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Analysis Report

PRELIMINARY DESIGN ASSIST PSA LEVEL 1 - SELECTED FULL POWER EVENT TREES

ACR-700

10810-03660-AR-001

Revision 1

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ACRONYMS IN TEXT

ACND	Auxiliary Condensate (system)
ACR	Advanced CANDU Reactor
AFW	Auxiliary Feed Water (subsystem)
AFW-IS	Auxiliary Feed Water – Isolation (failure)
ASDV	Atmospheric Steam Discharge Valve
ADW	Auto De-pressurization Water (system)
AV-DGS<3	Less Than Three Diesel Generator Sets are Available
BPCC	Boiler Pressure Control Cooldown (subsystem)
CC	Crash Cool-down (of steam generators)
CLPRV	No LOCA via D_2O Storage Tank (LRVs fail to reclose)
CLPS	Consequential LOCA via pump seals
CND	Condensate (system)
CSDV	Condenser Steam Dump Valve
D1SW	Division 1 Service Water
DECC	Dormant Emergency (Core) Cooling (same as ECI)
DG	Diesel Generator
DG-AV=1	1 Diesel Generator Set Available
ECC	Emergency Core Cooling
ECI	Emergency Coolant Injection (system)
EFW	Emergency Feed Water (supply from RWS to SG)
ET	Event Tree
FBIO	Feeder Break
FSB	Feeder Stagnation Break
FW	Feed Water systems (includes MFW and AFW subsystem)
FWBA	Asymmetric feed water line break downstream of the SG check valve
FWBS	Symmetric feed water line break upstream of the feed water LCVs
HEP	Human Error Probability
HT	Heat Transport
HTS	Heat Transport System (same as RCS)

IE	Initiating Event
IE-FBIO	Initiating Event Feeder Break
IE-FSB	Initiating Event Feeder Stagnation Break
IE-FWBA	Initiating Event Asymmetric feed water line break downstream of the SG check valve
IE-FWBS	Initiating Event Symmetric Feed Water Line Break Upstream of the Feed Water LCVs
IE-LCL4	Initiating Event Total Loss Of Class IV Power Supply
IE-LOR	Initiating Event Loss of Regulation
IE-MSL3	Initiating Event Small Steam Discharge Causing Low Level in the Deaerator
IE-PCTR	Initiating Event Small LOCA - Pressure Tube & Calandria Tube Rupture
IE-PTR	Initiating Event Small LOCA - Pressure Tube Rupture
IE-SCB	Initiating Event Loss of Inventory in the Shield Cooling System
IE-SWD2	Initiating Event – Total Loss of Service Water of Division 2
LCL4	Consequential Loss of Class IV (4) power supply
LCV	Level Control Valve
LOCA	Loss of Coolant Accident
LOECC	Loss of Emergency Core Cooling
LOR	Loss of Regulation
LRV	Liquid Relief Valve
LTC	Long Term Cooling (system)
LTC-ECC	Long Term Cooling system - Emergency Core Cooling function
LTC-SDC	Long Term Cooling system - Shutdown Cooling function
MFW	Main Feed Water (subsystem)
MHS	Moderator (system) acting as active Heat Sink
MSL3	Small Steam Line Break – causing Low Deaerator Level
MSSV	Main Steam Safety Valve
OBPCC	Operator Starts Boiler Pressure Control Cooldown
OEFW	Operator opens motorized valves for EFW water supply to steam generators
OMSSV	Operator Opens MSSVs
OLTC-SDC	Operator Starts Long Term Cooling System in Shutdown Cooling Mode

ORS	Operator Shuts Down the Reactor from the Main Control Room
ORWS-HTS	Operator Starts Reserve Water Make-up to the Heat Transport System
PDS	Plant Damage State
PTHT	Heat Transport Pumps trip on high upper bearing temperature
PSA	Probabilistic Safety Assessment
PTR	Pressure Tube Rupture
PCTR	Pressure Tube / Calandria Tube Rupture
RCS	Reactor Coolant System (same as HTS)
ROP	Regional Overpower Protection
RRS	Reactor Regulating System (Setback and /or Stepback)
R60E4	Operator Action – Switchyard restored in 60 minutes
RS	Reactor Shutdown by SDS1 /SDS2
RWS	Reserve Water System
RWS-HTS	Water makeup from Reserve Water System into the Heat Transport System
RWT	Reserve Water Tank (a component of RWS)
SCB	Shield Cooling Break
SCDF	Severe Core Damage Frequency
SDS1	Shutdown System #1
SDS2	Shutdown System #2
SG	Steam Generator
SGPR	Steam Generator Pressure Relief (system; includes ASDVs, CSDVs and MSSVs)
SWD1&D2	Service Water (system) Division 1 & Division 2
TBD	To Be Determined

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1. INTRODUCTION

This preliminary event tree (ET) analysis report examines the responses of Advanced CANDU Reactor™ (ACR™)* to selected internal initiating events (IEs) that significantly impact Severe Core Damage Frequency (SCDF) in existing CANDU®† reactors. Design targets for SCDF values in the ACR are < 10^{-7} per year for an individual sequence, < 10^{-6} per year for summed internal events and < 10^{-5} per year for summed internal and external events to be evaluated for a mission time of 24 hours. The first two targets are guides, which are expected to yield the summed SCDF for internal and external events prescribed by Reference [1]. This is why the focus of this report is on the SCDF.

The main purpose of this report is to provide early inputs to the design teams regarding the reliability/unavailability requirements on the ACR systems that are used for accident mitigation as well as feedbacks on some of the system performance requirements. These inputs/feedbacks are a part of iterative process in which the reactor design is finalized and optimized without compromising nuclear safety.

The ACR is an evolutionary CANDU reactor design. Active systems used for accident mitigation are functionally similar to those in the existing CANDU reactors. However, improvements to the redundancy and reliability of these systems have been made and some of the system performance characteristics have been enhanced (Reference [2]). A new design feature of the ACR is the Reserve Water System (RWS), which delivers emergency water by gravity into various process systems to provide and/or facilitate an essentially passive, interim[1] heat sink.

At the time when the event trees in this report were developed, some of the ACR parameters were tentative. In particular, the passive water supplies from the RWS to process systems other than steam generators were not finalized. As a result, the passive accident mitigation features of the ACR are not treated systematically and comprehensively in this report. The Emergency Feed Water (EFW) supply from the RWS is modelled, but the gravity water supplies to the other process volumes (i.e., to Heat Transport System (HTS)[2], Calandria Vessel, End Shield Tanks and Shield Tank) are not credited unless an ‘individual-sequence’ SCDF is close to its acceptance value. In two of the event trees, the event trees were expanded to include the gravity water supply to the HTS. Models of all passive water supplies from the RWS will be included in the

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* ACR™ (Advanced CANDU Reactor™) is a trademark of Atomic Energy of Canada Limited (AECL).

† CANDU® (CANada Deuterium Uranium) is a registered trademark of Atomic Energy of Canada Limited.

1 An interim heat sink would typically last for the mission time of this report (24 hours) and its timely availability is treated as ‘success’ in preventing fuel damage. However, an active heat sink is required in the long term, which is implicitly assumed to be provided.

2 The US terminology uses the Reactor Coolant System (RCS) instead of HTS. This report uses the Canadian term and acronym.

future. Meanwhile, the preliminary analyses in this report should be viewed as 'screening assessments' that identify accident sequences leading to Severe Core Damage (SCD) and the features that contribute most to these sequences (e.g., hardware failures, system cross-link failure or post-accident human error).

In the spirit of screening assessment, uncertainties associated with actions of active mitigating systems are treated conservatively. With only few deterministic analyses of accident progression being available when this work was performed and gaps in some equipment performance data, judgements guided by analyses for existing CANDU reactor are employed. For circumstances where unambiguous judgements are difficult, the worst event outcome is presumed. An example is assuming that the Emergency Core Cooling (ECC) function is unavailable following a small break when the steam generator cool down is not available to reduce the HTS pressure[3]. Design-assist deterministic analyses are in progress to facilitate realistic modelling in the future. Meanwhile, the preliminary analyses in this report intentionally employ conservative assumptions.

Preliminary system reliability/unavailability design targets are defined for use in event tree analysis in order to estimate individual accident sequence frequencies. These targets are based on simple fault tree analyses of the early ACR design and on the experience with Probabilistic Safety Assessments (PSAs) of existing CANDU reactors. They are now being verified by detailed fault tree analysis of the latest ACR design configuration.

Results indicate that the ACR design can meet the prescribed SCDF targets.

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[3] Small breaks cover a broad range of coolant discharge rates, the larger of which would depressurize the HTS sufficiently to permit the ECIS injection without the SG cool down. The break discharge threshold for injection not assisted by SG cool down has not yet been determined for the ACR.

2. OBJECTIVE

The purpose of the preliminary event tree analysis is to identify which internal event sequences will likely dominate the SCDF in the ACR and which elements of the dominant sequences contribute most to the SCDF. This is a design-assist exercise performed in the early stages of design. Insights are useful for design finalization and optimization.

3. SCOPE

The scope of this work is limited to developing and analyzing event trees for selected initiating events that are judged to potentially produce high values of ‘individual-sequence’ SCDF and/or be major contributors to the ‘summed’ SCDF (Section 3.1). End states of interest pertinent to SCDF are Plant Damage States (PDS) 0 to 2 (Section 3.2), which involve significant amounts of fuel debris located beyond the HTS boundaries.

The event trees employ the standard methodology (summarized in Section 4) in conjunction with preliminary reliability targets for the mitigating systems and preliminary assumptions related to system or component performance (documented in Section 6).

Commensurate with the objective in Section 2, the analysis approach is that of a screening assessment which employs conservative assumptions and progressively expands the event tree models to identify and characterize the dominant sequences. The models initially include only the active mitigating systems and the passive EFW supply from the RWS. They are expanded as needed to represent the passive water supplies to other process systems. However, the RWS is ‘credited’ implicitly in the choice of initiating events. Certain initiating events, such as a loss of moderator cooling or a loss of shield water cooling not included in the list of Table 3-1 because extremely low SCDF values are anticipated when the passive design feature is taken into account.

3.1 Selected Initiating Events

Initiating events examined in this report are listed in Table 3-1. All these events are postulated to occur during the full-power operation of the ACR.

Table 3-1 Selected Initiating Events

No	IE ID	INITIATING EVENT DEFINITION	Frequency (yr^{-1})
1	IE-PTR	Small LOCA - pressure tube rupture (with calandria tube remaining intact)	4.0E-03
2	IE-PCTR	Small LOCA - pressure tube & calandria tube rupture	6.0E-05
3	IE-FBIO	Feeder break	2.0E-03
4	IE-FSB	Feeder stagnation break (with consequential channel rupture)	2.0E-04
5	IE-SWD2	Total loss of one service water division (division 2)	5.0E-02
6	IE-LCL4	Total loss of Class IV power supply to one ACR unit	3.0E-01
7	IE-SCB	Loss of inventory in shield cooling system	4.0E-04
8	IE-MSL3	Small steam discharge causing low deaerator level	1.0E-01
9	IE-FWBS	Symmetric feed water line break upstream of feed water level control valves	2.2E-03
10	IE-FWBA	Asymmetric feed water line break downstream of SG check valve	5.8E-05
11	IE-LOR	Loss of reactivity control leading to uncontrolled power increase	4.24E-02

Initiating Event frequencies are best estimate values for operating CANDU plants in Canada rounded up to the significant digit. These values are anticipated to be conservative (i.e., high) for the ACR.

Table 3-1 lists initiating events considered most significant from the viewpoint of their contribution to the summed SCDF in CANDU 6 and CANDU 9 reactor designs. They can potentially produce high values of individual-sequence SCDF by virtue of a high Initiating Event frequency and/or by virtue of unique post-accident conditions that constraint the options available for the deployment of active mitigating systems.

Initiating events 1 to 4 are small breaks, which have different characteristics in terms of accident mitigation. Pressure tube rupture, IE-PTR (No. 1), results in a leak through the channel bellows just in excess of the HTS make-up capacity that relies mostly on SG cool down for routine mitigation. Pressure tube calandria tube rupture, IE-PCTR (No. 2), is a larger in-core break that might affect the ability of the moderator to act as alternate, long-term heat sink. Feeder break, IE-FBIO (No. 3), is a prototypic small break, which occurs in a feeder of the HTS. Feeder stagnation break, IE-FSB (No. 4), is a unique small break that could interconnect both the HTS and the Calandria Vessel with the Containment, thereby potentially voiding both of these process volumes.

For the preliminary analysis, a conservatively high initiating frequency for the feeder stagnation break was assumed as 10% of any feeder break frequency. This high frequency value was deliberately selected to evaluate the robustness of the ACR mitigating system design. During the detailed PSA, a best estimate of the feeder stagnation break frequency will be calculated. For a stagnation break leading to fuel melting and channel failure, the frequency is expected to be at least an order of magnitude lower than 2E-4/yr value assumed in this report.

Initiating Event 5 (IE-SWD2), total loss of Division 2 service water, partially disrupts the gland seal and motor cooling of two HT pumps and at the same time, it also only leads to loss of half of the mitigating system heat sinks, as Division 1 service is available.

Initiating Event 6 (IE-LCL4) is a loss of normal electrical power supplies to one unit in a two-unit ACR station (presumed to have a relatively high frequency), which constrains some options available for providing active heat sinks after the accident.

Initiating Event 7 (IE-SCB) is a shield water loss, which does not immediately impact on fuel cooling, but could cause excessive thermal stresses in reactor structures if not mitigated by a timely reactor cool down.

Initiating events 8 to 10 are secondary-side breaks that disrupt the normal HTS heat sink. Small steam line discharge, IE-MSL3 (No. 8), is a high-frequency accident initiator that includes a number of operator actions for normal mitigation. Symmetric feedwater line break upstream of the feedwater level control valves in the turbine building, IE-FWBS (No. 9), could cause a consequential loss of normal electrical power supply to complicate the accident mitigation. Asymmetric feedwater line break downstream of SG check valve in the reactor building, IE-

FWBA (No. 10), is a unique break that cannot be isolated from the affected SG. The inability to isolate constrains the options that are available for accident mitigation.

Loss of regulation, Initiating Event 11 (IE-LOR), is a power excursion that would cause a power-cooling mismatch at high power and high HTS pressure if not automatically mitigated in timely manner.

3.2 Event Tree End States

Development of event trees includes the assessment of Plant Damage State (PDS) resulting from accident progression through a particular sequence in the event tree. The PDS used in this report are defined in Table 3-2. These states are a subset of standard states in Reference [3].

Table 3-2 Plant Damage States

PDS 0	Early loss of core integrity at high power and pressure as a result of a failure to shutdown when required.	This PDS is assigned to end states resulting from failure of all shutdown functions when the shutdown is required to mitigate a power-cooling mismatch. The reactor core disassembles at high internal pressure
PDS 1	Late loss of core integrity at decay power starting from high HTS pressure caused by a loss of all primary and backup heat sinks.	Loss of primary heat sinks at HTS high pressure (e.g., loss of FW + SW + RWS make-up to SGs). A small number of channels fail to relieve HTS pressure, but ECC and moderator heat sinks are unavailable. The reactor core disassembles at low internal pressure. The core debris can be retained in the calandria if the shield water heat sink is available.
PDS 2	Late loss of core integrity at decay power starting from low HTS pressure caused by a loss of all primary and backup heat sinks.	LOCA + LOECC + loss of moderator heat sink. The reactor core disassembles at low internal pressure. The core debris can be retained in the calandria if the shield water heat sink is available.
PDS 3	Early, widespread fuel and channel damage at decay power starting from low HTS pressure caused by a loss of primary heat sinks + a failure of ECCS.	LOCA + LOECC cause rapid core voiding (e.g., large LOCA + failure of ECIS and LTCS). The moderator heat sink is available to maintain the fuel within the fuel channels, which are deformed but intact.
PDS 4	Late, widespread fuel and channel damage at decay power starting from low HTS pressure caused by a loss of primary heat sinks + a failure of ECCS.	LOCA + LOECC cause slow core voiding (e.g., a small LOCA + failure of ECIS & LTCS or any size LOCA + failure of LTCS). Moderator heat sink is available to maintain the fuel within the fuel channels, which are deformed but intact.
PDS 5	Early, limited fuel damage at decay power starting from low HTS pressure caused by a loss of primary heat sinks.	LOCA with ECCS performing as intended. No temperature-induced fuel failures, but some incipient cladding defects open. All pressure tubes remain intact.

PDS 6	Late, limited fuel and channel damage at decay power staring from high HTS pressure caused by a loss of primary heat sinks.	A small LOCA (leak) + a loss of SG cool-down or a loss of all feed water supplies. The HTS voids gradually at a high pressure. A small number of pressure tubes + bellows (or a few fuel channels) fail to depressurize the HTS. The ECIS activates while the fuel temperatures are moderate. The LTCS provides the long-term heat sink. The fuel damage is mainly mechanical.
PDS 7	Early but limited fuel damage caused by a single channel LOCA + containment pressurization.	Inlet feeder or end fitting breaks with ECCS performing as intended. Up to whole-channel FP inventory could be released into the containment.
PDS 8	Early but limited fuel and channel damage caused by a single channel LOCA + no containment pressurization.	In-core LOCAs (pressure tube rupture + calandria tube rupture) with ECCS and moderator system performing as intended (i.e., no significant steam discharge into containment, FP release into moderator).
PDS 9	Tritium release	Moderator spills or boiling, but no fuel damage.

Wording of PDS definitions in Table 3-2 is refined relative to that in Reference [3] to improve clarity, but the substance of the definition is unchanged. PDS 10 included in Reference [3] is not brought into Table 3-2, because it does not pertain to the set of selected initiating events (it is relevant to fuelling machine accidents).

PDS 0 to 2 are Severe Core Damage states used for SCDF quantification. The PDSs 3 to 6 are Limited Core Damage states, which are not used to enumerate the SCDF.

4. METHODOLOGY

The methodology used for this preliminary event tree analysis is described in Section 4 of Reference [3].

4.1 Overview

Event tree analysis is carried out for each initiating event in Table 3-1. The event tree depicts various possible sequences, which could occur after the initiating event, by modeling combinations of mitigating system availabilities or unavailabilities.

Each event tree starts with the initiating event, and then develops through a logical set of branch points. Each branch point represents the success (upward direction) or failure (downward direction) of a pertinent mitigating system. The event tree is horizontally oriented, and is read from left (the initiating event) to right (sequence endpoints).

Each sequence in a tree concludes when one of the following conditions exist;

- The reactor has been shut down and decay heat is being adequately removed. No significant plant damage has resulted. Such sequences are labelled “S” (success).
- Failures have resulted in some degree of plant damage. Depending on the initiating event, whether shutdown has occurred or not, and how (if at all) decay heat is being removed, a label is assigned from the listing of PDS in Table 3-2.
- The estimated frequency of the sequence is so low that further study is not meaningful. These sequences are labelled “NDF” (not developed further). A sequence is terminated and labelled “NDF” when its estimated frequency is lower than 1.0×10^{-9} events per year and further mitigating systems are available.

The preliminary event trees for ACR are of intermediate to large size. Separate branch points are assigned not only to heat sinks, but also to the operator actions and services which are required to support the heat sinks (e.g., electrical power and service water). The software used to draw and evaluate the trees is the personal-computer-based program “ETA-II” (Reference [4]).

4.2 Mitigating Systems

During event tree development, questions are asked about the success/failure of various mitigating functions identified in Section 5.

Class I, Class II control power and the Distributed Control systems are not shown in the event trees, since this would have made the trees unmanageable. Dependencies unaccounted for in the preliminary event trees due to these systems will be considered in a latter PSA phase, using fault tree analysis and accident sequence quantification.

For the screening assessments in this report, the functions of active mitigating systems are treated comprehensively, but the passive water make up from the RWS is modelled on as-required basis (see Section 1).

4.3 Operator Actions

Modeling of operator actions in present analysis is largely based on a simplified Human Error Probabilities (HEP) quantification. Table 4-1 sets the basic rules for assigning HEP values for diagnostic errors based on previous CANDU 6 PSA. It should be recognized that the detailed ACR PSA would incorporate errors of diagnosis as well as execution based on the ASEP procedure (Reference [5])).

Table 4-1 Human Error Probabilities Associated to Operator Action Times

Action Time	HEP
0 to 15 minutes	1 (no credit)
15 to 30 minutes	1.0 E-01
30 to 60 minutes	1.0 E-02
1 to 2 hours	1.0 E-03
2 to 4 hours	1.0 E-04
4 to 8 hours	1.0 E-05

Two event trees (IE-PCTR describing the rupture of pressure and calandria tubes and IE-FWBA describing the asymmetric feed water line break downstream of SG check valve) employ more detailed human error analysis methodology rules defined in Reference [3] because operator interventions have a considerable impact on the SCDF in these accidents.

4.4 Acceptance Criteria

All accident sequences are checked against the following acceptance criteria:

- Summed SCDF for all sequences evolving from internal events at power should be lower than 1.0 E-06 ev./year.
- Any one of the individual sequences ending up in a severe core damage state (i.e., in PDS0, PDS1 or PDS2 defined in Table 3-2) should have a frequency lower than 1.0 E-07 ev./year.

These acceptance criteria are guidelines that are expected to yield a summed SCDF for all internal and external events (including shutdown state events) of less than 1.0 E-05 ev./year. The latter is a design requirement used for advanced reactors, which is also adopted for the ACR (Reference [1]).

5. RELIABILITY TARGETS

To help meet the ACR target for summed SCDF, the system reliability targets for ACR were established based on simple fault tree analysis for ACR systems, previous CANDU 6 and CANDU 9 PSA experience and/or engineering judgment. These targets (see Table 5-1 and Table 5-2) were set at the start of the ACR design assist PSA work with the objective to satisfy the acceptance criteria in Section 4.4 and they are used in this report.

The reliability targets in Table 5-1 and Table 5-2 are currently being verified by detailed fault tree analysis of the latest ACR design configuration.

Table 5-1
System Reliability /Unavailability Targets (dependent on DG availability)

Electric Power Supply Status / Systems' Unavailability	Electric Power Supply (Composite Unav.)	AFW	SWD1&D2	D1SW	MHS	LTC-SDC	DECC	LTC-ECC
Class IV Available	-	3.0E-03	1.0E-05	5.0E-03	5.0E-03	1.0E-02	7.0E-04	5.0E-03
All DGs Available (Class IV lost)	-	6.0E-03	1.0E-05	5.0E-03	8.0E-03	1.0E-02	7.0E-04	5.0E-03
Class IV & 1 DG Unavailable	7.00E-02	1.0E-02	1.0E-03	1.0E-02	1.0E-02	5.0E-02	2.0E-03	8.0E-03
Class IV & 2 DGs Unavailable	7.75E-03	5.0E-02	5.0E-03	5.0E-02	5.0E-02	1.0E-01	1.0E-02	1.5E-02
Class IV & 3 DGs Unavailable	6.60E-04	1.0E-01	1.0E-02	x	1.0E-01	5.0E-01	2.0E-02	5.0E-02
Class IV & 4 DGs Unavailable	5.75E-05	x	x	x	x	x	5.0E-02	x
Class IV Unavailable and 1 SW Division lost			x	x	1.0E-02	x		

Auxiliary Condensate Extraction Pump is going to be supplied from an MCC connected to the Class III "F" (EVEN) bus.

Note: "-" sign means it is not applicable to place a value in that spot;
"x" sign means the system is unavailable or no credit is given to its function.

Table 5-2
System Reliability /Unavailability Targets (all DGs assumed available)

System	Acronym	Electric Power Supply Status	Unavailability
Auto De-pressurization Water system	ADW	Not dependent of Class IV power supply status	1.0 E-04
Main Feed Water (excludes auxiliary feed water pumps sub-system)	MFW	Class IV power available	3.5 E-03
Feed Water (includes MFW and AFW)	FW	Class IV power available	5.0 E-04
Feed Water (includes MFW and AFW)	FW	One SW Division available; Class IV power available	1.0 E-03
Emergency Feed Water Supply	EFW	Class IV power available	1.0 E-04
Emergency Feed Water Supply	EFW	Class IV power unavailable; Class III DGs all available	1.0 E-04
Emergency Feed Water System	EFW	One SG unavailable	7.0 E-04
Consequential Loss of Class IV Power Supply	LCL4	Not applicable	5.0 E-02
Crash Cool (overall Crash Cool unav. – including CC1 + CC2)	CC	Not dependent of Class IV power supply status	1.0 E-05
Condensate System	CND	Class IV power available	5.0 E-04
Auxiliary Condensate Extraction System	ACND	Any failure combination of 2 DGs or less.	5.0E-02
Auxiliary feed water or SGs (break) Isolation Failure	AFW-IS	Class IV power available	1.0 E-02
Boiler Pressure Control Cooldown	BPCC	Class IV power available	5.0 E-03
Steam Generator Pressure Relief	SGPR	Class IV available	1.0 E-06
Steam Generator Pressure Relief	SGPR	Class IV unavailable	1.0 E-05

System	Acronym	Electric Power Supply Status	Unavailability
Main Steam Safety Valves failure to open on demand	MSSV	Not dependent of Class IV power status	1.0 E-05
Reserve Water Make-up into the Heat Transport System	RWS-HTS	Not dependent on Class IV power supply status	1.0 E-02
HT pumps seals integrity maintained.	CLPS	Applicable only when pumps are running (Class IV should be available)	8.0 E-05
HT Pumps Trip on High Upper Bearing Temperature	PTHT	Only on sequences in which Class IV power supply is preserved.	5.0 E-03
Liquid Relief Valves Fail to Reclose after an HTS Overpressure Transient	CLPRV	Not dependent of Class IV power supply status	4.0 E-06
Reactor Shutdown by SDS1 & SDS2	RS	Not dependent on power supply	1.0 E-06
Operator Action – Switchyard restored in 60 minutes	R60E4	Not applicable	4.0 E-01
All Operator actions other than R60E4, PCTR and FWBA	OEFW, ORS, OLTC-SDC, etc.	Not applicable	Table 4-1

6. ASSUMPTIONS

Assumptions made in the preliminary event trees are documented in this section. Generic assumptions applicable to all initiating events are described first, followed by assumptions that are specific to individual accident scenarios.

6.1 Generic Assumptions

The following assumptions apply to all event trees in this report:

1. The reactor operates at 100% FP prior to the accident.
2. All on-site diesel generators, auxiliary condensate-extraction pump, auxiliary feed water pumps and instrument–air compressors are air-cooled or are capable of operating for 24 hours without cooling.
3. Auxiliary feed water pumps have sufficient head to supply the minimum required flow with the MSSVs relieving steam by lifting against their spring load.
4. Auto de-pressurization of steam generator secondary side is performed by four MSSVs.
5. One Class III diesel generator can run 1-RSW pump and 1-RCW pump plus other essential loads[4].
6. Moderator and LTC systems require cooling water to operate, which may be supplied from Division 1 or Division 2 RCW system.
7. A failure of instrument air system does not impact service water supplies. This holds when:
   a. All critical valves fail to their safe-state (which may be open or closed, depending on the valve).
   b. All valves that need to be operated after the initiating event have a back-up, gas supply (e.g., gas bottles).
8. The emergency feed water supply from reserve water system to steam generators is available for the mission time of 24 hours. This holds when:
   a. Isolating valves can be actuated without any dependency of Class IV and Class III power for approximately 3 hours[5].

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[4] This means that load shedding does no affect the essential safety loads in any of the accident scenarios, since one RSW pump and one RCW pump are fully capable to supply all necessary safety loads for any accident sequence.

[5] Class I batteries are sized for a minimum of 1 hour capacity for plant control after loss of Class IV and Class III power. It is expected that actual capacity of the batteries will be sufficient to allow operator to modulate the valves that supply RWS water to the steam generators for about 3 hours. The valves are supplied with Class II power.

   b. The same valves can be manually operated on long term basis (beyond 3 hours) in order to prevent spilling of water from the steam generators, thereby ensuring that RWS inventory can last for the mission time of 24 hours[6].

   c. The initiating event does not cause a discharge of HT coolant into the reactor building. When the reserve water tank and the open steam generators are interconnected, the containment envelope could be "bypassed" (i.e., the containment barrier would be provided only by the liquid pool in the reserve water tank and the associated piping)[7].

9. Moderator pony motors are automatically provided with Class III power within a few minutes following a loss of Class IV power.

10. For all events that involve a small LOCA as an initiating event or a consequential failure, it is presumed that the post-accident break discharge is not large enough to remove decay heat from the HTS as liquid at saturation temperature or less. In this report, this is interpreted to mean that the ECC function of the LTC is not sufficient to act as a heat sink and that a steam generator heat sink is also required for these events.

11. It the absence of forced HTS circulation, steam generators provide effective heat sink only if both of them are available (i.e., it is presumed that thermosyphoning breaks down when only one steam generator is available)[8].

6.2 Event Specific Assumptions

The following assumptions were applied to the event trees of selected initiating events:

6.2.1 Pressure Tube Rupture with Intact Calandria Tube

1. Coolant discharge through channel bellows is ~ 20 kg/sec and is beyond the capacity of H_2O feed pump.
2. Manual reactor shutdown occurring before the first automatic trip is[9] not credited, even though much more than 15 minutes would be available for the manual action before the automatic reactor trip on HTS low pressure comes in.
3. Steam generators are required to provide heat sink in conjunction with water make-up into the HTS (see Item 10 in Section 6.1).
4. The EFW supply from the RWS cannot be used (see Item 8c in Section 6.1).

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6 This implies that provisions for manual operation of the RWS valves are available.

7 Opening of isolation valves between the RWS and the steam generators is inhibited by an elevated reactor building pressure.

8 This is a conservative assumption made in the absence of deterministic analyses for relevant HTS configuration.

9 This is a conservative assumption.

6.2.2 Pressure Tube/Calandria Tube Rupture

1. The initial discharge rate is > 100 kg/sec and thus well beyond the capacity of the H_2O feed pumps.
2. The reactor trips automatically either on low HTS flow or on high moderator level.
3. Steam generators are required to provide heat sink in conjunction with water make-up into the HTS (see Item 10 in Section 6.1).
4. The EFW supply from the RWS cannot be used (see Item 8c in Section 6.1).
5. Moderator heat sink is not credited because a consequential hole is postulated to develop through lattice tube, which, in conjunctions with the broken channel, provides a path for calandria vessel draining down to the elevation of the affected channel.
6. Passive water supplies from RWS to HTS or calandria vessel are not credited[10]. Therefore, if this event coincides with a loss of both ECI and LTC, the moderator is not credited as heat sink because forced circulation through moderator heat exchangers may be impaired and no water make-up is available.
7. In order to defend calandria tube failure probability following a pressure tube rupture, the R&D program needs to demonstrate that the calandria tube will survive all relevant loading conditions. The program also needs to demonstrate that the calandria tube has a high creep rupture resistance. The latter is the ability of the calandria tube to withstand the elevated pressure and temperature environments after a pressure tube failure for long enough time so that operator action can be relied upon to reduce the HTS pressure. To afford high reliability credit for this operator action, the calandria tube needs to survive for about 2 hours or longer.

6.2.3 Feeder Break

1. No operator action is credited for reactor shutdown.
2. The RRS maintains approximately constant reactor power until the automatic reactor shutdown.
3. Steam generators are required to provide heat sink in conjunction with water make-up into the HTS (see Item 10 in Section 6.1).
4. The EFW supply from the RWS cannot be used (see Item 8c in Section 6.1).
5. Should steam generators be unavailable, the active heat removal by moderator system is credited as the back-up heat sink (provided a service water system is available). The passive heat removal in calandria vessel (by boiling and make-up from the RWS) is not credited.

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[10] Section 1 explains the rationale for not crediting the passive water supplies from the RWS.

6.2.4 Feeder Stagnation Break with Consequential Channel Rupture

1. The break causes a severe power/cooling mismatch, resulting in a consequential channel rupture (i.e., an in-core break) at full power.
2. Reactor trips automatically on low HTS pressure, low HTS flow or high moderator level.
3. RRS maintains approximately constant power up to the reactor trip.
4. ECC systems (ECIS and LTCS) are automatically activated to refill the HTS and to maintain it full of water.
5. Steam generators are required to provide heat sink in conjunction with water make-up into the HTS (see Item 10 in Section 6.1).
6. The EFW supply from the RWS cannot be used (see Item 8c in Section 6.1).
7. Should the ECC make-up fail, water can be supplied from the RWS into the HTS for the duration of mission time at a sufficient rate to maintain the fuel channels flooded with water[11].
8. Moderator heat sink is not credited because the ruptured bellows and the hole in the feeder in conjunction with the broken channel provide a path for calandria vessel draining down to the elevation of the affected channel[12].
9. Should the HTS make-up or the supplementary steam generator heat sink be unavailable, the affected sequence is not developed any further and a Severe Core Damage is presumed[13].

6.2.5 Loss of One Service Water Division

1. Loss of RCW cooling water to the main feed water and main condensate pumps causes them to trip early. Therefore, these pumps are not credited. The auxiliary feed water pumps are available.
2. The gland seals and bearings of the HT pumps are cooled by RCW water in the following configuration:
    a. Each of the two HT pumps located on either side of the reactor are supplied with cooling water from one separate RCW division.

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11 This assumption implies that the HTS remains depressurized for the gravity make-up to work. It may not be consistent with earlier assumption (Item 5 in Section 6.2.4). Future analysis will explore and confirm this assumption.

12 In the event tree, the RWS make-up to the HTS was credited in case of failure of ECC, but success of crash cooldown.

13 This is an expedient and conservative assignment of PDS. The sequences could be developed further, likely resulting in one of the Limited Core Damage states. However, the conservative assignment of PDS is permissible for screening analysis (Section 1).

b. HT Pumps #1 and #3 are taken to be supplied from SW Division #1. HT Pumps #2 and #4 are taken to be supplied from SW Division #2.

3. Because the bleed cooler is presumed to be supplied by Division #2 service water, the cooling water to gland seals of HT pumps is not available. Failure of bleed control valves to close and bottle up the bleed condenser (CLPS) after the initiating event causes ingress of hot water into the suction of the pressure and inventory control (feed) pumps and into the HT pumps seals, in time causing consequential small LOCAs at two HTS locations due to failures of pump seals.

4. Affected HT pumps trip on high upper bearing temperature or are tripped manually within 1 hour of accident initiation.

5. Service water from Division #2 provides cooling water to LTC heat exchanger #2 that supplies water through RIH #2 and ROH #2. However, if needed, the operator can supply LTC heat exchangers from both service water divisions.

6. The discharge flow from the two-point, induced small LOCAs (caused by failure of HTS pumps seals) is assumed to exceed the capacity of HTS feed pumps.

7. For small LOCAs, steam generators are required to provide heat sink in conjunction with water make-up into the HTS (see Item 10 in Section 6.1).

8. The EFW supply from the RWS cannot be used (see Item 8c in Section 6.1).

9. Instrument air compressors continue running on loss of RCW (see Item 2 in Section 6.1).

10. A failure to trip pumps automatically or manually is presumed to cause a two-point large LOCA. ECC systems are ineffective for two-point large LOCA[14] and cannot be credited in such sequence.

11. The inter-unit service water supply back-up is not credited.

6.2.6 Loss of Class IV Power Supply

1. The reactor shuts down either on low HT flow or high HT pressure.

2. The following equipment is affected:

    a. Turbine generator trips.

    b. HT pumps stop running. In the absence of forced HT flow, both steam generators are required to provide the heat sink (see Item 11 in Section 6.1).

    c. Main feed water pumps stop running.

    d. Main condensate extraction pumps stop running.

    e. Condenser cooling water pumps stop running.

    f. Pressurizer heaters cannot be energized.

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14 All injected water could drain out from the HTS without reaching the fuel.

   g. Boiler pressure control cool-down program is available. This program uses ASDVs and CSDVs to cool down the steam generators, but CSDVs do not open when Class IV power is unavailable.

3. All four diesel generators supplying Class III power start automatically following the loss of Class IV power. These diesel generators do not require external cooling water to operate (Item 2 in Section 6.1).

4. When Class III power is available:

   a. All auxiliary feed water pumps are energized (distributed as one pump per Class III bus).

   b. Auxiliary condensate pump is available (see Item 2 in Section 6.1).

   c. Instrument air system is available (see Item 2 in Section 6.1).

   d. Two of the four LTC pumps are energized.

5. If Class III power is also lost:

   a. Auto depressurization water system (ADW) will depressurize steam generators on a low steam generator level plus timer. Isolation valves between RWS and steam generators open automatically on this signal, unless inhibited by a signal of high reactor building pressure (see Item 8c in Section 6.1).

   b. MSSVs are remotely ‘gagged open’ from the Main Control Room.

   c. Water can be supplied from the RWS to the steam generators for the mission time of 24 hours (see Item 8 in Section 6.1).

6. A failure of a liquid relief valve to re-close constitutes a consequential small LOCA. This means that:

   a. The EFW supply from the RWS cannot be used (see Item 8c in Section 6.1).

7. Multiple failures of ASDVs, CSDVs and MSSVs to open produce a Main Steam Line Break[15].

6.2.7 Loss of Inventory in Shield Cooling System

1. The system is designed such that the End Shields cannot be drained for any pipe break location.

2. Fuel channel integrity (and thus HTS integrity) is maintained for at least 8 hours for any shield water draining transient[16].

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[15] The sequence is not developed beyond the steam line break, since it will be covered by the Event Tree for the Main Steam Line Break.

[16] This time is based on analyses of CANDU-9 design and needs to be verified for the ACR design.

3. EFW supply from the RWS into the steam generators is available if required. It would be manually stopped once the shutdown cooling function of the LTC system is operational.
4. A water make-up from the RWS into the shield tank or the end shields is not credited. It is presumed that the make-up to the shield tank would be ineffective due to the draining through the break. Analyses are not available of thermal stresses due to a make-up of the end shields only.

6.2.8 Small Steam Discharge Causing Low Deaerator Level

The following assumptions are engineering judgements based on 'Balance of Plant' features of existing CANDU reactors. When this preliminary event tree analysis was performed, ACR-specific design details were not available.

The discharge rate for this event is chosen using the following considerations:

a) it should be high enough to cover an opening of a MSSV; and
b) it should be low enough to be within the supply capacity from the demineralized water tank (including the capacity of feed water regulating valves), such that the entire feed water inventory in the plant can be credited.

A steam discharge rate of 180 kg/s would be most challenging. It is presumed that the capacity of supply from the demineralized water tank can match this rate.

1. The turbine generator is unloaded at about 85% full power by the overall plant control programs.
2. Deaerator contains water inventory equivalent to 90 full power minutes at the steam discharge rate of 180 kg /sec before a signal for reactor power setback comes in on low deaerator level.
3. For this event tree, feed water supply from the reserve feed water tank (turbine building) or demineralised water storage tank to the suction of the auxiliary feed water pumps is not credited[17]. Therefore, a failure of condensate system or failures of valves in the gravity supply line from the demineralized water storage tank to the condenser hotwell constitutes a failure of auxiliary feed water system.
4. EFW supply from the RWS to steam generators can be manually established within ~3 hours without opening the MSSVs. The implicit assumption is that the small steam line break depressurizes the secondary side of steam generators to below the injection pressure of gravity make-up from the RWS in less than 3 hours.

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[17] This supply line is available in existing CANDU reactors, but its existence in the ACR was not confirmed at the time when this analysis was performed.

6.2.9 Symmetric Feed Water Line Break Upstream of Feed Water LCVs

1. The reactor trips on low steam generator level.
2. Class IV power supply fails consequentially because the switchgear is located near the broken feed water line in the turbine building and it fails due to harsh environment resulting from a feed water discharge.
3. Following this feed water break:
   a. The auxiliary feed water pumps remain available (they are environmentally qualified for this event, Reference [6]).
   b. The condensate system is not available (it is not environmentally qualified).
4. The auto de-pressurization water system is available and will come in on sustained low steam generator level.

6.2.10 Asymmetric Feed Water Line Break Downstream of SG Check Valve

1. At least one of the HT pumps continues to operate for ~ 60 minutes into the accident[18], providing forced circulation of HT coolant.
2. Auxiliary feed water is available to the intact steam generator when auxiliary feed water level control valves automatically close and isolate the steam generator with the break upon sensing a level discrepancy between the two steam generators.
3. The ECI system is poised to compensate for HTS coolant shrinkage that is not accommodated by the pressurizer[19].
4. The EFW supply to steam generators is not available in the early stages of accident because the opening of RWS flow paths to steam generators is inhibited by a signal of high pressure in the reactor building (see Item 8d in Section 6.1). However, the operator can provide the EFW supply to steam generators upon confirming that HTS boundaries are intact. This could happen at 15 minutes into the accident or later.

6.2.11 Loss of Reactivity Control Leading to Core Power Excursion

This initiating event postulates power excursion which, if not arrested by timely shutdown, invariably leads to severe core damage in existing CANDU reactors (which all have a positive

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18 HT pumps are not environmentally qualified for harsh environment in the reactor building. This assumption is a best-estimate judgement that the actual pumps can operate for a period of time in harsh environment as long as the FW discharge does not impact the pump directly and the pumps do not cavitate. Both these preconditions are satisfied in this accident.

19 The pressurizer acting alone cannot accommodate the rapid HT coolant shrinkage in this accident, so HTS would depressurize below the injection pressure of the ECI system.

void reactivity coefficient). ACR is significantly different from existing CANDU reactors in this aspect by virtue of having a negative void reactivity coefficient. At the time when this event tree analysis was performed, the ACR response to power excursions had not been analyzed in any detail. This analysis presumes that the ACR would also end up is a severe core damage if the engineered shutdown systems were to fail. In this context, the only assumptions required for event tree developments are as follows:

1. Neutronic power measurements and calibration are carried out by devices offering at least the same level of performance as those employed in CANDU 6 plants.
2. The level of quality implemented in the design, manufacturing and commissioning of the reactor regulating system components for ACR will ensure the same reliability/unavailability for the systems functions as that of the CANDU 6. The frequency of loss of reactivity events is therefore assumed to be the same for ACR as the observed frequency for the operating CANDU 6 plants (4.24E-02 ev./year).

7. EVENT TREES

The selected initiating events and the associated event tree models discussed and analyzed in this section are all produced for an ACR-700 Unit, operating at full power and benefiting from the shared support services available from the twin Units.

7.1 Pressure Tube Rupture with Intact Calandria Tube

A failed pressure tube discharges HT coolant into the annulus, pressurizing it to approximately the HTS pressure. A few fuel rods in the affected pressure tube may be damaged during the initiating event due to mechanical forces imposed on the fuel by the pressure boundary failure.

7.1.1 Plant Response

The calandria tube withstands the hydrodynamic loading by the coolant[20], but the bellows on both channel ends invariably fail. Instruments that monitor the annulus gas system produce several alarms in the MCR, which announce the pressure tube failure early into the accident. The Emergency Operating Procedures (EOPs) will call for an orderly, manual shutdown of the reactor when these alarms are produced[21]. However, an early manual shutdown is not credited.

The two coolant discharge paths from the HTS are through small clearances past the end fitting bearings at each end of affected channel, so the coolant flow rates will be small. An engineering judgment is that the total leak rate from the ACR would be on the order of 20 kg/s, which is assumed to be beyond the make-up capacity of HT pressurizing pumps.

With a very small net loss of coolant from the HTS, water levels in the storage tank and the pressurizer decrease very slowly. After a certain time (which depends on the net coolant loss rate), the reactor automatically trips on low HT pressure. The reactor is shut down automatically before the pressurizer reaches its low level setpoint and before the storage tank becomes empty (without employing the back-up supply to the storage tank). Low level alarms from the pressurizer and the storage tank also give the operator ample time (>> 15 minutes) to shut down the reactor before any of HTS volumes would start to void due to the loss of coolant.

In the time period before the reactor is shut down, the RRS will maintain the power at preset value. Following the shutdown, the ECI system is activated on a sustained low HT pressure signal. A crash cool down of steam generators, which accompanies the ECI system activation, quickly reduces the HTS pressure to below the injection pressure. The injection of emergency coolant from the pressurized accumulator tanks refills the HTS and stabilizes the HTS pressure. The HT pumps are likely to still be running, producing a considerable heat load in addition to

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[20] Should the calandria tube fail, this constitutes a more or less simultaneous pressure tube and calandria tube rupture which is analyzed as a separate Initiating Event in Section 7.2.

[21] This is an expectation based on existing CANDU reactors. The EOPs for the ACR have not yet been developed.

fuel decay heat. Steam generators provide the heat sink for the post-accident heat load (i.e., heat discharged through the bellows is negligible relative to heat taken in by the feed water).

During a slow HTS depressurization that follows, the HT pumps are turned off and the ECI accumulator tanks drain. When the inventory in these tanks is depleted, the tanks are isolated and a pumped water supply from the LTC-ECC system commences. At this juncture, the HTS leak rate is very small because the driving pressure is low. The LTC-ECC system acts primarily as the make-up system for the HTS. Its heat exchangers deal only with a very small fraction of HTS heat that enters the containment as part of coolant discharge. The vast majority of HTS heat is dissipated to the steam generators. The passive make-up of steam generators from the RWS cannot be used in this accident (see Item 8c in Section 6.1).

The plant conditions at the end of mission time (at 24 hours) are stable. The fuel damage has been limited to the early fuel failures caused by the initiating event (i.e., no additional fuel failures). Long term actions (beyond the scope of this report) will remove the damaged fuel from the reactor and replace the damaged fuel channel.

7.1.2 Event Tree

The initiating event label is IE-PTR (Table 3-1). Event tree details are presented in Appendix A.

Severe core damage plant damage states are assigned for:

- A failure to shut down the reactor. A failure to depressurize the HTS such that a coolant make-up is not possible.
- A failure to provide service water to active mitigating systems.
- Failures of ECC systems to provide make-up for the depressurized HTS in conjunction with a failure of moderator system to provide an alternate heat sink. Note that only the ‘active’ heat sink mode of moderator system is modelled (i.e., the pumps and the heat exchangers). The ‘passive’ mode of ‘moderator as a heat sink’, which involves boiling off water in the calandria vessel and water make-up from the RWS is well suited for this type of accident, so the severe core damage frequency will be lower when these passive ACR features will be modelled.

7.2 Pressure Tube/Calandria Tube Rupture

Calandria tubes of the ACR are designed to withstand the hydrodynamic loads imposed by pressure tube failure. This initiating event thus involves a gradual creeping of calandria tube wall to failure. This can be mitigated by operator actions, because several hours are available before the calandria tube would creep to failure. The initiating event frequency (Table 3-1) is based on the operator failing to prevent the calandria tube rupture by creep. Given that more than two hours would be available for this action[22], a combined failure probability of 1.50E-2 can be

[22] This is based on a conservative assessment of the minimum time to creep failure and will be confirmed.

assigned[23]. This initiating event frequency is thus based on the pressure tube failure frequency and the preceding operator failure probability.

7.2.1 Plant Response

The failure of both pressure and calandria tubes discharges high-enthalpy HT coolant and a small amount of mechanically broken fuel (up to 12 fuel bundles) into the calandria vessel. One or more calandria rupture disks burst to provide overpressure protection for the calandria vessel. Broken fuel releases some of its volatile fission products, which are subjected to a ‘pool scrubbing’ before being released into the containment. The energy release into the containment depends on the elevation of ruptured channel. A channel at high elevation could release much of the energy contained in the two-phase HT coolant discharge. A channel at low elevation would release little energy into the containment, because steam is condensed within the calandria vessel. A small energy release into the containment is considered in this analysis.

The rupture constitutes a small LOCA with an initial discharge rate in excess of 100 kg/s[24]. A generic presumption for all small LOCAs is that these breaks require the steam generator heat sink to be available in order to avoid additional fuel failures (see Item 10 in Section 6.1 and its associated footnotes). The passive make-up of steam generators from the RWS cannot be used in this accident because of containment bypass considerations (see Item 8c in Section 6.1).

The reactor trips automatically shortly after the in-core rupture . There is some damage to the in-core devices caused by the channel rupture, but this damage does not impair the ability of either shutdown system to quickly reduce the reactor power to decay power level and keep the reactor shut down thereafter[25].

In this analysis, a consequential opening of a sizeable flow path between the calandria vessel and the containment is presumed at a low core elevation, which prevents the moderator from acting as a heat sink (see Item 5 in Section 6.2.2). This events assumes an end fitting ejection. Its effect is that severe core damage is assigned when either of the ECI or LTC functions are lost.

The HTS depressurizes, it is refilled by water injection from the ECI system and maintained full of water in the long term by water injection from the LTC-ECC subsystem. The HTS pumps are

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23 This value accounts for event diagnosis as well as the execution errors related to shutting down the reactor and reducing the HTS pressure and temperature by means of ‘boiler pressure control cool-down’ program and is based on the methodology in Reference [3]).

24 This is based on analyses for existing CANDU reactors . The range of break discharge rates for the ACR is yet to be quantified.

25 Consequences of in-core ruptures have not yet been evaluated for the ACR. However, the basic prerequisite for all CANDU reactor designs is that an in core rupture does not impair either of the shutdown functions. It is safe to assume that the ACR will meet this requirement.

tripped or turned off before the injection from the ECI accumulator tanks stops. HTS heat removal is shared between the steam generators and the LTC heat exchangers. In this case, the LTC heat exchangers dissipate a considerable fraction of the HTS heat.

The plant conditions at the end of mission time (at 24 hours) are stable. The fuel damage has been limited to the early fuel damage caused by the initiating event (i.e., no additional fuel damage). Long term actions (beyond the scope of this report) will remove the damaged fuel from the reactor and replace the damaged fuel channel.

7.2.2 Event Tree

The initiating event label is IE-PCTR (Table 3-1). Event tree details are presented in Appendix B.

Severe core damage plant damage states are assigned for:

- A failure to shut down the reactor
- A failure to depressurize the HTS by engineered means (i.e., steam generator crash cool-down) such that a coolant make-up is not possible.
- A failure to provide service water to active mitigating systems..
- Failures of ECC systems to provide make-up for the depressurized HTS, since the alternate moderator heat sink is presumed unavailable due to draining of the calandria vessel (Item 5 in Section 6.2.2).
- Failures of active steam generator heat sinks, since the LTC-ECC may not be able to remove the HTS heat (Item 10 in Section 6.1) the passive RWS supply to steam generators cannot be used (Item 8c in Section 6.1) and the alternate moderator heat sink is presumed unavailable due to draining of the calandria vessel (Item 5 in Section 6.2.2).

7.3 Feeder Break

Inlet or outlet feeder pipes would typically fail without causing any appreciable power-cooling mismatch in the affected fuel channel. Any outlet feeder break can only accelerate the flow through the affected channel. An inlet feeder break could reduce the forward flow (very small break), deteriorate the channel flow to very low values (so-called feeder stagnation break analyzed in Section 7.4) or reverse the channel flow (larger breaks). This section analyzes the off-stagnation breaks, which include all outlet feeder breaks and the vast majority of break sizes and locations on the inlet feeders. In the absence of local power-cooling mismatch, these breaks are not appreciably different from small LOCAs in reactor headers or other HTS piping. All pressure tubes and calandria tubes are intact after the initiating event.

7.3.1 Plant Response

The reactor is tripped on low heat transport system pressure, low heat transport system flow or high reactor building pressure.

The ECC conditioning signal is generated on sustained low heat transport system pressure. It activates the crash cool-down of steam generators, opens the isolation valves of ECI accumulator tanks and readies the LTC-ECC subsystem for longer term injection.

The injection from ECI accumulator tanks refills the HTS. Running HT pumps provide forced flow through steam generators to maintain the heat sink. Some fraction of HTS heat (which depends on the break size) is carried into the containment by the discharging HT coolant. The HT pumps are tripped or turned off before the ECI accumulator inventory depletes and the long – term injection by the LTC-ECC system commences. The steam generators continue to serve as heat sinks (Item 10 in Section 6.1) in conjunction with LTC heat exchangers. The fractions of heat transferred to these two heat sinks depend on the break size.

The plant conditions at the end of mission time (at 24 hours) are stable. The fuel damage has been limited to incipient fuel defects which had opened during the HTS depressurization. Long term actions (beyond the scope of this report) will isolate the affected channel (e.g., freeze plugs), de-fuel it and repair the broken feeder.

7.3.2 Event Tree

The initiating event label is IE-FBIO ((Table 3-1). Event tree details are given in Appendix C.

The assignments of severe core damage states are identical to those for the pressure tube rupture with intact calandria tube (Section 7.1), since both these accidents are small LOCAs on the lower end of break size spectrum with no fuel damage after the initiating event, other than the mechanically damaged fuel in the channel with the ruptured pressure tube. The rationale for assignments of severe care damage states and the qualifying comments in Section 7.1 apply here as well.

7.4 Feeder Stagnation Break

Certain breaks in inlet feeder pipes can deteriorate (i.e., essentially stagnate) the coolant flow in the affected channel while the reactor is at power. For a severe power cooling mismatch, the fuel would overheat to induce a consequential pressure tube rupture. The pressure tube ruptures either because its wall heats up by convection from a superheated steam, or because the hot fuel materials come into contact with pressure tube wall. The conditions at consequential tube rupture are different from those during a spurious tube rupture. Hot gases or fuel materials impacting on the calandria tube make its consequential failure quite likely, while the flashing HT coolant at saturation temperature does not pose an immediate challenge to the calandria tube integrity.

This section analyzes an inlet feeder break that induces a severe power cooling mismatch and leads to a more-or-less simultaneous rupture of both pressure and calandria tubes.

7.4.1 Plant Response

In terms of plant response, this accident is closely similar to the delayed channel rupture in Section 7.2. Automatic reactor trip times on some process parameters will differ because the pressurizer and storage tank inventories are different at the time of in-core rupture. However, this does not impact the potential for propagation to severe core damage. The damage to the in-core devices could be more severe for this accident, because superheated steam and some molten fuel materials are discharged following the severe power-cooling mismatch in the affected channel. However, this does not impact the shutdown capability, which must be maintained for either accident (see Footnote 25 on Page 7-3).

After the initial transient, the plant response is the same as that described in Section 7.2.1. However, for the feeder stagnation break, the passive water supply from the RWS to the HTS is modelled (which is not modelled for the in-core rupture in Section 7.2). This is done in order to reduce the individual-sequence SCDF to below the target value of 1E-7 events per year.

The credited water make-up by gravity into the depressurized HTS is feasible for this accident as well as for the delayed in-core rupture in Section 7.2. In both these accidents, the moderator heat sink is uncertain because of the postulated hole in the calandria vessel boundary at low core elevation ((Item 5 in Section 6.2.2 and Item 8 in 6.2.4). The RWS make-up to steam generators cannot be used because of the containment bypass considerations (see (Item 8c in Section 6.1). Hence, RWS water inventory is not used for other purposes (other than to provide a sufficient net pumps suction head for the LTC pumps) and is available for HTS make-up. The HTS make-up ends up in the calandria vessel in any case, so there is no conflict as to which process volume to supply with gravity make-up.

7.4.2 Event Tree

The initiating event label is IE-FSB. Event tree details are presented in Appendix D.

The assignments of severe core damage states are identical to those for the delayed in-core rupture (Section 7.2.2) except for the failure of the LTC-ECC subsystem. Sequences for this failure are developed further to consider the HTS make-up from the RWS. When LTC-ECC is unavailable to provide long term make-up and partial heat removal, a severe core damage state is assigned when:

- Gravity supply from the RWS is not established.
- Gravity supply from the RWS is successfully established, but a steam generator heat sink is not available to satisfy the conditions of Item 10 in Section 6.1.

7.5 Total Loss of One Service Water Division

A total loss of service water in a division means that there is no RCW flow, or there is no cooling of RCW system (i.e., no RSW flow through the heat exchangers between the RSW and RCW systems), in the division. This section examines the total loss of Division 2 service water system, which supplies the bleed condenser cooler.

A random failure of the expansion joints is not modelled. The plant response for this failure differs from that for the normal loss of service water, because consequential flooding becomes an issue. Flooding analyses are not within the scope of present work[26].

7.5.1 Plant Response

Following the total loss of Division 2 service water, the moderator temperature rises. The temperature-induced swell triggers a reactor power setback within minutes. If the setback fails and the temperature-induced swell continues, the reactor will trip on high moderator level (SDS1 and/or SDS2) shortly thereafter.

Shutting down the reactor significantly reduces the moderator heat load (the nuclear heating in the dominant component of the heat load). The moderator temperature continues to rise slowly providing a number of hours before heavy water would start to boil[27]. The operator can further reduce the heat load to moderator by cooling down the HTS. The HTS cool-down slows down, arrests or reverses the moderator temperature rise[28].

The HTS response depends on whether or not the initiating event induced a consequential LOCA (i.e., the pump seal failures) and on the responses of numerous systems as described by the event tree in Appendix E.

7.5.2 Event Tree

The Initiating Event label is IE-SWD2. Event tree details are presented in Appendix E.

Whether or not the consequential LOCA develops, a severe core damage plant damage state is assigned for a failure to shut down the reactor. Sequences for a failure of steam generator pressure relief in a shutdown reactor are not developed. Such failure will result in a main steam

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26 This failure will be addressed in flooding PSA.

