NUREG/CR-4674 ORNL/NOAC-232 Vol. 6

Precursors to Potential Severe Core Damage Accidents: 1986 A Status Report

Appendixes D, E, and F

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Prepared for U.S. Nuclear Regulatory Commission

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NOTE

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This document is bound in two volumes: Volume 5 contains the main report and Appendixes A, B, and C; Volume 6 contains Appendixes D, E, and F.

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LIST OF ACRONYMS

.

AFW	auxiliary feedwater
AFWS	auxiliary feedwater system
ATWS	anticipated transient without scram
BWR	boiling-water reactor
CC	component cooling
CCW	core cooling water
CRT	condensate return tank
CSR	condensate storage tank
CSS	core spray system
CST	condensate storage tank
CVCS	chemical and volume control system
DG	diesel generator
DHR	decay heat removal
ECC	emergency core cooling
ECCS	emergency core cooling system
ECCW	emergency condenser cooling water
EDG	emergency diesel generator
EFW	emergency feedwater
EPS	emergency power system
FSAR	final safety analysis report
HPCI	high-pressure cooling injection
HPI	high-pressure injection
HVAC	heating, ventilation, and air conditioning
LER	licensee event report
LOCA	loss-of-coolant accident
LOFW	loss of main feedwater
LOOP	loss of offsite power
L.PC1	low-pressure coolant injection
LFCS	low-pressure core spray
LPI	low-pressure injection
MFW	main feedwater
MTWP	main feedwater pump
MSIV	main steam isolation valve
MSRV	main steam relief valve
NRC	Nuclear Regulatory Commission
PORV	power- or pilot-operated relief valve
PWR	pressurized-water reactor
RCIC	reactor core isolation cooling
RCF	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
RHRS	residual heat removal system
RPS	reactor protection system
SDC	shutdown cooling
SG	steam generator
SI	safety injection
SLB	steam-line break
SRV	safety relief valve
SS	secondary-side

APPENDIX D

PRECURSOR DOCUMENTATION

APPENDIX D

PRECURSOR DOCUMENTATION

Reactor plant operational events for 1986 were selected for documentation as precursors to potential severe core damage based on the selection criteria described in this report. These events are documented here.

For each precursor, a precursor description sheet and a conditional core-damage calculation are included. The precursor description sheet briefly describes the event sequence, provides plant and event data pertinent to the evaluation, and documents the modeling considerations and decisions made. Included with the conditional core-damage calculations are individual sequences probabilities for the more significant core-damage (CD), core-vulnerability (CV), and anticipated-transient-without-scram (ATWS) sequences; identification of dominant sequences for each end state; and a listing of the branch probabilities and frequencies utilized. Individual sequences were listed if their probability was >0.03 times the probability of the dominant sequence for each end state.

Probabilities for sequences that reflect a decrease in conditional probability are enclosed in parentheses. A decrease in core-damage, core-vulnerability, or ATWS conditional probability for an individual sequence can occur in sequences containing success branches when an unavailability is modeled. For example, consider two sequences involving an initiator A, an observed degraded system B, plus another C. Sequence 1 includes success of B; sequence 2 includes failure of B.

A occurs, B succeeds, C fails (sequence 1) A occurs, B fails (sequence 2)

The probability of sequence 1 is probability $(A) \times [1 - failure probability (B)] \times failure probability (C); the probability of sequence 2 is probability (A) × failure probability (B).$

In assessing the significance of an unavailable system, the likelihood of core damage calculated without any observed failures and over the same period of time is subtracted from the value calculated considering the unavailable system so as to estimate only the additional impact of the unavailability. Applying this procedure to the above sequences with the likelihood of initiator A assumed to be 0.1, the likelihood of B failing (given that it has been degraded) assumed to be 0.5, the likelihood of C failing assumed to be 0.03, the probability of sequences 1 and 2, respectively, is calculated as follows:

$$[0.1 \times (1 - 0.5) \times 0.03] - [0.1 \times (1 - 0.01)$$
(1)

 \times 0.03] = -1.47 \times 10⁻³

$$[0, 1 \times 0, 5] - [0, 1 \times 0, 01] = 4.9 \times 10^{-2}$$
⁽²⁾

In this case, the differential probability for sequence 1 is negative, indicating a decrease in probability for the sequence compared with the same time period without the unavailability.

Each event is identified by its unique docket-LER number. Table D.1 provides an index to the documentation for each precursor and an index to conditional core damage calculations performed for a postulated nonspecific reactor trip and a postulated LOFW a the different BWR and PWR plant classes defined in this report. These calculations are included following the documentation of precursors. The LERs associated with each precursor event are included in Appendix E. Table E.1 in Appendix E provides an index to the corresponding LERs.

LER No.	Event title	Plant name	Page number
247/86-017	Open condenser dump valves cause trip, and one safeguards train	Indian Point 2	D-6
diana -	fails to start	Indian Point 2	D-11
47/86-035	Trip, LOFW, and two AFW train failures occur		
49/86-013	HPC1 and one train of the core spray and LPC1 systems are inoperable	Dresden 3	D-16
50/86-036	Unavailability of DGs	Turkey Point Units 3 and 4	D-21
250/86-038	AFW system is unavailable	Turkey Point Units 3 and 4	D-26
250/86-039	Trip occurs with stuck-open PORV	Turkey Point 3	D-31
261/86-005	Bue failure causes a trip followed by a LOOP with a DG unavailability	Robinson 2	D-36
269/86-001	TRIP, LOFW, and a stuck-open MSRV occur	Oconee 1	D-42
269/86-011	Emergency condenser cooling system is unavailable	Oconee Station Units 1,2, and 3	D-47
277/86-003	DG trip in test causes scram	Peach Bottom 2	D-52
280/86-029	Charging pump service-water gumps are unavailable	Surry 1	D-57
280/86-031	Righ-head injection system is unavailable	Surry 1	D-65
281/86-010	High-head injection system is unavailable	Surry 2	D-67
282/86-006	Emergency power system is unavailabile	Prairie Island Units 1 and 2	D-72
282/86-011	Emergency power system is unavailabile	Prairie Island Units 1 and 2	D-77
285/86-001	Trip occurs, and automatic depressurization and turbine bypass system fails to open	Ft. Calhoun	D-82
293/80-027	LOOP occurs due to winter storm	Pilgrim 1	D-87
301/86-004	MSIVs fail to close on demand	Point Beach 2	0-92
318/86-006	Trip occurs, and one atmospheric dump valve fails to close	Calvert Cliffs Unit 2	D-96
341/86-048	RCIC and HPCI are unavailable	Fermi 2	0-10
362/86-011	Saltwater and CCW systems are unavailable	San Onofre 3	D-10
366/86-035	LPCS system is unavailable	Hatch 2	D-11
370/86-006	High-head injection system is unavailable and DG A is out	McGuire 2	D-11
389/86-011	of service Emergency power system is	St. Lucie	D-12
409/86-023	unavailable LOOP occurs due to lightning	LaCrosse	D-12
	strike at coal-fired unit Small LOCA forces plant trip	Catawba 1	D-13
413/86-031 414/86-028	SG PORVS open inadvertently in test, and trip when other failures occur	Catawba 2	D-13
458/86-002	Hand-held ratio causes LOOP	River Bend 1	D-14
458/86-002 458/86-047	Emergency power, LPCS, RHR train A, and RCIC systems are degraded twice	River Bend 1	D-14

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Table D.1. Index to precursor descriptions and conditional core-damage calculations

PRECURSOR DESCRIPTION SHEET

LER No.:	247/86-017
Event Description:	Open condenser dump valves cause trip, and one safeguards train fails to start
Date of Event: Plant:	May 28, 1986 Indian Point 2

EVENT DESCRIPTION

Sequence

Unit 2 was operating at 30% power. The condenser steam dump control system was switched from the temperature mode to the pressure mode at 1455 h because of erratic behavior observed on temperature controller TC-412J. At 1556 h. all 12 condenser steam dump valves received an open signal as a result of faulty steam dump controller PC-404. This resulted in an increased steam flow and a reduction in reactor coolant temperature and a subsequent SI actuation.

SI train A actuated, resulting in a reactor trip and safeguards actuation; however, train B did not actuate. SI train A signal resulted in closure of the MSIVs ~2.5 s after the reactor trip, effectively ending the high steam-flow condition.

The required functions that did not fully actuate because train B did not function were containment isolation phase A, train B, and some of the required redundant valving required for SI.

SI train B was successfully actuated at 1607 h, when the control room operators reset SI. Resetting SI consists of manually locking in another SI signal and depressing SI reset buttons. This action actuates parallel contacts in both trains of the SI logic. Because train B had not been actuated by the first SI signal, the incroduction of the second signal initiated a separate SI sequence. SI equipment was stripped and automatically restarted; all required redundant valves (trains A and B) then operated normally.

Corrective Action

The steam dump control system repairs were as follows.

- 1. All electrolytic capacitors and the auto/manual relays were replaced on condenser steam dump controller PC-404, and the controller was recalibrated.
- Current-to-pneumatic converter PM-404 was replaced with a new unit and calibrated.
- Temperature controller TC-412J was replaced with a new unit and calibrated.
- 4. Pressure transmitter PT-404 was calibrated.

Corrective action for the SI system actuation circuit was as follows.

 Relays SIA-2, SL-2, TR-2, and TR-2X were replaced. The SI actuation log-in was tested and verified operational.

Plant/Event Data

Systems Involved: Turbine bypass and HPI

Components and Failure Modes Involved: Twelve dump valves failed to open during operation HPI, one train failed to autostart

Component Unavailability Duration: NA Plant Operating Mode: 1 (30% power) Discovery Method: Operational event Reactor Age: 13.0 years Plant Type: PWR

Comments

The event was modeled using a steam line break event tree, consistent with similar events identified in the ASP Program. Because of the limited information in the LEK, HPI redundancy was assumed lost. The model assumed local operator action would be required to close the dump valves had the MSIVs failed to close. If this was not the case (i.e., the dump valves could have been closed from the room) then the core damage probability estimate would have been lower by a factor of at least three.

