October 24, 1996

Mr. William R. McCollum
Site Vice President
Catawba Nuclear Station
Duke Power Company
4800 Concord Road
York, South Carolina 29745-9635

SUBJECT: CATAWBA NUCLEAR STATION - FINAL ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF EVENT AT CATAWBA NUCLEAR STATION, UNIT 2 (TAC M95254)

Dear Mr. McCollum:

Enclosed for your information is a copy of the final Accident Sequence Precursor (ASP) analysis of the operational event at Catawba Nuclear Station, Unit 2, reported in Licensee Event Report (LER) No. 414/96-001. 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, transmitted by your letter of June 11, 1996. 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 1996.

Please contact me at 301-415-1451 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:
Peter S. Tam, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-414

Enclosures: (1) Final ASP
(2) Review of DPC's 6/11/96 response

cc: See next page

DISTRIBUTION
Docket File
PUBLIC
PDII-2 Reading
S. Varga
J. Zwolinski
E. Merschoff, RII
RCrlenjak, RII
OGC, O-15B18
ACRS, TWF

250043

NRC FILE CENTER COPY

1/1
DF01

DOCUMENT NAME: G:\CATAWBA\CAT95254.LTR
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	DRPE\PDII-2\PM		PDII-2\LA		PDII-2\D			
NAME	PTam:cdw PST		LBerry		HBerkow			
DATE	10/23/96		10/23/96		10/23/96		10/ /96	

OFFICIAL RECORD COPY

9610250138 961024
PDR ADOCK 05000414
S PDR

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

October 24, 1996

Mr. William R. McCollum
Site Vice President
Catawba Nuclear Station
Duke Power Company
4800 Concord Road
York, South Carolina 29745-9635

SUBJECT: CATAWBA NUCLEAR STATION - FINAL ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF EVENT AT CATAWBA NUCLEAR STATION, UNIT 2 (TAC M95254)

Dear Mr. McCollum:

Enclosed for your information is a copy of the final Accident Sequence Precursor (ASP) analysis of the operational event at Catawba Nuclear Station, Unit 2, reported in Licensee Event Report (LER) No. 414/96-001. 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, transmitted by your letter of June 11, 1996. 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 1996.

Please contact me at 301-415-1451 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,

Peter S. Tam

Peter S. Tam, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-414

Enclosures: (1) Final ASP
(2) Review of DPC's 6/11/96 response

cc: See next page

Mr. W. R. McCollum
Duke Power Company

Catawba Nuclear Station

cc:
Mr. M. S. Kitlan
Regulatory Compliance Manager
Duke Power Company
4800 Concord Road
York, South Carolina 29745

Mr. Paul R. Newton
Legal Department (PB05E)
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242-0001

J. Michael McGarry, III, Esquire
Winston and Strawn
1400 L Street, NW
Washington, DC 20005

North Carolina Municipal Power
Agency Number 1
1427 Meadowwood Boulevard
P. O. Box 29513
Raleigh, North Carolina 27626-0513

Mr. Peter R. Harden, IV
Account Sales Manager
Westinghouse Electric Corporation
Power Systems Field Sales
P. O. Box 7288
Charlotte, North Carolina 28241

County Manager of York County
York County Courthouse
York, South Carolina 29745

Richard P. Wilson, Esquire
Assistant Attorney General
South Carolina Attorney General's
Office
P. O. Box 11549
Columbia, South Carolina 29211

Piedmont Municipal Power Agency
121 Village Drive
Greer, South Carolina 29651

Mr. T. Richard Puryear
Owners Group (NCEMC)
Duke Power Company
4800 Concord Road
York, South Carolina 29745

North Carolina Electric Membership
Corporation
P. O. Box 27306
Raleigh, North Carolina 27611

Senior Resident Inspector
4830 Concord Road
York, South Carolina 29745

Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street, NW. Suite 2900
Atlanta, Georgia 30323

Max Batavia, Chief
Bureau of Radiological Health
South Carolina Department of
Health and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Mr. G. A. Copp
Licensing - EC050
Duke Power Company
526 South Church Street
Charlotte, North Carolina 28242-0001

Saluda River Electric
P. O. Box 929
Laurens, South Carolina 29360

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of Justice
P. O. Box 629
Raleigh, North Carolina 27602

Elaine Wathen, Lead REP Planner
Division of Emergency Management
116 West Jones Street
Raleigh, North Carolina 27603-1335

Dayne H. Brown, Director
Division of Radiation Protection
N.C. Department of Environment,
Health and Natural Resources
P. O. Box 27687
Raleigh, North Carolina 27611-7687

LER No. 414/96-001

Event Description: Loss of Offsite Power (LOOP) with Emergency Diesel Generator (EDG) B Unavailable

Date of Event: February 6, 1996

Plant: Catawba 2

Event Summary

At 1231 hours, on February 6, 1996, Unit 2 was at 100% power when ground faults on the 2A main transformer X-phase and 2B main transformer Z-phase potential transformers resulted in a loss of offsite power (LOOP). The reactor scrammed and emergency diesel generator (EDG) 2A (train A) started and loaded. EDG 2B was out-of-service because of a faulty ac capacitor in the battery charger for that diesel generator. EDG 2B (train B) was returned to service and its emergency bus (2ETB) energized at 1522 hours (2 h 51 min into the event). Using parts from the 2A main transformer, the 2B main transformer was repaired and offsite power restored to Unit 2 on February 8, 1996, at 0120 hours (36 h 49 min into the event). The conditional core damage probability estimated for this event is 2.1×10^{-3}.

Event Description

At 1231 hours, on February 6, 1996, Unit 2 was at 100% power when ground faults on the resistor bushings for the 2A main transformer X-phase and 2B main transformer Z-phase potential transformers resulted in a LOOP. The reactor scrammed, and EDG 2A started and loaded. EDG 2B was out-of-service because of a faulty ac capacitor in the diesel battery charger. EDG 2B was returned to service and its emergency bus (2ETB) energized at 1522 hours (2 h 51 min into the event). Not all emergency loads on bus 2ETB were energized due to activities in progress to implement a cross-tie to Unit 1. At 1800 hours (5 h 29 min into the event), the cross-tie activities for train B were completed and the source for bus 2ETB was transferred to transformer SATB (a Unit 1 B-train offsite power source supplied power to Unit 2 transformer SATB). Initial efforts to complete the cross-tie to bus 2ETB were unsuccessful because of a procedural inadequacy. At 2000 (7 h 29 min into the event), cross-tie activities were completed for EDG Train A and power was transferred to transformer SATA (a Unit 1 A-train offsite power source supplied power to Unit 2 transformer SATA). Personnel repaired the 2B main transformer using parts from the 2A transformer and restored offsite power to Unit 2 on February 8, 1996, at 0120 hours (36h 49 min into the event). Repairs on the 2A main transformer were not completed until 0327 hours on February 11, 1996 (62 h 56 min from the start of the event).

At 1236 hours, or 5 min after the LOOP, operators manually closed the Main Steam Isolation Valves (MSIVs). At 1238 hours, a safety injection (SI) actuation occurred because of low steam line pressure in the 2A steam generator (SG). At 1247 hours, the pressurizer power operated relief valve (PORV) 2NC34A began to cycle; at 1310 hours (39 min into the event), the pressurizer level went off-scale high as the reactor coolant system (RCS) became water solid. At 1320 hours, the pressurizer relief tank (PRT) pressure increased and the PRT rupture disc ruptured as PORV 2NC34A continued to cycle. A steam bubble was reestablished in the pressurizer at 1926 hours (6 h 55 min into the event or 6 h 16 min after becoming water solid). PORV 2NC34A fully stroked approximately 43 times on steam and an additional 31 times on water. A Nuclear Regulatory Commission (NRC) Inspection Team estimated that this PORV came off its closed seat about 110 times. (Evaluations by the licensee of stroke-time tests and visual external inspection concluded that no damage to PORV 2NC34A occurred. The PORV was supplied by Control Components Incorporated. The PRT rupture disc was replaced on February 9, 1996, at 1428 hours.)

