MDP-B fails to start
Failure to actuate components controlled by SGHARS division II amplifier
Loss of recirculation fails pump
S452
S52PMBS
S900B
S459
Loss of power from bus 12NIE003B
Fails to run
Low flow condtion exists
Loss of MDP recirculation line
EM228
S52PMBR
S52LFCE
S470


Loss of suction to MDP-B
S450
S460
Pipe/valve rupture
XV113B transfers closed
Loss of PWST
S52P12033Y
S52XV113BT
S885


Loss of MDPB recirculation line
S459
S470
EV108C fails closed
XV139C transfers closed
CV152C fails
S472
S53XV139CT
S471
Transfers closed
Fails to open
Failure to actuate components controlled by SGHARS division II amplifier
Fails to open
Transfers closed
S52EV108CT
S52EV108CO
S900B
S52CV152CO
S52CV152CT

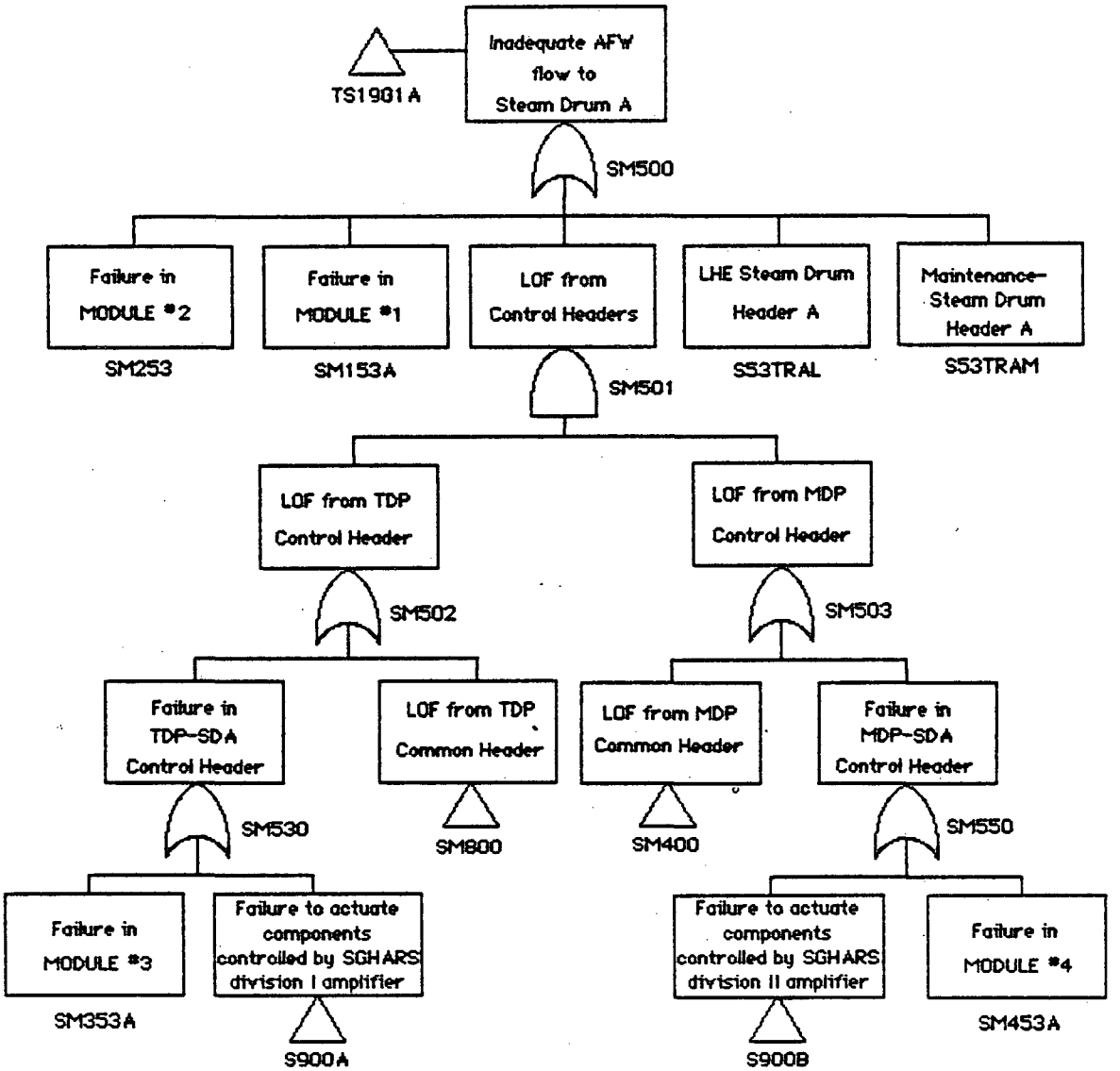
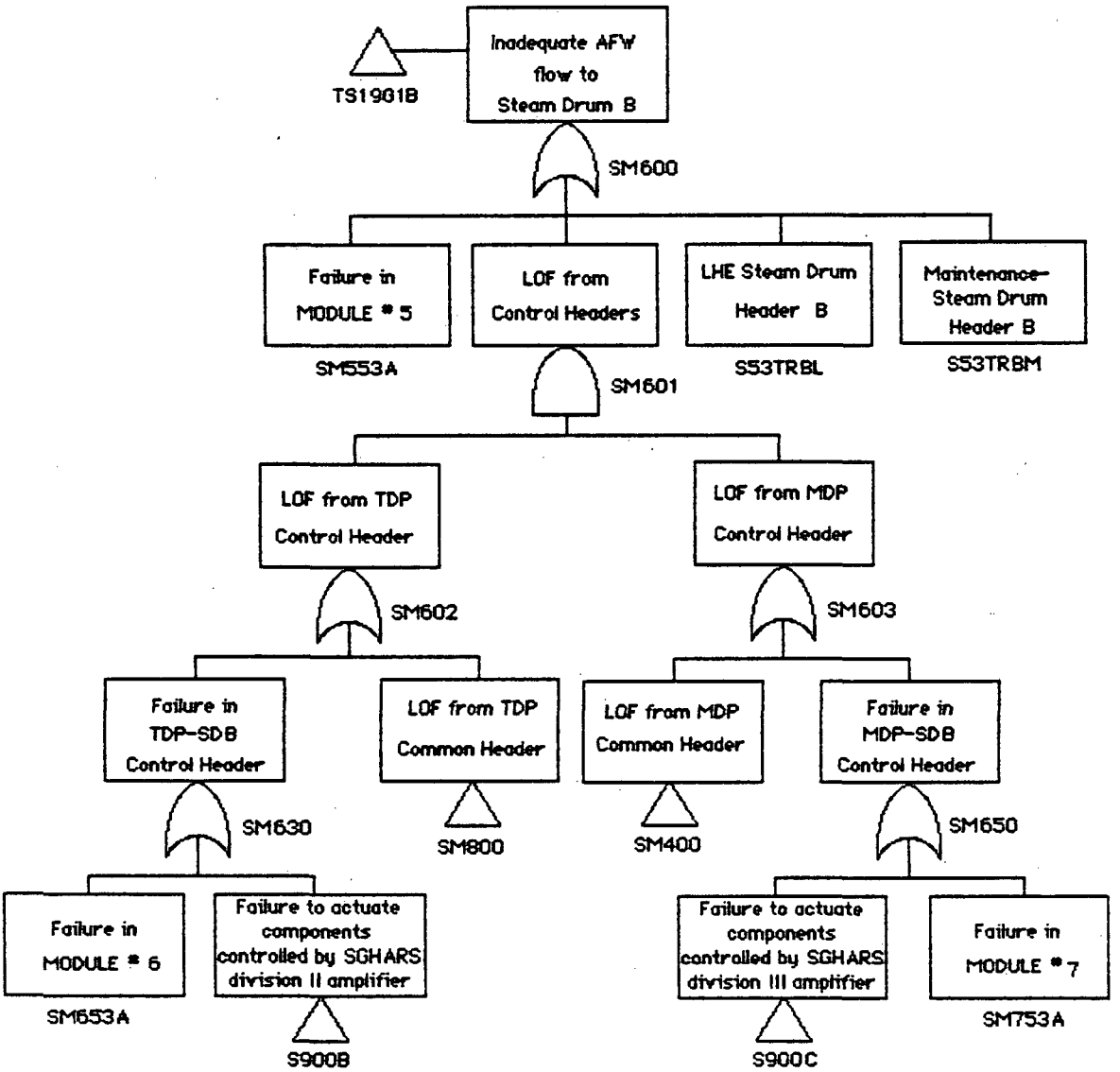
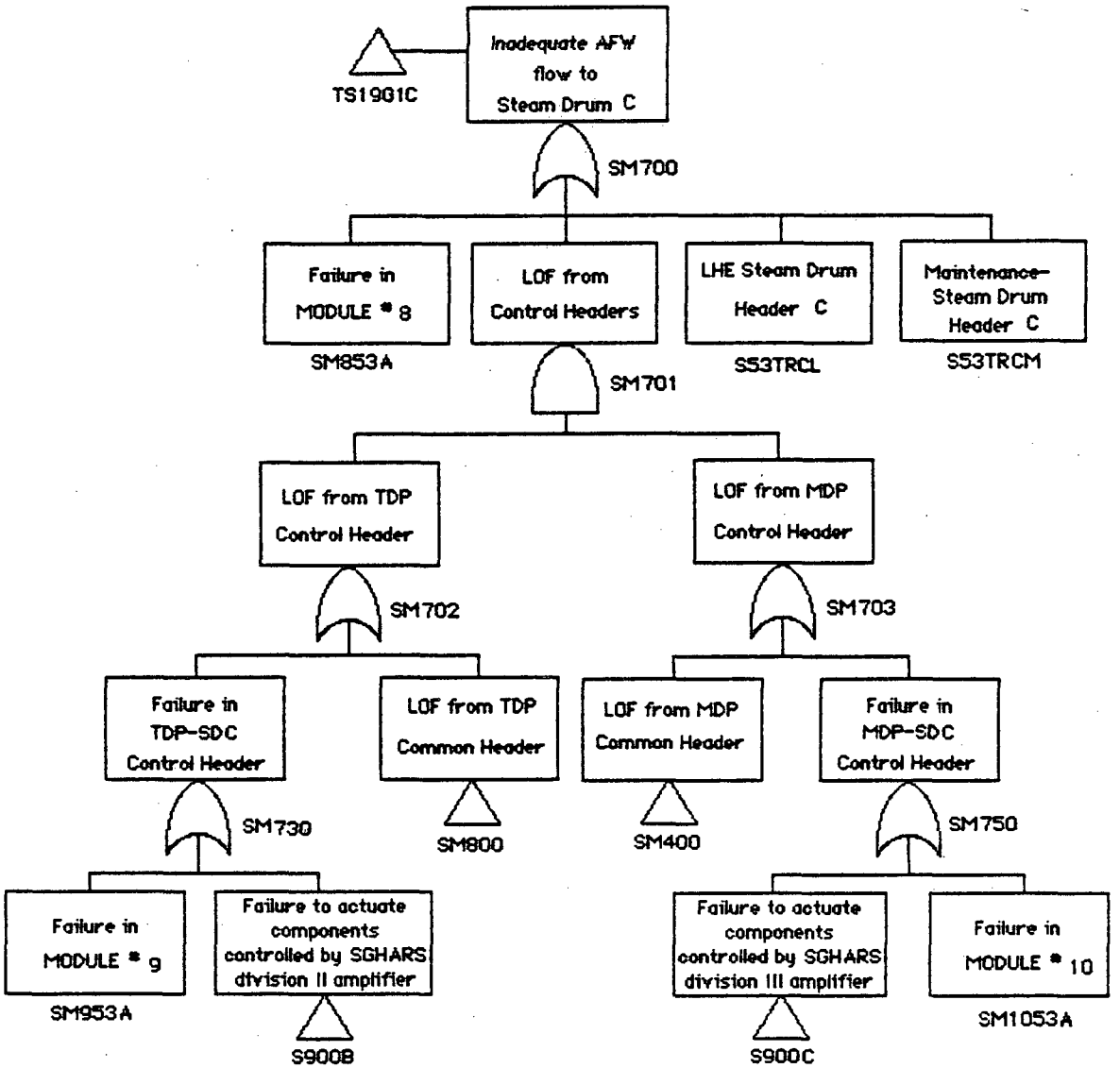
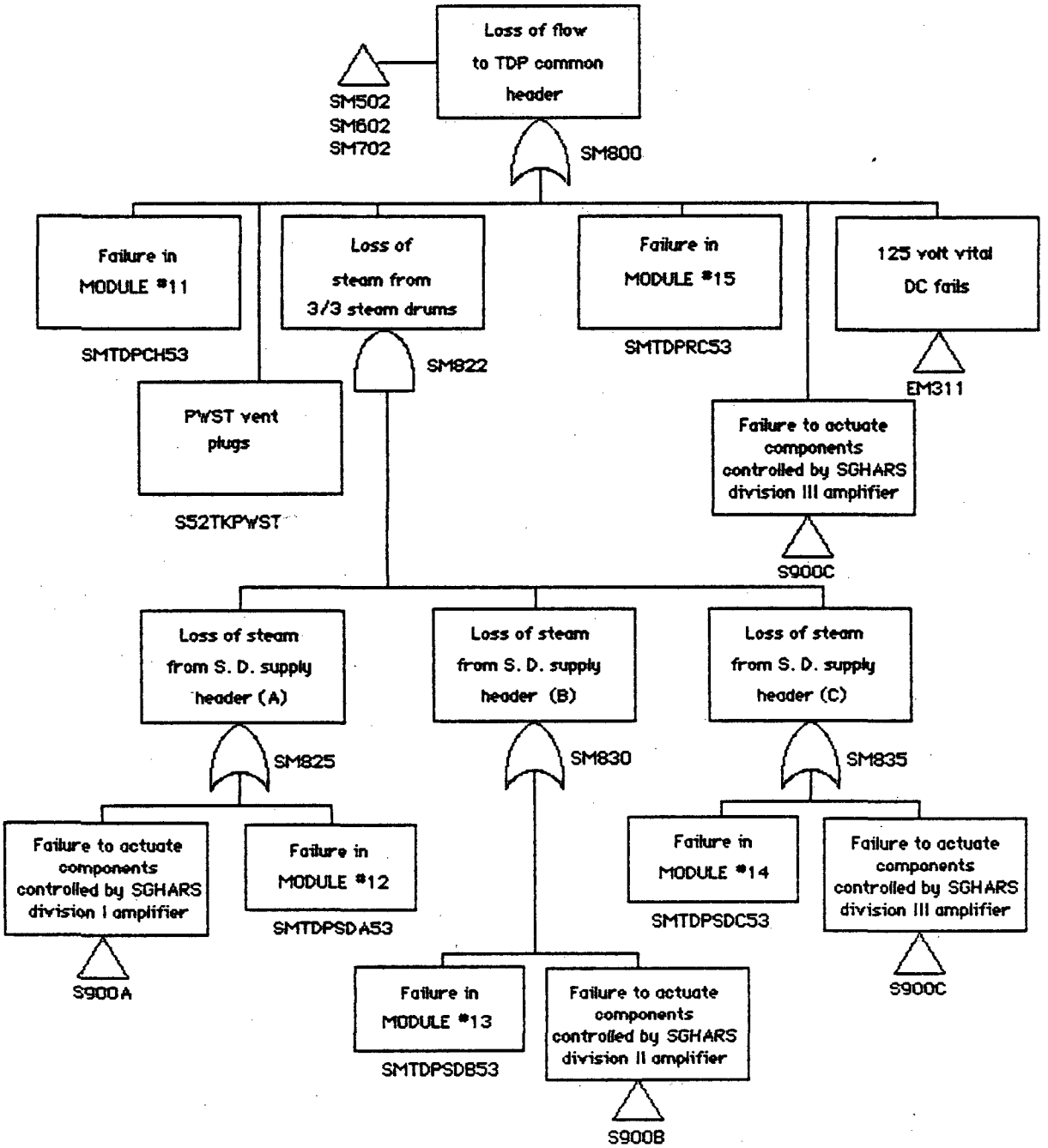
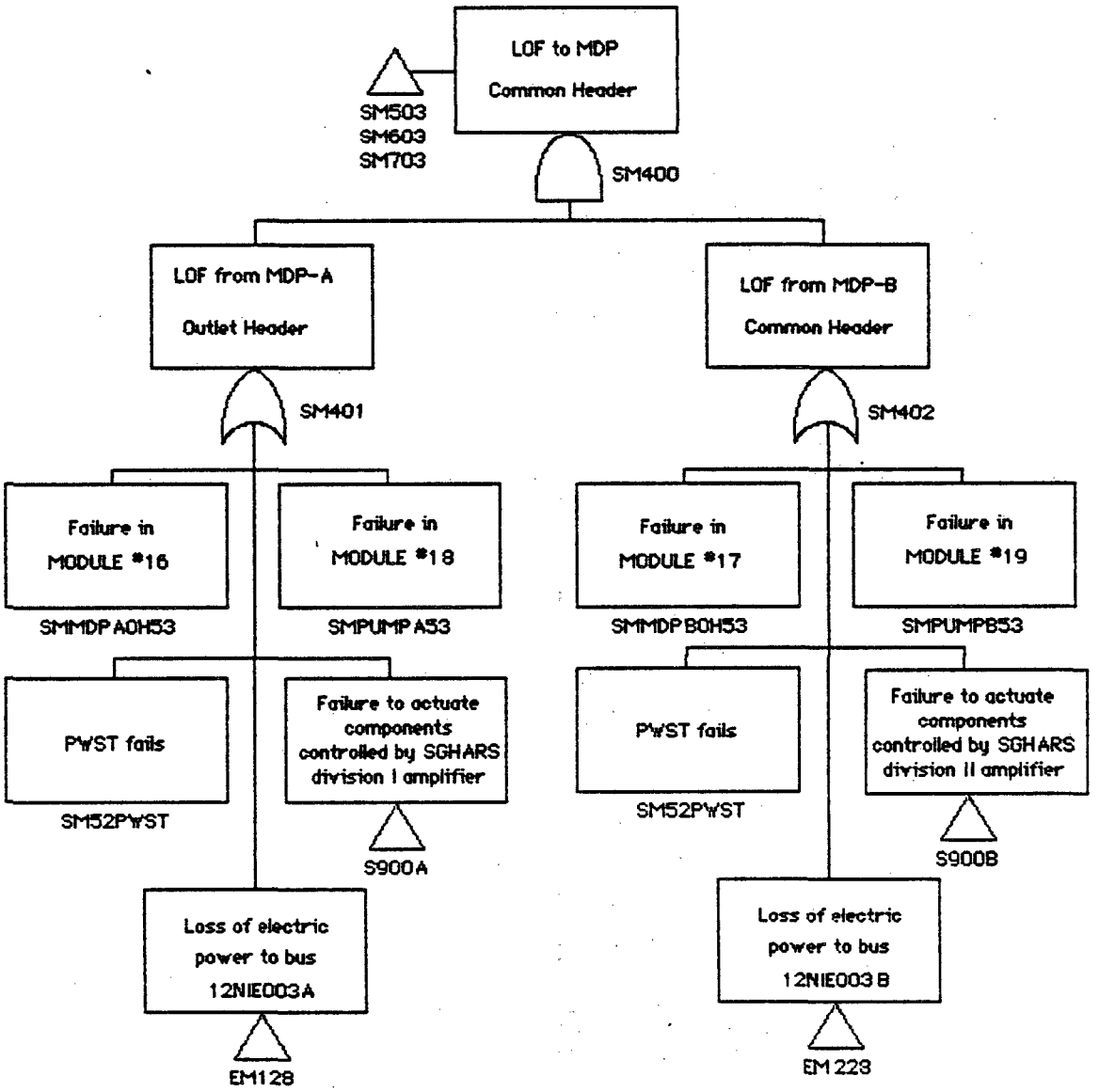


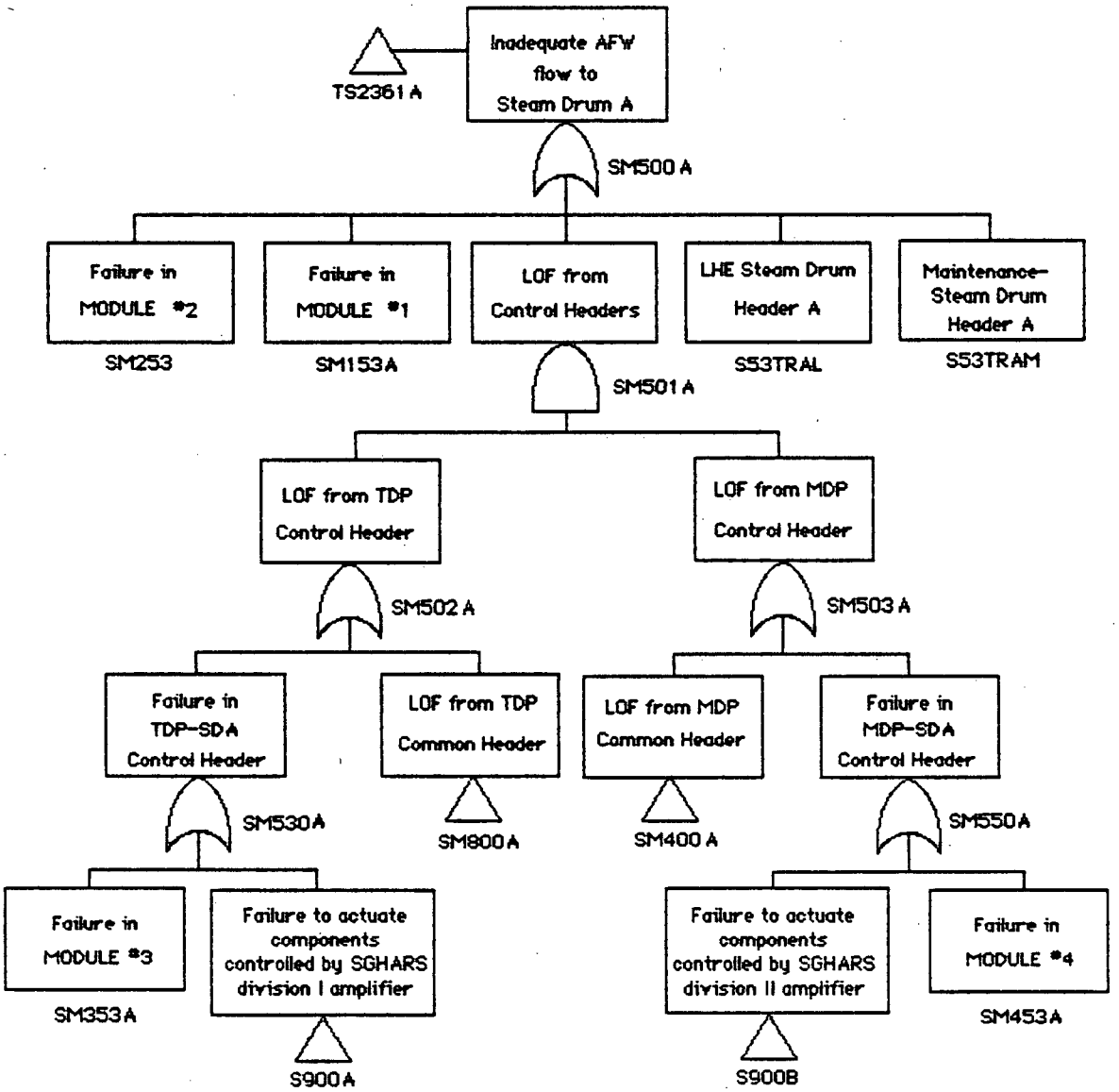
Figure A2-3. Auxiliary Feedwater Modularized Fault Tree.

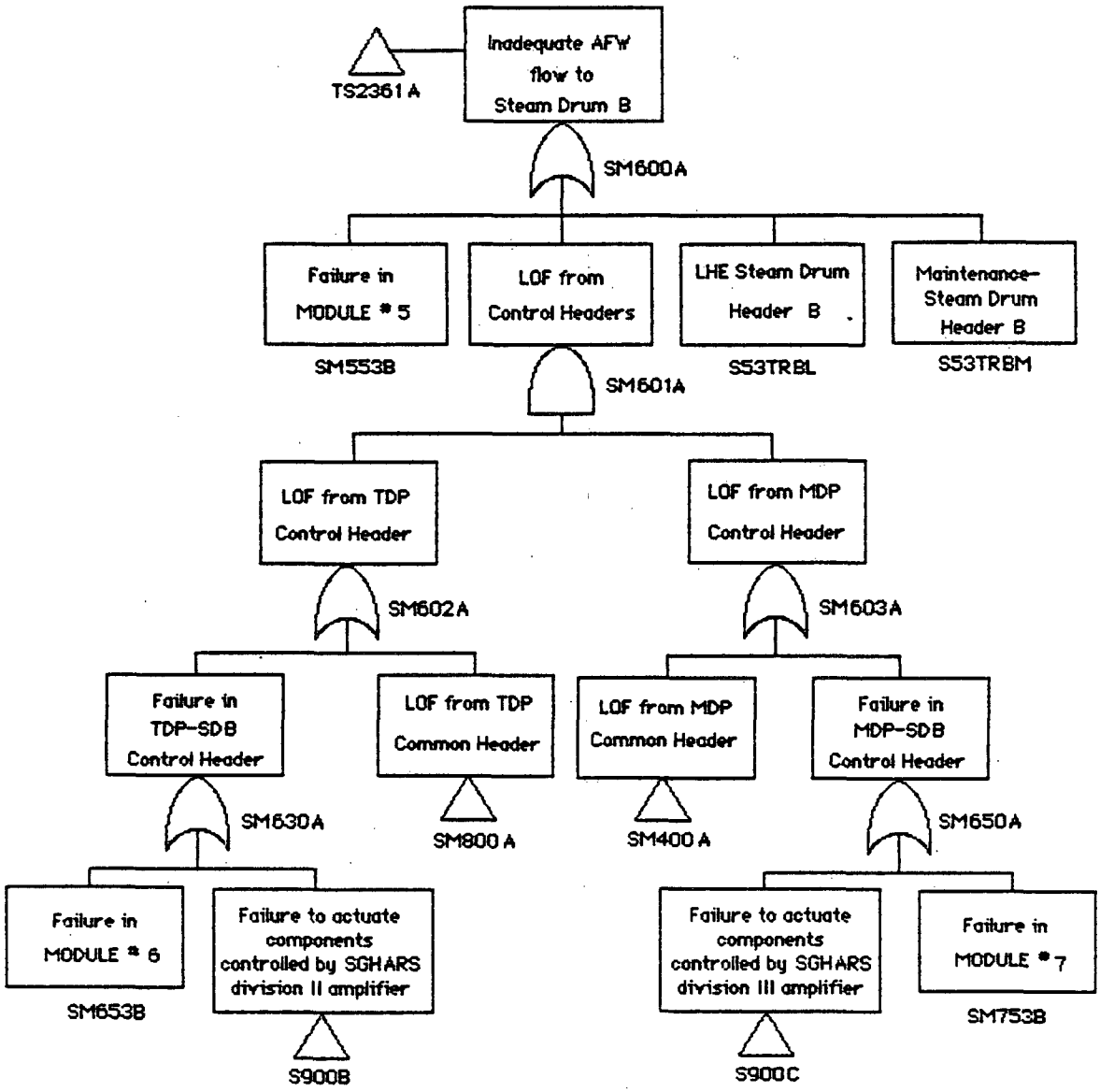

Inadequate AFW flow to Steam Drum B
TS1961B
SM600
Failure in MODULE # 5
SM553A
LOF from Control Headers
SM601
LHE Steam Drum Header B
S53TRBL
Maintenance-Steam Drum Header B
S53TRBM
LOF from TDP Control Header
SM602
LOF from MDP Control Header
SM603
Failure in TDP-SDB Control Header
SM630
LOF from TDP Common Header
SM800
LOF from MDP Common Header
SM400
Failure in MDP-SDB Control Header
SM650
Failure in MODULE # 6
SM653A
Failure to actuate components controlled by SGHARS division II amplifier
S900B
Failure to actuate components controlled by SGHARS division III amplifier
S900C
Failure in MODULE # 7
SM753A

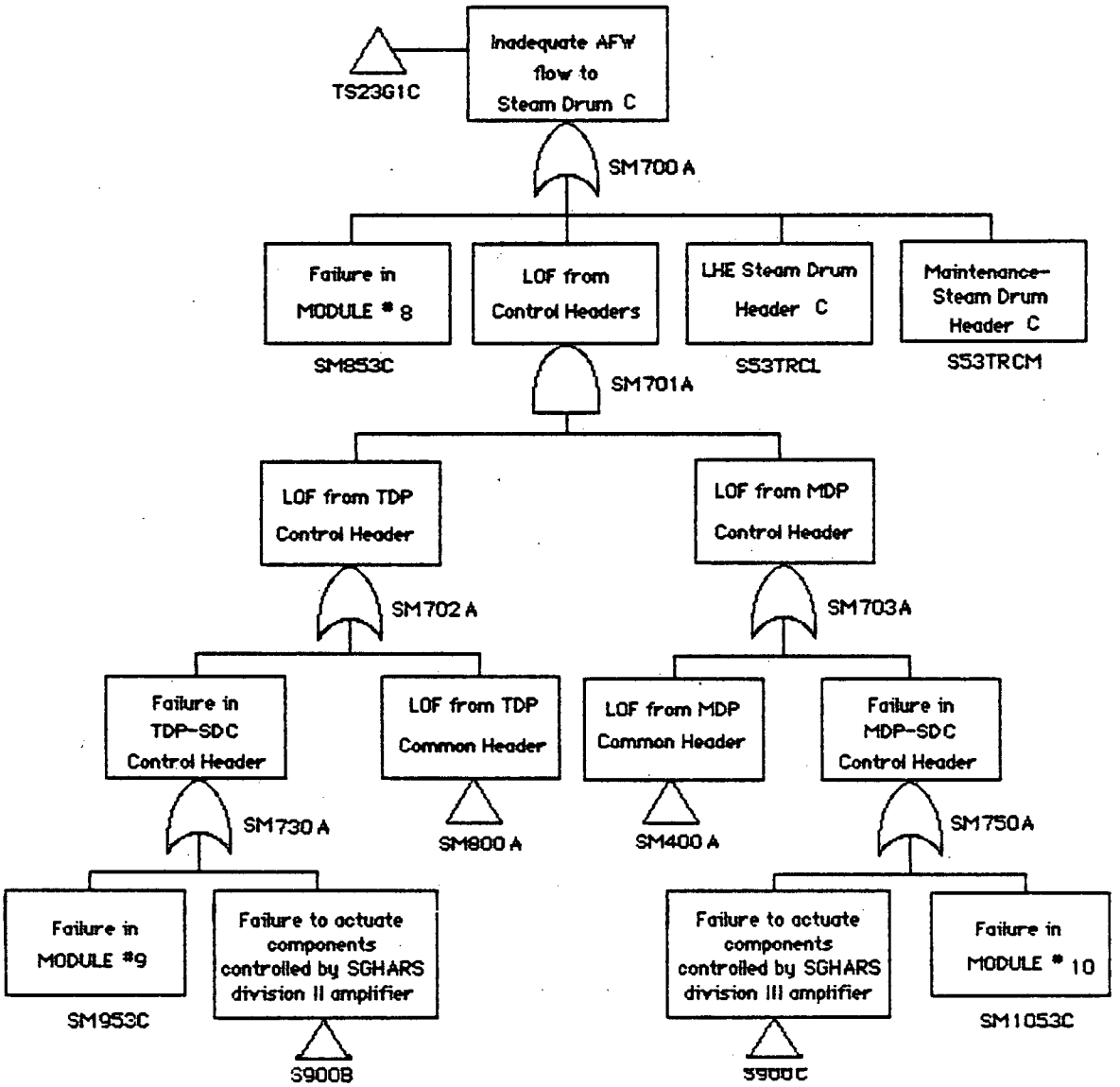

Inadequate AFW flow to Steam Drum C
TS1961C
SM700
Failure in MODULE # 8
SM853A
LOF from Control Headers
SM701
LHE Steam Drum Header C
S53TRCL
Maintenance-Steam Drum Header C
S53TRCM
LOF from TDP Control Header
SM702
LOF from MDP Control Header
SM703
Failure in TDP-SDC Control Header
SM730
LOF from TDP Common Header
SM800
LOF from MDP Common Header
SM400
Failure in MDP-SDC Control Header
SM750
Failure in MODULE # 9
SM953A
Failure to actuate components controlled by SGHARS division II amplifier
S900B
Failure to actuate components controlled by SGHARS division III amplifier
S900C
Failure in MODULE # 10
SM1053A

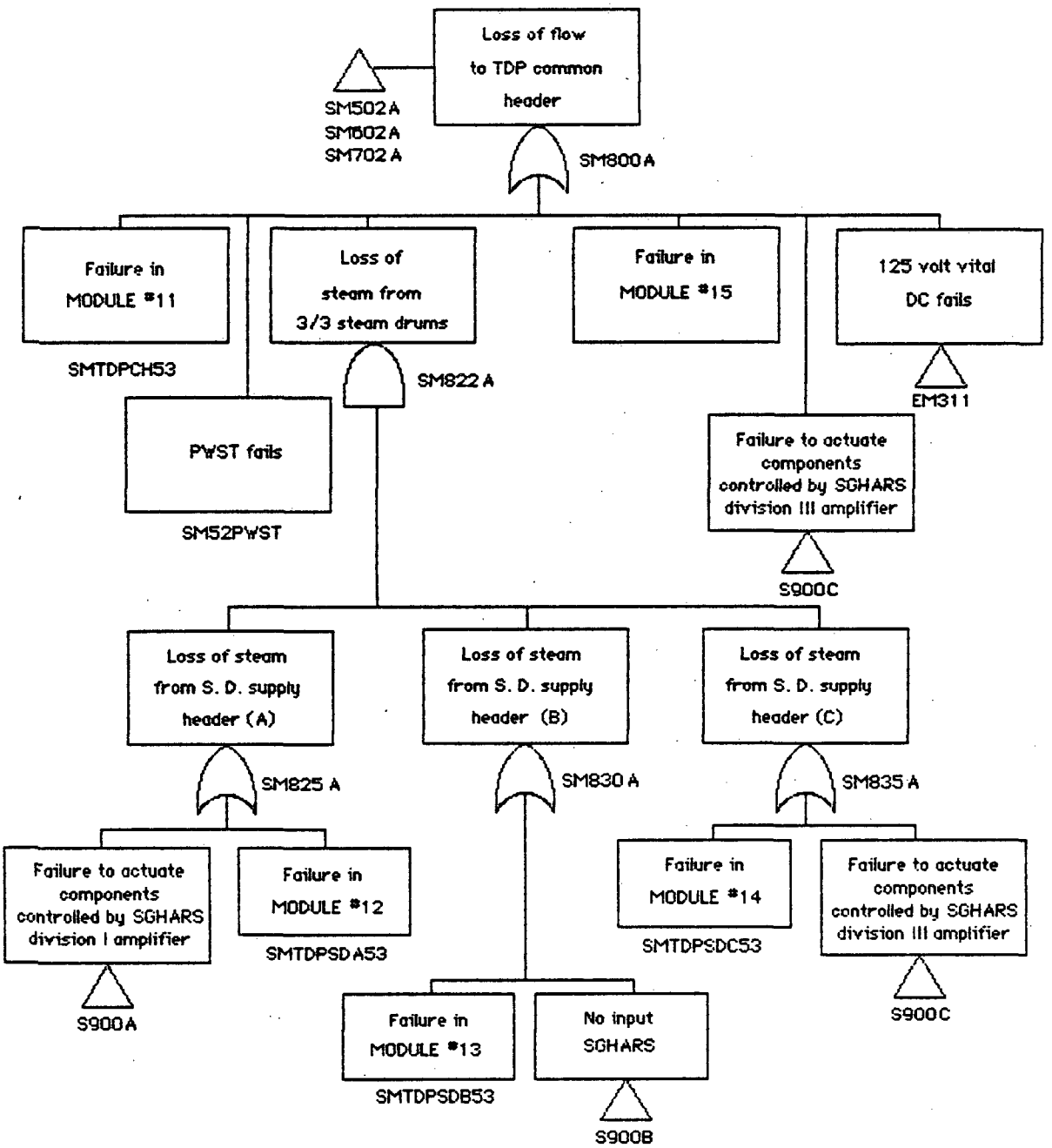

Loss of flow to TDP common header
SM502
SM602
SM702
SM800
Failure in MODULE #11
SMTDPCH53
Loss of steam from 3/3 steam drums
SM822
Failure in MODULE #15
SMTDPRC53
125 volt vital DC fails
EM311
PWST vent plugs
S52TKPWST
Failure to actuate components controlled by SGHARS division III amplifier
S900C
Loss of steam from S. D. supply header (A)
SM825
Loss of steam from S. D. supply header (B)
SM830
Loss of steam from S. D. supply header (C)
SM835
Failure to actuate components controlled by SGHARS division I amplifier
S900A
Failure in MODULE #12
SMTDPSDA53
Failure in MODULE #14
SMTDPSDC53
Failure to actuate components controlled by SGHARS division III amplifier
S900C
Failure in MODULE #13
SMTDPSDB53
Failure to actuate components controlled by SGHARS division II amplifier
S900B


LOF to MDP
Common Header
SM503
SM603
SM703
SM400
LOF from MDP-A
Outlet Header
LOF from MDP-B
Common Header
SM401
SM402
Failure in
MODULE #16
SMMDPA0H53
Failure in
MODULE #18
SMPUMPA53
Failure in
MODULE #17
SMMDPB0H53
Failure in
MODULE #19
SMPUMPB53
PWST fails
SM52PWST
Failure to actuate
components
controlled by SGHARS
division I amplifier
S900A
PWST fails
SM52PWST
Failure to actuate
components
controlled by SGHARS
division II amplifier
S900B
Loss of electric
power to bus
12NIE003A
EM128
Loss of electric
power to bus
12NIE003B
EM223


Inadequate AFW flow to Steam Drum A
TS2361 A
SM500 A
Failure in MODULE #2
SM253
Failure in MODULE #1
SM153 A
LOF from Control Headers
LHE Steam Drum Header A
S53TRAL
Maintenance-Steam Drum Header A
S53TRAM
SM501 A
LOF from TDP Control Header
LOF from MDP Control Header
SM502 A
SM503 A
Failure in TDP-SD A Control Header
LOF from TDP Common Header
LOF from MDP Common Header
Failure in MDP-SD A Control Header
SM530 A
SM800 A
SM400 A
SM550 A
Failure in MODULE #3
SM353 A
Failure to actuate components controlled by SGHARS division I amplifier
S900 A
Failure to actuate components controlled by SGHARS division II amplifier
S900B
Failure in MODULE #4
SM453 A


Inadequate AFW flow to Steam Drum B
TS2361A
SM600A
Failure in MODULE # 5
SM553B
LOF from Control Headers
SM601A
LHE Steam Drum Header B
S53TRBL
Maintenance-Steam Drum Header B
S53TRBM
LOF from TDP Control Header
SM602A
LOF from MDP Control Header
SM603A
Failure in TDP-SDB Control Header
SM630A
LOF from TDP Common Header
SM800A
LOF from MDP Common Header
SM400A
Failure in MDP-SDB Control Header
SM650A
Failure in MODULE # 6
SM653B
Failure to actuate components controlled by SGHARS division II amplifier
S900B
Failure to actuate components controlled by SGHARS division III amplifier
S900C
Failure in MODULE # 7
SM753B


TS23G1C
Inadequate AFW flow to Steam Drum C
SM700 A
Failure in MODULE # 8
SM853C
LOF from Control Headers
SM701A
LHE Steam Drum Header C
S53TRCL
Maintenance-Steam Drum Header C
S53TRCM
LOF from TDP Control Header
SM702 A
LOF from MDP Control Header
SM703A
Failure in TDP-SDC Control Header
SM730 A
LOF from TDP Common Header
SM800 A
LOF from MDP Common Header
SM400 A
Failure in MDP-SDC Control Header
SM750A
Failure in MODULE #9
SM953C
Failure to actuate components controlled by SGHARS division II amplifier
S900B
Failure to actuate components controlled by SGHARS division III amplifier
S900C
Failure in MODULE # 10
SM1053C


Loss of flow to TDP common header
SM502A
SM602A
SM702A
SM800A
Failure in MODULE #11
SMTDPCH53
Loss of steam from 3/3 steam drums
SM822A
Failure in MODULE #15
125 volt vital DC fails
EM311
PWST fails
SM52PWST
Failure to actuate components controlled by SGHARS division III amplifier
S900C
Loss of steam from S. D. supply header (A)
SM825A
Loss of steam from S. D. supply header (B)
SM830A
Loss of steam from S. D. supply header (C)
SM835A
Failure to actuate components controlled by SGHARS division I amplifier
S900A
Failure in MODULE #12
SMTDPSDA53
Failure in MODULE #13
SMTDPSDB53
No input SGHARS
S900B
Failure in MODULE #14
SMTDPSDC53
Failure to actuate components controlled by SGHARS division III amplifier
S900C

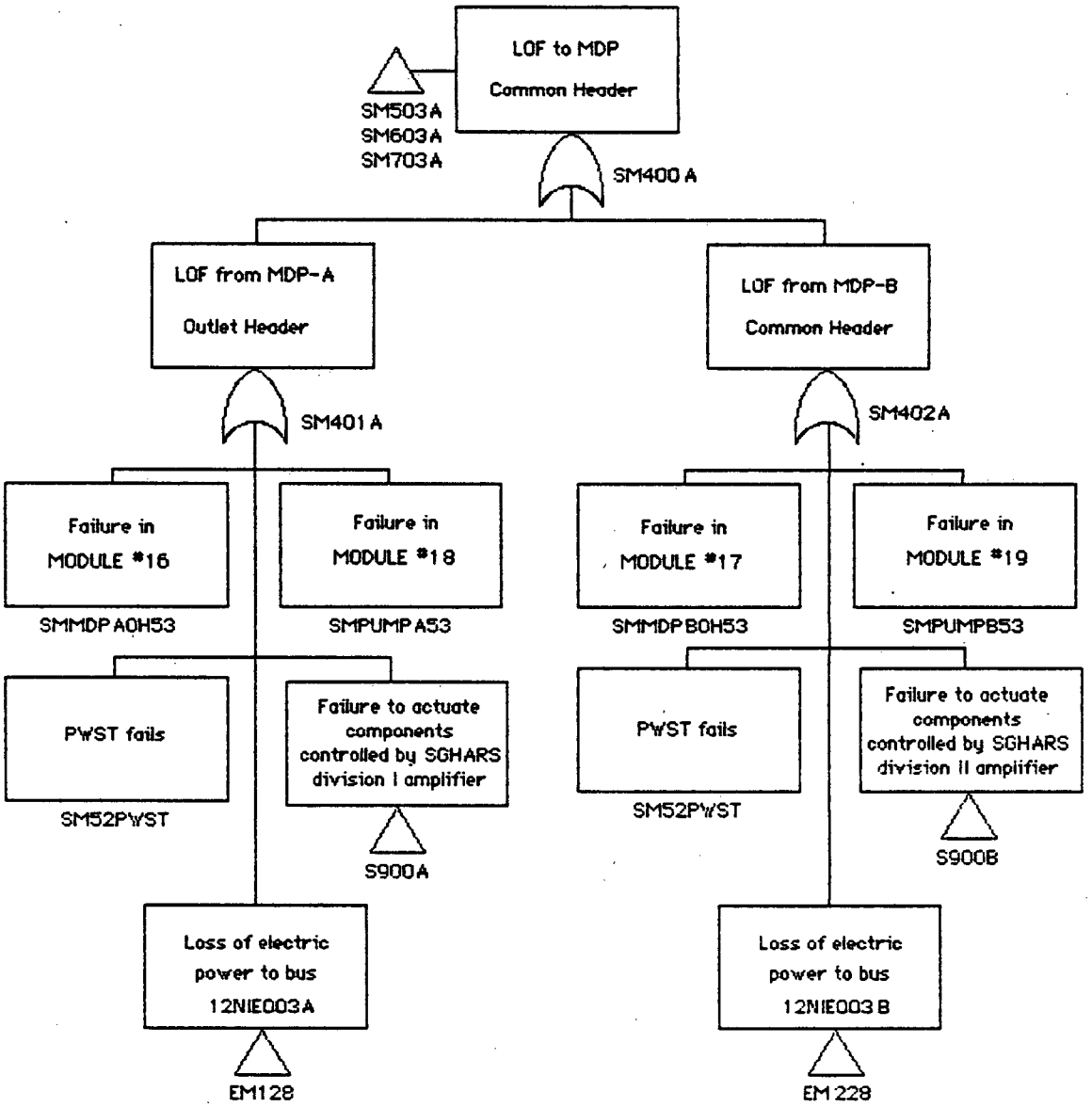

LOF to MDP
Common Header
SM503A
SM603A
SM703A
SM400 A
LOF from MDP-A
Outlet Header
LOF from MDP-B
Common Header
SM401 A
SM402A
Failure in
MODULE #16
SMMDPA0H53
Failure in
MODULE #18
SMPUMPA53
Failure in
MODULE #17
SMMDPB0H53
Failure in
MODULE #19
SMPUMPB53
PWST fails
SM52PWST
Failure to actuate
components
controlled by SGHARS
division I amplifier
S900A
PWST fails
SM52PWST
Failure to actuate
components
controlled by SGHARS
division II amplifier
S900B
Loss of electric
power to bus
12NIE003A
EM128
Loss of electric
power to bus
12NIE003 B
EM228

Table A2-1

DATA TABLE

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SM153A	Steam Drum A Inlet Header Failure	2.38-4			TAB-OR
S53XV201AT		3.64-5	1.0-7	364	
S52CV105AO		1.0-4		-	
S52CV105AT		9.2-7	2.3-7	4	
S52CV105BO		1.0-4		-	
S52CV105BT		9.2-7	2.3-7	4	
S52PI1090Y		ε			
SM253	Test Line Failure	1.32-9			S52EV155AT * S52EV155T + S52PI2083Y
S52EV155AT		3.64-5	1.0-7	364	
S52EV155BT		3.64-5	1.0-7	364	
S52PI2083Y		ε			
SM353A	Turbine Driven Pump Control Header Failure	1.07-3			TAB-OR
S52XV101AT		3.64-5	1.0-7	364	
S52EV104AT		4.0-7	1.0-7	4	
S52EV103AT		4.0-7	1.0-7	4	
S52EV103AI		1.12-6	2.8-7	4	
S52XV125AT		3.64-5	1.0-7	364	
S52EV103AO		1.0-3			
S52EV104F		4.0-6	1.0-6	4	
SM453A	Motor Driven Pump Control Header Failure	2.07-3			TAB-OR
S52XV101BT		3.64-5	1.0-7	364	
S52EV104BT		4.0-7	1.0-7	4	
S52EV104BO		1.0-3		-	
S52EV103BT		4.0-7	1.0-7	4	
S52EV103BI		1.12-6	2.8-7	4	
S52XV125BT		3.64-5	1.0-7	364	
S52EV104BF		4.0-6	1.0-6	4	
S52EV103BO		1.0-3	1.0-3	-	

Table A2-1 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SM853A	Steam Drum C Inlet	2.38-4			TAB-OR
S53XV201CT	Header Failure	3.64-5	1.0-7	364	
S52CV105EO		1.0-4		-	
S52CV105ET		9.2-7	2.3-7	4	
S52CV105FO		1.0-4		-	
S52CV105FT		9.2-7	2.3-7	4	
S52PI3090Y		ε	-	-	
SM953A	Steam Drum C Control	1.07-3			TAB-OR
S52XV101ET	Header Failure	3.64-5	1.0-7	364	
S52EV104ET	Turbine Driven Pump	4.0-7	1.0-7	4	
S52EV104EF		4.0-6	1.0-6	4	
S52EV103ET		4.0-7	1.0-7	4	
S52EV103EI		1.12-6	2.8-7	4	
S52XV125ET		3.64-5	1.0-7	364	
S52EV103EO		1.0-3	1.0-3	-	
SM1053A	Steam Drum C Control	1.07-3			TAB-OR
S52XV101FT	Header Failure Motor	3.64-5	1.0-7	364	
S52EV104FT	Driven Pump	4.0-7	1.0-7	4	
S52EV104FF		4.0-6	1.0-6	4	
S52EV103FT		4.0-7	1.0-7	4	
S52EV103FI		1.12-6	2.8-7	4	
S52XV125FT		3.64-5	1.0-7	364	
S52EV103FO		1.0-3	1.0-3	-	

Table A2-1 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SM553A	Steam Drum B Inlet	2.38-4			TAB-OR
S53XV201BT	Header Failure	3.64-5	1.0-7	364	
S52CV105CO		1.0-4		-	
S52CV105CT		9.2-7	2.3-7	4	
S52CV105DO		1.0-4		-	
S52CV105DT		9.2-7	2.3-7	4	
S52PI2090Y		ε			
SM653A	Steam Drum B Control	1.07-3			TAB-OR
S52XV101CT	Header Failure	3.64-5	1.0-7	364	
S52EV104CT	Turbine Driven Pump	4.0-7	1.0-7	4	
S52EV104CF		4.0-6	1.0-6	4	
S52EV103CT		4.0-7	1.0-7	4	
S52EV103CI		1.12-6	2.8-7	4	
S52XV125CT		3.64-5	1.0-7	364	
S52EV103CO		1.0-3	1.0-3	-	
SM753A	Steam Drum B Control	1.07-3			TAB-OR
S52XV101DT	Header Failure Motor	3.64-5	1.0-7	364	
S52EV104DT	Driven Pump	4.0-7	1.0-7	-	
S52EV104DF		4.0-6	1.0-6	4	
S52EV103DT		4.0-7	1.0-7	4	
S52EV103DI		1.12-6	2.8-7	4	
S52XV125DT		3.64-5	1.0-7	364	
S52EV103DO		1.0-3	-	-	

Table A2-1 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMTDPSDC53		1.17-3			TAB-OR
S53XV200CT	Loss of Steam to	3.64-5	1.0-7	364	
S52EV118CT	Turbine Driven Pump	4.0-7	1.0-7	4	
S52EV118CO	From Steam Drum C	1.0-3	1.0-3	-	
S52XV120CT		3.64-5	1.0-7	364	
S52CV119CO		1.0-4	1.0-4	-	
S52CV119CT		9.2-7	2.3-7	4	
S52PI3110Y		ε		-	
SMTDPRC53	Loss of TDP Recir-	1.17-3			TAB-OR
S52XV139AT	culation Line	3.64-5	1.0-7	364	
S52CV152AT		9.2-7	2.3-7	4	
S52CV152AO		1.0-4	1.0-4	-	
S52EV108AT		4.0-7	1.0-7	4	
S52EV108AO		1.0-3	1.0-3	-	
S52XV124T		3.64-5	1.0-7	364	
S52PI2031Y		ε		-	
SMMDPAOH53	Loss of Flow From	1.37-4			TAB-OR
S52CV107BT	MDP-A Outlet Header	9.2-7	2.3-7	4	
S52CV107BO		1.0-4	1.0-4	-	
S52XV106AT		3.64-5	1.0-7	364	
S52PI2052Y		ε		-	

Table A2-1 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMTDPCH53	Flow Block or Turbine Failures in TDP Control Header	5.14-3			TAB-OR
S52CV107AO		1.0-4	1.0-4	-	
S52CV107AT		9.2-7	2.3-7	4	
S52PTR		4.0-5	1.0-5	4	
S52PTS		4.0-3	4.0-3	-	
S52EV121T		4.0-7	1.0-7	4	
S52EV121O		1.0-3	1.0-3	-	
S52PI2051Y		ε		-	
S52PI2113Y		ε		-	
SMTDPSDA53	Loss of Steam to Turbine Driven Pump From Steam Drum A	1.17-3			TAB-OR
S53XV200AT		3.64-5	1.0-7	364	
S52EV118AT		4.0-7	1.0-7	4	
S52EV118AO		1.0-3	1.0-3	-	
S52XV120AT		3.64-5	1.0-7	364	
S52CV119AO		1.0-4	1.0-4	-	
S52CV119AT		9.2-7	2.3-7	4	
S52PI1110Y		ε		-	
SMTDPSDB53	Loss of Steam to Turbine Driven Pump From Steam Drum B	1.17-3			TAB-OR
S53XV200BT		3.64-5	1.0-7	364	
S52EV118BT		4.0-7	1.0-7	4	
S52EV118BO		1.0-3	1.0-3	-	
S52XV120BT		3.64-5	1.0-7	364	
S52CV119BO		1.0-4	1.0-4	-	
S52CV119BT		9.2-7	2.3-7	4	
S52PI2110Y		ε		-	

Table A2-1 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMMDPBOH53	Loss of Flow From MDP-B Outlet Header	1.37-4			TAB-OR
S52CV107CT		9.2-7	2.3-7	4	
S52CV107CO		1.0-4	1.0-4	-	
S52XV106BT		3.64-5	1.0-7	364	
S52PI2053Y		ε			
SMPUMPA53	Pump, Valve, and Electrical Faults Which Fail Motor Driven Pump A	5.21-3			TAB-OR
S52PMAS		4.0-3	4.0-3	-	
S52PMAR		4.0-5	1.0-5	4	
S52XV113AT		3.64-5	1.0-7	364	
S52XV139BT		3.64-5	1.0-7	364	
S52CV152BT		9.2-7	2.3-7	4	
S52CV152BO		1.0-4	1.0-4	-	
S52EV108BT		4.0-7	1.0-7	4	
S52EV108BO		1.0-3	1.0-3	-	
S52PI2032Y		ε		-	
SMPUMPB53	Pump, Valve, and Electrical Faults Which Fail Motor Driven Pump B	5.21-3			TAB-OR
S52PMBS		4.0-3	4.0-3	-	
S52PMBR		4.0-5	1.0-5	4	
S52XV113BT		3.64-5	1.0-7	364	
S52XV139CT		3.64-5	1.0-7	364	
S52CV152CT		9.2-7	2.3-7	4	
S52CV152CO		1.0-4	1.0-4	-	
S52EV108CT		4.0-7	1.0-7	4	
S52EV108BO		1.0-3	1.0-3	-	
S52PI2032Y		ε		-	

Table A2-1 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
S52TRAM	Main. Train A	1.45-3	7.6-5	19	
S52TRBM	Main. Train B	1.45-3	7.6-5	19	
S52TRCM	Main. Train C	1.45-3	7.6-5	19	
S52TRAL	LHE Train A	1.0-3			
S52TRBL	LHE Train B	1.0-3			
S52TRCL	LHE Train C	1.0-3			
S25TKPWST		1.02-4			TAB-OR
S52TKPWSTP	PWST Vent Plugs	1.02-4	2.8-7	364	
S52TKPWSTY	PWST Rupture	ε	-	-	

Appendix A, Section 3

DIRECT HEAT REMOVAL SERVICE

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

Appendix A, Section 3

DIRECT HEAT REMOVAL SERVICE (DHRS)

A3.1 SYSTEM DESCRIPTION

A3.1.1 Function

DHRS is an emergency mode of operation of the auxiliary liquid metal system (ALMS). DHRS uses part of the primary sodium storage and processing system (PSSPS) and part of the ex-vessel storage processing system (EVSPS). The function of DHRS is to remove decay heat directly from the reactor primary system in case of a loss of all secondary heat removal paths. Its initiation is per emergency operating instruction, EOI 10-23.

A3.1.2 Design

DHRS is not a system itself but uses two extant subsystems of ALMS by opening what are referred to as the DHRS "crossover" lines. The PSSPS used in DHRS operation includes redundancy of active components and the overflow heat exchanger (OHX) that transfer heat from the sodium in the PSSPS to the NaK in the DHRS crossover piping. The DHRS crossover piping is routed into two headers through active valves, one each to a normally operating EVSPS NaK loop. The NaK loops then each remove heat by exhausting to airblast heat exchangers (ABHXs).

Figure A3-1 is a schematic of the PSSPS (adapted from Ref. 3) and Figure A3-2 is a schematic of the EVSPS (adapted from Ref. 4). The PSSPS is completely contained in the reactor containment building. The EVSPS is housed in the reactor service building. Major sodium-bearing components are housed in inerted (N_2) cells, and major NaK components are housed in cells cooled by HVAC.

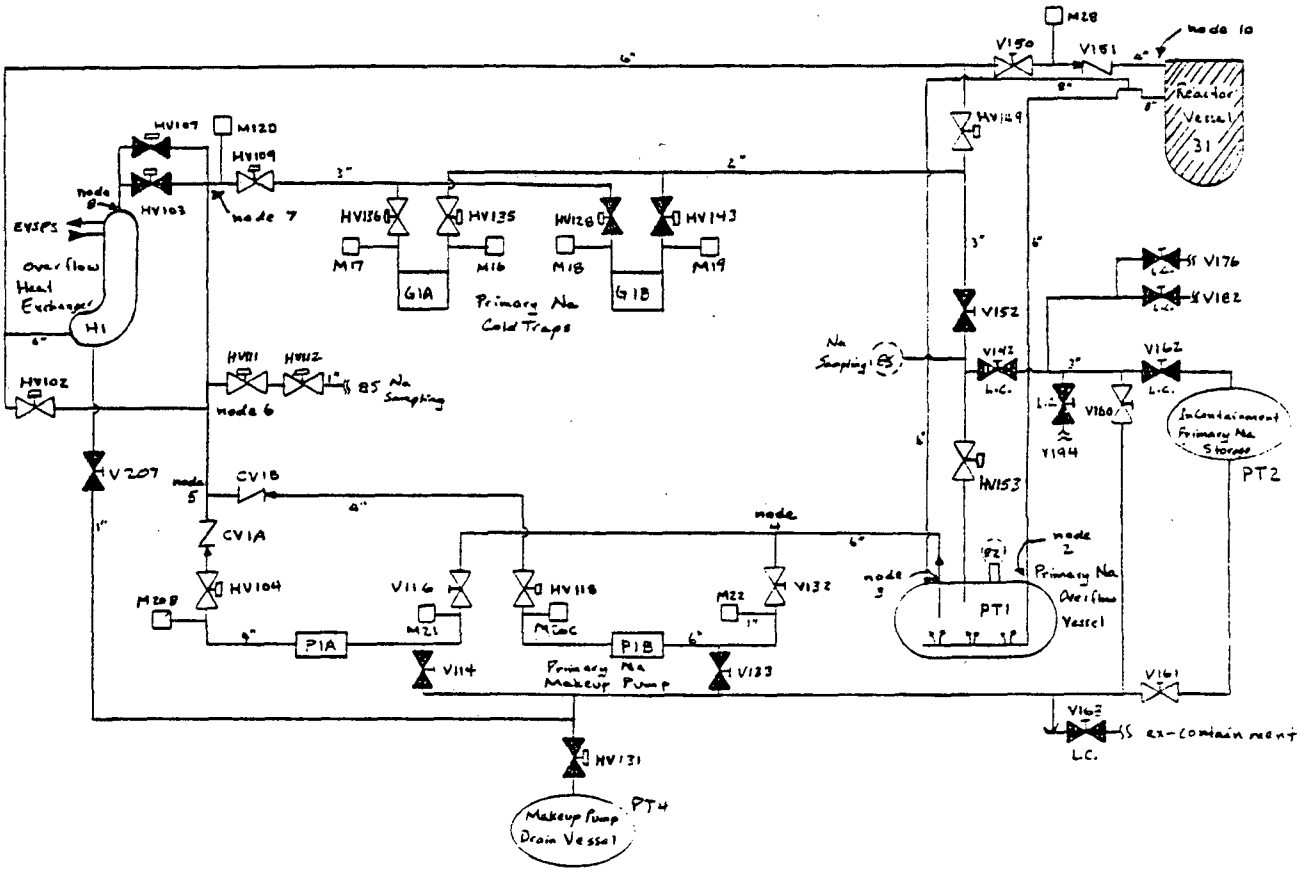


Figure A3-1. Primary Sodium Storage and Processing System.

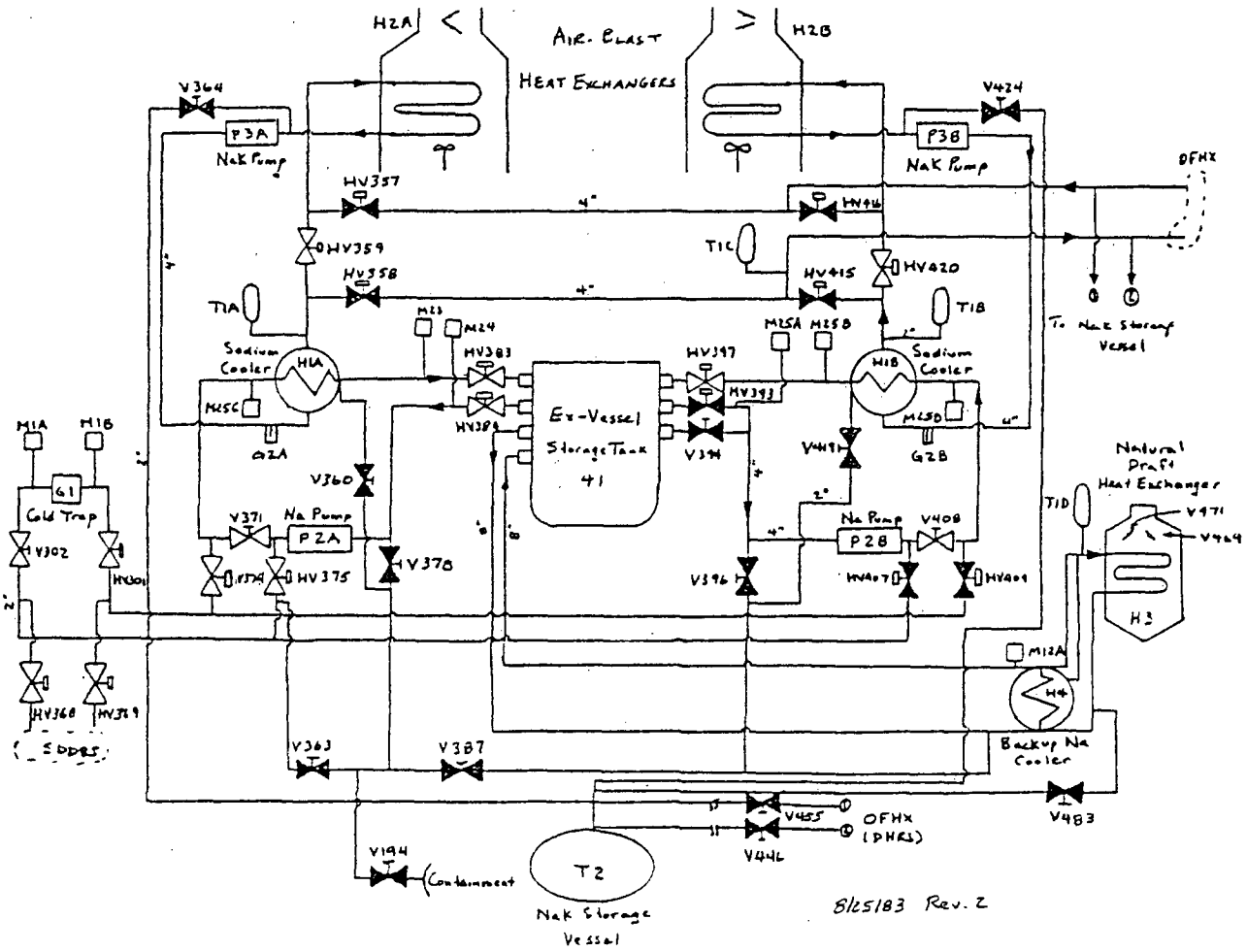


Figure A3-2. Ex-Vessel Storage Processing System.

Two electromagnetic (EM) pumps, each designed to provide 400 gpm sodium flow, provide 160 gpm (total) makeup flow through the PSSPS under normal conditions. One EM pump is located in each NaK loop and has a 400 gpm capacity.

An overflow heat exchanger in the PSSPS is isolated during normal operation but has redundant active inlet valves. The OHX is a shell and tube type, 300 stainless steel heat exchanger.

PSSPS piping also includes paths to the impurity monitoring and analysis system and to the ALMS cold trapping. There is a path from the primary sodium overflow tank to the reactor vessel inlet header that is used for header drainage. All other paths that exit normal PSSPS piping are closed by at least two normally closed valves.

The EVSPS NaK loops physically include the two sodium coolers that are used in EVST cooling but are not used for DHRS cooling. Each NaK loop transfers heat to an airblast heat exchanger, which supplies forced air cooling from the HVAC system, by a fan through an upper and lower damper and an upper and lower tornado* damper.

A3.1.3 Interface with Other Systems

DHRS, that is, ALMS, is connected to the reactor vessel and requires sodium flow from at least one PHTS pony motor. DHRS can be controlled from the main control room or from local panels in the RSB.

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*As their name implies, the tornado dampers are shut upon "tornado warning" per EOI 81-16 so that the ABHX structure is protected from tornados.

A3.1.3.1 Compressed gas. Service air supplies two accumulators for actuation of the DHRS crossover valves, one for the "357 set," HV357, HV358, HV359; and one for the "415 set," HV415, HV416, and HV420.

A3.1.3.2 HVAC. The nuclear island HVAC system provides cell cooling in cells 352A and 353A. HVAC in cell 352A cools NaK pump 3A and provides air supply for ABHX A. HVAC in cell 353A cools NaK pump 3B and provides air supply for ABHX B.

A3.1.3.3 Gas cooling. The recirculating gas cooling system provides nitrogen to cool the primary makeup pumps and their cells.

A3.1.3.4 Preheating. The equipment electrical heating system provides preheat, particularly to the OHX, the DHRS crossover components, and the NaK loop in the EVSPS.

A3.1.3.5 Inert gas. Argon cover gas is provided to all sodium and NaK vessels, expansion tanks, and freeze vents, but the ALMS/cover gas interface is usually isolated by a closed valve. Cover gas is not used in an active control capacity. The inert gas system also provides N_2 actuation to all other pneumatic valves. Valves HV104, HV118, HV103, and HV107 also are supplied by a common accumulator, T005.

A3.1.3.6 Electric power. All A pumps are supplied electric power from bus 12NIE041A; B pumps from 12NIE041B. The airblast heat exchangers (A and B) take power from buses 12NIE032 A and B.

A3.1.4 Operation

A3.1.4.1 Normal. The PSSPS has a normal sodium makeup function. The primary makeup pumps will be running normally, providing 160 gpm flow from the overflow vessel; 10 gpm is diverted for impurity analysis, 60 gpm is sent through one of two cold traps to the reactor, and 90 gpm returns to the reactor, bypassing the OHX (which is isolated).

One of the two EVSPS NaK loops will normally be used in removing heat from the EVST. The second NaK loop will circulate 10 gpm normally, when the loop is in standby. The DHRS crossover equipment is isolated but preheated to prevent NaK freezing.

A3.1.4.2 Emergency. The emergency operation of ALMS/DHRS is as a backup to secondary heat removal through the PHTS, IHTS, and SGS. DHRS is manually initiated, although the operator can use a sequencer in the main control room (MCR) to align and start the active DHRS components. Active components can be initiated from local motor control centers, and all valves have accessible reach rods or handwheels.

DHRS needs one of two full makeup pump's sodium flow and one of two NaK/ABHX cooling loops. By using active valves, the pumps or ABHXs can be switched as needed (EVST cooling can be postponed or switched to the backup circuit).

The assumed initial valve alignment can be inferred from valve status shown in Figures A3-1 and A3-2.

A3.1.5 Test and Operation Requirements

Alarmed instrumentation indicates loss of preheating in DHRS components. Testing in an EVSPS NaK loop must comply with heatup allowance requirements for the EVST, e.g., 5 hr to reach 775°F with no heat rejection and with an EVST load of 1800 kW. Testing in any other portion of the DHRS components cannot disable the components. It is assumed for this analysis that DHRS must always be available at power operation.*

A3.1.6 Maintenance Requirements

A3.1.6.1 Corrective maintenance. Loss of DHRS capability does not require plant shutdown if three full HTS loops are available (p. 6-4, Ref. 1). However, maintenance of PSSPS components would generally require containment building access.

Thus it is assumed for this analysis that any failure or other maintenance requirement in PSSPS requires plant shutdown.** EVSPS maintenance must be made to comply with the 5-hr EVST cooling requirement.

A3.1.6.2 Preventive maintenance. There is no scheduled maintenance on the EVSPS during the demand for EVST cooling or DHRS (p. 6-6, Ref. 1). It is assumed that no scheduled maintenance is performed on the PSSPS during reactor operation.

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*Reference 2 lists a "hold" on the number of hours within which DHRS must be able to provide heat removal.

**Actually a time allowance before shutdown would have been specified but was not available for this analysis.

A3.2 SYSTEM MODEL

A3.2.1 Modeling Assumptions

A. The portion of ALMS depicted in Figures A3-1 and A3-2 relevant to DHRS operation excludes the EVST and its two sodium loops but includes the two EVSPS sodium coolers.

B. Pipe break anywhere in the ALMS as shown in Figures A3-1 and A3-2 is assumed to fail DHRS.

C. The makeup circuit is assumed to be operating (both pumps) in the cold-trapping mode with the OHX isolated. EVSPS NaK loop A is assumed operating, and loop B is on standby. A trickle flow (10 gpm) in loop B is assumed when loop B is not in maintenance.

D. Diversion paths into the remainder of ALMS are isolated by at least two normally closed valves, and these are not modeled.

E. Diversion for impurity monitoring is not considered sufficient to fail DHRS.

F. Loss of cover gas (e.g., in PSSPS overflow vessel, EVSPS NaK expansion tanks, or reactor vessel/PHTS) is not modeled based on FMEA of cover gas/DHRS interface (Ref. 5). Overpressure from cover gas during a DHRS mission would require faults in the cover gas system and failure of relief valves. This contingency is estimated to produce a DHRS unavailability contribution of less than 1×10^{-6}, and therefore is not modeled.

G. A sequencer automatically aligns DHRS components; but the sequencer must be initiated by operators (from the MCR), and any failure to actuate can be overcome by manual operation. Thus, only a system-level human error is modeled.

H. The mission time for the analysis is assumed to be 24 hrs.

I. Check valves in each of the makeup pump discharge lines were assumed to be installed as indicated by engineering change subsequent to the study freeze date.

Further detailed assumptions are listed in the notes to the fault tree, Table A3-1.

A3.2.2 Detailed Fault Tree Model

A3.2.2.1 Success criteria. DHRS success requires operator action (with or without the alignment sequencer), the isolation of cold trapping in

Table A3-1

DHRS FAULT TREE NOTES

Notes

1. Includes sequencer failure scenario.

2. Applies only to OHX and piping isolated from normal makeup function which is continuous.

3. Escape path through an inadvertently open V142 is stopped by at least one other normally closed valve at every option (e.g., V194). Thus this event is not modeled.

4. Assume success criterion for DHRS is 1/2 EVST loops. Reference 1, p. 2-41, Fig. 2.2.3.2-1 shows preferred path is to cool both EVST and DHRS loads using all of circuit A but only NaK loop B. If circuit A fails, DHRS is assumed unaffected, but ultimately NDHX circuit may be needed. If loop B fails, DHRS can use loop A and EVST NaK loop A can be turned off. Again, NDHX circuit may be ultimately needed for EVST cooling.

5. EVST loss of cooling tree for NaK and ABHX loops (gates L33, L38, L59, L64) are appropriate for R5 and R6, except crossover valve development.

6. Reference 1, p. 2-41, makeup loop is used continuously, thus preheating failures have no effect.

7. Diversion path through N.O. valve V153, N.C. valve 152, and N.O. valve N149 would be closed by normal ALMS operation assumption.

8. Loss of preheating to makeup loop A assumed no problem since makeup loops are normally running. Diversion through N.C. valve V114 transferring open protected by another N.C. valve in any direction; not modelled.

9. Same as 8 for makeup loop B.

10. Deleted.

11. Backflow through a failed makeup pump is possible if the check valve at the discharge of the running pump does not close.

12. Deleted.

Table A3-1 (Continued)

Notes

13. Deleted.

14. HV103 and HV107 have N_2 accumulators as well as direct N_2 feed to their solenoids. These valves can be cycled once in 10 hrs after losing nitrogen.

15. Deleted.

16. HV104 and HV118 are assumed open in normal operation, are FAI, and have a common N_2 accumulator (81PPT005). This accumulator also supplies HV103 and HV107.

17. Transfer open of any interface valve (e.g., V364) is backed up by another normally closed valve downstream.

18. Deleted.

19. Assumed loss of HVAC cooling to cold trap would not result in pipe blockage or leak.

20. Loss of cover gas in freeze vents is assumed not a problem.

21. Natural draft in one ABHX is not sufficient for DHRS loads, but 2/2 ABHX on natural draft is. The latter will be treated as a recovery option, at cut set level.

22. Damper faults due to control logic or control power not developed.

23. Deleted.

24. The DHRS sequencer does not call for closing HV149 or HV136, but the operator is instructed to do so. Either would back up HV109.

25. Since 160 gpm flow is routed through V151 normally, it is assumed to be open already.

26. Generic loss of preheating for DHRS. Stagnant piping, OHX and DHRS crossovers have alarmed, redundant trace heaters.

27. N099TI810109 (Ref. 5) says HV103, 102, and 109 draw division 1 power, and HV107 draws division 2.

Table A3-1 (Continued)

Notes

28. Deleted.

29. Assuming valve set HV357, -58, -59 take power off bus 12NIE041A and valve set HV415, -16, -20 off of bus 12NIE041B.

30. Deleted.

31. Damper failures fail HVAC cooling but are already modeled.

32. Deleted.

the PSSPS, operation of both makeup pumps, and operation of one of two EVSPS/ABHX loops. Part of DHRS alignment requires reconfiguring two valve sets (HV357, HV358, HV359, and HV415, HV416, HV420) to allow heat transfer from the OHX in the PSSPS to the NaK loops in EVSPS through the DHRS crossover piping.

A3.2.2.2 Top events. There is only one DHRS top event: DHRS fails to remove decay heat. This event feeds into the "or" gate, TD1G1, in event D of the CRBRP PRA event tree.

The fault tree first separates a system-level human interaction (error), failure in the crossover valve sets, failure in PSSPS, and failure in both EVSPS NaK loops. The EVSPS part of the tree is borrowed from the fault tree developed for loss of EVST cooling and modified to reflect the appropriate DHRS crossover valve status for DHRS operation.

A3.2.2.3 Transfers to other systems. Preheating interfaces were modeled as developed events rather than transfers because DHRS preheating has alarmed, redundant heaters. Valves were assumed to be powered from buses that powered "analogous" (i.e., closest) pumps.

A3.2.2.4 Symbology. PSSPS events list node numbers that are shown on Figure A3-1. Numbers to the right of an intermediate event description refer to notes in Table A3-1.

A3.2.2.5 Fault tree. Figure A3-3 is the detailed fault tree for DHRS.

A3.2.3 Modularized Fault Tree Model

The detailed fault tree model for DHRS was simplified (i.e., modularized) based on the following ground rules:

1. All failure modes of a specific component were recombined.
2. Any groups of components that had identical support systems were combined.
3. Piping system failures were combined as much as possible, including heat tracing.
4. The system distinctions of two EVSPS trains and the PSSPS were retained distinct.
5. A modular gate was named in correspondence to the appropriate gate in the original tree, where possible, e.g., R1 became RM1.

The modularized fault tree is shown in Figure A3-4.

A3.3 FAILURE DATA

A3.3.1 Basic Event Data

Hardware data for DHRS were taken from generic data in Table 5-2 of the main report. Circuit 2 of EVSPS was assumed to have a maintenance outage contribution of $(1.1 \times 10^{-4}/hr)$ (19 hr) = 2.1×10^{-3}, according to Table 5-3. Human error data were screened and quantified according to Section 6.

Data sheets are provided following the fault trees in Table A3-2.

A3.3.2 Module Data

Module data were derived from basic event data according to the individual module's logic. These data also follow the fault trees.

A3.4 SYSTEM-LEVEL RESULTS AND INSIGHTS

DHRS is one of the few CRBRP safety-related systems that must be manually actuated. The relative importance of this actuation varied, depending on sequence context, from dominant (67%) to minor (2%). Credit was always given for using the sequencer to simplify the actuation.

Other than operations considerations, DHRS availability was dominated by the failure of valves that lead to bypass of the OHX, one of which is recoverable, given time.

A3.5 REFERENCES

1. CRBRP System Design Description, Auxiliary Liquid Metal System, Rockwell International, Rev. 32, June 23, 1983.

2. CRBRP Overall Plant Design Description, Westinghouse, OPPD 10, Rev. 146, August 1, 1983.

3. Piping and Instrumentation Drawing, Primary Sodium Storage and Processing System, N099811022, Rev. 29, April 18, 1983.

4. P&ID, EVS Processing System, N099811023, Rev. 25, April 18, 1983.

5. Reliability Design Support Document for the Interface Between the Inert Gas Receiving and Process System and DHRS, Rockwell International, N099TI810109, Rev. A, January 13, 1983.

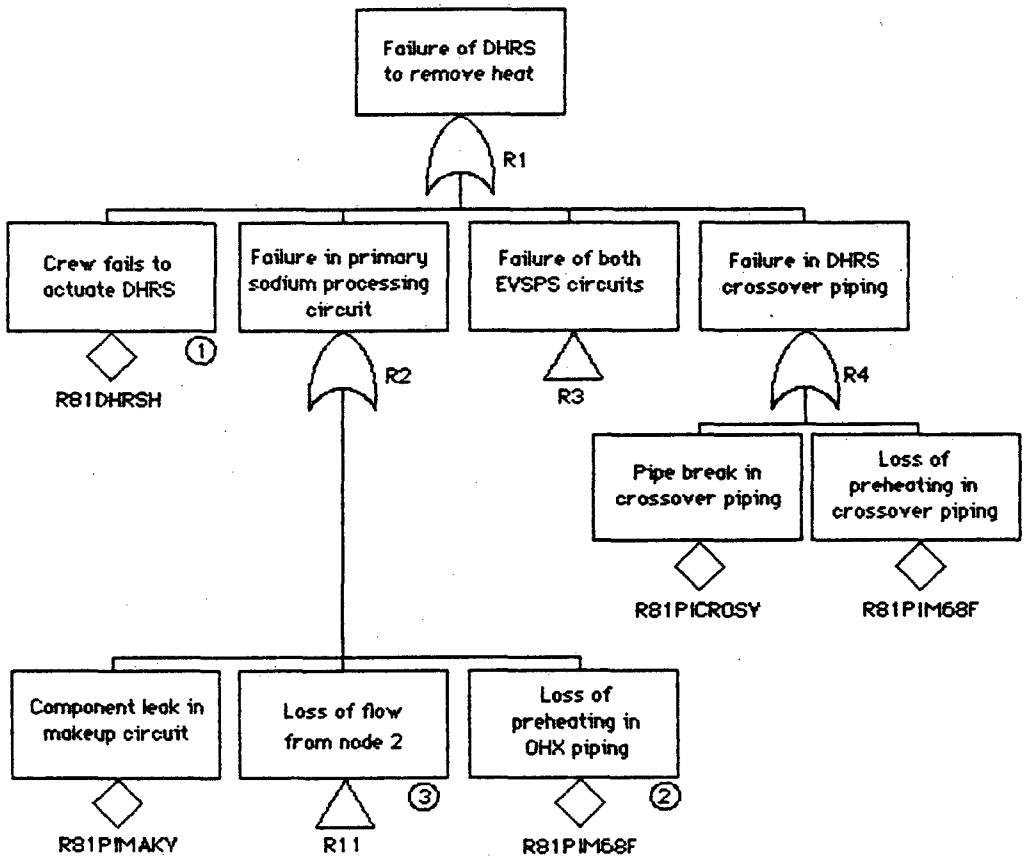
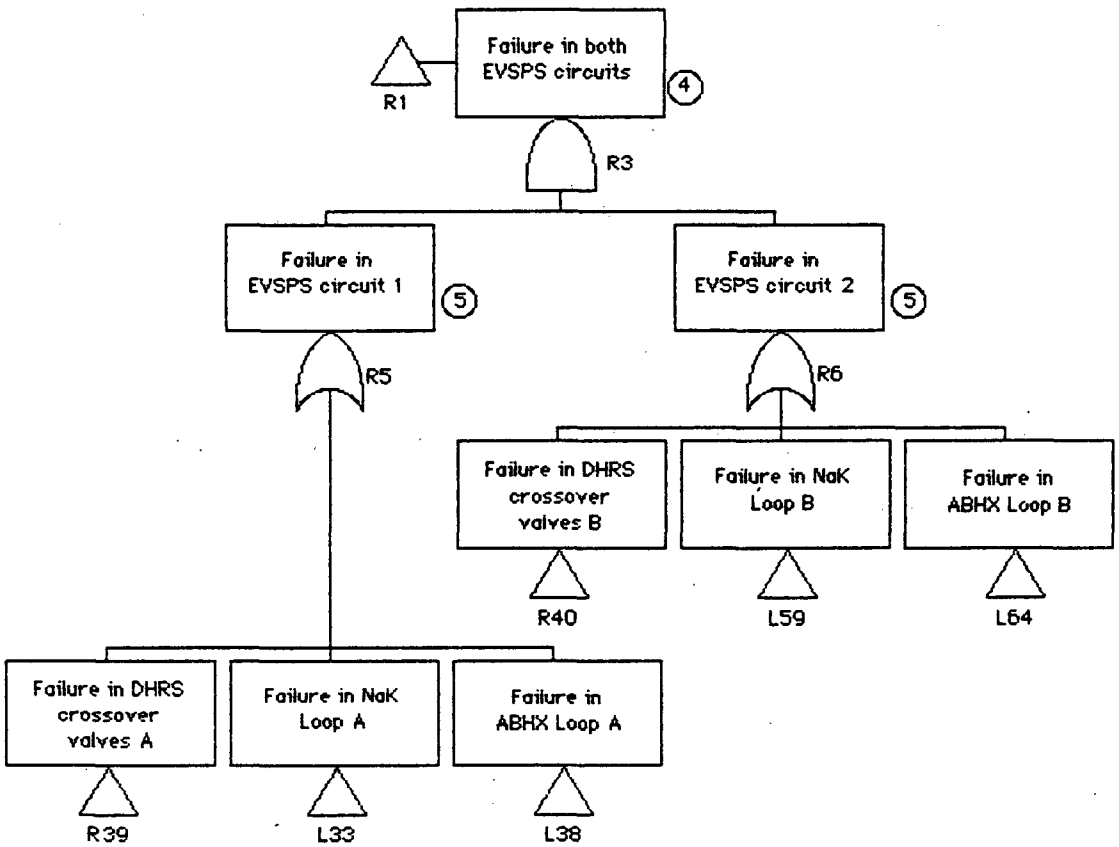
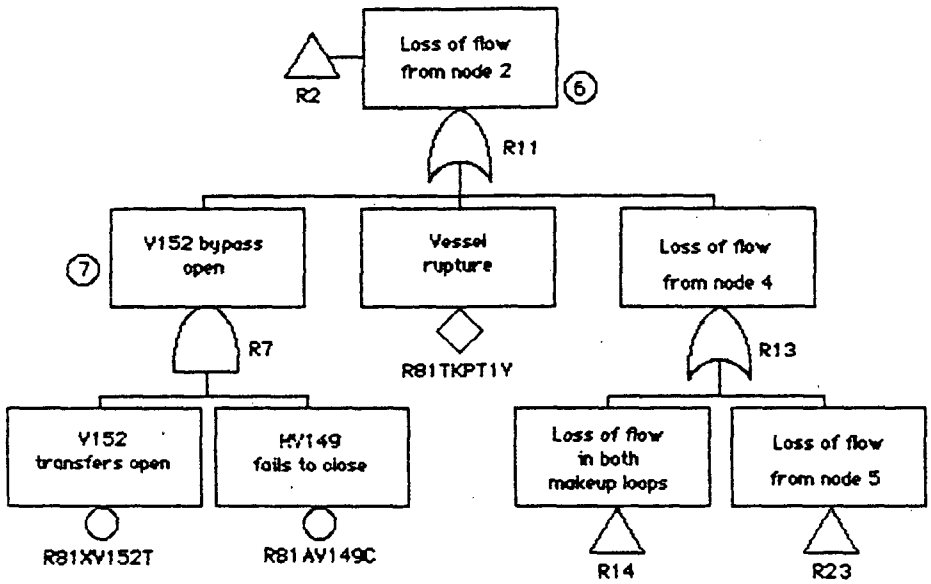
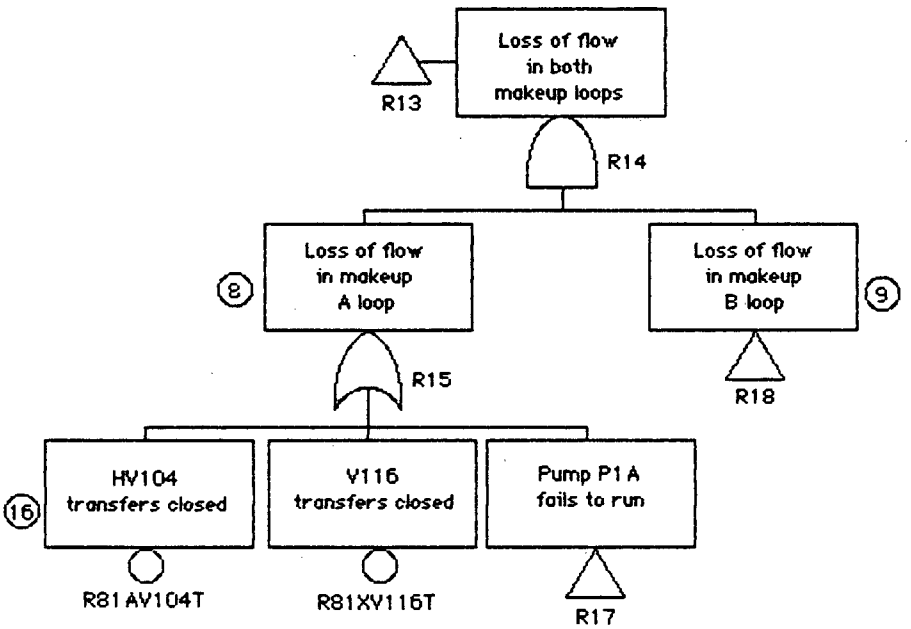
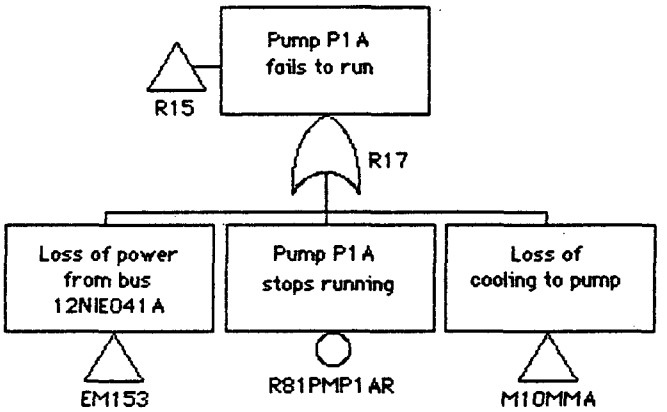


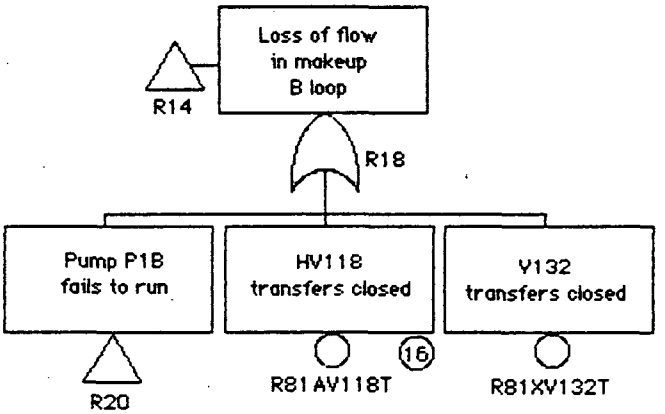
Figure A3-3. Direct Heat Removal Service Detailed Fault Tree.

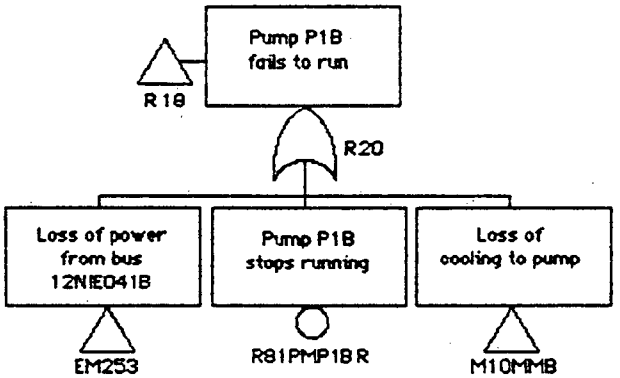

Failure in both EVSPS circuits
4
R1
R3
Failure in EVSPS circuit 1
5
Failure in EVSPS circuit 2
5
R5
R6
Failure in DHRS crossover valves B
Failure in NaK Loop B
Failure in ABHX Loop B
R40
L59
L64
Failure in DHRS crossover valves A
Failure in NaK Loop A
Failure in ABHX Loop A
R39
L33
L38

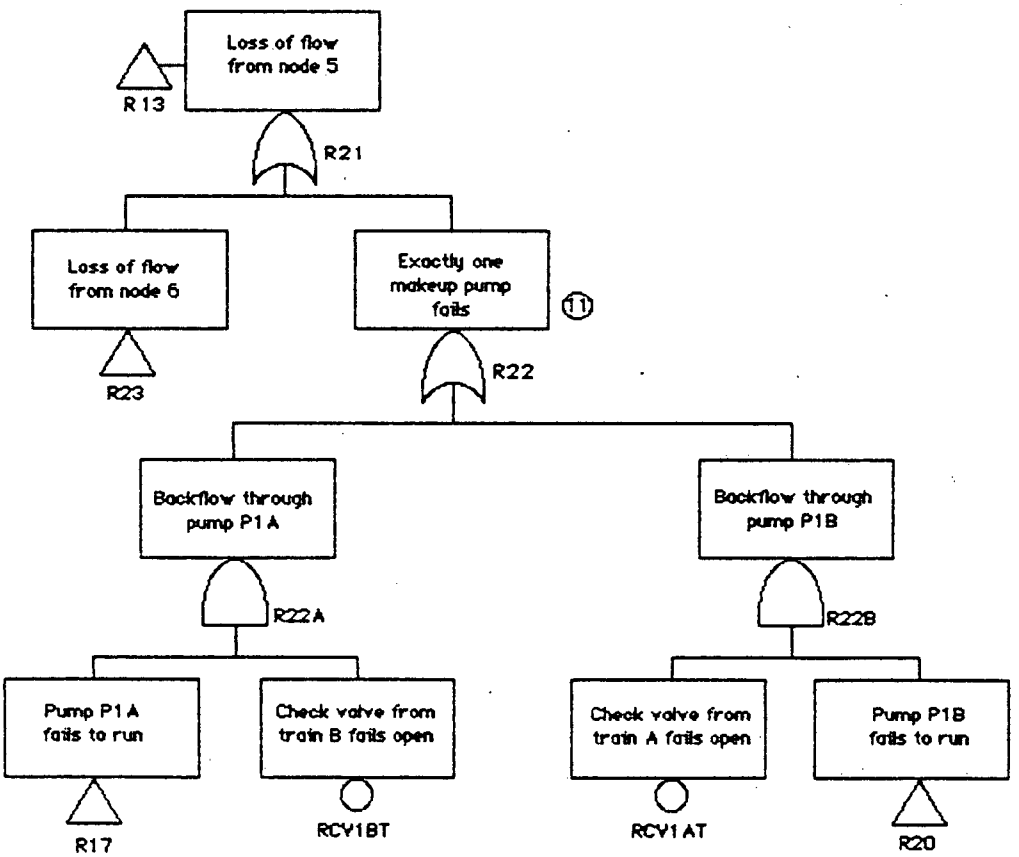

Loss of flow
from node 2
6
R2
R11
V152 bypass
open
7
Vessel
rupture
Loss of flow
from node 4
R7
R81TKPT1Y
R13
V152
transfers open
HV149
fails to close
Loss of flow
in both
makeup loops
Loss of flow
from node 5
R81XV152T
R81AV149C
R14
R23

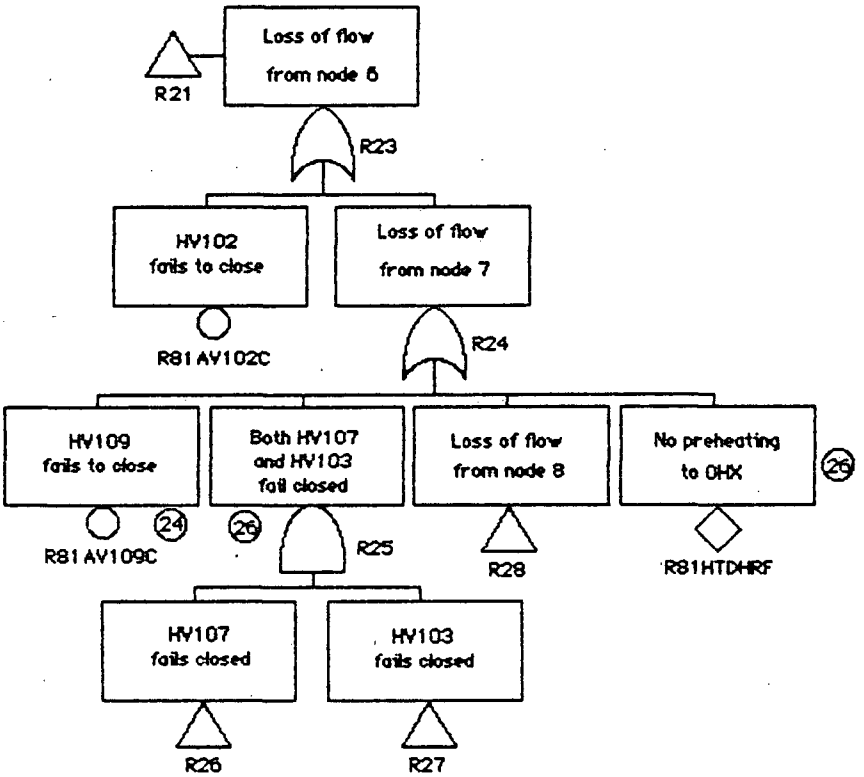

Loss of flow in both makeup loops
R13
R14
Loss of flow in makeup A loop
8
Loss of flow in makeup B loop
9
R15
R18
HV104 transfers closed
16
V116 transfers closed
Pump P1A fails to run
R81AV104T
R81XV116T
R17

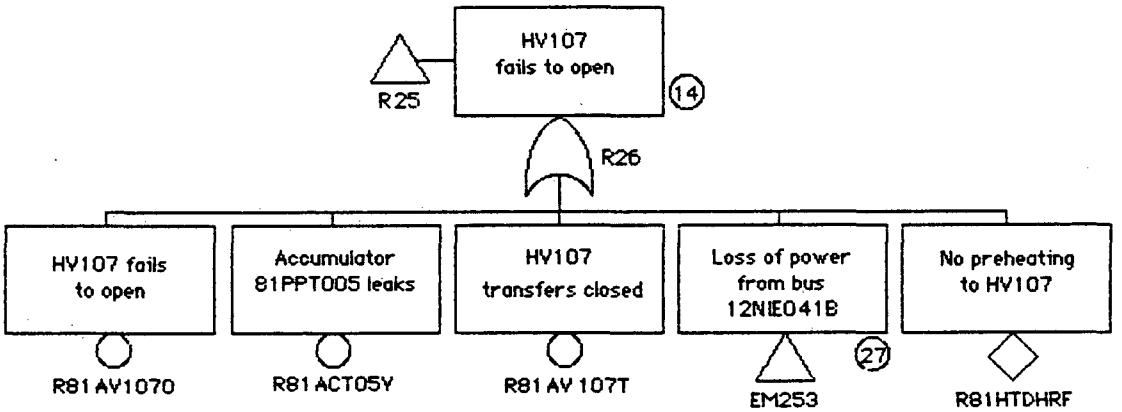

Pump P1 A
fails to run
R15
R17
Loss of power
from bus
12NIE041 A
Pump P1 A
stops running
Loss of
cooling to pump
EM153
R81PMP1 AR
M10MMA

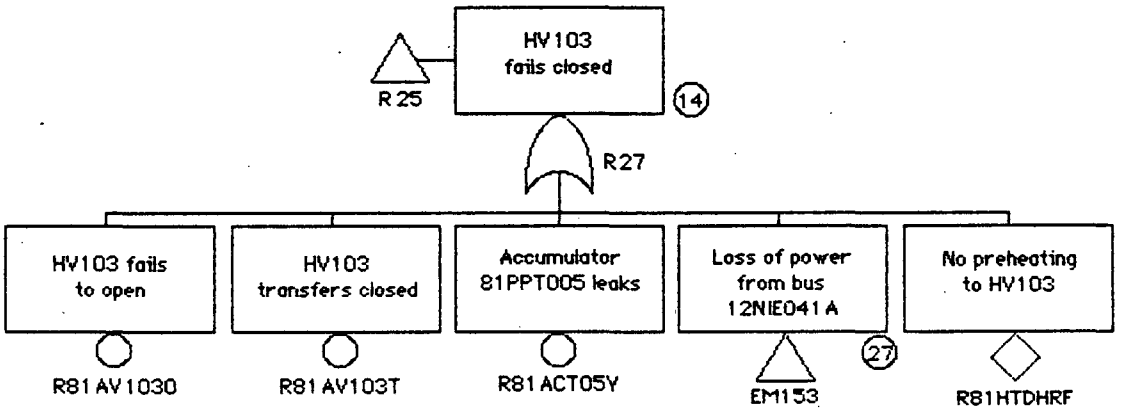

Loss of flow
in makeup
B loop
R14
R18
Pump P1B
fails to run
HV118
transfers closed
V132
transfers closed
16
R20
R81AV118T
R81XV132T

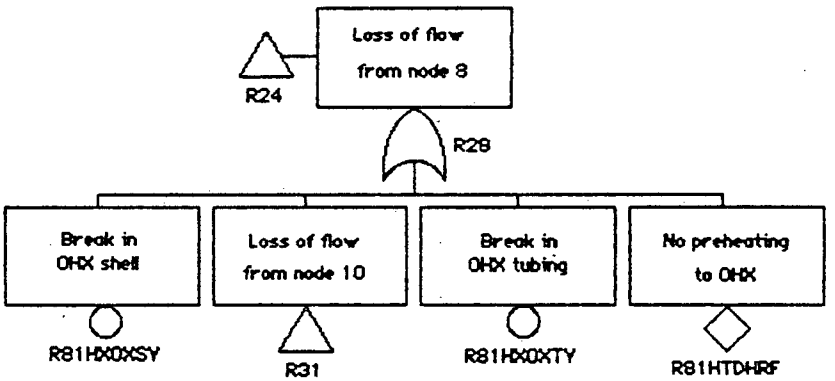

Pump P1B fails to run
R18
R20
Loss of power from bus 12NIE041B
Pump P1B stops running
Loss of cooling to pump
EM253
R81PMP1BR
M10MMB

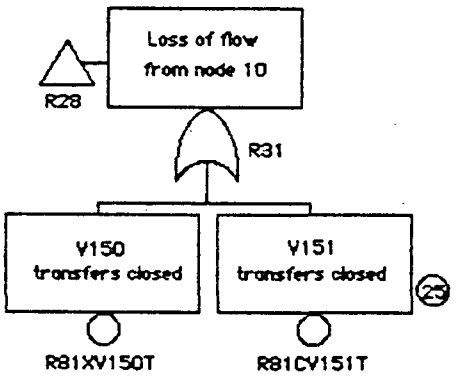

Loss of flow from node 5
R13
R21
Loss of flow from node 6
Exactly one makeup pump fails
11
R23
R22
Backflow through pump P1A
Backflow through pump P1B
R22A
R22B
Pump P1A fails to run
Check valve from train B fails open
Check valve from train A fails open
Pump P1B fails to run
R17
RCV1BT
RCV1AT
R20

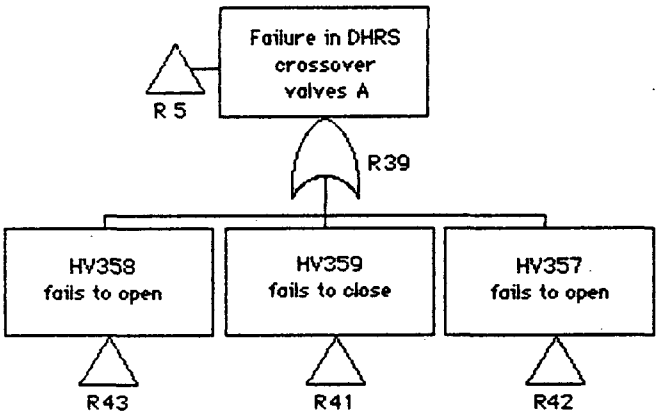

Loss of flow
from node 6
R21
R23
HV102
fails to close
Loss of flow
from node 7
R81AV102C
R24
HV109
fails to close
Both HV107
and HV103
fail closed
Loss of flow
from node 8
No preheating
to OHX
26
R81AV109C
24
26
R25
R28
R81HTDHRF
HV107
fails closed
HV103
fails closed
R26
R27

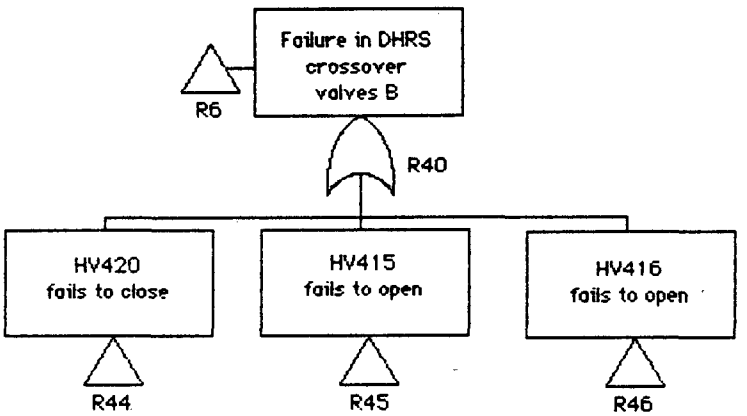

HV107
fails to open
14
R25
R26
HV107 fails
to open
Accumulator
81PPT005 leaks
HV107
transfers closed
Loss of power
from bus
12NIE041B
No preheating
to HV107
R81AV1070
R81ACT05Y
R81AV107T
27
EM253
R81HTDHRF

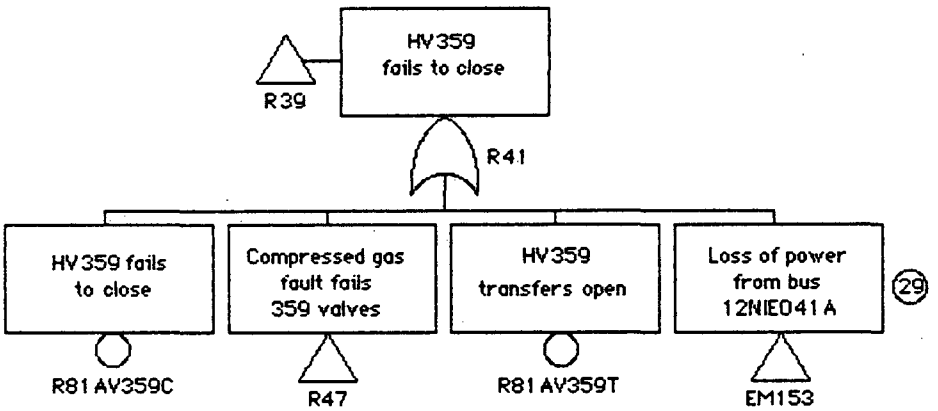

HV103
fails closed
14
R25
R27
HV103 fails
to open
HV103
transfers closed
Accumulator
81PPT005 leaks
Loss of power
from bus
12NIE041A
No preheating
to HV103
R81AV1030
R81AV103T
R81ACT05Y
27
EM153
R81HTDHRF

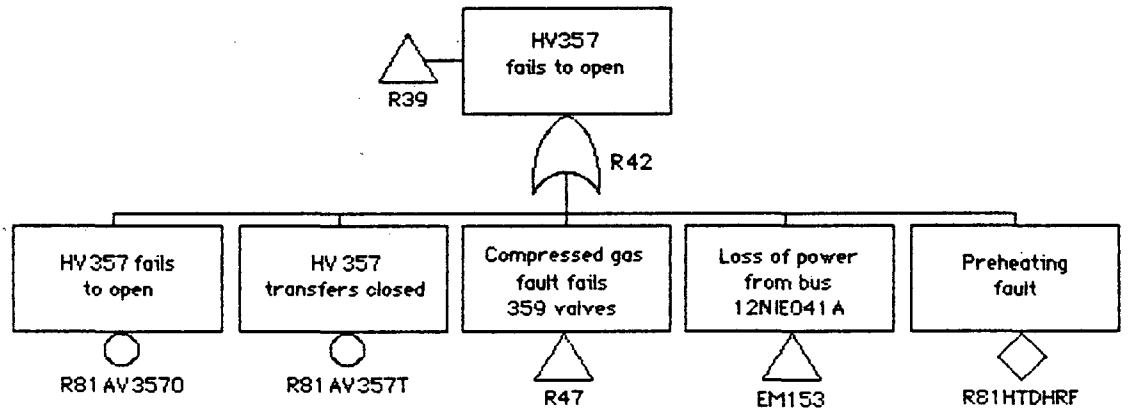

Loss of flow
from node 8
R24
R28
Break in
OHX shell
Loss of flow
from node 10
Break in
OHX tubing
No preheating
to OHX
R81HXOXSY
R31
R81HXOXTY
R81HTDHRF

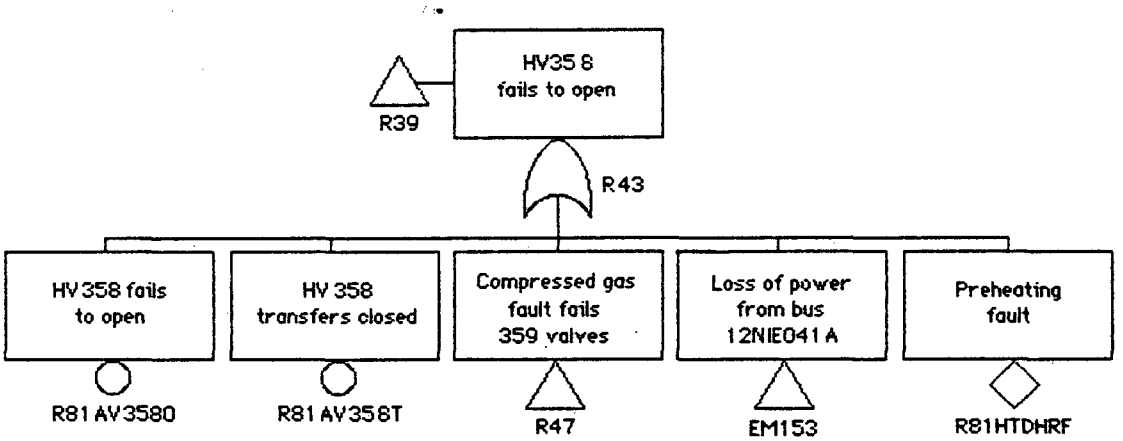

Loss of flow
from node 10
R28
R31
V150
transfers closed
V151
transfers closed
23
R81XV150T
R81CV151T

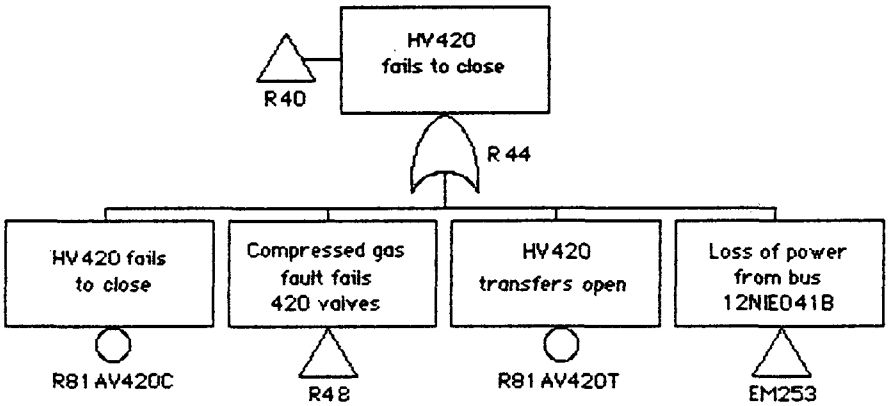

Failure in DHRS crossover valves A
R 5
R39
HV358 fails to open
HV359 fails to close
HV357 fails to open
R43
R41
R42