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

SLB 0.34 Steam line isolated on MSIV closure

Branches Impacted and Branch Nonrecovery Estimate

0.04

HPI

One train failed (recoverable from control room)

Plant Models Utilized

PWR plant Class F

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 247/86-017 Event Description: Open Condenser Dump Valve Causes Trip and ESF Train Fails Event Date: 5/28/86

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

SLB		3.4E-01
SEQUENCE	CONDITIONAL PROBABILITY SUMS	
End	State/Initiator	Probability
CD		

SL B	1.0E-04
Total	1.0E-04
ATWS	
SLB	1.0E-05
Total	1.0E-05

```
DOMINANT SEQUENCES
```

End State: CD	Conditional	Probability	9.1E-05
101 SLB -RT -REQ. 56. ISO -AFW -HP1	PORV. OPEN. DUE	TO.HP1 POR	V.CLOSURE HPR/-HPI
and an electron and	Conditional	Prehability	1.05-05

End State: ATWS Conditional Probability: 1.02-0;

112 SLB RT

SEQUENCE CONDITIONAL PROBABILITIES

	Sequence	End State	Prob	N Rect+
101	SLB -RT -REQ.SG.ISO -AFW -HPI PORV.OPEN.DUE.TO.HPI PORV.CLOS	CD	9,18-05 +	1.9E-01
102 10 4	URE HPR/-HPI SLB -RT -REQ.SG.ICO AFW -HPI(F/B) -PORV.OPEN HPR/-HPI SLB -RT -REQ.SG.ISO AFW HPI(F/B)	CD CD	5.0E-06 4.2E-06	5.1E-02 4.8E-02

112 SL& RT

ATWS 1.0E-05 + 4.1E-02

* dominant sequence for end state
** non-recovery credit for edited case

MODEL: c:\asp\newmodel\pwrbmslb.txt DATA:

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fai
SLB	1.1E-07 > 1.0E+00 +++	1.0E+00 > 3.4E-01	
Branch Model: INITOR		1146.44 / 2146-01	
Initiator Freq:	1.1E-07		
ŔŢ	2.5E-04	1.2E-01	
REQ. 56, 150	6.4E-04	1.0E+00	
AFW	1.0E-03	2.7E-01	
HP1	1.0E-03 > 1.0E-02	5,28-01	
Branch Model: 1.0F.2			
Train 1 Cond Prob:	1.02-02		
Train 2 Cund Prob:	1.0E-01 > Unavailable		
HPI(F/B)	1.0E-03 > 1.0E-02	5.2E-01	4.0E-02
Branch Model: 1.GF.2+opr			4.06-02
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
HFR/-HPI	3.0E-03 > 3.0E-02	3.6E-01	4.0E-02
Branch Hodel: 1.05.2+opr			4.05-02
Train 1 Cond Prob:	3.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
PORV. OPEN	1.0E-02	1,0E+00	
REP. BA. ADDITION	8.3E-04	1.05+00	
PORV. OPEN. DUE. TO. HPI	8.0E-01	1.0€+00	
PORV, CLOSURE	6.0E-03	1.0E+00	

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Austin 09-11-1987 13:46:40

PRECURSOR DESCRIPTION SHEET

LER No.: 247/ Event Description: Trip Date of Event: Octo Plant: Indi

247/86-035 Trip, LOFW, and two AFW train failures occur October 20, 1986 Indian Point 2

EVENT DESCRIPTION

Sequence

At 0936 h the Unit 2 reactor tripped from 100% power when reactor trip breaker B unexpectedly opened because of loose wires in the relay racks. Breakers RT3 and 4 were deenergized. One of the reactor protection relays had also been deenergized while a monthly SI surveillance test was being performed in a nearby equipment rack.

Following the trip, SG levels dropped rapidly as expected. Both motor-driven auxiliary feed pumps started on low-low SG level. While following the emergency recovery procedure, a control room operator discovered that auxiliary feed pump 21 had tripped when its breaker tripped for an unknown reason. The pump was then successfully restarted from the control room. AFW was used to maintain the SG water levels.

The steam-driven auxiliary-feed-pump steam relief valve had also popped open following the plant trip when its steam-pressure control valve opened because its set point was out of calibration on the low side. The steam-pressure control valve received an automatic open signal on low-low steam generator level in two of the four SGs, admitting steam up to the turbine governor valve. The auxiliary-feed-pump speed changer setting was at minimum as designed, but response by the pressure control valve was too slow, which caused the relief valve to lift.

Corrective Action

Repairs were made.

Plant/Event Data

Systems Involved: AFW, MFW

Components and Failure Modes Involved: Two trains of AFW failed in operation

Component Unavailability Duration: NA Plant Operating Mode: 1 (100% power) Discovery Method: Operational event Reactor Age: 13.4 years Plant Type: PWR

Comments

The FSAR states that upon SI actual signa¹, the MFW system will isolate. An SI actual signal actuation is not reported to have occurred. Upon RPS actuation, the MFW regulating valves should fully open, yet the LER states that the AFW pumps started as a result of low SG level and were used to maintain SG level. Therefore, MFW is assumed to have tripped, even though the LER does not appear to say so.

The event was modeled assuming that the turbine-driven AFW pump was made unavailable when its steam side relief valve opened during pump start. This, in combination with the also unavailable motor-driven train produced the calculated core damage probability estimate. It is possible that the turbine-driven AFW pump was available for steam generator cooling; if this was the case, the core damage probability estimate would be lower by a factor of ~10. However, the LER does state that steam generator inventory was maintained by one of the motor-driven AFW pumps and does nor clarify that the turbine-driven pump would have been available if required.

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Transient

Base case nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

AFW

MFW

Base case Two of three trains failed on demand; one train was nonrecoverable Base case Assumed failed in operation

Plant Models Utilized

PWR plant Class F

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 247/86-035 Event Description: Trip, LOFW, and Two AFW Train Failures Event Date: 10/20/86 Plant: Indian Point 2

INITIATING EVENT

KON-RECOVERABLE INITIATING EVENT PROBABILITIES 1.0E+00 TRANS SEQUENCE CONDITIONAL PROBABILITY SUMS Probability End State/Initiator CV 5,15-04 TRANS 5.1E-04 Total CD 2.9E-04 TRANS 2.9E-04 Total ATHS 3.4E-05 TRANS 3.4E-05 Total DOMINANT SEQUENCES Conditional Probability: 2.3E-04 End State: CV 125 TRANS -RT AFW MFW HPI(F/B) -SS.DEPRESS -COND/MFW End State: CD Conditional Probability: 1.2E-04

126 TRANS -KT AFM MF# HPI(F/B) -SS.DEPRESS COND/NFW

End State: ATWS Conditional Probability: 3.4E-05

128 TRANS RT

SEQUENCE CONDITIONAL PROBABILITIES

	Sequence	End State	Prob	N Rect+
119	IRANS -RT AFW MFW -HPI(F/B) -HPR/-HPI PORV.OPEN -SS.DEPRESS -COND/MFW	CV	5.2E-05	5.8E-02
120	TRANS -RT AFW MFW -HPI(F/B) -HPR/-HPI PORV.OPEN -SS.DEPRESS COND/MFW	CD	2.7E-05	3.0E-02
122 123 124 125 126 127 128	TRANS -RT AFW MFW -HP!(F/B) HPR/-HPI SS.DEPRESS -COND/MFW TRANS -RT AFW MFW -HPI(F/B) HPR/-HPI SS.DEPRESS COND/MFJ TRANS -RT AFW MFW -HPI(F/B) HPR/-HPI SS.DEPRESS TRANS -RT AFW MFW HPI(F/B) -SS.DEPRESS -COND/MFW TRANS -RT AFW MFW HPI(F/B) -SS.DEPRESS COND/MFW TRANS -RT AFW MFW HPI(F/B) SS.DEPRESS TRANS RT	CV CD CD CV CD CD CD ATWS	2.2E-04 1.1E-04 1.2E-05 2.7E-04 * 1.27-04 * 1.37-05 3.4E-05 *	5.8E-02 3.0E-02 8.8E-02 4.9E-02 2.5E-02 7.4E-02 1.2E-01

dominant sequence for end state ## non-recovery credit for edited case

SEQUENCE MODEL:	c:\aspineweodel\pwrbtree.cmp
BRANCH MODEL:	c:\asp\newmodel\indpsint.txt
PROBABILIT: FILE:	<pre>c:\asp\newmrdel\pwr_b.pro</pre>

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	4.8E-04	1.0E+00	
LOOP	4.6E-06	3.9E-01	
LOCA	2.4E-08	4.32-01	
RT	2.8E-04	1.2E-01	
RT/LOOP	0.0E+00	1.0E+00	
EMERG. POWER	2.98-03		
AFW	3.8E-04 > 1.0E-01	8.0E-01	
Branch Model: 1.0F.3+ser	0.05-04 / 1.05-01	2.6E-01	
	5 AF 45 1 Fuller		
Train 1 Cond Prob:	2.0E-02 > Failed		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	5.0E-02 > Failed		
Serial Component Prob:	2.85-04		
AFW/EMERG. POWER	5.0E-02 > 1.0E+00	3.4E-01	
Branch Model: 1.0F.1			
Train 1 Cond Prob:	5.0E-02 > Failed		
MFW	2.0E-01 > 1.0E+00	3.45-01	
Branch Model: 1.0F.1			
Train 1 Cond Probi	2.0E-01 > Failer		

PORV. OR, SRV. CHALL	4.0E-02	1.0E+00	
PORV. OR. SRV. RESEAT	2.0E-02	5.0E-02	
PORV, OR. SRV. RESEAT/EMERS. POWER	2.0E-02	1.0E+00	
SS. RELEAS. TERM	1.5E-02	3.4E-01	
SS. RELEAS. TERM /-MFW	1.5E-02	3.4E-01	
HP1	1.5E-03	8.4E-01	
HP1(F/B)	1.5E-03	8.4E-01	4.0E-02
HPR/-HPI	1.5E-04	1.0E+00	4.0E-02
PORV. OPEN	1.0E-02	1.0E+00	
SS. DEPRESS	3.6E-02	1.0E+00	
COND/MFW	1.0E+00	3.4E-01	
LP1/HP1	1.5E-04	3.4E-01	
LPR/-HPI.HPR	6.7E-01	1.0E+00	
LPR/HP1	1.5E-04	1.0E+00	

branch model file

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Austin 09-11-1987 14:20:03

PRECURSOR DESCRIPTION SHEET

LER No .: 249/86-013

Event Description: HPCI and one train of the core spray and LPCI systems are inoperable August 27, 1986 Dresden 3

EVENT DESCRIPTION

Date of Event:

Sequence

Plant:

Dresden 3 was in the run mode at 19% power with the HPCI system declared inoperable for repairs (reason not stated). At 0030 h during surveillance testing, the train B CSS full-flow-test valve (3-1042-4B) was discovered to be damaged, so the valve would not close; the B core spray subsystem was also unpressurized. In addition, the LFCI system minimum-flow valve (3-1501-13A) showed a double position indication the valve was in midposition. The 2/3 DG failed to close manually onto bus 33-1; however, the generator was able to be synchronized manually to bus 23-1 without incident. In the event of a LOCA, the DG would have closed automatically on bus 33-1. A unit shutdown was begun.