Enclosure 1

At 1641 hours (4 h 11 min into the event), control room operators received a report of a leak in the penetration room. [It was subsequently determined that three pit sump check valves from the turbine-driven auxiliary feedwater pump (TDAFWP) were leaking into the penetration room.] The TDAFWP was secured at 1759 hours (5 h 29 min into the event). Water in the pit sump for the TDAFWP was pumped to the turbine building sump. Back leakage through check valves 2WL894, 2WL836, and 2WL834 allowed the discharge from the sump for the TDAFWP to fill floor drain sump "C," which overflowed onto the Auxiliary Feedwater (AFW) pump room floor to a level of several inches. (This area is separated from the AFW pump pits by a concrete curb approximately 18 inches high. The floor drain sump "C" is not powered by emergency power and, therefore, would be unavailable until offsite power is restored.) Operators manually closed valves 2WL835 and 2WL836, thereby stopping the water leakage.

Because of a leak in the instrument air system, the containment was purged by using the Containment Air Release and Addition System on February 7, 1996, at 1033 hours. Air leakage was a recurring problem, as shown by venting data. This data shows that Unit 2 was being vented every 12 h prior to this event. During this event, containment temperature increased in response to the loss of containment chilled water to the ventilation units (containment chilled water is not a diesel-backed load). When the PRT rupture disc ruptured, containment pressure increased further (pressure peaked at 0.9 psig). This pressure increase was sufficient to partially open some—but not all—of the ice condenser lower inlet doors. Energy absorption was limited to contact with ice in the lowest portion of the ice condenser. Because there was no flow through the ice condenser, the intermediate and upper deck doors did not open.

Additional Event-Related Information

Each AFW pump is mounted in a separate pit for Net Positive Suction Head (NPSH) requirements. To prevent flooding of these pits, each motor-driven pump pit is supplied with a 50-gpm sump pump that discharges to the Liquid Radwaste System. For the TDAFWP, the turbine oil is cooled through the lube oil cooler by a small portion of the discharge flow. The TDAFWP turbine oil cooler flow and a portion of the turbine seal water empty directly into the pit for the TDAFWP. If the sump is not drained, failure of the TDAFWP could occur in as early as three hours. To provide extra assurance that the TDAFWP will not fail as a result of flooding, the pit for the TDAFWP is outfitted with two 50-gpm sump pumps. One of these sump pumps can be powered during a LOOP from either EDG 2A or from the standby shutdown facility; the other sump pump is powered from EDG 2B.

A standby shutdown facility (SSF) is located in a separate building on the Catawba site. This facility, which is not normally manned, is capable of providing limited high-pressure injection for RCS makeup and reactor coolant pump (RCP) seal cooling [provided an RCP seal loss-of-coolant accident (LOCA) does not occur]. The SSF includes a separate diesel generator that can power SSFB loads in the event of a station blackout. The diesel generator for the SSFC can also power one of the sump pumps for the TDAFWP. The SSF systems are single trains and, therefore, are susceptible to a single failure. In conjunction with the TDAFWP and the availability of SGs, the SSFs can maintain hot standby conditions for both units. An operator was sent to man the SSF facility during this event; however, the SSF was never started.

The licensee evaluated the flooding of the AFW pump room in its Individual Plant Examination (IPE). (Recall, the AFW pump room was in danger of being flooded by operating the TDAFWP.) The IPE flood analysis for the AFW pump room evaluates a break in a pipe outside of the sump pits. Water will reach the base of the Auxiliary Shutdown Panel at the same time it reaches the top of the curb around the AFW pump pits. (The curb walls around the pit are 18 inches high.) The lowest point of switches, fuses, or terminal strips within the Auxiliary Shutdown Panel is 8 inches from the base. When water reaches this level, the IPE assumes that equipment controlled from the Auxiliary Shutdown Panel is unavailable. Because the floor area outside the AFW pump room is about 2,231.6 square feet and the curb is 18 in. high, the estimated time to flood the turbine-driven pump pit area is about 33 h. The leakage into the turbine-driven pump pit is within the capability of the operating sump pump. If this pump failed, an additional 3 hours would be available before

the leakage or the water accumulation in the turbine driven pump pit could fail the pump. After the pump pits are flooded, there is an additional area of 1,110 square feet in the room for water to cover. The IPE further estimates that there is 41.6 min available to isolate a flood of 2,429 gpm. Therefore, flooding of the TDAFWP is not considered credible because considerable time was available to mitigate the flooding.

Modeling Assumptions

This event was modeled as a LOOP initiator with failure of train B of the emergency power system. Because offsite power was not restored for about 1½ days and both offsite power transformers required major repairs before power could be restored through these transformers, it was assumed that operators could not have restored offsite power during the event. Therefore, the following basic events were set to "TRUE" (i.e., failed):

(1) *operator fails to recover offsite power within 2 h* (OEP-XHE-NOREC-2H),
(2) *operator fails to recover offsite power within 6 h* (OEP-XHE-NOREC-6H),
(3) *operator fails to recover offsite power before battery depletion* (OEP-XHE-NOREC-BD) , and
(4) *operator fails to recover offsite power given a seal LOCA* (OEP-XHE-NOREC-SL).

In addition, the probability that the PORVs open during a transient (PORV) was set to "TRUE" because one PORV (2NC34A) lifted more than 74 times.

AC power to the emergency buses was assumed to be potentially recoverable to the emergency buses by implementing a cross-tie to Unit 1 and by recovering EDG 2B. These actions were assumed to be independent for this analysis given that the event occurred during the day shift. (This assumption would have to be confirmed for an event occurring outside the day shift because it was unknown if sufficient personnel would be available during the period between 5:00 pm and 8:00 am to perform all the actions in parallel that were performed during this event.)

The LOOP event tree for Catawba is shown in Fig. 1. Credit for the SSF at Catawba was accounted for by adding the fault tree shown in Fig. 2 at the SSF branch point in the event tree shown in Fig. 1. The failure probabilities for the basic events SSFB and SSFC were obtained from the Catawba IPE. Basic event SSFB is the failure to provide seal cooling to the reactor coolant pumps; basic event SSFC is the failure to provide power to the sump pump for the TDAFWP.

The recovery of power by implementing a cross-tie to Unit 1 was modeled by adding the basic event *failure to cross-tie EDG B emergency bus within 3 h* [OEP-XHE-NOCRS-B] to the Catawba fault trees for failure to recover power prior to core uncovery given an RCP seal LOCA (OP-SL, see Fig. 3) and prior to battery depletion given no seal LOCA (OP-BD, see Fig. 4). Failure to cross-tie to Unit 1 was modeled as a time-reliability correlation (TRC) as described in "Human Reliability Analysis," E. M. Dougherty and J. R. Fragola, John Wiley and Sons, New York, 1988. Because sequences of concern in the analysis involve a station blackout, the "recovery with hesitancy" TRC—as described in Chapter 11 of the reference—was used in the analysis. The probability distribution for this TRC is lognormal, with an error factor of 6.4. To reflect the observed time to implement the cross-tie, a median response of 60 min was assumed, following a 30 minute delay. The probability of crew failure at 3 h, estimated using this TRC and response time, is 0.27.

A single sump pump must be available (requiring emergency power or power from the SSF) within 3 h to prevent failure of the TDAFWP as a result of flooding.

The probability of a seal LOCA was obtained from NUREG-1032, "Evaluation of Station Blackout at Nuclear Power Plants," and RCP seal LOCA models were developed as part of the NUREG-1150 probabilistic risk assessment efforts, as described in "Revised LOOP Recovery and PWR Seal LOCA Models" (ORNL/NRC/LTR-89/11, August 1989). This model assumes that it would take 2 h to uncover the core given a seal LOCA and that the seal LOCA would occur within

1 h of the station blackout. Therefore, the assumption was made that the core would be uncovered 3 h after the initiation of the station blackout.

The basic event for failing to recover EDG 2B was developed with an exponential repair model with a median repair time of 140 min and a delay of 30 min (EPS-XHE-EDGB-NOR). (EDG 2B was recovered within 3 h.) Based on this repair model, the probability of failing to recover EDG 2B is 0.48.

To account for the longer run time of EDG 2A during this event, the failure probability was modified from 0.042 to 0.045. Hence, the mission time for this event was increased from 6 h to 7.5 h while maintaining the same failure to start probability (0.03) and the same failure to run failure rate (0.002/h), as reported in the "ASP Models, PWR B, Catawba Units 1 and 2," Revision 1, November 1994.