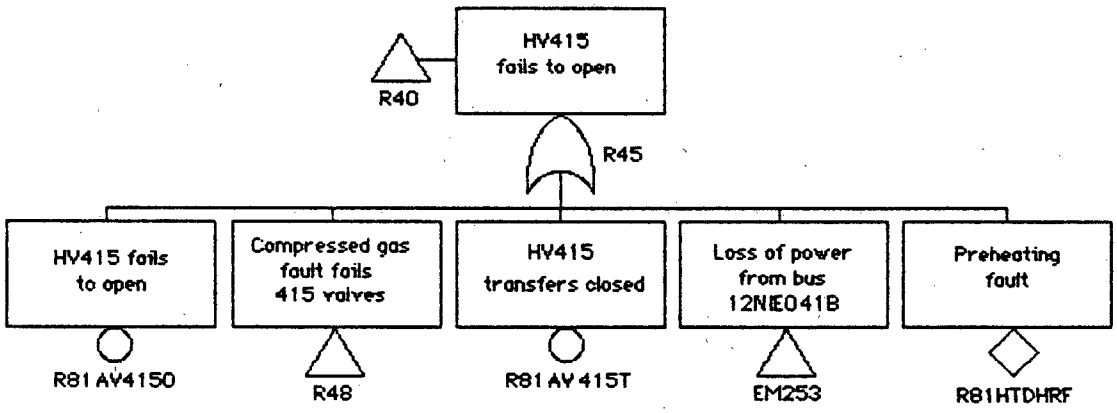

Failure in DHRS crossover valves B
R6
R40
HV420 fails to close
HV415 fails to open
HV416 fails to open
R44
R45
R46

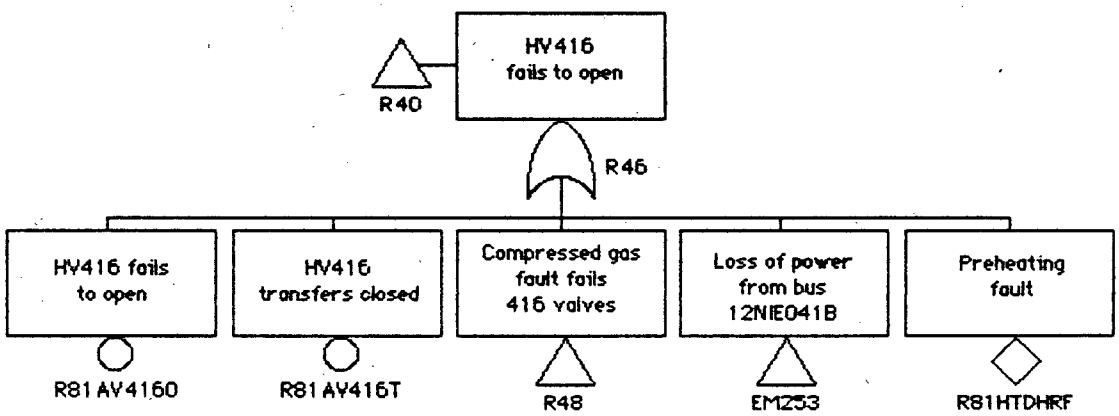

HV359
fails to close
R39
R41
HV359 fails
to close
Compressed gas
fault fails
359 valves
HV359
transfers open
Loss of power
from bus
12NIE041A
29
R81AV359C
R47
R81AV359T
EM153

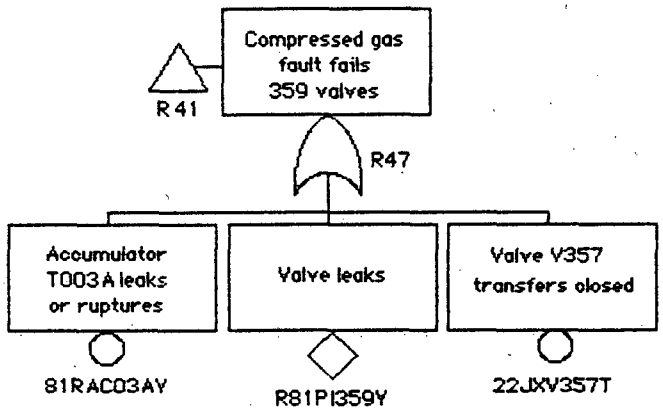

HV357
fails to open
R39
R42
HV357 fails
to open
HV357
transfers closed
Compressed gas
fault fails
359 valves
Loss of power
from bus
12NIE041A
Preheating
fault
R81AV357O
R81AV357T
R47
EM153
R81HTDHRF

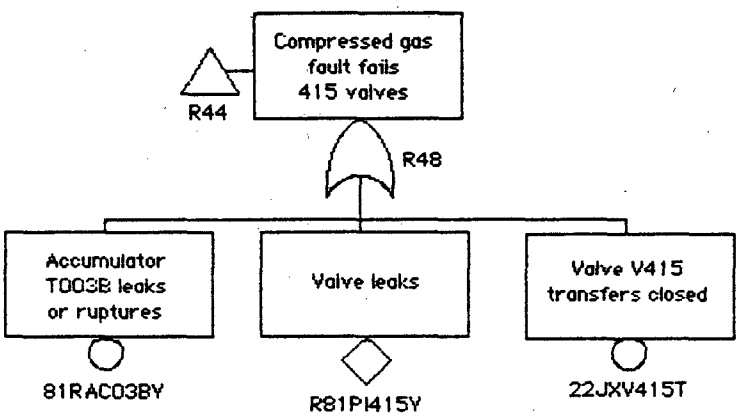

HV358
fails to open
R39
R43
HV358 fails
to open
HV 358
transfers closed
Compressed gas
fault fails
359 valves
Loss of power
from bus
12NIE041A
Preheating
fault
R81AV3580
R81AV358T
R47
EM153
R81HTDHRF

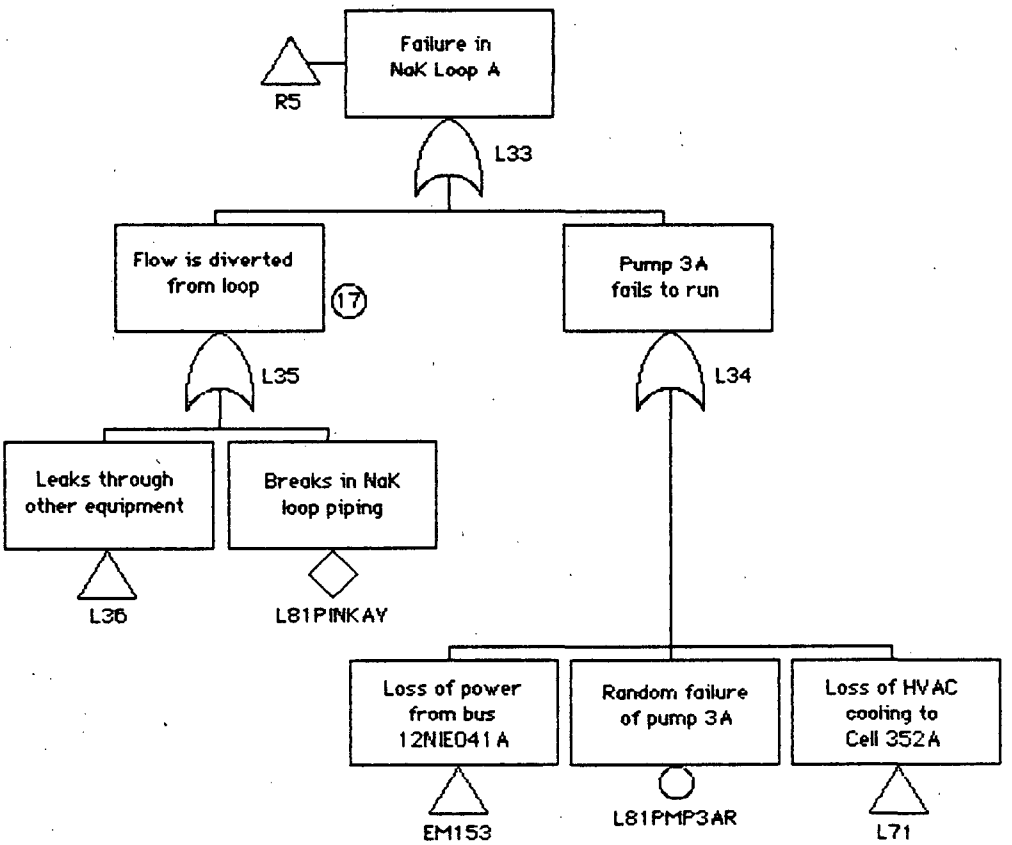

HV420
fails to close
R40
R44
HV420 fails
to close
Compressed gas
fault fails
420 valves
HV420
transfers open
Loss of power
from bus
12NIE041B
R81AV420C
R48
R81AV420T
EM253

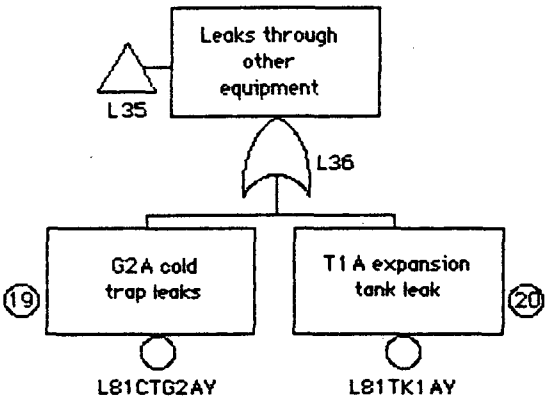

HV415
fails to open
R40
R45
HV415 fails
to open
Compressed gas
fault fails
415 valves
HV415
transfers closed
Loss of power
from bus
12NIE041B
Preheating
fault
R81AV415O
R48
R81AV415T
EM253
R81HTDHRF

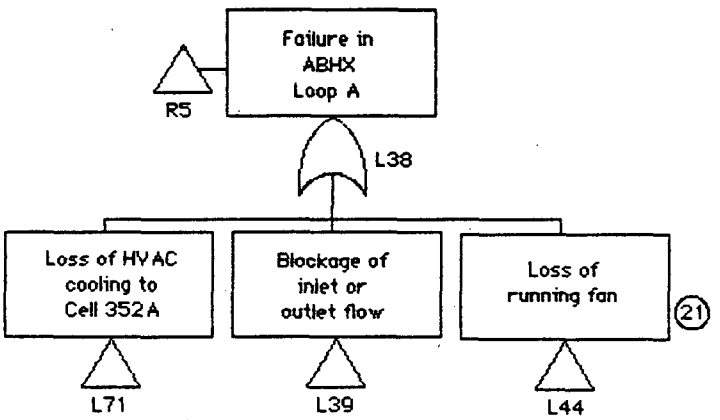

HV416
fails to open
R40
R46
HV416 fails
to open
HV416
transfers closed
Compressed gas
fault fails
416 valves
Loss of power
from bus
12NIE041B
Preheating
fault
R81AV416O
R81AV416T
R48
EM253
R81HTDHRF

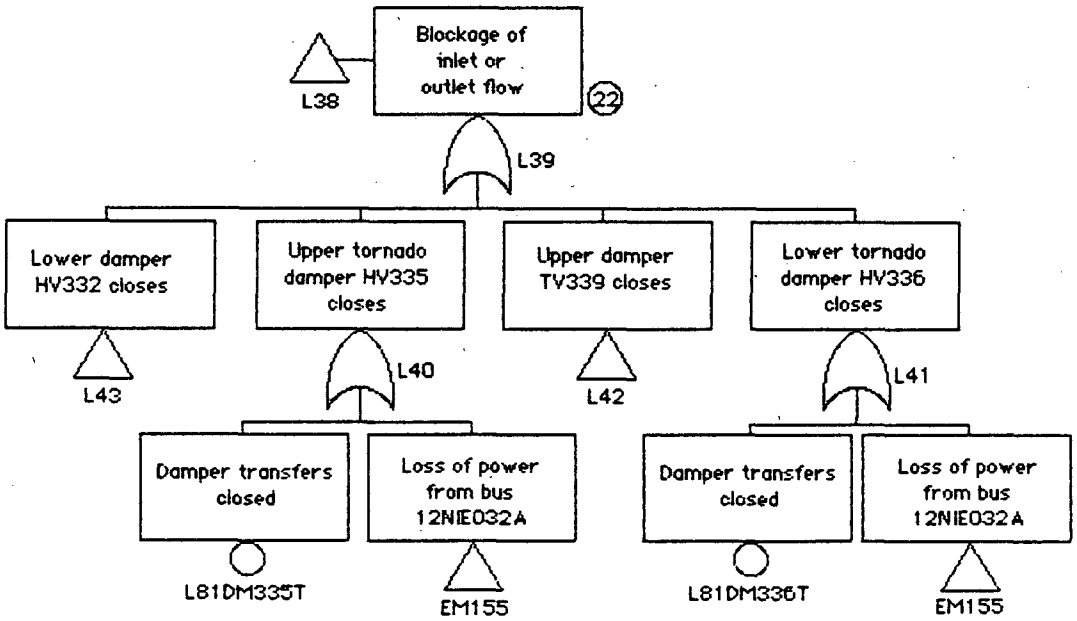

Compressed gas
fault fails
359 valves
R41
R47
Accumulator
T003A leaks
or ruptures
Valve leaks
Valve V357
transfers closed
81RAC03AY
R81PI359Y
22JXV357T

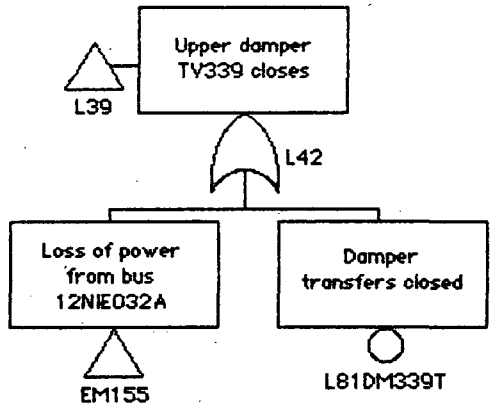

Compressed gas
fault fails
415 valves
R44
R48
Accumulator
T003B leaks
or ruptures
Valve leaks
Valve V415
transfers closed
81RAC03BY
R81PI415Y
22JXV415T

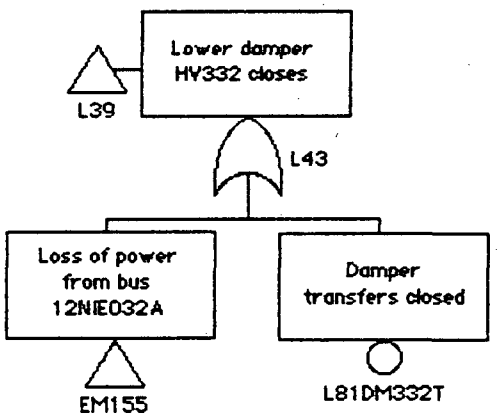

Failure in NaK Loop A
R5
L33
Flow is diverted from loop
17
Pump 3A fails to run
L35
L34
Leaks through other equipment
Breaks in NaK loop piping
L36
L81PINKAY
Loss of power from bus 12NIE041A
Random failure of pump 3A
Loss of HVAC cooling to Cell 352A
EM153
L81PMP3AR
L71

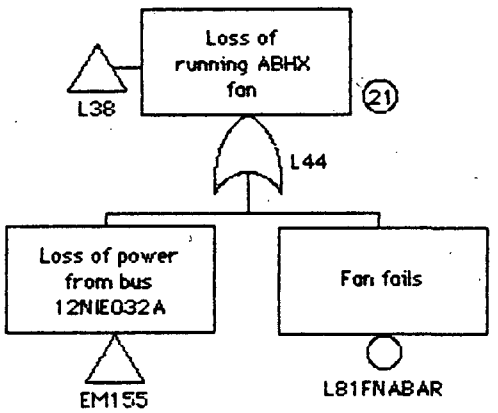

Leaks through other equipment
L35
L36
G2A cold trap leaks
19
T1A expansion tank leak
20
L81CTG2AY
L81TK1AY

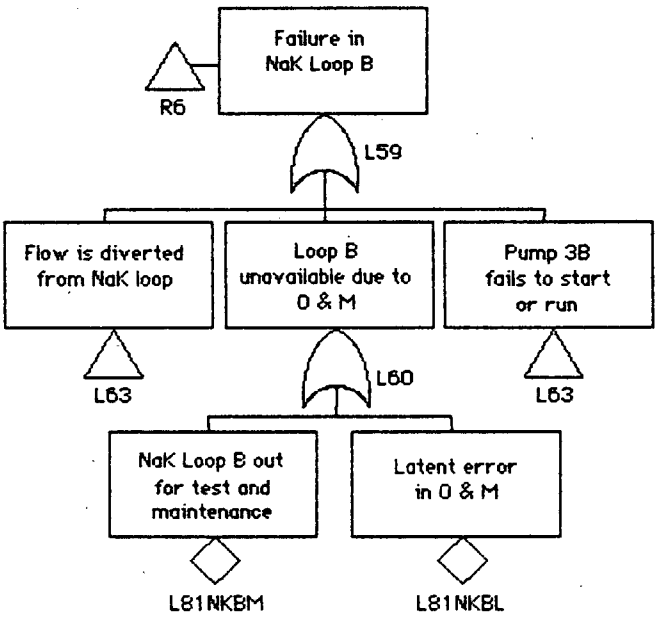

Failure in ABHX Loop A
R5
L38
Loss of HVAC cooling to Cell 352A
Blockage of inlet or outlet flow
Loss of running fan
21
L71
L39
L44

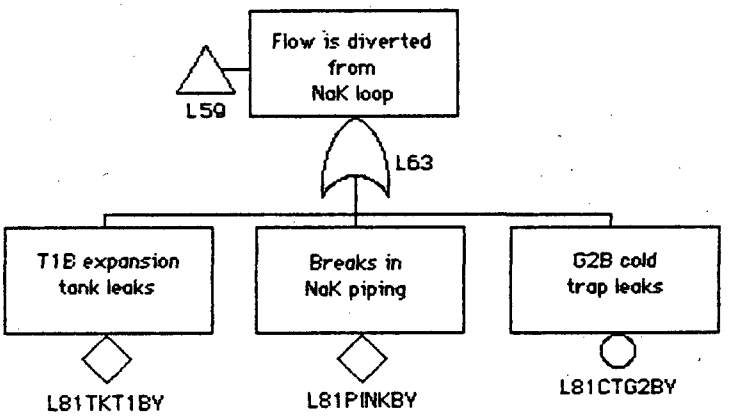

Blockage of inlet or outlet flow
22
L38
L39
Lower damper HV332 closes
Upper tornado damper HV335 closes
Upper damper TV339 closes
Lower tornado damper HV336 closes
L43
L40
L42
L41
Damper transfers closed
Loss of power from bus 12NIE032A
Damper transfers closed
Loss of power from bus 12NIE032A
L81DM335T
EM155
L81DM336T
EM155

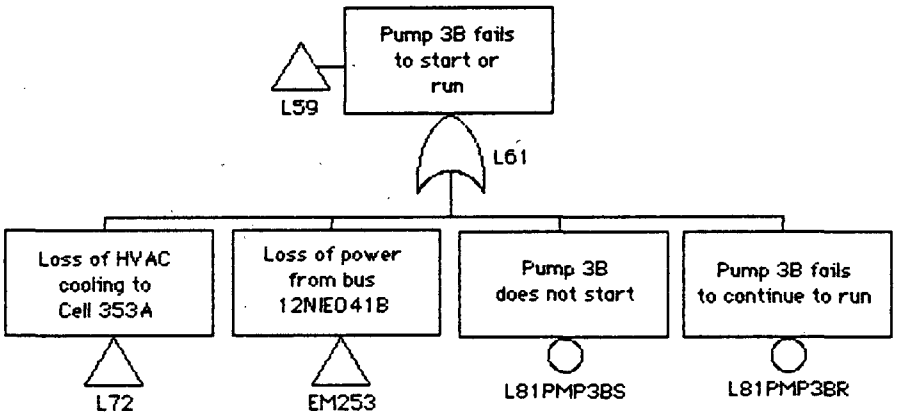

Upper damper
TV339 closes
L39
L42
Loss of power
from bus
12NIE032A
Damper
transfers closed
EM155
L81DM339T

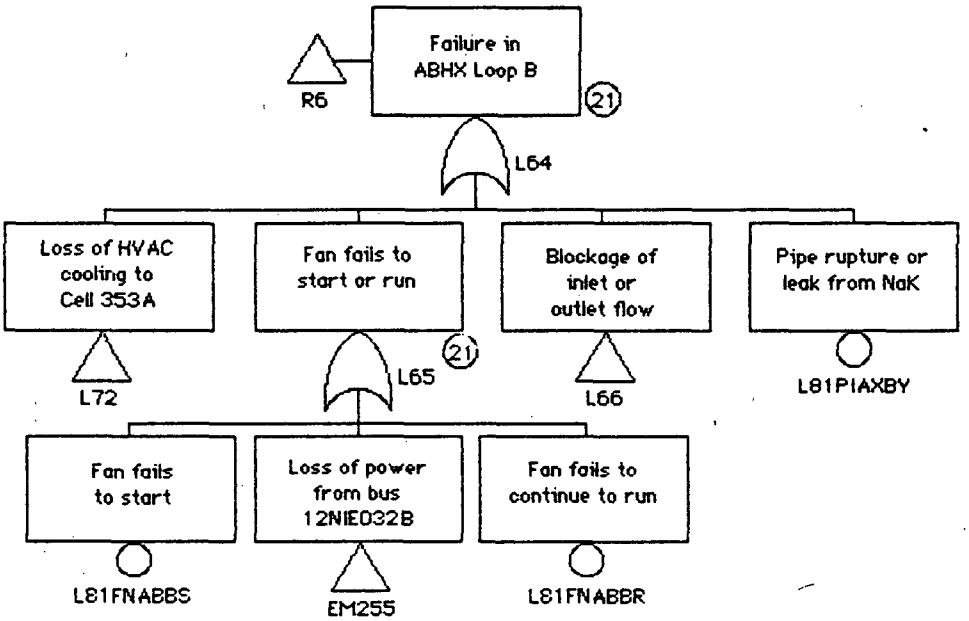

Lower damper
HV332 closes
L39
L43
Loss of power
from bus
12NIE032A
Damper
transfers closed
EM155
L81DM332T

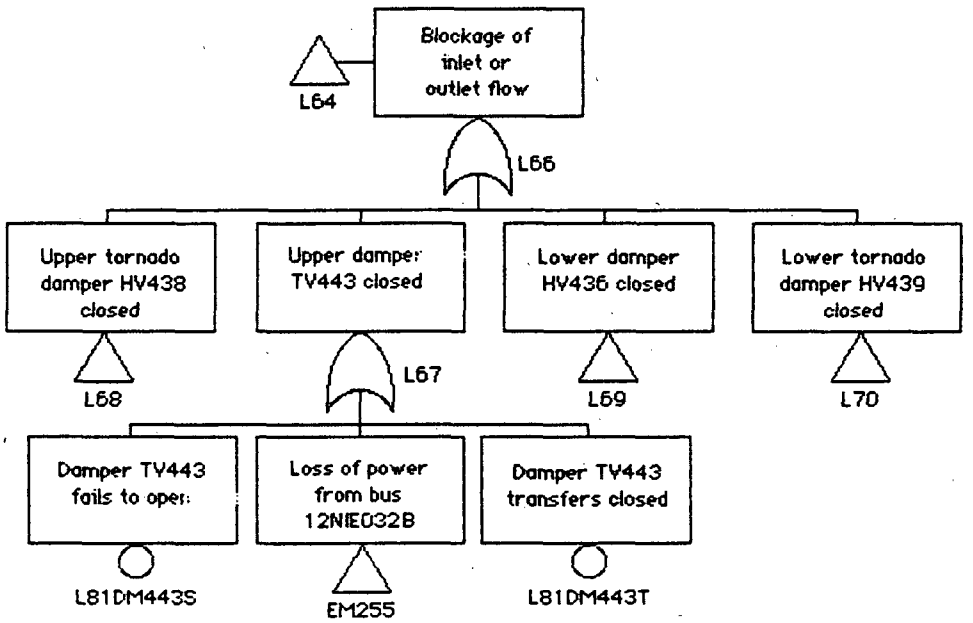

Loss of
running ABHX
fan
21
L38
L44
Loss of power
from bus
12NIE032A
Fan fails
EM155
L81FNABAR

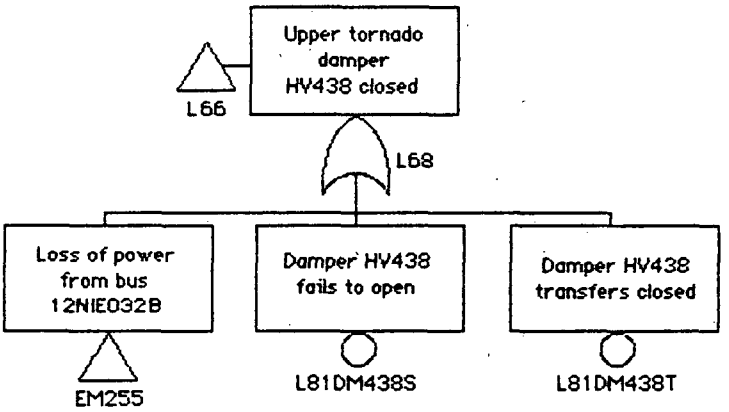

Failure in
NaK Loop B
R6
L59
Flow is diverted
from NaK loop
Loop B
unavailable due to
O & M
Pump 3B
fails to start
or run
L63
L60
L63
NaK Loop B out
for test and
maintenance
Latent error
in O & M
L81NKBM
L81NKBL

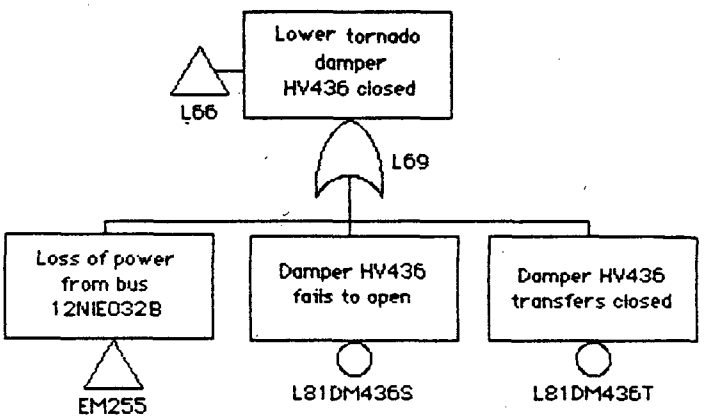

Flow is diverted from NaK loop
L59
L63
T1B expansion tank leaks
Breaks in NaK piping
G2B cold trap leaks
L81TKT1BY
L81PINKBY
L81CTG2BY

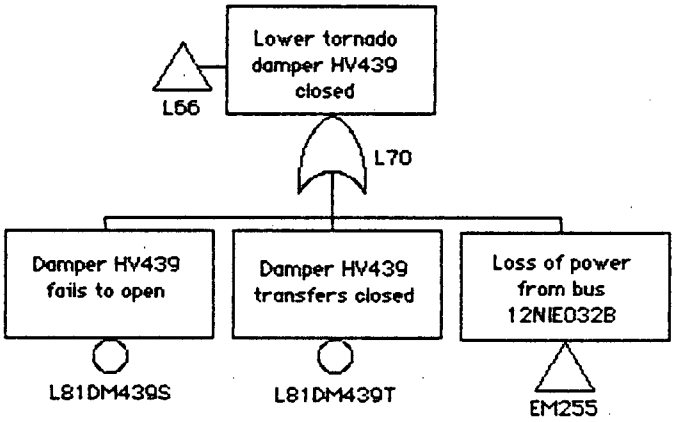

Pump 3B fails to start or run
L59
L61
Loss of HVAC cooling to Cell 353A
Loss of power from bus 12NIE041B
Pump 3B does not start
Pump 3B fails to continue to run
L72
EM253
L81PMP3BS
L81PMP3BR

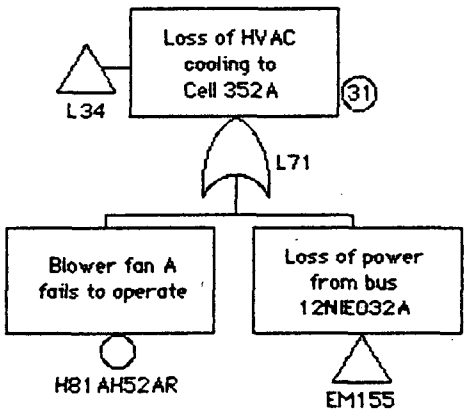

Failure in ABHX Loop B
21
R6
L64
Loss of HVAC cooling to Cell 353A
Fan fails to start or run
Blockage of inlet or outlet flow
Pipe rupture or leak from NaK
L72
L65
21
L66
L81PIAXBY
Fan fails to start
Loss of power from bus 12NIE032B
Fan fails to continue to run
L81FNABBS
EM255
L81FNABBR

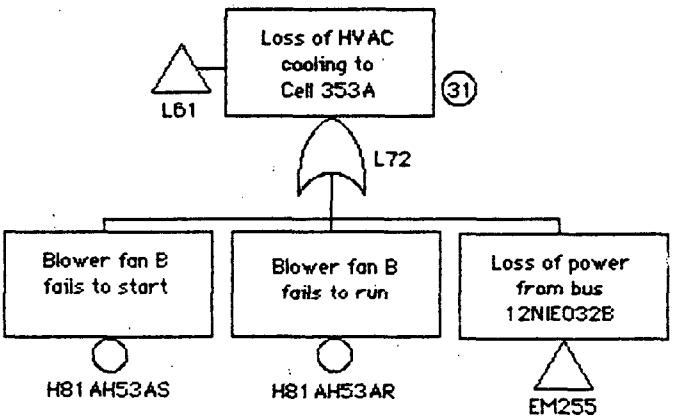

Blockage of inlet or outlet flow
L64
L66
Upper tornado damper HV438 closed
Upper damper TV443 closed
Lower damper HV436 closed
Lower tornado damper HV439 closed
L68
L67
L69
L70
Damper TV443 fails to open
Loss of power from bus 12NIE032B
Damper TV443 transfers closed
L81DM443S
EM255
L81DM443T


Upper tornado damper HV438 closed
L66
L68
Loss of power from bus 12NIE032B
Damper HV438 fails to open
Damper HV438 transfers closed
EM255
L81DM438S
L81DM438T


Lower tornado damper HV436 closed
L66
L69
Loss of power from bus 12NIE032B
Damper HV436 fails to open
Damper HV436 transfers closed
EM255
L81DM436S
L81DM436T


Lower tornado damper HV439 closed
L66
L70
Damper HV439 fails to open
Damper HV439 transfers closed
Loss of power from bus 12NIE032B
L81DM439S
L81DM439T
EM255


Loss of HVAC cooling to Cell 352A
31
L34
L71
Blower fan A fails to operate
Loss of power from bus 12NIE032A
H81AH52AR
EM155


Loss of HVAC cooling to Cell 353A
31
L61
L72
Blower fan B fails to start
Blower fan B fails to run
Loss of power from bus 12NIE032B
H81AH53AS
H81AH53AR
EM255

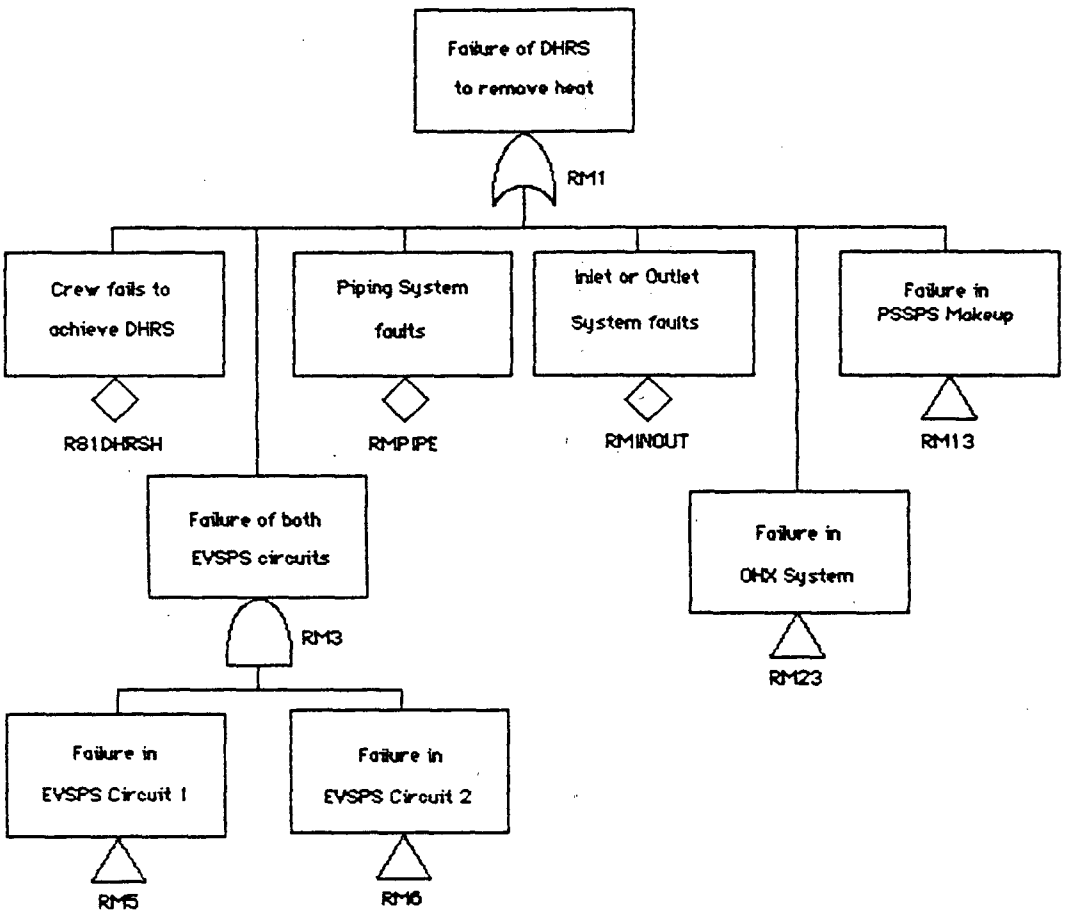
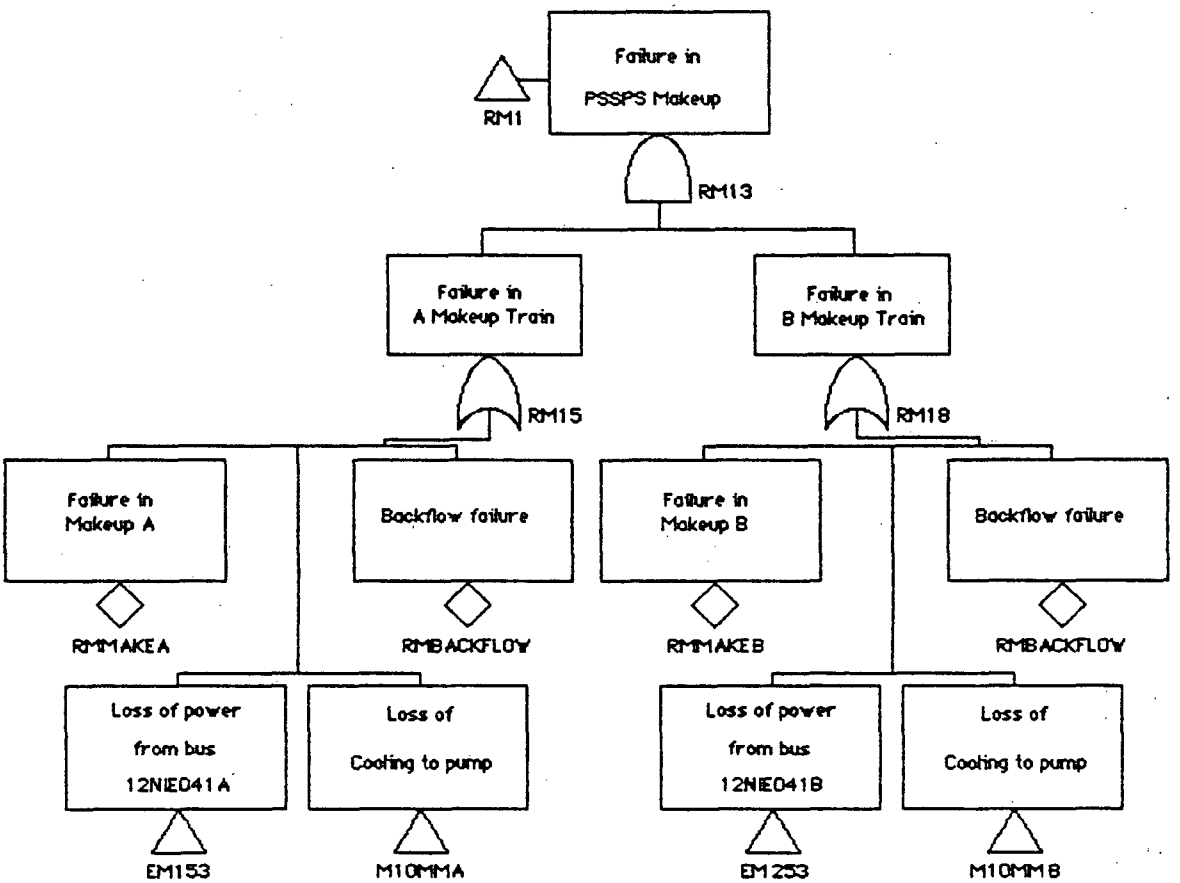
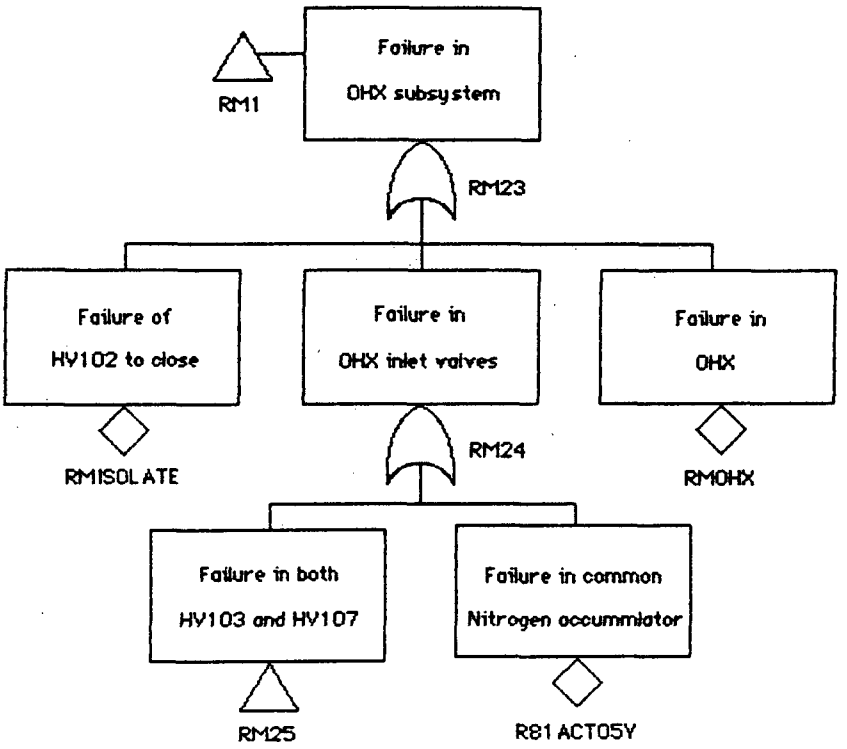
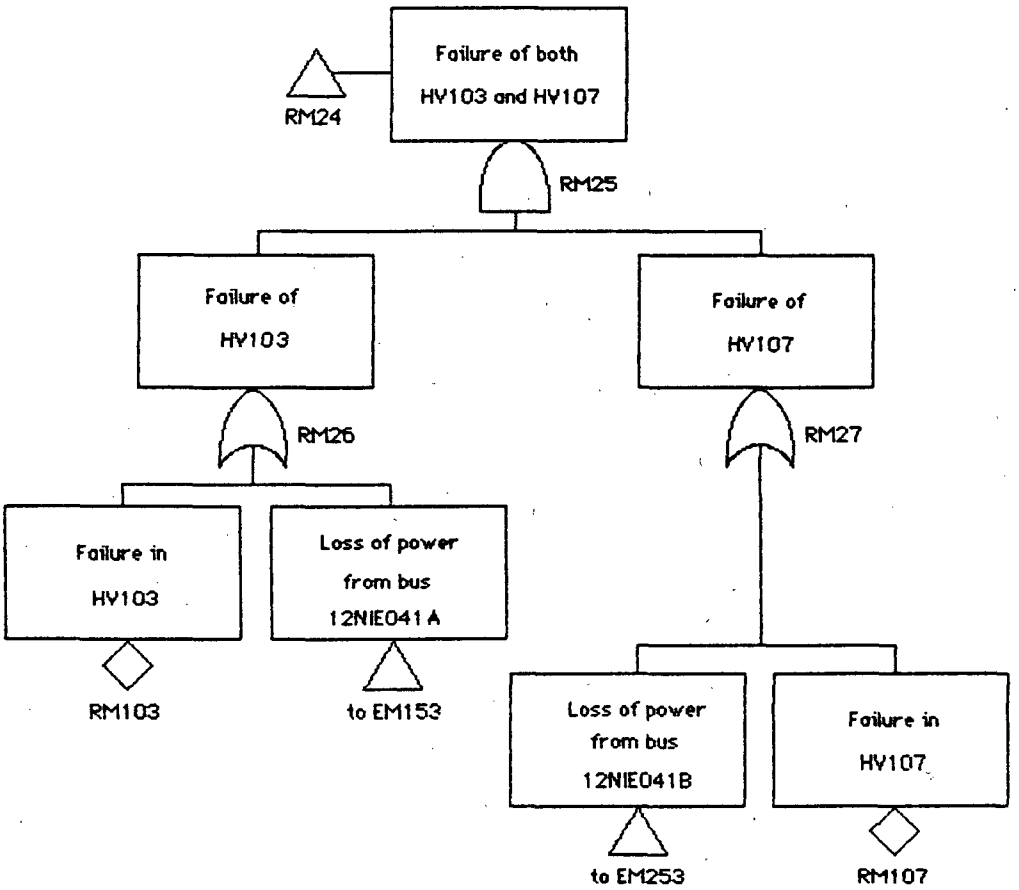
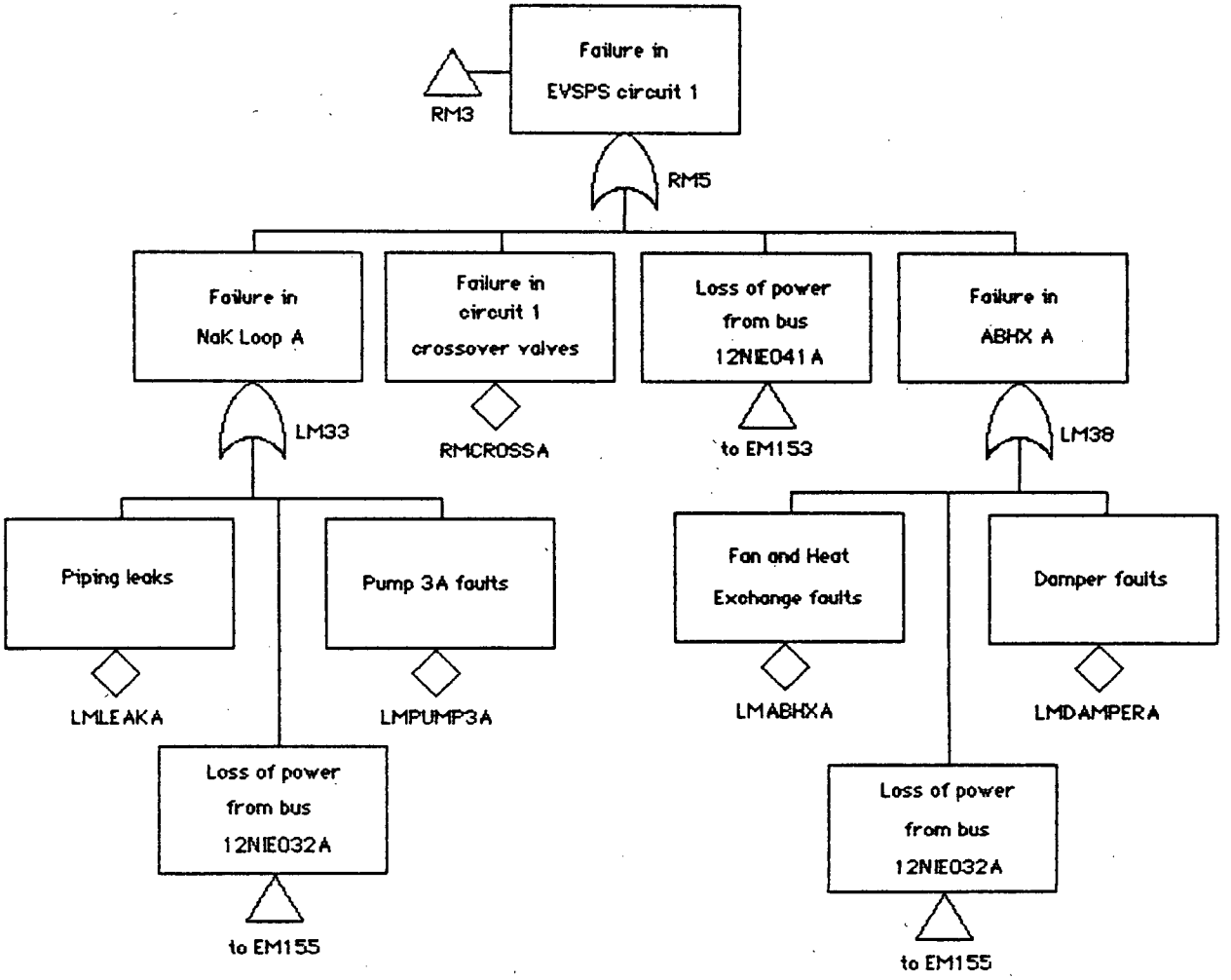


Figure A3-4. Direct Heat Removal Service Modularized Fault Tree.


Failure in
PSSPS Makeup
RM1
RM13
Failure in
A Makeup Train
Failure in
B Makeup Train
RM15
RM18
Failure in
Makeup A
Backflow failure
Failure in
Makeup B
Backflow failure
RMMAKEA
RMBACKFLOW
RMMAKEB
RMBACKFLOW
Loss of power
from bus
12NIE041A
Loss of
Cooling to pump
Loss of power
from bus
12NIE041B
Loss of
Cooling to pump
EM153
M10MMA
EM253
M10MMB


Failure in
OHX subsystem
RM11
RM23
Failure of
HV102 to close
Failure in
OHX inlet valves
Failure in
OHX
RM1SOLATE
RM24
RM0HX
Failure in both
HV103 and HV107
Failure in common
Nitrogen accummlator
RM25
R81ACT05Y


Failure of both
HV103 and HV107
RM24
RM25
Failure of
HV103
Failure of
HV107
RM26
RM27
Failure in
HV103
Loss of power
from bus
12NIE041A
RM103
to EM153
Loss of power
from bus
12NIE041B
Failure in
HV107
to EM253
RM107


Failure in
EVSPS circuit 1
RM3
RM5
Failure in
NaK Loop A
Failure in
circuit 1
crossover valves
Loss of power
from bus
12NIE041A
Failure in
ABHX A
LM33
RMCROSSA
to EM153
LM38
Piping leaks
Pump 3A faults
Fan and Heat
Exchange faults
Damper faults
LMLEAKA
LMPUMP3A
LMABHXA
LMDAMPERA
Loss of power
from bus
12NIE032A
Loss of power
from bus
12NIE032A
to EM155
to EM155

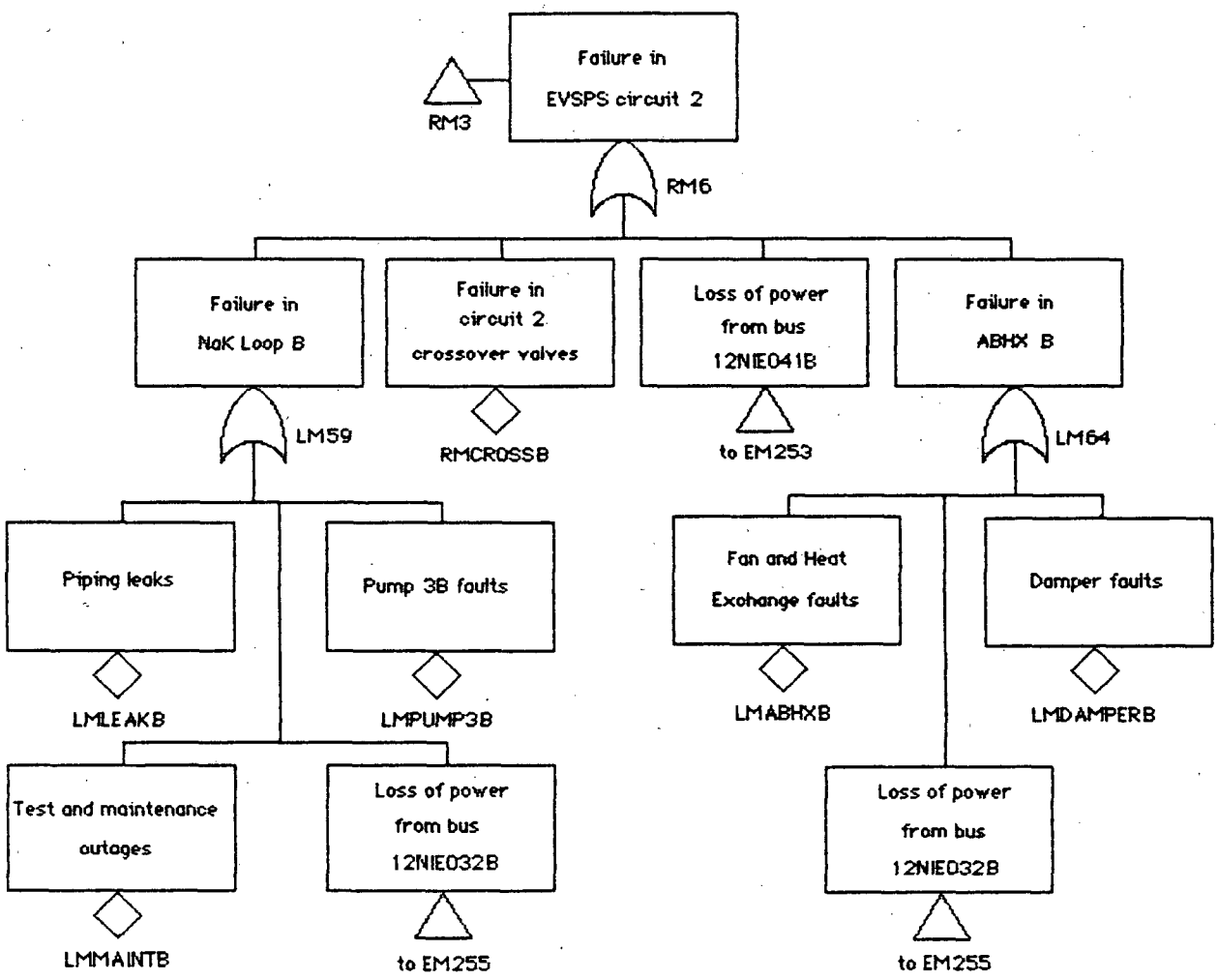

Failure in
EVSPS circuit 2
RM3
RM6
Failure in
NaK Loop B
Failure in
circuit 2
crossover valves
Loss of power
from bus
12NIE041B
Failure in
ABHX B
LM59
RMCROSSB
to EM253
LM64
Piping leaks
Pump 3B faults
Fan and Heat
Exchange faults
Damper faults
LMLEAKB
LMPUMP3B
LMABHXB
LMDAMPERB
Test and maintenance
outages
Loss of power
from bus
12NIE032B
Loss of power
from bus
12NIE032B
LMMAINTB
to EM255
to EM255

Table A3-2

DHRS DATA SHEETS

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
RMPIPE	DHRS primary pipe leak, or ruptures, or preheating failures	7.9-6	1.1-9 * 300*	24	TAB-OR
R81PICROSY	Pipe & component leaks/ruptures				
R81PIMAKY					
R81PIM68F					
R81HTDHRF	Preheating failures	ε			
RMINOUT		5.0-6			
R81XV152T	Valve transfers open	2.4-6	1.0-7	24	R81XV152T*R81AV149C
R81AV149C	Valve transfers open	2.4-6	1.0-7	24	+ R81TKPT1Y
R81TKPT1Y	Vessel rupture	2.4-7	1.0-8	24	+ R81XV150T
R81XV150T	Valve transfers closed	2.4-6	1.0-7	24	+ R81CV151T
R81CV151T	Valve transfers closed	2.4-6	1.0-7	24	
R81DHRSH	Crew fails to achieve DHRS	1.0-2**			

*Estimated total PSSPS piping to be 300 ft and used failure rate of 1.1×10^{-9}/hr-ft.

**This is a screening number. The actual estimate is sequence dependent--see Section 10 or Section 6.

Table A3-2 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
RMMAKEA		9.4-5			TAB-OR
R81XV116T	Valve transfers closed	2.4-6	1.0-7	24	
R81AV104T	Valve transfers closed	2.4-6	1.0-7	24	
R81PMP1AR	Pump 1A fails to run	8.9-5	3.7-6	24	
RMMAKEB		9.4-5			TAB-OR
R81XV132T	Valve transfers closed	2.4-6	1.0-7	24	
R81AV118T	Valve transfers closed	2.4-6	1.0-7	24	
R81PMP1BR	Pump 1B fails to run	8.9-5	3.7-6	24	
RMBACKFLOW		2.6-10			
R81PMP1AR	Pump 1A fails to run	8.9-5	3.7-6	24	R81PMP1AER * R81CV1BT +
R81PMP1BR	Pump 1B fails to run	8.9-5	3.7-6	24	
R81CV1AT	Check valve transfers open	1.5-6	6.1-8	24	R81PMP1BR * R81CV1AT
R81CV1BT	Check valve transfers open	1.5-6	6.1-8	24	

Table A3-2 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
RMISOLATE		2.0-3			TAB-OR
R81AV102C	Valve fails to close	1.0-3	1.0-3		
R81AV109C	Valve fails to close	1.0-3	1.0-3		
R81ACT05Y	Accumulator leaks	7.7-8	2.1-10	365	
RMOHX		5.2-7			TAB-OR
R81HXOXSY	Shell leaks	2.6-7	1.1-8	24	
R81HXOXTY	Tube leaks	2.6-7	1.1-8	24	
RM107		1.0-3			TAB-OR
R81AV1070	Valve fails to open	1.0-3	1.0-3		
R81AV107T	Valve transfers closed	2.4-6	1.0-7	24	
RM103		1.0-3			TAB-OR
R81AV1030	Valve fails to open	1.0-3	1.0-3		
R81AV103T	Valve transfers closed	2.4-6	1.0-7	24	

Table A3-2 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
RMCROSSA		3.0-3			TAB-OR
R81AV359C	Valve fails to open	1.0-3	1.0-3		
R81AV359T	Valve transfers open	2.4-6	1.0-7	24	
RM81AC03AY	Accumulator leaks	7.7-8	2.1-10	365	
R81PI359Y	Valve leaks	2.4-7	1.0-8	24	
J22XV357T	Air valve transfers	2.4-6	1.0-7	24	
R81AV357O	Valve fails to open	1.0-3	1.0-3		
R81AV357T	Valve transfers closed	2.4-6	1.0-7	24	
R81AV358O	Valve fails to open	1.0-3	1.0-3		
R81AV358T	Valve transfers	2.4-6	1.0-7	24	
LMLEAKA		7.9-6	1.1-9*300	24	TAB-OR
L81PINKAY	Leaks in EVSPS A				
L81CTG2AY					
L81TKT1AY					
LMPUMP3A		2.3-4			TAB-OR
L81PMP3AR	Pump fails to run	8.9-5	3.7-6	24	
H81AH52AR	Blower fails to run	1.4-4	5.9-6	24	

Table A3-2 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
LMABHXA		1.4-4			TAB-OR
L81FNABAR	Fan fails to run	1.4-4	5.9-6	24	
LMDAMPERA		7.6-6			TAB-OR
L81DM335T	Damper transfers closed	2.4-6	1.0-7	24	
L81DM336T		2.4-6	1.0-7	24	
L81DM339T		2.4-6	1.0-7	24	
L81DM332T		2.4-6	1.0-7	24	
RMCROSSB		3.0-3			TAB-OR
R81AV420C	Valve fails to close	1.0-3	1.0-3		
R81AV420T	Valve transfers open	2.4-6	1.0-7	24	
R81AC03BY	Leaks	7.7-8	2.1-10	365	
R81PI415Y	Leaks	7.7-8	2.1-10	365	
J22XV415T	Valve transfers	2.4-6	1.0-7	24	
R81AV415O	Valve fails to open	1.0-3	1.0-3		
R81AV415T	Valve transfers	2.4-6	1.0-7	24	
R81AV416O	Valve fails to open	1.0-3	1.0-3		
R81AV416T	Valve transfers	2.4-6	1.0-7	24	

Table A3-2 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
LMLEAKB		7.9-6	1.1-9*300*	24	TAB-OR
L81PINKBY	Leaks				
L81TKT1BY					
L81CTG2BY					
LMPUMP3B		5.4-4			TAB-OR
L81PMP3BS	Pump fails to start	1.0-5	1.0-5		
L81PMP3BR	Pump fails to run	8.9-5	3.7-6	24	
H81AH35AS	Blower fails to start	3.0-4	3.0-4		
H81AM53AR	Blower fails to run	1.4-4	5.9-6	24	
LMMAINTB		3.1-3			TAB-OR
L81NKBM	Maintenance outage	2.1-3	1.1-4	19	
L81NKBL	Latent error	1.0-3	1.0-3		

*Estimated total EVSPS piping as 300 ft and used failure rate of 1.1×10^{-9}/hr-ft.

Table A3-2 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
LMABHXB		4.4-4			TAB-OR
L81FIAXBY	Leak	ε			
L81FNABBS	Fan fails to start	3.0-4	3.0-4		
L81FNABBR	Fan fails to run	1.4-4	5.9-6	24	
LMDAMPERB		4.0-3			TAB-OR
L81DM443S	Damper fails to open	1.0-3	1.0-3		
L81DM443T	Damper transfers closed	2.4-6	1.0-7	24	
L81DM438S		1.0-3	1.0-3		
L81DM438T		2.4-6	1.0-7	24	
L81DM436S		1.0-3	1.0-3		
L81DM436T		2.4-6	1.0-7	24	
L81DM439S		1.0-3	1.0-3		
L81DM439T		2.4-6	1.0-7	24	

Appendix A, Section 4

STEAM GENERATOR SYSTEM (SGS)

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

Section 4

STEAM GENERATOR SYSTEM (SGS)

A4.1 DESCRIPTION

The following sections summarize the information input to the system analysis of the Steam Generator System (SGS). The topics covered are: function, design, interfaces with other systems, operation, test and technical specification requirements, and maintenance requirements. The system description will focus on the use of the SGS in response to an accident initiator although routine full power operation is also described.

A4.1.1 Function

During routine full power operation, the SGS uses the heat energy supplied by the intermediate heat transport system (IHTS) to convert high pressure, sub-cooled water into superheated steam at the pressure and temperature required by the turbine. It also provides the necessary cooling capacity to maintain the IHTS sodium at temperature levels required to cool the reactor when it is shut down.

During the plant response to an accident initiator the SGS serves to transfer heat from the IHTS sodium to a heat sink. The available heat sinks may include the release of steam directly to the atmosphere or to the main condenser, or the flow of steam through the protected air-cooled condensers. (See Section 3.4.3, Summary of Success Criteria and Top Logic, for discussion of the circumstances under which each of

these heat sinks is available and sufficient.) Since the generated steam is no longer supplying the main turbine, the temperature and pressure conditions are not as strict.

Other functions of the SGS are to detect and respond to water-to-sodium and steam-to-sodium leaks in the steam generator modules.

A4.1.2 Design

Each of the three heat transport systems incorporates an independent SGS. Figure A4-1 is a simplified piping and instrumentation diagram of one SGS.

Each SGS consists of the following components:

- 2 evaporator modules,
- 1 superheater module,
- 1 steam drum,
- 1 recirculation pump, and
- water/steam piping and valves.

Note that the steam generator auxiliary heat removal system (SGAHRS) vent valves and condenser are treated as part of that system and not as a part of the SGS.

Each SGS also includes the following subsystems:

- Sodium-water reaction pressure relief subsystem,
- Leak detection subsystem,
- Sodium dump subsystem, and
- Water dump subsystem.

All equipment of the SGS and its subsystems which has an impact on the SGS system analysis is shown in Figure A4-1.

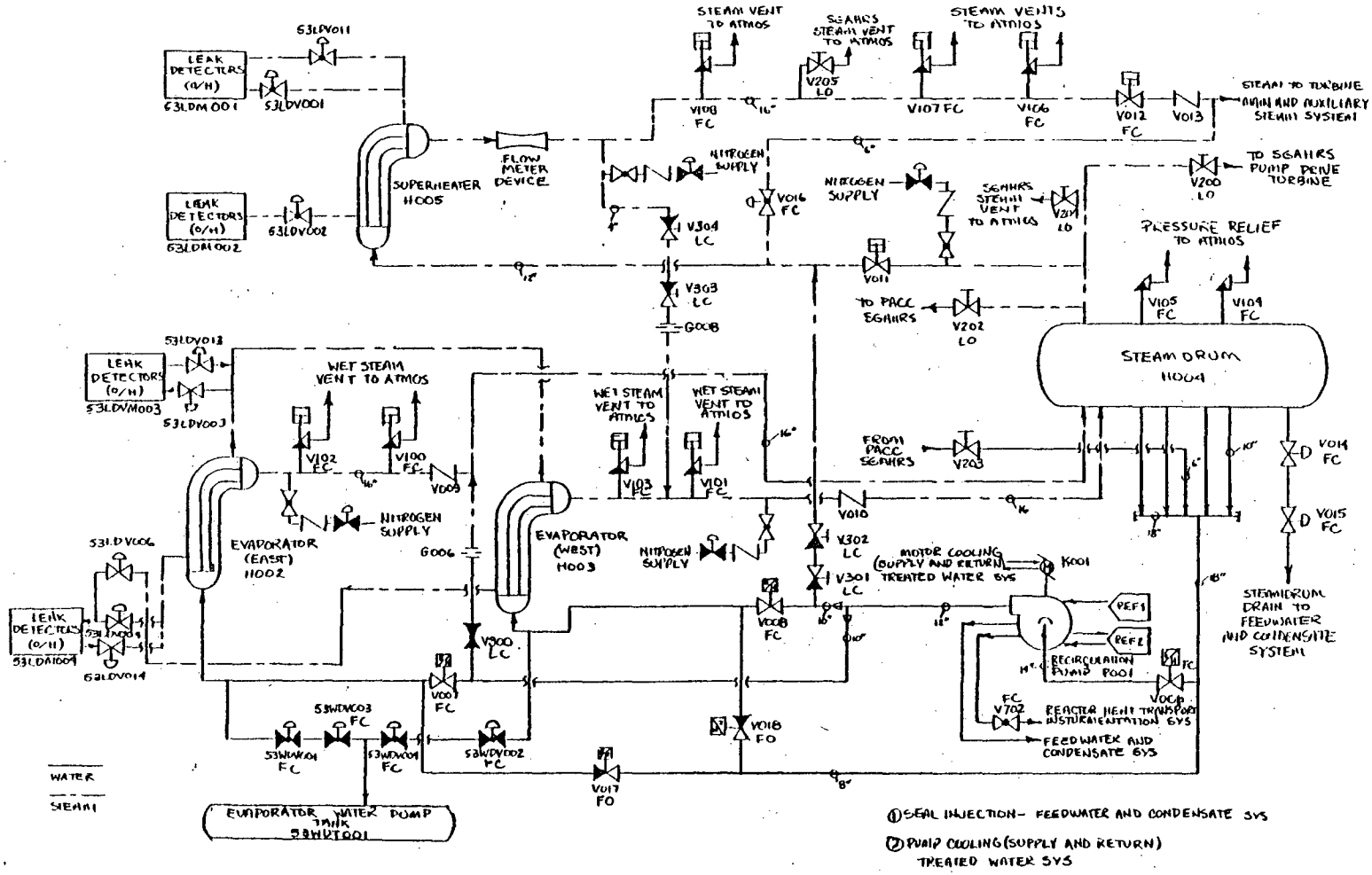


Figure A4-1. Steam Generator System.

A4.1.2.1 Components. The steam drum provides sub-cooled water to the evaporators and saturated steam to the superheater. Separators and dryers, internal to the steam drum, process the saturated water/steam from the evaporators. Saturated steam is sent to the superheater, while the saturated water is subcooled by incoming feedwater and recirculated through the evaporators. The feedwater is provided by the main feedwater system (MFW) or by the auxiliary feedwater system (AFW). The steam drum has a capacity in excess of 400 cubic feet of water at normal water level. This capacity is sufficient to prevent dryout of the steam drum for 10 minutes following a loss of flow from MFW and AFW (see Section A4.2.1, Modeling Assumptions).

The recirculating pump circulates water from the steam drum to the evaporators. The pump operates at a constant speed, 1800 rpm, and delivers a flow of 5920 gpm. This flow rate is twice the rated steam flow. The pump head is 397 ft, and the discharge pressure is 2019 psig.

The recirculation pump is normally provided with seal injection water from the MFW system. When the MFW system is not operating and the recirculation pump is operating, the recirculation pump will provide its own seal injection water, but it depends on support systems for seal water cooling and motor cooling.

If the recirculation pump seals fail, inventory loss can be halted by isolating the recirculation pump. A recirculation pump bypass line is included in the system to allow continued flow of water by natural circulation to the evaporators in the event that the recirculation pump is isolated.

The steam generator modules are the evaporators and the superheater. The evaporators convert the water from the steam drum into saturated water/steam which is returned to the steam drum. The superheater superheats the saturated steam from the steam drum. Although they perform different functions, the evaporators and superheater are of the same design. The steam generator modules are shell and tube heat exchangers. Water/steam flow is on the tube side, and sodium from the IHTS flows through the shell side. A 90° bend in the shell and tube bundle provides flexibility to accomodate differential thermal expansion.

Each SGS is equipped with valves capable of isolating either of the evaporators, the superheater, or the whole SGS loop. There are also valves to isolate the steam drum from main feedwater (MFW) and auxiliary feedwater (AFW). The feedwater isolation valves and control valves, although nominally included in the SGS, have been included in the analysis of the systems they serve.

The steam generator modules all have an inlet isolation valve. These valves are electro-hydraulically operated. The valves on the evaporator inlets are designed to fail in a closed position while the superheater inlet valve is designed to fail as is. Each evaporator outlet line includes a check valve, located downstream of the evaporator power relief valves.

A SGS loop may be isolated from the balance of plant (BOP) by closing the superheater outlet isolation valve (SOIV). The SOIV is an electro-hydraulic valve designed to fail closed. The outlet line is equipped with a check valve located downstream from the SOIV.

The SGS loops include several secondary water and steam lines and their associated valves. There is an evaporator bypass line, a superheater recirculation inlet line and a superheater recirculation outlet line. These lines are included for use during startup, maintenance or off-normal operation. However, use of these lines requires local operator action and is not included in procedures. For these reasons, the use of the secondary flow paths is considered a recovery action and is not included in the analysis of the SGS.

Each SGS is equipped with nine pressure relief valves. There are three safety relief valves between the superheater outlet and the superheater outlet isolation valve, two safety relief valves on the steam drum, and two at the outlet of each evaporator. The set points of the safety relief valves are staggered so that not all valves will open under all circumstances. The first superheater safety relief valve will open at 1800 psig; the last evaporator safety relief valve opens at 2280 psig. Table 3.4-2 displays the set points, flow capacities, and heat removal capacities of all SGS safety relief valves.

The steam drum relief valves function only in the event of over pressurization of the SGS. The superheater and evaporator relief valves, however, are opened in the event of a sodium-water reaction. This function is automatic and independent of the pressure set points. The relief valves reclose automatically when the pressure falls to 300 psig.

A4.1.2.2 Subsystems

Sodium-Water Reaction Pressure Relief Subsystem (SWRPRS)

The sodium-water reaction pressure relief subsystem (SWRPRS) mitigates the effects of a sodium-water reaction in one of the steam generator modules. If a sodium-water reaction takes place, the SGS system containing the reaction will be disabled. This would be true whether or not the SWRPRS response is successful. Successful SWRPRS operation shuts down a SGS, and unsuccessful SWRPRS operation exposes a SGS to an unmitigated sodium-water reaction. Thus the failure of SWRPRS to respond to a sodium-water reaction is not included in the development of the SGS. For this reason, the SWRPRS components are not included in Figure A4-1. The effects of an unmitigated sodium-water reaction are included in the common-cause analysis.

The inadvertent actuation of the SWRPRS shuts down a loop of the SGS. This possibility is included in the model both as a failure of one SGS loop and as an initiator.

The SWRPRS consists of reaction products separator tanks, rupture disk assemblies, and piping. The rupture disk assemblies consist of two rupture disks in series. The assemblies will be located in piping off the evaporator sodium outlets and the superheater sodium inlet. Another assembly will be located in a line bypassing the stop valves of the pressure equalizing line between the sodium dump tank and IHTS sodium expansion tank and pump. The rupture disks are arranged in pairs so that the spurious rupturing of one disk will not actuate SWRPRS and shut down the plant. Pressure monitors and sodium leak detectors will be

installed between the rupture disks to sound an alarm if a leak develops in the sodium-containing rupture disk.

Once the rupture disks have broken, pressure sensors in the lines downstream of the disk assemblies will sound an alarm and automatically isolate the steam generator modules and the steam drum. Rupture of the rupture disk on the equalization line will also initiate a plant trip and will actuate both the power relief valves and the evaporator water dump valves in the affected SGS loop. Reaction products will be conducted by the SWRPRS piping from the steam generator modules to the reaction product separator tanks. The rupture disks on the pressure equalization line are designed to rupture first and will conduct the reaction products from an intermediate sized water/sodium leak to the sodium dump tank.

Leak Detection Subsystem

The leak detection subsystem (LDS) warns the operator of water/sodium leaks small enough that SWRPRS is not immediately actuated. Leaks are detected by in-sodium oxygen and hydrogen detectors which respond to the increase in oxygen and sodium concentrations caused by a sodium water interaction.

A leak detection module consists of three instruments, one diffusion membrane hydrogen detector, and two electrochemical oxygen detectors. A module also contains an electromagnetic sodium pump, a regenerative heat exchanger, a magnet type flow meter, piping, and valves.

There are four leak detection modules on each SGS. One is located on the superheater vent to the IHTS expansion tank, another is located on

the superheater sodium outlet line. The other two are positioned so that one samples sodium in both evaporator vents to the expansion tank, and the other samples sodium from both evaporator sodium outlets.

Any one of the three leak detectors in a leak detector module can generate a signal on the control panel. It should be noted that no automatic actions are initiated when a detector senses a leak.

Sodium Dump Subsystem

The sodium dump subsystem (SDS) drains a loop of IHTS affected by a large sodium/water reaction. Sodium from the IHTS piping and components as well as the sodium side of the steam generator modules is directed to a sodium dump tank. Operator action is required to actuate the SDS.

Inadvertent sodium dump is a serious failure mode. Spurious operation of this subsystem fails a loop of IHTS. Development of this event may be found in the IHTS analysis.

Water Dump Subsystem

In the event of a large sodium water reaction in an evaporator, the amount of water available to be admitted to the IHTS is reduced by the water dump subsystem (WDS). Water dump valves, located near the water inlet of each evaporator, open on a signal from the instrumentation located downstream of the SWRPRS rupture disk assemblies. The water dump system operation also reduces the pressure in the evaporator.

The water dump valves opening allows flow through the water dump piping to an evaporator water dump tank. The tank is equipped with a steam

vent to the atmosphere to relieve the flashed steam. There is one evaporator water dump tank for each SGS. There are two water dump valves, arranged in series, per evaporator. The water dump valves are air-diaphragm valves designed to fail in a closed position.

A4.1.2.3 Interlocks. If both of one evaporator's water dump valves open, the evaporator's inlet valve will close. Similarly, if both safety relief valves on one evaporator open, the evaporator's inlet valve will close.

A4.1.3 Interfaces With Other Systems

The SGS interfaces with many other plant systems. Failures of other systems which affect the operation of the SGS in response to an initiator are summarized below. The detailed and modularized fault trees illustrate the transfers to other fault trees at the equipment requiring support systems.

A4.1.3.1 Electric power. Electric power is supplied to the following pieces of equipment in the SGS.

Equipment Number	Bus Number
53SGV012A	12NIE050A
53SGV012B	12NIE050B
53SGV012C	12NIE044C
53SGV008A	12NIE050A
53SGV008B	12NIE050B
53SGV008C	12NIE044C
53SGK001A	12NIE003C
53SGK001B	12NIE003D
53SGK001C	12NIE003C

A4.1.3.2 Control power. Control power is assumed to come from the same bus as the electric power.

A4.1.3.3 Cooling water. Cooling water is required for the recirculating pump. The treated water system provides motor cooling and seal water cooling. If the internally supplied seal water is unavailable, the main feedwater (MFW) system can supply seal injection.

A4.1.3.4 Actuation systems. The SGS operates during normal plant operation; thus it does not need to be actuated following an initiator. However, the actuation of the steam generator auxiliary heat removal system (SGAHRS) affects SGS performance (see Section A4.1.4.2, Emergency Operation).

A4.1.3.5 Heating, ventilating, and air conditioning. Heating, ventillating, and air conditioning systems are required to cool cells 241, 242, and 243. These cells contain the MFW flow element, superheater inlet and outlet valves, steam drum drain valves and all SGS safety relief valves. One cell contains all the equipment corresponding to a particular loop; cells 241, 242, and 243 contain equipment from loop A, B, and C respectively.

A loss of cooling to one of these cells during a plant blackout may produce temperatures in the room in excess of 125°F after 2 hours and 140°F after 30 hours (Ref. 5). Damage to the electrical equipment in these cells could affect the performance of an SGS loop. This failure mode was found to be a probabilisticall insignificant contributor to the failure of an SGS loop, therefore a detailed analysis on the effect of high temperature in the cell was not undertaken.

A4.1.3.6 Main feedwater system. In addition to providing water to the steam drum, main feedwater also provides seal injection to the recirculation pumps. MFW seal injection is parallel to internal seal water provided by the recirculation pumps.

A4.1.3.7 Treated water system. The treated water system provides motor cooling for the recirculation pumps and cools the internal seal water.

A4.1.4 Operation

The SGS is in operation during both normal full-power conditions, and during emergency conditions.

A.4.1.4.1 Normal operation. During normal operation, the SGS converts water to steam for the turbine. Water is supplied to the steam drum by the main feedwater system. The recirculation pump delivers water to both evaporators. The recirculation pump bypass valves are closed.

Water and steam returns to the steam drum where the water and steam are separated. The water is recirculated to the evaporators while the steam is sent to the superheater and then to the turbine.

A4.1.4.2. Emergency operation. Emergency operation is divided into two phases, short term and long term. The short term involves steam venting, either to the atmosphere through SGAHRS or SGS vents, or to the main condenser by way of the turbine bypass valves. The long term requires the SGS to act as an element in a closed loop, either delivering steam to the SGAHRS condensers or to the main condenser. Makeup to the steam drum is provided by MFW or AFW.

Short Term

As with normal operation, water is converted to steam using the energy from the IHTS. As steam leaves the superheater, it is directed through the turbine bypass valves to the main condenser. The short term begins at the time of the initiator and lasts for 30 minutes to a couple of hours.

If the steam drum water level falls or the ratio of steam flow to feedwater flow is high, then the steam generator auxiliary heat removal system (SGAHRS) is actuated. SGAHRS could be actuated by events such as a loss of MFW to a steam drum or by a break in steam piping. Upon SGAHRS actuation, the superheater outlet isolation valve closes and AFW pumps are started. The closing of the superheater outlet isolation valve precludes continued use of the turbine bypass. Cooling is provided by the SGAHRS steam vents or the SGS relief valves. Although AFW pumps are started, operating MFW pumps are not shut off. On the other hand, if there is sufficient flow from the MFW, the operator may shut off some of the AFW pumps.

Long Term

After the decay and sensible heat has decreased sufficiently, the SGS mission changes to provide cooling through a closed loop system. This would be accomplished using the protected air-cooled condensers (PACCs). Steam from the steam drum, instead of passing through the superheater, is sent to the PACCs. The condensed water is returned to the evaporator inlet header.

Transfer to the long-term mode of operation does not require operator action. The PACCs are actuated at the same time as reactor trip. Steam venting to the main condenser or the atmosphere will cease when the pressure in the SGS system falls below the set points of the turbine bypass valves, SGAHRS vent valves, and SGS relief valves. The pressure falls as core power decays and sensible heat is removed.

If the PACCs are not available and MFW is still operating, long-term cooling may be provided by dumping steam to the main condenser. This mode of cooling is not possible while the superheater outlet isolation valve is closed.

A4.1.4.3 SWRPRS actuation. Regardless of the mode of operation being used by an SGS loop, SWRPRS actuation in that loop will disable the loop. If the plant is at full power when the rupture disks break, the reactor will scram and the sodium pumps will trip. When SWRPRS is actuated, the following actions occur:

- Superheater outlet isolation valve closes,
- Superheater inlet isolation valve closes,
- Evaporator inlet isolation valves close,
- Main feedwater steam drum isolation valve closes,
- Feedwater SGS isolation valve closes,
- Auxiliary feedwater supply valves from turbine and motor driven pumps close,
- Recirculation pump trips,
- SGS power relief valves (on evaporators and superheater) open,
- Water dump valves open, and
- Steam drum drain valves close.

A4.1.4.4 Leak detection. Operator action following some types of leak alarms will disable an SGS loop. Table A4-1 displays the actions to be taken for the various types of leak alarms. A confirmed leak alarm is generated by signals from any two or more leak detectors on the same SGS loop. A single leak detector generates a leak alarm (unconfirmed).

An intermediate confirmed leak alarm requires the operator to blowdown the affected steam generator. Blowdown is affected by isolating the module and opening water dump and power relief valves. If the leaking module cannot be identified or if there is a confirmed high leak, then the operator is required to blowdown all three modules in the affected SGS loop.

A4.1.4.5 Other trip functions. Certain other conditions result in automatic actions for the SGS.

High Steam Drum Level

If the steam drum level rises to eight inches above the normal water level, MFW flow to the steam drum and flow from the motor driven AFW pumps to the steam drum are terminated by closing isolation valves. If the steam drum level rises to 12 inches above the normal water level, the feedwater steam generator building isolation valve closes and the steam drum is isolated from the AFW turbine driven pump.

Low Superheater Outlet Pressure

If the superheater outlet pressure falls to 110 psig, the superheater outlet isolation valve closes. The superheater bypass valve closes as well.

Table A4-1

OPERATOR ACTION FOLLOWING LEAK DETECTION SUBSYSTEM ALARMS

Alarm	Leak Size (lb/Sec)	Operator Action
Leak Alarm		
Low	Less than 2×10^{-5}	1. Monitor Leak Data 2. Initiate Leak Location Procedure
Intermediate	2×10^{-5} to 6.5×10^{-3}	1. Monitor Leak Data 2. Initiate/Continue Leak Location Procedure
High	$>6.5 \times 10^{-3}$	1. Monitor Leak Data 2. Prepare to Initiate Shutdown or Scram 3. Initiate/Continue Leak Location Procedure
Confirmed Leak		
Low	Less than 2×10^{-5}	1. Monitor Leak Data 2. Reduce to 40% Power 3. Continue Leak Location Procedure
Intermediate	2×10^{-5} to 6.5×10^{-3}	1. Initiate Shutdown 2. Continue Leak Location Procedure 3. Blowdown Affected Module*
High	$>6.5 \times 10^{-3}$	1. Initiate Reactor Scram 2. Initiate Loop Blowdown of All 3 Modules

*If affected module cannot be identified, blowdown all 3 modules in the affected loop.

Low Steam Drum Pressure

If the pressure in the steam drum falls to 500 psig, design features to limit the effects of a main feedwater pipe break are actuated. The superheater inlet isolation valve closes, the steam drum is isolated from MFW, and the steam drum drain valves close. If the pressure continues to fall to 200 psig, the steam drum is isolated from AFW.

Cell Temperature and Humidity

If the temperature and humidity in one of the four SGS cells exceed 150°F and 80% RH, then the SGB feedwater isolation valve and the feedwater control valves close.

A4.1.5 Test and Technical Specifications

A4.1.5.1 Testing. In-service inspection of all non-sodium components will be performed in compliance with the ASME Boiler and Pressure Vessel Code, Section XI, Division I requirements.

Following an accident, all SGS components and piping must be pneumatically or hydrostatically tested. Each component which can be isolated is tested separately. Since the steam drum and evaporators have no leaktight isolation valves, they are tested as a single component.

A4.1.5.2 Technical specifications. The following list of specifications is from SDD-53.[2]

a. During reactor operation, at least two SGS loops shall be operable, corresponding to the two operable PHTS and IHTS loops.

b. During reactor operation, the steam drum water level in each operating loop shall not be more than seven inches below, nor more than nine inches above, the drum centerline.

c. During reactor operation, all safety and power relief valves on the operating loops shall be operable.

d. During reactor operation, the steam drum pressure in each operating loop shall not be less than 200 psig.

e. Whenever a SGS loop is operating, its associated recirculation pump shall be operating.

f. At least one recirculation loop shall be operable when the reactor is shutdown but reactor criticality is possible.

g. The water chemistry of each operating SGS loop shall meet the requirements shown in Sections A-1-1 and A-1-2 of Appendix C.

h. Whenever there is water/steam on the tube side of the evaporator/superheater and sodium on the shell side of the evaporator/superheater, the SWRPRS shall be operable. Also during these conditions, the power relief valves shall be operable.

i. Whenever the SWRPRs system is required to be operable, the evaporator water dump subsystem shall be operable.

j. Whenever the water dump subsystem is required to be operable, the water dump tank shall not contain more than six inches of liquid.

k. Whenever the SWRPRS is required to be operable, the sodium dump tank (SDT) shall be operable.

l. When the sodium dump tank is required to be operable, the piping leading to the dump tank rupture disc shall be trace heated to 400°F ± 50°F.

A4.1.6 Maintenance Requirements

Scheduled maintenance will be performed during reactor outages.

Unscheduled maintenance will also be performed during outages, whenever practical.

A4.2 SYSTEM MODEL

The following sections describe the model of the SGS system. Both the detailed and the modularized fault trees are described.

A4.2.1 Modeling Assumptions

a. Electric components will fail when the temperature in their cell reaches 120°F. Although the actual tolerance of 1E power systems may be higher in some cells, this temperature was used as a screening value. If HVAC failures become a major contributor to plant risk, a more detailed analysis is possible.

b. With the normal steam drum water level, there is 10 minutes worth of water in the steam drum following a loss of all makeup. This information was transferred in a meeting with Westinghouse on October 5, 1983.

c. Natural circulation is effective both through the recirculation pump bypass lines and through the pump itself when it is not operating.

d. SGS relief valves are a viable heat sink when steam drum makeup is provided by one MFW pump or the turbine driven AFW pump or by both motor driven pumps operating together.

e. An SGS loop may not be reflooded following dryout.

A4.2.2 Detailed Fault Tree Model

The SGS fault tree is constructed in a cascading mission fashion. In this way each of the missions the SGS may need to perform may be modeled by transferring to the appropriate level of the tree. For example, at the top of the fault tree is the event "superheater to turbine bypass flow blocked or diverted," and lower in the tree is the event "insufficient flow to west evaporator." The first event would be used when modeling turbine bypass operation, but would be omitted when using the second event to model steam venting through west evaporator safety relief valves.

A4.2.2.1 Success criteria. The SGS serves several functions in an emergency. The success criteria depend upon which function is being performed. In general, the SGS must convert water from the steam drum into steam, and deliver the steam to a prescribed heat sink. Possible

heat sinks include the main condenser, PACCs, SGAHRS vents, and SGS relief valves.

Whichever of the missions is performed, the SGS requires one evaporator and either forced or natural circulation of water from the steam drum to the evaporator. Missions which involve venting steam, either to the main condenser or to the atmosphere, require feedwater makeup to the steam drum. Long-term closed-loop missions require that no significant leaks exist.

A.4.2.2.2 Top events. Depending on what mission the SGS is required to fill, the top logic fault tree transfers either to the entire SGS fault tree or a subtree. Table A4-2 lists all the gates to which the top logic transfers.

A4.2.2.3 Transfers to other systems. Some of the systems required by the SGS are not modeled in the SGS fault tree. These dependencies are included in the top logic fault tree. One example of such a dependency is makeup to the steam drum. Since the SGS sometimes operates along with MFW, sometimes with AFW, and possibly with both MFW and AFW, the dependence was not built into the SGS fault tree. Instead, the top logic keeps track of the operating environment of the SGS.

A4.2.2.4 Description of fault tree. The detailed fault tree is shown in Figure A4-2. The top gate "Loop α superheater to turbine bypass flow blocked or diverted" (α = A, B, or C. Each gate is repeated in the fault tree of all three SGS loops. Since each gate name ends in alpha, the symbol α will be omitted in this description), Q100, is the gate to

Table A4-2

TOP GATES DEVELOPED
FOR THE STEAM GENERATING SYSTEM

Gate No.	Gate Description
Q100A	Loop A superheater to turbine bypass flow blocked or diverted
Q100B	Loop B superheater to turbine bypass flow blocked or diverted
Q100C	Loop C superheater to turbine bypass flow blocked or diverted
Q400A	No flow to superheater from Steam Drum Loop A
Q400B	No flow to superheater from Steam Drum Loop B
Q400C	No flow to superheater from Steam Drum Loop C
Q300A	No flow to Steam Drum from Evaporators Loop A
Q300B	No flow to Steam Drum from Evaporators Loop B
Q300C	No flow to Steam Drum from Evaporators Loop C
Q303A	Insufficient flow to West A Evaporator Inlet
Q303B	Insufficient flow to West B Evaporator Inlet
Q303C	Insufficient flow to West C Evaporator Inlet
Q304A	Insufficient flow to East A Evaporator Inlet
Q304B	Insufficient flow to East B Evaporator Inlet
Q304C	Insufficient flow to East C Evaporator Inlet
Q700A	Steam Drum A Inventory Lost Through Steam Drum Drain
Q700B	Steam Drum B Inventory Lost Through Steam Drum Drain
Q700C	Steam Drum C Inventory Lost Through Steam Drum Drain

which the top logic fault tree transfers when developing failure of balance of plant cooling. Superheater to turbine bypass flow may fail due to failure of the check valve, superheater outlet isolation valve, or failures precluding flow to the superheater.

Gate Q101, superheater outlet isolation valve closed, may be caused by SWRPRS actuation, SGAHRS actuation, or by failures of the valve and its support systems.

Gate Q200, SWRPRS actuates, is caused by a steam generator tube rupture, an inadvertent SWRPRS actuation either as an initiator or during the accident, or overpressurization of the IHTS cover gas which breaks the SWRPRS rupture disks.

Gate Q400, no flow to superheater from steam drum, is the gate to which the top logic fault tree would transfer when developing failure of steam venting through superheater SGAHRS vents or SGS relief valves. The event may be caused by failure of the superheater inlet isolation valve, or HVAC failures which raise the temperature in the cell to the point where electrical failures close the valve. The inlet isolation valve may also be closed by SWRPRS actuation or a low steam drum pressure signal. The last input to gate Q400 is no flow to steam drum from evaporators.

Gate Q401, superheater inlet valve closed by low steam drum pressure signal, may be caused by a spurious low pressure signal or by a real leak in the SGS loop. A leak may be a break in a SGS component or piping, or a failure of a vent valve.

Gate Q300, no flow to steam drum from evaporators, is the gate used by top logic when the failure of PACC operation or steam venting from the steam drum is developed. Loss of flow to the steam drums from the evaporators occurs only if flow ceases from both evaporators. Flow from one evaporator is lost if the check valve at the evaporator outlet fails, or if there is no flow to the evaporator from the steam drum.

Gates Q303 and Q304, no flow to west (for Q303) and east (for Q304) evaporators from the steam drum, are the gates called by top logic when developing the failure of steam venting from evaporator safety relief valves. Loss of flow to an evaporator can occur by diverting the flow to the evaporator water dump tank by way of the water dump valves, or by insufficient flow to the evaporator inlet line.

Gates Q307, and Q309, flow diverted to water dump tank, have as inputs SWRPRS actuation, spurious water dump signal, and the inadvertent opening of both in-line water dump valves.

Gates Q500 and Q600, insufficient flow to evaporator inlet, is true only if both the forced flow through the recirculation pump and the natural circulation through the pump bypass lines are insufficient. Natural circulation through the pump is treated as a recovery action because the operator will isolate the pump following pump failure.

Forced flow may be failed by closing the pump inlet valve, by pump failures, or by insufficient flow from the pump to the evaporator. Flow between the pump and evaporator may be failed by closing the evaporator inlet valve or by establishing a flow loop which bypasses the evaporators. The evaporator inlet will close due to interlocks with the safety

relief valves and water dump valves on that evaporator, or by SWRPRS actuation, or by failures of the valve and its support systems. Since the evaporator inlet valve is designed to fail in a closed position, loss electric power to the valve will cause it to close.

A flow loop bypassing the evaporators, gates Q507 and Q607, could be established only if the pump is operating and the valves blocking such a loop are open. The pump must be running because, without the evaporator, no natural convection effects will drive water back up to the steam drum. Water exiting the pump may return directly to the steam drum through evaporator bypass line or to the pump suction through one of the pump bypass lines.

The recirculation pump can be failed (gate Q511) by a loss of power or by mechanical and support system failures. The pump must have motor cooling and seal integrity. Seal integrity will only fail when both the internal seal water and the external seal injection fail. External seal injection is provided by MFW. Internal seal water fails if the pump stops operating or if the treated water system fails to cool the seal water.

Natural circulation through the recirculation pump bypass line may fail (Q502 and Q602) if the bypass valve fails to open or transfers closed after opening.

Gate Q700, steam drum inventory lost through steam drum drain, is the gate to which top logic transfers when developing failure of PACC cooling or long-term turbine bypass operation. The drum drain valve may

be left open by HVAC failures which raise the cell temperature enough to cause electrical faults thus preventing the drum drain valves from closing, or by failures of both valves to close or stay closed.

A4.2.2.5 Notes on SGS detailed fault tree.

1. Superheater recirculation return line (containing valves 304 and 303) is not included as a flow diversion for "no flow from west evaporator to steam drum loop α." This flow diversion would not fail the cooling mission unless there were a stuck open SRV or SGAHRS vent. This failure is already modeled under loss of feed flow caused by stuck open steam relief lowering the steam drum pressure. An open superheater recirculation line would, however, defeat the isolation of the superheater in such a scenario. If failure of the SH outlet isolation valve to close is important, this line could defeat SH isolation in this case as well.

2. Evaporator SRVs were not included as a flow diversion for "no flow from west evaporator to steam drum loop α." This flow diversion would result in SGS isolation from feedwater due to lost steam drum pressure. This failure is already included in the top logic for event S.

3. Superheater inlet recirculation line (containing valves V301 and V302) was not included under "insufficient flow from recirculation pump α to west (or east) evap α." This flow diversion is only a failure when the superheater has been isolated following a loss of integrity--SRV stuck open or SGAHRS vent stuck open etc. This has been considered where appropriate in event C and S top logic.

A4.2.3 Modularized Fault Tree Model

Table A4-3 shows the definition of the modules. Because of the many different ways in which the SGS is used, components have several purposes. Thus the modules tend to be small.

Some simplifications were made on the modularized tree. The transfer to TS1G4 from gate Q101 was eliminated. The top logic already keeps track of the environment in which the SGS must operate. If SGAHRS has been

actuated, then the turbine bypass mission has failed in the top logic and the transfer to SGS will not occur.

All events concerning the circuit breakers and cables to the electric buses were eliminated. These faults have been accounted for in the electric power logic.

The recirculation pump failure and electric power failure were deleted from the inputs to gate Q513. These failures are redundant logically since they already appear as inputs to gate Q511.

A4.3 FAILURE DATA

Data was applied to the fault tree according to the rules and tables found in Chapter 5 and 6. The modules were calculated using the basic event probabilities and the module definitions. Results of these calculations can be found in Table A4.3.

A4.4 SYSTEM LEVEL RESULTS AND INSIGHTS

To obtain the system level results, the fault trees for the SGS were solved for gates Q100A, B and C. Thus the results apply primarily to scenarios in which SGS is supplying steam to the main condenser through the turbine bypass valves. The dominant contributor to the failure of this mission is the failure of one superheater relief valve to close combined with the failure to isolate the superheater. The relief valve set points are staggered such that one of the superheater relief valves will open first. This relief valve will open briefly following reactor trip due to the short delay in the opening of the turbine bypass valves.

The venting of steam to the atmosphere combined with the normal flow of steam to the condenser will result in a loss of pressure in the affected SGS loop. At 500 psig the loop will be automatically isolated from feedwater and the SOIV will be shut. The frequency of this failure was found to be 2×10^{-2} per demand of an SGS loop.

The next highest frequency failure were the spurious operation of the water dump subsystem or the SWRPRS. Either of these failures could occur during any phase of the SGS operation. The frequency of each of these failures was assessed to be 1×10^{-3}.

Other significant contributors to the loss of a SGS loop are events which depressurize the loop to the point when the loop will be automatically isolated. These events are failures or mis-calibrations of the SGS safety relief valves. Failures leading to closing of the superheater inlet valve are also significant. In particular, a spurious signal from the steam drum pressure monitors would close the valves. Such an event would not affect PACC operation or steam venting from the steam drum or evaporators.

An important insight into the design of the SGS is that the presence of the safety relief valves is both a significant contributor to safety and the unavailability of the SGS loops. With the relief valves, the ability to cool the plant using steam venting is assured. However, if the valves are stuck open when they are not required, depressurization of the SGS loop will result.

A4.5 REFERENCES

1. Project Management Corporation, CRBRP Preliminary Safety Analysis Report.

2. General Electric Company, A Reliability Assessment of the CRBRP Shutdown Heat Removal System, CRBRP-GEFR-00552, Revision 2, January 1983.

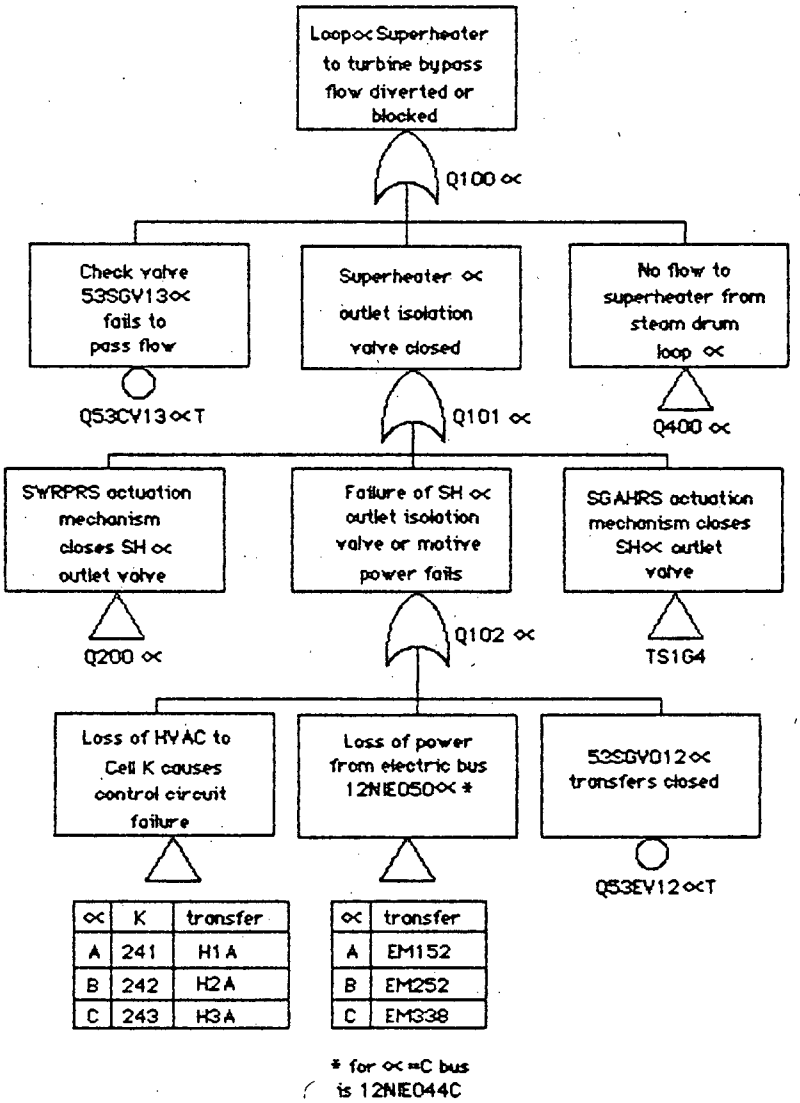
3. Letter from A. T. Dajani, Burns and Roe Inc. to Sharon Z. Bruske, E.G.&G. Idaho, Inc., BZ30272, February 23, 1983.

4. Burns and Roe, Inc., Auxiliary Cooling Systems Internal Design Review, Final report draft, February 1981.

5. General Electric Company, CRBRP System Design Description - Steam Generator System, SDD53, Revision 100, July 1983.

6. General Electric Company, CRBRP System Design Description - Reactor Heat Transport Instrumentation System, SDD 56, Revision 99, July 1983.

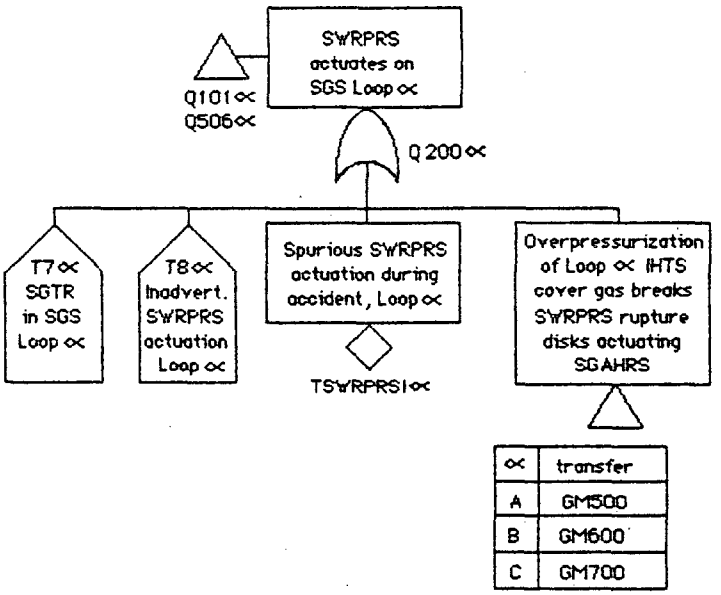
7. Burns and Roe, Inc., CRBRP System Design Description - Building Electrical Power System, SDD 12, Revision 19, February 1983.

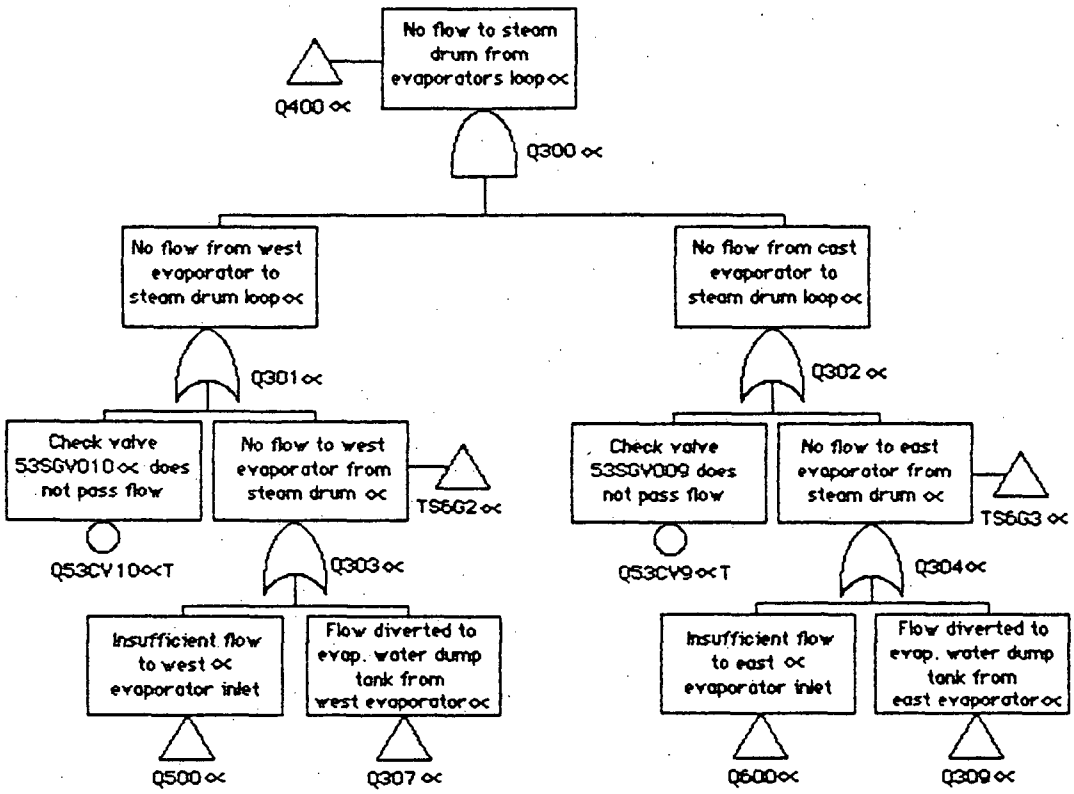


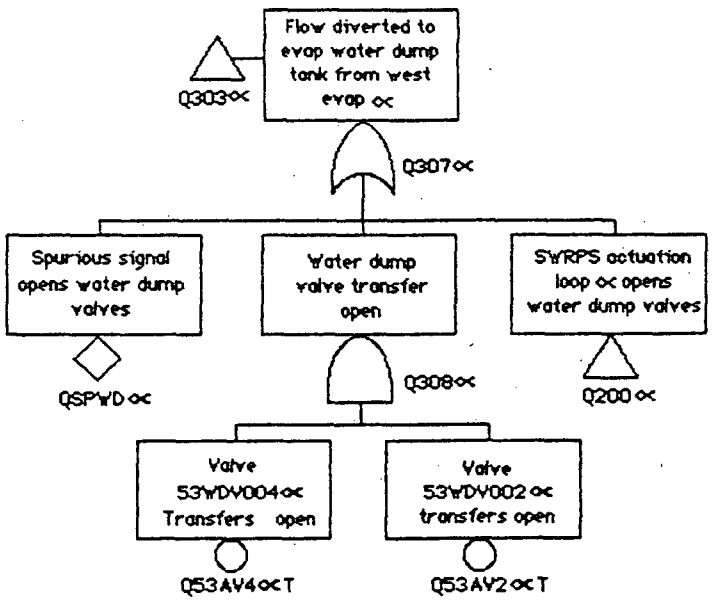
∝	K	transfer
A	241	H1A
B	242	H2A
C	243	H3A

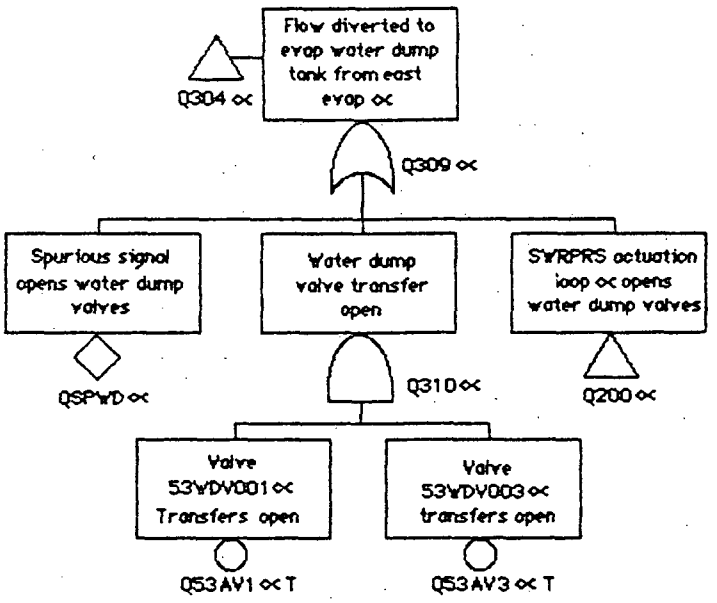
∝	transfer
A	EM152
B	EM252
C	EM338

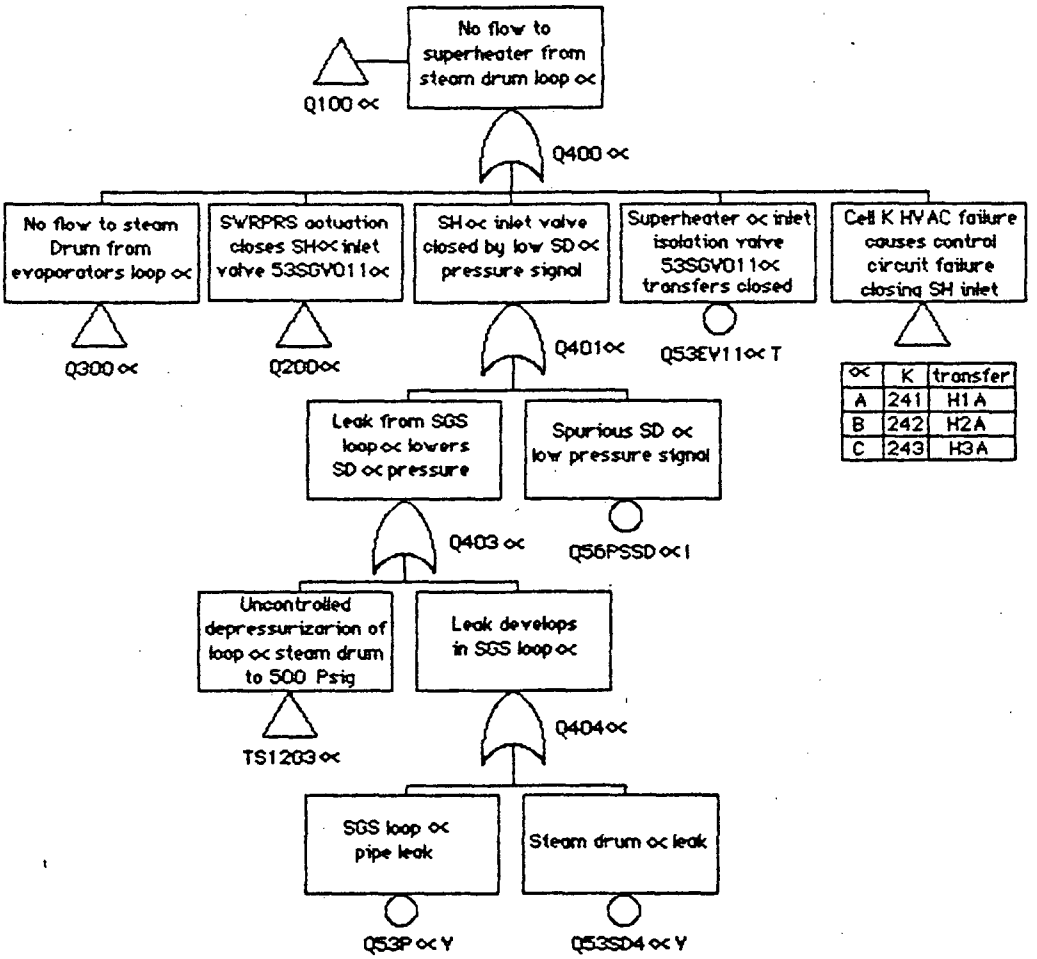
Figure A4-2. Steam Generator System Detailed Fault Tree.