Investigation revealed that valve 3-1042-4B (the "B" pump CSS fullflow-test valve) had a fractured motor-operator housing. The torque switch failed and allowed the motor to drive the valve disk into the valve seat until the motor housing was fractured. The torque switch was incorrectly installed in the reverse direction.

Investigation revealed that the handwheel retaining-ring was disengaged and resting atop the handwheel bearing of the Limitorque motoroperator for the LPCI system minimum-flow valve (3-1501-13A). The valve was opened manually.

Investigation revealed that DG 2/3 failed to close onto bus 33-1 because a terminal block screw was loose in junction box 3TB-187. Cold shutdown was achieved at 2007 h.

Corrective Action

The torque switch on valve 3-1042-4B (CSS full-flow-test valve) was installed correctly, and the motor housing was replaced. The handwheel retaining ring for the LPCI system minimum-flow valve (3-1501-13A) was correctly installed. The loose terminal block screw in the DG 2/3 junction box 3TB-187 was tightened.

Plant/Event Data

Systems Involved: LFCI, core spray, emergency power, and HPCI

Components and Failure Modes Involved: Pump B CSS full-flow-test valve — failed to close in test LPCI system minimum-flow valve — failed in midposition in test DG 2/3 — failed to close onto bus 33-1 in manual mode operation in test HPCI — inoperable (reason not stated)

Component Unavailability Duration: 15 d Plant Operating Mode: 1 (19% power) Discovery Method: Testing Reactor Age: 15.6 years Plant Type: BWR

Comments

Dresden station has three DGs. Each unit has one dedicated DG and the third is a swing DG (2/3) between both. One train of each unit's ECCS is supported by DG 2/3. Failure of DG 2/3 would prevent emergency power to one ECCS train. The SDC system is independent of LPCI so it was not affected.

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Postulated transient Postulated LOOP Postulated LOCA Base case nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

HPCI	1.0	Out of service and assumed unavailable
LPCS	Base case	Assumed one of two trains fails in
		test
LPCI	Base case	Assumed one of two trains fails in
		test

Plant Models Utilized

BWR plant Class B

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 249/86-013 Event Description: HPCI and One Train of LPCS and LPCI Are Inoperable Event Date: 8/27/86 Plant: Dresden 3

UNAVAILABILITY, DURATION= 360

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	3.1E-01
LOOP	2.0E-03
LOCA	5.9E-04

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator		Probability
CV		
TRANS LOOP LOCA		1.4E-06 6.1E-07 2.5E-09
Total		2.0E-06
CD		
TRANS LOOP LOCA		6.8E-07 1.7E-06 3.4E-07
Total		2.7E-06
ATWS		
LOOP LOCA		0.0E+00 0.0E+00 0.0E+00
Total		0.0E+00
DOMINANT SEQUENCES		
End State: CV	Conditional Probability:	9.2E-07

130 TRANS SCRAM -SLC.OR.RODS PCS/TRANS FW/PCS.TRANS HPCI -SRV.ADS -COND/FW.PCS -SDC

End State: CD Conditional Probability: 1.3E-06

213 LOOP -EMERG. POWER -SCRAM SRV. CHALL/LOOP. -SCRAM SRV. CLOSE HPCI SRV. ADS

SEQUENCE CONDITIONAL PROBABILITIES

Sequence	End State	Prob	N Rec**
109 TRANS -SCRAM PCS/TRANS SRV.CHALL/TRANSSCRAM -SRV.CLOSE IS OL.COND FW/PCS.TRANS HPC1 CRD SRV.ADS	CD	1.8E-07	2.4E-01
117 TRANS -SCRAM PCS/TRANS SRV.CHALL/TRANSSCRAM SRV.CLOSE FW /PCS.LDCA HPCI SRV.ADS	CD	4.28-07	2.4E-01
130 TRANS SCRAM -SLC.OR.RODS PCS/TRANS FW/PCS.TRANS HPCI -SRV. ADS -COND/FW.PCS -SDC	CV	9.28-07 *	2.2E-01
134 TRANS SCRAM -SLC.DC., RODS PCS/TRANS FW/PCS, TRANS HPCI -SRV. ADS COND/FW, PCS -LPCS -SDC	CV	4.6E-07	1.12-01
147 TRANS SCRAM -SLC.OR.RODS PCS/TRANS FW/PCS.TRANS HPCI SRV.	CD	6.2E-0E	2.4E-01
207 LOOP -EMERG.POWER -SCRAM SRV.CHALL/LOOPSCRAM -SRV.CLOSE IS OL.COND HPCI CRD SRV.ADS	CD	7.98-08	2.3E-01
212 LOOP -EMERG.POWER -SCRAM SRV.CHALL/LOOPSCRAM SRV.CLOSE HP CI -SRV.ADS LPCS LPCI FIREWTR.OR.OTHER/LPCS.LPCI/LOOP	CD	7.4E-08	7.7E-02
213 LOOP -EMERG.POWER -SCRAM SRV.CHALL/LOOPSCRAM SRV.CLOSE HP CI SRV.ADS	CD	1.3E-06 *	2.3E-01
222 LOOP -EMERG.FOWER SCRAM -SLC.OR.RODS HPCI -SRV.ADS -LPCS -SD	CV	5.9E-07	3.1E-01
238 LOOP EMERG.POWER -SCRAM SRV.CHALL/LOOPSCRAM -SRV.CLOSE IS OL.COND HFCI	CD	8.65-08	2.6E-01
240 LOOP EMERG.POWER -SCRAM SRV.CHALL/LOOPSCRAM SRV.CLOSE HP CI	CD	7.2E-08	2.68-01
309 LOCA -SCRAM PCS/LOCA FW/PCS,LOCA HPCI SRV.ADS	CD	3,4E-07	1.2E-01

* dominant sequence for end state

** 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:\asp\newmode)	\bwrbtree.cmp
BRANCH MODEL :	c:\asp\newmodel	\dresden.txt
PROBABILITY FILE:	c:\asp\newmode!	bwr_c.pro

No Recovery Limit

BRANCH FREDUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	8.6E-04	1.0E+00	
LOOP	1.7E-05	3.2E-01	
LOCA	3.3E-06	5.0E-01	
SCRAM	3.5E-04	1.0E+00	
SLC.OR.RODS	1.0E-02	1.0E+00	4.0E-02
PCS/TRANS	1.7E-01	1.0E+00	TIVE VE
PCS/LOCA	1.0E+00	1.0E+00	
SRV.CHALL/TRANSSCRAM	1.0E+00	1.0E+00	
SRV.CHALL/TRANS.SCRAM	1.0E+00	1.0E+00	
SRV.CHALL/LOOP SCRAM	1.0E+00	1.0E+00	
SRV.CHALL/LOOP.SCRAM	1.0E+00	1.0E+00	
SRV. CLOSE	1.6E-02	1.0E+00	
EMERG . POWER	2.9E-03	8.0E-01	
FW/PCS.TRANS	2.9E-01	3.4E-01	
FW/PCS.LOCA	4.0E-02	3.4E-01	
HPCI	2.9E-02 > 1.0E+CO	7.0E-01 > 1.0E+00	
Branch Model: 1.0F.1		TIVE VE C LIVE VV	
Train 1 Cond Prob:	2.9E-02 > Unavailable		
ISOL.COND	2.0E-02	1.0E+00	
CRD	1.0E-02	1,0E+00	4.0E-02
SRV.ADS	3.7E-03	7.1E-01	4.0E-02
COND/FW.PCS	1.0E+00	3.4E-01	TIVE VE
LPCS	2.0E-03 > 1.0E-01	3.4E-01	
Branch Model: 1.0F.2			
Train 1 Cond Prob:	2.0E-02 > Failed		
Train 2 Cond Prob:	1.0E-01		
LPCI	1.0E-03 > 1.0E-01	7.1E-01	
Branch Model: 1.0F.2			
Train 1 Cond Frob:	1.0E-02 > Failed		
Train 2 Cond Prob:	1.0E-01		
FIREWTR.OR.OTHER/LPCS.LPCI/TRA	1.0E+00	1.0E+00	
FIREWTR.OR.OTHER/LPCS.LPC1/LOO	1.0E+00	1.0E+00	
FIREWTR .OR .OTHER/LPCS .LPCI/LOC	1.0E+00	1.0E+00	
SDC	2.9E-03	3.4E-01	
LPCI (CC)	1.0E-03	3.4E-01	
LPCI (CC) /LPCI	1.0E+00	1.0E+00	
C.I.AND.V/LPCI	1.0E+00	3.4E-01	
* branch model file			
** forced			

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PRECURSOR DESCRIPTION SHEET

LER No.: 250/86-036 Event Description: Unavailability of DGs Date of Event: November 6, 1986 Plant: Turkey Point 3 and 4

EVENT DESCRIPTION

Sequence

The DG B was out of service for testing and instrument calibration. The DG A was in testing when the discovery was made that it would not shut off. It was removed from service for repairs. The DG B was restored to service in 1.5 h. Because Units 3 and 4 both share the two DGs, both units were affected.

Corrective Action

DG B was restored to service in 1.5 h. DG "A" was repaired.

Plant/Event Data

Systems Involved: Emergency power

Components and Failure Modes Involved: DG B — was out for maintenance DG A — failed in test

Component Unavailability Duration: 1.5 h Plant Operating Mode: 1 (100% power) Discovery Method: Testing Reactor Age: 14.1 and 13.4 years, respectively Plant Type: PWR

Comments

Because the entire station has only two DGs, both units were affected by this unavailability.

