Docket File

50-254/265

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

April 25, 1996

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 OPUS Place, Suite 500
Downers Grove, IL 60515

SUBJECT: DRAFT 1982-83 PRECURSOR REPORT

Dear Mr. Farrar:

Enclosed for your information are excerpts from the draft Accident Sequence Precursor (ASP) Report for 1982-83. This report documents the Accident Sequence Precursor (ASP) Program analyses of operational events which occurred during the period 1982-83. We are providing the appropriate section[s] of this draft report to each licensee with a plant which had an event in 1982 or 1983 that has been identified as a precursor. At least one of these precursors occurred at Quad Cities Nuclear Power Station. Also enclosed for your information are copies of Section 2.0 and Appendix A from the 1982-83 ASP Report. Section 2.0 discusses the ASP Program event selection criteria and the precursor quantification process; Appendix A describes the models used in the analyses. We emphasize that you are under no licensing obligation to review and comment on the enclosures.

The analyses documented in the draft ASP Report for 1982-83 were performed primarily for historical purposes to obtain the two years of precursor data for the NRC's ASP Program which had previously been missing. We realize that any review of the precursor analyses of 1982-83 events by affected licensees would necessarily be limited in scope due to: (1) the extent of the licensee's corporate memory about specific details of an event which occurred 13-14 years ago, (2) the desire to avoid competition for internal licensee staff resources with other, higher priority work, and (3) extensive changes in plant design, procedures, or operating practices implemented since the time period 1982-83, which may have resulted in significant reductions in the probability of (or, in some cases, even precluded) the occurrence of events such as those documented in this report.

The draft report contains detailed documentation for all precursors with conditional core damage probabilities $\geq 1.0 \times 10^{-5}$. However, the relatively large number of precursors identified for the period 1982-83 necessitated that only summaries be provided for precursors with conditional core damage probabilities between 1.0×10^{-6} and 1.0×10^{-5}.

300041

DF01 /1

We will begin revising the report about May 31, 1996, to put it in final form for publication. We will respond to any comments on the precursor analyses which we receive from licensees. The responses will be placed in a separate section of the final report. Commonwealth Edison Company is on distribution for the final report. Please contact me at (301) 415-3016 if you have any questions regarding this letter. Any response to this letter on your part is entirely voluntary and does not constitute a licensing requirement.

Sincerely,

Original signed by:

Robert M. Pulsifer, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-254, 50-265

Enclosures: 1) B.4 LER No. 254/82-007
2) B.5 LER No. 254/82-012, 254/82-013, and 254/82-018
3) C.4 LER No. 265/82-010
4) 2.0 Selection Criteria and Quantification
5) Appendix A: ASP MODELS

cc w/encls: see next page

Distribution:
Docket File
PUBLIC
PDIII-2 R/F
J. Roe (JWR)
R. Capra
C. Moore
R. Pulsifer
OGC O-15 B18
ACRS T-2 E26
B. Clayton, RIII
P. O'Reilly, T-4 A9

DOCUMENT NAME: PULSIFER\ASP.LTR

To receive a copy of this document, indicate in the box: "C" = Copy without enclosures "E" = Copy with enclosures "N" = No copy

OFFICE	LA;PDIII-2	C	PM;PDIII-2	E	D;PDIII-2	E			
NAME	CMOORE		RPULSIFER		RCAPRA				
DATE	04/25/96		04/25/96		04/25/96				

OFFICIAL RECORD COPY

D. L. Farrar
Commonwealth Edison Company

Quad Cities Nuclear Power Station
Unit Nos. 1 and 2

cc:

Michael I. Miller, Esquire
Sidley and Austin
One First National Plaza
Chicago, Illinois 60603

Mr. L. William Pearce
Station Manager
Quad Cities Nuclear Power Station
22710 206th Avenue North
Cordova, Illinois 61242

U.S. Nuclear Regulatory Commission
Quad Cities Resident Inspectors Office
22712 206th Avenue North
Cordova, Illinois 61242

Chairman
Rock Island County Board
of Supervisors
1504 3rd Avenue
Rock Island County Office Bldg.
Rock Island, Illinois 61201

Illinois Department of Nuclear Safety
Office of Nuclear Facility Safety
1035 Outer Park Drive
Springfield, Illinois 62704

Regional Administrator
U.S. NRC, Region III
801 Warrenville Road
Lisle, Illinois 60532-4351

Richard J. Singer
Manager - Nuclear
MidAmerican Energy Company
907 Walnut Street
P.O. Box 657
Des Moines, Iowa 50303

Brent E. Gale, Esq.
Vice President - Law and
Regulatory Affairs
MidAmerican Energy Company
One RiverCenter Place
106 East Second Street
P.O. Box 4350
Davenport, Iowa 52808

Document Control Desk-Licensing
Commonwealth Edison Company
1400 Opus Place, Suite 400
Downers Grove, Illinois 60515

B.4 LER No. 254/82-007

Event Description: Transient with RHRSW train B inoperable

Date of Event: April 15, 1982

Plant: Quad Cities 1

B.4.1 Summary

During normal operation on April 15, 1982, Residual Heat Removal Service Water (RHRSW) pump D outboard bearing was found to be failed due to excessive leakage of water from the adjacent packing to the oil in the bearing. On April 30th, RHRSW pump C was taken out of service for maintenance on the pump seal packing. Water which leaked from adjacent seal packing was found in the bearing oil reservoir. Three plant trips had occurred around the time of the faults in the pumps (April 17, 19, and 30). The conditional core damage probability estimated for this event is 7.2×10^{-4}.

B.4.2 Event Description

During normal operation on April 15, 1982, RHRSW pump D outboard bearing was found to be failed during a surveillance test. Investigation revealed that the pump bearing failed due to excessive leakage of water from the adjacent packing to the oil in the bearing. The bearing and packing was replaced and the pump was returned to service on April 22nd. A few days later, on April 30th, RHRSW pump C was taken out of service for maintenance on the pump seal packing. Water which leaked from adjacent seal packing was found in the bearing oil reservoir. The licensee stated that while there was insufficient water to cause bearing damage due to a loss of lubrication, continued operation could have possibly resulted in bearing damage. The pump was declared inoperable. The pump seals were repacked and the oil in the bearing oil reservoir was replaced. The pump was returned to service later that day.

Three plant trips occurred around the time of the discovery of the bearing faults in the pumps (April 17, 19, and 30). The plant trip on April 17th involved a reactor scram due to low condenser vacuum due to a condensate demineralizer valve failure. The plant trip on April 19th involved a reactor scram due to high main steam line flow. The plant trip on April 30th involved a trip on low reactor water level due to a B reactor feedpump discharge valve closure (ref: NUREG-0200).

B.4.3 Additional Event-Related Information

The Residual Heat Removal Service water system provides cooling water to the Residual Heat Removal (RHR) system heat exchangers. RHR is a two train system (A and B) which provides three functions: Suppression Pool Cooling, Containment Spray, and Shutdown cooling. Each train has two RHR pumps and one heat exchanger. Suppression Pool Cooling is used to remove heat from the suppression pool whenever the water temperature exceeds 95 F. Containment Spray is used in the event of a nuclear system break within the primary containment to prevent excessive containment pressure and temperature by condensing steam and cooling non-condensable gases. Shutdown Cooling can be used during normal shutdown and cooldown to remove decay heat once the reactor coolant temperature is low enough that the steam supply pressure is not sufficient to maintain turbine shaft gland seals or vacuum in the main condensor. RHR requires the use of one pump and one functioning heat exchanger (and thus one train of RHRSW) for Suppression Pool Cooling, Containment Spray, and Shutdown Cooling. RHRSW is a two train system (A and B). Each RHRSW train has two pumps and one heat exchanger. Pumps A and B supply heat exchanger A for RHR train A. Pumps C and D supply heat exchanger B for RHR train B. RHRSW also has a crosstie which enables the RHRSW pumps to provide coolant to the RHR system for use as an alternate injection system. Two RHRSW pumps supplying flow to one heat exchanger is sufficient for all RHR modes. One RHRSW pump is sufficient to provide the alternate injection source for RHR.

B.4.4 Modeling Assumptions

RHRSW and thus RHR were assumed to be degraded at the time of the trip on April 17, 1982. The event was modeled as a transient with Feedwater (FW) inoperable and degraded RHR. Assuming that the water was present in the lube oil for both pumps C and D at the time of the transient, two of the four RHRSW pumps were assumed to fail during their mission time, and potiential failure of the other two pumps from similar causes was assumed. The potential for common cause failure exists, even when a component is failed. Therefore, the conditional probability of a common-cause failure was included in the analysis for those components that were assumed to have been failed as part of the postulated event. Since the ASP model assumes that common cause failure of the RHR pumps dominate the failure of RHR and does not directly account for the failure of RHRSW pumps leading to RHR failure, the RHR failure probability was modified to reflect the degraded state of RHRSW in this event. The conditional train probabilities for RHRSW pumps shown in Table 1 were combined and added to the probability of RHR failure as follows

$$P(RHRSW) = P(A|DC)*P(B|ADC)$$

$$P(RHR)_{NEW} = P(RHR)_{OLD} + P(RHRSW)$$

$$P(RHR)_{NEW} = P(RHR)_{OLD} + 0.15.$$

Table 1. RHRSW Pump Train Failure to Start and Run Conditional Failure Probabilities

Train	Conditional Failure Probability
P(1)	0.01
P(2|1)	0.1
P(3|12)	0.3
P(4|123)	0.5