Although the AFW pump room was flooding and the source of the flooding was bearing and oil cooling water from the TDAFWP via the sump to the TDAFWP, the TDAFWP was considered operable with no change in the failure probability because (1) operators isolated the leak, and (2) operators would have had at least 33 h to isolate the leak if the TDAFWP had been required to run. The 33 h estimate was obtained by multiplying the time provided in the IPE for isolating a flood (41.6 min) by the assumed flooding rate in the IPE (2,429 gpm) and dividing by the maximum sump pump flow for the TDAFWP. This result is then converted to hours by dividing by 60 min/h [i.e. (41.6 min) × (2429 gpm / 50 gpm) / (60 min/h) ≈ 33 h].

Analysis Results

The conditional core damage probability estimated for this event is 2.1×10^{-3}. The dominant core damage sequence, highlighted as sequence number 39 on the event tree in Fig. 1 involves the following:

- given the loss of offsite power, the reactor successfully trips,
- both trains of emergency power fail,
- AFW provides sufficient flow,
- the PORVs open and then successfully reseat,
- the safe shutdown facility fails,
- the RCP seals fail, and
- offsite power is not recovered after the RCP seal failure.

The second highest core damage sequence (No. 41) involves the following:

- given the loss of offsite power, the reactor successfully trips,
- both trains of emergency power fail, and
- AFW fails to provide sufficient flow.

Definitions and probabilities for selected basic events are shown in Table 1. The conditional probabilities associated with the highest probability sequences are shown in Table 2. 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. Minimal cut sets associated with the dominant sequences are shown in Table 5.

Acronyms

ac	alternating current
AFW	Auxiliary Feedwater

EDG	Emergency Diesel Generator
HPI	High Pressure Injection
HPR	High Pressure Recirculation
IPE	Individual Plant Examination
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
MOV	Motor-Operated Valve
MSIV	Main Steam Isolation Valve
NPSH	Net Positive Suction Head
PORV	Power Operated Relief Valve
PRT	Pressurizer Relief Tank
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
SG	Steam Generator
SI	Safety Injection
SSF	Standby Shutdown Facility
TRC	Time Reliability Correlation
TDAFWP	Turbine Driven Auxiliary Feedwater Pump

References

1. Memorandum from S. D. Ebneter, Regional Administrator, to E. L. Jordan, Director, Office for Analysis and Evaluation of Operational Data, transmitting "Supporting Documents for the Catawba Loss of Offsite Power Event (February 6 - 8, 1996)," February 15, 1996.

2. U.S. Nuclear Regulatory Commission, "NRC Inspection Report Nos. 50-413/96-03 and 50-414/96-03 and Notice of Violation," March 12, 1996.

3. U.S. Nuclear Regulatory Commission, "Preliminary Notification of Event or Unusual Occurrence PNO-II-96-006," February 6, 1996.

4. U.S. Nuclear Regulatory Commission, "Preliminary Notification of Event or Unusual Occurrence PNO-II-96-006A," February 7, 1996.

5. U.S. Nuclear Regulatory Commission, "Preliminary Notification of Event or Unusual Occurrence PNO-II-96-006B," February 7, 1996.

6. U.S. Nuclear Regulatory Commission, "Preliminary Notification of Event or Unusual Occurrence PNO-II-96-006C," February 8, 1996.

7. 50.72 Report Number 29945, February 6, 1996.

8. LER 414/96-001, "Loss of Offsite Power Due to Electrical Component Failures," March 7, 1996.

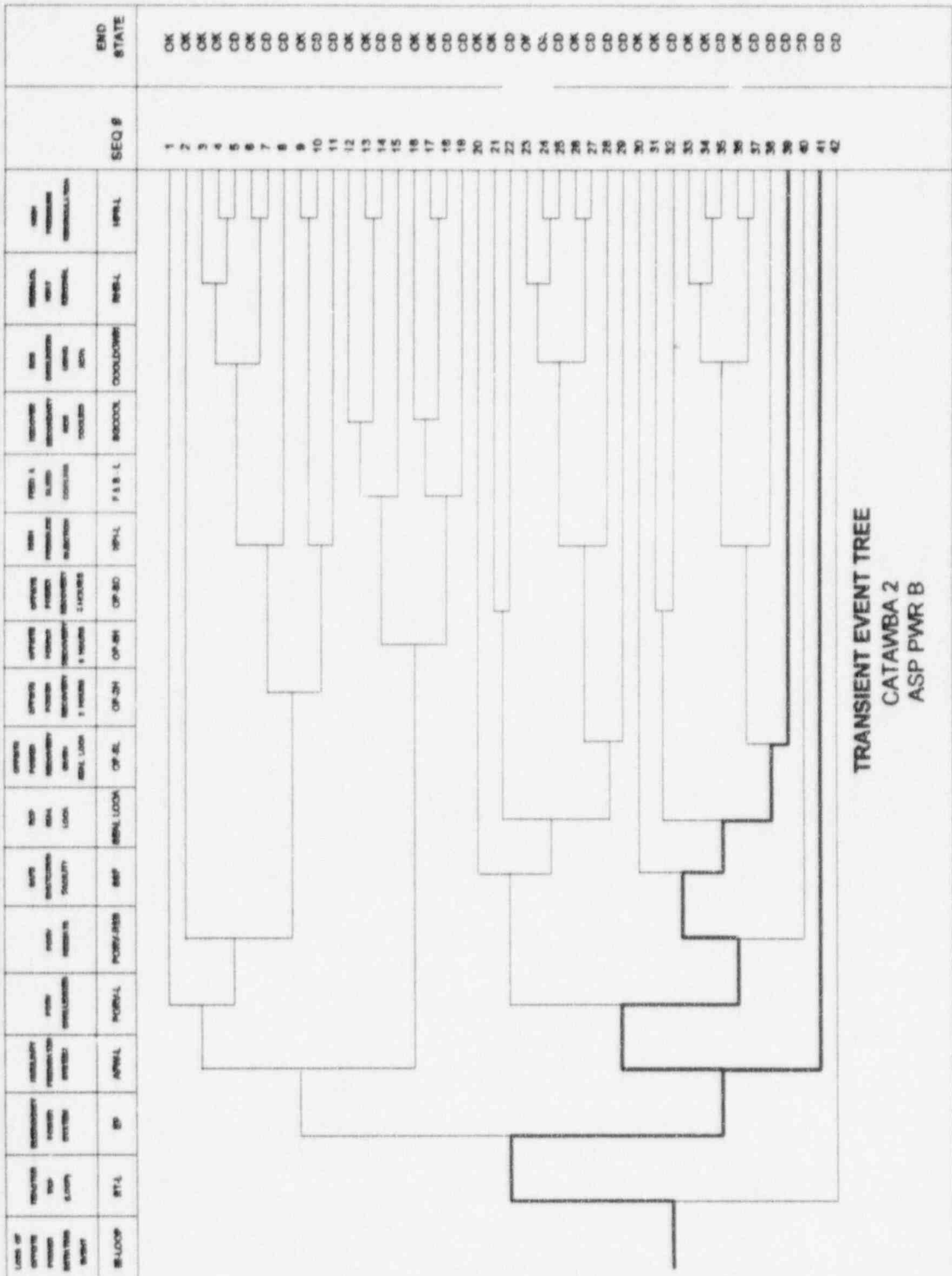

Fig. 1 Dominant core damage sequences for LER No. 414/96-001.