SWRPRS actuates on SGS Loop α
Q101α
Q506α
Q200α
T7α SGTR in SGS Loop α
T8α Inadvert. SWRPRS actuation Loop α
Spurious SWRPRS actuation during accident, Loop α
TSWRPRS1α
Overpressurization of Loop α IHTS cover gas breaks SWRPRS rupture disks actuating SGAHRS
α transfer
A GM500
B GM600
C GM700

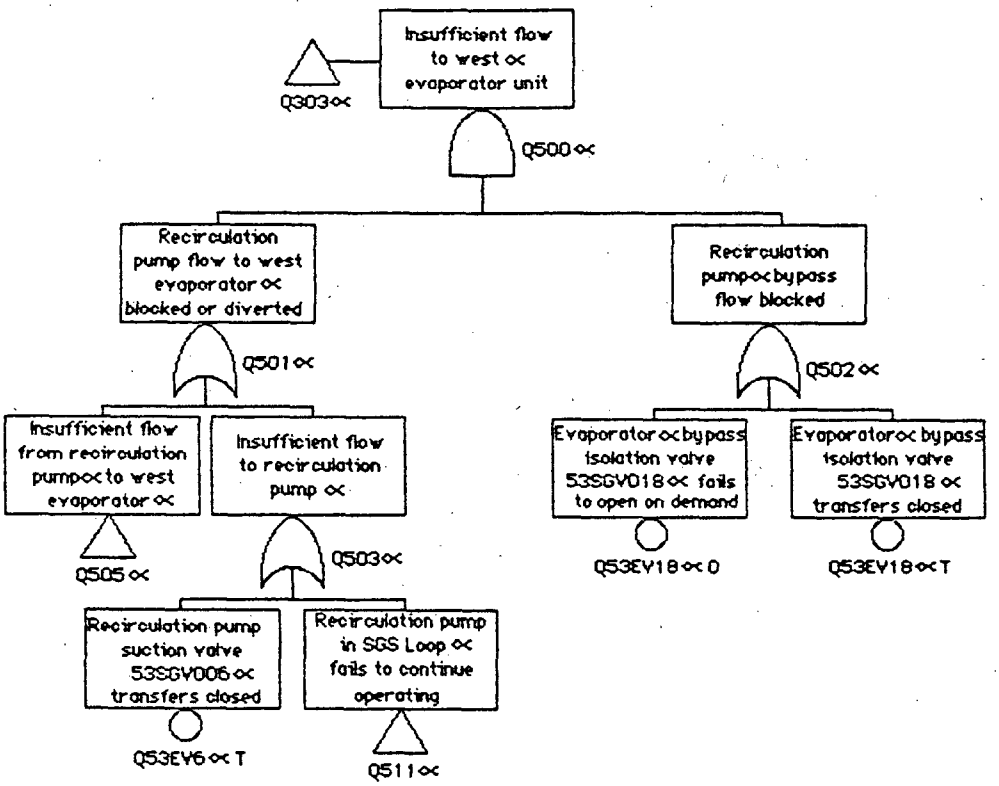

No flow to steam drum from evaporators loop∝
Q400∝
Q300∝
No flow from west evaporator to steam drum loop∝
No flow from east evaporator to steam drum loop∝
Q301∝
Q302∝
Check valve 53SGV010∝ does not pass flow
No flow to west evaporator from steam drum ∝
TS6G2∝
Check valve 53SGV009 does not pass flow
No flow to east evaporator from steam drum ∝
TS6G3∝
Q53CV10∝T
Q303∝
Q53CV9∝T
Q304∝
Insufficient flow to west ∝ evaporator inlet
Flow diverted to evap. water dump tank from west evaporator∝
Insufficient flow to east ∝ evaporator inlet
Flow diverted to evap. water dump tank from east evaporator∝
Q500∝
Q307∝
Q600∝
Q309∝

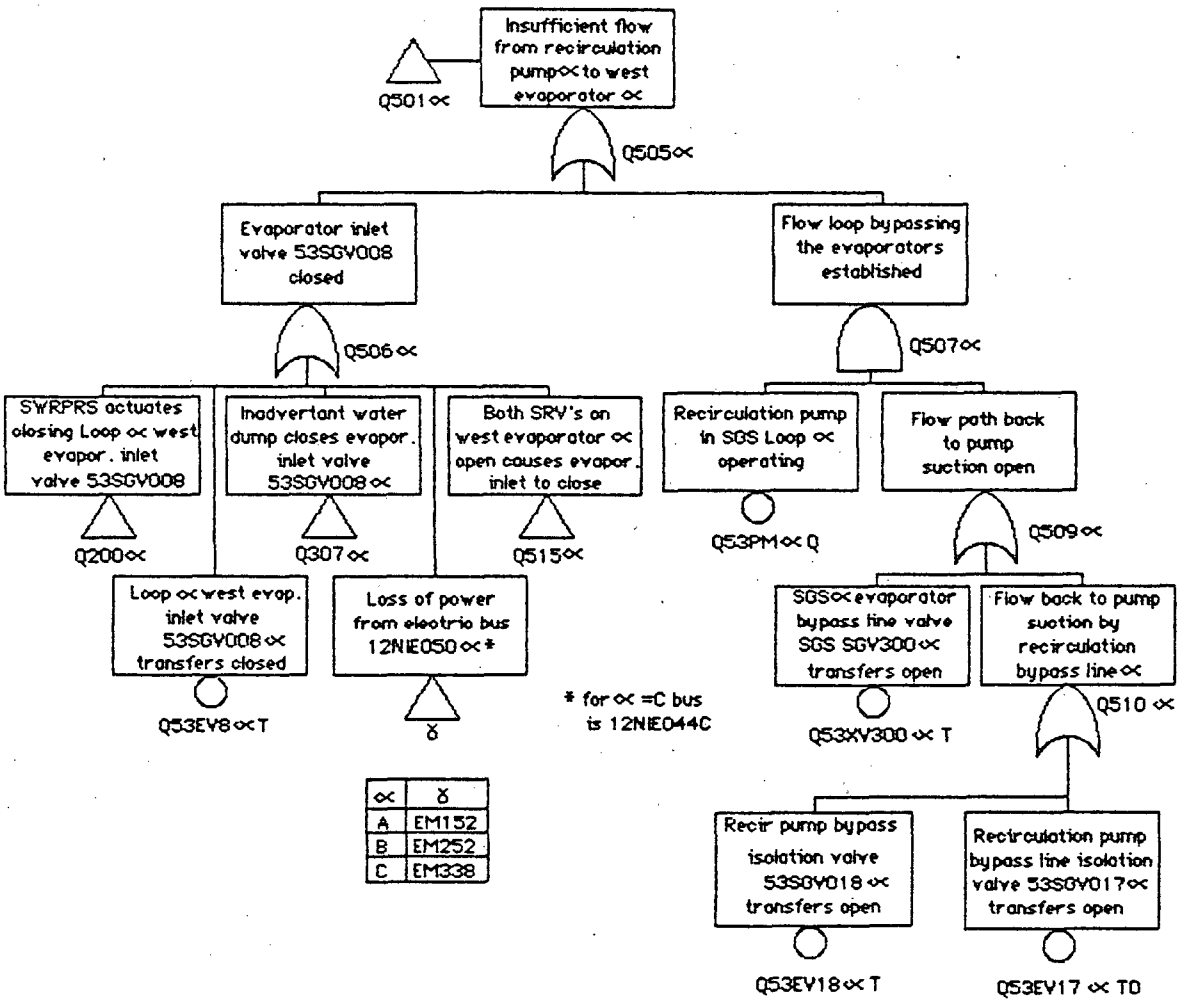

Flow diverted to evap water dump tank from west evap oc
Q303oc
Q307oc
Spurious signal opens water dump valves
Water dump valve transfer open
SWRPS actuation loop oc opens water dump valves
QSPWD oc
Q308oc
Q200 oc
Valve 53WDV004oc Transfers open
Valve 53WDV002 oc transfers open
Q53AV4ocT
Q53AV2ocT


Flow diverted to evap water dump tank from east evap ∝
Q304 ∝
Q309 ∝
Spurious signal opens water dump valves
Water dump valve transfer open
SWRPRS actuation loop ∝ opens water dump valves
QSPWD ∝
Q310 ∝
Q200 ∝
Valve 53WDV001 ∝ Transfers open
Valve 53WDV003 ∝ transfers open
Q53AV1 ∝ T
Q53AV3 ∝ T



∝	K	transfer
A	241	H1A
B	242	H2A
C	243	H3A


Insufficient flow to west ∝ evaporator unit
Q303∝
Q500∝
Recirculation pump flow to west evaporator ∝ blocked or diverted
Recirculation pump∝bypass flow blocked
Q501∝
Q502∝
Insufficient flow from recirculation pump∝to west evaporator ∝
Insufficient flow to recirculation pump ∝
Evaporator∝bypass isolation valve 53SGV018∝ fails to open on demand
Evaporator∝bypass isolation valve 53SGV018 ∝ transfers closed
Q505∝
Q503∝
Q53EV18∝O
Q53EV18∝T
Recirculation pump suction valve 53SGV006∝ transfers closed
Recirculation pump in SGS Loop ∝ fails to continue operating
Q53EV6∝T
Q511∝



∝	δ
A	EM152
B	EM252
C	EM338

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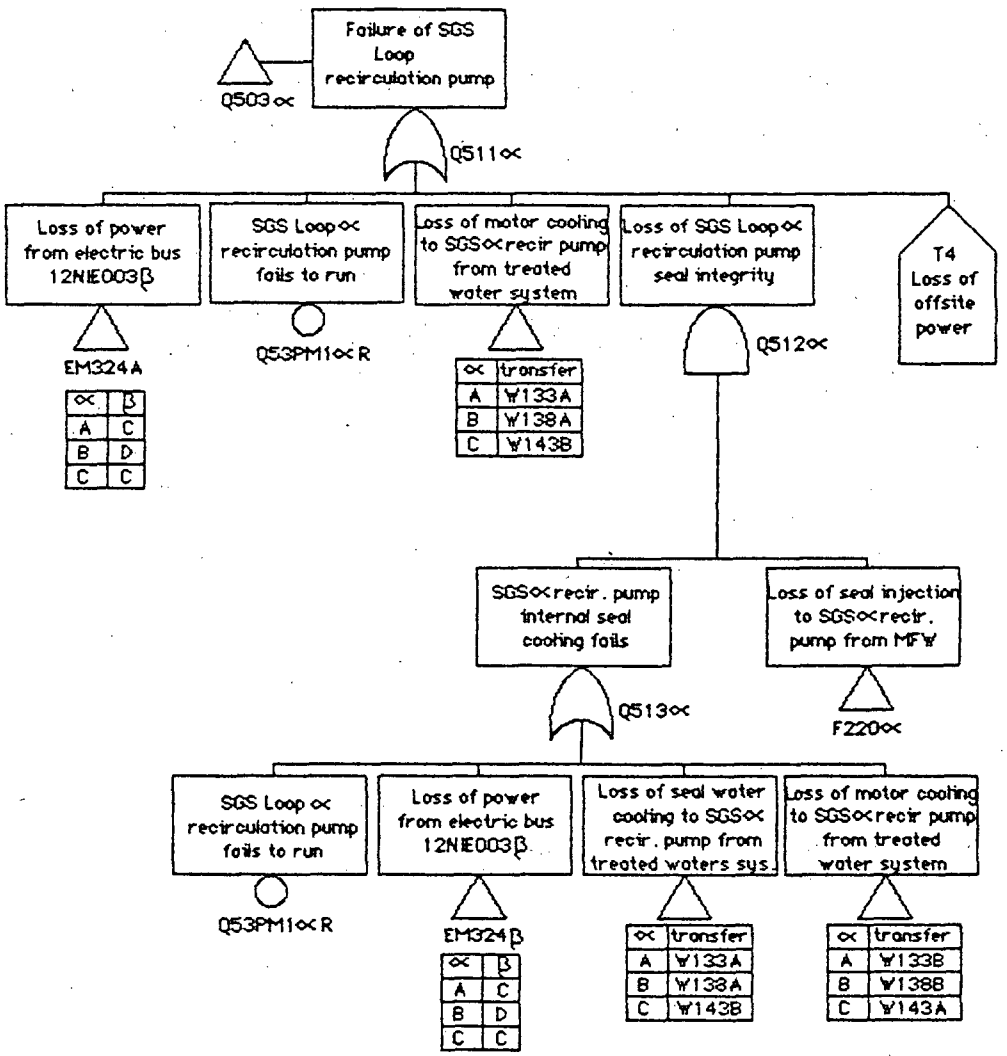
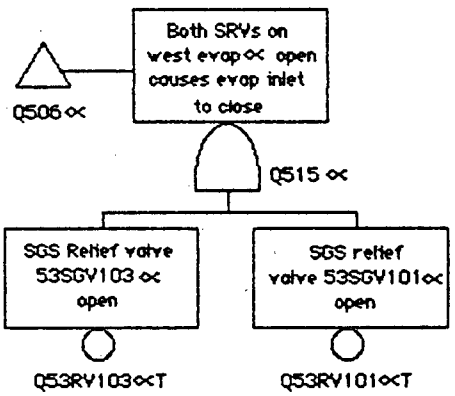
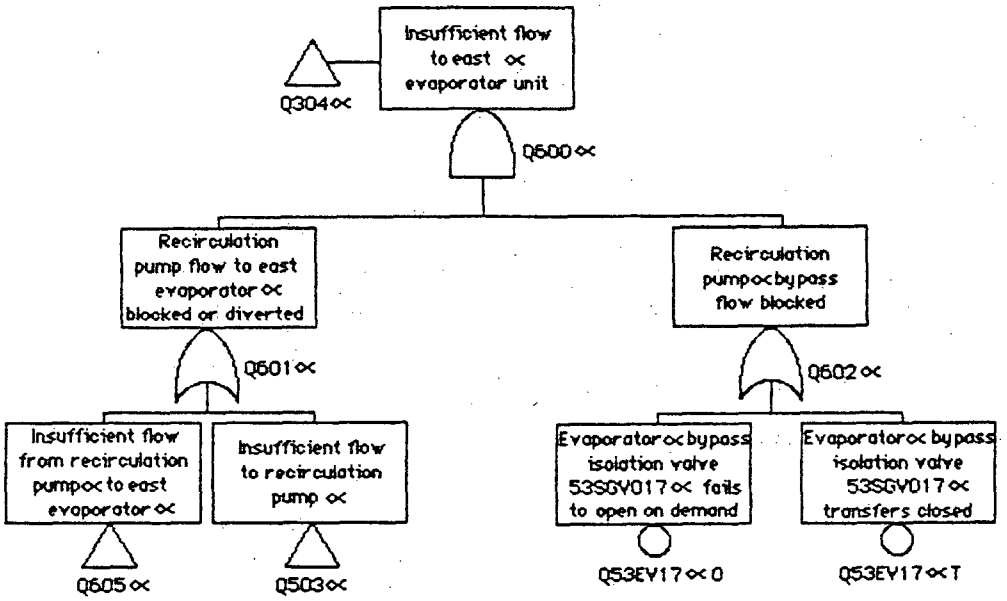
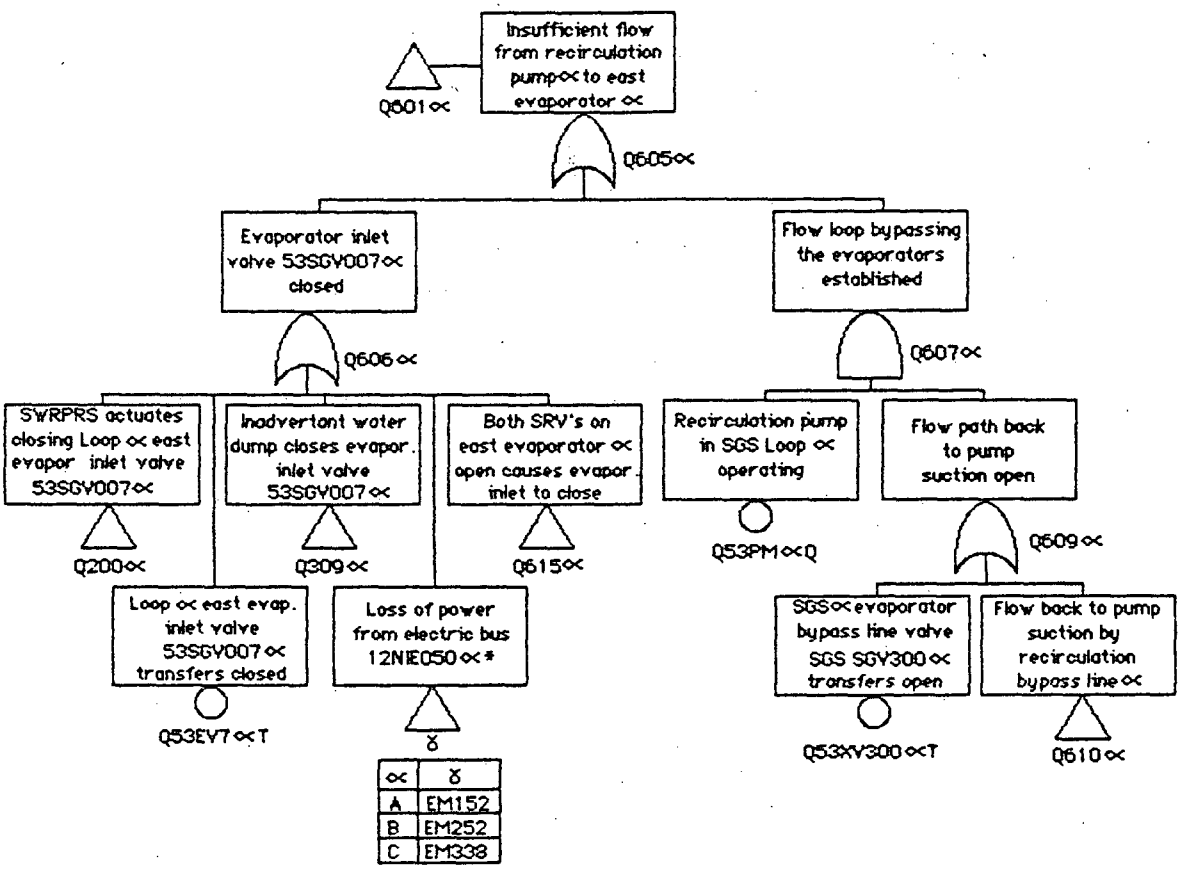
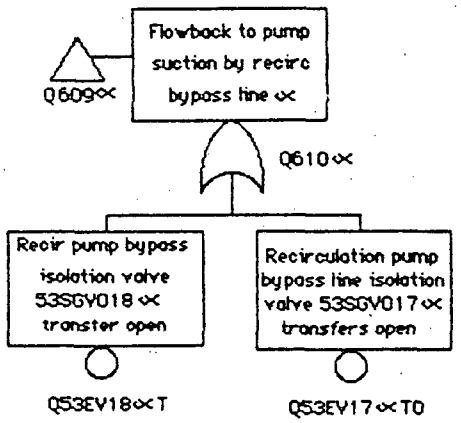
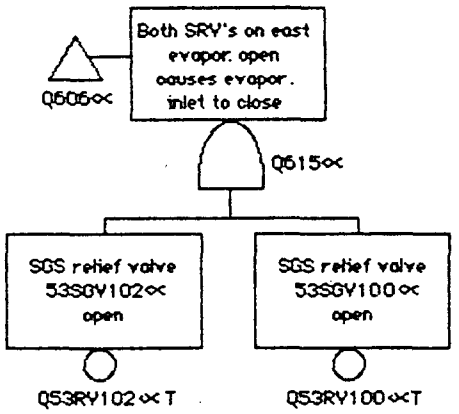

Failure of SGS Loop recirculation pump
Q503∝
Q511∝
Loss of power from electric bus 12NIE003β
SGS Loop∝ recirculation pump fails to run
Loss of motor cooling to SGS∝recir pump from treated water system
Loss of SGS Loop∝ recirculation pump seal integrity
T4 Loss of offsite power
EM324A
Q53PM1∝R
Q512∝
∝ β
A C
B D
C C
∝ transfer
A W133A
B W138A
C W143B
SGS∝recir. pump internal seal cooling fails
Loss of seal injection to SGS∝recir. pump from MFW
Q513∝
F220∝
SGS Loop ∝ recirculation pump fails to run
Loss of power from electric bus 12NIE003β
Loss of seal water cooling to SGS∝ recir. pump from treated waters sys.
Loss of motor cooling to SGS∝recir pump from treated water system
Q53PM1∝R
EM324β
∝ β
A C
B D
C C
∝ transfer
A W133A
B W138A
C W143B
∝ transfer
A W133B
B W138B
C W143A

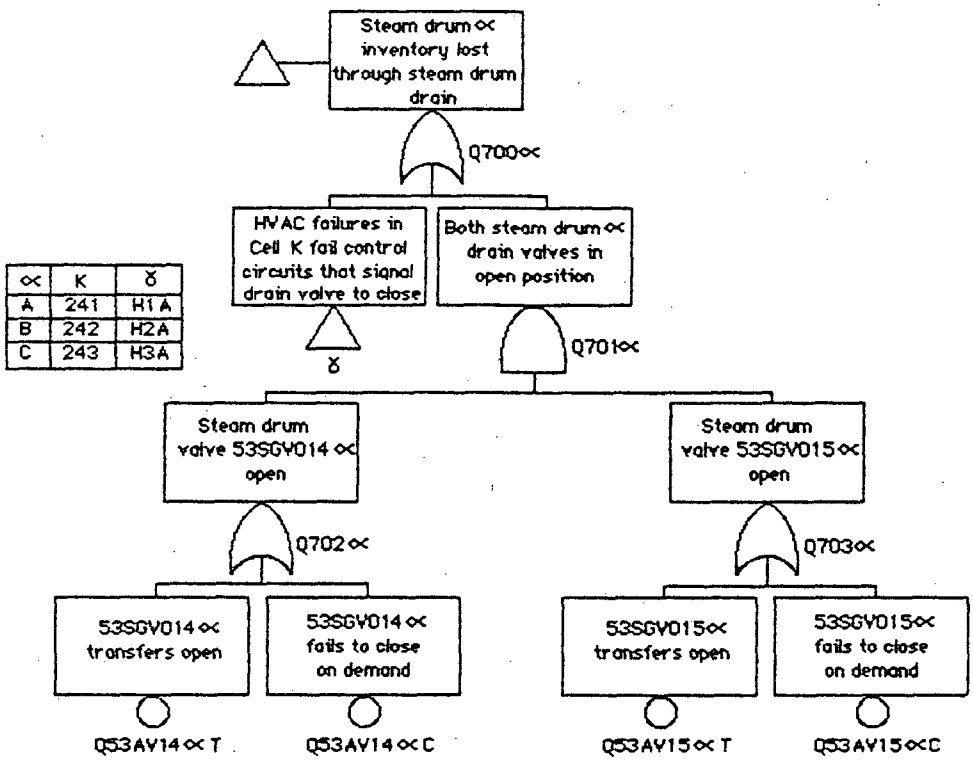

Both SRVs on west evap∝ open causes evap inlet to close
Q506∝
Q515∝
SGS Relief valve 53SGV103∝ open
SGS relief valve 53SGV101∝ open
Q53RV103∝T
Q53RV101∝T


Insufficient flow to east ∝ evaporator unit
Q304∝
Q600∝
Recirculation pump flow to east evaporator∝ blocked or diverted
Recirculation pump∝bypass flow blocked
Q601∝
Q602∝
Insufficient flow from recirculation pump∝ to east evaporator∝
Insufficient flow to recirculation pump ∝
Evaporator∝bypass isolation valve 53SGV017∝ fails to open on demand
Evaporator∝ bypass isolation valve 53SGV017∝ transfers closed
Q605∝
Q503∝
Q53EV17∝O
Q53EV17∝T

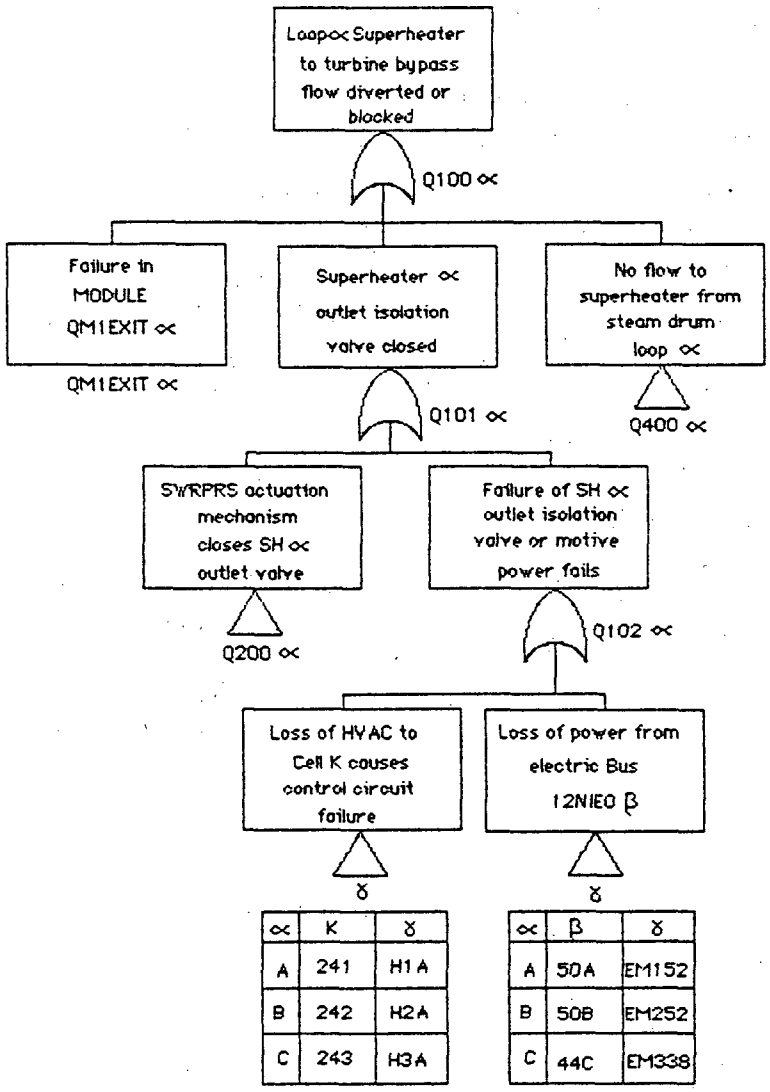

Insufficient flow from recirculation pump∝ to east evaporator ∝
Q601∝
Q605∝
Evaporator inlet valve 53SGV007∝ closed
Flow loop bypassing the evaporators established
Q606∝
Q607∝
SWRPRS actuates closing Loop ∝ east evapor inlet valve 53SGV007∝
Inadvertant water dump closes evapor. inlet valve 53SGV007∝
Both SRV's on east evaporator ∝ open causes evapor. inlet to close
Recirculation pump in SGS Loop ∝ operating
Flow path back to pump suction open
Q200∝
Q309∝
Q615∝
Q53PM∝Q
Q609∝
Loop ∝ east evap. inlet valve 53SGV007∝ transfers closed
Loss of power from electric bus 12NIE050∝*
SGS∝ evaporator bypass line valve SGS SGV300∝ transfers open
Flow back to pump suction by recirculation bypass line∝
Q53EV7∝T
γ
Q53XV300∝T
Q610∝
∝ γ
A EM152
B EM252
C EM338
* for ∝ =C bus is 12NIE044C


Flowback to pump suction by recirc bypass line ∝
Q609∝
Q610∝
Recir pump bypass isolation valve 53SGV018 ∝ transfer open
Recirculation pump bypass line isolation valve 53SGV017 ∝ transfers open
Q53EV18∝T
Q53EV17∝T0


Both SRV's on east evapor. open causes evapor. inlet to close
Q606∝
Q615∝
SGS relief valve 53SGV102∝ open
SGS relief valve 53SGV100∝ open
Q53RV102∝T
Q53RV100∝T



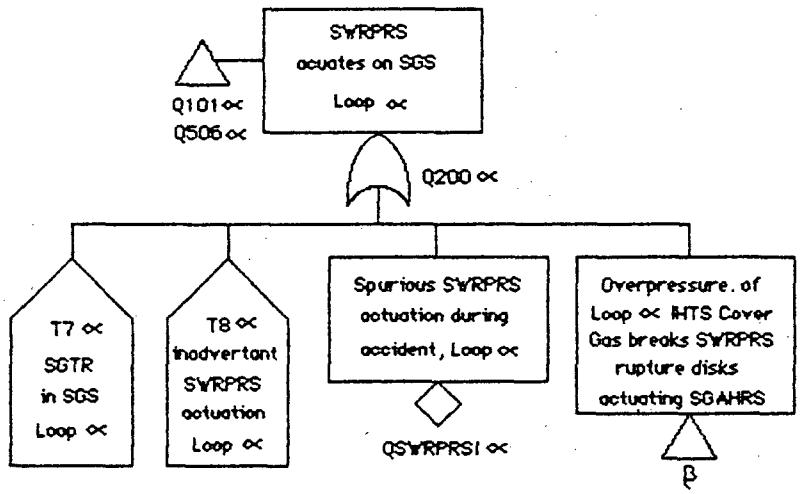
∝	K	ϑ
A	241	H1A
B	242	H2A
C	243	H3A



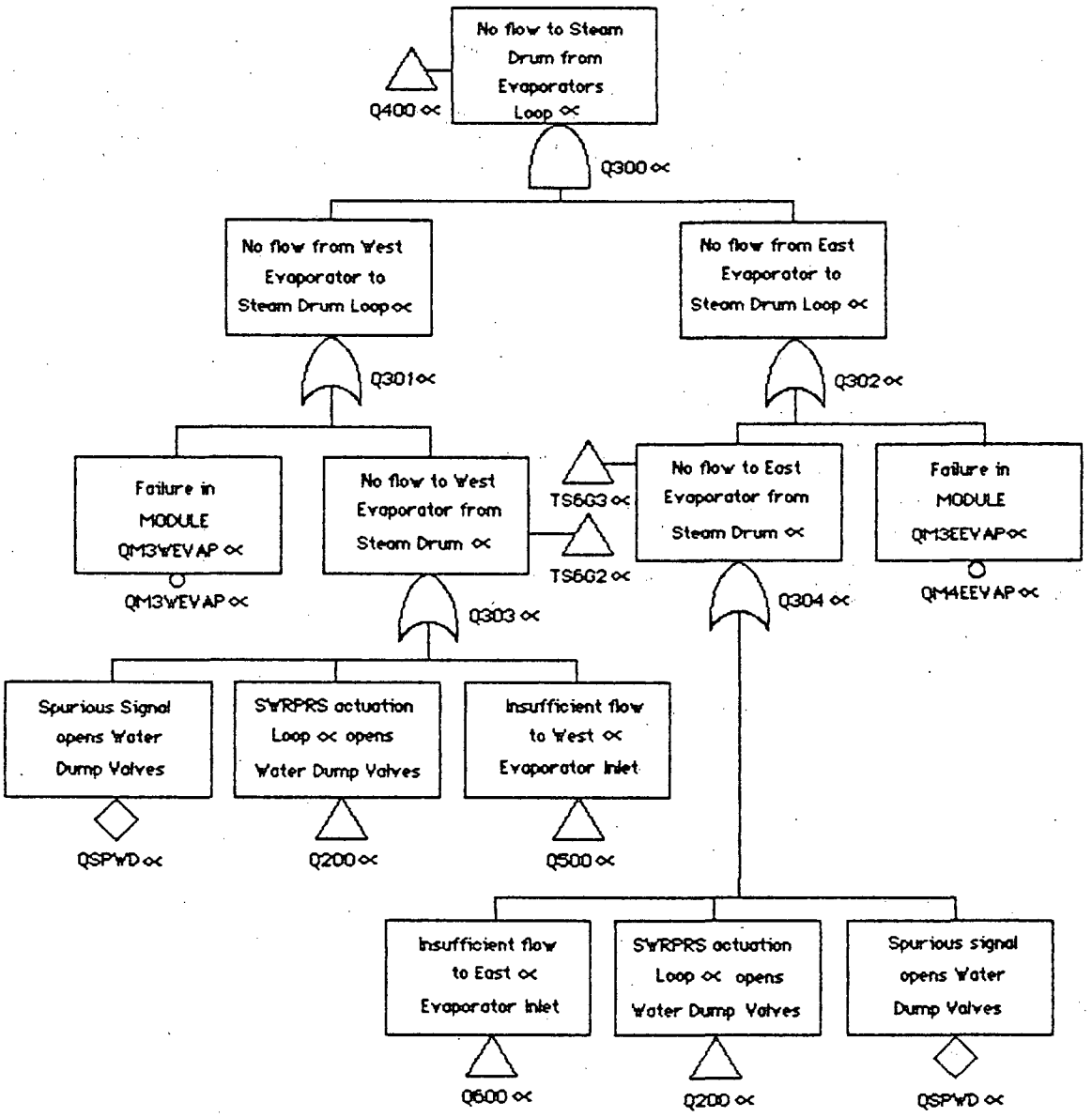
∝	K	γ
A	241	H1A
B	242	H2A
C	243	H3A

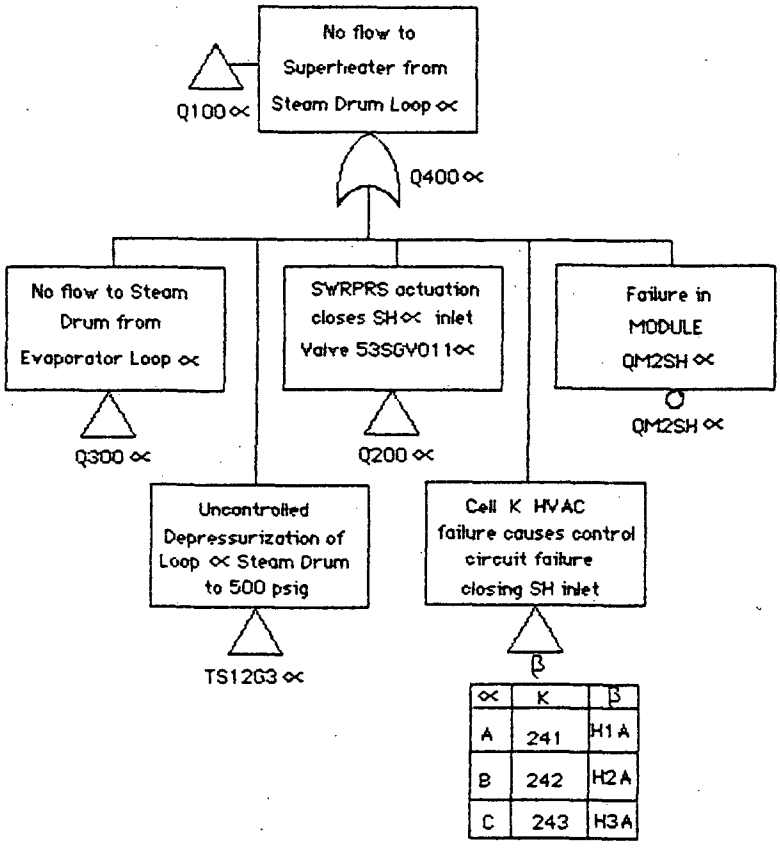
∝	β	γ
A	50A	EM152
B	50B	EM252
C	44C	EM338

Figure A4-3. Steam Generator System Modularized Fault Tree.

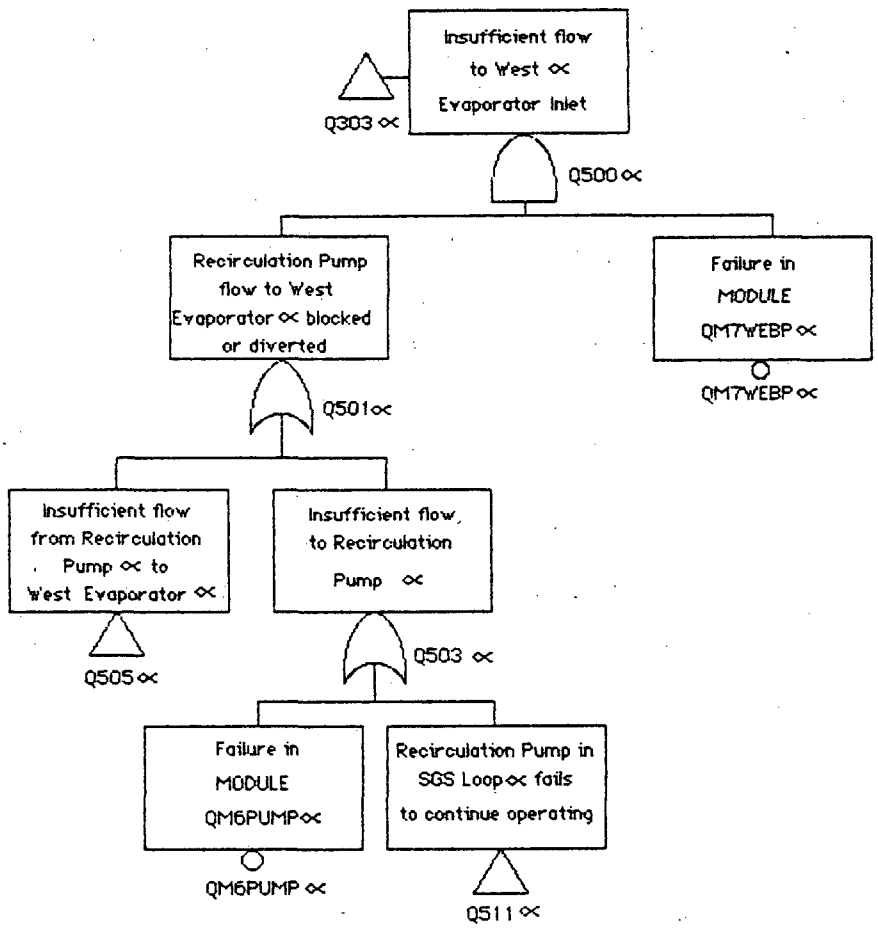


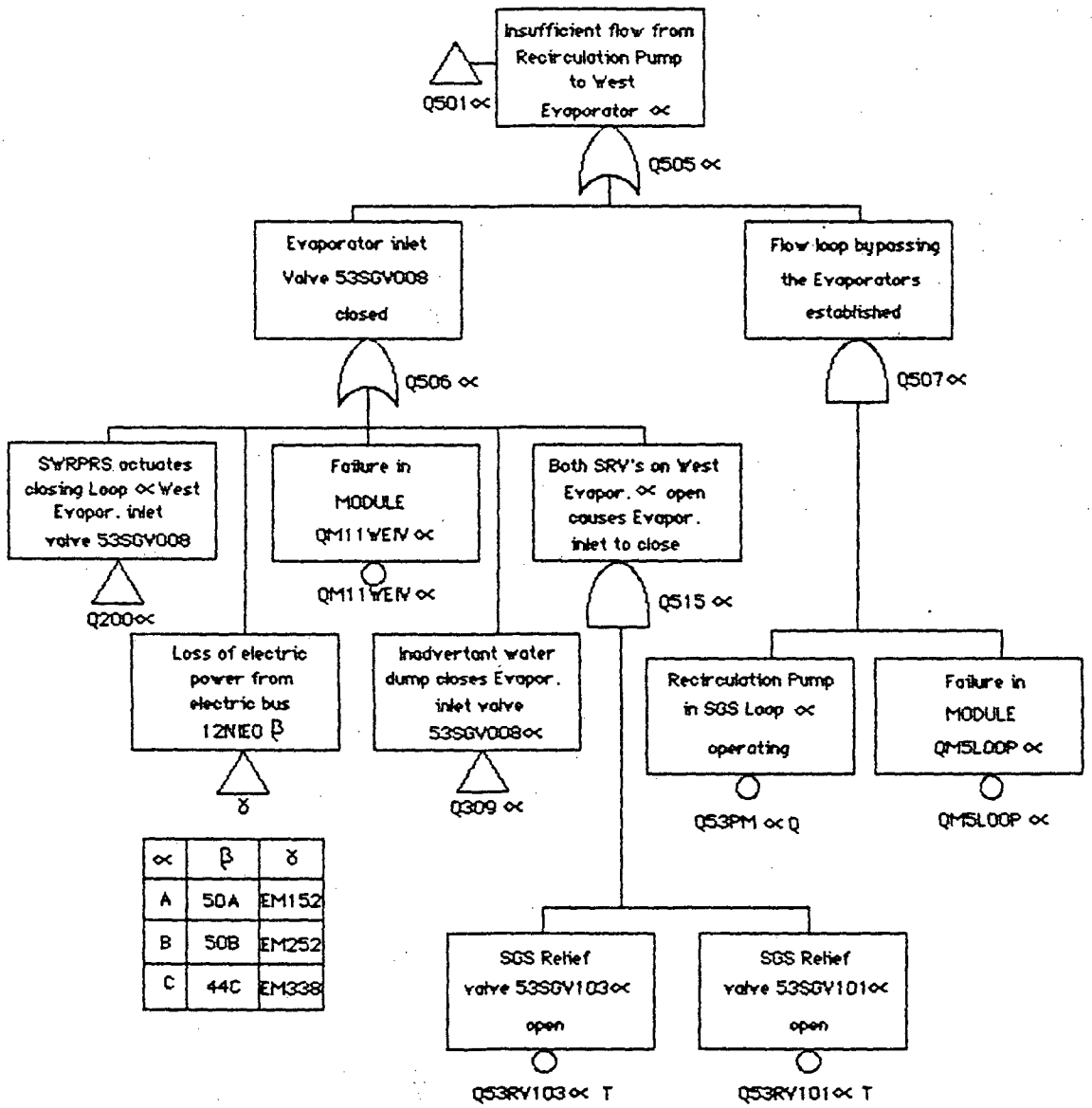
α	β
A	GM500
B	GM600
C	GM700


No flow to Steam Drum from Evaporators Loop ∝
Q400 ∝
Q300 ∝
No flow from West Evaporator to Steam Drum Loop ∝
No flow from East Evaporator to Steam Drum Loop ∝
Q301 ∝
Q302 ∝
Failure in MODULE QM3WEVAP ∝
QM3WEVAP ∝
No flow to West Evaporator from Steam Drum ∝
TS6G3 ∝
TS6G2 ∝
No flow to East Evaporator from Steam Drum ∝
Failure in MODULE QM3EEVAP ∝
QM4EEVAP ∝
Q303 ∝
Q304 ∝
Spurious Signal opens Water Dump Valves
SWRPRS actuation Loop ∝ opens Water Dump Valves
Insufficient flow to West ∝ Evaporator Inlet
QSPWD ∝
Q200 ∝
Q500 ∝
Insufficient flow to East ∝ Evaporator Inlet
SWRPRS actuation Loop ∝ opens Water Dump Valves
Spurious signal opens Water Dump Valves
Q600 ∝
Q200 ∝
QSPWD ∝

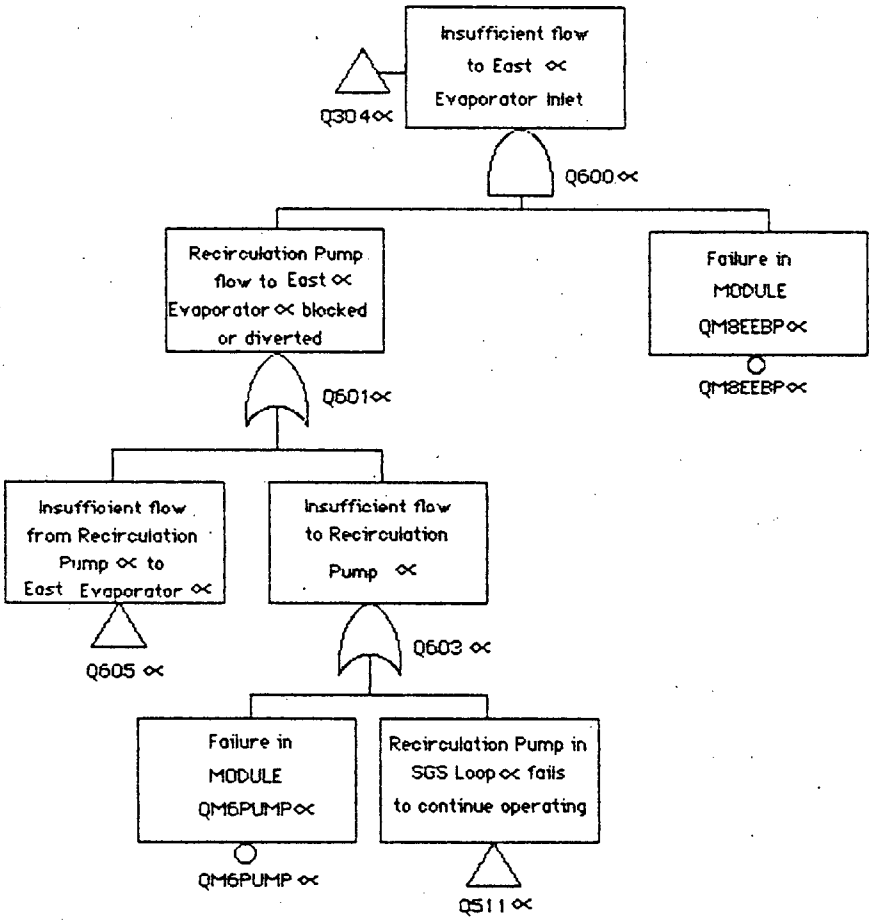


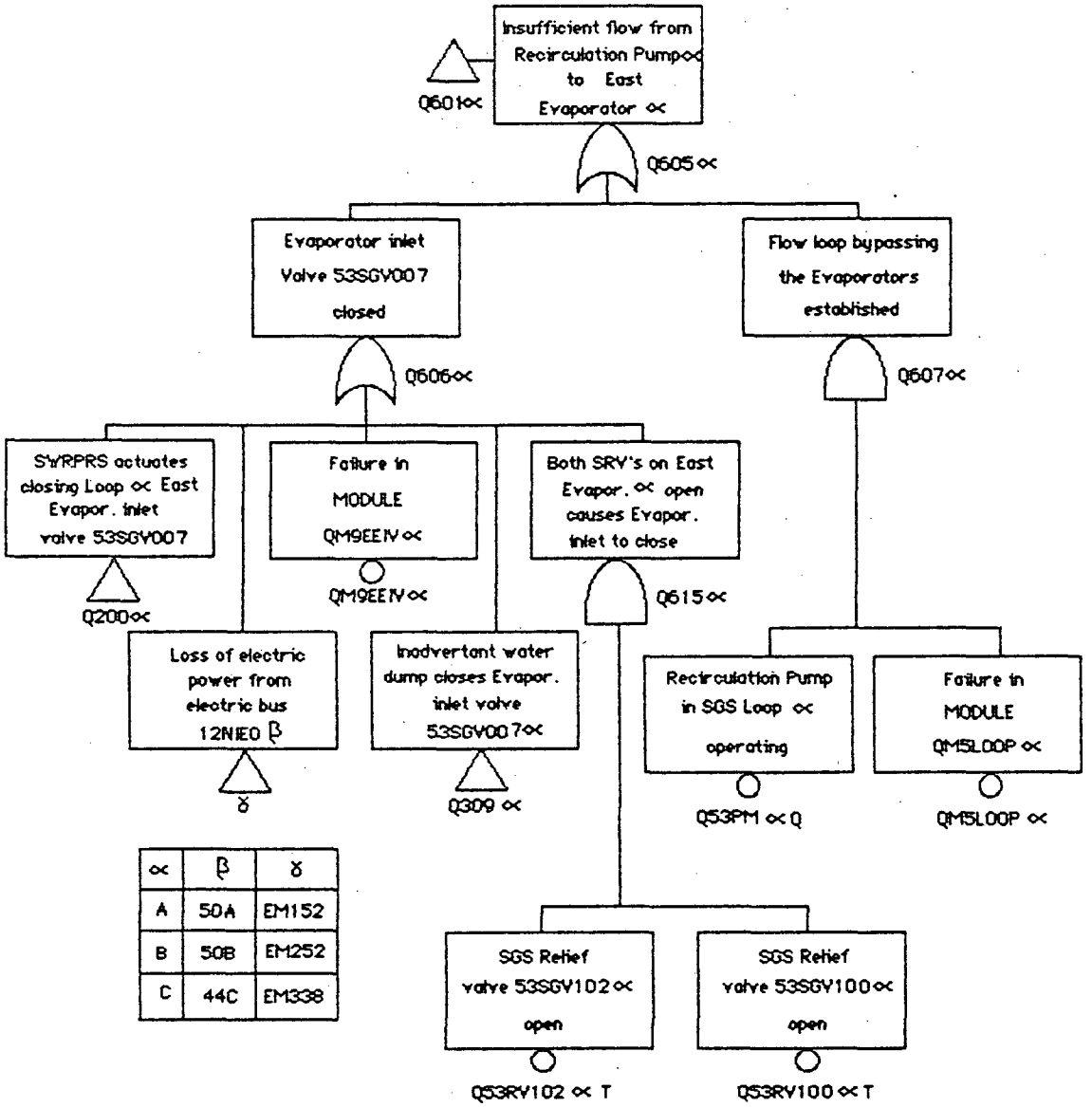
∝	K	β
A	241	H1A
B	242	H2A
C	243	H3A


Insufficient flow to West ∝ Evaporator Inlet
Q303 ∝
Q500 ∝
Recirculation Pump flow to West Evaporator ∝ blocked or diverted
Failure in MODULE QM7WEBP ∝
QM7WEBP ∝
Q501 ∝
Insufficient flow from Recirculation Pump ∝ to West Evaporator ∝
Insufficient flow to Recirculation Pump ∝
Q505 ∝
Q503 ∝
Failure in MODULE QM6PUMP ∝
Recirculation Pump in SGS Loop ∝ fails to continue operating
QM6PUMP ∝
Q511 ∝

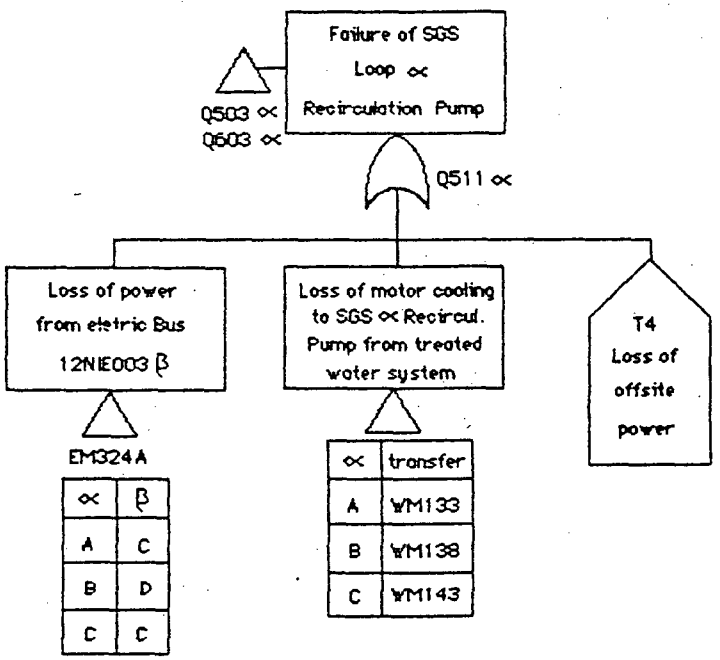


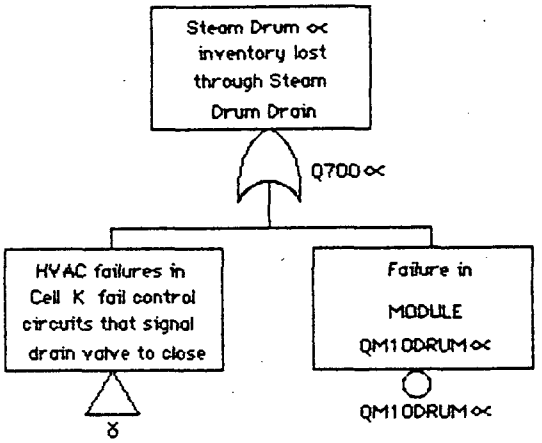
∝	β	γ
A	50A	EM152
B	50B	EM252
C	44C	EM338


Insufficient flow to East ∝ Evaporator inlet
Q304∝
Q600∝
Recirculation Pump flow to East ∝ Evaporator ∝ blocked or diverted
Failure in MODULE QM8EEBP∝
QM8EEBP∝
Q601∝
Insufficient flow from Recirculation Pump ∝ to East Evaporator ∝
Insufficient flow to Recirculation Pump ∝
Q605 ∝
Q603 ∝
Failure in MODULE QM6PUMP∝
Recirculation Pump in SGS Loop ∝ fails to continue operating
QM6PUMP ∝
Q511 ∝



∝	β	γ
A	50A	EM152
B	50B	EM252
C	44C	EM338


Failure of SGS
Loop ∝
Recirculation Pump
Q503 ∝
Q603 ∝
Q511 ∝
Loss of power
from eletric Bus
12NIE003 β
Loss of motor cooling
to SGS ∝ Recircul.
Pump from treated
water system
T4
Loss of
offsite
power
EM324A
∝ β
A C
B D
C C
∝ transfer
A WM133
B WM138
C WM143



α	K	γ
A	241	H1A
B	242	H2A
C	243	H3A

Table A4-3

DATA TABLE FOR SGS

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
QM1EXITA	SGS Loop A Superheater Steam Outlet	7.9-6			
Q53CV13AT	No flow through CV 13A	5.5-6	2.3-7/h	24	Q53CV13AT + Q53EV12AT
Q53EV12AT	No flow through EV 12A	2.4-6	1.0-7/h	24	
QSWRPRSIA	Inadvertent SWRPRS actuation in SGS Loop A	1.0-3	4.0-5/h	24	
QM3WEVAPA	W. Evap A to Steam drum flow insufficient	5.5-6			
Q53CV10AT	W. Evap A outlet CV closed	5.5-6	2.3-7/h	24	Q53CV10AT + (Q53AV4AT * Q53AV3AT)
Q53AV4AT	WD Valve 4A Opens	2.4-6	1.0-7/h	24	
Q53AV2AT	WD Valve 2A Opens	2.4-6	1.0-7/h	24	
QM4EEVAPA	E Evap A to Steam drum flow insufficient	5.5-6			
Q53CV9AT	E Evap A outlet CV closed	5.5-6	2.3-8/h	24	Q53CV9AT + (Q53AV1AT * Q53AV3AT)
Q53AV1AT	WD Valve 1A opens	2.4-6	1.0-7/h	24	
Q53AV3AT	WD Valve 3A opens	2.4-6	1.0-7/h	24	

Table A4-3 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
QSPWDA	Spurious signal opens water Dump, loop A	1.0-3	4.0-5/h	24	
QM2SHA	Superheater A inlet valve closed	1.0-3			
Q56PSSDAI	Spurious low steam drum A pressure signal	1.0-3	4.0-5/h	24	Q56PSSDAI + Q53PAY + Q53SD4AY + Q53EV11AT
Q53PAY	SGS Loop A pipe leak	7.9-6	1.1-9/h*ft	24	
Q53SD4AY	Steam Drum A leak	2.4-5	1.0-6/h	24	
Q53EV11AT	Superheater A inlet valve closed	2.4-6	1.0-7/h	24	
Q53RPBAH	Operator fails to open recirc. pump by-pass	1.0-2	1.0-2		
QM6PUMPA	SGS A Recirc. pump failures	4.8-4			
Q53EV6AT	Pump suction valve closed	2.4-6	1.0-7/h	24	Q53EV6AT + Q53PM1AR
Q53PM1AR	Pump fails to run	4.8-4	2.0-5/h	24	
Q12CBP1AT	Circuit breaker opens	9.1-5	3.8-6/h	24	
Q12CAP1AF	Cable faults	8.9-5	3.7-6/h	24	

Table A4-3 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
QM7WEBPA	Pump bypass to W Evap A blocked	1.0-3			
Q53EV18AO	Bypass isolation valve fails to open	1.0-3	1.0-3/d		Q53EV18AO + Q53EV18AT
Q53EV18ATC	Bypass isolation valve transfers closed	2.4-6	1.0-7/h	24	
QM11WEIVA	W Evap A inlet valve	2.4-6			
Q53E8AT	Valve transfers closed	2.4-6	1.0-7/h	24	
QM5LOOPA	Flow bypasses W Evap A	7.2-6			Q53XV300AT + Q53EV18ATO + Q53EV17ATO
Q53XV300AT	W Evap bypass open	2.4-6	1.0-7/h	24	
Q53EV18ATO	Pump bypass open	2.4-6	1.0-7/h	24	
Q53EV17ATO	Pump bypass open	2.4-6	1.0-7/h	24	
Q53PMAQ	Flag-Recirc pump A running	1.0			

Table A4-3 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
Q53RV103AT	W Evap A Relief valve open	2.4-4	1.0-5/h	24	
Q53RV101AT	W Evap A Relief valve open	2.4-4	1.0-5/h	24	
QM8EEBPA	Pump bypass to E Evap A blocked	1.0-3			Q53EV17AO + Q53EV17ATC
Q53EV17AO	Bypass isolation valve fails to open	1.0-3	1.0-3/d		
Q53EV17ATC	Bypass isolation valve transers closed	2.4-6	1.0-7/h	24	
QM9EEIVA	E Evap A Inlet valve	2.4-6			
Q53EV7AT	Valve transfers closed	2.4-6	1.0-7/h	24	
Q53RV102AT	E Evap A relief valve open	2.4-4	1.0-5/h	24	

Table A4-3 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
Q53RV100AT	E Evap A Relief valve open	2.4-4	1.0-5/h	24	
QM10DRUMA	Steam drum A drain valves	1.0-6			
Q53AV14AT	Drain valve 14A opens	2.4-6	1.0-7/h	24	(Q53AV14AT + Q53AV14AC)
Q53AV14AC	Drain valve 14A fails to close	1.0-3	1.0-3/d		* (Q53AV15AT + Q53AV15AC)
Q53AV15AT	Drain valve 15A opens	2.4-6	1.0-7/h		
Q53AV15AC	Drain valve 15A fails to close	1.0-3	1.0-3/d	24	

Table A4-3 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
QM1EXITB	SGS Loop B Superheater Steam Outlet	7.9-6			
Q53CV13BT	No flow through CV 13B	5.5-6	2.3-7/h	24	Q53CV13BT + Q53EV12BT
Q53EV12BT	No flow through EV 12B	2.4-6	1.0-7/h	24	
QSWRPRSIB	Inadvertent SWRPRS actuation in SGS Loop B	1.0-3	4.0-5/h	24	
QM3WEVAPB	W. Evap B to Steam drum flow insufficient	5.5-6			
Q53CV10BT	W. Evap B outlet CV closed	5.5-6	2.3-7/h	24	Q53CV10BT + (Q53AV4BT * Q53AV3BT)
Q53AV4BT	WD Valve 4B Opens	2.4-6	1.0-7/h	24	
Q53AV2BT	WD Valve 2B Opens	2.4-6	1.0-7/h	24	
QM4EEVAPB	E Evap B to Steam drum flow insufficient	5.5-6			
Q53CV9BT	E Evap B outlet CV closed	5.5-6	2.3-8/h	24	Q53CV9BT + (Q53AV1BT * Q53AV3BT)
Q53AV1BT	WD Valve 1B opens	2.4-6	1.0-7/h	24	
Q53AV3BT	WD Valve 3B opens	2.4-6	1.0-7/h	24	

Table A4-3 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
QSPWDB	Spurious signal opens water Dump, loop B	1.0-3	4.0-5/h	24	
QM2SHB	Superheater B inlet valve closed	1.0-3			
Q56PSSDBI	Spurious low steam drum B pressure signal	1.0-3	4.0-5/h	24	Q56PSSDBI + Q53PBY + Q53SD4BY + Q53EV11BT
Q53PBY	SGS Loop B pipe leak	7.9-6	1.1-9/h*ft	24	
Q53SD4BY	Steam Drum B leak	2.4-5	1.0-6/h	24	
Q53EV11BT	Superheater B inlet valve closed	2.4-6	1.0-7/h	24	
Q53RPBBH	Operator fails to open recirc. pump by-pass	1.0-2	1.0-2		
QM6PUMPB	SGS B Recirc. pump failures	4.8-4			
Q53EV6BT	Pump suction valve closed	2.4-6	1.0-7/h	24	Q53EV6BT + Q53PM1BR
Q53PM1BR	Pump fails to run	4.8-4	2.0-5/h	24	

Table A4-3 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
QM7WEBPB	Pump bypass to W Evap B blocked	1.0-3			
Q53EV18BO	Bypass isolation valve fails to open	1.0-3	1.0-3/d		Q53EV18BO + Q53EV18BT
Q53EV18BTC	Bypass isolation valve transfers closed	2.4-6	1.0-7/h	24	
QM11WEIVB	W Evap B inlet valve	2.4-6			
Q53E8BT	Valve transfers closed	2.4-6	1.0-7/h	24	
QM5LOOPB	Flow bypasses W Evap B	7.2-6			Q53XV300BT + Q53EV18BTO + Q53EV17BTO
Q53XV300BT	W Evap bypass open	2.4-6	1.0-7/h	24	
Q53EV18BTO	Pump bypass open	2.4-6	1.0-7/h	24	
Q53EV17BTO	Pump bypass open	2.4-6	1.0-7/h	24	
Q53PMBQ	Flag-Recirc pump B running	1.0			

Table A4-3 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
Q53RV103BT	W Evap B Relief valve open	2.4-4	1.0-5/h	24	
Q53RV101BT	W Evap B Relief valve open	2.4-4	1.0-5/h	24	
QM8EEBPB	Pump bypass to E Evap B blocked	1.0-3			Q53EV17BO + Q53EV17BTC
Q53EV17BO	Bypass isolation valve fails to open	1.0-3	1.0-3/d		
Q53EV17BTC	Bypass isolation valve transers closed	2.4-6	1.0-7/h	24	
QM9EEIVB Q53EV7BT	E Evap B Inlet valve Valve transfers closed	2.4-6 2.4-6	1.0-7/h	24	
Q53RV102BT	E Evap B relief valve open	2.4-4	1.0-5/h	24	

Table A4-3 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
Q53RV100BT	E Evap B Relief valve open	2.4-4	1.0-5/h	24	
QM10DRUMB	Steam drum B drain valves	1.0-6			
Q53AV14BT	Drain valve 14B opens	2.4-6	1.0-7/h	24	(Q53AV14BT + Q53AV14BC)
Q53AV14BC	Drain valve 14B fails to close	1.0-3	1.0-3/d		* (Q53AV15BT + Q53AV15BC)
Q53AV15BT	Drain valve 15B opens	2.4-6	1.0-7/h		
Q53AV15BC	Drain valve 15B fails to close	1.0-3	1.0-3/d	24	

Table A4-3 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
QM1EXITC	SGS Loop C Superheater Steam Outlet	7.9-6			
Q53CV13CT	No flow through CV 13C	5.5-6	2.3-7/h	24	Q53CV13CT + Q53EV12CT
Q53EV12CT	No flow through EV 12C	2.4-6	1.0-7/h	24	
QSWRPRSIC	Inadvertent SWRPRS actuation in SGS Loop C	1.0-3	4.0-5/h	24	
QM3WEVAPC	W. Evap C to Steam drum flow insufficient	5.5-6			
Q53CV10CT	W. Evap C outlet CV closed	5.5-6	2.3-7/h	24	Q53CV10CT + (Q53AV4CT * Q53AV3CT)
Q53AV4CT	WD Valve 4C Opens	2.4-6	1.0-7/h	24	
Q53AV2CT	WD Valve 2C Opens	2.4-6	1.0-7/h	24	
QM4EEVAPC	E Evap C to Steam drum flow insufficient	5.5-6			
Q53CV9CT	E Evap C outlet CV closed	5.5-6	2.3-8/h	24	Q53CV9CT + (Q53AV1CT * Q53AV3CT)
Q53AV1CT	WD Valve 1C opens	2.4-6	1.0-7/h	24	
Q53AV3CT	WD Valve 3C opens	2.4-6	1.0-7/h	24	

Table A4-3 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
QSPWDC	Spurious signal opens water Dump, loop C	1.0-3	4.0-5/h	24	
QM2SHC	Superheater C inlet valve closed	1.0-3			
Q56PSSDCI	Spurious low steam drum C pressure signal	1.0-3	4.0-5/h	24	Q56PSSDCI + Q53PCY + Q53SD4CY + Q53EV11CT
Q53PCY	SGS Loop C pipe leak	7.9-6	1.1-9/h*ft	24	
Q53SD4CY	Steam Drum C leak	2.4-5	1.0-6/h	24	
Q53EV11CT	Superheater C inlet valve closed	2.4-6	1.0-7/h	24	
Q53RPBCH	Operator fails to open recirc. pump by-pass	1.0-2	1.0-2		
QM6PUMPC	SGS C Recirc. pump failures	4.8-4			
Q53EV6CT	Pump suction valve closed	2.4-6	1.0-7/h	24	Q53EV6CT + Q53PM1CR
Q53PM1CR	Pump fails to run	4.8-4	2.0-5/h	24	

Table A4-3 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
QM7WEBPC	Pump bypass to W Evap C blocked	1.0-3			
Q53EV18CO	Bypass isolation valve fails to open	1.0-3	1.0-3/d		Q53EV18CO + Q53EV18CT
Q53EV18CTC	Bypass isolation valve transfers closed	2.4-6	1.0-7/h	24	
QM11WEIVC	W Evap C inlet valve	2.4-6			
Q53E8CT	Valve transfers closed	2.4-6	1.0-7/h	24	
QM5LOOPC	Flow bypasses W Evap C	7.2-6			Q53XV300CT + Q53EV18CTO + Q53EV17CTO
Q53XV300CT	W Evap bypass open	2.4-6	1.0-7/h	24	
Q53EV18CTO	Pump bypass open	2.4-6	1.0-7/h	24	
Q53EV17CTO	Pump bypass open	2.4-6	1.0-7/h	24	
Q53PMCQ	Flag-Recirc pump C running	1.0			

Table A4-3 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
Q53RV103CT	W Evap C Relief valve open	2.4-4	1.0-5/h	24	
Q53RV101CT	W Evap C Relief valve open	2.4-4	1.0-5/h	24	
QM8EEBPC	Pump bypass to E Evap C blocked	1.0-3			Q53EV17CO + Q53EV17CTC
Q53EV17CO	Bypass isolation valve fails to open	1.0-3	1.0-3/d		
Q53EV17CTC	Bypass isolation valve transers closed	2.4-6	1.0-7/h	24	
QM9EEIVC Q53EV7CT	E Evap C Inlet valve Valve transfers closed	2.4-6 2.4-6	1.0-7/h	24	
Q53RV102CT	E Evap C relief valve open	2.4-4	1.0-5/h	24	

Table A4-4 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
Q53RV100CT	E Evap C Relief valve open	2.4-4	1.0-5/h	24	
QM10DRUMC	Steam drum C drain valves	1.0-6			
Q53AV14CT	Drain valve 14C opens	2.4-6	1.0-7/h	24	(Q53AV14CT + Q53AV14CC)
Q53AV14CC	Drain valve 14C fails to close	1.0-3	1.0-3/d		* (Q53AV15CT + Q53AV15CC)
Q53AV15CT	Drain valve 15C opens	2.4-6	1.0-7/h		
Q53AV15CC	Drain valve 15C fails to close	1.0-3	1.0-3/d	24	

Appendix A, Section 5

PROTECTED AIR-COOLED CONDENSERS AND SGAHRS VENTS

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Appendix A, Section 5

PROTECTED AIR-COOLED CONDENSERS AND SGAHRS VENTS

A5.1 SYSTEM DESCRIPTION

The steam generator auxiliary heat removal system (SGAHRS) is a safety related system designed to provide the ultimate heat sink for all postulated loss of feedwater or loss of normal heat sink incidents. The SGAHRS consists of two subsystems used during two missions: (1) the auxiliary feedwater (AFW) system and SGAHRS vents, which provide a short-term heat sink, and (2) the protected air-cooled condensers (PACCs), which provide a long term heat sink. This section discusses the function, design, interfaces with other systems, operation, test and technical specifications, and maintenance requirements of the PACCs and SGAHRS vents. (The AFW system is discussed in Appendix A, Section 2).

A5.1.1 Function

Sensible and decay heat removal from the CRBRP following shutdown is accomplished through the primary and intermediate heat transport loops. With pony motor flow, each loop is capable of removing all short and long term decay heat from the reactor and transporting it to the steam generating system (SGS). Each SGS loop is adequate to remove all short- and long-term heat provided that the SGAHRS is available to that loop.

Whenever the normal heat removal path (through the main feedwater system and turbine bypass system) is not available, activation of SGAHRS will occur automatically with the AFW, SGAHRS vents, and PACC subsystems

brought into service. Short-term heat removal is accomplished by venting steam from the SGS steam drums to the atmosphere; steam drum inventory is replenished by the AFW system.

In addition, heat is also removed from the SGS via the PACCs. When the heat load has dropped to the PACC capacity, the SGAHRS vents close, and continued long-term heat removal is accomplished in a closed loop manner via the PACCs.

A5.1.2 Design

The following section discribes the design of the PACCs and the SGAHRS vents.

A5.1.2.1 PACCs. Each PACC is a tube-type steam condenser constructed of carbon steel. Heat is rejected to the atmosphere by condensing saturated steam from the steam drums by forced or natural circulation of air over the tube bundles.

Each PACC unit consists of two half-size tube bundles, two variable blade pitch axial fans, and two sets of variable position louvers to control airflow and, therefore, heat rejection. The arrangement of the PACC is illustrated in Figure A5-1. Air is delivered from the axial fans (one per tube bundle) into the insulated plenum surrounding each tube bundle. Air flows circumferentially around the tube bundle, then radially inward through the finned tube bundle into a central core. Air then flows upward through the central core and exhausts through louvers to an air exhaust stack. The tube bundles are independent of each

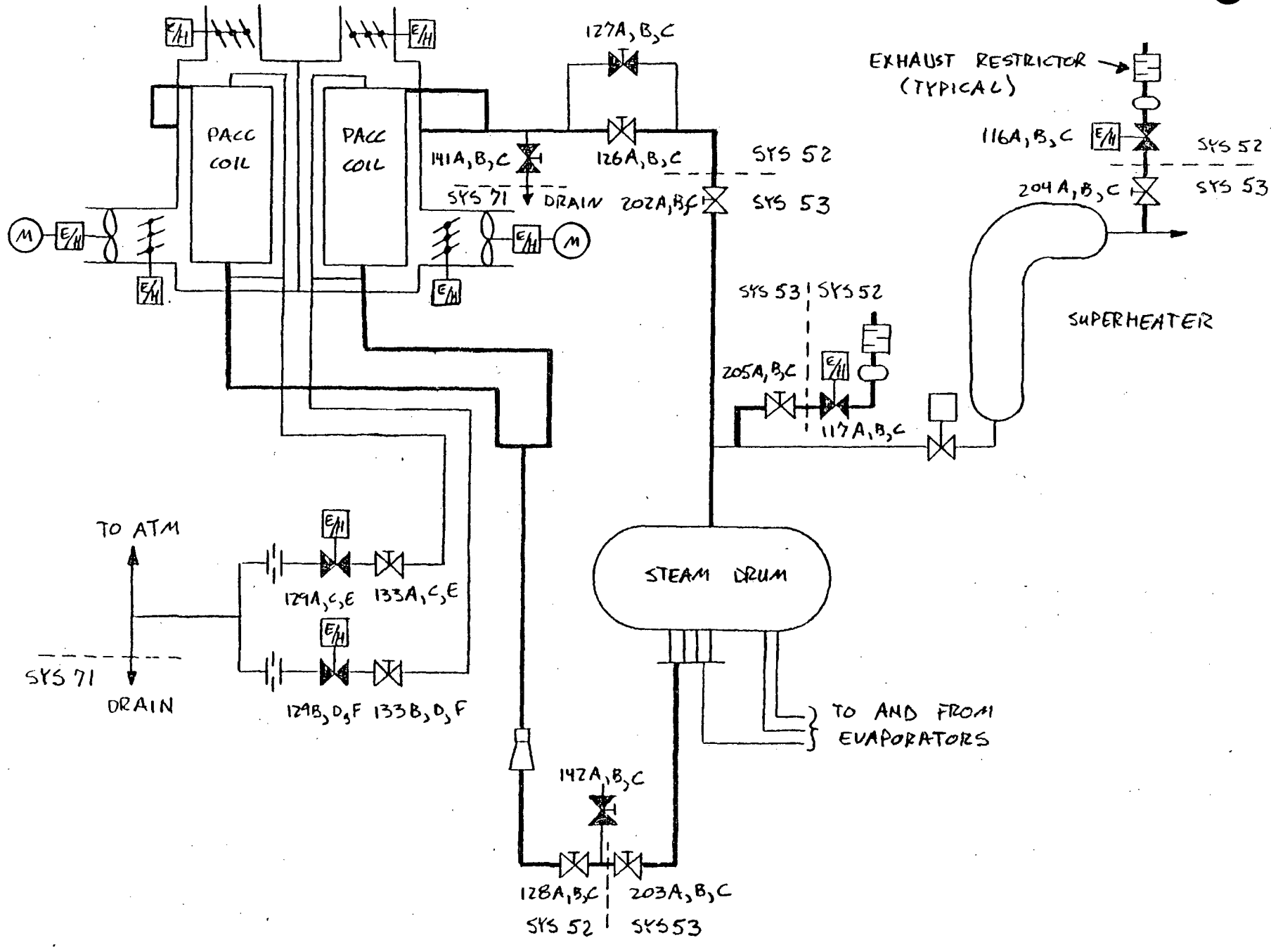


Figure A5-1. PACC and SGAHR Vents Arrangement.

other; hence, failure to remove heat from one bundle does not necessarily imply failure to remove heat from the other bundle in the PACC.

Saturated steam flows from the steam drum through an 8 inch diameter carbon steel pipe. There are three such parallel pipes, one for each PACC, which are separated by the steam generator building containment walls. Each line has two locked-open, manually-operated isolation valves. Before entering the PACC, each 8-inch line tees into two 6-inch lines which lead to the two half-size tube bundles. Condensate from each of the half-size tube bundles is piped in a separate 8 inch line down to an elevation 3 feet below normal water level in the steam drum. These separate lines assure that each PACC tube bundle is isolated from the other by a water seal. At an elevation 3 feet below normal steam drum water level, the 8 inch lines combine into a 6 inch line which continues down to the recirculation header 19 feet below normal steam drum level. This common condensate line contains two locked-open, manually-operated isolation valves and a venturi flowmeter. A thermocouple well is also provided to sense condensate return temperature.

During SGAHRS operation, the PACCs are periodically vented of non-condensible gases to assure maximum heat transfer in the tube bundles. This venting occurs automatically on PACC startup to sweep any non-condensibles which may have collected while the PACCs were in a standby mode. Thereafter, venting occurs whenever the difference in vent line temperature and steam drum saturation temperature exceeds a preset value. As gases are vented, saturated steam enters the collection

pipe, thus eliminating the temperature drop and resulting in non-condensible vent valve closure.

Each PACC bundle has a separate noncondensible venting system. Non-condensibles from the top of the saturated steam inlet piping and the condensate return line are concentrated in a collection line. The collection line contains a locked-open manually-operated isolation valve, an electrohydraulically operated vent valve (controlled by temperature difference) and an orifice to control the vent rate. (The vent rate is 190,000 lbm/hr, which is approximately twice the PACC's rated steam flow.) Downstream of the orifices, the vent piping tees to a common line which leads to an exhaust stack. Condensate in the exhaust stack is passed through a bucket-type steam trap to the feedwater and condensate system.

Automatic control of the heat rejection from each PACC is accomplished by varying the inlet louver and fan blade pitch in response to the difference between actual steam drum pressure and a preset setpoint. The difference between desired and actual steam drum pressure is converted to a desired heat removal signal by the control circuit. The desired heat removal signal is then compared to the actual heat removal rate of the PACC (determined by steam drum pressure, return condensate flow rate, and return condensate temperature). This signal (desired to actual heat removal) is then sent to the inlet louver and fan blade pitch position controllers. (As heat load is reduced on the PACC, fan blade pitch is reduced first. Once the fan blades are at minimum pitch, the inlet louver is throttled from its fully-open position.) In addition to the automatic control circuit (which is actuated on each reactor trip

and backed-up by the SGAHRS actuation logic), manual control of fan blade pitch and inlet louver position is provided.

Sensors are provided at the PACC airside inlet to detect sodium aerosols in the event of an IHTS loop sodium leak. When sodium aerosols are detected at any PACC inlet, the affected PACC will automatically shut down for a 5,000 second period and then will automatically restart. A manual overide is provided for the operator to restart the PACC before this time period has ended if the operator has determined this can be done without adverse effect on PACC operation.

Each PACC unit is capable of removing 15 MWt (7.5 MWt per tube bundle) under conditions of forced air circulation and natural circulation flow on the steam/water side. If no fans are operating (natural draft), each PACC unit is capable of removing 4.5 MWt (2.25 MWt per tube bundle).

A5.1.2.2 SGAHRS vents. Two SGAHRS vent lines are provided per SGS loop: one SGAHRS vent line is connected to the outlet piping from the loop's superheater module (valves 205A,B,C in Figure A5-1); the other is connected to the steam drum outlet piping (valves 116A,B,C in Figure A5-1). Both vent lines contain a locked-open manually operated isolation valve, a normally closed electrohydraulically operated vent valve, and an exhaust restrictor that leads to the atmosphere. Both vent valves are armed by SGAHRS actuation. The superheat SGAHRS vent valve opens when steam drum pressure exceeds 1475 psig; the steam drum SGAHRS vent valve opens when steam drum pressure exceeds 1550 psig.

A5.1.3 Interfaces With Other Systems

The PACCs interface with the following systems:

1. Building electric power system.

   This system provides motive power to the PACC fan motors, the fan blade pitch electro-hydraulic controller, the inlet and outlet louver electro-hydraulic controllers, the non-condensible vent valves, and the control circuits.

2. Steam generating system.

   This system provides saturated steam to the PACC tube bundles and receives condensate from the PACC tube bundles.

3. Reactor heat transport instrumentation system.

   This system provides PACC heat removal rate control, interlocks, and process monitoring instrumentation.

4. Feedwater and condensate system.

   This system collects condensate from the noncondensible vent piping.

The SGAHRS vent valves interface with the following systems:

1. Building electric power system.

   This system provides motive power to the SGAHRS vent valves.

2. Steam generating system.

   This system supplies steam to each vent valve.

3. Reactor heat transport instrumentation system.

   This system provides SGAHRS vent valve control.

A5.1.4 Operation

During power operation, the PACC fans are deenergized and the inlet and outlet louvers are closed. The flow path between the PACC tube bundles and the SGS is open, allowing a small amount of natural circulation flow. (The PACC louvers are insulated to minimize heat losses from the PACCs during power operations.) The SGAHRS vent valves are closed.

Following reactor trip, the PACC louvers open and the PACC fans start to provide some short term cooling. A majority of the short term heat load is dissipated by the main condenser via the turbine bypass system. The turbine bypass setpoint of 1450 psig is maintained while sodium and water temperatures are reduced to about 600°F. When the heat load drops to the PACC capacity (up to two hours following reactor trip, depending on the power history of the reactor and the number of operable PACCs), the turbine bypass valves close; continued heat removal is accomplished in a closed-loop manner using the PACCs.

In the event that the normal heat removal system is not available following reactor trip, actuation of SGAHRS occurs automatically with both the SGAHRS vent valves and PACCs brought into service. Specifically, SGAHRS is actuated if any one of three SGS loops experience either a high steam-to-feed ratio or a low steam drum water level. The following events occur following SGAHRS actuation:

1. The superheater outlet isolation valves (SOIVs) close, blocking steam flow from the steam drums to the turbine bypass system.

2. The AFW system is started to maintain steam drum inventory.

3. The SGAHRS vent valves open, venting steam from the SGS to atmosphere. This venting provides a majority of short term cooling. Initially, both SGAHRS vent valves open so that steam drum pressure is rapidly reduced to 1550 psig in about 30 seconds. Then, the steam drum SGAHRS vent valve closes; the superheater SGAHRS vent valve remains open to maintain short term cooling by steam venting.

4. The PACC fans are started and the louvers open to supplement short term cooling via steam venting.

As in the case of a routine reactor trip, the PACCs assume the entire heat load once sensible and decay heat has dropped to the PACC capacity. At this time, the SGAHRS superheater vent valve is closed.

A5.1.5 Test and Technical Specifications

The following sections summarize the test and technical specifications given in the applicable plant documentation for the SGAHRS vent valves and PACCs.

A5.1.5.1 SGAHRS vents. Normally, two SGAHRS vent valves per loop are available for service during power operations. Maintenance can be performed on only one valve of one loop at any given time. If core power is above 5%, a valve can be taken out of service for no more than 8 hours. Routine procedures, such as visual inspection, can be performed without taking the valve out of service.

Once every 3 months, the SGAHRS vent valves are cycled to assure their operability. This test momentarily disables the associated vent valve since the isolation valve to the vent is closed.

A5.1.5.2 PACCs. At least two PACC loops must be in service at all times during plant operation. A single PACC unit may be taken out of service not to exceed 24 hours when core power exceeds 5%. Prior to removing a PACC from service, the remaining PACCs must be tested on both normal and emergency diesel power (within 8 hours of removing the PACC from service).

Once every 3 months during plant operation above 40% power, a PACC periodic component test is performed. This test checks the operability of the PACC instrumentation, the louvers, and the fans. Tests on each PACC are performed back-to-back, i.e., all PACCs are tested at about the same time within the limitations of the Technical Specifications.

A5.1.6 Maintenance Requirements

Excepting routine visual inspections, all maintenance performed on the SGAHRS vents and PACCs requires that these components be taken out of service. No periodic maintenance (other than inservice inspections) has been specified for either the SGAHRS vents or the PACCs. Maintenance requirements for the SGAHRS control circuits have also not been specified.

A5.2 SYSTEM MODEL

The SGAHRS vent valve and PACC fault trees are directly input to the core-damage event tree top logic. (SGAHRS vent valves pertain to event S; PACCs pertain to event C). Since the SGAHRS vent valves are relatively simple to model, their fault trees actually comprise part of the top logic and are not further discussed in this appendix. The following sections detail the modeling assumptions, fault trees, and modularized fault trees for the PACCs.

A5.2.1 Modeling Assumptions

The following boundary and initial conditions were assumed during fault tree construction:

1. A reactor trip signal has been generated.
2. Pipe breaks or pipe plugging were not considered.
3. Wiring faults (open circuits, shorts to ground, etc.) were not considered.
4. Control circuit failures and faults have been explicitly modeled due to the uniqueness of the CRBRP SGAHRS control circuit.
5. Repair and recovery actions are not modeled; such actions were treated on a cut-set basis as necessitated by the sequence quantification tasks.

6. Detector sensing element and transmitter failures are represented by a single basic event.

7. The control logic for the noncondensible vent lines on the PACCs has been taken from a verbal description of the circuit's operation since instrumentation drawings were not available.

8. Maintenance requirements for the control circuits (reactor heat transport instrumentation) have not been specified; it was assumed that these requirements would be similar to requirements of the plant protection system.

A5.2.2 Detailed Fault Tree Model

A detailed fault tree model was developed for each PACC tube bundle and each PACC fan. The core-damage event tree top logic then combines these fault trees to model loss of either forced or natural draft heat removal from a single PACC tube bundle. The following sections discuss the success criteria, top events, transfers to other fault trees, and other related aspects of the PACC fault trees. Figures A5-2 through A5-13 (following text) present the detailed PACC fault trees.

A5.2.2.1 Success criteria. The core-damage event tree top logic specifies the following PACC success criteria for event C (long term core cooling):

1. Two of six PACC bundles operating in forced draft; or

2. Six of six PACC bundles operating in natural draft; or

3. One of six PACC bundles operating in forced draft and three of the remaining five PACC bundles operating in natural draft.

Table 5.6-7 of the PSAR shows that three PACCs operating in natural draft will assume the entire decay heat load in about 1.6 hr following a loss of all bulk ac power. Based on the above information, two of six PACC bundles operating in forced draft or six of six PACC bundles

operating in natural draft provide adequate long-term cooling capability.

Certain other combinations of PACCs will also provide adequate long-term cooling. Consideration of the above information suggests that any combination of PACCs where total heat removal capacity is above 13.5 MW (6 PACC bundles in natural draft times 2.25 MW per PACC bundle in natural draft) would provide adequate long-term cooling. There is, in fact, only one such combination of PACCs where total heat removal capability exceeds this 13.5 MW criteria: one of six PACC bundles operating in forced draft (7.5 MW) and three of the remaining five PACC bundles operating in natural draft (2.25 MW each, for a total of 6.75 MW). This combination provides a total heat removal capacity of 14.25 MW.

A5.2.2.2 Top events. Fault trees were developed for the following top events of the PACCs:

1. Loss of all heat removal from PACC bundle 52ACH001A1, 52ACH001A2, 52ACH001B1, 52ACH001B2, 52ACH001C1, or 52ACH001C2. The top gates corresponding to each of these top events are S200A1, S200A2, S200B1, S200B2, S200C1, and S200C2, respectively.

2. PACC blower 52ACH001A1, 52ACH001A2, 52ACH001B1, 52ACH001B2, 52ACH001C1, or 52ACH001C2 fails off. The top gates corresponding to each of these top events are S100A1, S100A2, S100B1, S100B2, S100C1, and S100C2, respectively.

A5.2.2.3 Transfers to other systems. Table A5-1 lists the transfers from the PACC fault trees to other system fault trees.

Table A5-1

LIST OF PACC FAULT TREE TRANSFERS

Description	Transfers to Gate	Input for Gate
Loss of power from bus 12NIE003A	EM128	S100A1 S100C2 S200A1 S200C2
Loss of power from bus 12NIE003B	EM228	S100A2 S100B1 S200A2 S200B1
Loss of power from bus 12NIE003E	EM324	S100B2 S100C1 S200B2 S200C1
Failure to actuate components controlled by SGAHRS division I amplifier	S900A	S224A
Failure to actuate components controlled by SGAHRS division II amplifier	S900B	S224B
Failure to actuate components controlled by SGAHRS division III amplifier	S900C	S224C

A5.2.2.4 Discussion on fault trees. The fault tree structure is identical for each PACC tube bundle and fan; only the basic event descriptions vary according to the location of the component being modeled (i.e., loop 1 verses loop 2, etc.). Further, it was noted that components controlled by a particular SGAHRS actuation logic division (I, II, or III) received electrical power from the corresponding electrical power division (I, II, or III).

The PACC fan fault tree (top gates S100A1 through S100C2) models hardware failures of the fan; faults of the fan due to testing, maintenance, and human errors; failure of the fan due to loss of motive power; and failure of the fan due to control circuit malfunctions. Control circuit faults may be due to SGAHRS actuation system faults, sodium aerosol detector faults, fan blade pitch control faults, or fan circuit breaker faults. The fan blade pitch control circuit also controls inlet louver position; hence, the fault trees for the fan and the tube bundle share much of the same logic. Circuit breakers for the fans in a loop are interlocked so that the opening of one fan breaker will trip open the circuit breaker of the alternate fan in the same loop.

The PACC tube bundle fault tree (top gates S200A1 through S200C2) models hardware failures of the louvers, hardware failures of the steam/water flow path between the tube bundle and its associated steam drum, faults of the tube bundle due to the collection of non-condensible gases (hardware failures and control circuit failures of the non-condensible vent lines are developed), faults due to testing and maintenance, and control circuit failures. Control circuit failures may be due to either

SGAHRS actuation system faults, sodium aerosol detector faults, or inlet louver position control circuit faults. (As previously discussed, inlet louver position control is closely coupled to fan blade pitch control).

A5.2.3 Modularized Fault Tree Model

In order to minimize the time required to solve each fault tree and gain insight into system behavior, the PACC fault trees were modularized. Modularization combines several events according to the fault tree's logic structure, thus reducing the number of gates and basic events in the tree. Figures A5-14 through A5-24 (following text) present the modularized PACC fault trees.

Modularization was performed on a division or loop basis; hence, the modules created applied to both the PACC fan fault tree and the PACC tube bundle fault tree. The following general types of modules were created (each module including its basic events and logical definition are given in Table A5-2):

1. Loop modules (3 total) that represent blockage of the steam/water flow paths between the PACC tube bundles and the steam drums.

2. Sodium aerosol modules (3 total) that represent faults due to the sodium aerosol detection system in each PACC cell.

3. Reactor trip modules (2 total) that represent faults in the circuits that actuate the PACCs following reactor trip.

4. PACC modules (6 total) that represent hardware faults of the inlet louvers, outlet louvers, and noncondensible vent lines.

5. PACC instrumentation modules (3 total) that represent hardware faults of the loop instrumentation used to control the fan blade pitch position and inlet louver position.

6. PACC control logic modules (6 total) that represent hardware faults of the control system for each fan blade pitch and inlet louver.

7. PACC fan modules (6 total) that represent hardware failures of the fan motor circuit breakers and fan blade pitch control mechanism.

8. PACC fan common modules (3 total) that represent hardware failures of the fans and motor circuit breakers that are common to each PACC unit (both half-size bundles).

It should be noted that human errors and test/maintenance basic events were not modularized.

A5.3 FAILURE DATA

Failure data for each basic event and module is given in Table A5-2 (following the fault trees). All data was taken from the generic list developed for the PRA (Section 5) except as noted.

In most cases, the failure probability for each basic event represents a component unreliability (i.e., the probability that the component has suffered at least one failure in a specified mission time). The mission time for the PACCs was assumed to be 24 hours. A few basic events required determination of a component's unavailability (i.e., the probability that the component is failed at a specified point in time); in these cases, the mean fault duration was taken to be the mean time to repair of the component.

A5.4 SYSTEM LEVEL RESULTS AND INSIGHTS

Upon solution of the core-damage event tree, several components in the PACC and SGAHRS vents subsystems appeared in the list of core-damage minimal cut sets. These risk significant failures include:

1. Both PACC tube bundles in a loop unavailable due to maintenance. (Both inservice inspection and corrective maintenance require that the PACCs be taken out of service by isolating the affected PACCs from their steam drums.)

2. Failure of a single PACC tube bundle to remove heat due to:

   a. Inlet or outlet louver closure,

   b. Noncondensible vent control system malfunction, or

   c. Noncondensible vent flow path blockage (vent valve fails to open, vent isolation valve transfers shut, or vent orifice plugged), or

   d. Heat rejection control system failure prior to the start of the accident.

3. Failure of the SGAHRS steam drum vent to open following an accident where the superheater inlet isolation valve is closed. (Isolation of the superheater fails flow to the SGAHRS superheater vent.)

4. Uncontrolled depressurization of a steam drum due to a stuck-open SGAHRS vent valve or a PACC noncondensible vent valve.

A5.5 REFERENCES

The following documentation was consulted during the construction of the PACC fault trees:

1. General Electric Company, Steam Generator Auxiliary Heat Removal System, SDD-52, Rev. 83.

2. General Electric Company, Steam Generating System, SDD-53, Rev. 97.

3. General Electric Company, Reactor Heat Transport Instrumentation System, SDD-56, Rev. 99.

4. Project Management Corporation, Clinch River Breeder Reactor Preliminary Safety Analysis Report.

5. General Electric Company, System 52 Modification to PACC Piping, ECP G1088, September 27, 1982.

6. PI&D: Steam Generator Auxiliary Heat Removal System, Dwg. 852EGG1, Rev. 31.

7. PI&D: Steam Generating System, Dwg. 261R121, Rev. 39.

8. Logic Diagram: Steam Generator Auxiliary Heat Removal System, System 56, Dwg. 852E355, Rev. 12.

9. Logic Diagram: SGAHRS Initiation, System 56, Dwg. 852E886, Rev. 2.

10. Instrument Drawing: Steam Generator Auxiliary Heat Removal System, System 56, Dwg. 908E803, Rev. 11.

11. Secondary RSS Logic, Dwg. 1440E26, Rev. 6.

12. Primary RSS Logic, Dwg. 1440E42, Rev. 6.

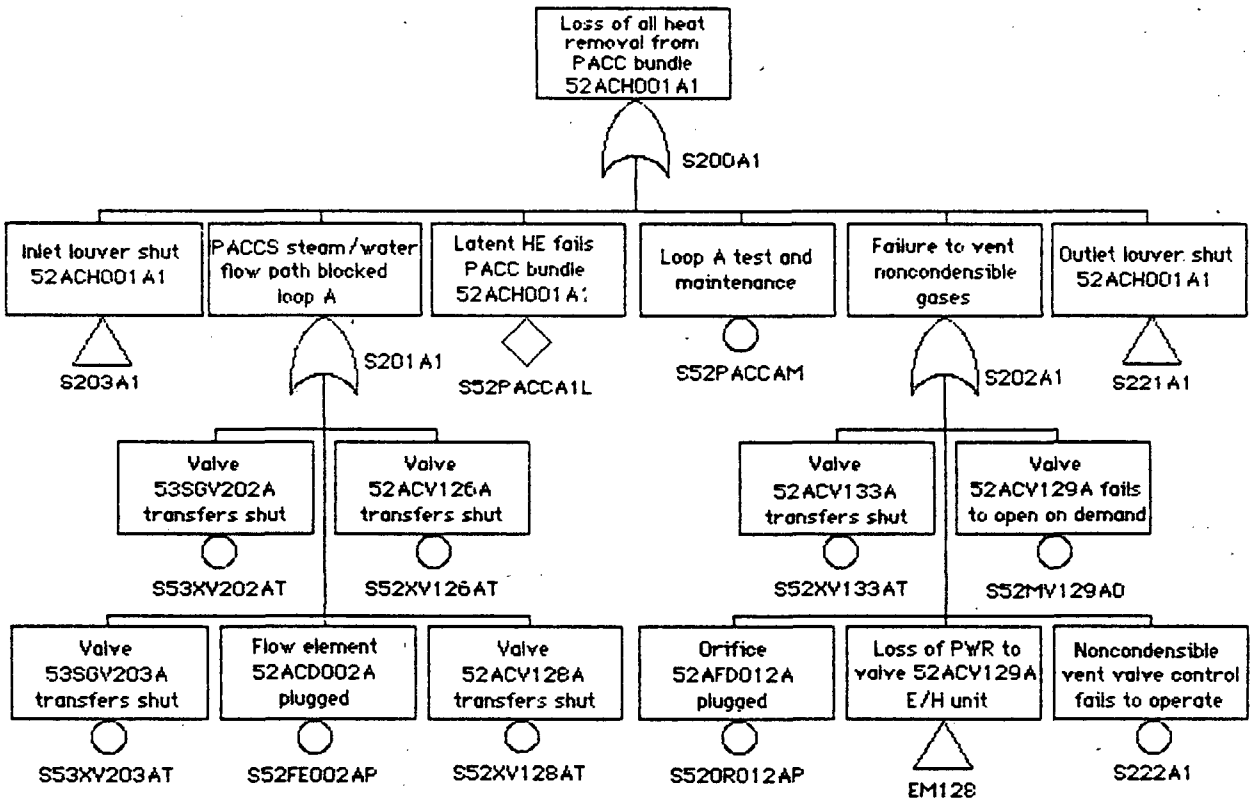
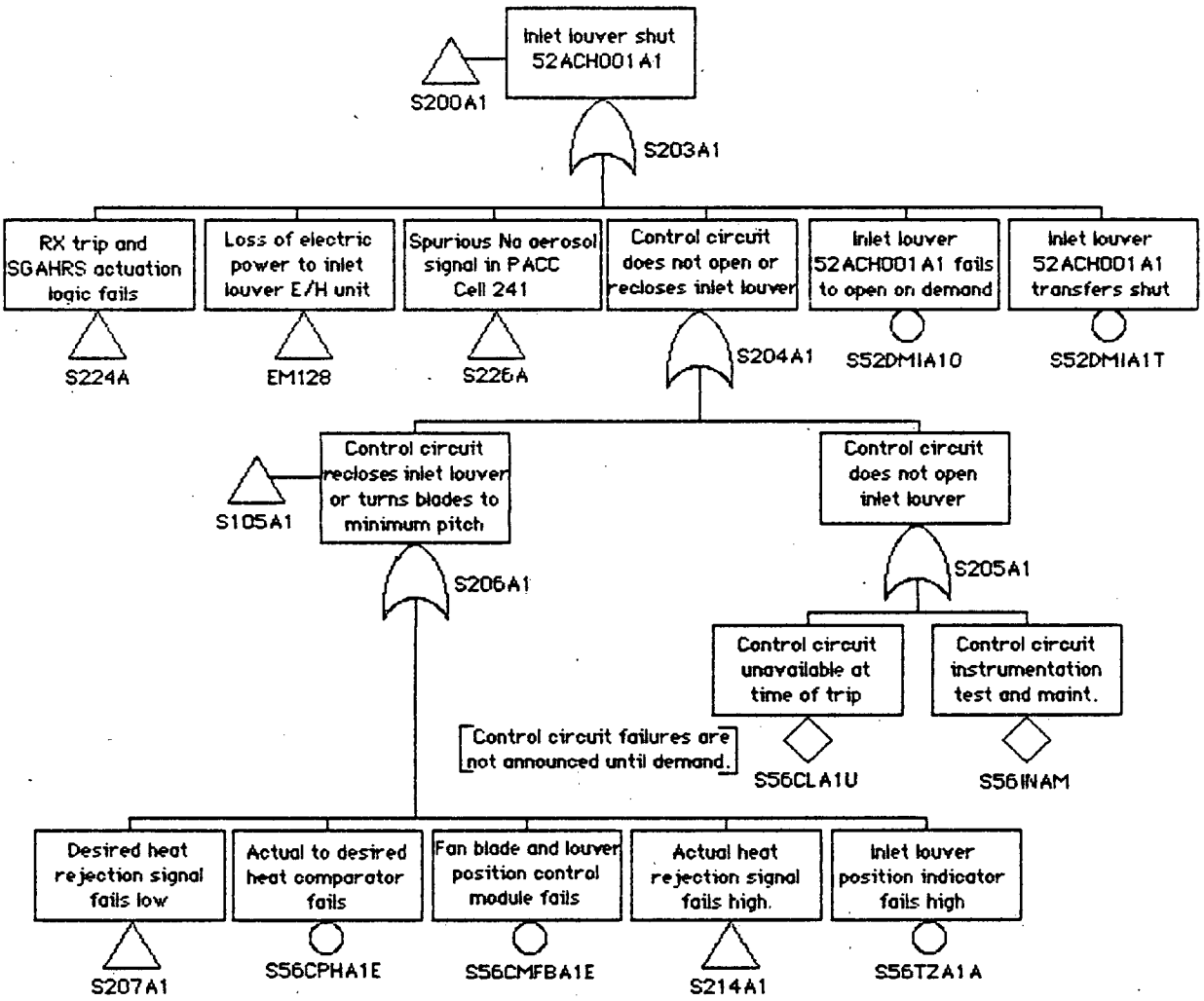
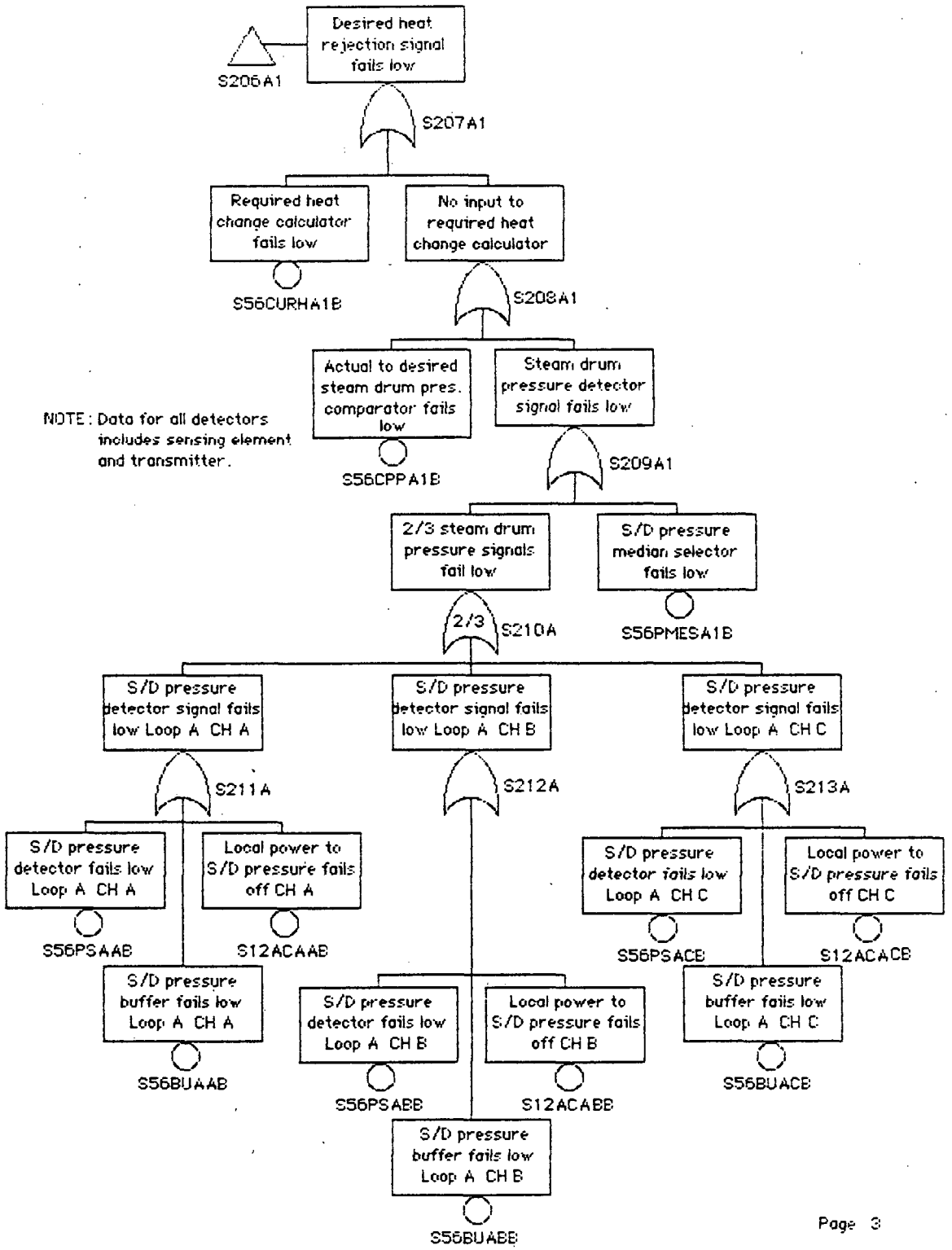
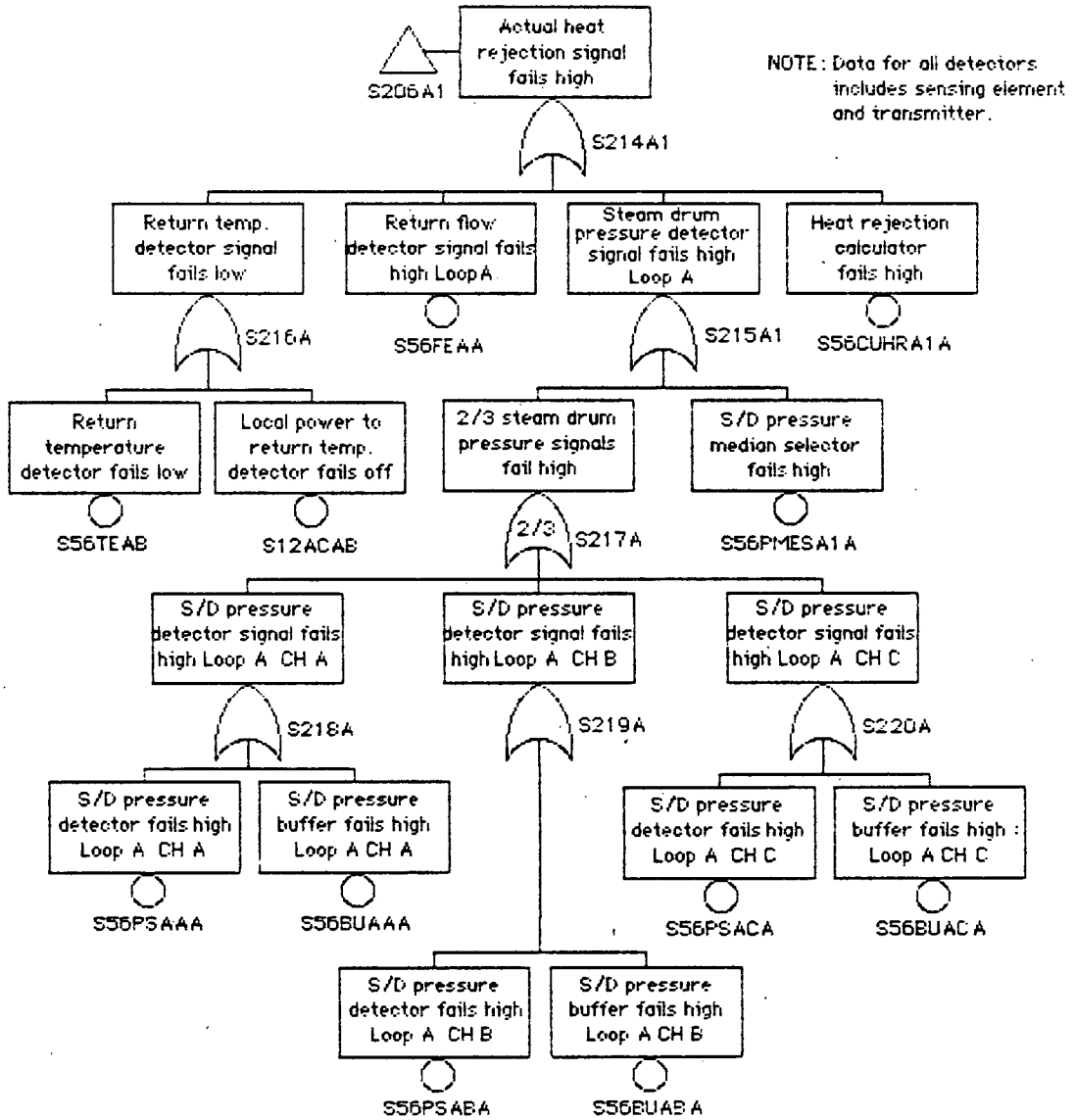
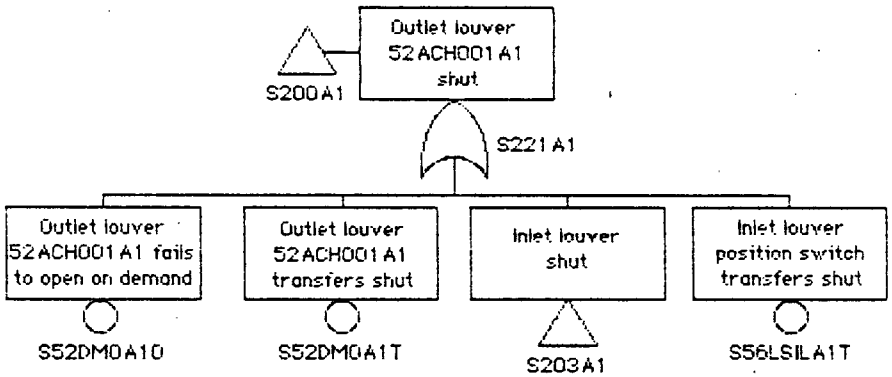


Figure A5-2. PACC Fault Tree (Detailed) for Tube Bundle 52ACH001A1.