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Postulated LOOP Base case nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

Postulated EPS 0.34

Recoverable locally at equipment; 0.34 assumed because DG A did start but failed to stop and DG B was only out for testing

Plant Models Utilized

PWR plant Class E

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Dates 1	50/86-036 naveilability of Diesel Generators 1/6/86 urkey Point 3	
UNAVAILABILITY, DURA	110N= 1.5	
NON-RECOVERABLE INIT	IATING EVENT PROBABILITIES	
LOCP		2.7E-06
SEQUENCE CONDITIONAL	PROBABILITY SUMS	
End State/Initia	ator	Probability
CV		
LOOP		4,62-09
Total		4.6E-09
CD		
LOOP		1.12-09
Total		1.1E-09
ATWS		
LOOP		0.0E+00
Total		0.0E+00
DOMINANT SEQUENCES		
End State: CV	Conditional Probability:	4.52-09
217 LOOP -RT/LOOP	EMERG. POWER -AFW/EMERG. POWER -PORV. OR. SRV. CH	ALL SS.RELEAS.TERM
End State: CD	Conditional Probability:	7.3E-10
216 LOOP -RT/LOOP POWER	EMERB.POWER -AFW/EMERB.POWER PORV.OR.SRV.CH	ALL PORV.OR.SRV.RESEAT/EMERS.
SEQUENCE CONDITIONAL	PROBABILITIES	
Event Identifier: 250	1/86~036	

	Sequence	End State	Prob	N Rec++
215	LOOP -RT/LOOP EMERS.POWER -AFW/EMERS.POWER PORV.OR.SRV.CHALL -PORV.OR.SRV.RESEAT/EMERS.POWER SS.RELEAS.TERM	CV	1.8E-10	4.5E-02
216	LOOP -RT/LOOP EMERS.POWER -AFW/EMERS.POWER PORV.OR.SRV.CHALL PORV.OR.SRV.RESEAT/EMERS.POWER	CD	7.3E-10 •	1.3E-01
217	LOOP -RT/LOOP EMERG.POWER -AFW/EMERG.POWER -PORV.OR.SRV.CHALL SS.RELEAS.TERM	CV	4.58-09 •	4.58-02
218	LOOP -RT/LOOP EMERG.POWER AFW/EMERG.POWER	CD	3.7E-10	3.6E-02

dominant sequence for end state ## 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:\asp\newmode]\pwrbtree.cmp
BRANCH MODEL:	c:\asp\newmodel\turkey.txt
PROBABILITY FILE:	c:\asp\newsodel\pwr_b.pro

No Recovery Limit

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BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	4.8E-04	1.0E+00	
LOOP	4.6E-06	3.9E-01	
LOCA	2.4E-06	4.3E-01	
RT	2.8E-04	1.2E-01	
RT/LOOP	0.0E+00	1.0E+00	
EMERG. POWER	2.9E-03 > 1.0E+00	8.0E-01 > 3.4E-01	
Branch Model: 1.0F.2			
Train 1 Cond Prob:	5.0E-02 > Failed		
Train 2 Cond Prob:	5.7E-02 > Unavailable		
AFW	1.5E-03	2.7E-01	
AFW/EMERG, POWER	1.5E-03	2.7E-01	
MSW	1.9E-01	3.4E-01	
PORV. OR. SRV. CHALL	4.0E-02	1.0E+00	
PORV. OR. SRV. RESEAT	2.0E-02	5.0E-02	
PORV. OR. SRV. RESEAT / EMERG. POWER	2.0E-02	1.0E+00	
SS.RELEAS. TERM	1.58-02	3.4E-01	
SS. RELEAS, TERM/-MFW	1.5E-02		
HP]		3.46-01	
NP1(F/B)	3.0E-04	8.46-01	1.12.12
	3.0E-04	8.4E-01	4.0E-02
HPR/-HP1	1.5E-04	1.0E+00	4.0E-02
PORV. OPEN	1.0E-02	1.07+00	

SS. DEPRESS	3.6E-02	1.0E+00
COND/MFW	1.0E+00	3.4E-01
LP1/HP1	1.5E-04	3.4E-01
LPR/-HP1.HPR	6.7E-01	1.0E+00
LPR/HP1	1.5E-04	1.0E+00

* branch model file
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Event Identifier: 250/86-036

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PRECURSOR DESCRIPTION SHEET

LER No.: Event Description: Date of Event: Plant:

250/86-038 System is unavailable AFW December 4, 1986 Turkey Points 3 and 4

EVENT DESCRIPTION

Sequence

During routine testing, AFW pump B of AFW train 2 unexpectedly tripped off on an overspeed trip (the set point had drifted). Because the station AFW consists of three turbine-driven pumps (pumps B and C on train 2 and pump A on train 1), pump C was placed in service. Personnel then discovered pump C's steam supply valve (MOV-3-1403) failed to open. Train A was available but could not service both Units 3 and 4.

After trains B and C were restored to service, unit 4 AFW motor valves were inspected. The alignment of the valves rendered one train inoperable.

Corrective Action

The AFW pump B trip set point was adjusted. The valve MOV-3-1403 motor was replaced.

Plant/Event Data

Systems Involved: AFW

Components and Failure Modes Involved: Train B — failed in testing Train A — failed in testing

Component Unavailability Duration: 360 h Plant Operating Mode: 1 (100% power) Discovery Method: Testing Reactor Age: 14.2 (Unit 3) and 13.5 years (Unit 4), respectively Plant Type: PWR

Comments

1

Both units were affected by this event; however, this unavailability is a problem only if both MFW systems fail simultaneously, as in a station LOOP event. In a LOOP only one unit could be provided with sufficient AFW.

Event Identifier. 250/86-038

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Postulated LOOP Base case nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

1.0

AFW

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Recovery impossible without repair and local troubleshooting

Plant Models Utilized PWR plant Class E

Event Identifier: 250/86-038

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

6.5E-04

0.0E+00

Event Identifier: 250/86-038 Event Description: Unavailability of Auxiliary Feedwater Event Date: 12/4/86 Plant: Turkey Point 3

UNAVAILABILITY, DURATION= 360

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator Probability
CV
LOOP (7.9E-09)
Total (7.9E-09)
CD

LOOP 5.8E-05 Total 5.8E-05 ATWS LOOP 0.0E+00 Total

DOMINANT SEQUENCES

End State: CD Conditional Probability: 2.6E-05

214 LOOP -RT/LOOP -EMERS. POWER AFM HPI(F/B)

SEQUENCE CONDITIONAL PROBABILITIES

Sequence				End State	Prob	N Rec++		
212 213	LOOP -RT/LOOP	-EMERS. POWER	AFN -HPI(F/B) AFN -HPI(F/B)	-HPR/-HP1 HPR/-HP1	PORV. OPEN	CD CD	5.9E-06 2.5E-05	3.9E-01 3.9E-01

214	1000 - 97/1009	-ENERS, POWER	AFW HPI(F/B)	CD	2.6E-05 +	3.3E-01	
				CD.	1.5E-06	3.1E-01	
218	LOOP -RT/LOOP	EMERG. POWER	AFW/EMERB. POWER		1.05.00	V116 V1	

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tominant sequence for end state to 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:\asp\newmodel\pwrbtree.cmp SRANCH MODEL: c:\asp\newmodel\turkey.txt PROBABILITY FILE: c:\asp\newmodel\pwr_b.pro

No Recovery Limit

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BRANCH FREQUENCIES/PROBABILITIES

System	Non-Recov	Opr Fail
4.85-04	1.0E+00	
4.68-06	3.9E-01	
2.4E-06	4.3E-01	
35.25.05.	1.2E-01	
	1.0E+00	
2.9E-03	8.0E-01	
	2.7E-01 > 1.0E+00	
5.0E-02 > Unavailable		
1.5E-03 > 1.0E+00	2.7E-01 > 1.0E+00	
5.0E-02 > Unavailable		
1.0E-01 > Unavailable		
3.0E-01 > Unavailable		
1.9E-01	3.4E-01	
4.0E-02	1.0E+00	
2.0E-02	5.0E-02	
2.0E-02	1.0E+00	
1.5E-02	3.4E-01	
1.5E-02	3.4E-01	
3.0E-04	8.4E-01	
3.0E-04	8.4E-01	4.0E-02
1.5E-04	1.0E+00	4.0E-02
1.0E-02	1.0E+00	
3.6E-02	1.0E+00	
1.0E+00	3.4E-01	
1.5E-04	3.4E-01	
	4.8E-04 4.6E-06 2.4E-06 2.8E-04 0.0E+00 2.9E-03 1.5E-03 > 1.0E+00 5.9E-02 > Unavailable 1.0E-01 > Unavailable 1.5E-03 > 1.0E+00 5.0E-02 > Unavailable 1.0E-01 > Unavailable 1.0E-01 > Unavailable 1.9E-01 > Unavailable 1.9E-01 > Unavailable 1.9E-01 4.0E-02 2.0E-02 1.5E-02 1.5E-02 1.5E-02 3.0E-04 3.0E-04 1.0E-02 3.6E-02 1.0E+00	4.8E-04 1.0E+00 4.6E-06 3.9E-01 2.4E-06 4.3E-01 2.8E-04 1.2E-01 0.0E+00 1.0E+00 2.9E-03 8.0E-01 1.5E-03 > 1.0E+00 2.7E-01 > 1.0E+00 5.0E-02 > Unavailable 1.0E+00 2.0E-01 > Unavailable 1.0E+00 1.9E-01 = Unavailable 3.4E-01 1.9E-02 = 1.0E+00 2.0E-02 2.0E-02 = 1.0E+00 3.4E-01 1.5E-02 = 3.4E-01 3.0E-04 8.4E-01 3.0E-04 9.4E-01 3.0E+00 1.0E+00 1.0E+00 1.0E+00 3.4E-01 1.0E+00 1.0E+00

LPR/-HPI.HPR LPR/HP1	6.7E-01	1.0E+00	
	1.56-04	1.0E+00	

* branch model file of forced

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PRECURSOR DESCRIPTION SHEET

LER No.: 250/86-039 Event Description: Trip occurs with stuck-open PORV Date of Event: December 27, 1986 Plant: Turkey Point 3

EVENT DESCRIPTION

Sequence

Unit 3 was tripped manually following a loss of turbine governor oil system pressure and a subsequent rapid electrical load decrease from 730 to 0 MW(e). No automatic control rod insertion occurred. The reactor control operator, noting that the coolant temperature was increasing above the reference temperature, placed the rods under manual control, and initiated rod insertion. Concurrently, a second reactor control operator attempted to raise the oil pressure, unsuccessfully. At this time (~24 s into the transient) it became clear that the unit could not be recovered, and the unit was tripped manually. During the transient, a PORV opened but then would not fully close, necessitating closure of the associated block valve. The unit was stabilized in <5 min. The most probable cause of the drop in oil pressure was the clearing of blockage of the governor impeller orifice, resulting in the auxiliary governor dumping control oil. The control rods failed to insert automatically because of two cold solder joints in the final variable gain summator of the power mismatch circuit. The cause of the PORV failure to close was under investigation. The PORV, turbine governor impeller, and associated components were inspected; and no problems were found. The cold solder joints were repaired. The control, lube, and seal oil piping were to be cleaned.