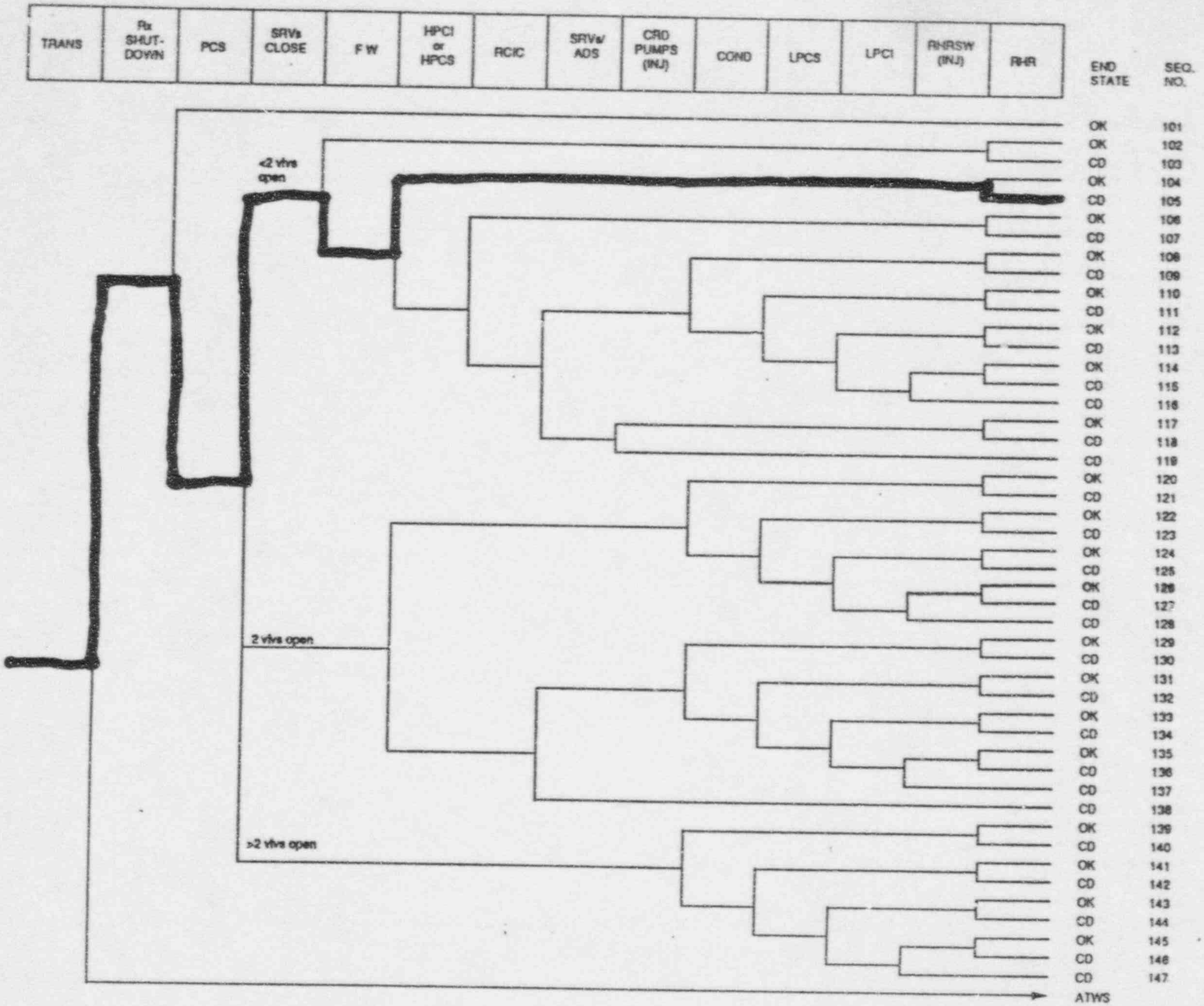
The suppression pool cooling mode of RHR would also be affected in the same manner. Thus, P(RHRSW) was added to the branch probability for RHR(SPCOOL) in the same manner as described above. The same modifications were made to RHR/-LPCI and RHR(SPCOOL)/-LPCI. Since there would still be ample time to recover RHR given LPCI success, the non-recovery probability for RHR/-LPCI was set to the same nominal non-recovery probability as that for RHR.

A sensitivity study was done assuming that the water leak into the bearing oil reservoir for pump C was not sufficient to cause pump C to fail. RHR, RHR(SPCOOL), RHR/-LPCI, and RHR(SPCOOL)/-LPCI were modified to reflect only one failed RHRSW pump ($p = 0.015$).

The nonrecovery probability for RHR was revised to 0.054 to reflect the RHRSW failure (based on data included in "Faulted Systems Recovery Experience," NSAC-161, May 1992). For sequences involving potential RHR or PCS recovery, the nonrecovery estimate was revised to 0.054 x 0.52 (PCS nonrecovery), or 0.028.

B.4.5 Analysis Results

The estimated conditional core damage probability is 7.2×10^{-4}. The dominant sequence involves a successful reactor shutdown, failure of the power conversion system, failure of feedwater, and failure of RHR and is highlighted in the event tree in Figure B.4.1 (to be provided in final report). The estimated conditional core damage probability for the sensitivity study (with RHRSW pump C operable) is 7.5×10^{-5}. The dominant sequence remained the same.



TRANS
Rx SHUT-DOWN
PCS
SRVs CLOSE
FW
HPCI or HPCS
RCIC
SRVs/ ADS
CRD PUMPS (INJ)
COND
LPCS
LPCI
RHRSW (INJ)
RHR
END STATE
SEQ. NO.
<2 vlvs open
2 vlvs open
>2 vlvs open
OK 101
OK 102
CD 103
OK 104
CD 105
OK 106
CD 107
OK 108
CD 109
OK 110
CD 111
OK 112
CD 113
OK 114
CD 115
CD 116
OK 117
CD 118
CD 119
OK 120
CD 121
OK 122
CD 123
OK 124
CD 125
OK 126
CD 127
CD 128
OK 129
CD 130
OK 131
CD 132
OK 133
CD 134
OK 135
CD 136
CD 137
CD 138
OK 139
CD 140
OK 141
CD 142
OK 143
CD 144
OK 145
CD 146
CD 147
ATWS

Fig B.4.1

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 254/82-007
Event Description: Transient with RHRSW train B inop
Event Date: April 15, 1982
Plant: Quad Cities 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
TRANS	7.2E-04
Total	7.2E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

Sequence	End State	Prob	N Rec**
105 trans -rx.shutdown pcs srv.ftc.<2 FW -hpci RHR.AND.PCS.NREC	CD	7.0E-04	2.8E-02

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

Sequence	End State	Prob	N Rec**
105 trans -rx.shutdown pcs srv.ftc.<2 FW -hpci RHR.AND.PCS.NREC	CD	7.0E-04	2.8E-02

** non-recovery credit for edited case

SEQUENCE MODEL: d:\asp\models\bwrc8283.cmp
BRANCH MODEL: d:\asp\models\quadcit1.82
PROBABILITY FILE: d:\asp\models\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	1.5E-03	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
pcs	1.7E-01	1.0E+00	
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	

FW	2.9E-01 > 1.0E+00	3.4E-01 > 1.0E+00	
Branch Model: 1.0F.1			
Train 1 Cond Prob:	2.9E-01 > 1.0E+00		
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
RHR	1.5E-04 > 1.5E-01 **	1.6E-02 > 5.4E-02	1.0E-05
Branch Model: 1.0F.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR.AND.PCS.NREC	1.5E-04 > 1.5E-01 **	8.3E-03 > 2.8E-02	1.0E-05
Branch Model: 1.0F.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
RHR/-LPCI	0.0E+00 > 1.5E-01 **	1.0E+00 > 5.4E-02	1.0E-05
Branch Model: 1.0F.1+opr			
Train 1 Cond Prob:	0.0E+00		
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
RHR(SPCOOL)	2.1E-03 > 1.5E-01 **	1.0E+00	1.0E-03
Branch Model: 1.0F.4+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Train 4 Cond Prob:	5.0E-01		
Serial Component Prob:	2.0E-03		
RHR(SPCOOL)/-LPCI	2.0E-03 > 1.5E-01 **	1.0E+00	1.0E-03
Branch Model: 1.0F.1+ser+opr			
Train 1 Cond Prob:	0.0E+00		
Serial Component Prob:	2.0E-03		
ep	2.9E-03	8.7E-01	
ep.rec	4.9E-02	1.0E+00	
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

Dolan
01-08-1996
20:10:18

Event Identifier: 254/82-007

B.5 LER No. 254/82-012, -013, and -018

Event Description:	Postulated LOOP with 2 EDGs inoperable (Unit 1) and Plant-centered LOOP with one EDG inoperable (Unit 2)
Date of Event:	June 22, 1982
Plant:	Quad Cities 1 and 2

B.5.1 Summary

During normal operation on June 22, 1982, Unit 2 reactor experienced a trip while the reserve auxiliary transformer 22 was being removed from service for maintenance. Upon the loss of off-site power (LOOP), both the Unit 2 and swing emergency diesel generators (EDGs) loaded to their respective emergency buses. The swing diesel generator tripped when the A Residual Heat Removal Service Water (RHRSW) pump was started. One day prior to the Unit 2 loss of off-site power, the Unit 1 EDG was removed from service due to the failure of the diesel generator cooling water pump to provide flow to the EDG during a HPCI flow rate surveillance test. Thus, when the Unit 2 LOOP occurred, Unit 1 began operating without any EDGs available. The estimated increase in core damage probability over the duration of the postulated LOOP at Unit 1 with both EDGs inoperable is 2.2×10^{-5}. The conditional core damage probability estimate for a plant-centered LOOP with one EDG inoperable at Unit 2 is 1.3×10^{-4}.

B.5.2 Event Description

During normal operation on June 22, 1982, at 0526 hours, Unit 2 reactor experienced a trip due to a reactor feedwater pump trip and subsequent low water level due to a loss of bus 22 while the reserve auxiliary transformer 22 was being removed from service for maintenance. An equipment operator mistakenly pulled out the fuses for a 4-kilovolt bus instead of pulling the transformer fuses. The error disconnected power to the 2B reactor feedwater pump which caused a low water level and initiated a trip. The Unit 2 main generator subsequently tripped and all normal AC power to Unit 2 was lost. Upon the loss of off-site power (LOOP), both the Unit 2 and swing emergency diesel generators (EDGs) loaded to their respective emergency buses. The swing diesel generator tripped when the A Residual Heat Removal Service Water (RHRSW) pump was started, approximately 22 minutes after the fuses were pulled, due to under-excitation trip. The EDG underexcitation relay was unblocked and thus tripped when the RHRSW pump was initiated. Actuation of the underexcitation relay tripped the EDG lock-out relay as well. To restart the EDG, the relay had to be manually reset by the equipment operator. The resetting of the lock-out relay was delayed since the equipment operator had been sent to the switchyard to expedite the restoration of offsite power. The trip of the unblocked relay was attributed to a design flaw. The under-excitation relays were temporarily removed on all three diesel generators until a permanent design change could be completed.