27 Analyses for the ACR are not yet available. Extrapolating CANDU-6 data and taking into account the smaller moderator volume in the ACR, the adiabatic heat-up would take 3 to 4 hours to the onset of boiling. Heat losses to the cooler shield water on the outside of calandria vessel would extend this time considerably.

28 In principle, the HTS can be used as a heat sink for the moderator if the HT coolant temperature is reduced below the saturation temperature of heavy water at calandria relief setpoint, which is feasible.

line break, which will be analysed in a separate event tree in conjunction with a loss of service water.

Assignments of severe core damage states following loss of Division 2 service water depend on the mitigating systems availabilities. For consequential LOCA sequences, the rationale for assignments of severe core damage states and the qualifying comments in Section 7.1 apply here as well. For non LOCA sequences, where both Division 2 and 1 service water are lost, PDS1 is assigned. If division 1 service water is available then PDS6 is assigned.

7.6 Loss of Class IV Power to One Unit

This initiating event is the total loss of Class IV electrical power supply distribution system to one unit of a two-unit ACR station. The total loss of Class IV power supply is defined as the loss of power to both 5314-BUA (and hence 5324-BUC, 5323-BUE) and 5314-BUB (and hence 5324-BUD, 5323-BUF) Class IV buses. It is assumed that, following a loss of grid, the second unit remains operational.

All four diesel generators supplying Class III power start automatically following this initiating event (Item 3 in Section 6.2.6). Class III power supply would normally be available within a few minutes of losing Class IV power.

7.6.1 Plant Response

The equipment affected by this initiation event is listed in Section 6.2.6.

The steam generator inventory can provide “passive” HTS heat sink for approximately 2 hours provided the steam relief via spring-loaded MSSVs is available. This analysis assumes that the passive steam generator heat sink would last for 1 hour into the accident.

Heat sinks designated to mitigate this initiating event are:

1. The steam generators supplied by main or auxiliary feed water, when Class IV power is restored quickly.
2. The steam generators supplied by auxiliary feed water when Class III power is available.
3. The LTC system when Class III power is automatically restored by the diesel generators.
4. The steam generators supplied from RWS, when Class III power cannot be restored and only Class I and II power supplies are available.

The plant response depends on the number and timing of power supplies availability. This response can be altered by early failures which affect power generation (reactor shutdown), the passive steam generator heat sink (steam relief via MSSVs) and pressure boundary integrity (re-closure of HT liquid relief valves).

7.6.2 Event Tree

The initiating event label is IE-LCL4 (Table 3-1). Event tree details are presented in Appendix F.

Severe core damage plant damage states are assigned for:

- A failure to shut down the reactor.
- A failure to provide service water to active mitigating systems. A loss of Class IV and Class III electrical power and loss of RWS to the steam generators.

7.7 Loss of Inventory in the Shield Cooling System

The inventory loss is due to a pipe break in the shield cooling system. This break drains the inventory in the shield tank, but not in the end shields (Item 1 in Section 6.2.7). Up to 8 hours is available to prevent excessive stresses in the reactor assembly (Item 2 in Section 6.2.7).

7.7.1 Plant Response

The reactor is shutdown by a power setback on a low level in the head tank. The operator confirms and ensures that the reactor is shutdown within 8 hours with a success probability of 1.0.

For normal mitigation, operator manually cools down the reactor using the boiler pressure control cool-down program (via ASDVs and CSDVs) and employs the shutdown cooling subsystem of LTC system.

If normal mitigation cannot be preformed, the operator employs other available systems as described in the event tree.

7.7.2 Event Tree

The initiating event label is IE-SCB (Table 3-1). Event tree details are given in Appendix G.

7.8 Small Steam Discharge Causing Low Level in Deaerator

This event is a small steam discharge of 180 kg/s resulting (see Section 6.2.8 for rationale for choosing this discharge rate). The contributing events are (a) spurious opening of one MSSV and (b) small steam line failures. The dominant contributor is the spurious MSSV opening.

7.8.1 Plant Response

Following the initiating event, the deaerator level falls causing levels in the reserve feed water storage tank and in the hotwell to fall. The main condensate extraction pumps trip and auxiliary condensate extraction pump starts to continue depletion of feed water inventory.

Operator manually shuts down the reactor within 60 minutes with probability of 1.0E-04 in response to:

- power is automatically reduced to about 85% shortly after the initiating event due to falling steam generator pressure (Item 1 in Section 6.2.8).
- alarm on feed water/steam mismatch in steam generators;
- alarm on reactor power / turbine power mismatch;
- low-level indication in the condenser hotwell.
- Following the manual shutdown at 60 minutes, feed water inventory left in the deaerator is equivalent to ~ 3 hours of decay heat dissipation. The operator uses the boiler pressure controlled cool-down program (i.e., steam relief via ASDVs and CSDVs) to activate LTC-SDC system within this time in order to provide the long-term heat sink.

If no action is taken within 60 minutes, the deaerator level drops further and a reactor power setback on low deaerator level comes in at about 90 minutes into the accident (Item 2 in Section 6.2.8). The reactor is in a shutdown state in 3 minutes after the automatic setback. At this juncture, feed water inventory left in the deaerator is equivalent to ~ 1.5 hours of decay heat dissipation. As for the manual cooldown, the operator uses the boiler pressure controlled cool-down program to activate LTC-SDC system within this time in order to provide the long-term heat sink.

If the above normal mitigating actions cannot be performed because of additional failures, the operator uses the available systems as described in the event tree. The ultimate line of defence is the passive (gravity) make-up of steam generators from the RWS, which is invariably facilitated by the 'passive' steam generator depressurization via the postulated hole in the steam line boundary (see Item 4 in Section 6.2.8).

7.8.2 Event Tree

The initiating event label is IE-MSL3 (Table 3-1). Event tree details are given in Appendix H.

Severe core damage states are assigned to the following sequences:

- A failure to shut down the reactor; and
- Multiple failures which include a failure to establish the gravity EFW supply from the RWS.

For modelling this event tree, the condensate system was asked at the beginning of the ET. The availability of the condensate system, allows sufficient time for the operator to take action as feedwater is available (prior to reactor setback). This modelling simplified the ET development. The condensate was assigned a low unavailability value because for the initial period (3 hours) Class IV power is unlikely to fail. Some sequences (loss of Class IV electrical power) should be higher but the impact is not significant.

7.9 Symmetric Feed Water Line Break Upstream of Feed Water Control Valves

This break is located in the turbine building. It is assumed that Class IV electrical power fails consequentially (Item 2 in Section 6.2.9).

7.9.1 Plant Response

Steam generators quickly loose feedwater supply via the break and water level decreases. The reactor trips on low level in steam generators.

The equipment affected by the loss of Class IV electrical power is listed in Section 6.2.6. Additional equipment fails due to the harsh environment as listed in Item 3 of Section 6.2.9. Decay heat is transferred from the reactor core to the steam generators by thermosyphoning.

The automatic actions to mitigate this accident in a shutdown reactor is to auto-depressurize the steam generators (Item 4 in Section 6.2.9) and open the isolation valves for the gravity water supply from the RWS into the steam generators (Item 8 in Section 6.1).

If the gravity EFW supply is not available, but auto-depressurization is successful, it is assumed that the LTC-SDC subsystem can be manually activated.

7.9.2 Event Tree

The initiating event label is IE-FWBS (Table 3-1). Event tree details are presented in Appendix I.

The assignment of plant damage states is similar to other transient events.

7.10 Asymmetric Feed Water Line Break Downstream of the SG Check Valve

This break is located in the reactor building and it cannot be isolated from the affected steam generator.

7.10.1 Plant Response

The reactor trips automatically on a high reactor building pressure. The automatic trip on the low pressure in the feed line to steam generators would come in.

The affected steam generator drains and ceases to provide heat sink for the HTS.

Forced circulation of HT coolant is available for at least one hour (Item 1 in Section 6.2.10). This allows a single, unaffected steam generator to act as heat sink for the HTS (Item 11 in Section 6.1).

Since the affected steam generator is not taken to act as heat sink from the initiating event, the controlled cool-down (using the boiler pressure controlled cool-down program) cannot be executed. The auto-depressurization signal will come in on a sustained low steam generator level (in the broken steam generator).

The gravity supply of EFW from RWS into steam generators is not activated automatically, but can be provided by manual operator actions (Item 4 in Section 6.2.10). However, this manual action is not credited in this analysis. Auxiliary feed water supply is available (Item 2 in Section 6.2.10).

HTS cools down and ECI system accommodates the coolant shrinkage (Item 3 in Section 6.2.10). If the ECI make-up were impaired, thermosyphoning flow through steam generators would be unstable. In this analysis, the impacted sequence is not developed any further and severe core damage is presumed. Implicit in this rule is that the shutdown cooling function of LTC system is credited only with successful ECI make-up

7.10.2 Event Tree

The initiating event label is IE-FWBA (Table 3-1). Event tree details are presented in Appendix J.

Severe core damage plant damage states are assigned for:

- A failure to shut down the reactor
- A failure to depressurize the steam generators (which prevents the ECI make-up and triggers the above described rule for the assignment of severe core damage).
- A failure of ECI system to make-up the depressurized HTS (which triggers the same rule, but at a higher frequency.

7.11 Loss of Reactivity Control Leading to Core Power Excursion

The initiating event is reactor power excursion.

7.11.1 Plant Response

The power excursion is presumed to result in a severe core damage state if it is not terminated in a timely manner by the automatic action of engineered shutdown systems (see Section 6.2.11).

7.11.2 Event Tree

The event tree label is IE-LOR (Table 3-1). The tree (Appendix K) is simple and does not require any explanation.

8. RESULTS AND DISCUSSION

8.1 Overview

The following table shows the preliminary summed frequencies of relevant plant damage states. Severe core damage states that involve a core disassembly include PDS0, PDS1 and PDS2. Limited core damage states PDS4 and PDS6 maintain the fuel within the HTS boundaries. However, the fuel channels are damaged (deformed).

Table 8-1 Summed Frequency of Plant Damage States

PDS State	**Abbreviated Definition** (see Table 3-2 for full definition)	**Frequency without recovery (events/yr)**
PDS0	Early loss of core integrity at high power and pressure	5.6E-08
PDS1	Late loss of core integrity at decay power starting from high HTS pressure	8.7E-08
PDS2	Late loss of core integrity at decay power starting from low HTS pressure	1.5E-06
Summed Severe Core Damage Frequency (PDS0, PDS1, PDS2)		**1.6E-06**
PDS4	Late, widespread fuel & channel damage at decay power starting from low HTS pressure	5.3E-05
PDS6	Late, limited fuel & channel damage at decay power staring from high HTS pressure	6.6E-06

8.1.1 Summed Severe Core Damage Frequency

The preliminary estimate of SCDF is 1.6E-06 events per year, which is marginally higher than the AECL guideline for internal events of 1.0E-06 events per year (Section 4.4). This estimate can be reduced to 1.3E-06 events per year by taking additional credits discussed below. Nevertheless, detailed PSA analysis will need to remove conservative simplifications made in this preliminary analysis and investigate various options to reduce the SCDF values, including:

a) Developing a case for a reduced pressure tube rupture frequency (from the current value of 4.0E-03 events per year to a best estimate value of 1.0E-03 events per year). Industry-wide consensus for such a reduction is now emerging, based on a number of factors (e.g.: improvements in the manufacturing process and improvements in refueling and operating practices contribute to avoiding of in-service flaws from debris-fretting, crevice corrosion flaws, tooling-induced flaws, and formation of hydride blisters).

b) Revising the frequency of a loss of Class IV power from the current value of 3.0E-1 events per year to 1E-1 events per year. The current value is representative of New Brunswick, Canada grid (Pt. Lepreau site) while the revised value applies to Ontario, Canada grid.

c) Developing a case for effective heat removal by a single steam generator in the absence of forced flow. Thermosyphoning with single steam generator will need to be analyzed.

d) Revisiting the generic simplification made in this preliminary analysis that all small LOCAs require both the ECC make-up and the steam generator heat sink to successfully mitigate an accident. In some cases, LTC alone may provide make-up and decay heat removal functions depending on the size of the break.

e) Revisiting credits taken for operator recovery actions. This preliminary analysis has taken only minimum credit.

8.1.2 Summed Limited Core Damage Frequencies

The summed frequency of PDS4, which corresponds to 'moderator acting as heat sink', is 5.3E-05 events per year. This satisfies the current CNSC regulatory guideline (Reference [7]) that *any use of the moderator as a heat sink should be lower than 1.0E-4*. Resolving the above topics will have a beneficial effect on this estimate.

The summed frequency of PDS6 is 6.6E-06. PDS6 approximately corresponds to a loss of all HTS heat sinks, causing a very limited number of consequential channel ruptures that depressurize the HTS and facilitate mitigation by ECI, LTC and RWS systems. Almost all of this summed PDS6 frequency is due to one event only, namely – the asymmetric feed water line break that contributes 90% to the estimated value. Work on above topics c) to e) is relevant to PDS6. In addition, the inhibition of water supply from the RWS to the steam generator needs to be examined more detail.

8.1.3 Top Contributors to SCDF for Internal Events

Table 8-2 list the 10 top-most contributors to the summed SCDF. Three of these sequences have frequencies marginally higher than 1.0E-07 events per year (i.e., above the AECL guideline for the individual sequence frequency in Section 4.4) prior to application of recovery factors. Following application of recovery factors, there are only two sequences above the guideline.

The results are conservative since recovery actions were not applied to all sequences where appropriate. One such function is the gravity make-up from the RWS into process volumes, which is explained in Section 1. In this preliminary analysis, this ACR specific feature is not credited unless the individual sequence frequency is above the guideline frequency of 1.0E-07 events per year. Table 8-2 that this latter intent was not followed consistently in this preliminary analysis (i.e., we have several sequences with frequencies > 1.0E-07 events per year that were not expanded to consider the RWS make-up).

Table 8-2
Significant Contributors to Summed SCDF

No	Initiating Event	Sequence Index	Sequence Description	Plant Damage State	Frequency Without Recovery events/yr	Frequency With Recovery events/yr
1	IE-PCTR	PCTR-4	Pressure Tube and Calandria Tube Rupture followed by Loss of LTC	PDS2	2.85E-07	2.85E-07
2	IE-PCTR	PCTR-A12	Pressure Tube and Calandria Tube Rupture followed by Loss of Class IV Power and Loss of Auxiliary Condensate Supply to the Deaerator	PDS2	1.38E-07	1.38E-08
3	IE-FSB	FSB-7	Feeder Stagnation Break followed by Loss of Dormant ECC Injection	PDS2	1.33E-07	1.33E-07
4	IE-PTR	PTR-7	Pressure Tube Rupture followed by Loss of LTC and Loss of Moderator Cooling	PDS2	9.49E-08	9.49E-09
5	IE-FSB	FSB-2	Feeder Stagnation Break followed by Loss of Main and Auxiliary Feedwater Supply to SGs	PDS2	9.45E-08	9.45E-08
6	IE-PTR	PTR-A13	Pressure Tube Rupture followed by Loss of Class IV Power, Loss of Auxiliary Condensate to Deaerator and Loss of Moderator as a Heat Sink	PDS2	7.35E-08	7.35E-09
7	IE-LCL4	LCL4-B52	Loss of Class IV Power Supply and Failure to Start of all Standby DGs, followed by Operator failure to Actuate Open the Main Steam Safety Valves	PDS2	6.91E-08	6.91E-08

No	Initiating Event	Sequence Index	Sequence Description	Plant Damage State	Frequency Without Recovery events/yr	Frequency With Recovery events/yr
8	IE-FSB	FSB-A12	Feeder Stagnation Break followed by Loss of Class IV Power and Loss of Auxiliary Feedwater Supply to SGs	PDS2	5.55E-08	5.55E-08
9	IE-FWBA	FWBA4	Asymmetric Feedwater Line Break followed by Loss of Dormant ECC Injection	PDS2	5.51E-08	5.51E-08
10	IE-FBIO	FBIO-5	Feeder Break followed by Loss of LTC and Loss of Moderator Cooling	PDS2	4.75E-08	4.75E-09

A simple post-analysis review was performed to get a "feel" of how much impact additional recovery actions by the operator could have on the SCDF. Some sequences could all be mitigated by water supply from the RWS into the steam generators, HTS calandria vessel or reserve water tank or demineralised back-up to the auxiliary feedwater pump suction. The frequencies of these sequences were reduced by an order of magnitude (based on engineering judgement) by crediting moderator make-up or reserve feedwater tank (or demineralised storage tank to AFW). The recovery values are in the " frequency with recovery" column in Table 8-2.

It is not a surprise that this simple exercise shows a significant impact of long-term operator actions on SCDF. The severe accident management, which is at issue here, is performed by people. By not crediting the long-term operator interventions as was intentionally done in this preliminary analysis the SCDF estimates are unduly conservative. Clearly, the detailed PSA will model the operator interventions more comprehensively (see Item e in Section 8.1.1).

8.2 SCDF Contributors

This section examines and discusses accident sequences related to severe core damage (PDS0, PDS1 and PDS2) for individual initiating events. Only PDS1 and PDS2 with sequences greater than 1E-09 are shown.

8.2.1 Pressure Tube Rupture (Calandria Tube Intact)

The SCDF contributors identified by the PTR event tree model in Appendix A are listed below, sorted in descending order.

PDS No	Sequence No	Frequency (events/yr)
PDS2	PTR-7	9.49E-08
PDS2	PTR-A13	7.35E-08
PDS1	PTR-11	4.00E-08
PDS2	PTR-10	3.80E-08
PDS2	PTR-9	1.33E-08
PDS2	PTR-A21	1.25E-08
PDS1	PTR-A01	1.15E-08
PDS0	PTR-12	4.00E-09

The two top-most sequences involve a loss of moderator heat sink, which could be mitigated by water supply from the RWS into the calandria vessel.

Recovery action can be applied to sequence PTR-A21: Since this sequence stems from a total loss of power, a recovery of Class IV power within 60 minutes can be applied to it (R60E4 = 0.4). This would reduce the PDS2 frequency to 5.0 E-09 events per year.

8.2.2 Pressure Tube and Calandria Tube Rupture

The SCDF contributors identified by the PCTR ET model in Appendix B are listed below.

PDS No	Sequence No	Frequency (events/yr)
PDS2	PCTR-4	2.85E-07
PDS2	PCTR-A12	1.38E-07
PDS2	PCTR-5	3.99E-08
PDS2	PCTR-2	2.83E-08
PDS2	PCTR-3	2.83E-08
PDS2	PCTR-A13	1.66E-08
PDS2	PCTR-A14	1.39E-08
PDS0	PCTR-8	4.00E-11

The two topmost sequences are above the guideline criterion of 1.0E-07 events per year for an individual sequence. They involve:

- A failure of the long term LTC-ECC system (PCTR-4)
- A loss of Class IV power and a failure of auxiliary condensate system (PCTR-A12). This sequence could credit the water supply from the reserve feed water tank (turbine building) to the suction of auxiliary feed water pumps (as shown in Table 8-2), or gravity make-up of steam generators from the RWS.

The sequences 4 to 6 are all related to the stipulation that all small LOCAs require an ECC make-up in conjunction with a steam generator heat sink to avoid damage (Item 10 in Section 6.1 and its footnote) As explained in the cited item, this may be unduly conservative for in-core LOCAs. PCTR-2 could be mitigated by gravity make-up of steam generators from the RWS.

The moderator heat sink could not be credited because of the postulate of an end fitting ejection, which drains the calandria vessel (Item 5 in Section 6.2.2).

During the detailed fault tree analysis work, options to reduce this sequence frequency will be explored. Such options include:

- Review PTR frequency data to reduce PTR frequency to below 4.0E-03.
- Detailed analysis to reduce human error probability to below 1E-2 for actions associated with preventing CT creep rupture.

8.2.3 Feeder Breaks

The SCDF contributors identified by the IE-FBIO event tree model in Appendix C are listed below.

PDS No	Sequence No	Frequency (events/year)
PDS2	FBIO5	4.75E-08
PDS1	FBIO9	1.90E-08
PDS2	FBIO8	1.90E-08

The top-most sequence involves a loss of moderator heat sink, which could be mitigated by water supply from the RWS into the calandria vessel.

8.2.4 Feeder Stagnation Break with Consequential Channel Rupture

The SCDF contributors identified by the IE-FBIO event tree model in Appendix D are listed below.

PDS No	Sequence No	Frequency (events/year)
PDS2	FSB2	9.45E-08
PDS2	FSB-A12	5.55E-08
PDS0	FSB10	2.00E-10

The top-most sequence is related to the stipulation that all small LOCAs require an ECC make-up in conjunction with a steam generator heat sink to avoid damage (Item 10 in Section 6.1 and its footnote). As explained in the cited item, this may be unduly conservative for in-core LOCAs, which is the consequence of this break.

8.2.5 Total Loss of One Service Water Division

The SCDF contributors identified by the IE-SW-D2 event tree model in Appendix E are listed below. Sequence SW2-A13 could be mitigated by water supply from the RWS into the calandria vessel.

PDS No	Sequence No	Frequency (events/year)
PDS2	SW2-A23	1.89E-08
PDS2	SW2-A14	1.19E-08
PDS2	SW2-A13	1.18E-08
PDS0	SWD2-01	1.50E-09

8.2.6 Loss of Class IV Power to One Unit

The SCDF contributors identified by the IE-LCL4 event tree model in Appendix F are listed below. The frequencies meet the acceptance criteria and are not reviewed any further.

PDS No	Sequence No	Frequency (events/year)
PDS2	LCL4-B52	6.91E-08
PDS0	LCL4-3	3.00E-09

8.2.7 Loss of Inventory in Shield Cooling System

The SCDF contributors identified by the IE-SCB event tree model in Appendix G are listed below. The frequencies meet the acceptance criteria and are not reviewed any further.

PDS No	Sequence No	Frequency (events/year)
PDS0	SCB-5	1.20E-10
PDS0	SCB-4	1.20E-11

8.2.8 Small Steam Discharge Causing Low Level in Deaerator

The SCDF contributor identified by the IE-MSL3 event tree model in Appendix H is listed below. The frequency meets the acceptance criteria and is not reviewed any further.

PDS No	Sequence No	Frequency (events/year)
PDS0	MSL3-1	5.0E-12

8.2.9 Symmetric FW Line Break Upstream of FW Control Valves

The SCDF contributor identified by the IE-FWBS event tree model in Appendix I is listed below. The frequency meets the acceptance criteria and is not reviewed any further.

PDS No	Sequence No	Frequency (events/year)
PDS0	FWBS1	2.20E-09

8.2.10 Asymmetric FW Line Break Downstream of Steam Generator Check Valve

The SCDF contributors identified by the IE-FWBA event tree model in Appendix J are listed below. The frequencies meet the acceptance criteria and are not reviewed any further.

PDS No	Sequence No	Frequency (events/year)
PDS0	FWBA7	5.80E-11
PDS2	FWBA4	5.51E-08

8.2.11 Loss of Reactivity Control Leading to Core Power Excursion

The SCDF contributor identified by the IE-LOR event tree model in Appendix K is listed below. The frequency meets the acceptance criteria and is not reviewed any further.

PDS No	Sequence No	Frequency (events/year)
PDS0	LOR1	4.24E-08

8.3 Role of Reserve Water System

The preliminary event tree analyses have not comprehensively examined the role of the RWS in accident mitigation. Nevertheless, the limited credits for this system that were made are indicating a major impact on the SCDF. A sensitivity assessment was performed by removing all RWS-related credits from the events trees presented in this report. The summed SCDF value for the 11 selected initiating events then increases from 1.60E-06 events per year to 1.68E-05 events per year. This is an order of magnitude effect for credits that are far from being comprehensive. This sensitivity assessment only covers the prevention of Severe Core Damage and not the role of the RWS in severe accident mitigation.

Not counting the implicit credit for the reliability of the LTC-ECC system, the credits in this report involve only the steam generator make-up (in most, but not all event trees) and the HTS make-up (in only few event trees). A make-up of the moderator or the shield water is not credited in any of the trees.

The role of the RWS will be comprehensively assessed in the detailed PSA. A few observations stemming from the work performed so far are as follows.

- In order to support a sufficiently long mission time (24 hours), the operator needs to be able to actuate the RWS isolation valves within 3 hours and then modulate these valves for the remainder of the mission time. The design implications are that:
  - The batteries that provide the Class II power in case of the total loss of Class IV and III supplies should last for 3 hours (as best-estimate value).
  - Provisions to modulate RWS valves manually can be provided.

8.4 Key PSA Assumptions

The event tree analysis utilized many assumptions related to design and plant response. As part of the ACR detailed design development process, these assumptions are required to be supported by analysis and/or equipment supplier's test records as appropriate. A list of key PSA support assumptions and risk items is provided below:

- A connection from reserve feed water tank (turbine building) or demineralised water storage tank to the auxiliary feed water pumps suction header is important. This connection will enable the auxiliary feed water pump to maintain supply to the steam generators in the event of a loss of auxiliary condensate pump.
- A reliable, automatic closure of the auxiliary feed water level control valves when discrepancy between the two steam generator levels is sensed is important to crediting the auxiliary feed water in certain accidents.
- In order to defend calandria tube failure probability following a pressure tube rupture, the R&D program needs to demonstrate that the calandria tube will survive all relevant loading conditions. The program also needs to demonstrate that the calandria tube has a high creep rupture resistance. The latter is the ability of the calandria tube to withstand the elevated pressure and temperature environments after a pressure tube failure for long enough time so that operator action can be relied upon to reduce the HTS pressure. To afford high reliability credit for this operator action, the calandria tube needs to survive for about 2 hours or longer.
- Operator plays a crucial role following shield cooling accidents. For highly reliable actions, long times need to be available for manual actions. The current assumption based on CANDU-9 analysis is that 8 hours is available before the HTS pressure boundary, or any other boundary that holds water, could be threatened following a loss of shield water inventory. Analyses need to ascertain that long times are available as well.
- Although the HT pumps are not formally environmentally qualified, it is assumed (as best estimate) that at least one pump can run for up to 60 minutes after a feed water line break in the reactor building. This assumption is reasonably supported by the fact that the HTS pumps are not exposed to cavitation conditions (HTS remains pressurized and it is re-filled by emergency coolant) and that the discharge from the broken line will not directly impact the pumps. As far as practicable, the layout of the piping needs to minimize a harsh environment around the HT pumps.

9. CONCLUSIONS

This preliminary event tree analysis report has provided early inputs to the design teams regarding the reliability/unavailability requirements on the ACR systems that are used for accident mitigation as well as feedbacks on some of the system performance requirements. Completion of this design assist event tree analysis is an important milestone in that it has provided insights into adequacy of the safety design to help meet the acceptance criteria. The role of PSA is ongoing. These reliability targets are intended to guide the designers in the development of the design.

The analyses in this report identify which internal event sequences will likely dominate the SCDF in the ACR and which elements of the dominant sequences contribute most to the SCDF. Conservative simplifications are employed in this report, which will be reviewed, updated and removed during the detailed PSA.

This preliminary analysis provides a high degree of confidence that the design target on the summed SCDF for all internal and external events can be met.

10. REFERENCES

[1] AECL – “Licensing Basis for Advanced CANDU Reactor”, 108-00580-LBD-001, Rev. 0, 2002 July.

[2] AECL – ACR Technical Outline, 10810-01372-TED-001, Rev. 1, 2002 August.

[3] AECL - Probabilistic Safety Assessment Methodology , 108-03660-AB-001, Rev. 1, 2003 July.

[4] Science Applications International Corporation - “ETA-II Users Manual For Version 2.1”, now DS&S) 1992 August.

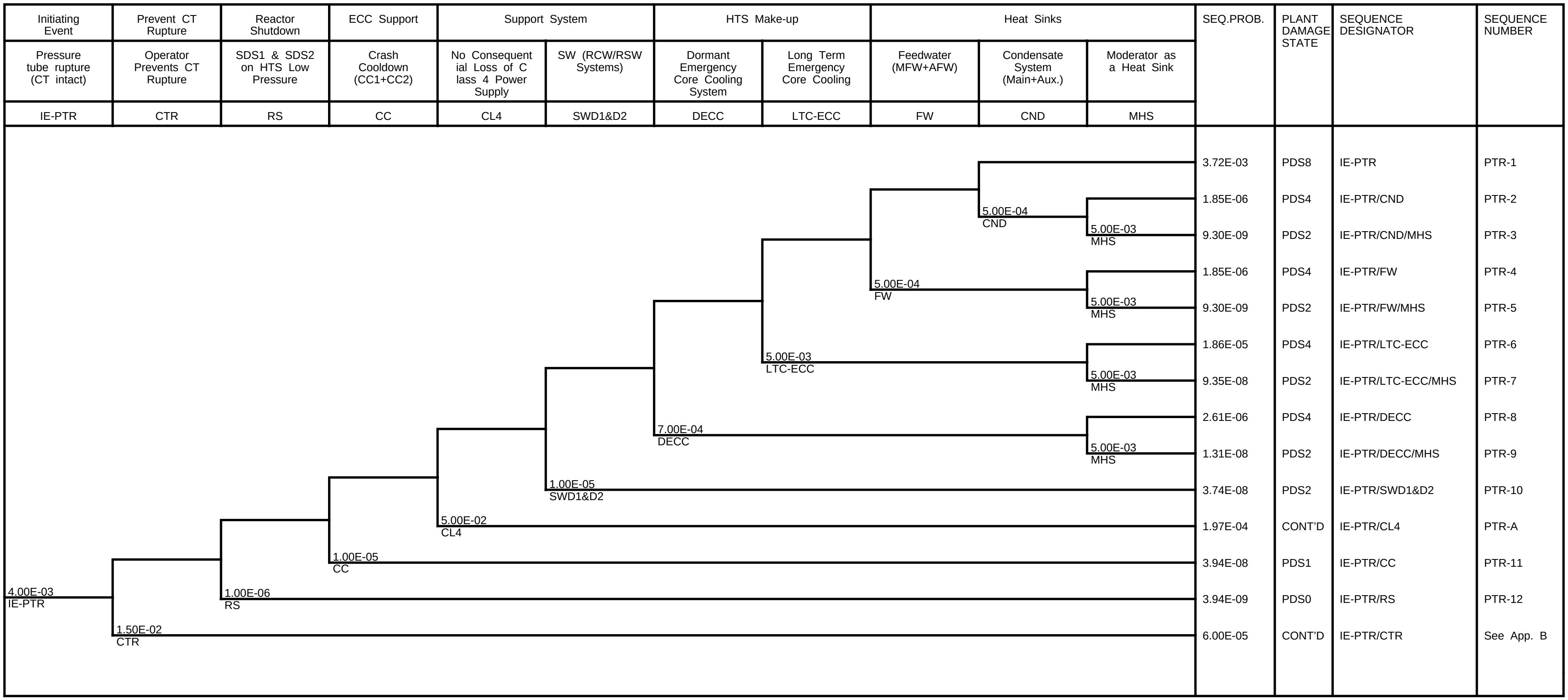
[5] US NRC, Accident Sequence Evaluation Program Human reliability Analysis Procedure, NUREG/CR-4772, Prepared for the US NRC by Sandia National Laboratories (SAND86-1996), Albuquerque, NM, February 1987.

[6] AECL – “Environmental Qualification”, 108-03650-SDG-003, Rev. 1, October 2002.

[7] CNSC - Safety Analysis of CANDU Nuclear Power Plant – Draft Regulatory Guide C-006, Rev. 1, 1999 September.

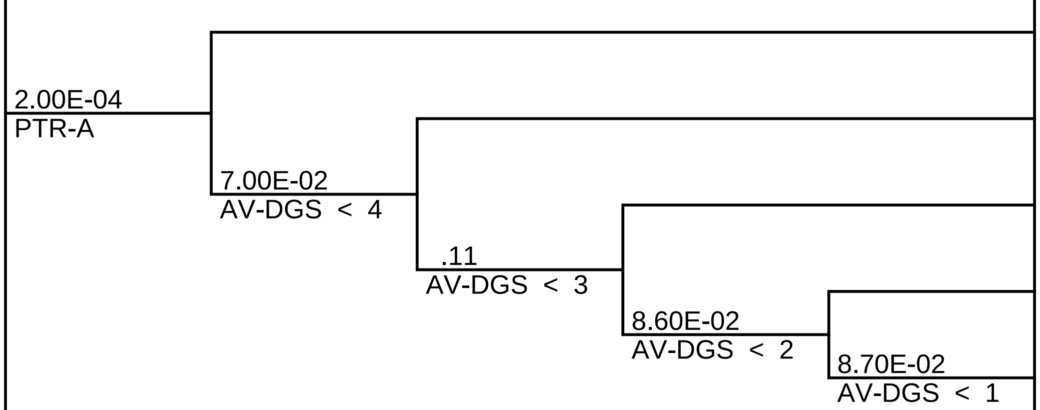
Appendix A

Event Tree for Pressure Tube Rupture

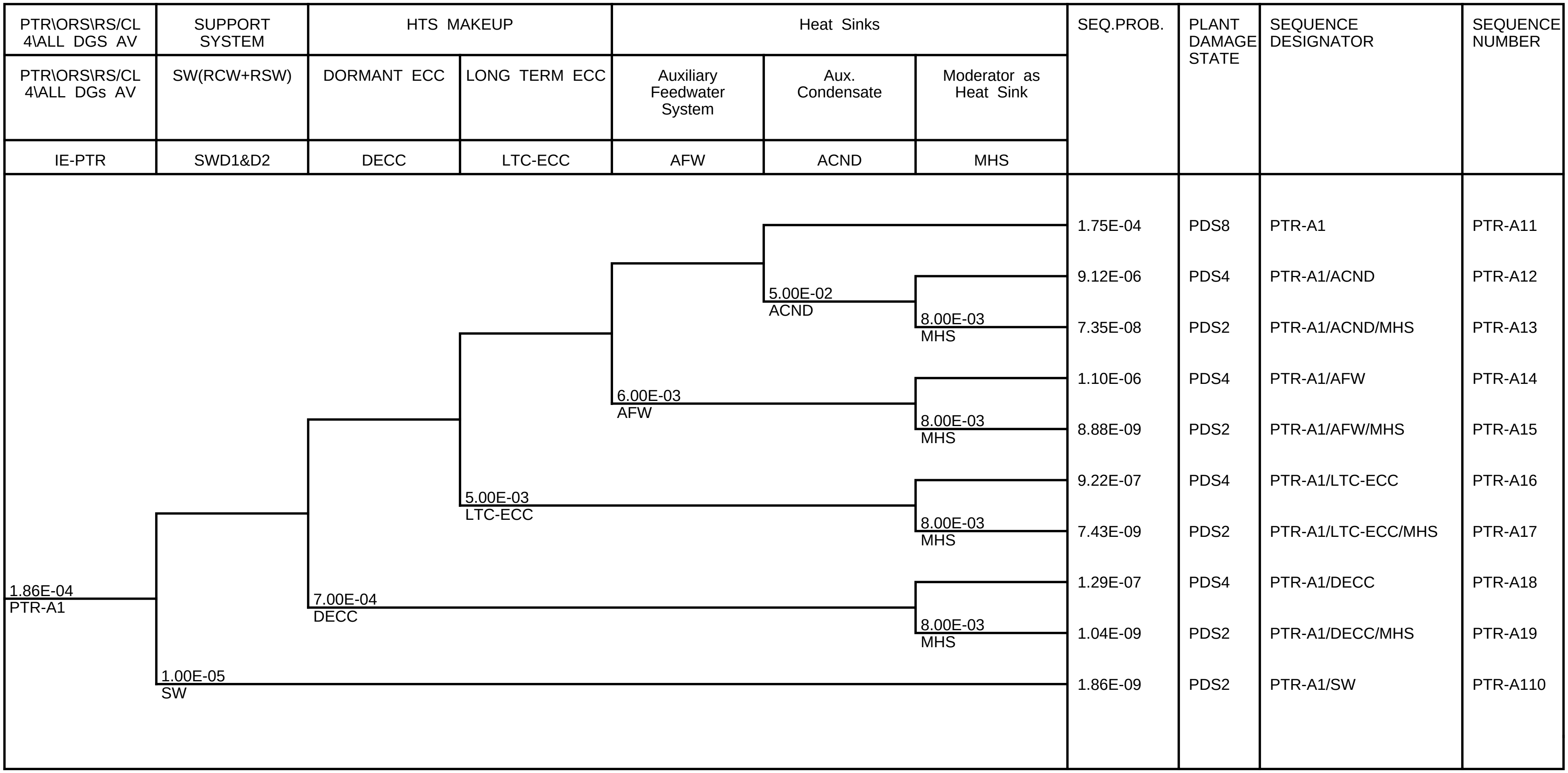


PRESSURE TUBE RUPTURE (CALANDRIA TUBE INTACT) C:\CAFTA\TREE(ET)\IE-PTR.TRE 23-01-04

PTR\ORS\RS /LOCL4	Class III Power Supply				SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE NUMBER	SEQUENCE DESCRIPTION
PT Rupture\ORS\RS/LOCL4	4 DGs Available for 24h	3 DGs Available for 24h	2 DGs Available for 24h	1 DG Available for 24h				
IE-PTR	DG-AV=4	DG-AV=3	DG-AV=2	DG-AV=1				
					1.86E-04	Cont'd	PTR-A1	IE-PTR/CL4/UNAV-DGS=0
					1.25E-05	Cont'd	PTR-A2	IE-PTR/CL4/UNAV-DGS=1
					1.41E-06	Cont'd	PTR-A3	IE-PTR/CL4/UNAV-DGS=2
					1.21E-07	Cont'd	PTR-A4	IE-PTR/CL4/UNAV-DGS=3
					1.15E-08	PDS1	PTR-A01	IE-PTR/CL4/UNAV-DGS=4

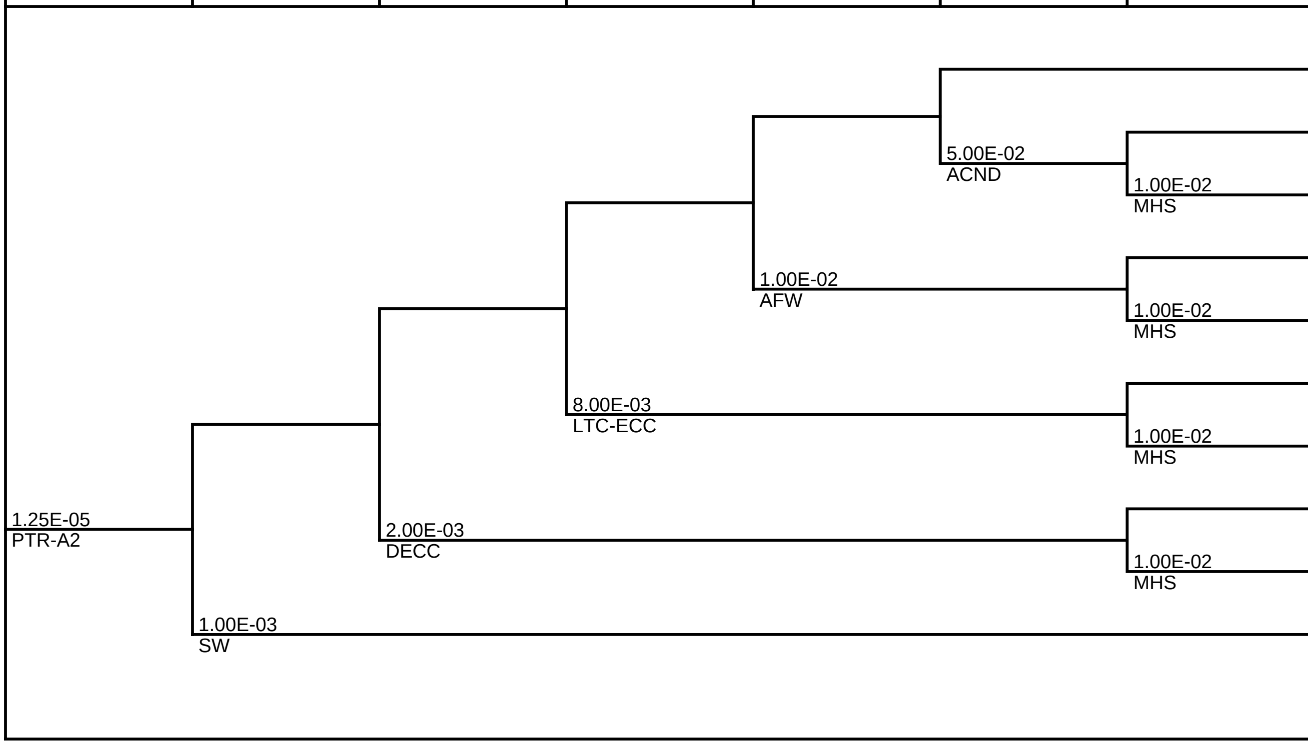


PRESSURE TUBE RUPTURE (CALANDRIA TUBE INTACT) C:\CAFTA\TREE(ET)\PTR-A.TRE 16-06-03



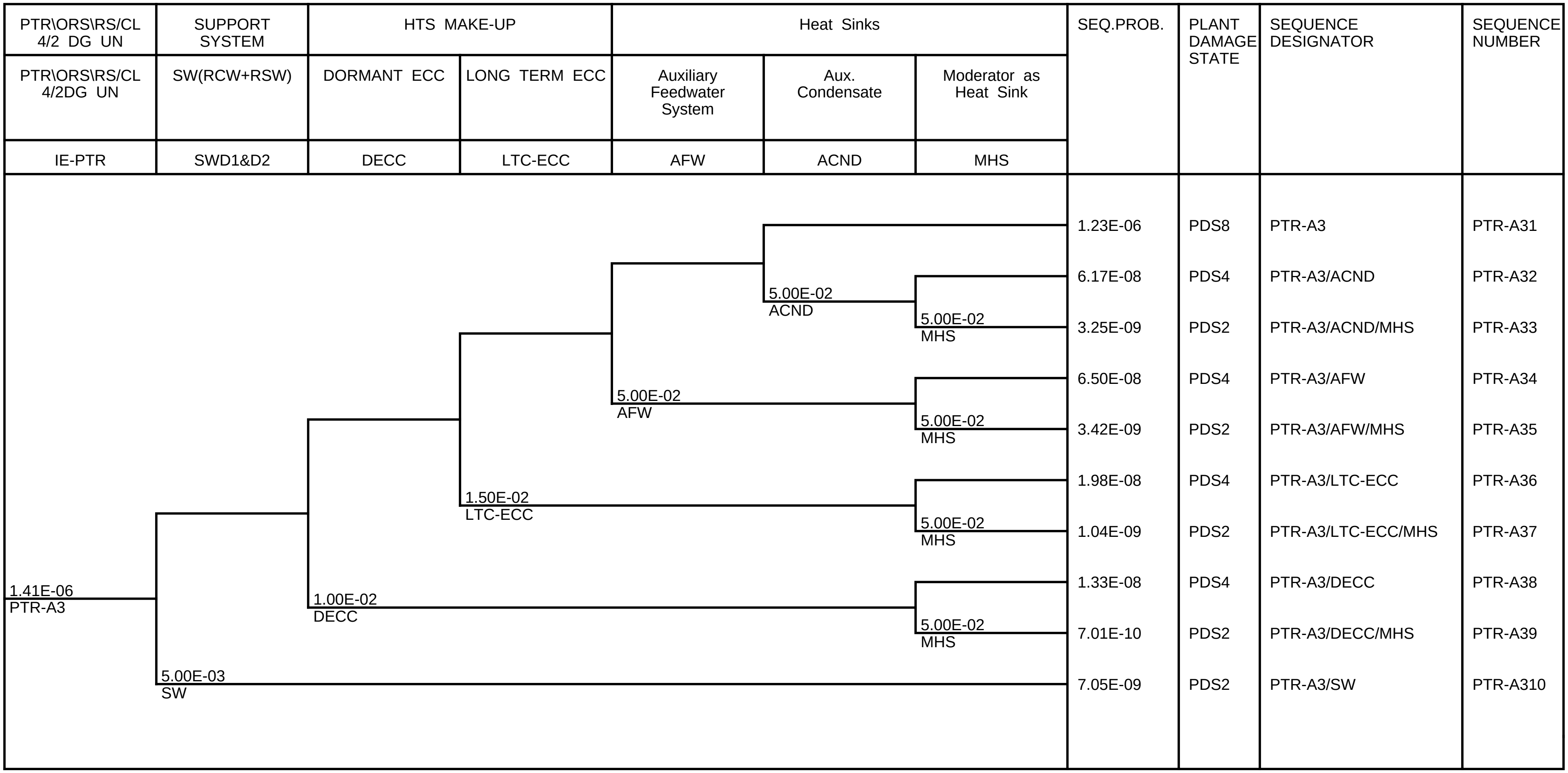
PRESSURE TUBE RUPTURE (CALANDRIA TUBE INTACT) C:\CAFTA\TREE(ET)\PTR-A1.TRE 16-06-03

PTR/ORS\RS/CL 4/1 DG UN	SUPPORT SYSTEM	HTS MAKE-UP		Heat Sinks		
PTR/ORS\RS/CL 4/1DG UN	SW(RCW+RSW)	DORMANT ECC	LONG TERM ECC	Auxiliary Feedwater System	Aux. Condensate	Moderator as Heat Sink
IE-PTR	SWD1&D2	DECC	LTC-ECC	AFW	ACND	MHS



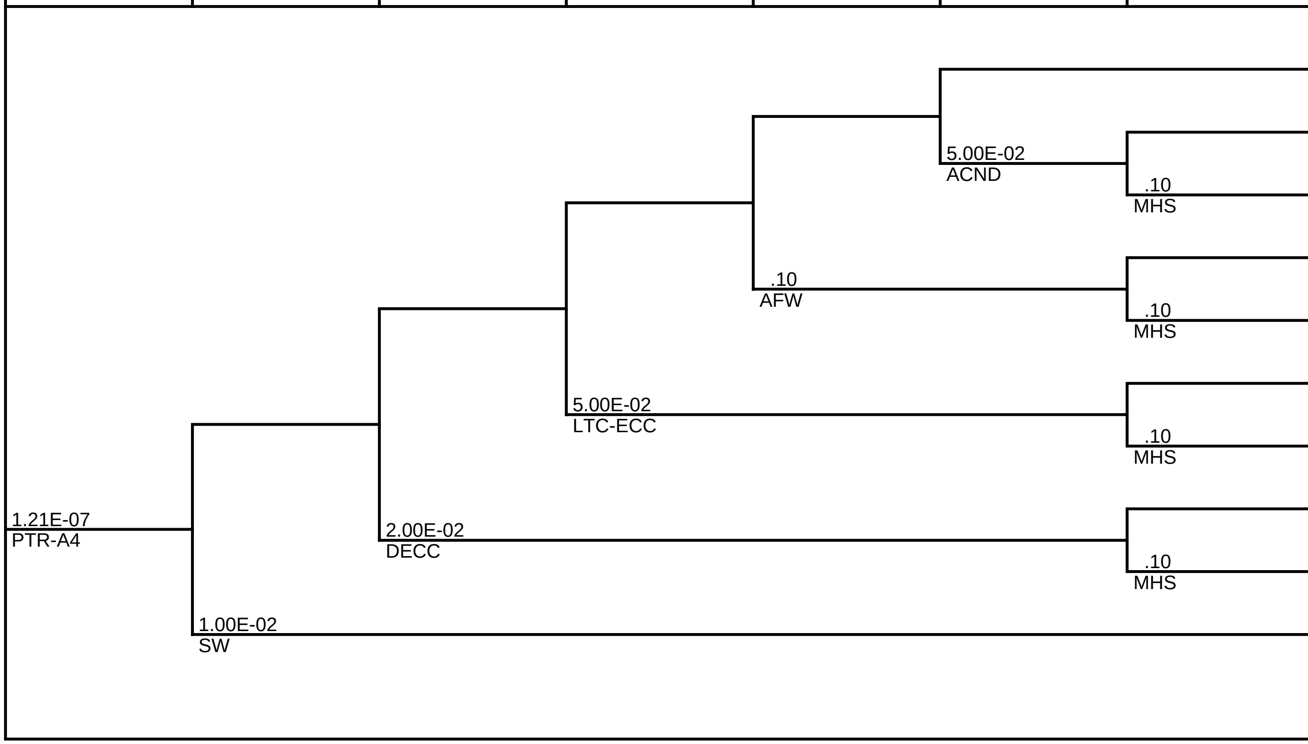
SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
1.16E-05	PDS8	PTR-A2	PTR-A21
6.06E-07	PDS4	PTR-A2/ACND	PTR-A22
6.12E-09	PDS2	PTR-A2/ACND/MHS	PTR-A23
1.22E-07	PDS4	PTR-A2/AFW	PTR-A24
1.24E-09	PDS2	PTR-A2/AFW/MHS	PTR-A25
9.87E-08	PDS4	PTR-A2/LTC-ECC	PTR-A26
9.97E-10	PDS2	PTR-A2/LTC-ECC/MHS	PTR-A27
2.47E-08	PDS4	PTR-A2/DECC	PTR-A28
2.50E-10	PDS2	PTR-A2/DECC/MHS	PTR-A29
1.25E-08	PDS2	PTR-A2/SW	PTR-A210

PRESSURE TUBE RUPTURE (CALANDRIA TUBE INTACT) C:\CAFTA\TREE(ET)\PTR-A2.TRE 16-06-03



PRESSURE TUBE RUPTURE (CALANDRIA TUBE INTACT) C:\CAFTA\TREE(ET)\PTR-A3.TRE 16-06-03

PTR\ORS\RS/CL 4/3 DG UN	SUPPORT SYSTEM	HTS MAKE-UP		Heat Sinks		
PTR\ORS\RS/CL 4/3DG UN	SW(RCW+RSW)	DORMANT ECC	LONG TERM ECC	Auxiliary Feedwater System	Aux. Condensate	Moderator as Heat Sink
IE-PTR	SWD1&D2	DECC	LTC-ECC	AFW	ACND	MHS

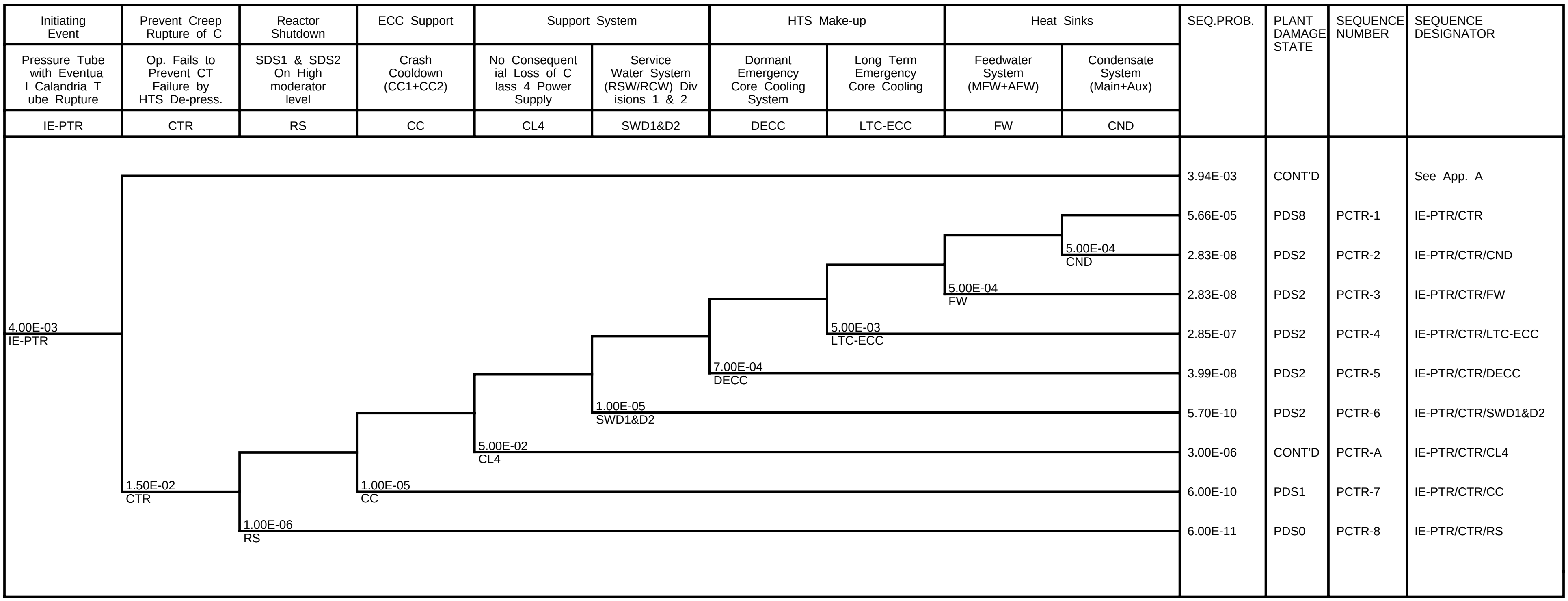


SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
9.54E-08	PDS8	PTR-A4	PTR-A41
4.52E-09	PDS4	PTR-A4/ACND	PTR-A42
5.02E-10	PDS2	PTR-A4/ACND/MHS	PTR-A43
1.00E-08	PDS4	PTR-A4/AFW	PTR-A44
1.12E-09	PDS2	PTR-A4/AFW/MHS	PTR-A45
5.28E-09	PDS4	PTR-A4/LTC-ECC	PTR-A46
5.87E-10	PDS2	PTR-A4/LTC-ECC/MHS	PTR-A47
2.16E-09	PDS4	PTR-A4/DECC	PTR-A48
2.40E-10	PDS2	PTR-A4/DECC/MHS	PTR-A49
1.21E-09	PDS2	PTR-A4/SW	PTR-A50

PRESSURE TUBE RUPTURE (CALANDRIA TUBE INTACT) C:\CAFTA\TREE(ET)\PTR-A4.TRE 16-06-03

Appendix B

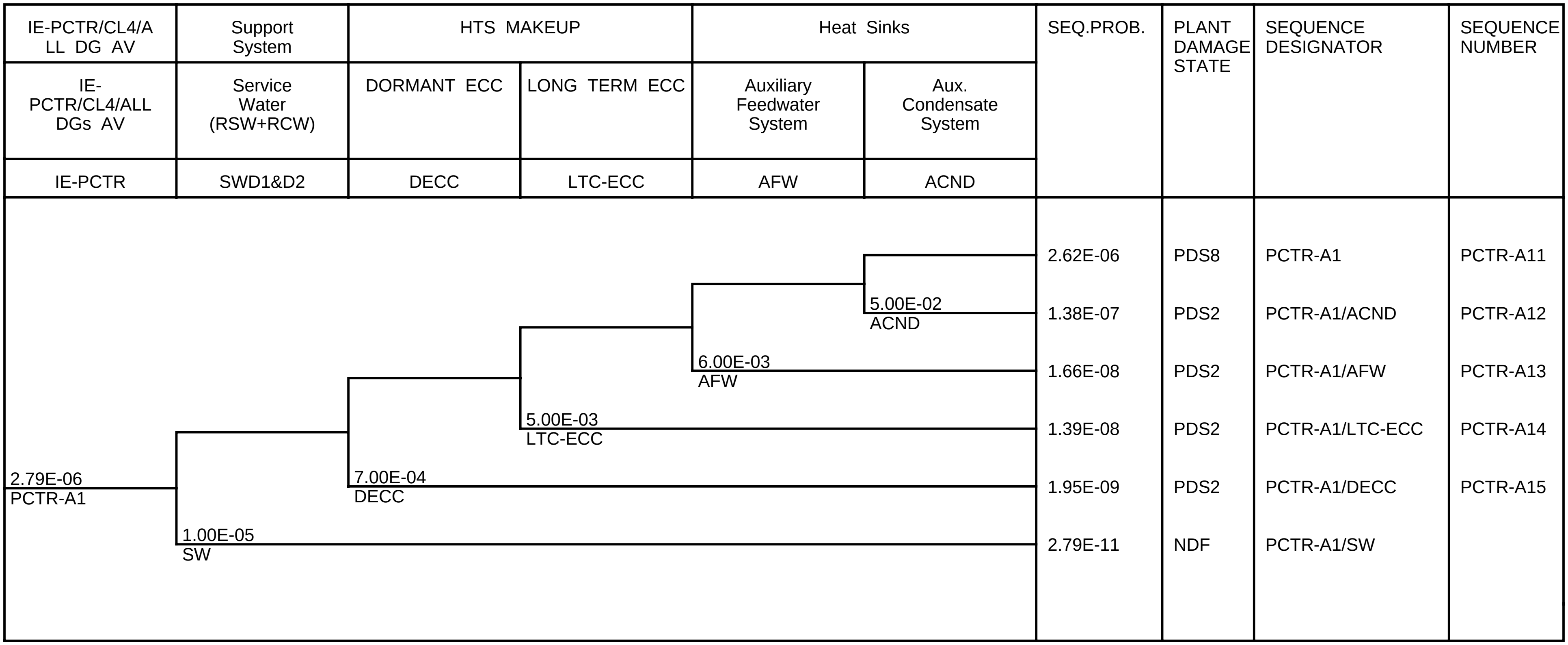
Event Tree for Pressure Tube and Calandria Tube Rupture