Corrective Action

The PORV block valve was closed.

Plant/Event Data

Systems Involved: Pressurizer relief

Components and Failure Modes Involved: PORV - failed to open in operation

Component Unavailability Duration: NA Plant Operating Mode: 1 (100% power) Discovery Method: operational event Reactor Age: 14.1 years Plant Type: PWR

Comments

Manual control rod insertion occurred before any trip signal actuations.

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Transient 1.0 No recovery

Branches Impacted and Branch Nonrecovery Estimate

PORV/SRV reseat Base case Recoverable from the control room by closing the block valve

Plant Models Utilized

Plant Class E

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

1.0E+00

Probability

6.5E-04

6.5E-04

1.4E-03

1.4E-03

Event Identifier: 250/86-039 Event Description: Trip and Stuck Open PORV Event Date: 12/27/86 Plant: Turkey Point 3

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator

CV

TRANS

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CD

TRANS

Total

ATWS

TRANS 3.4E-05 Total 3.4E-05

DOMINANT SEQUENCES

- End State: CV Conditional Probability: 6.4E-04
- 102 TRAMS -RT -AFW PORV.OR.SRV.CHALL PORV.OR.SRV.RESEAT -HPI HPR/-HPI -SS.DEPRESS -LPR/-HP 1.HPR

End State: CD Conditional Probability: 1.3E-03

103 TRANS -RT -AFW PORV.OR.SRV.CHALL PORV.OR.SRV.RESEAT -HP1 HPR/-HP1 -SS.DEPRESS LPR/-HP 1.HPR

End State: ATWS Conditional Probability: 3.4E-05

128 TRANS RT

SEQUENCE CONDITIONAL PROBABILITIES

	Sequence			End State	Prob	N Rr.++
102	TRANS -RT -AFN PORV.OR.SRV.CHALL R/-HPI -SS.DEPRESS -LPR/-HPI.HPR	PORV.OR.SRV.RESEAT -HPI	КP	CV	6.48-04 +	5.0E-02
103		PORV.OR.SRV.RESEAT -HPI	HP	CD	1.3E-03 *	5.0E-02
104		PORV. OR. SRV. RESEAT -HPI	HP	00	7.28-05	5.0E-02
128	TRANS RT			ATWS	3.4E-05 *	1.28-01

+ dominant sequence for end state ## non-recovery credit for edited case

SEQUENCE MODEL:	c:\asp\newsode]\pwritree.cmp
BRANCH MODEL:	c:\asp\newmodel\turkey.txt
PROBABILITY FILES	c:\asp\newmodel\pwr_b.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	4.8E-04	1.0E+00	
LOOP	4.65-06	3.98-01	
LOCA	2.48-06		
RT	2.8E-04	4.3E-01	
RT/LOOP		1.2E-01	
EMERG. POWER	0.0E+00	1.0E+00	
AFN	2.9E-03	8.0E-01	
	1.5E-03	2.7E-01	
AFW/ENERB. POWER	1.5E-03	2.7E-01	
MFW	1.9E-01	3.4E-01	
PORV. DR. SRV. CHALL	4.0E-02 > 1.0E+00 **	1.0E+00	
Branch Model: 1.0F.1			
Train 1 Cond Probi	4.0E-02		
PORV. OR. SRV. RESEAT	2.0E-02 > 1.0E+00	5.0E-02	
Branch Model: 1.0F.1		0.06-02	
Train 1 Cond Prob:	2.0E-02 > Failed		
PORV. OR. SRV. RESEAT/EMERS. POWER	2.0E-02	1.0E+00	
SS. RELEAS. TERM	1.5E-02	3.4E-01	
SS. RELEAS. TERM/-MFW	1.58-02		
HP1	3.0E-04	3.4E-01	
HP1(F/B)		8.4E-01	
	3.0E-04	8.4E-01	4.0E-02

HPR/-HP1	1.5E-04	1.0E+00
PORV. OPEN	1.0E-02	1.0E+00
	3.68-02	1.0E+00
SS. DEPRESS	1.0F+00	3.4E-01
COND/MFW	1.56-04	3.4E-01
LP1/HP1	4.72-01	1.0E+00
LPR/-HP1.HPR	1.58-04	1.0E+00
LPR/HP1	1.25-04	

* branch model file
** forced

Austin 09-11-1987 11:16:50 4.0E-02

PRECURSOR DESCRIPTION SHEET

LER No.: Event Description:	261/86-005 Bus failure causes DG unavailability	a trip	followed	by a	LOOP	with	a
Date of Event: Plant:	January 28, 1986 Robinson 2						

EVENT DESCRIPTION

Sequence

The plant was operating at ~80% power. EDG B had just been taken out of service to install a solid state overcurrent trip device on its output breaker. This breaker upgrade was being performed on all Westinghouse type DB safety-related breakers and had been completed on EDG A the week before. At 0917 h, the EDG B output breaker had just been "racked out" when emergency bus E-2 was lost as a result of a blown fuse. This also resulted in the loss of instrument bus 4 (IB-4), which is supplied by motor control center MCC-6. Nuclear instrumentation system power range channel N-44 (fed from IB-4) was lost, which initiated a turbine runback. The automatic-rod-control and steam-dumpcontrol systems would not function properly. As a result, a reactor trip was received on "Hi Pressurizer Pressure" ~21 s after bus E-2 was lost.

One minute after the reactor trip, the main generator oil circuit breakers opened, and the plant auxiliaries (those powered by the auxiliary transformer during operation) shifted to the startup transformer as part of the normal turbine generator lockout feature. Approximately 1 s later, a west bus lockout occurred in the 115-kV switchyard; this deenergized the Unit 2 startup transformer, resulting in a loss of offsite ac power. EDG A started automatically and loaded emergency bus E-1. Approximately 67 s after the west bus lockout was received, an SI and MSIV signal were received. These were caused by high steam-line flow coincident with low Tave. The low Tave signal was caused by the plant cooldown as a result of the reactor trip. The high steam-line flow signal was present due to loss of bus IB-4. During the attempt to restore bus E-2, an operator accidentally disabled HPI train B.

At 1027 h power was restored to bus E-2 by manually starting and loading the B EDG.

At 1115 h after investigation, offsite ac power was restored to the plant's nonvital electrical distribution system.

At 1228 h, a second SI signal was received. It was caused by steam-line high differential pressure, which resulted when frozen sensing lines caused "C" SG's PORV to stick open. The "C" PORV was closed by isolating the air supply to the PORV.

Corrective Action

The investigations concluded that two major events (loss of emergency bus E-2 and the loss of offsite ac power) were separate and independent from one another. An extensive investigation of the EDG B output breaker, bus E-2 control cabinet, associated circuits, and wiring was performed. No unusual conditions were found that would have caused the blown fuse. Later, while in the process of energizing E-2 via E-1 (cross tie E-1 and E-2), degraded voltage relay actuation caused the E-2 normal supply breaker to trip open.

Plant/Event Data

Systems Involved: AFW, emergency power, HPI/recirculation, and LPI/ recirculation

Components and Failure Modes Involved: Diesel generator — unavailable due to maintenance

Component Unavailability Duration: NA Plant Operating Mode: 1 (80% power) Discovery Method: Operational event Reactor Age: 15.4 years Plant Type: PWR

Comments

None

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nontacovery Estimate

LOOP Base case Normal recovery assumed

Branches Impacted and Branch Nonrecovery Estimate

EPS HPI/HPR PI/LPR	Base case Base case Base case	DG B out of service for repairs Train B disabled by error Train B unavailable because DG B was
AFW	Base case	unavailable Motor train B unavailable because DG B
SS release terminated	Base case	was unavailable "C" PORV required local action to isolate the valve

Plant Models Utilized

PWR plant Class B

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

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Event Identifier:	261/86-005	1.41.
Event Description:	Bus Failure Causes Trip and LOOP with D6 Unavai	lable
Event Date:	1/28/66	
Plants	Robinson 2	
INITIATING EVENT		
NON-RECOVERABLE IN	ITTATING EVENT PROBABILITIES	
		3.9E-01
LOOP		
SEQUENCE CONDITION	AL PROBABILITY SUMS	
SERVENCE CONFILTER		
End State/Ini	tistor	Probability
CV		
LOOP		5.3E-03
Total		5.3E-03
CD		
LOOP		3.0E-04
Total		3.0E-04
ATWS		
LOOP		0.0E+00
Total		0.0E+00
DOMINANT SEQUENCE		
DOUTHHUL DEBOUND		
End State: CV	Conditional Probability:	5.0E-03
	THE REPORT FOR A DUP A DOU NO COU P	
217 LOOP -RT/LO	DOP EMERG.POWER -AFM/EMERG.POWER -PORV.DR.SRV.C	MALL SPINCLEMPITERN
End State: CD	Conditional Probability:	2.7E-04
218 LOOP -RT/LO	DOP EMERS. POWER AFW/EMERG. POWER	
SEQUENCE CONDITII	DWAL PROBABILITIES	

