TABLE 3.1.1-1
SOURCES OF INITIATING EVENT INFORMATION

1. LERs for North Anna Unit 1 and 2 for 1986 - 1990.

2. NUREG/CR-3862, "Development of Transient Initiating Event Frequencies for use in PRA, May 1988".

3. NUREG/CR-4550, Volume 3.

4. Review of Support System Drawings for North Anna Units 1 and 2.

5. North Anna monthly operating reports 1980 - 1990.

6. Review of past PRAs on Westinghouse PWRs.

TABLE 3.1.1-2
LIST OF INITIATING EVENTS

Abbreviations	Descriptions
T1	Loss of Offsite Power
T2	Transients with non-recoverable Loss of Main Feedwater
T2A	Transients with recoverable loss of Main Feedwater following FW Isolation
T3	Transients with Main Feedwater initially available
T4	Loss of RC Pump Seal Injection and Thermal Barrier Cooling
T5A	Non-recoverable Loss of DC Bus 1-I
T5B	Non-recoverable Loss of DC Bus 1-III
T6	Loss of Service Water System
T7	Steam Generator Tube Rupture
T8	Loss of Emergency Switchgear Room Cooling
T9A	Loss of 4160 V Bus 1H
T9B	Loss of 4160 V Bus 1J
A	Large LOCA 6" - 31"
S1	Medium LOCA 2" - 6"
S2	Small LOCA 3/8" - 2"
VX	Interfacing System LOCA
RX	Reactor Vessel Rupture

TABLE 3.1.1-7
TRANSIENT INITIATING EVENT T2 SUB-GROUP

Initiating Event Group	Representative Initiators	Comments
T2 Non-recoverable Loss of Main FW	Failure of Main Feedwater. Loss of Instrument Air (IA) system. Main Feed-water Line Break.	Includes MFW failures (i.e., disabled pumps), failure in hotwell FW flow path, and insufficient condensate inventory, loss of IA.
T2A Recoverable Loss of MFW	Steam Generator Hi Hi Level. Inadvertent SI. Main Steamline Break.	FW recovered by start of 1 MFW pump and flow through 1 FRV or bypass valve.
	Lo Tavg coincident with Reactor Trip.	FW recovered by flow through 1 FW bypass valve and 1 MFW pump maintained on recirculation.

TABLE 3.1.1-8
CORRESPONDENCE BETWEEN NORTH ANNA IE GROUP AND EPRI CATEGORIES

North Anna IE Group(s)	EPRI Category	Title and Definition
T2A, T3	1	**Loss of RCS Flow (1 loop)** - An inadvertent hardware or human error interrupts the flow in one loop of the reactor coolant system. **SI***
T3	2	**Uncontrolled Rod Withdrawal** - One or more control rods are withdrawn inadvertently.
T3*	3	**CRDM Problems and/or Rod Drop** - Failures in the control rod drive mechanism (CRDM) occur that lead to out-of-tolerance conditions in the primary system. The transient may include dropping of one or more control rods into the core as part of the CRDM failure. [Assumes no turbine runback-use category 33 with turbine runback].
T3*	4	**Leakage from Control Rods** - Primary system leakage around the control rod drive mechanism is excessive and reactor shutdown is required.
T3*	5	**Leakage in Primary System** - Primary system leakage through various piping components is excessive and reactor shutdown is required. This transient does not include: No. 4 - Leakage from control rods No. 7 - Pressurizer leakage No. 26 - Steam generator leakage
T3	6	**Low Pressurizer Pressure** - Pressurizer pressure falls below the lower operating limit.
T3*	7	**Pressurizer Leakage** - Pressurizer components allow excessive primary system leakage and reactor shutdown is required.

TABLE 3.1.1-8 (Continued)
CORRESPONDENCE BETWEEN NORTH ANNA IE GROUP AND EPRI CATEGORIES

North Anna IE Group(s)	EPRI Category	Title and Definition
T3	8	**High Pressurizer Pressure** - Pressurizer pressure climbs above the upper operating limit.
T2A	9	**Inadvertent Safety Injection Signal** - Hardware or operator error initiates a safety injection.
T3	10	**Containment Pressure Problems** - Hardware or operator error results in containment pressure exceeding limits.
T3	11	**CVCS Malfunction - Boron Dilution** - Hardware or operator error results in a CVCS malfunctions such that reactor power is affected.
T3	12	**Pressure/Temperature/Power Imbalance - Rod Position Error** - Poor control rod positioning from mechanical or operator error causes a scram based on a pressure, temperature, or power imbalance.
Not Applicable	13	**Startup of Inactive Coolant Pump** - An inactive coolant pump is started at an improper power and flow condition. [Unit operation with inactive coolant loop is precluded by Technical specifications.]
T3	14	**Total Loss of RCS Flow** - A hardware or operator error causes a loss of reactor coolant system flow.
T3*	15	**Loss or Reduction in Feedwater Flow (1 loop)** - One feedwater pump trips or another occurrence results in an overall decrease in feedwater flow.

TABLE 3.1.1-8 (Continued)
CORRESPONDENCE BETWEEN NORTH ANNA IE GROUP AND EPRI CATEGORIES

North Anna IE Group(s)	EPRI Category	Title and Definition
T2, T2A	16	**Total Loss of Feedwater Flow (all loops)** - A simultaneous loss of all main feedwater occurs, excluding that due to loss of all offsite power (Category 35).
T2A	17	**Full or Partial Closure of MSIV (1 loop)** -One main steam isolation valve (MSIV) closes, the rest remaining open, or partial closure of one or more MSIV occurs. [Can result in Steam Generator Lo-Lo Level reactor trip.]
T3	18	**Closure of all MSIV** - One of various steam line or nuclear system malfunctions requires termination o f steam flow from the vessel. The closure of one MSIV may cause an immediate closure of all other MSIVs; this occurrence is also included in this transient definition. However, any closure that is the result of another initiator is not included. [Can result in Steam Generator Lo-Lo Level reactor trip.]
T3*	19	**Increase in Feedwater Flow (1 loop)** - An increase in feedwater flow occurs in one loop. [Steam Generator Hi-Hi Level Turbine Trip/Reactor Trip causes Feedwater Isolation.]
T3*	20	**Increase in Feedwater Flow (All Loops)** - An increase in feedwater flow occurs in one loop.[Steam Generator Hi-Hi Level Turbine Trip/Reactor Trip causes Feedwater Isolation.]