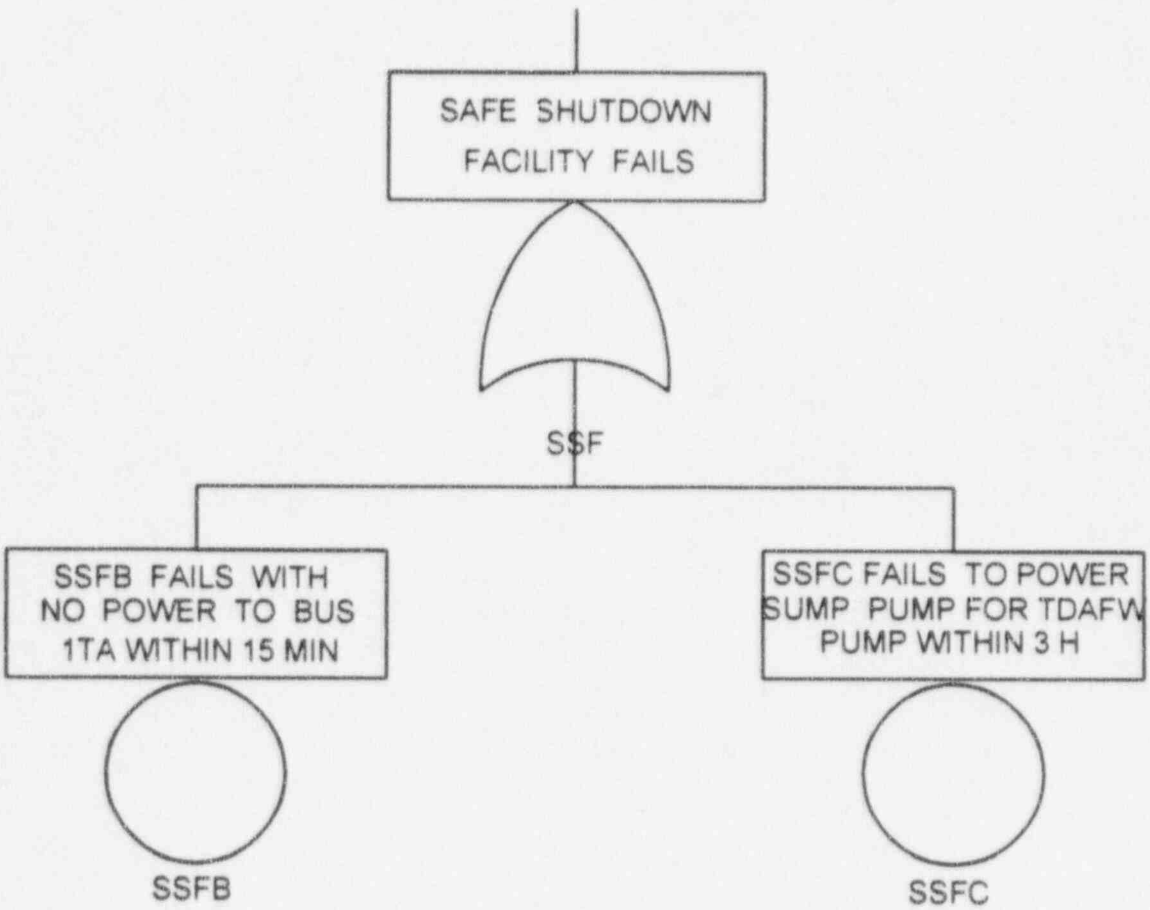

Fig. 2 Fault tree modeling the Standby Shutdown Facility (SSF).

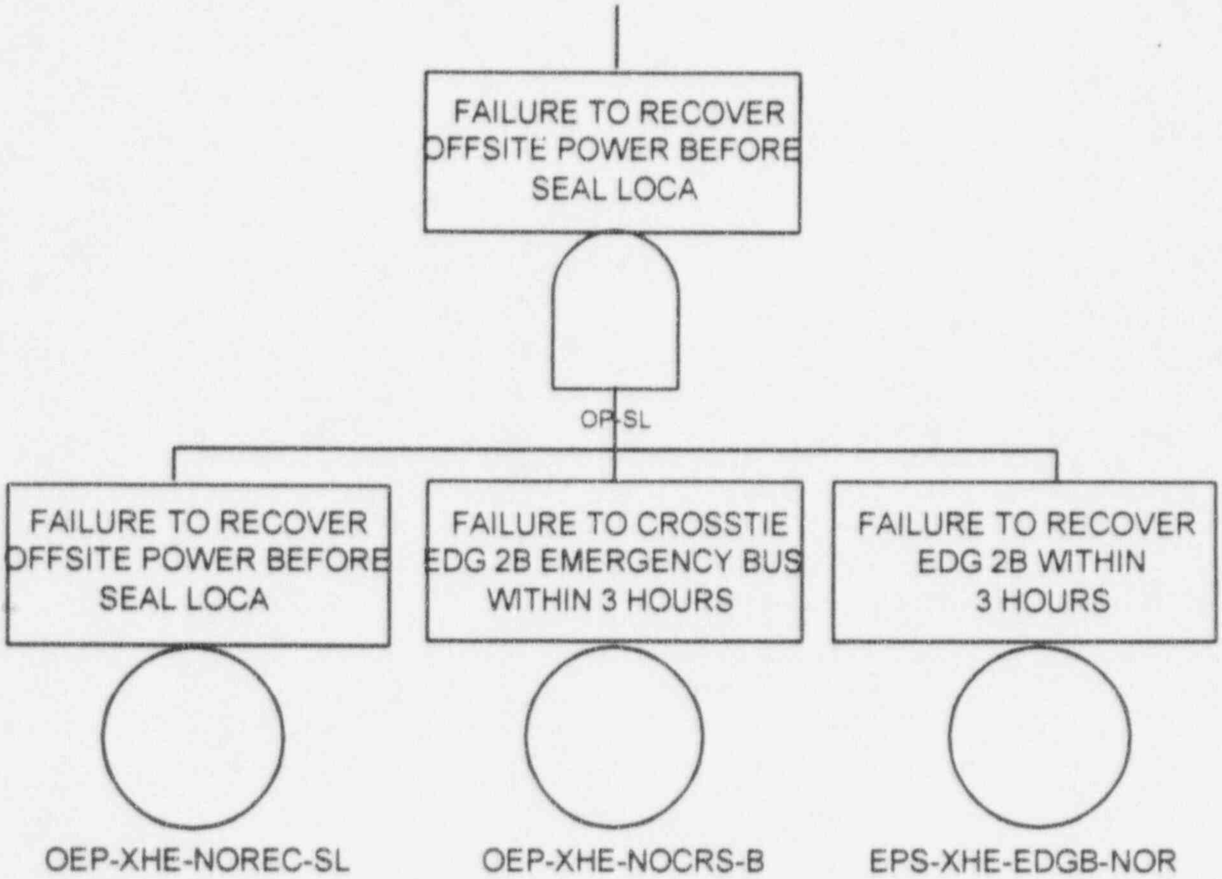

Fig. 3 Fault tree modeling the recovery of offsite power before the core becomes uncovered given a seal LOCA (OP-SL).

