October 18, 1996

Mr. Michael B. Sellman
Vice President Operations
Entergy Operations, Inc.
P. O. Box B
Killona, LA 70066

SUBJECT: REVIEW OF PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF EVENT AT WATERFORD STEAM ELECTRIC GENERATING STATION, UNIT 3

Dear Mr. Sellman:

Enclosed for your information is a copy of the final Accident Sequence Precursor (ASP) analysis of the operational event at Waterford Steam Electric Generating Station, Unit 3 reported in Licensee Event Report (LER) No. 382/95 002. This final analysis (Enclosure 1) was prepared by our contractor at the Oak Ridge National Laboratory (ORNL), based on review and evaluation of your comments on the preliminary analysis and comments received from the NRC staff and from our independent contractor, Sandia National Laboratories (SNL). Enclosure 2 contains our responses to your specific comments. Our review of your comments employed the criteria contained in the material which accompanied the preliminary analysis. The results of the final analysis indicate that this event is a precursor for 1995.

Please contact me at (301) 415-3025 if you have any questions regarding the enclosures. We recognize and appreciate the effort expended by you and your staff in reviewing and providing comments on the preliminary analysis.

Sincerely,

ORIGINAL SIGNED BY:

Chandu P. Patel, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures: As stated

cc w/encls: See next page

DISTRIBUTION:

Docket File	PUBLIC	PD4-1 r/f
C. Patel	W. Beckner	C. Hawes
J. Roe	J. Dyer, RIV	E. Adensam (EGA1)
P. O'Reilly	S. Mays	

Document Name: WATASPFN.LTR

OFC	PM:PD4-1 gop	(A)LA/PD4-1
NAME	CPatel:sp	CHawes CMH
DATE	10/18/96	10/18/96
COPY	YES/NO	YES/NO

OFFICIAL RECORD COPY

DFO1 1/1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 18, 1996

Mr. Michael B. Sellman
Vice President Operations
Entergy Operations, Inc.
P. O. Box B
Killona, LA 70066

SUBJECT: REVIEW OF PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF EVENT AT WATERFORD STEAM ELECTRIC GENERATING STATION, UNIT 3

Dear Mr. Sellman:

Enclosed for your information is a copy of the final Accident Sequence Precursor (ASP) analysis of the operational event at Waterford Steam Electric Generating Station, Unit 3 reported in Licensee Event Report (LER) No. 382/95 002. This final analysis (Enclosure 1) was prepared by our contractor at the Oak Ridge National Laboratory (ORNL), based on review and evaluation of your comments on the preliminary analysis and comments received from the NRC staff and from our independent contractor, Sandia National Laboratories (SNL). Enclosure 2 contains our responses to your specific comments. Our review of your comments employed the criteria contained in the material which accompanied the preliminary analysis. The results of the final analysis indicate that this event is a precursor for 1995.

Please contact me at (301) 415-3025 if you have any questions regarding the enclosures. We recognize and appreciate the effort expended by you and your staff in reviewing and providing comments on the preliminary analysis.

Sincerely,

Chandu P. Patel

Chandu P. Patel, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosures: As stated

cc w/encls: See next page

cc:

Administrator
Louisiana Radiation Protection Division
Post Office Box 82135
Baton Rouge, LA 70884-2135

Vice President, Operations
Support
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286

Director
Nuclear Safety
Entergy Operations, Inc.
P. O. Box B
Killona, LA 70066

Wise, Carter, Child & Caraway
P. O. Box 651
Jackson, MS 39205

General Manager Plant Operations
Entergy Operations, Inc.
P. O. Box B
Killona, LA 70066

Licensing Manager
Entergy Operations, Inc.
P. O. Box B
Killona, LA 70066

Winston & Strawn
1400 L Street, N.W.
Washington, DC 20005-3502

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

Resident Inspector/Waterford NPS
Post Office Box 822
Killona, LA 70066

Parish President Council
St. Charles Parish
P. O. Box 302
Hahnville, LA 70057

Executive Vice-President
and Chief Operating Officer
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

Chairman
Louisiana Public Service Commission
One American Place, Suite 1630
Baton Rouge, LA 70825-1697

LER No. 382/95-002

Event Description: Reactor trip, breaker failure and fire, degraded offsite power, and degraded shutdown cooling

Date of Event: June 10, 1995

Plant: Waterford 3

Event Summary

A switchyard lightning arrestor failure caused a trip from 100% power at Waterford 3. Delayed opening of the 4.16-kV unit auxiliary transformer (UAT) feeder breaker paralleled the grid with the main generator which was speeding up. The resulting out-of-phase condition caused an overvoltage and fault-level currents that started a fire that damaged cables and switchgear for nonvital Train A. Power wa initially lost to Train A safety loads, but was recovered when emergency diesel generator (EDG) A started and loaded. Condenser vacuum was subsequently lost as a result of loss of power to balance of plant Train A equipment and the unexpected bypass of circulating water flow around the condenser. Plant cooldown was delayed when low hydraulic fluid levels prevented proper operation of shutdown cooling (SDC) system isolation valves. The conditional core damage probability (CCDP) estimated for this combined event is 2.5×10^{-5}. The increase in CCDP over a one-year period because of the unavailability of the SDC isolation valves is 1.7×10^{-5}.

Event Description

Waterford 3 was operating at 100% power on June 10, 1995. At 0858 h, a lightning arrestor failed at the Waterford Substation. The resulting grid disturbance caused the sudden pressure relay on Main Transformer A to actuate the main generator lockout relays. This actuation resulted in the trip of the main generator output breaker and exciter field breaker initiation of a fast dead bus transfer, and trip of the main turbine.

The B 6.9-kV and 4.16-kV buses successfully transferred to Startup Transformer (SUT) B. However, during the transfer of 4.16-kV bus A2 to SUT A, the A2 SUT feeder breaker closed before the A2 UAT breaker opened. The UAT and SUT breakers tripped and power was lost to bus A2.

The reactor tripped on low Departure from Nucleate Boiling Ratio (DNBR) signals, caused by low reactor coolant pump speed. Bus A1 (6.9-kV) deenergized, which tripped two reactor coolant pumps, circulating water pumps, condensate pumps, and condenser air evacuation pumps. Main feedwater (MFW) pump A also tripped, apparently from loss of power to the pump speed pickups.

Vital 4.16-kV bus A3 deenergized when power was lost to bus A2. EDG A started and reenergized the required safety-related loads via the load sequencer. Emergency feedwater (EFW) actuated, and within 13 min both MFW isolation valves had been closed as a result of high steam generator (SG) level.

Approximately one min after the trip, all turbine generator building (TGB) switchgear room fire alarm annunciators actuated. The TGB operator reported heavy smoke coming from the switchgear room 7 min later. Two auxiliary operators were directed to set up blowers to help dissipate the smoke, don protective clothing, and enter the switchgear room to investigate the cause of the smoke.