TABLE 3.1.1-8 (Continued)
CORRESPONDENCE BETWEEN NORTH ANNA IE GROUP AND EPRI CATEGORIES

North Anna IE Group(s)	EPRI Category	Title and Definition
T2A*, T3*,	21	**Feedwater Flow Instability - Operator Error** - Feedwater is being controlled manually, usually during startup or shutdown, and excessive or insufficient feedwater flow occurs.
T2A, T3*	22	**Feedwater Flow Instability - Miscellaneous Mechanical Causes** - Excessive or insufficient feedwater flow results form hardware failures in the feedwater system.
T2, T3	23	**Loss of Condensate Pumps (1 loop)** - One condensate pump fails, reducing feedwater flow. [Can result in Feedwater pump trip on low suction pressure]
T2	24	**Loss of Condensate Pumps (all loops)** - All condensate pumps fail, causing a loss of feedwater flow.
T3	25	**Loss of Condenser Vacuum** - Either a complete loss or decrease in condenser vacuum results from hardware or human error.Can use atmospheric steam dump without condenser, Feedwater pumps will not trip as long as hotwell inventory lasts.
T3+, T7	26	**Steam Generator Leakage** - Excessive primary system to secondary leakage occurs in the steam generator.
T3	27	**Condenser Leakage** - Excessive secondary system leakage occurs in the condenser. [Feedwater heater level Turbine Trip].

TABLE 3.1.1-8 (Continued)
CORRESPONDENCE BETWEEN NORTH ANNA IE GROUP AND EPRI CATEGORIES

North Anna IE Group(s)	EPRI Category	Title and Definition
T3	28	**Miscellaneous Leakage in Secondary System** - Excessive leakage occurs in the secondary system other than in the condenser (see Category 27).
T2A, T3*	29	**Sudden Opening of Steam Relief Valves** - A secondary system steam relief valve opens inadvertently, causing an unacceptably low pressure in the secondary system. [Can result in Feedwater Isolation from SI or Steam Generator Hi-Hi Level Turbine Trip/Reactor Trip.]
T2A*, T3*	30	**Loss of Circulating Water** - Circulating water is not available to the plant. [Can result in loss of condenser vacuum - see Category 25.]
T3	31	**Loss of Component Cooling** - Excessive temperature of critical components is a result of a loss or decrease in component cooling water flow.
T3	32	**Loss of Service Water System** - The service water system fails to perform its function.
T2A*, T3*	33	**Turbine Trip, Throttle Valve Closure, EHC Problems** - A turbine trip occurs; or turbine problems occur which in effect decrease steam flow to the turbine, causing a rapid change in the amount of energy removed from the primary system. [Turbine runback can result in Steam Generator Hi-Hi Level or Steam Generator Lo-Lo Level, causing Feedwater Isolation.]

TABLE 3.1.1-8 (Continued)
CORRESPONDENCE BETWEEN NORTH ANNA IE GROUP AND EPRI CATEGORIES

North Anna IE Group(s)	EPRI Category	Title and Definition
T3*	34	**Generator Trip or Generator Caused Faults** - The generator is tripped due to electrical grid disturbances or generator faults.
T1	35	**Loss of All Offsite Power** - All power to the plant from external sources (the grid or a dedicated transmission line to another plant) is lost.
T3	36	**Pressurizer Spray Failure** - The pressurizer spray system spuriously actuates or fails upon demand.
T3	37	**Loss of Power to Necessary Plant Systems** -Power is lost to a component or group of components such that plant shutdown is necessary. It does not include loss of power to those components whose failure causes another defined transient to occur.
T3	38	**Spurious Trips - Cause Unknown** - A scram occurs and no out-of-tolerance condition can be detected; the cause of the scram cannot be determined. [Use Category 9 if scram by SI reactor trip (and SI is spurious).]
T3*	39	**Automatic Trip - No Transient Condition** - An auto scram is initiated by a hardware failure in instrumentation or logic circuits and no out-of-tolerance condition exists.

TABLE 3.1.1-8 (Continued)
CORRESPONDENCE BETWEEN NORTH ANNA IE GROUP AND EPRI CATEGORIES

North Anna IE Group(s)	EPRI Category	Title and Definition
T3	40	**Manual Trip - No Transient Condition** - The operator initiates a scram for any reason when no our-of-tolerance condition exists.
T3	41	**Fire Within Plant** - A plant shutdown is necessitated by a fire in some part of the plant.

* Evidenced in North Anna data
+ Manual reactor trip only
[ ] North Anna specific

TABLE 3.1.1-9
SOURCES OF DATA FOR PLANT-SPECIFIC INITIATORS

1. North Anna Licensee Event Reports (LERs) for the period 1986 - 1990[1],

2. North Anna Power Station "Monthly Operating Report" for the period 1986 - 1990,

3. NUREG/CR-3862 for reactor trips within the interval 1978 through 1981, and for the power level of some reactor trip events over the interval 1982 through 1983. Note that North Anna "Monthly Operating Reports" were scanned to identify any unusual initiating events for the interval 1980 through 1990.

1 North Anna LERs were reviewed for the period 1984-1990 for the T9-related precursors involving loss of feeder power to the 4160 V buses 1H and 1J.