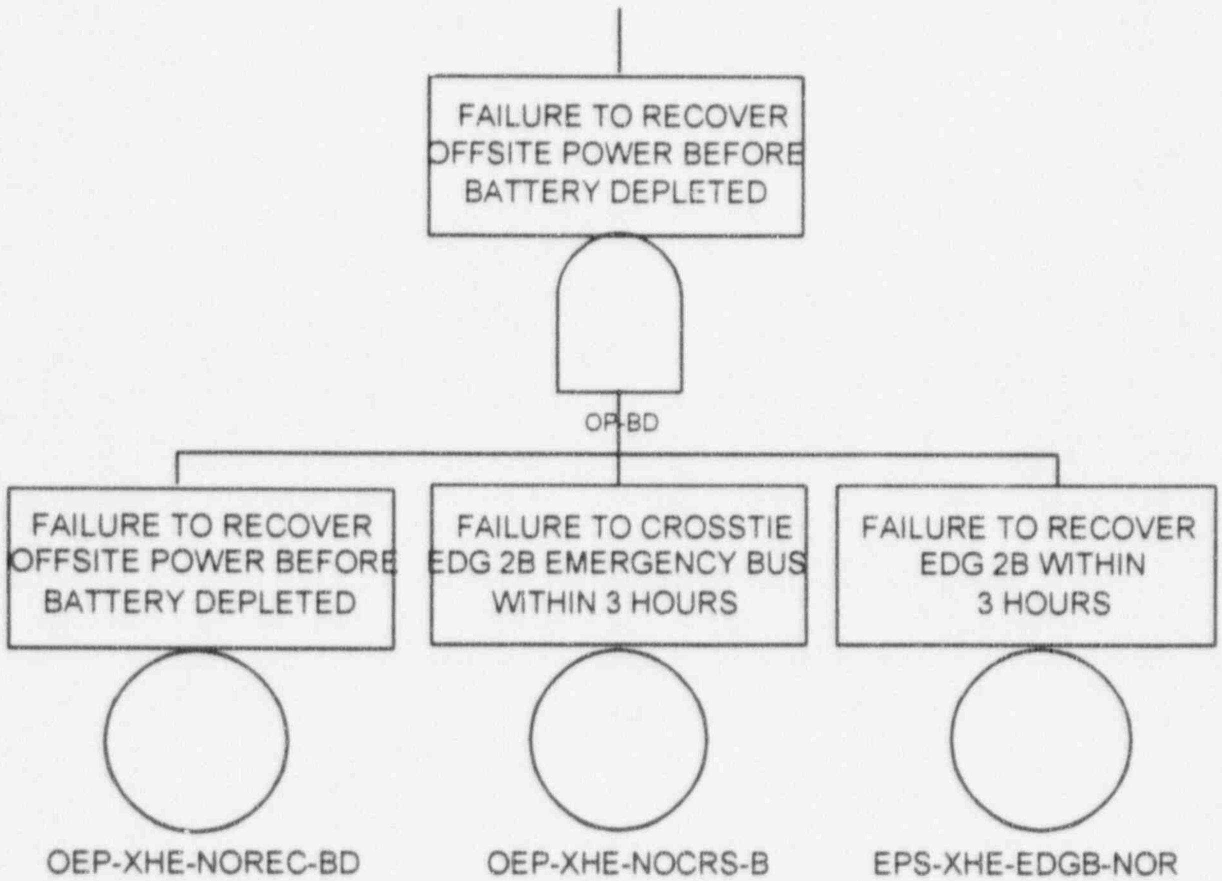

Fig. 4 Fault tree modeling the recovery of offsite power before the batteries are depleted (OP-BD).

Table 1. Definitions and Probabilities for Selected Basic Events for LER 414/96-001

Event name	Description	Base probability	Current probability	Type	Modified for this event
IE-LOOP	Loss of Offsite Power Initiating Event	6.9 E-006	1.0 E+000	TRUE	Yes
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	5.3 E-004	0.0 E+000	IGNORE	No
AFW-TDP-FC-1A	AFW Turbine-Driven Pump Fails	3.2 E-002	3.2 E-002		No
AFW-XHE-NOREC-EP	Operator Fails to Recover AFW During Station Blackout	3.4 E-001	3.4 E-001		No
AFW-XHE-XA-NWS	Operator Fails to Align Nuclear Service Water	1.0 E-003	1.0 E-003		No
EPS-DGN-CF-ALL	Common Cause Failure of Diesel Generators	1.1 E-003	1.1 E-003		No
EPS-DGN-FC-1A	Diesel Generator A Fails	4.2 E-002	4.5 E-002		Yes
EPS-DGN-FC-1B	Diesel Generator B Fails	4.2 E-002	1.0 E+000	TRUE	Yes
EPS-XHE-EDGB-NOR	Operator Fails to Recover EDG B Within 3 Hours	1.0 E+000	3.1 E-001	NEW	Yes
EPS-XHE-NOREC	Operator Fails to Recover Emergency Power	8.0 E-001	1.0 E+000		Yes
HPI-MDP-CF-ALL	Common Cause Failure of the High Pressure Injection (HPI) Pumps	7.8 E-004	7.8 E-004		No
HPI-MDP-FC-1A	HPI Motor-Driven Pump Train A Fails	4.0 E-003	4.0 E-003		No
HPI-MOV-CC-DISCH	HPI Cold Leg Injection Valve Fails	3.0 E-003	3.0 E-003		No
HPR-MOV-CC-RHRB	Residual Heat Removal (RHR) Discharge Motor-Operated Valve (MOV) into HPI Train B Fails	3.1 E-003	3.1 E-003		No
HPR-MOV-CC-SMPA	Sump Isolation MOV 185A Fails to Open	3.0 E-003	3.0 E-003		No
HPR-MOV-CF-SUCA	High Pressure Recirc (HPR) Suction MOVs from RHR Train A Fail to Open due to Common Cause	1.0 E+000	1.0 E+000		No
HPR-XHE-NOREC-L	Operator Fails to Recover the HPR System During a LOOP	1.0 E+000	1.0 E+000		No

Table 1. Definitions and Probabilities for Selected Basic Events for LER 414/96-001

Event name	Description	Base probability	Current probability	Type	Modified for this event
HPR-XHE-XM-L	Operator Fails to Initiate HPR During a LOOP	1.0 E-003	1.0 E-003		No
OEP-XHE-NOCRS-B	Failure to Cross-tie EDG B Emergency Bus Within 3 Hours	1.0 E+000	2.7 E-001	NEW	Yes
OEP-XHE-NOREC-2H	Operator Fails to Recover Offsite Power Within 2 Hours	1.4 E-001	1.0 E+000	TRUE	Yes
OEP-XHE-NOREC-6H	Operator Fails to Recover Offsite Power Within 6 Hours	9.9 E-004	1.0 E+000	TRUE	Yes
OEP-XHE-NOREC-BD	Operator Fails to Recover Offsite Power Before Battery Depleted	2.3 E-002	1.0 E+000	TRUE	Yes
OEP-XHE-NOREC-SL	Operator Fails to Recover Offsite Power (Seal LOCA)	4.8 E-001	1.0 E+000	TRUE	Yes
PORV	PORVs Open During Transient	7.0 E-001	1.0 E+000	TRUE	Yes
PPR-SRV-OO-PRV1	PORV 1 Fails to Reclose After Opening	2.0 E-003	2.0 E-003		No
PPR-SRV-OO-PRV2	PORV 2 Fails to Reclose After Opening	2.0 E-003	2.0 E-003		No
PPR-SRV-OO-PRV3	PORV 3 Fails to Reclose After Opening	2.0 E-003	2.0 E-003		No
RCS-MDP-LK-SEALS	RCP Seals Fail Without Cooling and Injection	2.4 E-001	7.0 E-001		Yes
RHR-MDP-CF-ALL	RHR Pump Common Cause Failures	4.5 E-004	4.5 E-004		No
RHR-MDP-FC-1A	RHR MDP 1A Fails	4.1 E-003	4.1 E-003		No
SEALLOCA	RCP Seals Fail During LOOP	2.4 E-001	7.0 E-001		Yes
SSFB	SSF Fails with No Power to Bus 1TA	2.2 E-001	2.2 E-001	NEW	Yes
SSFC	SSF Fails to Power Sump Pump of TDAFW Pump	9.5 E-002	9.5 E-002	NEW	Yes

Table 2. Sequence Conditional Probabilities for LER 414/96-001

Event tree name	Sequence name	Conditional core damage probability (CCDP)	Percent Contribution
LOOP	39	8.5 E-004	40.1
LOOP	41	5.3 E-004	25.0
LOOP	32	3.6 E-004	17.2
LOOP	40	2.7 E-004	13.0
LOOP	10	7.8 E-005	3.7
Total (all sequences)		2.1 E-003	

Table 3. Sequence Logic for Dominant Sequences for LER 414/96-001

Event tree name	Sequence name	Logic
LOOP	39	/RT-L, EP, /AFW-L, PORV-L, /PORV-RES, SSF, SEALLOCA, OP-SL
LOOP	41	/RT-L, EP, AFW-L-EP
LOOP	32	/RT-L, EP, /AFW-L, PORV-L, /PORV-RES, SSF, /SEALLOCA, OP-BD
LOOP	40	/RT-L, EP, /AFW-L, PORV-L, PORV-EP
LOOP	10	/RT-L, /EP, /AFW-L, PORV-L, PRV-L-EP, OP-2H, /HPI-L, HPR-L

Table 4. System Names for LER 414/96-001

System name	Logic
AFW-L	No or Insufficient AFW Flow During LOOP
AFW-L-EP	No or Insufficient AFW Flow During Station Blackout
EP	Failure of Both Trains of Emergency Power
HPI-L	No or Insufficient Flow from HPI System During a LOOP
HPR-L	No or Insufficient Flow from HPR System During a LOOP
OP-2H	Operator Fails to Recover Offsite Power Within 2 Hours
OP-BD	Operator Fails to Recover Offsite Power Before Battery Is Depleted
OP-SL	Operator Fails to Recover Offsite Power (Seal LOCA)
PORV-EP	PORVs Fail to Reclose (no Electric Power)
PORV-L	PORVs Open During LOOP
PORV-RES	PORVs Fail to Reseat
PRV-L-EP	PORVs and Block Valves Fail to Reclose [Electric Power (EP) succeeds]
RT-L	Reactor Fails to Trip During LOOP
SEALLOCA	RCP Seals Fail During LOOP
SSF	Safe Shutdown Facility Fails