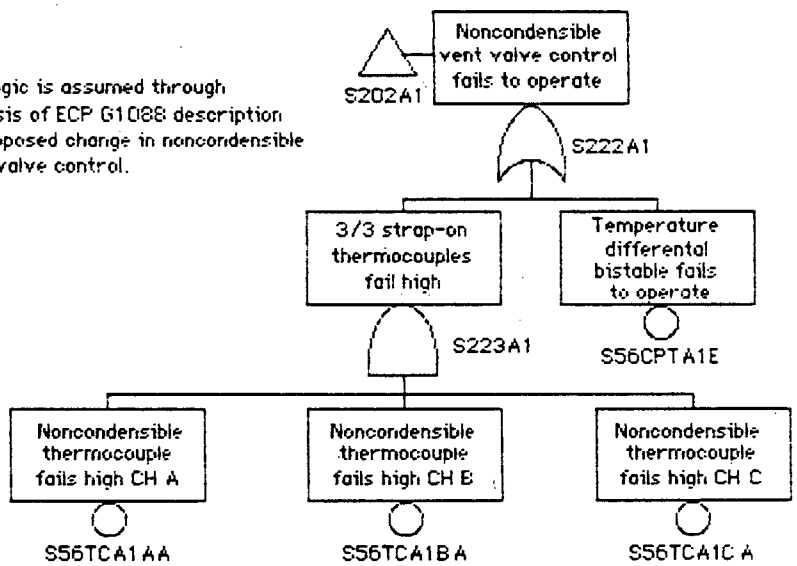

Inlet louver shut 52ACH001A1
S200A1
S203A1
RX trip and SGAHRS actuation logic fails
Loss of electric power to inlet louver E/H unit
Spurious Na aerosol signal in PACC Cell 241
Control circuit does not open or recloses inlet louver
Inlet louver 52ACH001A1 fails to open on demand
Inlet louver 52ACH001A1 transfers shut
S224A
EM128
S226A
S204A1
S52DMIA10
S52DMIA1T
Control circuit recloses inlet louver or turns blades to minimum pitch
Control circuit does not open inlet louver
S105A1
S206A1
S205A1
Control circuit unavailable at time of trip
Control circuit instrumentation test and maint.
Control circuit failures are not announced until demand.
S56CLA1U
S56INAM
Desired heat rejection signal fails low
Actual to desired heat comparator fails
Fan blade and louver position control module fails
Actual heat rejection signal fails high.
Inlet louver position indicator fails high
S207A1
S56CPHA1E
S56CMFBA1E
S214A1
S56TZA1A

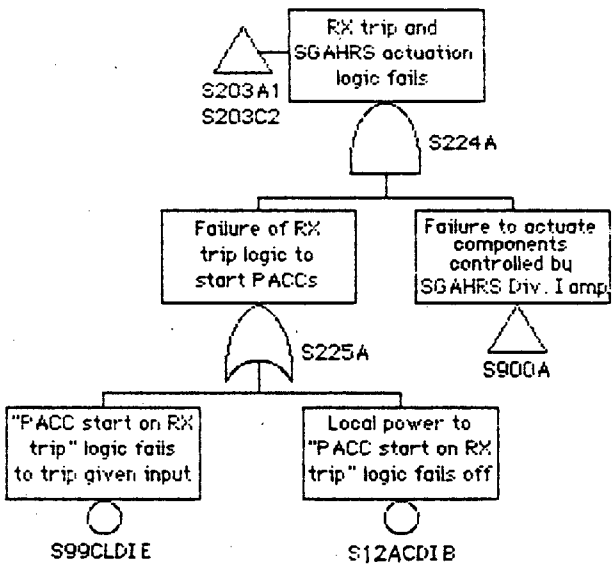

Desired heat rejection signal fails low
S206A1
S207A1
Required heat change calculator fails low
S56CURHA1B
No input to required heat change calculator
S208A1
Actual to desired steam drum pres. comparator fails low
S56CPPA1B
Steam drum pressure detector signal fails low
S209A1
NOTE: Data for all detectors includes sensing element and transmitter.
2/3 steam drum pressure signals fail low
2/3
S210A
S/D pressure median selector fails low
S56PMESA1B
S/D pressure detector signal fails low Loop A CH A
S211A
S/D pressure detector signal fails low Loop A CH B
S212A
S/D pressure detector signal fails low Loop A CH C
S213A
S/D pressure detector fails low Loop A CH A
S56PSAAB
Local power to S/D pressure fails off CH A
S12ACAAB
S/D pressure buffer fails low Loop A CH A
S56BUAAB
S/D pressure detector fails low Loop A CH B
S56PSABB
Local power to S/D pressure fails off CH B
S12ACABB
S/D pressure buffer fails low Loop A CH B
S56BUABB
S/D pressure detector fails low Loop A CH C
S56PSACB
Local power to S/D pressure fails off CH C
S12ACACB
S/D pressure buffer fails low Loop A CH C
S56BUACB


Actual heat rejection signal fails high
S206A1
NOTE: Data for all detectors includes sensing element and transmitter.
S214A1
Return temp. detector signal fails low
Return flow detector signal fails high Loop A
Steam drum pressure detector signal fails high Loop A
Heat rejection calculator fails high
S216A
S56FEAA
S215A1
S56CUHRA1A
Return temperature detector fails low
Local power to return temp. detector fails off
2/3 steam drum pressure signals fail high
S/D pressure median selector fails high
S56TEAB
S12ACAB
2/3
S217A
S56PMESA1A
S/D pressure detector signal fails high Loop A CH A
S/D pressure detector signal fails high Loop A CH B
S/D pressure detector signal fails high Loop A CH C
S218A
S219A
S220A
S/D pressure detector fails high Loop A CH A
S/D pressure buffer fails high Loop A CH A
S/D pressure detector fails high Loop A CH C
S/D pressure buffer fails high Loop A CH C
S56PSAAA
S56BUAAA
S56PSACA
S56BUACA
S/D pressure detector fails high Loop A CH B
S/D pressure buffer fails high Loop A CH B
S56PSABA
S56BUABA

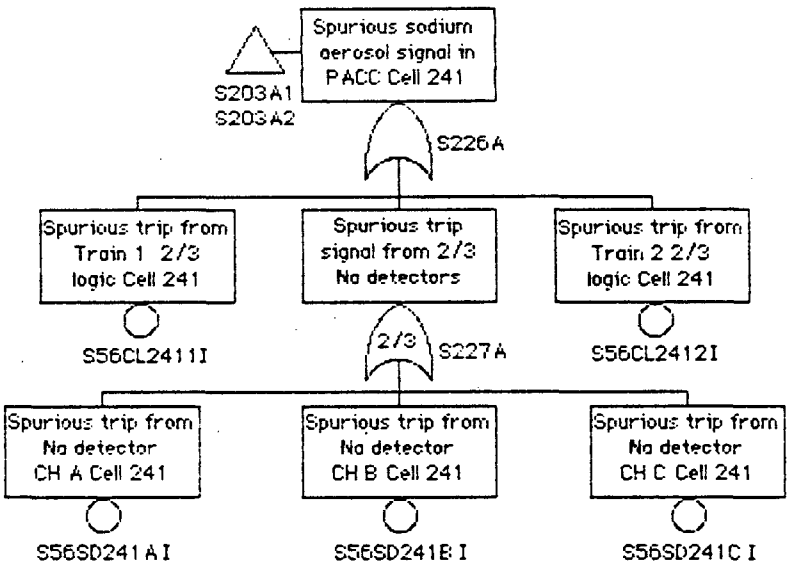


NOTE: This logic is assumed through analysis of ECP G1088 description of proposed change in noncondensible vent valve control.





NOTE: Primary and secondary RSS logic output splits: one signal to rod control and one signal to auxiliaries (SGAHRS and PHTS pumps). Therefore, given a scram, the only way to prevent actuation of PACCs is failure of SGAHRS logic modules.
REF: Drawings- 1440E26 and 1440E42



NOTE: No detectors are photoelectric smoke detectors; data includes current amplifier.

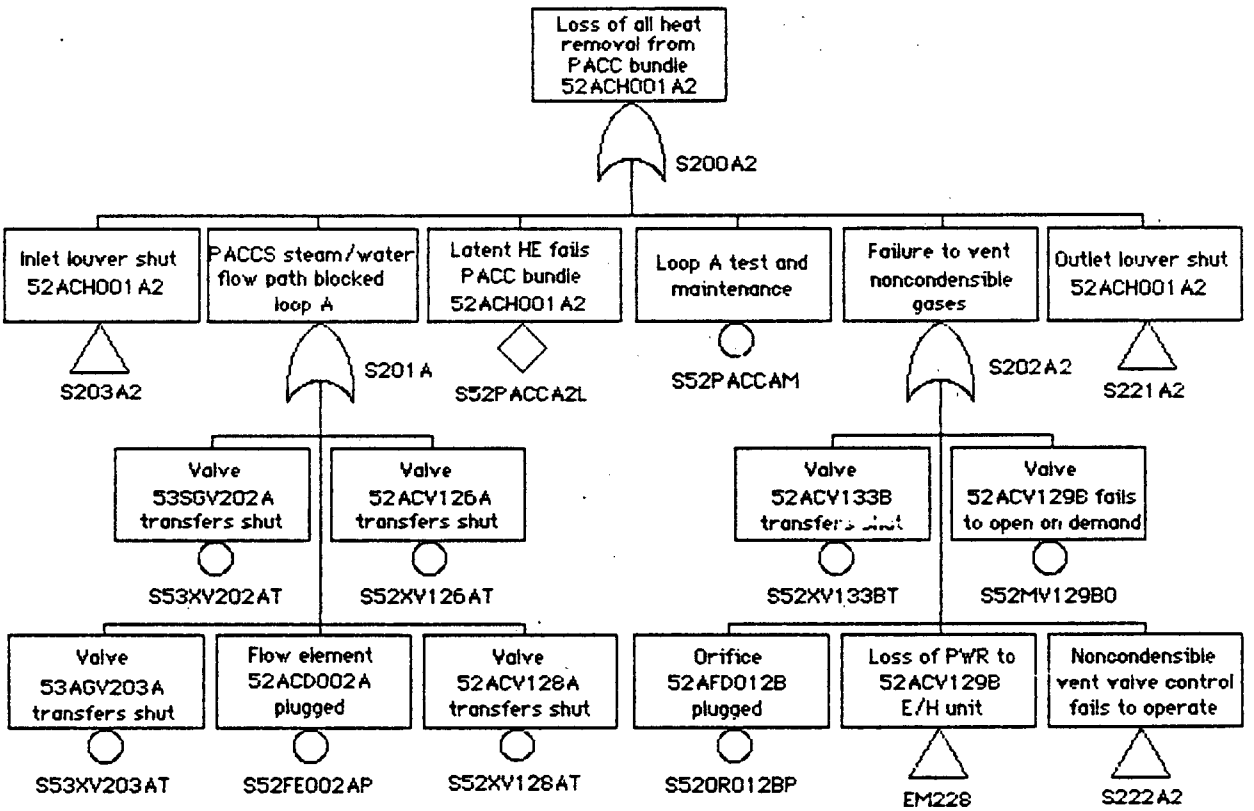
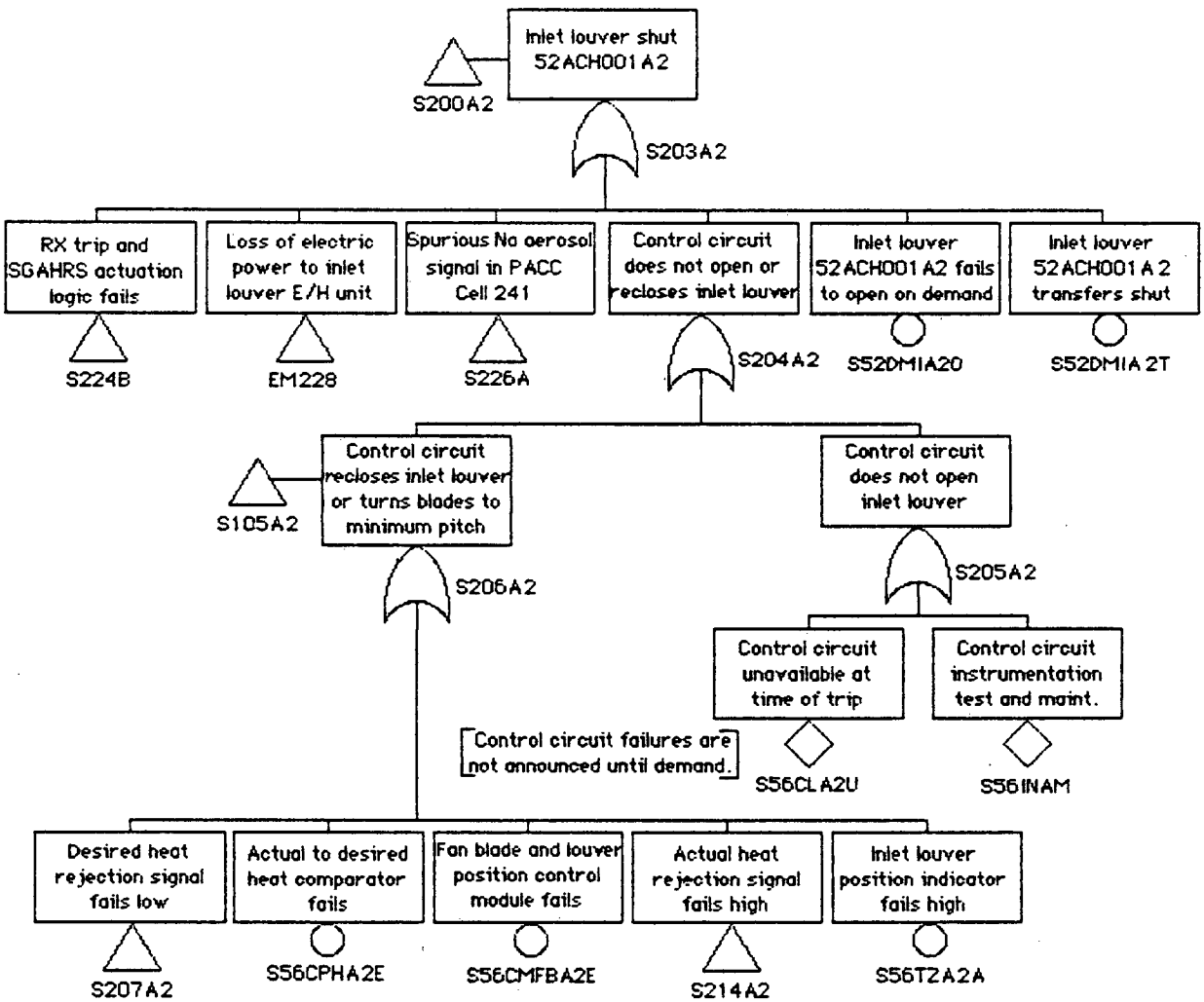
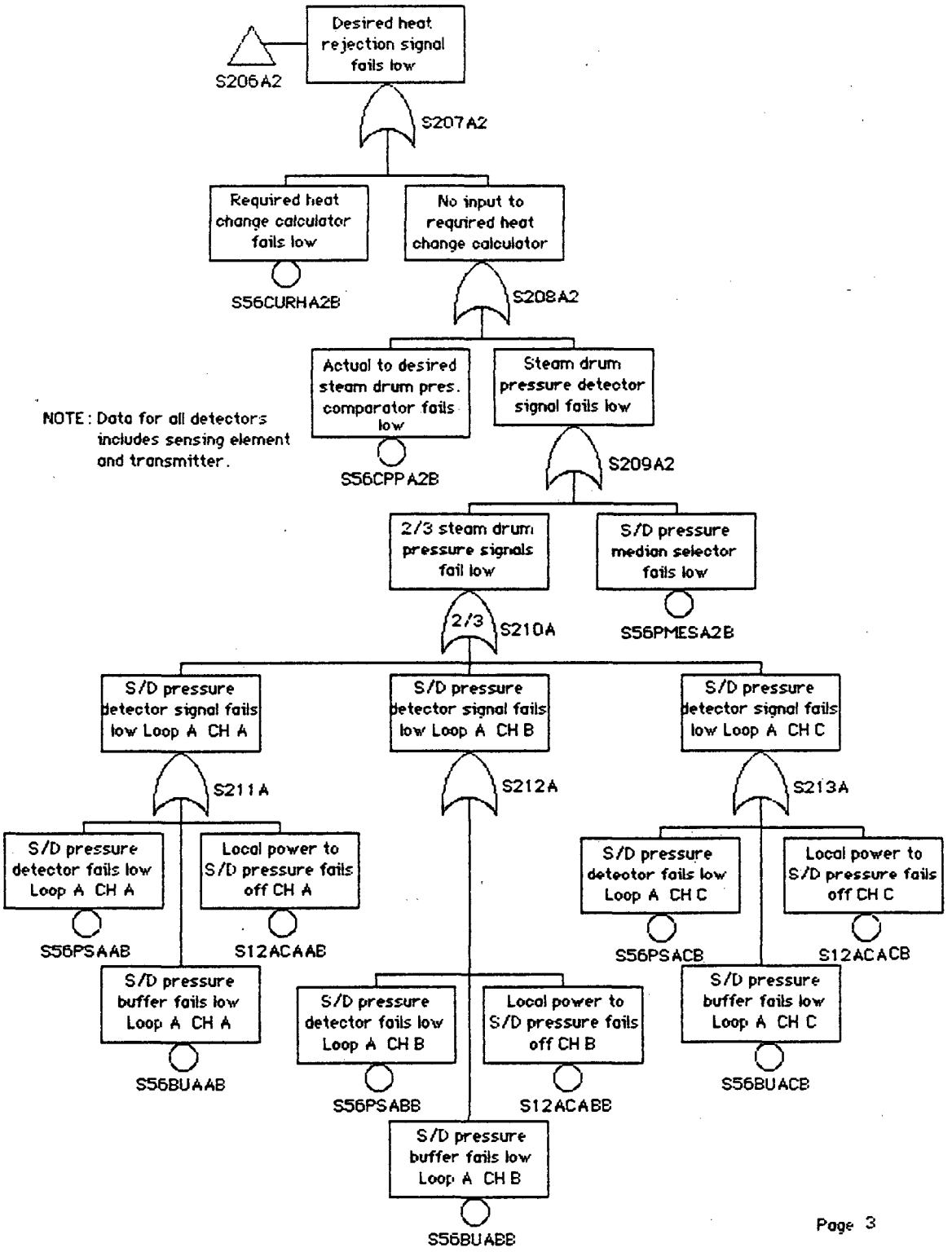
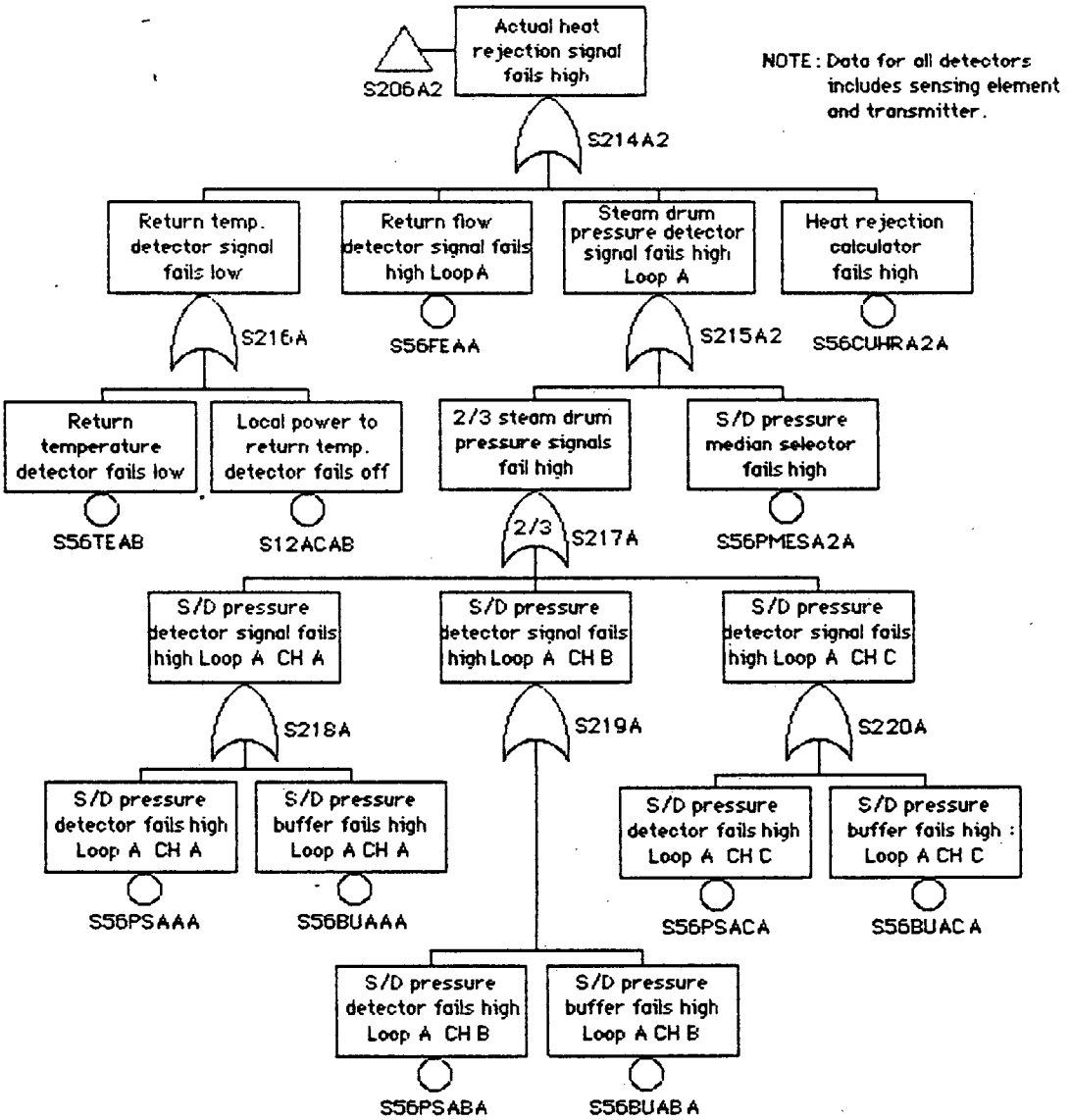
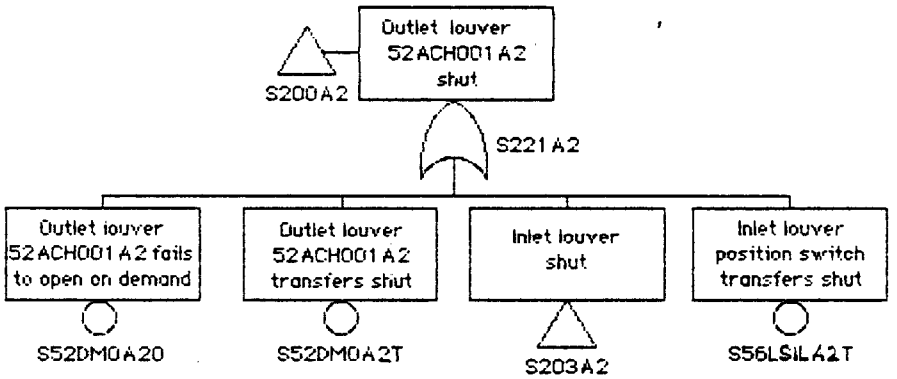


Figure A5-3. PACC Fault Tree (Detailed) for Tube Bundle 52ACH001A2.

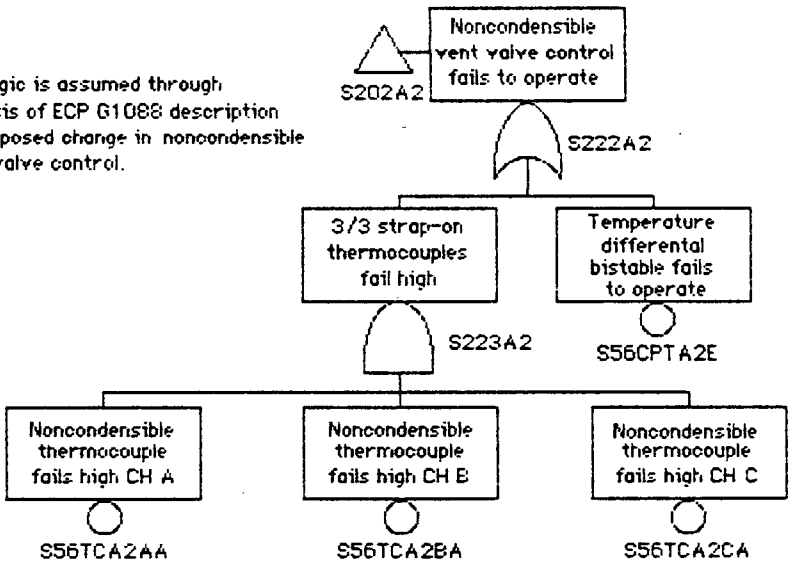

Inlet louver shut 52ACH001A2
S200A2
S203A2
RX trip and SGAHRS actuation logic fails
Loss of electric power to inlet louver E/H unit
Spurious No aerosol signal in PACC Cell 241
Control circuit does not open or recloses inlet louver
Inlet louver 52ACH001A2 fails to open on demand
Inlet louver 52ACH001A2 transfers shut
S224B
EM228
S226A
S204A2
S52DMIA20
S52DMIA2T
Control circuit recloses inlet louver or turns blades to minimum pitch
Control circuit does not open inlet louver
S105A2
S206A2
S205A2
Control circuit unavailable at time of trip
Control circuit instrumentation test and maint.
Control circuit failures are not announced until demand.
S56CLA2U
S56INAM
Desired heat rejection signal fails low
Actual to desired heat comparator fails
Fan blade and louver position control module fails
Actual heat rejection signal fails high
Inlet louver position indicator fails high
S207A2
S56CPHA2E
S56CMFBA2E
S214A2
S56TZA2A

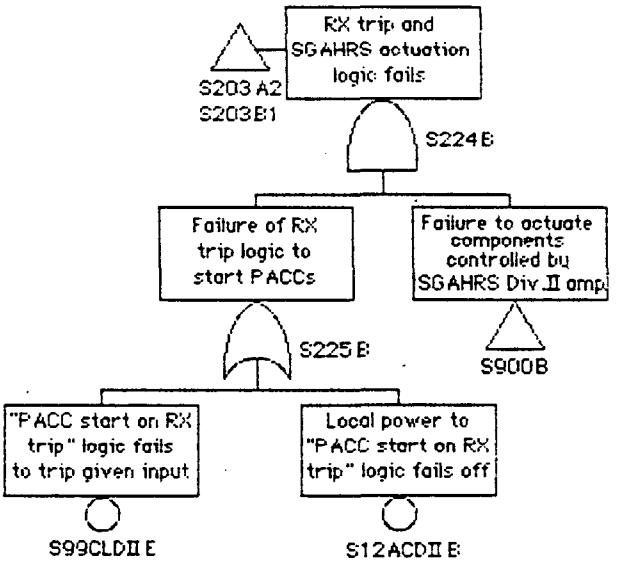

Desired heat rejection signal fails low
S206A2
S207A2
Required heat change calculator fails low
S56CURHA2B
No input to required heat change calculator
S208A2
Actual to desired steam drum pres. comparator fails low
S56CPPA2B
Steam drum pressure detector signal fails low
S209A2
NOTE: Data for all detectors includes sensing element and transmitter.
2/3 steam drum pressure signals fail low
2/3
S210A
S/D pressure median selector fails low
S56PMESA2B
S/D pressure detector signal fails low Loop A CH A
S211A
S/D pressure detector signal fails low Loop A CH B
S212A
S/D pressure detector signal fails low Loop A CH C
S213A
S/D pressure detector fails low Loop A CH A
S56PSAAB
Local power to S/D pressure fails off CH A
S12ACAAB
S/D pressure buffer fails low Loop A CH A
S56BUAAB
S/D pressure detector fails low Loop A CH B
S56PSABB
Local power to S/D pressure fails off CH B
S12ACABB
S/D pressure buffer fails low Loop A CH B
S56BUABB
S/D pressure detector fails low Loop A CH C
S56PSACB
Local power to S/D pressure fails off CH C
S12ACACB
S/D pressure buffer fails low Loop A CH C
S56BUACB


Actual heat rejection signal fails high
S206A2
NOTE: Data for all detectors includes sensing element and transmitter.
S214A2
Return temp. detector signal fails low
Return flow detector signal fails high Loop A
Steam drum pressure detector signal fails high Loop A
Heat rejection calculator fails high
S216A
S56FEAA
S215A2
S56CUHRA2A
Return temperature detector fails low
Local power to return temp. detector fails off
2/3 steam drum pressure signals fail high
S/D pressure median selector fails high
S56TEAB
S12ACAB
2/3
S217A
S56PMESA2A
S/D pressure detector signal fails high Loop A CH A
S/D pressure detector signal fails high Loop A CH B
S/D pressure detector signal fails high Loop A CH C
S218A
S219A
S220A
S/D pressure detector fails high Loop A CH A
S/D pressure buffer fails high Loop A CH A
S/D pressure detector fails high Loop A CH C
S/D pressure buffer fails high : Loop A CH C
S56PSAAA
S56BUAAA
S56PSACA
S56BUACA
S/D pressure detector fails high Loop A CH B
S/D pressure buffer fails high Loop A CH B
S56PSABA
S56BUABA



NOTE: This logic is assumed through analysis of ECP G1088 description of proposed change in noncondensible vent valve control.





NOTE: Primary and secondary RSS logic output splits: one signal to rod control and one signal to auxiliaries (SGAHRS and PHTS pumps). Therefore, given a scram, the only way to prevent actuation of PACCs is failure of SGAHRS logic modules.
REF: Drawings- 1440E26 and 1440E42

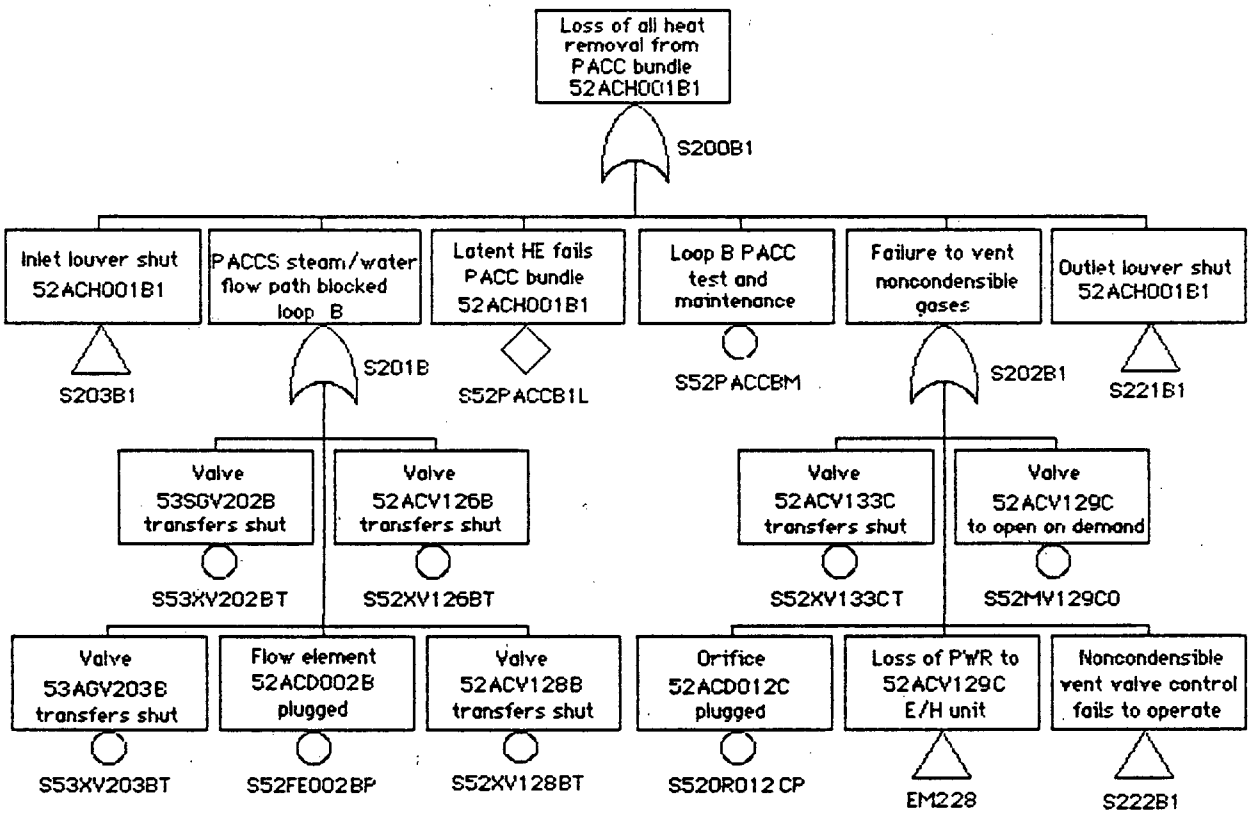
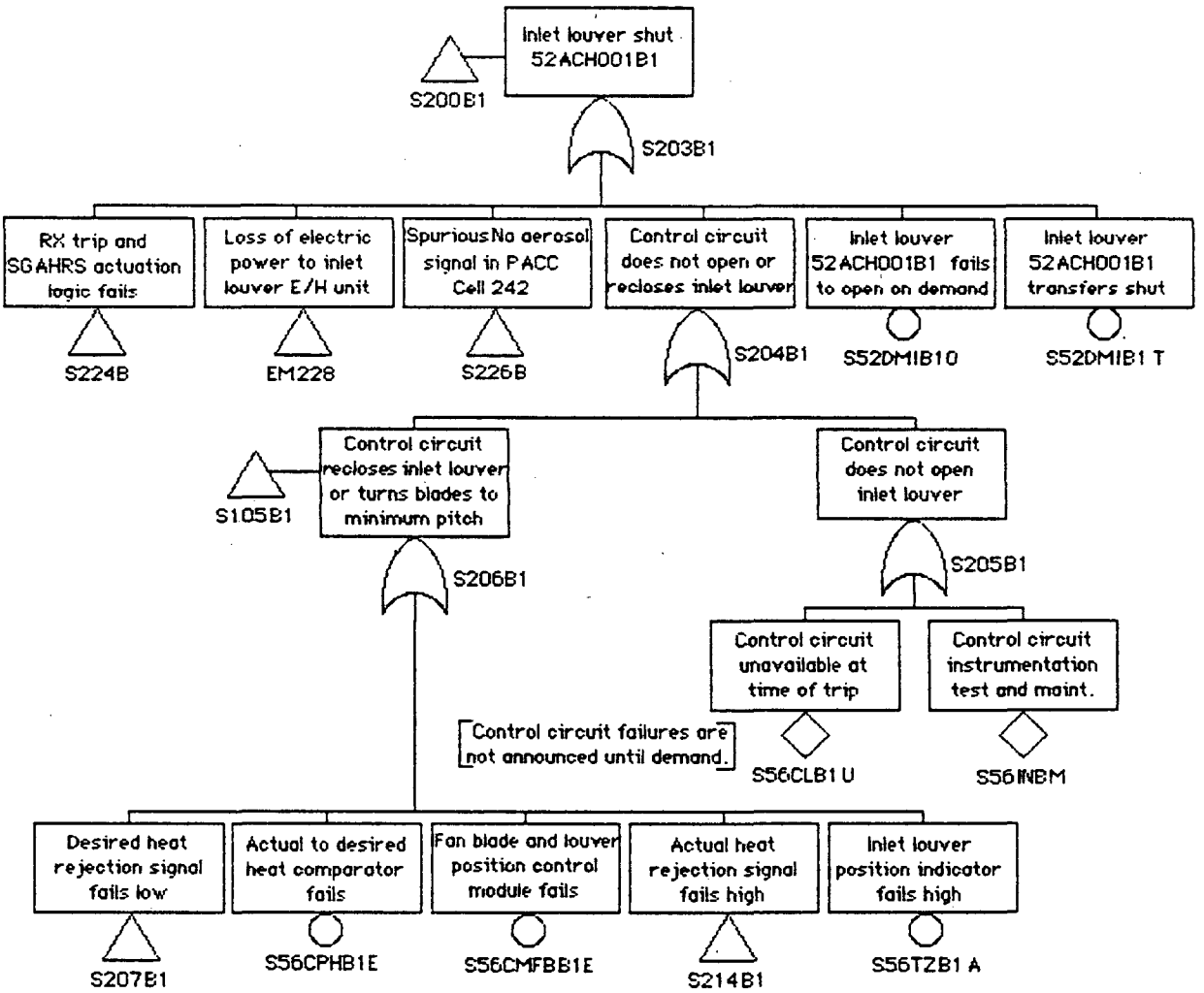
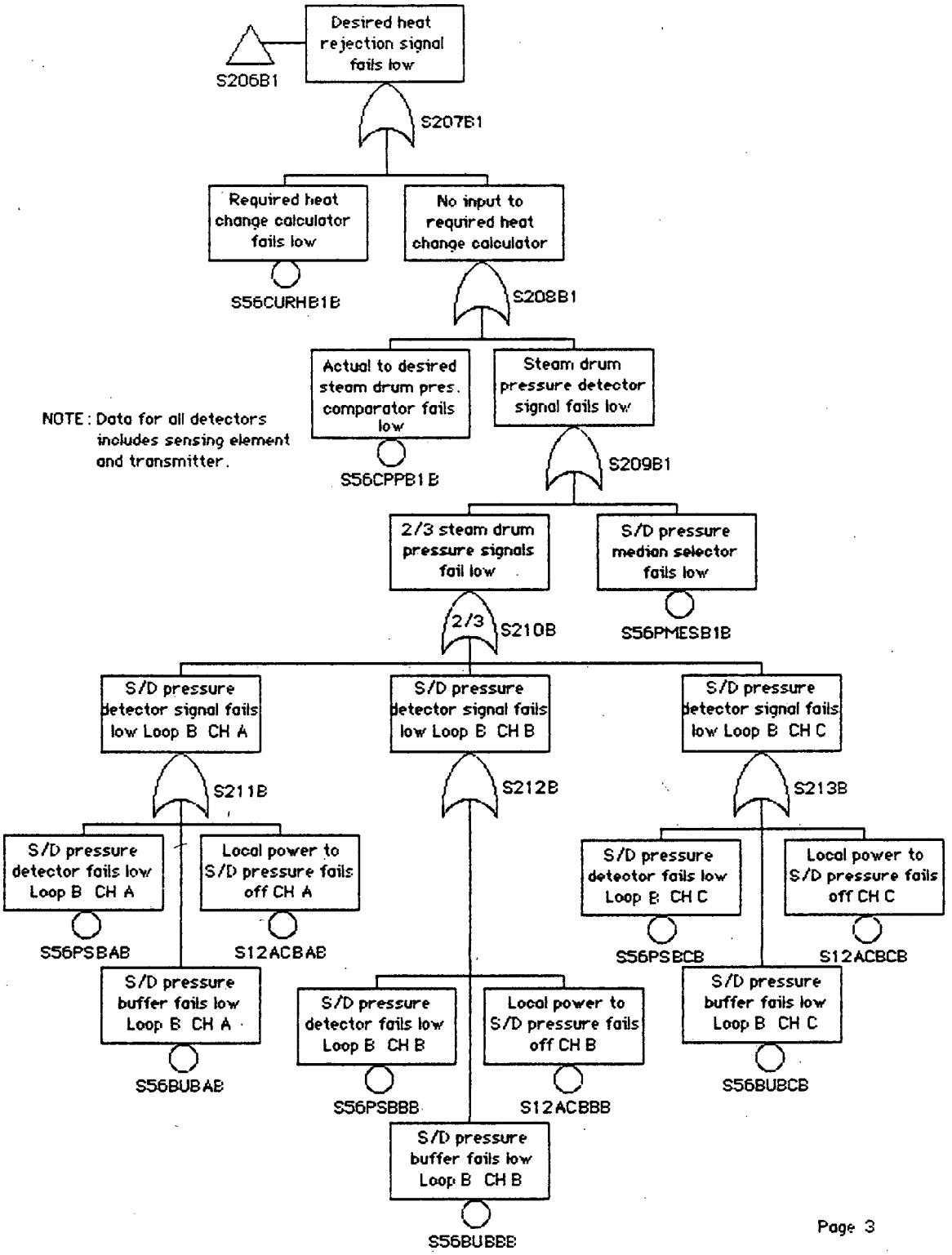
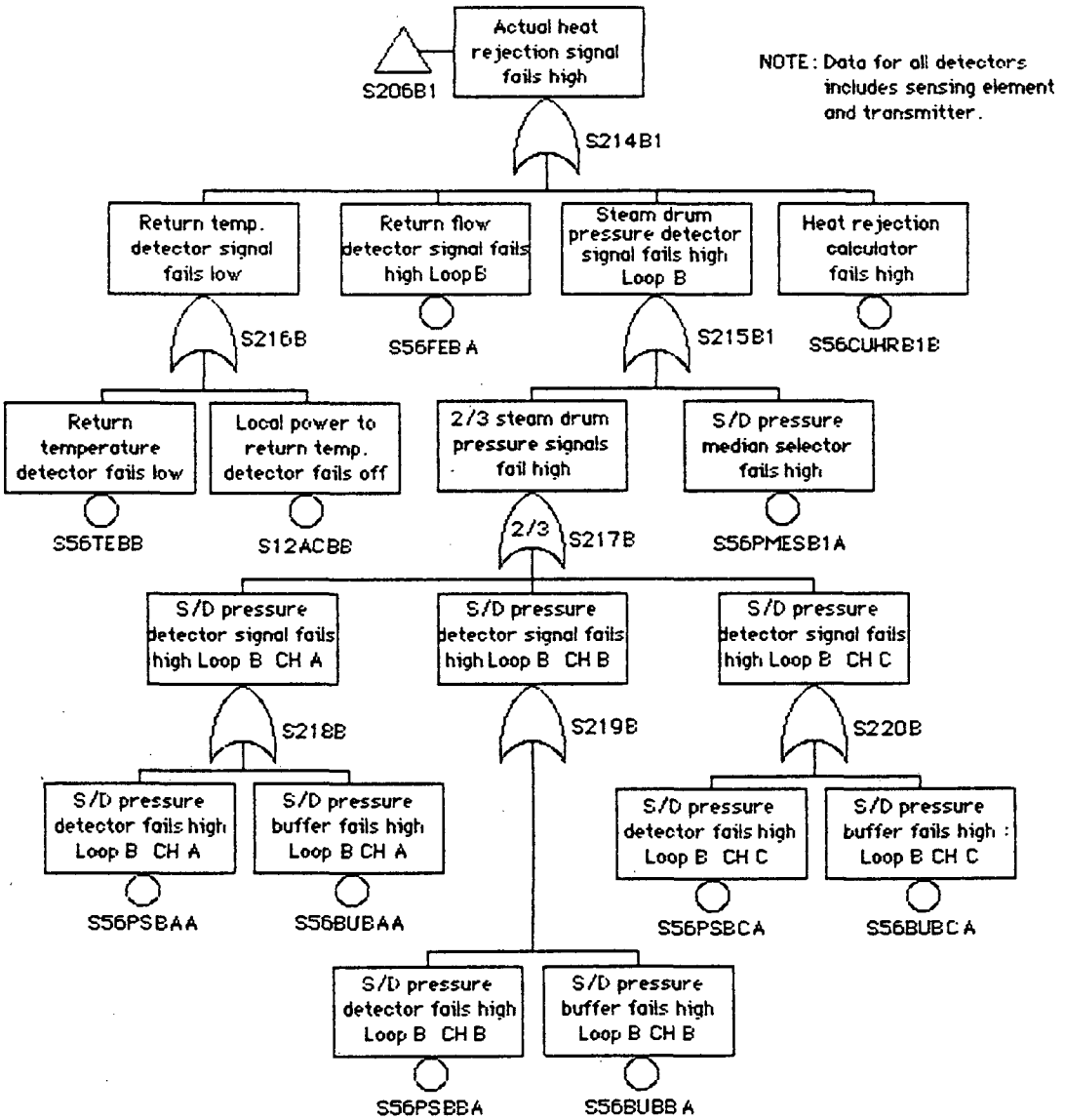


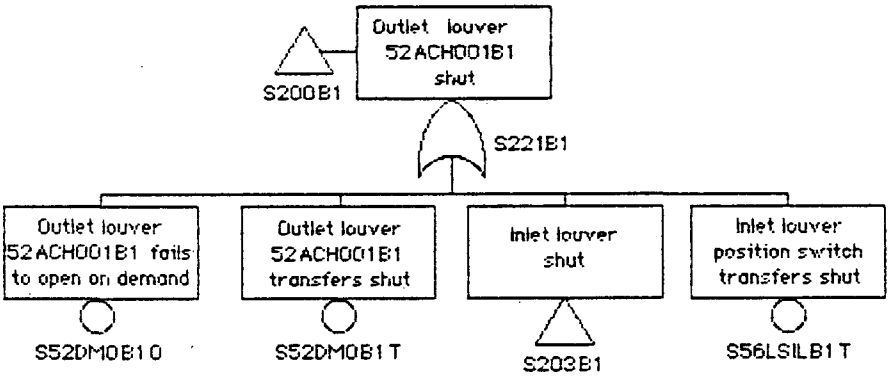
Figure A5-4. PACC Fault Tree (Detailed) for Tube Bundle 52ACH001B1.


Inlet louver shut 52ACH001B1
S200B1
S203B1
RX trip and SGAHRS actuation logic fails
Loss of electric power to inlet louver E/H unit
Spurious No aerosol signal in PACC Cell 242
Control circuit does not open or recloses inlet louver
Inlet louver 52ACH001B1 fails to open on demand
Inlet louver 52ACH001B1 transfers shut
S224B
EM228
S226B
S204B1
S52DMIB10
S52DMIB1 T
Control circuit recloses inlet louver or turns blades to minimum pitch
S105B1
S206B1
Control circuit does not open inlet louver
S205B1
Control circuit unavailable at time of trip
Control circuit instrumentation test and maint.
Control circuit failures are not announced until demand.
S56CLB1 U
S56INBM
Desired heat rejection signal fails low
Actual to desired heat comparator fails
Fan blade and louver position control module fails
Actual heat rejection signal fails high
Inlet louver position indicator fails high
S207B1
S56CPHB1E
S56CMFBB1E
S214B1
S56TZB1 A

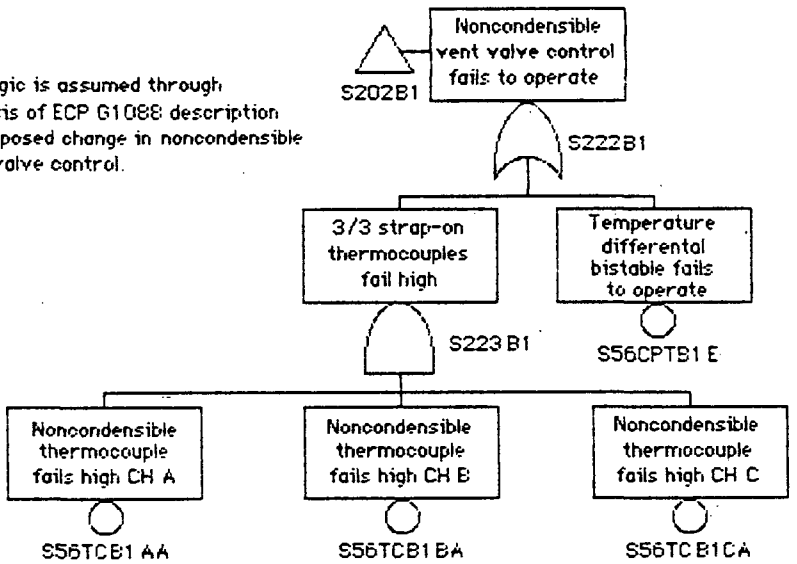


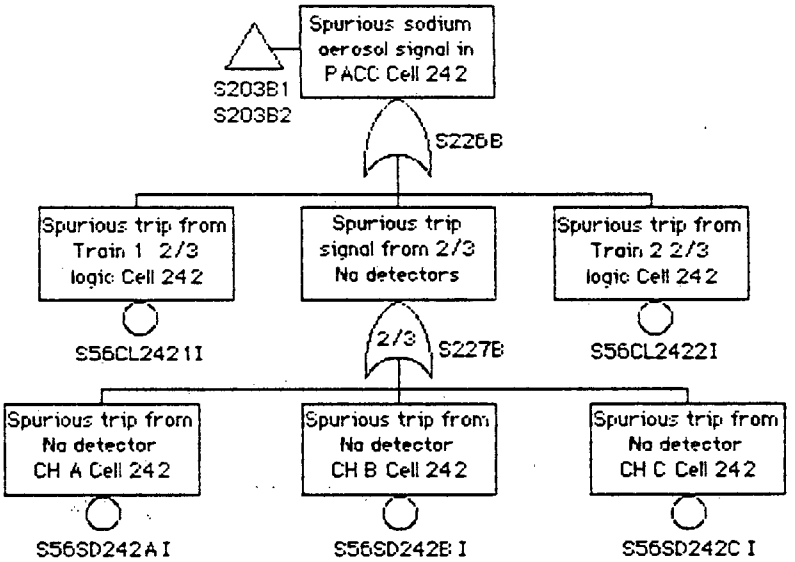
NOTE: Data for all detectors includes sensing element and transmitter.


Actual heat rejection signal fails high
S206B1
NOTE: Data for all detectors includes sensing element and transmitter.
S214B1
Return temp. detector signal fails low
Return flow detector signal fails high Loop B
Steam drum pressure detector signal fails high Loop B
Heat rejection calculator fails high
S216B
S56FEB A
S215B1
S56CUHRB1B
Return temperature detector fails low
Local power to return temp. detector fails off
2/3 steam drum pressure signals fail high
S/D pressure median selector fails high
S56TEBB
S12ACBB
2/3
S217B
S56PMESB1A
S/D pressure detector signal fails high Loop B CH A
S/D pressure detector signal fails high Loop B CH B
S/D pressure detector signal fails high Loop B CH C
S218B
S219B
S220B
S/D pressure detector fails high Loop B CH A
S/D pressure buffer fails high Loop B CH A
S/D pressure detector fails high Loop B CH C
S/D pressure buffer fails high Loop B CH C
S56PSBAA
S56BUBAA
S56PSBCA
S56BUBCA
S/D pressure detector fails high Loop B CH B
S/D pressure buffer fails high Loop B CH B
S56PSBBA
S56BUBBA



NOTE: This logic is assumed through analysis of ECP G1088 description of proposed change in noncondensible vent valve control.





NOTE: Na detectors are photo-electric smoke detectors; data includes current amplifier.

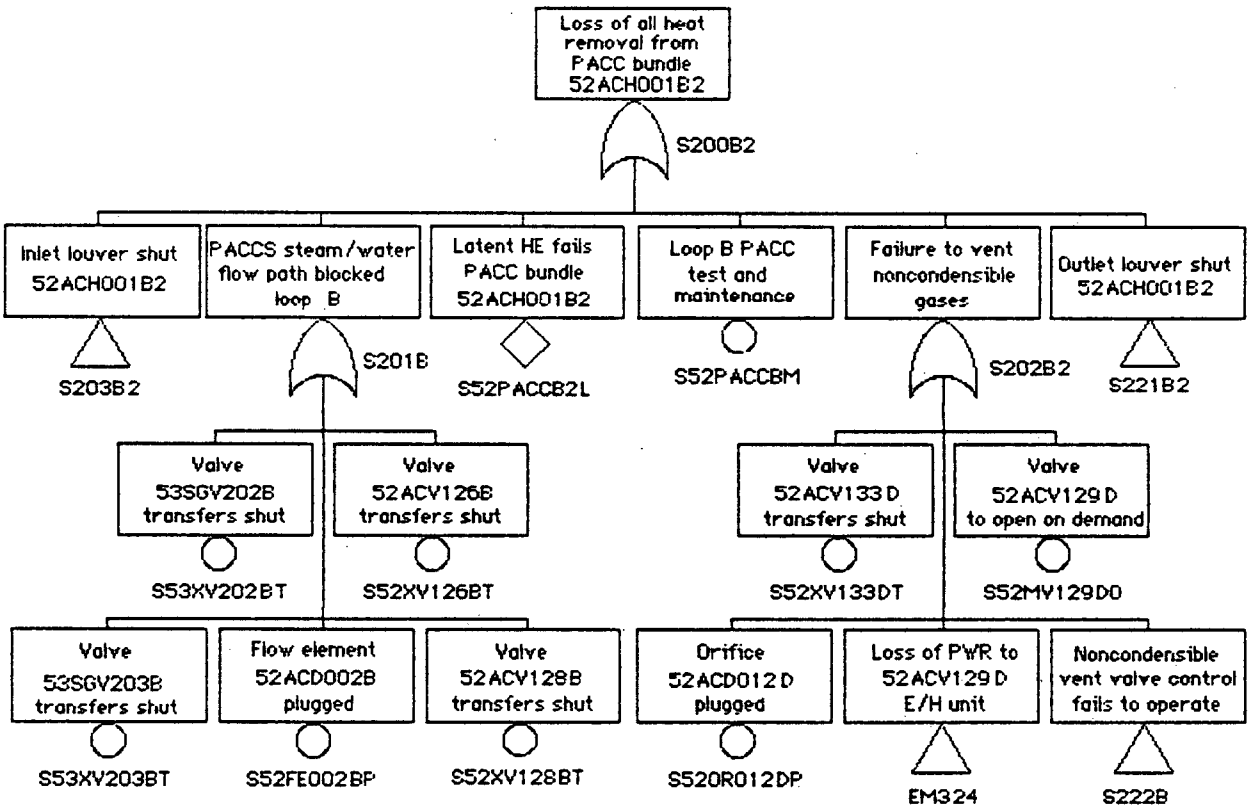
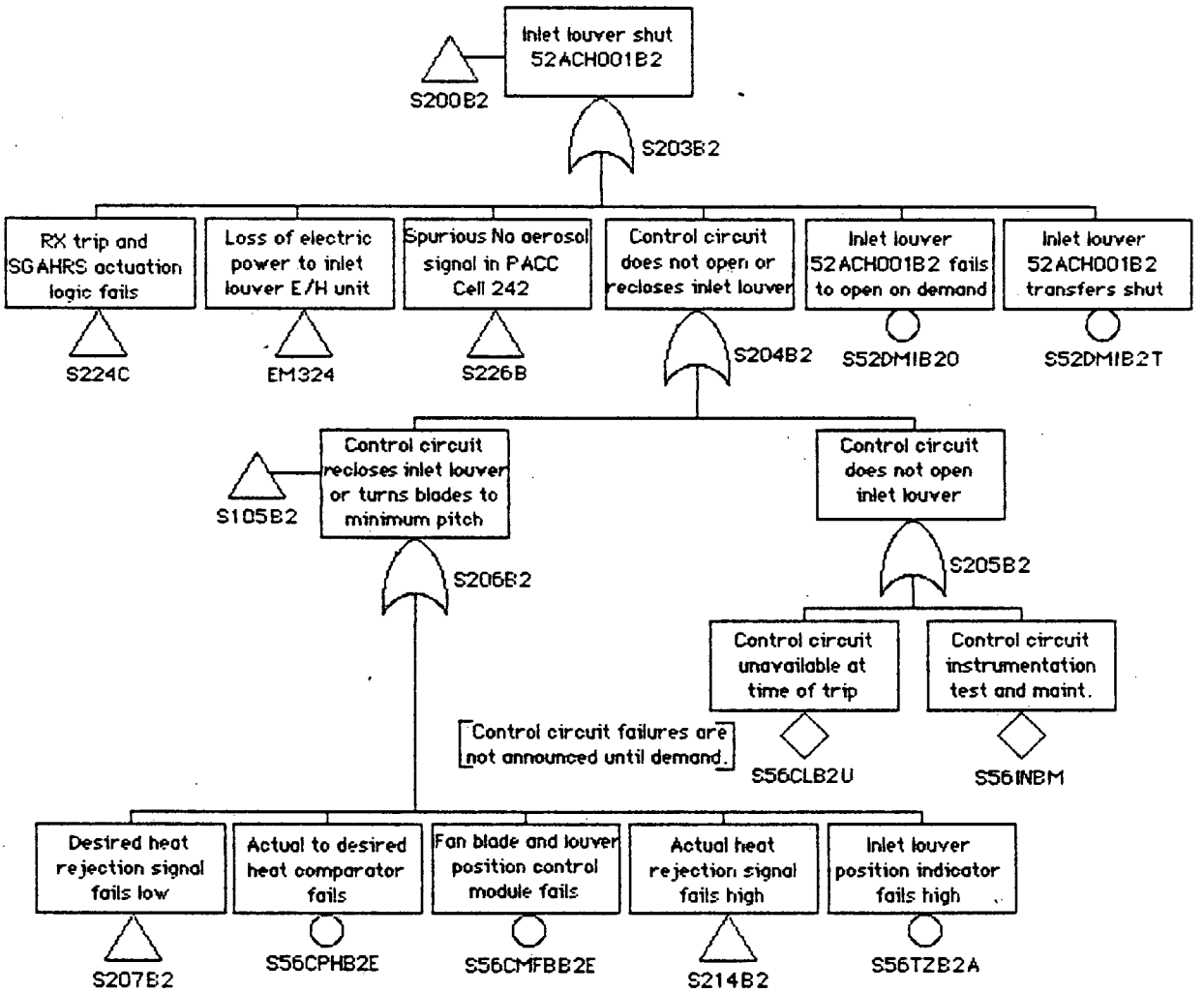
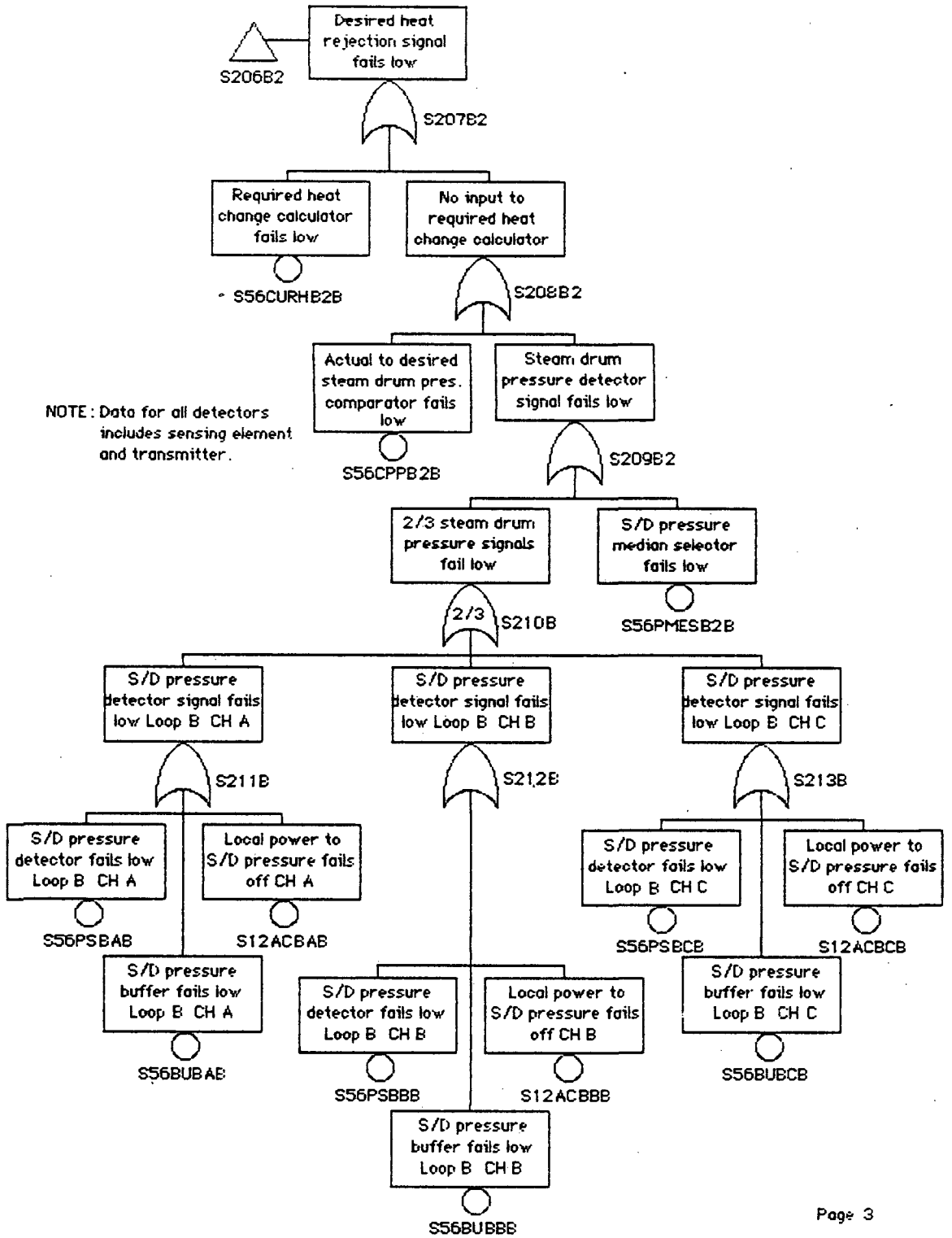
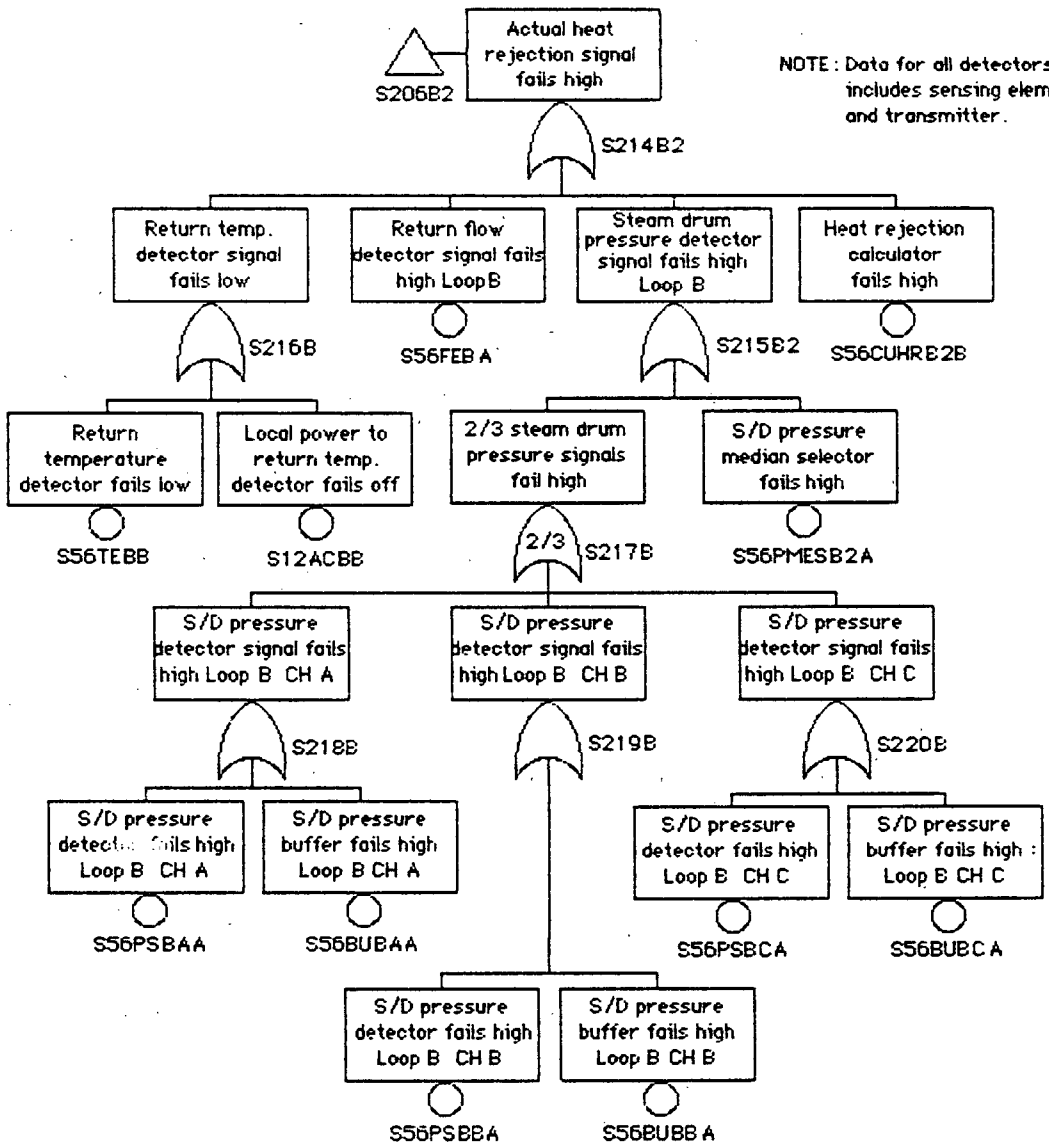
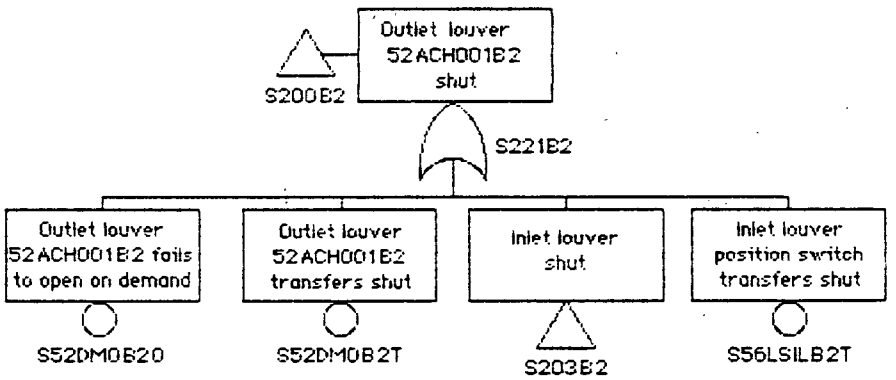


Figure A5-5. PACC Fault Tree (Detailed) for Tube Bundle 52ACH001B2.

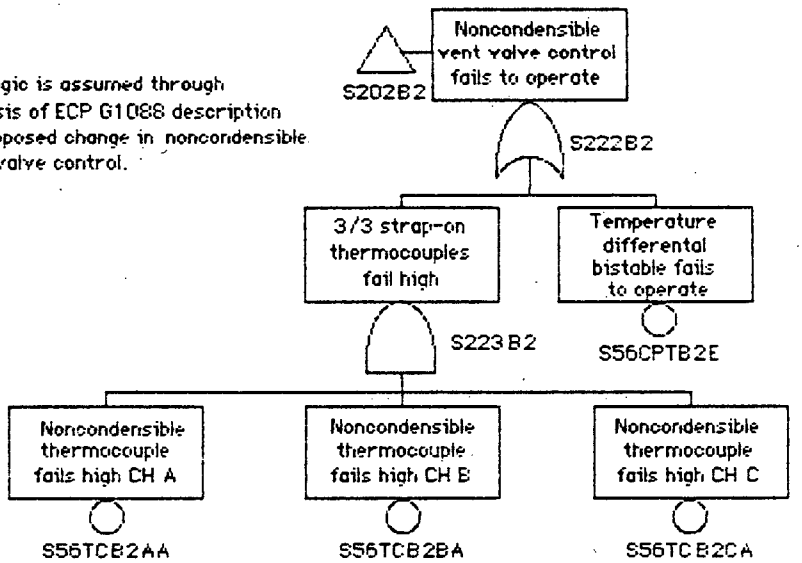

Inlet louver shut 52ACH001B2
S200B2
S203B2
RX trip and SGAHRS actuation logic fails
Loss of electric power to inlet louver E/H unit
Spurious No aerosol signal in PACC Cell 242
Control circuit does not open or recloses inlet louver
Inlet louver 52ACH001B2 fails to open on demand
Inlet louver 52ACH001B2 transfers shut
S224C
EM324
S226B
S204B2
S52DMIB20
S52DMIB2T
Control circuit recloses inlet louver or turns blades to minimum pitch
Control circuit does not open inlet louver
S105B2
S206B2
S205B2
Control circuit unavailable at time of trip
Control circuit instrumentation test and maint.
Control circuit failures are not announced until demand.
S56CLB2U
S56INBM
Desired heat rejection signal fails low
Actual to desired heat comparator fails
Fan blade and louver position control module fails
Actual heat rejection signal fails high
Inlet louver position indicator fails high
S207B2
S56CPHB2E
S56CMFBB2E
S214B2
S56TZB2A

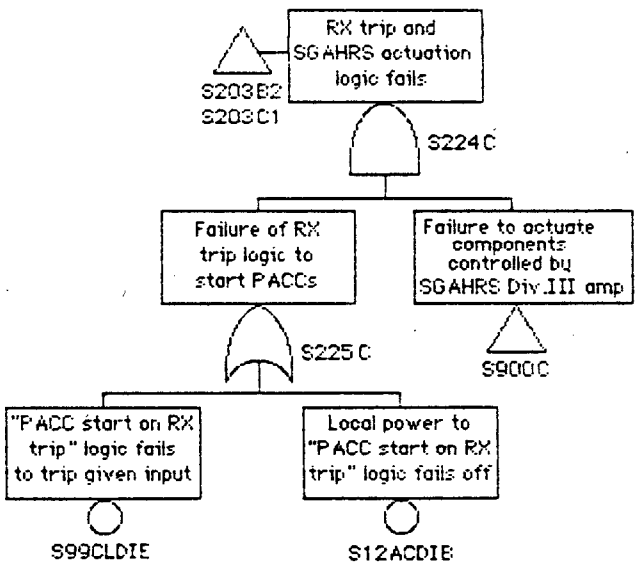

Desired heat rejection signal fails low
S206B2
S207B2
Required heat change calculator fails low
S56CURHB2B
No input to required heat change calculator
S208B2
Actual to desired steam drum pres. comparator fails low
S56CPPB2B
Steam drum pressure detector signal fails low
S209B2
NOTE: Data for all detectors includes sensing element and transmitter.
2/3 steam drum pressure signals fail low
2/3
S210B
S/D pressure median selector fails low
S56PMESB2B
S/D pressure detector signal fails low Loop B CH A
S211B
S/D pressure detector signal fails low Loop B CH B
S212B
S/D pressure detector signal fails low Loop B CH C
S213B
S/D pressure detector fails low Loop B CH A
S56PSBAB
Local power to S/D pressure fails off CH A
S12ACBAB
S/D pressure buffer fails low Loop B CH A
S56BUBAB
S/D pressure detector fails low Loop B CH B
S56PSBBB
Local power to S/D pressure fails off CH B
S12ACBBB
S/D pressure buffer fails low Loop B CH B
S56BUBBB
S/D pressure detector fails low Loop B CH C
S56PSBCB
Local power to S/D pressure fails off CH C
S12ACBCB
S/D pressure buffer fails low Loop B CH C
S56BUBCB


Actual heat rejection signal fails high
S206B2
NOTE : Data for all detectors includes sensing element and transmitter.
S214B2
Return temp. detector signal fails low
Return flow detector signal fails high Loop B
Steam drum pressure detector signal fails high Loop B
Heat rejection calculator fails high
S216B
S56FEB A
S215B2
S56CUHRB2B
Return temperature detector fails low
Local power to return temp. detector fails off
2/3 steam drum pressure signals fail high
S/D pressure median selector fails high
S56TEBB
S12ACBB
2/3
S217B
S56PMESB2A
S/D pressure detector signal fails high Loop B CH A
S/D pressure detector signal fails high Loop B CH B
S/D pressure detector signal fails high Loop B CH C
S218B
S219B
S220B
S/D pressure detector fails high Loop B CH A
S/D pressure buffer fails high Loop B CH A
S/D pressure detector fails high Loop B CH C
S/D pressure buffer fails high Loop B CH C
S56PSBAA
S56BUBAA
S56PSBCA
S56BUBC A
S/D pressure detector fails high Loop B CH B
S/D pressure buffer fails high Loop B CH B
S56PSBBA
S56BUBB A



NOTE : This logic is assumed through analysis of ECP G1088 description of proposed change in noncondensible vent valve control.





NOTE: Primary and secondary RSS logic output splits: one signal to rod control and one signal to auxiliaries (SGAHRS and PHTS pumps). Therefore, given a scram, the only way to prevent actuation of PACCs is failure of SGAHRS logic modules.
REF: Drawings- 1440E26 and 1440E42

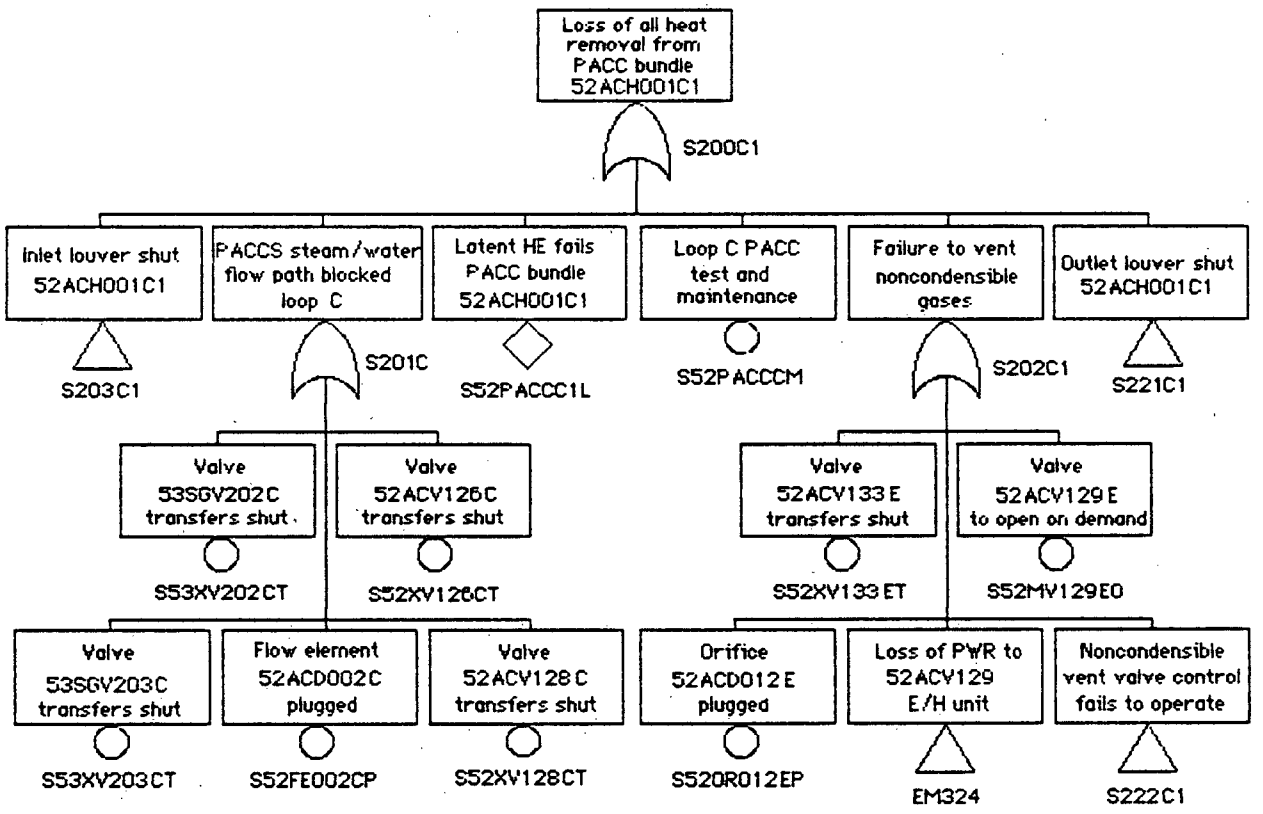
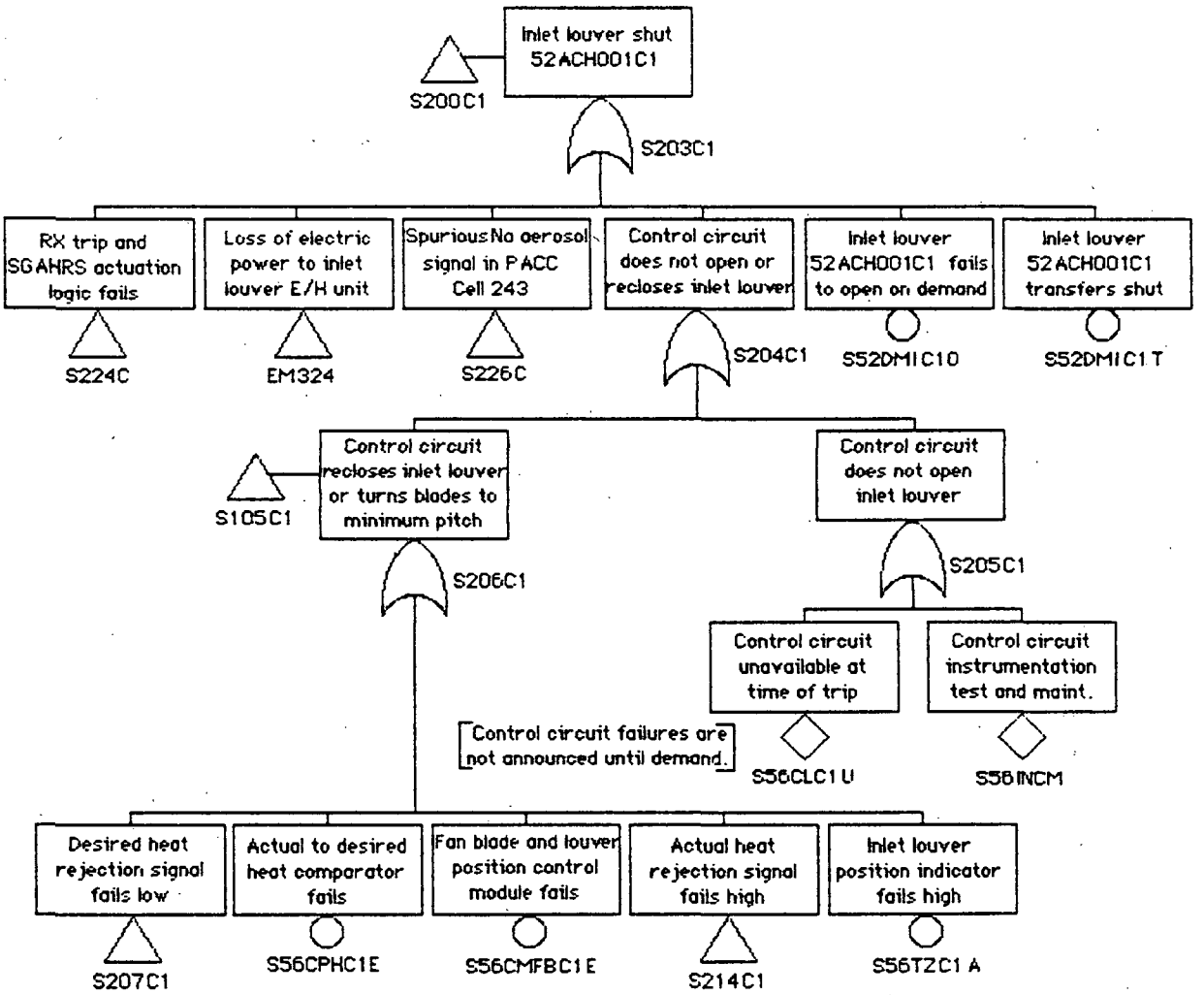
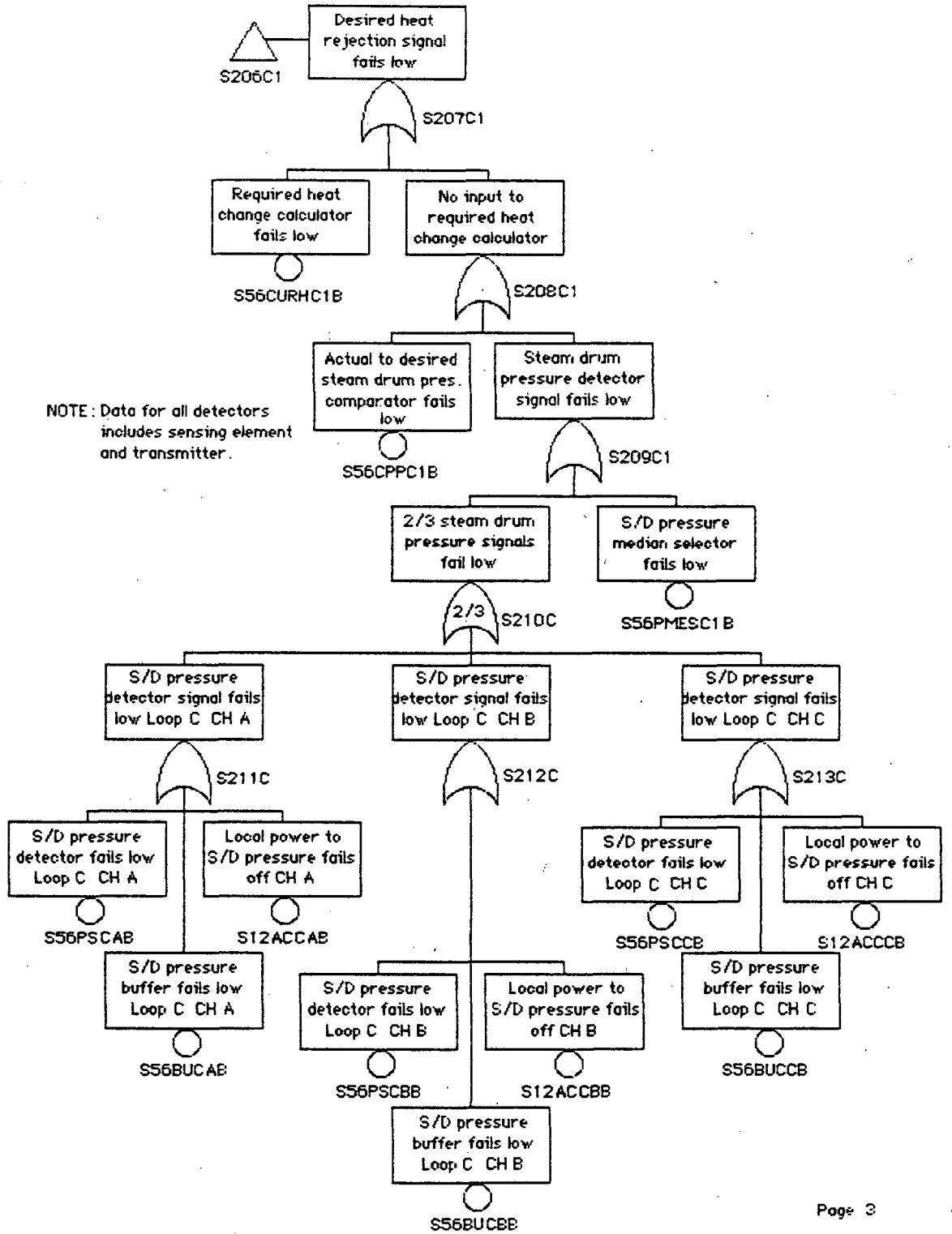
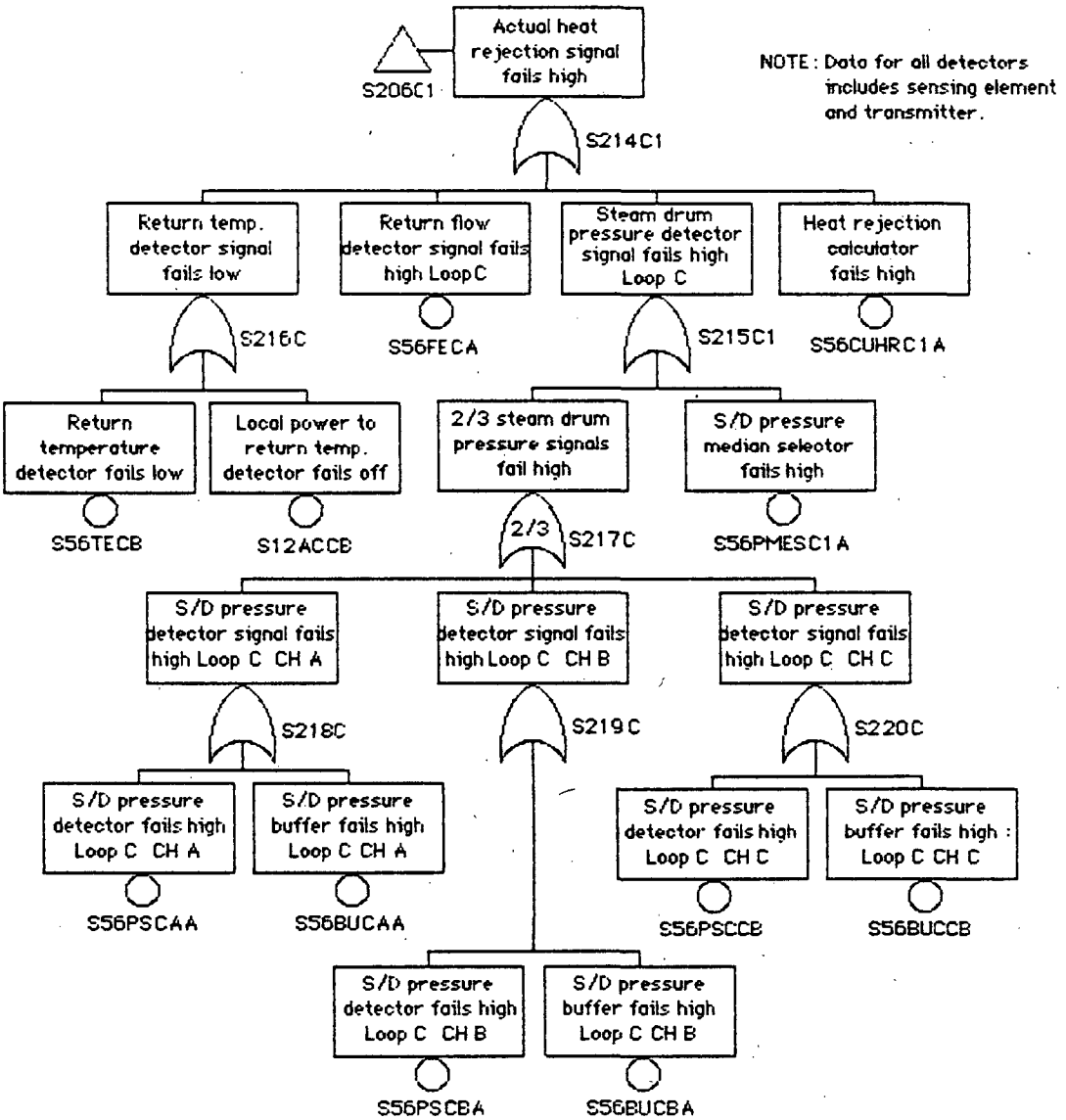
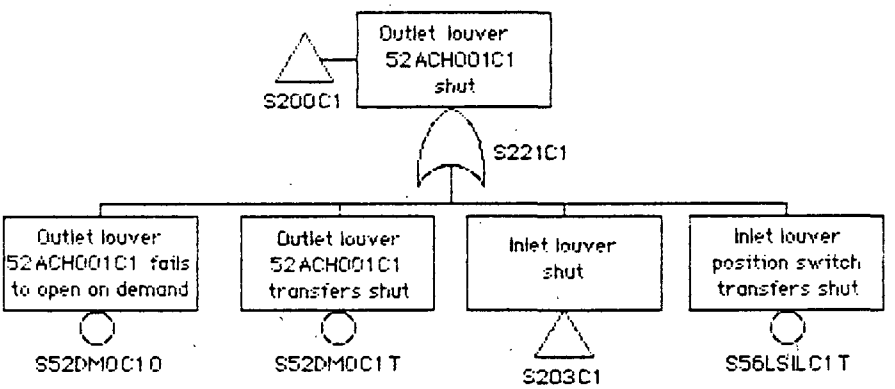


Figure A5-6. PACC Fault Tree (Detailed) for Tube Bundle 52ACH001C1.

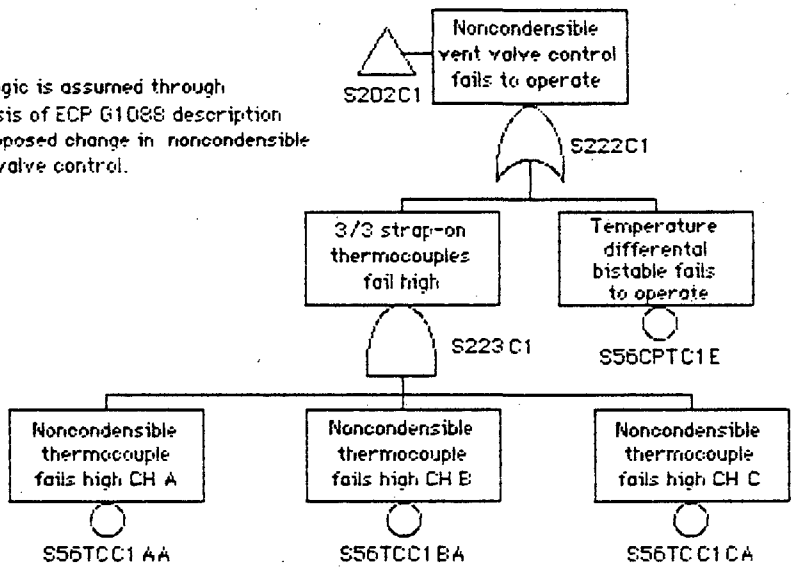

Inlet louver shut 52ACH001C1
S200C1
S203C1
RX trip and SGAHRS actuation logic fails
Loss of electric power to inlet louver E/H unit
Spurious Na aerosol signal in PACC Cell 243
Control circuit does not open or recloses inlet louver
Inlet louver 52ACH001C1 fails to open on demand
Inlet louver 52ACH001C1 transfers shut
S224C
EM324
S226C
S204C1
S52DMIC10
S52DMIC1T
Control circuit recloses inlet louver or turns blades to minimum pitch
Control circuit does not open inlet louver
S105C1
S206C1
S205C1
Control circuit unavailable at time of trip
Control circuit instrumentation test and maint.
Control circuit failures are not announced until demand.
S56CLC1U
S56INCM
Desired heat rejection signal fails low
Actual to desired heat comparator fails
Fan blade and louver position control module fails
Actual heat rejection signal fails high
Inlet louver position indicator fails high
S207C1
S56CPHC1E
S56CMFBC1E
S214C1
S56TZC1A

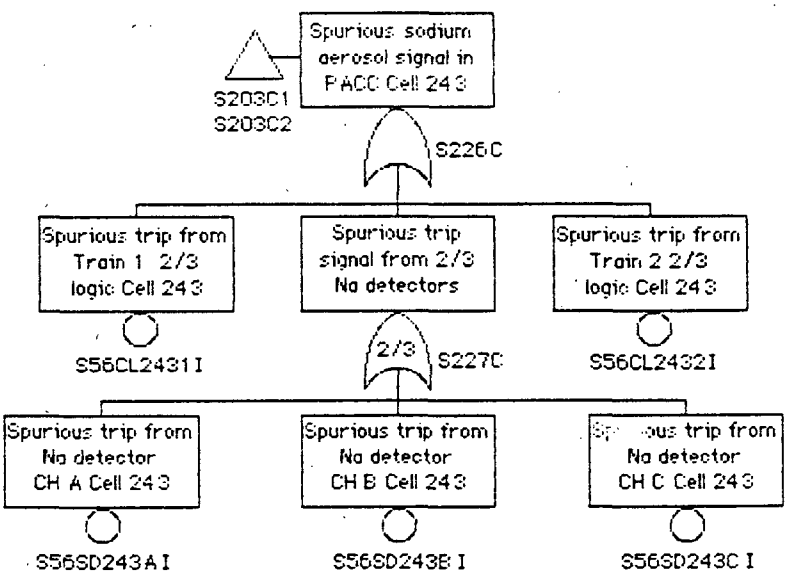

Desired heat rejection signal fails low
S206C1
S207C1
Required heat change calculator fails low
S56CURHC1B
No input to required heat change calculator
S208C1
Actual to desired steam drum pres. comparator fails low
S56CPPC1B
Steam drum pressure detector signal fails low
S209C1
NOTE: Data for all detectors includes sensing element and transmitter.
2/3 steam drum pressure signals fail low
2/3
S210C
S/D pressure median selector fails low
S56PMESC1B
S/D pressure detector signal fails low Loop C CH A
S211C
S/D pressure detector signal fails low Loop C CH B
S212C
S/D pressure detector signal fails low Loop C CH C
S213C
S/D pressure detector fails low Loop C CH A
S56PSCAB
Local power to S/D pressure fails off CH A
S12ACCAB
S/D pressure buffer fails low Loop C CH A
S56BUCAB
S/D pressure detector fails low Loop C CH B
S56PSCBB
Local power to S/D pressure fails off CH B
S12ACCBB
S/D pressure buffer fails low Loop C CH B
S56BUCBB
S/D pressure detector fails low Loop C CH C
S56PSCCB
Local power to S/D pressure fails off CH C
S12ACCCB
S/D pressure buffer fails low Loop C CH C
S56BUCCB


Actual heat rejection signal fails high
S206C1
NOTE: Data for all detectors includes sensing element and transmitter.
S214C1
Return temp. detector signal fails low
Return flow detector signal fails high Loop C
Steam drum pressure detector signal fails high Loop C
Heat rejection calculator fails high
S216C
S56FECA
S215C1
S56CUHRC1A
Return temperature detector fails low
Local power to return temp. detector fails off
2/3 steam drum pressure signals fail high
S/D pressure median selector fails high
S56TECB
S12ACCB
2/3
S217C
S56PMESC1A
S/D pressure detector signal fails high Loop C CH A
S/D pressure detector signal fails high Loop C CH B
S/D pressure detector signal fails high Loop C CH C
S218C
S219C
S220C
S/D pressure detector fails high Loop C CH A
S/D pressure buffer fails high Loop C CH A
S/D pressure detector fails high Loop C CH C
S/D pressure buffer fails high Loop C CH C
S56PSCAA
S56BUCAA
S56PSCCB
S56BUCCB
S/D pressure detector fails high Loop C CH B
S/D pressure buffer fails high Loop C CH B
S56PSCBA
S56BUCBA



NOTE: This logic is assumed through analysis of ECP G1088 description of proposed change in noncondensible vent valve control.





NOTE: Na detectors are photoelectric smoke detectors; data includes current amplifier.

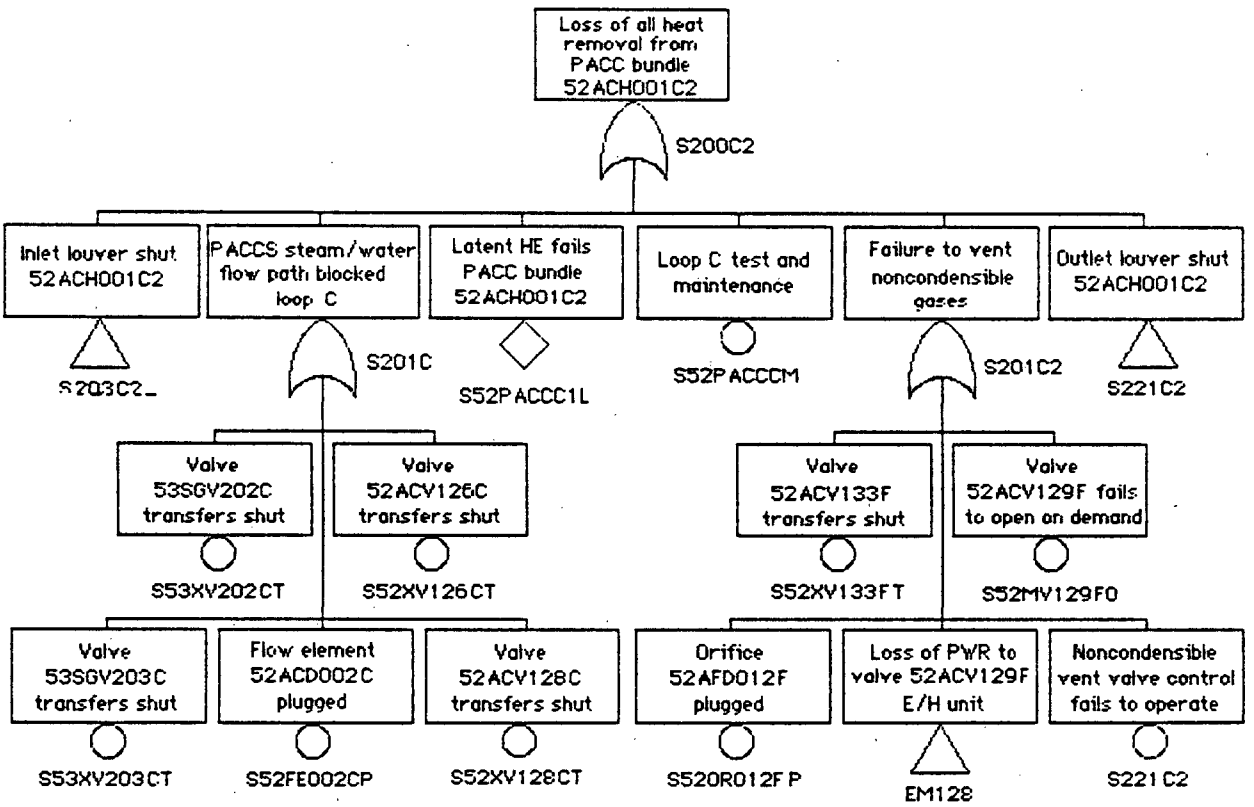
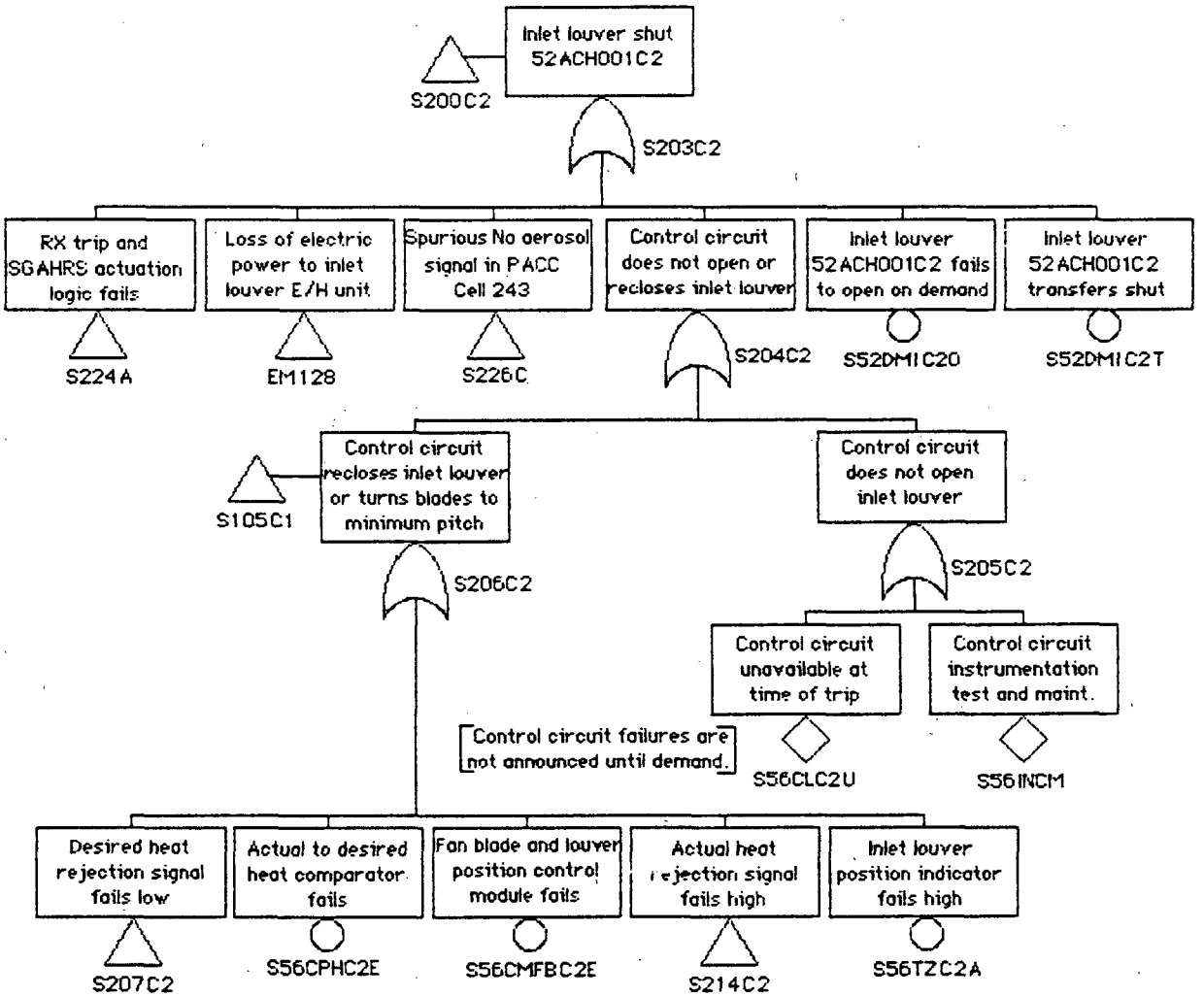
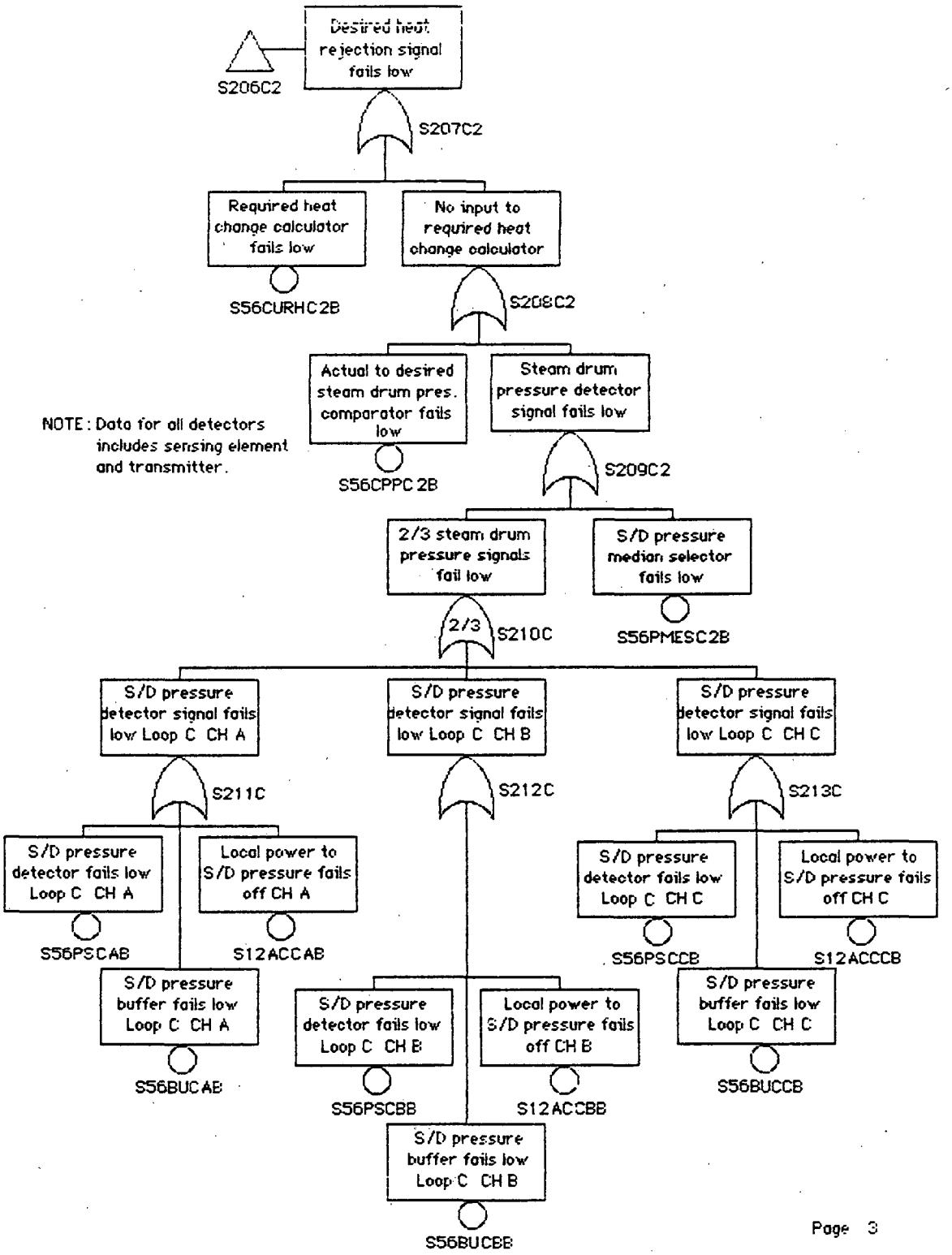
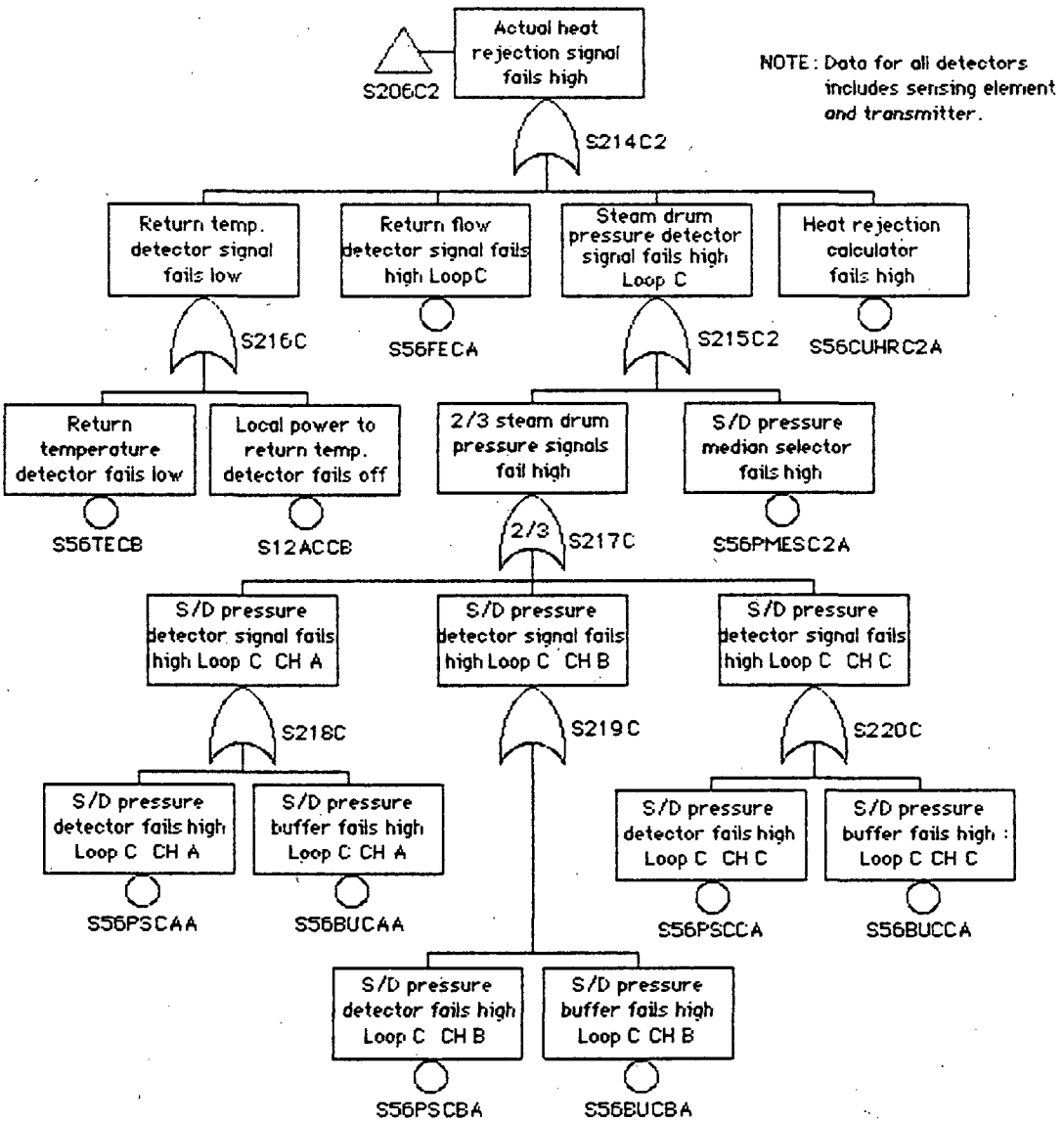
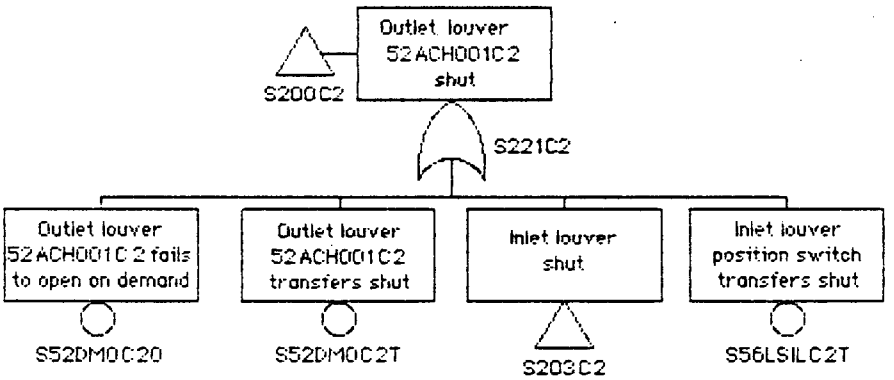


Figure A5-7. PACC Fault Tree (Detailed) for Tube Bundle 52ACH001C2.