One day prior to the Unit 2 LOOP, the Unit 1 EDG was removed from service due to the failure of the diesel generator cooling water pump to provide flow to the EDG during a HPCI flow rate surveillance test. Investigation revealed that the pump was air bound due to air which entered the suction line while RHRSW A was being drained to install system modifications. The rotating element of the pump was replaced and the pump was returned to service in the late afternoon of June 22nd. When the Unit 2 LOOP occurred, Unit 1 began operating without any EDGs available.

On June 26th, the Unit 1 EDG cooling water pump was again removed from service to reduce the vibration of the pump due to misalignment of the motor and pump. The motor and pump were re-aligned, and the pump was returned to service.

B.5.3 Additional Event-Related Information

Quad Cities Units 1 and 2 each have one EDG (EDG 1 and EDG 2) dedicated to that unit. They share a common swing EDG (EDG ½). EDGs 1 and 2 supply emergency power buses 14-1 and 24-1, respectively, which power containment spray pumps 1B and 2B, RHR pumps 1C, 1D, 2C, and 2D, and RHRSW pumps 1C, 1D, 2C, and 2D. The swing EDG provides emergency power to buses 13-1 or 23-1 which power Containment Spray pump 1A or 2A, RHR pumps 1A and 1B or 2A and 2B, and RHRSW pumps 1A and 1B or 2A and 2B. The emergency power buses are automatically fed from the EDGs on the loss of off-site power. Unit 1 bus 14-1 and Unit 2 bus 24-1 can be cross-tied by closing two normally open breakers.

Two 250-V dc and two 125-V dc batteries are shared between both Units. Each battery is sized to power its respective loads for 4 hours. Unit 1 batteries are charged from bus 14-1 through bus 19, and Unit 2 batteries are charged from bus 24-1 through bus 29. An alternate charger can be powered from bus 13-1 and 23-1 and can charge either unit's battery. The 480-V ac buses power the battery chargers on each unit and can also be cross-tied.

B.5.4 Modeling Assumptions

This event was modeled as two separate events. The first analysis considers a postulated LOOP with two EDGs inoperable for Unit 1 and assumes that both of the EDGs were inoperable for up to half the surveillance period on the EDGs, 15 days. One train of emergency power (EP) was set to failed to reflect the failure of EDG ½, and the other train was set to unavailable to reflect EDG 1's unavailability due to maintenance. Recovery of power to the Unit 1 buses was assumed to occur following the recovery of offsite power. Recovery of power to bus 14-1 was also assumed possible from Unit 2 bus 24-1 by the closure of the normally open breakers 2429 and 1421. The probability of failing to open the breakers before battery depletion was assumed to be 0.10. This value was taken from Table X in section XXX of this report and reflects the operators ability of performing the required non-routine actions in the required time from the control room. Thus , the non-recovery factor for power recovery prior to battery depletion (EP.REC) was set to 0.10 to incorporate recovery by the opening the breakers.

The second analysis considers the plant-centered LOOP which occurred at Unit 2 and the inoperability of the swing EDG. The LOOP frequency and the probabilities of failing to recover offsite power in the short-term and before battery depletion were modified for a plant-centered LOOP using the models described in *Revised LOOP Frequency and PWR Seal LOCA Models*, ORNL/NRC/LTR-89/11, August 1989. One train of emergency power was set to failed to reflect the failure of EDG ½, and all associated equipment powered by the swing EDG was set to unavailable. The non-recovery factor for the recovery of offsite power prior to battery depletion was set to 0.10 to reflect the ability to recovery power to bus 24-1 from Unit 1 bus 14-1.

In this event, Unit 1 remained operating during the Unit 2 LOOP. Had Unit 1 tripped and experienced a LOOP during the Unit 2 LOOP, Unit 1 would have experienced station blackout.

B.5.5 Analysis Results

The estimated increase in core damage probability over the duration of the postulated LOOP at Unit 1 is 2.2×10^{-5}. The dominant sequence highlighted on the event tree in Figure B.5.1, involved a successful reactor shutdown, failure of emergency power, and failure to recover off-site power prior to battery depletion. The estimated conditional core damage probability for the Unit 2 plant-centered LOOP with one EDG inoperable is 1.3×10^{-4}. The dominant sequence involves a successful reactor shutdown, successful emergency power, successful HPCI, and failure of RHR.

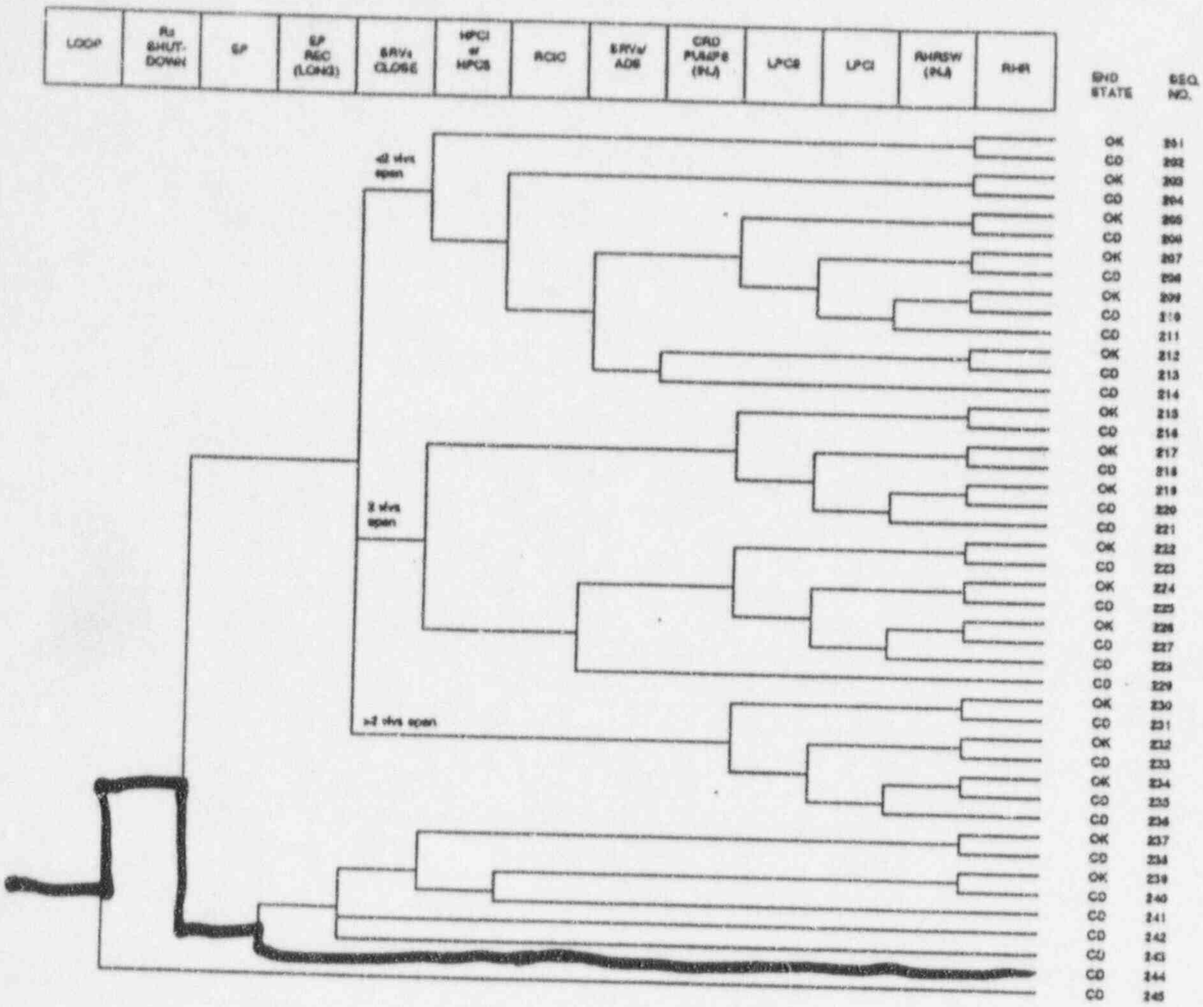

LOOP
Rx SHUT-DOWN
EP
EP REC (LONG)
SRVs CLOSE
HPCI or HPCS
RCIC
SRVs/ ADS
CRD PUMPS (INJ)
LPCS
LPCI
RHRSW (INJ)
RHR
END STATE
SEQ. NO.
<2 vlvs open
2 vlvs open
>2 vlvs open
OK 201
CD 202
OK 203
CD 204
OK 205
CD 206
OK 207
CD 208
OK 209
CD 210
CD 211
OK 212
CD 213
CD 214
OK 215
CD 216
OK 217
CD 218
OK 219
CD 220
CD 221
OK 222
CD 223
OK 224
CD 225
OK 226
CD 227
CD 228
CD 229
OK 230
CD 231
OK 232
CD 233
OK 234
CD 235
CD 236
OK 237
CD 238
OK 239
CD 240
CD 241
CD 242
CD 243
CD 244
CD 245