Table 5. Conditional Cut Sets for Higher Probability Sequences for LER 414/96-001

Cut set No.	Percent Contribution	Conditional Probability[a]	Cut sets[b]
LOOP Sequence 39		8.5 E-004	
1	68.1	5.8 E-004	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, SSFB, OEP-XHE-NOCRS-B, PORV, SEALLOCA, OEP-XHE-NOREC-SL, EPS-XHE-EDGB-NOR
2	29.4	2.5 E-004	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, SSFC, OEP-XHE-NOCRS-B, PORV, SEALLOCA, OEP-XHE-NOREC-SL, EPS-XHE-EDGB-NOR
3	1.7	1.4 E-005	EPS-DGN-CF-ALL, EPS-XHE-NOREC, SSFB, OEP-XHE-NOCRS-B, PORV, SEALLOCA, OEP-XHE-NOREC-SL, EPS-XHE-EDGB-NOR
LOOP Sequence 41		5.3 E-004	
1	94.6	5.0 E-004	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, AFW-TDP-FC-1A, AFW-XHE-NOREC-EP
2	2.8	1.5 E-005	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC AFW-XHE-NOREC-EP, AFW-XHE-XA-NWS
3	2.4	1.2 E-005	EPS-DGN-CF-ALL, EPS-XHE-NOREC, AFW-TDP-FC-1A, AFW-XHE-NOREC-EP
LOOP Sequence 32		3.6 E-004	
1	68.1	2.4 E-004	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, SSFB, OEP-XHE-NOCRS-B, /SEALLOCA, OEP-XHE-NOREC-BD, EPS-XHE-EDGB-NOR
2	29.4	1.0 E-004	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, SSFC, OEP-XHE-NOCRS-B, /SEALLOCA, OEP-XHE-NOREC-BD, EPS-XHE-EDGB-NOR
3	1.7	6.3 E-006	EPS-DGN-CF-ALL, EPS-XHE-NOREC, SSFB, OEP-XHE-NOCRS-B, /SEALLOCA, OEP-XHE-NOREC-BD, EPS-XHE-EDGB-NOR
LOOP Sequence 40		2.7 E-004	
1	32.5	9.0 E-005	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, PORV, PPR-SRV-OO-PRV1
2	32.5	9.0 E-005	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, PORV, PPR-SRV-OO-PRV2
3	32.5	9.0 E-005	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-NOREC, PORV PPR-SRV-OO-PRV3

Table 5. Conditional Cut Sets for Higher Probability Sequences for LER 414/96-001

Cut set No.	Percent Contribution	Conditional Probability[a]	Cut sets[b]
LOOP Sequence 10		7.8 E-005	
1	9.9	7.8 E-006	/EPS-DGN-FC-1A, PORV, PPR-SRV-OO-PRV1, OEP-XHE-NOREC-2H, EPS-DGN-FC-1B, RHR-MDP-FC-1A, HPR-XHE-NOREC-L
2	9.9	7.8 E-006	/EPS-DGN-FC-1A, PORV, PPR-SRV-OO-PRV3, OEP-XHE-NOREC-2H, EPS-DGN-FC-1B, RHR-MDP-FC-1A, HPR-XHE-NOREC-L
3	9.7	7.6 E-006	/EPS-DGN-FC-1A, PORV, PPR-SRV-OO-PRV1, OEP-XHE-NOREC-2H, EPS-DGN-FC-1B, HPI-MDP-FC-1A, HPR-XHE-NOREC-L
4	9.7	7.6 E-006	/EPS-DGN-FC-1A, PORV, PPR-SRV-OO-PRV3, OEP-XHE-NOREC-2H, EPS-DGN-FC-1B, HPI-MDP-FC-1A, HPR-XHE-NOREC-L
5	7.5	5.9 E-006	/EPS-DGN-FC-1A, PORV, PPR-SRV-OO-PRV1, OEP-XHE-NOREC-2H, EPS-DGN-FC-1B, HPR-MOV-CF-SUCA, HPR-MOV-CC-RHRB, HPR-XHE-NOREC-L
6	7.5	5.9 E-006	/EPS-DGN-FC-1A, PORV, PPR-SRV-OO-PRV3, OEP-XHE-NOREC-2H, EPS-DGN-FC-1B, HPR-MOV-CF-SUCA, HPR-MOV-CC-RHRB, HPR-XHE-NOREC-L
7	7.2	5.7 E-006	/EPS-DGN-FC-1A, PORV, PPR-SRV-OO-PRV1, OEP-XHE-NOREC-2H, EPS-DGN-FC-1B, HPR-MOV-CC-SMPA, HPR-XHE-NOREC-L
8	7.2	5.7 E-006	/EPS-DGN-FC-1A, PORV, PPR-SRV-OO-PRV3, OEP-XHE-NOREC-2H, EPS-DGN-FC-1B, HPR-MOV-CC-SMPA, HPR-XHE-NOREC-L
9	7.2	5.7 E-006	/EPS-DGN-FC-1A, PORV, PPR-SRV-OO-PRV1, OEP-XHE-NOREC-2H, EPS-DGN-FC-1B, HPI-MOV-CC-DISCH, HPR-XHE-NOREC-L
10	7.2	5.7 E-006	/EPS-DGN-FC-1A, PORV, PPR-SRV-OO-PRV3, OEP-XHE-NOREC-2H, EPS-DGN-FC-1B, HPI-MOV-CC-DISCH, HPR-XHE-NOREC-L
11	2.4	1.9 E-006	/EPS-DGN-FC-1A, PORV, PPR-SRV-OO-PRV1, OEP-XHE-NOREC-2H, EPS-DGN-FC-1B, HPR-XHE-XM-L
12	2.4	1.9 E-006	/EPS-DGN-FC-1A, PORV, PPR-SRV-OO-PRV3, OEP-XHE-NOREC-2H, EPS-DGN-FC-1B, HPR-XHE-XM-L
13	1.8	1.4 E-006	/EPS-DGN-FC-1A, PORV, PPR-SRV-OO-PRV1, OEP-XHE-NOREC-2H, EPS-DGN-FC-1B, HPI-MDP-CF-ALL, HPR-XHE-NOREC-L

Table 5. Conditional Cut Sets for Higher Probability Sequences for LER 414/96-001

Cut set No.	Percent Contribution	Conditional Probability[a]	Cut sets[b]
14	1.8	1.4 E-006	/EPS-DGN-FC-1A, PORV, PPR-SRV-OO-PRV3, OEP-XHE-NOREC-2H, EPS-DGN-FC-1B, HPI-MDP-CF-ALL, HPR-XHE-NOREC-L
15	1.0	8.6 E-007	/EPS-DGN-FC-1A, PORV, PPR-SRV-OO-PRV1, OEP-XHE-NOREC-2H, EPS-DGN-FC-1B, RHR-MDP-CF-ALL, HPR-XHE-NOREC-L
16	1.0	8.6 E-007	/EPS-DGN-FC-1A, PORV, PPR-SRV-OO-PRV3, OEP-XHE-NOREC-2H, EPS-DGN-FC-1B, RHR-MDP-CF-ALL, HPR-XHE-NOREC-L
Total (all sequences)		**2.1 E-003**	

[a] The conditional probability for each cut se[illegible] is determined by multiplying the probability of the initiating event by the probabilities of the basic events in that minimal cut set. The probability o[illegible] the i[illegible]tiating 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 IE-LOOP, EPS-DGN-FC-1B, OPE-XHE-NOREC-2H, OEP-XHE-NOREC-6H, OEP-XHE-NOREC-BD, OEP-XHE-NOREC-SL, and PORV 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.

LER No. 414/96-001

Event Description: Loss of Offsite Power (LOOP) with Emergency Diesel Generator (EDG) B Unavailable

Date of Event: February 6, 1996

Plant: Catawba 2

Licensee Comments

Reference: Letter from W. R. McCollum, Jr., Catawba Nuclear Station, to U. S. Nuclear Regulatory Commission, transmitting "Response to the Preliminary Accident Sequence Precursor Analysis of Loss of Offsite Power Event at Catawba Unit 2 (TAC M95254)," Duke Power, June 11, 1996.