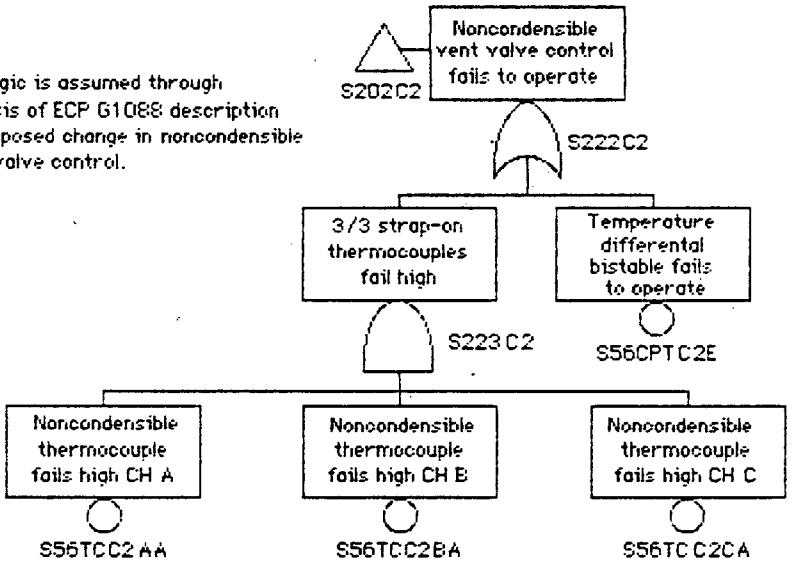

Inlet louver shut 52ACH001C2
S200C2
S203C2
RX trip and SGAHRS actuation logic fails
Loss of electric power to inlet louver E/H unit
Spurious No aerosol signal in PACC Cell 243
Control circuit does not open or recloses inlet louver
Inlet louver 52ACH001C2 fails to open on demand
Inlet louver 52ACH001C2 transfers shut
S224A
EM128
S226C
S204C2
S52DMIC20
S52DMIC2T
Control circuit recloses inlet louver or turns blades to minimum pitch
Control circuit does not open inlet louver
S105C1
S206C2
S205C2
Control circuit unavailable at time of trip
Control circuit instrumentation test and maint.
Control circuit failures are not announced until demand.
S56CLC2U
S56INCM
Desired heat rejection signal fails low
Actual to desired heat comparator fails
Fan blade and louver position control module fails
Actual heat rejection signal fails high
Inlet louver position indicator fails high
S207C2
S56CPHC2E
S56CMFBC2E
S214C2
S56TZC2A

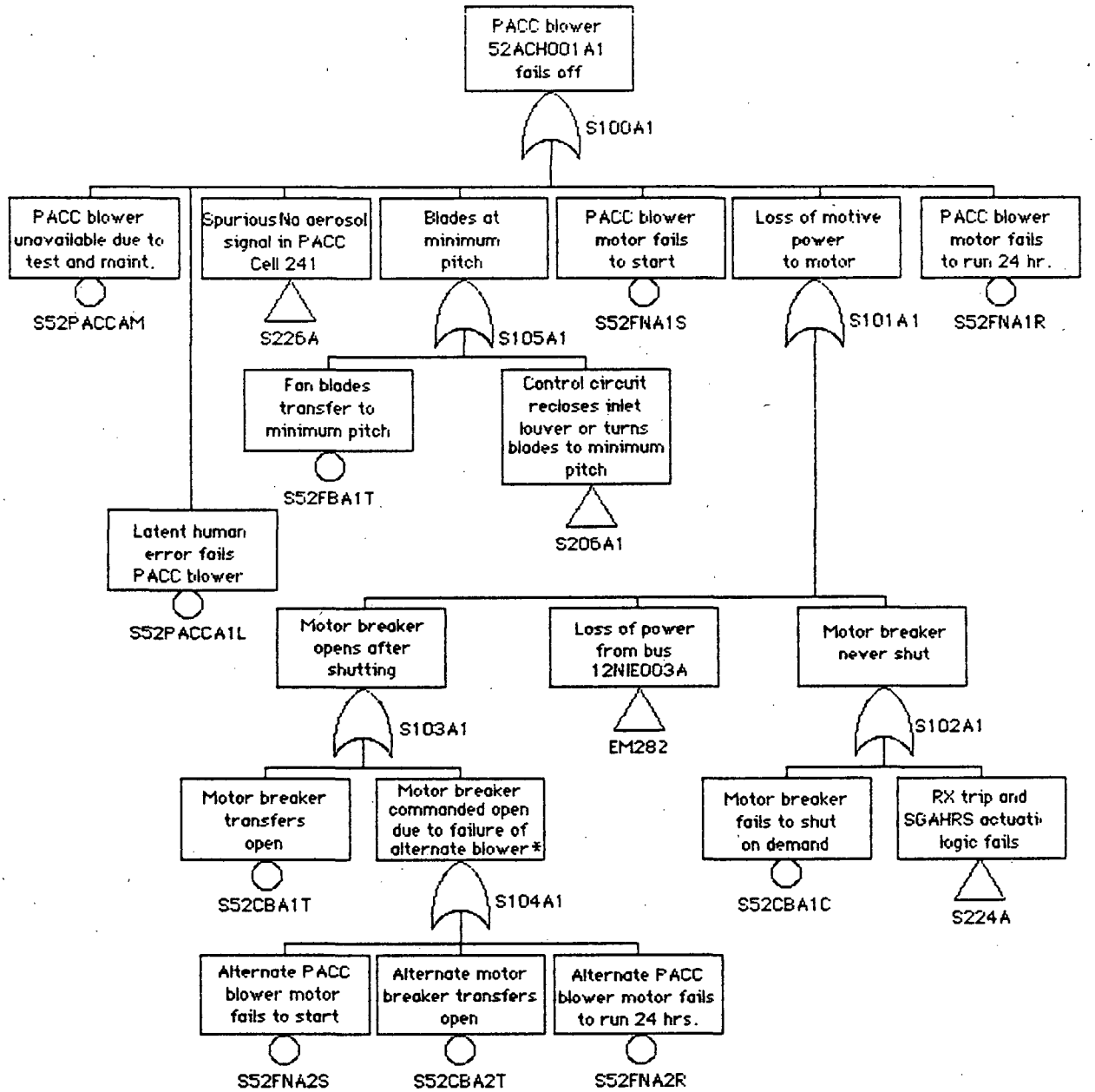

Desired heat rejection signal fails low
S206C2
S207C2
Required heat change calculator fails low
S56CURHC2B
No input to required heat change calculator
S208C2
Actual to desired steam drum pres. comparator fails low
S56CPPC 2B
Steam drum pressure detector signal fails low
S209C2
NOTE: Data for all detectors includes sensing element and transmitter.
2/3 steam drum pressure signals fail low
2/3
S210C
S/D pressure median selector fails low
S56PMESC2B
S/D pressure detector signal fails low Loop C CH A
S211C
S/D pressure detector signal fails low Loop C CH B
S212C
S/D pressure detector signal fails low Loop C CH C
S213C
S/D pressure detector fails low Loop C CH A
S56PSCAB
Local power to S/D pressure fails off CH A
S12ACCAB
S/D pressure buffer fails low Loop C CH A
S56BUCAB
S/D pressure detector fails low Loop C CH B
S56PSCBB
Local power to S/D pressure fails off CH B
S12ACCBB
S/D pressure buffer fails low Loop C CH B
S56BUCBB
S/D pressure detector fails low Loop C CH C
S56PSCCB
Local power to S/D pressure fails off CH C
S12ACCCB
S/D pressure buffer fails low Loop C CH C
S56BUCCB


Actual heat rejection signal fails high
S206C2
NOTE: Data for all detectors includes sensing element and transmitter.
S214C2
Return temp. detector signal fails low
Return flow detector signal fails high Loop C
Steam drum pressure detector signal fails high Loop C
Heat rejection calculator fails high
S216C
S56FECA
S215C2
S56CUHRC2A
Return temperature detector fails low
Local power to return temp. detector fails off
2/3 steam drum pressure signals fail high
S/D pressure median selector fails high
S56TECB
S12ACCB
2/3
S217C
S56PMESC2A
S/D pressure detector signal fails high Loop C CH A
S/D pressure detector signal fails high Loop C CH B
S/D pressure detector signal fails high Loop C CH C
S218C
S219C
S220C
S/D pressure detector fails high Loop C CH A
S/D pressure buffer fails high Loop C CH A
S/D pressure detector fails high Loop C CH C
S/D pressure buffer fails high Loop C CH C
S56PSCAA
S56BUCAA
S56PSCCA
S56BUCCA
S/D pressure detector fails high Loop C CH B
S/D pressure buffer fails high Loop C CH B
S56PSCBA
S56BUCBA



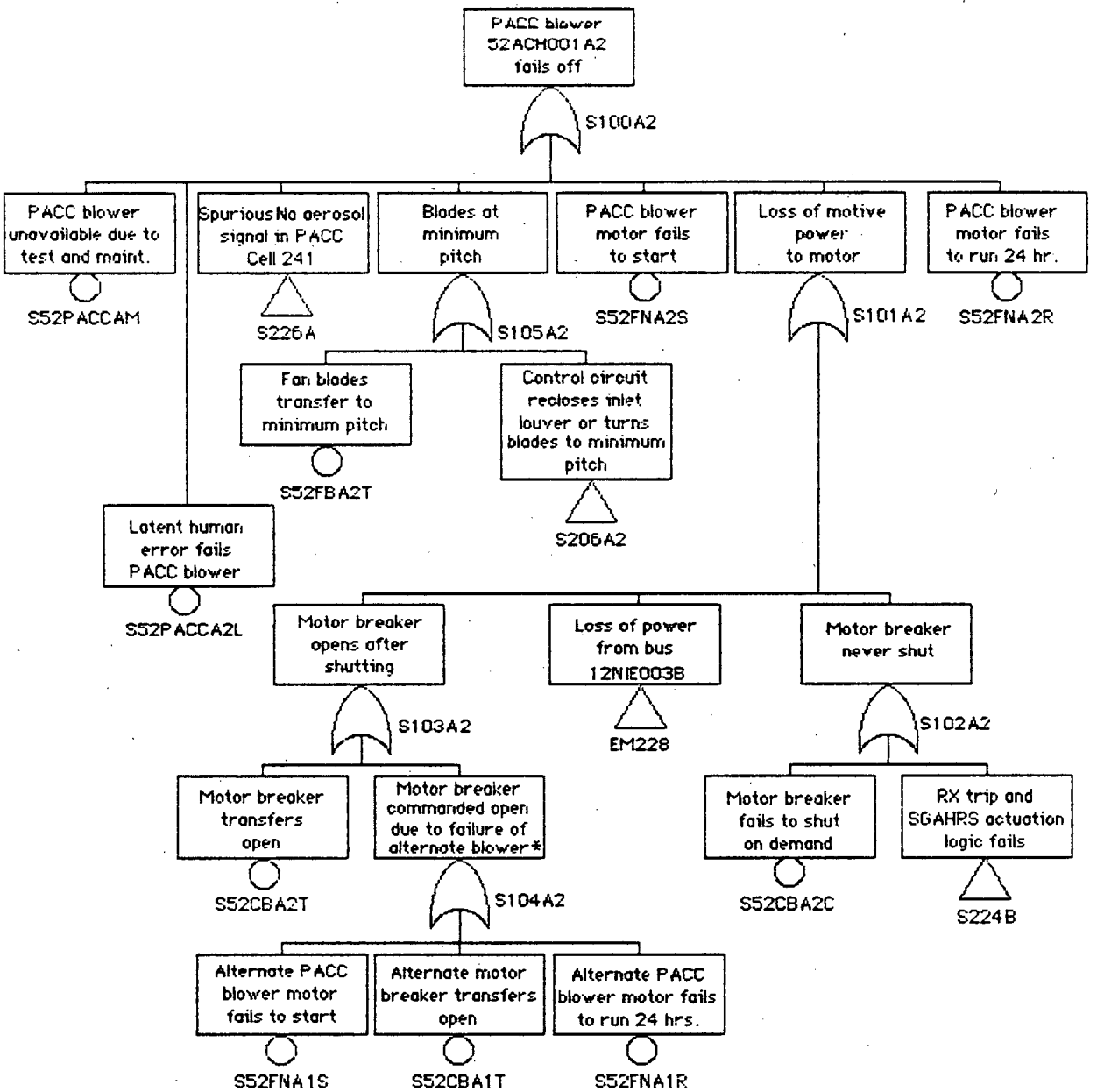
NOTE: This logic is assumed through analysis of ECP G1088 description of proposed change in noncondensible vent valve control.





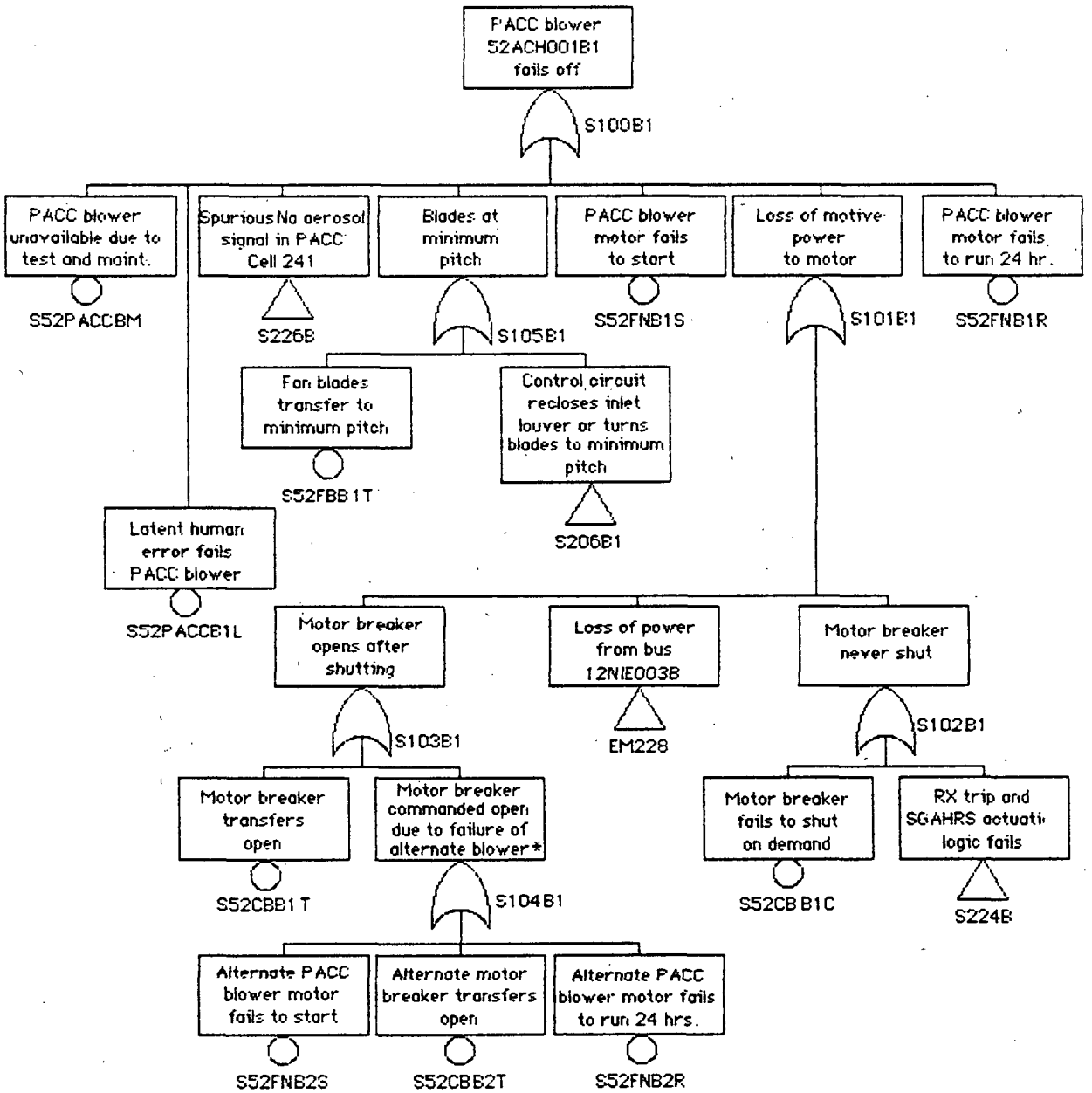
* NOTE: PACC blower motor breakers are interlocked such that if one opens, then the other is opened. Assumed that motor faults would cause overcurrent, thus opening the breaker.

Figure A5-8. PACC Fault Tree (Detailed) for Blower 52ACH001A1.



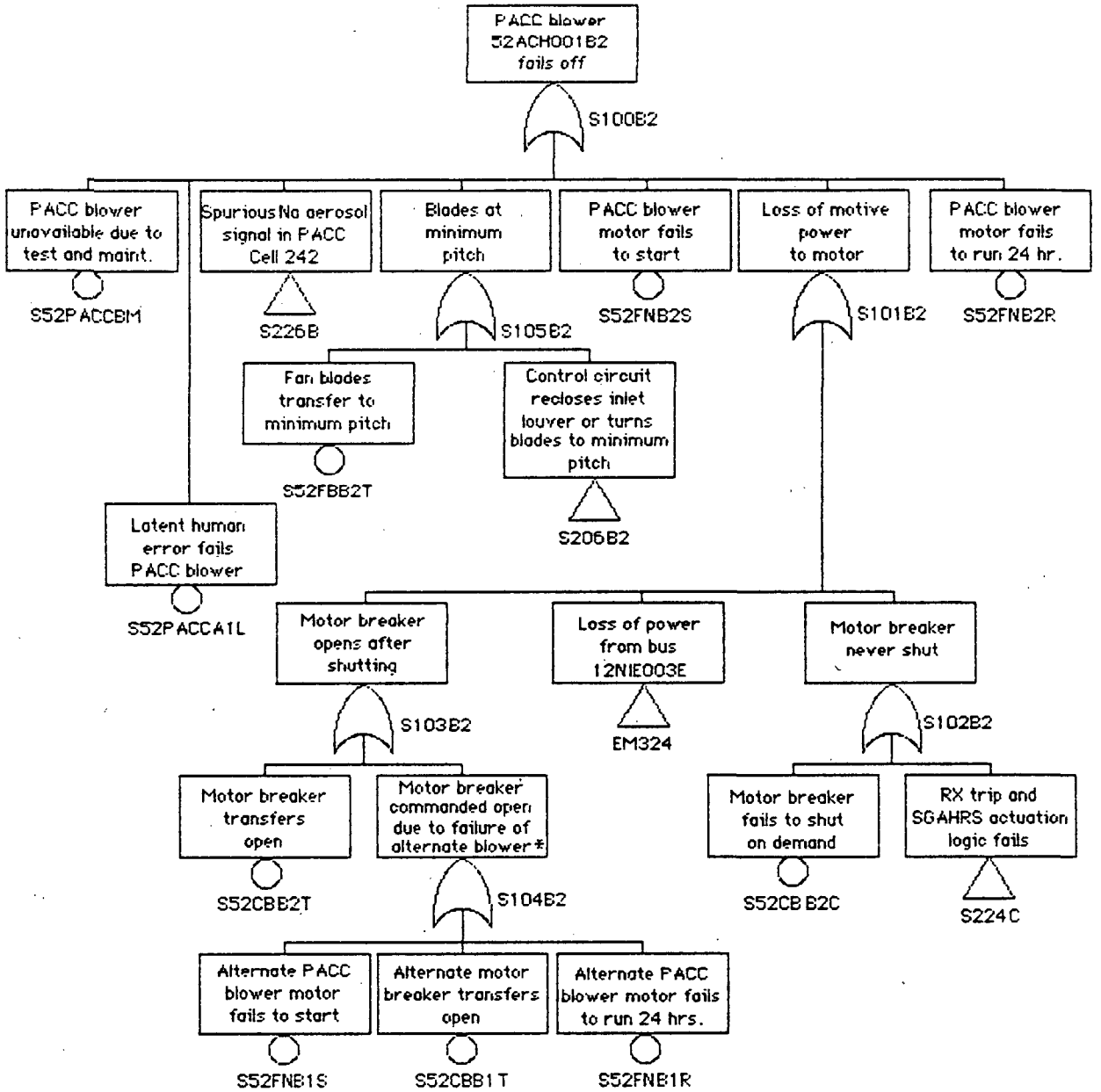
*NOTE: PACC blower motor breakers are interlocked such that if one opens, then the other is opened. Assumed that motor faults would cause overcurrent, thus opening the breaker.

Figure A5-9. PACC Fault Tree (Detailed) for Blower 52ACH001A2.



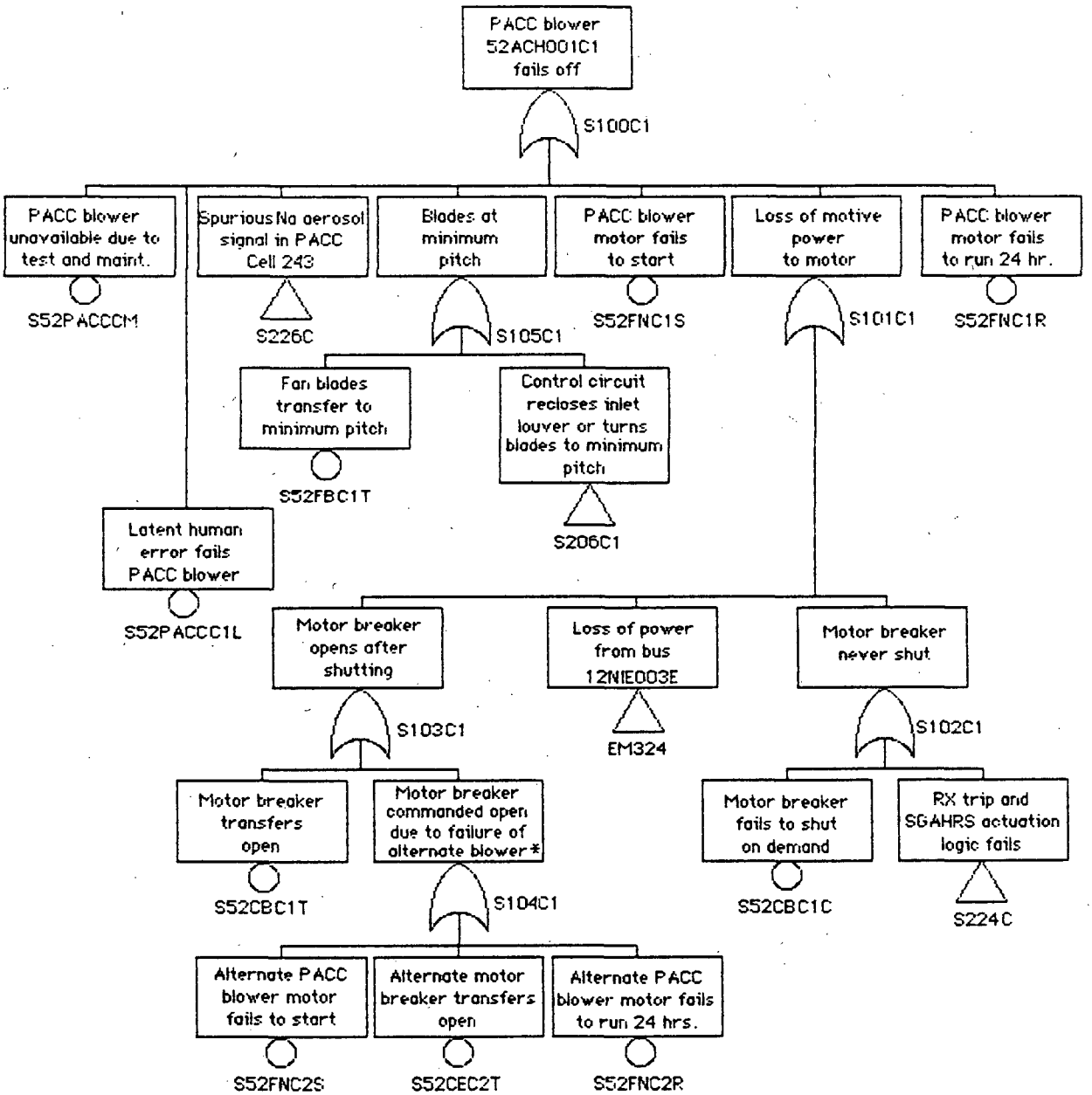
* NOTE: PACC blower motor breakers are interlocked such that if one opens, then the other is opened. Assumed that motor faults would cause overcurrent, thus opening the breaker.

Figure A5-10. PACC Fault Tree (Detailed) for Blower 52ACH001B1.



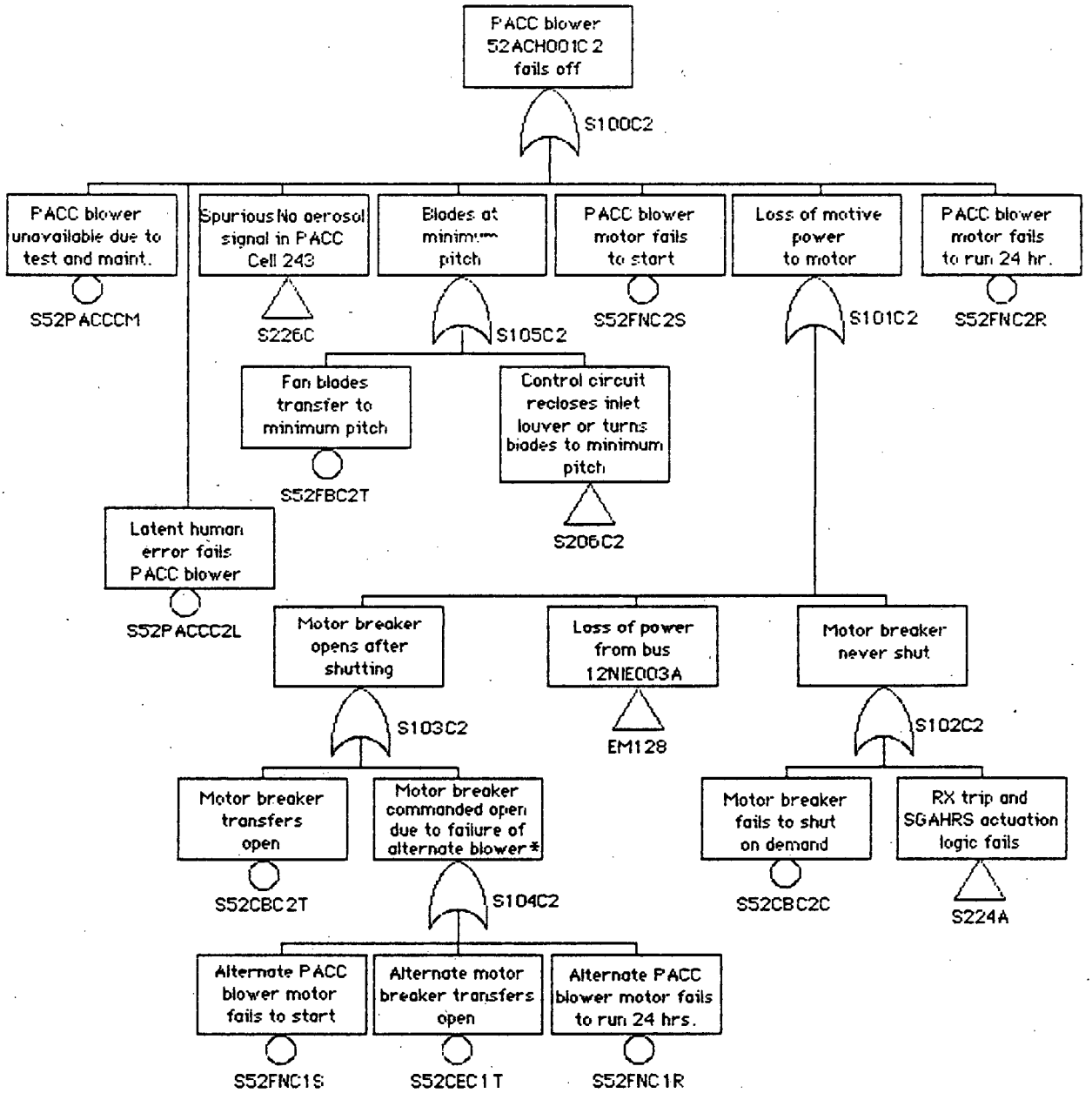
* NOTE: PACC blower motor breakers are interlocked such that if one opens, then the other is opened. Assumed that motor faults would cause overcurrent, thus opening the breaker.

Figure A5-11. PACC Fault Tree (Detailed) for Blower 52ACH001B2.



* NOTE: PACC blower motor breakers are interlocked such that if one opens, then the other is opened. Assumed that motor faults would cause overcurrent, thus opening the breaker.

Figure A5-12. PACC Fault Tree (Detailed) for Blower 52ACH001C1.



*NOTE: PACC blower motor breakers are interlocked such that if one opens, then the other is opened. Assumed that motor faults would cause overcurrent, thus opening the breaker.

Figure A5-13. PACC Fault Tree (Detailed) for Blower 52ACH001C2.

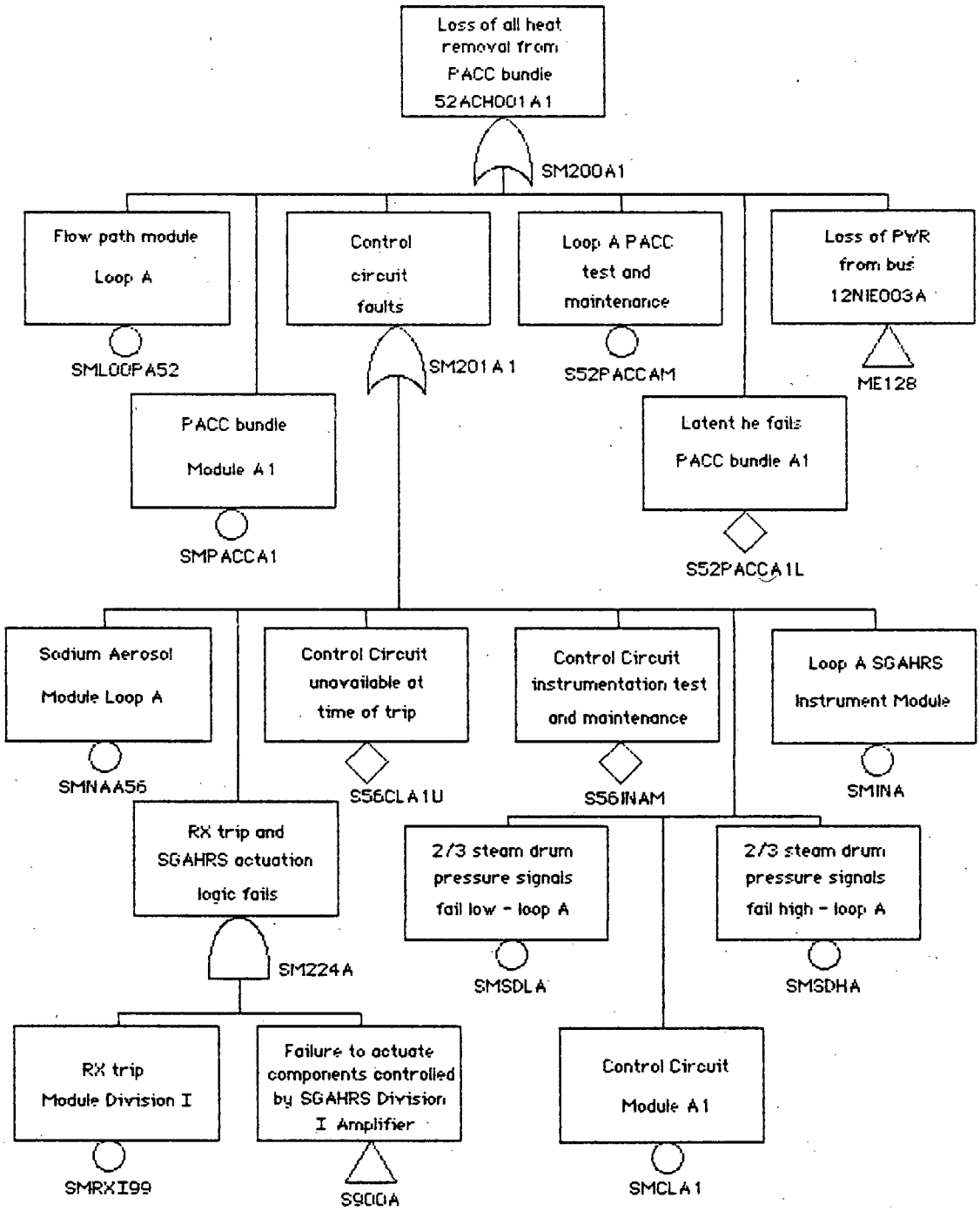


Figure A5-14. PACC Fault Tree (Modularized) for Tube Bundle 52ACH001A1.

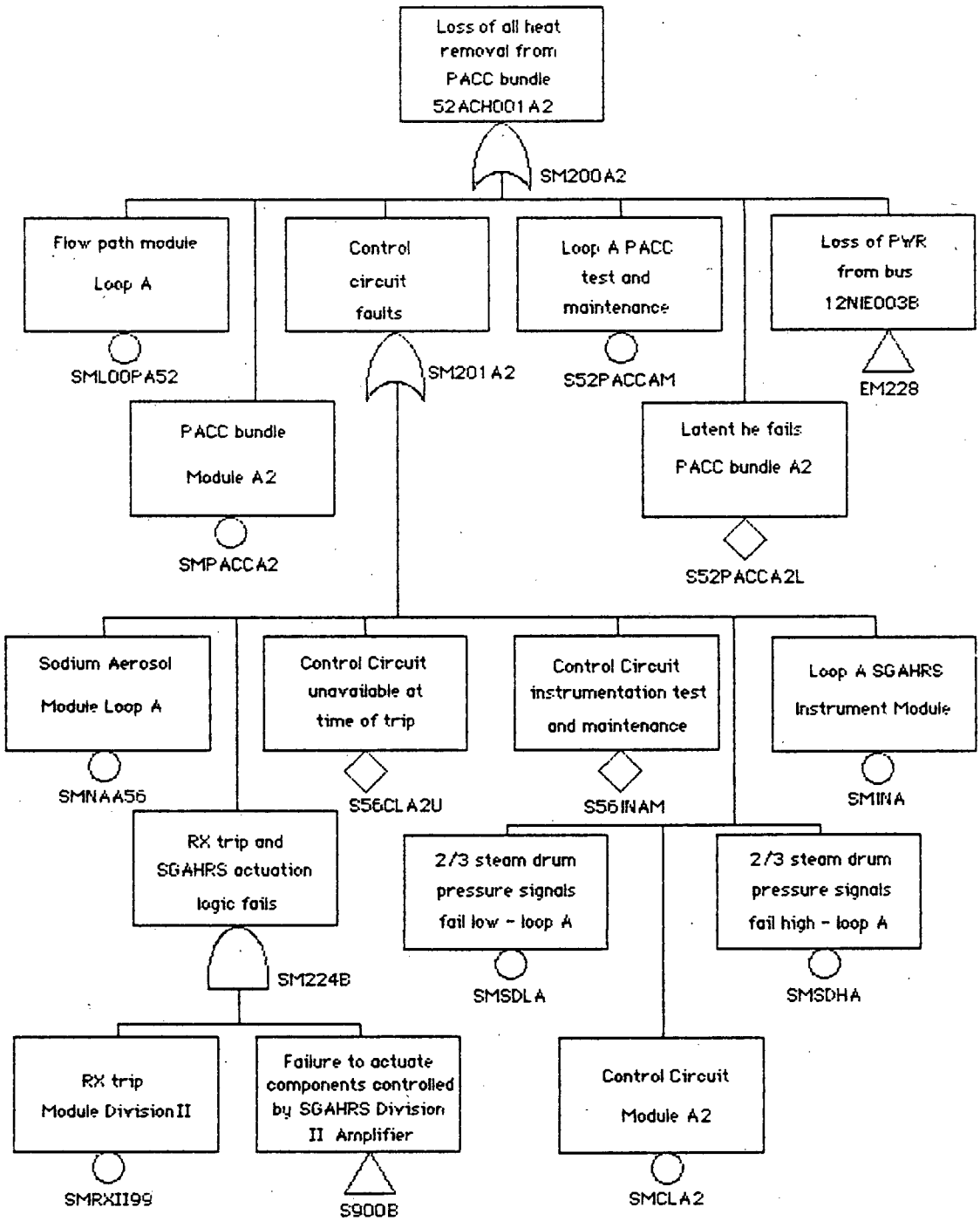


Figure A5-15. PACC Fault Tree (Modularized) for Tube Bundle 52ACH001A2.

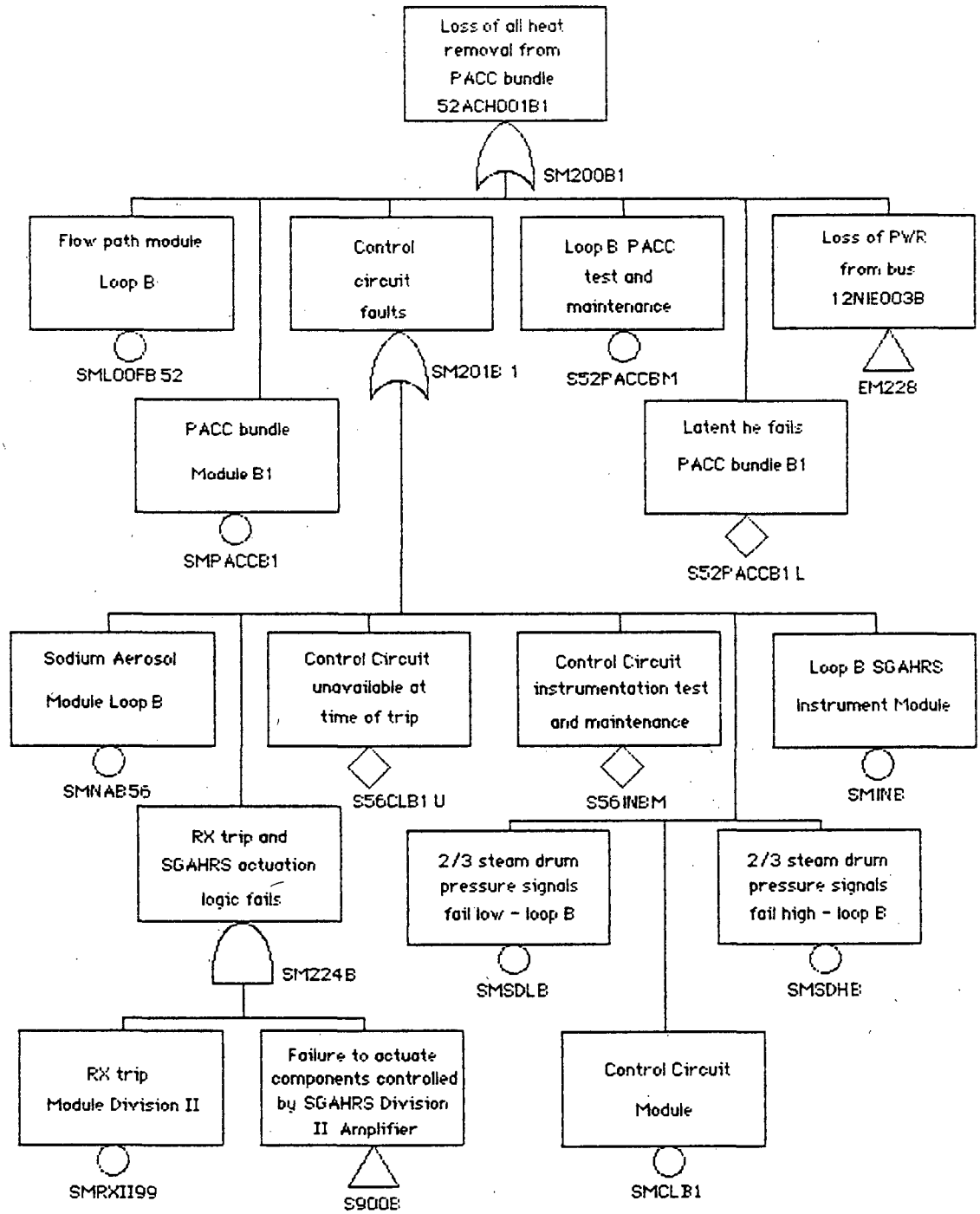


Figure A5-16. PACC Fault Tree (Modularized) for Tube Bundle 52ACH001B1.

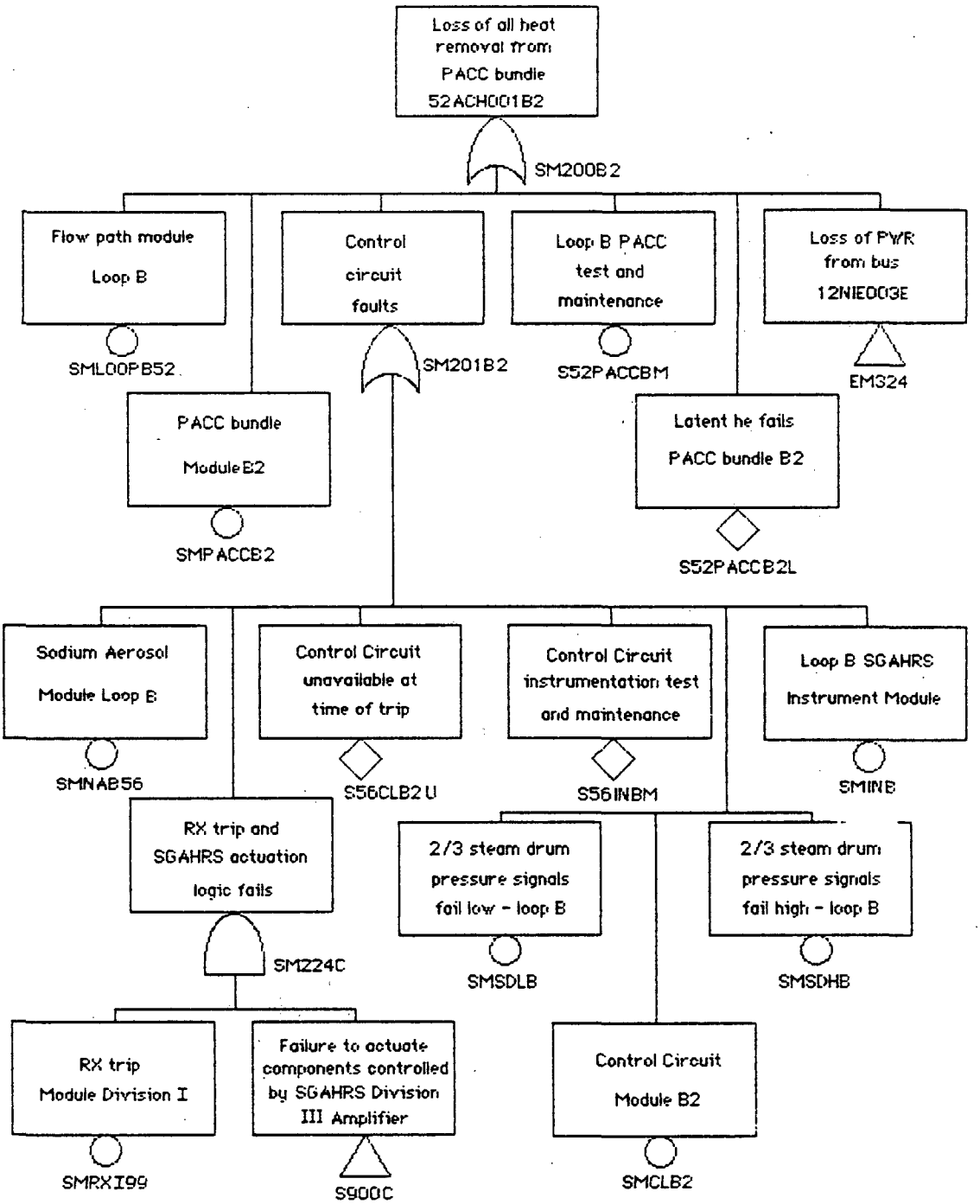


Figure A5-17. PACC Fault Tree (Modularized) for Tube Bundle 52ACH001B2.

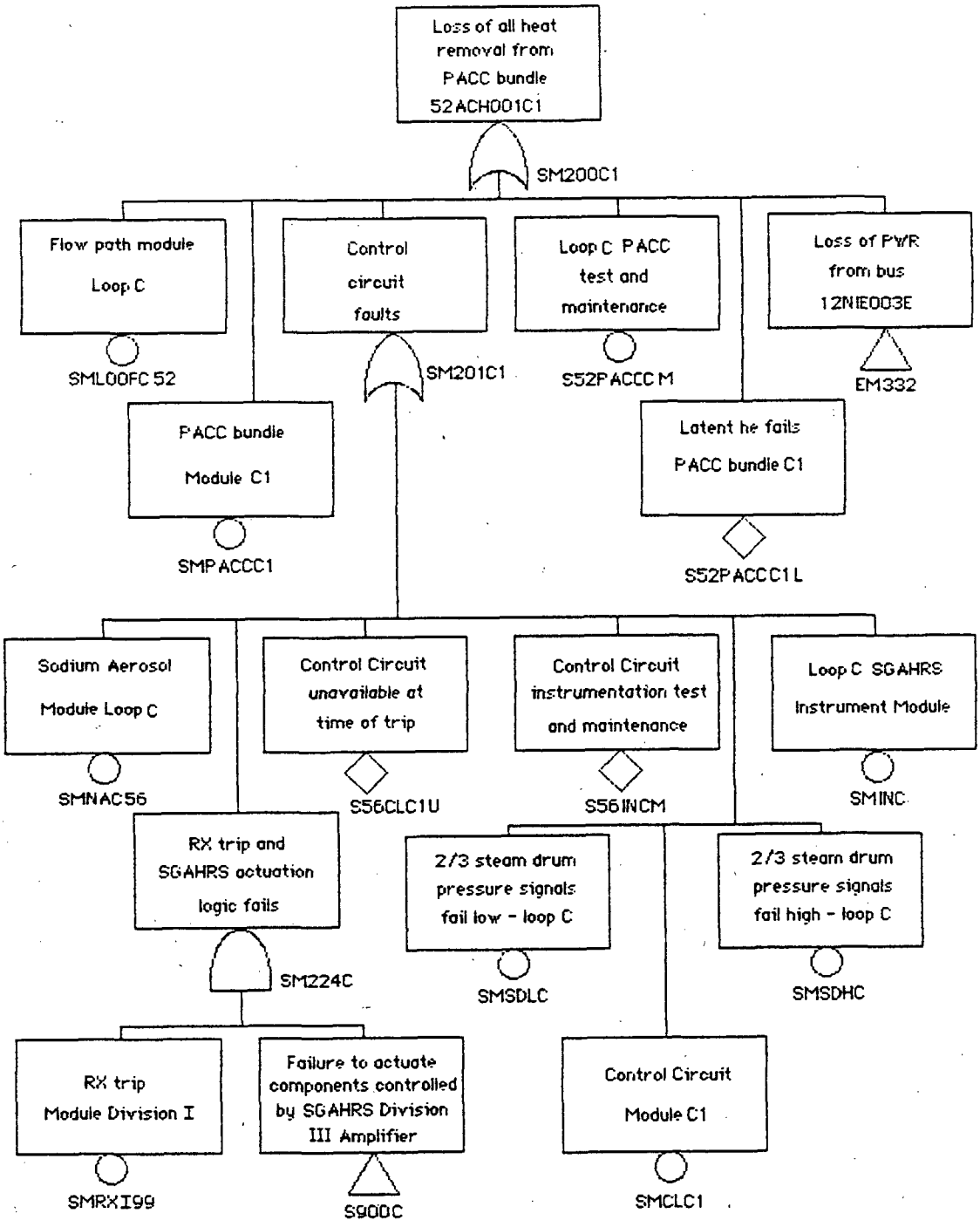


Figure A5-16. PACC Fault Tree (Modularized) for Tube Bundle 52ACH001C1.

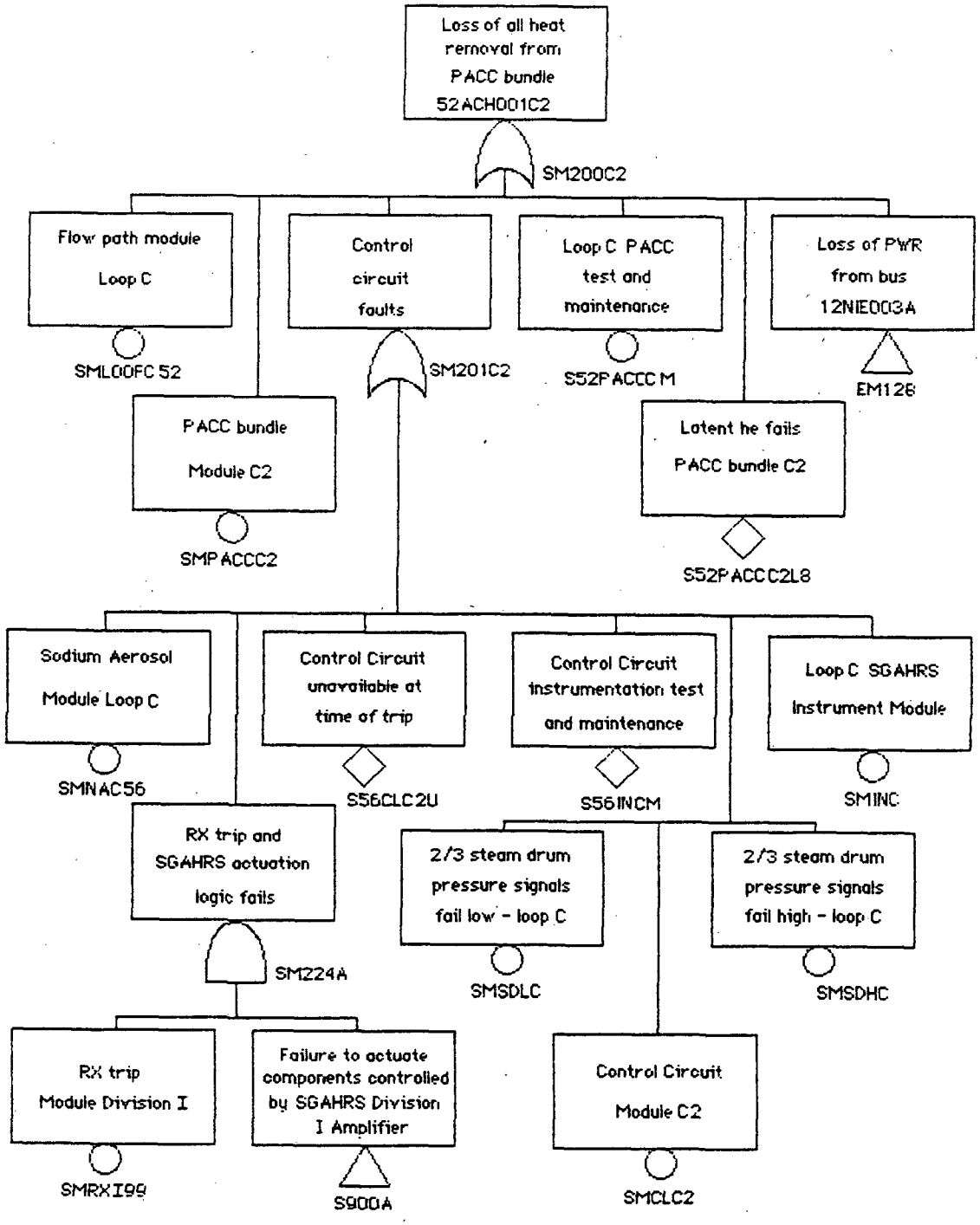


Figure A5-19. PACC Fault Tree (Modularized) for Tube Bundle 52ACH001C2.

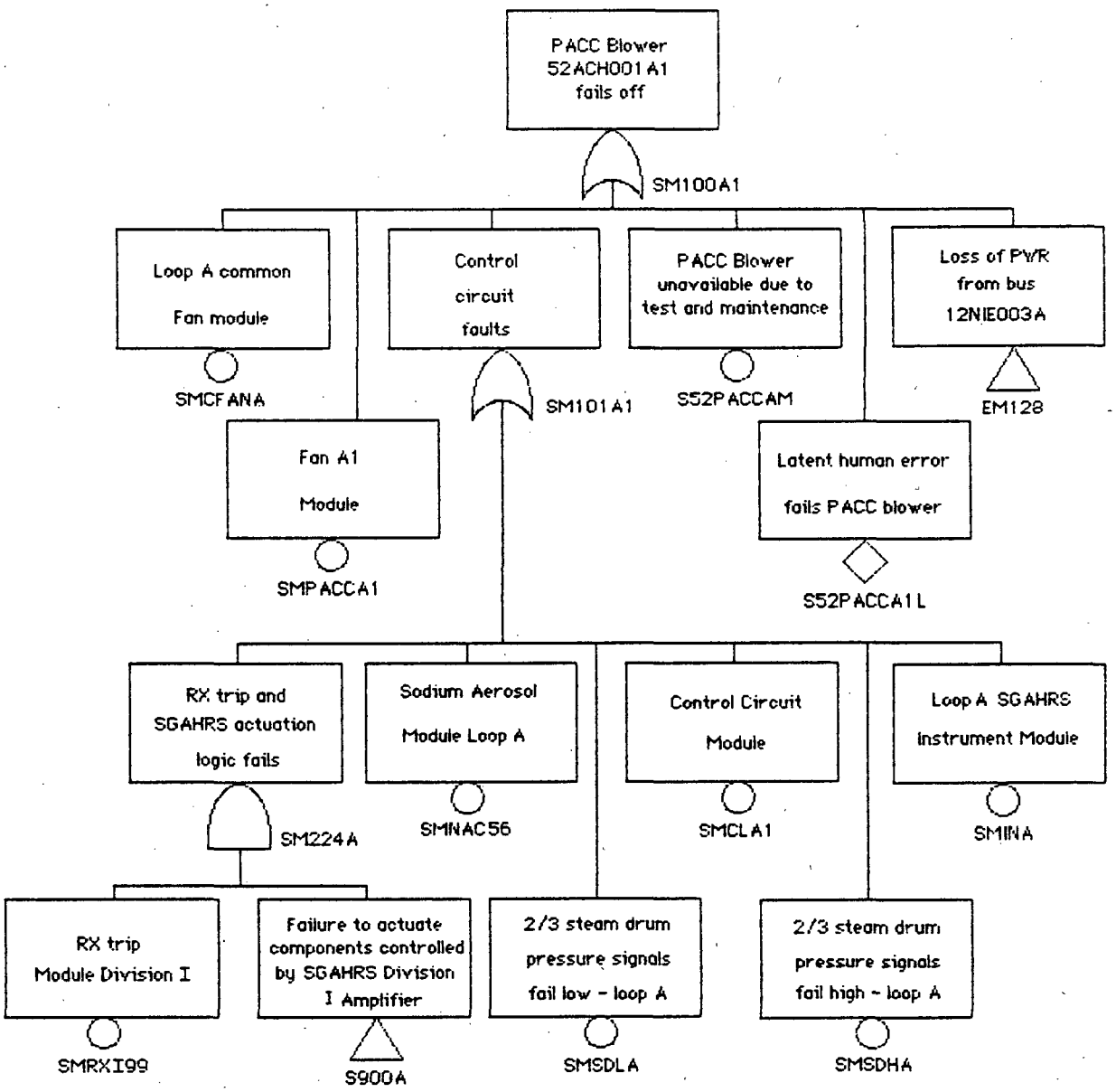


Figure A5-20. PACC Fault Tree (Modularized) for Blower 52ACH001A1.

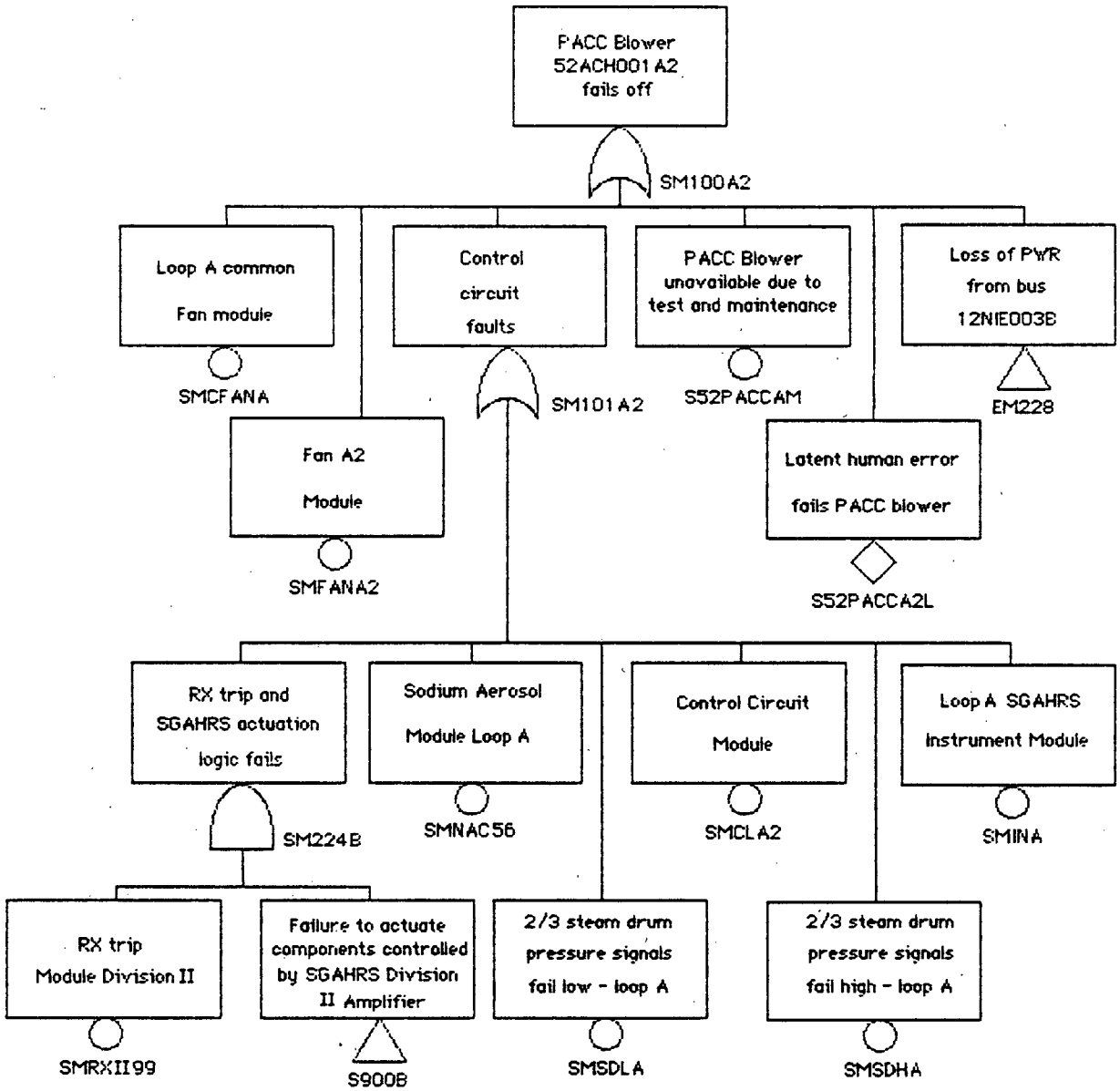


Figure A5-21. PACC Fault Tree (Modularized) for Blower 52ACH001A2.

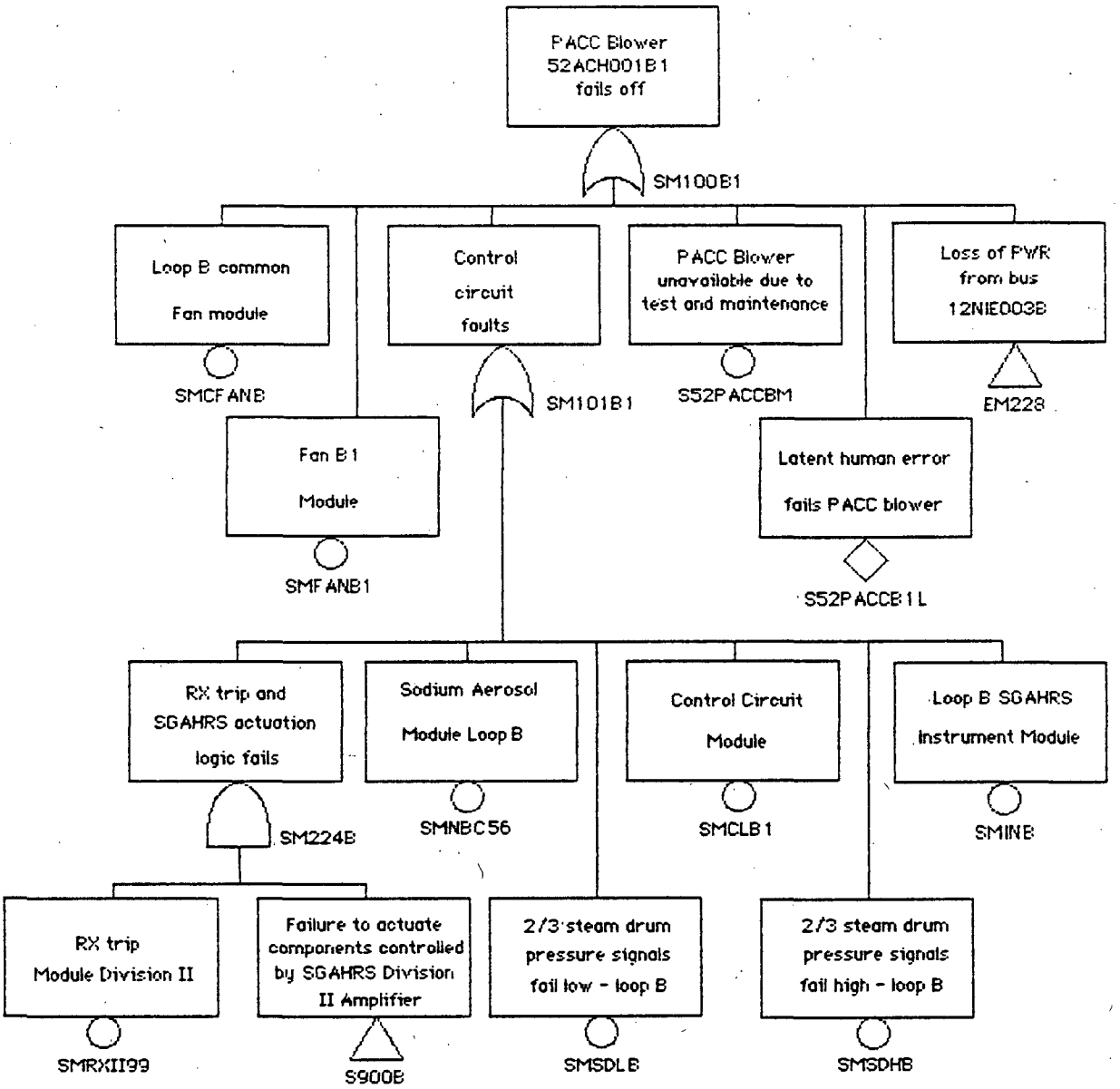


Figure A5-22. PACC Fault Tree (Modularized) for Blower 52ACH001B1.

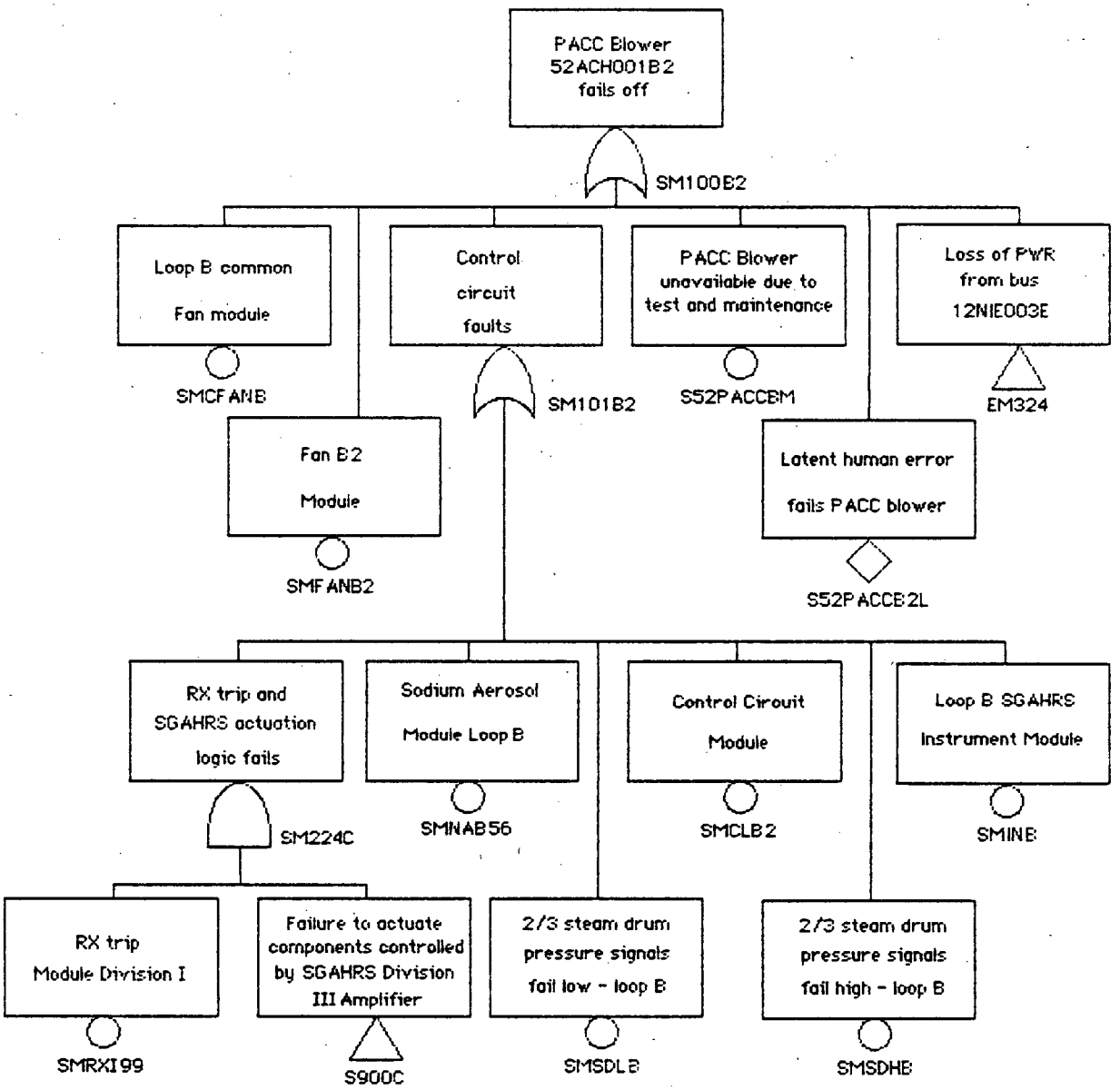


Figure A5-23. PACC Fault Tree (Modularized) for Blower 52ACH001B2.

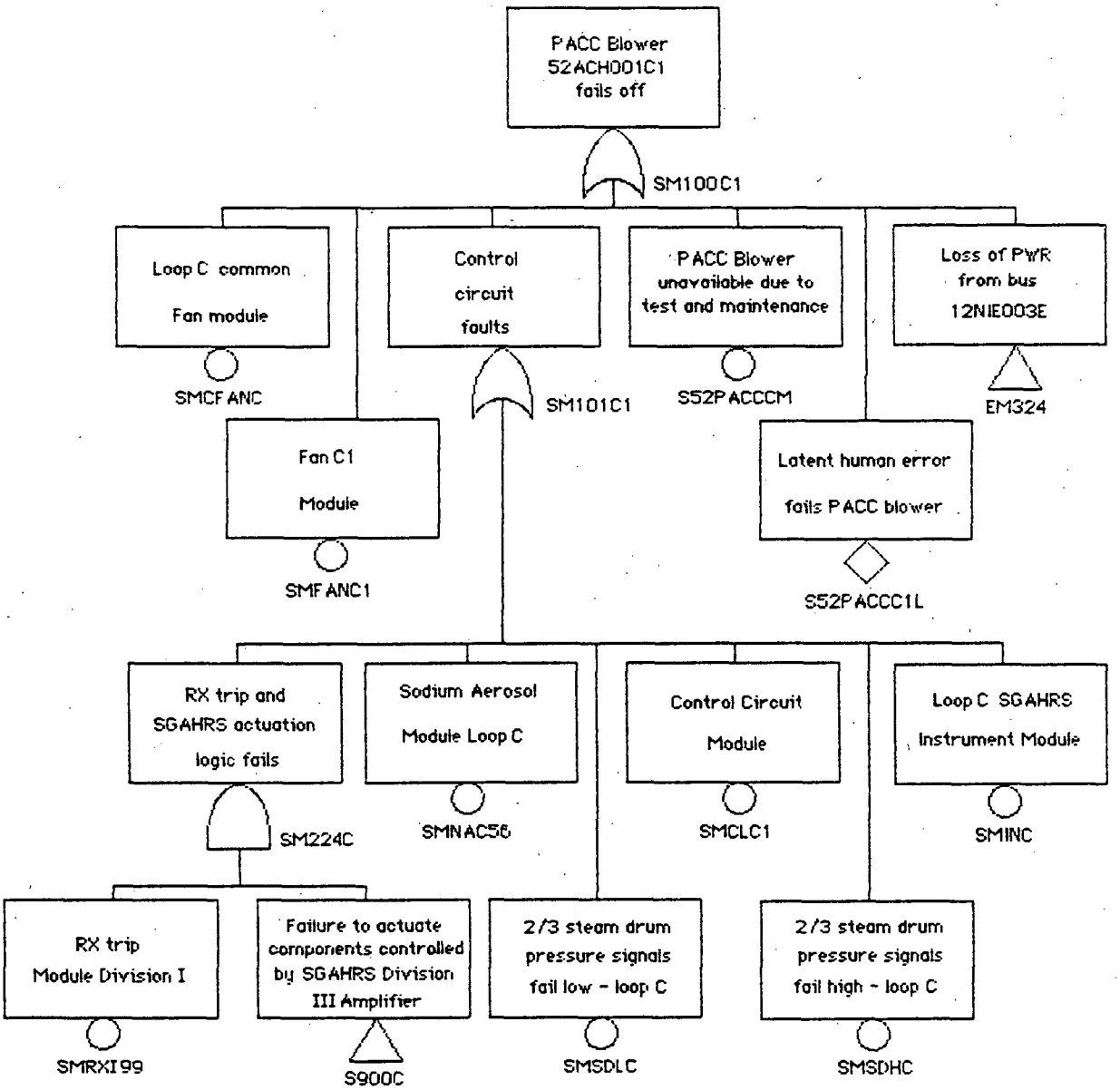


Figure A5-24. PACC Fault Tree (Modularized) for Blower 52ACH001C1.

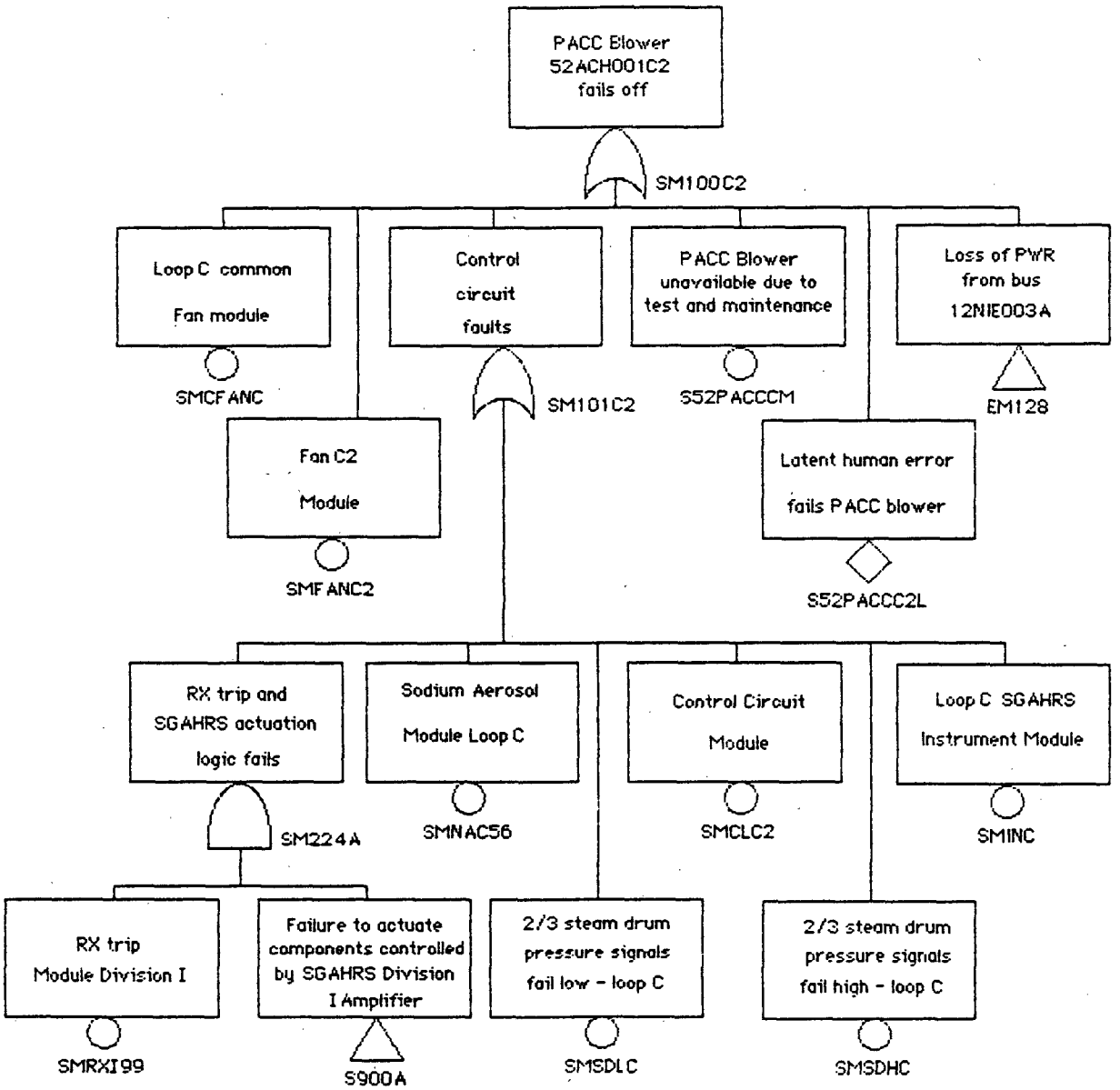


Figure A5-25. PACC Fault Tree (Modularized) for Blower 52ACH001C2.

Table A5-2

PACC FAULT TREE DATA

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMLOOPA52		1.6-5			TAB-OR
S53XV202AT	Valve 53SGV202A Transfers Shut	2.4-6	1.0-7	24	
S52XV126AT	Valve 52ACV126A Transfers Shut	2.4-6	1.0-7	24	
S53XV203AT	Valve 53SGV203A Transfers Shut	2.4-6	1.0-7	24	
S53XV128AT	Valve 52ACV128A Transfers Shut	2.4-6	1.0-7	24	
S52FE002AP	Flow Element 52ACD002A Plugged	6.7-6	2.8-7	24	
SMNAA56		8.2-6			SMNAA56 = S56CL2411I
S56CL2411I	Spurious Trip From Train 1 2/3 Logic Cell 241	4.1-6	1.7-7[a]	24	+ S56CL2412I + S56SM241AI * S56SM241BI + S56SM241AI * S56SM241CI + S56SM241BI * S56SM241CI
S56CL2412I	Spurious Trip From Train 2 2/3 Logic Cell 241	4.1-6	1.7-7[a]	24	
S56SD741AI	Spurious Trip From NA Detector CHA Cell 241	3.4-6	1.4-7[b]	24	
S56SD241BI	Spurious Trip From NA Detector CHB Cell 241	3.4-6	1.4-7	24	
S56SD241CI	Spurious Trip From NA Detector CHC Cell 241	3.4-6	1.4-7	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMLOOPB52		1.6-5			TAB-OR
S53XV202BT	Valve 53SGV202B Transfers Shut	2.4-6	1.0-7	24	
S52XV126BT	Valve 52ACV126B Transfers Shut	2.4-6	1.0-7	24	
S53XV203BT	Valve 53SGV203B Transfers Shut	2.4-6	1.0-7	24	
S53XV128BT	Valve 52ACV128B Transfers Shut	2.4-6	1.0-7	24	
S52FE002BP	Flow Element 52ACD002B Plugged	6.7-6	2.8-7	24	
SMNAB56		8.2-6			SMNAB56 = S56CL2421I + S56CL2422I + S56SM242AI * S56SM242BI + S56SM242AI * S56SM242CI + S56SM242BI * S56SM242CI
S56CL2421I	Spurious Trip From Train 1 2/3 Logic Cell 242	4.1-6	1.7-7[a]	24	
S56CL2422I	Spurious Trip From Train 2 2/3 Logic Cell 242	4.1-6	1.7-7[a]	24	
S56SD742AI	Spurious Trip From NA Detector CHA Cell 242	3.4-6	1.4-7[b]	24	
S56SD242BI	Spurious Trip From NA Detector CHB Cell 242	3.4-6	1.4-7[b]	24	
S56SD242CI	Spurious Trip From NA Detector CHC Cell 242	3.4-6	1.4-7[b]	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SML00PC52		1.6-5			TAB-OR
S53XV202CT	Valve 53SGV202B Transfers Shut	2.4-6	1.0-7	24	
S52XV126CT	Valve 52ACV126B Transfers Shut	2.4-6	1.0-7	24	
S53XV203CT	Valve 53SGV203B Transfers Shut	2.4-6	1.0-7	24	
S53XV128CT	Valve 52ACV128B Transfers Shut	2.4-6	1.0-7	24	
S52FE002CP	Flow Element 52ACD002B Plugged	6.7-6	2.8-7	24	
SMNAC56		8.2-6			SMNAC56 = S56CL2431I
S56CL2431I	Spurious Trip From Train 1 2/3 Logic Cell 243	4.1-6	1.7-7[a]	24	+ S56CL2432I + S56SM243AI * S56SM243BI + S56SM243AI * S56SM243CI + S56SM243BI * S56SM243CI
S56CL2432I	Spurious Trip From Train 2 2/3 Logic Cell 243	4.1-6	1.7-7[a]	24	
S56SD743AI	Spurious Trip From NA Detector CHA Cell 243	3.4-6	1.4-7[b]	24	
S56SD243BI	Spurious Trip From NA Detector CHB Cell 243	3.4-6	1.4-7[b]	24	
S56SD243CI	Spurious Trip From NA Detector CHC Cell 243	3.4-6	1.4-7[b]	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMRXI99		2.9-5			TAB-OR
S99CLDIE	"PACC Start on RX Trip" Logic Fails to Trip Given Input	2.4-5	2.4-5/d	--	
S12ACDIB	Local PWR to "PACC Start on RX Trip" Logic Transfers Off	5.0-6	1.0-6	5c	
SMRXII99		2.9-5			
S99CLDIIE	"PACC Start on RX Trip" Logic Fails to Trip Given Input	2.4-5	2.4-5/d	--	
S12ACDIIB	Local PWR to "PACC Start on RX Trip" Logic Transfers Off	5.0-6	1.0-6	5c	
SMPACCA1		3.1-3			SMPACCA1 = (S56TCHA1AA * S56TCA1BA * S56TCA1CA) + TAB-OR of remaining components
S52DMIA10	Inlet Louver 52ACH001A1 Fails to Open on Demand	1.0-3	1.0-3/d	--	
S52DMIA1T	Inlet Louver 52ACH001A1 Transfers Shut	2.4-6	1.0-7	24	
S52DMOA10	Outlet Louver 52SCH001A1 Fails to Open on Demand	1.0-3	1.0-3/d	--	
S52DMOA1T	Outlet Louver 52ACH001A1 Transfers Shut	2.4-6	1.0-7	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMPACCA1 (Cont.)					
S56LSILA1T	Inlet Louver Position Switch Transfers Shut	1.0-4	4.2-6	24	
S56CPTA1E	Temperature Differential Bistable Fails to Operate	4.1-6	1.7-7[a]	24	
S56TCA1AA	Non-Condensable Thermocouple Fails High CH A	2.4-5	1.0-6	24	
S56TCA1BA	Non-Condensable Thermocouple Fails High CH B	2.4-5	1.0-6	24	
S56TCA1CA	Non-Condensable Thermocouple Fails High CH C	2.4-5	1.0-6	24	
S52XV133AT	Valve 52ACV133A Transfers Shut	2.4-6	1.0-7	24	
S52MV129AO	Valve 52ACV129A Fails to Open on Demand	1.0-3	1.0-3/d	--	
S52ORO12AP	Orifice 52AFDO12A Plugged	6.7-6	2.8-7	24	
SMPACCA2					
S52DMIA2O	Inlet Louver 52ACHOO1A2 Fails to Open on Demand	3.1-3 / 1.0-3	1.0-3/d	--	SMPACCA2 = (S56TCHA2AA * S56TCA2BA * S56TCA2CA) + TAB-OR of remaining components

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMPACCA2 (Cont.)					
S52DMIA2T	Inlet Louver 52ACH001A2 Transfers Shut	2.4-6	1.0-7	24	
S52DMOA2O	Inlet Louver 52SCH001A2 Fails to Open on Demand	1.0-3	1.0-3/d	--	
S52DMOA2T	Outlet Louver 52ACH001A2 Transfers Shut	2.4-6	1.0-7	24	
S56LSILA2T	Inlet Louver Position Switch Transfers Shut	1.0-4	4.2-6	24	
S56CPTA2E	Temperature Differential Bistable Fails to Operate	4.1-6	1.7-7[a]	24	
S56TCA2AA	Non-Condensable Thermocouple Fails High CH A	2.4-5	1.0-6	24	
S56TCA2BA	Non-Condensable Thermocouple Fails High CH B	2.4-5	1.0-6	24	
S56TCA2CA	Non-Condensable Thermocouple Fails High CH C	2.4-5	1.0-6	24	
S52XV133BT	Valve 52ACV133B Transfers Shut	2.4-6	1.0-7	24	
S52MV129BO	Valve 52ACV129B Fails to Open on Demand	1.0-3	1.0-3/d	--	
S52OR012BP	Orifice 52AFD012B Plugged	6.7-6	2.8-7	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMPACCB1		3.1-3			SMPACCB1 = (S56TCHB1AA * S56TCB1BA * S56TCB1CA) + TAB-OR of remaining components
S52DMIB10	Inlet Louver 52ACH001B1 Fails to Open on Demand	1.0-3	1.0-3/d	--	
S52DMIB1T	Inlet Louver 52ACH001B1 Transfers Shut	2.4-6	1.0-7	24	
S52DMOB10	Inlet Louver 52SCH001B1 Fails to Open on Demand	1.0-3	1.0-3/d	--	
S52DMOB1T	Outlet Louver 52ACH001B1 Transfers Shut	2.4-6	1.0-7	24	
S56LSILB1T	Inlet Louver Position Switch Transfers Shut	1.0-4	4.2-6	24	
S56CPTB1E	Temperature Differential Bistable Fails to Operate	4.1-6	1.7-7[a]	24	
S56TCB1AA	Non-Condensable Thermocouple Fails High CH A	2.4-5	1.0-6	24	
S56TCB1BA	Non-Condensable Thermocouple Fails High CH B	2.4-5	1.0-6	24	
S56TCB1CA	Non-Condensable Thermocouple Fails High CH C	2.4-5	1.0-6	24	
S52XV133CT	Valve 52ACV133C Transfers Shut	2.4-6	1.0-7	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMPACCB1 (Cont.)					
S52MV129CO	Valve 52ACV129C Fails to Open on Demand	1.0-3	1.0-3/d	--	
S52OR012CP	Orifice 52AFD012C Plugged	6.7-6	2.8-7	24	
SMPACCB2		3.1-3			SMPACCB2 = (S56TCHB2AA * S56TCB2BA * S56TCB2CA) + TAB-OR of remaining components
S52DMIB2O	Inlet Louver 52ACH001B2 Fails to Open on Demand	1.0-3	1.0-3/d	--	
S52DMIB2T	Inlet Louver 52ACH001B2 Transfers Shut	2.4-6	1.0-7	24	
S52DMOB2O	Inlet Louver 52SCH001B2 Fails to Open on Demand	1.0-3	1.0-3/d	--	
S52DMOB2T	Outlet Louver 52ACH001B2 Transfers Shut	2.4-6	1.0-7	24	
S56LSILB2T	Inlet Louver Position Switch Transfers Shut	1.0-4	4.2-6	24	
S56CPTB2E	Temperature Differential Bistable Fails to Operate	4.1-6	1.7-7[a]	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMPACCB2 (Cont.)					
S56TCB2AA	Non-Condensable Thermocouple Fails High CH A	2.4-5	1.0-6	24	
S56TCB2BA	Non-Condensable Thermocouple Fails High CH B	2.4-5	1.0-6	24	
S56TCB2CA	Non-Condensable Thermocouple Fails High CH C	2.4-5	1.0-6	24	
S52XV133DT	Valve 52ACV133D Transfers Shut	2.4-6	1.0-7	24	
S52MV129DO	Valve 52ACV129D Fails to Open on Demand	1.0-3	1.0-3/d	--	
S52ORO12DP	Orifice 52AFD012D Plugged	6.7-6	2.8-7	24	
SMPACCC1		3.1-3			SMPACCC1 = (S56TCHC1AA * S56TCC1BA * S56TCC1CA) + TAB-OR of remaining components
S52DMIC1O	Inlet Louver 52ACH001C1 Fails to Open on Demand	1.0-3	1.0-3/d	--	
S52DMIC1T	Inlet Louver 52ACH001C1 Transfers Shut	2.4-6	1.0-7	24	
S52DMOC1O	Inlet Louver 52SCH001C1 Fails to Open on Demand	1.0-3	1.0-3/d	--	
S52DMOC1T	Outlet Louver 52ACH001C1 Transfers Shut	2.4-6	1.0-7	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMPACCC1 (Cont.)					
S56LSILC1T	Inlet Louver Position Switch Transfers Shut	1.0-4	4.2-6	24	
S56CPTC1E	Temperature Differential Bistable Fails to Operate	4.1-6	1.7-7[a]	24	
S56TCC1AA	Non-Condensable Thermocouple Fails High CH A	2.4-5	1.0-6	24	
S56TCC1BA	Non-Condensable Thermocouple Fails High CH B	2.4-5	1.0-6	24	
S56TCC1CA	Non-Condensable Thermocouple Fails High CH C	2.4-5	1.0-6	24	
S52XV133ET	Valve 52ACV133E Transfers Shut	2.4-6	1.0-7	24	
S52MV129EO	Valve 52ACV129E Fails to Open on Demand	1.0-3	1.0-3/d	--	
S52ORO12EP	Orifice 52AFD012E Plugged	6.7-6	2.8-7	24	
SMPACCC2					
S52DMIC2O	Inlet Louver 52ACH001C2 Fails to Open on Demand	3.1-3 / 1.0-3	1.0-3/d	--	SMPACCC2 = (S56TCHC2AA * S56TCC2BA * S56TCC2CA) + TAB-OR of remaining components

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMPACCC2 (Cont.)					
S52DMIC2T	Inlet Louver 52ACH001C2 Transfers Shut	2.4-6	1.0-7	24	
S52DMOC20	Inlet Louver 52SCH001C2 Fails to Open on Demand	1.0-3	1.0-3/d	--	
S52DMOC2T	Outlet Louver 52ACH001C2 Transfers Shut	2.4-6	1.0-7	24	
S56LSILC2T	Inlet Louver Position Switch Transfers Shut	1.0-4	4.2-6	24	
S56CPTC2E	Temperature Differential Bistable Fails to Operate	4.1-6	1.7-7[a]	24	
S56TCC2AA	Non-Condensable Thermocouple Fails High CH A	2.4-5	1.0-6	24	
S56TCC2BA	Non-Condensable Thermocouple Fails High CH B	2.4-5	1.0-6	24	
S56TCC2CA	Non-Condensable Thermocouple Fails High CH C	2.4-5	1.0-6	24	
S52XV133FT	Valve 52ACV133F Transfers Shut	2.4-6	1.0-7	24	
S52MV129FO	Valve 52ACV129F Fails to Open on Demand	1.0-3	1.0-3/d	--	
S52ORO12FP	Orifice 52AFD012F Plugged	6.7-6	2.8-7	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMINA		7.2-5			TAB-OR
S56FEAA	Return Flow Detector Signal Fails High Loop A	2.4-5	1.0-6	24	
S56TEAB	Return Temp Detector Fails Low Loop A	2.4-5	1.0-6	24	
S12ACAB	Local PWR to Return Temp Detector Fails Off Loop A	2.4-5	1.0-6	24	
SMSDPHA		6.9-9			SMSDPHA = [(S56PSAAA + S56BUAAA) * (S56PSABA + S56BUABA)] + [(S56PSAAA + S56BUAAA) * (S56PSACA + S56BUACA)] + [(S56PSABA + S56BUABA) * (S56PSACA + S56BUACA)]
S56PSAAA	S/D PRES Detector Fails High Loop A CH A	2.4-5	1.0-6	24	
S56BUAAA	S/D PRES Buffer Fails High Loop A CH A	2.4-5	1.0-6	24	
S56PSABA	S/D PRES Detector Fails High Loop A CH B	2.4-5	1.0-6	24	
S56BUABA	S/D PRES Buffer Fails High Loop A CH B	2.4-5	1.0-6	24	
S56PSACA	S/D PRES Detector Fails High Loop A CH C	2.4-5	1.0-6	24	
S56BUACA	S/D PRES Buffer Fails High Loop A CH C	2.4-5	1.0-6	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMINB		7.2-5			TAB-OR
S56FEBA	Return Flow Detector Signal Fails High Loop B	2.4-5	1.0-6	24	
S56TEBB	Return Temp Detector Fails Low Loop B	2.4-5	1.0-6	24	
S12ACBB	Local PWR to Return Temp Detector Fails Off Loop B	2.4-5	1.0-6	24	
SMSDPHB		6.9-9			SMSDPHB = [(S56PSBAA + S56BUBAA) * (S56PSBBA + S56BUBBA)] + [(S56PSAAA + S56BUBAA) * (S56PSBCA + S56BUBCA)] + [(S56PSBBA + S56BUBBA) * (S56PSBCA + S56BUBCA)]
S56PSBAA	S/D PRES Detector Fails High Loop B CH A	2.4-5	1.0-6	24	
S56BUBAA	S/D PRES Buffer Fails High Loop B CH A	2.4-5	1.0-6	24	
S56PSBBA	S/D PRES Detector Fails High Loop B CH B	2.4-5	1.0-6	24	
S56BUBBA	S/D PRES Buffer Fails High Loop B CH B	2.4-5	1.0-6	24	
S56PSBCA	S/D PRES Detector Fails High Loop B CH C	2.4-5	1.0-6	24	
S56BUBCA	S/D PRES Buffer Fails High Loop B CH C	2.4-5	1.0-6	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMINC		7.2-5			TAB-OR
S56FECA	Return Flow Detector Signal Fails High Loop C	2.4-5	1.0-6	24	
S56TECB	Return Temp Detector Fails Low Loop C	2.4-5	1.0-6	24	
S12ACCB	Local PWR to Return Temp Detector Fails Off Loop C	2.4-5	1.0-6	24	
SMSDPHC		6.9-9			SMSDPHC = [(S56PSCAA + S56BUCAA) * (S56PSCBA + S56BUCBA)] + [(S56PSCAA + S56BUCAA) * (S56PSCCA + S56BUCCA)] + [(S56PSCBA + S56BUCBA) * (S56PSCCA + S56BUCCA)]
S56PSCAA	S/D PRES Detector Fails High Loop C CH A	2.4-5	1.0-6	24	
S56BUCAA	S/D PRES Buffer Fails High Loop C CH A	2.4-5	1.0-6	24	
S56PSCBA	S/D PRES Detector Fails High Loop C CH B	2.4-5	1.0-6	24	
S56BUCBA	S/D PRES Buffer Fails High Loop C CH B	2.4-5	1.0-6	24	
S56PSCCA	S/D PRES Detector Fails High Loop C CH C	2.4-5	1.0-6	24	
S56BUCCA	S/D PRES Buffer Fails High Loop C CH C	2.4-5	1.0-6	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMSDPLA		1.6-8			SMSDPLA =
S56PSAAB	S/D PRES Detector Fails Low Loop A CH A	2.4-5	1.0-6	24	[(S56PSAAB + S56BUAAB + S12ACAAB) * (S56PSABB + S56BUABB + S12ACABB)] + [(S56PSAAB + S56BUAAB + S12ACAAB) * (S56PSACB + S56BUACB + S12ACACB)] + [(S56PSABB + S56BUABB + S12ACABB) * (S56PSACB + S56BUACB + S12ACACB)]
S56BUAAB	S/D PRES Buffer Fails Low Loop A CH A	2.4-5	1.0-6	24	
S12ACAAB	Local PWR to S/D PRES Fails Off Loop A CH A	2.4-5	1.0-6	24	
S56PSABB	S/D PRES Detector Fails Low Loop A CH B	2.4-5	1.0-6	24	
S56BUABB	S/D PRES Buffer Fails Low Loop A CH B	2.4-5	1.0-6	24	
S12ACABB	Local PWR to S/D PRES Fails Off Loop A CH A	2.4-5	1.0-6	24	
S56PSACB	S/D PRES Detector Fails Low Loop A CH C	2.4-5	1.0-6	24	
S56BUACB	S/D PRES Buffer Fails Low Loop A CH C	2.4-5	1.0-6	24	
S12ACACB	Local PWR to S/D PRES Fails Off Loop A CH A	2.4-5	1.0-6	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMSDPLB		1.6-8			SMSDPLB =
S56PSBAB	S/D PRES Detector Fails Low Loop B CH A	2.4-5	1.0-6	24	[(S56PSBAB + S56BUBAB + S12ACBAB) * (S56PSBBB + S56BUBBB + S12ACBBB)] + [(S56PSBAB + S56BUBAB + S12ACBAB) * (S56PSBCB + S56BUBCB + S12ACBCB)] + [(S56PSBBB + S56BUBBB + S12ACBBB) * (S56PSBCB + S56BUBCB + S12ACBCB)]
S56BUBAB	S/D PRES Buffer Fails Low Loop B CH A	2.4-5	1.0-6	24	
S12ACBAB	Local PWR to S/D PRES Fails Off Loop B CH A	2.4-5	1.0-6	24	
S56PSBBB	S/D PRES Detector Fails Low Loop B CH B	2.4-5	1.0-6	24	
S56BUBBB	S/D PRES Buffer Fails Low Loop B CH B	2.4-5	1.0-6	24	
S12ACBBB	Local PWR to S/D PRES Fails Off Loop B CH A	2.4-5	1.0-6	24	
S56PSBCB	S/D PRES Detector Fails Low Loop B CH C	2.4-5	1.0-6	24	
S56BUBCB	S/D PRES Buffer Fails Low Loop B CH C	2.4-5	1.0-6	24	
S12ACBCB	Local PWR to S/D PRES Fails Off Loop B CH A	2.4-5	1.0-6	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMSDPLC		1.6-8			SMSDPLC =
S56PSCAB	S/D PRES Detector Fails Low Loop C CH A	2.4-5	1.0-6	24	[(S56PSCAB + S56BUCAB + S12ACCAB) * (S56PSCBB + S56BUCBB + S12ACCBB)] + [(S56PSCAB + S56BUCAB + S12ACCAB) * (S56PSCCB + S56BUCCB + S12ACCCB)] + [(S56PSCBB + S56BUCBB + S12ACCBB) * [(S56PSCCB + S56BUCCB + S12ACCCB)]
S56BUCAB	S/D PRES Buffer Fails Low Loop C CH A	2.4-5	1.0-6	24	
S12ACCAB	Local PWR to S/D PRES Fails Off Loop C CH A	2.4-5	1.0-6	24	
S56PSCBB	S/D PRES Detector Fails Low Loop C CH B	2.4-5	1.0-6	24	
S56BUCBB	S/D PRES Buffer Fails Low Loop C CH B	2.4-5	1.0-6	24	
S12ACCBB	Local PWR to S/D PRES Fails Off Loop C CH A	2.4-5	1.0-6	24	
S56PSCCB	S/D PRES Detector Fails Low Loop C CH C	2.4-5	1.0-6	24	
S56BUCCB	S/D PRES Buffer Fails Low Loop C CH C	2.4-5	1.0-6	24	
S12ACCCB	Local PWR to S/D PRES Fails Off Loop C CH A	2.4-5	1.0-6	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMCLA1		5.3-5			TAB-OR
S56CPHA1E	Actual to Desired Heat Comparator Fails	4.1-6	1.7-7[a]	24	
S56CMFBA1E	Fan Blade and Louver Position Control Module Fails	4.1-6	1.7-7[a]	24	
S56TZA1A	Inlet Louver Position Indicator Fails High	2.4-5	1.0-6	24	
S56CURHA1B	Required Heat Change Calculator Fails Low	4.1-6	1.7-7[a]	24	
S56CPPA1B	Actual to Desired Steam Drum Pressure Comparator Fails Low	4.1-6	1.7-7[a]	24	
S56CUHRA1A	Heat Rejection Calculator Fails High	4.1-6	1.7-7[a]	24	
S56PMESA1B	S/D PRES Median Selector Fails Low	4.1-6	1.7-7[a]	24	
S56PMESA1A	S/D PRES Median Selector Fails High	4.1-6	1.7-7[a]	24	
SMCLA2		5.3-5			TAB-OR
S56CPHA2E	Actual to Desired Heat Comparator Fails	4.1-6	1.7-7[a]	24	
S56CMFBA2E	Fan Blade and Louver Position Control Module Fails	4.1-6	1.7-7[a]	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMCLA2 (Cont.)					
S56TZA2A	Inlet Louver Position Indicator Fails High	2.4-5	1.0-6	24	
S56CURHA2B	Required Heat Change Calculator Fails Low	4.1-6	1.7-7[a]	24	
S56CPPA2B	Actual to Desired Steam Drum Pressure Comparator Fails Low	4.1-6	1.7-7[a]	24	
S56CUHRA2A	Heat Rejection Calculator Fails High	4.1-6	1.7-7[a]	24	
S56PMESA2B	S/D PRES Median Selector Fails Low	4.1-6	1.7-7[a]	24	
S56PMESA2A	S/D PRES Median Selector Fails High	4.1-6	1.7-7[a]	24	
SMCLB1		5.3-5			TAB-OR
S56CPHB1E	Actual to Desired Heat Comparator Fails	4.1-6	1.7-7[a]	24	
S56CMFBB1E	Fan Blade and Louver Position Control Module Fails	4.1-6	1.7-7[a]	24	
S56TZB1A	Inlet Louver Position Indicator Fails High	2.4-5	1.0-6	24	
S56CURHB1B	Required Heat Change Calculator Fails Low	4.1-6	1.7-7[a]	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMCLB1 (Cont.)					
S56CPPB1B	Actual to Desired Steam Drum Pressure Comparator Fails Low	4.1-6	1.7-7[a]	24	
S56CUHRB1A	Heat Rejection Calculator Fails High	4.1-6	1.7-7[a]	24	
S56PMESB1B	S/D PRES Median Selector Fails Low	4.1-6	1.7-7[a]	24	
S56PMESB1A	S/D PRES Median Selector Fails High	4.1-6	1.7-7[a]	24	
SMCLB2		5.3-5			TAB-OR
S56CPHB2E	Actual to Desired Heat Comparator Fails	4.1-6	1.7-7[a]	24	
S56CMFBB2E	Fan Blade and Louver Position Control Module Fails	4.1-6	1.7-7[a]	24	
S56TZB2A	Inlet Louver Position Indicator Fails High	2.4-5	1.0-6	24	
S56CURHB2B	Required Heat Change Calculator Fails Low	4.1-6	1.7-7[a]	24	
S56CPPB2B	Actual to Desired Steam Drum Pressure Comparator Fails Low	4.1-6	1.7-7[a]	24	
S56CUHRB2A	Heat Rejection Calculator Fails High	4.1-6	1.7-7[a]	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMCLB2 (Cont.)		5.3-5			TAB-OR
S56PMESB2B	S/D PRES Median Selector Fails Low	4.1-6	1.7-7[a]	24	
S56PMESB2A	S/D PRES Median Selector Fails High	4.1-6	1.7-7[a]	24	
SMCLC1		5.3-5			TAB-OR
S56CPHC1E	Actual to Desired Heat Comparator Fails	4.1-6	1.7-7[a]	24	
S56CMFBC1E	Fan Blade and Louver Position Control Module Fails	4.1-6	1.7-7[a]	24	
S56TZC1A	Inlet Louver Position Indicator Fails High	2.4-5	1.0-6	24	
S56CURHC1B	Required Heat Change Calculator Fails Low	4.1-6	1.7-7[a]	24	
S56CPPC1B	Actual to Desired Steam Drum Pressure Comparator Fails Low	4.1-6	1.7-7[a]	24	
S56CUHRC1A	Heat Rejection Calculator Fails High	4.1-6	1.7-7[a]	24	
S56PMESC1B	S/D PRES Median Selector Fails Low	4.1-6	1.7-7[a]	24	
S56PMESC1A	S/D PRES Median Selector Fails High	4.1-6	1.7-7[a]	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMCLC2		5.3-5			TAB-OR
S56CPHC2E	Actual to Desired Heat Comparator Fails	4.1-6	1.7-7[a]	24	
S56CMFBC2E	Fan Blade and Louver Position Control Module Fails	4.1-6	1.7-7[a]	24	
S56TZC2A	Inlet Louver Position Indicator Fails High	2.4-5	1.0-6	24	
S56CURHC2B	Required Heat Change Calculator Fails Low	4.1-6	1.7-7[a]	24	
S56CPPC2B	Actual to Desired Steam Drum Pressure Comparator Fails Low	4.1-6	1.7-7[a]	24	
S56CUHRC2A	Heat Rejection Calculator Fails High	4.1-6	1.7-7[a]	24	
S56PMESC2B	S/D PRES Median Selector Fails Low	4.1-6	1.7-7[a]	24	
S56PMESC2A	S/D PRES Median Selector Fails High	4.1-6	1.7-7[a]	24	
S52PACCAM	Loop A PACC Test and Maintenance	1.8-3	7.6-5[d]	24[d]	
S52PACCBM	Loop B PACC Test and Maintenance	1.8-3	7.6-5[d]	24[d]	
S52PACCBM	Loop C PACC Test and Maintenance	1.8-3	7.6-5[d]	24[d]	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
S52PACCA1L	Latent HE Fails PACC Bundle A1	1.0-3[e]			
S52PACCA2L	Latent HE Fails PACC Bundle A2	1.0-3[e]			
S52PACCB1L	Latent HE Fails PACC Bundle B1	1.0-3[e]			
S52PACCB2L	Latent HE Fails PACC Bundle B2	1.0-3[e]			
S52PACCC1L	Latent HE Fails PACC Bundle C1	1.0-3[e]			
S52PACCC2L	Latent HE Fails PACC Bundle C2	1.0-3[e]			
S56INAM	Control Circuit Instrumentation Test and Maintenance	5.6-4	--	--	S56INAM = [(A1 + A2 + A3) * (B1 + B2 + B3)] + [(A1 + A2 + A3) * (C1 + C2 + C3)] + [(B1 + B2 + B3) * (C1 + C2 + C3)] + D + E + F + G
D	Steam Drum Pressure Median Selector Fails High or Low	1.4-4	1.7-7	Note g	
A1	Steam Drum Pressure Detector Fails High or Low - CH A	1.4-4	1.0-6	Note g	
A2	Steam Drum Pressure Detector Fails High or Low - CH A	1.4-4	1.0-6	Note g	
A3	Steam Drum Pressure Detector Fails High or Low - CH A	1.4-4	1.0-6	Note g	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
S56INAM (Cont.)					
B1	Steam Drum Pressure Detector Fails High or Low - CH B	1.4-4	1.0-6	Note g	
B2	Steam Drum Pressure Detector Fails High or Low - CH B	1.4-4	1.0-6	Note g	
B3	Steam Drum Pressure Detector Fails High or Low - CH B	1.4-4	1.0-6	Note g	
C1	Steam Drum Pressure Detector Fails High or Low - CH C	1.4-4	1.0-6	Note g	
C2	Steam Drum Pressure Detector Fails High or Low - CH C	1.4-4	1.0-6	Note g	
C3	Steam Drum Pressure Detector Fails High or Low - CH C	1.4-4	1.0-6	Note g	
E	Return Flow Detector Signal Fails High	1.4-4	1.0-6	Note g	
F	Return Temperature Detector Fails Low	1.4-4	1.0-6	Note g	
G	Local PWR to Return Temperature Detector Fails Off	1.4-4	1.0-6	Note g	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
S56INBM	Control Circuit Instrumentation Test and Maintenance	5.6-4	--	--	S56INBM = [(A1 + A2 + A3) * (B1 + B2 + B3)] + [(A1 + A2 + A3) * (C1 + C2 + C3)] + [(B1 + B2 + B3) * (C1 + C2 + C3)] + D + E + F + G
D	Steam Drum Pressure Median Selector Fails High or Low	1.4-4	1.7-7	Note g	
A1	Steam Drum Pressure Detector Fails High or Low - CH A	1.4-4	1.0-6	Note g	
A2	Steam Drum Pressure Detector Fails High or Low - CH A	1.4-4	1.0-6	Note g	
A3	Steam Drum Pressure Detector Fails High or Low - CH A	1.4-4	1.0-6	Note g	
B1	Steam Drum Pressure Detector Fails High or Low - CH B	1.4-4	1.0-6	Note g	
B2	Steam Drum Pressure Detector Fails High or Low - CH B	1.4-4	1.0-6	Note g	
B3	Steam Drum Pressure Detector Fails High or Low - CH B	1.4-4	1.0-6	Note g	
C1	Steam Drum Pressure Detector Fails High or Low - CH C	1.4-4	1.0-6	Note g	
C2	Steam Drum Pressure Detector Fails High or Low - CH C	1.4-4	1.0-6	Note g	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
S56INBM (Cont.)					
C3	Steam Drum Pressure Detector Fails High or Low - CH C	1.4-4	1.0-6	Note g	
E	Return Flow Detector Signal Fails High	1.4-4	1.0-6	Note g	
F	Return Temperature Detector Fails Low	1.4-4	1.0-6	Note g	
G	Local PWR to Return Temperature Detector Fails Off	1.4-4	1.0-6	Note g	
S56INCM	Control Circuit Instrumentation Test and Maintenance	5.6-4	--	--	S56INCM = [(A1 + A2 + A3) * (B1 + B2 + B3)] + [(A1 + A2 + A3) * (C1 + C2 + C3)] + [(B1 + B2 + B3) * (C1 + C2 + C3)] + D + E + F + G
D	Steam Drum Pressure Median Selector Fails High or Low	1.4-4	1.7-7	Note g	
A1	Steam Drum Pressure Detector Fails High or Low - CH A	1.4-4	1.0-6	Note g	
A2	Steam Drum Pressure Detector Fails High or Low - CH A	1.4-4	1.0-6	Note g	
A3	Steam Drum Pressure Detector Fails High or Low - CH A	1.4-4	1.0-6	Note g	
B1	Steam Drum Pressure Detector Fails High or Low - CH B	1.4-4	1.0-6	Note g	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
S56INCM (Cont.)					
B2	Steam Drum Pressure Detector Fails High or Low - CH B	1.4-4	1.0-6	Note g	
B3	Steam Drum Pressure Detector Fails High or Low - CH B	1.4-4	1.0-6	Note g	
C1	Steam Drum Pressure Detector Fails High or Low - CH C	1.4-4	1.0-6	Note g	
C2	Steam Drum Pressure Detector Fails High or Low - CH C	1.4-4	1.0-6	Note g	
S56CLA1U	Control Circuit Unavailable at Time of Trip	4.0-3	--	--	TAB-OR
	200 S Cycle Sample and Hold Module Fails to Pass Signal	3.2-4	1.7-7	Note f	
	Required Heat Change Comparator Fails	3.2-4	1.7-7	Note f	
	Required Heat Change Calculator Fails	3.2-4	1.7-7	Note f	
	Actual to Desired Steam Drum Pressure Comparator Fails	3.2-4	1.7-7	Note f	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
S56CLA1U (Cont.)					
	Actual to Desired Heat Comparator Fails	3.2-4	1.7-7	Note f	
	Inlet Lower Position Detector Fails High	2.1-3	1.0-6	Note f	
	Heat Rejection Calculator Fails High	3.2-4	1.7-7	Note f	
S56CLA2U	Control Circuit Unavailable at Time of Trip	4.0-3	--	--	TAB-OR
	200 S Cycle Sample and Hold Module Fails to Pass Signal	3.2-4	1.7-7	Note f	
	Required Heat Change Comparator Fails	3.2-4	1.7-7	Note f	
	Required Heat Change Calculator Fails	3.2-4	1.7-7	Note f	
	Actual to Desired Steam Drum Pressure Comparator Fails	3.2-4	1.7-7	Note f	
	Actual to Desired Heat Comparator Fails	3.2-4	1.7-7	Note f	
	Inlet Lower Position Detector Fails High	2.1-3	1.0-6	Note f	
	Heat Rejection Calculator Fails High	3.2-4	1.7-7	Note f	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
S56CLB1U	Control Circuit Unavailable at Time of Trip	4.0-3	--	--	TAB-OR
	200 S Cycle Sample and Hold Module Fails to Pass Signal	3.2-4	1.7-7	Note f	
	Required Heat Change Comparator Fails	3.2-4	1.7-7	Note f	
	Required Heat Change Calculator Fails	3.2-4	1.7-7	Note f	
	Actual to Desired Steam Drum Pressure Comparator Fails	3.2-4	1.7-7	Note f	
	Actual to Desired Heat Comparator Fails	3.2-4	1.7-7	Note f	
	Inlet Lower Position Detector Fails High	2.1-3	1.0-6	Note f	
	Heat Rejection Calculator Fails High	3.2-4	1.7-7	Note f	
S56CLB2U	Control Circuit Unavailable at Time of Trip	4.0-3	--	--	TAB-OR
	200 S Cycle Sample and Hold Module Fails to Pass Signal	3.2-4	1.7-7	Note f	
	Required Heat Change Comparator Fails	3.2-4	1.7-7	Note f	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
	Required Heat Change Calculator Fails	3.2-4	1.7-7	Note f	
	Actual to Desired Steam Drum Pressure Comparator Fails	3.2-4	1.7-7	Note f	
	Actual to Desired Heat Comparator Fails	3.2-4	1.7-7	Note f	
	Inlet Lower Position Detector Fails High	2.1-3	1.0-6	Note f	
	Heat Rejection Calculator Fails High	3.2-4	1.7-7	Note f	
S56CLC1U	Control Circuit Unavailable at Time of Trip	4.0-3	--	--	TAB-OR
	200 S Cycle Sample and Hold Module Fails to Pass Signal	3.2-4	1.7-7	Note f	
	Required Heat Change Comparator Fails	3.2-4	1.7-7	Note f	
	Required Heat Change Calculator Fails	3.2-4	1.7-7	Note f	
	Actual to Desired Steam Drum Pressure Comparator Fails	3.2-4	1.7-7	Note f	
	Actual to Desired Heat Comparator Fails	3.2-4	1.7-7	Note f	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
	Inlet Lower Position Detector Fails High	2.1-3	1.0-6	Note f	
	Heat Rejection Calculator Fails High	3.2-4	1.7-7	Note f	
S56CLC2U	Control Circuit Unavailable at Time of Trip	4.0-3	--	--	TAB-OR
	200 S Cycle Sample and Hold Module Fails to Pass Signal	3.2-4	1.7-7	Note f	
	Required Heat Change Comparator Fails	3.2-4	1.7-7	Note f	
	Required Heat Change Calculator Fails	3.2-4	1.7-7	Note f	
	Actual to Desired Steam Drum Pressure Comparator Fails	3.2-4	1.7-7	Note f	
	Actual to Desired Heat Comparator Fails	3.2-4	1.7-7	Note f	
	Inlet Lower Position Detector Fails High	2.1-3	1.0-6	Note f	
	Heat Rejection Calculator Fails High	3.2-4	1.7-7	Note f	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMFANA1		1.0-3			TAB-OR
S52CBA1C	Motor Breaker Fails to Shut on Demand	1.0-3	1.0-3/d	--	
S52FBA1T	Fan Blades Transfer to Minimum Pitch	2.9-5	1.2-6/h	24	
SMFANA2		1.0-3			TAB-OR
S52CBA2C	Motor Breaker Fails to Shut on Demand	1.0-3	1.0-3/d	--	
S52FBA2T	Fan Blades Transfer to Minimum Pitch	2.9-5	1.2-6/h	24	
SMFANB1		1.0-3			TAB-OR
S52CBB1C	Motor Breaker Fails to Shut on Demand	1.0-3	1.0-3/d	--	
S52FBB1T	Fan Blades Transfer to Minimum Pitch	2.9-5	1.2-6/h	24	
SMFANB2		1.0-3			TAB-OR
S52CBB2C	Motor Breaker Fails to Shut on Demand	1.0-3	1.0-3/d	--	
S52FBB2T	Fan Blades Transfer to Minimum Pitch	2.9-5	1.2-6/h	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMFANC1		1.0-3			TAB-OR
S52CBC1C	Motor Breaker Fails to Shut on Demand	1.0-3	1.0-3/d	--	
S52FBC1T	Fan Blades Transfer to Minimum Pitch	2.9-5	1.2-6/h	24	
SMFANC2		1.0-3			TAB-OR
S52CBC2C	Motor Breaker Fails to Shut on Demand	1.0-3	1.0-3/d	--	
S52FBC2T	Fan Blades Transfer to Minimum Pitch	2.9-5	1.2-6/h	24	
SMCFANA		3.1-3			TAB-OR
S52FNA1S	PACC Blower Fails To Start - A1	1.0-3	1.0-3/d	--	
S52FNA1R	PACC Blower Fails to Run 24 H - A1	4.8-4	2.0-5	24	
S52FNA2S	PACC Blower Fails to Start - A2	1.0-3	1.0-3/d	--	
S52FNA2R	PACC Blower Fails to Run 24 H - A2	4.8-4	2.0-5	24	
S52CBA1T	Motor Breaker Transfers Open - A1	9.1-5	3.8-6	24	
S52CBA2T	Motor Breaker Transfers Open - A2	9.1-5	3.8-6	24	

Table A5-2 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
SMCFHNB		3.1-3			TAB-OR
S52FNB1S	PACC Blower Fails To Start - B1	1.0-3	1.0-3/d	--	
S52FNB1R	PACC Blower Fails to Run 24 H - B1	4.8-4	2.0-5	24	
S52FNB2S	PACC Blower Fails to Start - B2	1.0-3	1.0-3/d	--	
S52FNB2R	PACC Blower Fails to Run 24 H - B2	4.8-4	2.0-5	24	
S52CBB1T	Motor Breaker Transfers Open - B1	9.1-5	3.8-6	24	
S52CBB2T	Motor Breaker Transfers Open - B2	9.1-5	3.8-6	24	
SMCFANC		3.1-3			TAB-OR
S52FNC1S	PACC Blower Fails To Start - C1	1.0-3	1.0-3/d	--	
S52FNC1R	PACC Blower Fails to Run 24 H - C1	4.8-4	2.0-5	24	
S52FNC2S	PACC Blower Fails to Start - C2	1.0-3	1.0-3/d	--	
S52FNC2R	PACC Blower Fails to Run 24 H - C2	4.8-4	2.0-5	24	
S52CBC1T	Motor Breaker Transfers Open - C1	9.1-5	3.8-6	24	
S52CBC2T	Motor Breaker Transfers Open - C2	9.1-5	3.8-6	24	

NOTES FOR TABLE A5-2

a. Data from IEEE Std. 500-1977 for all types of bistables.

b. Data from IEEE Std. 500-1977 for all types of signal modifiers (zero or maximum output).

c. Failures are immediately announced. Mean duration is mean time to repair given in MOP 99-003.

d. Values from Data Chapter Table 5-3.

e. Screening value assigned during preliminary HRA.

f. Failures of the control circuit are not announced; hence, they are detected only when the circuit is used or tested.

$$\bar{a} = \frac{1}{T}\left[\int_{0}^{T-t} \lambda t' dt' + \int_{T-t}^{T} 1.0\, dt'\right]$$

where:

$\bar{a}$ = mean unavailability averaged over the test interval,

T = test interval (2190 hrs. or 4 per year),

t = test duration (0.3 hrs), and

λ = failure rate (per Table A5.3-1).