PRESSURE TUBE RUPTURE (CALANDRIA TUBE RUPTURE) C:\CAFTA\TREE(ET)\IE-PT-CT.TRE 23-01-04

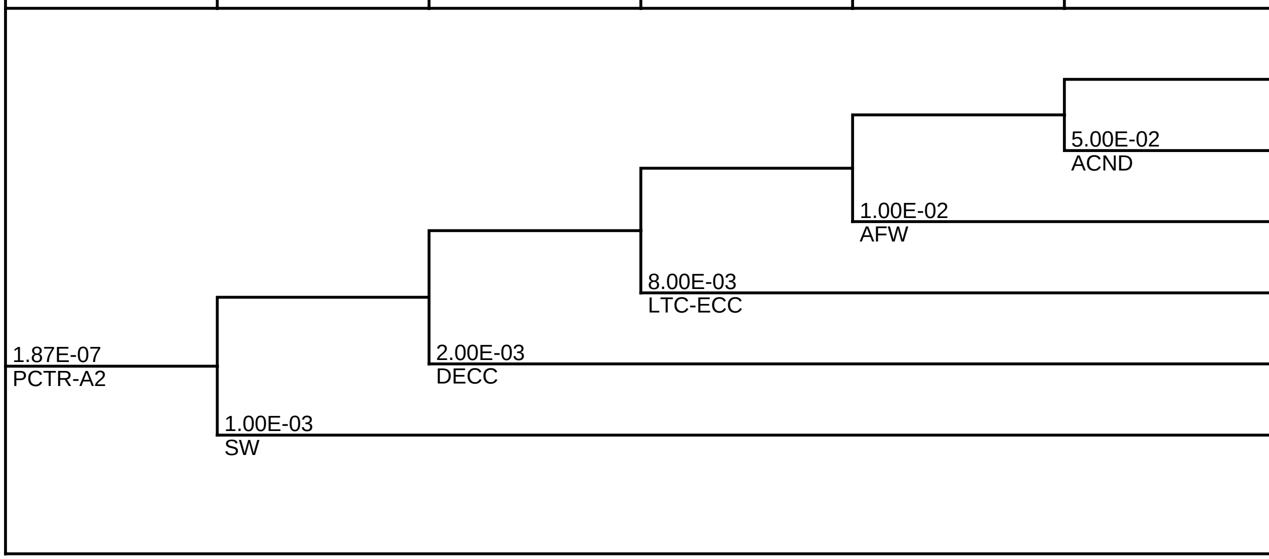
Transfer	Class III Power Supply				SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE NUMBER	SEQUENCE DESCRIPTION
PT/CT Rupture & Consequential Loss of Cl	4 DGs Available FOR 24h	3 DGs Available for 24h	2 DGs Available for 24h	1 DG Available for 24h				
IE-PCTR	DG-AV=4	DG-AV=3	DG-AV=2	DG-AV=1				
3.00E-06 PCTR-A					2.79E-06	CONT'D	PCTR-A1	IE-PCTR/UNAV-DGS=0
	7.00E-02 AV-DGS < 4				1.87E-07	CONT'D	PCTR-A2	IE-PCTR/UNAV-DGS=1
		.11 AV-DGS < 3			2.11E-08	CONT'D	PCTR-A3	IE-PCTR/UNAV-DGS=2
			8.60E-02 AV-DGS < 2		1.81E-09	CONT'D	PCTR-A4	IE-PCTR/UNAV-DGS=3
				8.70E-02 AV-DGS < 1	1.73E-10	PDS1	PCTR-A01	IE-PCTR/UNAV-DGS=4

PRESSURE TUBE RUPTURE (CALANDRIA TUBE RUPTURE) C:\CAFTA\TREE(ET)\PCTR-A.TRE 03-12-03



PRESSURE TUBE RUPTURE (CALANDRIA TUBE RUPTURE) C:\CAFTA\TREE(ET)\PCTR-A1.TRE 03-12-03

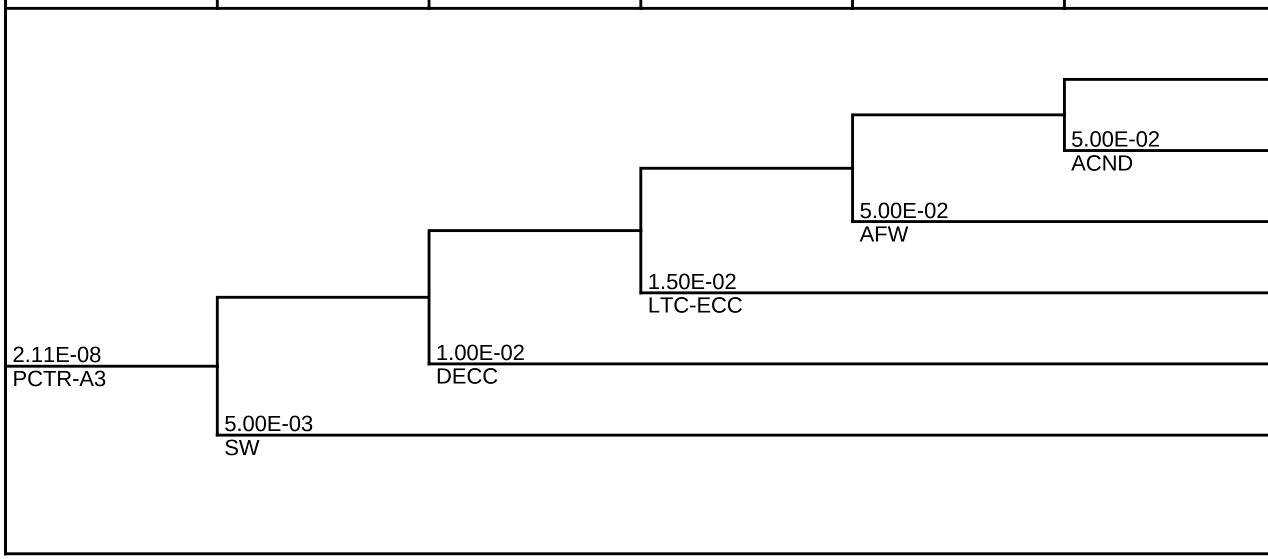
PCTR-A2	Support System	HTS Make-up		Heat Sinks	
IE-PCTR/CL4/UNAV-DGS=1	Service water (RSW+RCW)	DORMANT ECC	LONG TERM ECC	Auxiliary Feedwater System	Aux. Condensate System
IE-PCTR	SWD1&D2	DECC	LTC-ECC	AFW	ACND



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
1.74E-07	PDS8	PCTR-A2	PCTR-A21
9.15E-09	PDS2	PCTR-A2/ACND	PCTR-A22
1.85E-09	PDS2	PCTR-A2/AFW	PCTR-A23
1.49E-09	PDS2	PCTR-A2/LTC-ECC	PCTR-A24
3.74E-10	PDS2	PCTR-A2/DECC	PCTR-A25
1.87E-10	PDS2	PCTR-A2/SW	PCTR-A26

PRESSURE TUBE RUPTURE (CALANDRIA TUBE RUPTURE) C:\CAFTA\TREE(ET)\PCTR-A2.TRE 03-12-03

PCTR-A3	Support System	HTS Make-up		Heat Sinks	
IE-PCTR/CL4/UNAV-DGS=2	Service Water System (RSW+RCW) Divisions 1 & 2	DORMANT ECC	LONG TERM ECC	Auxiliary Feedwater System	Aux. Condensate System
IE-PCTR	SWD1&D2	DECC	LTC-ECC	AFW	ACND



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
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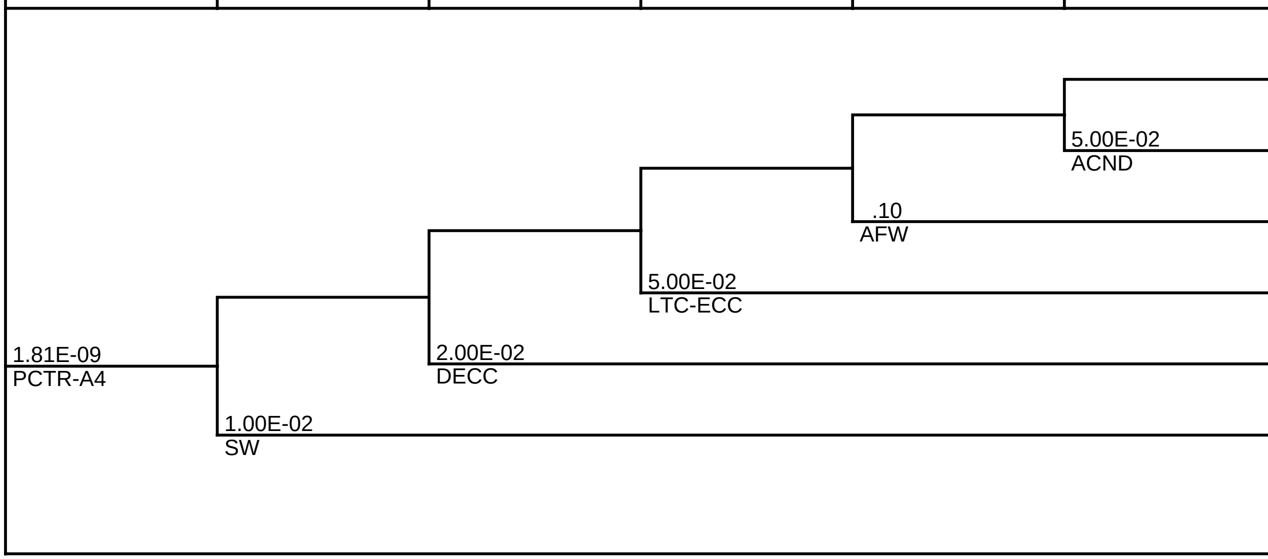
9.72E-10	PDS2	PCTR-A3/ACND	PCTR-A32
1.02E-09	PDS2	PCTR-A3/AFW	PCTR-A33
3.12E-10	PDS2	PCTR-A3/LTC-ECC	PCTR-A34
2.10E-10	PDS2	PCTR-A3/DECC	PCTR-A35
1.05E-10	NDF	PCTR-A3/SW	

PRESSURE TUBE RUPTURE (CALANDRIA TUBE RUPTURE) C:\CAFTA\TREE(ET)\PCTR-A3.TRE 03-12-03

PCTR-A4	Support System	HTS Make-up		Heat Sinks	
IE-PCTR/CL4/3 DG UN	Service Water System (RSW+RCW) Divisions 1 & 2	DORMANT ECC	LONG TERM ECC	Auxiliary Feedwater System	Aux. Condensate System
IE-PCTR	SWD1&D2	DECC	LTC-ECC	AFW	ACND



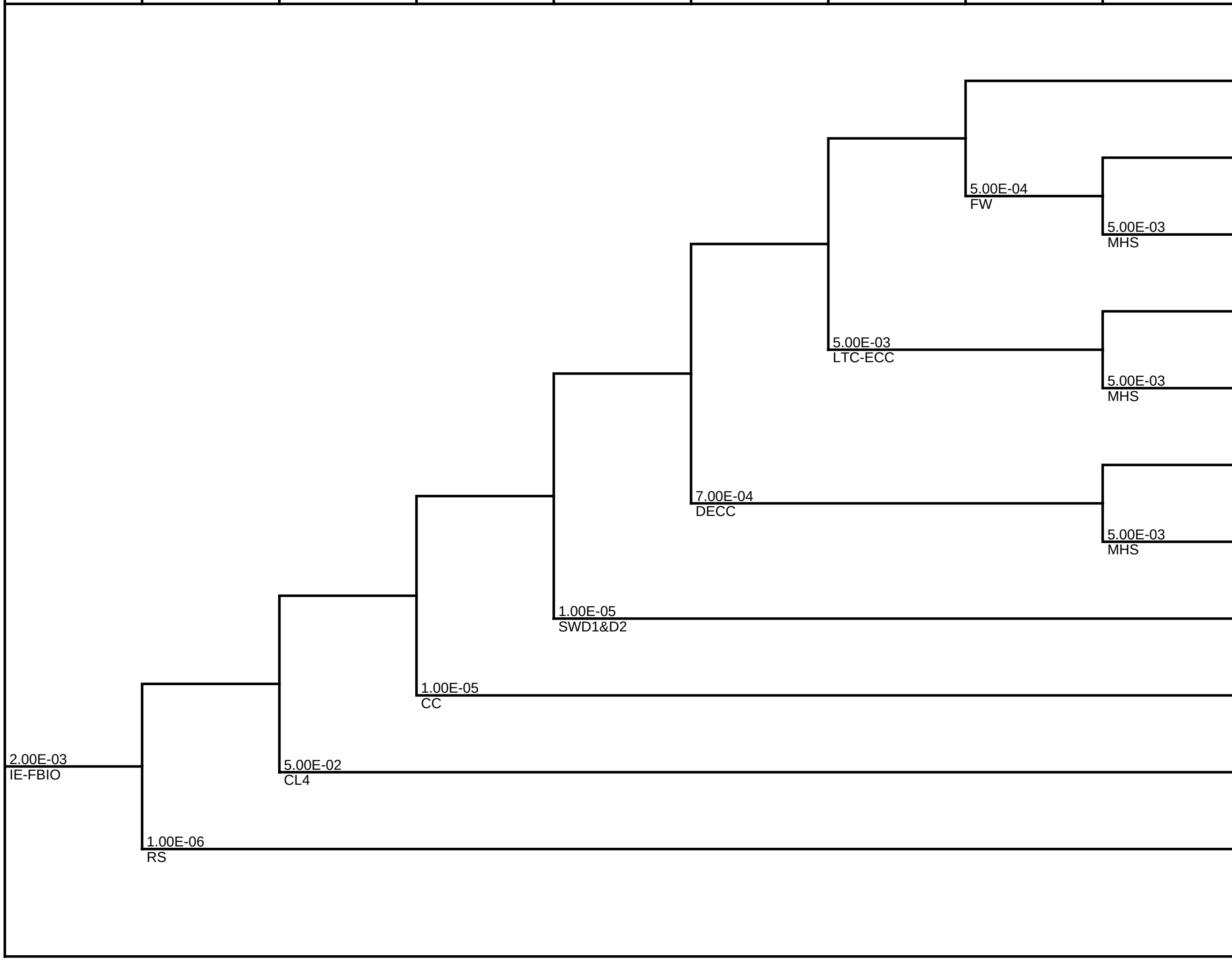
SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
1.43E-09	PDS8	PCTR-A4	PCTR-A41
7.51E-11	PDS2	PCTR-A4/ACND	PCTR-A42
1.67E-10	PDS2	PCTR-A4/AFW	PCTR-A43
8.78E-11	PDS2	PCTR-A4/LTC-ECC	PCTR-A44
3.58E-11	PDS2	PCTR-A4/DECC	PCTR-A45
1.81E-11	NDF	PCTR-A4/SW	

PRESSURE TUBE RUPTURE (CALANDRIA TUBE RUPTURE) C:\CAFTA\TREE(ET)\PCTR-A4.TRE 06-11-03

Appendix C

Event Tree for Feeder Break

Initiating Event	Reactor Shutdown	Support Systems			HTS Makeup		Heat Sink	
Feeder Break	SDS1&SDS2	No Consequential loss of CL4 Supply	Crash Cooldown	SWD1&D2 (RSW/RCW) System (CL4 available)	HTS Makeup	Long Term Cooling System (ECC function)	FW Supply to SGs	Moderator Acts as Heat Sink
IE-FBIO	RS	CL4	CC	SWD1&D2	DECC	LTC-ECC	FW	MHS



SEQ.FREQ.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
1.89E-03	PDS7	IE-FBIO	FBIO1
9.40E-07	PDS4	IE-FBIO/FW	FBIO2
4.72E-09	PDS2	IE-FBIO/FW/MHS	FBIO3
9.45E-06	PDS4	IE-FBIO/LTC-ECC	FBIO4
4.75E-08	PDS2	IE-FBIO/LTC-ECC/MHS	FBIO5
1.32E-06	PDS4	IE-FBIO/DECC	FBIO6
6.65E-09	PDS2	IE-FBIO/DECC/MHS	FBIO7
1.90E-08	PDS2	IE-FBIO/SWD1&D2	FBIO8
1.90E-08	PDS1	IE-FBIO/CC	FBIO9
1.00E-04	CONT'D	IE-FBIO/CL4	FBIO-A
2.00E-09	PDS0	IE-FBIO/RS	FBIO10

Fig.1 (Feeder Break) Page(1) C:\CAFTA\TREE(ET)\FBIO-01.TRE 28-01-04

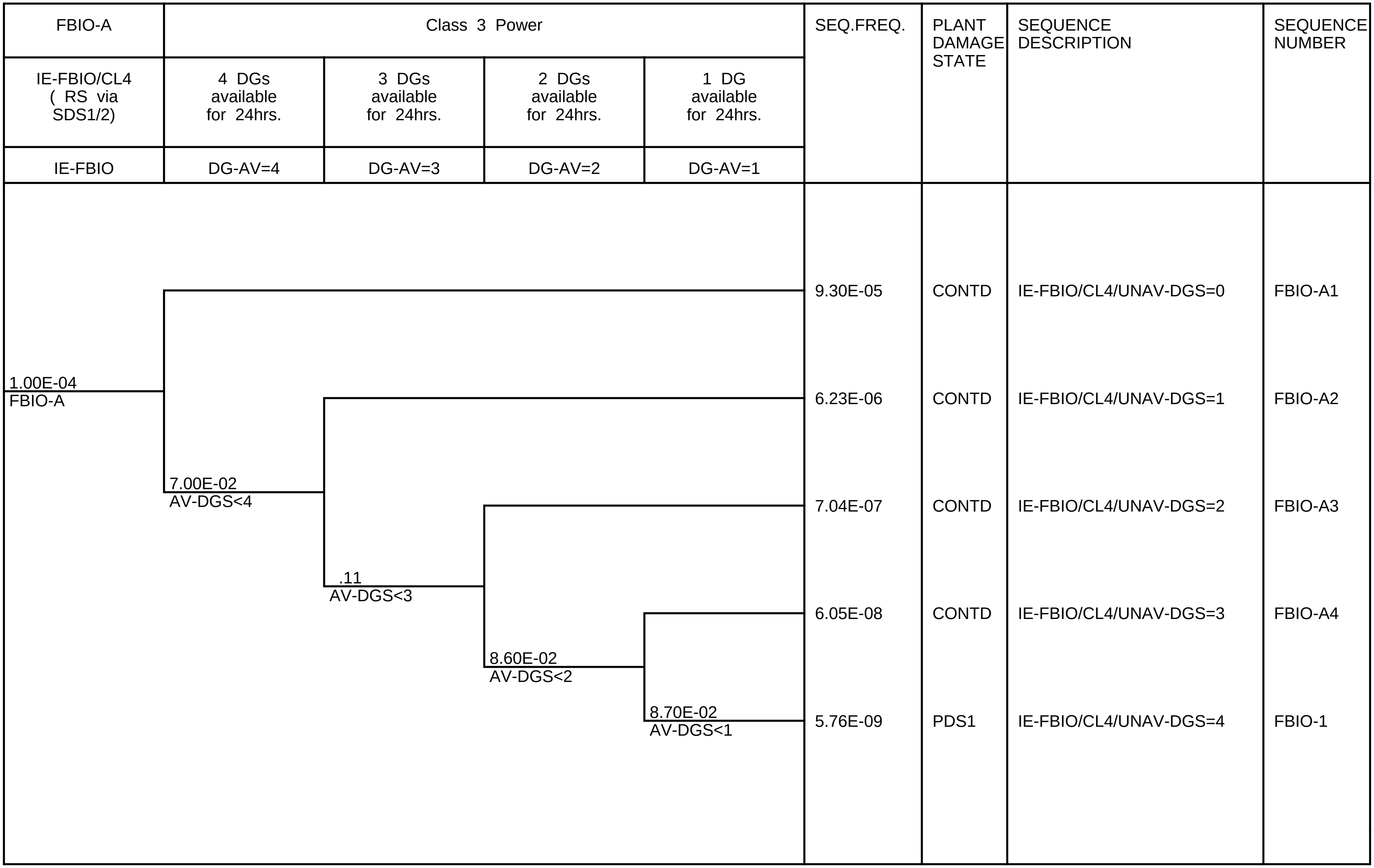


Fig.2 (Feeder Break) Page(2) C:\CAFTA\TREE(ET)\FBIO-02.TRE 20-01-04

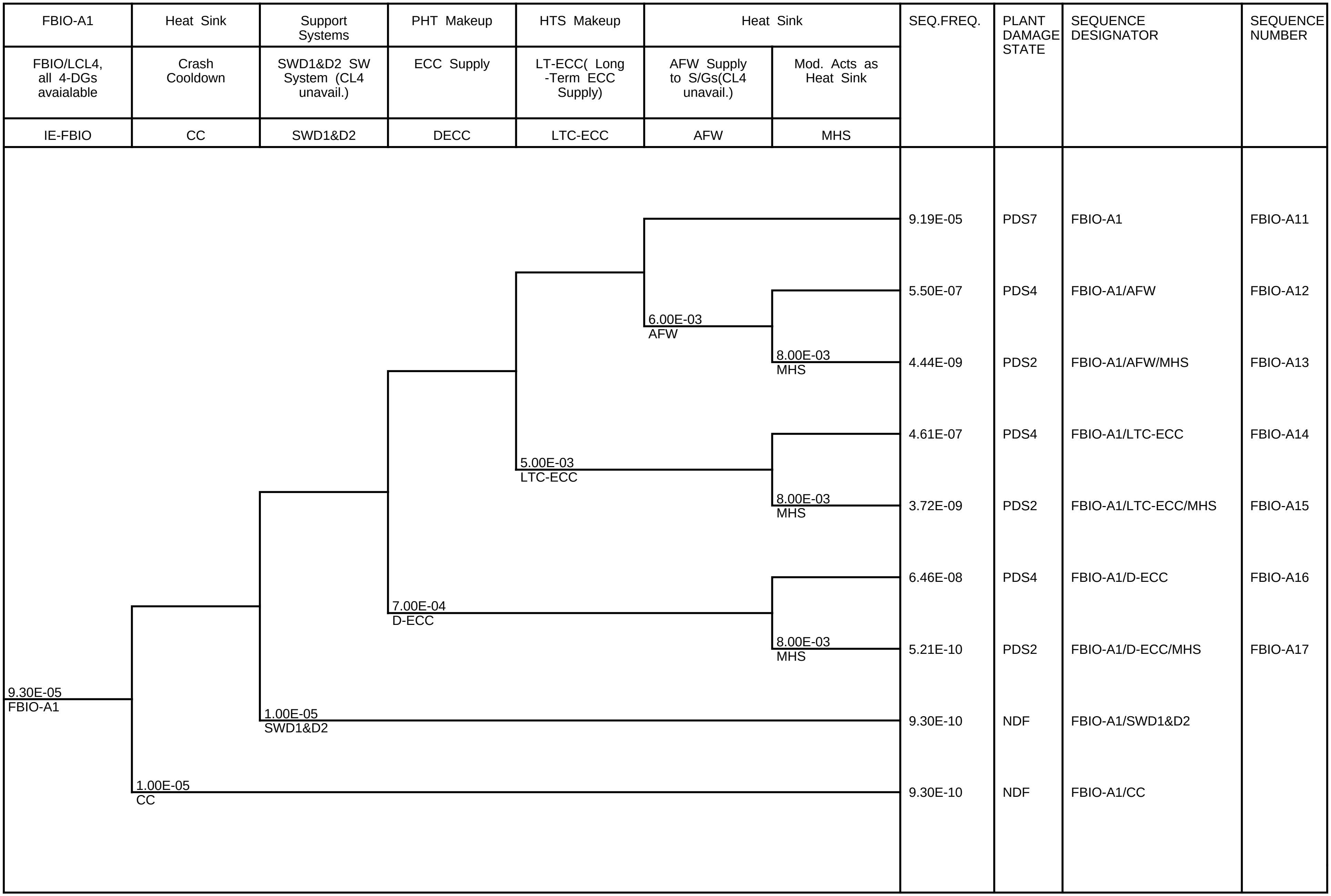
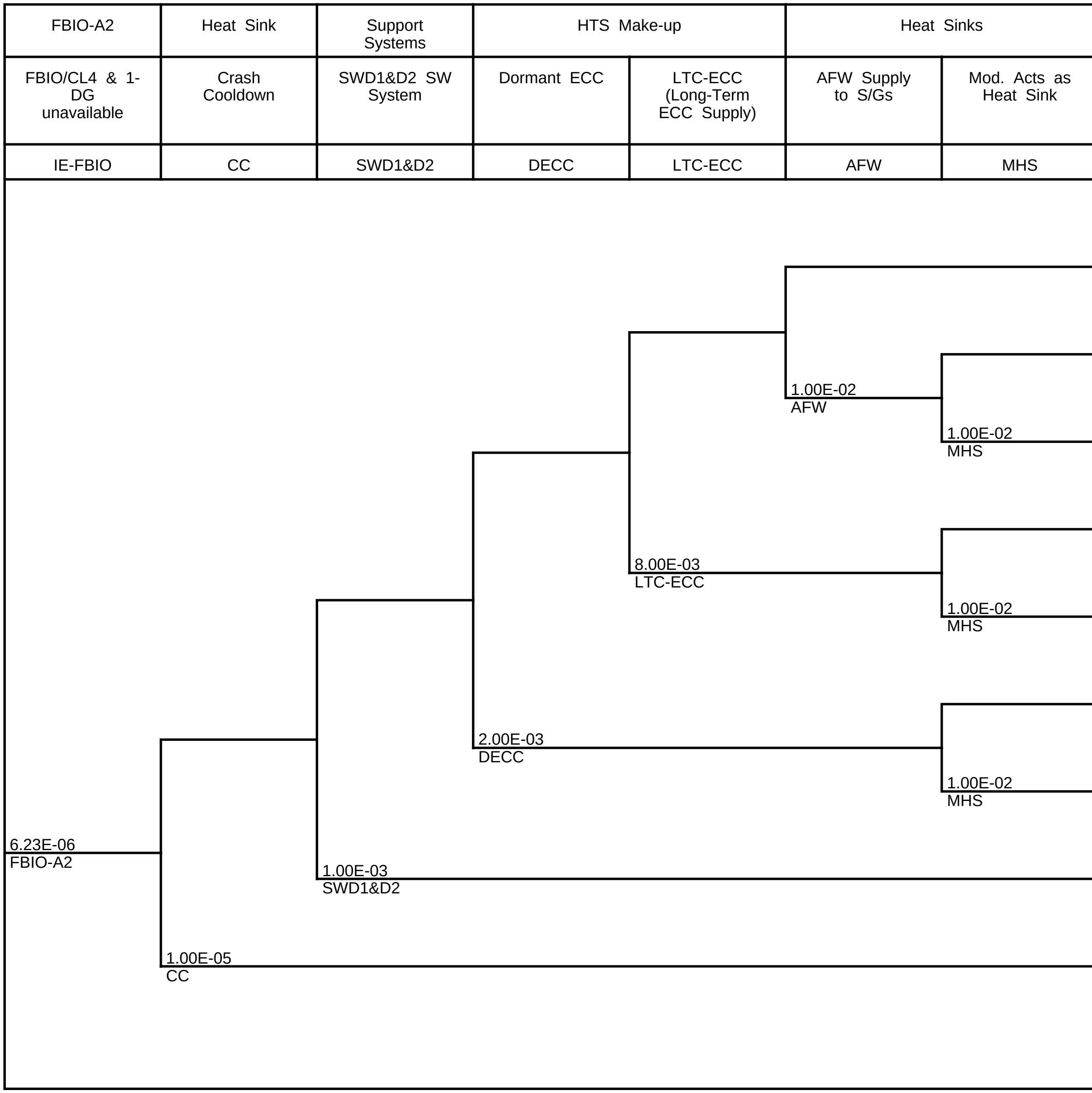
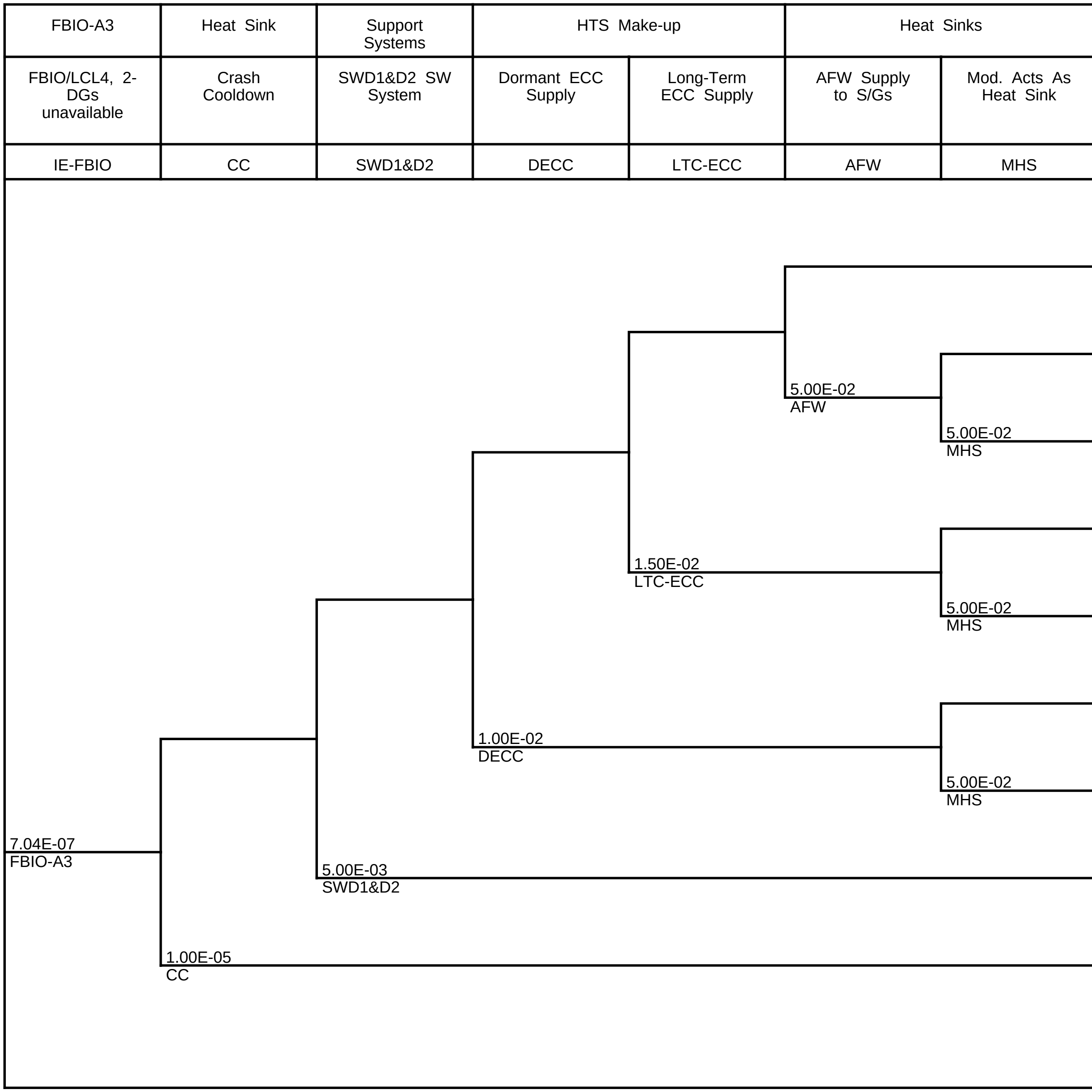


Fig.3 (Feeder Break)



SEQ.FREQ.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
6.10E-06	PDS7	FBIO-A2	FBIO-A21
6.10E-08	PDS4	FBIO-A2/AFW	FBIO-A22
6.16E-10	PDS2	FBIO-A2/AFW/MHS	FBIO-A23
4.92E-08	PDS4	FBIO-A2/LTC-ECC	FBIO-A24
4.97E-10	PDS2	FBIO-A2/LTC-ECC/MHS	FBIO-A25
1.23E-08	PDS4	FBIO-A2/DECC	FBIO-A26
1.24E-10	PDS2	FBIO-A2/DECC/MHS	FBIO-A27
6.23E-09	PDS1	FBIO-A2/SWD1&D2	FBIO-A28
6.23E-11	NDF	FBIO-A2/CC	

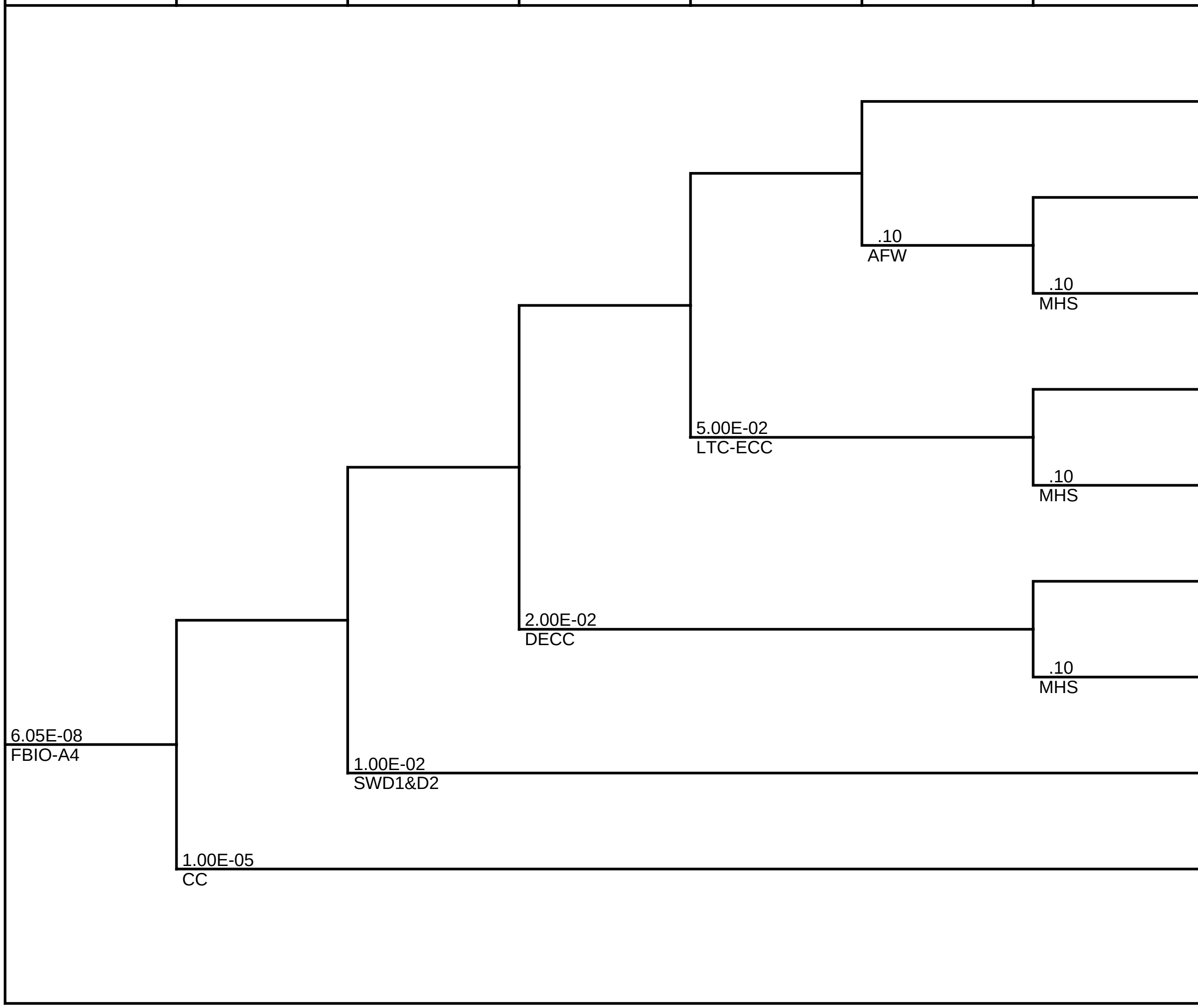
Fig.4 (Feeder Break) Page(4) C:\CAFTA\TREE(ET)\FBIO-04.TRE 03-06-03



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
6.49E-07	PDS7	FBIO-A3	FBIO-A31
3.24E-08	PDS4	FBIO-A3/AFW	FBIO-A32
1.71E-09	PDS2	FBIO-A3/AFW/MHS	FBIO-A33
9.88E-09	PDS4	FBIO-A3/LTC-ECC	FBIO-A34
5.20E-10	PDS2	FBIO-A3/LTC-ECC/MHS	FBIO-A35
6.65E-09	PDS4	FBIO-A3/DECC	FBIO-A36
3.50E-10	PDS2	FBIO-A3/DECC/MHS	FBIO-A37
3.52E-09	PDS2	FBIO-A3/SWD1&D2	FBIO-A38
7.04E-12	NDF	FBIO-A3/CC	

Fig.5 (Feeder Break)

FBIO-A4	Heat Sink	Support Systems	PHT Makeup	HTS Makeup	Heat Sink	
FBIO/LCL4, 3-DGs unavaialable	Crash Cooldown	SWD1&D2 SW (RSW/RCW) System	ECC System	Long-Term ECC Supply	AFW Supply to S/Gs	Moderator Acts as Heat Sink
IE-FBIO	CC	SWD1&D2	DECC	LTC-ECC	AFW	MHS



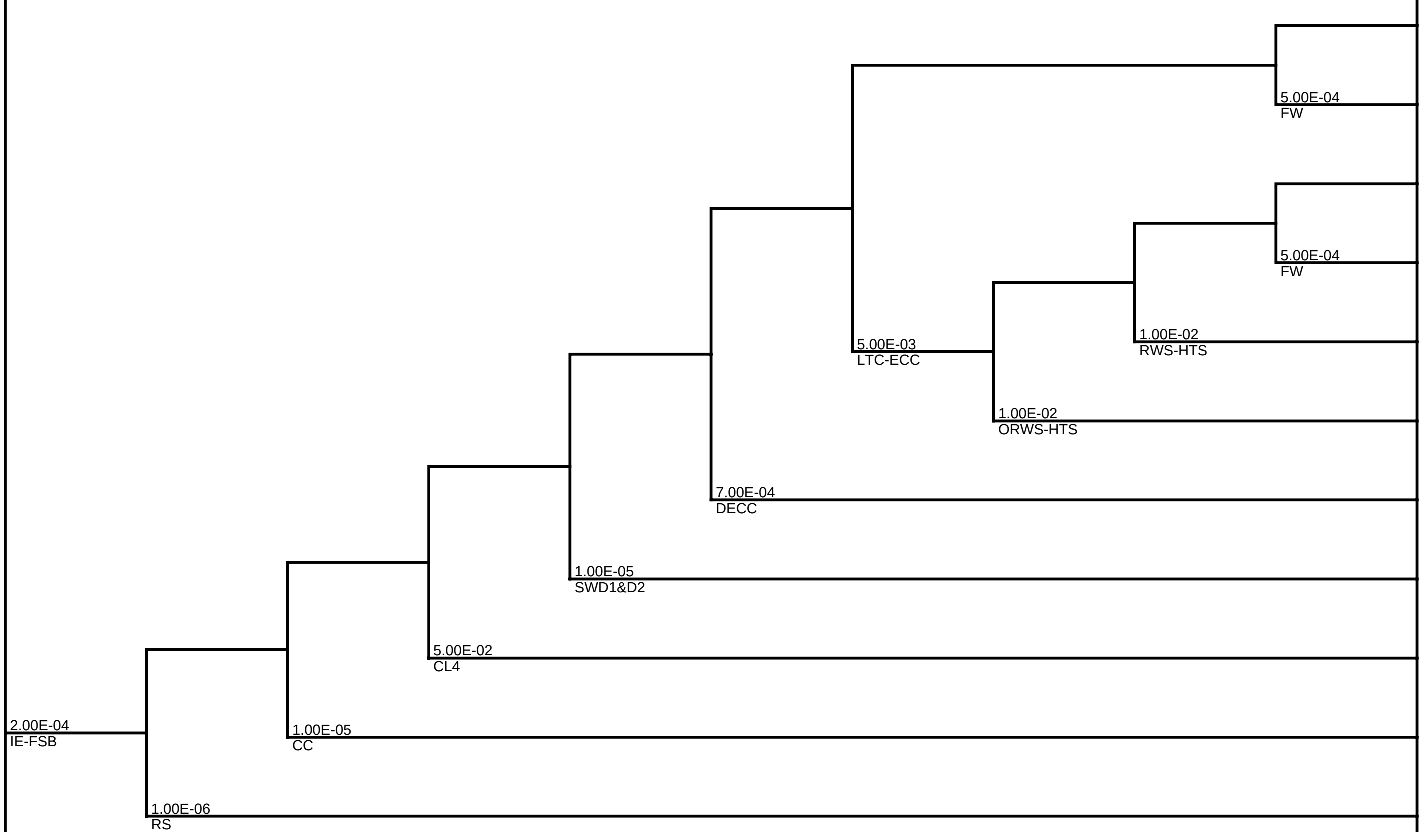
SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
5.02E-08	PDS7	FBIO-A4	FBIO-A41
5.02E-09	PDS4	FBIO-A4/AFW	FBIO-A42
5.58E-10	PDS2	FBIO-A4/AFW/MHS	FBIO-A43
2.64E-09	PDS4	FBIO-A4/LTC-ECC	FBIO-A44
2.93E-10	PDS2	FBIO-A4/LTC-ECC/MHS	FBIO-A45
1.08E-09	PDS4	FBIO-A4/DECC	FBIO-A46
1.20E-10	PDS2	FBIO-A4/DECC/MHS	FBIO-A47
6.05E-10	NDF	FBIO-A4/SWD1&D2	
6.05E-13	NDF	FBIO-A4/CC	

Fig.6 (Feeder Break)

Appendix D

Event Tree for Feeder Stagnation Break

Initiating Event	Reactor Shutdown	Heat Sink	Support Systems		HTS Makeup		Operator Action	HTS Make-up	Heat Sink
Feeder Stagnation Break	Reactor Shutdown by SDS1/SDS2	Crash Cooldown	No Consequential Loss of CL4 Power Supply	Divisions 1 & 2 Service Water System	ECC Supply	LT-ECC Supply to HTS	Operator Starts Reserve Water Make-up to HTS	Reserve Water System Make-up to HTS	FW Supply to S/Gs
IE-FSB	RS	CC	CL4	SWD1&D2	DECC	LTC-ECC	ORWS-HTS	RWS-HTS	FW



SEQ.FREQ.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
1.89E-04	PDS8	IE-FSB	FSB1
9.45E-08	PDS2	IE-FSB/FW	FSB2
9.30E-07	PDS8	IE-FSB/LTC-ECC	FSB3
4.65E-10	PDS2	IE-FSB/LTC-ECC/FW	FSB4
9.40E-09	PDS2	IE-FSB/LTC-ECC/RWS-HTS	FSB5
9.49E-09	PDS2	IE-FSB/LTC-ECC/ORWS-HTS	FSB6
1.33E-07	PDS2	IE-FSB/DECC	FSB7
1.90E-09	PDS2	IE-FSB/SWD1&D2	FSB8
1.00E-05	CONT'D	IE-FSB/CL4	FSB-A
2.00E-09	PDS1	IE-FSB/CC	FSB9
2.00E-10	PDS0	IE-FSB/RS	FSB10

Fig.1 (Feeder Stagnation Break) Page(1) C:\CAFTA\TREE(ET)\IE-FSB.TRE 22-01-04

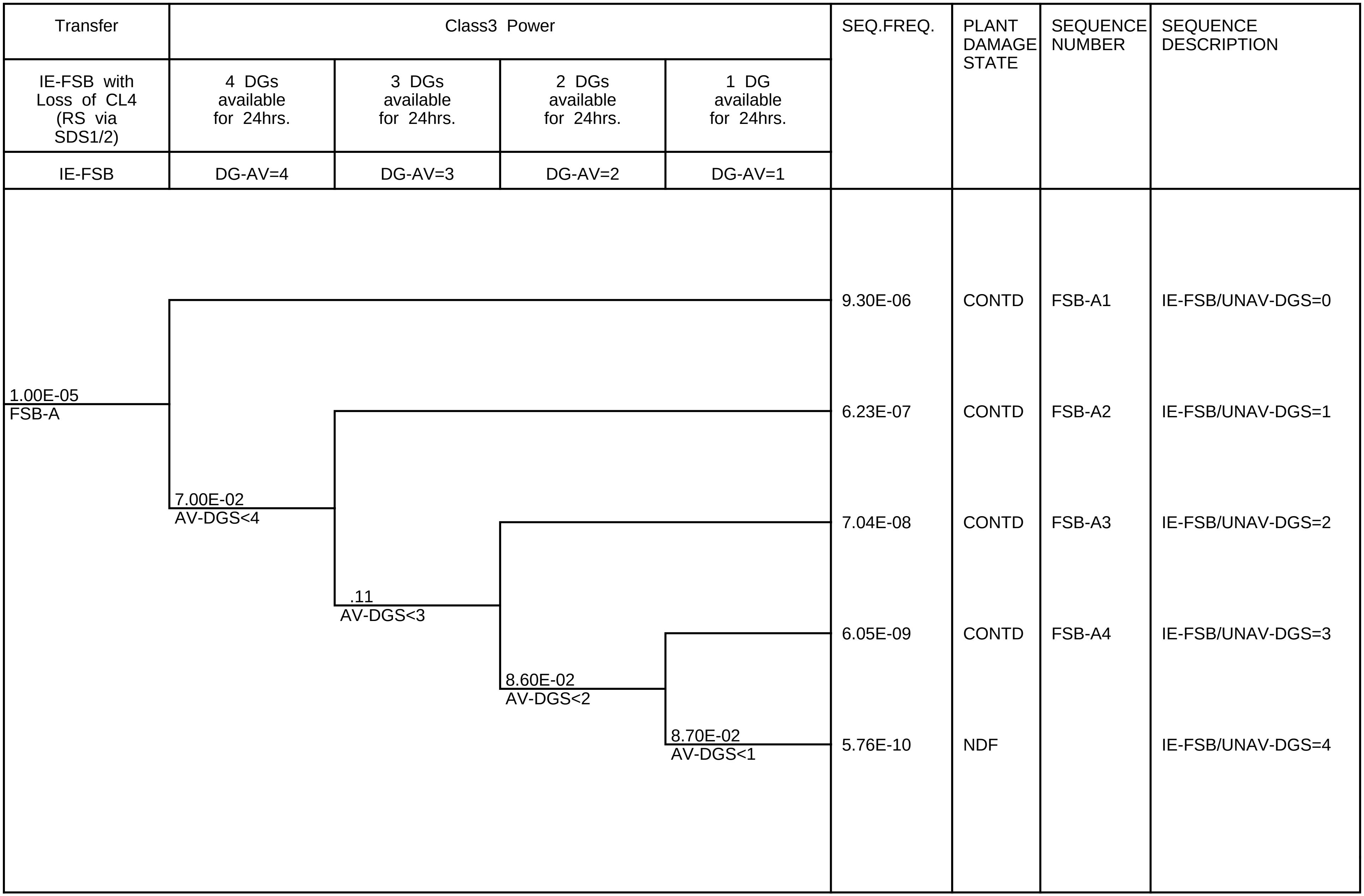
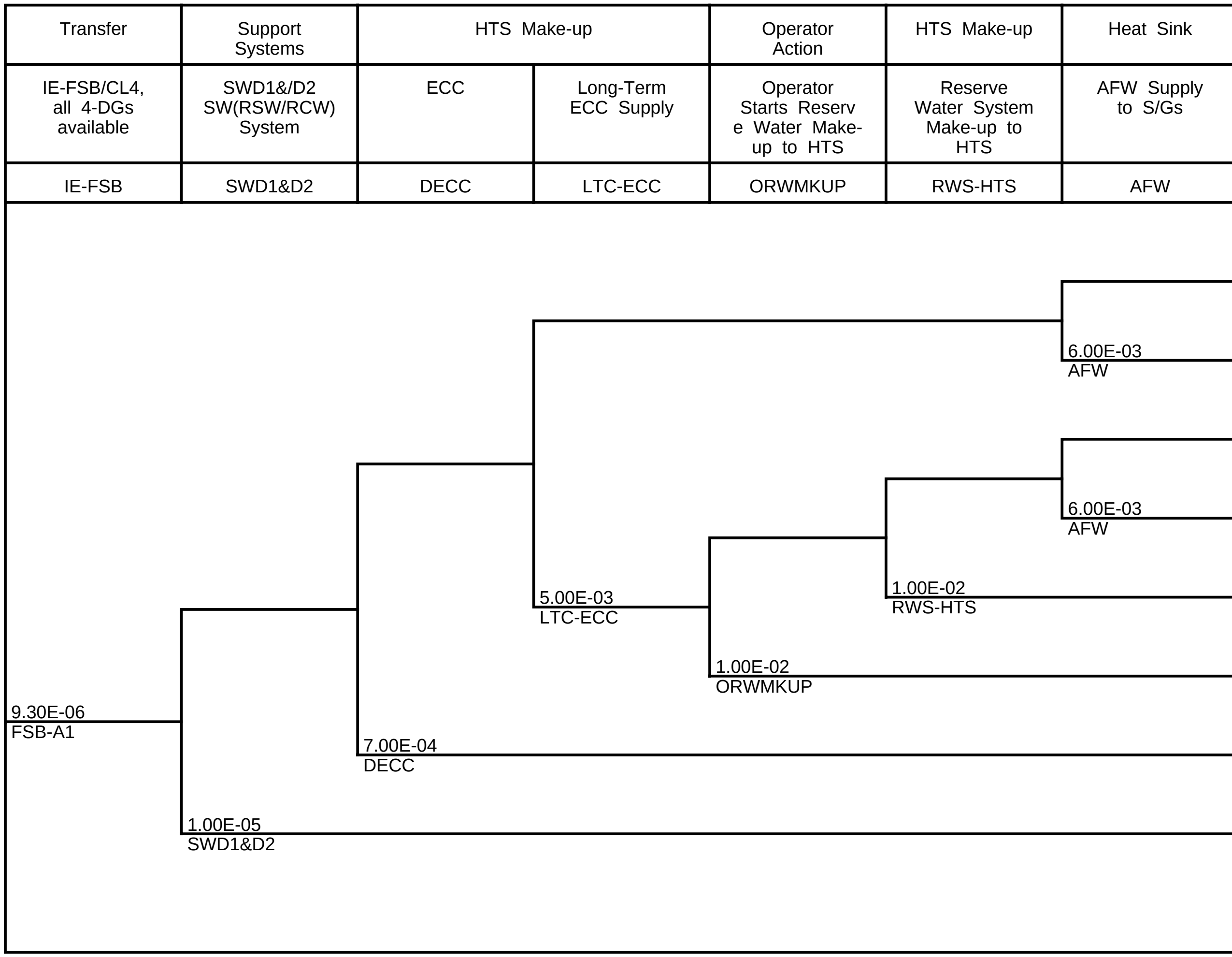


Fig.2 (Feeder Stagnation Break) Page(2) C:\CAFTA\TREE(ET)\FSB-A.TRE 16-06-03



SEQ.FREQ.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
9.19E-06	PDS8	FSB-A1	FSB-A11
5.55E-08	PDS2	FSB-A1/AFW	FSB-A12
4.53E-08	PDS8	FSB-A1/LTC-ECC	FSB-A13
2.73E-10	PDS2	FSB-A1/LTC-ECC/AFW	FSB-A14
4.60E-10	PDS2	FSB-A1/LTC-ECC/RWS-HTS	FSB-A15
4.65E-10	PDS2	FSB-A1/LTC-ECC/ORWMKUP	FSB-A16
6.51E-09	PDS2	FSB-A1/DECC	FSB-A17
9.30E-11	NDF	FSB-A1/SWD1&D2	

Fig.3 (Feeder Stagnation Break) Page(3) C:\CAFTA\TREE(ET)\FSB-A1.TRE 25-07-03

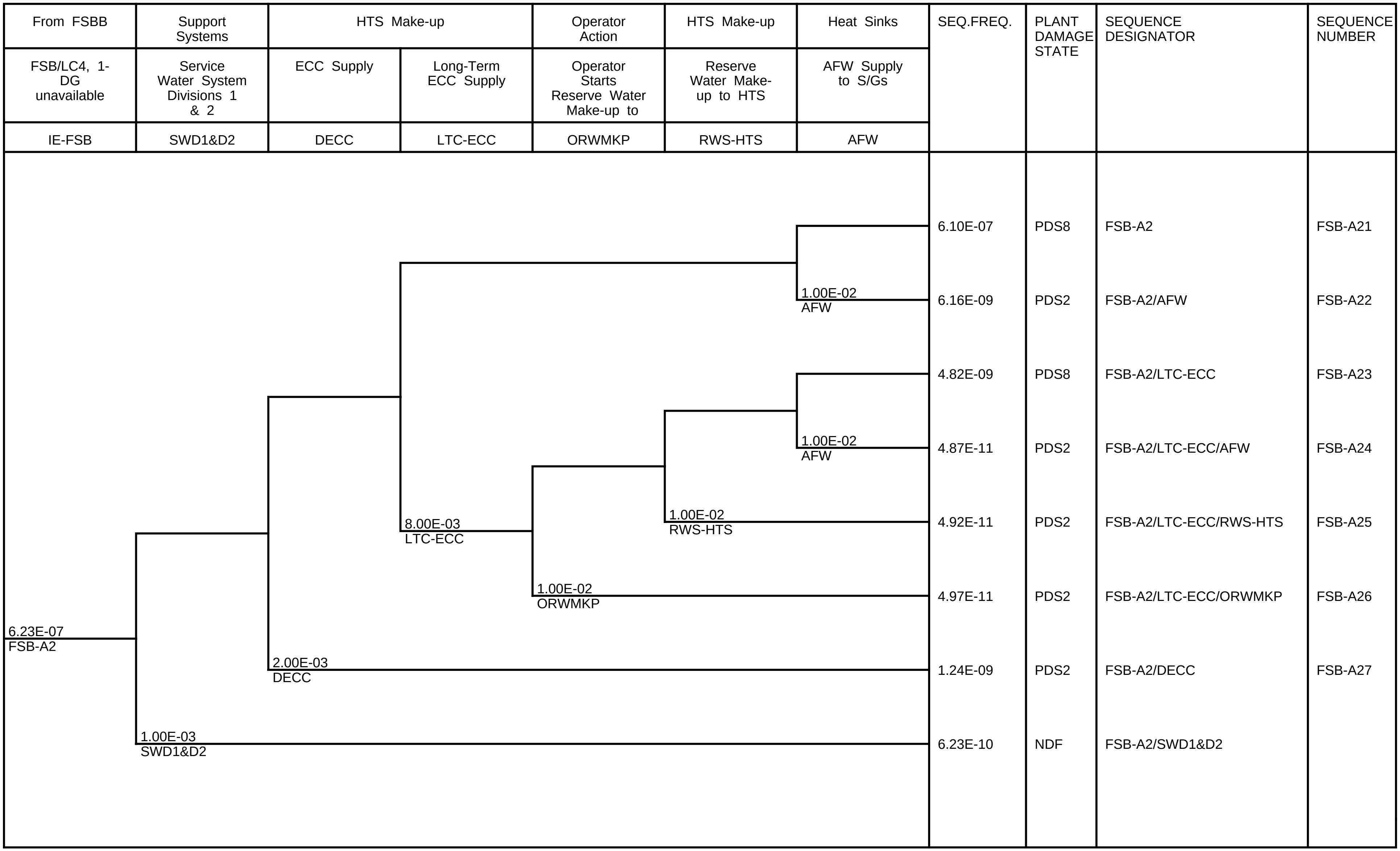
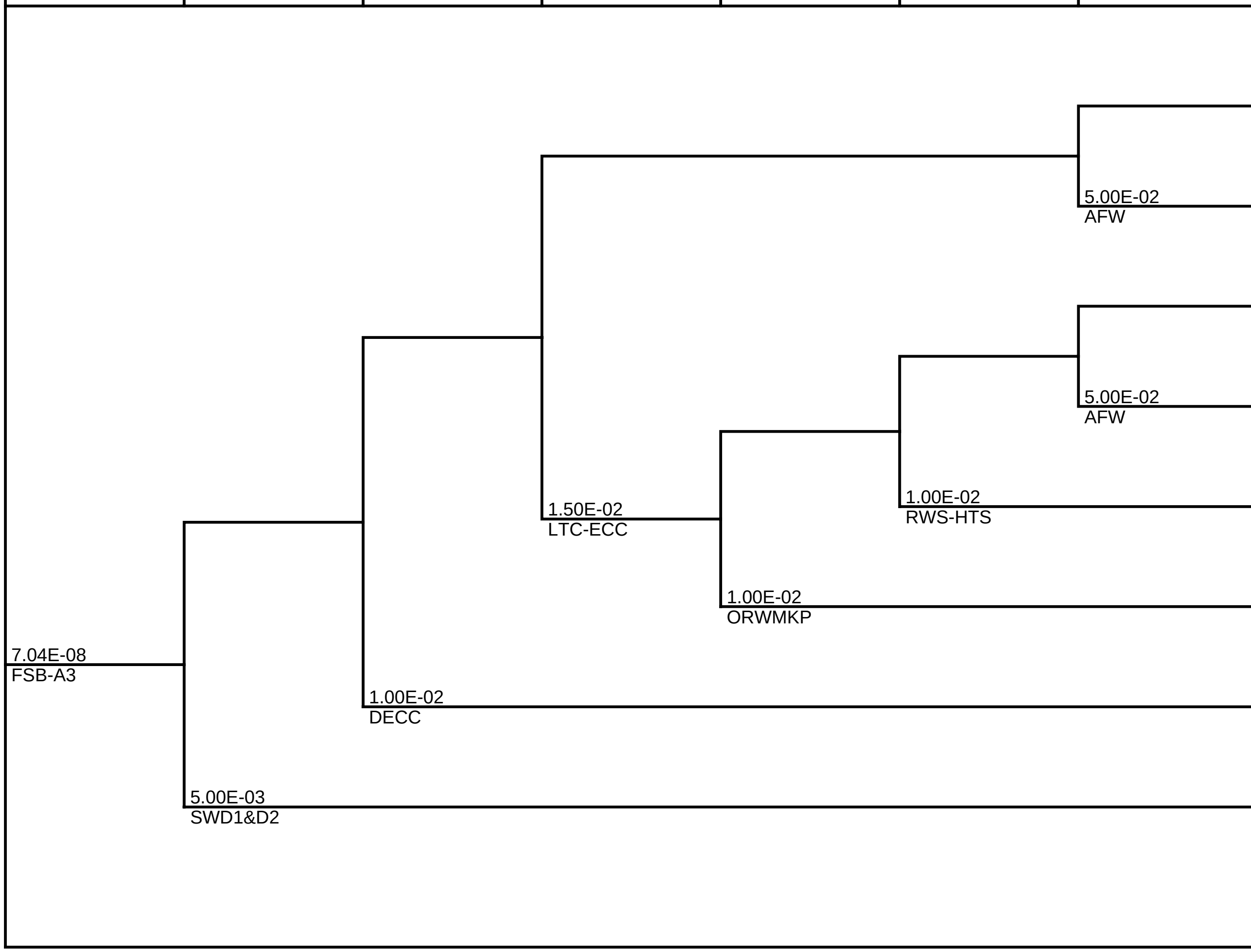