TABLE 3.1.1-10
LIST OF NORTH ANNA REACTOR TRIP EVENTS, 1986-1990

VaP Unit	EPRI Cat	Yr	Date	ID	IE Group	Pwr Lev	Bkr Cls	Description	Cause	SI	References
N1	15	90	01/23/90	1	T3	100	Y	RT ON STEAM/FEEDWATER FLOW MISMATCH DUE TO A FAILED DRIVER CARD ON A FRV.	NA 1 EXPERIENCED AN AUTO Rx TRIP FROM 100% POWER DUE TO LOW LEVEL IN THE C SG WITH STEAM FLOW/FW FLOW MISMATCH. THE MISMATCH RESULTED FROM THE CLOSURE OF THE C MF REG. VLV DUE TO A FA LED PCB DRIVER CARD IN THE VALVE CONTROLLER	N	LER 90-001-00
N1	21	89	12/05/89	2	T2A	90	Y	AUTO REACTOR TRIP RESULTING FROM EHC SYSTEM TRANSIENT. REACTOR WAS INITIALLY AT 90% POWER AND RAMPED DOWN UNTIL TRIP.	UNIT 1 EXPERIENCED AN AUTO REACTOR TRIP FROM 7% POWER DUE TO A LO LO LEVEL IN THE B SG CAUSED BY FW ISOLATION. PRIOR TO THE REACTOR TRIP, THE POWER WAS BEING REDUCED DUE TO EHC SYSTEM PRESSURE TRANSIENTS WHICH WAS CAUSED BY LEAKING TURBINE OPC VLVS.	N	LER 89-017-00
N1	33	89	07/19/89	3	T3	90	Y	REACTOR TRIP DUE TO A LOSS OF EHC SYSTEM PRESSURE.	UNIT 1 EXPERIENCED AN AUTO Rx TRIP FROM 90% POWER DUE TO A LOSS OF EHC SYSTEM PRESSURE WHICH WAS CAUSED BY A FAILED O-RING ON THE TURBINE TRIP SOV 20-ET, RESULTING IN THE CLOSURE OF THE TURBINE STOP VALVES GENERATING THE TURBINE TRIP SIGNAL.	N	LER 89-014-00
N1	15	89	02/25/89	4	T3	76	Y	REACTOR TRIP DUE TO A MAIN FEEDWATER REGULATING VALVE CLOSURE AND SUBSEQUENT SG TUBE LEAK.	UNIT 1 EXPERIENCED AN AUTO Rx TRIP FROM 76% POWER DUE TO 'C' SG STEAM FLOW/FW FLOW MISMATCH COINCIDENT WITH A LOW SG LEVEL. THE MISMATCH WAS CAUSED BY THE CLOSURE OF THE C MF REG. VALVE, ON THE LOSS OF AIR.	N	LER 89-005-00

TABLE 3.1.1-10 (Continued)
LIST OF NORTH ANNA REACTOR TRIP EVENTS, 1986-1990

VaP Unit	EPRI Cat	Yr	Date	ID	IE Group	Pwr Lev	Bkr Cls	Description	Cause	SI	References
N1	15	88	08/06/88	5	T9	100	Y	REACTOR TRIP ON STEAM FLOW/FEED FLOW MISMATCH COINCIDENT WITH A LOW LEVEL DUE TO MFRV CLOSURE.	AUTO Rx TRIP FROM 100% POWER DUE TO THE MISMATCH OF SG FEED FLOW/SG COINCIDING WITH A LOW LEVEL. THE MISMATCH RESULTED FROM A CLOSURE OF THE 'B' MF REG VLV WHICH WAS CAUSED BY A DEGRADED VOLTAGE CONDITION ON THE 1J EMERGENCY BUS, CAUSED BY AN RSST (RESERVE STATION SERVICE TRANSFORMER) FAULT.	N	LER 88-020-00
N1	33	88	03/19/88	6	T3	004	N	TURBINE TRIP/REACTOR TRIP-EHC SYSTEM MALFUNCTION. NOT INCLUDED BECAUSE OF LOWER POWER LEVEL.	UNIT 1 EXPERIENCED AN AUTO Rx TRIP FROM 3.5% POWER DUE TO SPIKE IN THE TURBINE IMPULSE PRESSURE WHICH CAUSED A TURBINE TRIP & ENABLED THE LOGIC FOR A REACTOR TRIP WHEN A TURBINE TRIP CONDITION EXISTED.	N	LER 88-013-00
N1	33	88	01/13/88	7	T2A	015	Y	AUTOMATIC REACTOR TRIP DUE TO HI-HI STEAM GENERATOR LEVEL.	AUTO TURBINE TRIP/Rx TRIP FROM 15% POWER DUE TO A TURBINE SOLENOID TRIP WHICH RESULTED WHEN A HI-HI LEVEL (>75%) WAS DETECTED ON 2/3 LEVEL CHANNELS IN THE B SG. THE HI-HI LEVEL CAUSED FW ISOLATION AND WAS THE RESULT OF SG LEVEL OSCILLATIONS.	N	LER 88-005-00
N1	30	88	01/08/88	8	T2A	100	Y	MANUAL REACTOR TRIP IN ANTICIPATION OF LOSS OF THE MAIN CONDENSER.	Rx WAS MANUALLY TRIPPED FROM 100% POWER IN ANTICIPATION OF LOSS OF THE MAIN CONDENSER AFTER THE THREE RUNNING CW PUMPS TRIPPED SIMULTANEOUSLY & CONDENSER VACUUM WAS OBSERVED TO BE DECREASING RAPIDLY. CAUSE OF PUMPS FAILURE COULD NOT BE FOUND.	N	LER 88-002-00

TABLE 3.1.1-10 (Continued)
LIST OF NORTH ANNA REACTOR TRIP EVENTS, 1986-1990

VaP Unit	EPRI Cat	Yr	Date	ID	IE Group	Pwr Lev	Bkr Cls	Description	Cause	SI	References
N1	22	87	11/23/87	9	T3	100	Y	REACTOR TRIP GENERATED FROM 5A FEEDWATER HI-HI LEVEL SIGNAL.	REACTOR TRIPPED FROM 100% POWER DUE TO A TURBINE SOLENOID TRIP WHICH RESULTED FROM A 5A FEEDWATER HEATER HI-HI LEVEL SIGNAL WHICH WAS GENERATED WHEN A LEVEL SWITCH FAILED.	N	LER 87-020-00
N1	26	87	07/15/87	10	T2A	100	Y	MANUAL REACTOR TRIP DUE TO INDICATIONS OF EXCESSIVE RCS LEAKAGE THROUGH STEAM GENERATOR TUBE.	REACTOR WAS MANUALLY TRIPPED FROM 100% POWER DUE TO INDICATIONS OF A SG TUBE LEAKAGE IN THE C SG. -20 MIN. LATER SAFETY INJECTION SYSTEM WAS AUTOMATICALLY INITIATED. THE ROOT CAUSE HAS BEEN LABELED A SG TUBE RUPTURE; HOWEVER, CONSIDERING SG REPLACEMENT, THIS EVENT WAS CATEGORIZED T3 AS A SG TUBE LEAK REQUIRING MANUAL REACTOR TRIP.	Y	LER 87-017-01
N1	33	87	06/29/87	11	T3	018	Y	REACTOR TRIP DUE TO 5A FEEDWATER HEATER HI-HI LEVEL.	Rx TRIPPED FROM 18% POWER DUE TO A TURBINE SOLENOID TRIP WHICH RESULTED FROM A 5A FW HEATER HI-HI LEVEL SIGNAL. THE HI-HI LEVEL IN THE 5A FW HEATER WAS CAUSED BY AN IMPROPER VLV LINE-UP FOLLOWING A REFUELING OUTAGE.	N	LER 87-015-01
N1	3	87	04/19/87	12	T3	067	Y	REACTOR TRIP CAUSED BY DROPPED CONTROL ROD.	REACTOR TRIPPED FROM 67% POWER DURING A CONTROLLED RAMPDOWN INTO A REFUELING OUTAGE DUE TO NUCLEAR INSTRUMENTATION SYSTEM POWER RANGE HIGH NEGATIVE FLUX RATE CAUSED BY A SINGLE DROPPED ROD.	N	LER 87-004-00