Fig B.5.1

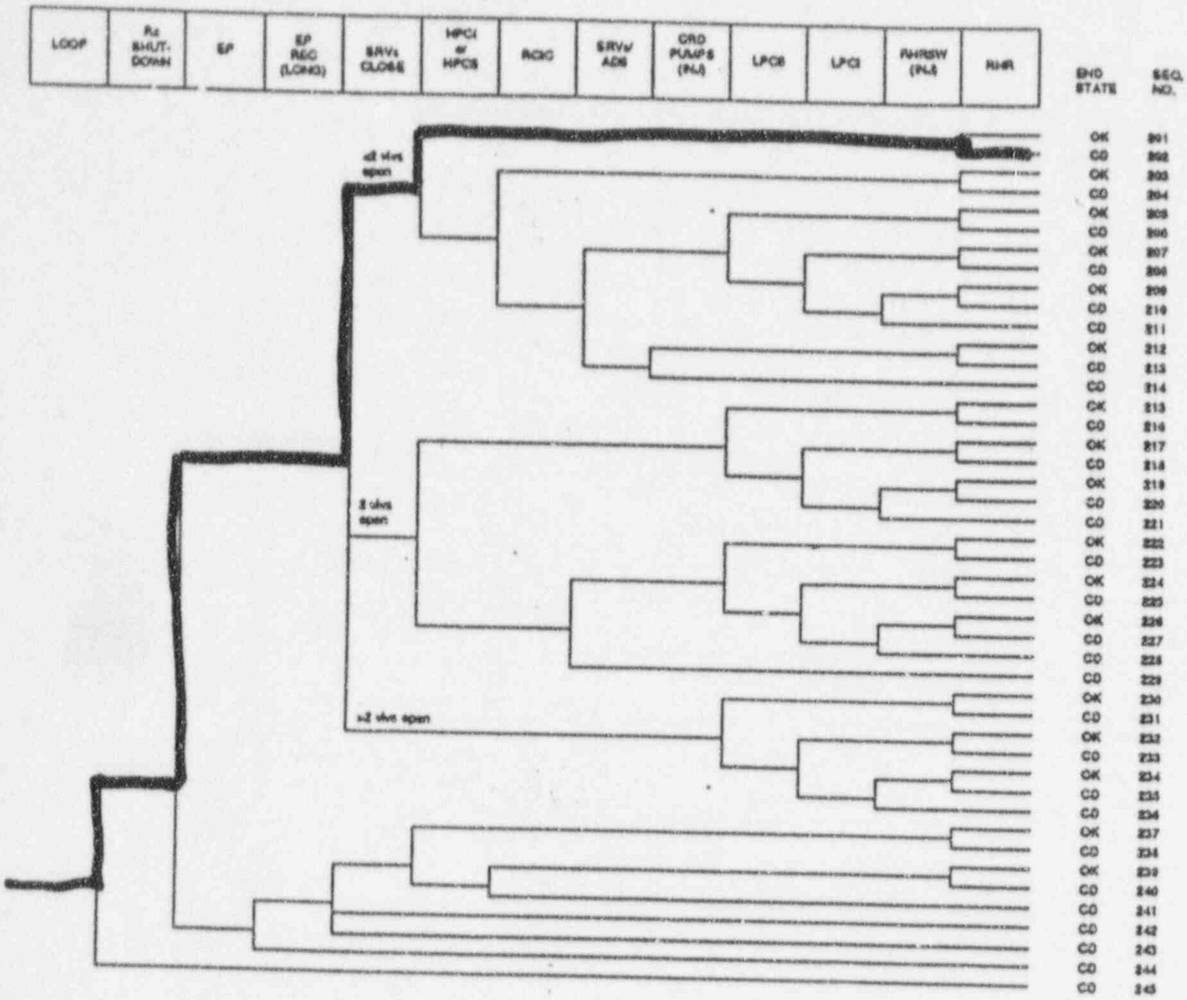

LOOP
Rx SHUT-DOWN
EP
EP REC (LONG)
SRVs CLOSE
HPCI or HPCS
RCIC
SRVs/ ADS
CRD PUMPS (INJ)
LPCS
LPCI
RHRSW (INJ)
RHR
END STATE
SEQ. NO.
<2 vlvs open
2 vlvs open
>2 vlvs open
OK 201
CD 202
OK 203
CD 204
OK 205
CD 206
OK 207
CD 208
OK 209
CD 210
CD 211
OK 212
CD 213
CD 214
OK 215
CD 216
OK 217
CD 218
OK 219
CD 220
CD 221
OK 222
CD 223
OK 224
CD 225
OK 226
CD 227
CD 228
CD 229
OK 230
CD 231
OK 232
CD 233
OK 234
CD 235
CD 236
OK 237
CD 238
OK 239
CD 240
CD 241
CD 242
CD 243
CD 244
CD 245

Figure B.5.2

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 254/82-012, -013, and -018
Event Description: Postulated LOOP with two EDGs inoperable (Unit 1)
Event Date: June 22, 1982
Plant: Quad Cities 1

UNAVAILABILITY, DURATION= 360

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	3.1E-03

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	2.2E-05
Total	2.2E-05

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
244	loop -rx.shutdown EP EP.REC	CD	1.5E-05	5.3E-02
242	loop -rx.shutdown EP -EP.REC srv.ftc.2	CD	4.0E-06	5.3E-01
241	loop -rx.shutdown EP -EP.REC srv.ftc.<2 hpci rcic	CD	2.6E-06	2.6E-01
243	loop -rx.shutdown EP -EP.REC srv.ftc.>2	CD	6.8E-07	5.3E-01

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
241	loop -rx.shutdown EP -EP.REC srv.ftc.<2 hpci rcic	CD	2.6E-06	2.6E-01
242	loop -rx.shutdown EP -EP.REC srv.ftc.2	CD	4.0E-06	5.3E-01
243	loop -rx.shutdown EP -EP.REC srv.ftc.>2	CD	6.8E-07	5.3E-01
244	loop -rx.shutdown EP EP.REC	CD	1.5E-05	5.3E-02

** non-recovery credit for edited case

Note: For unavailabilities, conditional probability values are differential values which reflect the added risk due to failures associated with an event. Parenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

SEQUENCE MODEL: c:\aspcode\models\bwrc8283.cmp
BRANCH MODEL: c:\aspcode\models\quadcit1.82
PROBABILITY FILE: c:\aspcode\models\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail

trans	1.5E-03	1.0E+00	
loop	1.6E-05	5.3E-01	
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
pcs	1.7E-01	1.0E+00	
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
mfw	2.9E-01	3.4E-01	
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
lpcs	2.0E-03	1.0E+00	
lpci	1.1E-03	1.0E+00	
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
rhr	1.5E-04	7.0E-02	1.0E-05
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.1E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
EP	2.9E-03 > 1.0E+00	8.7E-01 > 1.0E+00	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	5.0E-02 > Failed		
Train 2 Cond Prob:	5.7E-02 > Unavailable		
EP.REC	4.9E-02 > 4.9E-02	1.0E+00 > 1.0E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.9F-02		
rpt	1.9E-02	1.0E+00	
[illegible]	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

Heather Schriner
09-25-1995
13:18:26

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 254/82-012, -013, and -018
Event Description: Plant-centered LOOP with one EDG inoperable (Unit 2)
Event Date: June 22, 1982
Plant: Quad Cities 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP	5.0E-01

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CD	
LOOP	1.3E-04
Total	1.3E-04

SEQUENCE CONDITIONAL PROBABILITIES (PROBABILITY ORDER)

	Sequence	End State	Prob	N Rec**
202	LOOP -rx.shutdown -EP srv.ftc.<2 -hpci RHR	CD	3.7E-05	3.5E-02
242	LOOP -rx.shutdown EP -EP.REC srv.ftc.2	CD	3.2E-05	4.3E-01
241	LOOP -rx.shutdown EP -EP.REC srv.ftc.<2 hpci rcic	CD	2.1E-05	2.1E-01
245	LOOP rx.shutdown	CD	1.8E-05	5.0E-02
244	LOOP -rx.shutdown EP EP.REC	CD	1.6E-05	4.3E-02
243	LOOP -rx.shutdown EP -EP.REC srv.ftc.>2	CD	5.5E-06	4.3E-01
238	LOOP -rx.shutdown EP -EP.REC srv.ftc.<2 -hpci RHR	CD	1.9E-06	3.0E-02

** non-recovery credit for edited case

SEQUENCE CONDITIONAL PROBABILITIES (SEQUENCE ORDER)

	Sequence	End State	Prob	N Rec**
202	LOOP -rx.shutdown -EP srv.ftc.<2 -hpci RHR	CD	3.7E-05	3.5E-02
238	LOOP -rx.shutdown EP -EP.REC srv.ftc.<2 -hpci RHR	CD	1.9E-06	3.0E-02
241	LOOP -rx.shutdown EP -EP.REC srv.ftc.<2 hpci rcic	CD	2.1E-05	2.1E-01
242	LOOP -rx.shutdown EP -EP.REC srv.ftc.2	CD	3.2E-05	4.3E-01
243	LOOP -rx.shutdown EP -EP.REC srv.ftc.>2	CD	5.5E-06	4.3E-01
244	LOOP -rx.shutdown EP EP.REC	CD	1.6E-05	4.3E-02
245	LOOP rx.shutdown	CD	1.8E-05	5.0E-02

** non-recovery credit for edited case

SEQUENCE MODEL: c:\aspcode\models\bwrc8283.cmp
BRANCH MODEL: c:\aspcode\models\quadcit2.82
PROBABILITY FILE: c:\aspcode\models\bwr8283.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
trans	6.9E-04	1.0E+00	
LOOP	1.6E-05 > 1.4E-05	5.3E-01 > 5.0E-01	
Branch Model: INITOR			
Initiator Freq:	1.6E-05 > 1.4E-05		
loca	3.3E-06	6.7E-01	
rx.shutdown	3.5E-04	1.0E-01	
pcs	1.7E-01	1.0E+00	
srv.ftc.<2	1.0E+00	1.0E+00	
srv.ftc.2	1.3E-03	1.0E+00	
srv.ftc.>2	2.2E-04	1.0E+00	
mfw	2.9E-01	3.4E-01	
hpci	2.9E-02	7.0E-01	
rcic	6.0E-02	7.0E-01	
srv.ads	3.7E-03	7.0E-01	1.0E-02
crd(inj)	1.0E-02	1.0E+00	1.0E-02
cond	1.0E+00	3.4E-01	1.0E-03
LPCS	2.0E-03 > 2.0E-02	1.0E+00	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	2.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
LPCI	1.1E-03 > 2.0E-03	1.0E+00	
Branch Model: 1.OF.4+ser			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01 > Unavailable		
Train 4 Cond Prob:	5.0E-01 > Unavailable		
Serial Component Prob:	1.0E-03		
rhrsw(inj)	2.0E-02	1.0E+00	1.0E-02
RHR	1.5E-04 > 1.0E-03	7.0E-02	1.0E-05
Branch Model: 1.OF.4+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01 > Unavailable		
Train 4 Cond Prob:	5.0E-01 > Unavailable		
rhr/-lpci	0.0E+00	1.0E+00	1.0E-05
rhr/lpci	1.0E+00	1.0E+00	1.0E-05
rhr(spcool)	2.1E-03	1.0E+00	1.0E-03
rhr(spcool)/-lpci	2.0E-03	1.0E+00	1.0E-03
EP	2.9E-03 > 5.7E-02	8.7E-01	
Branch Model: 1.OF.2			
Train 1 Cond Prob:	5.0E-02 > Failed		
Train 2 Cond Prob:	5.7E-02		
EP.REC	4.9E-02 > 6.4E-03	1.0E+00 > 1.0E-01	
Branch Model: 1.OF.1			
Train 1 Cond Prob:	4.9E-02 > 6.4E-03		
rpt	1.9E-02	1.0E+00	
slcs	2.0E-03	1.0E+00	1.0E-02
ads.inhibit	0.0E+00	1.0E+00	1.0E-02
man.depress	3.7E-03	1.0E+00	1.0E-02

* branch model file
** forced

Heather Schriner
09-25-1995
13:21:17

C.4 LER No. 265/82-010

Event Description: Transient with HPCI inoperable

Date of Event: June 24, 1982

Plant: Quad Cities 2

Summary

On June 24, 1982, with the plant increasing power from 21% in preparation for rolling the turbine and placing the unit on-line, HPCI pump discharge motor-operated valve 2-2301-8 failed to open when given a signal from the control room during a HPCI valve operability surveillance test. HPCI was declared inoperable. The valve was manually opened and taken out of service. Investigation revealed that the open torque switch in the motor operator had a broken arm. The arm was replaced and the operator reassembled the valve. The valve was opened successfully three times, and HPCI was returned to service the next day.