Fig.4 (Feeder Stagnation Break) Page(4) C:\CAFTA\TREE(ET)\FSB-A2.TRE 25-07-03

Transfer	Support Systems	HTS Make-up		Operator Action	HTS Make-up	Heat Sinks	SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
IE-FSB/CL4, 2-DGs Unavailable	SWD1&D2 SW(RSW/RCW) System	ECC Supply	Long-Term ECC Supply	Operator Starts Reserve Water Make-up	Reserve Water Make-up to HTS	AFW Supply to S/Gs				
IE-FSB	SWD1&D2	DECC	LTC-ECC	ORWMKP	RWS-HTS	AFW				



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
6.49E-08	PDS8	FSB-A3	FSB-A31
3.42E-09	PDS2	FSB-A3/AFW	FSB-A32
9.69E-10	PDS8	FSB-A3/LTC-ECC	FSB-A33
5.10E-11	PDS2	FSB-A3/LTC-ECC/AFW	FSB-A34
1.03E-11	PDS2	FSB-A3/LTC-ECC/RWS-HTS	FSB-A35
1.04E-11	PDS2	FSB-A3/LTC-ECC/ORWMKP	FSB-A36
7.00E-10	NDF	FSB-A3/DECC	
3.52E-10	NDF	FSB-A3/SWD1&D2	

Fig.5 (Feeder Stagnation Break) Page(5) C:\CAFTA\TREE(ET)\FSB-A3.TRE 25-07-03

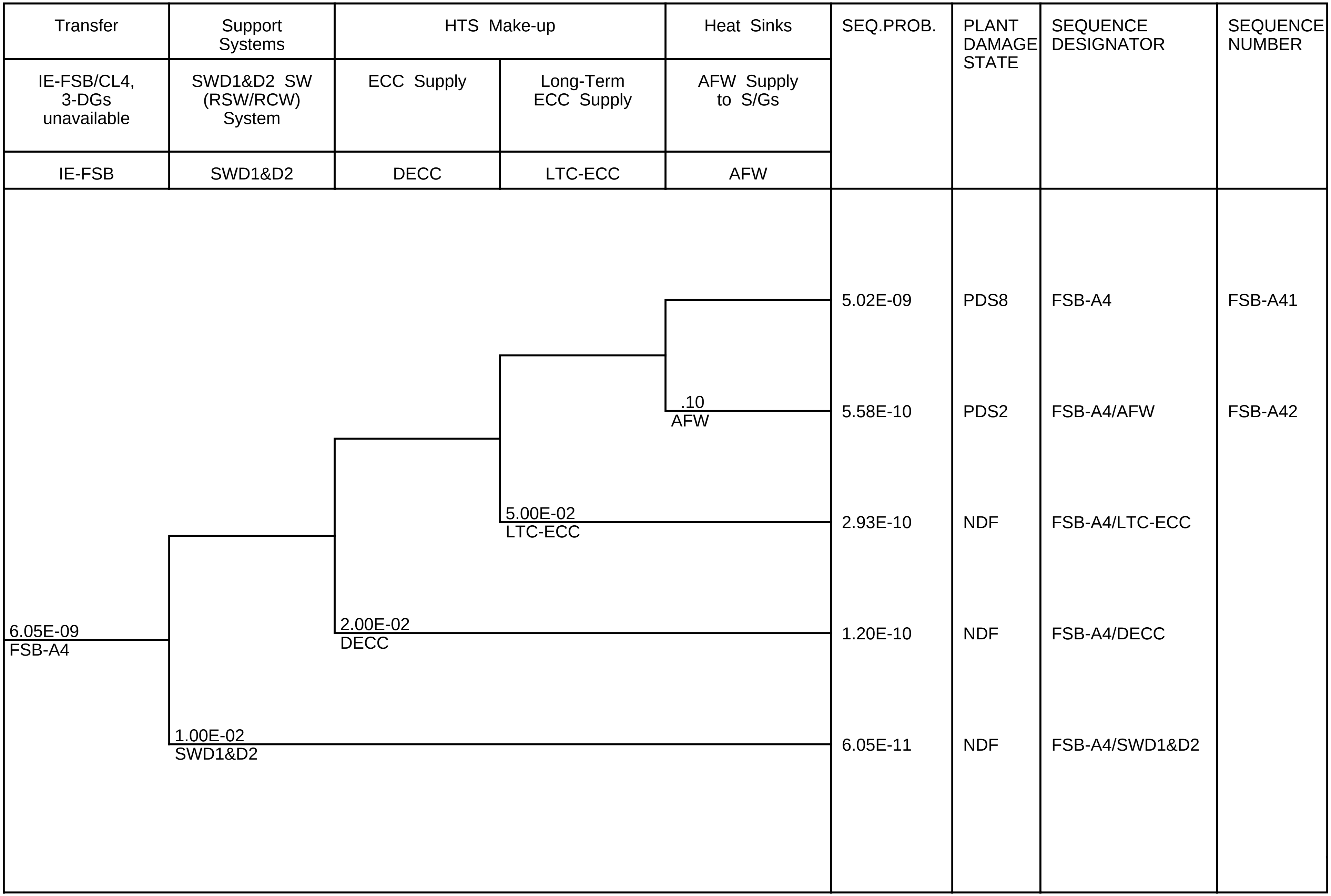
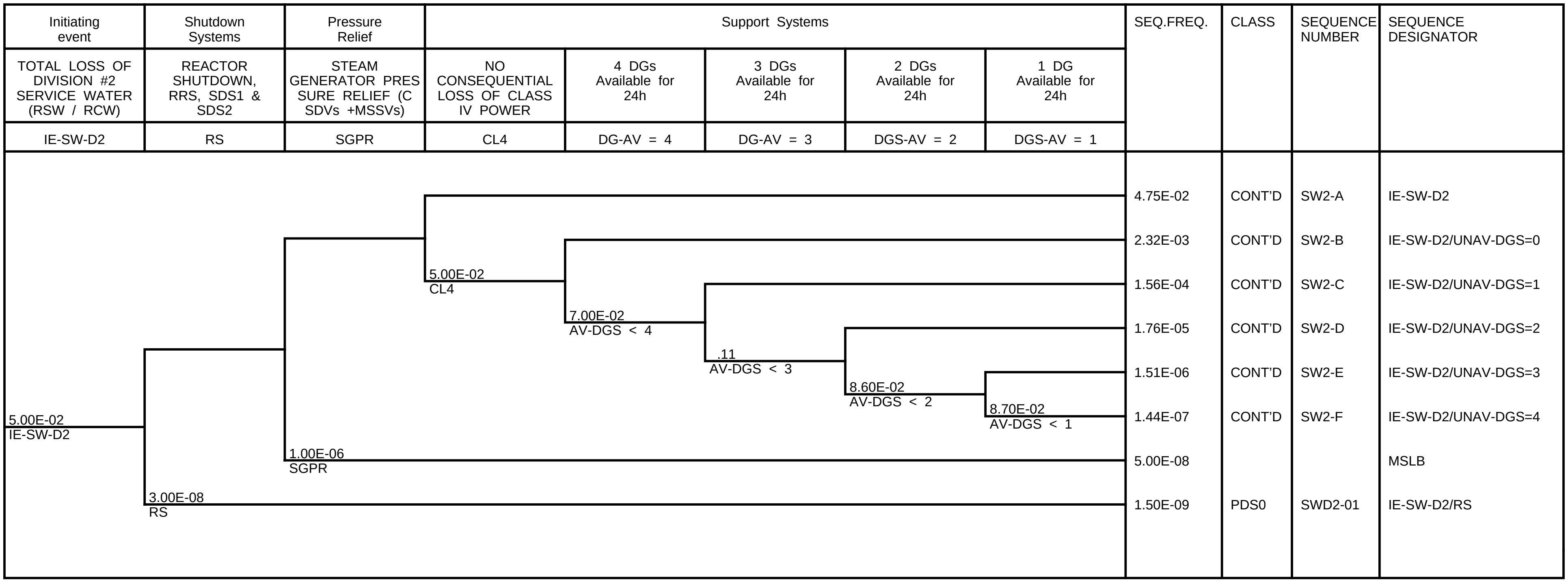


Fig.6 (Feeder Stagnation Break) Page(6) C:\CAFTA\TREE(ET)\FSB-A4.TRE 05-06-03

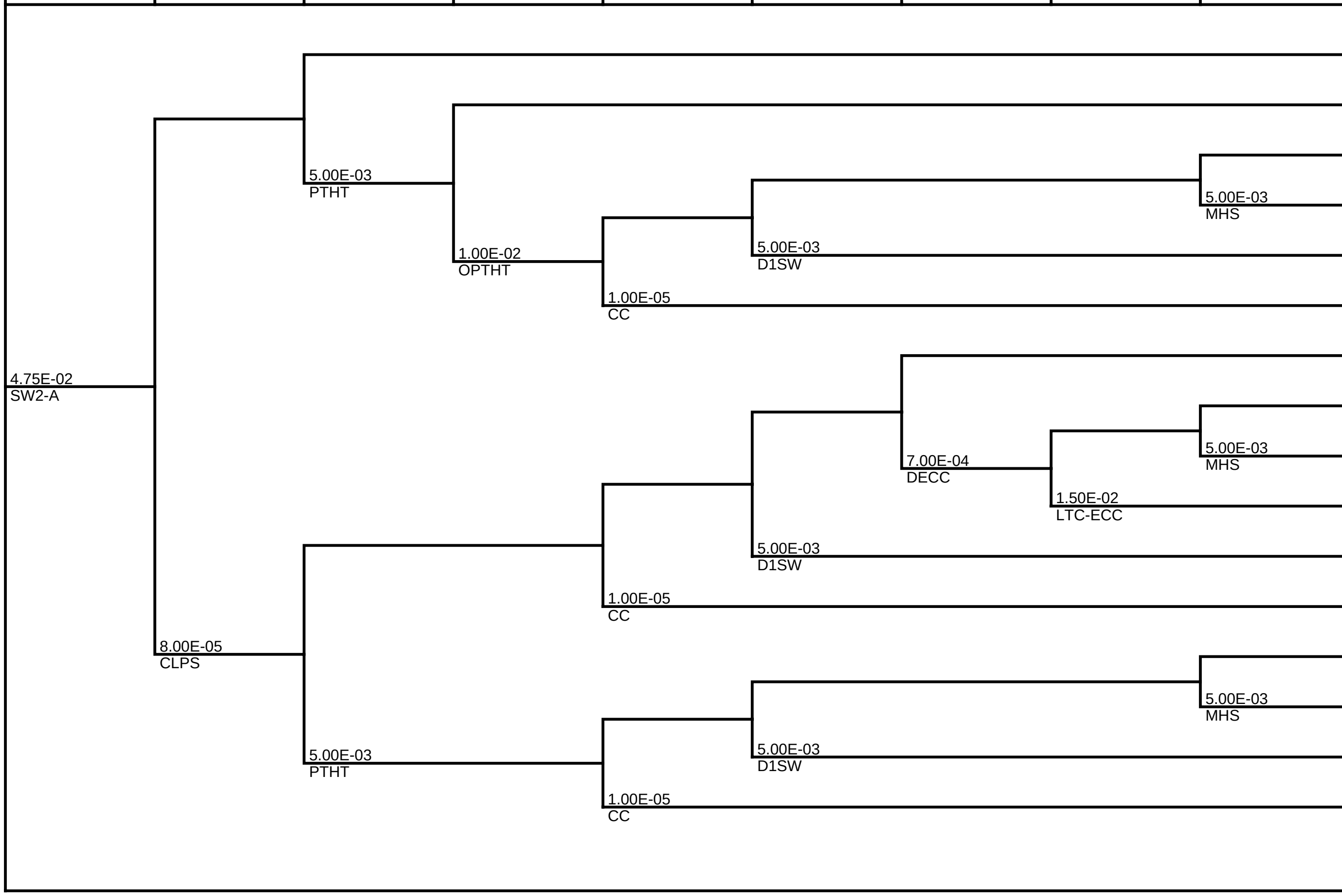
Appendix E

Event Tree for Total Loss of One Service Water Division



TOTAL LOSS OF DIV.#2 SERVICE WATER FOR ACR C:\CAFTA\TREE(ET)\IE-SW-D2.TRE 20-01-04

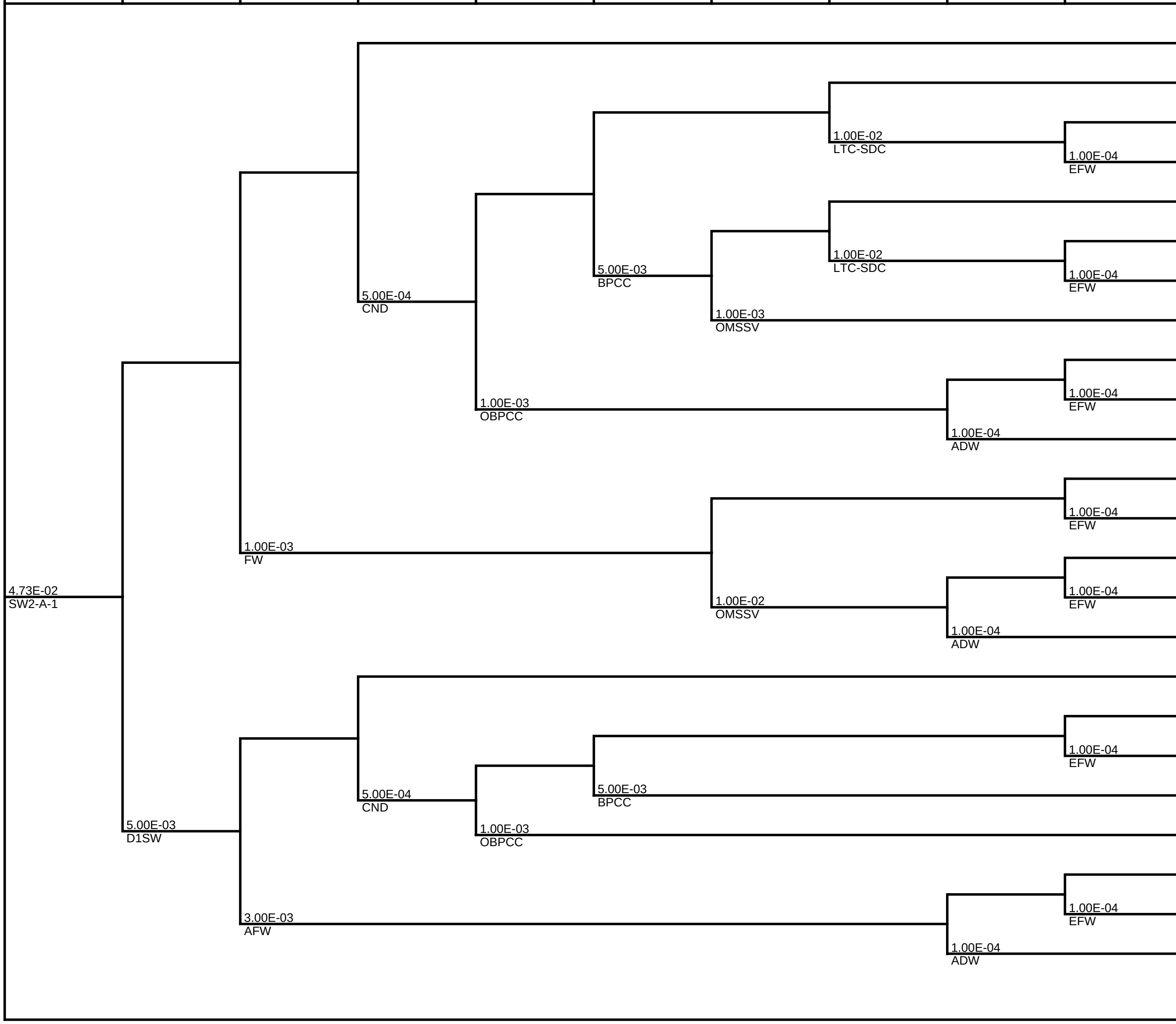
Initiating Event Subset	HT System Integrity		Support Actions	Support Systems		HTS Makeup		Ultimate Heat Sink
Total Loss of Div. 2 SW (RS & Cl4 Power Available)	HTS PUMP SEAL INTEGRITY MAINTAINED	HTS PUMPS TRIP ON HIGH UPPER BEARING TEMP.	Operator Trips HTS Pumps	CRASH COOLDOWN	DIVISION #1 SERVICE WATER	Emergency Core Cooling System	Long Term Cooling ECC System (includes reserve water	MODERATOR SYSTEM AS HEAT SINK
IE-SW2-A	CLPS	PTHT	OPTHT	CC	D1SW	DECC	LTC-ECC	MHS



SEQ.FREQ.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
4.73E-02	CONT'D	SW2-A	SW2-A-1
2.35E-04	CONT'D	SW2-A/PTHT	SW2-A-2
2.35E-06	PDS4	SW2-A/PTHT/OPTHT	SW2-A12
1.18E-08	PDS2	SW2-A/PTHT/OPTHT/MHS	SW2-A13
1.19E-08	PDS2	SW2-A/PTHT/OPTHT/D1SW	SW2-A14
2.37E-11	NDF	SW2-A/PTHT/OPTHT/CC	
3.76E-06	CONT'D	SW2-A/CLPS	SW2-A-3
2.58E-09	PDS4	SW2-A/CLPS/DECC	SW2-A20
1.30E-11	PDS2	SW2-A/CLPS/DECC/MHS	SW2-A21
3.95E-11	PDS1	SW2-A/CLPS/DECC/LTC-ECC	SW2-A22
1.89E-08	PDS2	SW2-A/CLPS/D1SW	SW2-A23
3.78E-11	NDF	SW2-A/CLPS/CC	
1.88E-08	PDS4	SW2-A/CLPS/PTHT	SW2-A24
9.45E-11	PDS2	SW2-A/CLPS/PTHT/MHS	SW2-A25
9.50E-11	PDS2	SW2-A/CLPS/PTHT/D1SW	SW2-A26
1.90E-13	NDF	SW2-A/CLPS/PTHT/CC	

TOTAL LOSS OF DIVISION 2 SERVICE WATER FOR ACR C:\CAFTA\TREE(ET)\IE-SW2-A.TRE 22-01-04

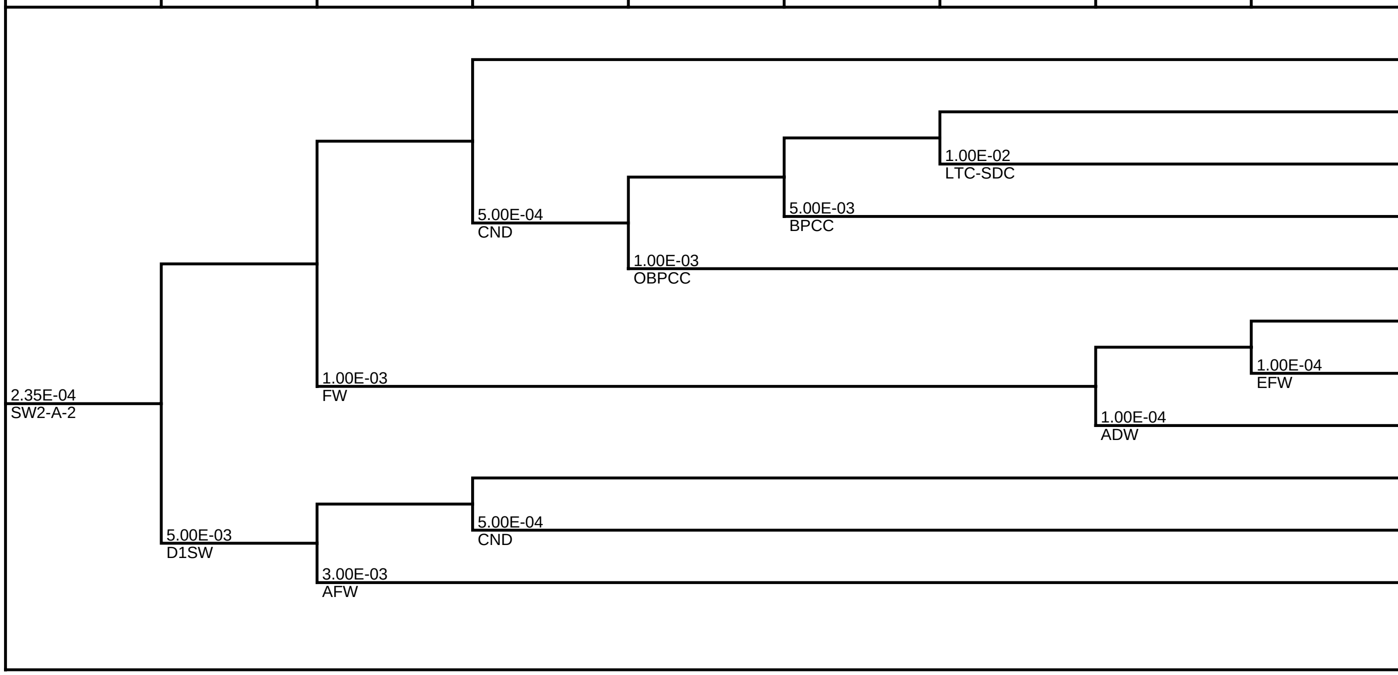
Initiating Event Subset	Support System	Heat Sinks		Operator Action	Heat Sink	Op. Action / Passive H.S.	Heat Sinks			SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
LOSS OF DIV.# 2 SW (HTS PUMPS INTEGRITY PRESERVED)	DIV. #1 SERVICE WATER	FEEDWATER SYSTEM AVAILABLE	CONDENSATE WATER SYSTEM (Main + Aux.)	OPERATOR STARTS BPCC AND SDC SYSTEM	BOILER PRESSURE CONTROL COOLDOWN	OP. OPENS MSSVs /STEAM DISCHARGED PASSIVELY BY MSSVs	LONG TERM (SHUTDOWN) COOLING SYSTEM	AUTO-DEPRESSURIZATION WATER SYSTEM	EMERGENCY FEEDWATER SUBSYSTEM				
IE-SW-D2	D1SW	FW	CND	OBPCC	BPCC	OMSSV / SGPR	LTC-SDC	ADW	EFW				



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
4.70E-02	S	SW2-A-1	
2.31E-05	S	SW2-A-1/CND	
2.34E-07	S	SW2-A-1/CND/LTC-SDC	
2.34E-11	PDS6	SW2-A-1/CND/LTC-SDC/EFW	SW2-A1
1.16E-07	S	SW2-A-1/CND/BPCC	
1.17E-09	S	SW2-A-1/CND/BPCC/LTC-SDC	
1.17E-13	PDS6	SW2-A-1/CND/BPCC/LTC-SDC/EFW	SW2-A2
1.17E-10	NDF	SW2-A-1/CND/BPCC/OMSSV	
2.35E-08	S	SW2-A-1/CND/OBPCC	
2.35E-12	PDS6	SW2-A-1/CND/OBPCC/EFW	SW2-A3
2.35E-12	PDS6	SW2-A-1/CND/OBPCC/ADW	SW2-A4
4.66E-05	S	SW2-A-1/FW	
4.66E-09	PDS6	SW2-A-1/FW/EFW	SW2-A5
4.71E-07	S	SW2-A-1/FW/OMSSV	
4.71E-11	PDS6	SW2-A-1/FW/OMSSV/EFW	SW2-A6
4.71E-11	PDS6	SW2-A-1/FW/OMSSV/ADW	SW2-A7
2.36E-04	S	SW2-A-1/D1SW	
1.17E-07	S	SW2-A-1/D1SW/CND	
1.17E-11	PDS6	SW2-A-1/D1SW/CND/EFW	SW2-A8
5.89E-10	NDF	SW2-A-1/D1SW/CND/BPCC	
1.18E-10	NDF	SW2-A-1/D1SW/CND/OBPCC	
7.09E-07	S	SW2-A-1/D1SW/AFW	
7.09E-11	PDS1	SW2-A-1/D1SW/AFW/EFW	SW2-A8
7.09E-11	PDS1	SW2-A-1/D1SW/AFW/ADW	SW2-A9

TOTAL LOSS OF DIVISION #2 SERVICE WATER C:\CAFTA\TREE(ET)\SW2-A-1.TRE 15-06-03

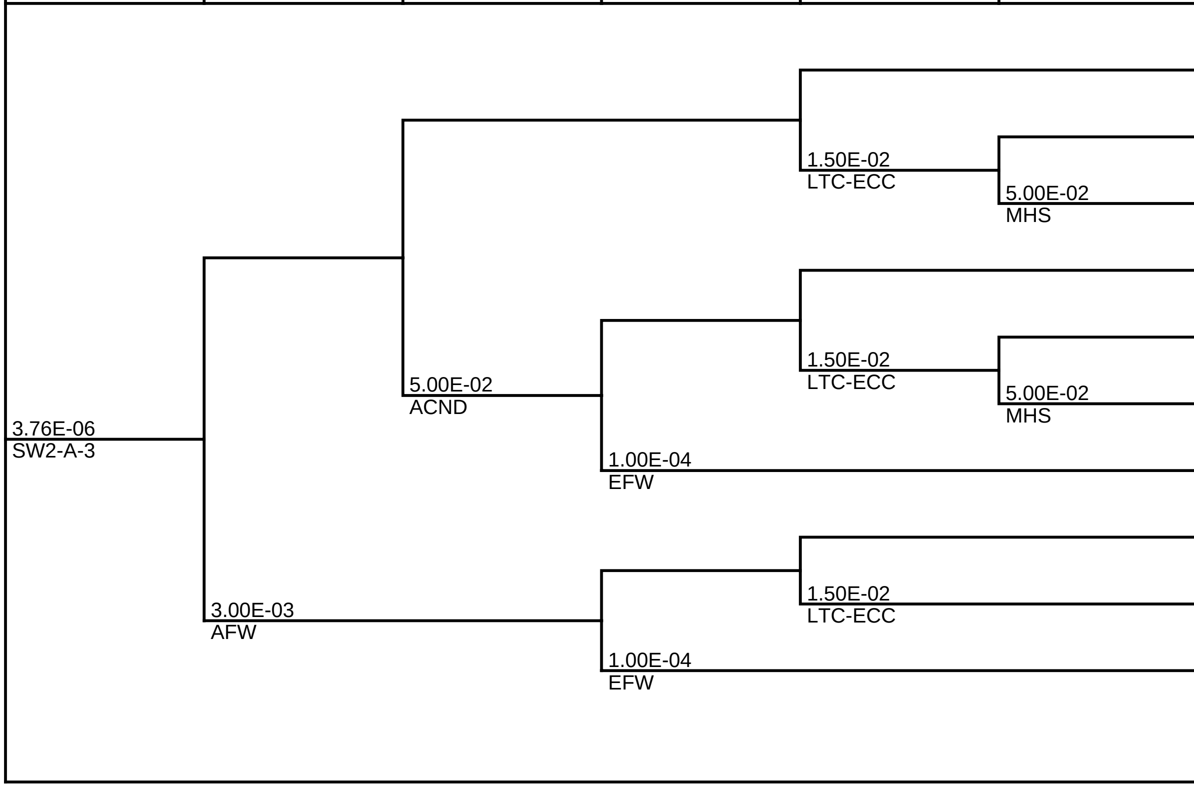
Initiating Event Subset	Support System	Heat Sinks		Operator Action	Heat Sink	Heat Sinks		
LOSS OF DIV.# 2 SW (HTS PUMPS TRIPPED BY OPERATOR)	DIV. #1 SERVICE WATER	FEEDWATER SYSTEM AVAILABLE	CONDENSATE WATER SYSTEM (Main + Aux.)	OPERATOR STARTS BPCC AND SDC SYSTEM	BOILER PRESSURE CONTROL COOLDOWN	LONG TERM (SHUTDOWN) COOLING SYSTEM	AUTO-DEPRESSURIZATION WATER SYSTEM	EMERGENCY FEEDWATER SUBSYSTEM
IE-SW-D2	D1SW	FW	CND	OBPCC	BPCC	LTC-SDC	ADW	EFW



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
2.33E-04	S	SW2-A-2	
1.15E-07	S	SW2-A-2/CND	
1.16E-09	NDF	SW2-A-2/CND/LTC-SDC	
5.83E-10	NDF	SW2-A-2/CND/BPCC	
1.17E-10	NDF	SW2-A-2/CND/OBPCC	
2.34E-07	S	SW2-A-2/FW	
2.34E-11	PDS6	SW2-A-2/FW/EFW	SW2-A10
2.34E-11	PDS6	SW2-A-2/FW/ADW	SW2-A11
1.17E-06	S	SW2-A-2/D1SW	
5.86E-10	NDF	SW2-A-2/D1SW/CND	
3.52E-09	NDF	SW2-A-2/D1SW/AFW	

TOTAL LOSS OF DIVISION #2 SERVICE WATER C:\CAFTA\TREE(ET)\SW2-A-2.TRE 15-06-03

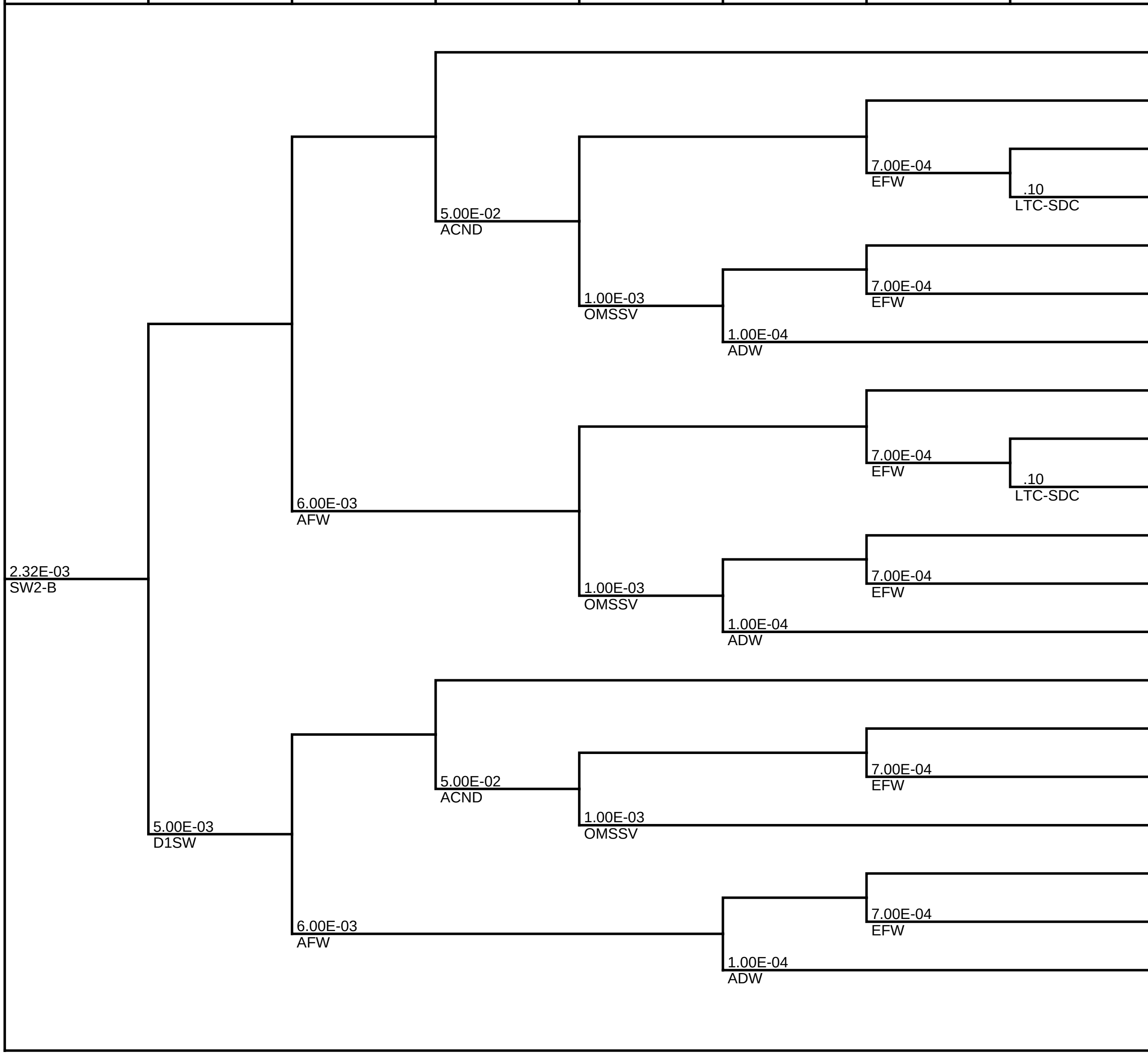
Initiating Event	Heat Sinks				
Loss of Div.2 SW -HT Pump Seals Damaged (CC Avail.)	AUX. FEEDWATER SYSTEM AVAILABLE	AUXILIARY CONDENSATE EXTRACTION	EMERGENCY FEEDWATER SUBSYSTEM	LONG TERM ECC SYSTEM	MODERATOR SYSTEM AS A HEAT SINK
IE-SW-D2	AFW	ACND	EFW	LTC-ECC	MHS



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
3.51E-06	S	SW2-A-3	
5.07E-08	PDS4	SW2-A-3/LTC-ECC	SW2-A15
2.67E-09	PDS2	SW2-A-3/LTC-ECC/MHS	SW2-A16
1.85E-07	S	SW2-A-3/ACND	
2.67E-09	PDS4	SW2-A-3/ACND/LTC-ECC	SW2-A17
1.41E-10	PDS2	SW2-A-3/ACND/LTC-ECC/MHS	SW2-A18
1.87E-11	NDF	SW2-A-3/ACND/EFW	
1.11E-08	S	SW2-A-3/AFW	
1.69E-10	NDF	SW2-A-3/AFW/LTC-ECC	
1.13E-12	NDF	SW2-A-3/AFW/EFW	

TOTAL LOSS OF DIVISON #2 SERVICE WATER C:\CAFTA\TREE(ET)\SW2-A-3.TRE 22-01-04

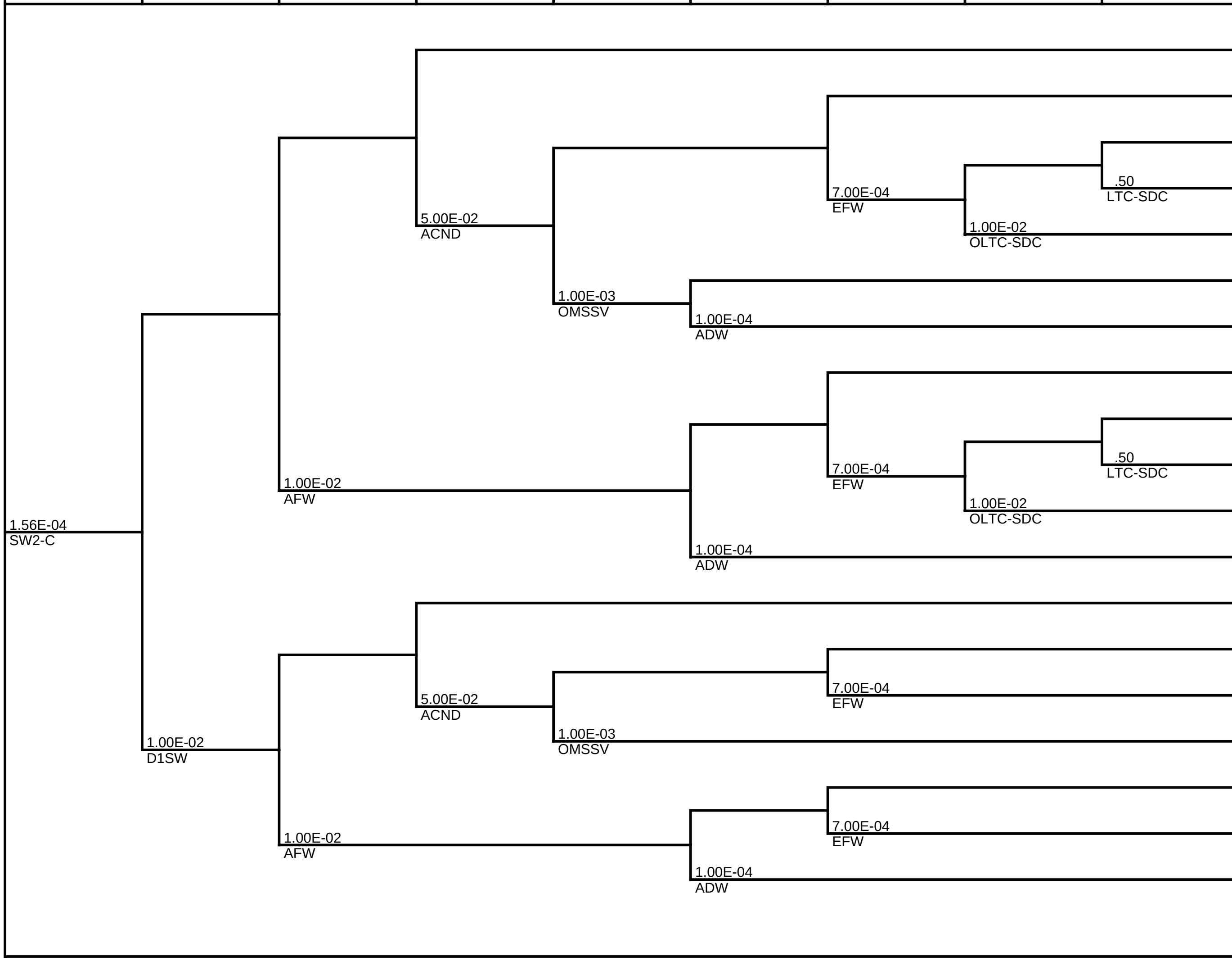
Initiating Event	Support System	Heat Sinks		Operator Action	Heat Sinks		
LOSS OF DIVISION#2 SW WITH 4 CL3 D Gs AVAILABLE	DIVISION#1 SERVICE WATER	AUXILIARY FEEDWATER SYSTEM	Auxiliary Condensate Extraction	OPERATOR OPENS MSSVs & STARTS SD COOLING	AUTO DE-PRESSURIZATIO N WATER SYSTEM	Emergency Feedwater Sub-System	LONG TERM (SHUTDOWN) COOLING SYSTEM
IE-SW-D2	D1SW	AFW	ACND	OMSSV	ADW	EFW	LTC-SDC



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
2.18E-03	S	SW2-B	
1.15E-04	S	SW2-B/ACND	
7.22E-08	S	SW2-B/ACND/EFW	
8.02E-09	PDS6	SW2-B/ACND/EFW/LTC-SDC	SW2-B1
1.15E-07	S	SW2-B/ACND/OMSSV	
8.03E-11	NDF	SW2-B/ACND/OMSSV/EFW	
1.15E-11	NDF	SW2-B/ACND/OMSSV/ADW	
1.38E-05	S	SW2-B/AFW	
8.72E-09	S	SW2-B/AFW/EFW	
9.69E-10	PDS6	SW2-B/AFW/EFW/LTC-SDC	SW2-B2
1.38E-08	S	SW2-B/AFW/OMSSV	
9.69E-12	NDF	SW2-B/AFW/OMSSV/EFW	
1.39E-12	PDS6	SW2-B/AFW/OMSSV/ADW	SW2-B3
1.10E-05	S	SW2-B/D1SW	
5.76E-07	S	SW2-B/D1SW/ACND	
4.03E-10	PDS1	SW2-B/D1SW/ACND/EFW	SW2-B4
5.77E-10	NDF	SW2-B/D1SW/ACND/OMSSV	
6.95E-08	S	SW2-B/D1SW/AFW	
4.87E-11	PDS1	SW2-B/D1SW/AFW/EFW	SW2-B5
6.96E-12	NDF	SW2-B/D1SW/AFW/ADW	

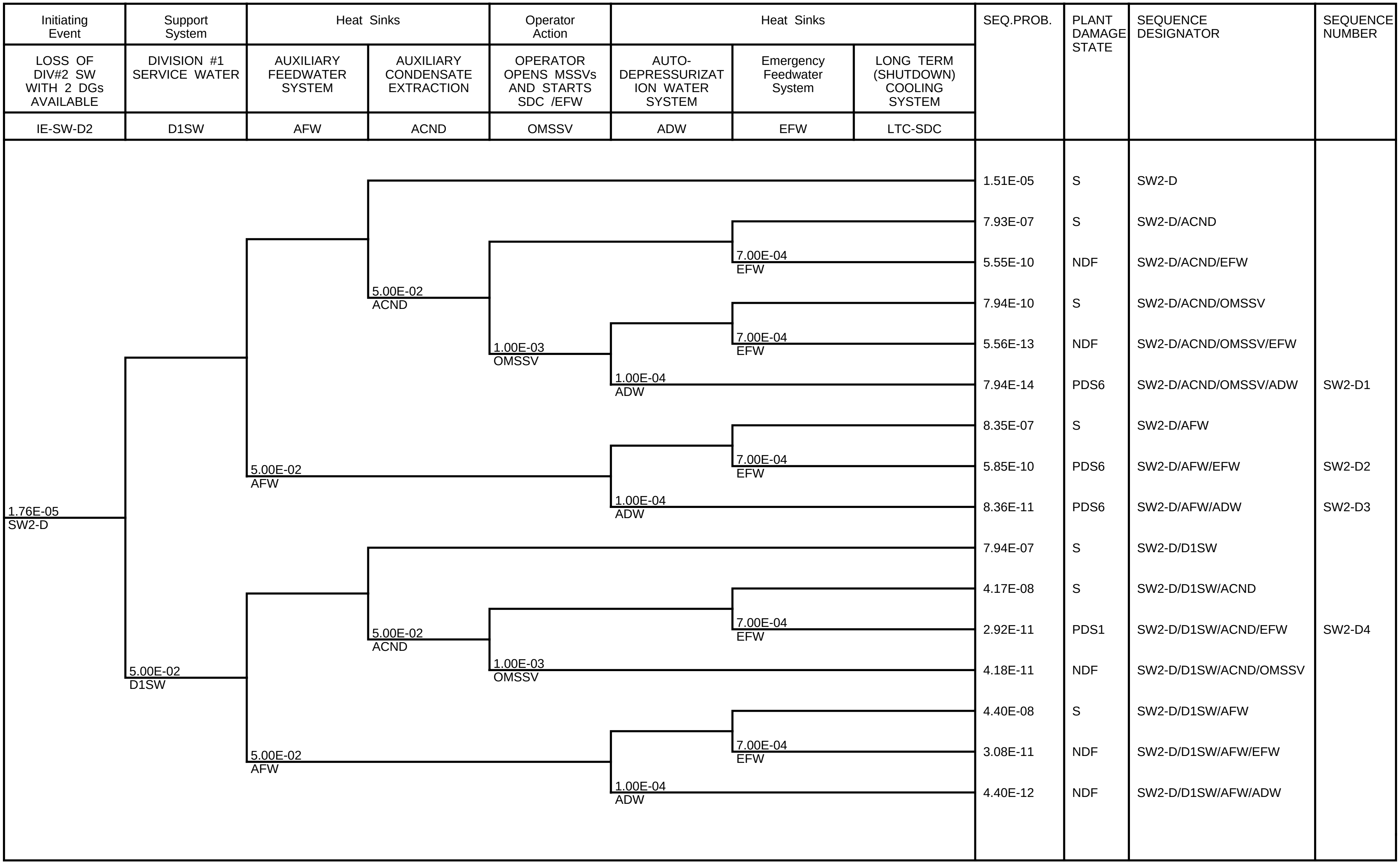
TOTAL LOSS OF DIVISION 2 SERVICE WATER FOR ACR   C:\CAFTA\TREE(ET)\IE-SW2-B.TRE   16-06-03

Initiating Event Subset	Supprt System	Heat Sinks		Operator Action	Heat Sinks		Operator Action	Heat Sinks
LOSS OF DIV#2 SW WITH 3 DGs AVAILABLE	DIVISION #1 SERVICE WATER	AUXILIARY FEEDWATER SYSTEM	AUXILIARY CONDENSATE EXTRACTION	OP. OPENS MSSVs AND STARTS SDC / EFW	AUTO-DEPRESSURIZATION WATER SYSTEM	Emergency Feedwater System	OPERATOR INTIATES SDC NORMAL MODE	SHUTDOWN COOLING SYSTEM
IE-SW-D2	D1SW	AFW	ACND	OMSSV	ADW	EFW	OLTC-SDC	LTC-SDC



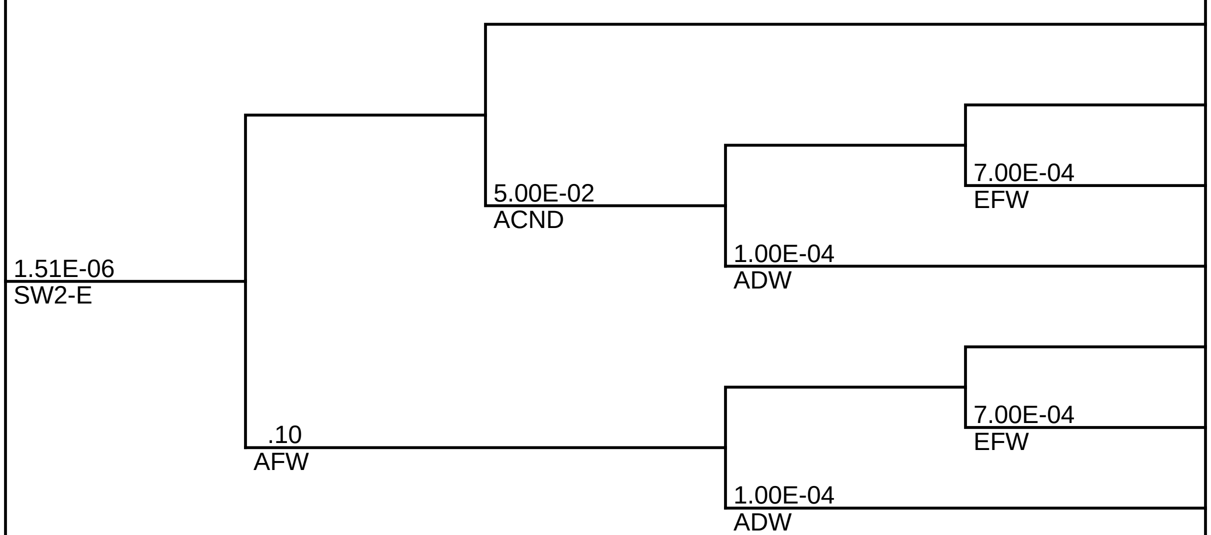
SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
1.45E-04	S	SW2-C	
7.63E-06	S	SW2-C/ACND	
2.65E-09	S	SW2-C/ACND/EFW	
2.65E-09	PDS6	SW2-C/ACND/EFW/LTC-SDC	SW2-C1
5.35E-11	PDS6	SW2-C/ACND/EFW/OLTC-SDC	SW2-C2
7.64E-09	S	SW2-C/ACND/OMSSV	
7.64E-13	PDS6	SW2-C/ACND/OMSSV/ADW	SW2-C3
1.54E-06	S	SW2-C/AFW	
5.35E-10	S	SW2-C/AFW/EFW	
5.35E-10	PDS6	SW2-C/AFW/EFW/LTC-SDC	SW2-C4
1.08E-11	PDS6	SW2-C/AFW/EFW/OLTC-SDC	SW2-C5
1.54E-10	PDS6	SW2-C/AFW/ADW	SW2-C6
1.47E-06	S	SW2-C/D1SW	
7.71E-08	S	SW2-C/D1SW/ACND	
5.40E-11	PDS1	SW2-C/D1SW/ACND/EFW	SW2-C7
7.72E-11	NDF	SW2-C/D1SW/ACND/OMSSV	
1.56E-08	S	SW2-C/D1SW/AFW	
1.09E-11	PDS1	SW2-C/D1SW/AFW/EFW	SW2-C8
1.56E-12	PDS1	SW2-C/D1SW/AFW/ADW	SW2-C9

TOTAL LOSS OF DIVISION #2 SERVICE WATER C:\CAFTA\TREE(ET)\IE-SW2-C.TRE 16-06-03



TOTAL LOSS OF DIVISION #2 SERVICE WATER C:\CAFTA\TREE(ET)\IE-SW2-D.TRE 16-06-03

Initiating Event	Heat Sinks				SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
LOSS OF DIV#2 SW WITH 1 DG AVAILABLE	AUXILIARY FEEDWATER SYSTEM	AUXILIARY CONDENSATE EXTRACTION	AUTO-DEPRESSURIZATION WATER SYSTEM	Emergency Feedwater Sub-System				
IE-SW-D2	AFW	ACND	ADW	EFW				



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
1.29E-06	S	SW2-E	
6.79E-08	S	SW2-E/ACND	
4.76E-11	NDF	SW2-E/ACND/EFW	
6.79E-12	NDF	SW2-E/ACND/ADW	
1.51E-07	S	SW2-E/AFW	
1.06E-10	NDF	SW2-E/AFW/EFW	
1.51E-11	NDF	SW2-E/AFW/ADW	

TOTAL LOSS OF DIVISION #2 SERVICE WATER C:\CAFTA\TREE(ET)\IE-SW2-E.TRE 16-06-03

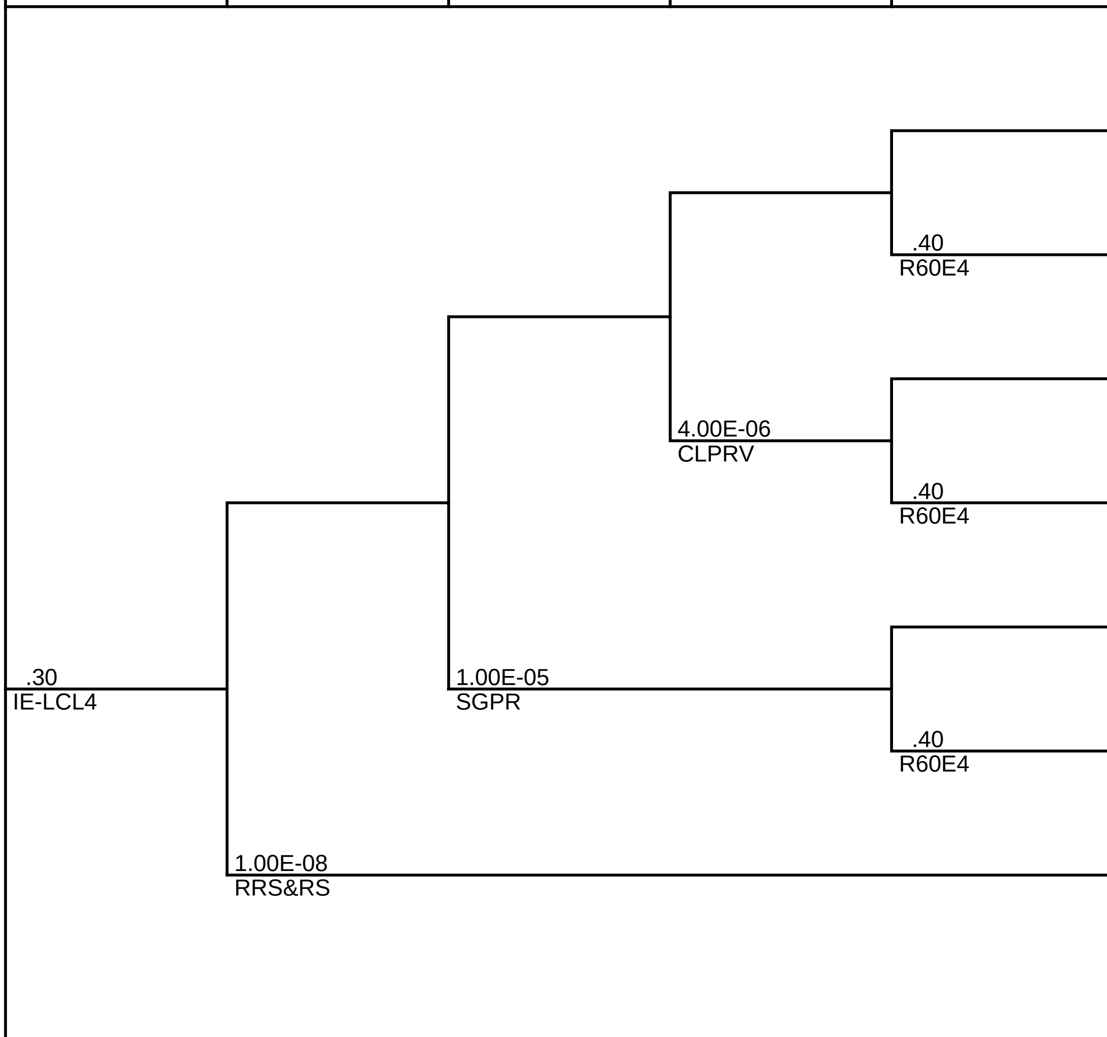
Initiating Event	Heat Sinks			SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
LOSS OF DIV#2 SW (with Cl3 DGs unavailable)	AUXILIARY FEEDWATER SYSTEM	AUTO-DEPRESSURIZATION WATER SYSTEM	Emergency Feedwater Subsystem				
IE-SW-D2	AFW	ADW	EFW				
				1.44E-07	S	SW2-F	
1.44E-07 SW2-F			7.00E-04 EFW	1.01E-10	PDS1	SW2-F/EFW	SW2-F1
		1.00E-04 ADW		1.44E-11	PDS2	SW2-F/ADW	SW2-F2

TOTAL LOSS OF DIVISION #2 SERVICE WATER C:\CAFTA\TREE(ET)\IE-SW2-F.TRE 16-06-03

Appendix F

Event Tree for Loss of Class IV Power to One Unit

Initiating Event	Reactor Trip	Heat Sink	HTS Integrity	Operator Action	SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE DESCRIPTION
Total Loss of Class IV Power Supply	Reactor Shutdown	Steam Generator Pressure Relief	No Conseq. LOCA (HTS invent.loss via LRV)	Switchyard Restored in 60 minutes				
IE-LCL4	RS	SGPR	CLPRV	R60E4				



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE DESCRIPTION
1.80E-01	CONTD	IE-LCL4	LCL4-A
1.20E-01	CONTD	IE-LCL4/R60E4	LCL4-B
7.20E-07	CONTD	IE-LCL4/CLPRV	LCL4-C
4.80E-07	CONTD	IE-LCL4/CLPRV/R60E4	LCL4-D
1.80E-06	MSLB	IE-LCL4/SGPR	LCL4-1
1.20E-06	MSLB/CL4	IE-LCL4/SGPR/R60E4	LCL4-2
3.00E-09	PDS0	IE-LCL4/RRS&RS	LCL4-3