TABLE 3.1.1-10 (Continued)
LIST OF NORTH ANNA REACTOR TRIP EVENTS, 1986-1990

VaP Unit	EPRI Cat	Yr	Date	ID	IE Group	Pwr Lev	Bkr Cls	Description	Cause	SI	References
N1	33	86	08/27/86	13	T3	100	Y	MANUAL TURBINE/REACTOR TRIP DUE TO HIGH TURBINE/GENERATOR VIBRATION.	TURBINE/REACTOR WERE MANUALLY TRIPPED WHEN NA1 WAS AT 100% POWER DUE TO HIGH VIBRATION OF TURBINE/GENERATOR BEARING VIBRATION. VIBRATION CAUSE WAS BREAKAGE OF A 13 INCH PIECE OF TURBINE BLADE FROM THE LAST STAGE OF THE 'A' LOW PRESSURE TURBINE.	N	LER 86-015-00
N1	16	86	05/20/86	14	T3	100	Y	REACTOR TRIP FROM STEAM FLOW/FEED MISMATCH COINCIDENT WITH LOW STEAM GENERATOR LEVEL.	Rx TRIP OCCURRED FROM 100% POWER DUE TO A TRIP SIGNAL GENERATED FROM A STEAM FLOW/FEED FLOW MISMATCH (ALL 3 FW REG VLVS CLOSED BY SPURIOUS FW ISOLATION SIGNAL TO FRVS ONLY) CONCURRENT WITH A LOW LEVEL (2/3 LESS THAN/EQUAL TO 25% N.R. LEVEL) IN THE SG.	N	LER 86-008-00
N1	17	86	03/26/86	15	T2A	100	Y	REACTOR TRIP DUE TO A SAFETY INJECTION TRIP SIGNAL.	Rx TRIPPED FROM 100% POWER DUE TO A SI CAUSED BY THE CLOSURE OF THE B MAIN STEAM LINE TRIP VALVE. THIS RESULTED IN REACTOR AND TURBINE TRIP. THE SI WAS INITIATED DUE TO HIGH STEAM FLOW COINCIDENT WITH LOW STEAM LINE PRESSURE IN 'A' & 'C' SGs.	Y	LER 86-006-00
N1	39	86	05/31/86	16	T3	100	Y	REACTOR TRIP DUE TO LOSS OF A POWER TO 120 VAC VITAL BUS.	Rx TRIPPED FROM 100% POWER DUE TO FAILURE OF VITAL BUS WHICH POWERS THE RELAY THAT SENSES THE BREAKER POSITION OF 'A' RCP. DE-ENERGIZED RELAY, LEAD TO Rx TRIP SIGNAL BECAUSE THE RPS SENSED THE 'A' RCP BREAKER OPEN COINCIDENT WITH REACTOR POWER >30%.	N	LER 86-009-00

TABLE 3.1.1-10 (Continued)
LIST OF NORTH ANNA REACTOR TRIP EVENTS, 1986-1990

VaP Unit	EPRI Cat	Yr	Date	ID	IE Group	Pwr Lev	Bkr Cls	Description	Cause	SI	References
N1	33	86	02/23/86	17	T3	100	Y	REACTOR/TURBINE TRIP - TURBINE CONTROL SYSTEM MALFUNCTION.	Rx TRIP/TURBINE TRIP OCCURRED FROM 100% POWER. THE REACTOR TRIP SIGNAL WAS GENERATED BY A LO-LO LEVEL IN 'B' SG, DUE TO CLOSURE OF THE TURBINE GOVERNOR VALVES, CAUSING SHRINKAGE IN ALL SG WITH 'B' SG REACHING THE Rx TRIP SETPOINT FIRST.	N	LER 86-002-00
N1	33	86	01/19/86	18	T3	004	Y	REACTOR/TURBINE TRIP DUE TO A TURBINE FIRST-STAGE IMPULSE PRESSURE SPIKE. NOT INCLUDED BECAUSE OF LOW POWER LEVEL.	TURBINE TRIP/REACTOR TRIP OCCURRED FROM 4% POWER DUE TO A TURBINE FIRST-STAGE IMPULSE PRESSURE SPIKE AS PLANT PERSONNEL WERE SETTING UP FOR A TURBINE-GENERATOR OVERSPEED TRIP TEST.	N	LER 86-001-00
N2	21	90	11/02/90	19	T3	15	Y	REACTOR TRIP FROM 9% POWER DUE TO LOSS OF NORMAL FEEDWATER. REACTOR WAS INITIALLY AT 15 % POWER.	AUTO REACTOR TRIP OCCURRED FROM 9% POWER DUE TO A LO-LO LEVEL IN 'A' SG WHILE RETURNING TO POWER OPER. THE REACTOR TRIP OCCURRED ~8 MIN. FOLLOWING AN AUTO TURBINE TRIP FROM ~15% POWER. THE CAUSE OF EVENT WAS PERSONNEL ERROR TO RESET FW BYPASS VALVE.	N	LER 90-010-00
N2	15	86	06/29/86	20	T9	100	Y	REACTOR TRIP DUE TO LOW STEAM GENERATOR LEVEL COINCIDENT WITH A STEAM FLOW/FEED FLOW MISMATCH.	Rx TRIP OCCURRED FROM 100% POWER DUE TO LOW SG LEVEL COINCIDENT WITH A STEAM FLOW/FEED FLOW MISMATCH DURING EMERGENCY RAMPDOWN, DUE TO LOSS OF 2/3 MFW PUMPS CAUSED BY A LOSS OF POWER TO 1 OF 2 500kV SWITCHYARD BUSSES.	N	LER 86-009-00