4

D

Event Identifier: 261/86-005

D-39

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	Sequence	End Sta	te Prob N Reces
215 LOOP -RT -PORV.OR	/LOOP EMERG. POWER - AFW/EMERG. POWER I .SRV.RESEAT/EMERG. POWER SS. RELEAS. TEM	PORV. OR. SRV. CHALL CV	2.0E-04 1.0E-01
216 LOOP -RT	/LOOP EMERS. POWER -AFW/EMERS. POWER F SRV. RESEAT/EMERS. POWER	PORV. OR. SRV. CHALL CD	1.82-05 3.12-01
	LOOP EMERS. POWER -AFW/EMERS. POWER -F	PORV. DR. SAV. CHALL CV	5.0E-03 + 1.0E-01
	LOOP EMERS. POWER AFW/EMERS. POWER	CD	2.7E-04 + 1.1E-01
** non-recovery SEQUENCE MODEL:	and the second of the second second		
BRANCH MODEL: PROBABILITY FIL	C:\asp\newmodel\robinson.txt E: C:\asp\newmodel\pwr_b.pro		
No Recovery Lis	it		
BRANCH FREQUENC	IES/PROBABILITIES		
Branch	System	Non-Recov	Opr Fail
TRANS LOOP LOCA RT	4.8E-04 4.6E-06 2.4E-06 2.8E-04	1.0€+00 3.9E-01 4.3E-01	
RT/LOOP	0.06+00	1.28-01	

1.0E+00 0,05+00 EMERG. POWER 2.9E-03 > 5.0E-02 8.0E-01 Branch Models 1.0F.2 Train 1 Cond Prob: 5.0E-02 Train 2 Cond Prob: 5.7E-02 > Unavailable AFM 3.8E-04 > 1.3E-03 2.6E-01 Branch Model: 1.DF.3+ser Train 1 Cond Prob: 2.0E-02 Train 2 Cond Probi 1.0E-01 > Unavailable Train 3 Cond Prob: 5.0E-02 Serial Component Prob: 2.8E-04 AFW/EMERG. POWER 5.0E-02 3.4E-01 RFW 2.0E-01 3.4E-01 PORV. DR. SRV. CHALL 4.0E-02 1.0E+00 PORV. OR. SRV. RESEAT 3.0E-02 5.0E-02 FORV. OR. SRV. RESEAT / EMERG. POWER 3.08-02 1.0E+00 SS. RELEAS. TERM 1.5E-02 > 1.0E+00 3.4E-01 Branch Model: 1.0F.1 Train 1 Cond Prob: 1.5E-02 > Failed SS. RELEAS. TERM / - MP 3 1.5E-02 > 1.0E+00 3.4E-01 Branch Model: 1.0F.1 Train 1 Cond Prob: 1.5E-02 > Failed

HP1	1.0E-03 > 1.0E-02	8.4E-01	
Branch Model: 1.0F.2			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Probi	1.0E-01 > Unavailable		· · · · · · · · ·
HP1(F/B)	1.0E-03 > 1.0E-02	8.4E-01	4.0E-02
Branch Model: 1.0F.2+opr			
Train 1 Cond Probi	1.0E-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
HPR/-HPI	1.5E-04	1.0E+00	4.0E-02
PORV. OPEN	1.0E-02	1.0E+00	
SS. DEPRESS	3.68-02	1.0E+00	
	1.0E+00	3.4E-01	
COND/HFM	1.5E-04 > 1.0E-02	3.4E-01	
LP1/HP1			
Branch Models 1.0F.2	1.0E-02		
Train 1 Cond Probi	1.5E-02 > Unavailable		
Train 2 Cond Prob:	6.7E-01	1.0E+00	
LPR/-HP1.HPR	1.5E-04 > 1.0E-02	1.0E+00	
LPR/HP1	1.05-04 / 1.05-04		
Branch Hodels 1.0F.2			
Train 1 Cond Probi	1.05-02		
Train 2 Cond Probi	1.5E-02 > Unavailable		
branch model file			
** forced			

Austin

09-11-1987

PRECURSOR DESCRIPTION SHEET

LER No.: 269/86-001 Event Description: Trip, LOFW, and a stuck-open MSRV occur Date of Event: January 31, 1986 Plant: Oconee 1

EVENT DESCRIPTION

Sequence

At 1546 h during troubleshooting, circuit breaker PCB-24 was manually closed without the reset of the generator lockout relays. When the breaker was closed, the relay logic was satisfied, causing a yellow bus lockout. Consequently, all the 230-kV yellow-bus tie breakers opened.

One of the breakers that opened was PCB-21, the generator to the yellow-bus tie breaker. When it opened, the only path for current flow from the generator to the switchyard was via PCB-20. PCB-20 faulted, undergoing an explosion, ~17 seconds after PCB-21 opened. At 1547 h the turbine/generator tripped, initiating an anticipatory reactor trip from 100% stable power conditions.

Following the reactor trip, both MFWPs tripped on high dischar e pressure. A preliminary investigation showed that the MFWP speed demar: did not run back as expected. All three EFW pumps started immediately to supply feedwater flow for DHR.

Following the reactor trip, one of the MSRVs (1MS-8) opened and stuck open for 11 min. It reseated when the SG pressure decreased to 975 psig, the minimum RCS pressure was 1750 psig.

Unit 1 was stabilized at hot shutdown conditions with no actuations of engineering safeguard systems or pressurizer relief values, and no RCS leakage was induced.

Corrective Action

PCB-20 was repaired. The MFWP trip set points were reviewed. The MSRV (1MS-8) was repaired.

Plant/Event Data

Systems Involved: MFW, main steam relief, electrical

Components and Failure Modes Involved: MSRV — stuck open in operation MFWPS — tripped off in operation

Component Unavailability Duration: NA Plant Operating Mode: 1 (100% power) Discovery Method: Operational event Reactor Age: 12.8 years Plant Type: PWR

Comments

None

MODELING CONSIDERATIONS AND DECISIONS

1.0

Initiators Modeled and Initiator Nonrecovery Estimate

Transient

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No recovery

Branches Impacted and Branch Nonrecovery Estimate

MEW	1.0	No recovery	possible	in the	short
SS release	0.12	term MSRV stuck	open but	closed on	lower
terminated		pressure			

Plant Models Utilized

PWR plant Class D

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

1.0E+00

Event Identifier: 269/86-001 Event Description: Trip, LOFW, and Stuck Open MSRV Event Date: 1/31/86 Plant: Oconee 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CV	
TRANS	3.2E-05
Total	3.2E-05
CD	
TRANS	2.1E-06
Total	2.16-06
ATWS	
TRANS	3.4E-05
Total	3.4E-05

DOMINANT SEQUENCES

End	State:	CV	Conditional Probability:	2.86-05	
109	TRANS	-RT -AFW	-PORV.OR. SRV. CHALL SS. RELEAS. TERM HPI		
End	State:	CD	Conditional Probability:	1.0E-06	
103	TRANS 1. HPR	-RT -AFW	PORV.OR.SRV.CHALL PORV.OR.SRV.RESSAT -HPI	HPR/-HPI -SS.DEPRESS LPR/-HP	
End	State:	ATWS	Conditional Probability:	3.4E-05	

128 TRANS RT

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100

SEQUENCE CONDITIONAL PROBABILITIES

Sequence	End State	Prob	N Rectt
101 TRANS -RT -AFW PORV.OR.SRV.CHALL -PORV.OR.SRV.RESEAT SS.RELE	CV	2.4E-06	1.0E-01
AS.TERM HP1 103 TRANS -RT -AFW PORV.OR.SRV.CHALL PORV.OR.SRV.RESEAT -HP1 HP	CD	1.0E-06 *	5.0E-02
R/-HPI -SS.DEPRESS LPR/-HPI.HPR 104 TRANS -RT -AFW PORV.OR.SRV.CHALL PORV.OR.SRV.RESEAT -HPI HP	CD	5.88-08	5.0E-02
R/-HPI SS.DEPRESS 109 TRANS -RT -AFW -PORY.DR.SRV.CHALL SS.RELEAS.TERM HPI 123 TRANS -RT AFW MFW -HPI(F/B) HPR/-HPI -SS.DEPRESS COND/MFW 124 TRANS -RT AFW MFW -HPI(F/B) HPR/-HPI SS.DEPRESS 125 TRANS -RT AFW MFW HPI(F/B) -SS.DEPRESS -COND/MFW 126 TRANS -RT AFW MFW HPI(F/B) -SS.DEPRESS COND/MFW 127 TRANS -RT AFW MFW HPI(F/B) SS.DEPRESS 128 TRANS RT	CV CD CD CV CD CD CD CD	2.8E-05 * 4.2E-07 4.7E-08 8.6E-07 4.4E-07 4.9E-08 3.4E-05 *	1.0E-01 3.0E-02 8.8E-02 4.9E-02 2.5E-02 7.4E-02 1.2E-01

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* dominant sequence for end state
** non-recovery credit for edited case

SEQUENCE MODEL:	cilasp\newsodel\pwrotree.cap
BRANCH MODEL:	c:\asp\newmodel\oconee.txt
PROBABILITY FILE:	c:\asp\newmodel\pwr_b.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS LOOP LOCA RT RT/LOOP EMER8.POWER AFW AFW/EMER6.POWER	4.8E-04 4.6E-06 2.4E-06 2.8E-04 0.0E+00 2.9E-03 3.8E-04 5.0E-02	1.0E+00 3.9E-01 4.3E-01 1.2E-01 1.0E+00 8.0E-01 2.6E-01 3.4E-01	
MFW Branch Model: 1.0F.1 Train 1 Cond Prob: PORV.OR.SRV.CHALL PORV.OR.SRV.RESEAT PORV.OR.SRV.RESEAT/EMERG.POWER SS.RELEAS.TERM	2.0E-01 > 1.0E+00 2.0E-01 > Failed 8.0E-02 1.0E-02 1.0E-02 1.5E-02 > 1.0E+00	3.45-01 1.0E+00 5.0E+00 3.4E-01 ⇒ 1.2E-01	

Branch Model: 1.0F.1			
Train 1 Cond Prob:	1.5E-02 > Failed		
SS. RELEAS. TERM / -MFW	1.5E-02 > 1.0E+00	3.4E-01 > 1.2E-0	
Branch Model: 1.0F.1			
Train 1 Cond Prob:	1.5E-02 > Failed		
HPI	3.0E-04	8.4E-01	
HPI(F/B)	3.0E-04	8.4E-01	4.0E-02
HPR/-HPI	1.5E-04	1.0E+00	4.0E-02
SS. DEPRESS	3.6E-02	1.0E+00	
COND/MFN	1.0E+00	3.4E-01	
LP1/HP1	1.5E-04	3.4E-01	
LPR/-HPI.HPR	6.7E-01	1.0E+00	
LPR/HP1	1.5E-04	1.0E+00	

+ branch model file ## forced

Austin 09-11-1987 11:25:59

PRECURSOR DESCRIPTION SHEET

LER No.: 269/86-011 Event Description: Emergency condenser cooling system is unavailable Date of Event: October 1, 1986 Plant: Oconee 1, 2, and 3

EVENT DESCRIPTION

Sequence

The design basis of the ECCW system is to provide water to the condenser for the removal of decay heat during a loss of all ac power event (station blackout). The station blackout scenario is limiting in that CCW siphon flow through the main condenser is used to remove decay heat. Decay heat is transferred to the main condenser via the turbine bypass valves. Feedwater is delivered to the SGs via the turbine-driven EFW pumps.