Fig.1 (Total Loss of CLIV supply) Page(1) C:\CAFTA\TREE(ET)\IELCL401.TRE 13-06-03

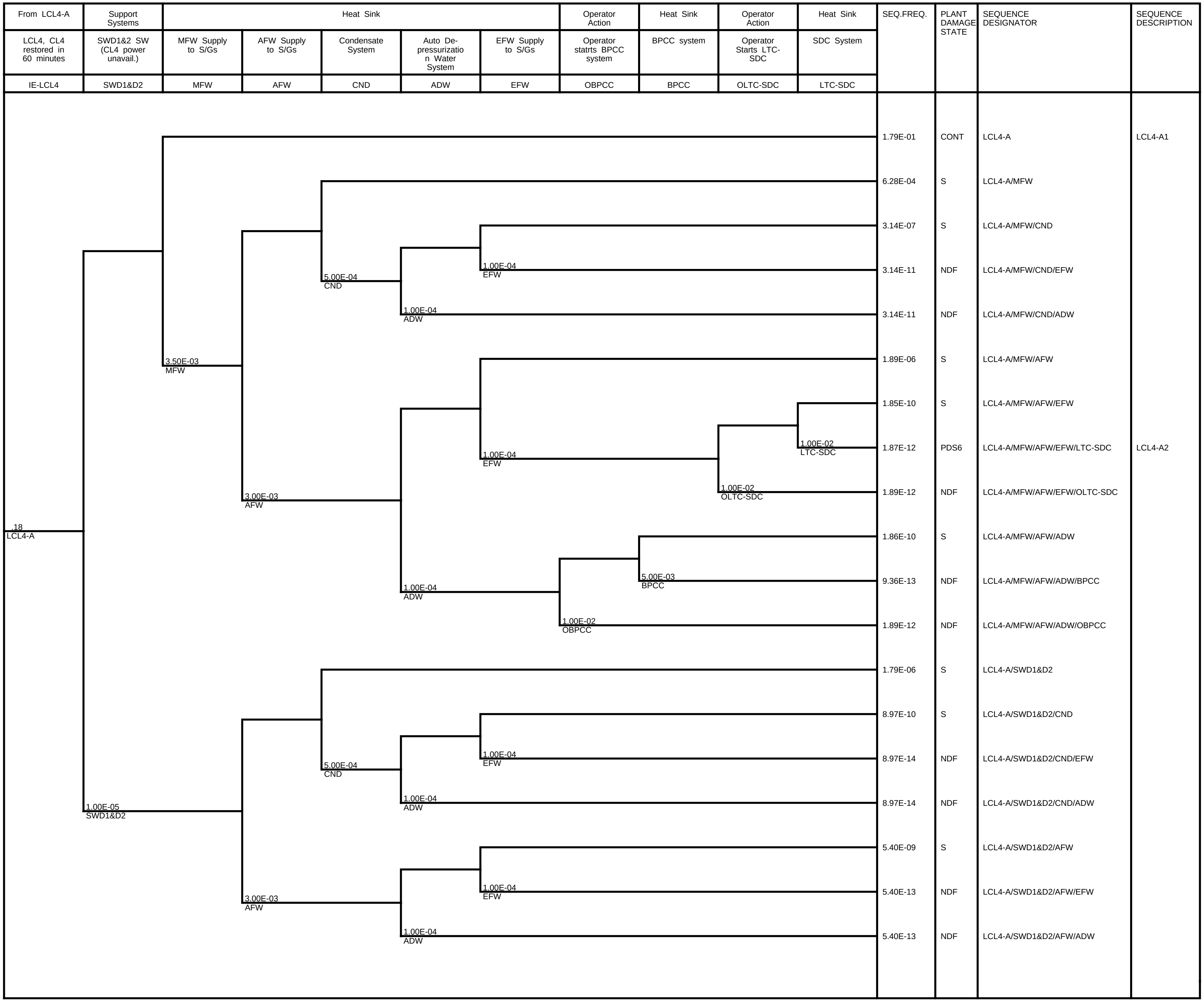
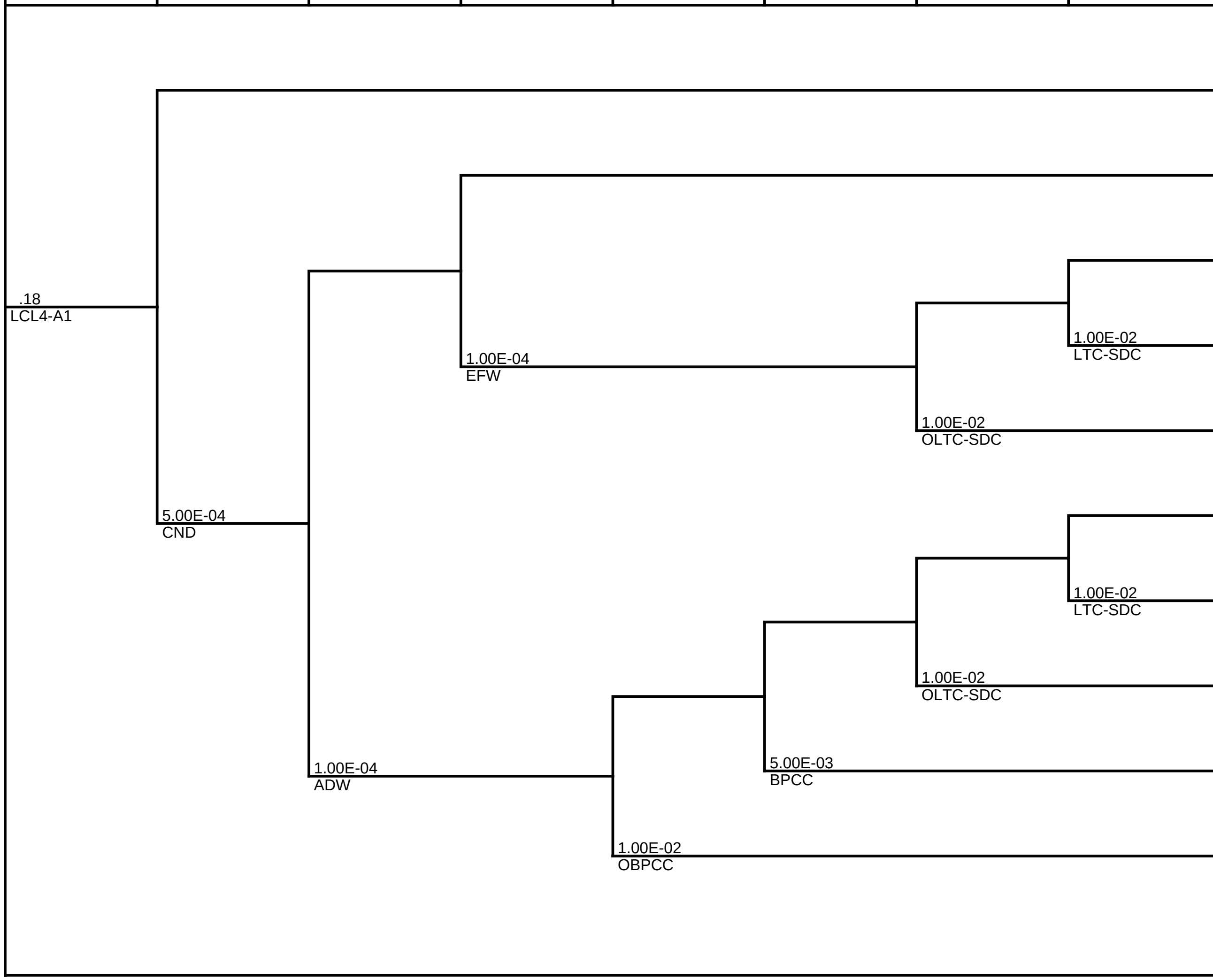


Fig.2 (Total Loss of CL4 supply) Page(2) C:\CAFTA\TREE(ET)\IELCL402.TRE 28-01-04

From LCL4-A1	Heat Sink			Operator Action	Heat Sink	Operator Action	Heat Sink
LCL4, Class4 restored in 60 minutes	Main Condensate System Fails	Auto De-pressurization Water System	EFW Supply to S/Gs	Operator Starts BPCC System	Boiler Pres. Control Cooling System	Operator Starts LTC-SDC	Long Term Cooling System (SDC Mode)
IE-LCL4	CND	ADW	EFW	OBPCC	BPCC	OLTC-SDC	LTC-SDC



SEQ.FREQ.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE DESCRIPTION
1.80E-01	S	LCL4-A1	
9.00E-05	S	LCL4-A1/CND	
8.82E-09	S	LCL4-A1/CND/EFW	
8.91E-11	PDS6	LCL4-A1/CND/EFW/LTC-SDC	LCL4-A11
9.00E-11	PDS6	LCL4-A1/CND/EFW/OLTC-SDC	LCL4-A12
8.69E-09	S	LCL4-A1/CND/ADW	
8.78E-11	PDS6	LCL4-A1/CND/ADW/LTC-SDC	LCL4-A13
8.87E-11	NDF	LCL4-A1/CND/ADW/OLTC-SDC	
4.46E-11	NDF	LCL4-A1/CND/ADW/BPCC	
9.00E-11	NDF	LCL4-A1/CND/ADW/OBPCC	

Fig.3 (Total Loss of CL4 supply) Page(3) C:\CAFTA\TREE(ET)\IELCL403.TRE 28-01-04

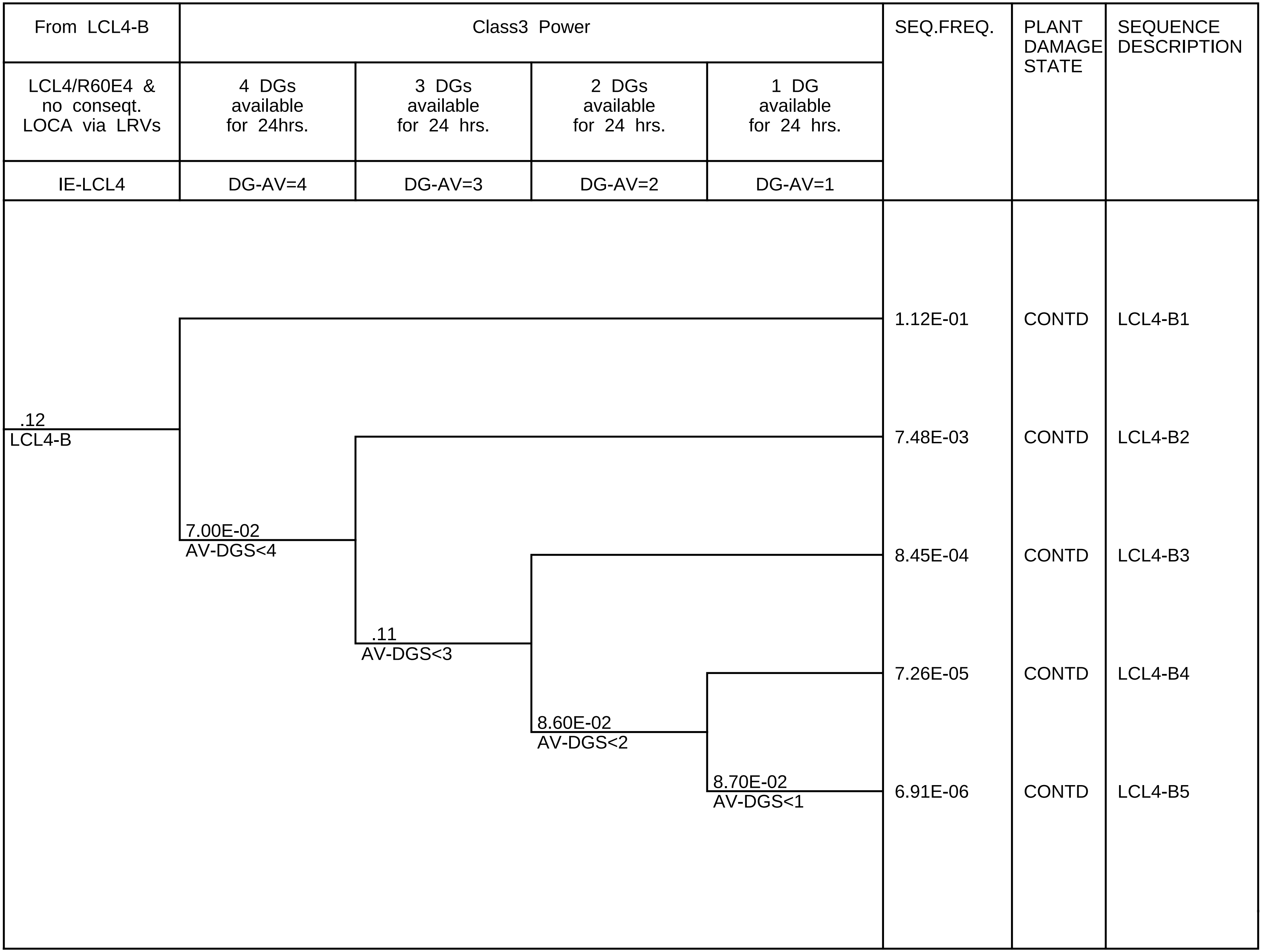
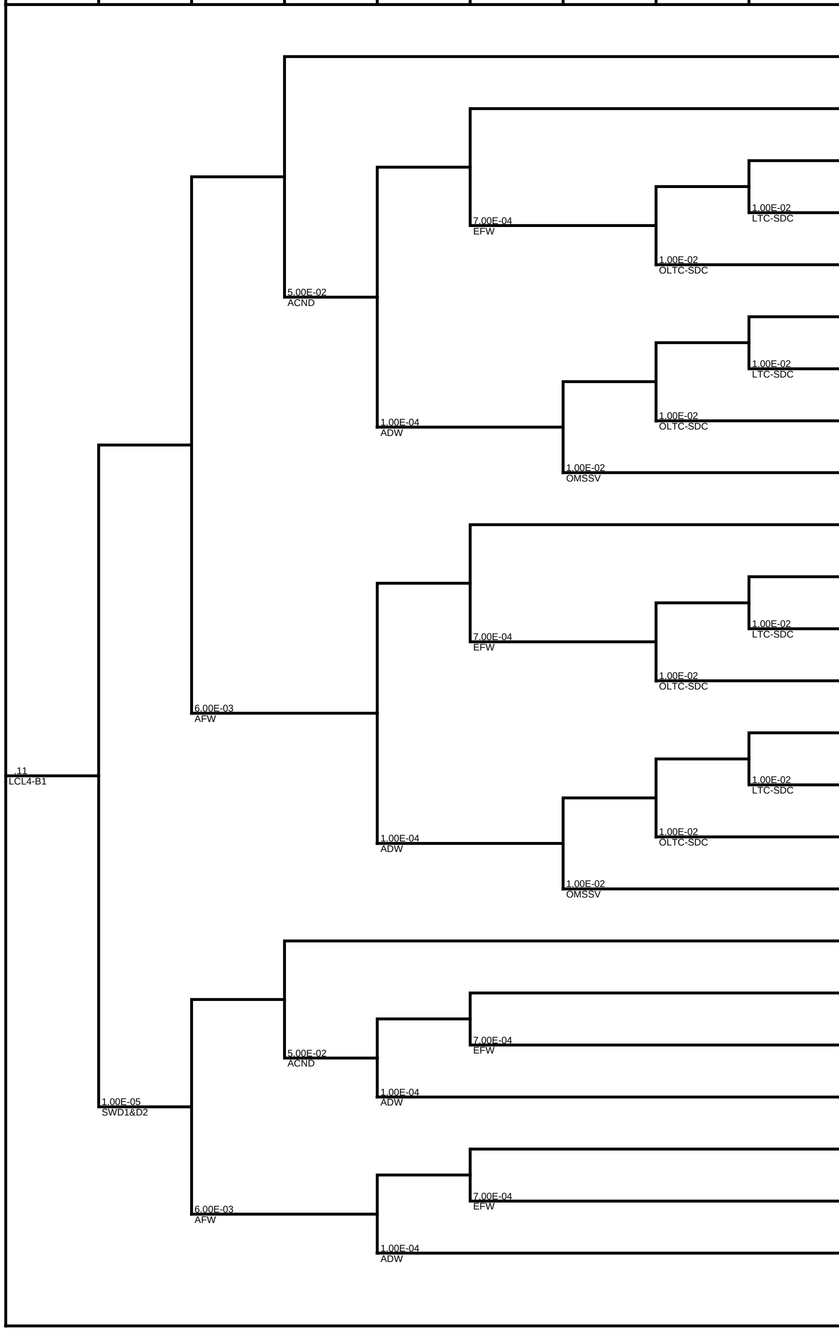


Fig.4 (Total Loss of CL4 supply) Page(4) C:\CAFTA\TREE(ET)\IELCL404.TRE 13-06-03

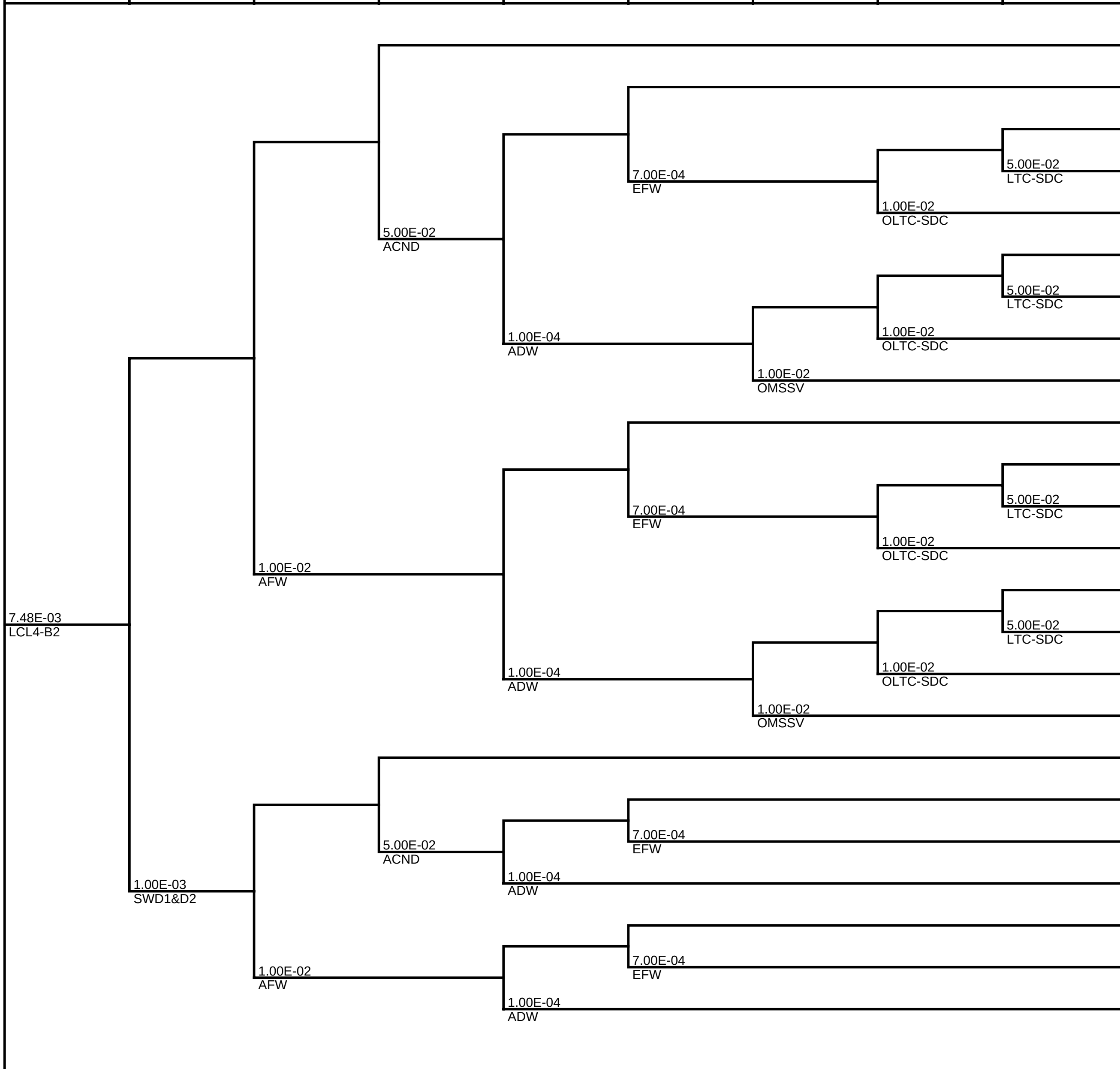
From LCL4-B1	Support Systems	Heat Sink				Operator Action		Heat Sink
LCL4/R60E4 - at least 4-DGs available	SWD1&D2 SW (CL4 power unavail.)	AFW Supply to SGs	Auxiliary Condensate Pump	auto-depressurization water system	EFW Supply to S/Gs	Operator Opens MSSVs to De-pressurize SGs	Operator Starts LTC-SDC	SDC System
IE-LCL4	SWD1&D2	AFW	ACND	ADW	EFW	OMSSV	OLTC-SDC	LTC-SDC



SEQ.FREQ.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE DESCRIPTION
1.06E-01	S	LCL4-B1	
5.56E-03	S	LCL4-B1/ACND	
3.82E-06	S	LCL4-B1/ACND/EFW	
3.86E-08	PDS6	LCL4-B1/ACND/EFW/LTC-SDC	LCL4-B11
3.90E-08	PDS6	LCL4-B1/ACND/EFW/OLTC-SDC	LCL4-B12
5.40E-07	S	LCL4-B1/ACND/ADW	
5.46E-09	PDS6	LCL4-B1/ACND/ADW/LTC-SDC	LCL4-B13
5.51E-09	PDS6	LCL4-B1/ACND/ADW/OLTC-SDC	LCL4-B14
5.57E-09	PDS6	LCL4-B1/ACND/ADW/OMSSV	LCL4-B15
6.71E-04	S	LCL4-B1/AFW	
4.61E-07	S	LCL4-B1/AFW/EFW	
4.66E-09	PDS6	LCL4-B1/AFW/EFW/LTC-SDC	LCL4-B16
4.70E-09	PDS6	LCL4-B1/AFW/EFW/OLTC-SDC	LCL4-B17
6.52E-08	S	LCL4-B1/AFW/ADW	
6.59E-10	PDS6	LCL4-B1/AFW/ADW/LTC-SDC	LCL4-B18
6.65E-10	NDF	LCL4-B1/AFW/ADW/OLTC-SDC	
6.72E-10	NDF	LCL4-B1/AFW/ADW/OMSSV	
1.06E-06	S	LCL4-B1/SWD1&D2	
5.56E-08	S	LCL4-B1/SWD1&D2/ACND	
3.90E-11	NDF	LCL4-B1/SWD1&D2/ACND/EFW	
5.57E-12	NDF	LCL4-B1/SWD1&D2/ACND/ADW	
6.71E-09	S	LCL4-B1/SWD1&D2/AFW	
4.70E-12	NDF	LCL4-B1/SWD1&D2/AFW/EFW	
6.72E-13	NDF	LCL4-B1/SWD1&D2/AFW/ADW	

Fig.5 (Total Loss of CL4 supply) Page(5) C:\CAFTA\TREE(ET)\IELCL405.TRE 28-01-04

From LCL4-B2	Support Systems	Heat Sink				Operator Action		Heat Sink
LCL4/R60E4, at least 3-DGs available	SWD1&D2 SW System	AFW Supply to SGs	Aux. Condensate Pump System	Auto De-pressurization Water System	Emergency FW Supply to SGs	Operator Opens MSSVs to De-pressurize SGs	Operator Starts LTC-SDC	LTC-SDC System
IE-LCL4	SWD1&D2	AFW	ACND	ADW	EFW	OMSSV	OLTC-SDC	LTC-SDC



SEQ.FREQ.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE DESCRIPTION
7.03E-03	S	LCL4-B2	
3.70E-04	S	LCL4-B2/ACND	
2.43E-07	S	LCL4-B2/ACND/EFW	
1.28E-08	PDS6	LCL4-B2/ACND/EFW/LTC-SDC	LCL4-B21
2.59E-09	PDS6	LCL4-B2/ACND/EFW/OLTC-SDC	LCL4-B22
3.44E-08	S	LCL4-B2/ACND/ADW	
1.81E-09	PDS6	LCL4-B2/ACND/ADW/LTC-SDC	LCL4-B23
3.66E-10	NDF	LCL4-B2/ACND/ADW/OLTC-SDC	
3.70E-10	NDF	LCL4-B2/ACND/ADW/OMSSV	
7.47E-05	S	LCL4-B2/AFW	
4.92E-08	S	LCL4-B2/AFW/EFW	
2.59E-09	PDS6	LCL4-B2/AFW/EFW/LTC-SDC	LCL4-B24
5.23E-10	NDF	LCL4-B2/AFW/EFW/OLTC-SDC	
6.96E-09	S	LCL4-B2/AFW/ADW	
3.66E-10	PDS6	LCL4-B2/AFW/ADW/LTC-SDC	LCL4-B25
7.40E-11	NDF	LCL4-B2/AFW/ADW/OLTC-SDC	
7.47E-11	NDF	LCL4-B2/AFW/ADW/OMSSV	
7.03E-06	S	LCL4-B2/SWD1&D2	
3.70E-07	S	LCL4-B2/SWD1&D2/ACND	
2.59E-10	NDF	LCL4-B2/SWD1&D2/ACND/EFW	
3.70E-11	NDF	LCL4-B2/SWD1&D2/ACND/ADW	
7.47E-08	S	LCL4-B2/SWD1&D2/AFW	
5.24E-11	NDF	LCL4-B2/SWD1&D2/AFW/EFW	
7.48E-12	NDF	LCL4-B2/SWD1&D2/AFW/ADW	

Fig.6 (Total Loss of CL4 supply) Page(6) C:\CAFTA\TREE(ET)\IELCL406.TRE 28-01-04

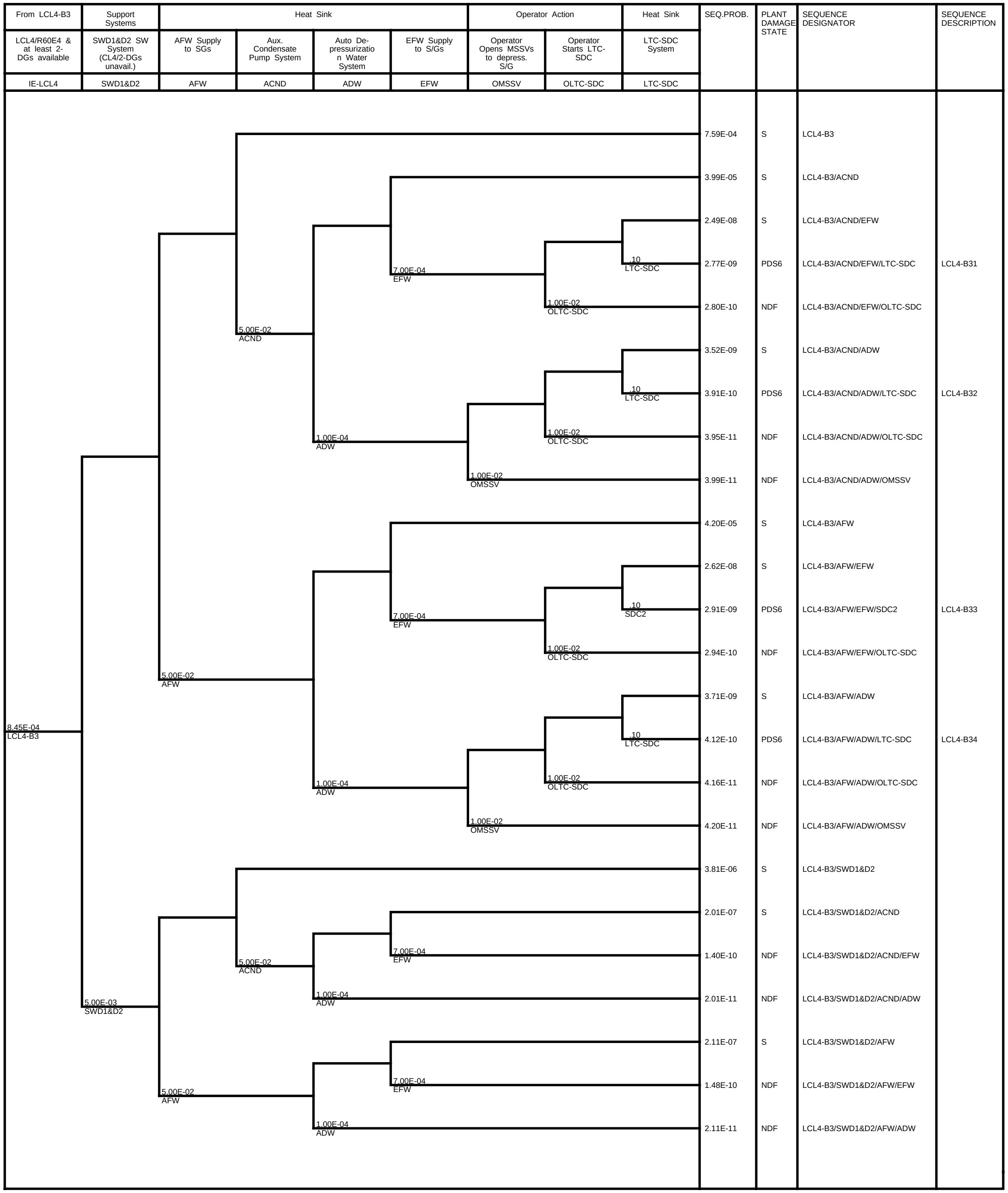
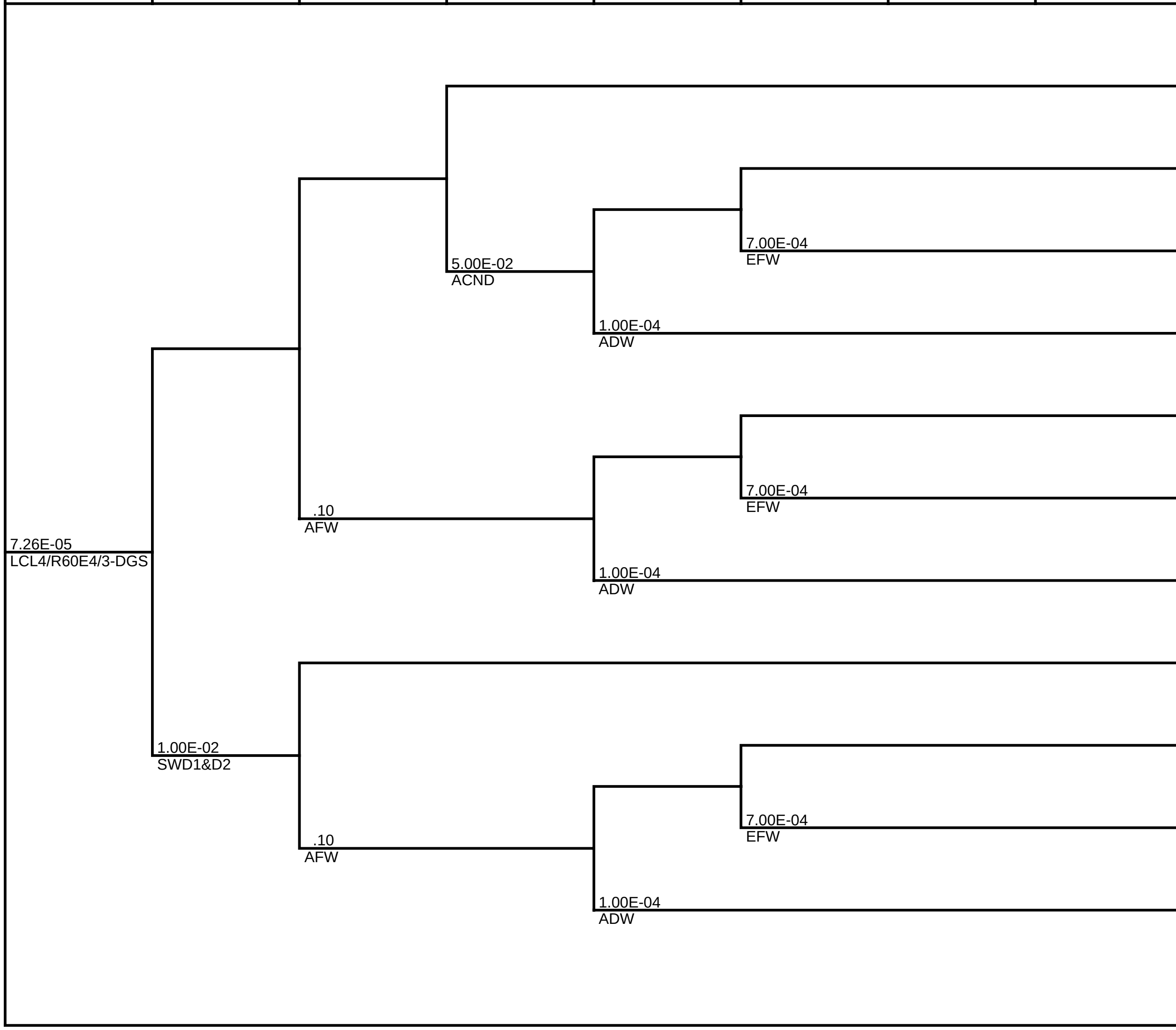


Fig.7 (Total Loss of CL4 supply) Page(7) C:\CAFTA\TREE(ET)\IELCL407.TRE 28-01-04

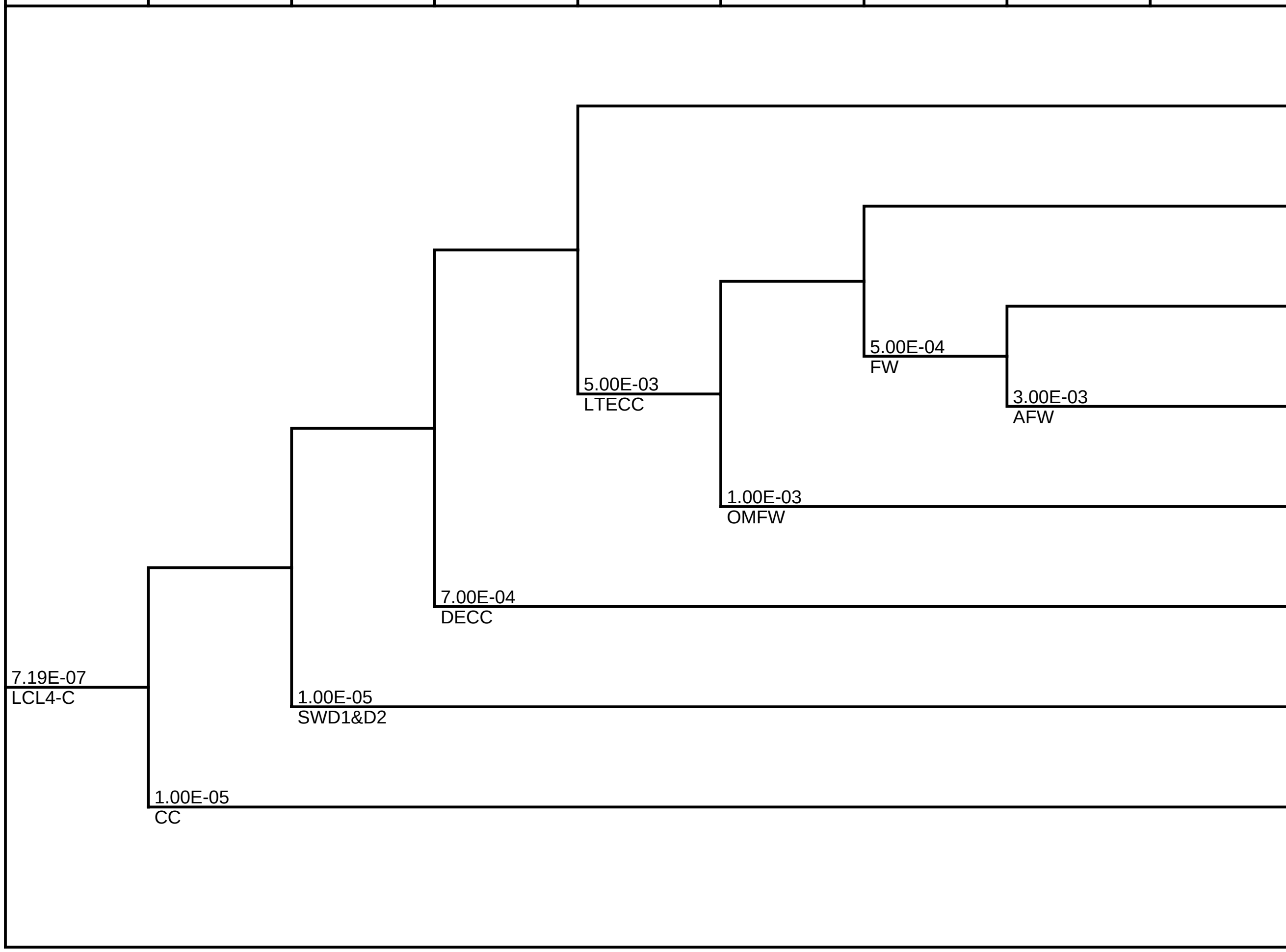
From LCL4-B4	Support Systems	Heat Sink				Operator Action	Heat Sink
LCL4/R60E4, at least 1-DG available	SWD1&D2 SW System	AFW Supply to SGs	Aux. Condensate Pump System	Auto De-pressurization Water System	EFW Supply to S/Gs	Operator Starts LTC-SDC	LTC-SDC System
LCL4-B4	SWD1&D2	AFW	ACND	ADW	EFW	OLTC-SDC	LTC-SDC



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
6.15E-05	S	LCL4/R60E4/3-DGS	
3.23E-06	S	LCL4/R60E4/3-DGS/ACND	
2.26E-09	NDF	LCL4/R60E4/3-DGS/ACND/EFW	
3.23E-10	NDF	LCL4/R60E4/3-DGS/ACND/ADW	
7.18E-06	S	LCL4/R60E4/3-DGS/AFW	
5.03E-09	NDF	LCL4/R60E4/3-DGS/AFW/EFW	
7.19E-10	NDF	LCL4/R60E4/3-DGS/AFW/ADW	
6.53E-07	S	LCL4/R60E4/3-DGS/SWD1&D2	
7.25E-08	S	LCL4/R60E4/3-DGS/SWD1&D2/AFW	
5.08E-11	NDF	LCL4/R60E4/3-DGS/SWD1&D2/AFW/EFW	
7.26E-12	NDF	LCL4/R60E4/3-DGS/SWD1&D2/AFW/ADW	

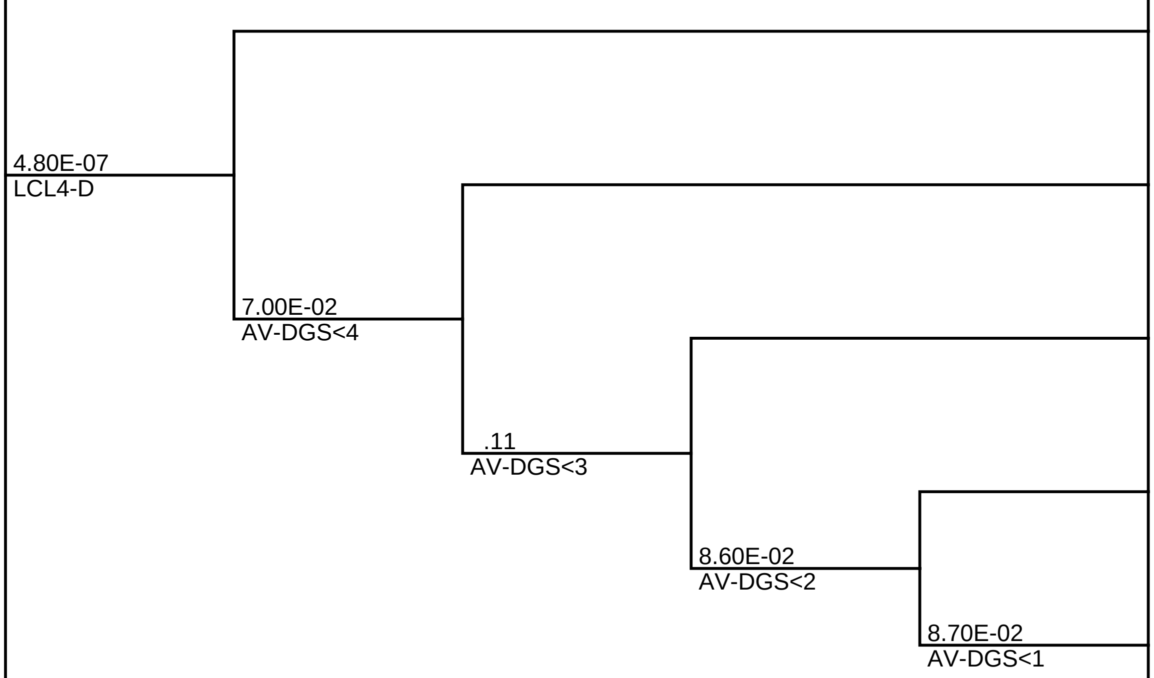
Fig.8 (Total Loss of CL4 supply) Page(8) C:\CAFTA\TREE(ET)\IELCL408.TRE 28-01-04

From LCL4-C	Heat Sink	Support System	HTS Make-up	Heat Sink	Operator Action	Heat Sink			SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
IE-CLIV/CLPRV(conseq.LOCA), CL 4 restored	Crash Cooldown	(RSW/RCW) System	D-ECC Supply to HTS	Long-term ECC supply	Operator starts MFW supply	FW Supply to S/Gs	AFW supply to S/Gs	EFW supply to S/Gs				
IE-LCL4	CC	SWD1&D2	DECC	LTECC	OMFW	FW	AFW	EFW				



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
7.15E-07	S	LCL4-C	
3.59E-09	S	LCL4-C/LTECC	
1.79E-12	S	LCL4-C/LTECC/FW	
5.38E-15	NDF	LCL4-C/LTECC/FW/AFW	
3.59E-12	NDF	LCL4-C/LTECC/OMFW	
5.03E-10	NDF	LCL4-C/DECC	
7.19E-12	NDF	LCL4-C/SWD1&D2	
7.19E-12	NDF	LCL4-C/CC	

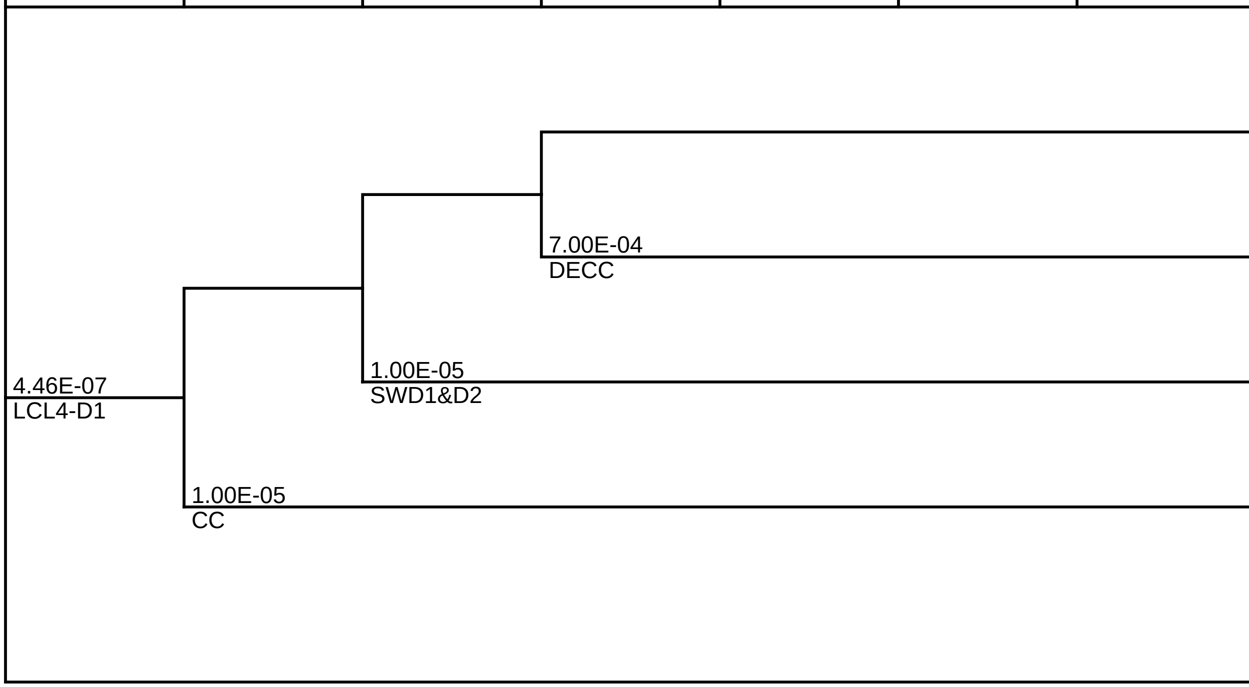
From LCL4-D	Class3 Power			
LCL4/R60E4, conseqt. LOCA via LRVs	4 DGs available for 24hrs.	3 DGs available for 24hrs.	2 DGs available for 24 hrs.	1 DG available for 24hrs.
IE-LCL4	DG-AV=4	DG-AV=3	DG-AV=2	DG-AV=1



SEQ.FREQ.	PLANT DAMAGE STATE	SEQUENCE DESCRIPTION	SEQUENCE DESIGNATOR
4.46E-07	CONTD	LCL4-D1	LCL4-D/UNAV-DGS=0
2.99E-08	CONTD	LCL4-D2	LCL4-D/UNAV-DGS=1
3.38E-09	CONTD	LCL4-D3	LCL4-D/UNAV-DGS=2
2.90E-10	NDF		LCL4-D/UNAV-DGS=3
2.77E-11	NDF		LCL4-D/UNAV-DGS=4

Fig.10 (Total Loss of CL4 supply) Page(10) C:\CAFTA\TREE(ET)\IELCL410.TRE 13-06-03

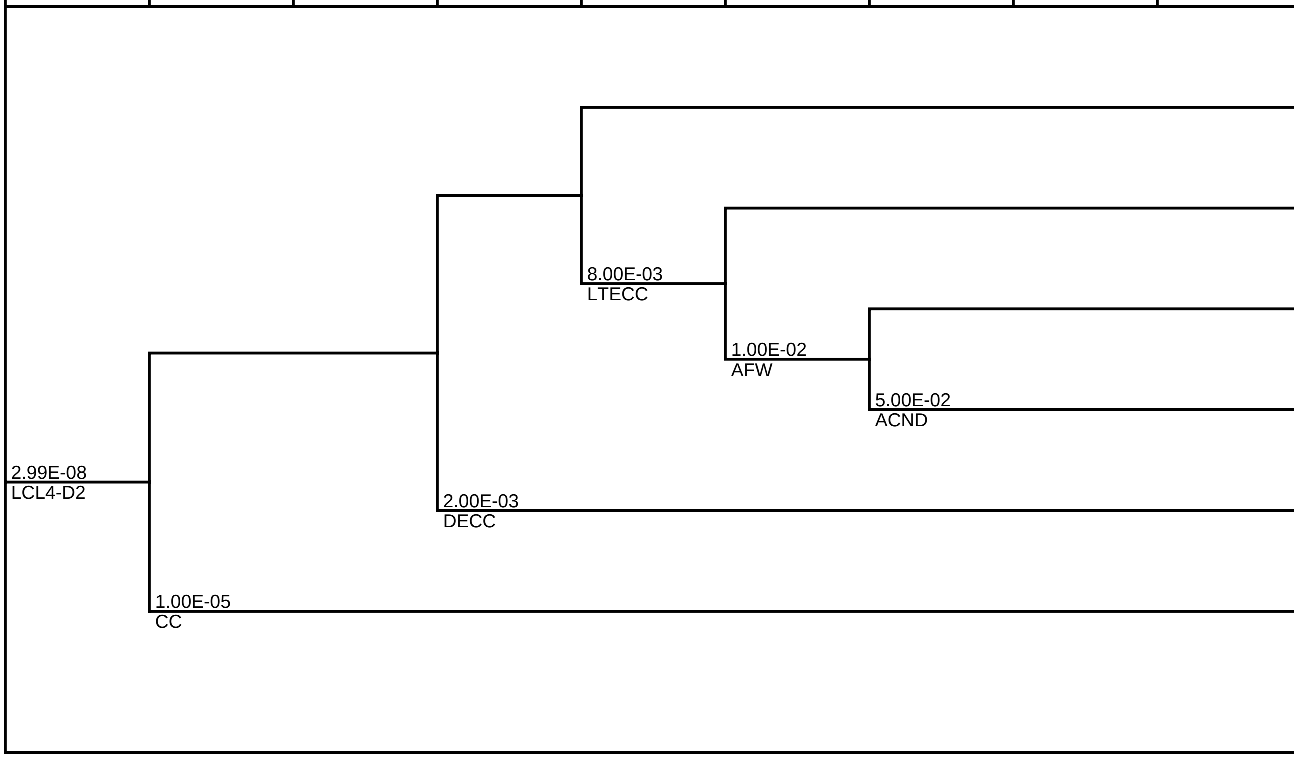
From LCL4-D1	Heat Sink	Support System	HTS Makeup	Heat Sink			SEQ.PROB.	CLASS	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
LCLIV/CLPR V/R60E4 (4-DGs available)	Crash Cooldown	SWD1&D2 SW (RSW/RCW) System	D-ECC Supply to HTS	Long-term ECC Supply	AFW Supply to S/Gs	Aux. Condensate System				
IE-LCL4	CC	SWD1&D2	DECC	LTECC	AFW	ACND				



SEQ.PROB.	CLASS	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
4.46E-07	S	LCL4-D1	
3.12E-10	NDF	LCL4-D1/DECC	
4.46E-12	NDF	LCL4-D1/SWD1&D2	
4.46E-12	NDF	LCL4-D1/CC	

Fig.11 (Total Loss of CL4 supply) Page(11) C:\CAFTA\TREE(ET)\IELCL411.TRE 16-06-03

From LCL4-D2	Heat Sink	Support System	HTS Makeup	Heat Sink					SEQ.PROB.	CLASS	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
LCLIV/CLPRV/R60E4(3-DGs available)	Crash Cooldown	SWD1&D2 SW(RSW/RCW) System	Emergency Core Cooling Supply to HTS	Long-term ECC Supply	AFW Supply to S/Gs	Auxiliary Condensate System	EFW Supply to SGs	Moderator Acts as Heat Sink				
IE-LCL4	CC	SWD1&D2	DECC	LTECC	AFW	ACND	EFW	MHS				



SEQ.PROB.	CLASS	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
2.96E-08	S	LCL4-D2	
2.36E-10	S	LCL4-D2/LTECC	
2.27E-12	S	LCL4-D2/LTECC/AFW	
1.19E-13	NDF	LCL4-D2/LTECC/AFW/ACND	
5.98E-11	NDF	LCL4-D2/DECC	
2.99E-13	NDF	LCL4-D2/CC	

Fig.12 (Total Loss of CL4 supply) Page(12) C:\CAFTA\TREE(ET)\IELCL412.TRE 16-06-03

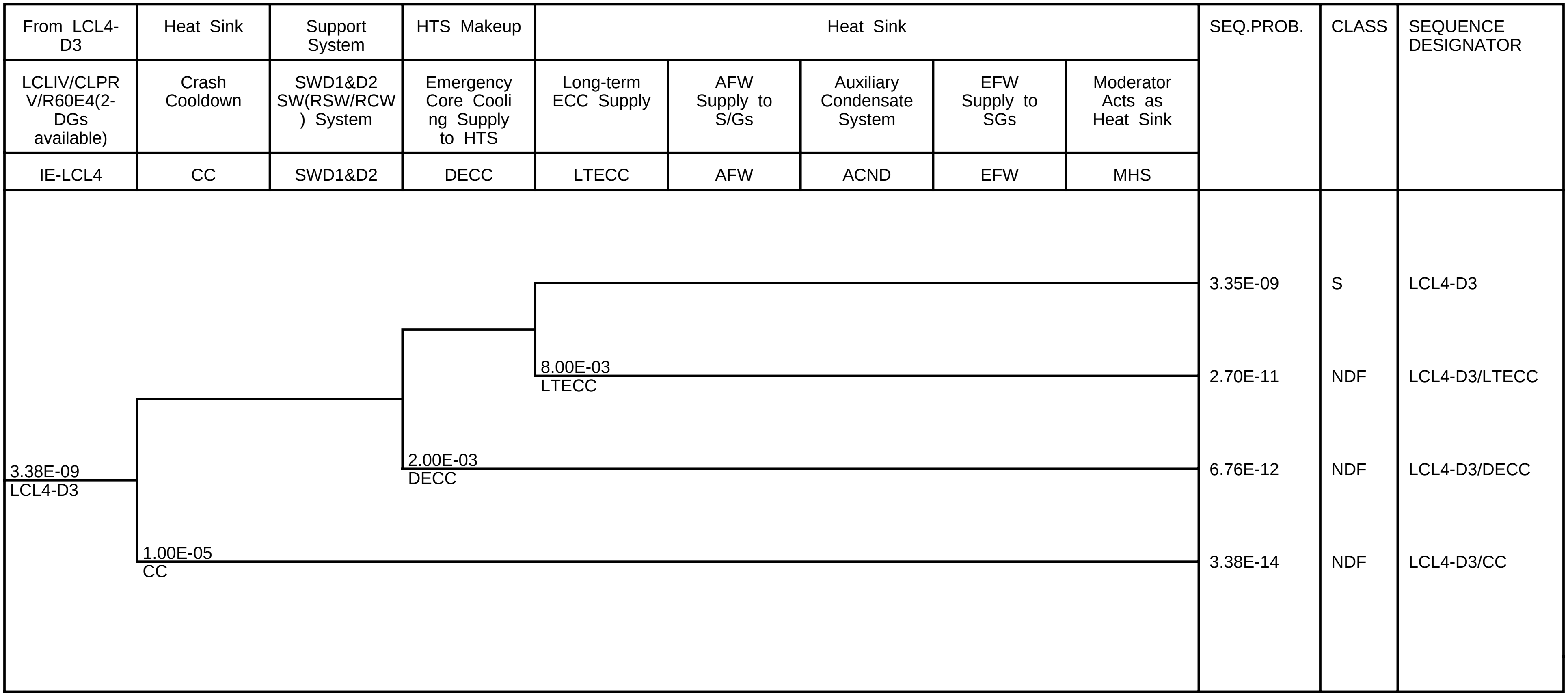


Fig.13 (Total Loss of CL4 supply) Page(13) C:\CAFTA\TREE(ET)\IELCL413.TRE 03-12-03