At 0935 (+37 min), the TGB auxiliary operator reported a fire in the 2A switchgear and in the cables above the switchgear. The fire was caused by the delayed opening of the A2 UAT breaker, which resulted in a voltage across the

breaker during opening beyond the breaker's design and a subsequent high-energy fault. The breaker failed internally and caused the fire (the breaker failure and fire are described in more detail in Additional Event-Related Information).

Upon notification of an actual fire in the switchgear room, the shift supervisor sounded the plant fire alarm (post-event review indicated that the fire alarm should have been sounded when smoke was first detected), dispatched the fire brigade, and directed the motor-operated disconnect for SUT A to be opened to ensure electrical isolation of the A2 bus. The control room supervisor left the control room to serve as fire brigade leader.

The fire brigade attempted to extinguish the fire using halon, carbon dioxide and dry-chemical fire extinguishers. When the fire brigade leader arrived at the fire scene, he immediately notified the control room to request offsite fire department assistance. The Hahnville Fire Department was contacted at 0941 h (+43 min) via 911 for support.

The Hahnville Fire Department arrived on-site 17 min later and recommended that water be used to extinguish the fire. Carbon dioxide and dry chemical extinguishers were being unsuccessfully used by the fire brigade to fight the fire (although experience gained from the 1976 Browns Ferry fire and other fires indicated that the use of water was necessary on large cable fires). The use of water was delayed for an additional 20 min. (Ref. 2 noted that interviews conducted with plant operators after the event indicated a general reluctance on the part of the operators to apply water to an electrical fire, based on previous training that had emphasized the use of water was a last resort on electrical fires.) The fire was extinguished within 4 min, once water was used.

At 1112 h (+2.2 h), condenser vacuum was broken after it had fallen to 20 in. Hg. A condenser low vacuum alarm had actuated at 0940 h, shortly after the fire was reported. The loss of vacuum was initially attributed to the unavailability of the two circulating water and condenser air evacuation pumps, resulting from the deenergization of bus A1 at the beginning of the event, combined with several steam loads that were still discharging to the condenser, and the operators made a decision not to divert resources from fighting the fire to attempt to recover condenser vacuum. In actuality, when the two circulating water pumps deenergized at 0858 h, their associated motor-operated discharge valves also deenergized and remained open, resulting in a bypass of circulating water flow.

At 1147 h (+2.8 h), the main steam isolation valves were closed and the atmospheric dump valves used for decay heat removal. At 2348 h (14.8 h after the event began), EFW was secured, and Condensate Pump B (the operable condensate pump) was used to supply water to Steam Generator B.

By 1257 h on June 11, 1996, the plant had been cooled down and depressurized to shutdown cooling entry conditions. At 1311 h, shutdown cooling suction header isolation valve SI-405B was commanded open while placing the shutdown cooling system in service. This valve closed after only partially opening and was declared inoperable. The equivalent valve in Train A, SI-405A was then opened. Several hours later, this valve's hydraulic pump was observed to be continually running instead of cycling as designed. Valve SI-405A was also closed and declared inoperable.

A containment entry was made to inspect the two valves, and low hydraulic fluid levels were found in both valve actuator reservoirs. Approximately 200 $in.^3$ of hydraulic fluid were added to the reservoir for SI-405B, and the valve operated satisfactorily. Shutdown cooling loop B was placed in service between 1800 h and 2400 h on June 12, 1996.

When valve SI-405A was tested after fluid had been added to its reservoir, the valve opened slowly. Additional troubleshooting indicated that the valve's hydraulic pump had been damaged by the continuous operation caused by the low hydraulic fluid level. The pump was replaced and the valve was returned to service shortly after midnight on June 13, 1995. Cooldown to Mode 5 began, with Train A components still powered by EDG A.

Additional Event-Related Information

The Waterford 3 fast dead bus transfer scheme consists of automatic or manual transfer of in-house loads from the UATs to the SUTs. During a fast dead bus transfer, the UAT feeder breakers to the A1 and B1 6.9-kV and the A2 and B2 4.16-kV buses are designed to open in five cycles, and the SUT feeder breakers are designed to close in seven cycles, resulting in a two-cycle nominal deadband on the respective buses.

This scheme is a "simultaneous" (simultaneous trip and close signals with no interlock) bus transfer scheme (zero to two-cycle deadband) instead of the "sequential" (the tripping breaker interlocked with the closing breaker) bus transfer (greater than six-cycle deadband) commonly used in the United States. The simultaneous bus transfer scheme is used in all Swedish nuclear power plants. To prevent exceeding the fault duty of associated equipment and buses when two sources are in parallel, the Swedish design includes an interlock that limits the time period during which both breakers are permitted to remain closed to less than 0.1 sec. The Waterford 3 design does not include the interlock, and both breakers appeared to have remained closed for about 0.3 sec during the event.

During the time that the two breakers were simultaneously closed, the A2 bus connected SUT A to the main generator, which then provided power to the grid via the UAT and bus A2. During this time the main generator was rotating faster than the system frequency due to the load rejection. When the UAT breaker opened, the main generator was approaching 180 degrees out of phase with the system (~8 kV across the breaker). The interrupted current was ~28,800 A. This overvoltage due to the out-of-phase condition and the overcurrent resulted in an internal breaker failure and the creation of ionizing gases, which caused the fire in the A2 switchgear. A preliminary investigation indicated that the most probable cause for the slow opening time of the UAT breaker was restricted movement of the trip latch roller bearing.

The amount of damage to the breaker and surrounding equipment indicates that (1) the fault current through the breaker was extremely high and (2) significant arcing occurred for some period of time. The arc chutes and main contacts on all phases were destroyed, and the contact structures, breaker frame, and cubicle were also significantly damaged. The main bus and bus enclosure also appeared to have experienced severe arcing damage.

The fire that resulted from the breaker failure damaged the bus and surrounding cables and components. Two cubicles (the failed breaker was an end cubicle) were heavily damaged, and approximately 10 ft of the cable bus duct was destroyed. Cables in approximately a 5-ft diameter column above the breaker had visible fire damage over their entire 10-ft vertical run. At the top of the vertical run, the cables were routed through a horizontal cable tray. Approximately 8 ft of cable in the horizontal tray had visible fire damage. General smoke and slight heat damage to the exterior of the remaining cubicles in the A2 bus occurred. In addition, damage included external heat to the jackets of four of the 15 feeder cables from the SUT to the A2 bus, and burn marks on the conduit of the cables that supply 6.9-kV power to the reactor coolant pump 1A and 2A motors.

The TGB switchgear room contains both the A and B trains of nonvital switchgear. The ceiling of the room is approximately 25 ft above the floor; the tops of the switchgear cubicles are approximately 7 ft high. A 10-ft-high concrete block radiant heat shield, located 6 ft from the front of each set of cubicles, separates the two trains. The fire did not affect the Train B switchgear or cables.

The TGB switchgear room had an ionization-type fire detection system, with detectors mounted on the ceiling, but no fire suppression system. The fire detection computer recorded the first fire alarm 55 sec after the reactor trip. Within 7 sec, all 36 fire detectors in the room had alarmed. Twenty-six min after the trip, the first detector went into "device communication error"; it apparently failed at that time and melted. By 0942 h (+44 min), all detectors in the room had apparently failed.