A plant trip occurred approximately two days prior to the discovery of the faulty HPCI pump discharge valve. Thus, this event was modeled as a transient with HPCI assumed inoperable. The HPCI train probability was set to failed and the HPCI non-recovery probability was set to 0.55 to reflect the ability of the operators to recover HPCI locally within the allowable recovery time (see Appendix A). The estimated conditional core damage probability for this event is 4.7×10^{-6}. The dominant sequence involves the trip with a postulated failure of the power conversion system, successful operation of main feedwater, and the failure of the residual heat removal system.

C.5 LER No. 265/82-017 and -018

Event Description: HPCI and one EDG inoperable

Date of Event: October 1, 1982

Plant: Quad Cities 2

Summary

On October 1, 1982, during a routine surveillance a small leak was discovered in the High Pressure Coolant Injection (HPCI) system supply steam line break flange due to a failed flange gasket. The licensee stated that the steam leakage may have been sufficient to cause HPCI isolation on high HPCI area temperature following prolonged operation. A few days later on October 6, 1982, following monthly preventative maintenance on Emergency Diesel Generator (EDG) 2, the EDG tripped on high temperature ten minutes after loading due to fouled EDG cooling water system heat exchangers. Thus, this event was modeled as an unavailability of HPCI and one EDG. Assuming that both HPCI and the EDG were faulted for a period of half their surveillence periods prior to the discovery of the faults, the duration of the unavailability was estimated to be ten days (240 hours). To reflect the failure of EDG 2, one train of emergency power was set to failed and all system trains which rely on EDG 2 (bus 24-1) given a LOOP were set to unavailable. Since Unit 2 bus 24-1 can be fed by Unit 1 bus 14-1 through cross-connection, recovery of power to bus 24-1 was assumed possible from Unit 1 bus 14-1 by the closure of the normally open breakers 2429 and 1421 for plant-centered LOOPs. Thus, this event was modeled as two cases. The first case examines the likelihood of the occurrence of a plant-centered LOOP during the unavailability with credit given for the ability to recover power through the use of the cross-connect. In this case, the LOOP frequency was revised to 1.39×10^{-5} with a short-term non-recovery probability of 0.5, and off-site power recovery prior to battery depletion (EP.REC) was modified to 6.4×10^{-3} to reflect values for plant-centered LOOPs determined from the models described in *Revised LOOP Frequency and PWR Seal LOCA Models*, ORNL/NRC/LTR-89/11, August 1989. The probability of failing to close breakers before battery depletion was assumed to be 0.10 (see Table XX of section XX of this report) and reflects the operators ability of performing the required non-routine actions in the required time from the control room. The probability of failing to recover power prior to battery depletion was revised to be 0.29 (0.10 non-recovery probability for closing the breakers + 0.19 probability of EDG1 failing given EDG 2 and the swing EDG were failed). To reflect the inoperability of HPCI, HPCI was set to failed, and the non-recovery probability for HPCI was set to 1.0 to reflect the likelihood that operators would not be able to recover HPCI within the allotted recovery time.

The second case examines the likelihood of the occurrence of a dual Unit LOOP from grid or weather-related LOOPs. In this case, the LOOP frequency was revised to 2.78×10^{-6} with a short-term non-recovery probability of 0.66, and off-site power recovery prior to battery depletion (EP REC) was modified to 0.21 to reflect values for grid and weather-related LOOPs determined from the models described in *Revised LOOP Frequency and PWR Seal LOCA Models*, ORNL/NRC/LTR-89/11, August 1989. Since both Units would need their designated EDGs, no credit is given for recovery using the breakers, and the probability of failing to recover power prior to battery depletion was left at 1.0. To reflect the inoperability of HPCI, HPCI was set to failed, and the non-recovery probability for HPCI was set to 1.0 to reflect the likelihood that operators would not be able to recover HPCI within the allotted recovery time.

The increase in core damage probability over the event duration for the first case is 3.6×10^{-6}. The dominant sequence involved a postulated plant-centered LOOP with the failure of emergency power, recovery of offsite power, the failure of HPCI and the failure of RCIC. The increase in core damage probability over the event duration for the second case is 5.1×10^{-6}. The dominant sequence involved a postulated grid/weather-related LOOP with the failure of emergency power and failure to recover off-site power prior to battery depletion.

2.0 Selection Criteria and Quantification

2.1 Accident Sequence Precursor Selection Criteria

The Accident Sequence Precursor (ASP) Program identifies and documents potentially important operational events that have involved portions of core damage sequences and quantifies the core damage probability associated with those sequences.

Identification of precursors requires the review of operational events for instances in which plant functions that provide protection against core damage have been challenged or compromised. Based on previous experience with reactor plant operational events, it is known that most operational events can be directly or indirectly associated with four initiators: trip [which includes loss of main feedwater (LOFW) within its sequences], loss-of-offsite power (LOOP), small-break loss-of-coolant accident (LOCA), and steam generator tube ruptures (SGTR) (PWRs only). These four initiators are primarily associated with loss of core cooling. ASP Program staff members examine licensee event reports (LERs) and other event documentation to determine the impact that operational events have on potential core damage sequences.

2.1.1 Precursors

This section describes the steps used to identify events for quantification. Figure 2.1 illustrates this process.

A computerized search of the SCSS data base at the Nuclear Operations Analysis Center (NOAC) of the Oak Ridge National Laboratory was conducted to identify LERs that met minimum selection criteria for precursors. This computerized search identified LERs potentially involving failures in plant systems that provide protective functions for the plant and those potentially involving core damage-related initiating events. Based on a review of the 1984–1987 precursor evaluations and all 1990 LERs, this computerized search successfully identifies almost all precursors and the resulting subset is approximately one-third to one-half of the total LERs. It should be noted, however, that the computerized search scheme has not been tested on the LER database for the years prior to 1984. Since the LER reporting requirements for 1982-83 were different than for 1984 and later, the possibility exists that some 1982-83 precursor events were not included in the selected subset. Events described in NUREG -0900[20] and in issues of *Nuclear Safety* that potentially impacted core damage sequences were also selected for review.

Those events selected for review by the computerized search of the SCSS data base underwent at least two independent reviews by different staff members. The independent reviews of each LER were performed to determine if the reported event should be examined in greater detail. This initial review was a bounding review, meant to capture events that in any way appeared to deserve detailed review and to eliminate events that were clearly unimportant. This process involved eliminating events that satisfied predefined criteria for rejection and accepting all others as either potentially significant and requiring analysis, or potentially significant but impractical to analyze. All events identified as impractical to analyze at any point in the study are documented in Appendix E. Events were also eliminated from further review if they had little impact on core damage sequences or provided little new information on the risk impacts of plant operation—for example, short-term single failures in redundant systems, uncomplicated reactor trips, and LOFW events.

ENCLOSURE 4

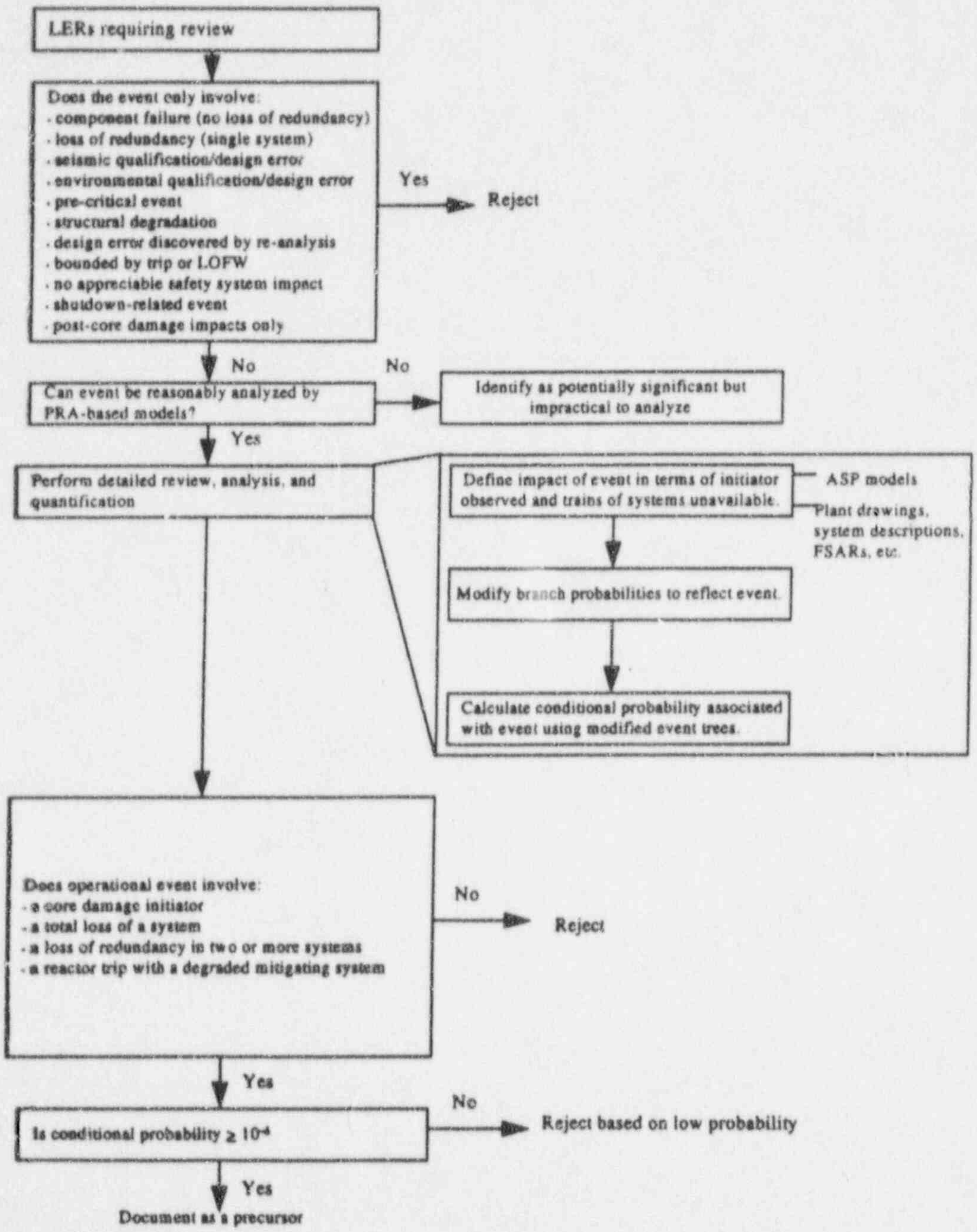


Figure 2.1 ASP Analysis Process

LERs were eliminated from further consideration as precursors if they involved, at most, only one of the following:

- a component failure with no loss of redundancy,
- a short-term loss of redundancy in only one system,
- a seismic design or qualification error,
- an environmental design or qualification error,
- a structural degradation,
- an event that occurred prior to initial criticality,
- a design error discovered by reanalysis,
- an event bounded by a reactor trip or LOFW,
- an event with no appreciable impact on safety systems, or
- an event involving only post core-damage impacts.

Events identified for further consideration typically included the following:

- unexpected core damage initiators (LOOP, SGTR, and small-break LOCA);
- all events in which a reactor trip was demanded and a safety-related component failed;
- all support system failures, including failures in cooling water systems, instrument air, instrumentation and control, and electric power systems;
- any event in which two or more failures occurred;
- any event or operating condition that was not predicted or that proceeded differently from the plant design basis; and
- any event that, based on the reviewers' experience, could have resulted in or significantly affected a chain of events leading to potential severe core damage.

Events determined to be potentially significant as a result of this initial review were then subjected to a thorough, detailed analysis. This extensive analysis was intended to identify those events considered to be precursors to potential severe core damage accidents, either because of an initiating event, or because of failures that could have affected the course of postulated off-normal events or accidents. These detailed reviews were not limited to the LERs; they also used final safety analysis reports (FSARs) and their amendments, individual plant examinations (IPEs), and other information related to the event of interest.

The detailed review of each event considered the immediate impact of an initiating event or the potential impact of the equipment failures or operator errors on readiness of systems in the plant for mitigation of off-normal and accident conditions. In the review of each selected event, three general scenarios (involving both the actual event and postulated additional failures) were considered.

1. If the event or failure was immediately detectable and occurred while the plant was at power, then the event was evaluated according to the likelihood that it and the ensuing plant response could lead to severe core damage.

2. If the event or failure had no immediate effect on plant operation (i.e., if no initiating event occurred), then the review considered whether the plant would require the failed items for mitigation of potential severe core damage sequences should a postulated initiating event occur during the failure period.

3. If the event or failure occurred while the plant was not at power, then the event was first assessed to determine whether it impacted at-power or hot shutdown operation. If the event could only occur at cold shutdown or refueling shutdown, or the conditions clearly did not impact at-power operation, then its impact on continued decay heat removal during shutdown was assessed; otherwise it was analyzed as if the plant were at power. (Although no cold shutdown events were analyzed in the present study, some potentially significant shutdown-related events are described in Appendix D).

For each actual occurrence or postulated initiating event associated with an operational event reported in an LER or multiple LERs, the sequence of operation of various mitigating systems required to prevent core damage was considered. Events were selected and documented as precursors to potential severe core damage accidents (accident sequence precursors) if the conditional probability of subsequent core damage was at least 1.0×10^{-6} (see section 2.2). Events of low significance are thus excluded, allowing attention to be focused on the more important events. This approach is consistent with the approach used to define 1988–1993 precursors, but differs from that of earlier ASP reports, which addressed all events meeting the precursor selection criteria regardless of conditional core damage probability.

As noted above, 115 operational events with conditional probabilities of subsequent severe core damage $\geq 1.0 \times 10^{-6}$ were identified as accident sequence precursors.

2.1.2 Potentially Significant Shutdown-Related Events

No cold shutdown events were analyzed in this study because the lack of information concerning plant status at the time of the event (e.g., systems unavailable, decay heat loads, RCS heat-up rates, etc.) prevented development of models for such events. However, cold shutdown events such as a prolonged loss of RHR cooling during conditions of high decay heat can be risk significant. Sixteen shutdown-related events which may have potential risk significance are described in Appendix D.

2.1.3 Potentially Significant Events Considered Impractical to Analyze

In some cases, events are impractical to analyze due to lack of information or inability to reasonably model within a probabilistic risk assessment (PRA) framework, considering the level of detail typically available in PRA models and the resources available to the ASP Program.

Forty-three events (some involving more than a single LER) identified as potentially significant were considered impractical to analyze. It is thought that such events are capable of impacting core damage sequences. However, the events usually involve component degradations in which the extent of the degradation could not be determined or the impact of the degradation on plant response could not be ascertained.

For many events classified as impractical to analyze, an assumption that the affected component or function was unavailable over a 1-year period (as would be done using a bounding analysis) would result in the conclusion that a very significant condition existed. This conclusion would not be supported by the specifics of the event as reported in the LER(s) or by the limited engineering evaluation performed in the ASP Program. Descriptions of events considered impractical to analyze are provided in Appendix E.

Selection Criteria and Quantification

2.1.4 Containment-Related Events

In addition to accident sequence precursors, events involving loss of containment functions, such as containment cooling, containment spray, containment isolation (direct paths to the environment only), or hydrogen control, identified in the reviews of 1982-83 LERs are documented in Appendix F. It should be noted that the SCSS search algorithm does not specifically search for containment related events. These events, if identified for other reasons during the search, are then examined and documented.

2.1.5 "Interesting" Events

Other events that provided insight into unusual failure modes with the potential to compromise continued core cooling but that were determined not to be precursors were also identified. These are documented as "interesting" events in Appendix G.

2.2 Precursor Quantification

Quantification of accident sequence precursor significance involves determination of a conditional probability of subsequent severe core damage, given the failures observed during an operational event. This is estimated by mapping failures observed during the event onto the ASP models, which depict potential paths to severe core damage, and calculating a conditional probability of core damage through the use of event trees and system models modified to reflect the event. The effect of a precursor on event tree branches is assessed by reviewing the operational event specifics against system design information. Quantification results in a revised probability of core damage failure, given the operational event. The conditional probability estimated for each precursor is useful in ranking because it provides an estimate of the measure of protection against core damage that remains once the observed failures have occurred. Details of the event modeling process and calculational results can be found in Appendix A of this report.

The frequencies and failure probabilities used in the calculations are derived in part from data obtained across the light-water reactor (LWR) population for the 1982-86 time period, even though they are applied to sequences that are plant-specific in nature. Because of this, the conditional probabilities determined for each precursor cannot be rigorously associated with the probability of severe core damage resulting from the actual event at the specific reactor plant at which it occurred. Appendix A documents the accident sequence models used in the 1982-83 precursor analyses, and provides examples of the probability values used in the calculations.

The evaluation of precursors in this report considered equipment and recovery procedures believed to have been available at the various plants in the 1982-83 time frame. This includes features addressed in the current (1994) ASP models that were not considered in the analysis of 1984-91 events, and only partially in the analysis of 1992-93 events. These features include the potential use of the residual heat removal system for long-term decay heat removal following a small-break LOCA in PWRs, the potential use of the reactor core isolation cooling system to supply makeup following a small-break LOCA in BWRs, and core damage sequences associated with failure to trip the reactor (this condition was previously designated "ATWS," and not developed). In addition, the potential long-term recovery of the power conversion system for BWR decay heat removal has been addressed in the models.

Because of these differences in the models, and the need to assume in the analysis of 1982-83 events that equipment reported as failed near the time of a reactor trip could have impacted post-trip response (equipment response following a reactor trip was required to be reported beginning in 1984), the evaluations for these years may not be directly comparable to the results for other years.

Another difference between earlier and the most recent (1994) precursor analyses involves the documentation of the significance of precursors involving unavailable equipment without initiating events. These events are termed unavailabilities in this report, but are also referred to as condition assessments. The 1994 analyses distinguish a precursor conditional core damage probability (CCDP), which addresses the risk impact of the failed equipment as well as all other nominally functioning equipment during the unavailability period, and an importance measure defined as the difference between the CCDP and the nominal core damage probability (CDP) over the same time period. This importance measure, which estimates the increase in core damage probability because of the failures, was referred to as the CCDP in pre-1994 reports, and was used to rank unavailabilities.

For most unavailabilities that meet the ASP selection criteria, observed failures significantly impact the core damage model. In these cases, there is little difference between the CCDP and the importance measure. For some events, however, nominal plant response dominates the risk. In these cases, the CCDP can be considerably higher than the importance measure. For 1994 unavailabilities, the CCDP, CDP, and importance are all provided to better characterize the significance of an event. This is facilitated by the computer code used to evaluate 1994 events (the GEM module in SAPHIRE), which reports these three values.

The analyses of 1982-83 events, however, were performed using the event evaluation code (EVENTEVL) used in the assessment of 1984-93 precursors. Because this code only reports the importance measure for unavailabilities, that value was used as a measure of event significance in this report. In the documentation of each unavailability, the importance measure value is referred to as the increase in core damage probability over the period of the unavailability, which is what it represents. An example of the difference between a conditional probability calculation and an importance calculation is provided in Appendix A.

2.3 Review of Precursor Documentation

With completion of the initial analyses of the precursors and reviews by team members, this draft report containing the analyses is being transmitted to an NRC contractor, Oak Ridge National Laboratories (ORNL), for an independent review. The review is intended to (1) provide an independent quality check of the analyses, (2) ensure consistency with the ASP analysis guidelines and with other ASP analyses for the same event type, and (3) verify the adequacy of the modeling approach and appropriateness of the assumptions used in the analyses. In addition, the draft report is being sent to the pertinent nuclear plant licensees for review and to the NRC staff for review. Comments received from the licensees within 30 days will be considered during resolution of comments received from ORNL and NRC staff.