Comment 1: In the preliminary ORNL analysis only Emergency Diesel Generator (EDG) 2A was considered available for the mission time (7.5 hours) of interest. EDG 2B, which was in maintenance at the time of the event but returned to service at about 3 hours, is treated as in maintenance but potentially recoverable, with a recovery probability of 0.48. [This is basic event EPS-XHE-EDGB-NOR.]

An alternate approach, which perhaps more closely resembles the actual post-event condition, would be to consider only one EDG to be available during the first three hours, with potential recovery of the other EDG, and both EDGs to be available subsequently. Attachment B [of McCollum's letter] presents the development of the dominant related sequence.

At the time of the event, EDG 2B was out of service to perform maintenance on the EDG 2B battery charger. In the event EDG 2B was needed after the LOOP event because no other source of ac power was readily available, plant personnel would have attempted to place EDG 2B into service by clearing the out-of-service tags and closing the breakers. EDG 2B battery is considered to have adequate capacity to start the EDG without the charger. The estimated time to place EDG 2B into a functional status for this scenario is estimated to be in the range of 1 to 1.5 hours. Since EDG 2A was supplying the load, EDG 2B was not needed. It was placed into operation at 2 hours and 51 minutes after the initiating event.

Response 1: The modeling approach taken assumes that EDG 2A is available with a failure probability of 0.045 (EPS-DGN-FC-1A). Not only is the ability to recover EDG 2B within the first 3 h of the event considered (EPS-XHE-EDGB-NOR), it also credits the potential for restoring offsite power (OEP-XHE-NOREC-SL and OPE-XHE-NOREC-BD) and for powering the Unit 2 emergency bus by EDG 2B from the corresponding Unit 1 emergency power bus (OEP-XHE-NOCRS-B) (see Figs. 3 and 4). The Unit 1 emergency power bus remained powered from its normal offsite ac power source.

The nonrecovery probability of 0.48 for EPS-XHE-EDGB-NOR was obtained using a value of 140 min as the median time to repair EDG 2B with a 30-min delay to allow for the decision process to decide to repair EDG 2B. Based on Comment 1, this repair time was adjusted to 90 min with a 30-min delay required for the decision process to reach completion. The value of 3 h for the time available to restore EDG 2B (in case EDG 2A becomes unavailable) was based on the ability of the Standby Shutdown Facility (SSF) to prevent a seal-LOCA and the turbine-driven AFW pump to provide feedwater for decay heat removal. Using the exponential repair model with a repair time of 90 min and a 30-min delay, in combination with the time available to restore EDG 2B (3 h), results in a revised nonrecovery probability of 0.31 for EPS-XHE-EDGB-NOR.

Enclosure 2

Modeling the event in two phases would require revising all basic event (super component) failure probabilities that are dominated by failure to start and failure to run to account for the shorter mission times for the first phase (0 to 3 h). The second phase analysis (3 to 7.5 h) would require revising all basic event failure probabilities that are dominated by failure to start and failure to run to account for this mission time and removing the failure to start probability for those basic events that would be running at the end of the first phase. The final core damage probability would be the sum of the probabilities for each phase developed. Based on an approach *identical* to that provided in Attachment B of McCollum's letter, cut set number 1 in Sequence 39 would be

First 3 h following LOOP event		Next 4.5 h		7.5 h mission
Basic Event	Probability	Basic Event	Probability	Current Value
EPS-DGN-FC-1A EDG 2A fails to start or run .03 + (3 h * 0.002/h)	0.036	EPS-DGN-FC-1A EDG 2A fails to run (4.5 h * 0.002/h)	0.009	0.045
EPS-XHE-EDGB-NOR recovery of EDG 2B	0.31	EPS-XHE-EDGB-NOR recovery of EDG 2B	n/a	0.31
EPS-DGN-FC-1B EDG 2B is in maintenance	1.0	EPS-DGN-FC-1B EDG 2B fails to start or run .03 + (4.5 h * 0.002/h)	0.039	1.0
OEP-XHE-NOCRS-B failure to recover using Unit 1 power	0.27	OEP-XHE-NOCRS-B failure to recover using Unit 1 power	0.27	0.27
SSFB SSF fails to provide seal injection	0.22	SSFB SSF fails to provide seal injection	0.22	0.22
PORV	1.0	PORV	1.0	1.0
SEALLOCA	0.7	SEALLOCA	0.7	0.7
OEP-XHE-NOREC-SL failure to recover offsite power	1.0	OEP-XHE-NOREC-SL failure to recover offsite power	1.0	1.0
	4.6 E-004		1.5 E-005	
	4.8 E-004			5.8 E-004

The difference in the total cut set probability from an unphased approach (5.8×10^{-4}) to a phased approach (4.8×10^{-4}) for this cut set is 17%. Because the other cut sets that are affected by this phased approach are similar to the above cut set, it is expected that the CCDP would be affected similarly (e.g., about 17%). The sequences affected by this approach (sequences 39 and 32) contribute 57.3% to the overall CCDP. A phased approach then, would result in a new CCDP of

$$\text{CCDP} = 1.9 \times 10^{-3} = [(1 - 0.17)(0.573) + (1 - 0.573)] \times 2.1 \times 10^{-3}.$$

Using the Catawba IPE values for EDG start and run probabilities (0.007 and 0.0046/h) with a phased approach reduces the CCDP by 46% for cut set 1 for Sequence 39 (to 2.9×10^{-4}). Because the other cut sets that are affected by this phased approach are similar to the above cut set, it is expected that the CCDP would be affected similarly (e.g., about 46%). The sequences affected by this approach (sequences 39 and 32) contribute 57.3% to the overall CCDP. A phased approach then, would result in a new CCDP of

$$\text{CCDP} = 1.6 \times 10^{-3} = [(1 - 0.46)(0.573) + (1 - 0.573)] \times 2.1 \times 10^{-3}.$$

However, if IPE values are accepted, then the failure probability of the TDAFWP must be adjusted to the IPE value of 0.083 versus the 0.032 used in this analysis. This affects LOOP Sequence 41, cut sets 1 and 3. The CCDP for Sequence 41 becomes 1.3×10^{-3} versus 5.3×10^{-4}. The total CCDP then becomes

$$\text{TOTAL CCDP} = 2.9 \times 10^{-3} = 1.6 \times 10^{-3} + 1.3 \times 10^{-3}.$$

Hence, using a "phased" approach appears to provide no more accurate a core damage probability, yet requires considerably more effort.

Comment 2. The preliminary ORNL analysis uses the EDG start and run failure probabilities of 0.03 and 0.002/hr, respectively. The estimated Catawba EDG start and run failure probabilities, as reported in the Catawba IPE are 0.007 and 0.0046/hr, respectively. In fact, the current 3-year average values, as shown in Attachment C, are 0.003 and 0.0015/hr. Use of the current plant-specific values should change the preliminary conditional core damage probability from 0.0033 to approximately 0.001 without any other changes.

Response 2: The basic event for EDG 2A in the ASP model (EPS-DGN-FC-1A) has an EDG failure probability based on its failure to start (0.03/d) and its failure to run [(0.002/h)(7.5h)]. However, this is really a "super component" because the failure to start probability includes the contribution of all other major components in the safety system train and the contribution from any support systems (e.g., EDG in maintenance, fuel unavailabilities, load sequencer, etc.). Therefore, the 0.03/d is appropriate. The failure to run probability of 0.002/h is consistent with the current 3-year average and about one-half the Catawba IPE value. Regardless, use of the IPE values themselves provides a failure probability consistent with the ORNL analysis (0.0415 vs. 0.045).

Comment 3a: The preliminary ORNL analysis attempts to include the Standby Shutdown Facility (SSF) feature to mitigate a loss of all ac power condition. However, inclusion of both the SSFB and SSFC logic is not correct. The SSFC cut sets are a subset of the SSFB cut sets. (Please see Table A. 18-7 of the Catawba IPE.) The main differences between SSFB and SSFC are that (i) SSFB requires operator action within 15 minutes, while for the SSFC case, action can be delayed for up to 3 hours, and (ii) SSFB contains additional failure modes involving the Reactor Coolant pump seal injection components. In the Catawba Probabilistic Risk Assessment (PRA), success of SSFC does not prevent a core melt; it simply changes the plant damage state.