Subsequent to the fire, the licensee found tape over the fire alarm annunciator buzzer located on the fire detection computer in the control room. Because of the tape, the alarm volume was low and nonintrusive. Due to the alarm panel's placement in the control room, alarm lights were also not readily visible. These factors, combined with the fact that the fire was not declared until after the auxiliary operators entered the switchgear room and observed it (36 min after the fire alarm annunciators actuated), contributed to the delay in responding to the fire.

Unlike many PWRs, the Waterford primary pressure relief system includes only code safety valves; no power-operated relief valves (PORVs) are incorporated in the design. The lack of PORVs prevents the use of feed and bleed for core cooling in the event both main and emergency feedwater systems are unavailable. If both of these systems were to fail at Waterford, safety-related secondary-side atmospheric dump valves could be used to depressurize the steam generators to below the shutoff head of the condensate pumps. These pumps could then be used for decay heat removal.

Modeling Assumptions

The event was modeled both as (1) a reactor trip, loss of main feedwater (caused by the loss of condenser vacuum 2.2 h after the trip), loss of offsite power to Train A safety-related components, and unavailability of SDC isolation valves SI-405A and SI-405B during the cooldown (initiating event assessment) and (2) a long-term unavailability of the SDC isolation valves (condition assessment).

Reactor trip, loss of feedwater, and unavailable SDC isolation valves (initiating event assessment)

The ASP model for Waterford 3 was revised to address the potential failure of the main feedwater isolation valves (MFIVs) to open. These valves were closed because of high SG levels shortly into the event. Failure of these valves to open would prevent use of the auxiliary feedwater (AFW) system and the condensate system for SG makeup. Short-term ex-control room recovery of EFW (beyond the use of the AFW pump), high-pressure injection (HPI), and the condensate system, had these systems failed, was not considered feasible because significant crew resources were being used to fight the fire.

Redundant shutdown cooling isolation valves SI-405A and SI-405B were both assumed to have failed. This assumption may be conservative for SI-405A because it initially operated. However, the licensee determined that the valve's hydraulic motor was sufficiently damaged to require replacement before the plant cooldown continued.

The ASP models for a transient do not currently address the potential unavailability of offsite power to an individual train, as was observed in this event. During the event, power to safety-related Train A loads was provided by EDG A. The potential failure of the EDG to power Train A was modeled by adding a basic event to the model, EPS-DGN-FC-3AFR, to represent the potential failure of the EDG to start and run following the breaker failure.

The mission time for the initiating event assessment was assumed to be the time from the reactor trip until shutdown cooling was established, ~60 h. EDG A continued to supply Train A loads beyond this time. However, the added risk is considered to be small compared with the risk before shutdown cooling was established. [The Accident Sequence Precursor (ASP) program addresses shutdown-related events that are considered unusual and significant. Events such as this one, where one train is powered from its EDG, are not typically selected for analysis.]

The following changes were made to basic events to reflect conditions observed during the event:

Basic event	Revised probability	Description (reason for change)
AFW-TRAIN-FC-ALL	9.8×10^{-3}	Nonsafety AFW system fails to provide flow to SGs (revised to reflect extended mission time)
COND-PFS-FC-SYS	7.8×10^{-3}	Secondary heat removal using condensate system fails (revised to reflect extended mission time and low probability of initial condensate system failure)
EFW-MDP-FC-A, B	5.0×10^{-3}	EFW motor-driven pump train failures (revised to reflect extended mission time)
EFW-PMP-CF-ALL	2.0×10^{-4}	Common cause failure of EFW pumps (revised to reflect extended mission time)
EFW-TDP-FC-TDP	4.1×10^{-2}	EFW turbine-driven pump train failures (revised to reflect extended mission time)
EFW-XHE-NOREC	TRUE	Ex-control room resources required for recovery utilized to fight fire
EPS-DGN-FC-3AFR	1.4×10^{-1}	EDG A fails to start and run (revised to reflect extended mission time)
HPI-XHE-NOREC	TRUE	Ex-control room resources required for recovery utilized to fight fire
MFW-SYS-TRIP	TRUE	MFW system trips (MFW unavailable because of loss of condenser vacuum)
MFW-VLV-CF-MFIV	2.6×10^{-4}	Common cause failure of the MFW isolation valves to open (basic event added to model because these valves affect both the AFW and the condensate systems)
MFW-XHE-NOREC	TRUE	Operator fails to recover MFW (MFW not recoverable because of loss of vacuum)
RHR-MOV-CF-SUCT	TRUE	Common cause failure of residual heat removal (RHR) suction valves (set to TRUE to reflect the failure of SI-405A and SI-405B)

The mission time for the HPI pumps was not revised to reflect the 60-h mission time. If a transient-induced loss-of-coolant accident (LOCA) had occurred, the modeled plant response would have been accomplished in less than 24 h. With the SDC isolation valves unavailable following a transient-induced (small-break) LOCA, the operators would have transferred to high-pressure recirculation (HPR) once the refueling water storage pool was depleted. This transfer would have occurred ~6 h following the LOCA.

The licensee addressed this specific switchgear room fire in the Waterford Individual Plant Examination for External Events (IPEEE) (Ref. 3). In that document the licensee concluded that the fire – while extensive and not suppressed until

the cables from the UAT to the switchgear were fully involved - did not cause significant damage outside the plume/ceiling jet. Fire modeling also confirmed that a large TGB switchgear fire would not generate a hot gas layer that could fail cables outside the plume. Because of this, the IPEEE assumed that TGB switchgear fires would only cause damage to one train of offsite power. This assumption was used in this analysis as well. A sensitivity analysis, described in the Analysis Results, addresses the potential impact if the fire, or common cause breaker problems, had also resulted in a nonrecoverable loss of offsite power to Train B.

Long-term unavailability of the SDC isolation valves (condition assessment)

The SDC isolation valves were assumed to have been unavailable since the last refueling outage, in Spring 1994. The longest time period used to assess a condition (unavailability) in the ASP Program is one year, during which the plant is typically assumed to have been at power 70% of the time. In this event, however, Waterford was at power for the full 1-year period, resulting in an unavailability of 8760 h. (Because a duration of 8760 h is longer than that used in the analysis of a typical long-term condition, the analysis results cannot be directly compared with those of other long-term condition assessments.) This assumption presumes that the loss of hydraulic fluid from the valve actuators occurs during valve operation (not when the valves are inoperative) and that the fluid level during the previous use of the valves was barely acceptable. If the hydraulic fluid was lost when the valves were in standby, then the analysis duration is overestimated (the valves would then become unavailable at one-half of the duration since last use; this would result in a 50% reduction in the increase in core damage probability caused by the failed valves).