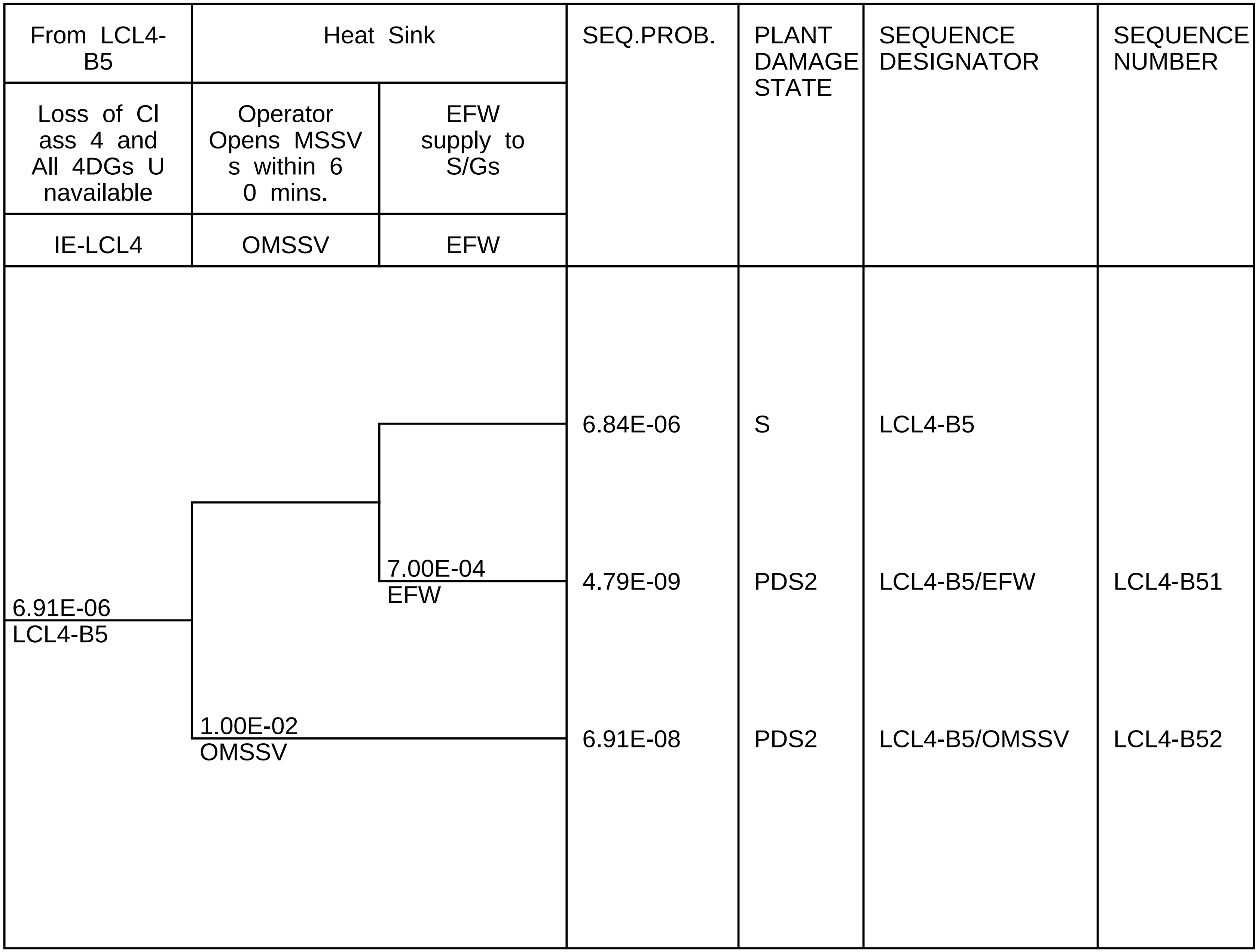
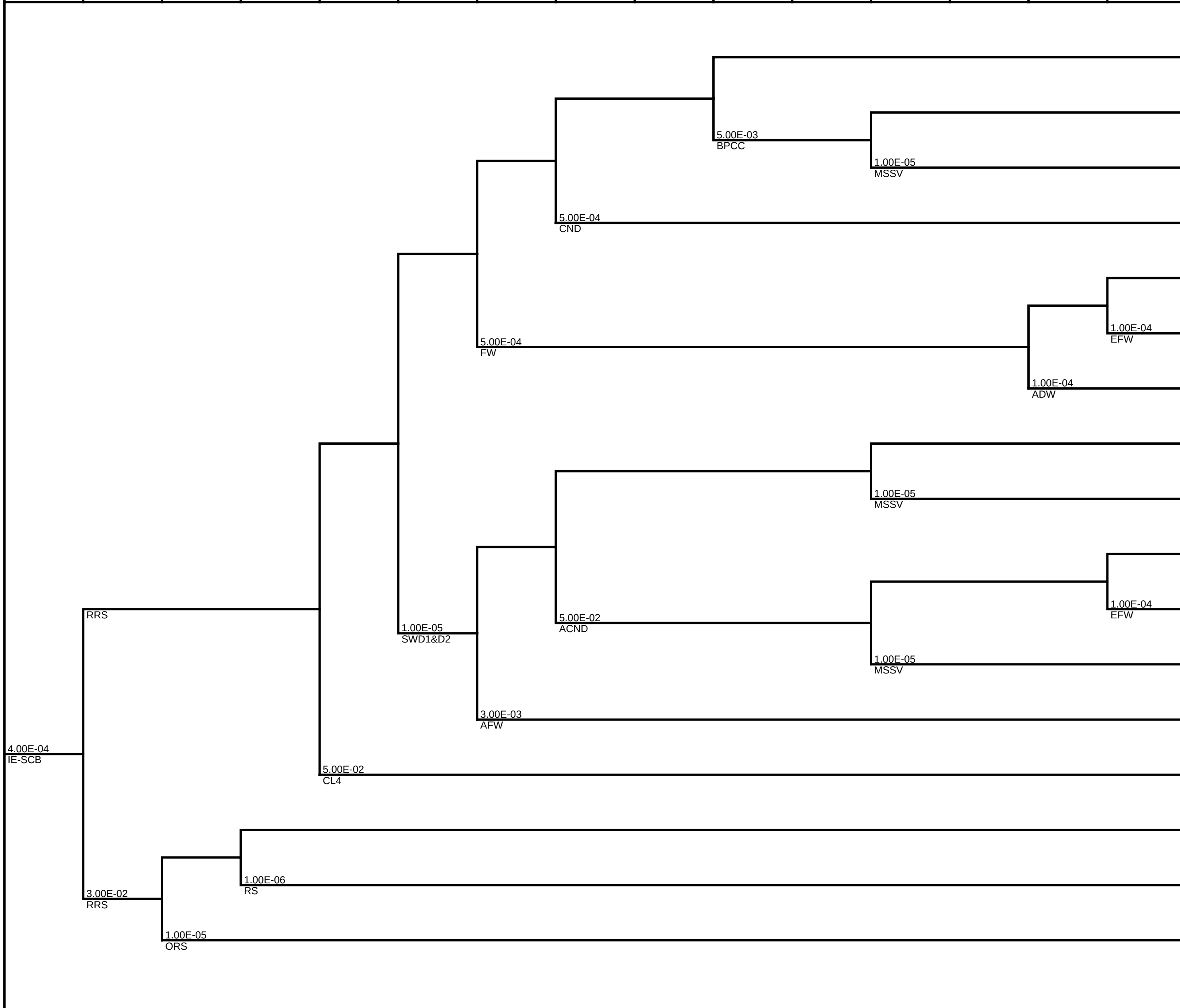


Fig.14 (Loss of CL4&CL3 power supplies) Page(14) C:\CAFTA\TREE(ET)\IELCL414.TRE 16-06-03

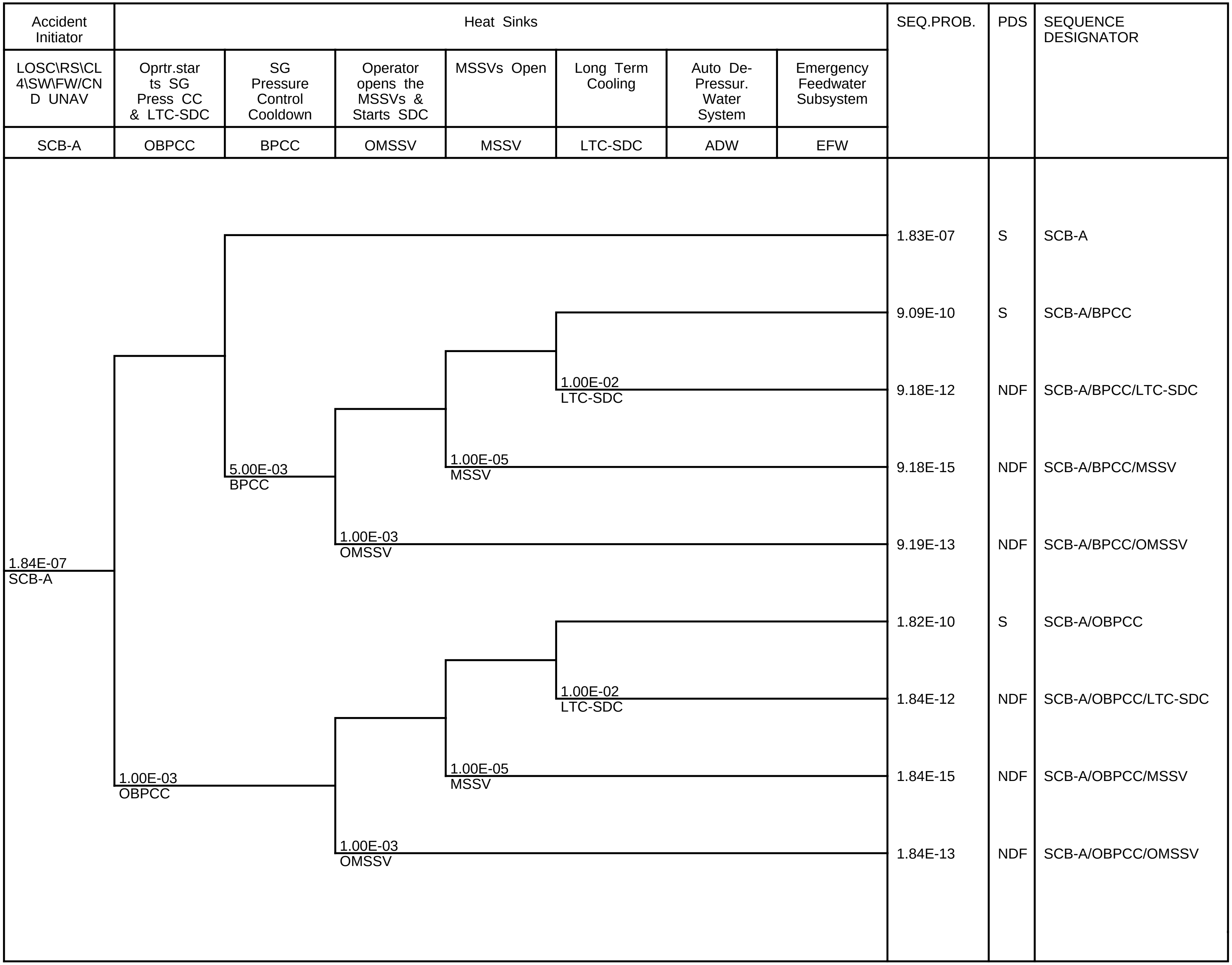
Appendix G

Event Tree for Loss of Inventory in Shield Cooling System

Initiating Event	Reactor Shutdown			Support System		Heat Sinks								
Loss of SC inventory via pipe break	RS via SETB on low SC Level	Operator Initiates RS via SDS1/SDS2	Reactor Shutdown (SDS1/SDS2)	No Consequential Loss of CL4 Power Suppl	Divisions #1 & #2 Service Water System	Feedwater (MFW+AFW)	Condensate System (Main+Aux.)	Oprtr starts SG Press CC & LTC-SDC	SG Pressure Control Cooldown	Operator Starts MSSVs	MSSVs open	Long term cooling	Auto De-Pressurisation Water System	Emergency Feedwater Subsystem
IE-SCB	RRS	ORS	RS	CL4	SWD1&D2	FW	CND	OBPCC	BPCC	OMSSV	MSSV	LTC-SDC	ADW	EFW



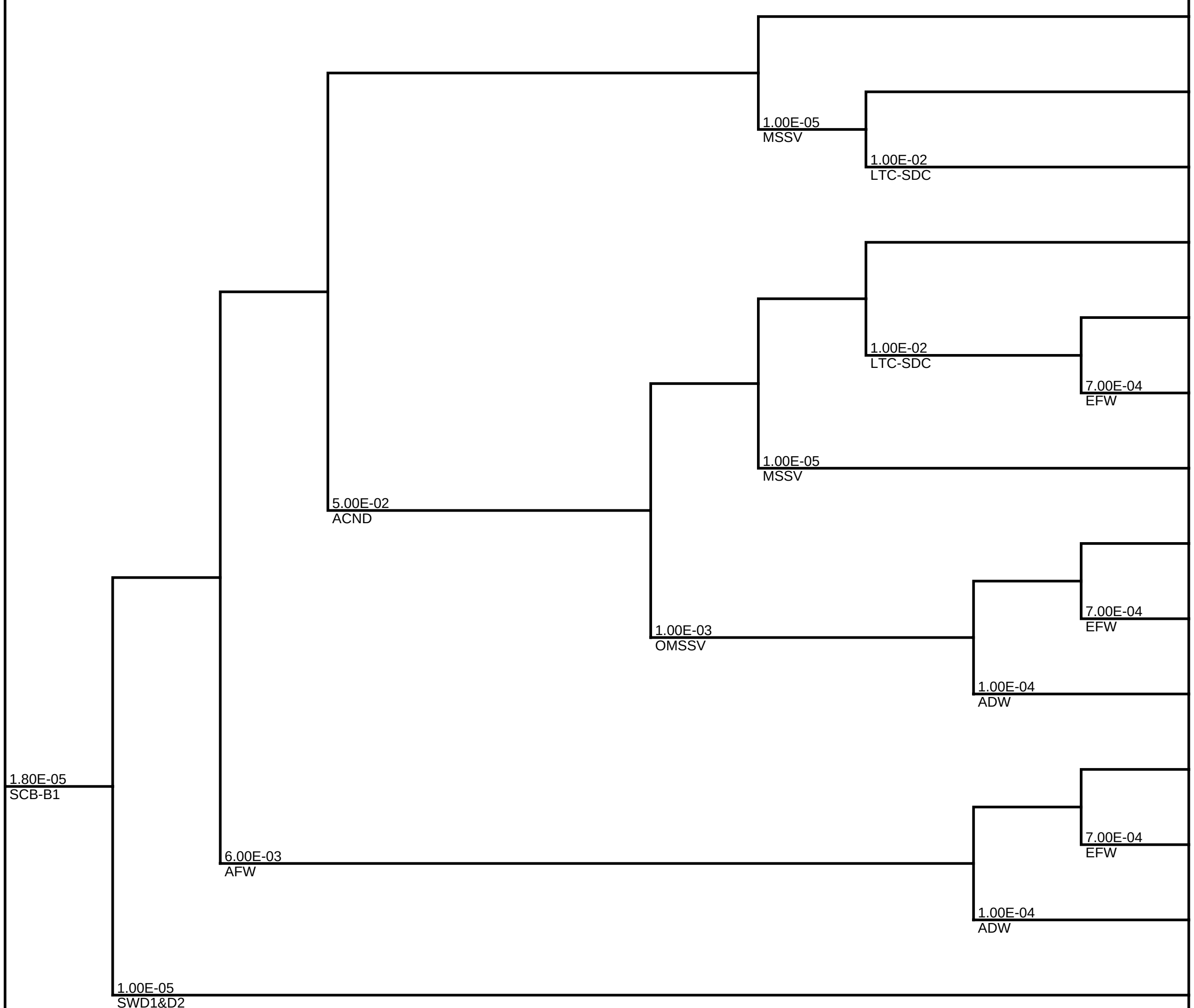
SEQ.PROB.	PDS	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
3.66E-04	S	IE-SCB	
1.84E-06	S	IE-SCB/BPCC	
1.84E-11	NDF	IE-SCB/BPCC/MSSV	
1.84E-07	CONT'D	IE-SCB/CND	SCB-A
1.84E-07	S	IE-SCB/FW	
1.84E-11	PDS6	IE-SCB/FW/EFW	SCB-1
1.84E-11	PDS6	IE-SCB/FW/ADW	SCB-2
3.49E-09	S	IE-SCB/SWD1&D2	
3.49E-14	NDF	IE-SCB/SWD1&D2/MSSV	
1.84E-10	S	IE-SCB/SWD1&D2/ACND	
1.84E-14	PDS6	IE-SCB/SWD1&D2/ACND/EFW	SCB-3
1.84E-15	NDF	IE-SCB/SWD1&D2/ACND/MSSV	
1.11E-11	NDF	IE-SCB/SWD1&D2/AFW	
1.94E-05	CONT'D	IE-SCB/CL4	SCB-B
1.20E-05	CONT'D	IE-SCB/RRS	SCB-C
1.20E-11	PDS0	IE-SCB/RRS/RS	SCB-4
1.20E-10	PDS0	IE-SCB/RRS/ORS	SCB-5



SCB-Loss of SC inventory via pipe break C:\CAFTA\TREE(ET)\SCB-A.TRE 03-12-03

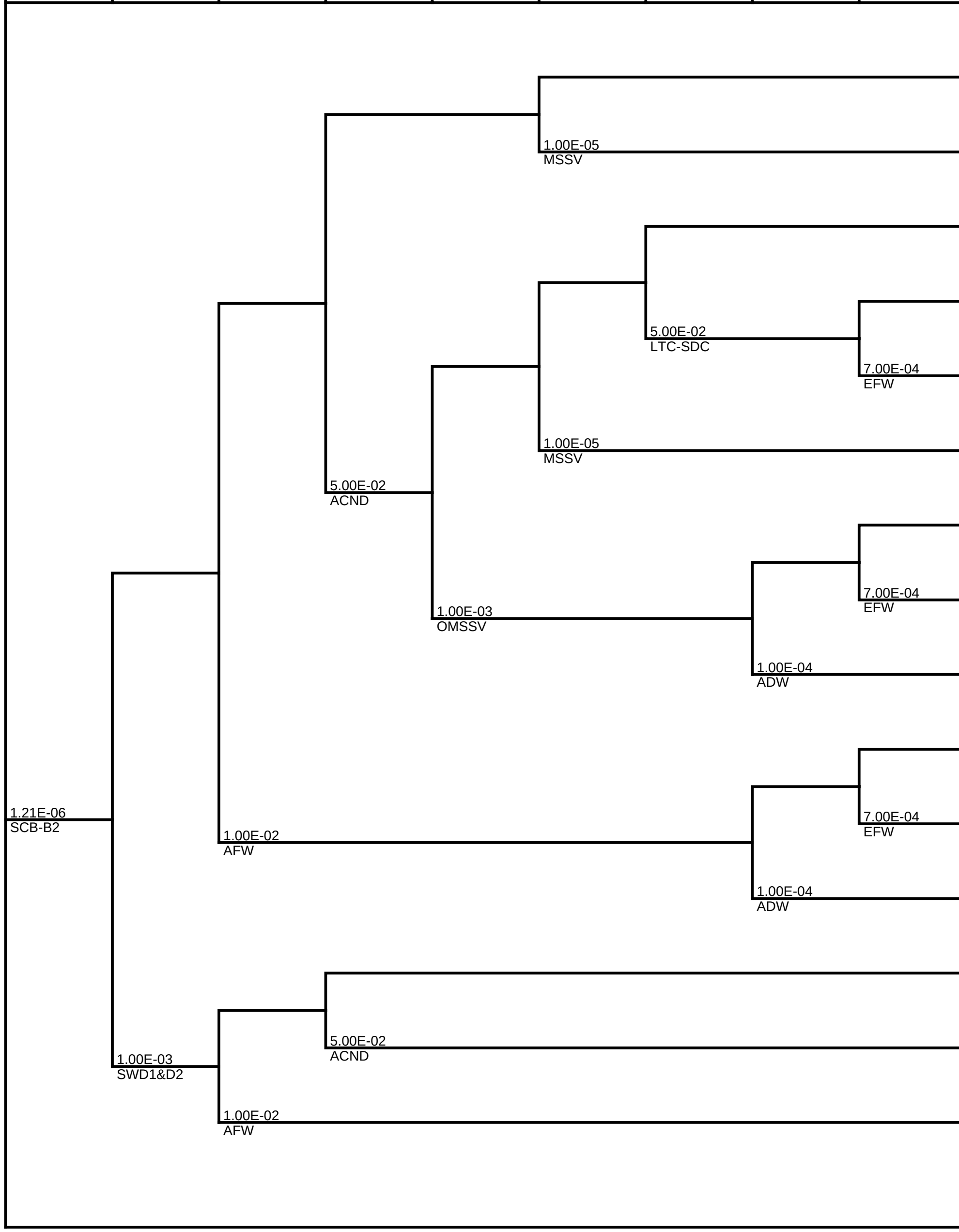
Initiating Event	Class III Power Supply				SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE NUMBER	SEQUENCE DESIGNATOR
LOSC\RS/CL 4	4 DGs Available for 24h	3 DGs Available for 24h	2 DGs Available for 24h	1 DG Available for 24h				
SCB-B	DG-AV=4	DG-AV=3	DG-AV=2	DG-AV=1				
					1.80E-05	CONT'D	SCB-B1	IE-SCB/CL4/UNAV-DGS=0
1.94E-05 SCB\RS/CL4					1.21E-06	CONT'D	SCB-B2	IE-SCB/CL4/UNAV-DGS=1
	7.00E-02 AV-DGS < 4				1.37E-07	CONT'D	SCB-B3	IE-SCB/CL4/UNAV-DGS=2
		.11 AV-DGS < 3			1.17E-08	CONT'D	SCB-B4	IE-SCB/CL4/UNAV-DGS=3
			8.60E-02 AV-DGS < 2	8.70E-02 AV-DGS < 1	1.12E-09	PDS2	SCB-B-1	IE-SCB/CL4/UNAV-DGS=4

Initiating Event	Support System	Heat Sinks									SEQ.PROB.	PDS	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
LOSC\RS/CL 4\ALL DGS AV	Divisions #1 & #2 Service Water System	Auxiliary Feedwater System	Auxiliary Condensate System	Oprtr.starts SG Press CC & LTC-SDC	SG Pressure Control Cooldown	Operator opens the MSSV & starts LTCSDC	MSSVs Open	Long Term Cooling	Auto De-Pressur. Water System	Emergency Feedwater Subsystem				
SCB-B1	SWD1&D2	AFW	ACND	OBPCC	BPCC	OMSSV	MSSV	LTC-SDC	ADW	EFW				



SEQ.PROB.	PDS	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
1.70E-05	S	SCB-B1	
1.68E-10	S	SCB-B1/MSSV	
1.70E-12	NDF	SCB-B1/MSSV/LTC-SDC	
8.85E-07	S	SCB-B1/ACND	
8.93E-09	S	SCB-B1/ACND/LTC-SDC	
6.26E-12	NDF	SCB-B1/ACND/LTC-SDC/EFW	
8.94E-12	NDF	SCB-B1/ACND/MSSV	
8.94E-10	S	SCB-B1/ACND/OMSSV	
6.26E-13	NDF	SCB-B1/ACND/OMSSV/EFW	
8.95E-14	NDF	SCB-B1/ACND/OMSSV/ADW	
1.08E-07	S	SCB-B1/AFW	
7.56E-11	PDS6	SCB-B1/AFW/EFW	SCB-B1-1
1.08E-11	PDS6	SCB-B1/AFW/ADW	SCB-B1-2
1.80E-10	NDF	SCB-B1/SWD1&D2	

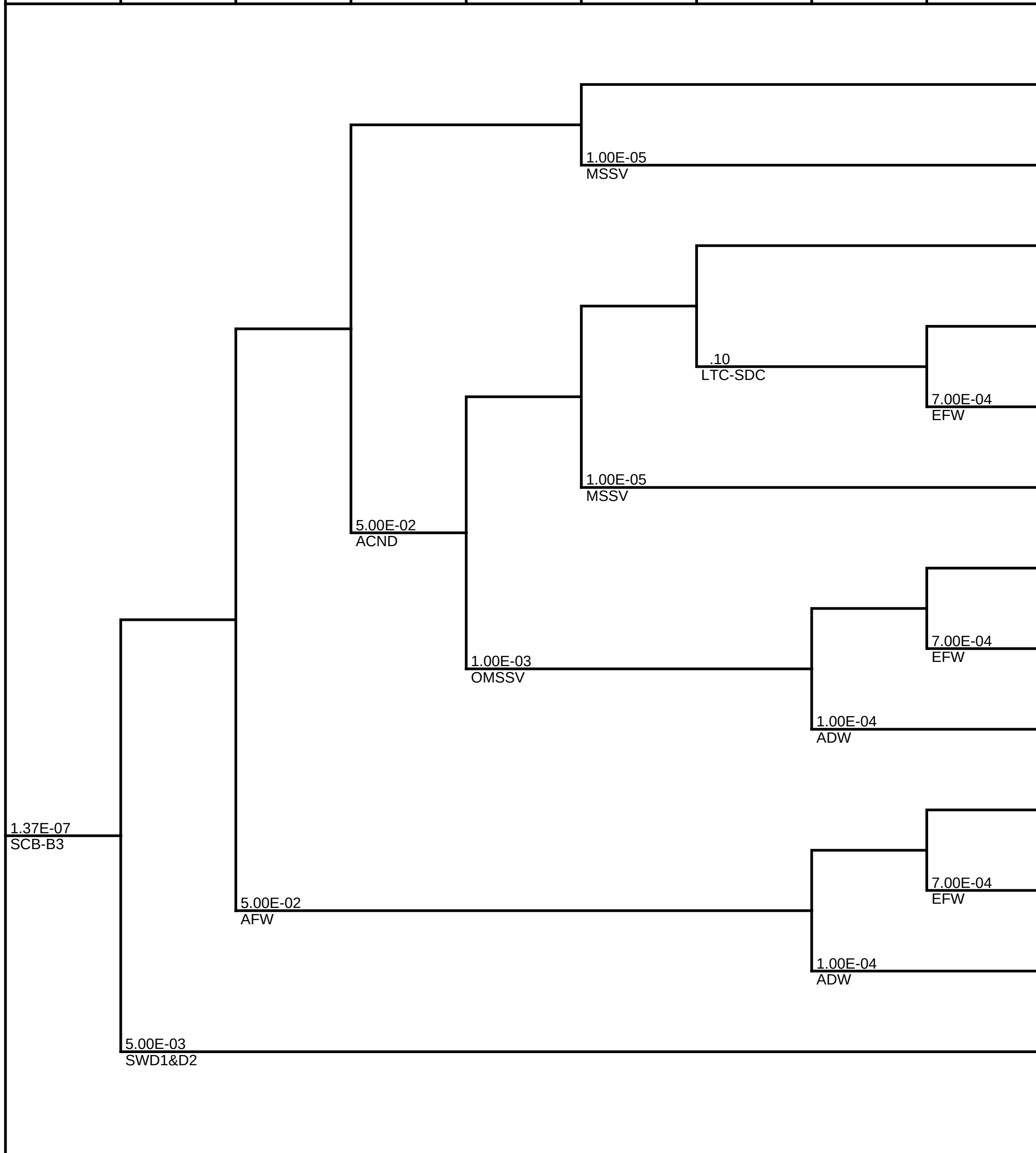
Initiating Event	Support System	Heat Sinks						
LOSC\RS/CL 4\1 DG UN	Divisions #1 & #2 Service Water System	Auxiliary Feedwater System	Auxiliary Condensate System	Operator opens the MSSV & starts LTCSDC	MSSVs OPEN	Long Term Cooling	Auto De-Pressur. Water System	Emergency Feedwater Subsystem
SCB-B2	SWD1&D2	AFW	ACND	OMSSV	MSSV	LTC-SDC	ADW	EFW



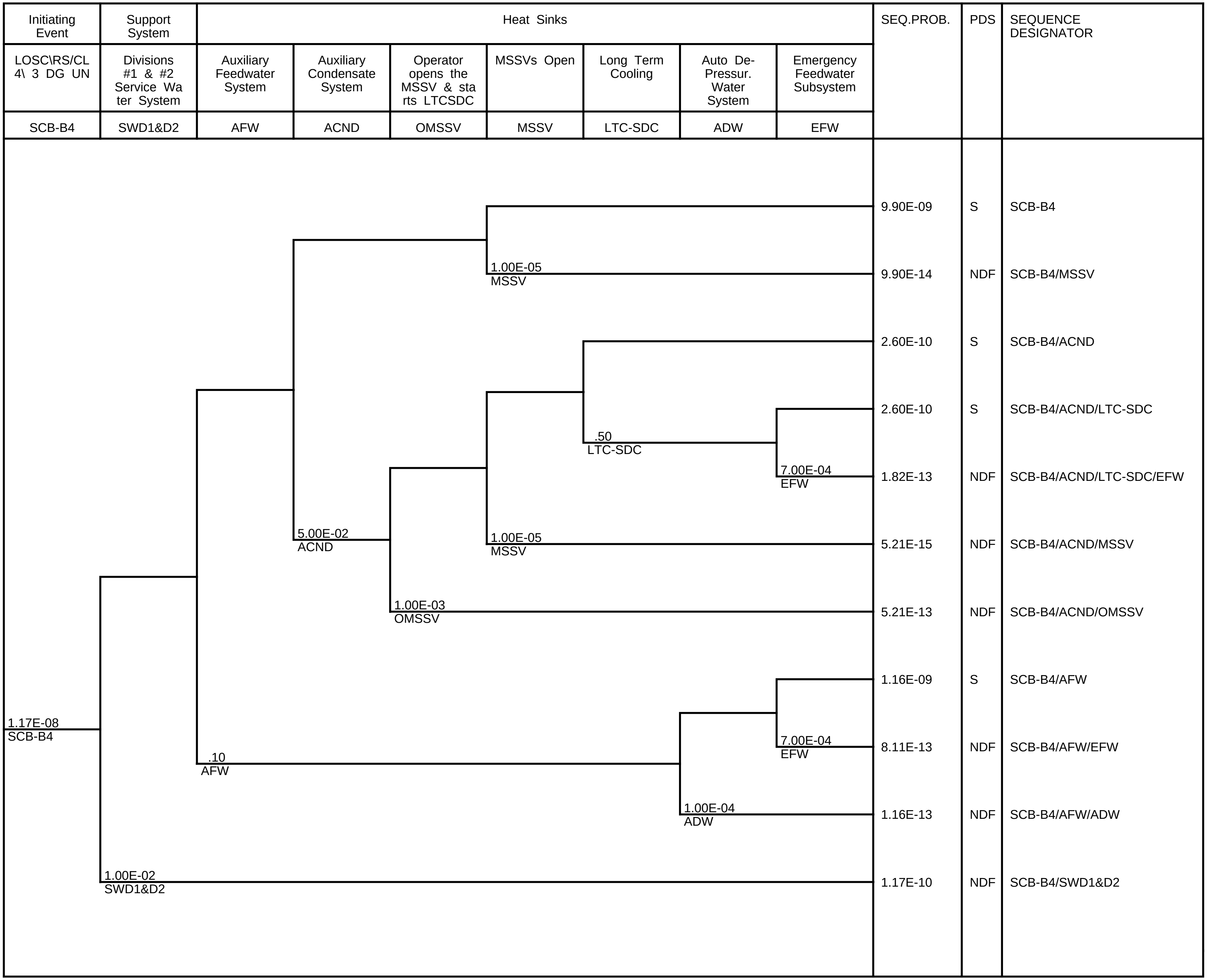
SEQ.PROB.	PDS	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
1.14E-06	S	SCB-B2	
1.14E-11	NDF	SCB-B2/MSSV	
5.68E-08	S	SCB-B2/ACND	
2.99E-09	S	SCB-B2/ACND/LTC-SDC	
2.09E-12	NDF	SCB-B2/ACND/LTC-SDC/EFW	
5.98E-13	NDF	SCB-B2/ACND/MSSV	
5.98E-11	S	SCB-B2/ACND/OMSSV	
4.19E-14	NDF	SCB-B2/ACND/OMSSV/EFW	
5.98E-15	NDF	SCB-B2/ACND/OMSSV/ADW	
1.21E-08	S	SCB-B2/AFW	
8.46E-12	PDS6	SCB-B2/AFW/EFW	SCB-B2-1
1.21E-12	PDS6	SCB-B2/AFW/ADW	SCB-B2-2
1.14E-09	S	SCB-B2/SWD1&D2	
5.99E-11	NDF	SCB-B2/SWD1&D2/ACND	
1.21E-11	NDF	SCB-B2/SWD1&D2/AFW	

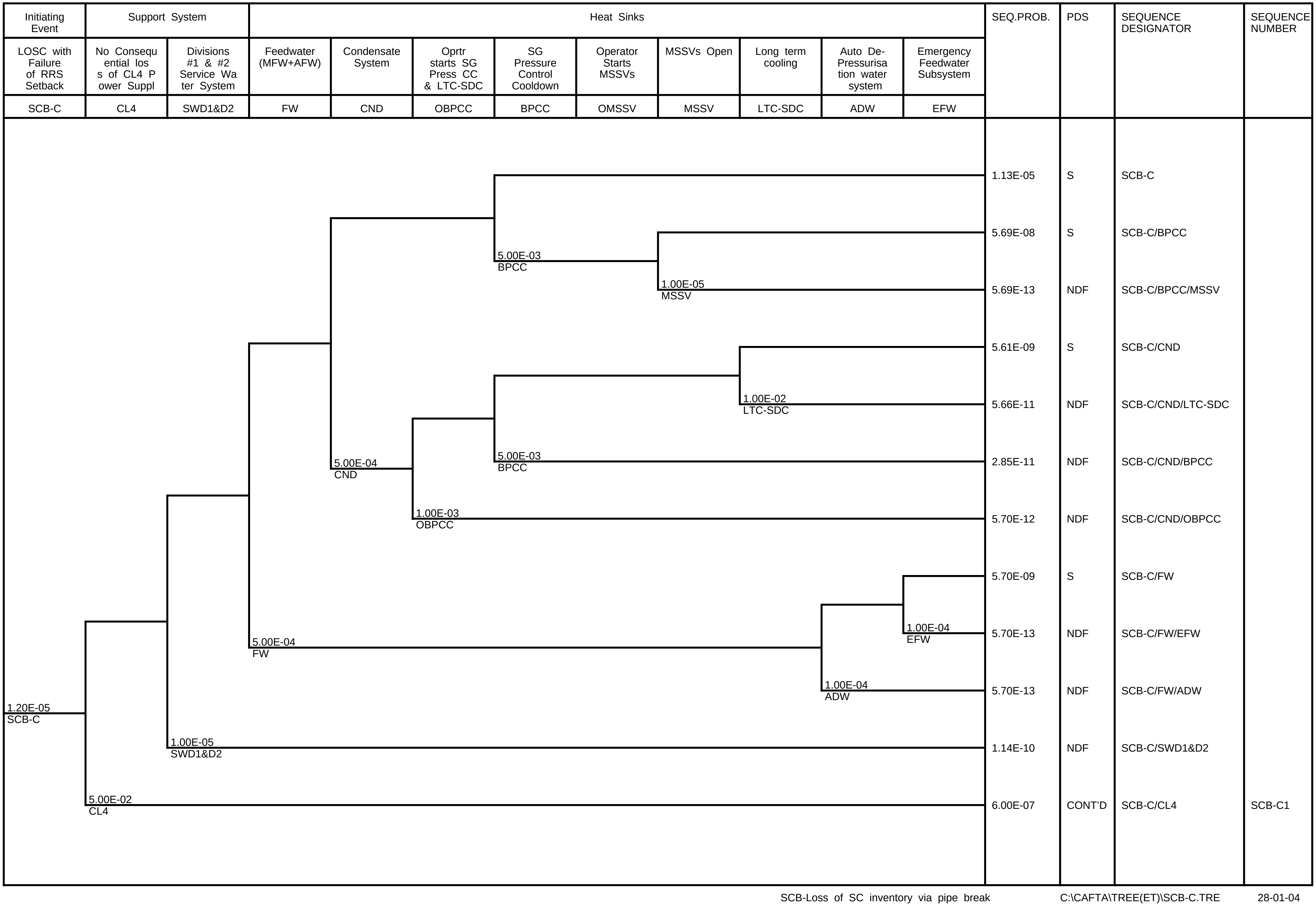
SCB-Loss of SC inventory via pipe break C:\CAFTA\TREE(ET)\SCB-B2.TRE 03-12-03

Initiating Event	Support System	Heat Sinks							SEQ.PROB.	PDS	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
LOSC\RS/CL 4\2 DG UN	Divisions #1 & #2 Service Water System	Auxiliary Feedwater System	Auxiliary Condensate System	Operator opens the MSSV & starts LTCSDC	MSSVs OPEN	Long Term Cooling	Auto De-Pressur. Water System	Emergency Feedwater Subsystem				
SCB-B3	SWD1&D2	AFW	ACND	OMSSV	MSSV	LTC-SDC	ADW	EFW				

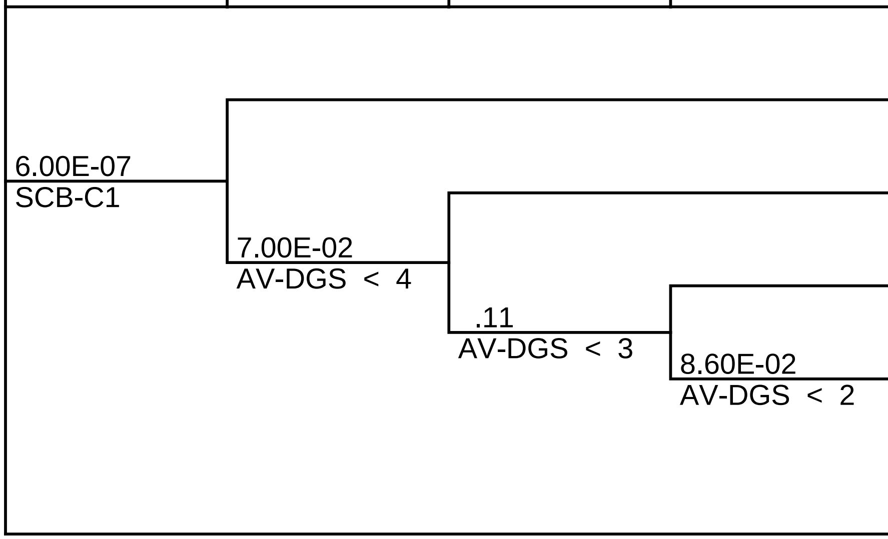


SEQ.PROB.	PDS	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
1.23E-07	S	SCB-B3	
1.23E-12	NDF	SCB-B3/MSSV	
5.82E-09	S	SCB-B3/ACND	
6.46E-10	S	SCB-B3/ACND/LTC-SDC	
4.53E-13	NDF	SCB-B3/ACND/LTC-SDC/EFW	
6.47E-14	NDF	SCB-B3/ACND/MSSV	
6.47E-12	S	SCB-B3/ACND/OMSSV	
4.53E-15	NDF	SCB-B3/ACND/OMSSV/EFW	
6.47E-16	NDF	SCB-B3/ACND/OMSSV/ADW	
6.81E-09	S	SCB-B3/AFW	
4.77E-12	NDF	SCB-B3/AFW/EFW	
6.82E-13	NDF	SCB-B3/AFW/ADW	
6.85E-10	NDF	SCB-B3/SWD1&D2	


Initiating Event
Support System
Heat Sinks
SEQ.PROB.
PDS
SEQUENCE DESIGNATOR
LOSC\RS/CL 4\ 3 DG UN
Divisions #1 & #2 Service Water System
Auxiliary Feedwater System
Auxiliary Condensate System
Operator opens the MSSV & starts LTCSDC
MSSVs Open
Long Term Cooling
Auto De-Pressur. Water System
Emergency Feedwater Subsystem
SCB-B4
SWD1&D2
AFW
ACND
OMSSV
MSSV
LTC-SDC
ADW
EFW
1.17E-08
SCB-B4
1.00E-02
SWD1&D2
.10
AFW
5.00E-02
ACND
1.00E-05
MSSV
1.00E-03
OMSSV
1.00E-05
MSSV
.50
LTC-SDC
7.00E-04
EFW
1.00E-04
ADW
7.00E-04
EFW
9.90E-09 S SCB-B4
9.90E-14 NDF SCB-B4/MSSV
2.60E-10 S SCB-B4/ACND
2.60E-10 S SCB-B4/ACND/LTC-SDC
1.82E-13 NDF SCB-B4/ACND/LTC-SDC/EFW
5.21E-15 NDF SCB-B4/ACND/MSSV
5.21E-13 NDF SCB-B4/ACND/OMSSV
1.16E-09 S SCB-B4/AFW
8.11E-13 NDF SCB-B4/AFW/EFW
1.16E-13 NDF SCB-B4/AFW/ADW
1.17E-10 NDF SCB-B4/SWD1&D2

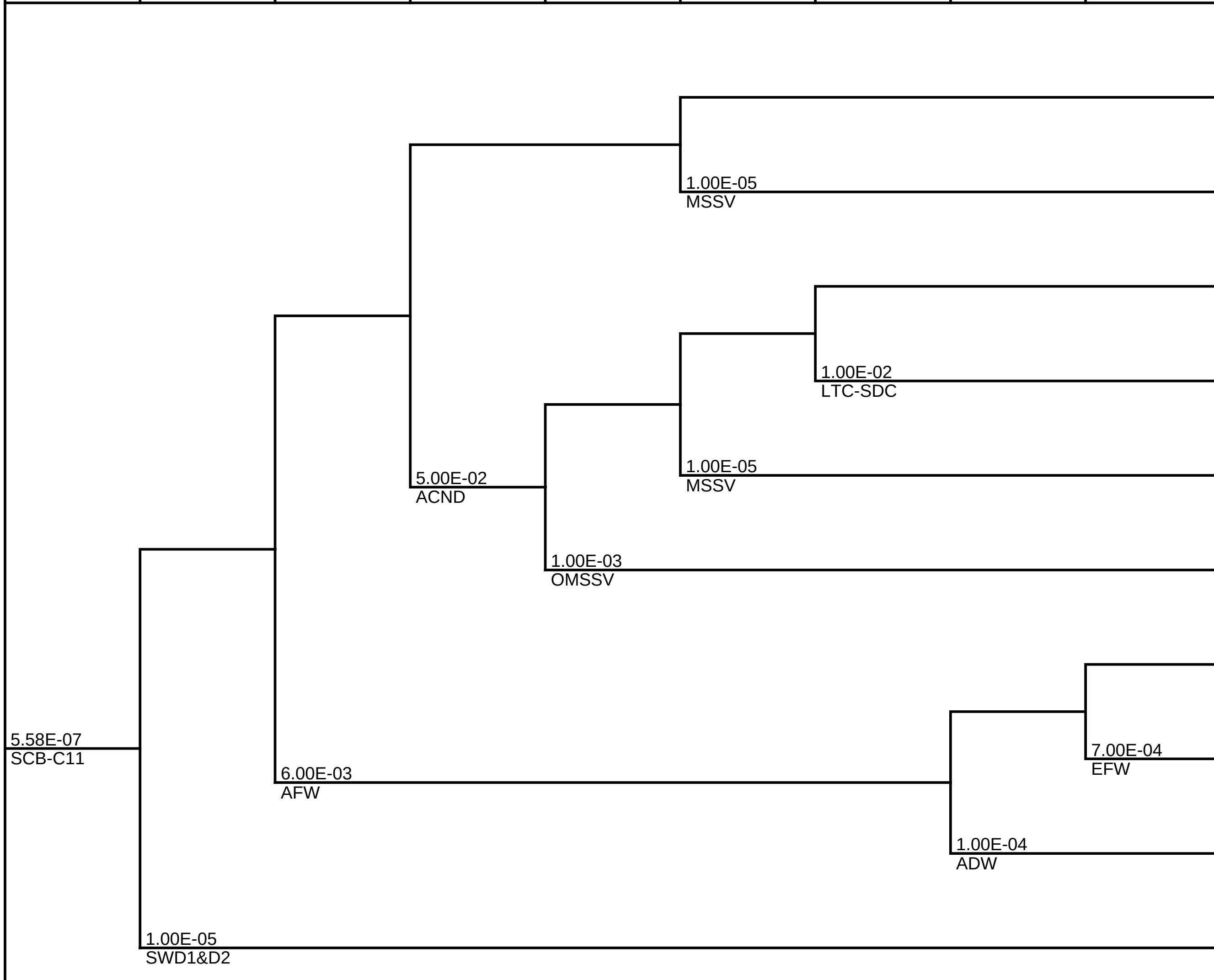

Initiating Event
Support System
Heat Sinks
SEQ.PROB.
PDS
SEQUENCE DESIGNATOR
SEQUENCE NUMBER
LOSC with Failure of RRS Setback
No Consequential loss of CL4 Power Suppl
Divisions #1 & #2 Service Water System
Feedwater (MFW+AFW)
Condensate System
Oprtr starts SG Press CC & LTC-SDC
SG Pressure Control Cooldown
Operator Starts MSSVs
MSSVs Open
Long term cooling
Auto De-Pressurisation water system
Emergency Feedwater Subsystem
SCB-C
CL4
SWD1&D2
FW
CND
OBPCC
BPCC
OMSSV
MSSV
LTC-SDC
ADW
EFW
1.20E-05
SCB-C
5.00E-02
CL4
1.00E-05
SWD1&D2
5.00E-04
FW
5.00E-04
CND
1.00E-03
OBPCC
5.00E-03
BPCC
5.00E-03
BPCC
1.00E-05
MSSV
1.00E-02
LTC-SDC
1.00E-04
ADW
1.00E-04
EFW
1.13E-05 S SCB-C
5.69E-08 S SCB-C/BPCC
5.69E-13 NDF SCB-C/BPCC/MSSV
5.61E-09 S SCB-C/CND
5.66E-11 NDF SCB-C/CND/LTC-SDC
2.85E-11 NDF SCB-C/CND/BPCC
5.70E-12 NDF SCB-C/CND/OBPCC
5.70E-09 S SCB-C/FW
5.70E-13 NDF SCB-C/FW/EFW
5.70E-13 NDF SCB-C/FW/ADW
1.14E-10 NDF SCB-C/SWD1&D2
6.00E-07 CONT'D SCB-C/CL4 SCB-C1
SCB-Loss of SC inventory via pipe break
C:\CAFTA\TREE(ET)\SCB-C.TRE
28-01-04

Initiating Event	Class III Power Supply			SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESCRIPTION	SEQUENCE NUMBER
LOSC with RRS Setbac k Failure & Loss of	4 DGs Available For 24h	3 DGs Available for 24h	2 DGs Available for 24h				
IE-SCB	DG-AV=4	DG-AV=3	DG-AV=2				
				5.58E-07	Cont’d	IE-SCB/RRS/CL4/UNAV-DGS=0	SCB-C11
				3.74E-08	Cont’d	IE-SCB/RRS/CL4/UNAV-DGS=1	SCB-C12
				4.22E-09	Cont’d	IE-SCB/RRS/CL4/UNAV-DGS=2	SCB-C13
				3.97E-10	NDF	IE-SCB/RRS/CL4/UNAV-DGS=3	

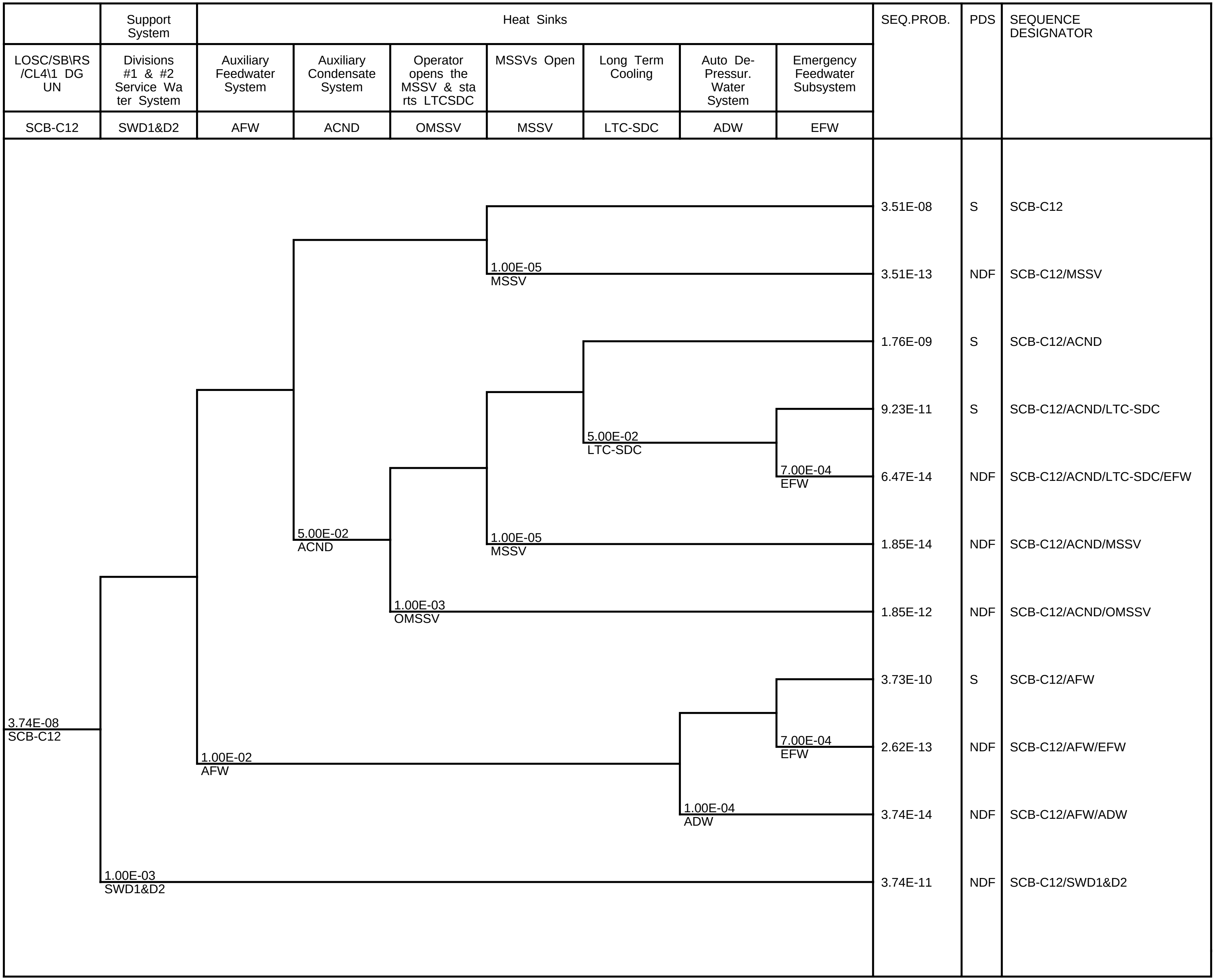


SCB-Loss of SC inventory via pipe break C:\CAFTA\TREE(ET)\SCB-C1.TRE 16-06-03

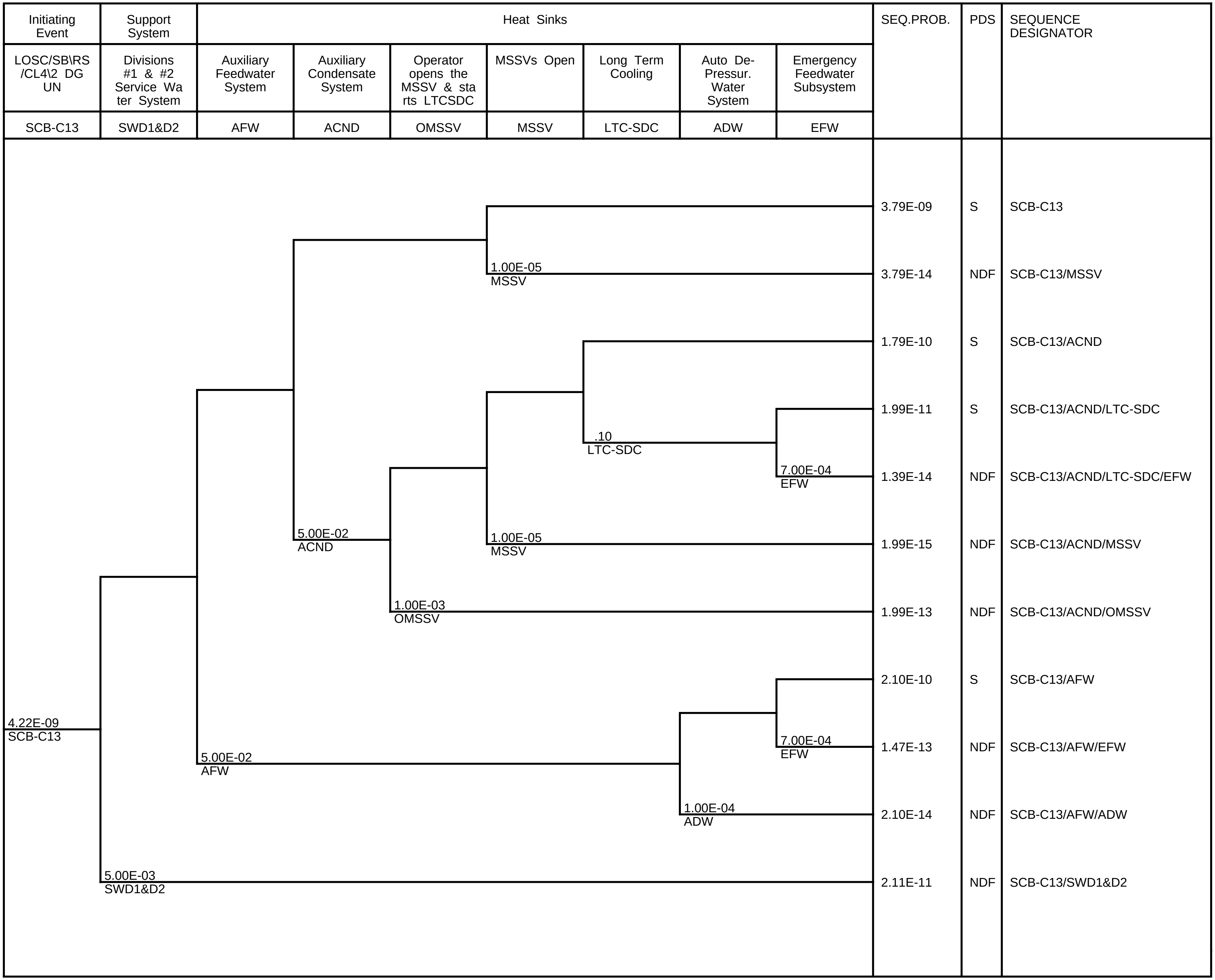
Initiating Event	Support System	Heat Sinks						
LOSC/SB\RS/CL4\ALL DGS AV	Divisions #1 & #2 Service Water System	Auxiliary Feedwater System	Auxiliary Condensate System	Operator opens the MSSV & starts LTCSDC	MSSVs Open	Long Term Cooling	Auto De-Pressur. Water System	Emergency Feedwater Subsystem
SCB-C11	SWD1&D2	AFW	ACND	OMSSV	MSSV	LTC-SDC	ADW	EFW



SEQ.PROB.	PDS	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
5.27E-07	S	SCB-C11	
5.27E-12	NDF	SCB-C11/MSSV	
2.74E-08	S	SCB-C11/ACND	
2.77E-10	NDF	SCB-C11/ACND/LTC-SDC	
2.77E-13	NDF	SCB-C11/ACND/MSSV	
2.77E-11	NDF	SCB-C11/ACND/OMSSV	
3.35E-09	S	SCB-C11/AFW	
2.34E-12	NDF	SCB-C11/AFW/EFW	
3.35E-13	NDF	SCB-C11/AFW/ADW	
5.58E-12	NDF	SCB-C11/SWD1&D2	



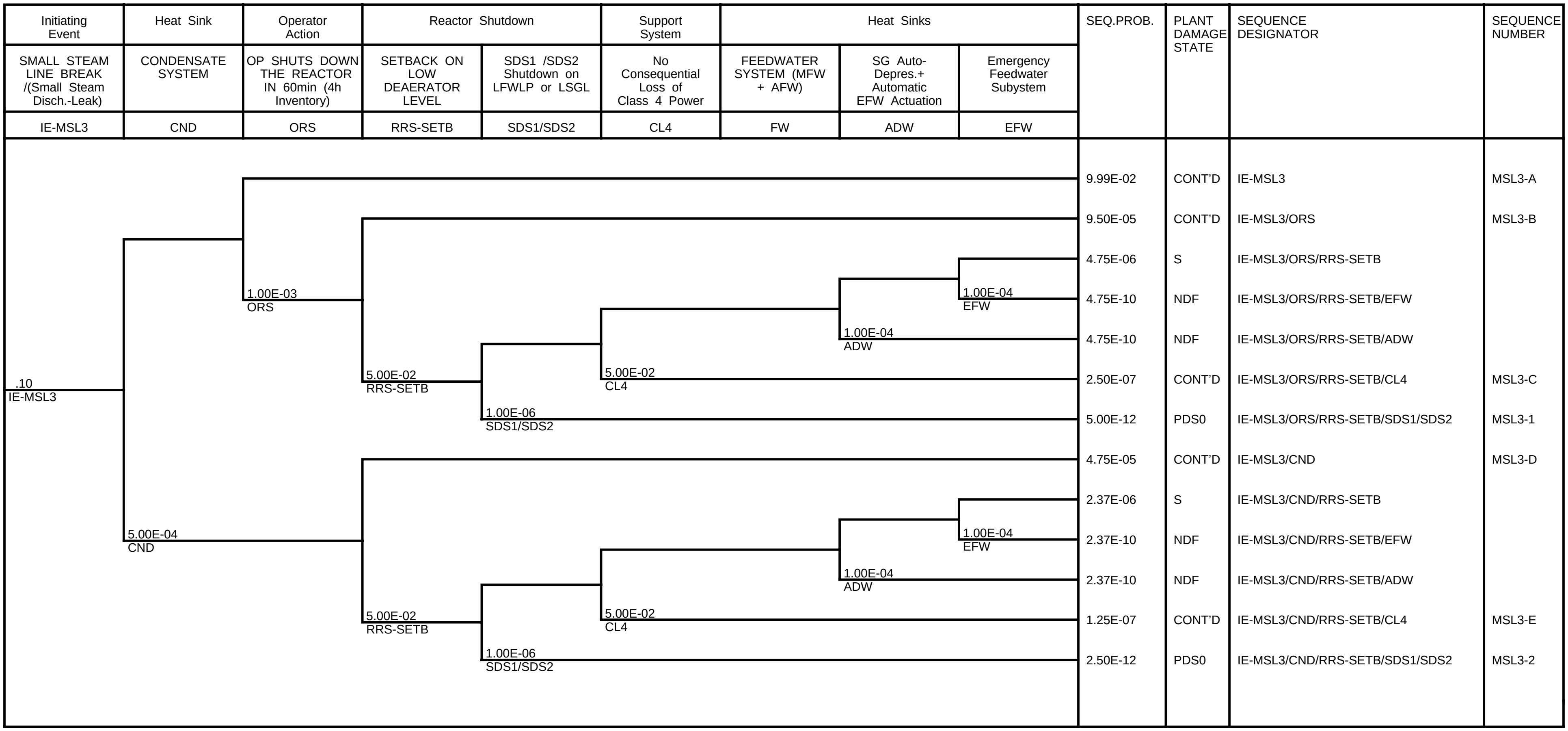
SCB-Loss of SC inventory via pipe break C:\CAFTA\TREE(ET)\SCB-C12.TRE 03-12-03