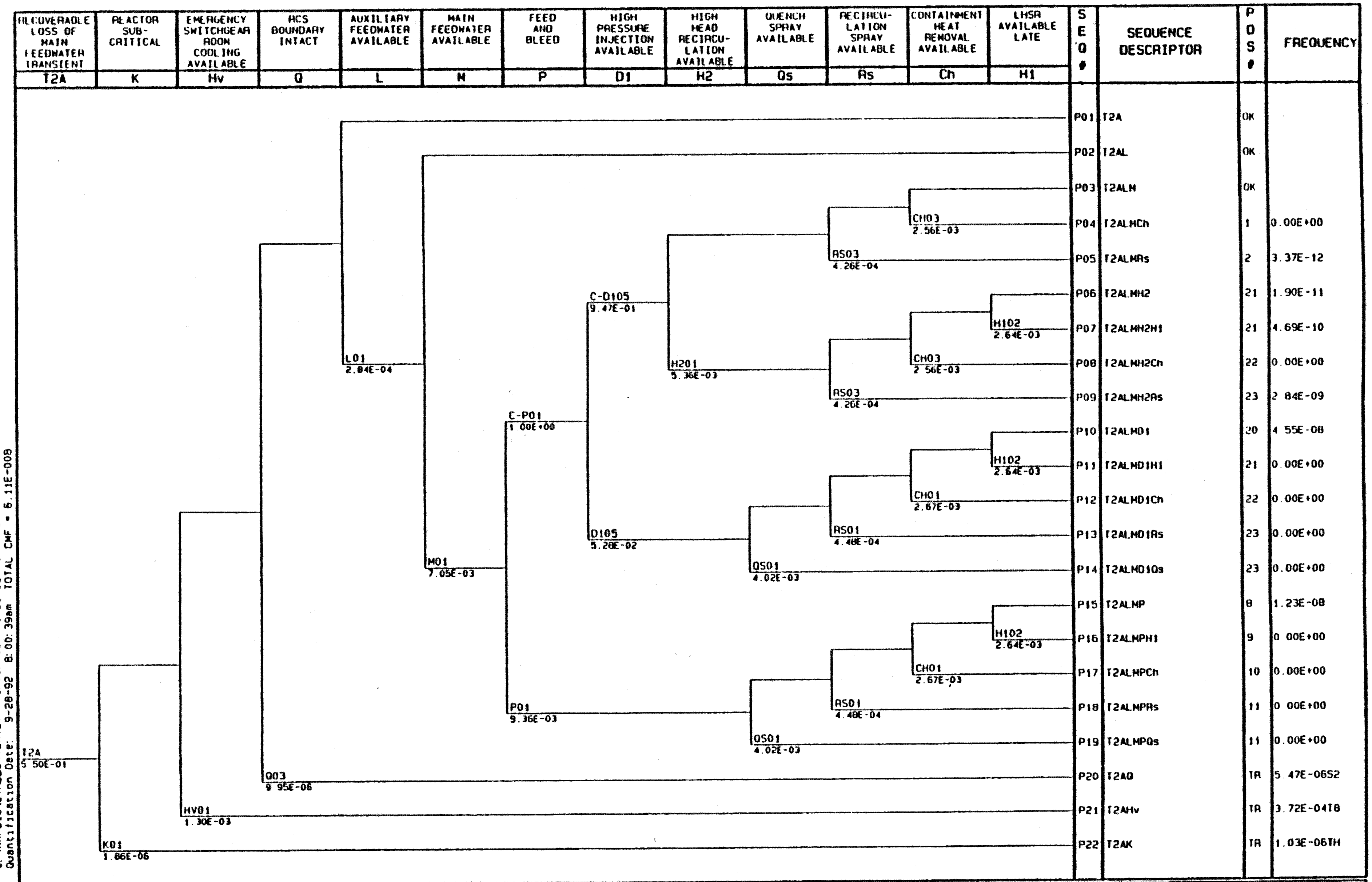
TABLE 3.1.1-10 (Continued)
LIST OF NORTH ANNA REACTOR TRIP EVENTS, 1986-1990

VaP Unit	EPRI Cat	Yr	Date	ID	IE Group	Pwr Lev	Bkr Cls	Description	Cause	SI	References
N2	34	86	04/11/86	21	T3	071	Y	UNIT 2 REACTOR TRIP DUE TO A TURBINE TRIP CAUSED BY A MAIN ELECTRICAL GENERATOR TRIP.	REACTOR TRIP OCCURRED FROM 71% POWER DUE TO A TURBINE TRIP CAUSED BY A MAIN ELECTRICAL GENERATOR TRIP, DUE TO ACTUATION OF A GENERATOR DIFFERENTIAL LOCKOUT RELAY UPON LOSS OF EXCITATION FIELD SIGNAL CAUSED BY FAILURE OF THE PERMANENT MAGNET GENERATOR.	N	LER 86-008-00
N2	33	86	04/16/86	22	T3	004	Y	REACTOR TRIP CAUSED BY TURBINE FIRST STAGE PRESSURE SPIKE. NOT INCLUDED BECAUSE OF LOW POWER LEVEL.	REACTOR TRIPPED FROM 4% POWER DUE TO TURBINE 1ST STAGE PRESS. SPIKE, CAUSED BY PERFORMING A THROTTLE VALVE/GOVERNOR VALVE TRANSFER WITH TURBINE IN AUTO CONTROL. THE PRESS. SPIKE CLEARED THE P-7 Rx TRIP BLOCKS CAUSING Rx TRIP DUE TO TURBINE TRIP.	N	LER 86-007-00
N2	3	86	05/29/86	23	T3	100	Y	UNIT 2 REACTOR TRIP OCCURRED FROM A NEGATIVE FLUX RATE TRIP.	Rx TRIP OCCURRED FROM 100% POWER DUE TO A NEGATIVE FLUX RATE CAUSED BY THE OPENING OF THE STATIONARY COIL POWER SUPPLY DISCONNECTED TO ROD CONTROL POWER DISTRIBUTION CABINET 1AC, CAUSING 12 RODS TO DROP INTO THE CORE. PERSONNEL ERROR CAUSED THE EVENT.	N	LER 86-005-00