Consistent with the previous assessment, shutdown cooling isolation valves SI-405A and SI-405B were both assumed to be failed. This assumption was reflected by setting basic event RHR-MOV-CF-SUCT to TRUE. Plant response to all initiators addressed in the ASP model was considered impacted by the unavailability of the SDC isolation valves.

Analysis Results

The CCDP estimated for trip, fire and resulting loss of offsite power to Train A, loss of feedwater, and unavailability of the SDC isolation valves is 2.5×10^{-5}. The dominant sequence, highlighted on the event tree in Fig. 1 (transient sequence 19), contributes about 83% to the conditional probability estimate for the initiating event and involves

- the successful reactor trip,
- failure of EFW (including the AFW pump) to provide secondary-side cooling,
- MFW unavailability, and
- failure of the condensate system as an alternate source of cooling water.

The dominant cut sets involve failure to provide an alternate source of water to the EFW pumps following depletion of the condensate storage pool within the 60 h mission time and failure of the condensate system to provide flow to the steam generators (failure to initiate and equipment failure both contribute).

Table 1 provides the definitions and probabilities for selected basic events for the initiating event assessment. The conditional probabilities associated with the highest probability sequences are shown in Table 2, while Table 3 lists the sequence logic associated with the sequences listed in Table 2. Table 4 describes the system names associated with the dominant sequences. The minimal cut sets associated with each sequence are shown in Table 5.

The calculation for the reactor trip and fire is sensitive to the assumption that the fire or potential common cause breaker failures would not affect the availability of offsite power to Train B. If the fire could have affected Train B, or if slow breaker opening also resulted in the loss of Train B switchgear (which is believed to be unlikely), then the event could have been more significant. For example, an assumption of a 0.03 probability of nonrecoverable loss of offsite power to Train B (similar to Train A) results in an estimated CCDP of 1.4×10^{-4} (such an event would be considered significant from an ASP standpoint).

The unavailable SDC isolation valves (the condition assessment) result in an overall increase in core damage probability for the assumed 1-year period of 1.7×10^{-5} over the nominal core damage probability (CDP) estimated for the same period of 8.8×10^{-5}. This is the sum of the changes to the sequence probabilities (importance) shown in Table 7, which are calculated by subtracting the total CDP sequence value from the total CCDP sequence value for each sequence. The dominant core damage sequence involves

- a small-break LOCA initiating event,
- successful EFW and HPI operation,
- successful depressurization,
- failure to initiate SDC (which would avoid the use of high-pressure sump recirculation), and
- failure of high-pressure recirculation.

For most ASP analyses of conditions (equipment failures over a period of time during which postulated initiating events could have occurred), sequences and cut sets associated with the observed failures dominate the CCDP (the probability of core damage over the unavailability period, given the observed failures). The increase in core damage probability (CDP) because of the failures is therefore essentially the same as the CCDP, and the CCDP can be considered a reasonable measure of the significance of the observed failures.

For this event, however, sequences unrelated to the SDC isolation valves dominate the CCDP estimate. The increase in CDP given the failed SDC isolation valves, 1.7×10^{-5}, is, therefore, a better measure of the significance of the SDC valve problems.

Definitions and probabilities for selected basic events for the condition assessment are shown in Table 6. The conditional probabilities associated with the highest probability sequences are shown in Table 7. Table 8 lists the sequence logic associated with the sequences listed in Table 7. Table 9 describes the system names associated with the dominant sequences. Cut sets associated with each sequence are shown in Table 10.

Acronyms

AFW	auxiliary feedwater
ASP	accident sequence precursor
ATWS	anticipated transient without scram
CCDP	conditional core damage probability
CDP	core damage probability
cd	core damage
DNBR	departure from nucleate boiling ratio
EDG	emergency diesel generator
EFW	emergency feedwater
HPI	high pressure injection
HPR	high pressure recirculation
IPEEE	individual plant examination for external events
kV	kilovolts
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
MFIV	main feedwater isolation valves
MFW	main feedwater
PORV	power-operated relief valve
RCS	reactor coolant system
RHR	residual heat removal
RWSP	refueling water storage pool

SDC	shutdown cooling
SG	steam generator
SRV	safety relief valve
SUT	startup transformer
TGB	turbine generator building
UAT	unit auxiliary transformer

References

1. LER 382/95-002, Rev. 0, "Reactor Trip and Non-Safety Related Switchgear Fire," July 7, 1995.

2. NRC Augmented Inspection Team Report 50-382/95-15, July 5, 1995

3. *Waterford 3 Individual Plant Examination for External Events*, July 1995.