During performance a load shed test on Unit 2 during refueling, the low-pressure service-water system pumps were found to have failed. A load shed of nonessential loads is initiated when emergency power is required via the underground feeder from Keowee through transformer CT-4. The load shed protects this power path from overload. When the load shed test was initiated, the condenser circulating water pumps were deenergized. Normally, this causes the gravity flow system to align automatically and to allow the flow of water from the intake structure through the condenser and discharging to the Keowee tailrace into Lake Hartwell. The elevation difference and a siphon effect are used to cause the condenser gravity drain to the Keowee tailrace was blocked because it was not part of the test. The pumps had started initially on loss of load to provide condenser cooling but began to cavitate after l h.

CCW flow was restored by restarting a CCW pump, and the plant was restored to its normal powered condition without any plant damage or system upsets. Before the occurrence, two low-pressure service-water pumps were operating with ~13,000 gal/min per pump. The low-pressure service-water pumps are supplied from the CCW crossover header, which was being supplied from Unit 2 at the time.

In the evening of October 1, 1986, the test was repeated; but this time the gravity drain feature was also cested. The results were the same with the loss of low-pressure service-water flow. The U.S. Nuclear Regulatory Commission (NRC), Region II, was advised of these results late in the evening, and NRC concurred that Units 1 and 3 could continue to operate until the test data could be fully evaluated. Units 1 and 3 were at 100% power. At 0900 on October 2, 1986, evaluation of the tests revealed that the operation of this design feature (the CCW siphon flow) was questionable for Units 1 and 3 and that this resulted in inoperability of the low-pressure service-water systems for Oconee. As a result, an orderly shutdown of the two operating units was begun as required by Technical Specification 3.3.7. Both units reached cold shutdown conditions by October 3, 1986. An investigation determined that the ECCW system (gravity flow system) was not working. The lake level was lower than normal, and the low-pressure service-water pump housings were exposed. Because the housings were not qualified for this duty, air leaked in and caused pump cavitation and loss of the condenser syphon. All three units were affected.

The root cause of this incident is the inadequate design and testing of the ECCW system. This led to a failure of the ECCW system to perform its intended function as described in the FSAR under all assumed conditions. Inadequate original design evaluation of the ECCW system and the lower-than-normal lake level of Keowee are contributing factors to the cause of this incident.

Corrective Action

Numerous design and procedure changes were made (see LER pp. 4-5).

Plant/Event Data

Systems Involved: AFW and SG emergency condenser cooling

Components and Failure Modes Involved: CCW pumps — failed in testing

Component Unavailability Duration: Assumed 120 d Plant Operating Mode: 1 (100% power) Discovery Method: Testing Reactor Age: 13.5, 12.9, and 12.1 years, respectively Plant Type: PWR

Comments

Because the system is shared by all units at the station, all three units were affected. AFW is affected given that an EPS failure has occurred.

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Postulated LOOP Base case nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

AFW and EPS 1.0 No recovery assumed possible in the short term

Plant Models Utilized

PWR plant Class D

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 269/86-011 Event Description: Station Emergency Condenser Cooling System is Unavailable Event Date: 10/1/86 Plant: Oconee 1 UNAVAILABILITY, DURATION= 2880 NON-RECOVERABLE INITIATING EVENT PROBABILITIES LOOP 5.2E-03 SEQUENCE CONDITIONAL PROBABILITY SUMS End State/Initiator Probability CV LOOP (5.98-08) Total (5.9E-08) CD LOOP 1.1E-05 Total 1.1E-05 ATWS L009 0.0E+00 Total 0.0E+00 DOMINANT SEQUENCES Conditional Probability: 1.2E-05 End State: CD 218 LOOP -RT/LOOP EMERS. POWER AFW/EMERS, POWER SEQUENCE CONDITIONAL PROBABILITIES Sequence End State Prob N Rec## 218 LOOP -RT/LOOP EMERS. POWER AFM/EMERS. POWER CD 1.2E-05 * 3.1E-01

* dominant sequence for end state

++ 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:\asp\newmodel\pwrdtree.cap
BRANCH MODEL:	c:\asp\newmodel\oconee.txt
PROBABILITY FILE:	c:\asp\newmodel\pwr_b.pro

No Recovery Ligit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	4.85-04	1.0E+00	
LOOP	4.6E-06	3.9E-01	
LOCA	2.45-06	4.3E-01	
RT	2.8E-04	1.2E-01	
RT/LOOP	0.0E+00	1,0E+00	
EMERG, POWER	2.9E-03	8.0E-01	
AFW	3.8E-04	2.6E-01	
AFW/EMERG, POWER	5.0E-02 > 1.0E+00	3.4E-01 > 1.0E+00	
Branch Model: 1.0F.1			
Train 1 Cond Prob:	5.0E-02 > Unavailable		
NEW	2.0E-01	3.4E-01	
PORV. OR. SRV. CHALL	8.0E-02	1.0E+00	
PORV. OR. SRV. RESEAT	1.0E-02	5.0E-02	
PORV. OR. SRV. RESEAT/EMERG. POWER		1.0E+00	
SS. RELEAS. TERM	1,5E-02	3.4E-01	
SS. RELEAS. TERM / - MFW	1.5E-02	3.4E-01	
HP]	3.0E-04	8.4E-01	
HP1(F/B)	3.0E-04	8.4E-01	4.0E-02
HPR/-HPI	1.5E-04	1.0E+00	4.0E-02
SS. DEPRESS	3.6E-02	1.0E+00	
COND/MEN	1.0E+00	3.4E-01	
LP1/HP1	1.58-04	3.4E-01	
LPR/-HPI.HPR	6.78-01	1.0E+00	
LPR/HPI	1.5E-04	1.0E+00	

t branch model file

** forced

Austin 09-11-1987

PRECURSOR DESCRIPTION SHEET

LER No .: Date of Event: Plant:

277/86-003 Event Description: DG trip in test causes scram January 24, 1986 Peach Bottom 2

EVENT DESCRIPTION

Sequence

49

Before the event, the DG E-2 was in service supplying the E-22 and E-23 emergency buses in preparation for a loss of power test on Unit 3.

At 0612 h, DG E-2 automatically tripped, thereby removing all power to the E-22 and E-23 buses. Loss of bus E-22 caused MSIVs A0-2-2-86B and AO-2-2-86D to close inadvertently (their solenoids deenergized). The redundant do solenoids were later found failed. Closure of these valves resulted in a high core-flux condition, which was sufficient to initiate a full reactor scram. Immediately following the scram, reactor water level decreased to -32 in. Group II and III isolations occurred properly at the O-in. water level. The speeds of all three reactor feed pumps automatically increased to recover reactor water level. At +45 in. the reactor feed pumps and the main turbine received trip signals indicating high reactor water level. The feed pumps and main turbine tripped properly. Both reactor recirculation pumps tripped properly during the 13.2-kV bus fast transfer. At 0634 h reactor feed pump C was reset from the high-water-level trip and placed in service to control reactor water level. Both recirculation pumps were returned to service by 0645 h.

Additionally, Group II and III outboard isolations occurred on Unit 3 as a result of this event.

Corrective Action

A review of the most recently completed surveillance test indicated that all MSIV ac and dc coils had satisfactory operating currents when tested 2 d before the event. The dc solenoids were replaced on January 25.

Plant/Event Data

Systems Involved: Emergency power, main steam isolation, and MFW

Components and Failure Modes Involved: DG — failed in test MSIVs — failed closed during operation

Component Unavailability Duration: NA Plant Operating Mode: 1 (95% power) Discovery Method: Testing Reactor Age: 12.3 years Plant Type: BWR

Comments

None

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Transient

Nonrecoverable

Branches Impacted and Branch Nonrecovery Estimate

MFW	0.04	Recoverable from control room (tripped off on high head after scram)
Emergency power system	Base case	DG train E-2 is unavailable
RHR/shutdown cooling	Base case	One train unavailable because of loss buses
RHR/SDC	Base case	One train unavailable because of loss buses
RHR service water	Base case	One train unavailable because of loss buses
SLC/rod insert	Base case	One train unavailable because of loss buses
LPCS	Base case	One train unavailable because of loss buses

Plant Models Utilized

BWR plant Class C

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

1.0E+00

Event Identifier: 277/86-003 Event Description: Diesel Generator Trip Test Causes Scram Event Date: 1/24/86 Plant: Peach Bottom 2

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS			

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability	ł

CV

TRANS	9.0E-08
Total	9.0E-08
CD	
TRANS	8.1E-05
Total	8.1E-05
ATWS	
TRANS .	1.7E-05
Total	1.7E-05

DOMINANT SEQUENCES

End State: CV Conditional probability: 5.2E-08

155 TRANS SCRAM -SLC.OR.RODS PCS/TRANS SRV.CLOSE FW/PCS.LOCA HPCI RCIC/LOCA -SRV.ADS -C OND/FW.PCS -RHR(SDC)

End State: CD Conditional Probability: 6.4E-05

101 TRANS -SCRAM PCS/TRANS SRV.CHALL/TRANS.-SCRAM -SRV.CLOSE -FW/PCS.TRANS RHR(SDC) RHR(S PCOOL) /~LPCI.RHR (SDC) C.I.AND.V/RHR (SDC).RHR (SPCOOL)