2.4 Precursor Documentation Format

The 1982-83 precursors are documented in Appendices B and C. The at-power events with conditional core damage probabilities (CCDPs) $\geq 1.0 \times 10^{-5}$ are contained in Appendix B and those with CCDPs between 1.0×10^{-5} and 1.0×10^{-6} are summarized in Appendix C. For the events in Appendix B, a description of the event is provided with additional information relevant to the assessment of the event, the ASP modeling assumptions and approach used in the analysis, and analysis results. The conditional core damage probability calculations are documented and the documentation includes probability summaries for end states, the conditional probabilities for the more important sequences and the branch probabilities used. A figure indicating the dominant core damage sequence postulated for each event will be included in the final report. Copies of the LERs are not provided with this draft report.

2.5 Potential Sources of Error

As with any analytic procedure, the availability of information and modeling assumptions can bias results. In this section, several of these potential sources of error are addressed.

1. *Evaluation of only a subset of 1982-83 LERs.* For 1969–1981 and 1984–1987, all LERs reported during the year were evaluated for precursors. For 1988–1994 and for the present ASP study of 1982-83 events, only a subset of the LERs were evaluated after a computerized search of the SCSS data base. While this subset is thought to include most serious operational events, it is possible that some events that would normally be selected as precursors were missed because they were not included in the subset that resulted from the screening process. *Reports to Congress on Abnormal Occurrences*[20] (NUREG-0900 series) and operating experience articles in *Nuclear Safety* were also reviewed for events that may have been missed by the SCSS computerized screening.

2. *Inherent biases in the selection process.* Although the criteria for identification of an operational event as a precursor are fairly well-defined, the selection of an LER for initial review can be somewhat judgmental. Events selected in the study were more serious than most, so the majority of the LERs selected for detailed review would probably have been selected by other reviewers with experience in LWR systems and their operation. However, some differences would be expected to exist; thus, the selected set of precursors should not be considered unique.

3. *Lack of appropriate event information.* The accuracy and completeness of the LERs and other event-related documentation in reflecting pertinent operational information for the 1982-83 events are questionable in some cases. Requirements associated with LER reporting at the time, plus the approach to event reporting practiced at particular plants, could have resulted in variation in the extent of events reported and report details among plants. In addition, only details of the sequence (or partial sequences for failures discovered during testing) that actually occurred are usually provided; details concerning potential alternate sequences of interest in this study must often be inferred. Finally, the lack of a requirement at the time to link plant trip information to reportable events required that certain assumptions be made in the analysis of certain kinds of 1982-83 events. Specifically, through use of the "Grey Books" (*Licensed Operating Reactors Status Report*, NUREG-0200)[19] it was possible to determine that system unavailabilities reported in LERs could have overlapped with plant trips if it was assumed that the component could have been out-of-service for ½ the test/surveillance period associated with that component. However, with the link between trips and events not being described in the LERs, it was often impossible to determine whether or not the component was actually unavailable during the trip or whether it was demanded

during the trip. Nevertheless, in order to avoid missing any important precursors for the time period, any reported component unavailability which overlapped a plant trip within ½ of the component's test/surveillance period, and which was believed not to have been demanded during the trip, was assumed to be unavailable concurrent with the trip. (If the component had been demanded and failed, the failure would have been reported; if it had been demanded and worked successfully, then the failure would have occurred after the trip). Since such assumptions may be conservative, these events are distinguished from the other precursors listed in Tables 3.1 - 3.6. As noted above, these events are termed "windowed" events to indicate that they were analyzed because the potential time window for their unavailability was assumed to have overlapped a plant trip.

4. *Accuracy of the ASP models and probability data.* The event trees used in the analysis are plant-class specific and reflect differences between plants in the eight plant classes that have been defined. The system models are structured to reflect the plant-specific systems, at least to the train level. While major differences between plants are represented in this way, the plant models utilized in the analysis may not adequately reflect all important differences. Modeling improvements that address these problems are being pursued in the ASP Program.

   Because of the sparseness of system failure events, data from many plants must be combined to estimate the failure probability of a multitrain system or the frequency of low- and moderate-frequency events (such as LOOPs and small-break LOCAs). Because of this, the modeled response for each event will tend toward an average response for the plant class. If systems at the plant at which the event occurred are better or worse than average (difficult to ascertain without extensive operating experience), the actual conditional probability for an event could be higher or lower than that calculated in the analysis.

   Known plant-specific equipment and procedures that can provide additional protection against core damage beyond the plant-class features included in the ASP event tree models were addressed in the 1982-83 precursor analysis for some plants. This information was not uniformly available; much of it was based on FSAR and IPE documentation available at the time this report was prepared. As a result, consideration of additional features may not be consistent in precursor analyses of events at different plants. However, analyses of multiple events that occurred at an individual plant or at similar units at the same site have been consistently analyzed.

5. *Difficulty in determining the potential for recovery of failed equipment.* Assignment of recovery credit for an event can have a significant impact on the assessment of the event. The approach used to assign recovery credit is described in detail in Appendix A. The actual likelihood of failing to recover from an event at a particular plant during 1982-83 is difficult to assess and may vary substantially from the values currently used in the ASP analyses. This difficulty is demonstrated in the genuine differences in opinion among analysts, operations and maintenance personnel, and others, concerning the likelihood of recovering from specific failures (typically observed during testing) within a time period that would prevent core damage following an actual initiating event.

6. Assumption of a 1-month test interval. The core damage probability for precursors involving

unavailabilities is calculated on the basis of the exposure time associated with the event. For failures discovered during testing, the time period is related to the test interval. A test interval of 1 month was assumed unless another interval was specified in the LER. See reference 1 for a more comprehensive discussion of test interval assumptions.

Appendix A:
ASP MODELS

A.0 ASP Models

This appendix describes the methods and models used to estimate the significance of 1982-83 precursors. The modeling approach is similar to that used to evaluate 1984-91 operational events. Simplified train-based models are used, in conjunction with a simplified recovery model, to estimate system failure probabilities specific to an operational event. These probabilities are then used in event tree models that describe core damage sequences relevant to the event. The event trees have been expanded beyond those used in the analysis of 1984-91 events to address features of the ASP models used to assess 1994 operational events (Ref. 1) known to have existed in the 1982-83 time period.

A.1 Precursor Significance Estimation

The ASP program performs retrospective analyses of operating experience. These analyses require that certain methodological assumptions be made in order to estimate the risk significance of an event. If one assumes, following an operational event in which core cooling was successful, that components observed failed were "failed" with probability 1.0, and components that functioned successfully were "successful" with probability 1.0, then one can conclude that the risk of core damage was zero, and that the only potential sequence was the combination of events that occurred. In order to avoid such trivial results, the status of certain components must be considered latent. In the ASP program, this latency is associated with components that operated successfully—these components are considered to have been capable of failing during the operational event.

Quantification of precursor significance involves the determination of a conditional probability of subsequent core damage given the failures and other undesirable conditions (such as an initiating event or an unexpected relief valve challenge) observed during an operational event. The effect of a precursor on systems addressed in the core damage models is assessed by reviewing the operational event specifics against plant design and operating information, and translating the results of the review into a revised model for the plant that reflects the observed failures. The precursors's significance is estimated by calculating a conditional probability of core damage given the observed failures. The conditional probability calculated in this way is useful in ranking because it provides an estimate of the measure of protection against core damage remaining once the observed failures have occurred.

A.1.1 Types of Events Analyzed

Two different types of events are addressed in precursor quantitative analysis. In the first, an initiating event such as a loss of offsite power (LOOP) or small-break loss of coolant accident (LOCA) occurs as a part of the precursor. The probability of core damage for this type of event is calculated based on the required plant response to the particular initiating event and other failures that may have occurred at the same time. This type of event includes the "windowed" events subsetted for the 1982-83 ASP program and discussed in Section 2.2 of the main report.

The second type of event involves a failure condition that existed over a period of time during which an initiating event could have, but did not occur. The probability of core damage is calculated based on the required plant response to a set of postulated initiating events, considering the failures that were observed. Unlike an initiating event assessment, where a particular initiating event is assumed to occur with probability 1.0, each initiating event is assumed to occur with a probability based on the initiating event frequency and the failure duration.

A.1.2 Modification of System Failure Probabilities to Reflect Observed Failures

The ASP models used to evaluate 1982-83 operational events describe sequences to core damage in terms of combinations of mitigating systems success and failure following an initiating event. Each system model represents those combinations of train or component failures that will result in system failure. Failures observed during an operational event must be represented in terms of changes to one or more of the potential failures included in the system models.

If a failed component is included in one of the trains in the system model, the failure is reflected by setting the probability for the impacted train to 1.0. Redundant train failure probabilities are conditional, which allows potential common cause failures to be addressed. If the observed failure could have occurred in other similar components at the same time, then the system failure probability is increased to represent this. If the failure could not simultaneously occur in other components (for example, if a component was removed from service for preventive maintenance), then the system failure probability is also revised, but only to reflect the "removal" of the unavailable component from the model.

If a failed component is not specifically included as an event in a model, then the failure is addressed by setting elements impacted by the failure to failed. For example, support systems are not completely developed in the 1982-83 ASP models. A breaker failure that results in the loss of power to a group of components would be represented by setting the elements associated with each component in the group to failed.

Occasionally, a precursor occurs that cannot be modelled by modifying probabilities in existing system models. In such a case, the model is revised as necessary to address the event, typically by adding events to the system model or by addressing an unusual initiating event through the use of an additional event tree.

A.1.3 Recovery from Observed Failures

The models used to evaluated 1982-83 events address the potential for recovery of an entire system if the system fails. This is the same approach that was used in the analysis of most precursors through 1991.[1] In this approach, the potential for recovery is addressed by assigning a recovery action to each system failure and initiating event. Four classes were used to describe the different types of short-term recovery that could be involved:

[1] Later precursor analyses utilize Time-Reliability Correlations to estimate the probability of failing to recover a failed system when recovery is dominated by operator action.