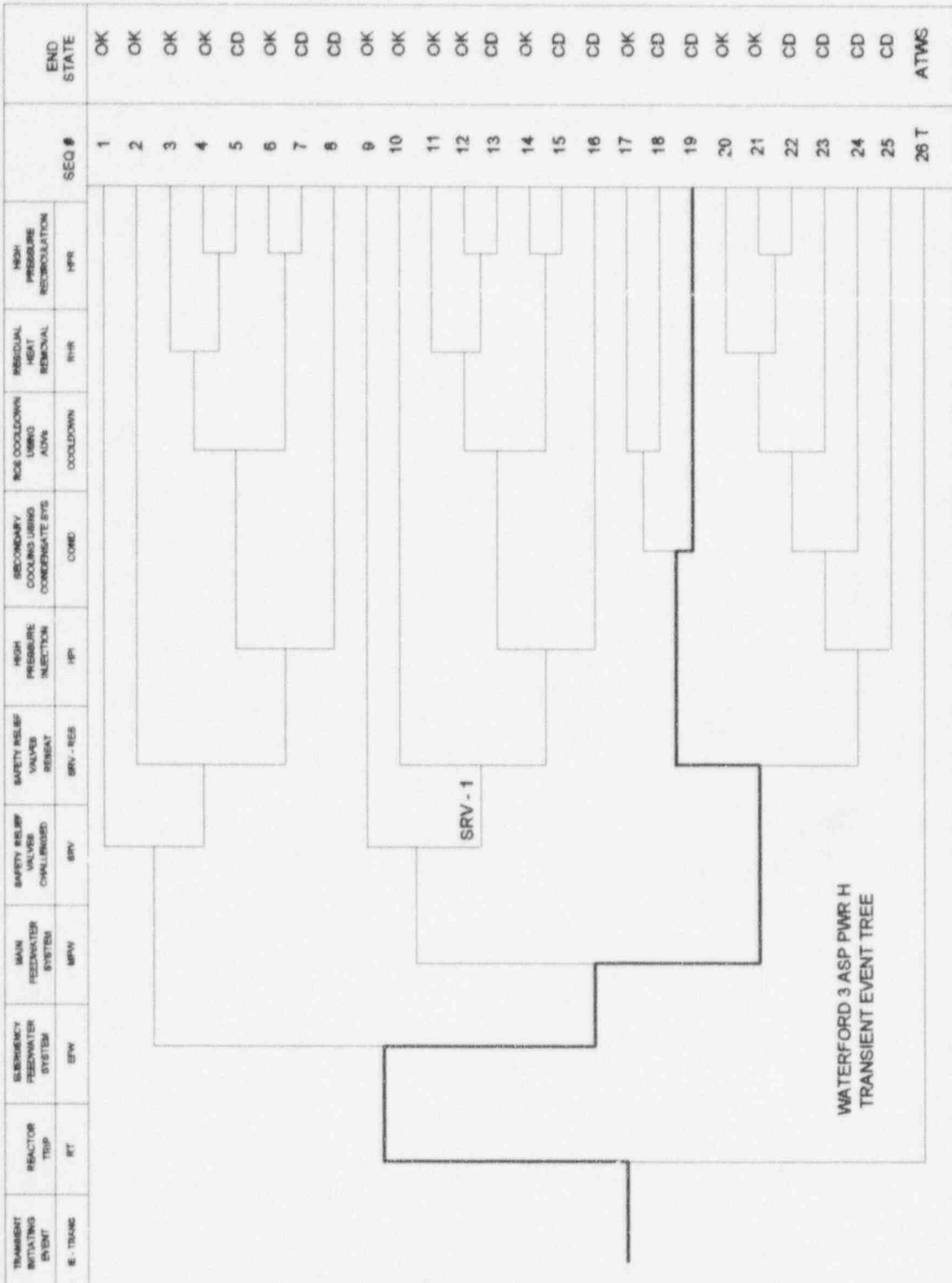

Fig. 1 Dominant core damage sequences for LER No. 382/95-002.

Table 1. Definitions and Probabilities for Selected Basic Events for the Initiating Event Assessment for LER 382/95-002

Event name	Description	Base probability	Current probability	Type	Modified for this event
IE-LOOP	Loss of Offsite Power Initiating Event	8.6 E-006	0.0 E+000	IGNORE	No
IE-SGTR	Steam Generator Tube Rupture Initiating Event	1.6 E-006	0.0 E+000	IGNORE	No
IE-SLOCA	Small Loss-of-Coolant Accident Initiating Event	1.0 E-006	0.0 E+000	IGNORE	No
IE-TRANS	Transient Initiating Event	6.8 E-004	1.0 E+000		Yes
AFW-TRAIN-FC-ALL	AFW Pump Train Fails to Provide Flow	8.7 E-003	9.8 E-003		Yes
COND-PFS-FC-SYS	Secondary Heat Removal Using Condensate System Fails	1.5 E-002	7.8 E-003		Yes
COND-XHE-XM	Operator Fails to Initiate Secondary Cooling	1.0 E-002	1.0 E-002		No
EFW-MDP-FC-A	EFW Motor-Driven Pump A Failures	3.9 E-003	3.9 E-003		No
EFW-MDP-FC-B	EFW Motor-Driven Pump B Failures	3.9 E-003	5.0 E-003		Yes
EFW-PMP-CF-ALL	Common Cause Failure of EFW Pumps	1.4 E-004	1.4 E-004		No
EFW-TDP-FC-TDP	EFW Turbine-Driven Pump Train Failures	3.6 E-002	4.0 E-002		Yes
EFW-XHE-NOREC	Operator Fails to Recover EFW System	2.6 E-001	1.0 E+000	TRUE	Yes
EFW-XHE-XA-CCW	Operator Fails to Initiate Backup Water Source	1.0 E-003	1.0 E-003		No
EPS-DGN-FC-3AFR	EDG 3A Fails to Start and Run	0.0 E+000	1.4 E-001	NEW	Yes
HPI-HDV-OC-SUCB	Refueling Water Storage Pool (RWSP) Suction Train B Failures	1.4 E-004	1.4 E-004		No
HPI-MDP-CF-ALL	Common Cause Failure of High Pressure Injection Motor-Driven Pumps	1.0 E-004	1.0 E-004		No

Table 1. Definitions and Probabilities for Selected Basic Events for the Initiating Event Assessment for LER 382/95-002

Event name	Description	Base probability	Current probability	Type	Modified for this event
HPI-MDP-FC-B	HPI Motor-Driven Pump B Train Failures	3.9 E-003	3.9 E-003		No
HPI-MOV-CF-ALL	Common Cause Failure of Injection Motor-Operated Valves	5.5 E-005	5.5 E-005		No
HPI-XHE-NOREC	Operator Fails to Recover the HPI System	8.4 E-001	1.0 E+000	TRUE	Yes
MFW-SYS-TRIP	Main Feedwater System Trips	2.9 E-001	1.0 E+000	TRUE	Yes
MFW-VLV-CF-MFIV	Common Cause Failure of MFIVs to Open	0.0 E+000	2.6 E-004	NEW	Yes
MFW-XHE-NOREC	Operator Fails to Recover Main Feedwater	3.4 E-001	1.0 E+000	TRUE	Yes
PCS-VCF-HW	Turbine Bypass Valves / Condensate / Circulation Failures	1.0 E-003	1.0 E-003		No
PCS-XHE-XM-CDOWN	Operator Fails to Initiate Cooldown	1.0 E-003	1.0 E-003		No
PPR-SRV-CO-TRAN	Safety Relief Valves (SRVs) Open During Transient	2.0 E-002	2.0 E-002		No
PPR-SRV-OO-1	SRV 1 Fails to Reseat	1.6 E-002	1.6 E-002		No
PPR-SRV-OO-2	SRV 2 Fails to Reseat	1.6 E-002	1.6 E-002		No
RHR-MOV-CF-SUCT	Common Cause Failure of RHR Suction Valves	1.2 E-003	1.0 E+000	TRUE	Yes

Table 2. Sequence Conditional Probabilities for the Initiating Event Assessment for LER 382/95-002

Event tree name	Sequence name	Conditional core damage probability (CCDP)	Percent contribution
TRANS	19	2.0 E-005	82.9
TRANS	18	2.2 E-006	9.1
TRANS	24	6.6 E-007	2.6
TRANS	08	4.7 E-007	1.9
Total (all sequences)		**2.5 E-005**	

Table 3. Sequence Logic for Dominant Sequences for the Initiating Event Assessment for LER 382/95-002

Event tree name	Sequence name	Logic
TRANS	19	/RT, EFW, MFW, /SRV-RES, COND
TRANS	18	/RT, EFW, MFW, /SRV-RES, /COND, COOLDOWN
TRANS	24	/RT, EFW, MFW, SRV-RES, /HPI, COND
TRANS	08	/RT, /EFW, SRV, SRV-RES, HPI

Table 4. System Names for the Initiating Event Assessment for LER 382/95-002

System name	Logic
COND	Secondary Heat Removal Using Condensate System Fails
COOLDOWN	RCS Cooldown to RHR Pressure Using Turbine-Bypass Valves, etc.
EFW	No or Insufficient EFW Flow
HPI	No or Insufficient HPI System Flow
MFW	Failure of the Main Feedwater System
RT	Reactor Fails to Trip During Transient
SRV	SRVs Open During Transient
SRV-RES	SRVs Fail to Reseat

Table 5. Conditional Cut Sets for Higher Probability Sequences for the Initiating Event Assessment for LER 382/95-002