Test interval and duration are not specified for SGAHRS control circuits; the above values are ones given for the RSS.

g. Instrumentation is continuously monitored in the Control Room; hence, failures are announced.

$$\bar{a} = \frac{1}{T}\left[\int_{0}^{T-t} \lambda t' dt' + \int_{T-t}^{T} 1.0\, dt'\right]$$

where:

T = mean time to repair (5 hrs), and $\bar{a}$, T, t, λ are defined as in Note f.

h. Data from IEEE Std. 500-1977 for low air flow from axial fans.

Appendix A, Section 6

NORMAL PLANT SERVICE WATER SYSTEM

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Appendix A, Section 6

NORMAL PLANT SERVICE WATER SYSTEM

A6.1 SYSTEM DESCRIPTION

A6.1.1 Function

The normal plant service water (NPSW) system consists of a nonredundant cooling loop with two 100% capacity circulating water pumps and associated piping, valves, instrumentation, and controls.[1]

The NPSW system is a Seismic Category III system designed to provide heat removal service for plant components during start-up and normal plant operation. The system is an open loop cooling system that operates in conjunction with the main cooling tower and cooling tower basin as a heat rejection medium and water source.

The NPSW system removes heat from the following equipment during normal plant operation:

1. Radioactive waste system evaporator condenser.
2. NI HVAC unit coolers for the reactor heat transport motor generator sets.
3. Reactor heat transport system motor generator set heat exchangers.
4. Normal chilled water system chiller condensers.
5. Secondary services closed cooling water heat exchangers.
6. Auxiliary steam system blowdown tank.
7. Reactor refueling system spent fuel shipping cask.

The NPSW system distributes cooling water to this equipment which is located in the following buildings:

- Steam generator intermediate bay
- Reactor service building radwaste area
- Control building
- Diesel generator building
- Turbine generator building
- Reactor service building

For the purposes of this analysis, equipment described under 1, 2, 3, 6, and 7 above will not be modeled. These pieces of equipment are deemed nonessential in the 24-hr period following off-normal conditions.

A6.1.2 Design

The NPSW system provides the cooling functions required during normal and upset conditions of plant operation, except for upset events involving the loss of all offsite power. For the purposes of this analysis, the NPSW system during normal operation is designed to remove heat from the following plant components:

a. Normal chilled water system chillers

   1. 23NCH001A
   2. 23NCH001B
   3. 23NCH001C
   4. 23NCH001D
   5. 23NCH001E
   6. 23NCH001F

b. Nuclear island HVAC MG set unit coolers

   1. 25ACA421
   2. 25ACA422
   3. 25ACA423
   4. 25ACA424
   5. 25ACA321
   6. 25ACA322

c. Lube-oil coolers for primary and intermediate heat transport system pump motors

1. 56PRH205A
2. 56PRH205B
3. 56PRH205C
4. 56INH205A
5. 56INH205B
6. 56INH205C

d. Secondary services closed cooling water heat exchangers

1. 75SWH001A
2. 75SWH001B

Figure A6-1[2] illustrates the design of the NPSW system.

A6.1.3 Interfaces With Other Systems

A6.1.3.1 Electric power. Electric power is required for the two motor-operated pumps 75NPP001A and P001B that receive power from buses 12BPE002A and 12BPE002B, respectively.

A6.1.3.2 Control power. Control power is assumed to come from the same bus as the motive power.

A6.1.3.3 Cooling water. The NPSW system delivers water to the components identified in Section 6.1.2. Cooling water supplied to these components comes from the main cooling tower and cooling tower basin.

A6.1.3.4 Instrument Air. Instrument air is necessary for the operation of two temperature control valves (TV3A and TV3B) and one pressure control valve (PV2).

A6.1.3.5 Actuation system. This system receives no interface actuation from other systems. Actuation and control logic internal to the NPSW system are described in the next section.

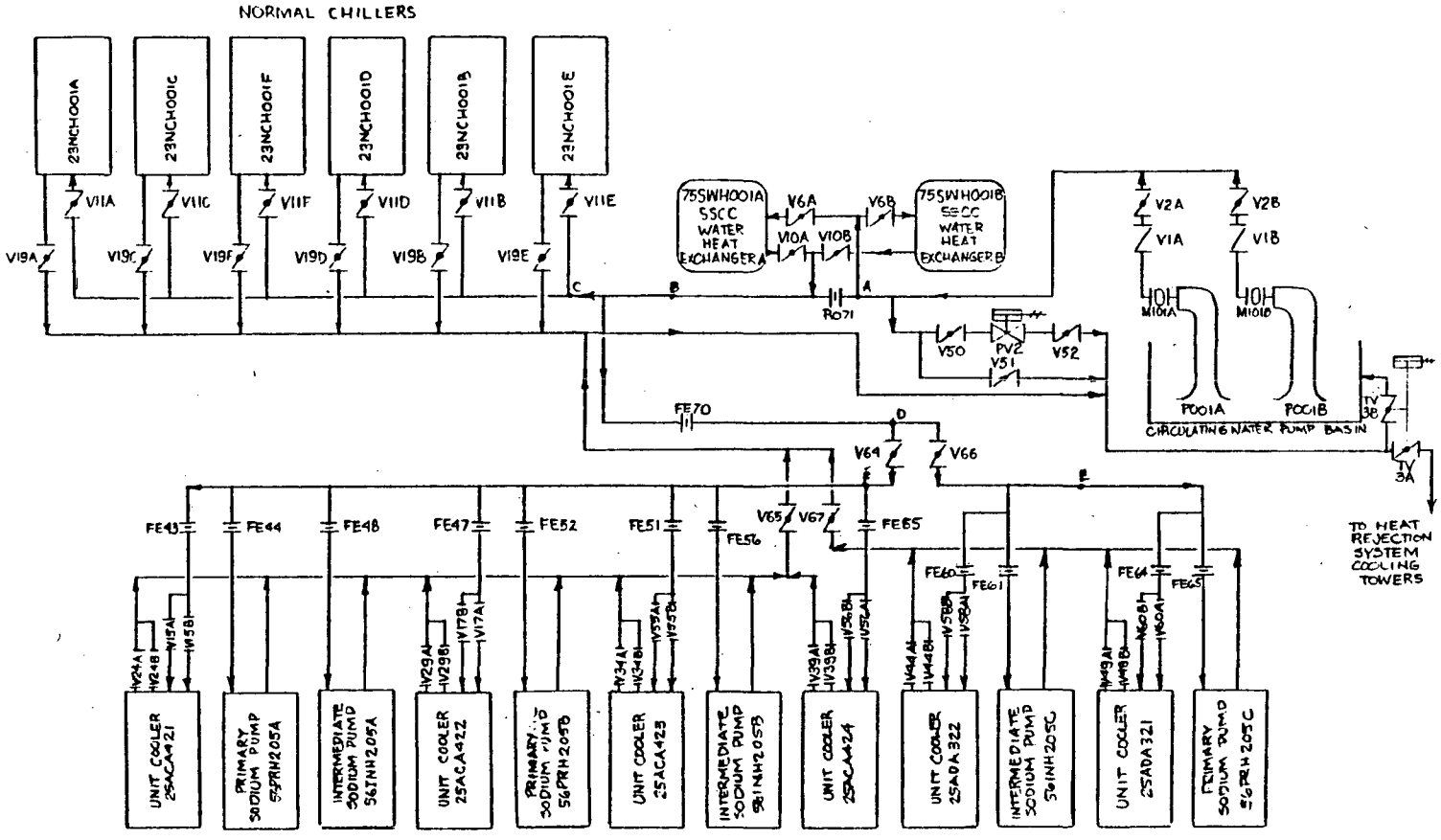


Figure A6-1. P&ID Normal Plant Service Water

A6.1.4 Operation

A6.1.4.1 Normal operation. Water is pumped from the circulating water basin by one operating NPSW pump and directed to the various plant components via supply mains and distribution piping. Cooling water is supplied at 87°F maximum to the secondary services closed cooling water (SSCCW) heat exchangers and 90°F to all other system components. After cooling the plant components, warm water is collected in a return header, and directed to the heat rejection system cooling towers and then into the circulating water basin. To account for seasonal temperature variations, two temperature control valves (TV-3A and TV-3B) served by a common operator, bypass a portion of the returning warm water back to the NPSW pump suction pit. A temperature indicator controller in the pump house automatically adjusts the valves as required to maintain supply temperature above 55°F. In addition a bypass line is provided for returning pump discharge water back to the pump sump through a pressure control valve (PV-2). The pressure control valve automatically operates to maintain a constant supply header pressure should a significant portion of the system load be isolated for preventive or corrective maintenance.

Valves are provided at each of the equipment connections throughout the system for component and branch line isolation and flow control adjustments. All distribution lines are sized to meet design flow requirements when the throttling valves have been set.

The seal water requirements for the NPSW pumps and the circulating water pumps are normally supplied from the NPSW supply header. Seal water

from the supply header is directed to the heat rejection system where it is filtered and strained. The clean seal water is then returned for use by the NPSW pumps. Potable water also supplies seal water to the NPSW pumps and circulating water pumps for system start-up and as a backup to the normal seal water supply. Flow switches are provided in the seal water line to each NPSW pump to prevent pump start-up when seal water is not available or to alarm if seal water flow decreases below an acceptable level during pump operation.

Each pump (75NPP001A and P001B) has a three-position selector switch on the back panel in the main control room. When the pump selector switch is in the "Auto" position the pump is considered to be in the standby mode. In this mode the pump will start automatically if the running pump is inadvertently stopped or experiences an electrical fault that trips the circuit breaker. A "Start" and a "Stop" position on each switch also allows manual control of each pump.

The back panel in the control room has indicating lights for the service water pumps 75NPP001A and P001B. The main control board contains a common alarm for the following conditions:

- Pump 75NPP001A/P001B discharge pressure low
- Normal plant supply temperature low
- Circulating water pump house basin/pump 75NPP001A pit level high
- Circulating water pump house basin/pump 75NPP001B pit level high

A6.1.4.2 <u>Off normal operation</u>. Normally, the NPSW system return header joins the heat rejection system (HRS) at a point where the warm water is distributed evenly to each of the two 50% capacity cooling towers. During a plant shutdown or no-load condition when the turbine-generator

is not operating, the NPSW system return is directed to only one of the cooling towers. In this mode, using one cell of one tower, the cooling equipment operates well within its design range.

When the cooling tower basin blowdown is not operating, an 8-in blowdown line from the NPSW pump discharge will be utilized to supply blowdown/dilution water for the process requirements of the waste water treatment system.

The NPSW system is not designed to operate under the loss of all offsite power conditions.

A6.1.5 Test and Operational Requirements

A6.1.5.1 Testing. The normal plant service system will not be periodically tested.

A6.1.5.2 Technical specification. The NPSW system is designed to provide heat removal service for plant components during startup, shutdown, and normal plant operation. The only anticipated condition for which the NPSW system is not designed to operate is the loss of all offsite power condition.

A6.1.6 Maintenance Requirements

The NPSW system is designed to permit contact maintenance and direct access to all system components. Only one pump train at a time should be maintained. The other train, not being maintained, is assumed to be operating. Preventive maintenance that takes a pump out of service will be performed during refueling.

All system instrumentation may be isolated without shutdown of the NPSW system with the following exceptions:

1. Operating NPSW system pumps and water low-flow switches (FISHL-39A or 39B). These switches could not be found in the P&ID or the instrument list. It is assumed these switches shut off the pump in the event of low-seal water flow.

2. Pump pit low level (LSL-42A or 42B).

3. Supply header temperature transmitter (TT-3).

4 Supply header pressure transmitter (PT-2).

The loss of instruments (1 or 2 above) causes shutdown of the NPSW system. The NPSW system can be operated by manual control if only 3 or 4 is lost.

Most units serviced by the NPSW system contain multiple cooling coils with the capability of being removed independently. NPSW operation will not be affected by isolation and removal of an individual coil.

A6.2 SYSTEM MODEL

A6.2.1 Modeling Assumptions

a. There are two pump headers, "A" and "B", each individually capable of providing sufficient water flow to fulfill the load requirements of the NPSW system.

b. This analysis assumes the design engineer has properly sized the piping system to meet cooling requirements of the loads.

c. NPSW header "A" is assumed to be operating, and header "B" is assumed to be in the standby mode.

d. Mission time for the analysis is 24 hr. Because of the short mission time, repair of equipment, once failed, is not accounted for in the model.

e. Pipe, pump, and valve ruptures are included in the model as a conglomerate event.

f. The term "transfers" implies going from a desirable or required position to an undesirable or failed position.

g. Flow blockage to or from the secondary services closed cooling water heat exchangers will not fail the NPSW system. The restricting orifice R071 permits the flow of sufficient water to permit cooling of the other system components.

h. Dynamic human errors such as (1) operator turns off the redundant train after the operating train has failed, and (2) operator inadvertently changes a valve position causing failure of the NPSW system are treated in the model. Specifics are on the fault tree. Latent human errors such as operator failure to return the NPSW system components to operational readiness following maintenance activities are included in the model.

i. Unit coolers in the control building and the diesel generator building have not been modeled.

j. It is assumed that water supply temperature is above 55°F since mission involves previous operation of the plant. TV-3A and TV-3B bypass water back to pump basin if supply temperature (TT3) is below 55°F. This analysis assumes only one failure mode for these valves, that is, TV-3A transfers closed and TV-3B fails to open.

k. This analysis assumes the pump basin has a supply of cooling water at the start of the mission. This assumption is made because train A is assumed to be operating at start of the mission. Seismic events of sufficient magnitude to fail the pump basin will be modeled in the seismic sequence analysis and are not considered here.

l. Assumptions regarding operation of the HRS include the following:

   1. The circulating water basin, from which the NPSW system draws its cooling water, contains approximately 988,865 gallons.[3]

   2. In addition to this large volume of water, river water could be used to cool basin water in the event of loss of cooling tower function.

   3. For the purpose of this analysis, the heat rejection system will need to dissipate heat from the condenser for less than 4 hr. This represents a minimal load on the HRS.

   4. The NPSW System is only 10% of the HRS load.

   As a result of these assumptions, failure of normal plant service cooling by the HRS is not considered a credible event in this analysis.

m. Loss of cooling to the heat exchangers for the sodium pump motor-generator sets has not been modeled.

n. It is assumed that if either recirculation flow path for the NPSW pumps (via valve PV-2 or PV-51) is open, sufficient flow will be diverted to defeat the system.

A6.2.2 Detailed Fault Tree Model

The detailed fault tree for the NPSW system is depicted in Figure A6-2.

A6.2.2.1 Success criteria. Success of the NPSW system involves the distribution of cooling water to those loads listed in Section 6.1.2 (with the exception of the items noted in assumptions i and m above). In order for the system to successfully operate, one of two NPSW pumps must operate as designed.

A6.2.2.2 Top events. The top events in this analysis are the loss of cooling water to the NCW system chillers and the secondary services closed cooling water heat exchanger.

A6.2.2.3 Transfers to other systems.

	Electric Power		
Equipment	Bus	Breaker	Gate
75NPP001A	12BPE002A	YD-2A	EM324A
75NPP001B	12BPE002B	YD-2B	EM324A

A6.2.3 Modularized Fault Tree Model

NPSW system fault tree was simplified by combining sets of independent basic events into modules. The modularized fault tree is presented in Figure A6-3. The following modules include the equipment as noted:

Module	Contents of Module
WNM1	SSCC heat exchanger and discharge valves
WNM2	Pump 1A and discharge valves
WNM3	Pump 1B and discharge valves
WNM4	Normal chillers NCH001A-F
WNM5	Return line valves

A6.3 FAILURE DATA

A6.3.1 Basic Event Data

Data were applied to the events in the detailed fault tree according to the rules and tables found in Sections 5 and 6.

A6.3.2 Module Data

Failure probabilities for the modules were calculated by combining the basic event data according to the definitions of the modules. Results of these calculations can be found in Table A6-1.

A6.4 SYSTEM LEVEL RESULTS AND INSIGHTS

Failure of the NPSW system to supply cooling water to the SSCCW heat exchanger appears to be dominated by a failure of module WNM1 failure. Loss of NPSW system cooling water to the chillers appears to be dominated by failure of the chillers and their associated valves.

The predominant failures for the NPSW system include the loss of offsite power and NPSW system specific bus failures. The probabilities of these two failures dominate any faults that originate within the NPSW system.

A6.5 REFERENCES

1. System Design Description, Plant Service Water System (SDD-75A) Rev. 23 (7/83).

2. Drawing NN570, P&ID Normal Plant Service Water System, Rev. 12 (5/10/83).

3. Burns & Roe Draft Calculation, B-73CW-4.02 (6/15/82).

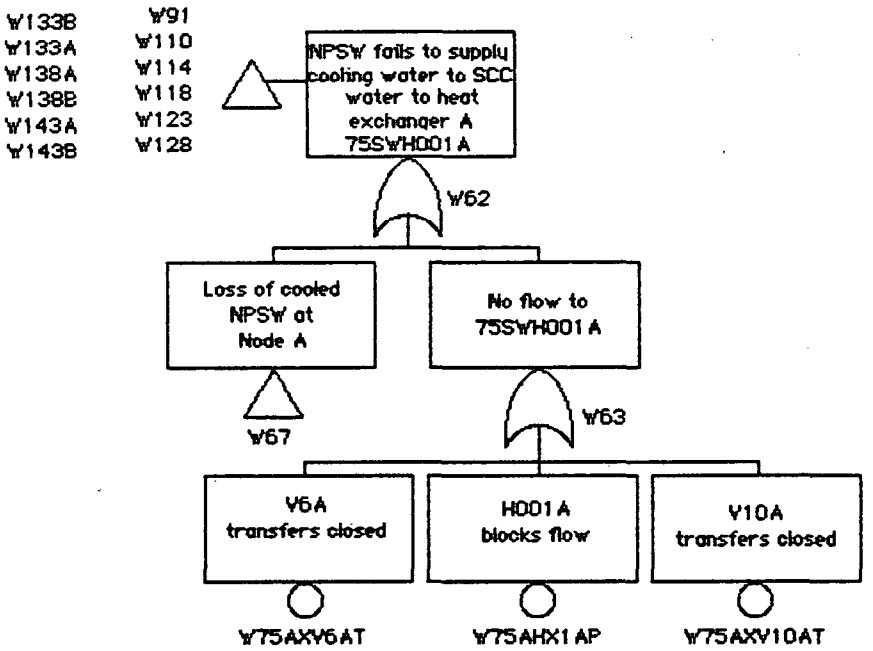
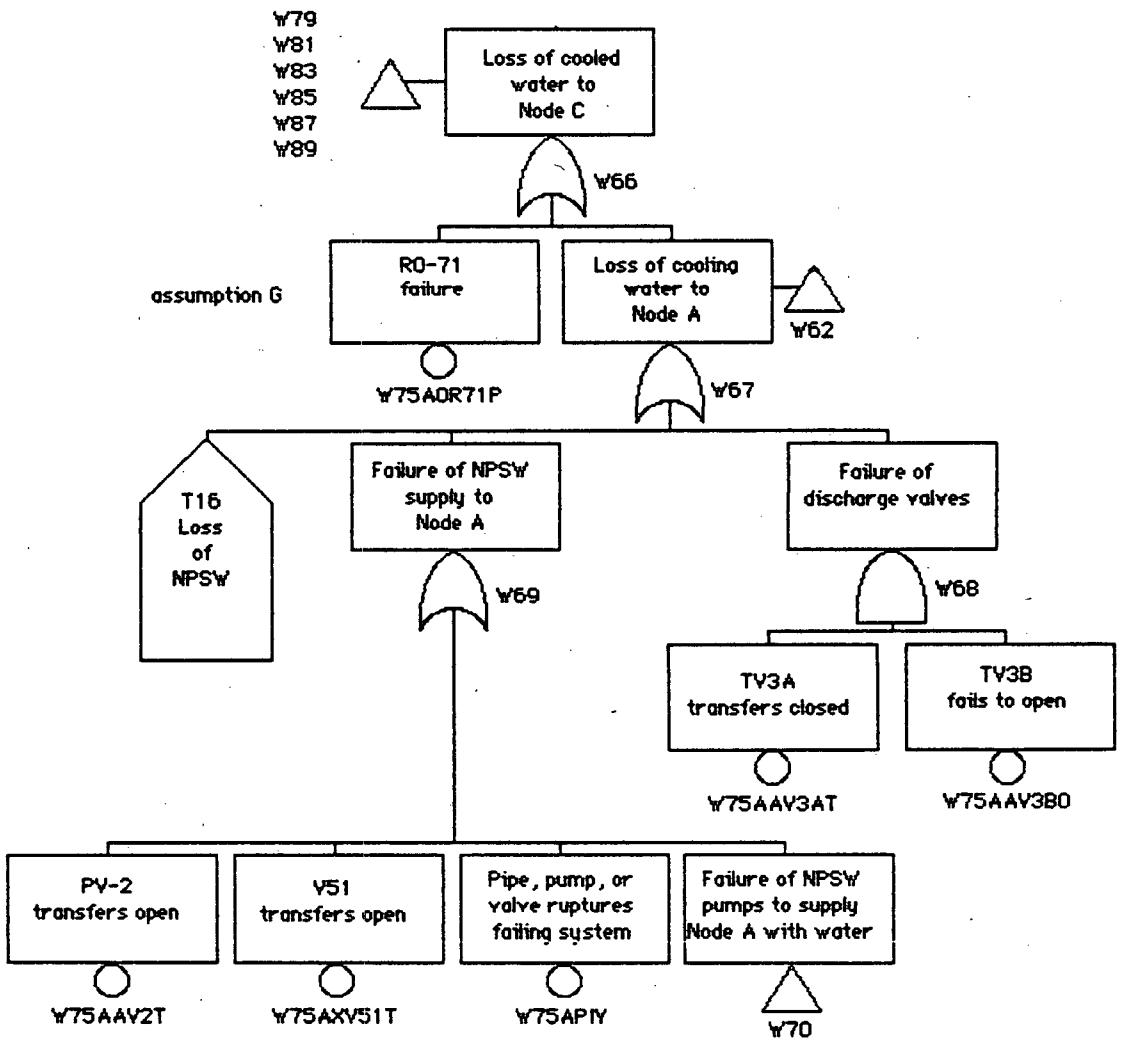
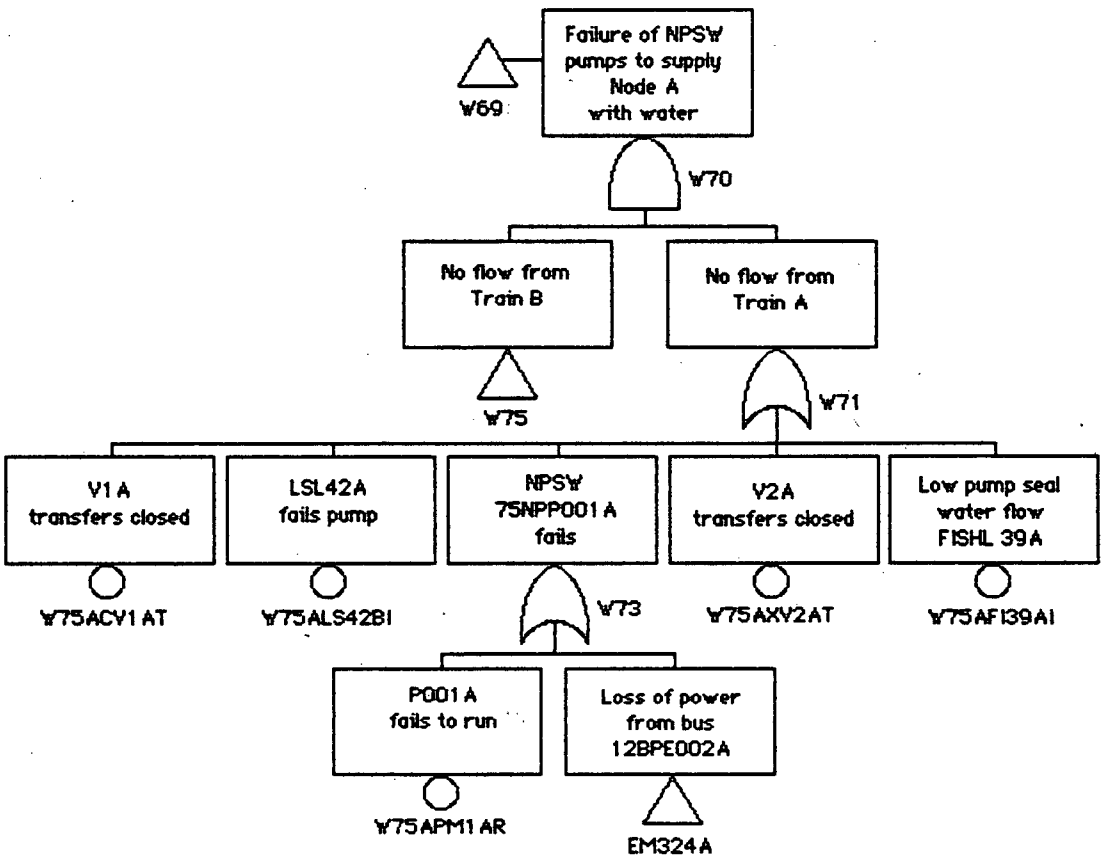


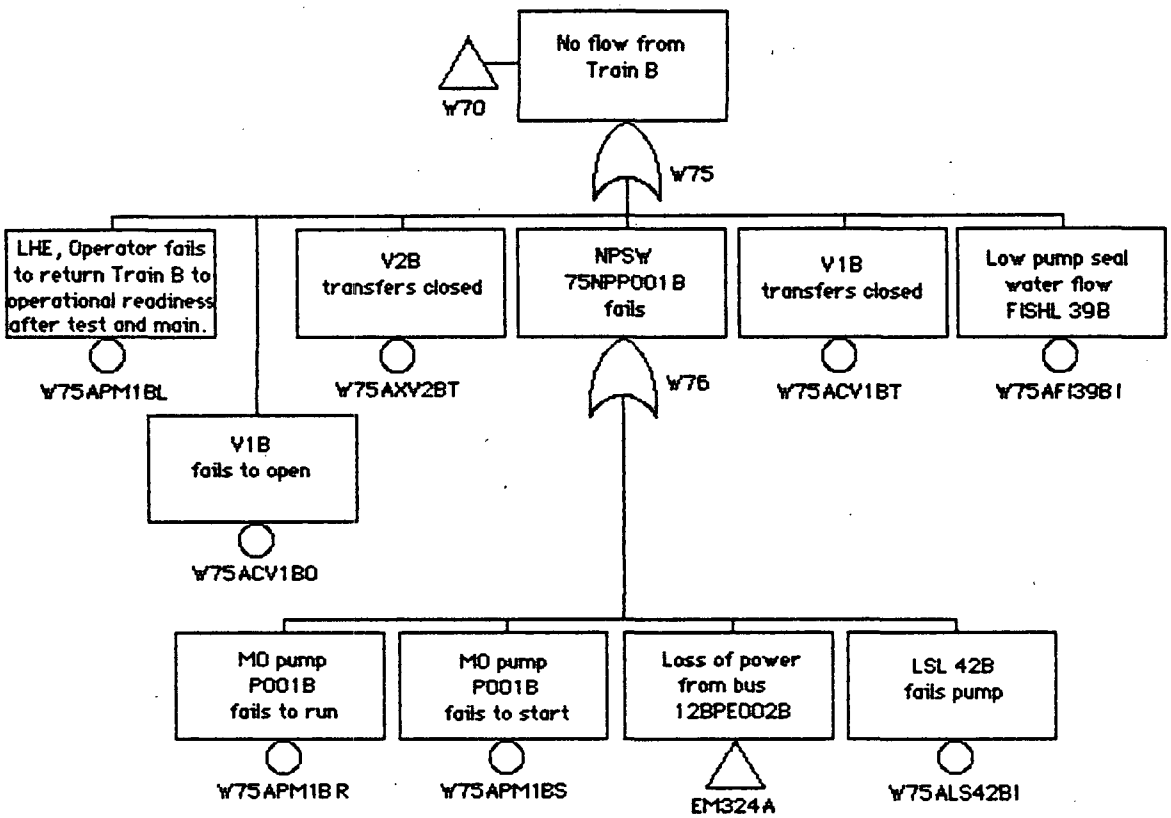
Figure A6-2. Normal Plant Service Water System

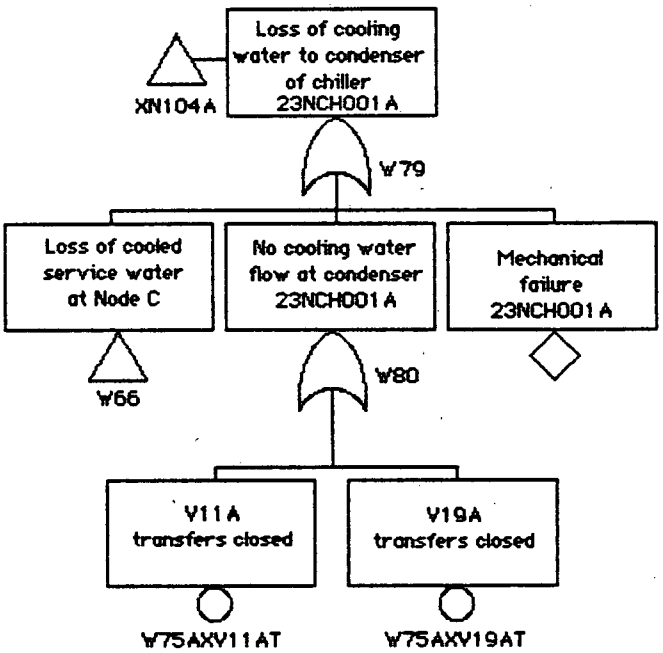
Detailed Fault Tree

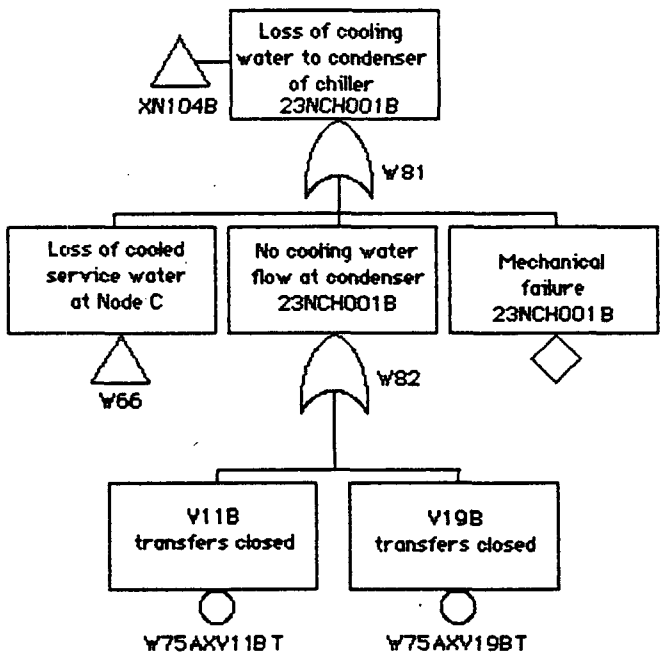

W79
W81
W83
W85
W87
W89
Loss of cooled water to Node C
W66
assumption G
RO-71 failure
W75AOR71P
Loss of cooling water to Node A
W62
W67
T16 Loss of NPSW
Failure of NPSW supply to Node A
W69
Failure of discharge valves
W68
TV3A transfers closed
W75AAV3AT
TV3B fails to open
W75AAV3BO
PV-2 transfers open
W75AAV2T
V51 transfers open
W75AXV51T
Pipe, pump, or valve ruptures failing system
W75APIY
Failure of NPSW pumps to supply Node A with water
W70

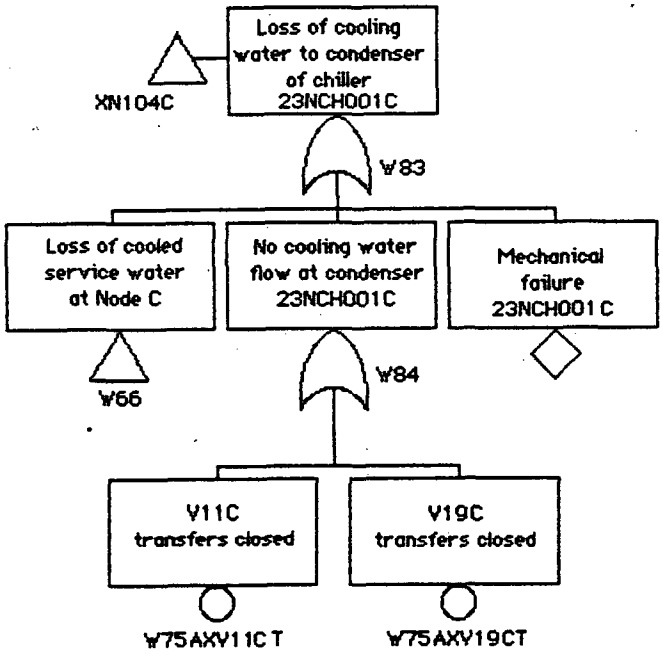
Page #2

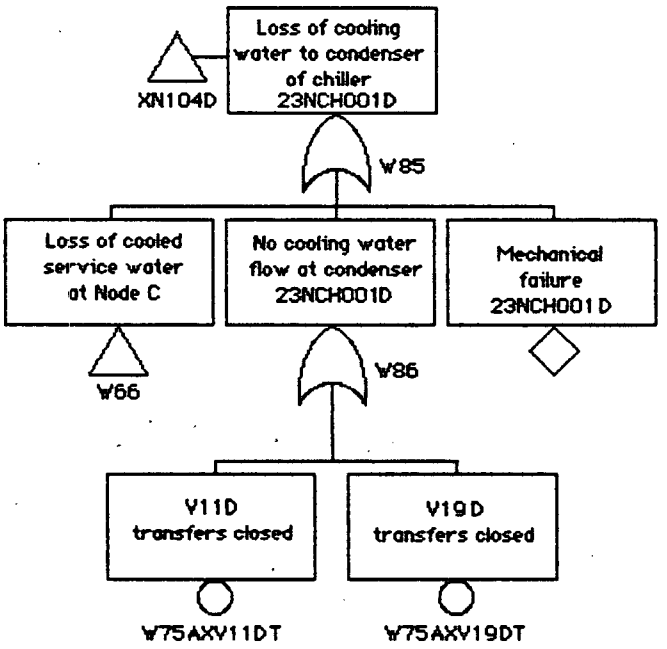

Failure of NPSW pumps to supply Node A with water
W69
W70
No flow from Train B
No flow from Train A
W75
W71
V1A transfers closed
LSL42A fails pump
NPSW 75NPP001A fails
V2A transfers closed
Low pump seal water flow FISHL 39A
W75ACV1AT
W75ALS42BI
W73
W75AXV2AT
W75AFI39AI
P001A fails to run
Loss of power from bus 12BPE002A
W75APM1AR
EM324A

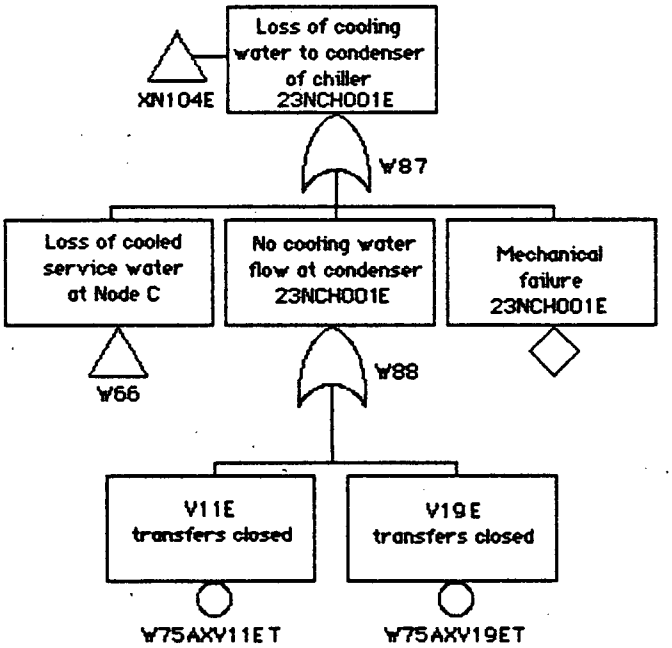

No flow from Train B
W70
W75
LHE, Operator fails to return Train B to operational readiness after test and main.
W75APM1BL
V2B transfers closed
W75AXV2BT
NPSW 75NPP001B fails
W76
V1B transfers closed
W75ACV1BT
Low pump seal water flow FISHL 39B
W75AFI39BI
V1B fails to open
W75ACV1BO
MO pump P001B fails to run
W75APM1BR
MO pump P001B fails to start
W75APM1BS
Loss of power from bus 12BPE002B
EM324A
LSL 42B fails pump
W75ALS42BI

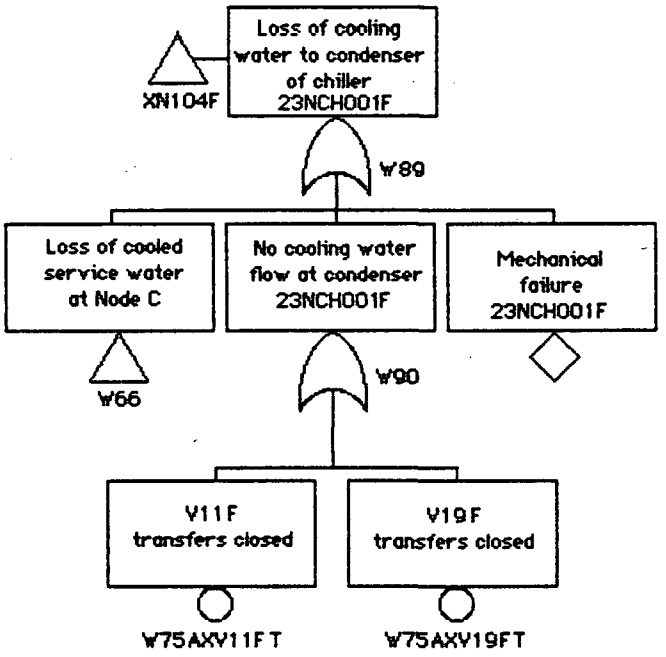

Loss of cooling water to condenser of chiller 23NCH001A
XN104A
W79
Loss of cooled service water at Node C
No cooling water flow at condenser 23NCH001A
Mechanical failure 23NCH001A
W66
W80
V11A transfers closed
V19A transfers closed
W75AXV11AT
W75AXV19AT


Loss of cooling water to condenser of chiller 23NCH001B
XN104B
W81
Loss of cooled service water at Node C
No cooling water flow at condenser 23NCH001B
Mechanical failure 23NCH001B
W66
W82
V11B transfers closed
V19B transfers closed
W75AXV11BT
W75AXV19BT


Loss of cooling water to condenser of chiller 23NCH001C
XN104C
W83
Loss of cooled service water at Node C
No cooling water flow at condenser 23NCH001C
Mechanical failure 23NCH001C
W66
W84
V11C transfers closed
V19C transfers closed
W75AXV11CT
W75AXV19CT


Loss of cooling water to condenser of chiller 23NCH001D
XN104D
W85
Loss of cooled service water at Node C
No cooling water flow at condenser 23NCH001D
Mechanical failure 23NCH001D
W66
W86
V11D transfers closed
V19D transfers closed
W75AXV11DT
W75AXV19DT


Loss of cooling water to condenser of chiller 23NCH001E
XN104E
W87
Loss of cooled service water at Node C
No cooling water flow at condenser 23NCH001E
Mechanical failure 23NCH001E
W66
W88
V11E transfers closed
V19E transfers closed
W75AXV11ET
W75AXV19ET


Loss of cooling water to condenser of chiller 23NCH001F
XN104F
W89
Loss of cooled service water at Node C
No cooling water flow at condenser 23NCH001F
Mechanical failure 23NCH001F
W66
W90
V11F transfers closed
V19F transfers closed
W75AXV11FT
W75AXV19FT

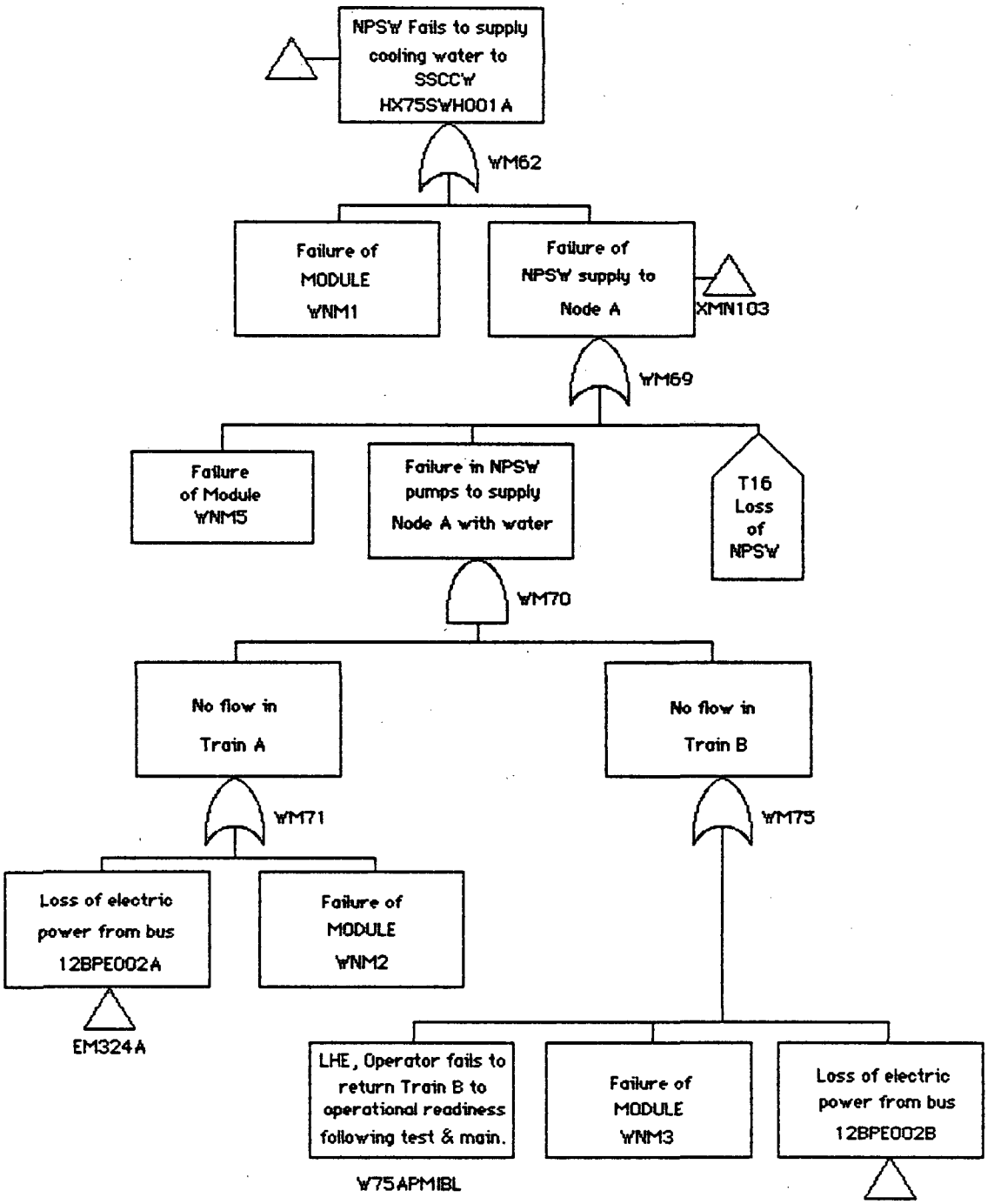


Figure A6-3 Normal Plant Service Water System

Modularized Fault Tree

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Table A6-1

Data Table

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hr)	Module Definition
WNM1	SSCCW heat exhanger A and valves	2.9-5			W75AXV6AT + W75AHX1AP + W75AXV10AT
W75AXV6AT		2.4-6	1.0-7	24	
W75AHX1AP		2.4-5	1.0-6	24	
W75AXV10AT		2.4-6	1.0-7	24	
WNM2	Pump A and valves	6.9-4			TAB-OR
W75ACV1AT		5.5-6	2.3-7	24	
W75ALS42AI		1.0-4	4.2-6	24	
W75AFI39AI		1.0-4	4.2-6	24	
W75APM1AR		4.8-4	2.0-5	24	
W75AXV2AT		2.4-6	1.0-7	24	
WNM3	Pump B and valves	1.9-4			TAB-OR
W75AXV2BT		2.4-6	1.0-7	24	
W75APM1BR		4.8-4	2.0-5	24	
W75APM1BS		1.0-3	1.0-3		
W75ALS42BI		1.0-4	4.2-6	24	
W75AFI39BI		1.0-4	4.2-6	24	
W75ACV1BO		1.0-4	1.0-4		
W75ACV1BT		5.5-6	2.3-7	24	

Table A6-1 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hr)	Module Definition
WNM5	Return line valves or system rupture	1.2-6			W75AAV3AT * W75AAV3BO + W75AAV2T + W75AXV51T + W75AR071P + W75APIY
W75AAV3AT		2.4-6	1.0-7	24	
W75AAV3BO		1.0-3	1.0-3		
W75AAV2T		2.4-6	1.0-7	24	
W75AXV51T		2.4-6	1.0-7	24	
W75AR071P	Orifice plugs	6.7-6	2.8-7	24	
W75APIY	Pipe leak	ε			
W75APM1BL*	Latent human error	1.0-3			
Events in NCW**	Normal chillers				
W75AXV11AT		2.4-6	1.0-7	24	
W75AXV11BT		2.4-6	1.0-7	24	
W75AXV11CT		2.4-6	1.0-7	24	
W75AXV11DT		2.4-6	1.0-7	24	
W75AXV11ET		2.4-6	1.0-7	24	
W75AXV11FT		2.4-6	1.0-7	24	
W75AXV19AT		2.4-6	1.0-7	24	
W75AXV19BT		2.4-6	1.0-7	24	
W75AXV19CT		2.4-6	1.0-7	24	
W75AXV19DT		2.4-6	1.0-7	24	

Table A6-1 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hr)	Module Definition
W75AXV19ET		2.4-6	1.0-7	24	
W75AXV19FT		2.4-6	1.0-7	24	
W75ACH1AP		2.4-5	1.0-6	24	
W75ACH1BP		2.4-5	1.0-6	24	
W75ACH1CP		2.4-5	1.0-6	24	
W75ACH1DP		2.4-5	1.0-6	24	
W75ACH1EP		2.4-5	1.0-6	24	
W75ACH1FP		2.4-5	1.0-6	24	

*This is a screening number. Some human events may have been assessed in detail, and their reassessed probability can be found in Section 6.

**To simplify the modularized fault trees, these faults were included with the respective chillers they affect in the NCWS fault tree.

Appendix A, Section 7

EMERGENCY PLANT SERVICE WATER SYSTEM

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EMERGENCY PLANT SERVICE WATER SYSTEM

A7.1 SYSTEM DESCRIPTION

A7.1.1 Function

The emergency plant service water (EPSW) system is designed to provide sufficient cooling water to permit the safe shutdown of the plant and the maintenance of the safe shutdown condition in the event of an accident resulting in the loss of normal plant service water NPSW or plant ac power supply and all offsite ac power supply.[1] The system provides the emergency chilled water system chiller condensers and the standby diesel generators with cooling water.

A7.1.2 Design

The EPSW system consists of two 100% capacity cooling loops and a third cooling loop for the Division 3 diesel generator. Each cooling loop includes one circulating pump, one emergency cooling tower, and associated piping, valves, instrumentation, and controls. The EPSW system is designed to Seismic Category I requirements. Pumps, valving, and piping required for the safe shutdown of the plant are designed to ASME Section III, Class 3 requirements. All electric motors serving the system are connected to the Class 1E onsite power supply. In case of loss of plant and offsite power, these motors are switched automatically to the standby diesel generator. The piping and equipment for each redundant loop of the system is physically separated or protected with a barrier.

Cooled water from the emergency cooling tower storage basin is pumped via underground supply mains to the emergency loads described below:

a. EPSW system loop "A"

   1. 23ECH001A (emergency chiller)
   2. 12NIE022A (standby diesel generator)
   3. 25ADA300A (unit cooler)
   4. 25AEA100A (unit cooler)

b. EPSW system loop "B"

   1. 23ECH001B (emergency chiller)
   2. 12NIE022B (standby diesel generator)
   3. 25ADA300B (unit cooler)
   4. 25AEA100B (unit cooler)

c. EPSW system loop "C"

   1. 12NIE022C (standby diesel generator)
   2. 25ADA300C (unit cooler)
   3. 25AEA100C (unit cooler)

After cooling the emergency chillers and the standby diesel generators, warm water is returned from trains A and B via the underground mains, to the emergency cooling tower. Temperature control valves served by a common electro-hydraulic operator bypass a portion of the returning warm water back to the EPSW pump suction under control of a temperature indicator controller that automatically adjusts the valves as required to maintain supply temperature above 55°F, the minimum required for chiller operation. Train C operates similarly, but it does not supply a chiller.

The EPSW system loop "A" is supplied ac power from Class 1E Division 1. Loop "B" is supplied from Class 1E Division 2, and Loop "C" is supplied from Class 1E Division 3.

The emergency cooling tower structure consists of three pump houses located directly above the storage water basin. The storage basin has sufficient storage capacity for 30 days of operation, including 30,000 gallons of water storage for the non-sodium fire protection system plus adequate allowance for drift and evaporation losses. The total volume of the basin is approximately 2,896,000 gallons.[2]

The emergency cooling towers are of a counter-flow, wet-type, mechanically induced draft design. The internal distribution piping distributes the intake water evenly over the fill area so that sufficient water area is exposed to the counter air flow to provide evaporation for the required heat removal. The counter air flow is provided by the induced draft fans; three fans for tower "A" three fans for tower "B", and two fans for tower "C."

The EPSW pump can receive the start signal from a number of actuators. Normally the pump switch on the back panel of the control room is in the "auto" mode. In this mode pumps A and B automatically start upon receipt of a signal from either: the emergency diesel generator loading program, or loss of normal chilled water.

Pump C starts upon receipt of a signal from emergency diesel generator C loading program.

All three loops can also receive system-level initiation from push buttons in the main control room. All three loops also have a local panel switch that can be energized in the event the main control room becomes uninhabitable. This switch is similar to that on the control room back panel.

Trains "A" and "B" of the EPSW system have three cooling tower fans. The system is designed such that two out of three of the fans must operate to supply the required cooling. Switches similar to those for the EPSW pumps are provided for the cooling tower fans. The cooling tower fans receive their start signal from their respective EPSW pump. Train "C" of the EPSW system has two cooling tower fans, both of which must operate to supply the required cooling.

A simplified drawing (Figure A7-1) illustrates the piping and instrumentation arrangement of loop "A." Loops "B" and "C" are illustrated in Figures A7-2 and A7-3 respectively.

Common alarms are provided in the control room for the following equipment in the fault tree:

Emergency cooling tower A fans	(3) A002A, C, E
Emergency cooling tower B fans	(3) A002B, D, F
Emergency cooling tower C fans	(2) A002G, H
Storage basin level low	(3) LSL 90A, B, C

A7.1.3 Interfaces With Other Systems

A7.1.3.1 Electric power. The EPSW System receives electric power from the following buses:

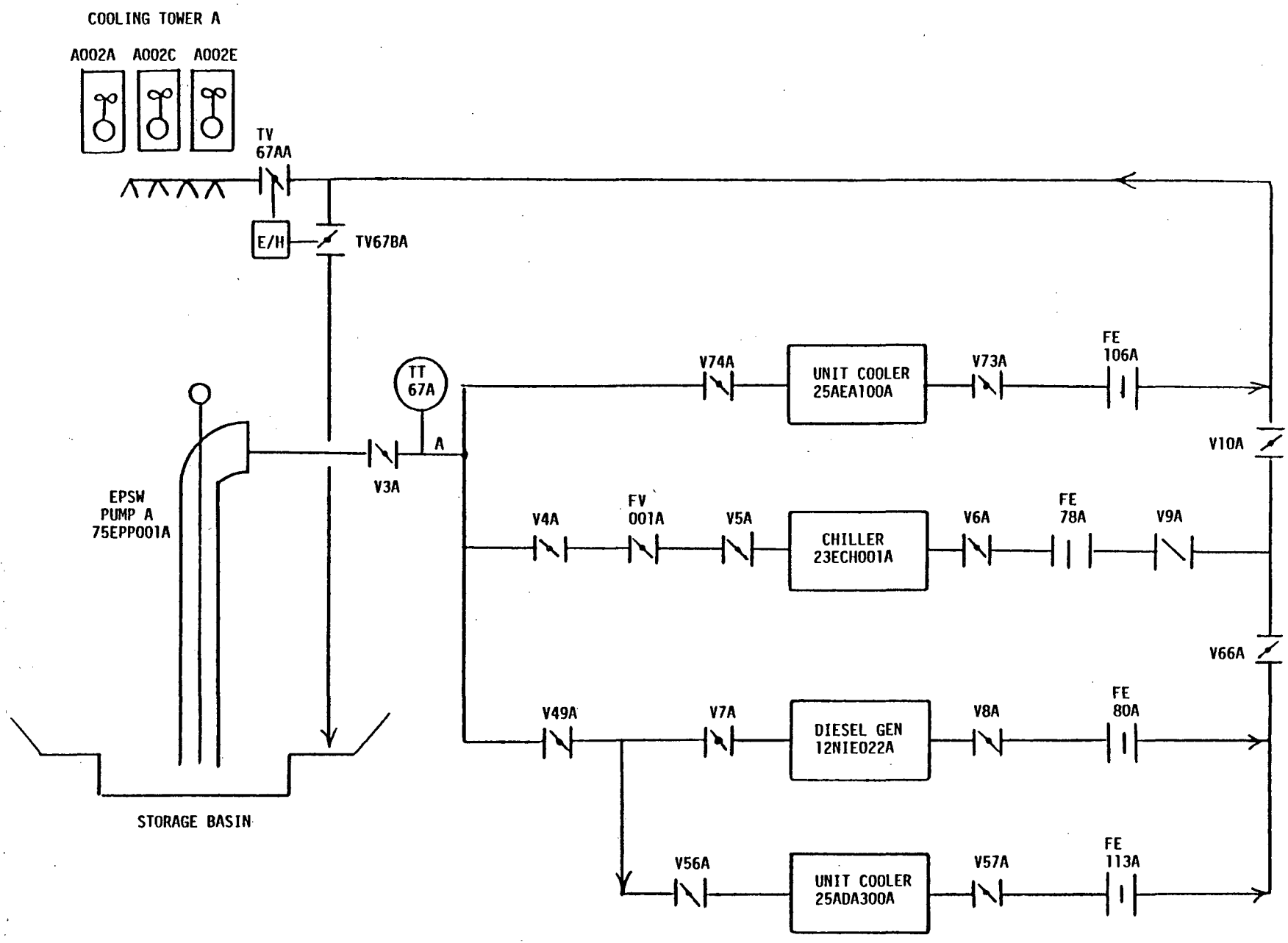


Figure A7-1. EPSW Loop A

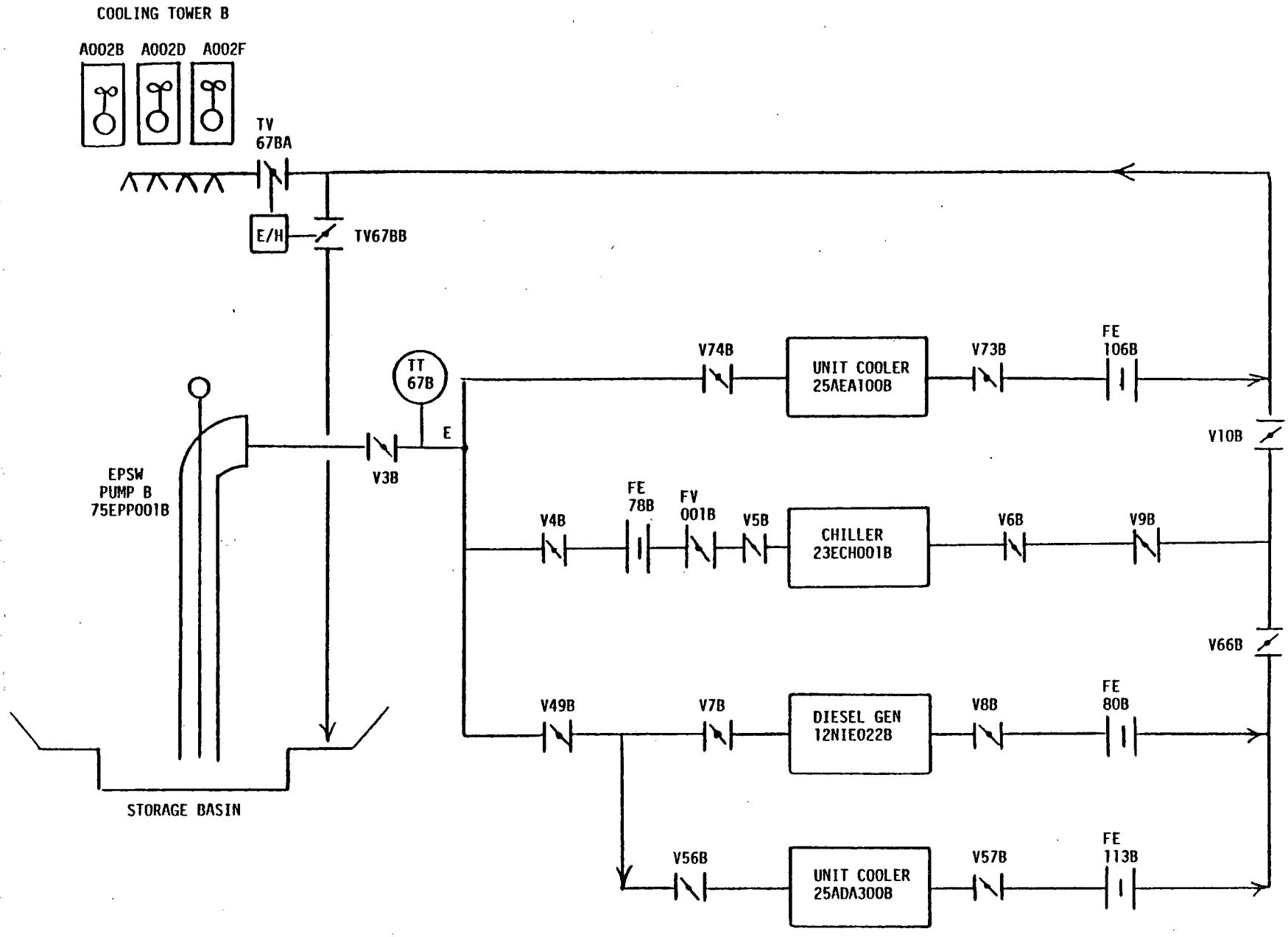


Figure A7-2. EPSW Loop B

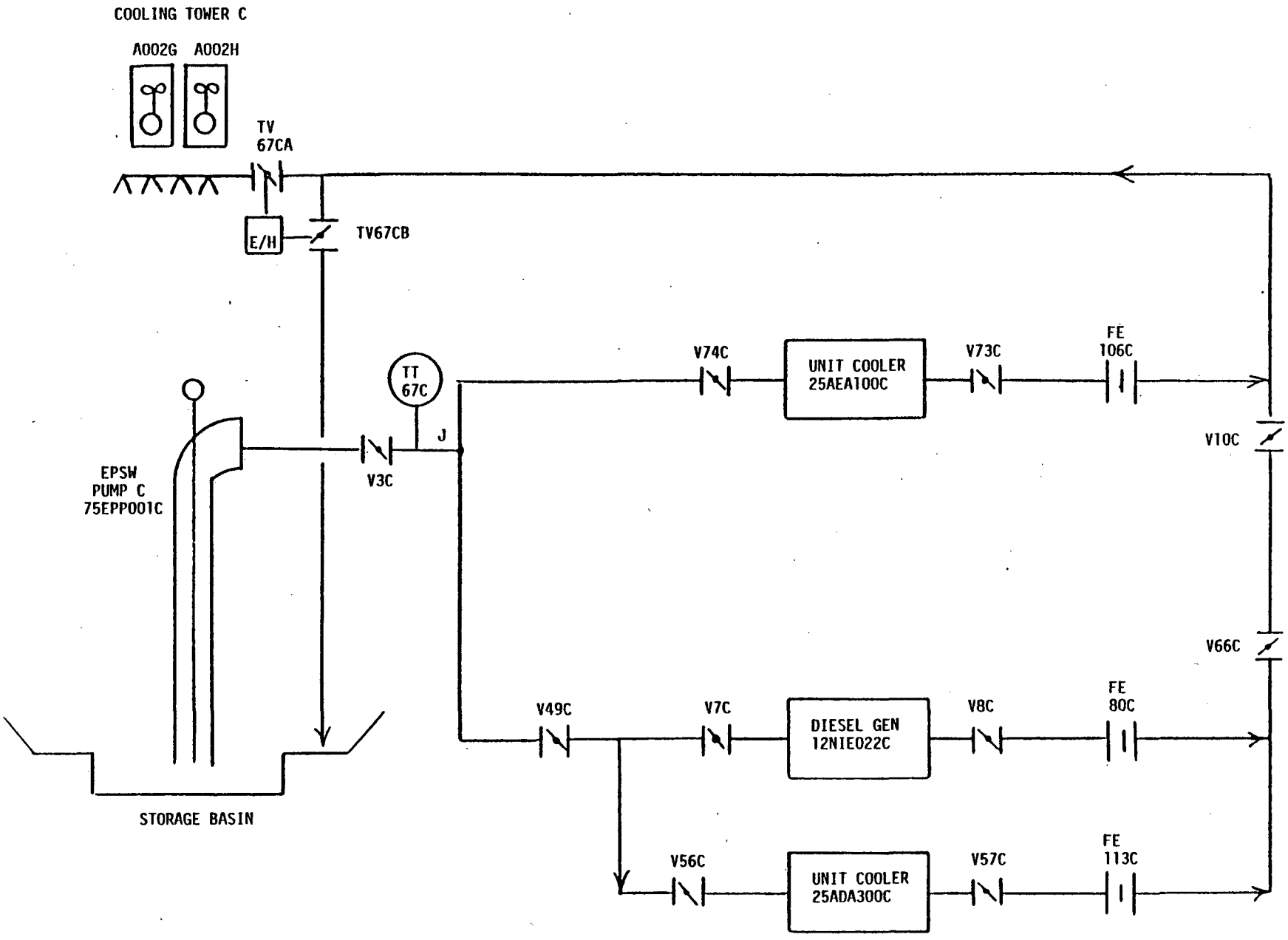


Figure A7-3. EPSW Loop C

Equipment	Equipment Number	Bus
EPSW pump A	75EPP001A	12NIE022A
EPSW pump B	75EPP001B	12NIE022B
EPSW pump C	75EPP001C	12NIE022C
Cooling tower A fans	75EPA002A,C,E	TBD
Cooling tower B fans	75EPA002B,D,F	TBD
Cooling tower C fans	75EPA002G,H	TBD

A7.1.3.2 Control power. Control power and electric power are served from a common bus.

A7.1.3.3 Hydraulic power. Hydraulic power is required for operation of the temperature control valves.

TV67AA, TV67AB
TV67BA, TV67BB
TV67CA, TV67CB

The hydraulic actuators for these valves are assumed to be self-contained units, requiring external connections only for electric power and control signals.

A7.1.3.4 Instrument air. Instrument air is required for the operation of the EPSW system instrumentation.

A7.1.3.5 Actuation systems. Normally the pump switches on the back panel of the control room are in the "auto" mode. In this mode the pumps automatically start upon receipt of a signal from either the emergency diesel generator loading program, or the loss of normal chilled water.

A7.1.4 Operation

A7.1.4.1 Normal operation. The EPSW system is only used during off-normal operations. During normal operation, this system is in standby.

A7.1.4.2 Emergency operation. Upon loss of normal chilled water or upon start of the standby diesel generators A or B, the EPSW pumps A or B and cooling tower A or B fans will start and provide cooling water at 90°F maximum. Loss of normal chilled water or low chilled water flow starts either emergency chiller A or B which sends a cascading start signal to the respective EPSW pump. Upon start of standby diesel generator C, EPSW pump C (75EPP001C) starts and provides cooling water to the diesel generator cooling jacket. All pumps are aligned to the "auto" position permitting an automatic start upon receipt of the cascading signal.

A7.1.5 Test and Technical Specification Requirements

A7.1.5.1 Testing. The EPSW system will be tested with the standby diesel generators. The EPSW pumps P001A, P001B, and P001C will be tested on a monthly basis to assure functional operation of the pumps. The temperature control valves TV67AA, TV67AB, TV67BA, TV67BB, TV67CA, and TV67CB will be tested quarterly to measure stroke time and to exercise the valves open. These valves will also be tested bi-annually to determine leakage.

A7.1.5.2 Technical specification. The reactor shall not be made critical unless the following conditions are satisfied:

1. The emergency cooling tower basin contains the required water capacity for 30 days uninterrupted operation of the emergency plant service water system,

2. The emergency cooling tower fans are operable,

3. The water temperature in the basin is within design limits,

4. The EPSW pumps are operable, and

5. The manual isolation valves provided, at all equipment served by the EPSW system, are in the "locked open" position.

A7.1.6 Maintenance

Maintenance is allowed, during power operation of the plant, on any components in the emergency plant service water system that will not remove more than two interrelated trains from service.

A7.2 SYSTEM MODEL

A7.2.1 Modeling Assumptions

A. This analysis assumes the design engineer has properly sized the piping system to meet cooling requirements of the loads.

B. The term "transfers" implies going from a desirable or required position to an undesirable or failed position.

C. Mission time for this analysis is 24 hrs. Because of the short mission time, repair of failed equipment is not accounted for in the model.

D. Pipe, pump, and valve ruptures are included in the model as a conglomerate event whose frequency will be calculated by summing the frequencies for rupture of each component as appropriate.

E. This analysis assumes the pump basin has a 30-day supply of cooling water at the start of the mission. This assumption comes from the technical specifications for operation of the reactor. Seismic events of sufficient magnitude to fail the pump basin will be modeled in the seismic analysis and are not considered here. The basin capacity is approximately 2,896,000 gallons.

F. Dynamic human errors such as (1) operator turns off the pump failing the system, and (2) operator inadvertently changes a valve position failing the EPSW System are treated in the model. Latent human errors such as operator failure to return the EPSW System components to operational readiness following maintenance activities are included in the model.

G. There are three pump headers "A," "B," and "C" that are assumed to be in the standby mode.

H. Two of the three fans in cooling towers "A" and "B" must operate to successfully cool the EPSW system discharge water. Both of the fans in cooling tower C must operate to successfully cool the EPSW System discharge water.

I. When the supply water temperature is above 55°F, the temperature control valves can fail the system by TV67AA transferring closed or TV67AB transferring open, similarly for TV67BA and TV67BB, also TV67CA and TV67CB.

J. This analysis assumes that failure of the unit coolers in each loop will not adversely affect system operation to the other loads in the loop. Loss of cooling water to these unit coolers was not modeled.

K. Latent human errors such as operator fails to return system components to operational readiness following test and maintenance are modeled.

L. Electrical supply to cooling tower fans was assumed to be the same as the pump in the same loop.

A7.2.2 Detailed Fault Tree Model (Figure A7-4)

A7.2.2.1 Success criteria. Success of the emergency plant service water system involves the distribution of cooling water to those loads described in Section 7.1.2. The operation of the individual loops are tied in the overall logic model to the equipment requiring their support.

Operation of loop A involves successful operation of pump 75EPP001A and operation of two out of three of the cooling tower fans in cooling tower

A. Operation of loop B involves successful operation of pump 75EPP001B and operation of two out of three of the cooling tower fans in cooling tower B. Operation of loop C involves successful operation of pump 75EPP001C and operation of both cooling tower fans in cooling tower C.

A7.2.2.2 Top events. The top event in this analysis is the loss of cooling water to the three diesel generator heat exchangers and the two emergency chillers.

A7.2.2.3 Transfer to other systems.

Equipment	Bus	Breaker	Gate
75EPP001A	12NIE022A	TBD*	E128
75EPP001B	12NIE022B	TBD	E228
75EPP001C	12NIE022C	TBD	E324
75EPA002A,C,E	TBD	TBD	E128
75EPA002B,D,F	TBD	TBD	E228
75EPA002G,H	TBD	TBD	E324

* To be determined (TBD). Information unavailable at baseline date of study.

A7.2.3 Modularized Fault Tree Model (Figure A7-5)

The EPSW system fault tree was simplified by combining the following pieces of equipment to form modules:

Module	Contents of Module
WMA1	Emergency chiller A and discharge valves
WMA1A	Chiller upstream valves
WMA2	Diesel generator A and upstream and discharge valves
WMA3,S,L	Cooling tower A and discharge valves
WMA4,S,L	Pump A and discharge valve
WMB1	Emergency chiller B and discharge valves
WMB1A	Chiller upstream valves
WMB2	Diesel generator B and upstream and discharge valves

WMB3	Cooling tower B and discharge valves
WMB4	Pump B and discharge valve
WMC1	Diesel generator C and upstream and discharge valves
WMC2	Cooling tower C and discharge valves
WMC3	Pump C and discharge valve

A7.3 FAILURE DATA

A7.3.1 Basic Event Data

Data were applied according to the rules and tables found in Sections 5 and 6.

A7.3.2 Module Data

Module data were calculated using the basic event data of Sections 5 and 6 and the detailed fault tree model. Results of these calculations can be found in Table A7-1.

A7.4 SYSTEM LEVEL RESULTS AND INSIGHTS

The failure of the EPSW system (Trains A and B) is dominated by the combination of loss of offsite power and short-term diesel failures. The predominant system component failure is the failure of the EPSW pumps to start. Train C failures are dominated by the failure of the Train C pump to start. Latent human errors are also a creditable source of failures for the EPSW System.

A7.5 REFERENCES

1. System Design Description Plant Service Water System, (SDD-75A), Rev. 23 (7/83).

2. DeMonbrun, J. R., and J. P. Belk, CRBR Water Systems Analysis 3066-75-52 Emergency Cooling Tower and Accessories, CRBR-2, ORGDP Report, September 1982.

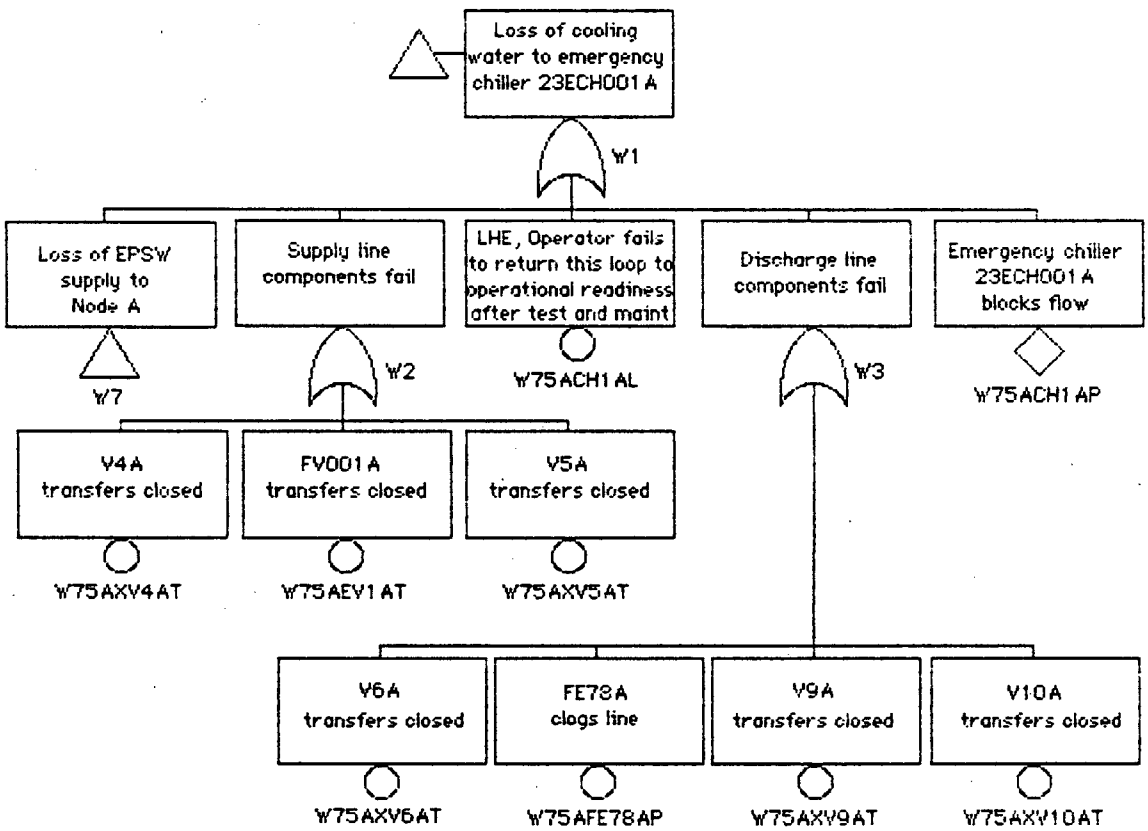
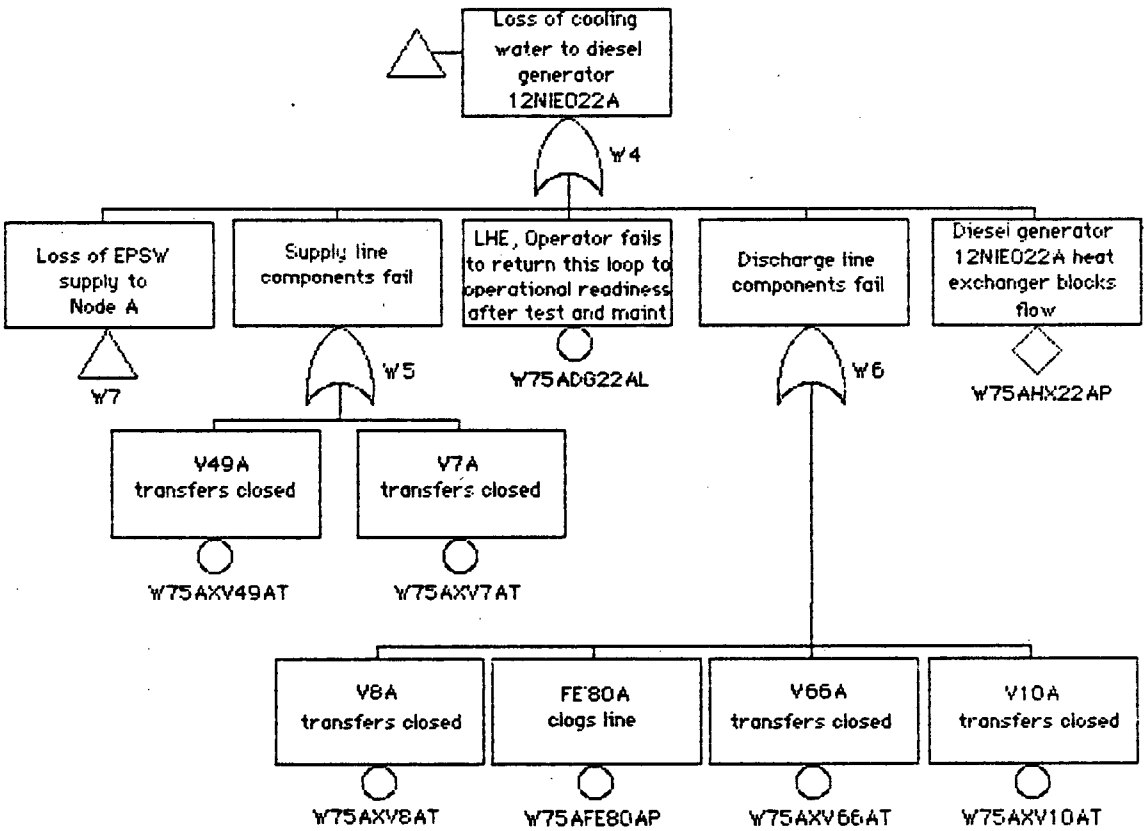
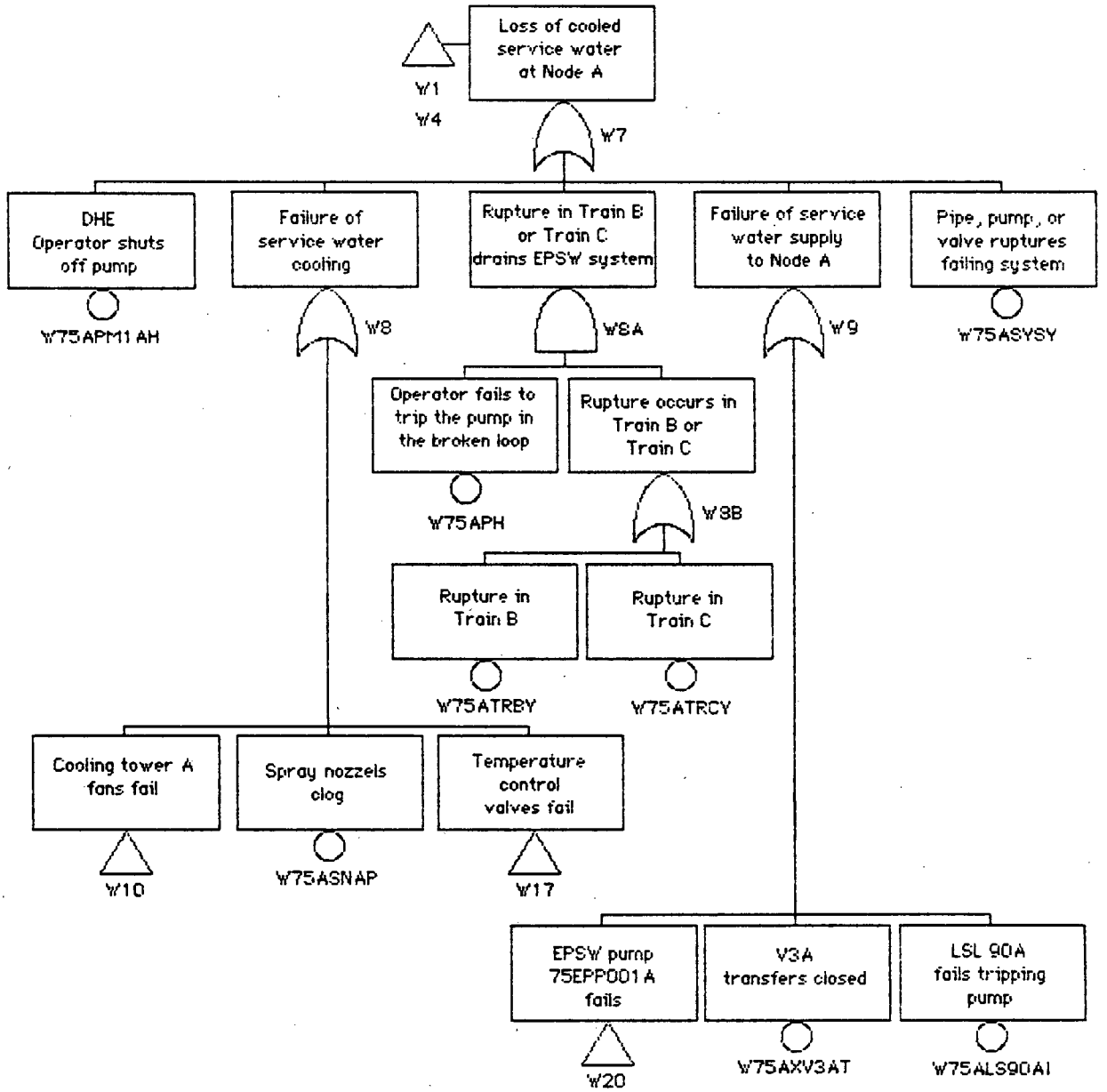


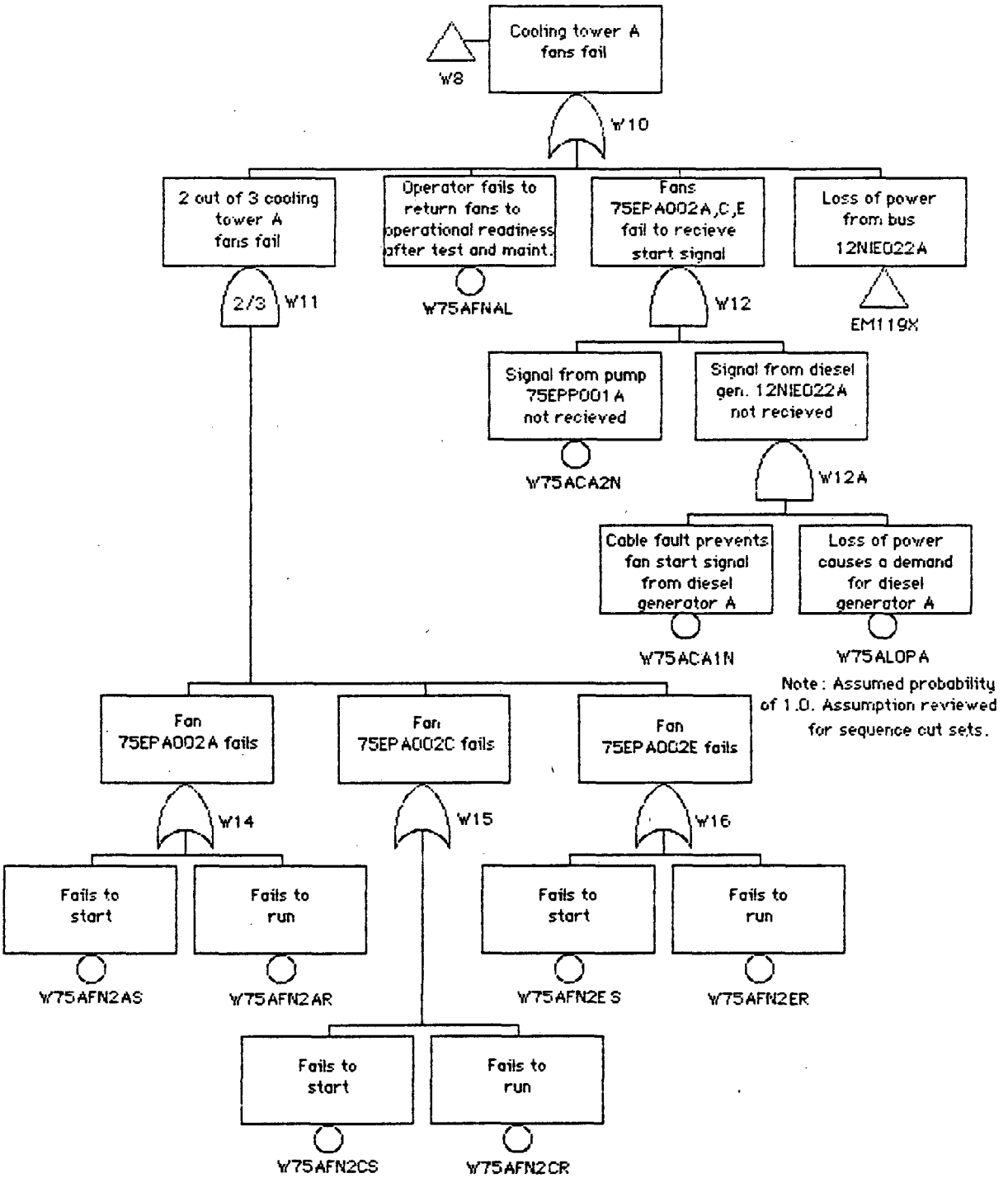
Figure A7-4. Emergency Plant Service Water System

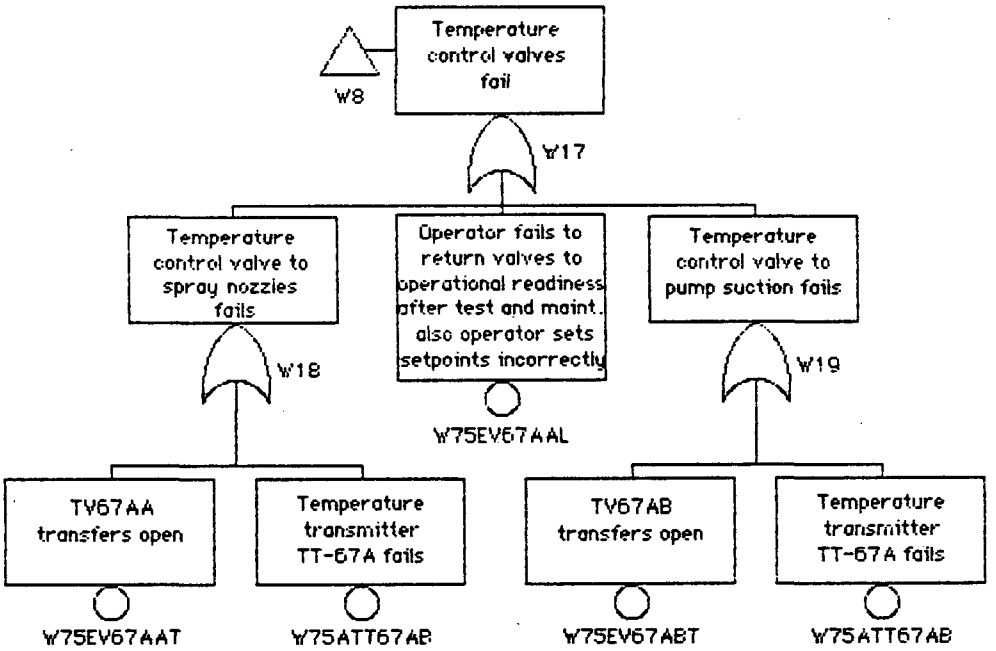
Detailed Fault Tree

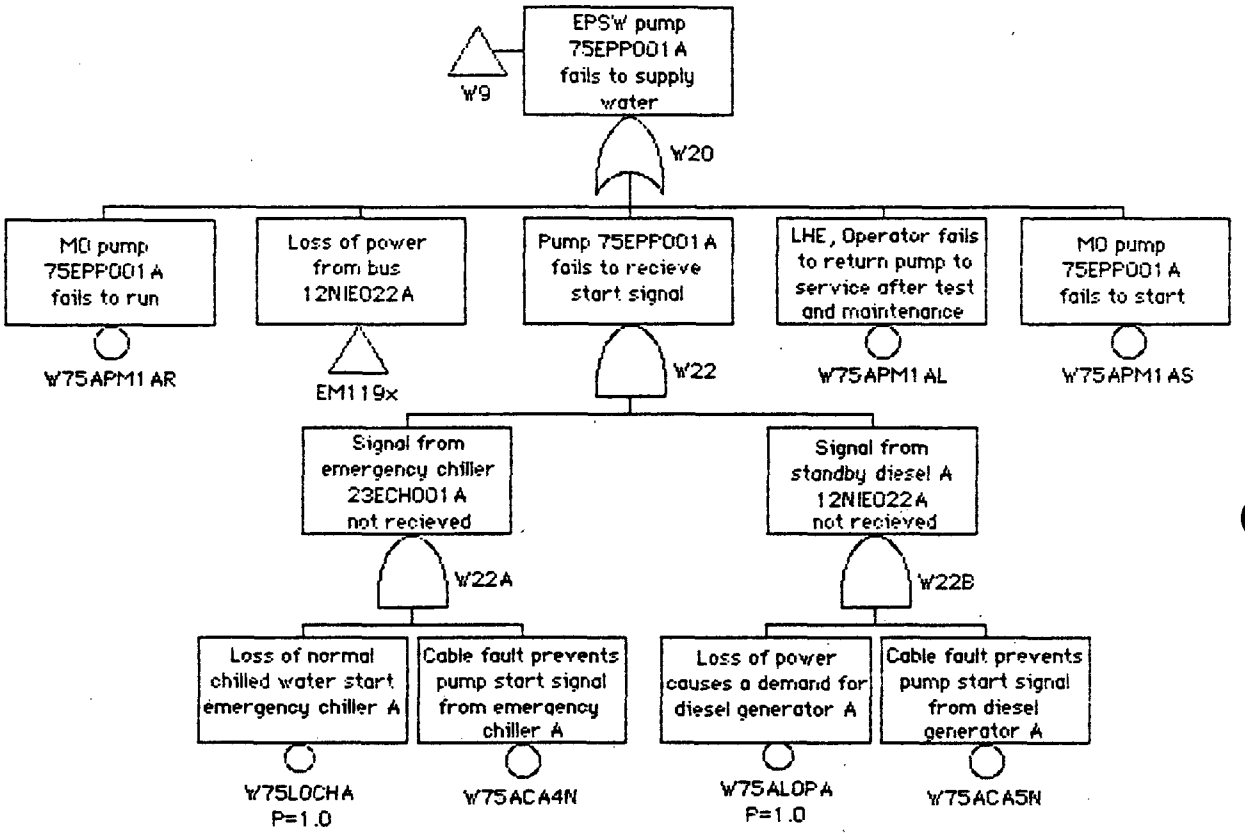
Page #1

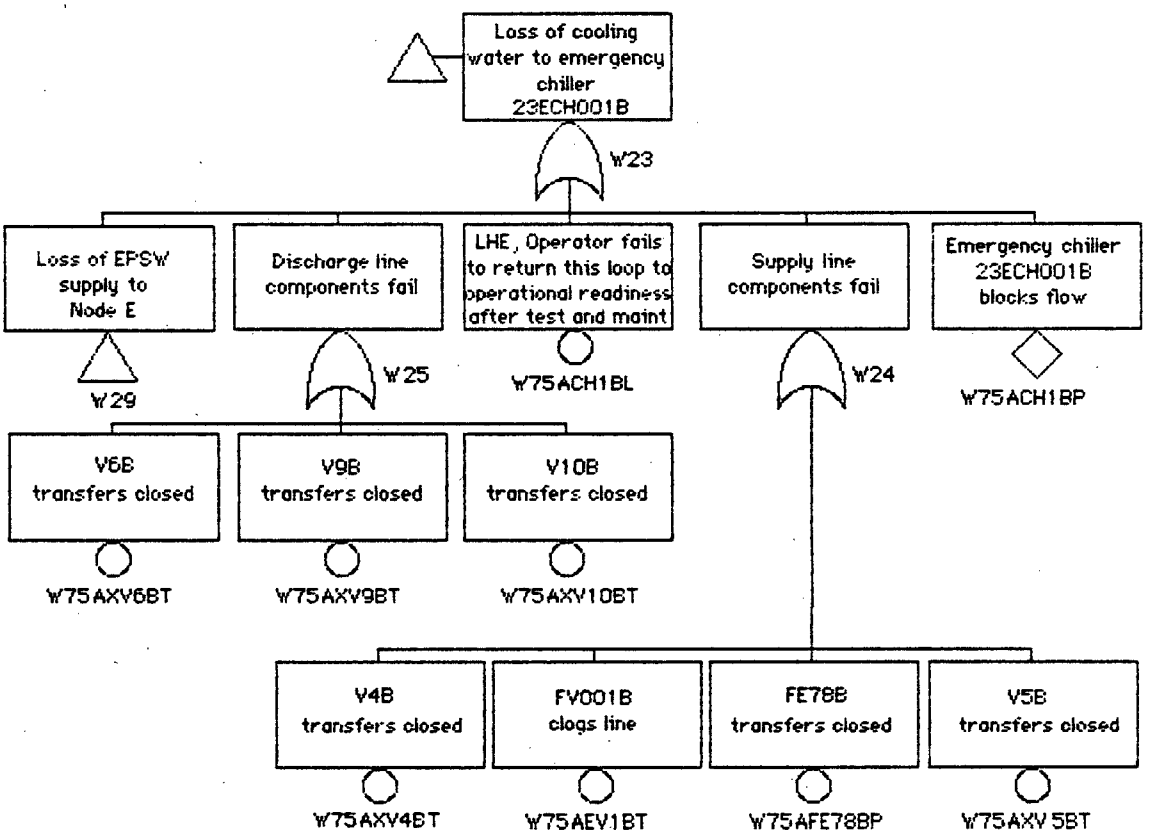

Loss of cooling water to diesel generator 12NIE022A
W4
Loss of EPSW supply to Node A
W7
Supply line components fail
W5
LHE, Operator fails to return this loop to operational readiness after test and maint
W75ADG22AL
Discharge line components fail
W6
Diesel generator 12NIE022A heat exchanger blocks flow
W75AHX22AP
V49A transfers closed
W75AXV49AT
V7A transfers closed
W75AXV7AT
V8A transfers closed
W75AXV8AT
FE80A clogs line
W75AFE80AP
V66A transfers closed
W75AXV66AT
V10A transfers closed
W75AXV10AT

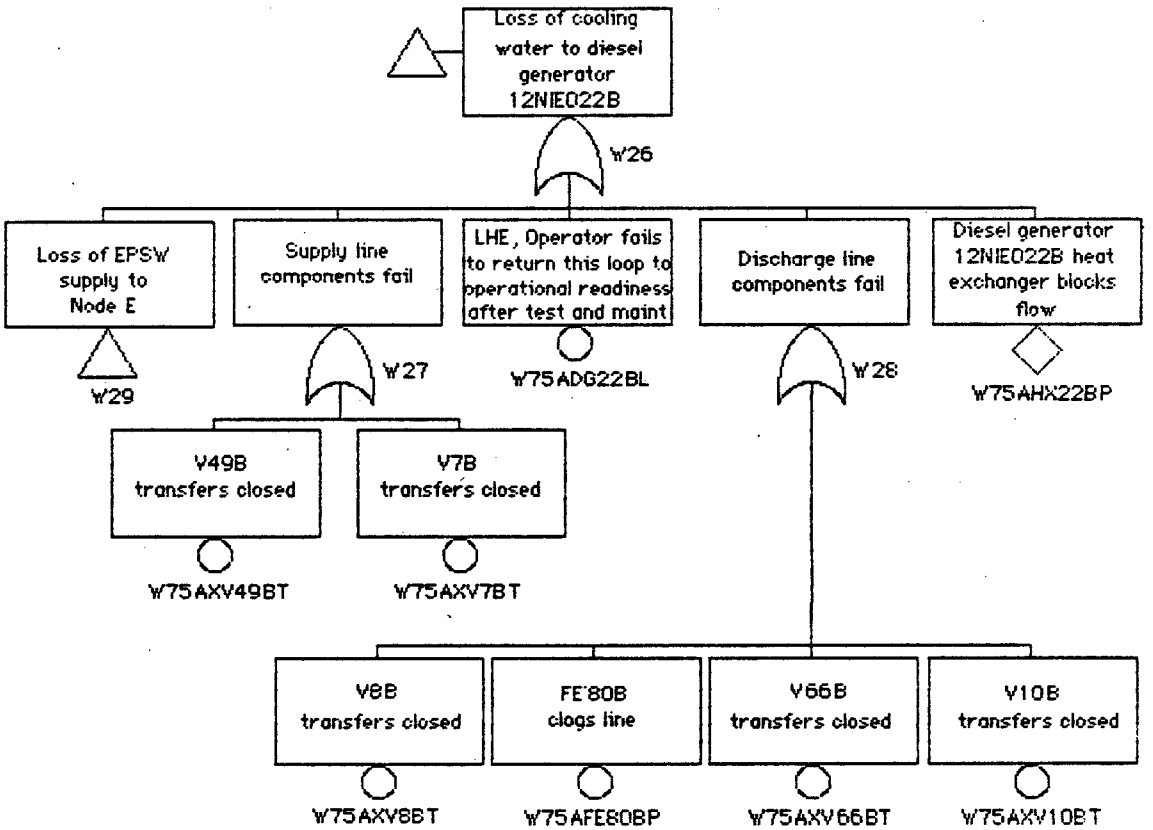

Loss of cooled service water at Node A
W1
W4
W7
DHE Operator shuts off pump
W75APM1AH
Failure of service water cooling
W8
Rupture in Train B or Train C drains EPSW system
W8A
Failure of service water supply to Node A
W9
Pipe, pump, or valve ruptures failing system
W75ASYSY
Operator fails to trip the pump in the broken loop
W75APH
Rupture occurs in Train B or Train C
W8B
Rupture in Train B
W75ATRBY
Rupture in Train C
W75ATRCY
Cooling tower A fans fail
W10
Spray nozzels clog
W75ASNAP
Temperature control valves fail
W17
EPSW pump 75EPP001A fails
W20
V3A transfers closed
W75AXV3AT
LSL 90A fails tripping pump
W75ALS90AI

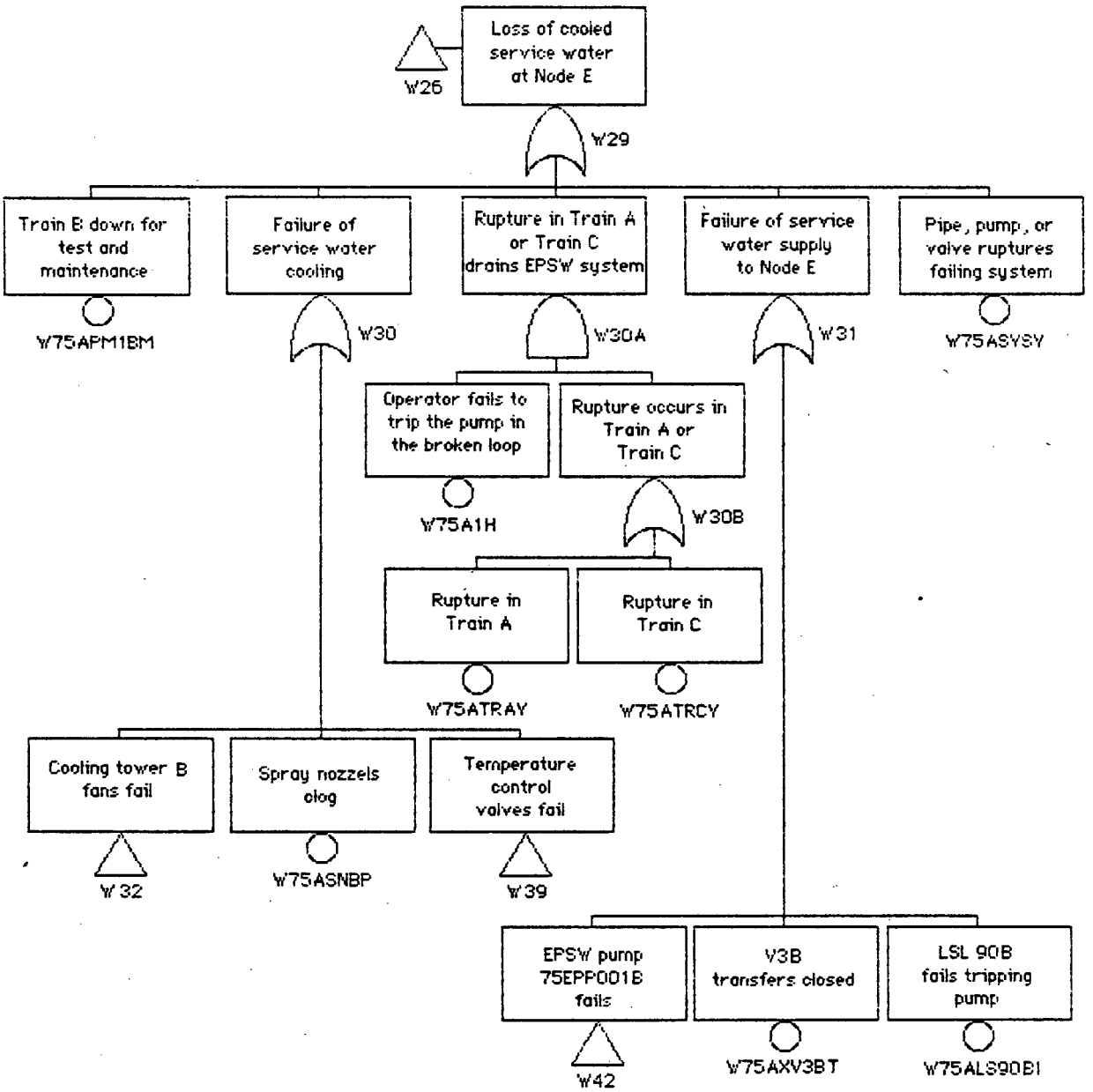

Cooling tower A fans fail
W8
W10
2 out of 3 cooling tower A fans fail
Operator fails to return fans to operational readiness after test and maint.
Fans 75EPA002A,C,E fail to recieve start signal
Loss of power from bus 12NIE022A
2/3
W11
W75AFNAL
W12
EM119X
Signal from pump 75EPP001A not recieved
Signal from diesel gen. 12NIE022A not recieved
W75ACA2N
W12A
Cable fault prevents fan start signal from diesel generator A
Loss of power causes a demand for diesel generator A
W75ACA1N
W75ALOPA
Note: Assumed probability of 1.0. Assumption reviewed for sequence cut sets.
Fan 75EPA002A fails
Fan 75EPA002C fails
Fan 75EPA002E fails
W14
W15
W16
Fails to start
Fails to run
Fails to start
Fails to run
W75AFN2AS
W75AFN2AR
W75AFN2ES
W75AFN2ER
Fails to start
Fails to run
W75AFN2CS
W75AFN2CR