TABLE 3.1.1-11
SUMMARY OF NORTH ANNA SYSTEM REVIEW FOR INITIATING EVENTS

System	System Symbol	Front line or Support	Detailed Analysis
Ambient Air Monitoring	AM	Neither	No
ATWS Mitigation System Actuation & Control (AMSAC)		Front line	No
Auxiliary Boilers	AB	Neither	No
Auxiliary Feedwater	AFW	Front Line	Yes
Auxiliary Steam	AS	Neither	No
Batteries, 125VDC	BY	Support	Yes
Bearing Cooling	BC	Support	Yes
Bearing Lube	BL	Neither	No
Blowdown	BD	Neither	No
Boron Recovery	BR	Neither	No
Building Structure	BLD	Neither	No
Chemical & Volume Control	CH	Front line	Yes
Chilled Water	CD	Neither	No
Circulating Water	CW	Support	Yes
Communications	CO	Neither	No
Component Cooling	CC	Support	Yes
Compressed Air	CA	Neither	No
Computer	CM	Neither	No
Condensate	CN	Support	Yes
Condensate Polishing	CP	Neither	No
Containment Access	CE	Neither	No
Containment Vacuum	CV	Neither	No
Control Rod Drive Power Supply	ED	Neither	No
Decay Heat Release	DHR	Neither	No
Decontamination	DC	Neither	No
Demineralizer Drain	WDR	Neither	No
Diesel Air	EB	Support	No
Drains (Aerated)	DA	Neither	No
Drains (Building Services)	DB	Neither	No
Drains (Gaseous)	DG	Neither	No
Domestic Water	DW	Neither	No
Early Warning	EW	Neither	No
Earthquake Reporting	ER	Neither	No
Electrical Calibration	EC	Neither	No
Electrical Equipment	PHP	Neither	No
Electrical Equipment (4KV & Above)	PH	Support	Yes
Electrical Equipment (600V & Below)	PL	Support	Yes
Electrical Hydraulic Control	EH	Neither	No
Electrical Instrumentation	EI	Neither	No
Electrical Power	EP	Support	Yes

C:\NAPS10\ETREES\T2A.EVT 8:00:40am 9-28-92 NUPRA 2.0 VPWR
Quantification Date: 9-28-92 8:00:39am TOTAL CMF = 6.11E-008



NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T2A: RECOVERABLE LOSS OF MAIN FEEDWATER EVENT TREE

FIGURE 3.1-T2A

TABLE 3.1.1-12
SUMMARY OF LOSS OF SUPPORT SYSTEMS AS INITIATORS

Support System Loss Considered	Impact on Normal Operation	Attendant Important System Failures	Initiating Event Group
4160 V Bus 1H	IRPI loss with total 4160 1H Bus loss could result in manual Reactor Trip or Shutdown	Charging Pump A ECCS Train A 480 V 1H 480 V 1H1 480 V 1H1-1 480 V 1H1-2S 480 V 1H1-4	Represented by the T9A Initiator. Impact on ESGR cooling also considered in the T8 Initiator.
4160 V Bus 1J	Isolation of RCP CC cooling could result in manual Reactor Trip or Shutdown	Charging Pump B ECCS Train B 480 V 1J 480 V 1J1 480 V 1J1-1 480 V 1J1-2S	Represented by the T9B Initiator. Impact on ESGR cooling also considered in the T8 Initiator.
480 V Bus 1H	IRPI loss with total 480 1H bus loss could result in manual Reactor Trip or Shutdown	Some ECCS Train A 480 V 1H1-1 480 V 1H1-4	Included within the T9A Initiator.
480 V Bus 1H1	No direct impact	Some ECCS Train A 480 V 1H1-2S	Not included as an Initiator. Disables some standby ECCS equipment, but doesn't cause transient or direct reactor trip.

TABLE 3.1.1-12 (Continued)
SUMMARY OF LOSS OF SUPPORT SYSTEMS AS INITIATORS

Support System Loss Considered	Impact on Normal Operation	Attendant Important System Failures	Initiating Event Group
480 V Bus 1H1-1	IRPI loss with total 480 1H1-1 Bus loss could result in manual Reactor	ESGR Chiller Trains A & C	Included within the T9A Initiator.
480 V Bus 1H1-2S	No direct impact	Some ECCS Train A Same as 480 V Bus 1H1.	
480 V Bus 1H1-4	No direct impact	ESGR AHU 6 ESGR Chiller Train C	Included within the T9A Initiator. Impact on ESGR cooling also considered in the T8 Initiator.
480 V Bus 1J	Isolation of RCP cooling could result in manual Reactor Trip or Shutdown	Some ECCS Train B 480 V 1J1-1 480 V 1J1-2S	Included within the T9B Initiator.
480 V Bus 1J1	No direct impact	Some ECCS Train B	Same as 480 V Bus 1H1.

TABLE 3.1.1-12 (Continued)
SUMMARY OF LOSS OF SUPPORT SYSTEMS AS INITIATORS

Support System Loss Considered	Impact on Normal Operation	Attendant Important System Failures	Initiating Event Group
480 V Bus 1J1-1	Isolation of RCP cooling could result in manual Reactor Trip or Shutdown	ESGR AHU 7 ESGR Chiller Train B	Included within the T9B Initiator.
480 V Bus 1J1-2S	No direct impact	Some ECCS Train B	Same as 480 V Bus 1H1.
120 VAC Vital Bus 1-I	Manual Reactor Trip on loss of RCP Cooling	MS Atmospheric Dump Valve A CC to RCP Thermal Barriers isolated	Included within the T3 Initiator.
120 VAC Vital Bus 1-II	No direct impact	MS Atmospheric Dump Valve B	None
120 VAC Vital Bus 1-III	Manual Reactor Trip on loss of RCP Cooling	MS Atmospheric Dump Valve C CC to RCP Thermal Barriers isolated	Included within the T3 Initiator.
125 VDC Bus 1-I	Reactor Trip on loss of MFW	ECCS Train A 4160 V switchgear MS Condenser Dump Valves	Represented by the T5A Initiator.