Cut set number	Percent contribution	Conditional probability[a]	Cut sets[b]
TRANS Sequence 19		2.0 E-005	
1	48.2	1.0 E-005	EFW-XHE-XA-CCW, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, COND-XHE-XM
2	37.6	7.8 E-006	EFW-XHE-XA-CCW, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, COND-PFS-FC-SYS
3	6.8	1.4 E-006	EFW-PMP-CF-ALL, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, COND-XHE-XM
4	5.3	1.1 E-006	EFW-PMP-CF-ALL, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, COND-PFS-FC-SYS
5	1.2	2.6 E-007	EFW-XHE-XA-CCW, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, MFW-VLV-CF-MFIV
TRANS Sequence 18		2.2 E-006	
1	43.6	1.0 E-006	EFW-XHE-XA-CCW, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PCS-XHE-XM-CDOWN
2	43.6	1.0 E-006	EFW-XHE-XA-CCW, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PCS-VCF-HW
3	6.1	1.4 E-007	EFW-PMP-CF-ALL, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PCS-XHE-XM-CDOWN
4	6.1	1.4 E-007	EFW-PMP-CF-ALL, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PCS-VCF-HW
TRANS Sequence 24		6.6 E-007	
1	24.1	1.6 E-007	EFW-XHE-XA-CCW, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-OO-1, COND-XHE-XM
2	24.1	1.6 E-007	EFW-XHE-XA-CCW, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-OO-2, COND-XHE-XM
3	18.8	1.2 E-007	EFW-XHE-XA-CCW, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-OO-1, COND-PFS-FC-SYS
4	18.8	1.2 E-007	EFW-XHE-XA-CCW, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-OO-2, COND-PFS-FC-SYS
5	3.4	2.2 E-008	EFW-PMP-CF-ALL, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-OO-1, COND-XHE-XM
6	3.4	2.2 E-008	EFW-PMP-CF-ALL, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-OO-2, COND-XHE-XM

Table 5. Conditional Cut Sets for Higher Probability Sequences for the Initiating Event Assessment for LER 382/95-002

Cut set number	Percent contribution	Conditional probability[a]	Cut sets[b]
7	2.6	1.7 E-008	EFW-PMP-CF-ALL, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-OO-1, COND-PFS-FC-SYS
8	2.6	1.7 E-008	EFW-PMP-CF-ALL, EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, PPR-SRV-OO-2, COND-PFS-FC-SYS
TRANS Sequence 08		4.7 E-007	
1	36.6	1.7 E-007	PPR-SRV-CO-TRAN, PPR-SRV-OO-1, EPS-DGN-FC-3AFR, HPI-MDP-FC-B, HPI-XHE-NOREC
2	36.6	1.7 E-007	PPR-SRV-CO-TRAN, PPR-SRV-OO-2, EPS-DGN-FC-3AFR, HPI-MDP-FC-B, HPI-XHE-NOREC
3	6.7	3.2 E-008	PPR-SRV-CO-TRAN, PPR-SRV-OO-1, HPI-MDP-CF-ALL, HPI-XHE-NOREC
4	6.7	3.2 E-008	PPR-SRV-CO-TRAN, PPR-SRV-OO-2, HPI-MDP-CF-ALL, HPI-XHE-NOREC
5	3.7	1.7 E-008	PPR-SRV-CO-TRAN, PPR-SRV-OO-1, HPI-MDP-CF-ALL, HPI-XHE-NOREC
6	3.7	1.7 E-008	PPR-SRV-CO-TRAN, PPR-SRV-OO-2, HPI-MDP-CF-ALL, HPI-XHE-NOREC
7	1.3	6.2 E-009	PPR-SRV-CO-TRAN, PPR-SRV-OO-1, EPS-DGN-FC-3AFR, HPI-MOV-OC-SUCB, HPI-XHE-NOREC
8	1.3	6.2 E-009	PPR-SRV-CO-TRAN, PPR-SRV-OO-2, EPS-DGN-FC-3AFR, HPI-MOV-OC-SUCB, HPI-XHE-NOREC
Total (all sequences)		2.5 E-005	

[a] The conditional probability for each cut set is determined by multiplying the probability of the initiating event by the probabilities of the basic events in that minimal cut set. The probability of the initiating events are given in Table 1 and begin with the designator "IE". The probabilities for the basic events are also given in Table 1.

[b] Basic events EFW-XHE-NOREC, MFW-SYS-TRIP, MFW-XHE-NOREC, and RHR-MOV-CF-SUCT are all type TRUE events which are not normally included in the output of fault tree reduction programs. These events have been added to aid in understanding the sequences to potential core damage associated with the event.

Table 6. Definitions and Probabilities for Selected Basic Events for the Condition Assessment for LER 382/95-002

Event name	Description	Base probability	Current probability	Type	Modified for this event
HPI-MDP-FC-B	HPI Motor-Driven Pump-B Train Failures	3.9 E-003	3.9 E-003		No
HPR-AOV-CC-SMPA	Containment Sump A Failures	1.1 E-003	1.1 E-003		No
HPR-AOV-CC-SMPB	Containment Sump B Failures	1.1 E-003	1.1 E-003		No
HPR-AOV-CF-SMP	Common Cause Failure (CCF) of Sump Air-Operated Valves	1.0 E-004	1.0 E-004		No
HPR-HDV-CF-RWSP	CCF of the Isolation Hydraulic Discharge Valves to the RWSP	2.0 E-004	2.0 E-004		No
HPR-HDV-OO-RWSPA	RWSP Train A Isolation Hydraulic Discharge Valve (HDV) Failures	2.0 E-003	2.0 E-003		No
HPR-HDV-OO-RWSPB	RWSP Train B Isolation HDV Failures	2.0 E-003	2.0 E-003		No
HPR-SMP-FC-SUMP	Containment Recirculation Sump Failures	5.0 E-005	5.0 E-005		No
HPR-XHE-NOREC	Operator Fails to Recover the HPR System	1.0 E+000	1.0 E+000	TRUE	No
HPR-XHE-NOREC-L	Operator Fails to Recover the HPR System During a LOOP	1.0 E+000	1.0 E+000	TRUE	No
MSS-VCF-HW-ISOL	Ruptured Steam Generator Isolation Failures	1.0 E-002	1.0 E-002		No
MSS-XHE-NOREC	Operator Recovery Action for Steam Generator Isolation	1.0 E-001	1.0 E-001		No
PPR-SRV-CO-L	SRVs Open During a LOOP	1.6 E-001	1.6 E-001		No
PPR-SRV-CO-TRAN	SRVs Open During Transient	2.0 E-002	2.0 E-002		No
PPR-SRV-OO-1	SRV 1 Fails to Reseat	1.6 E-002	1.6 E-002		No
PPR-SRV-OO-2	SRV 2 Fails to Reseat	1.6 E-002	1.6 E-002		No
RHR-MOV-CF-SUCT	Common Cause Failure of RHR Suction Valves	1.2 E-003	1.0 E+000	TRUE	Yes
RHR-XHE-NOREC	Operator Fails to Recover the RHR System	3.4 E-001	3.4 E-001		No
RHR-XHE-NOREC-L	Operator Fails to Recover the RHR System During a LOOP	3.4 E-001	3.4 E-001		No
RWSP-REFILL	Operator Fails to Refill RWSP	8.5 E-003	8.5 E-003		No