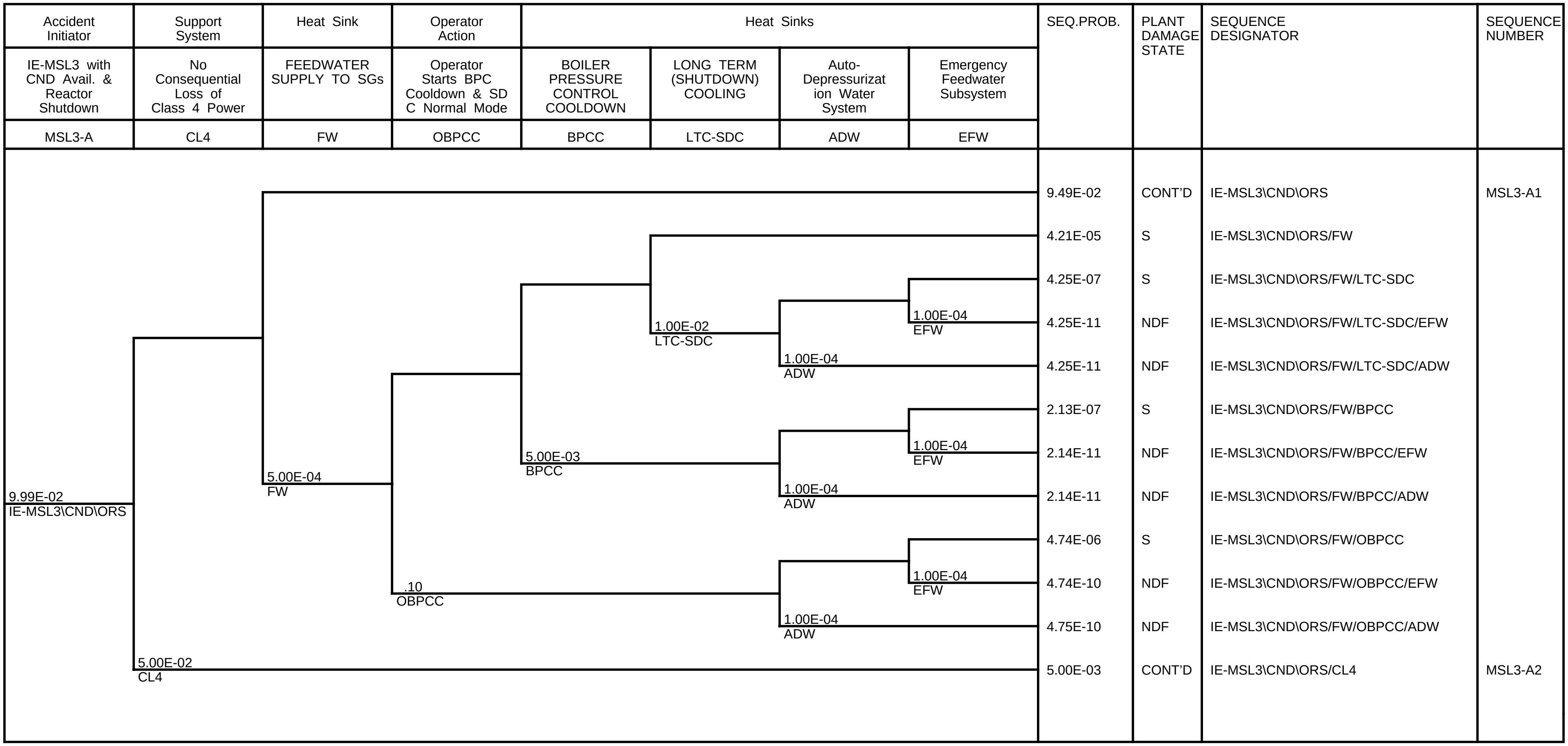
SCB-Loss of SC inventory via pipe break C:\CAFTA\TREE(ET)\SCB-C13.TRE 03-12-03

Appendix H

Event Tree for Small Steam Line Break Causing Low Level in the Deaerator

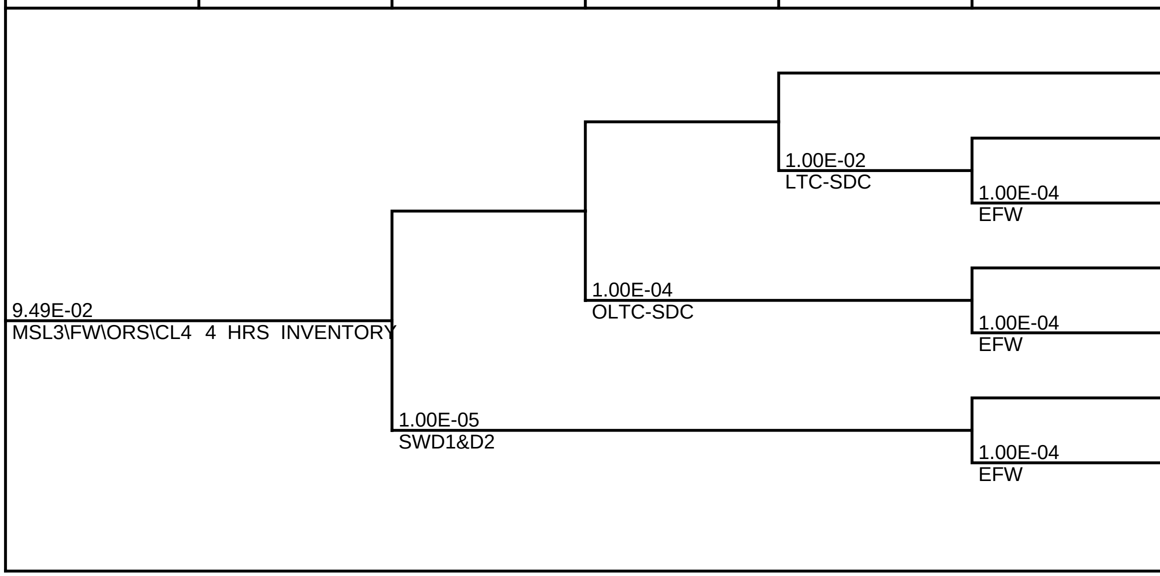


SMALL STEAM LINE BREAK CAUSING LOW DEAERATOR LEVEL C:\CAFTA\TREE(ET)\IE-MSL3.TRE 23-01-04



Main Steam Line Break (Small Discharge) C:\CAFTA\TREE(ET)\MSL3-A.TRE 23-01-04

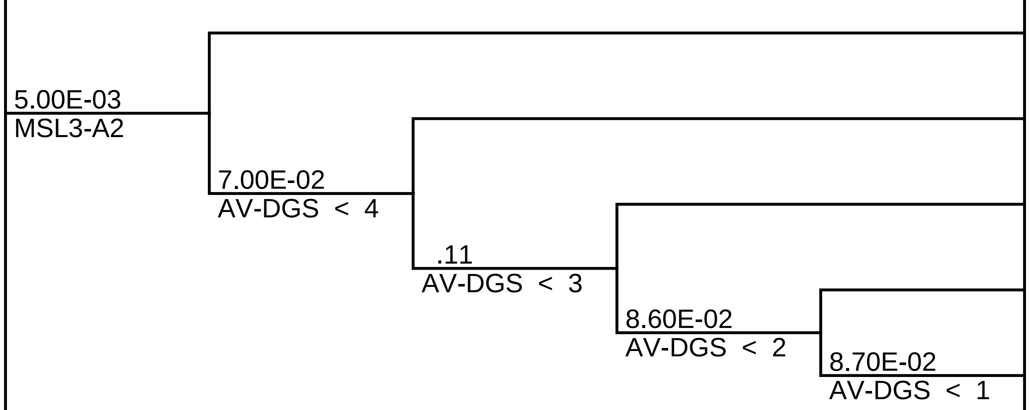
Accident Initiator	Heat Sink	Support System	Operator Action	Heat Sinks		SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
IE-MSL3 with Reactor Shutd own, CND, FW & Cl4 Availab	AUTO DE-PRESSURIZATIO N WATER SYSTEM	DIVISIONS #1 & #2 RSW / RCW SYSTEMS	OP. STARTS SDC / EFW SYSTEMS within 4h	LONG TERM (SHUTDOWN) COOLING SYSTEM	EMERGENCY FEEDWATER SUBSYSTEM				
MSL3-A1	ADW	SWD1&D2	OLTC-SDC	LTC-SDC	EFW				



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
9.39E-02	S	MSL3\FW\ORS\CL4	
9.49E-04	S	MSL3\FW\ORS\CL4/LTC-SDC	
9.49E-08	PDS6	MSL3\FW\ORS\CL4/LTC-SDC/EFW	MSL3A1-1
9.49E-06	S	MSL3\FW\ORS\CL4/OLTC-SDC	
9.49E-10	PDS6	MSL3\FW\ORS\CL4/OLTC-SDC/EFW	MSL3A1-2
9.49E-07	S	MSL3\FW\ORS\CL4/SWD1&D2	
9.49E-11	NDF	MSL3\FW\ORS\CL4/SWD1&D2/EFW	

Main Steam Line Break (Small Discharge) C:\CAFTA\TREE(ET)\MSL3-A1.TRE 22-01-04

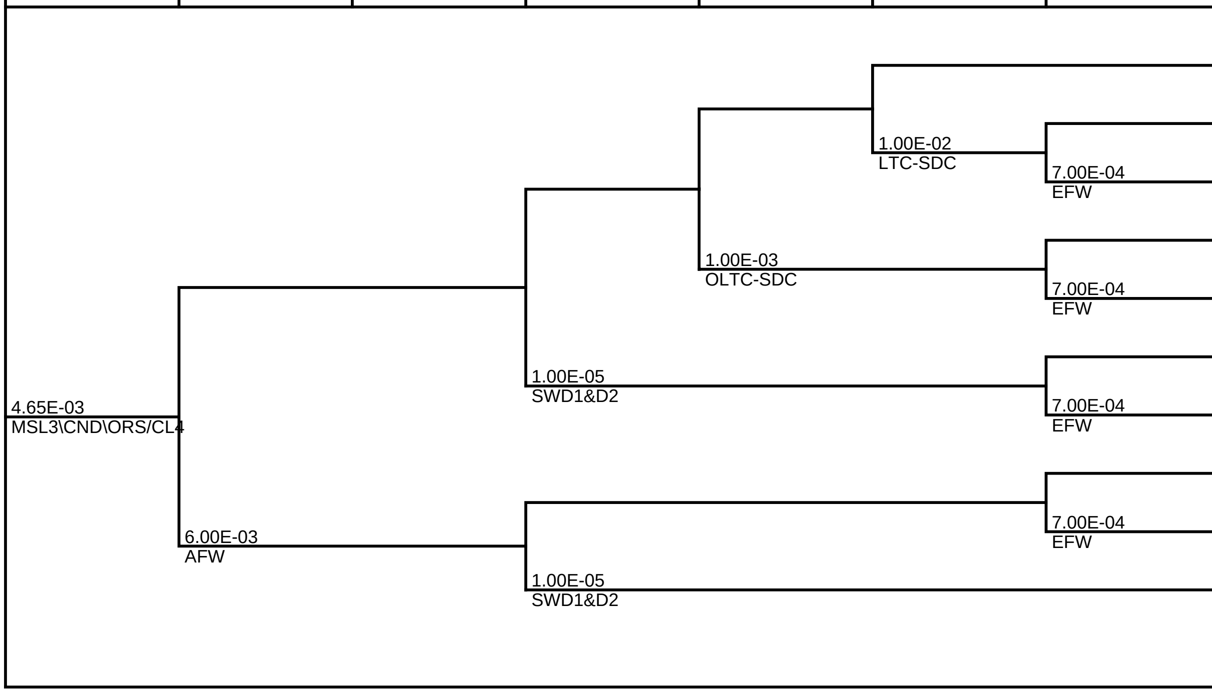
Initiating Event	Class III Power Supply				SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE NUMBER	SEQUENCE DESIGNATOR
Small Steam Line Break & Loss of CL4 (SD	4 DGs Available for 24h	3 DGs Available for 24h	2 DGs Available for 24h	1 DG Available for 24h				
IE-MSL3	DG-AV=4	DG-AV=3	DG-AV=2	DG-AV=1				



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE NUMBER	SEQUENCE DESIGNATOR
4.65E-03	CONT'D	MSL3A2-1	IE-MSL3/CL4/UNAV-DGS=0
3.11E-04	CONT'D	MSL3A2-2	IE-MSL3/CL4/UNAV-DGS=1
3.52E-05	CONT'D	MSL3A2-3	IE-MSL3/CL4/UNAV-DGS=2
3.02E-06	CONT'D	MSL3A2-4	IE-MSL3/CL4/UNAV-DGS=3
2.88E-07	CONT'D	MSL3A2-5	IE-MSL3/CL4/UNAV-DGS=4

SMALL STEAM LINE BREAK (Causing Low Deaerator Lvl) C:\CAFTA\TREE(ET)\MSL3-A2.TRE 17-06-03

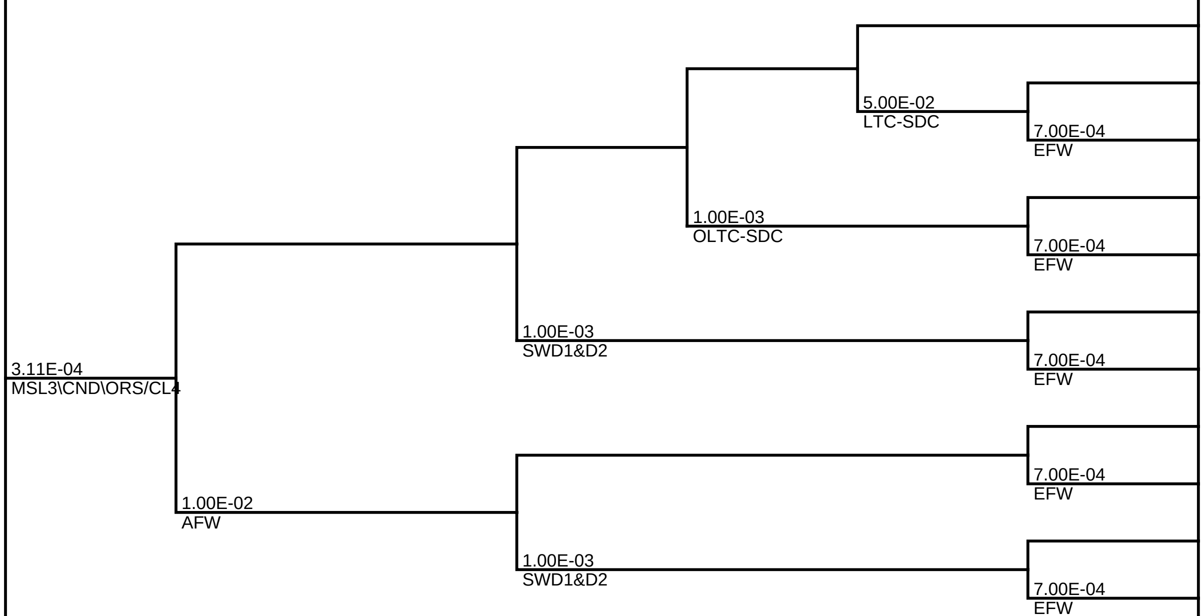
Accident Initiator	Heat Sinks			Operator Action	Heat Sinks	
IE-MSL3 - RS & CND Available *Cl4 Lost	AUXILIARY FEEDWATER SYSTEM	AUTO DE-PRESSURIZATION WATER SYSTEM	DIVISIONS #1 & #2 RSW / RCW SYSTEMS	OP. STARTS SDC /EFW SYSTEMS within 4h	LONG TERM (SHUTDOWN) COOLING SYSTEM	EMERGENCY FEEDWATER SUBSYSTEM
MSL3A2-1	AFW	ADW	SWD1&D2	OLTC-SDC	LTC-SDC	EFW



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
4.57E-03	S	MSL3\CND\ORS/CL4	
4.61E-05	S	MSL3\CND\ORS/CL4/LTC-SDC	
3.23E-08	PDS6	MSL3\CND\ORS/CL4/LTC-SDC/EFW	MSL3A2-11
4.62E-06	S	MSL3\CND\ORS/CL4/OLTC-SDC	
3.24E-09	PDS6	MSL3\CND\ORS/CL4/OLTC-SDC/EFW	MSL3A2-12
4.62E-08	S	MSL3\CND\ORS/CL4/SWD1&D2	
3.24E-11	NDF	MSL3\CND\ORS/CL4/SWD1&D2/EFW	
2.79E-05	S	MSL3\CND\ORS/CL4/AFW	
1.95E-08	PDS6	MSL3\CND\ORS/CL4/AFW/EFW	MSL3A2-13
2.79E-10	NDF	MSL3\CND\ORS/CL4/AFW/SWD1&D2	

Main Steam Line Break (Small Discharge) C:\CAFTA\TREE(ET)\MSL3A2-1.TRE 22-01-04

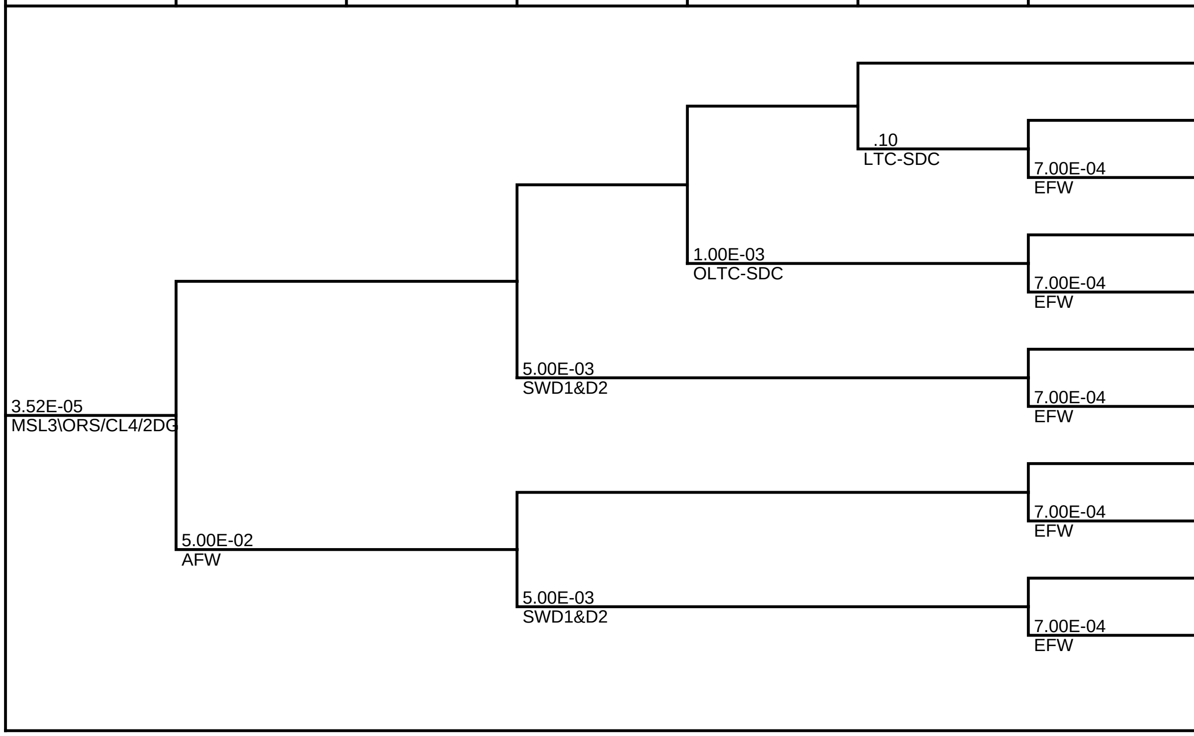
Accident Initiator	Heat Sinks		Support System	Operator Action	Heat Sinks	
IE-MSL3 - RS & CND Availab le *Cl4 & 1 DG Lost	AUXILIARY FEEDWATER SYSTEM	AUTO DE-PRESSURIZATIO N WATER SYSTEM	DIVISIONS #1 & #2 RSW / RCW SYSTEMS	OP. STARTS SDC /EFW SYSTEMS AFTER 4h	SHUTDOWN COOLING SYSTEM	EMERGENCY FEEDWATER SUBSYSTEM
MSL3A2-2	AFW	ADW	SWD1&D2	OLTC-SDC	LTC-SDC	EFW



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
2.92E-04	S	MSL3\CND\ORS/CL4	
1.54E-05	S	MSL3\CND\ORS/CL4/LTC-SDC	
1.08E-08	PDS6	MSL3\CND\ORS/CL4/LTC-SDC/EFW	MSL3A2-21
3.07E-07	S	MSL3\CND\ORS/CL4/OLTC-SDC	
2.15E-10	NDF	MSL3\CND\ORS/CL4/OLTC-SDC/EFW	
3.08E-07	S	MSL3\CND\ORS/CL4/SWD1&D2	
2.16E-10	NDF	MSL3\CND\ORS/CL4/SWD1&D2/EFW	
3.10E-06	S	MSL3\CND\ORS/CL4/AFW	
2.17E-09	PDS6	MSL3\CND\ORS/CL4/AFW/EFW	MSL3A2-22
3.11E-09	S	MSL3\CND\ORS/CL4/AFW/SWD1&D2	
2.18E-12	NDF	MSL3\CND\ORS/CL4/AFW/SWD1&D2/EFW	

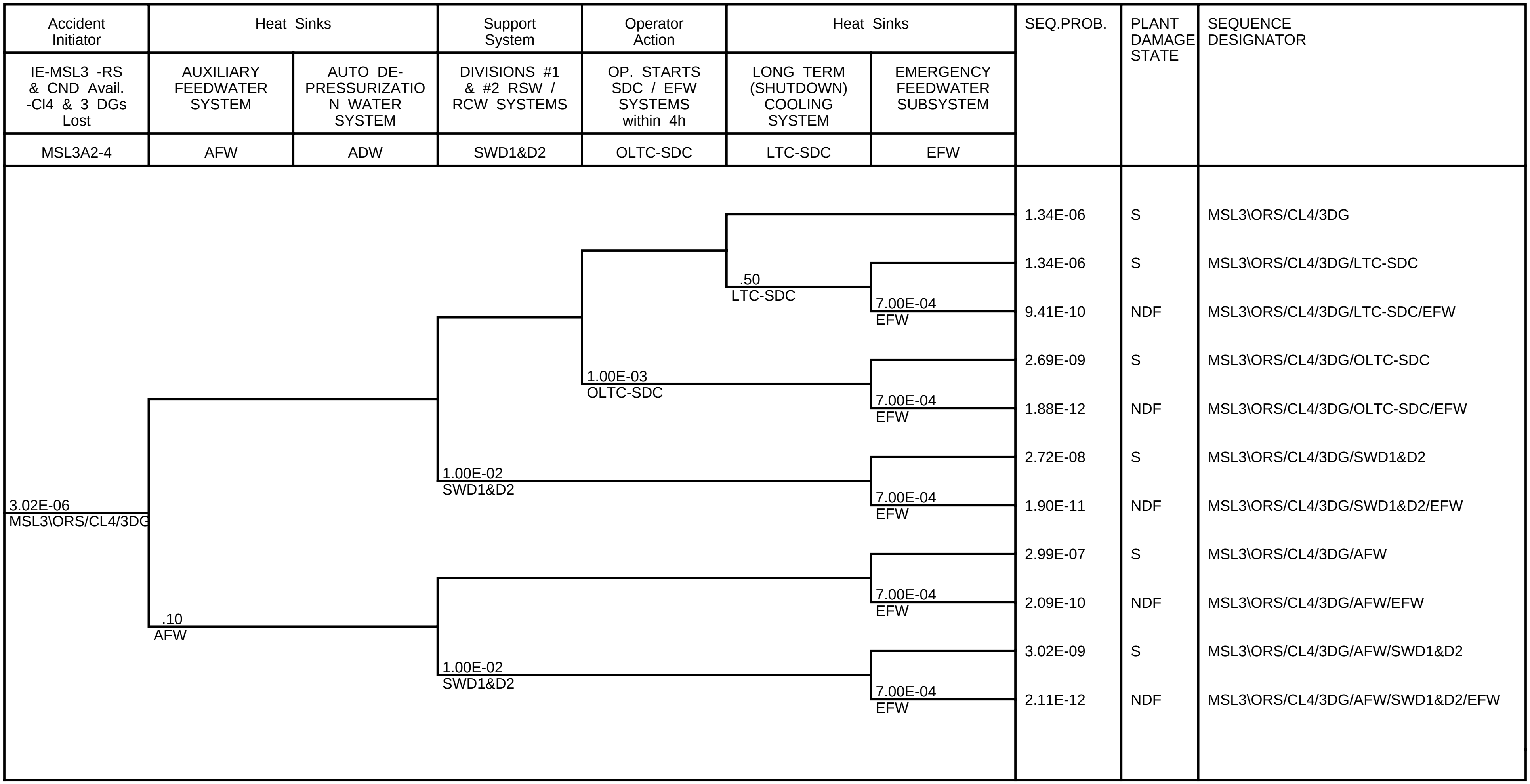
Main Steam Line Break (Small Discharge) C:\CAFTA\TREE(ET)\MSL3A2-2.TRE 22-01-04

Accident Initiator	Heat Sinks		Support System	Operator Action	Heat Sinks	
IE-MSL3 -RS & CND Avail *-Cl4 & 2 DGs Lost	AUX. FEEDWATER SYSTEM	AUTO DE-PRESSURIZATIO N WATER SYSTEM	DIVISIONS #1 & #2 RSW / RCW SYSTEMS	OP. STARTS SDC / EFW SYSTEMS AFTER 4h	SHUTDOWN COOLING SYSTEM	EMERGENCY FEEDWATER SUBSYSTEM
MSL3A2-3	AFW	ADW	SWD1&D2	OLTC-SDC	LTC-SDC	EFW



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
2.99E-05	S	MSL3\ORS/CL4/2DG	
3.32E-06	S	MSL3\ORS/CL4/2DG/LTC-SDC	
2.33E-09	PDS6	MSL3\ORS/CL4/2DG/LTC-SDC/EFW	MSL3A2-31
3.32E-08	S	MSL3\ORS/CL4/2DG/OLTC-SDC	
2.33E-11	NDF	MSL3\ORS/CL4/2DG/OLTC-SDC/EFW	
1.67E-07	S	MSL3\ORS/CL4/2DG/SWD1&D2	
1.17E-10	NDF	MSL3\ORS/CL4/2DG/SWD1&D2/EFW	
1.75E-06	S	MSL3\ORS/CL4/2DG/AFW	
1.23E-09	PDS6	MSL3\ORS/CL4/2DG/AFW/EFW	MSL3A2-32
8.79E-09	S	MSL3\ORS/CL4/2DG/AFW/SWD1&D2	
6.16E-12	NDF	MSL3\ORS/CL4/2DG/AFW/SWD1&D2/EFW	

Main Steam Line Break (Small Discharge) C:\CAFTA\TREE(ET)\MSL3A2-3.TRE 22-01-04

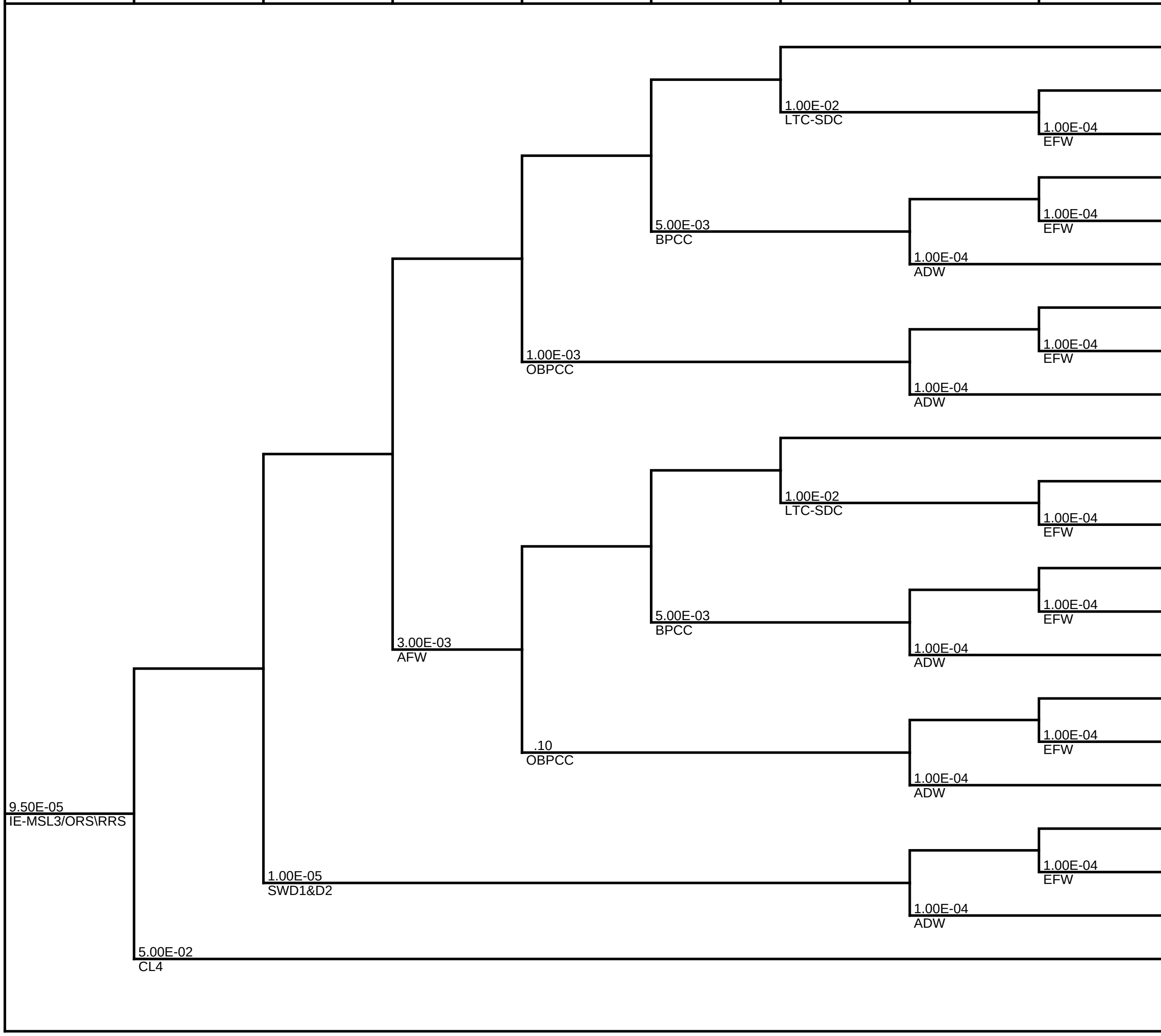


Main Steam Line Break (Small Discharge) C:\CAFTA\TREE(ET)\MSL3A2-4.TRE 22-01-04

Accident Initiator	Heat Sinks		Support System	Operator Action	Heat Sinks				SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
IE-MSL3 -RS & CND Avail. -*Cl4 & 4 DG Lost	AUXILIARY FEEDWATER SYSTEM	AUTO DE-PRESSURIZATION WATER SYSTEM	DIVISIONS #1 & #2 RSW / RCW SYSTEMS	OP. STARTS SDC / EFW SYSTEMS within 4h	LONG TERM (SHUTDOWN) COOLING SYSTEM	EMERGENCY FEEDWATER SUBSYSTEM	EMERGENCY CORE COOLING (DORMANT)	MODERATOR AS A HEAT SINK				
MSL3A2-5	AFW	ADW	SWD1&D2	OLTC-SDC	LTC-SDC	EFW	DECC	MHS				
2.88E-07 MSL3\ORS/CL4/4DG									2.88E-07	S	MSL3\ORS/CL4/4DG	
						7.00E-04 EFW			2.02E-10	NDF	MSL3\ORS/CL4/4DG/EFW	

Main Steam Line Break (Small Discharge) C:\CAFTA\TREE(ET)\MSL3A2-5.TRE 17-06-03

Accident Initiator	Support Systems		Heat Sinks	Operator Action	Heat Sinks		Operator Action	Heat Sink
IE-MSL3 with CND Avail. & Reactor Shutdown by Setbac	Consequential Loss of Class 4 Power	DIVISIONS #1 & #2 RSW / RCW SYSTEMS	AUX. FEEDWATER SUPPLY TO SGs	OP. STARTS BP. CNTRL. COOLDOWN /SD COOLING	BOILER PRESS. CONTROL COOLDOWN PROGRAM	LONG TERM SHUTDOWN COOLING SYSTEM	AUTO DE-PRESSURIZATION WATER SYSTEM	EMERGENCY FEEDWATER SUBSYSTEM
MSL3-B	CL4	SWD1&D2	AFW	OBPCC	BPCC	LTC-SDC	ADW	EFW

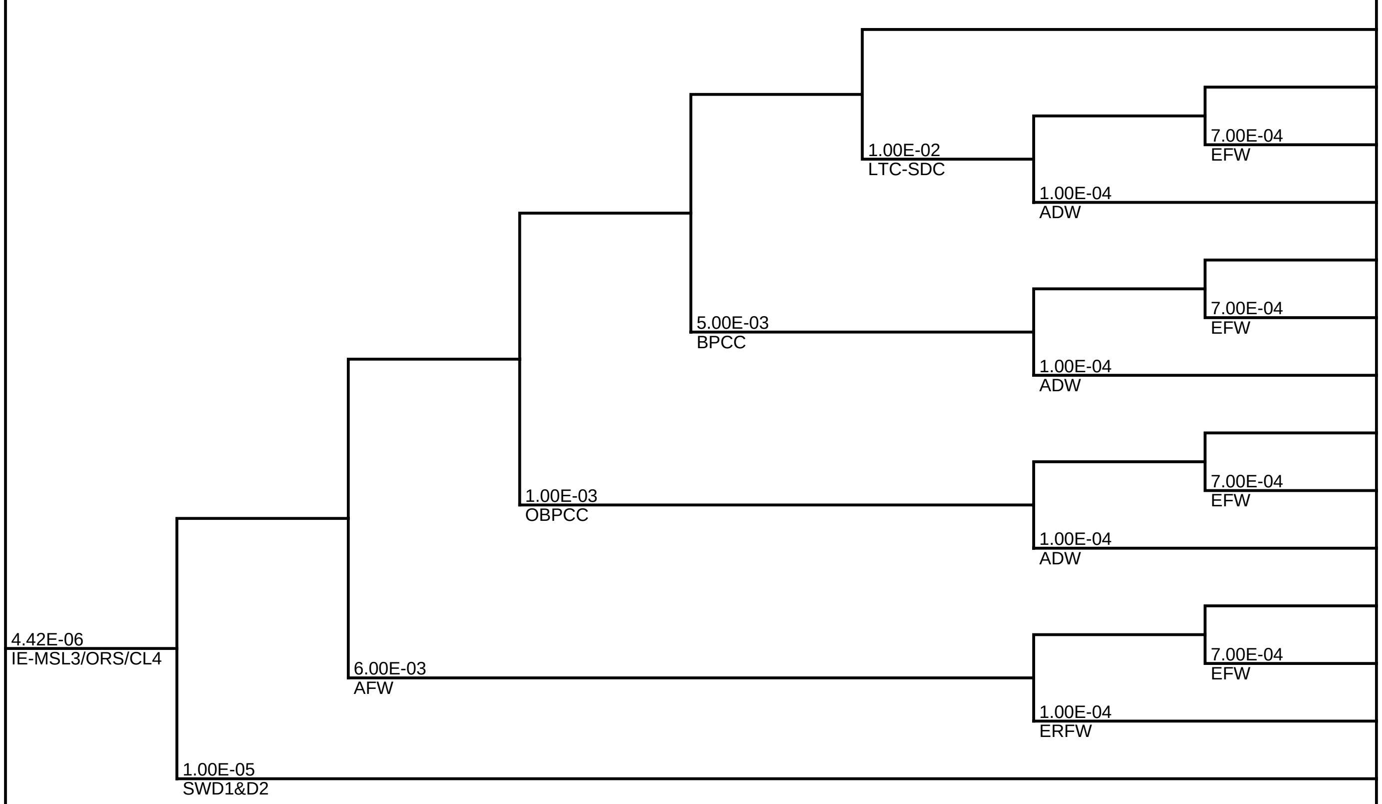


SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
8.85E-05	S	IE-MSL3/ORS\RRS	
8.94E-07	S	IE-MSL3/ORS\RRS/LTC-SDC	
8.94E-11	NDF	IE-MSL3/ORS\RRS/LTC-SDC/EFW	
4.49E-07	S	IE-MSL3/ORS\RRS/BPCC	
4.49E-11	NDF	IE-MSL3/ORS\RRS/BPCC/EFW	
4.49E-11	NDF	IE-MSL3/ORS\RRS/BPCC/ADW	
9.00E-08	S	IE-MSL3/ORS\RRS/OBPCC	
9.00E-12	NDF	IE-MSL3/ORS\RRS/OBPCC/EFW	
9.00E-12	NDF	IE-MSL3/ORS\RRS/OBPCC/ADW	
2.40E-07	S	IE-MSL3/ORS\RRS/AFW	
2.42E-09	S	IE-MSL3/ORS\RRS/AFW/LTC-SDC	
2.42E-13	NDF	IE-MSL3/ORS\RRS/AFW/LTC-SDC/EFW	
1.22E-09	S	IE-MSL3/ORS\RRS/AFW/BPCC	
1.22E-13	NDF	IE-MSL3/ORS\RRS/AFW/BPCC/EFW	
1.22E-13	NDF	IE-MSL3/ORS\RRS/AFW/BPCC/ADW	
2.71E-08	S	IE-MSL3/ORS\RRS/AFW/OBPCC	
2.71E-12	NDF	IE-MSL3/ORS\RRS/AFW/OBPCC/EFW	
2.71E-12	NDF	IE-MSL3/ORS\RRS/AFW/OBPCC/ADW	
9.02E-10	S	IE-MSL3/ORS\RRS/SWD1&D2	
9.02E-14	NDF	IE-MSL3/ORS\RRS/SWD1&D2/EFW	
9.02E-14	NDF	IE-MSL3/ORS\RRS/SWD1&D2/ADW	
4.75E-06	CONT'D	IE-MSL3/ORS\RRS/CL4	MSL3-B1

Steam Line Failure & Loss of C l4 Power (	4 DGs Available	3 DGs Available	2 DGs Available	1 DG Available	SEQ.PROB.	Plant Damage State	SEQUENCE NUMBER	SEQUENCE DESIGNATOR
IE-MSL3	DG-AV = 4	DG-AV = 3	DG-AV = 2	DG-AV = 1				
					4.42E-06	CONT'D	MSL3B1-1	IE-MSL3/CL4/UNAV-DGS=0
4.75E-06 MSL3-B1					2.96E-07	CONT'D	MSL3B1-2	IE-MSL3/CL4/UNAV-DGS=1
	7.00E-02 AV-DGS < 4				3.34E-08	CONT'D	MSL3B1-3	IE-MSL3/CL4/UNAV-DGS=2
		.11 AV-DGS < 3			2.87E-09	NDF		IE-MSL3/CL4/UNAV-DGS=3
			8.60E-02 AV-DGS < 2	8.70E-02 AV-DGS < 1	2.74E-10	NDF		IE-MSL3/CL4/UNAV-DGS=4

Main Steam Line Break (Small Discharge Rate) C:\CAFTA\TREE(ET)\MSL3-B1.TRE 17-06-03

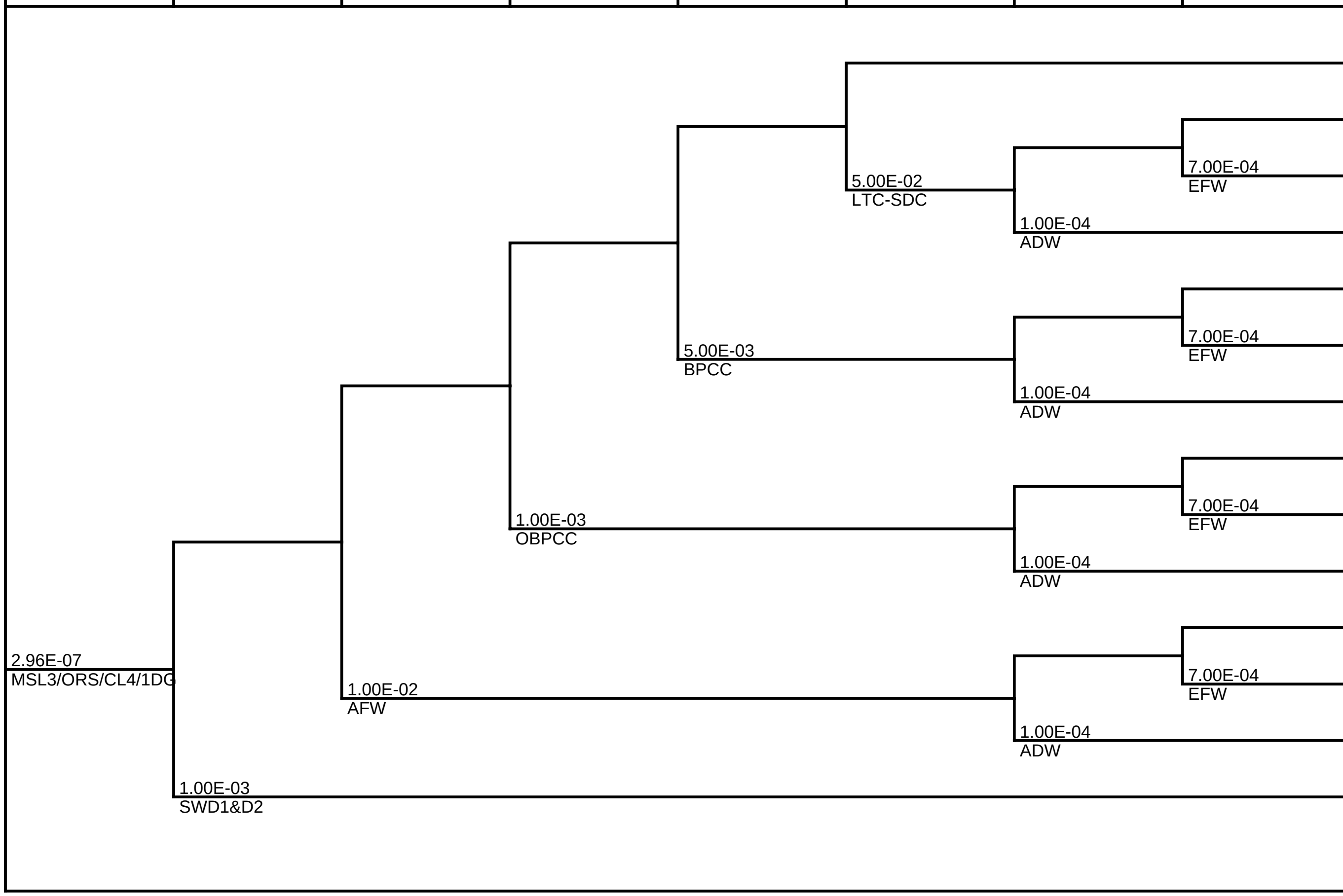
Accident Initiator	Support Systems	Heat Sinks	Operator Action	Heat Sinks			
IE-MSL3 with CND Avail & RS by Setback (CL4 Lost)	DIVISIONS #1 & #2 RSW / RCW SYSTEMS	AUX. FEEDWATER SUPPLY TO SGs	OP. STARTS SG PRES. CNTRL COOLDOWN & SDC SYSTEM	BOILER PRESS. CONTROL COOLDOWN PROGRAM	LONG TERM (SHUTDOWN) COOLING SYSTEM	AUTO DE-PRESSURIZATION WATER SYSTEM	EMERGENCY FEEDWATER SUB-SYSTEM
MSL3B1-1	SWD1&D2	AFW	OBPCC	BPCC	LTC-SDC	ADW	EFW



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR
4.32E-06	S	IE-MSL3/ORS/CL4
4.36E-08	S	IE-MSL3/ORS/CL4/LTC-SDC
3.06E-11	NDF	IE-MSL3/ORS/CL4/LTC-SDC/EFW
4.37E-12	NDF	IE-MSL3/ORS/CL4/LTC-SDC/ADW
2.19E-08	S	IE-MSL3/ORS/CL4/BPCC
1.54E-11	S	IE-MSL3/ORS/CL4/BPCC/EFW
2.19E-12	NDF	IE-MSL3/ORS/CL4/BPCC/ADW
4.39E-09	S	IE-MSL3/ORS/CL4/OBPCC
3.08E-12	NDF	IE-MSL3/ORS/CL4/OBPCC/EFW
4.39E-13	NDF	IE-MSL3/ORS/CL4/OBPCC/ADW
2.65E-08	S	IE-MSL3/ORS/CL4/AFW
1.86E-11	NDF	IE-MSL3/ORS/CL4/AFW/EFW
2.65E-12	NDF	IE-MSL3/ORS/CL4/AFW/ERFW
4.42E-11	NDF	IE-MSL3/ORS/CL4/SWD1&D2

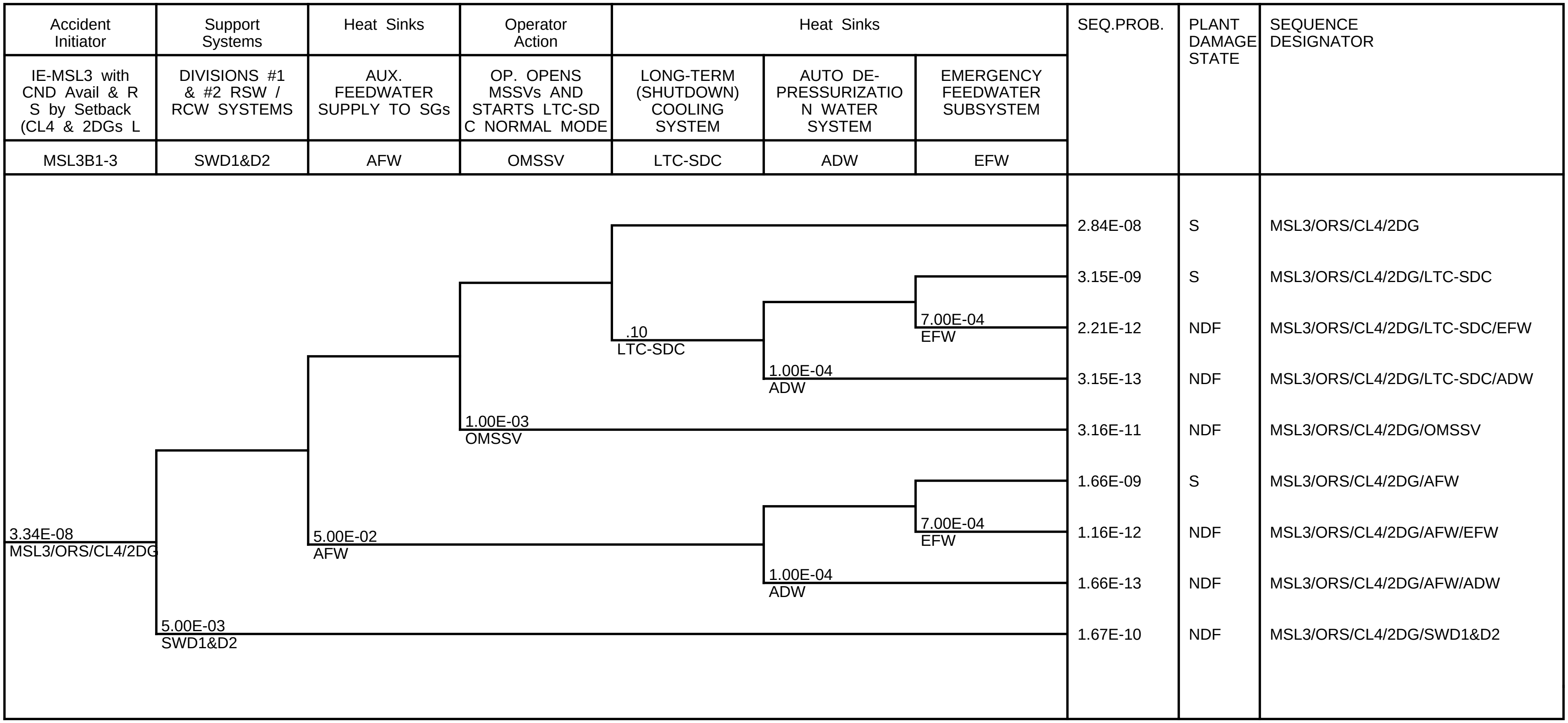
MAIN STEAM LINE BREAK CAUSING DEAERATOR LOW LEVEL C:\CAFTA\TREE(ET)\MSL3B1-1.TRE 17-06-03

Accident Initiator	Support Systems	Heat Sinks	Operator Action	Heat Sinks				SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR
IE-MSL3 with CND Avail & R S by Setback (CL4 & 1DG Lo	DIVISIONS #1 & #2 RSW / RCW SYSTEMS	AUX. FEEDWATER SUPPLY TO SGs	OP. STARTS SG PRESS. CNTRL COOLDOWN & S D COOLING SYS	BOILER PRESS. CONTROL COOLD OWN PROGRAM	LONG TERM (SHUTDOWN) COOLING SYSTEM	AUTO DE-PRESSURIZATIO N WATER SYSTEM	EMERGENCY FEEDWATER SUBSYSTEM			
MSL3B1-2	SWD1&D2	AFW	OBPCC	BPCC	LTC-SDC	ADW	EFW			



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR
2.76E-07	S	MSL3/ORS/CL4/1DG
1.45E-08	S	MSL3/ORS/CL4/1DG/LTC-SDC
1.02E-11	NDF	MSL3/ORS/CL4/1DG/LTC-SDC/EFW
1.45E-12	NDF	MSL3/ORS/CL4/1DG/LTC-SDC/ADW
1.46E-09	S	MSL3/ORS/CL4/1DG/BPCC
1.02E-12	NDF	MSL3/ORS/CL4/1DG/BPCC/EFW
1.46E-13	NDF	MSL3/ORS/CL4/1DG/BPCC/ADW
2.93E-10	S	MSL3/ORS/CL4/1DG/OBPCC
2.05E-13	NDF	MSL3/ORS/CL4/1DG/OBPCC/EFW
2.93E-14	NDF	MSL3/ORS/CL4/1DG/OBPCC/ADW
2.95E-09	S	MSL3/ORS/CL4/1DG/AFW
2.07E-12	NDF	MSL3/ORS/CL4/1DG/AFW/EFW
2.96E-13	NDF	MSL3/ORS/CL4/1DG/AFW/ADW
2.96E-10	NDF	MSL3/ORS/CL4/1DG/SWD1&D2

MAIN STEAM LINE BREAK CAUSING DEAERATOR LOW LEVEL C:\CAFTA\TREE(ET)\MSL3B1-2.TRE 17-06-03

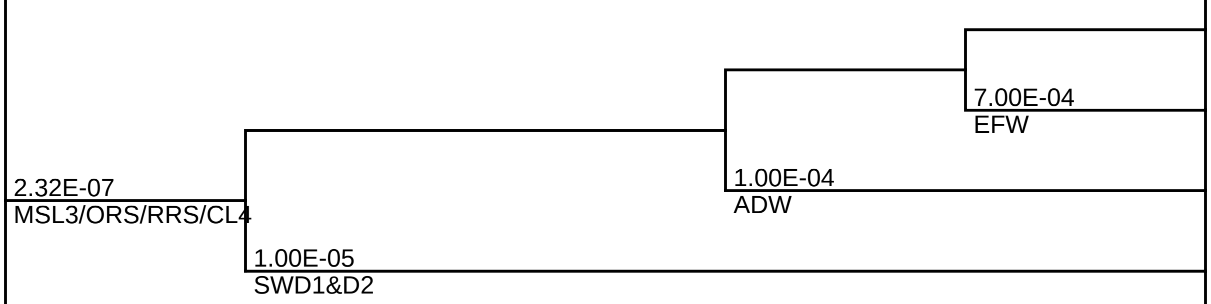


MAIN STEAM LINE BREAK CAUSING DEAERATOR LOW LEVEL C:\CAFTA\TREE(ET)\MSL3B1-3.TRE 10-06-03

Initiating Event	Class III Power Supply				SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE NUMBER	SEQUENCE DESIGNATOR
Small Steam Line Break & Loss of CL4 (SD	4 DGs Available for 24h	3 DGs Available for 24h	2 DGs Available for 24h	1 DG Available for 24h				
IE-MSL3	DG-AV=4	DG-AV=3	DG-AV=2	DG-AV=1				
2.50E-07 MSL3-C					2.32E-07	CONT'D	MSL3C-1	IE-MSL3/CL4/UNAV-DGS=0
	7.00E-02 AV-DGS < 4				1.56E-08	CONT'D	MSL3C-2	IE-MSL3/CL4/UNAV-DGS=1
		.11 AV-DGS < 3			1.76E-09	NDF		IE-MSL3/CL4/UNAV-DGS=2
			8.60E-02 AV-DGS < 2		1.51E-10	NDF		IE-MSL3/CL4/UNAV-DGS=3
				8.70E-02 AV-DGS < 1	1.44E-11	NDF		IE-MSL3/CL4/UNAV-DGS=4

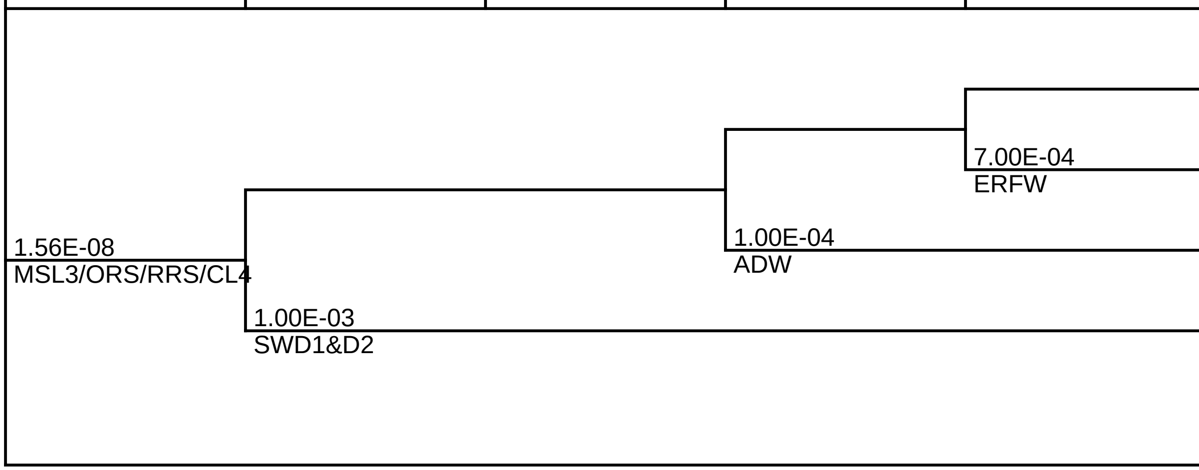
SMALL STEAM LINE BREAK (Causing Low Deaerator Lvl) C:\CAFTA\TREE(ET)\MSL3-C.TRE 17-06-03

Accident Initiator	Support System	Heat Sinks			SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR
IE-MSL3 +Loss of FW Inventory (RS by SDS1/2 & LOCl4)	DIVISIONS #1 & #2 RSW / RCW SYSTEMS	AUX. FEEDWATER SUPPLY TO SGs	AUTO DE-PRESSURISATION WATER SYSTEM	Emergency Feedwater Sub-System			
MSL3C-1	SWD1&D2	AFW	ADW	EFW			
					2.32E-07	S	MSL3/ORS/RRS/CL4
					1.62E-10	NDF	MSL3/ORS/RRS/CL4/EFW
					2.32E-11	NDF	MSL3/ORS/RRS/CL4/ADW
					2.32E-12	NDF	MSL3/ORS/RRS/CL4/SWD1&D2