Recovery Class	Likelihood of Non-Recovery[3]	Recovery Characteristic
R1	1.00	The failure did not appear to be recoverable in the required period, either from the control room or at the failed equipment.
R2	0.55	The failure appeared recoverable in the required period at the failed equipment, and the equipment was accessible; recovery from the control room did not appear possible.
R3	0.10	The failure appeared recoverable in the required period from the control room, but recovery was not routine or involved substantial operator burden.
R4	0.01	The failure appeared recoverable in the required period from the control room and was considered routine and procedurally based.

The assignment of an event to a recovery class is based on engineering judgment, which considers the specifics of each operational event and the likelihood of not recovering from the observed failure in a moderate to high-stress situation following an initiating event.

Substantial time is usually available to recover a failed residual heat removal (RHR) or BWR power conversion system (PCS). For these systems, the nonrecovery probabilities listed above are overly conservative. Data in Refs. 2 and 3 was used to estimate the following nonrecovery probabilities for these systems:

System	p(nonrecovery)
BWR RHR system	0.016 (0.054 if failures involve service water)
BWR PCS	0.52 (0.017 for MSIV closure)
PWR RHR system	0.057

It must be noted that the actual likelihood of failing to recover from an event at a particular plant is difficult to assess and may vary substantially from the values listed. This difficulty is demonstrated in the genuine differences in opinion among analysts, operations and maintenance personnel, etc., concerning the likelihood of recovering specific failures (typically observed during testing) within a time period that would prevent core damage following an actual initiating event.

A.1-4 Conditional Probability Associated with Each Precursor

As described earlier in this appendix, the calculation process for each precursor involves a determination of initiators that must be modeled, plus any modifications to system probabilities necessitated by failures observed

[3]These nonrecovery probabilities are consistent with values specified in M.B. Sattison *et al.*, "Methods Improvements Incorporated into the SAPHIRE ASP Models," *Proceedings of the U.S. Nuclear Regulatory Commission Twenty-Second Water Reactor Safety Information Meeting*, NUREG/CP-0140, Vol. 1, April 1995.

in an operational event. Once the probabilities that reflect the conditions of the precursor are established, the sequences leading to core damage are calculated to estimate the conditional probability for the precursor. This calculational process is summarized in Table A.1.

Several simplified examples that illustrate the basics of precursor calculational process follow. It is not the intent of the examples to describe a detailed precursor analysis, but instead to provide a basic understanding of the process.

The hypothetical core damage model for these examples, shown in Fig. A.1, consists of initiator I and four systems that provide protection against core damage: system A, B, C, and D. In Fig. A.1, the up branch represents success and the down branch failure for each of the systems. Three sequences result in core damage if completed: sequence 3 [I /A ("/" represents system success) B C], sequence 6 (I A /B C D) and sequence 7 (I A B). In a conventional PRA approach, the frequency of core damage would be calculated using the frequency of the initiating event I, $\lambda(I)$, and the failure probabilities for A, B, C, and D [p(A), p(B), p(C), and p(D)]. Assuming $\lambda(I) = 0.1\ yr^{-1}$ and $p(A|I) = 0.003$, $p(B|IA) = 0.01$, $p(C|I) = 0.05$, and $p(D|IC) = 0.1$,[3] the frequency of core damage is determined by calculating the frequency of each of the three core damage sequences and adding the frequencies:

$$0.1\ yr^{-1} \times (1 - 0.003) \times 0.05 \times 0.1 \text{ (sequence 3)} +$$
$$0.1\ yr^{-1} \times 0.003 \times (1 - 0.01) \times 0.05 \times 0.1 \text{ (sequence 6)} +$$
$$0.1\ yr^{-1} \times 0.003 \times 0.01 \text{ (sequence 7)}$$

$$= 4.99 \times 10^{-4}\ yr^{-1} \text{ (sequence 3)} + 1.49 \times 10^{-6}\ yr^{-1} \text{ (sequence 6)} + 3.00 \times 10^{-6}\ yr^{-1} \text{ (sequence 7)}$$

$$= 5.03 \times 10^{-4}\ yr^{-1}.$$

In a nominal PRA, sequence 3 would be the dominant core damage sequence.

The ASP program calculates a conditional probability of core damage, given an initiating event or component failures. This probability is different than the frequency calculated above and cannot be directly compared with it.

Example 1. Initiating Event Assessment. Assume that a precursor involving initiating event I occurs. In response to I, systems A, B, and C start and operate correctly and system D is not demanded. In a precursor initiating event assessment, the probability of I is set to 1.0. Although systems A, B, and C were successful, nominal failure probabilities are assumed. Since system D was not demanded, a nominal failure probability is assumed for it as well. The conditional probability of core damage associated with precursor I is calculated by summing the conditional probabilities for the three sequences:

$$1.0 \times (1 - 0.003) \times 0.05 \times 0.1 \text{ (sequence 3)} +$$
$$1.0 \times 0.003 \times (1 - 0.010) \times 0.05 \times 0.1 \text{ (sequence 6)} +$$
$$1.0 \times 0.003 \times 0.01 \text{ (sequence 7)}$$

[3] The notation $p(B|IA)$ means the probability that B fails, given I occurred and A failed.

$$= 5.03 \times 10^{-3}.$$

If, instead, B had failed when demanded, its probability would have been set to 1.0. The conditional core damage probability for precursor IB would be calculated as

$$1.0 \times (1 - 0.003) \times 0.05 \times 0.1 \text{ (sequence 3)} + 1.0 \times 0.003 \times 1.0 \text{ (sequence 7)} = 7.99 \times 10^{-3}.$$

Since B is failed sequence 6 cannot occur.

Example 2. Condition Assessment. Assume that during a monthly test system B is found to be failed, and that the failure could have occurred at any time during the month. The best estimate for the duration of the failure is one half of the test period, or 360 h. To estimate the probability of initiating event I during the 360 h period, the yearly frequency of I must be converted to an hourly rate. If I can only occur at power, and the plant is at power for 70% of a year, then the frequency for I is estimated to be $0.1\ yr^{-1}/(8760\ h/yr \times 0.7) = 1.63 \times 10^{-5}\ h^{-1}$.

If, as in example 1, B is always demanded following I, the probability of I in the 360 h period is the probability that at least one I occurs (since the failure of B will then be discovered), or

$$1 - e^{-\lambda(I) \times \text{failure duration}} = 1 - e^{-1.63E-5 \times 360} = 5.85 \times 10^{-3}.$$

Using this value for the probability of I, and setting p(B) = 1.0, the conditional probability of core damage for precursor B is calculated by again summing the conditional probabilities for the core damage sequences in Fig. A.1:

$$5.85 \times 10^{-3} \times (1 - 0.003) \times 0.05 \times 0.1 \text{ (sequence 3)} + 5.85 \times 10^{-3} \times 0.003 \times 1.0 \text{ (sequence 7)}$$

$$= 4.67 \times 10^{-5}.$$

As before, since B is failed, sequence 6 cannot occur. The conditional probability is the probability of core damage in the 360 h period, given the failure of B. Note that the dominant core damage sequence is sequence 3, with a conditional probability of 2.92×10^{-5}. This sequence is unrelated to the failure of B. The potential failure of systems C and D over the 360 h period still drive the core damage risk.

To understand the significance of the failure of system B, another calculation, an importance measure, is required. The importance measure that is used is equivalent to risk achievement worth on an interval scale (see Ref. 4). In this calculation, the increase in core damage probability over the 360 h period due to the failure of B is estimated: p(cd | B) - p(cd). For this example the value is $4.67 \times 10^{-5} - 2.94 \times 10^{5} = 1.73 \times 10^{5}$, where the second term on the left side of the equation is calculated using the previously developed probability of I in the 360 h period and nominal failure probabilities for A, B, C, and D.

For most conditions identified as precursors in the ASP program, the importance and the conditional core damage probability are numerically close, and either can be used as a significance measure for the precursor. However, for some events—typically those in which the components that are failed are not the primary mitigating plant features—the conditional core damage probability can be significantly higher than the importance. In such cases, it is important to note that the potential failure of other components, unrelated to the precursor, are still dominating the plant risk.

ASP MODELS

The importance measure for unavailabilities (condition assessments) like this example event were previously referred to as a "conditional core damage probability" in annual precursor reports before 1994, instead of as the increase in core damage probability over the duration of the unavailability. Because the computer code used to analyze 1982-83 events is the same as was used for 1984-93 evaluations, the results for 1982-83 conditions are also presented in the computer output in terms of "conditional probability," when in actuality the result is an importance.

A.2 Overview of 1982-83 ASP Models

Models used to rank 1982-83 precursors as to significance consist of system-based plant-class event trees and simplified plant-specific system models. These models describe mitigation sequences for the following initiating events: a nonspecific reactor trip [which includes loss of feedwater (LOFW) within the model], LOOP, small-break LOCA, and steam generator tube rupture [SGTR, pressurized water reactors (PWRs) only].

Plant classes were defined based on the use of similar systems in providing protective functions in response to transients, LOOPs, and small-break LOCAs. System designs and specific nomenclature may differ among plants included in a particular class; but functionally, they are similar in response. Plants where certain mitigating systems do not exist, but which are largely analogous in their initiator response, are grouped into the appropriate plant class. ASP plant categorization is described in the following section.

The event trees consider two end states: success (OK), in which core cooling exists, and core damage (CD), in which adequate core cooling is believed not to exist. In the ASP models, core damage is assumed to occur following core uncovery. It is acknowledged that clad and fuel damage will occur at later times, depending on the criteria used to define "damage," and that time may be available to recover core cooling once core uncovery occurs but before the onset of core damage. However, this potential recovery is not addressed in the models. Each event tree describes combinations of system failures that will prevent core cooling, and makeup if required, in both the short and long term. Primary systems designed to provide these functions and alternate systems capable of also performing these functions are addressed.

The models used to evaluate 1982-83 events consider both additional systems that can provide core protection and initiating events not included in the plant-class models used in the assessment of 1984-91 events, and only partially included in the assessment of 1992-93 events. Response to a failure to trip the reactor is now addressed, as is an SGTR in PWRs. In PWRs, the potential use of the residual heat removal system following a small-break LOCA (to avoid sump recirculation) is addressed, as is the potential recovery of secondary-side cooling in the long term following the initiation of feed and bleed. In boiling water reactors (BWRs), the potential use of reactor core isolation cooling (RCIC) and the control rod drive (CRD) system for makeup if a single relief valve sticks open is addressed, as is the potential long-term recovery of the power conversion system (PCS) for decay heat removal in BWRs. These models better reflect the capabilities of plant systems in preventing core damage.