Table 7. Sequence Conditional Probabilities for the Condition Assessment for LER 382/95-002

Event tree name	Sequence name	Conditional core damage probability (CCDP)	Core damage probability (CDP)	Importance (CCDP-CDP)	Percent contribution[a]
SLOCA	03	1.1 E-006	8.5 E-009	1.1 E-006	65.4
TRANS	05	4.8 E-007	3.4 E-009	4.8 E-007	28.7
LOOP	05	8.2 E-008	3.6 E-008	4.5 E-008	2.6
SGTR	04	4.1 E-008	2.9 E-010	4.1 E-008	2.4
Total (all sequences)		9.1 E-005	8.9 E-005	1.7 E-005	

[a] Percent contribution to the total Importance.

Table 8. Sequence Logic for Dominant Sequences for the Condition Assessment for LER 382/95-002

Event tree name	Sequence name	Logic
SLOCA	03	/RT, /EFW, /HPI, /COOLDOWN, RHR, HPR
TRANS	05	/RT, /EFW, SRV, SRV-RES, /HPI, /COOLDOWN, RHR, HPR
LOOP	05	/RT-L, /EP, /EFW-L, SRV-L, SRV-RES, /HPI-L, /COOLDOWN, RHR-L, HPR-L
SGTR	04	/RT, /EFW-SGTR, /HPI, /RCS-SG, SGISOL, /RCSCOOL, RHR, RWSPREFIL

Table 9. System Names for the Condition Assessment for LER 382/95-002

System name	Logic
COOLDOWN	RCS Cooldown to RHR Pressure Using Turbine-Bypass Valves, etc.
EFW	No or Insufficient EFW Flow
EFW-L	No or Insufficient EFW Flow During a LOOP
EFW-SGTR	No or Insufficient EFW Flow During a Steam Generator Tube Rupture event
EP	Failure of Both Trains of Emergency Power
HPI	No or Insufficient HPI System Flow
HPI-L	No or Insufficient HPI System Flow During a LOOP
HPR	No or Insufficient HPR Flow
HPR-L	No or Insufficient HPR Flow During a LOOP
RCS-SG	Failure to Lower RCS Pressure to Less Than Steam Generator Relief-Valve Set Point
RCSCOOL	Failure to Cooldown RCS to Less Than RCS Pressure
RHR	No or Insufficient RHR System Flow
RHR-L	No or Insufficient RHR System Flow During a LOOP
RT	Reactor Fails to Trip During a Transient
RT-L	Reactor Fails to Trip During a LOOP
RWSPREFIL	Operator Fails to Refill RWSP
SGISOL	Failure to Isolate Ruptured Steam Generator Before RWSP Depletion
SRV	SRVs Open During a Transient
SRV-L	SRVs Open During a LOOP
SRV-RES	SRVs Fail to Reseat

Table 10. Conditional Cut Sets for Higher Probability Sequences for the Condition Assessment for LER 382/95-002

Cut set number	Percent contribution	Change in CCDP (Importance)[a]	Cut sets[b]
SLOCA Sequence 03		1.1 E-006	
1	53.4	6.0 E-007	RHR-MOV-CF-SUCT, RHR-XHE-NOREC, HPR-HDV-CF-RWSP, HPR-XHE-NOREC
2	26.7	3.0 E-007	RHR-MOV-CF-SUCT, RHR-XHE-NOREC, HPR-AOV-CF-SMP, HPR-XHE-NOREC
3	13.3	1.5 E-007	RHR-MOV-CF-SUCT, RHR-XHE-NOREC, HPR-SMP-FC-SUMP, HPR-XHE-NOREC
4	2.0	2.3 E-008	RHR-MOV-CF-SUCT, RHR-XHE-NOREC, HPI-MDP-FC-B, HPR-HDV-OO-RWSPA, HPR-XHE-NOREC
5	1.1	1.2 E-008	RHR-MOV-CF-SUCT, RHR-XHE-NOREC, HPI-MDP-FC-B, HPR-AOV-CC-SMPA, HPR-XHE-NOREC
6	1.0	1.1 E-008	RHR-MOV-CF-SUCT, RHR-XHE-NOREC, HPR-HDV-OO-RWSPA, HPR-HDV-OO-RWSPB, HPR-XHE-NOREC
TRANS Sequence 05		4.8 E-007	
1	26.7	1.2 E-007	PPR-SRV-CO-TRAN, PPR-SRV-OO-1, RHR-MOV-CF-SUCT, RHR-XHE-NOREC, HPR-HDV-CF-RWSP, HPR-XHE-NOREC
2	26.7	1.2 E-007	PPR-SRV-CO-TRAN, PPR-SRV-OO-2, RHR-MOV-CF-SUCT, RHR-XHE-NOREC, HPR-HDV-CF-RWSP, HPR-XHE-NOREC
3	13.3	6.5 E-008	PPR-SRV-CO-TRAN, PPR-SRV-OO-1, RHR-MOV-CF-SUCT, RHR-XHE-NOREC, HPR-AOV-CF-SMP, HPR-XHE-NOREC
4	13.3	6.5 E-008	PPR-SRV-CO-TRAN, PPR-SRV-OO-2, RHR-MOV-CF-SUCT, RHR-XHE-NOREC, HPR-AOV-CF-SMP, HPR-XHE-NOREC
5	6.6	3.2 E-008	PPR-SRV-CO-TRAN, PPR-SRV-OO-1, RHR-MOV-CF-SUCT, RHR-XHE-NOREC, HPR-SMP-FC-SUMP, HPR-XHE-NOREC
6	6.6	3.2 E-008	PPR-SRV-CO-TRAN, PPR-SRV-OO-2, RHR-MOV-CF-SUCT, RHR-XHE-NOREC, HPR-SMP-FC-SUMP, HPR-XHE-NOREC
7	1.0	5.1 E-009	PPR-SRV-CO-TRAN, PPR-SRV-OO-1, RHR-MOV-CF-SUCT, RHR-XHE-NOREC, HPI-MDP-FC-B, HPR-HDV-OO-RWSPA, HPR-XHE-NOREC
8	1.0	5.1 E-009	PPR-SRV-CO-TRAN, PPR-SRV-OO-2, RHR-MOV-CF-SUCT, RHR-XHE-NOREC, HPI-MDP-FC-B, HPR-HDV-OO-RWSPA, HPR-XHE-NOREC

Table 10. Conditional Cut Sets for Higher Probability Sequences for the Condition Assessment for LER 382/95-002