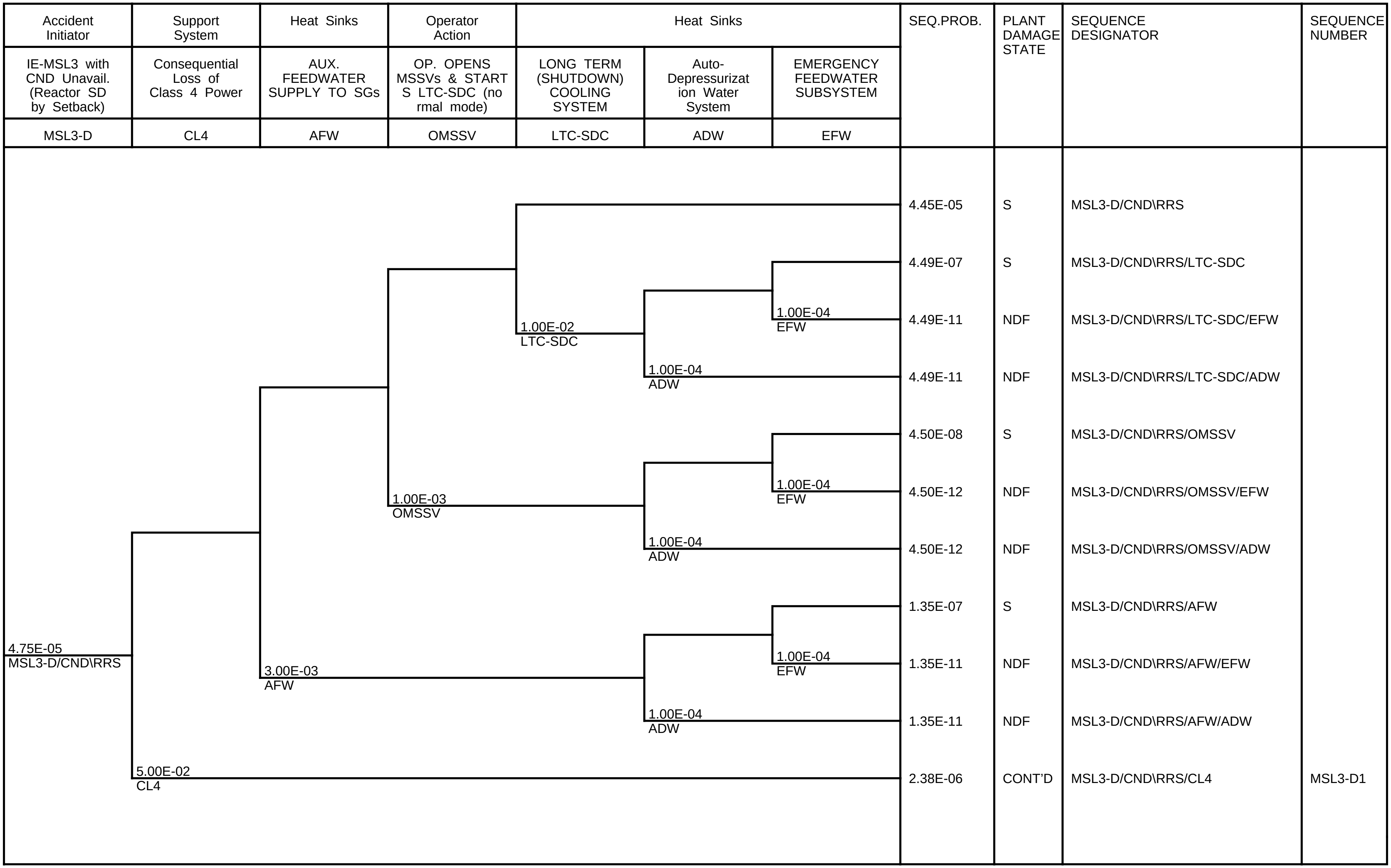
MAIN STEAM LINE BREAK (Causing Low Deaerator Lvl.)   C:\CAFTA\TREE(ET)\MSL3C-1.TRE   10-06-03

Accident Initiator	Heat Sinks				SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR
IE-MSL3 - Loss of Cl4 & 1 DG (RS by SDS1/SDS2)	DIVISIONS #1 & #2 RSW / RCW SYSTEMS	AUX. FEEDWATER SUPPLY TO SGs	AUTO-DEPRESSURISATION WATER SYS.	Emergency Reserve Feedwater System			
MSL3C-2	SWD1&D2	AFW	ADW	ERFW			

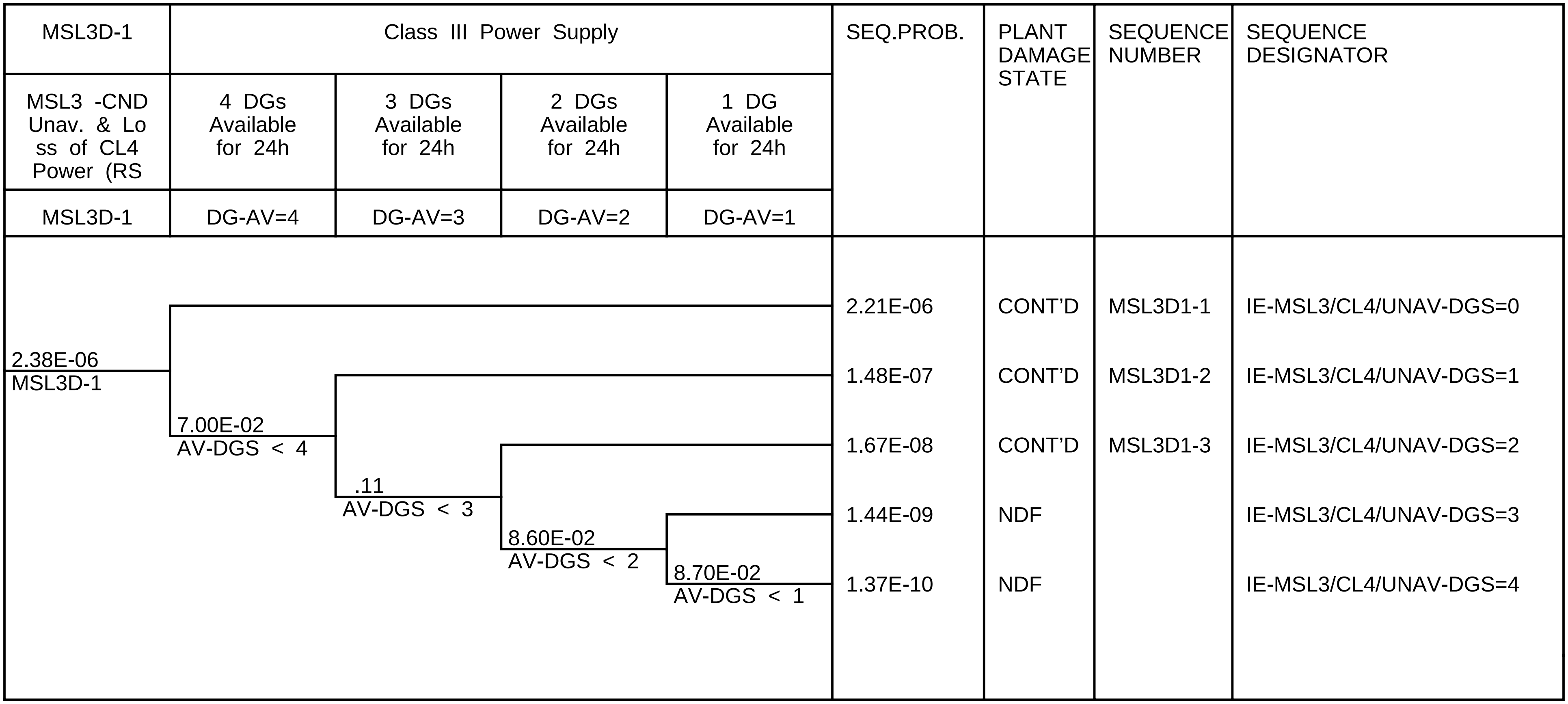


SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR
1.56E-08	S	MSL3/ORS/RRS/CL4
1.09E-11	NDF	MSL3/ORS/RRS/CL4/ERFW
1.56E-12	NDF	MSL3/ORS/RRS/CL4/ADW
1.56E-11	NDF	MSL3/ORS/RRS/CL4/SWD1&D2

MAIN STEAM LINE BREAK (Causing Low Deaerator Lvl.) C:\CAFTA\TREE(ET)\MSL3C-2.TRE 17-06-03

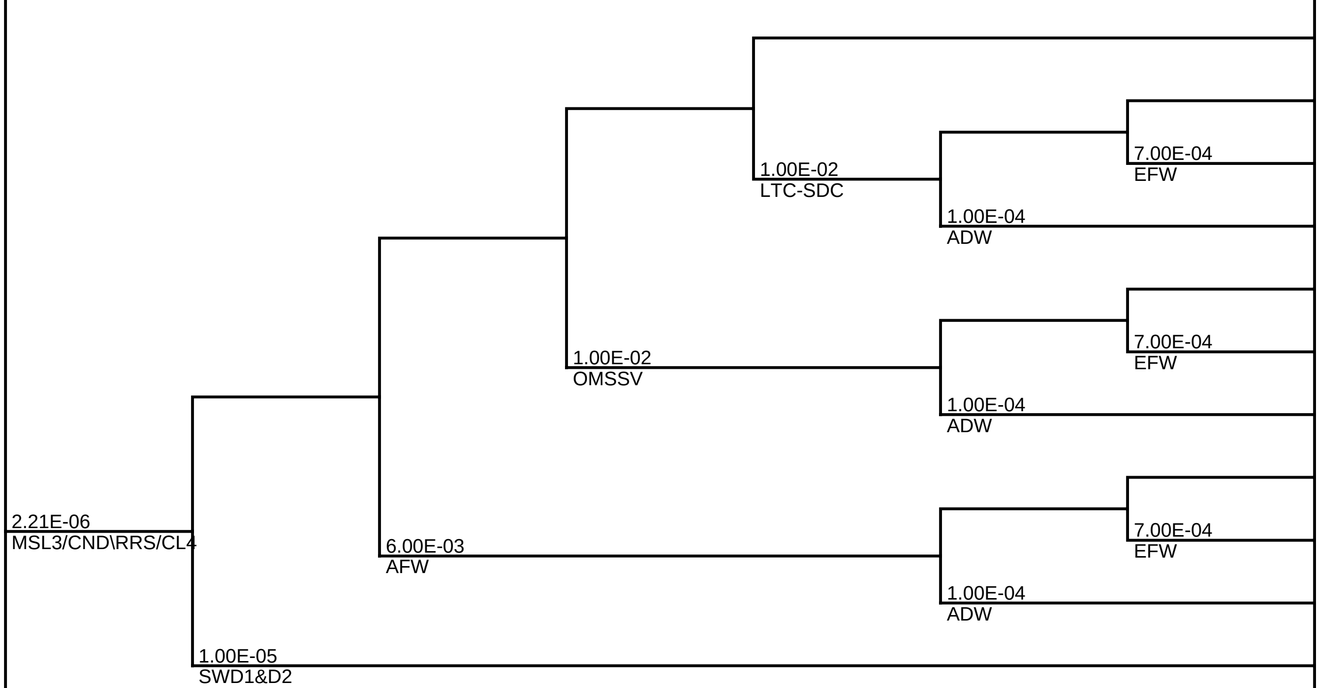


MAIN STEAM LINE BREAK (With Low Deaerator Level) C:\CAFTA\TREE(ET)\MSL3-D.TRE 10-06-03



SMALL STEAM LINE BREAK (Causing Low Deaerator Lvl) C:\CAFTA\TREE(ET)\MSL3D-1.TRE 17-06-03

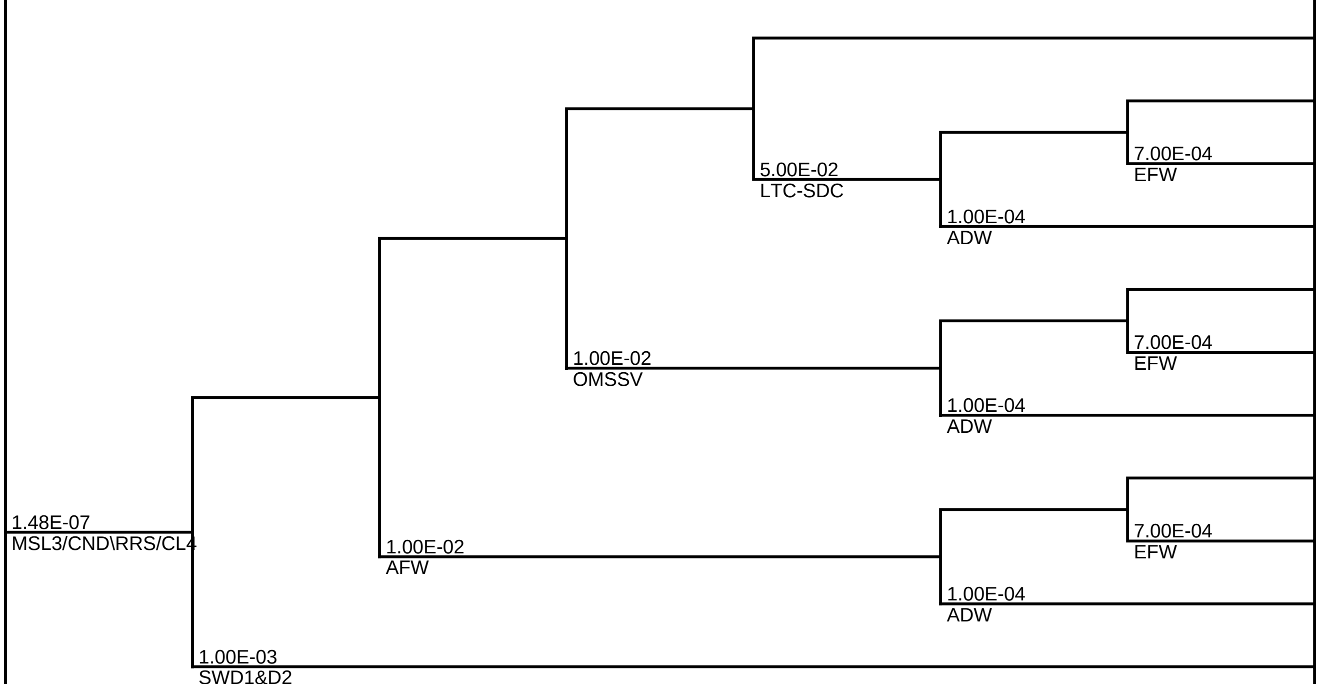
Accident Initiator	Heat Sinks		Operator Action	Heat Sinks			SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR
IE-MSL3 + Loss of CND (RS by Setback & Loss of CL4)	DIVISIONS #1 & #2 RSW / RCW SYSTEMS	AUX. FEEDWATER SUPPLY TO SGs	OP. OPENS MSSVs & STARTS LTC-SDC (normal mode)	LONG TERM (SHUTDOWN) COOLING SYSTEM	AUTO-DEPRESSURISATION WATER SYS.	Emergency Feedwater Sub-System			
MSL3D1-1	SWD1&D2	AFW	OMSSV	LTC-SDC	ADW	EFW			



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR
2.15E-06	S	MSL3/CND\RRS/CL4
2.17E-08	S	MSL3/CND\RRS/CL4/LTC-SDC
1.52E-11	NDF	MSL3/CND\RRS/CL4/LTC-SDC/EFW
2.17E-12	NDF	MSL3/CND\RRS/CL4/LTC-SDC/ADW
2.19E-08	S	MSL3/CND\RRS/CL4/OMSSV
1.54E-11	NDF	MSL3/CND\RRS/CL4/OMSSV/EFW
2.20E-12	NDF	MSL3/CND\RRS/CL4/OMSSV/ADW
1.32E-08	S	MSL3/CND\RRS/CL4/AFW
9.28E-12	NDF	MSL3/CND\RRS/CL4/AFW/EFW
1.33E-12	NDF	MSL3/CND\RRS/CL4/AFW/ADW
2.21E-11	NDF	MSL3/CND\RRS/CL4/SWD1&D2

SMALL STEAM LINE BREAK CAUSING LOW DEAERATOR LEVEL C:\CAFTA\TREE(ET)\MSL3D1-1.TRE 17-06-03

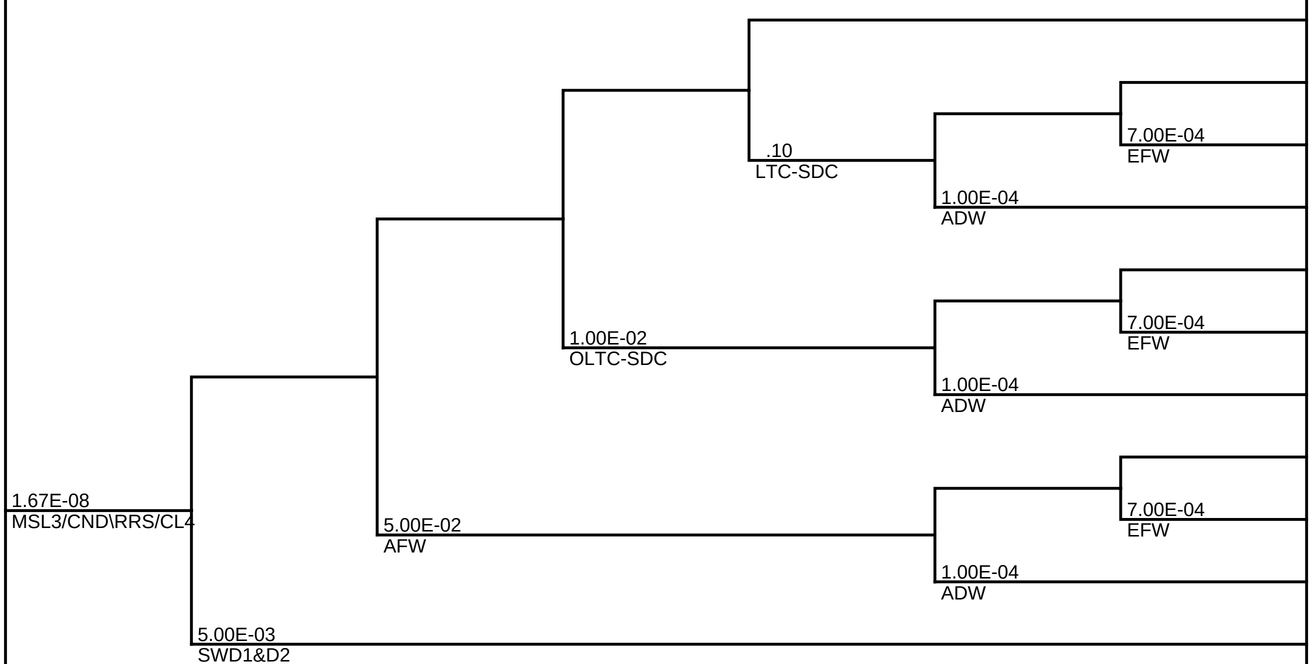
Accident Initiator	Support System	Heat Sinks	Operator Action	Heat Sinks			SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR
IE-MSL3 + Loss of CND, Cl4 & 1 DG	DIVISIONS #1 & #2 RSW / RCW SYSTEMS	AUX. FEEDWATER SUPPLY TO SGs	OP. OPENS MSSVs & START S LTC-SDC (no rmal mode)	LONG TERM (SHUTDOWN) COOLING SYSTEM	AUTO-DEPRESSURISAT ION WATER SYS.	Emergency Feedwater Subsystem			
MSL3D1-2	SWD1&D2	AFW	OMSSV	LTC-SDC	ADW	EFW			



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR
1.38E-07	S	MSL3/CND\RRS/CL4
7.24E-09	S	MSL3/CND\RRS/CL4/LTC-SDC
5.07E-12	NDF	MSL3/CND\RRS/CL4/LTC-SDC/EFW
7.25E-13	NDF	MSL3/CND\RRS/CL4/LTC-SDC/ADW
1.46E-09	S	MSL3/CND\RRS/CL4/OMSSV
1.02E-12	NDF	MSL3/CND\RRS/CL4/OMSSV/EFW
1.46E-13	NDF	MSL3/CND\RRS/CL4/OMSSV/ADW
1.48E-09	S	MSL3/CND\RRS/CL4/AFW
1.03E-12	NDF	MSL3/CND\RRS/CL4/AFW/EFW
1.48E-13	NDF	MSL3/CND\RRS/CL4/AFW/ADW
1.48E-10	NDF	MSL3/CND\RRS/CL4/SWD1&D2

SMALL STEAM LINE BREAK CAUSING LOW DEAERATOR LEVEL C:\CAFTA\TREE(ET)\MSL3D1-2.TRE 10-06-03

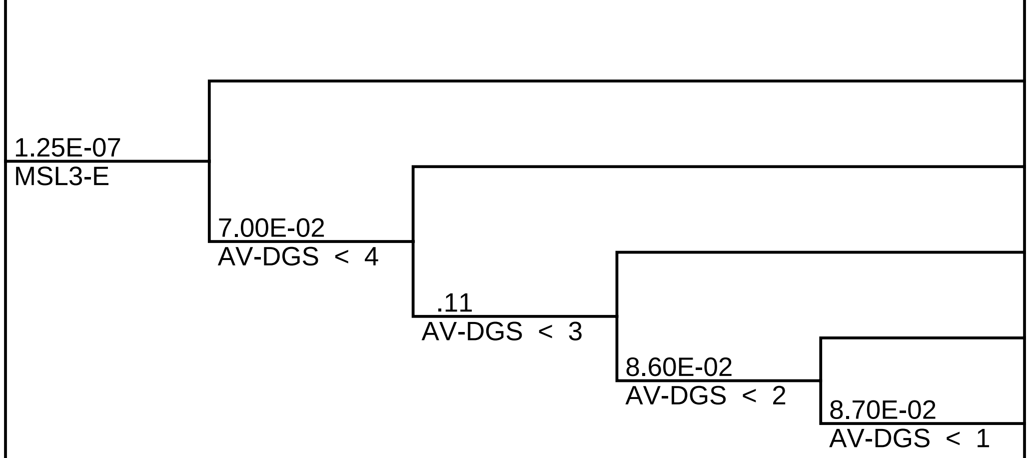
Accident Initiator	Support System	Heat Sinks	Operator Action	Heat Sinks		
IE-MSL3 + Loss of CND, Cl4 & 2 DGs	DIVISIONS #1 & #2 RSW / RCW SYSTEMS	AUX. FEEDWATER SUPPLY TO SGs	OPERATOR STARTS LONG TERM SHUTDOWN COOLING	LONG TERM (SHUTDOWN) COOLING SYSTEM	AUTO-DEPRESSURISATION WATER SYS.	Emergency Feedwater Sub-System
MSL3D1-3	SWD1&D2	AFW	OLTC-SDC	LTC-SDC	ADW	EFW



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR
1.41E-08	S	MSL3/CND\RRS/CL4
1.56E-09	S	MSL3/CND\RRS/CL4/LTC-SDC
1.09E-12	NDF	MSL3/CND\RRS/CL4/LTC-SDC/EFW
1.56E-13	NDF	MSL3/CND\RRS/CL4/LTC-SDC/ADW
1.58E-10	S	MSL3/CND\RRS/CL4/OLTC-SDC
1.10E-13	NDF	MSL3/CND\RRS/CL4/OLTC-SDC/EFW
1.58E-14	NDF	MSL3/CND\RRS/CL4/OLTC-SDC/ADW
8.30E-10	S	MSL3/CND\RRS/CL4/AFW
5.82E-13	NDF	MSL3/CND\RRS/CL4/AFW/EFW
8.31E-14	NDF	MSL3/CND\RRS/CL4/AFW/ADW
8.35E-11	NDF	MSL3/CND\RRS/CL4/SWD1&D2

SMALL STEAM LINE BREAK CAUSING LOW DEAERATOR LEVEL C:\CAFTA\TREE(ET)\MSL3D1-3.TRE 17-06-03

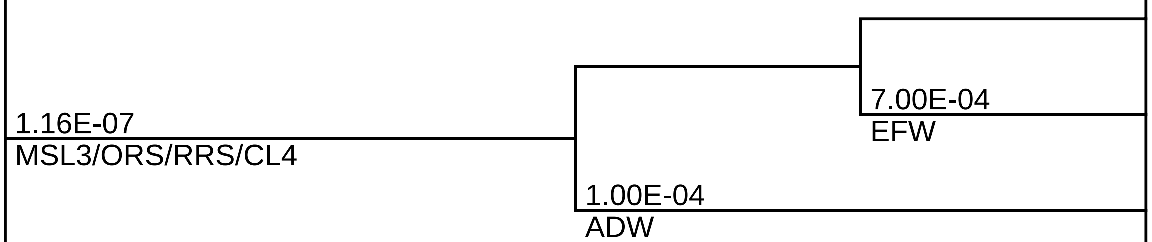
Initiating Event	Class III Power Supply				SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE NUMBER	SEQUENCE DESIGNATOR
Small Steam Line Break & Loss of CL4 (SD	4 DGs Available for 24h	3 DGs Available for 24h	2 DGs Available for 24h	1 DG Available for 24h				
IE-MSL3	DG-AV=4	DG-AV=3	DG-AV=2	DG-AV=1				



SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE NUMBER	SEQUENCE DESIGNATOR
1.16E-07	CONT'D	MSL3-E1	IE-MSL3/CL4/UNAV-DGS=0
7.79E-09	CONT'D	MSL3-E2	IE-MSL3/CL4/UNAV-DGS=1
8.80E-10	NDF		IE-MSL3/CL4/UNAV-DGS=2
7.56E-11	NDF		IE-MSL3/CL4/UNAV-DGS=3
7.20E-12	NDF		IE-MSL3/CL4/UNAV-DGS=4

SMALL STEAM LINE BREAK (Causing Low Deaerator Lvl) C:\CAFTA\TREE(ET)\MSL3-E.TRE 17-06-03

Accident Initiator	Heat Sinks			SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR
IE-MSL3 -CND & RRS Unavail. + Loss of Cl4	AUXILIARY FW SUPPLY TO SGs	AUTO-DEPRESSURISATION WATER SYS.	Emergency Feedwater Sub-System			
MSL3E-1	AFW	ADW	EFW			
				1.16E-07	S	MSL3/ORS/RRS/CL4
				8.12E-11	NDF	MSL3/ORS/RRS/CL4/EFW
				1.16E-11	NDF	MSL3/ORS/RRS/CL4/ADW



MAIN STEAM LINE BREAK (With Low Deaerator Level) C:\CAFTA\TREE(ET)\MSL3E-1.TRE 10-06-03

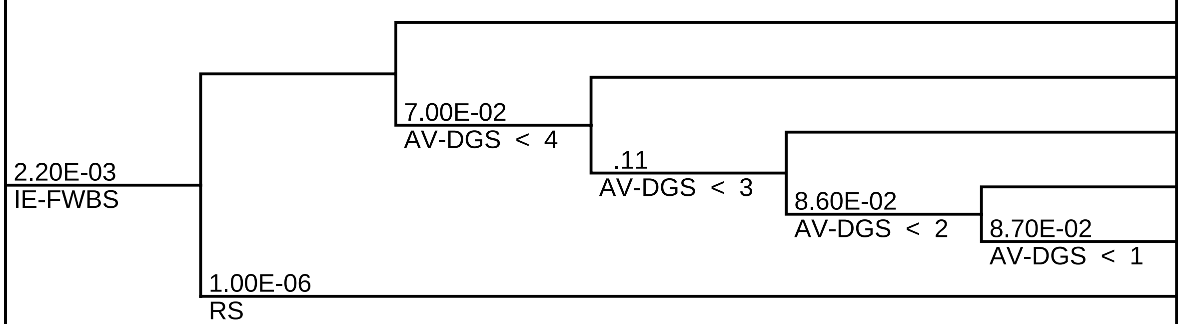
Accident Initiator	Heat Sinks			SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR
IE-MSL3 -CND Unavail., Los s of Cl4 & 1D G (RS by SDS1	AUXILIARY FW SUPPLY TO SGs	AUTO-DEPRESSURISAT ION WATER SYS.	Emergency Feedwater Sub-System			
MSL3E-2	AFW	ADW	EFW			
7.79E-09 MSL3/CND/RRS/CL4				7.78E-09	S	MSL3/CND/RRS/CL4
			7.00E-04 EFW	5.45E-12	NDF	MSL3/CND/RRS/CL4/EFW
		1.00E-04 ADW		7.79E-13	NDF	MSL3/CND/RRS/CL4/ADW

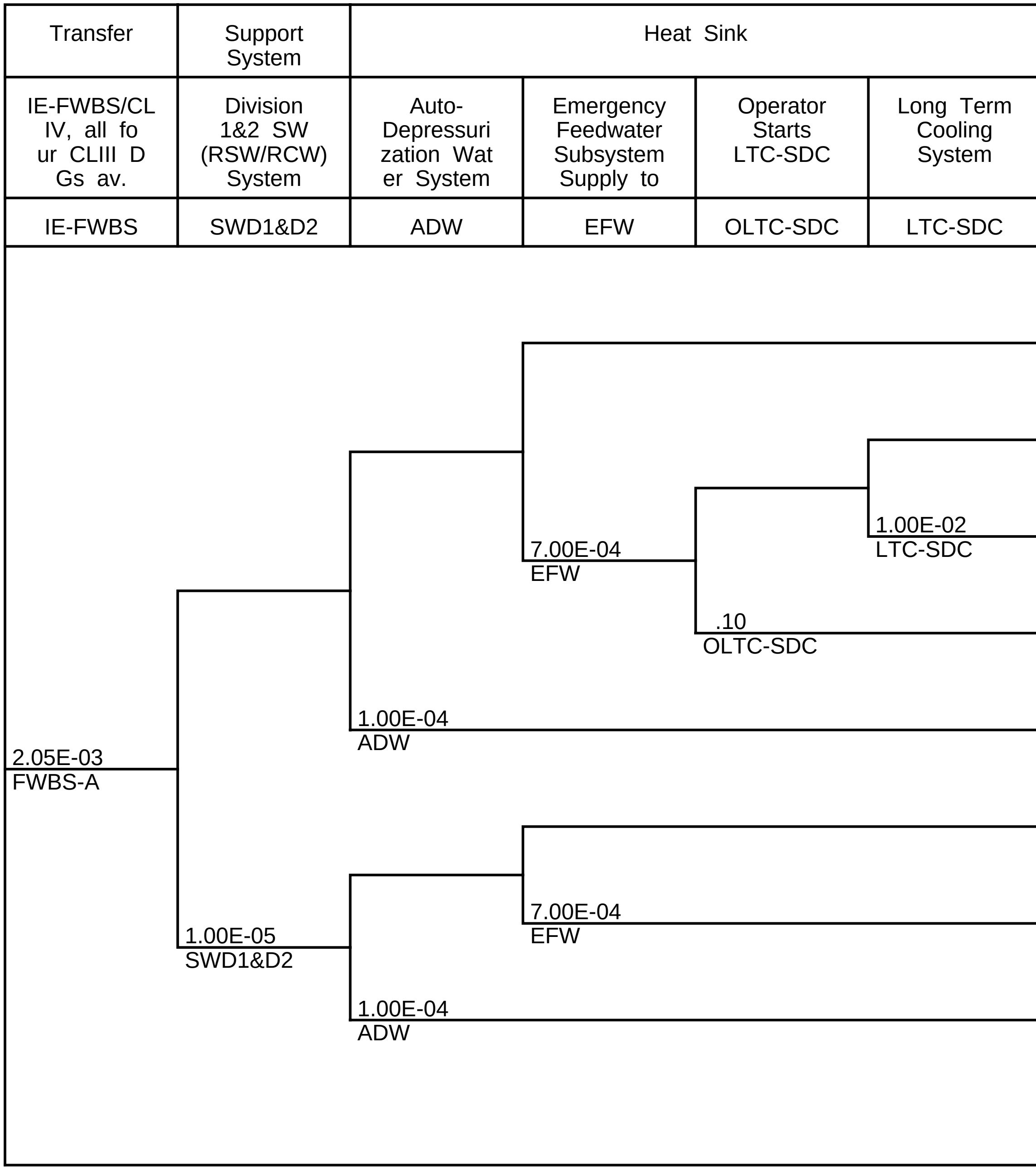
MAIN STEAM LINE BREAK (With Low Deaerator Level) C:\CAFTA\TREE(ET)\MSL3E-2.TRE 10-06-03

Appendix I

Event Tree for Symmetric FW Line Break Upstream of FW Control Valves

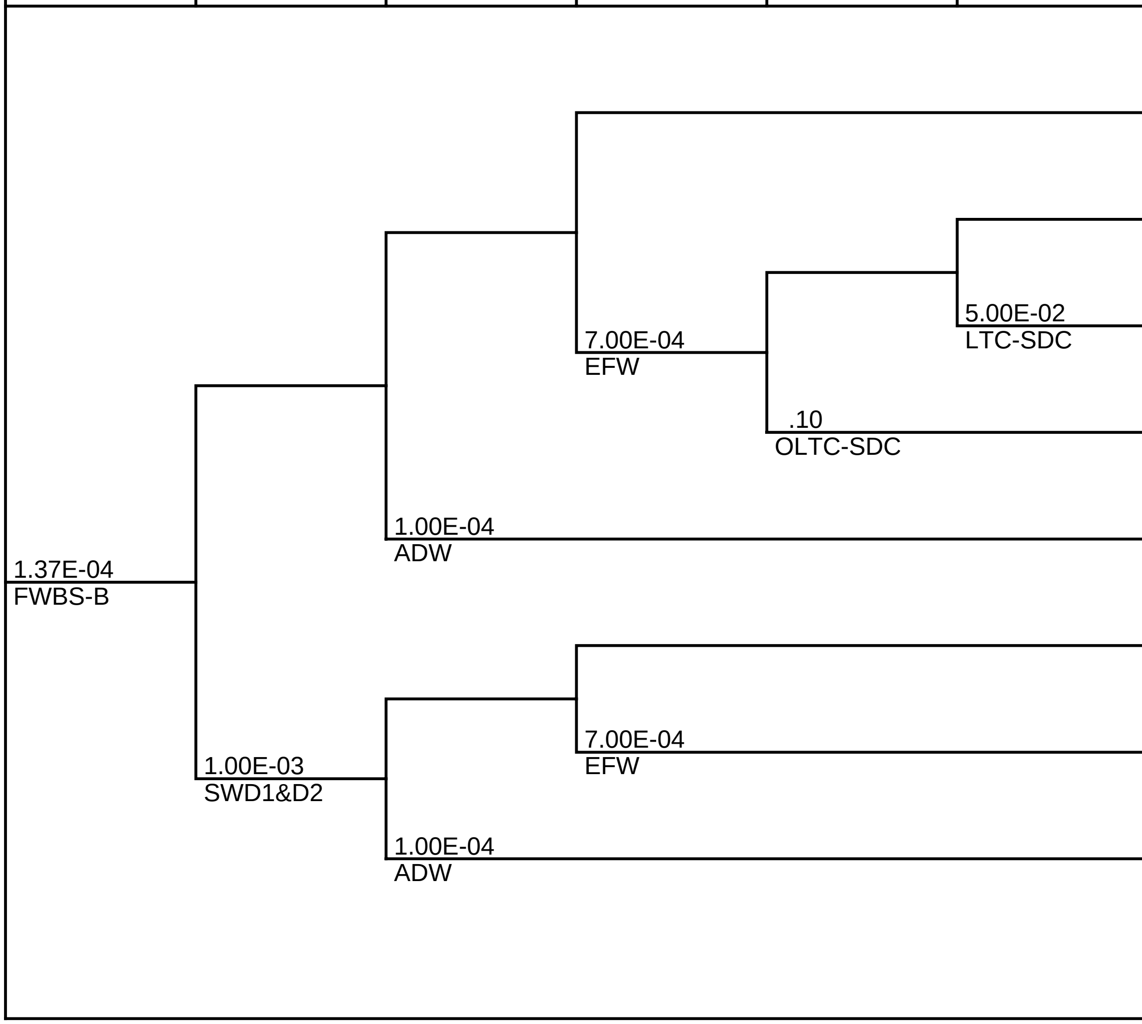
Sym. FWLB in TB (ups tr. FW reg . valves)	Reactor Shutdown via SDS1&SDS2	4 DGs Available for 24h	3 DGs Available for 24h	2 DGs Available for 24h	1 DG is Available for 24h	SEQ.PROB.	Plant Damage State	SEQUENCE NUMBER	SEQUENCE DESIGNATOR
IE-FWBS	RS	DG-AV = 4	DG-AV = 3	DG-AV = 2	DG-AV = 1				
						2.05E-03	CONT'D	FWBS-A	FWBS/UNAV-DGS=0
						1.37E-04	CONT'D	FWBS-B	FWBS/UNAV-DGS=1
						1.55E-05	CONT'D	FWBS-C	FWBS/UNAV-DGS=2
						1.33E-06	CONT'D	FWBS-D	FWBS/UNAV-DGS=3
						1.27E-07	CONT'D	FWBS-E	FWBS/UNAV-DGS=4
						2.20E-09	PDS0	FWBS1	





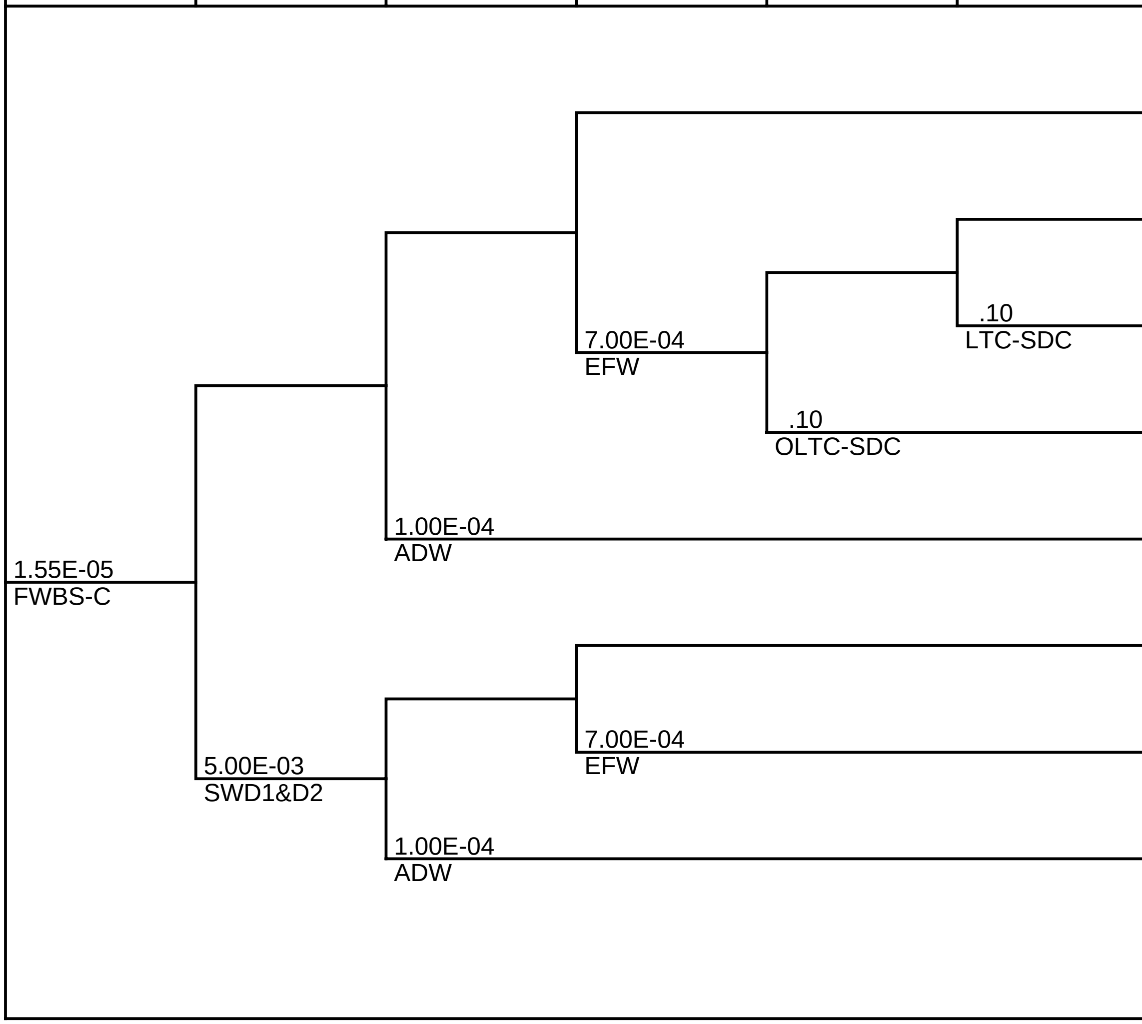
SEQ.PROB.	CLASS	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
2.05E-03	S	FWBS-A	
1.28E-06	S	FWBS-A/EFW	
1.29E-08	PDS6	FWBS-A/EFW/LTC-SDC	FWBS-A1
1.43E-07	PDS6	FWBS-A/EFW/OLTC-SDC	FWBS-A2
2.05E-07	PDS6	FWBS-A/ADW	FWBS-A3
2.05E-08	S	FWBS-A/SWD1&D2	
1.43E-11	NDF	FWBS-A/SWD1&D2/EFW	
2.05E-12	NDF	FWBS-A/SWD1&D2/ADW	

Transfer	Support System	Heat Sink			
IE-FWBS/CLIV, 3 DGs available	Division 1&2 SW (RSW/RCW) System	Auto-Depressurization Water System	Emergency Feedwater Subsystem Supply to	Operator Starts LTC-SDC	Long Term Cooling System
IE-FWBS	SWD1&D2	ADW	EFW	OLTC-SDC	LTC-SDC

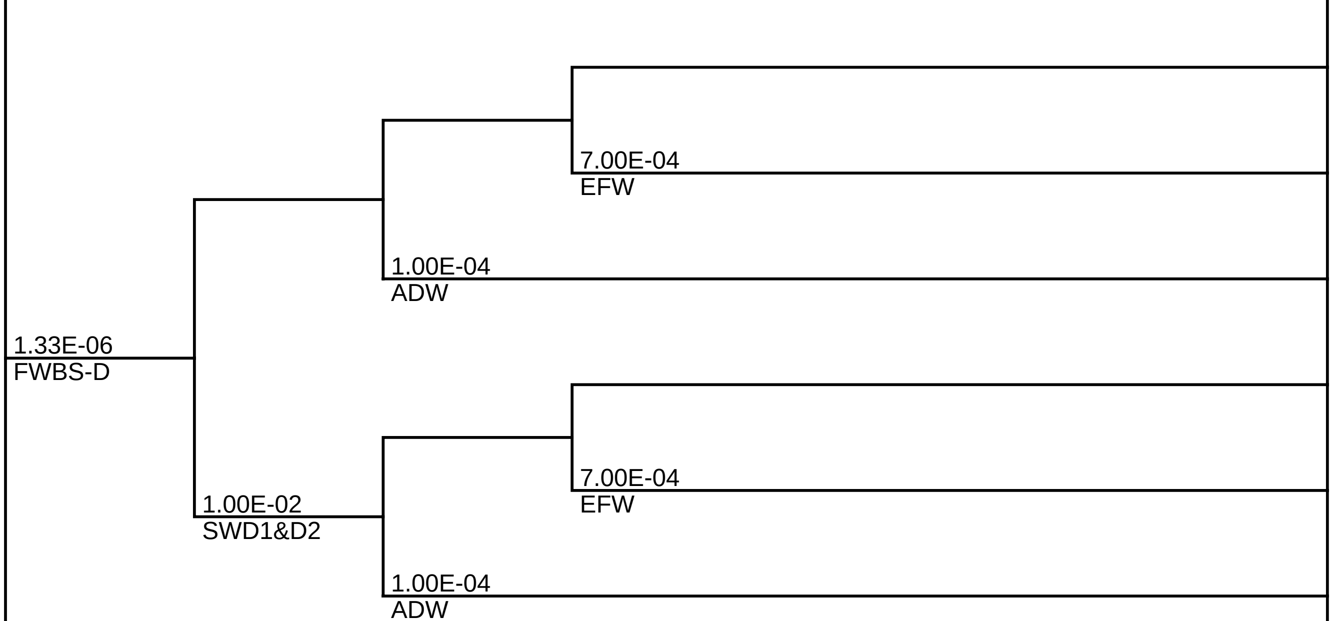


SEQ.PROB.	CLASS	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
1.37E-04	S	FWBS-B	
8.19E-08	S	FWBS-B/EFW	
4.31E-09	PDS6	FWBS-B/EFW/LTC-SDC	FWBS-B1
9.58E-09	PDS6	FWBS-B/EFW/OLTC-SDC	FWBS-B2
1.37E-08	PDS6	FWBS-B/ADW	FWBS-B3
1.37E-07	S	FWBS-B/SWD1&D2	
9.59E-11	NDF	FWBS-B/SWD1&D2/EFW	
1.37E-11	NDF	FWBS-B/SWD1&D2/ADW	

Transfer	Support System	Heat Sink				SEQ.PROB.	CLASS	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
IE-FWBS/CLIV, 2 DGs available	Division 1&2 SW (RSW/RCW) System	Auto-Depressurization Water System	Emergency Feedwater Subsystem Supply to	Operator Starts LTC-SDC	Long Term Cooling System				
IE-FWBS	SWD1&D2	ADW	EFW	OLTC-SDC	LTC-SDC				
						1.54E-05	S	FWBS-C	
						8.74E-09	S	FWBS-C/EFW	
						9.72E-10	NDF	FWBS-C/EFW/LTC-SDC	
						1.08E-09	PDS6	FWBS-C/EFW/OLTC-SDC	FWBS-C1
						1.54E-09	PDS6	FWBS-C/ADW	FWBS-C2
						7.74E-08	S	FWBS-C/SWD1&D2	
						5.42E-11	NDF	FWBS-C/SWD1&D2/EFW	
						7.75E-12	NDF	FWBS-C/SWD1&D2/ADW	

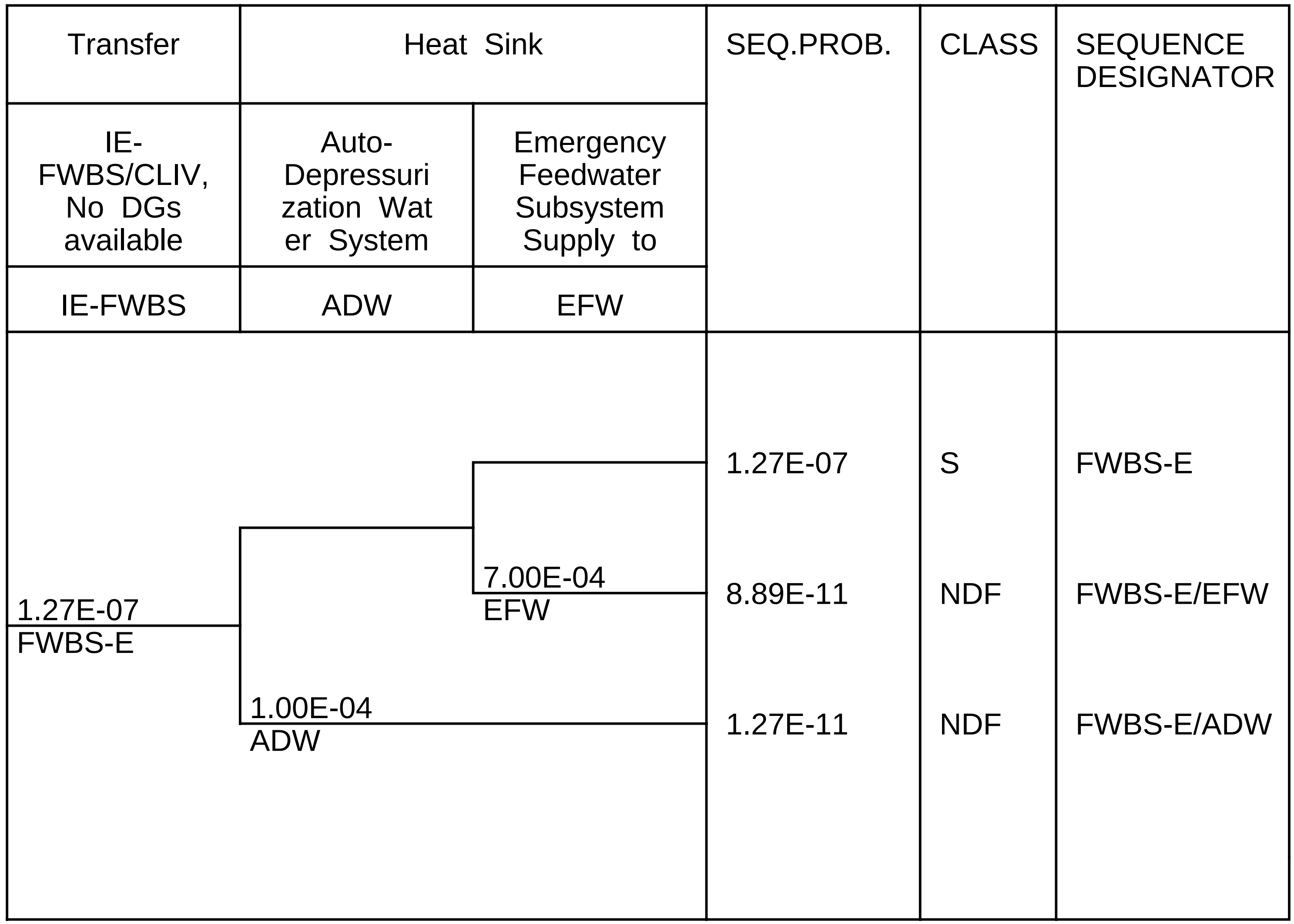


Transfer	Support System	Heat Sink					SEQ.PROB.	CLASS	SEQUENCE DESIGNATOR
IE-FWBS/CLIV, 1 DGs available	Division 1&2 SW (RSW/RCW) System	Auto-Depressurization Water System	Emergency Feedwater Subsystem Supply to	Operator Starts LTC-SDC	Long Term Cooling System	Dormant ECC			
IE-FWBS	SWD1&D2	ADW	EFW	OLTC-SDC	LTC-SDC	DECC			



SEQ.PROB.	CLASS	SEQUENCE DESIGNATOR
1.32E-06	S	FWBS-D
9.22E-10	NDF	FWBS-D/EFW
1.32E-10	NDF	FWBS-D/ADW
1.33E-08	S	FWBS-D/SWD1&D2
9.31E-12	NDF	FWBS-D/SWD1&D2/EFW
1.33E-12	NDF	FWBS-D/SWD1&D2/ADW

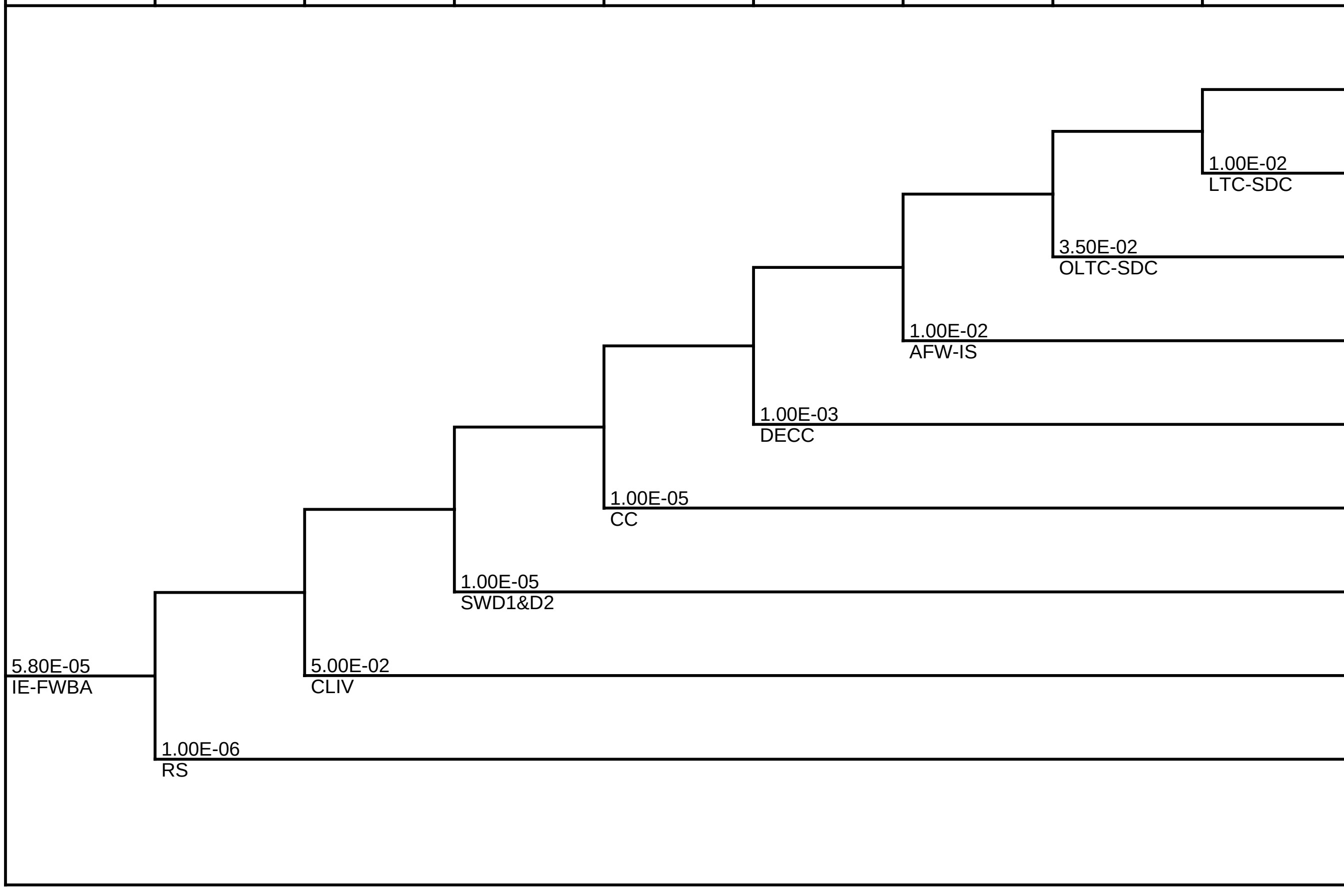
FWLB SYM. (upstrm. of FW reg. valves) T/B - P5 C:\CAFTA\TREE(ET)\FWBS-5.TRE 22-01-04


Transfer
Heat Sink
SEQ.PROB.
CLASS
SEQUENCE DESIGNATOR
IE-FWBS/CLIV, No DGs available
Auto-Depressurization Water System
Emergency Feedwater Subsystem Supply to
IE-FWBS
ADW
EFW
1.27E-07
FWBS-E
1.00E-04
ADW
7.00E-04
EFW
1.27E-07
S
FWBS-E
8.89E-11
NDF
FWBS-E/EFW
1.27E-11
NDF
FWBS-E/ADW

Appendix J

Event Tree for Asymmetric FW Line Break Downstream of SG Check Valve

Initiating Event	Reactor Shutdown	Support System		ECC Support	HTS Make-up	Heat Sinks	Op. Action	Heat Sinks
Asym. FWLB in R/B (d ownst. S/G check val	Reactor Shutdown via SDS1&SDS2	Class IV Power available	Service Wa ter System (RSW/RCW) Divisions	Crash Cooldown (CC1 + CC2)	Dormant ECC Injection	AFW or SG Isolation Failures (break isol	Operator Starts LTC -SDC (with in 30min.)	Long Term Cooling Sy stem - Shu tdown Cool
IE-FWBA	RS	CLIV	SWD1&D2	CC	DECC	AFW-IS	OLTC-SDC	LTC-SDC



SEQ.FREQ.	CLASS	SEQUENCE DESIGNATOR	Sequence Number
5.21E-05	S	IE-FWBA	
5.26E-07	PDS6	IE-FWBA/LTC-SDC	FWBA1
1.91E-06	PDS6	IE-FWBA/OLTC-SDC	FWBA2
5.50E-07	PDS6	IE-FWBA/AFW-IS	FWBA3
5.51E-08	PDS2	IE-FWBA/DECC	FWBA4
5.51E-10	PDS1	IE-FWBA/CC	FWBA5
5.51E-10	NDF	IE-FWBA/SWD1&D2	
2.90E-06	PDS6	IE-FWBA/CLIV	FWBA6
5.80E-11	PDS0	IE-FWBA/RS	FWBA7

Appendix K

Event Tree for Loss of Reactivity Control Leading to Core Power Excursion

Initiating Event	Reactor Shutdown		SEQ.PROB.	PLANT DAMAGE STATE	SEQUENCE DESIGNATOR	SEQUENCE NUMBER
LOSS OF REACTIVITY CONTROL	SHUT DOWN BY SDS1 ON ROP OR HLR	SHUTDOWN BY SDS2 ON ROP OR HLR				
IE-LOR	SDS1	SDS2				
4.24E-02 IE-LOR			4.24E-02	S	IE-LOR	
	1.00E-03 SDS1		4.24E-05	S	IE-LOR/SDS1	
		1.00E-03 SDS2	4.24E-08	PDS0	IE-LOR/SDS1/SDS2	LOR1

LOSS OF REACTIVITY CONTROL CAUSING POWER EXCURSION C:\CAFTA\TREE(ET)\ACR-LOR.TRE 21-07-03