TABLE 3.1.1-12 (Continued)
SUMMARY OF LOSS OF SUPPORT SYSTEMS AS INITIATORS

Support System Loss Considered	Impact on Normal Operation	Attendant Important System Failures	Initiating Event Group
125 VDC Bus 1-II	No direct impact	MFW Pump B Condensate Pump B	Not included as an Initiator. Standby MFW & Condensate Pumps available with autostart.
125 VDC Bus 1-III	Reactor Trip on loss of MFW	ECCS Train B 4160 V switchgear MS Condenser Dump Valves	Represented by the T5B Initiator.
Service Water	Manual Reactor Trip or Shutdown on loss of CC to RCPs, loss of Instrument Air or loss of ESGR cooling	Charging Pumps A/B/C CC Heat Exchangers ESGR Chillers A/B/C Instrument Air Compressors Recirculation Spray Heat Exchangers RCP Thermal Barriers RHR Pumps and Heat Exchangers cooling for SGTR	Represented by the T6 Initiator.
Component Cooling Water	Manual Reactor Trip or Shutdown on loss of RCP cooling	RCP Thermal Barriers RHR Pumps and Heat Exchangers cooling for SGTR	Impact on RCP Thermal Barriers considered in the T4 Initiator.

TABLE 3.1.1-12 (Continued)
SUMMARY OF LOSS OF SUPPORT SYSTEMS AS INITIATORS

Support System Loss Considered	Impact on Normal Operation	Attendant Important System Failures	Initiating Event Group
Emergency Switchgear Room Cooling	Manual Reactor Trip or Shutdown due to switchgear thermal overload	All AC ECCS switchgear in ESGR	Represented by the T8 Initiator.
Containment Instrument Air	Manual Reactor Trip or Shutdown on loss of RCP cooling	Pressurizer PORV (backup nitrogen supply) RCP Thermal Barriers	Impact on RCP Thermal Barriers considered in the T4 Initiator.
Instrument Air Outside Containment	Reactor Trip on loss of MFW or MS isolation	RCP Thermal Barriers RHR Pump and Heat Exchanger cooling for SGTR MS Condenser Dump Valves MS Atmospheric Dump Valves (backup air (supply)	Included within the T2 Initiator. Impact on RCP Thermal Barriers considered in the T4 Initiator.
Bearing Cooling Water	Reactor Trip on loss of MFW	MFW Pumps Condensate Pumps	Included within the T2 Initiator.

TABLE 3.1.1-15
TRANSIENT SUCCESS CRITERIA

Reactivity Control	Core Heat Removal Early	Core Heat Removal Late	Secondary Heat Removal	RCS (Integrity)	Containment Condition
RPS Scram with < 2 rod failure to insert[a]	RCS - Natural Circ.		1/3 MFW pumps[b,f] OR 1/3 AFW pumps to 1/3 SGs[c]	RCS PORV Closure Note 1	Not Required
RPS Scram	1/3 Charging Pumps AND 1 RCS PORV (Feed & Bleed)[e]	Recirc. through 1/3 charging pumps - AND 1/2 Lo Head SI Pumps[d] (Note 3)	Not Required	Note 2	Recirculation through 1/2 IRS OR 1/2 ORS[e]

Notes:

1. Failure of RCS Integrity by failure of RCS PORV to close transfers to S2 event tree.

2. Feed & Bleed operation fails RCS Integrity through continued RCS PORV use.

3. For Transients, RCS depressurization before recirculation is not certain, so only high head safety recirculation is modeled. Also, ORS can be manually aligned to act as a backup for Lo Head Recirc for NAPS Unit 1.

References:

a. WCAP-9691 p. A-11
b. WCAP-9691 p. A-12
c. WCAP-9691 p. A-15
d. WCAP-9744
e. Surry Analysis File 321MAF.1
f. NAPS UFSAR

TABLE 3.1.2-1 (Continued)
LIST OF INITIATING EVENT CLASSES

INITIATING EVENT GROUP	DESCRIPTIONS	EVENT TREE
A	Large LOCA 6" - 20"	A
S1	Medium LOCA 2" - 6"	S1
S2	Small LOCA 3/8" - 2"	S2
V	Interfacing System LOCA	Vx
R	Reactor Vessel Rupture	Rx
TL	Transient with failure to Scram at Power < 40 percent	TL
TH	Transient with failure to Scram at Power > 40 percent	TH

* These event trees are discussed in one section of the report, as they are very similar.

** T1A is not a true initiating event, but is a consequential event from T1.

TABLE 3.1.2-2
EVENT TREE HEADINGS

Abbreviation	Headings	Description of Event
A	Large LOCA	Initiating Event-large LOCA
B	Offsite Power Recovery	Failure to recover an ESF bus following station black-out by recovering offsite power.
Ch	Containment Heat Removal	Failure of Service Water to an operable Recirculation Spray heat exchanger.
DG	EDG 1H or 1J Available	Failure of at least one diesel generator to start and run following loss of offsite power leading to station blackout.
Dh	Hot Leg Recirculation	Failure of the operator to switch to hot leg recirculation following a large LOCA.
D1	High Pressure Injection	Failure of Charging Pumps to inject in the appropriate mode.
D2	Accumulators Inject	Failure of Accumulators to inject in the appropriate mode.
D3	Low Head SI	Failure of low head SI pumps to inject.
D4	Emergency Boration	Failure to shutdown following ATWS by boron addition.
Fm	Break Size Partition	Percentage of small breaks not causing a CDA Hi Hi signal.
Hv	ESGR Cooling	Failure to provide HVAC to the ESGR using 1/2 AHUs and 1/3 chillers.
H1	Low Head Recirculation	Failure of low head pumps in the recirculation mode.

TABLE 3.1.2-2 (Continued)
EVENT TREE HEADINGS

Abbreviation	Headings	Description of Event
H2	High Head Recirculation	Failure of low head and charging pumps in the high pressure recirculation mode.
K	Reactor Subcritical	Failure of control rods to insert as result of Reactor Protection System failure.
L	Auxiliary Feedwater System Available	Failure of Auxiliary Feedwater System for transients or small or medium LOCAs with reactor trip.
Lt	Turbine-Driven AFW available	Failure of the Turbine-Driven Auxiliary Feedwater Pump to start and run following station blackout.
M	Main Feedwater System Available	Failure of Main Feedwater.
MS1	Manual Scram	Failure of the operator to remove power from the control rod drive mechanisms.
O	Cooldown and Depressurize	Operator fails to cooldown and depressurize the reactor after a small break or in response to a loss of RCP seal cooling.
O2	Late Cooldown	Failure of operator to cooldown and depressurize in response to a ruptured steam generator.
P	Pressurizer PORVs	Failure of the operator to open 1/2 pressurizer PORVs to cause RCS feed and bleed.
Pr	Pressure Relief	Failure of adequate pressure relief following an ATWS event.