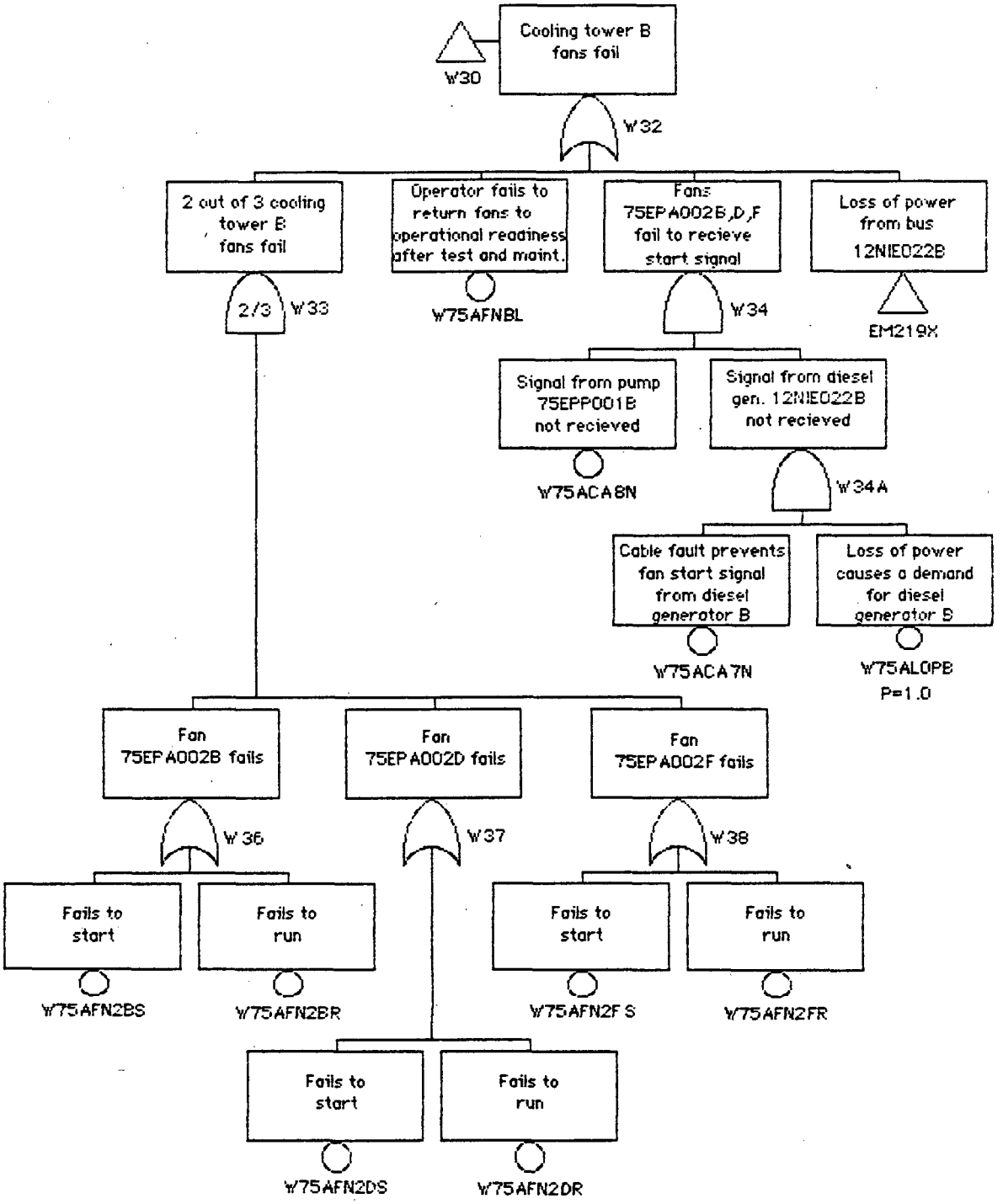

Temperature control valves fail
W8
W17
Temperature control valve to spray nozzles fails
Operator fails to return valves to operational readiness after test and maint. also operator sets setpoints incorrectly
Temperature control valve to pump suction fails
W18
W19
W75EV67AAL
TV67AA transfers open
Temperature transmitter TT-67A fails
TV67AB transfers open
Temperature transmitter TT-67A fails
W75EV67AAT
W75ATT67AB
W75EV67ABT
W75ATT67AB

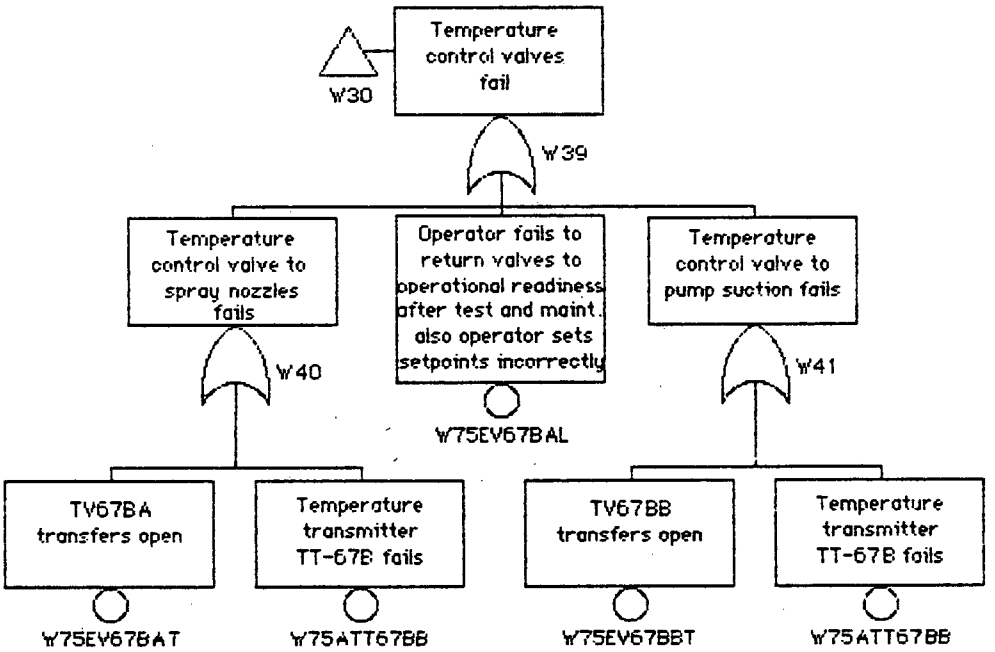

EPSW pump 75EPP001A fails to supply water
W9
W20
MO pump 75EPP001A fails to run
Loss of power from bus 12NIE022A
Pump 75EPP001A fails to recieve start signal
LHE, Operator fails to return pump to service after test and maintenance
MO pump 75EPP001A fails to start
W75APM1AR
EM119x
W22
W75APM1AL
W75APM1AS
Signal from emergency chiller 23ECH001A not recieved
Signal from standby diesel A 12NIE022A not recieved
W22A
W22B
Loss of normal chilled water start emergency chiller A
Cable fault prevents pump start signal from emergency chiller A
Loss of power causes a demand for diesel generator A
Cable fault prevents pump start signal from diesel generator A
W75LOCHA P=1.0
W75ACA4N
W75ALOPA P=1.0
W75ACA5N

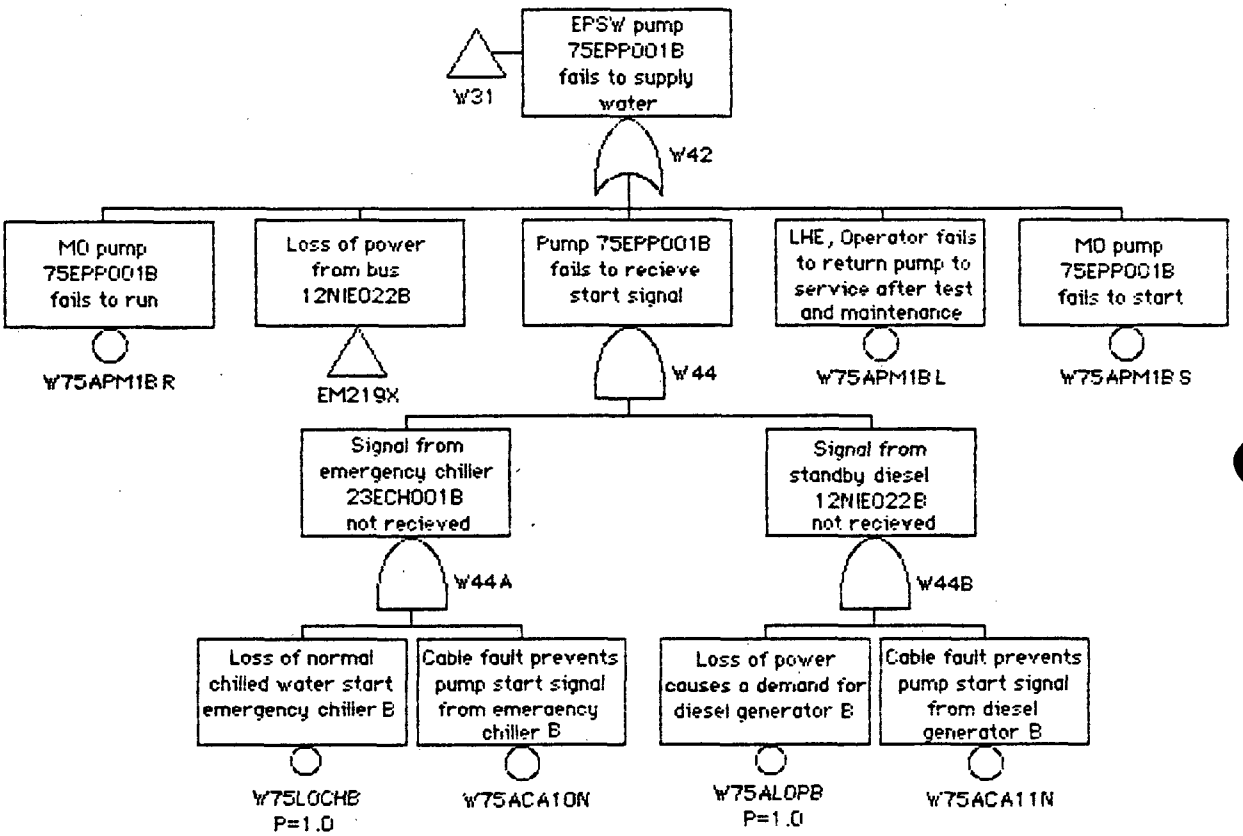

Loss of cooling water to emergency chiller 23ECH001B
W23
Loss of EPSW supply to Node E
Discharge line components fail
LHE, Operator fails to return this loop to operational readiness after test and maint
Supply line components fail
Emergency chiller 23ECH001B blocks flow
W29
W25
W75ACH1BL
W24
W75ACH1BP
V6B transfers closed
V9B transfers closed
V10B transfers closed
W75AXV6BT
W75AXV9BT
W75AXV10BT
V4B transfers closed
FV001B clogs line
FE78B transfers closed
V5B transfers closed
W75AXV4BT
W75AEV1BT
W75AFE78BP
W75AXV5BT

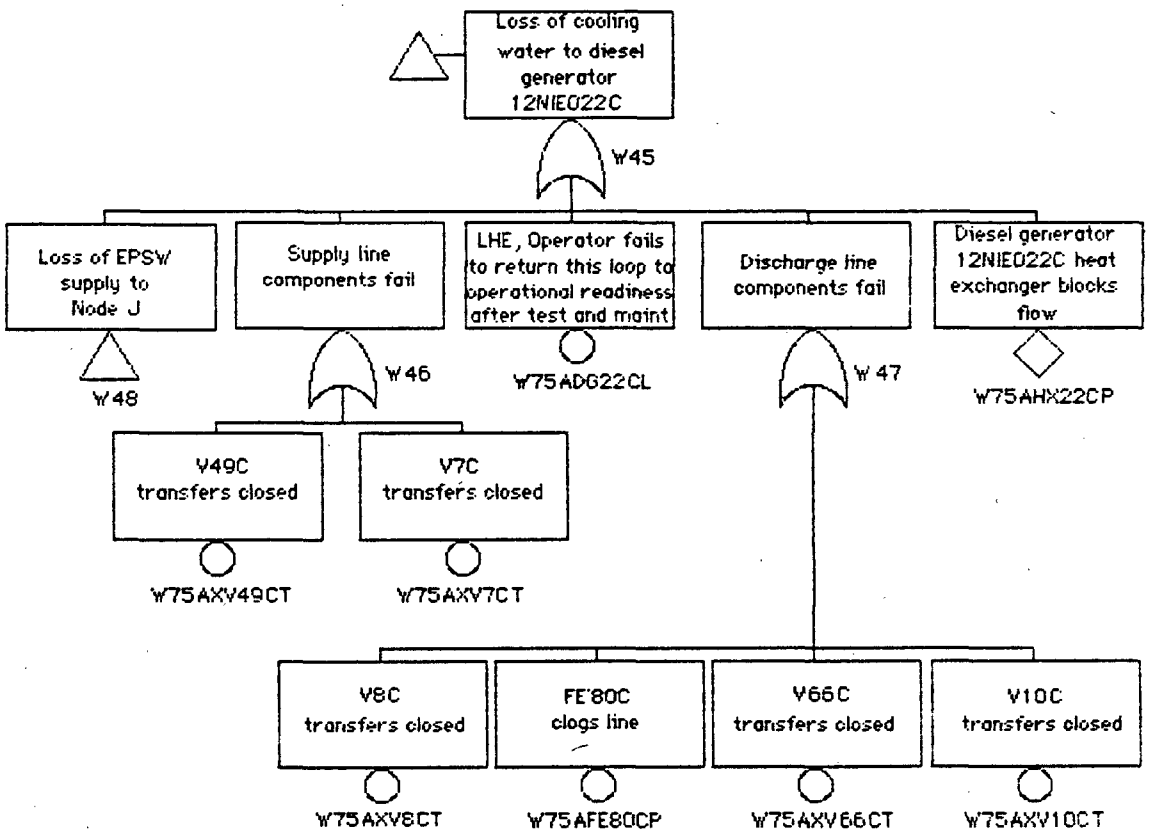

Loss of cooling water to diesel generator 12NIE022B
W26
Loss of EPSW supply to Node E
W29
Supply line components fail
W27
LHE, Operator fails to return this loop to operational readiness after test and maint
W75ADG22BL
Discharge line components fail
W28
Diesel generator 12NIE022B heat exchanger blocks flow
W75AHX22BP
V49B transfers closed
W75AXV49BT
V7B transfers closed
W75AXV7BT
V8B transfers closed
W75AXV8BT
FE80B clogs line
W75AFE80BP
V66B transfers closed
W75AXV66BT
V10B transfers closed
W75AXV10BT

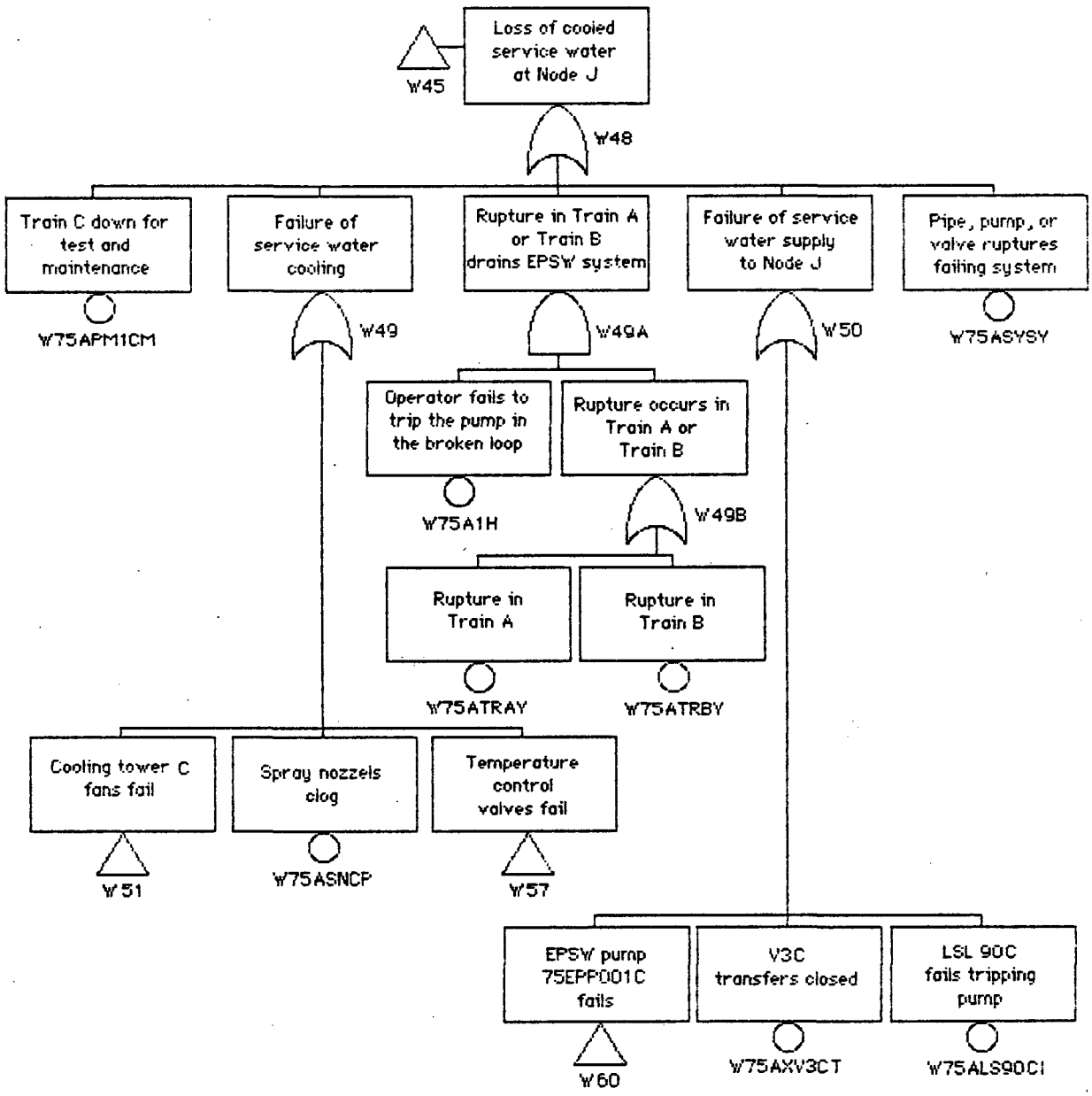

Loss of cooled service water at Node E
W26
W29
Train B down for test and maintenance
W75APM1BM
Failure of service water cooling
W30
Rupture in Train A or Train C drains EPSW system
W30A
Failure of service water supply to Node E
W31
Pipe, pump, or valve ruptures failing system
W75ASYSY
Operator fails to trip the pump in the broken loop
W75A1H
Rupture occurs in Train A or Train C
W30B
Rupture in Train A
W75ATRAY
Rupture in Train C
W75ATRCY
Cooling tower B fans fail
W32
Spray nozzels clog
W75ASNBP
Temperature control valves fail
W39
EPSW pump 75EPP001B fails
W42
V3B transfers closed
W75AXV3BT
LSL 90B fails tripping pump
W75ALS90BI

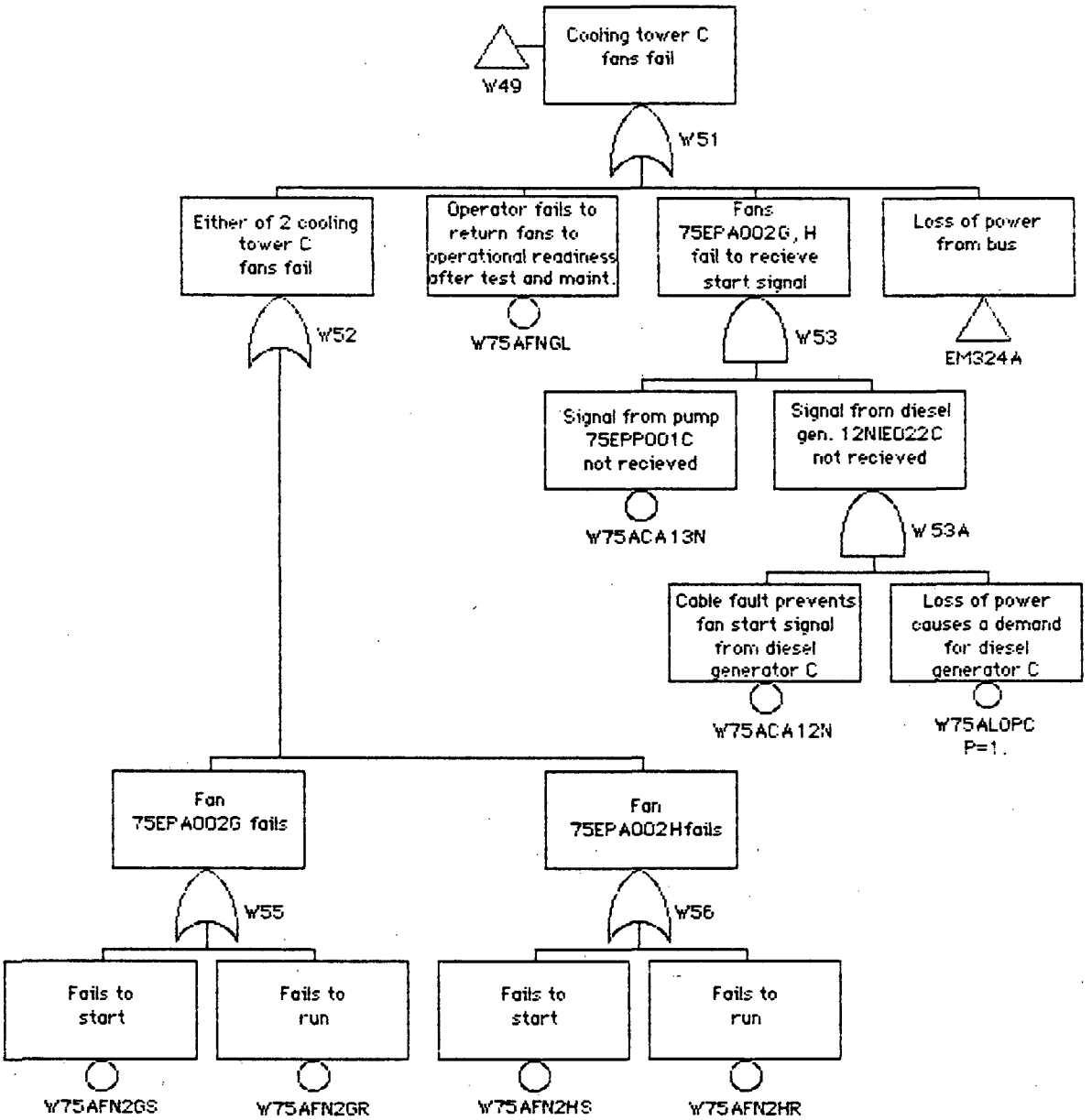

Cooling tower B fans fail
W30
W32
2 out of 3 cooling tower B fans fail
Operator fails to return fans to operational readiness after test and maint.
Fans 75EPA002B,D,F fail to recieve start signal
Loss of power from bus 12NIE022B
2/3
W33
W75AFNBL
W34
EM219X
Signal from pump 75EPP001B not recieved
Signal from diesel gen. 12NIE022B not recieved
W75ACA8N
W34A
Cable fault prevents fan start signal from diesel generator B
Loss of power causes a demand for diesel generator B
W75ACA7N
W75ALOPB
P=1.0
Fan 75EPA002B fails
Fan 75EPA002D fails
Fan 75EPA002F fails
W36
W37
W38
Fails to start
Fails to run
Fails to start
Fails to run
W75AFN2BS
W75AFN2BR
W75AFN2FS
W75AFN2FR
Fails to start
Fails to run
W75AFN2DS
W75AFN2DR

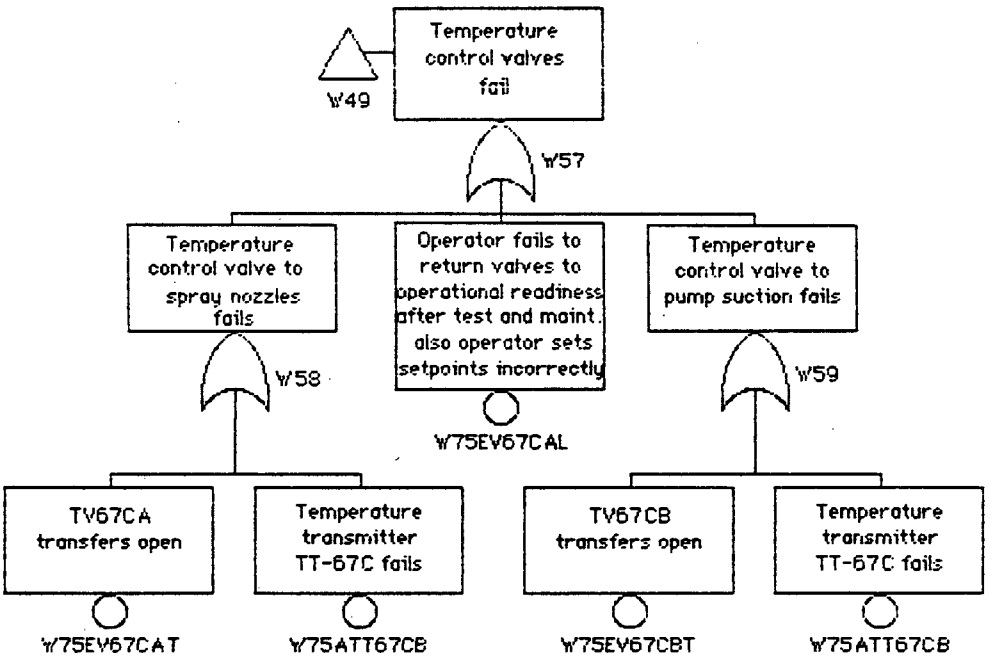

Temperature control valves fail
W30
W39
Temperature control valve to spray nozzles fails
Operator fails to return valves to operational readiness after test and maint. also operator sets setpoints incorrectly
Temperature control valve to pump suction fails
W40
W41
W75EV67BAL
TV67BA transfers open
Temperature transmitter TT-67B fails
TV67BB transfers open
Temperature transmitter TT-67B fails
W75EV67BAT
W75ATT67BB
W75EV67BBT
W75ATT67BB

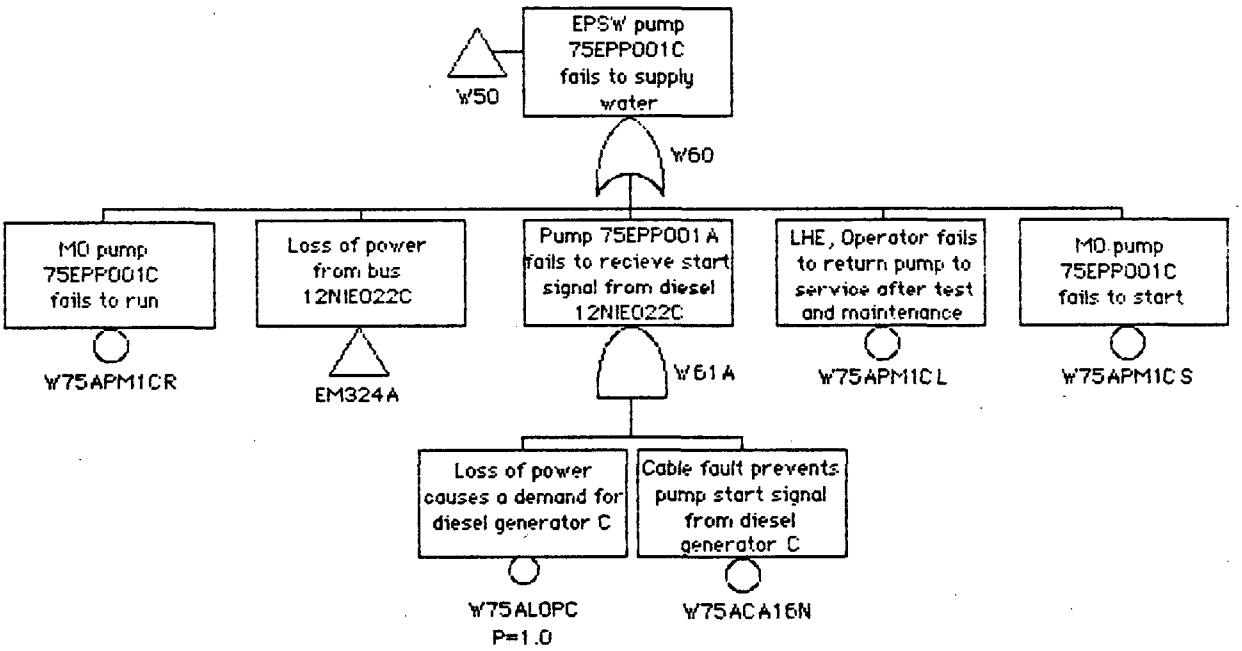

EPSW pump 75EPP001B fails to supply water
W31
W42
MO pump 75EPP001B fails to run
Loss of power from bus 12NIE022B
Pump 75EPP001B fails to recieve start signal
LHE, Operator fails to return pump to service after test and maintenance
MO pump 75EPP001B fails to start
W75APM1B R
EM219X
W44
W75APM1B L
W75APM1B S
Signal from emergency chiller 23ECH001B not recieved
Signal from standby diesel 12NIE022B not recieved
W44A
W44B
Loss of normal chilled water start emergency chiller B
Cable fault prevents pump start signal from emergency chiller B
Loss of power causes a demand for diesel generator B
Cable fault prevents pump start signal from diesel generator B
W75LOCHB P=1.0
W75ACA10N
W75ALOPB P=1.0
W75ACA11N


Loss of cooling water to diesel generator 12NIE022C
W45
Loss of EPSW supply to Node J
W48
Supply line components fail
W46
LHE, Operator fails to return this loop to operational readiness after test and maint
W75ADG22CL
Discharge line components fail
W47
Diesel generator 12NIE022C heat exchanger blocks flow
W75AHX22CP
V49C transfers closed
W75AXV49CT
V7C transfers closed
W75AXV7CT
V8C transfers closed
W75AXV8CT
FE80C clogs line
W75AFE80CP
V66C transfers closed
W75AXV66CT
V10C transfers closed
W75AXV10CT


Loss of cooled service water at Node J
W45
W48
Train C down for test and maintenance
W75APM1CM
Failure of service water cooling
W49
Rupture in Train A or Train B drains EPSW system
W49A
Failure of service water supply to Node J
W50
Pipe, pump, or valve ruptures failing system
W75ASYSY
Operator fails to trip the pump in the broken loop
W75A1H
Rupture occurs in Train A or Train B
W49B
Rupture in Train A
W75ATRAY
Rupture in Train B
W75ATRBY
Cooling tower C fans fail
W51
Spray nozzels clog
W75ASNCP
Temperature control valves fail
W57
EPSW pump 75EPP001C fails
W60
V3C transfers closed
W75AXV3CT
LSL 90C fails tripping pump
W75ALS90CI


Cooling tower C fans fail
W49
W51
Either of 2 cooling tower C fans fail
Operator fails to return fans to operational readiness after test and maint.
Fans 75EPA002G, H fail to recieve start signal
Loss of power from bus
W52
W75AFNGL
W53
EM324A
Signal from pump 75EPP001C not recieved
Signal from diesel gen. 12NIE022C not recieved
W75ACA13N
W53A
Cable fault prevents fan start signal from diesel generator C
Loss of power causes a demand for diesel generator C
W75ACA12N
W75ALOPC P=1.
Fan 75EPA002G fails
Fan 75EPA002H fails
W55
W56
Fails to start
Fails to run
Fails to start
Fails to run
W75AFN2GS
W75AFN2GR
W75AFN2HS
W75AFN2HR


Temperature control valves fail
W49
W57
Temperature control valve to spray nozzles fails
Operator fails to return valves to operational readiness after test and maint. also operator sets setpoints incorrectly
Temperature control valve to pump suction fails
W58
W59
W75EV67CAL
TV67CA transfers open
Temperature transmitter TT-67C fails
TV67CB transfers open
Temperature transmitter TT-67C fails
W75EV67CAT
W75ATT67CB
W75EV67CBT
W75ATT67CB


EPSW pump 75EPP001C fails to supply water
W50
W60
MO pump 75EPP001C fails to run
W75APM1CR
Loss of power from bus 12NIE022C
EM324A
Pump 75EPP001A fails to recieve start signal from diesel 12NIE022C
W61A
LHE, Operator fails to return pump to service after test and maintenance
W75APM1CL
MO pump 75EPP001C fails to start
W75APM1CS
Loss of power causes a demand for diesel generator C
W75ALOPC
P=1.0
Cable fault prevents pump start signal from diesel generator C
W75ACA16N

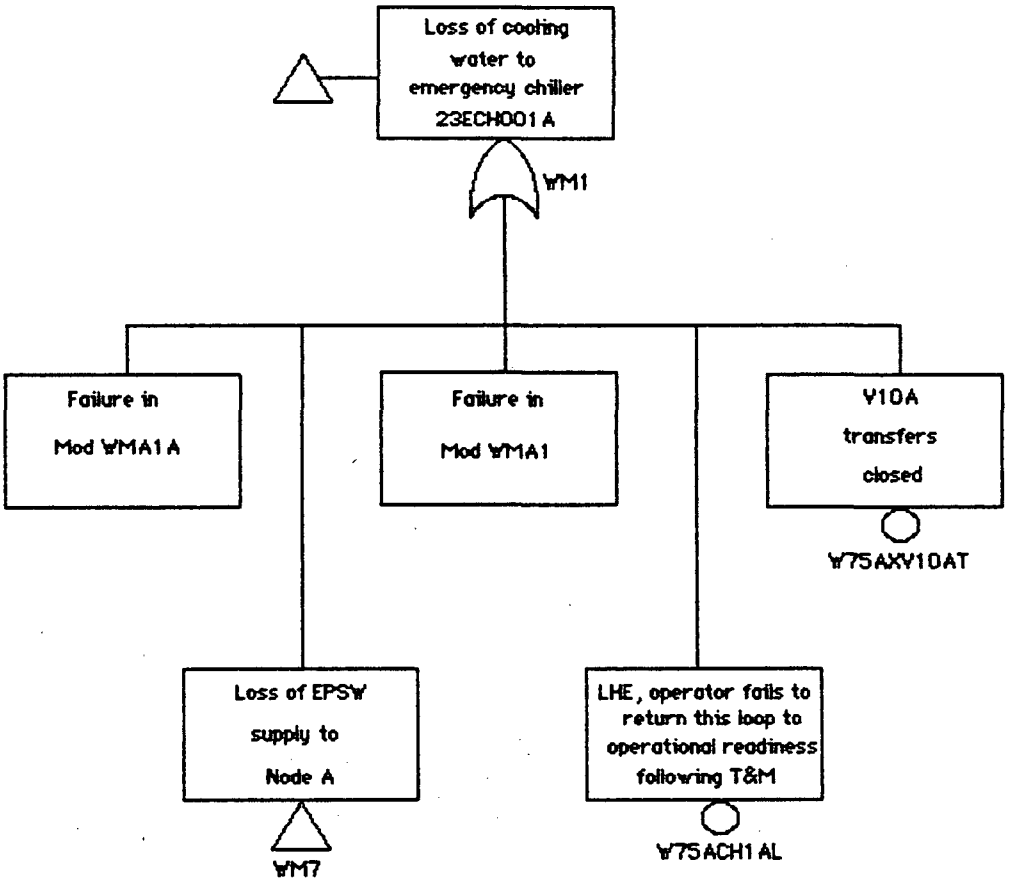
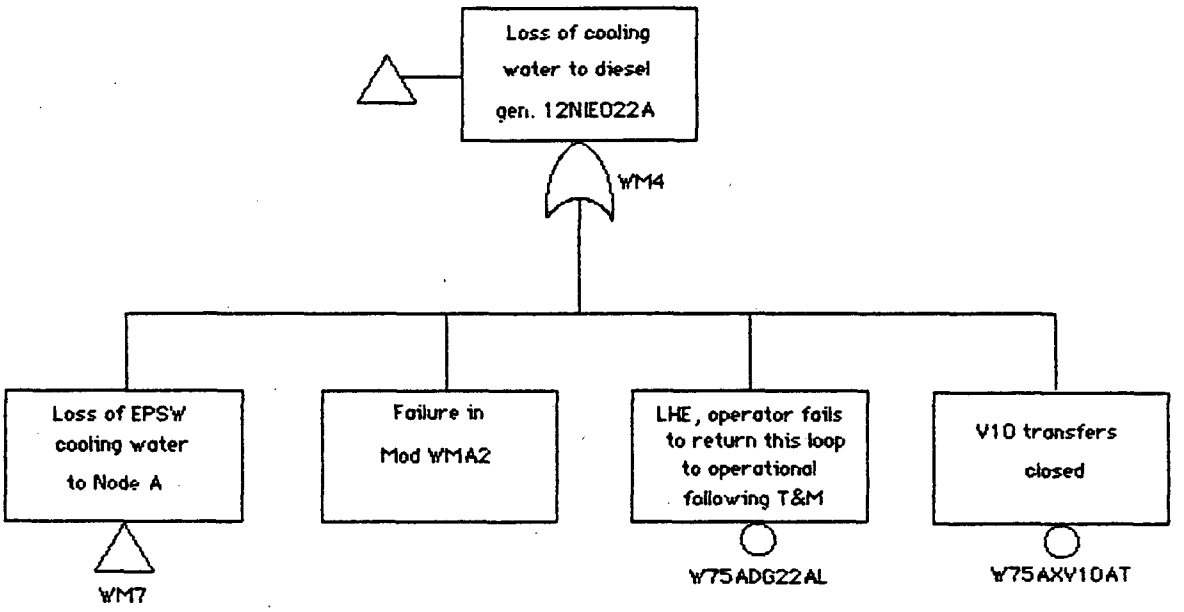
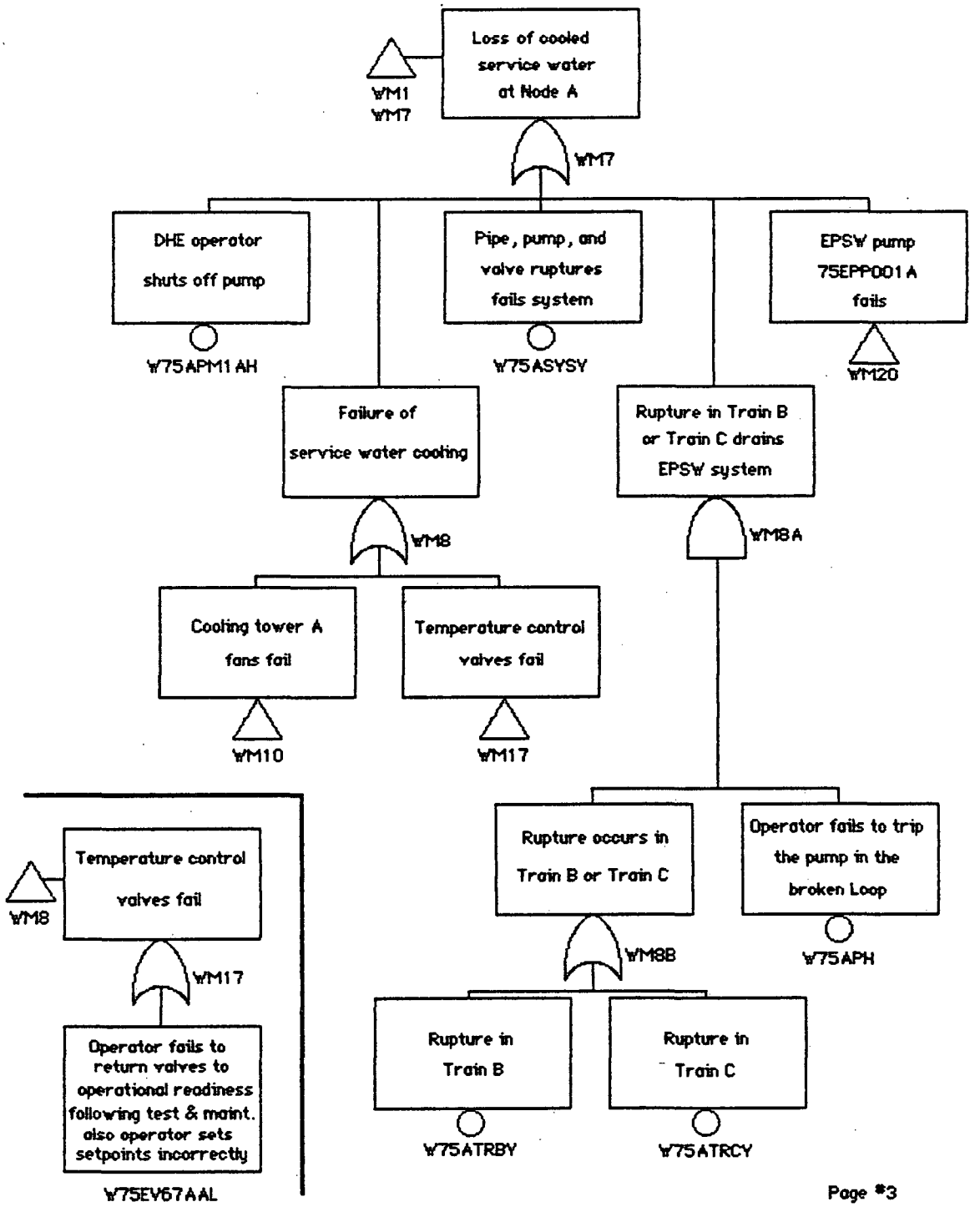


Figure A7-5. Emergency Plant Service Water System

Modularized Fault Tree


Loss of cooling water to diesel gen. 12NIE022A
WM4
Loss of EPSW cooling water to Node A
WM7
Failure in Mod WM42
LHE, operator fails to return this loop to operational following T&M
W75ADG22AL
V10 transfers closed
W75AXV10AT


Loss of cooled service water at Node A
WM1
WM7
WM7
DHE operator shuts off pump
W75APM1AH
Pipe, pump, and valve ruptures fails system
W75ASYSY
EPSW pump 75EPP001A fails
WM20
Failure of service water cooling
WM8
Rupture in Train B or Train C drains EPSW system
WM8A
Cooling tower A fans fail
WM10
Temperature control valves fail
WM17
Temperature control valves fail
WM8
WM17
Operator fails to return valves to operational readiness following test & maint. also operator sets setpoints incorrectly
W75EV67AAL
Rupture occurs in Train B or Train C
WM8B
Operator fails to trip the pump in the broken Loop
W75APH
Rupture in Train B
W75ATRBY
Rupture in Train C
W75ATRCY

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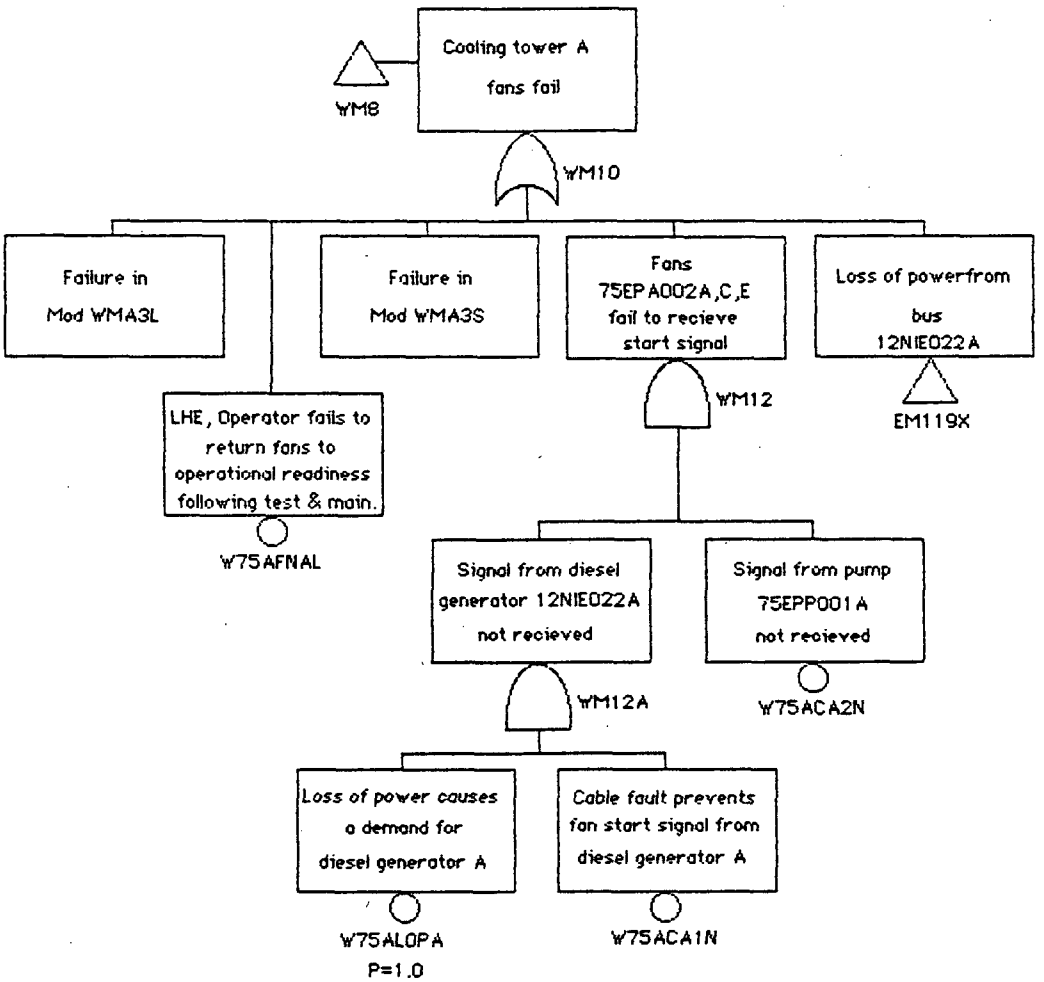
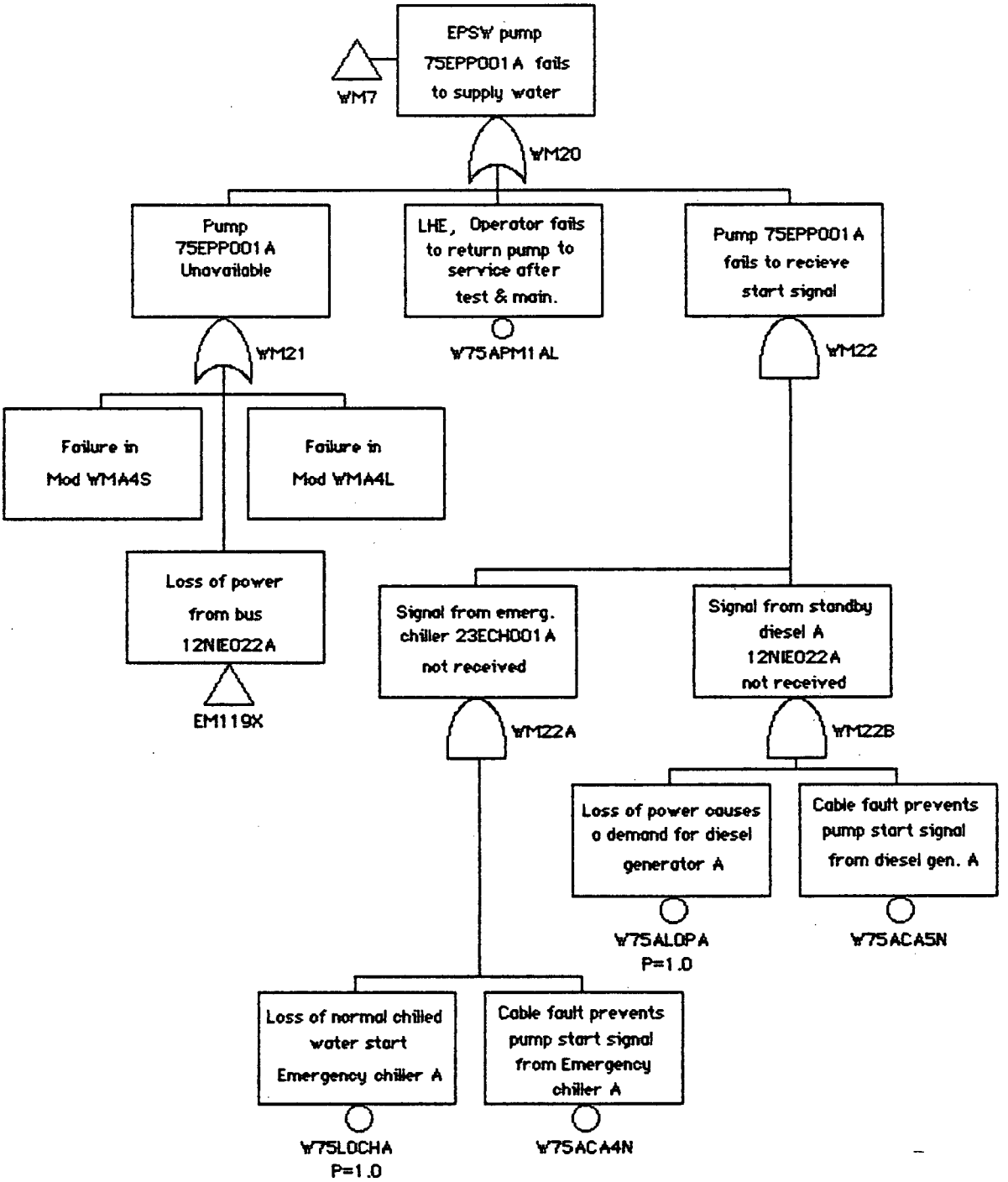

Cooling tower A
fans fail
WM8
WM10
Failure in
Mod WMA3L
Failure in
Mod WMA3S
Fans
75EPA002A,C,E
fail to recieve
start signal
Loss of powerfrom
bus
12NIE022A
EM119X
WM12
LHE, Operator fails to
return fans to
operational readiness
following test & main.
W75AFNAL
Signal from diesel
generator 12NIE022A
not recieved
Signal from pump
75EPP001A
not recieved
W75ACA2N
WM12A
Loss of power causes
a demand for
diesel generator A
Cable fault prevents
fan start signal from
diesel generator A
W75ALOPA
P=1.0
W75ACA1N


EPSW pump 75EPP001A fails to supply water
WM7
WM20
Pump 75EPP001A Unavailable
LHE, Operator fails to return pump to service after test & main.
Pump 75EPP001A fails to recieve start signal
W75APM11AL
WM21
WM22
Failure in Mod WMA4S
Failure in Mod WMA4L
Loss of power from bus 12NIE022A
EM119X
Signal from emerg. chiller 23ECH001A not received
Signal from standby diesel A 12NIE022A not received
WM22A
WM22B
Loss of power causes a demand for diesel generator A
Cable fault prevents pump start signal from diesel gen. A
W75ALOPA
P=1.0
W75ACA5N
Loss of normal chilled water start Emergency chiller A
Cable fault prevents pump start signal from Emergency chiller A
W75LOCHA
P=1.0
W75ACA4N

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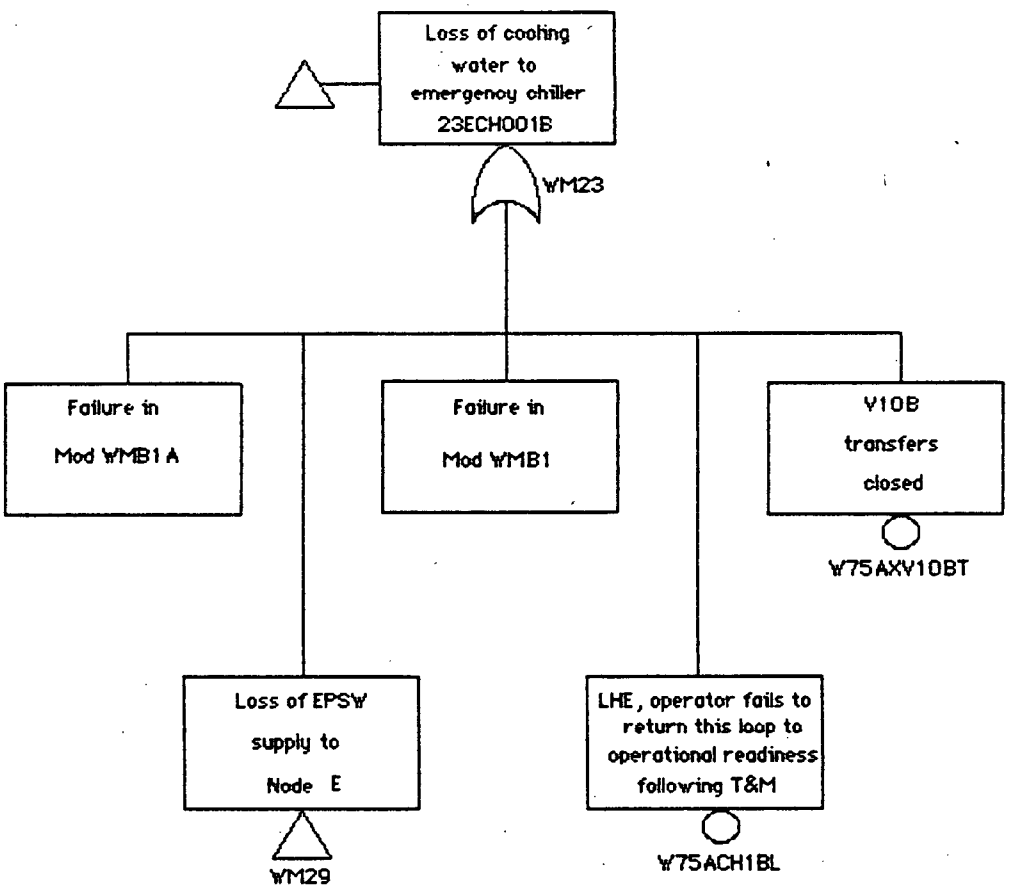
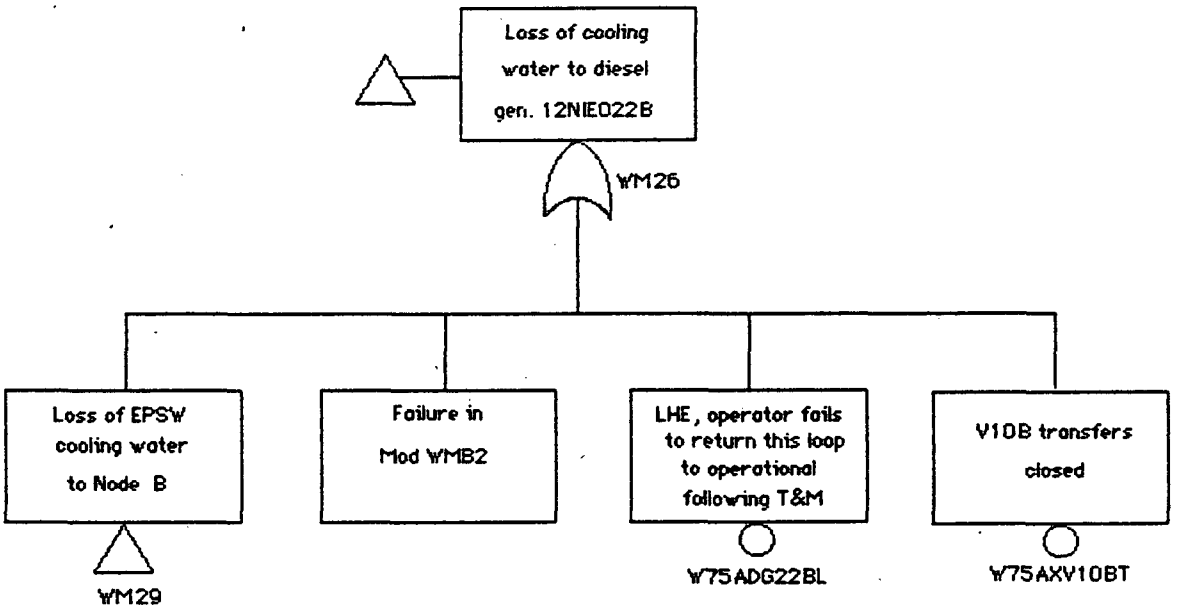
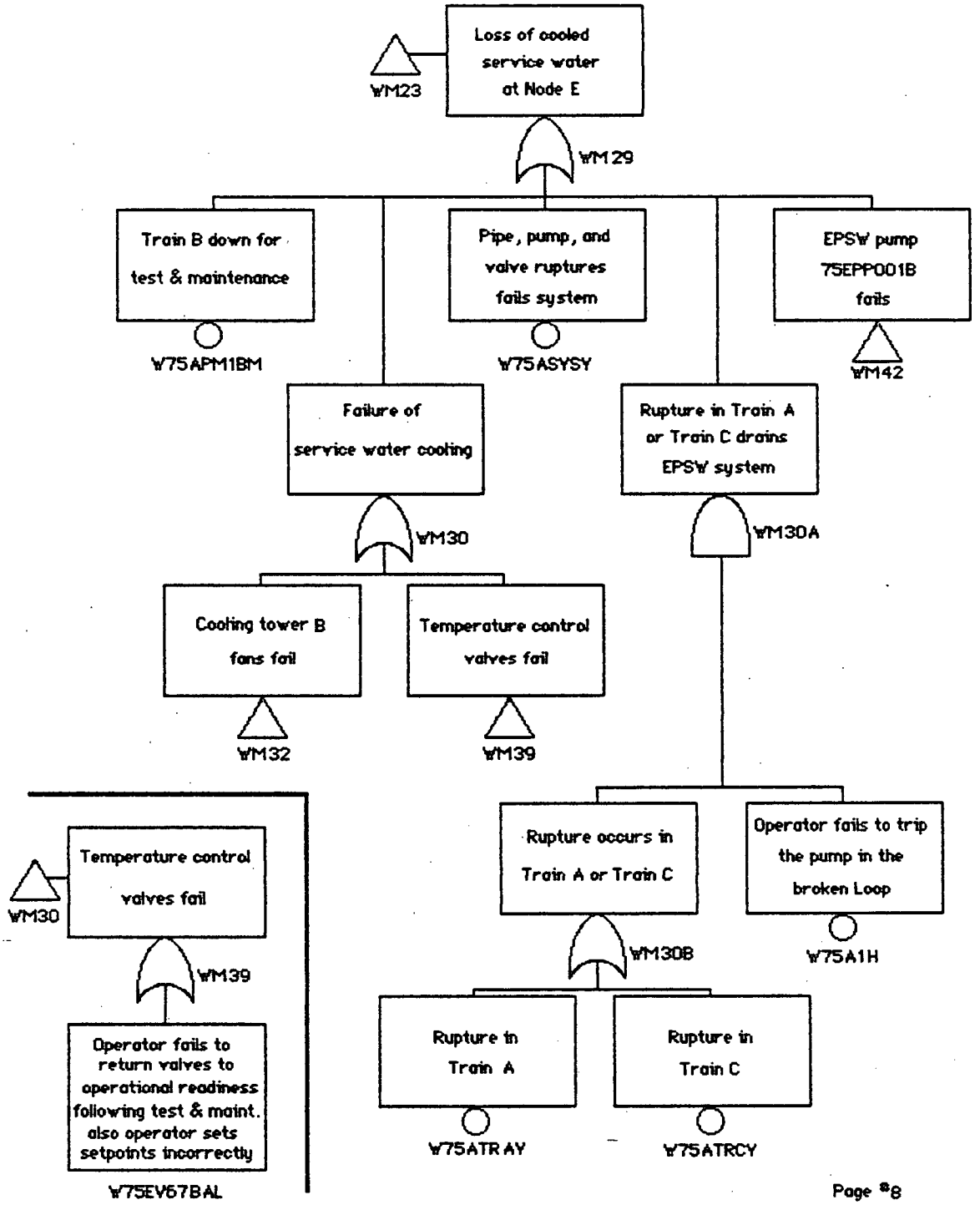

Loss of cooling water to emergency chiller 23ECH001B
WM23
Failure in Mod WMB1A
Failure in Mod WMB1
V10B transfers closed
W75AXV10BT
Loss of EPSW supply to Node E
WM29
LHE, operator fails to return this loop to operational readiness following T&M
W75ACH1BL


Loss of cooling water to diesel gen. 12NIE022B
WM26
Loss of EPSW cooling water to Node B
WM29
Failure in Mod WMB2
LHE, operator fails to return this loop to operational following T&M
W75ADG22BL
V10B transfers closed
W75AXV10BT


Loss of cooled service water at Node E
WM23
WM29
Train B down for test & maintenance
W75APM1BM
Pipe, pump, and valve ruptures fails system
W75ASYSY
EPSW pump 75EPP001B fails
WM42
Failure of service water cooling
WM30
Rupture in Train A or Train C drains EPSW system
WM30A
Cooling tower B fans fail
WM32
Temperature control valves fail
WM39
Temperature control valves fail
WM30
WM39
Operator fails to return valves to operational readiness following test & maint. also operator sets setpoints incorrectly
W75EV67BAL
Rupture occurs in Train A or Train C
WM30B
Operator fails to trip the pump in the broken Loop
W75A1H
Rupture in Train A
W75ATRAY
Rupture in Train C
W75ATRCY
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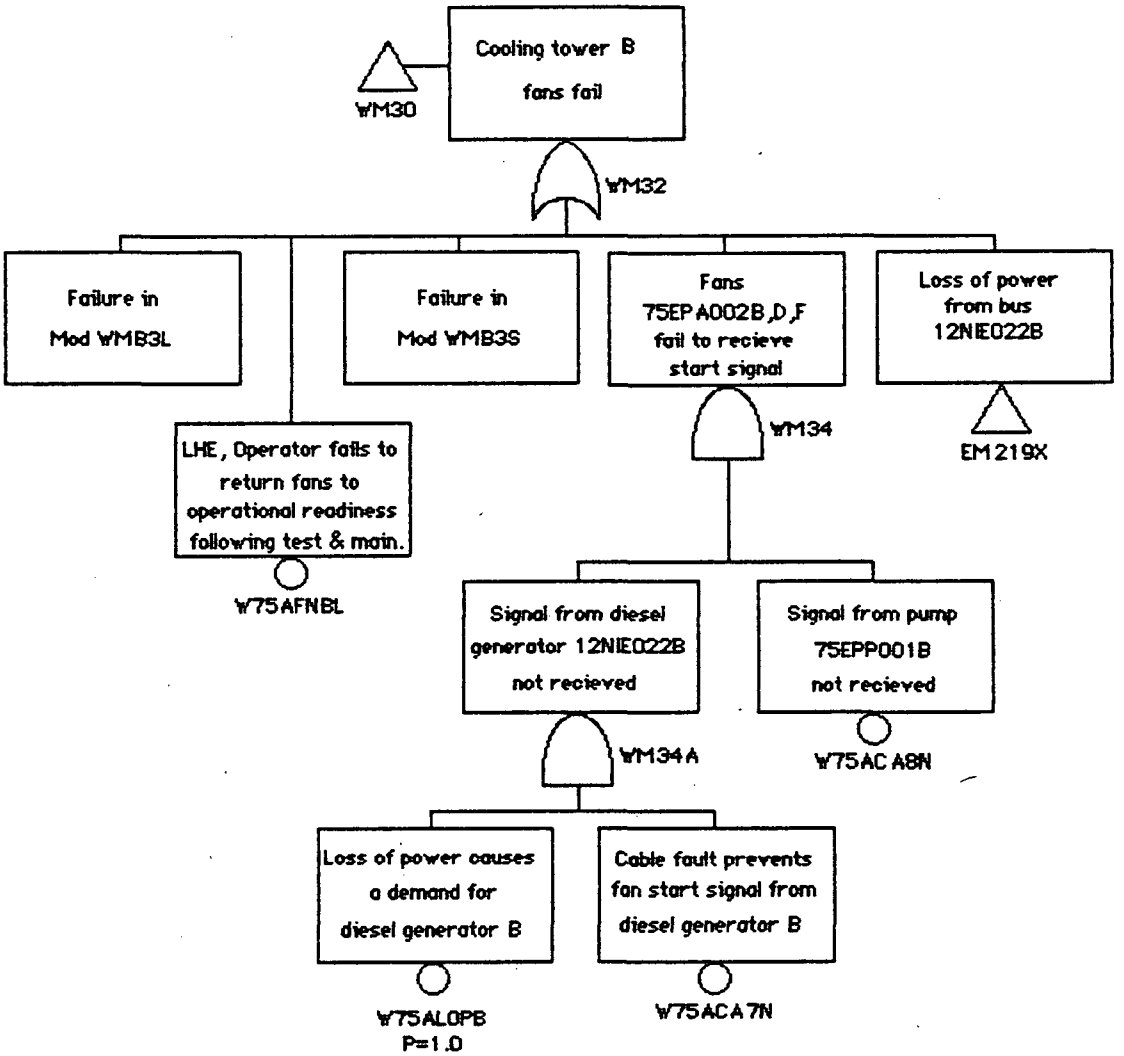
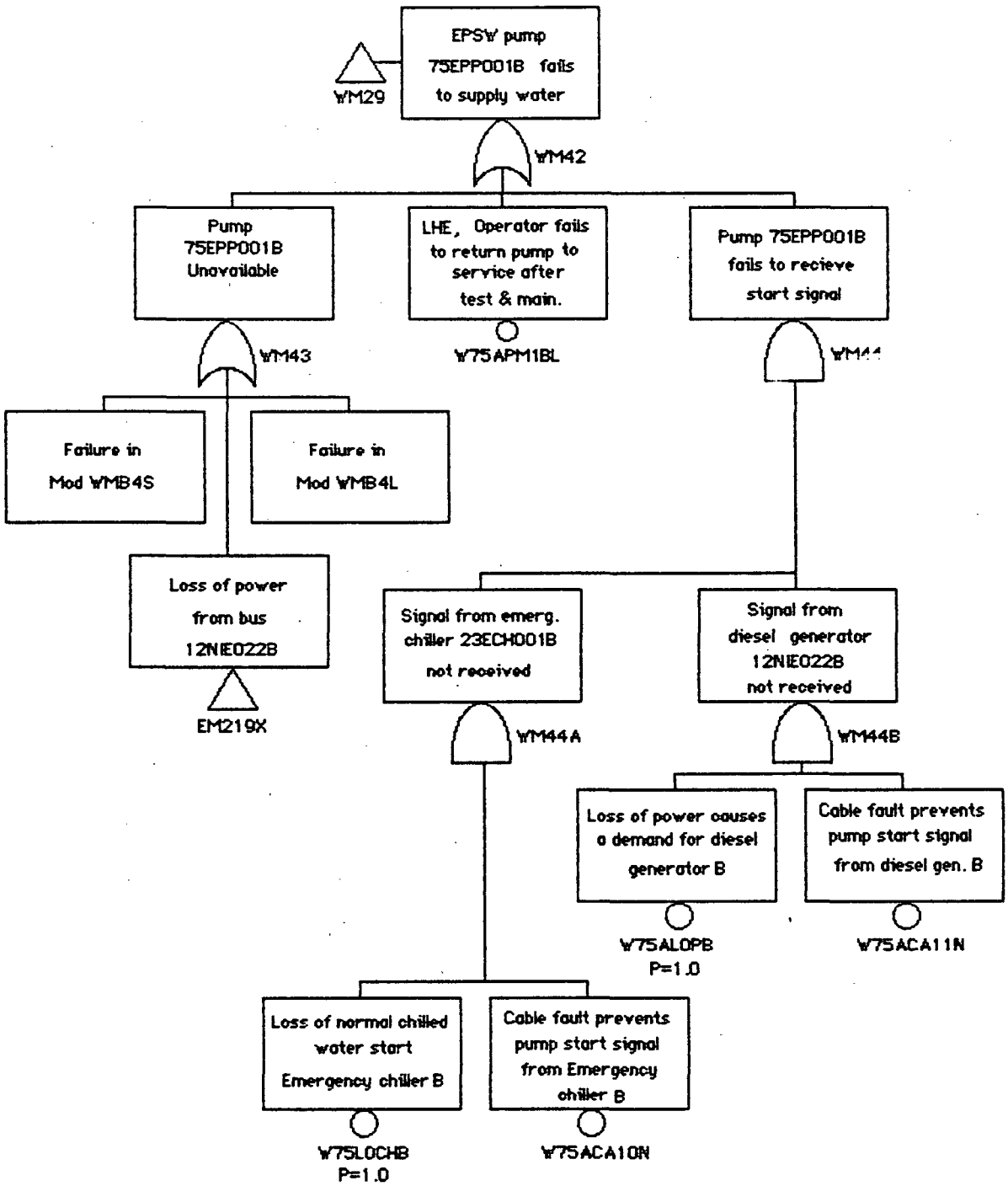

Cooling tower B
fans fail
WM30
WM32
Failure in
Mod WMB3L
Failure in
Mod WMB3S
Fans
75EPA002B,D,F
fail to recieve
start signal
Loss of power
from bus
12NIE022B
EM219X
WM34
LHE, Operator fails to
return fans to
operational readiness
following test & main.
W75AFNBL
Signal from diesel
generator 12NIE022B
not recieved
Signal from pump
75EPP001B
not recieved
WM34A
W75ACA8N
Loss of power causes
a demand for
diesel generator B
Cable fault prevents
fan start signal from
diesel generator B
W75ALOPB
P=1.0
W75ACA7N


EPSW pump 75EPP001B fails to supply water
WM29
WM42
Pump 75EPP001B Unavailable
LHE, Operator fails to return pump to service after test & main.
Pump 75EPP001B fails to recieve start signal
WM43
W75APM1BL
Failure in Mod WMB4S
Failure in Mod WMB4L
Loss of power from bus 12NIE022B
EM219X
Signal from emerg. chiller 23ECH001B not received
Signal from diesel generator 12NIE022B not received
WM44A
WM44B
Loss of power causes a demand for diesel generator B
Cable fault prevents pump start signal from diesel gen. B
W75ALOPB
P=1.0
W75ACA11N
Loss of normal chilled water start Emergency chiller B
Cable fault prevents pump start signal from Emergency chiller B
W75LOCHB
P=1.0
W75ACA10N

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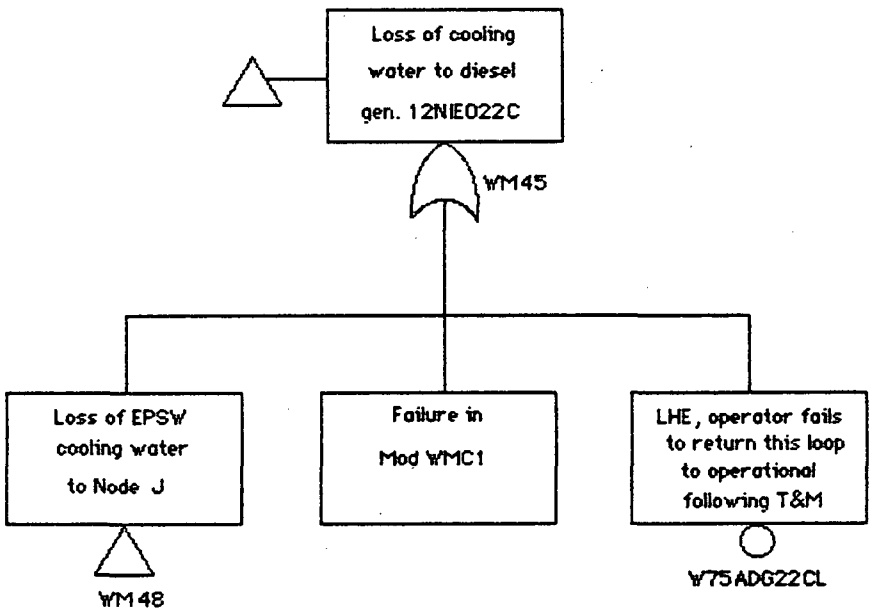
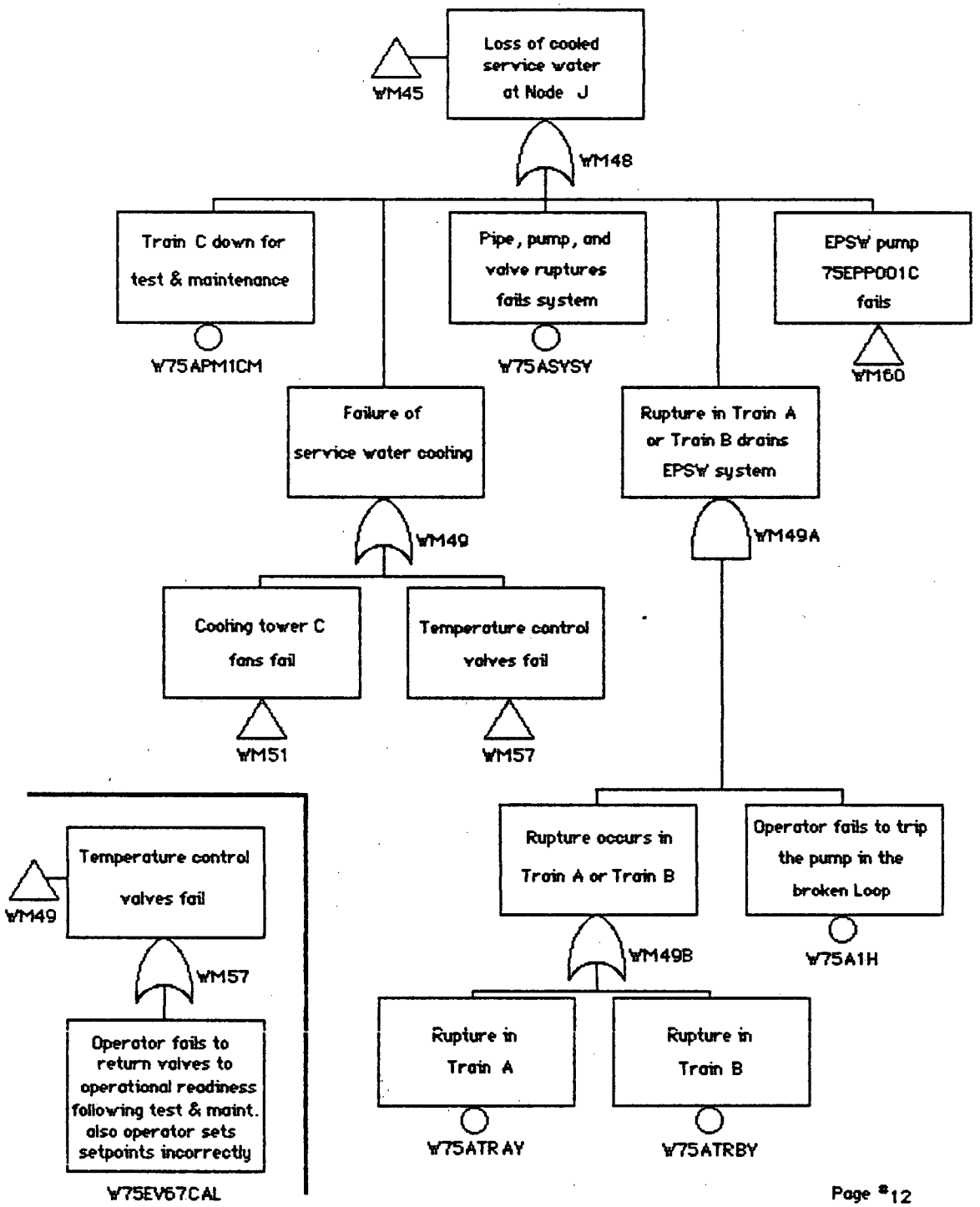

Loss of cooling water to diesel gen. 12NIE022C
WM45
Loss of EPSW cooling water to Node J
Failure in Mod WMC1
LHE, operator fails to return this loop to operational following T&M
WM48
W75ADG22CL


Loss of cooled service water at Node J
WM45
WM48
Train C down for test & maintenance
W75APM1CM
Pipe, pump, and valve ruptures fails system
W75ASYSY
EPSW pump 75EPP001C fails
WM60
Failure of service water cooling
WM49
Rupture in Train A or Train B drains EPSW system
WM49A
Cooling tower C fans fail
WM51
Temperature control valves fail
WM57
Temperature control valves fail
WM49
WM57
Operator fails to return valves to operational readiness following test & maint. also operator sets setpoints incorrectly
W75EV67CAL
Rupture occurs in Train A or Train B
WM49B
Operator fails to trip the pump in the broken Loop
W75A1H
Rupture in Train A
W75ATRAY
Rupture in Train B
W75ATRBY

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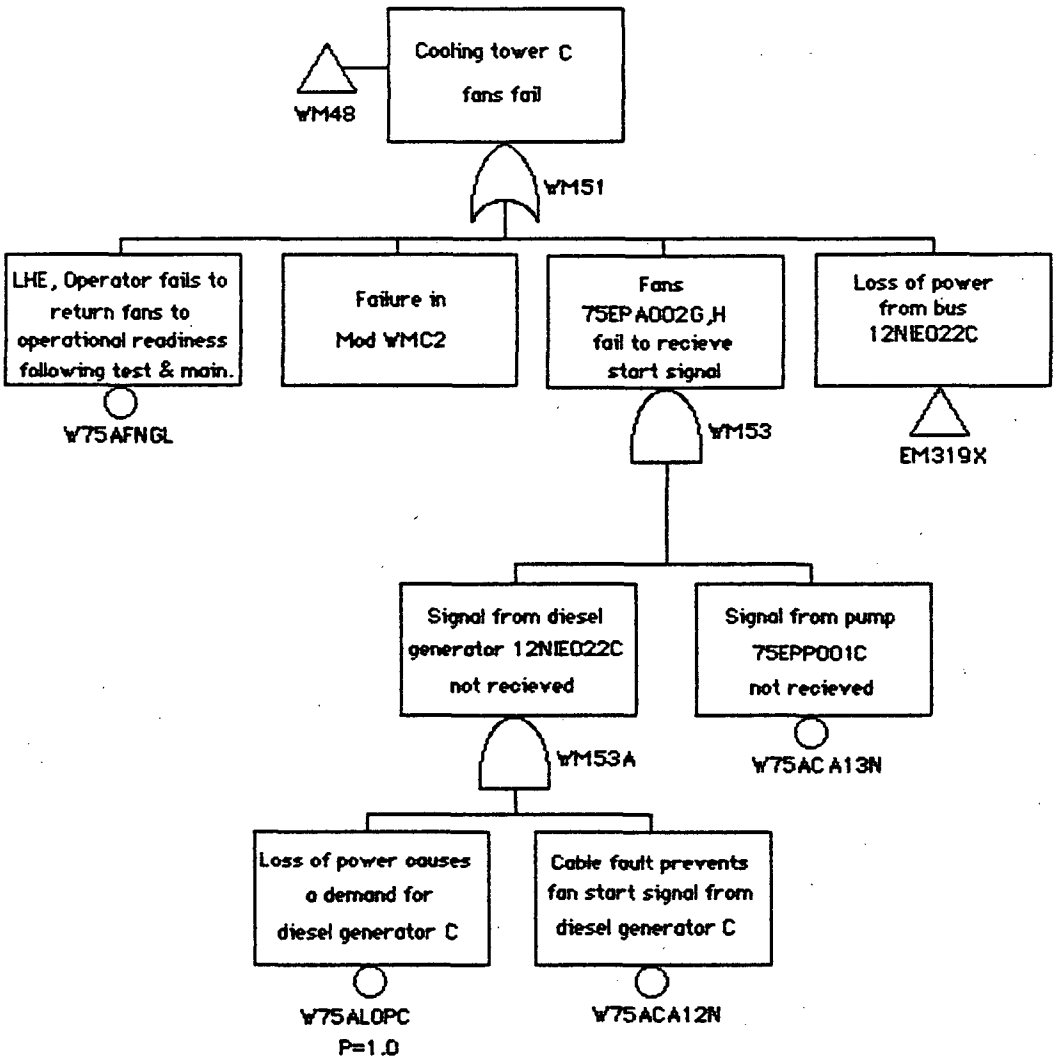
Cooling tower C
fans fail
WM48
WM51
LHE, Operator fails to
return fans to
operational readiness
following test & main.
W75AFNGL
Failure in
Mod WMC2
Fans
75EPA002G,H
fail to recieve
start signal
WM53
Loss of power
from bus
12NIE022C
EM319X
Signal from diesel
generator 12NIE022C
not recieved
WM53A
Signal from pump
75EPP001C
not recieved
W75ACA13N
Loss of power causes
a demand for
diesel generator C
W75ALOPC
P=1.0
Cable fault prevents
fan start signal from
diesel generator C
W75ACA12N

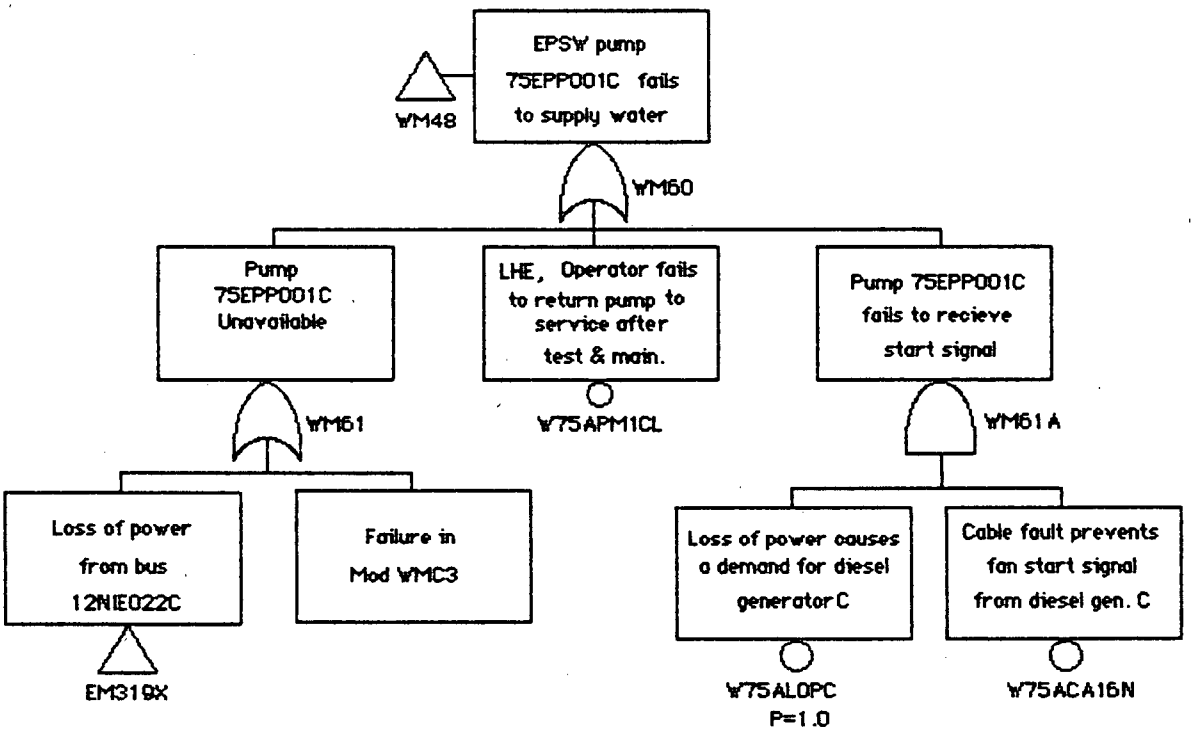
EPSW pump
75EPP001C fails
to supply water
WM48
WM60
Pump
75EPP001C
Unavailable
LHE, Operator fails
to return pump to
service after
test & main.
Pump 75EPP001C
fails to recieve
start signal
WM61
W75APM11CL
WM61A
Loss of power
from bus
12NIE022C
Failure in
Mod WMC3
Loss of power causes
a demand for diesel
generator C
Cable fault prevents
fan start signal
from diesel gen. C
EM319X
W75ALOPC
P=1.0
W75ACA16N

Table A7-1

DATA TABLE

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hr)	Module Definition
WMA1	Emergency chiller A and discharge valves	5.71-4			TAB-OR
W75AXV6AT		3.84-5	1.0-7	384	
W75AFE78AP		1.10-4	2.8-7	384	
W75AXV9AT		3.84-5	1.0-7	384	
W75ACH1AP		3.84-4	1.0-6	384	
WMA2	Diesel generator A and intake and discharge valves	3.02-4			TAB-OR
W75AXV49AT		3.84-5	1.0-7	384	
W75AXV7AT		3.84-5	1.0-7	384	
W75AXV8AT		3.84-5	1.0-7	384	
W75AFE80AP		1.10-4	2.8-7	384	
W75AXV66AT		3.84-5	1.0-7	384	
W75AHX22AP		3.84-5	1.0-7	384	
WMA1A	Valves upstream of chiller A	1.15-4			TAB-OR
W75AXV4AT		3.84-5	1.0-7	384	
W75AEV1AT		3.84-5	1.0-7	384	
W75AXV5AT		3.84-5	1.0-7	384	
WMB1A	Valves upstream of chiller B	2.25-4			TAB-OR
W75AXV4BT		3.84-5	1.0-7	384	
W75AEV1BT		3.84-5	1.0-7	384	
W75AFE78BP		1.10-4	2.8-7	384	
W75AXV5BT		3.84-5	1.0-7	384	

Table A7-1 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hr)	Module Definition
WMB1	Emergency chiller B and discharge valves	4.60-4			TAB-OR
W75AXVBT		3.84-5	1.0-7	384	
W75AXV9BT		3.84-5	1.0-7	384	
W75ACH1BP		3.84-4	1.0-6	384	
WMB2	Diesel Generator B upstream and downstream valves	6.48-4			TAB-OR
W75AXV49BT		3.84-5	1.0-7	384	
W75AXV7BT		3.84-5	1.0-7	384	
W75AXV8BT		3.84-5	1.0-7	384	
W75FE80BP		1.10-4	2.8-7	384	
W75AXV66BT		3.84-5	1.0-7	384	
W75AHX22BP		3.84-4	1.0-6	384	
WMC1	Diesel Generator C upstream and downstream valves	6.86-4			TAB-OR
W75AXV49CT		3.84-5	1.0-7	384	
W75AXV7CT		3.84-5	1.0-7	384	
W75AXV8CT		3.84-5	1.0-7	384	
W75AFE80CP		1.10-4	2.8-7	384	
W75AXV66CT		3.84-5	1.0-7	384	
W75AXV10CT		3.84-5	1.0-7	384	
W75AHX22CP		3.84-4	1.0-7	384	

Table A7-1 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hr)	Module Definition
WMC2	Cooling tower C and return valves	9.35-3			
W75ASNCP		1.10-4	2.8-7	384	TAB-OR
W75AFN2GS		3.0-4	3.0-4		
W75AFN2GR		3.84-3	1.0-5	384	
W75AFN2HS		3.0-4	3.0-4		
W75AFN2HR		3.84-3	1.0-5	384	
W75EV67CAT		3.84-5	1.0-7	384	
W75AT67CB		8.83-4	5.3-6	384	
W75EV67CBT		3.84-5	1.0-7	384	
WMC3	Pump C and discharge valve	1.03-2			
W75AXV3CT		3.84-5	1.0-7	384	TAB-OR
W75ALS90CI		1.6-3	4.2-6	384	
W75APM1CR		7.68-3	2.0-5	384	
W75APMICS		1.0-3	1.0-3	384	
W75ACH1AL*	Latent human error	1.0-3			
W75AXV10AT	Valve 10A transfers	3.84-5	1.0-7	384	
W75ADG22AL*	Latent human error	1.0-3			
W75AP1AH*	Dynamic human error	1.0			
W75APH*	Dynamic human error	1.0			
W75AFNAL*	Latent human error	1.0-3			
W75ACA2N	Cable failure	ε	3.7-6	384	
W75ACA1N	Cable failure	ε	3.7-6	384	
W75EV67AAL*	Latent human error	1.0-3			
W75APM1AL*	Latent human error	1.0-3			
W75ACA4N	Cable fault	ε	3.7-6	384	
W75ACA5N	Cable fault	ε	3.7-6	384	

Table A7-1 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hr)	Module Definition
W75ACH1BL*	Latent human error	1.0-3			
W75AXV10BT	Valve 10B transfers	3.84-5	1.0-7	384	
W75DG22BL*	Latent human error	1.0-3			
W75APM1BM	Maintenance	1.44-3	7.6-5	384	
W75AFNBL*	Latent human error	1.0-3			
W75ACA7N	Cable fault	ε	3.7-6	384	
W75ACA8N	Cable fault	ε	3.7-6	384	
W75EV67BAL*	Latent human error	1.0-3			
W75APM1BL*	Latent human error	1.0-3			
W75ACA10N	Cable fault	ε	3.7-6	384	
W75ACA11N	Cable fault	ε	3.7-6	384	
W75ADG22CL*	Latent human error	1.0-3			
W75APM1CM	Maintenance	1.44-3	7.6-5	384	
W75AFNGL*	Latent human error	1.0-3			
W75ACA13N	Cable fault	ε	3.7-6	384	
W75ACA12N	Cable fault	ε	3.7-6	384	
W75EV67CAL	Latent human error*	1.0-3			
W75APM1CL	Latent human error*	1.0-3			
W75ACA16N	Cable fault	ε	3.7-6	24	
W75ATRBY	Train B leak	ε			
W75ATRCY	Train C leak	ε			
W75ATRAY	Train A leak	ε			
W75ASYSY	System leak	ε			
W75A1H	Dynamic human error	1.0			

*This is a screening number. Some human events may have been assessed in detail and their reassessed probability can be found in Section 6.

Table A7-1 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hr)	Module Definition
WMA4L		2.12-3			TAB-OR
W75AXV3AT	Long Term	3.84-5	1.0-7	384	
W75ALS90AI	Pump A	1.61-3	4.2-6	384	
W75APM1AR		4.8-4	2.0-5	24	
WMA4S	Short Term	1.0-3			
W75APM1AS	Pump A	1.0-3	1.0-3		TAB-OR
WMB4L		2.12-3			TAB-OR
W75AXV3BT	Long Term	3.84-5	1.0-7	384	
W75ALS90BI	Pump B	1.61-3	4.2-6	384	
W75APM1BR		4.8-4	2.0-5	24	
WMB4S	Short Term	1.0-3			
W75APM1BS	Pump B	1.0-3	1.0-3		TAB-OR
WMA3S	Short Term A	2.7-7			
W75AFN2AS	Fan Operation	3.0-4	3.0-4		(W75AFN2AS * W75AFN2CS) +
W75AFN2CS		3.0-4	3.0-4		(W75AFN2CS * W75AFN2ES) +
W75AFN2ES		3.0-4	3.0-4		(W75AFN2AS * W75AFN2ES)
WMA3L	Long Term A	2.18-3			
W75ASNAP	Fan Operation	1.07-4	2.8-7	384	W75ASNAP + [W75AFN2AR *
W75AFN2AR		2.4-6	1.0-7	24	W75AFN2CR] + [W75AFN2CR *
W75AFN2CR		2.4-6	1.0-7	24	W75AFN2ER] + [W75AFN2AR *
W75AFN2ER		2.4-6	1.0-7	24	W75AFN2ER] + W75EV67AAT +
W75EV67AAT		3.84-5	1.0-7	384	W75ATT67AB + W75EV67ABT
W75ATT67AB		2.0-3	5.3-6	384	
W75EV67ABT		3.84-5	1.0-7	384	

Table A7-1 Continued.

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hr)	Module Definition
WMB3S	Short Term B	2.7-7			
W75AFN2BS	Fan Operation	3.0-4	3.0-4		(W75AFN2BS * W75AFN2DS) +
W75AFN2DS		3.0-4	3.0-4		(W75AFN2DS * W75AFN2FS) +
W75AFN2FS		3.0-4	3.0-4		(W75AFN2BS * W75AFN2FS)
WMB3L	Long Term B	2.18-3			
W75ASNBP	Fan Operation	1.07-4	2.8-7	384	W75ASNBP + [W75AFN2BR *
W75AFN2BR		2.4-6	1.0-7	24	W75AFN2DR] + [W75AFN2DR *
W75AFN2DR		2.4-6	1.0-7	24	W75AFN2FR] + [W75AFN2BR *
W75AFN2FR		2.4-6	1.0-7	24	W75AFN2FR] + W75EV67BAT +
W75EV67BAT		3.84-5	1.0-7	384	W75ATT67BB + W75EV67BBT
W75ATT67BB		2.0-3	5.3-6	384	
W75EV67BBT		3.84-5	1.0-7	384	

Appendix A, Section 8

HEAT REJECTION SYSTEM

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Appendix A, Section 8

HEAT REJECTION SYSTEM

A8.1 DESCRIPTION

The heat rejection system consists of two subsystems:

- Circulating water system
- Condenser air extraction system

A8.1.1 Function

A8.1.1.1 Circulating water system. The main functions of the circulating water system are as follows:

1. Provides the heat sink required for the turbine generator System turbine exhaust steam.

2. Deaerates condensate for the main feedwater and condensate system.

3. Provides the heat sink for the normal plant service water system.

4. Transports the rejected heat to the atmosphere.

A8.1.1.2 Condenser air extraction system. The main function of the condenser air extraction system is to increase the overall efficiency of the turbine generator system by removing air and noncondensable gases from the condenser.

A8.1.2 Design

A8.1.2.1 Circulating water system. The circulating water system, shown in Figure A8-1, is a closed-loop system utilizing two mechanical draft wet cooling towers. The water circuit includes the three circulating water pumps taking suction from the pump house channel.[1] The cold water flows

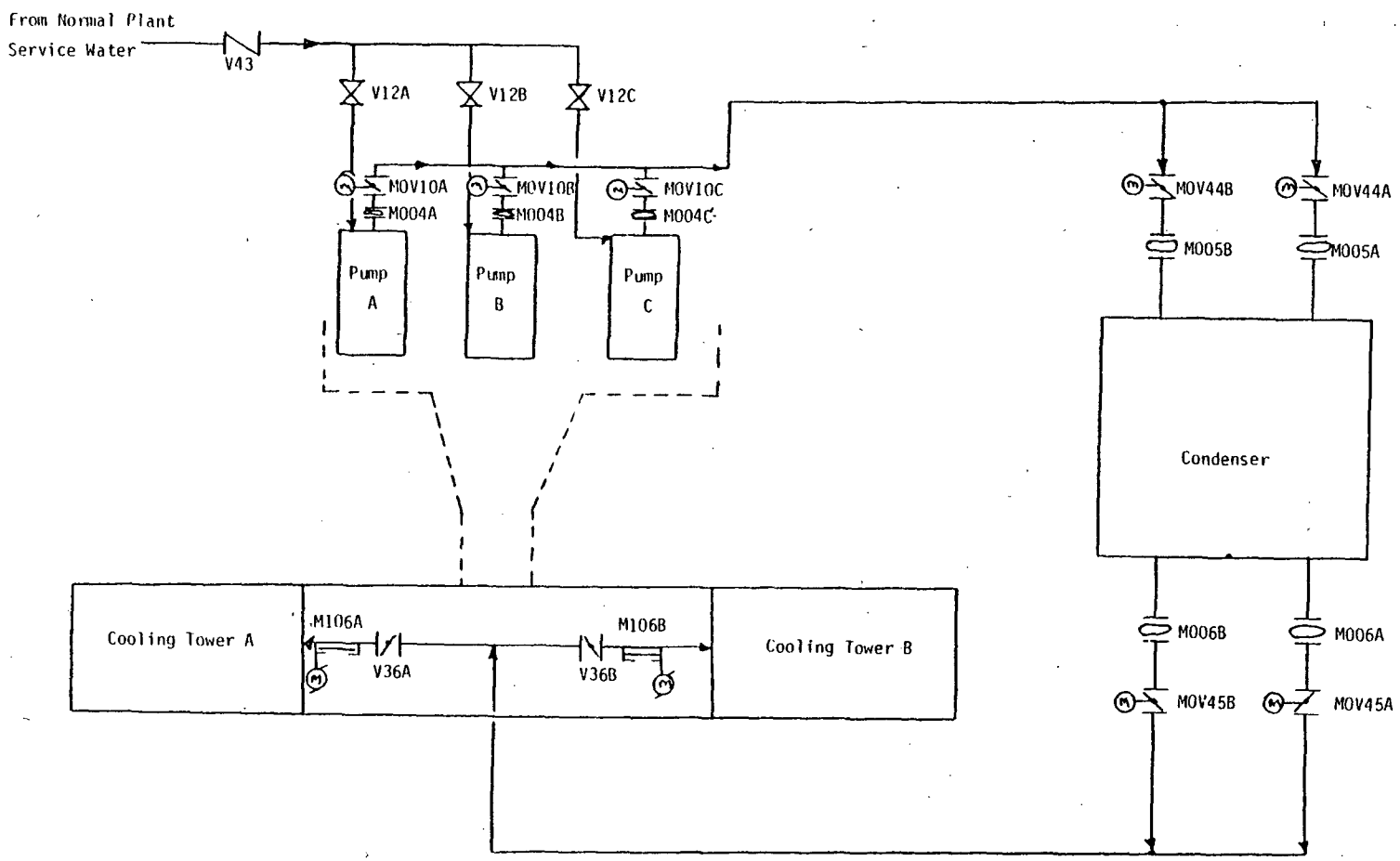


Figure A8-1. Circulating Water System Simplified P&ID.

to the condenser inlet waterboxes and through the condenser. Hot water returns to the distribution header of the "fill section" of the towers.

Each circulating water pump bay is equipped with a stop log at the suction and a motor-operated butterfly valve in the discharge piping. The inlet to and outlet from the condenser waterboxes are equipped with motor-operated isolation butterfly valves.

The hot effluents from the two outlet pipes from the condenser and the normal plant service water system are discharged into a single line. This discharge line takes the effluent to the tower distribution headers above the fill where the process of cooling is initiated.

Only part of the circulating water flowing to the cooling towers is returned to the system. The three major water loss paths are evaporation, drift, and blowdown. Cooling tower basin levels are automatically controlled by adding makeup from the river water service system through a level control valve.

A8.1.2.2 Condenser air extraction system. The condenser air extraction system consists of two full capacity vacuum pump units. Non-condensables are drawn from the condenser by the vacuum pumps after being monitored for tritium, and are then diluted with air and discharged to the atmosphere. Vacuum pump cooling water is supplied by the circulating water system. During the initial startup of the condenser air extraction system, both vacuum pump units are placed in service. During normal operation, one unit is in service and the other is maintained on standby with automatic starting capability. Automatic startup would

occur if the operating unit should fail or in the event of excessive air leakage into the condenser vacuum system as measured by a vacuum switch.

The condenser air extraction system components are located on the ground floor of the turbine generator building near the cold end of the condenser. The vacuum pumps take suction from a common header connected to the condenser shell at the cold end of the condenser as shown on Figure A8-2. The noncondensables flow through the vacuum pump to the separator where excess condensate is removed and the noncondensables are discharged to a common header. Cold circulating water from the condenser inlet water line branches off from a common header to the two vacuum pump coolers where heat is removed from the vacuum pump seal water system. The hot water returns to the condenser outlet water line through a common header.

A8.1.3 Interfaces with Other Systems

A8.1.3.1 Electric power. The building electric power system provides motive power to all pumps, fans, and motor-operated valves in the heat rejection system.

A8.1.3.2 Compressed gas. The compressed gas system provides air to heat rejection system components at 165±15 psig.

A8.1.3.3 Normal plant service water. The normal plant service water system provides 125 gpm seal water flow to the three circulating water pumps in the heat rejection system.

A8.1.3.4 River water system. The river water system provides makeup water to the two cooling tower basins at a maximum flow of 8500 gpm.

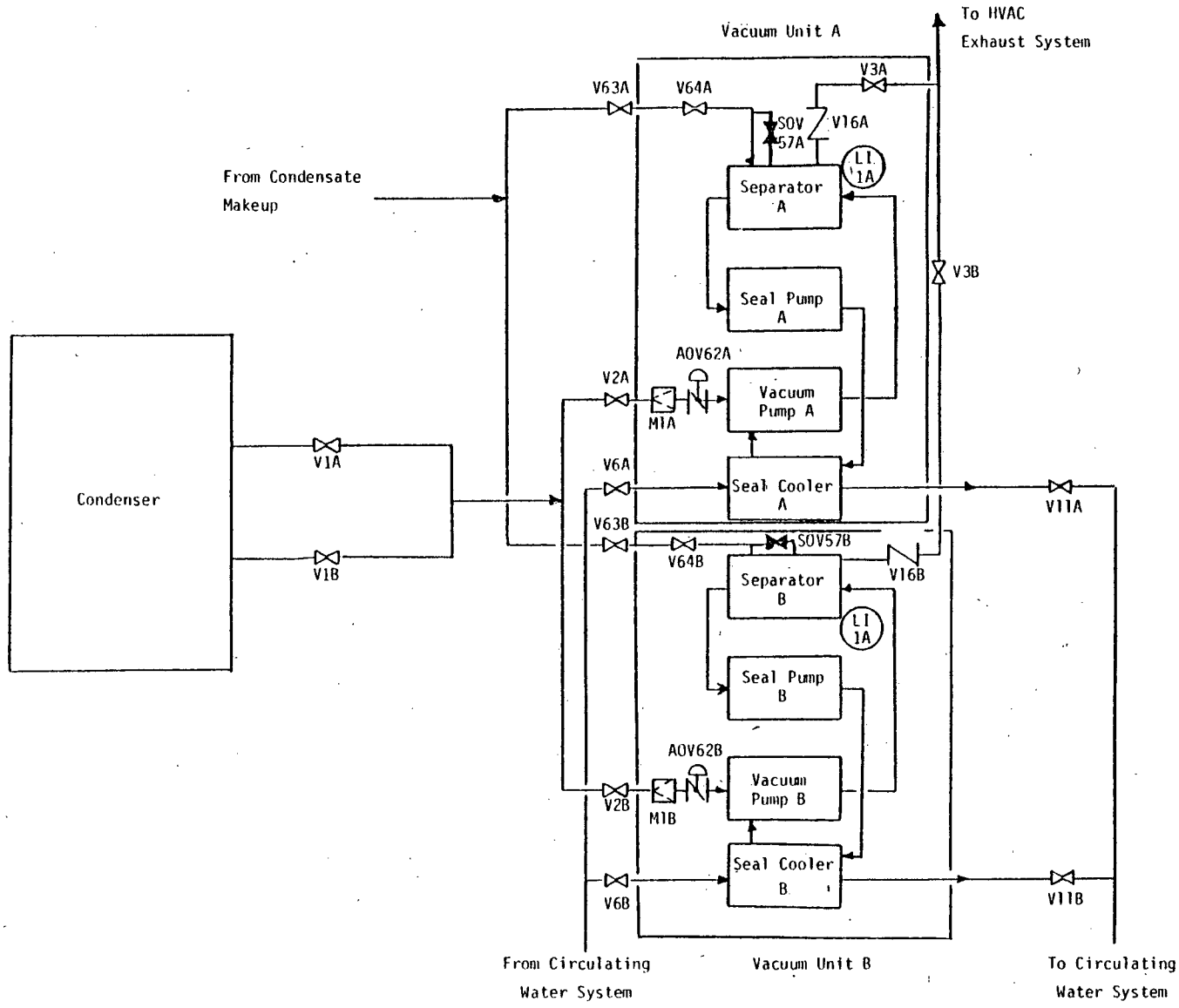


Figure A8-2. Condenser Air Extraction System Simplified P&ID.

A8.1.3.5 Control power. Complete control power information was not available. Temperature and pressure sensors were assumed to be provided with uninterruptible power supplies. Pneumatic control power information was not available.

A8.1.3.6 Actuation systems. The heat rejection system operates during both normal and off-normal conditions and therefore requires no external actuation systems.

A8.1.4 Operation

A8.1.4.1 Normal operation.

A8.1.4.1.1 Circulating water system. The three circulating water pumps and all cooling tower fans are running. The operator can manually remove pumps or fans from service to adjust for climate changes or to optimize the condenser vacuum conditions.

A8.1.4.1.2 Condenser air extraction system. One vacuum pump unit is in operation with the second unit in a standby mode.

A8.1.4.2 Off-normal operation.

A8.1.4.2.1 Circulating water system. Two circulating water pumps operating in the "run out" mode are capable of supplying the necessary circulating water to continue power operation. During a plant shutdown condition, one of the three running pumps may be turned off to reduce the flow through the condenser.

A8.1.4.2.2 Condenser air extraction system. If the operating vacuum unit fails, a standby vacuum unit starts, maintaining the condenser

vacuum. No change in normal operation is made during a shutdown plant condition.

A8.1.5 Test and Technical Specifications

No specific test and technical information was available for the heat rejection system.

A8.1.6 Maintenance Requirements

A8.1.6.1 Circulating water system.

A8.1.6.1.1 Condenser. The condenser is visually inspected once per shift for external leakage. It is drained and the tubes are inspected once per year.

A8.1.6.1.2 Circulating water pump motors. The lubrication levels are checked once per shift, and the vibration level is checked once per week. The bearing oil is changed once per year.

A8.1.6.1.3 Cooling tower fan. The motor bearings are lubricated, the gear oil changed, and the pillowblock inspected 4 times per year. The gear drive oil level is checked, and the pillowblock bearings are lubricated once per week. The fan driveshaft is inspected once per month, and the gear drive shaft seal is lubricated two times per year.

A8.1.6.1.4 Cooling tower. The distribution spray system nozzles are cleaned once per year. The fill material and mist eliminator are cleaned twice per year.

A8.1.6.1.5 Circulating water pump. The circulating water pumps are visually inspected once per shift.

A8.1.6.2 Condenser air extraction system.

A8.1.6.2.1 Separator. The low-level makeup valve and the overflow check valve are inspected once per year.

A8.1.6.2.2 Seal water cooler. The cooler is inspected visually once per shift. The tubes are checked for blockage once per year.

A8.1.6.2.3 Vacuum pump motor. The lubrication and vibration levels are checked once per shift. The drive coupling is lubricated twice per year.

A8.1.6.2.4 Seal water pump motor. The lubrication and vibration levels are checked once per shift.

A8.2 SYSTEM MODEL

A8.2.1 Modeling Assumptions

A. This analysis assumes the design of the piping is properly sized to meet the requirements of the system.

B. Mission time for which the analysis is conducted is 24 hours. Because of the short mission time, there is no allowance for the repair of failed equipment.

C. Pipe and valve ruptures are included in the model as a conglomerate event.

D. The term "transfers" implies going from a desirable or required position to an undesirable or failed position.

E. It is assumed that one circulating water pump, operating properly, can supply the required circulating water flow for heat removal following a plant trip.

F. Loss of seal water to any one circulating water pump causes that pump to fail.

G. It was assumed that one vacuum pump is operating with the second unit in the standby mode.

H. It was assumed that three circulating water pumps are running with no pump in the standby mode.

I. It was assumed that the loss of all cooling tower fans would not fail the heat rejection system under decay heat removal circumstances.

J. It was assumed that the cooling tower basin water level would not fall below the minimum operating level following loss of the river water system. This is due to the size of the basin, and the small evaporative and blowdown losses that occur during decay heat removal circumstances.

K. Control power was included as a conglomerate event.

A8.2.2 Detailed Fault Tree Model

The heat rejection system detailed fault tree is shown in Figure A8-3.

A8.2.2.1 Success criteria. One of three circulating water pumps and one vacuum unit must be operating together with sufficient-water in the cooling tower basins and no blockage of flow for the heat rejection system to operate.

A8.2.2.2 Top event. The Top event is: Failure of the heat rejection system to operate.

A8.2.2.3 Transfers to other systems.

A8.2.2.3.1 Electric power.

Equipment	Bus	Gate
73CEK102A	(TBD)*	EM324A
73CEK102B	(TBD)	EM324A
73CEK101A	12BP006A	EM324A
73CEK101B	(TBD)	EM324A
73CWK011A	12BPE002A	EM324A
73CWK011B	12BPE002B	EM324A
73CWK011C	12BPE002C	EM324A
73CWK101A	12BPE010A	EM324A
73CWK102A	"	EM324A
73CWK103A	"	EM324A
73CWK104A	"	EM324A
73CWK105A	"	EM324A

*To be determined. Information not available at the time of the study.