Comment 3b: In addition, cut sets representing unavailability of the SSF due to maintenance (2.57E-2) and maintenance on the SSF diesel generator (2.96E-3) could be deleted when determining the probability associated with SSFB, since this equipment was available during the event.

Comment 3c: Thus SSFC cut sets and SSF maintenance events should be deleted from the ORNL preliminary analysis. Incorporating these suggestions results in an SSF failure probability of 0.19.

Response 3a: The original fault tree model of the SSF was misleading. The original figure (Fig. 2) simply had "SSF" without identifying "B" or "C." This has been corrected. The placement of the SSF in the event tree (Fig. 1) is such that the SSF is not needed to mitigate core damage until the remaining EDG

fails (EDG 2A) and the turbine-driven AFW pump starts and runs successfully. As indicated on page 35 of Figure A.5-7 in the AFW system fault tree for the Catawba IPE, SSFC is used to ensure power to the turbine-driven pump (TDAFWP) pit sump pump (sump pump 1A1) given a station blackout (SBO) condition. A station blackout would have occurred if EDG 2A failed during this event. Because the motor-driven AFW pumps are unavailable, success of the AFW is dependent on the success of the TDAFWP. The success of the TDAFWP is now dependent on the SSFC to provide power to the TDAFWP pit sump pump because the TDAFWP oil cooler flow and a portion of the turbine seal water empties directly into the pit for the TDAFWP. SSFC must be available within 3 h before the leakage or the water accumulation in the TDAFWP pit would fail the pump. The SSFB, on the other hand, is modeled as providing seal cooling in the event of a station blackout. Specifically, page A.18-15 in the SSF insight section for the Catawba IPE indicates that SSFB is used to provide a means of seal cooling to prevent a seal LOCA in the event of an SBO. SSFB must be available within 15 min of station blackout. The ORNL analysis is consistent with the Catawba PRA.

Response 3b: Because a component successfully performed its function during an event is no reason to reduce or change the failure probability or unavailability of that component. Similarly, it is inappropriate to remove the unavailability due to potential maintenance activities on these components just because no maintenance was being performed on them during the event.

Response 3c: Based on the responses to 3a and 3b, the SSF model appears to be appropriate, and no changes were made to that portion of the analysis.

Comment 4: Catawba has two 4 kV transformers (SATA and SATB) which can power the two essential 4 kV switchgears in one unit from the ac power system from the other unit. The operator action to make use of this feature is contained in the plant emergency procedure, and operators are trained on this action. Catawba's estimate to perform this action is 30 minutes to 1 hour. Considering that this action is required in the emergency procedure, that the operators are given training on it, and that the available time is about 3 hours, the operator failure probability is not considered to be significant. The Duke calculations use 0.17 as the failure probability of this event, derived primarily from the data on LOOP events where more than one unit suffered a LOOP in multi-unit sites. Even with the assumption of a 30 minute cognitive time and a 90 minute action time, the Human Cognitive reliability model (Hannaman et. al., "Human Cognitive Reliability Model for PRA Analysis," NUS-4531, December 1994) yields the operator failure probability as 0.03, with the 3 hour available time. Therefore, the value of 0.35 used in the preliminary ORNL analysis for failure probability of using the Unit 1 ac power when it was indeed available seems a bit conservative. It should be noted that the procedural difficulty encountered during the event should not be viewed as a significant factor since one EDG was operating and supplying necessary loads and the use of Unit 1 ac power was not critically needed at that time.

The Duke analysis of this event, based on Catawba-specific EDG reliability data, the two distinct EDG availability representations, and the applicable SSF failure modes, yielded a conditional core damage probability of approximately 4 E-4. This calculation is based on a base case failure probability of Unit 1 power of 0.17. Since offsite power was available through Unit 1 throughout the mission time, the 0.17 value is conservative. As a sensitivity analysis, changing this value to 0.03 and then to 0.5 produces results of 7 E-5 and 1 E-3, respectively.

Response 4: ORNL assumed that the recovery of emergency power by cross-tieing to the other unit would have been necessary only following the failure of EDG 2A. However, the time required to perform the cross-tie was changed from 90 min to 60 min. The parameters of interest in determining the failure probability are

Parameters	ORNL
available time	3 h
action time	90 min
delay time	30 min
model	E. M. Dougherty and J. R. Fragola, *Human Reliability Analysis*, Ch. 10 & 11, John Wiley and Sons, New York, 1988
"Recovery without Hesitancy" time-reliability correlation	0.16
"Recovery with Hesitancy" time-reliability correlation	0.27

If the operator responds without hesitancy (i.e., although the problem is uncommon, procedures exist), the operator failure probability would be 0.16. This is essentially the same number reported by Duke (i.e., 0.17). The "recovery with hesitancy" time-reliability correlation is appropriate because it considers that although the problem is uncommon, the procedures are weak or sketchy. Because initial efforts to complete the cross-tie to bus 2ETB were unsuccessful due to a procedural inadequacy and cross-tie activities were not completed until 7 h 29 min into the event, the recovery with hesitancy is appropriate. The resultant failure probability is 0.27 given that the cross-tie is an uncommon problem and an error existed in the procedure.

Comment 5: The "Modeling Assumptions" section states that a mission time of 7.5 h was used for EDG 2B. It appears that this mission time was actually used to compute the failure probability of the EDG 2A.

Response 5: This is correct. The mission time was increased from 6 h to 7.5 h for calculating the failure rate of EDG 2A, not EDG 2B. EDG 2B was classified as a TRUE event because it was out-of-service due to a faulty ac capacitor in the battery charger for that EDG.

Comment 6: The following description of the auxiliary feedwater (CA) pump room and associated floor drain system is provided to understand the significance of leakage found in the floor drain sump.

The three CA pumps are located in the CA pump room, which has a floor drain sump (4x3x7 ft) with two load-shed floor drain sump pumps. Each CA pump is located in a pit (17x17x14.5 ft) with two sump pumps for the turbine driven pump and one sump pump for each of the motor driven CA pumps. The CA pump pit sump pumps (each with 50 gpm capacity) are powered by the essential ac power system and one of the turbine driven CA pump sump pumps is backed up by the SSF power.

The CA pump pit sump pumps and the CA pump room sump pumps discharge into a common header.

During the LOOP event some water was found to be accumulating in the CA pump room floor and is attributed to leakage of check valves between the room sump pumps and the common header for

this floor drain system. The leakage was estimated to be no greater than 20 gpm, the expected maximum floor drain requirement for the turbine driven CA pump pit. Considering that the floor area outside the CA pump pits is about 2231.6 square foot and that the curb is about 1.5 feet high, the estimated time for the leakage to spill into the CA pump pits is about 22 hours. The leakage into the turbine driven pump pit is within the capability of the operating sump pump. If this pump failed, an additional 3 hours would be available before the leakage or the water accumulation in the turbine driven pump pit could fail the pump.

For the motor driven CA pump B pit, the estimated time to fill the pump pit is about 26 hours once the leakage starts into the pit and if the sump pump is not operating.

Thus the leakage into the CA pump room from part of the CA pump pit floor discharge should not influence the mission times and success criteria for CA pumps or battery depletion time considerations.

Response 6: The ORNL analysis assumed that the CA pump room flooding had no influence on the mission times, success criteria for the CA pumps, or battery depletion time considerations. The **Additional Event-Related Information** section states that "Because the floor area outside the AFW pump room is about 2,231.6 square feet and the curb is 18 in. high, the estimated time to flood the turbine-driven pump pit area is about 33 h. The leakage into the turbine-driven pump pit is within the capability of the operating sump pump. If this pump failed, an additional 3 hours would be available before the leakage or the water accumulation in the turbine driven pump pit could fail the pump. After the pump pits are flooded, there is an additional area of 1,110 square feet in the room for water to cover. The IPE further estimates that there is 41.6 min available to isolate a flood of 2,429 gpm. Therefore, flooding of the TDAFWP is not considered credible because considerable time was available to mitigate the flooding." The concern with flooding the pump pit was that given a LOOP and failure of the remaining EDG, the only means of decay heat removal was use of the turbine-driven AFW pump to provide makeup to the steam generators. The LER and preliminary information about the CA pump room flooding were sketchy; however, it was determined that the CA pump room flooding was of little consequence to this event.