Cut set number	Percent contribution	Change in CCDP (Importance)[a]	Cut sets[b]
LOOP Sequence 05		4.5 E-008	
1	26.7	1.3 E-008	PPR-SRV-CO-L, PPR-SRV-OO-1, RHR-MOV-CF-SUCT, RHR-XHE-NOREC-L, HPR-HDV-CF-RWSP, HPR-XHE-NOREC-L
2	26.7	1.3 E-008	PPR-SRV-CO-L, PPR-SRV-OO-2, RHR-MOV-CF-SUCT, RHR-XHE-NOREC-L, HPR-HDV-CF-RWSP, HPR-XHE-NOREC-L
3	13.3	6.6 E-009	PPR-SRV-CO-L, PPR-SRV-OO-1, RHR-MOV-CF-SUCT, RHR-XHE-NOREC L, HPR-AOV-CF-SMP, HPR-XHE-NOREC-L
4	13.3	6.6 E-009	PPR-SRV-CO-L, PPR-SRV-OO-2, RHR-MOV-CF-SUCT, RHR-XHE-NOREC-L, HPR-AOV-CF-SMP, HPR-XHE-NOREC-L
5	6.6	3.2 E-009	PPR-SRV-CO-L, PPR-SRV-OO-1, RHR-MOV-CF-SUCT, RHR-XHE-NOREC-L, HPR-SMP-FC-SUMP, HPR-XHE-NOREC-L
6	6.6	3.2 E-009	PPR-SRV-CO-L, PPR-SRV-OO-2, RHR-MOV-CF-SUCT, RHR-XHE-NOREC-L, HPR-SMP-FC-SUMP, HPR-XHE-NOREC-L
7	1.0	5.1 E-010	PPR-SRV-CO-L, PPR-SRV-OO-1, RHR-MOV-CF-SUCT, RHR-XHE-NOREC-L, HPI-MDP-FC-B, HPR-HDV-OO-RWSPA, HPR-XHE-NOREC-L
8	1.0	5.1 E-010	PPR-SRV-CO-L, PPR-SRV-OO-2, RHR-MOV-CF-SUCT, RHR-XHE-NOREC-L, HPI-MDP-FC-B, HPR-HDV-OO-RWSPA, HPR-XHE-NOREC-L
SGTR Sequence 08		4.1 E-008	
1	99.7	4.1 E-008	MSS-VCF-HW-ISOL, MSS-XHE-NOREC, RHR-MOV-CF-SUCT, RHR-XHE-NOREC, RWSP-REFILL
Total (all sequences)		**1.9 E-005**	

[a] The change in conditional probability (importance) is determined by calculating the conditional probability for the period in which the condition existed and given the condition, and subtracting the conditional probability for the same period but with plant equipment assumed to be operating nominally. The conditional probability for each cut set within a sequence is determined by multiplying the probability that the portion of the sequence that makes the precursor visible (e.g., the system with a failure is demanded) will occur during the duration of the event by the probabilities of the remaining basic events in the minimal cut set. This can be approximated by $1 - e^{-p}$, where p is determined by multiplying the expected number of initiators that occur during the duration of the event by the probabilities of the basic events in that minimal cut set. The expected number of initiators is given by λt, where λ is the frequency of the initiating event (given on a per hour basis), and t is the duration time of the event (in this case, 8760 h). This approximation is conservative for precursors made visible by the initiating event. The frequencies of interest for this event are: $\lambda_{TRANS} = 6.8 \times 10^{-4}$/h, $\lambda_{LOOP} = 8.5 \times 10^{-6}$/h, $\lambda_{SLOCA} = 1.0 \times 10^{-6}$/h, and $\lambda_{SGTR} = 1.6 \times 10^{-6}$/h.

[b] Basic event RHR-MOV-CF-SUCT is a type TRUE event which is not normally included in the output of fault tree reduction programs. This events has been added to aid in understanding the sequences to potential core damage associated with the event.

LER No. 382/95-002

Event Description: Reactor trip, breaker failure and fire, degraded offsite power, and degraded shutdown cooling

Date of Event: June 10, 1995

Plant: Waterford 3

Licensee Comments

Reference: Letter from J. J. Fisicaro (Entergy Operations, Inc.) to the U.S. Nuclear Regulatory Commission, "Review of Preliminary Accident Sequence Precursor Analysis," W3F1-96-0140, August 15, 1996.

Comment 1: The licensee provided many specific comments on the text of the Event Summary, Event Description, and Additional Event-Related Information sections of the analysis documentation concerning the cause of the breaker failure, plant response to that failure, and the design of the bus transfer scheme at Waterford.

Response 1: With the exception of Comment 4, the clarifications and corrections provided by the licensee were incorporated into the analysis documentation. Because most of the comments were editorial in nature, they have not been repeated below. Those comments of a technical nature are discussed below.

Comment 2: The second paragraph, 3rd sentence in the Event Description is not correct. There is nothing to support the UAT tripping on overcurrent. The overcurrent relays were set to trip @ > 1 second for a current of 30000 amps. The event was less than 29000 amps for approximately 0.3 seconds. The power would not have been lost to the A2 bus unless the SUT breaker had also tripped.

Response 2: The reference to the UAT feeder breaker tipping on overcurrent was changed to indicate that both the UAT and the SUT breakers tripped, and power was lost to bus A2.

Comment 3: The 6th paragraph, 1st sentence in the Event Description states that the A1 bus de-energized and all of its loads de-energized at the beginning of the event. This sentence should be moved to the beginning of the event.

Response 3: To preserve the sequence of events, this sentence was moved to the 3rd paragraph in the Event Description.

Comment 4: The 9th paragraph in the Event Description discusses the use of water on the fire. The recommendation to use water was not made solely by the Volunteer Fire Department. The decision to use water was the result of a methodical analysis performed by the Waterford 3 Fire Brigade Leader and the Voluntary Fire Department Chief. Also, the Fire Brigade was not "reluctant" to use water. They had been trained to consider gas and dry chemical as the preferred options.

Response 4: This comment pertains to the reluctance of the fire brigade to use water on the fire when carbon dioxide and dry chemical fire extinguishers were proving to be ineffective. The AIT report for the event (Reference 2 to the analysis documentation) noted that all operators indicated in later interviews that they were reluctant to use water on the electrical fire. The applicable paragraph in the Event Description was reworded instead to indicate that the source of this information was the AIT report.

Comment 5: The 10th paragraph, 2nd sentence in the Event Description states that a condenser low vacuum alarm had actuated at 0940 hours, "42 min after the 6.9 kV . . ." The 6.9 kV bus de-energized at 0858 hours when the transfer to the SUT failed.

Response 5: This part of the sentence was deleted.

Comment 6: The 5th paragraph in the Additional Event-Related Information section discusses "fire stops." The design of the Calvert Bus used at Waterford 3 does not employ the use of "fire stops." Thus, the statement regarding the ineffectiveness of the vertical fire stops is inaccurate. Additionally, the statement that fire damage was limited by the fire stop in the horizontal section is also inaccurate.

Response 6: All references to fire stops at Waterford 3 have been removed.