Equipment	Bus	Gate
73CWK101B	12BPE010B	EM324A
73CWK102B	"	EM324A
73CWK103B	"	EM324A
73CWK104B	"	EM324A
73CWK105B	"	EM324A

A8.2.2.3.2 Normal plant service water.

Equipment	Gate
73CWP001A	W67
73CWP001B	W67
73CWP001C	W67

A8.2.3 Modularized Fault Tree

The heat rejection system fault tree was simplified by combining the independent pump, valve, and pipe failures into larger events called modules (Figure A8-4). These modules will be used when the fault tree is solved, as their use reduces the computational time required.

A8.3 FAILURE DATA

Data were applied according to the rules and tables found in Chapters 5 and 6. Module data were calculated using the basic event data and the module definitions. Results of these calculations can be found in Table A8-1.

A8.4 SYSTEM-LEVEL INSIGHTS AND RESULTS

The heat rejection system modularized fault tree was solved using the set equation transformation system (SETS) algorithm to determine the minimal cut sets of the system model. The cut sets with the highest probability of failure, as indicated from the SETS output, were those

involving offsite power failures. Included in these cut sets were nonessential ac bus and transformer failures. The condenser air extraction system had, in addition to these power failures, cut sets involving a failure in the running vacuum unit and a failure of the standby unit to start.

A8.5 REFERENCES

1. System Design Description, Heat Rejection System (SDD-73), Revision 12 (7/83)

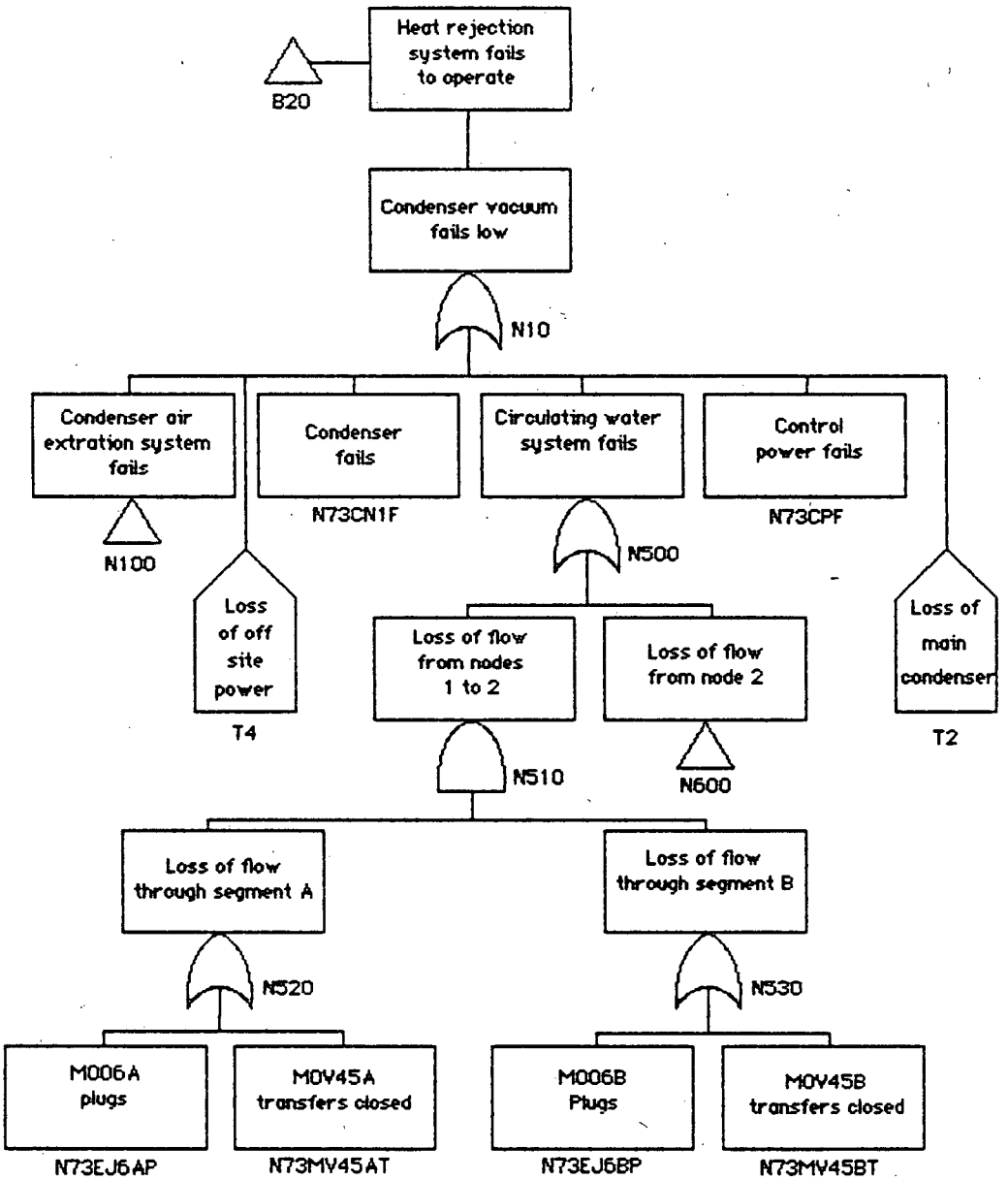
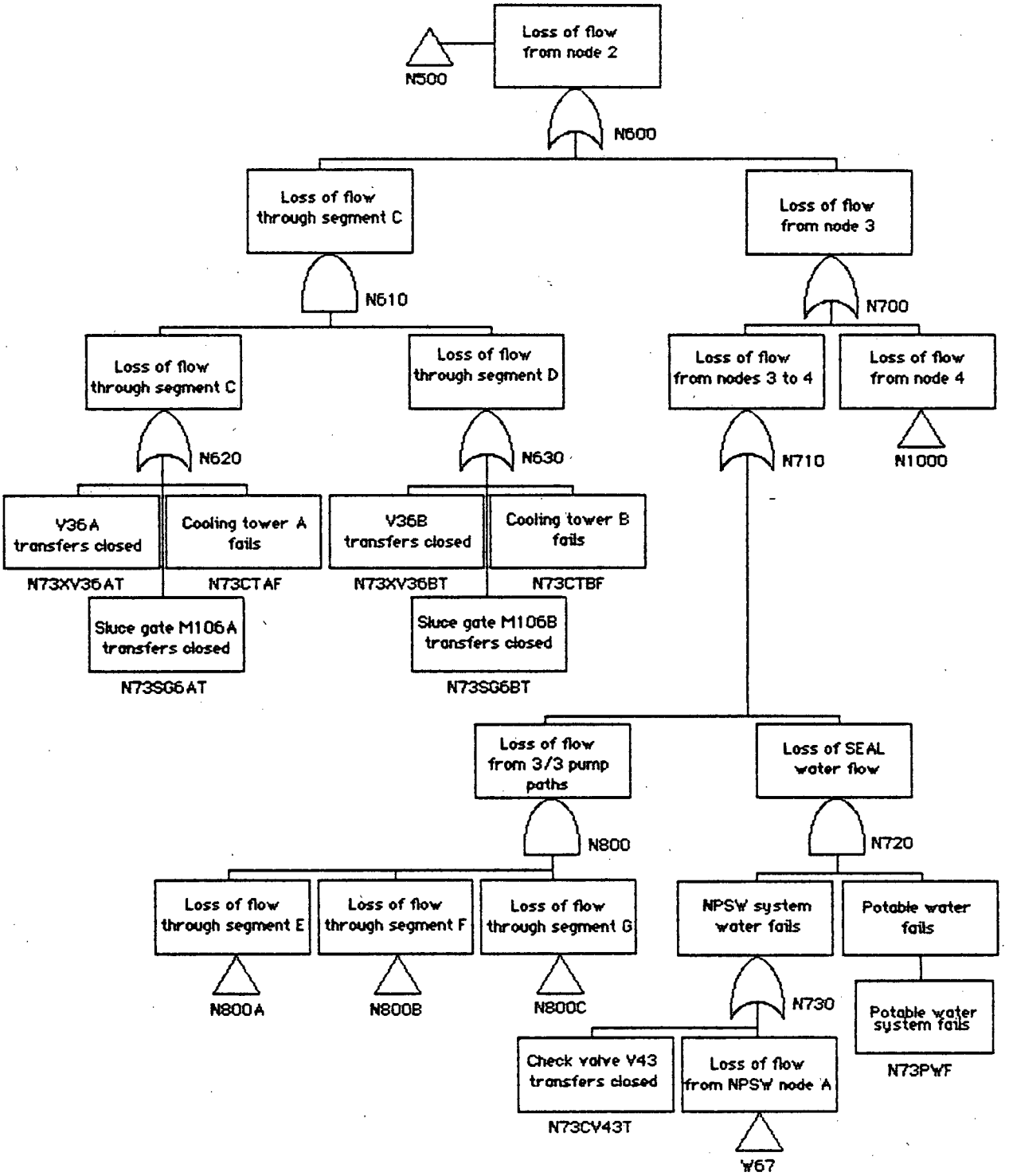
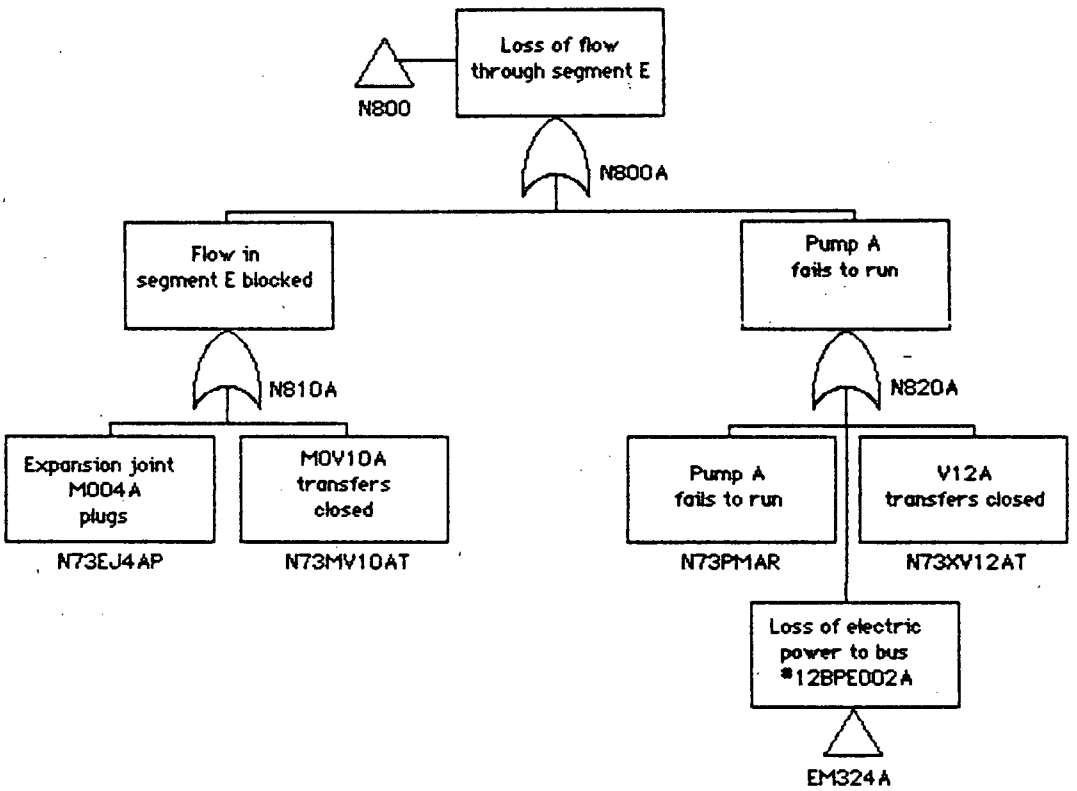
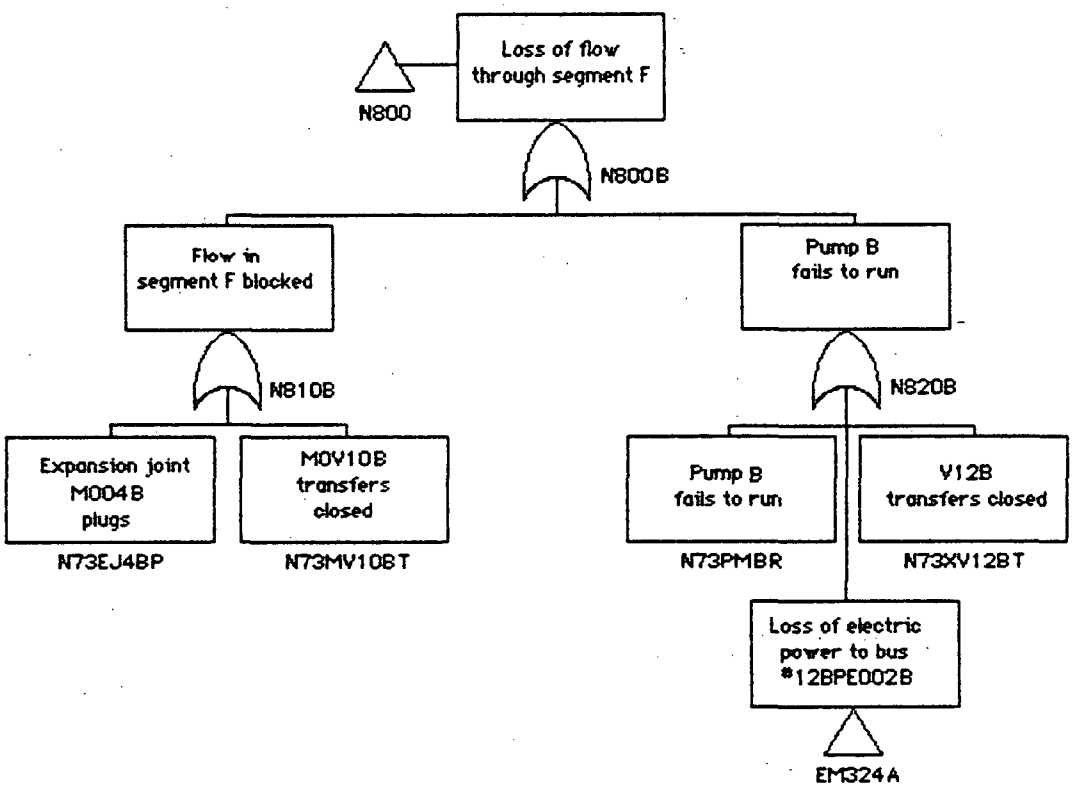
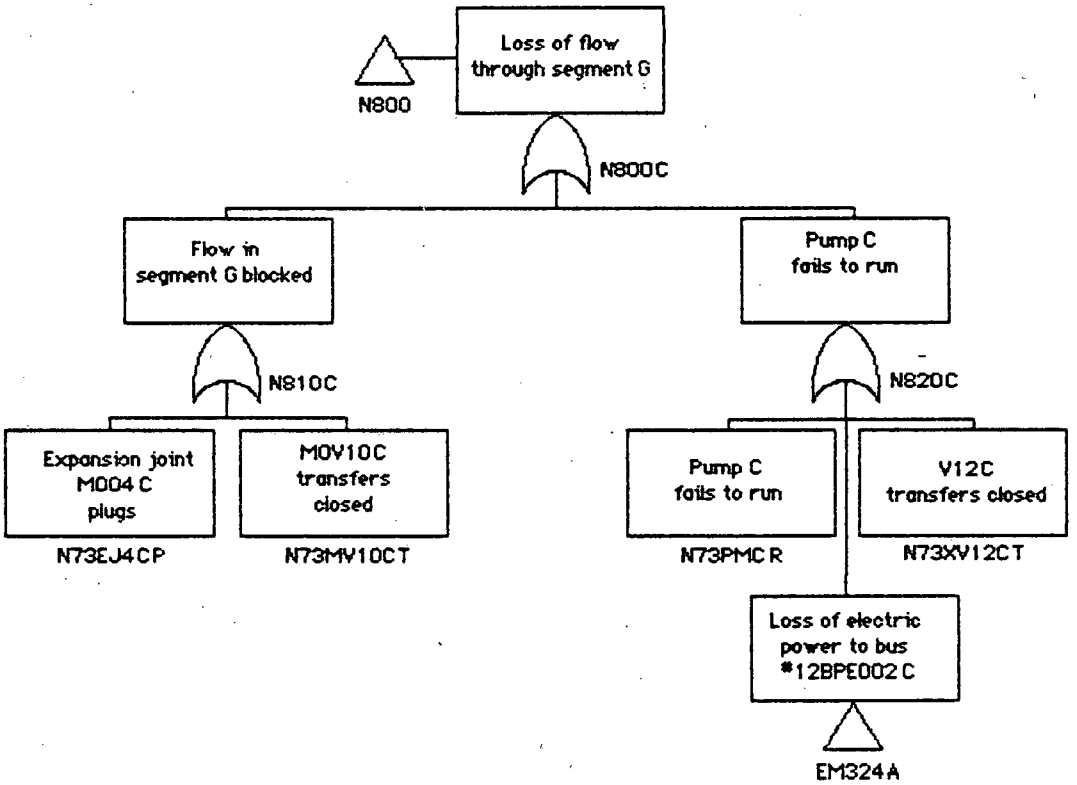


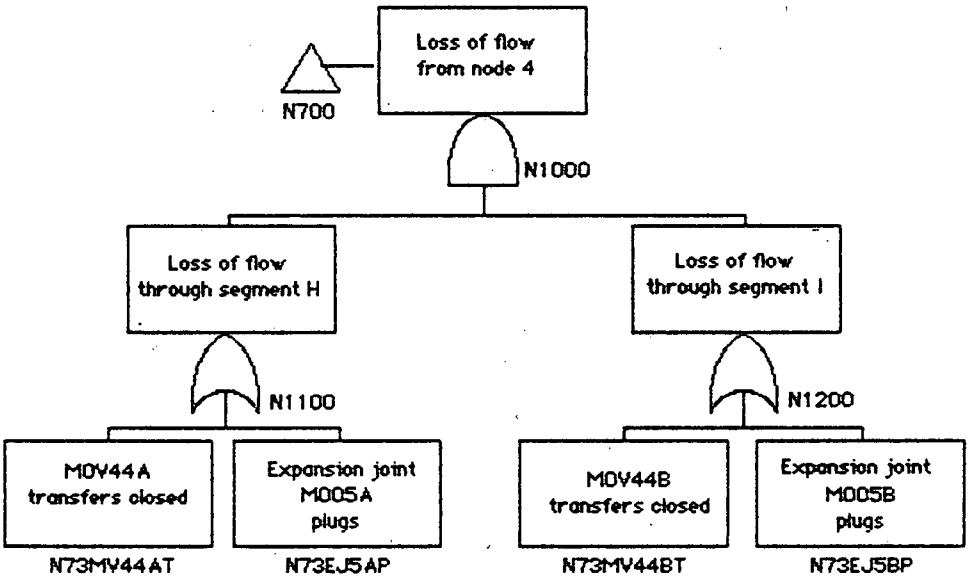
Figure A8-3. Heat Rejection Detailed Fault Tree.

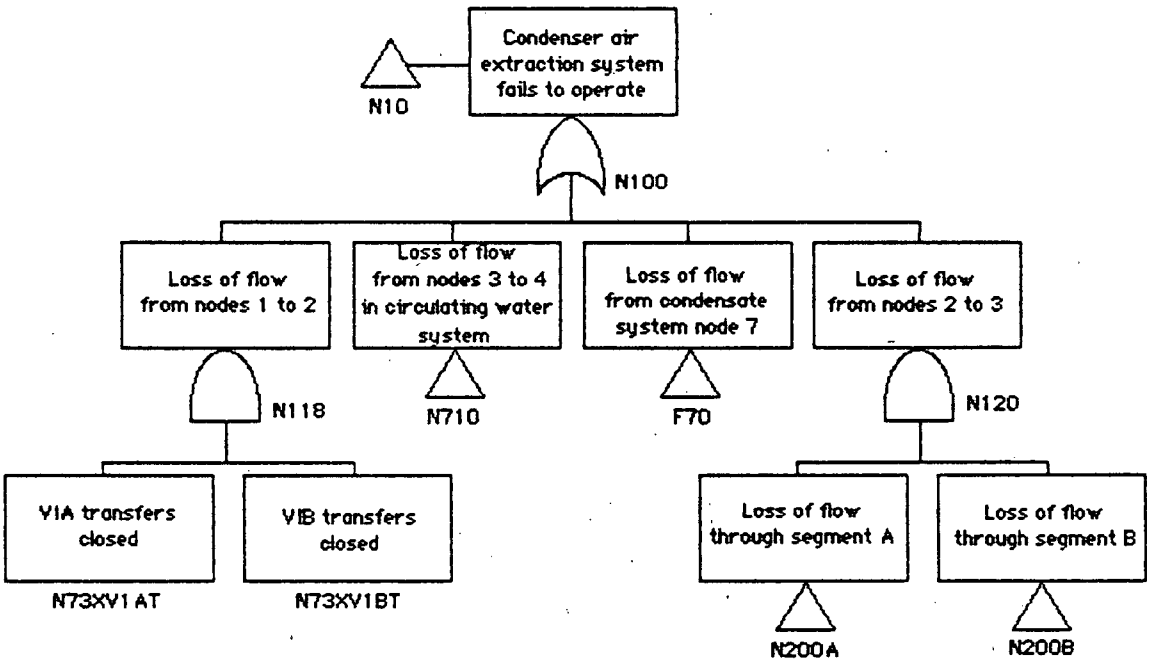

Loss of flow from node 2
N500
N600
Loss of flow through segment C
Loss of flow from node 3
N610
N700
Loss of flow through segment C
Loss of flow through segment D
Loss of flow from nodes 3 to 4
Loss of flow from node 4
N620
N630
N710
N1000
V36A transfers closed
Cooling tower A fails
V36B transfers closed
Cooling tower B fails
N73XV36AT
N73CTAF
N73XV36BT
N73CTBF
Sluce gate M106A transfers closed
Sluce gate M106B transfers closed
N73SG6AT
N73SG6BT
Loss of flow from 3/3 pump paths
Loss of SEAL water flow
N800
N720
Loss of flow through segment E
Loss of flow through segment F
Loss of flow through segment G
NPSW system water fails
Potable water fails
N800A
N800B
N800C
N730
Potable water system fails
Check valve V43 transfers closed
Loss of flow from NPSW node A
N73PWF
N73CV43T
W67

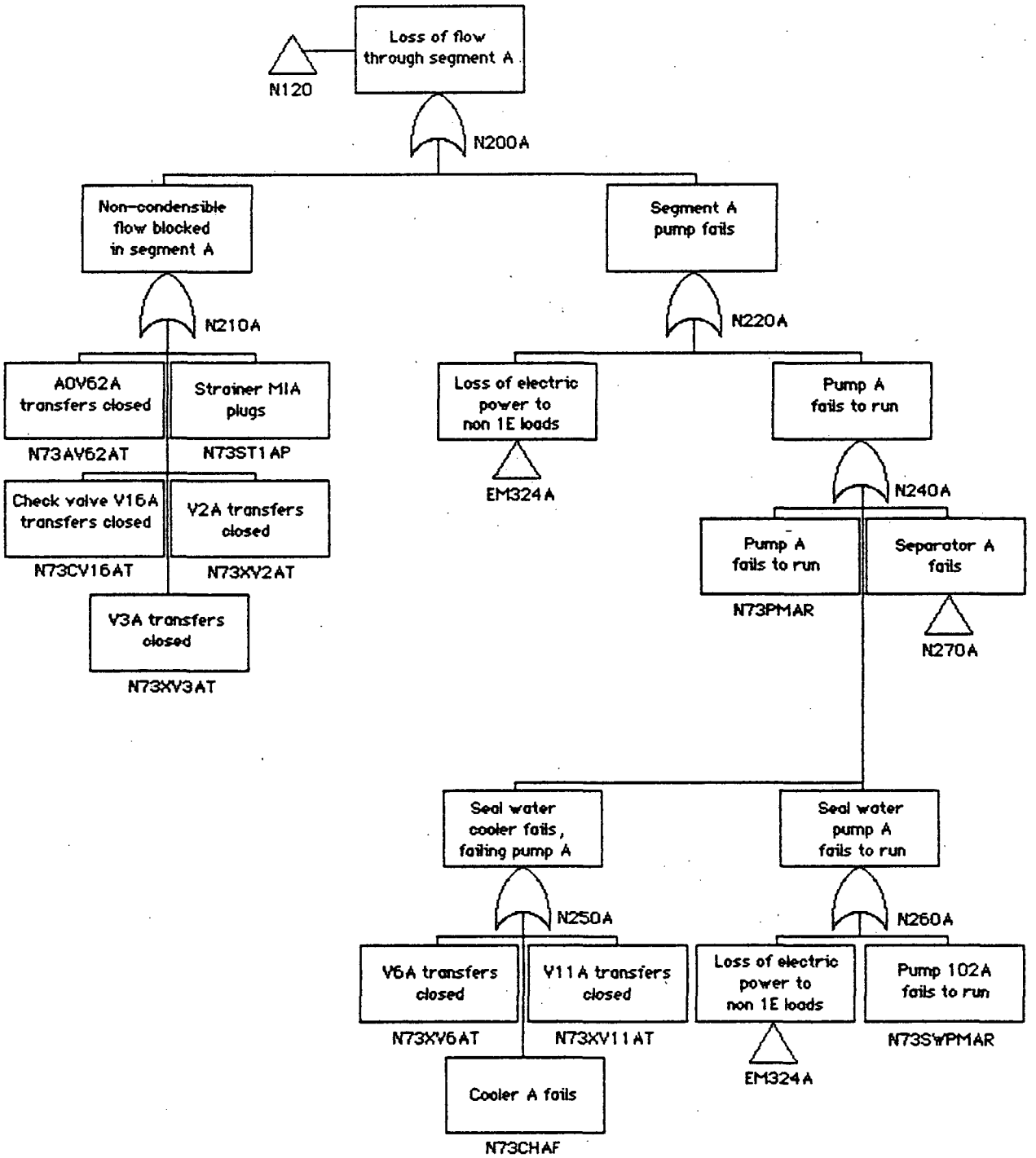

Loss of flow through segment E
N800
N800A
Flow in segment E blocked
Pump A fails to run
N810A
N820A
Expansion joint M004A plugs
MOV10A transfers closed
Pump A fails to run
V12A transfers closed
N73EJ4AP
N73MV10AT
N73PM1AR
N73XV12AT
Loss of electric power to bus #12BPE002A
EM324A

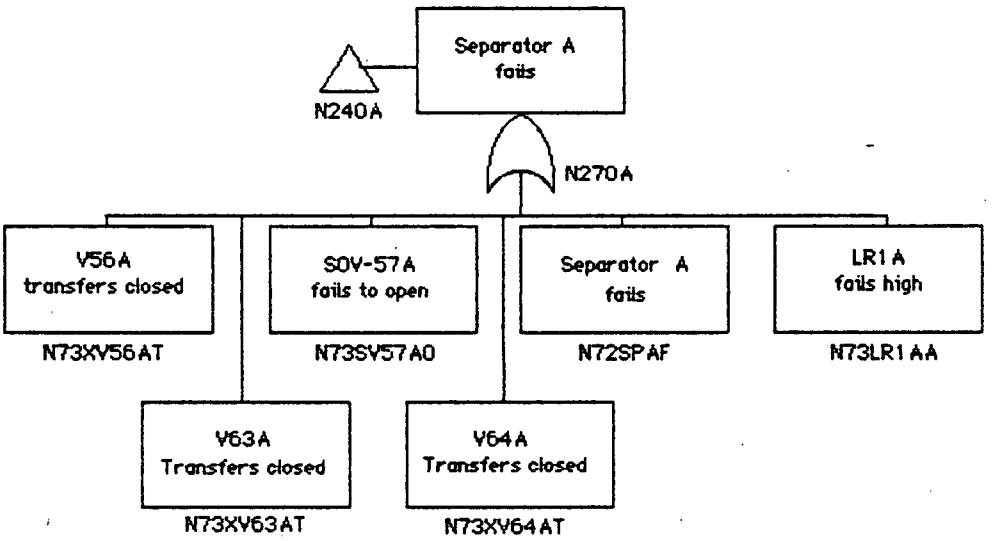

Loss of flow through segment F
N800
N800B
Flow in segment F blocked
Pump B fails to run
N810B
N820B
Expansion joint M004B plugs
N73EJ4BP
MOV10B transfers closed
N73MV10BT
Pump B fails to run
N73PMBR
V12B transfers closed
N73XV12BT
Loss of electric power to bus #12BPE002B
EM324A

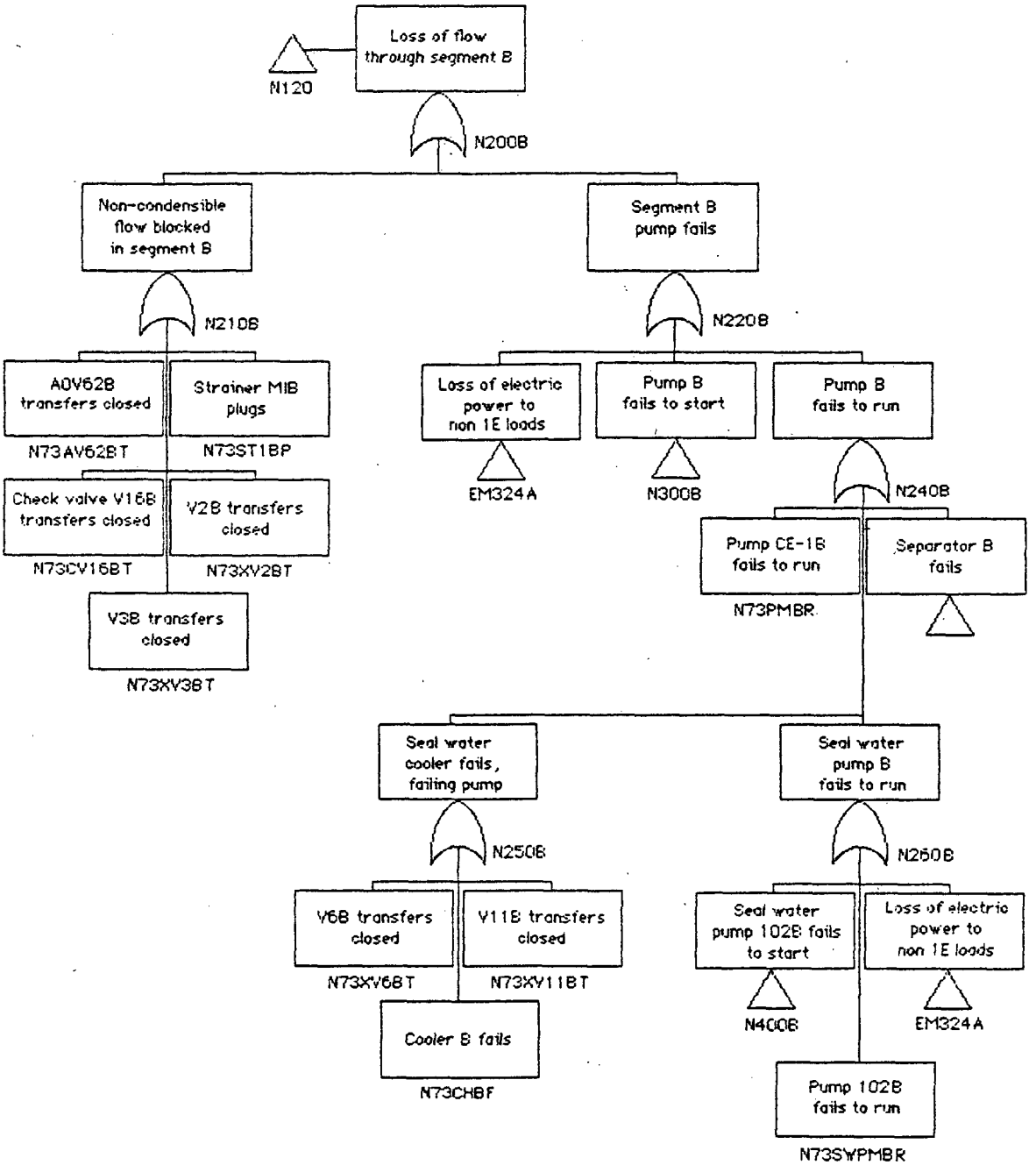

Loss of flow through segment G
N800
N800C
Flow in segment G blocked
Pump C fails to run
N810C
N820C
Expansion joint M004 C plugs
N73EJ4CP
MOV10C transfers closed
N73MV10CT
Pump C fails to run
N73PMC R
V12C transfers closed
N73XV12CT
Loss of electric power to bus *12BPE002 C
EM324A

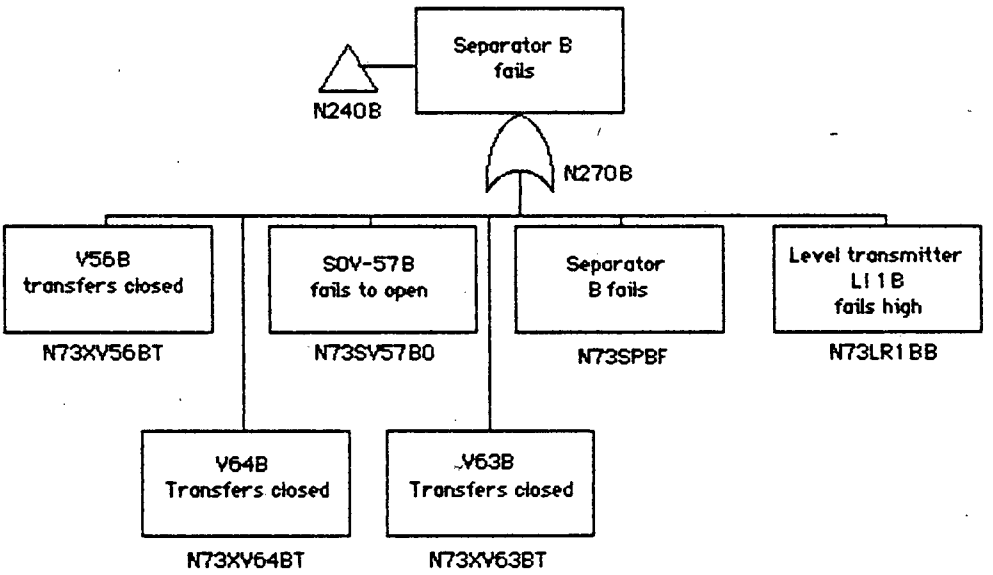

Loss of flow
from node 4
N700
N1000
Loss of flow
through segment H
Loss of flow
through segment I
N1100
N1200
MOV44A
transfers closed
Expansion joint
M005A
plugs
MOV44B
transfers closed
Expansion joint
M005B
plugs
N73MV44AT
N73EJ5AP
N73MV44BT
N73EJ5BP

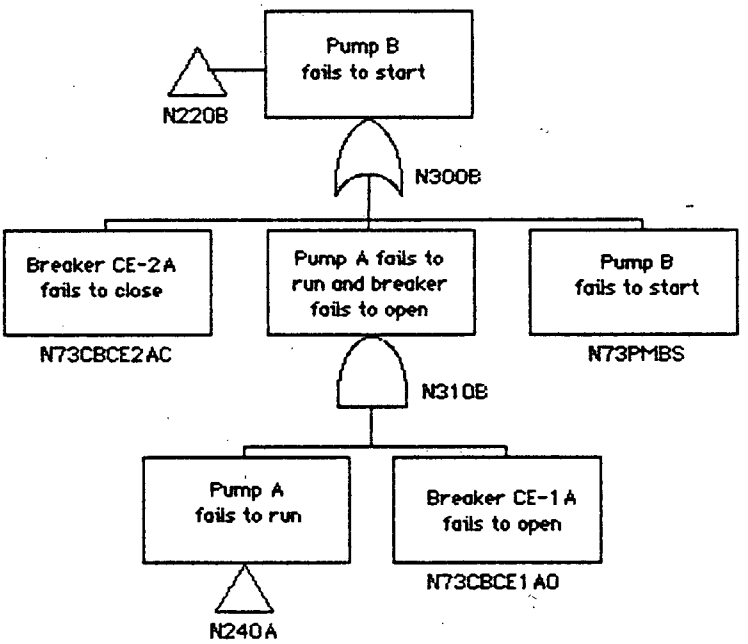

Condenser air extraction system fails to operate
N10
N100
Loss of flow from nodes 1 to 2
Loss of flow from nodes 3 to 4 in circulating water system
Loss of flow from condensate system node 7
Loss of flow from nodes 2 to 3
N118
N710
F70
N120
V1A transfers closed
V1B transfers closed
N73XV1AT
N73XV1BT
Loss of flow through segment A
Loss of flow through segment B
N200A
N200B

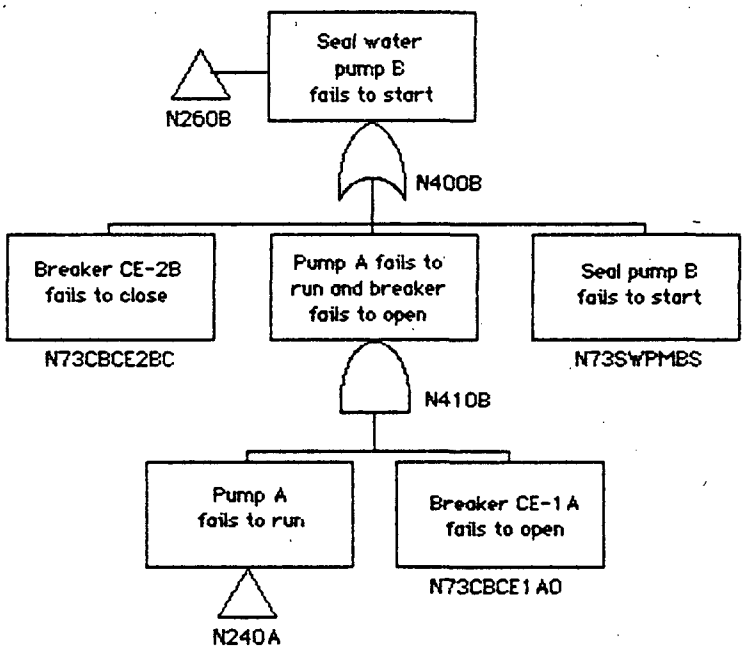

Loss of flow through segment A
N120
N200A
Non-condensible flow blocked in segment A
Segment A pump fails
N210A
N220A
AOV62A transfers closed
N73AV62AT
Strainer M1A plugs
N73ST1AP
Check valve V16A transfers closed
N73CV16AT
V2A transfers closed
N73XV2AT
V3A transfers closed
N73XV3AT
Loss of electric power to non 1E loads
EM324A
Pump A fails to run
N240A
Pump A fails to run
N73PMAR
Separator A fails
N270A
Seal water cooler fails, failing pump A
N250A
V6A transfers closed
N73XV6AT
V11A transfers closed
N73XV11AT
Cooler A fails
N73CHAF
Seal water pump A fails to run
N260A
Loss of electric power to non 1E loads
EM324A
Pump 102A fails to run
N73SWPMAR


Separator A
fails
N240A
N270A
V56A
transfers closed
N73XV56AT
SOV-57A
fails to open
N73SV57A0
Separator A
fails
N72SPAF
LR1A
fails high
N73LR1AA
V63A
Transfers closed
N73XV63AT
V64A
Transfers closed
N73XV64AT


Loss of flow through segment B
N120
N200B
Non-condensible flow blocked in segment B
Segment B pump fails
N210B
N220B
AOV62B transfers closed
N73AV62BT
Strainer M1B plugs
N73ST1BP
Check valve V16B transfers closed
N73CV16BT
V2B transfers closed
N73XV2BT
V3B transfers closed
N73XV3BT
Loss of electric power to non 1E loads
EM324A
Pump B fails to start
N300B
Pump B fails to run
N240B
Pump CE-1B fails to run
N73PMBR
Separator B fails
Seal water cooler fails, failing pump
N250B
Seal water pump B fails to run
N260B
V6B transfers closed
N73XV6BT
V11B transfers closed
N73XV11BT
Cooler B fails
N73CHBF
Seal water pump 102B fails to start
N400B
Loss of electric power to non 1E loads
EM324A
Pump 102B fails to run
N73SWPMBR


Separator B
fails
N240B
N270B
V56B
transfers closed
N73XV56BT
SOV-57B
fails to open
N73SV57B0
Separator
B fails
N73SPBF
Level transmitter
LI 1B
fails high
N73LR1BB
V64B
Transfers closed
N73XV64BT
V63B
Transfers closed
N73XV63BT


Pump B
fails to start
N220B
N300B
Breaker CE-2A
fails to close
N73CBCE2AC
Pump A fails to
run and breaker
fails to open
N310B
Pump B
fails to start
N73PMBS
Pump A
fails to run
N240A
Breaker CE-1A
fails to open
N73CBCE1AO


Seal water pump B fails to start
N260B
N400B
Breaker CE-2B fails to close
N73CBCE2BC
Pump A fails to run and breaker fails to open
N410B
Seal pump B fails to start
N73SWPMBS
Pump A fails to run
N240A
Breaker CE-1A fails to open
N73CBCE1A0

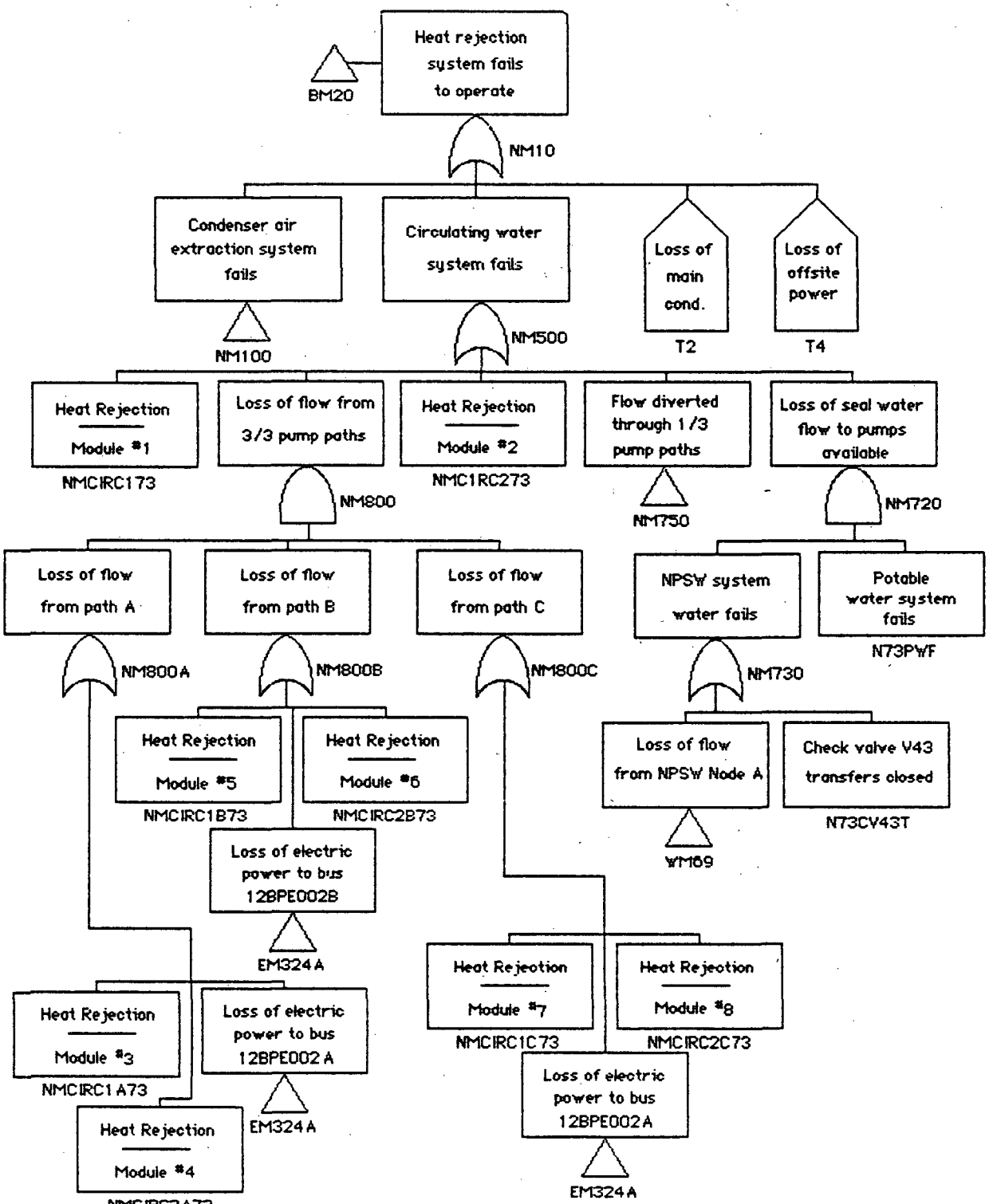
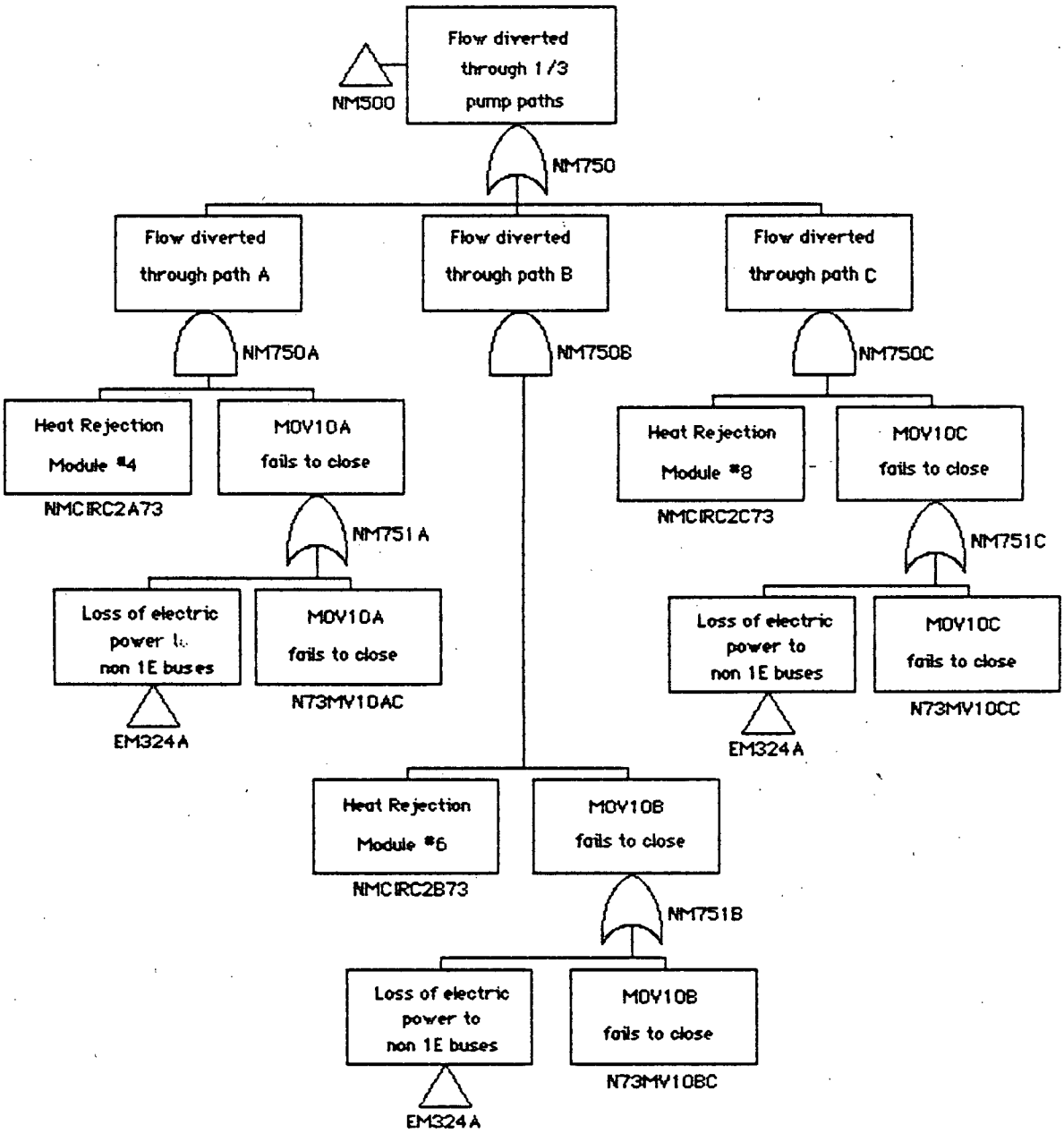


Figure A8-4. Heat Rejection Modularized Fault Tree

Page #1


Flow diverted through 1/3 pump paths
NM500
NM750
Flow diverted through path A
Flow diverted through path B
Flow diverted through path C
NM750A
NM750B
NM750C
Heat Rejection Module #4
NMCIRC2A73
MOV10A fails to close
NM751A
Loss of electric power to non 1E buses
EM324A
MOV10A fails to close
N73MV10AC
Heat Rejection Module #8
NMCIRC2C73
MOV10C fails to close
NM751C
Loss of electric power to non 1E buses
EM324A
MOV10C fails to close
N73MV10CC
Heat Rejection Module #6
NMCIRC2B73
MOV10B fails to close
NM751B
Loss of electric power to non 1E buses
EM324A
MOV10B fails to close
N73MV10BC

Page #2

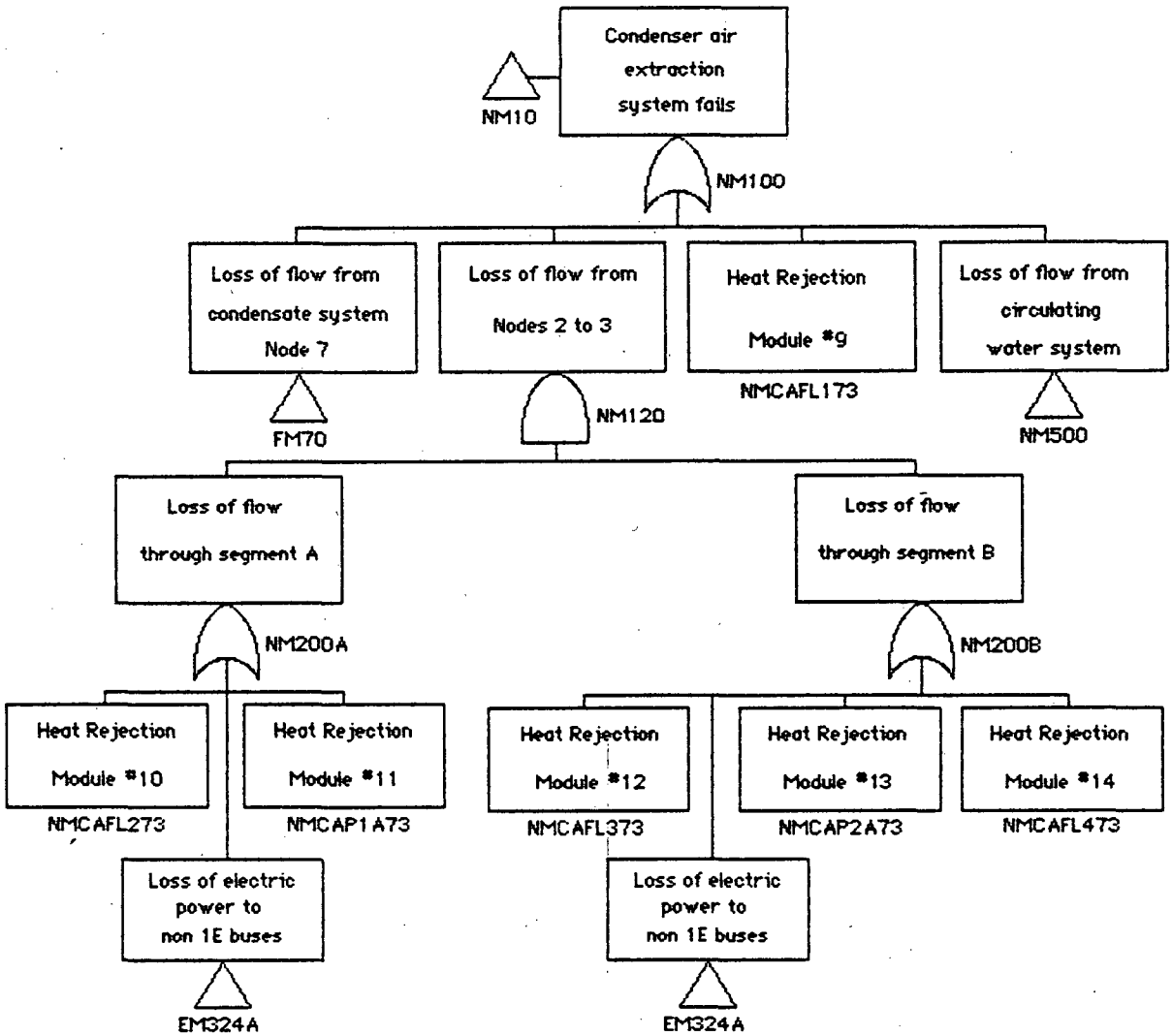

Condenser air extraction system fails
NM10
NM100
Loss of flow from condensate system Node 7
Loss of flow from Nodes 2 to 3
Heat Rejection Module #9
NMCAFL173
Loss of flow from circulating water system
FM70
NM120
NM500
Loss of flow through segment A
Loss of flow through segment B
NM200A
NM200B
Heat Rejection Module #10
NMCAFL273
Heat Rejection Module #11
NMCAP1A73
Heat Rejection Module #12
NMCAFL373
Heat Rejection Module #13
NMCAP2A73
Heat Rejection Module #14
NMCAFL473
Loss of electric power to non 1E buses
EM324A
Loss of electric power to non 1E buses
EM324A

Table A8-1

HEAT REJECTION SYSTEM MODULE DATA

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
NMCAP1A73		2.3-3			TAB-OR
N73XV63AT	Valve transfers	2.4-6	1.0-7	24	
N73PMAR	Pump fails to run	4.8-4	2.0-5	24	
N73XV6AT	Manual valve transfers	2.4-6	1.0-7	24	
N73XV11AT	Manual valve transfers	2.4-6	1.0-7	24	
N73CHAF	Chiller fails	2.4-5	1.0-6	24	
N73SWPMAR	Pump fails to run	4.8-5	2.0-5	24	
N73XV56AT	Manual valve transfers	2.4-6	1.0-7	24	
N73SV57AO	Sol. valve fails to open	1.0-3	1.0-3	-	
N73LR1AA	Level sensor fails high	7.2-4	3.0-5	24	
N73SPAF	Separator fails	ε	-	-	
N73XV64AT	Valve transfers	2.4-6	1.0-7	24	
NMCAFL373		2.7-3			TAB-OR
N73AV62BT	Air valve transfers	2.4-6	1.0-7	24	
N73CV16BT	Check valve transfers	5.5-6	2.3-7	24	
N73XV3BT	Manual valve transfers	2.4-6	1.0-7	24	
N73ST1BP	Strainer plugs	2.4-4	1.0-5	24	
N73XV2BT	Manual valve transfers	2.4-6	1.0-7	24	

Table A8-1 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
NMCAP2A73		4.7-3			TAB-OR
N73XV63BT	Valve transfers	2.4-6	1.0-7	24	
N73PMBR	Pump fails to run	4.8-4	2.0-5	24	
N73XV6BT	Manual valve transfers	2.4-6	1.0-7	24	
N73XV11BT	Manual valve transfers	2.4-6	1.0-7	24	
N73CHBF	Chiller fails	2.4-5	1.0-6	24	
N73SWPMBR	Pump fails to run	4.8-4	2.0-5	24	
N73XV56BT	Manual valve transfers	2.4-6	1.0-7	24	
N73SV57BO	Sol. valve fails to open	1.0-3	1.0-3	-	
N73LR1BB	Level sensor fails low	7.2-7	3.0-5	24	
N73PMBS	Pump (motor) fails to start	1.0-3	1.0-3	-	
N73SWPMBS	Pump fails to start	1.0-3			
N73XV64BT	Valve transfers	2.4-6	1.0-7	24	
NMCIRC1C73		9.1-6			TAB-OR
N73EJ4CP	Expansion joint plugged	6.7-6	2.8-7	24	
N73MV10CT	Motor valve transfers	2.4-6	1.0-7	24	

Table A8-1 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
NMCIRC2C73		3.0-4			TAB-OR
N73PMCR	Pump fails to run	4.8-4	2.0-5	24	
N73XV12CT	Manual valve transfers	2.4-6	1.0-7	24	
NMCRCFL273		1.8-5			
N73MV44AT	Motor valve transfers	2.4-6	1.0-7	24	(N73MV44AT+N73EJ5AP)*
N73EJ5AP	Expansion joint plugged	6.7-6	2.8-7	24	(N73MV44BT+N73EJ5BP)
N73MV44BT	Motor valve transfers	2.4-6	1.0-7	24	
N73EJ5BP	Expansion joint plugged	6.7-6	2.8-7	24	
NMCAFL173		4.8-6			
N73XV1AT	Manual valve transfers	2.4-6	1.0-7	24	N73XV1AT*N73XV1BT
N73XV1BT	Manual valve transfers	2.4-6	1.0-7	24	

Table A8-1 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
NMCAFL273		8.5-4			TAB-OR
N73AV62AT	Air valve transfers	2.4-6	1.0-7	24	
N73CV16AT	Check valve transfers	5.5-6	2.3-7	24	
N73XV3AT	Manual valve transfers	2.4-6	1.0-7	24	
N73ST1AP	Strainer plugged	2.4-4	1.0-5	24	
N73XV2AT	Manual valve transfers	2.4-6	1.0-7	24	
NMCIRC173		2.5-8			
N73CN1F	Condenser fails	ε	-	-	N73CNIF+N73CPF+
N73CPF	Control power fails	ε	-	-	[(N73EJ6AP+N73MV45AT)*
N73EJ6AP	Expansion joint plugs	6.7-6	2.8-7	24	(N73EJ6BP+N73MV45BT)]+
N73MV45AT	Motor valve transfers	2.4-6	1.0-7	24	[(N73XV36AT+N73SG6AT)+
N73EJ6BP	Expansion joint plugs	6.7-6	2.8-7	24	N73CTAF)*(N73XV36BT+ N73SG6BT+N73CTBF)]
N73MV45BT	Motor valve transfers	2.4-6	1.0-7	24	
N73XV36AT	Manual valve transfer	2.4-6	1.0-7	24	
N73SG6AT	Sluice gate transfers	2.4-6	1.0-7	24	
N73CTAF	Cooling tower fails to operate properly	ε	-	-	
N73XV36BT	Manual valve transfers	2.4-6	1.0-7	24	
N73SG6BT	Sluice gate transfers	2.4-6	1.0-7	24	
N73CTBF	Cooling tower fails to operate properly	ε	-	-	

Table A8-1 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
NMCIRC1A73		9.1-6			TAB-OR
N73EJ4AP	Expansion joint plugged	6.7-6	2.8-7	24	
N73MV10AT	Motor valve transfers	2.4-6	1.0-7	24	
NMCIRC2A73		4.8-4			TAB-OR
N73PMAR	Pump fails to run	4.8-4	2.0-5	24	
N73XV12AT	Manual valve transfers	2.4-6	1.0-7	24	
NMCIRC1B73		9.1-6			TAB-OR
N73EJ4BP	Expansion joint plugged	6.7-6	2.8-7	24	
N73MV10BT	Motor valve transfers	2.4-6	1.0-7	24	
NMCIRC2B73		4.81-4			TAB-OR
N73PMBR	Pump fails to run	4.8-4	2.0-5	24	
N73XV12BT	Manual valve transfers	2.4-6	1.0-7	24	

Table A8-1 (Continued)

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
N73PWF	Potable water fails	1.0	-	-	
N73CV43T	Check valve transfers	5.5-6	2.3-7	24	
N73MV10AC	MOV fails to close	1.0-3	1.0-3	-	
N73MV10BC	MOV fails to close	1.0-3	1.0-3	-	
N73MV10CC	MOV fails to close	1.0-3	1.0-3	-	

Appendix A, Section 9

NORMAL CHILLED WATER SYSTEM

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APPENDIX A, SECTION 9

NORMAL CHILLED WATER SYSTEM

A9.1 SYSTEM DESCRIPTION

The normal chilled water (NCW) system is designed as a closed chilled-water loop with water chillers, heat exchangers, recirculation pumps, an expansion tank, piping, valves, and instrumentation and controls.

A9.1.1 Function

The NCW system provides 42°F chilled water during normal plant operations for heat removal from nuclear island (NI) heating, ventilation, and air conditioning (HVAC); balance of plant (BOP) HVAC; recirculating gas cooling; NI general purpose maintenance equipment, inert gas receiving and processing; and BOP instrumentation and controls.

A9.1.2 Design

The NCW system is a closed chilled-water loop consisting of mechanical refrigeration water chillers that are cooled by the normal plant service water (NPSW) system, heat exchangers for the various components served by the NCW system, centrifugal circulating pumps, an expansion tank, piping, valves, and instrumentation and controls. The NCW system includes five normally operating pump/chiller sets and one standby pump/chiller set as shown in Figure A9-1. The NCW system has a minimum 10% refrigeration margin available during operation with simultaneous peak loads--42°F leaving chilled water temperature and 90°F entering service water temperature. The loads served by the NCW system are shown

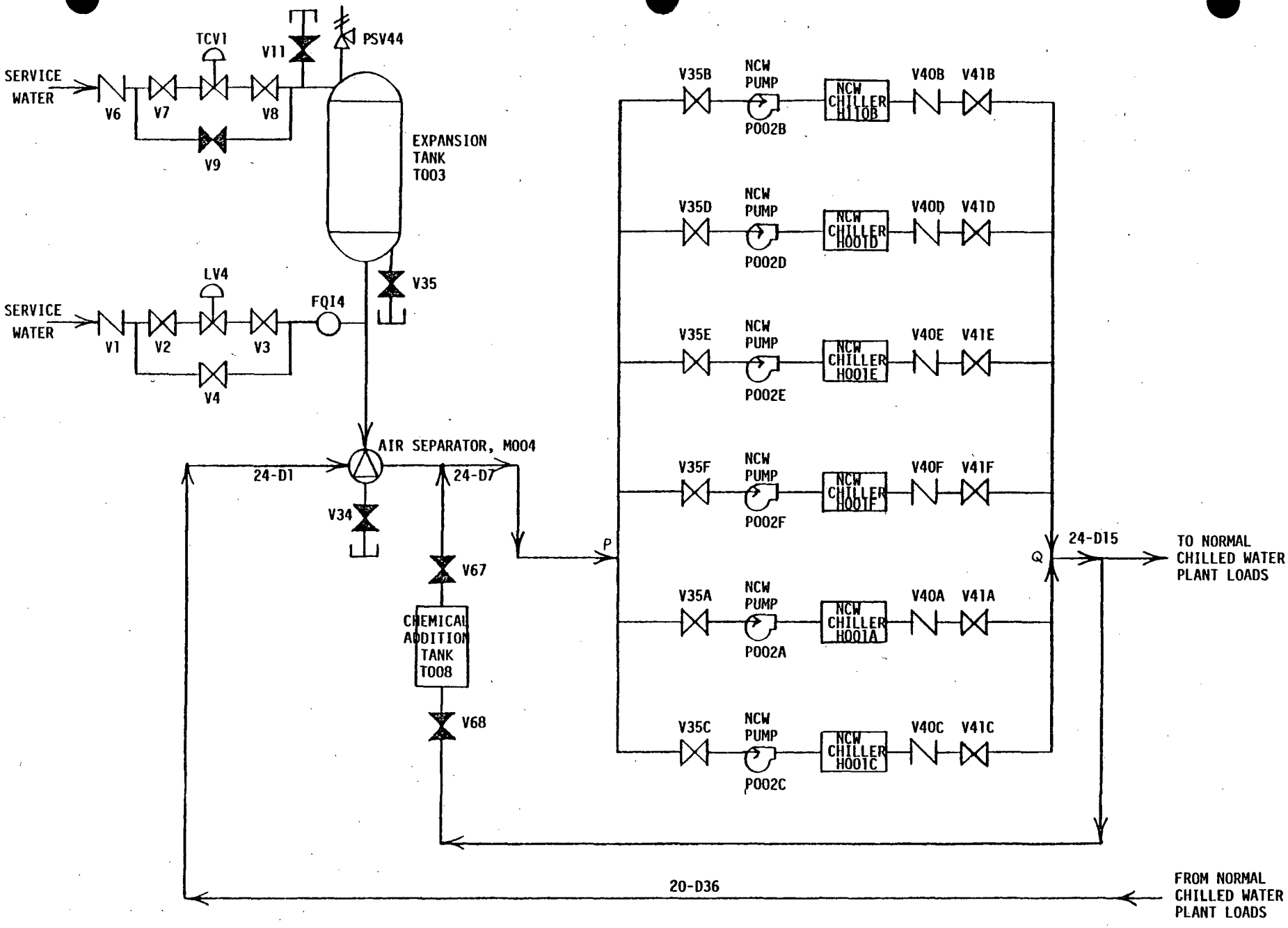

Figure A9-1. Normal Chilled Water System

in Figure A9-2. The above minimum margins are established on the basis of the initially baselined cooling load information from the various interfacing systems.

A9.1.3 Interfaces with Other Systems

The NCW system requires support from the following systems to satisfy its operational requirements.

A9.1.3.1 Electric power. The electric power system provides motive power, lighting, safety grounding, and cabling for the NCW system. Electric power is supplied to the following equipment in the NCW system:

Equipment	Bus No.	Breaker No.	Control Switch
NCW pump P002A	12NIE029A	PD-29A	HS-232A
NCW pump P002B	12NIE029B	PD-29B	HS-232B
NCW pump P002C	12NIE029A	PD-29A	HS-232C
NCW pump P002D	12NIE029B	PD-29B	HS-232D
NCW pump P002E	12NIE029A	PD-29A	HS-232E
NCW pump P002F	12NIE029B	PD-29B	HS-232F
NCW chiller H001A	12NIE003C	YH-3C	HS-281A
NCW chiller H001B	12NIE003D	YH-3D	HS-281B
NCW chiller H001C	12NIE003C	YH-3C	HS-281C
NCW chiller H001D	12NIE003D	YH-3D	HS-281D
NCW chiller H001E	12NIE003C	YH-3C	HS-281E
NCW chiller H001F	12NIE003D	YH-3D	HS-281F

The NCW pumps receive non-1E, 480-V power, and the chillers receive non-1E, 4160-V power from the normal electrical power system. The electric power system provides standard lighting levels for the NCW equipment rooms and permits communication with the control room.

A9.1.3.2 Control power. Control and instrumentation power for the NCW system is assumed to come from the same bus as the motive power. This provides power and instrumentation cabling and grounding for the NCW equipment.

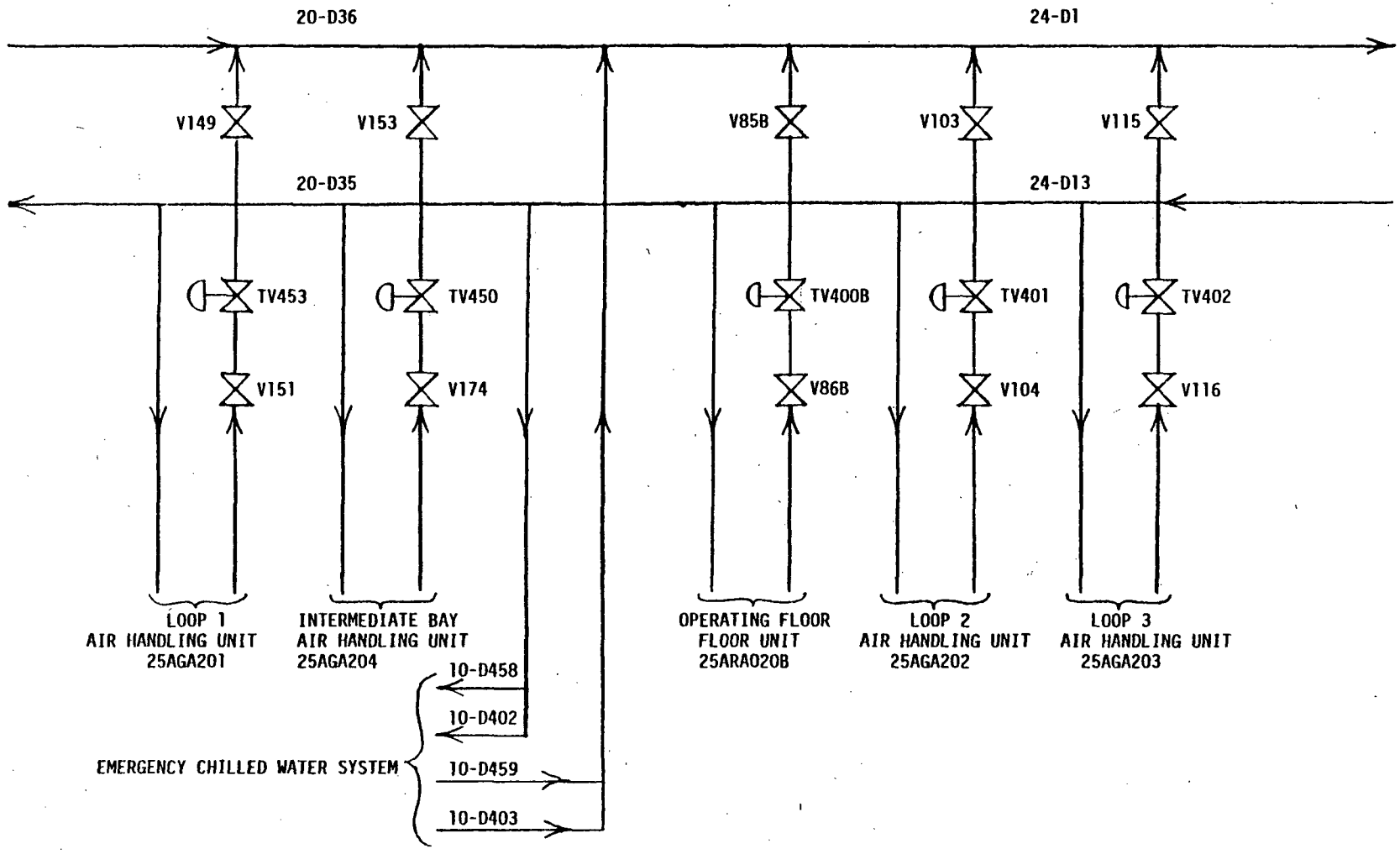

Figure A9-2. Normal Chilled Water System Plant Loads

A9.1.3.3 Compressed gas system. The compressed gas system provides dry, filtered, instrument air to all NCW pneumatic valve operators and instruments in the RCB, RSB, SGB, CB, and TGB. Service air is supplied to the expansion tank to regulate tank pressure, and air connections are provided for the maintenance of the NCW equipment.

A9.1.3.4 NI heating, ventilation, and air conditioning. The NI HVAC system provides the environmental conditions for a maximum normal plant operation heat removal rate of 301,900 btu/hr for cells containing major NCW system equipment.

A9.1.3.5 Normal plant service water system. Normal plant service water is supplied to all NCW chiller condensers with a maximum supply temperature of 90°F and a minimum supply temperature of 55°F.

A9.1.3.6 Balance of plant treated water system. The BOP treated water system supplies 80 gpm of chemically treated water to the NCW for system component flushing, initial fill, and makeup supply during NCW system operation.

A9.1.4 Operation

The NCW pumps take suction from the system return header. Five of the six pump/chiller sets operate during peak normal plant operation to provide the total design flow at the system design pressure drop. The sixth pump/chiller set serves as a standby and may be placed in service when required or on a rotating basis to equalize the operating time of all the sets. A minimum of three pump/chiller sets are required for successful operation. One butterfly isolation valve is installed in the

supply line to each pump. The discharge of each pump is directed through a pipe tee into the chilled water inlet of its associated chiller. On the outlet side of the chiller a check valve and a butterfly isolation valve are installed. All butterfly isolation valves are locked open.

The NCW system has the following active components:

- 20% capacity pumps (6)
- 20% capacity chillers (6)
- Dowtherm J circulation pump (1) for primary cold trap NaK cooler secondary loop
- Automatic valves

The NCW system is designed to have one standby pump and chiller that can be started manually upon failure of the operating equipment so the system can be operated at full-load capacity. Failure of more than one pump or chiller impairs the capability of the system to feed all loads.

Simultaneous failure of a pump and chiller not of the same pump/chiller set will not impair system performance because there is capability to cross-connect a pump and a chiller from different pump/chiller sets. The nonoperating pump or chiller is blanked off, and the cross-connected pump/chiller set restores system capability.

The NCW system cannot function if all the power supply sources of the normal ac distribution system fail. The NCW system is not a safety-related system, and therefore is not powered from an on-site, Class 1E power supply.

The NCW system cannot operate during a loss of the NPSW system. The NPSW system is designed with two 100 percent capacity pumps, making the loss of this system unlikely.

The NCW pumps and chillers are controlled manually from the local control panel. The number of pump/chiller sets to be kept operating at any time depends on the chilled water system load requirements. Chillers are provided with solid state controls, and the temperature of the water leaving the chillers is controlled to 42°F. Pump/chiller sets are placed in service and shutdown as indicated by the NCW system differential temperature indicator located on the master control panel in the main control room.

A9.1.5 Test and Technical Specification Requirements

A9.1.5.1 Testing. The NCW system is designed to be tested, as appropriate, to verify redundancy and electrical independence and to demonstrate that the system will perform as designed upon initial startup.

A9.1.5.2 Operational requirements. The system is designed to ensure adequate chilled water supply during plant operation and plant shutdown periods based on the following:

a. All equipment is operating at maximum design condition or load.

b. The chilled water pump/chiller sets and system piping shall supply 42°F chilled water to assigned loads.

A9.1.6 Maintenance Requirements

A9.1.6.1 Corrective maintenance. Most maintenance can be accomplished without affecting the system operation. Maintenance that can affect the

system operation should be accomplished on an individual component and subsystem basis. Corrective maintenance shall be performed in the following sequence to maximize plant availability:

a. Adjust or repair component in place.

b. Replace component with spares.

c. Remove, repair, and reinstall the component.

Corrective maintenance of components within a unit (or system) would be completed during shutdown.

A9.1.6.2 Preventive maintenance. Preventive maintenance consists of periodic, planned, and scheduled inspections and maintenance to determine and correct conditions prior to excessive wear or component failure. The flowmeters are checked quarterly for calibration and cleanliness. Temperature and pressure indicators are checked for calibration and adjusted as needed. Control panels and switches are examined for loose connections and contacts as operating conditions permit. Valves are inspected for packing leakage, scale buildup, and gasket leakage as required to maintain equipment in good operating condition.

A9.2 SYSTEM MODEL

The NCW fault tree details the potential failure modes of the system and is shown in Figure A9-3.

A9.2.1 Modeling Assumptions

a. The NCW system is assumed to be sufficiently charged, aligned, and operating satisfactorily with five of the six pump/chiller sets in operation and the sixth set in standby. The running pump/chiller sets are A through E, and F is in standby. When

standby pump/chiller set F is required, the operator must manually start both the pump and the chiller.

b. It is assumed that the design engineer has properly sized the piping system to meet the requirements of the plant loads.

c. Mission time for the analysis is assumed to be 24 hours. Most passive failures are neglected, based on the short mission time and low failure rate ($\leq 1 \times 10^{-6}$/hr).

d. The NCW system is treated as a closed loop system without considering makeup water from the plant service water system.

e. Strainers to pumps are used only during initial charging of the system; therefore, they are not modeled.

f. The secondary chilled water Dowtherm-J to water loop is not modeled. Failures in this loop will not cause the NCW system to fail to accomplish its mission under shutdown conditions.

g. The term "transfers" implies going from a desirable or required position to an undesirable or a failed position.

h. Low chilled water temperature is not modeled; however, the chillers have a protection device (TSL 40A-F) that trips on low chilled water temperature.

i. Normal plant operation shows chiller sets A-E in operation, with F in standby. Emergency plant operation requires operation of at least three chiller sets. A failure of four of the six chiller sets is modeled as a system failure.

j. Due to the failure criteria of four out of six chiller sets for system failure, testing and maintenance that would disable a chiller set are not modeled.

k. Electrical power for the NCW pumps is supplied by non-1E buses 12NIE003C and 12NIE003D, which are modeled by a transfer to gate EM324.

l. HVAC is modeled as common to pump trains A-F, as the pumps are located in a common cell.

m. Failure of the expansion tank is not modeled because of low failure rate and 24-hour mission time.

n. No common-mode failures exist in the return line of the NCW system.

o. Only valves in common headers are modeled, since their failure has a more serious impact on system performance than valves in isolated headers.

p. Chilled water temperature greater than 42°F is considered a loss of system capability to provide adequate quality chilled water. This failure is modeled in compressor failures as well as loss of service water to the chillers.

A9.2.2 Detailed Fault Tree Model

The detailed fault tree for the NCW system is shown in Figure A9-3.

A9.2.2.1 Success criteria. Success of the NCW system involves the distribution of adequate quality chilled water to normal plant loads via common "tie" piping (both supply and return). The term "adequate quality" was defined as the flow, pressure, and temperature required by system loads at peak operating and designed efficiency.

A9.2.2.2 Top events. The top events for the NCW system are "events that demand the emergency chilled water system" and "less than adequate chilled water to emergency loads from the NCW system." The first top event represents those support system failures that would degrade the NCW system's capability to supply chilled water to the emergency loads. The second top event represents failure or failures within the NCW system's components which, if failed, would degrade the system's capability.

A9.2.2.3 Transfers to other systems. The support-system requirements for the NCW system are modeled by transfers to the gates in the appropriate system fault trees as indicated below.

Equipment	Support System	Transfer
Chiller H001A	NPSW	W79
Chiller H001B	NPSW	W81
Chiller H001C	NPSW	W83
Chiller H001D	NPSW	W85
Chiller H001E	NPSW	W87
Chiller H001F	NPSW	W89
Pumps P002A-P002F	Electric power	EM324
Pumps P002A-P002F	HVAC	H9A

A9.2.3 Modularized Fault Tree Model

The NCW system fault tree model was simplified by combining groups of independent basic events to form modules. The modified structure is shown in the modularized fault tree of Figure A9-4.

A9.3 FAILURE DATA

A9.3.1 Basic Event Data

The probabilities of the basic events in the detailed fault tree were evaluated by applying the data from Sections 5 and 6. The events and their probabilities are summarized in Table A9-1.

A9.3.2 Module Data

Probabilities for the modules were calculated by combining the values for the basic events according to the definitions of the modules. These values are also reported in Table A9-1.

A9.4 SYSTEM-LEVEL RESULTS AND INSIGHTS

Failures in the NI HVAC, 4160/480 V ac electric power, and NPSW systems combine with failures in the NCW system to cause loss of cooling to the various NCW loads. Each air-handling unit cooled by NCW is served by an independent loop. Any failure defined above can cause simultaneous loss of cooling to air-handling units 24AGA201, 25AGA202, and 25AGA203, minimizing the benefits of a redundant system.

A9.5 REFERENCES

1. System Design Description, Chilled Water System (SDD 23), Revision 30 (1/83).

2. P&I Diagrams:

   NN550, Rev. 7
   NN551, Rev. 6
   NN552, Rev. 4
   NN553, Rev. 6
   NN554, Rev. 9
   NN555, Rev. 7
   NN556, Rev. 13
   NN557, Rev. 10
   NN556, Rev. 9

3. Logic Diagrams:

   NE4489, Rev. 0
   NE4490, Rev. 5
   NE4491, Rev. 5
   NE4492, Rev. 5
   NE4493, Rev. 5
   NE4494, Rev. 5
   NE4495, Rev. 5
   NE4496, Rev. 5
   NE4498, Rev. 5
   NE4499, Rev. 5
   NE4503, Rev. 4
   NE4504, Rev. 5
   NE4505, Rev. 5
   NE4506, Rev. 5
   NE4507, Rev. 5
   NE4508, Rev. 5
   NE4509, Rev. 4
   NE4510, Rev. 4
   NE4511, Rev. 4
   NE4512, Rev. 5
   NE4533, Rev. 5
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   NE4539, Rev. 5
   NE4540, Rev. 5
   NE4541, Rev. 5
   NE4542, Rev. 4

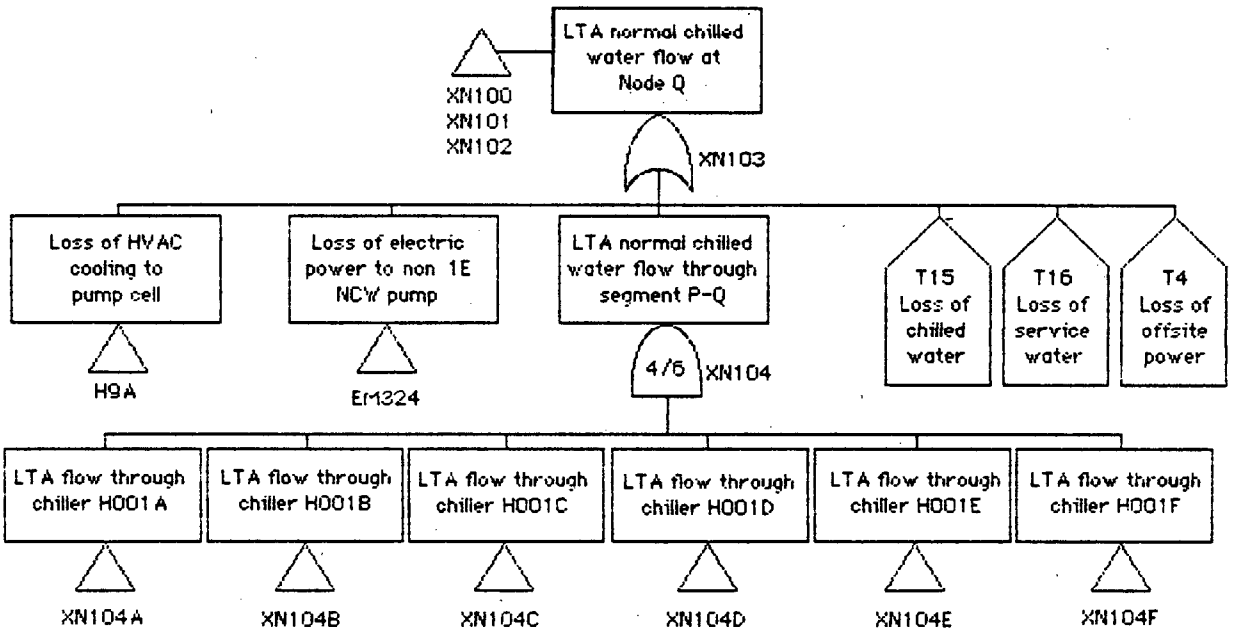
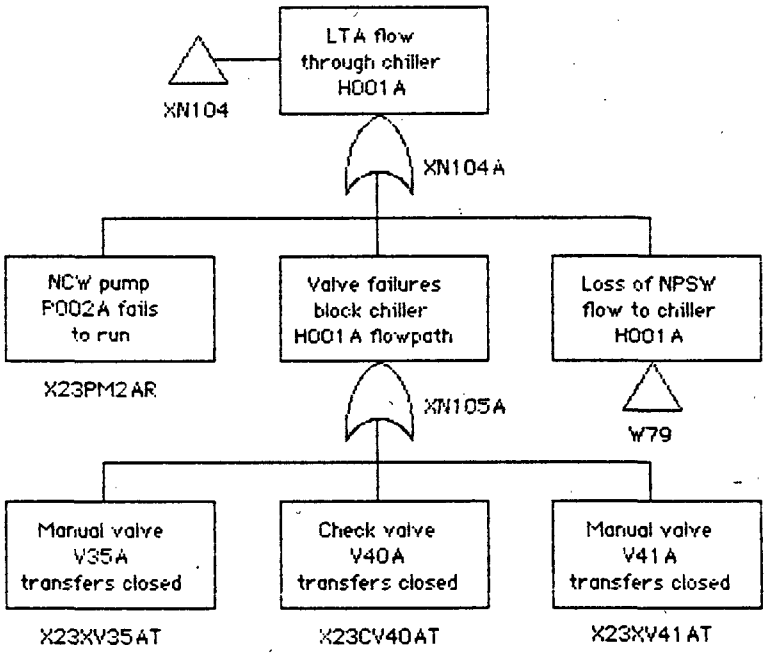
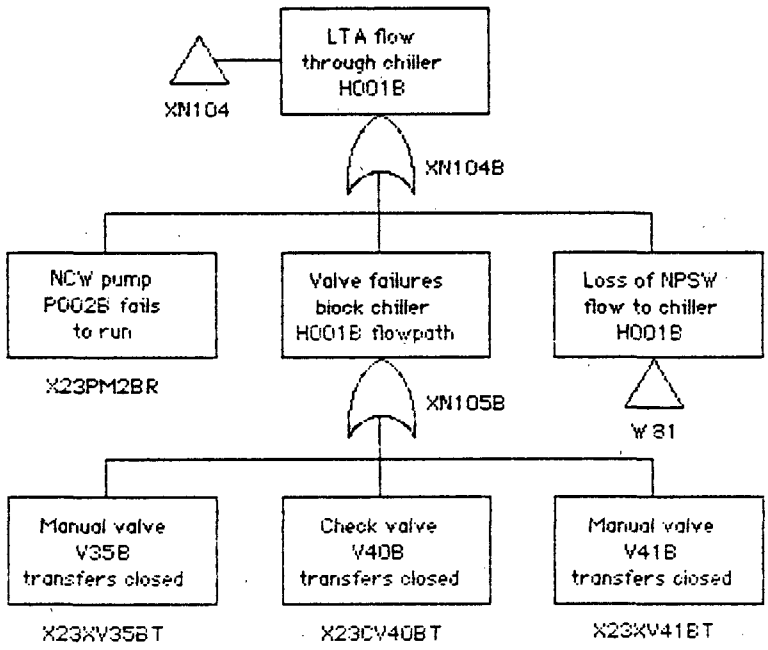


Figure A9-3. Detailed Fault Tree for Normal Chilled Water System.


LTA flow through chiller H001A
XN104
XN104A
NCW pump P002A fails to run
X23PM2AR
Valve failures block chiller H001A flowpath
XN105A
Loss of NPSW flow to chiller H001A
W79
Manual valve V35A transfers closed
X23XV35AT
Check valve V40A transfers closed
X23CV40AT
Manual valve V41A transfers closed
X23XV41AT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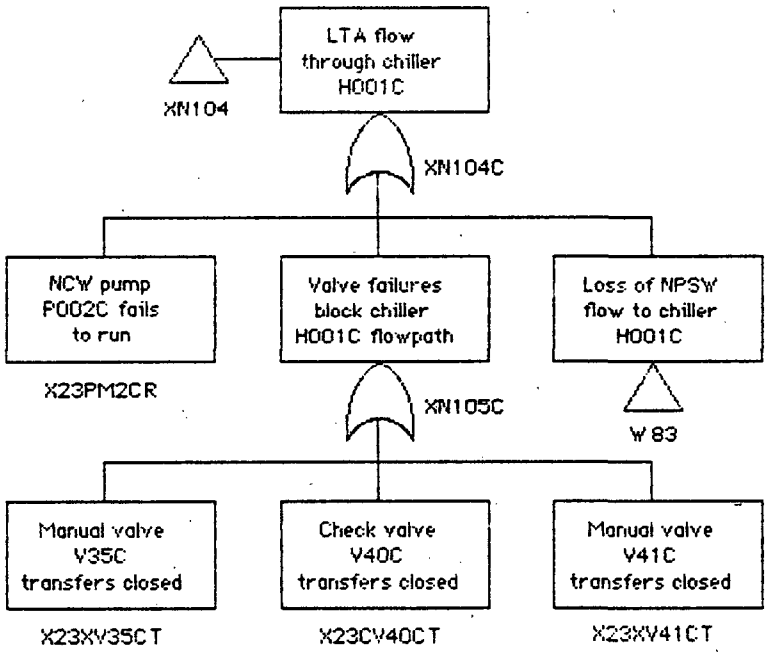
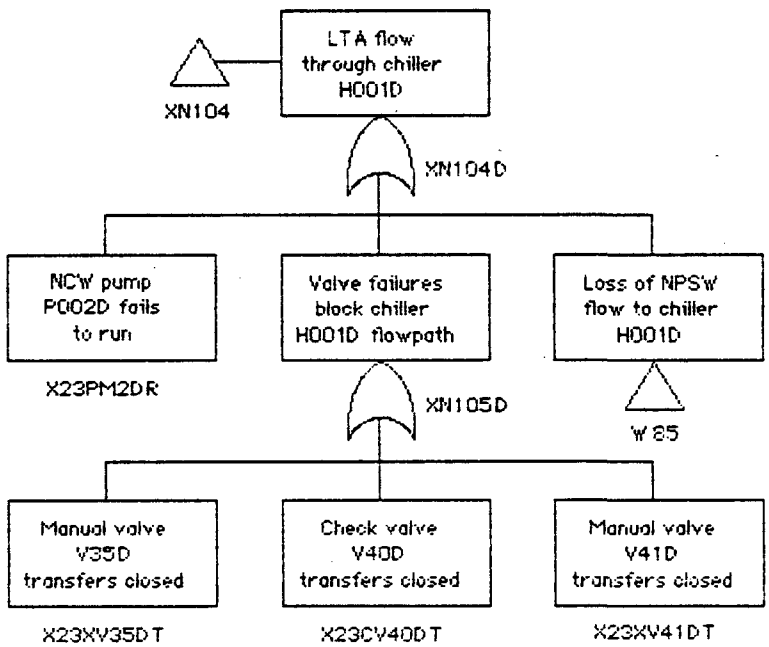
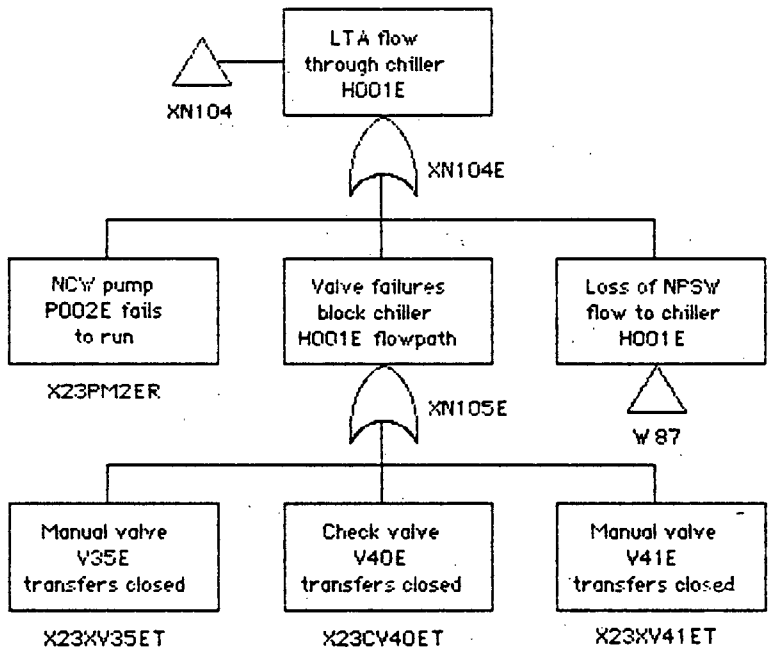
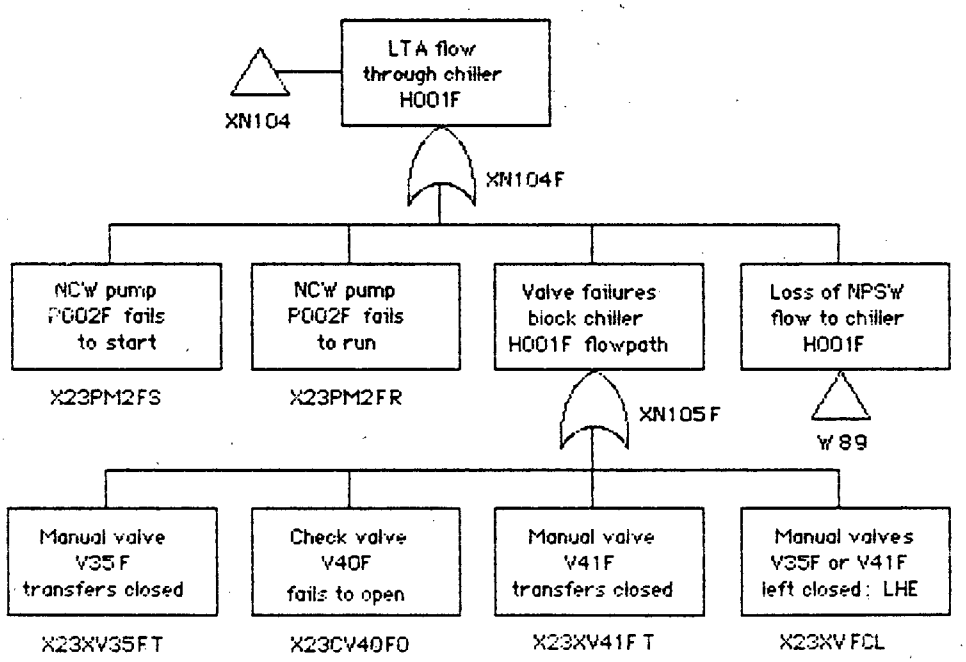
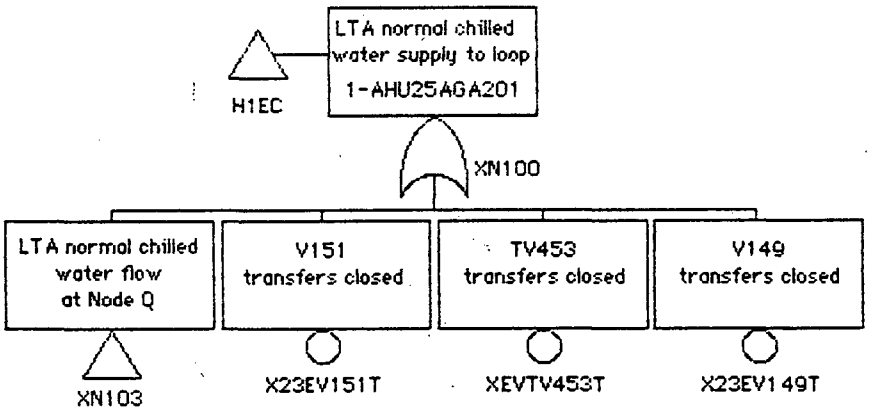
XN104
XN104B
NCW pump P002B fails to run
X23PM2BR
Valve failures block chiller H001B flowpath
XN105B
Loss of NPSW flow to chiller H001B
W81
Manual valve V35B transfers closed
X23XV35BT
Check valve V40B transfers closed
X23CV40BT
Manual valve V41B transfers closed
X23XV41BT

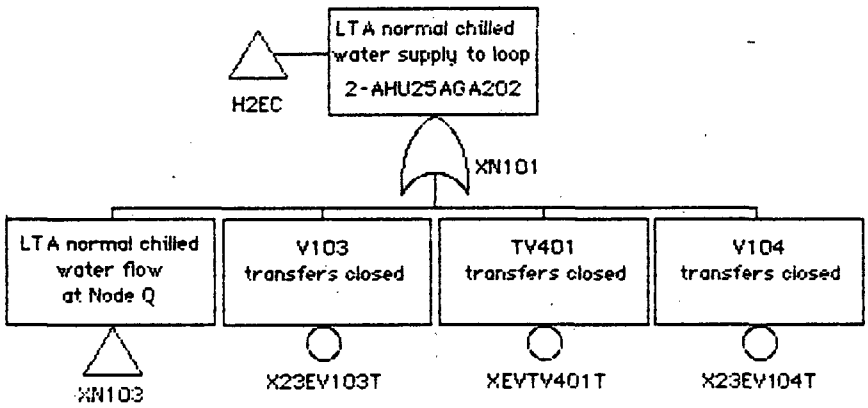

LTA flow through chiller H001C
XN104
XN104C
NCW pump P002C fails to run
X23PM2CR
Valve failures block chiller H001C flowpath
XN105C
Loss of NPSW flow to chiller H001C
W83
Manual valve V35C transfers closed
X23XV35CT
Check valve V40C transfers closed
X23CV40CT
Manual valve V41C transfers closed
X23XV41CT

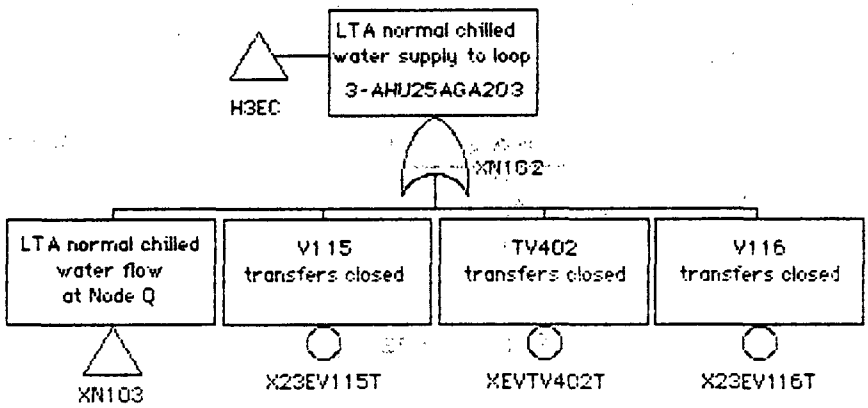

LTA flow through chiller H001D
XN104
XN104D
NCW pump P002D fails to run
X23PM2DR
Valve failures block chiller H001D flowpath
XN105D
Loss of NPSW flow to chiller H001D
W85
Manual valve V35D transfers closed
X23XV35DT
Check valve V40D transfers closed
X23CV40DT
Manual valve V41D transfers closed
X23XV41DT

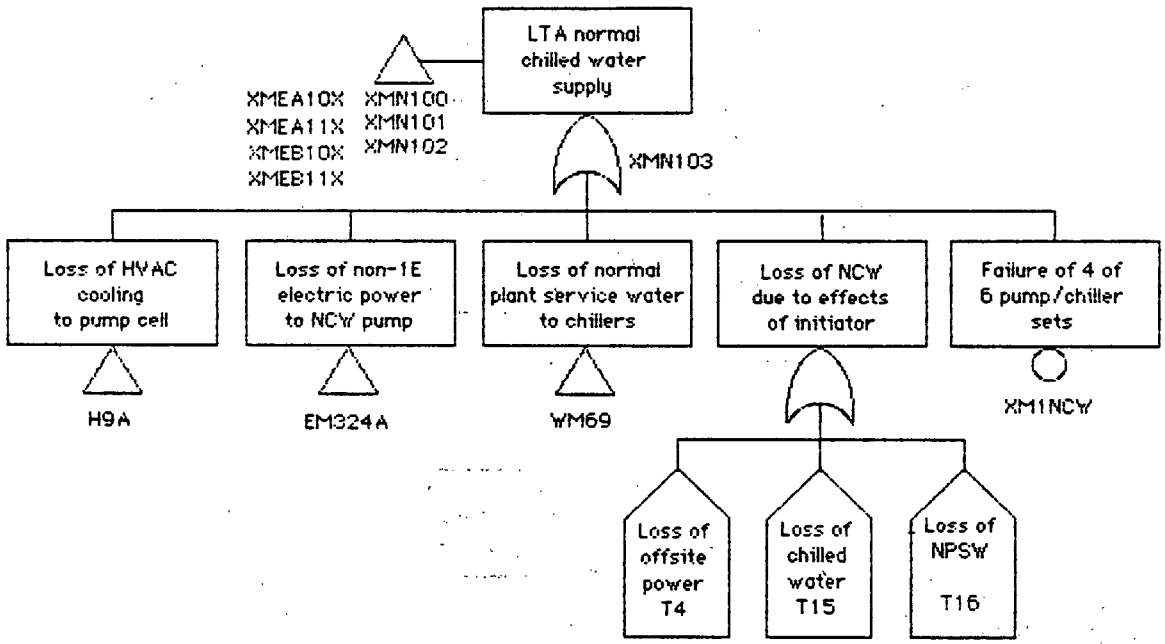

LTA flow through chiller H001E
XN104
XN104E
NCW pump P002E fails to run
X23PM2ER
Valve failures block chiller H001E flowpath
XN105E
Loss of NPSW flow to chiller H001E
W87
Manual valve V35E transfers closed
X23XV35ET
Check valve V40E transfers closed
X23CV40ET
Manual valve V41E transfers closed
X23XV41ET


LTA flow through chiller H001F
XN104
XN104F
NCW pump P002F fails to start
X23PM2FS
NCW pump P002F fails to run
X23PM2FR
Valve failures block chiller H001F flowpath
XN105F
Loss of NPSW flow to chiller H001F
W89
Manual valve V35F transfers closed
X23XV35FT
Check valve V40F fails to open
X23CV40FO
Manual valve V41F transfers closed
X23XV41FT
Manual valves V35F or V41F left closed: LHE
X23XVFCL


LTA normal chilled water supply to loop 1-AHU25AGA201
H1EC
XN100
LTA normal chilled water flow at Node Q
V151 transfers closed
TV453 transfers closed
V149 transfers closed
XN103
X23EV151T
XEVTV453T
X23EV149T


LTA normal chilled water supply to loop 2-AHU25AGA202
H2EC
XN101
LTA normal chilled water flow at Node Q
V103 transfers closed
TV401 transfers closed
V104 transfers closed
XN103
X23EV103T
XEVTV401T
X23EV104T


LTA normal chilled water supply to loop 3-AHU25AGA203
H3EC
XN102
LTA normal chilled water flow at Node Q
V115 transfers closed
TV402 transfers closed
V116 transfers closed
XN103
X23EV115T
XEVTV402T
X23EV116T



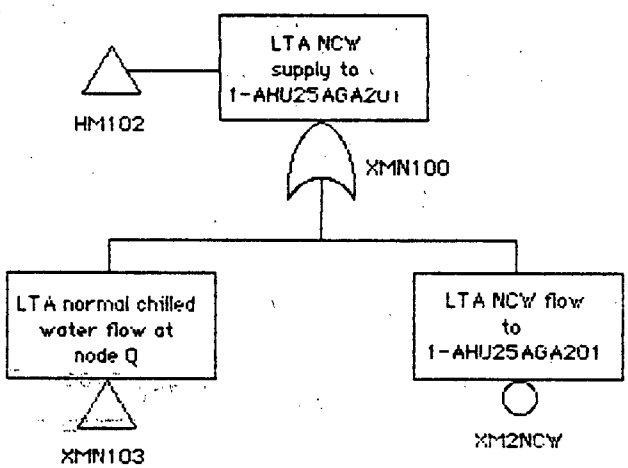
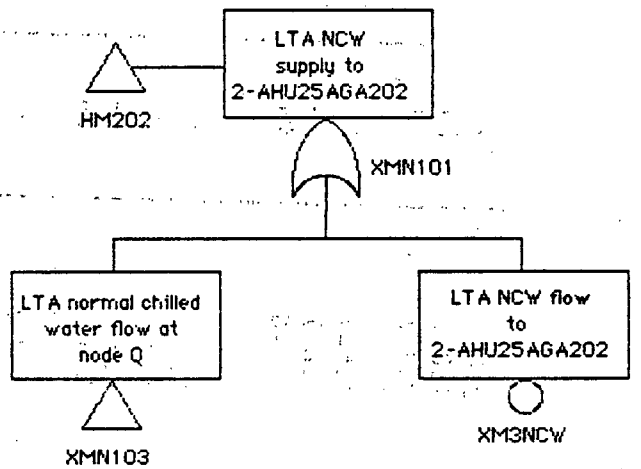


Figure A9-4. Modularized Fault Tree for Normal Chilled Water System.


LTA NCW supply to 2-AHU25AGA202
HM202
XMN101
LTA normal chilled water flow at node Q
LTA NCW flow to 2-AHU25AGA202
XMN103
XM3NCW

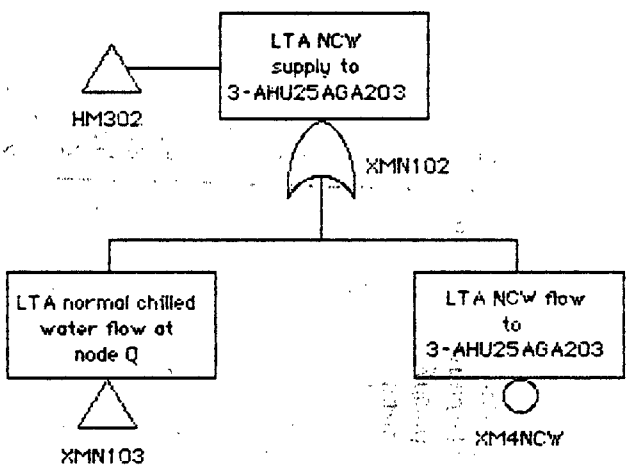

LTA NCW supply to 3-AHU25AGA203
HM302
XMN102
LTA normal chilled water flow at node Q
LTA NCW flow to 3-AHU25AGA203
XMN103
XM4NCW

Table A9-1

NORMAL CHILLED WATER DATA TABLE

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
XM1NCW					
A	Loss of flow through 4 of 6 pump/chiller sets	ε			Combination of 4 of 6 events, one each from A through F
X23PM2AR		4.8-4	2.0-5	24	
X23XV35AT		2.4-6	1.0-7	24	
X23CV40AT		5.5-6	2.3-7	24	
X23XV41AT		2.4-6	1.0-7	24	
W75AXV11AT		2.4-6	1.0-7	24	
W75AXV19AT		2.4-6	1.0-7	24	
W75ACH1AP		2.4-5	1.0-6	24	
B					
X23PM2BR		4.8-4	2.0-5	24	
X23XV35BT		2.4-6	1.0-7	24	
X23CV40BT		5.5-6	2.3-7	24	
X23XV41BT		2.4-6	1.0-7	24	
W75AXV19BT		2.4-6	1.0-7	24	
W75AXV11BT		2.4-6	1.0-7	24	
W75ACH1BP		2.4-5	1.0-6	24	
C					
X23PM2CR		4.8-4	2.0-5	24	
X23XV35CT		2.4-6	1.0-7	24	
X23CV40CT		5.5-6	2.3-7	24	
X23XV41CT		2.4-6	1.0-7	24	
W75AXV11CT		2.4-6	1.0-7	24	
W75AXV19CT		2.4-6	1.0-7	24	
W75ACH1CT		2.4-6	1.0-7	24	
		2.4-5	1.0-6	24	

Table A9-1

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
D					
X23PM2DR		4.8-4	2.0-5	24	
X23XV35DT		2.4-6	1.0-7	24	
X23CV40DT		5.5-6	2.3-7	24	
X23XV41DT		2.4-6	1.0-7	24	
W75AXV11DT		2.4-6	1.0-7	24	
W75AXV19DT		2.4-6	1.0-7	24	
W75ACH1DT		2.4-5	1.0-6	24	
E					
X23PM2ER		4.8-4	2.0-5	24	
X23XV35ET		2.4-6	1.0-7	24	
X23CV40ET		5.5-6	2.3-7	24	
X23XV41ET		2.4-6	1.0-7	24	
W75AXV11ET		2.4-6	1.0-7	24	
W75AXV19ET		2.4-6	1.0-7	24	
W75ACH1ET		2.4-6	1.0-7	24	
		2.4-5	1.0-6	24	
F					
X23PM2FS		1.0-3	1.0-3	-	
X23PM2FR		4.8-4	2.0-5	24	
X23XV35FT		2.4-6	1.0-7	24	
X23CV40FO		1.0-4	1.0-4	-	
X23XV41FT		2.4-6	1.0-7	24	
X23XVFCL		1.0-3	1.0-3	-	
W75AXV11FT		2.4-6	1.0-7	24	
W75AXV19FT		2.4-6	1.0-7	24	
W75ACH1ET		2.4-5	1.0-6	24	

Table A9-1

Designator	Description	Failure Probability	Mean Failure Rate (hr^{-1}) or Demand Failure	Mean Duration (hrs)	Module Definition
XM2NCW		7.2-6			TAB-OR
X23EV151T	Loop 1 - AHU	2.4-6	1.0-7	24	
X23EV149T	25AGA201	2.4-6	1.0-7	24	
XEVTV453T		2.4-6	1.0-7	24	
XM3NCW		7.2-6			TAB-OR
X23EV103T	Loop 2 - AHU	2.4-6	1.0-7	24	
XEVTV401T	25AGA202	2.4-6	1.0-7	24	
X23EV104T		2.4-6	1.0-7	24	
XM4NCW		7.2-6			TAB-OR
X23EV115T	Loop 3 - AHU	2.4-6	1.0-7	24	
XEVTV402T	25AGA203	2.4-6	1.0-7	24	
X23EV116T		2.4-6	1.0-7	24	