TABLE 3.1.2-2 (Continued)
EVENT TREE HEADINGS

Abbreviation	Headings	Description of Event
Q	RCS Boundary Intact	Failure of pressurizer PORV to close after opening during a transient.
Qs	Quench Spray	Failure of 1/2 trains of Quench Spray.
Rc	Room Cooling Restored	Recovery of ESGR cooling or SW (resulting in reactor trip and loss of emergency power) prior to core uncovery and vessel failure, or containment failure.
Rs	Recirculation Sprays Operable	Failure of at least one train of Recirculation Sprays to remove heat from Containment.
Rv	Reactor Vessel Integrity	Consideration of PTS following a rapid RCS cooldown.
RX	Reactor Vessel Rupture	Initiating event is a Reactor Vessel rupture.
SGI	Steam Generator Isolation	Failure to isolate the ruptured Steam Generator.
Slc	No Potential for RCP Seal Failure	Failure to establish seal cooling from operable Unit 2 CC pumps.
S1	Medium LOCA	Initiating event is a medium LOCA (2" to 6").
S2	Small LOCA	Initiating event is a small LOCA (3/8" to 2").
T	Transients	Representative initiating event for general transient event tree.
Tt	Turbine Trip	Turbine fails to trip.

TABLE 3.1.2-2 (Continued)
EVENT TREE HEADINGS

Abbreviation	Headings	Description of Event
T1	Loss of Offsite Power	Initiating event is Loss of of all Offsite Power.
T1A	Station Blackout	Loss of diesel generators 1H and 1J leading to station blackout at Unit 1.
T1Tr	Loss of ESGR Cooling Transfer from T1 Event Tree	Transfer of T1Hv sequence, Loss of Offsite Power with consequential loss of Emergency Switchgear Room Cooling.
T2	Loss of MFW	Initiating event is non-recoverable loss of Main Feed-water.
T2A	Recoverable Loss of MFW	Initiating event is recoverable loss of Main Feedwater following Feedwater isolation.
T2ATr	Loss of ESGR Cooling Transfer from T2A Event Tree	Transfer of T2AHv sequence, recoverable loss of Main Feedwater with coincidental loss of Emergency Switchgear Room Cooling.
T2Tr	Loss of ESGR Cooling Transfer from T2 Event Tree	Transfer of T2Hv sequence, non-recoverable loss of Main Feedwater with coincidental loss of Emergency Switchgear Room Cooling.
T3	Transient with MFW Available	Initiating event is Transient with Main Feedwater available.
T3Tr	Loss of ESGR Cooling Transfer from T3 Event Tree	Transfer of T3Hv sequence, transient with Main Feedwater available, with coincidental loss of Emergency Switchgear Room Cooling.
T4	Loss of RC Pump Seal Cooling	Initiating event is loss of RCP seal injection and thermal barrier cooling.

TABLE 3.1.2-2 (Continued)
EVENT TREE HEADINGS

Abbreviation	Headings	Description of Event
T5A	Loss of DC Bus I	Initiating event is loss of DC Bus 1-I.
T5B	Loss of DC Bus III	Initiating event is loss of DC Bus 1-III.
T6	Loss of Service Water	Service Water is lost from both the reservoir and Lake Anna.
T7	Steam Generator Tube Rupture	Initiating event is a steam generator tube rupture.
T8	Loss of Emergency Switch-gear Room Cooling	Loss of HVAC to the Emergency Switchgear Room.
T9A	Loss of Power from 4160 V Emergency Bus 1H	Loss of feeder power to or failure of 4160 V emergency bus 1H.
T9ATr	Loss of ESGR Cooling Transfer from T9A Event Tree	Transfer of T9AHv sequence, loss of feeder power to or failure of 4160 V Emergency Bus 1H, with consequential loss of Emergency Switchgear Room Cooling.
T9B	Loss of Power from 4160 V Emergency Bus 1J	Loss of feeder power to or failure of 4160V emergency bus 1J.
T9BTr	Loss of ESGR Cooling Transfer from T9B Event Tree	Transfer of T9BHv sequence, loss of feeder power to or failure of 4160 V Emergency Bus 1J, with consequential loss of Emergency Switchgear Room Cooling.
TL	Low power transients (for ATWS)	Initiating event is all transients at power lower than or equal to 40 percent.
TH	High power transients (for ATWS)	Initiating event is all transients at power greater than or equal to 40 percent.

TABLE 3.1.2-2 (Continued)
EVENT TREE HEADINGS

Abbreviation	Headings	Description of Event
VX	Interfacing System LOCA	Initiating event is an Interfacing System LOCA.
Vi	Isolation of LOCA	Failure to isolate interfacing LOCA.
W	RHR Cooling	Failure of 1/2 Residual Heat Removal Trains.
Y	Core Cooling Recovery	Failure of the operator to use steam to rapidly cooldown and depressurize the RCS as directed by 1-FR-C.1 or C.2.

TABLE 3.3.1-1
DEFINITION OF PROBABILITY MODELS AND THEIR PARAMETERS

Basic Event	Probability Models	Data Required
Initiating Event	Poisson Model $$P(r) = \frac{(ft)^{r} e^{-ft}}{r!}$$ f: frequency	Number of events r in time t
Standby component fails on demand	1) Constant probability failure on demand, or $U = \frac{n}{N}$	1) Number of events n in total number of demands N
Standby component fails in time, or component changes state between tests (faults revealed on functional test only)	2) Constant standby failure rate $$U = 1 - \frac{1-e^{-\lambda_s T_I}}{\lambda_s T_I}$$ T_I : Time between tests λ_s: Standby failure rate	2) Number of events n in total time in standby T_s
Component in operation fails to run, or component changes state during mission (state of component continuously monitored)	Constant failure rate $U = 1-\exp(-\lambda_o T_m)$ $\approx \lambda_o T_m$ T_m: Mission time λ_o: Operating failure rate	Number of events n in total exposure time T_e (Time standby component is operating, or time the component is on line)