End State: ATWS Conditional Probability: 1.7E-05

173 TRANS SCRAM SLC.OR.RODS

SEQUENCE CONDITIONAL PROBABILITIES

	Sequence	End State	Prob	N Rec**
101	TRANS -SCRAM PCS/TRANS SRV.CHALL/TRANSSCRAM -SRV.CLOSE -FW /PCS.TRANS RHR(SDC) RHR(SPCOOL)/-LPCI.RHR(SDC) C.I.AND. V/RHR(SDC).RHR(SPCOOL)	CD	6.4E-05 *	1.12-01
102	TRANS -SCRAM PCS/TRANS SRV.CHALL/TRANSSCRAM -SRV.CLOSE FW /PCS.TRANS -HPCI RHR(SDC) RHR(SPCOOL)/-LPCI.RHR(SDC) C. I.ANI/.V/RHR(SDC).RHR(SPCOOL)	CD	2.6E-06	4.62-03
119	TRANS -SCRAM PCS/TRANS SRV.CHALL/TRANSSCRAM SRV.CLOSE FW /PCS.LOCA HPCI RCIC/LOCA SRV.ADS	CD	1.12-05	1.7E-01
134	TRANS SCRAM -SLC.OR.RODS PCS/TRANS -SRY.CLOSE FW/PCS.TRANS HPCI RCIC/TRANS.OR.LOOP -SRY.ADS -COND/FW.PCS -RHR(SDC)	CV	6.8E-09	1.3E-02
138	TRANS SCRAM -SLC.OR.RODS PCS/TRANS -SRV.CLOSE FW/PCS.TRANS HPCI RCIC/TRANS.OR.LOOP -SRV.ADS COND/FW.PCS -LPCS -RHR(SDC)	CV	3.5E-09	6.6E-03
155	TRANS SCRAM -SLC.OR.RODS PCS/TRANS SRV.CLOSE FW/PCS.LOCA HPCI RCIC/LOCA -SRV.ADS -COND/FW.PCS -RHR(SDC)	CV	5.28-08 *	1.6E-01
159	TRANS SCRAM -SLC.OR.RODS PCS/TRANS SRY.CLOSE FW/PCS.LOCA HPCI RCIC/LOCA -SRY.ADS COND/FW.PCS -LPCS -RHR(SDC)	CV	2.7E-08	8.0E-02
173	TRANS SCRAM SLC.OR.RODS	ATWS	1.7E-05 *	1.0E+00

* dominant sequence for end state

** non-recovery credit for edited case

SEQUENCE MODEL :	c:\asp\newmodel	bwrctree.cmp
BRANCH MODEL :	c:\asp\newmodel)	peach.txt
PROBABILITY FILE:	c:\asp\newmodel	bwr_c.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	8.68-04	1.0E+00	
LOOP	1.7E-05	3.22-01	
LOCA	3.3E-06	5.0E-01	
SCRAM	3.5E-04	1.0E+00	
SLC.OR.RODS	1.0E-02	1.0E+00	4.0E-02
PCS/TRANS	1.7E-01 > 1.0E+00	1.0E+00	
Branch Model: 1.0F.1			
Train 1 Cond Prob:	1.7E-01 > Unavailable		
PCS/LOCA	1.0E+00	1.0E+00	

SRV.CHALL/TRANSSCRAM	1.05.00	v las st	
SRV.CHALL/TRANS.SCRAM	1.0E+00	1.0E+00	
SRY, CHALL/LOOP, -SCRAM	1.0E+00	1.0E+00	
SRV. CHALL/LOOP. SCRAM	1.0E+00	1.0E+00	
	1.0E+00	1.0E+00	
	3.6E-02	1.0E+00	
	2.7E-05	8.0E-01	
FW/PCS.TRANS	4.6E-01 > 1.0E+00	3.4E-01 > 4.0E-02	
Branch Model: 1.0F.1			
Train 1 Cond Prob:	The second		
FW/PCS.LOCA	1.0E+00	3.4E-01	
HPCI	2.9E-02	7.0E-01	
	6.0E-02	7.0E-01	
RCIC/LOCA	1.0E+00	1.0E+00	
CRD	1.0E-02 > 1.0E+00	1.0E+00	4.0E-02
Branch Model: 1.0F.1+opr			TIVE VE
Train 1 Cond Prob:	1.0E-02 > Unavailable		
SRV . ADS	3.7E-03	7.1E-01	4.0E-02
COND/FW.PCS	3.7E-03 1.0E+00	3.4E-01	4100 02
LPCS	3.0E-03 > 3.0E-02	3.4E-01	
Branch Model: 1.0F.2		0.46 01	
Train 1 Cond Prob:	3.0E-02		
Train 2 Cond Prob:			
LPCI (RHR) /LPCS	1.0E-03 > 1.0E-02	7 15-01	
Branch Model: 1.0F.2		7.666 14	
Train 1 Cond Prob:	1.0F-02		
	1.0E-01 > Unavailable		
	5.0E-01	1.0E+00	1 05 05
RHRSW/LPCS.LPC1.LOOP		1.0E+00	4.0E-02
RHRSW/LPCS.LPCI.LOCA	5.05-01	1.0E+00	4.0E-02
RHR (SDC)	2.1E-02 > 3.0E-02		4.0E-02
Branch Model: 1.0F.2+ser	2.12-02 7 3.02-02	3.42-01	
the second	1.0E-02		
Train 2 Cond Prob:	1.02-02		
Serial Component Prob:	1.0E-01 > Unavailable		
Serial Component Prop:			
		3.4E-01	
	1.0E+00	1.0E+00	
RHR(SPCOOL)/-LPCI.RHR(SDC)		1.0E+00	
RHR (SPCOOL) /LPC1.RHR (SDC)	5.2E-01	1.0E+00	
C.I.AND.V/RHR(SDC),RHR(SPCDOL)	1.0E+00	3.4E-01	

* branch model file
** forced

Minarick 02-24-1988 12:06:26

PRECURSOR DESCRIPTION SHEET

LER No.: 280/86-029 Event Description: Charging pump service-water pumps are unavailable Date of Event: September 29, 1986 Plant: Surry 1

EVENT DESCRIPTION

Sequence

All service-water flow to the charging pump service-water subsystem was lost because the pump became air bound. This abnormal condition affected the heat sink for the charging pump lubricating-oil coolers and the intermediate heat sink for the charging pump mechanical seals.

Earlier in the day, one of the redundant charging pumps servicewater pumps, 1-SW-P-10A, had been removed from service for replacement. Maintenance activities required that grinding be performed on a pump support prior to pump replacement. The grinding activity resulted in actuation of a smoke detector, which automatically closed a servicewater fire isolation valve. Due to a leak on a strainer blowdown line in the service-water supply line, the valve closure allowed air inleakage, which caused 1-SW-P-10B to become air bound.

The charging pumps continued to operate. The temperatures of the operating charging pump were monitored.

Corrective Action

The affected pump was vented, and the leak at the strainer blowdown line was repaired.

Plant/Event Data

Systems Involved: HPI and chemical and volume control

Components and Failure Modes Involved: Charging pump service water pumps — one failed in operation; another was out of service Charging pumps — degraded operation without service water

Component Unavailability Duration: 1 h assumed Plant Operating Mode: 1 (100% power) Discovery Method: Operational event Reactor Age: 14.2 years Plant Type: PWR

Comments

For routine charging operations the pump performance was acceptable. For transient conditions with the increased heat load associated with HPI operation, the lack of cooling is assumed to degrade performance.

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Postulated	the second s	Base	case	nonrecovery
Postulated				nonrecovery
Postulated	LOCA			nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

HPI 1.0 Service water for pump cooling not recoverable in the short term

Plant Models Utilized

PWR plant Class A

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 280/86-029 Event Description: Charging Pump Service Water Pumos Are Unavailable Event Date: 9/29/86 Plant: Surry 1

UNAVAILABILITY, DURATION= 1

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.8E-04
LOOP	1.8E-06
LOCA	1.0E-06

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CV	
TRANS	2.5E-06
LOOP	9.2E-09
LOCA	9,7E-07
Total	3.4E-06
CD	
TRANS	1.16-09
LOOP	1.6E-10
LOCA	9.0E-09
Total	1.0E-08
ATWS	
TRANS	0.0E+00
LOOP	0.0E+00
LOCA	0.0E+00
Total	0.0E+00

DOMINANT SEQUENCES

End State: CV Conditional Probabil	1 () 1	2.38-06
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109 TRANS -RT -AFM -PORV. OR. SRV. CHALL SS. RELEAS. TERM HPI

End State: CD

Conditional Probability: 3.7E-08

307 LOCA -RT -AFN HPI SS. DEPRESS

SEQUENCE CONDITIONAL PROBABILITIES

	Sequence	End State	Prob	N Rectt
101	TRANS -RT -AFW PORV.OR.SRV.CHALL -PORV.OR.SRV.RESEAT SS.RELE AS.TERM HPI	CV	9.85-08	3.4E-01
109 304 307	TRANS -RT -AFW -PORV.DR.SRV.CHALL SS.RELEAS.TERM HPI LOCA -RT -AFW HPI -SS.DEPRESS -LPI/HPI -LPR/HPI LOCA -RT -AFW HPI SS.DEPRESS	CV CV CD	2.3E-06 * 9.9E-07 3.7E-08 *	3.4E-01 4.3E-01 4.3E-01

dopinant sequence for end state ## 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:\asp\newmodel\pwratree.cmp
BRANCH MODEL:	C:\asp\newmodel\surry.txt
PROBABILITY FILE:	c:\asp\newmodel\pwr b.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	4.82-04	1.0E+00	
LOOP	4.68-06	3.9E-01	
LOCA	2.4E-06	4.3E-01	
RT	2.8E-04	1.2E-01	
RT/LOOP	0.0E+00	1.0E+00	
EMERG. POWER	5.4E-04	8.0E-01	
AFM	3.8E-04	2,6E-01	
AFW/EMERG. POWER	5.08-02	3.45-01	
MFW	1.98-01	3.4E-01	
PORV. OR, SRV. CHALL	4.0E-02	1.0E+00	
PORV. OR. SRV. RESEAT	2.0E-02	5.0E-02	
PORV. OR. SRV. RESEAT/EMERG. POWER	2.0E-02	1.0E+00	
SS. RELEAS. TERM	1.5E-02	3.4E-01	
SS.RELEAS. TERM/-MFW	1.5E-02	3.46-01	
SS. DEPRESS	3.6E-02	1.0E+00	
COND/MFW	1.0E+00	3.4E-01	

	1.5E-03 > 1.0E+00 **	8.4E-01 > 1.0E+00	
HP1	1102 00 / 1102 00		
Branch Model: 1.0F.3+ser			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Serial Component Prob:	1.2E-03		
HP1(F/B)	1.5E-03 > 1.0E+00 **	8.4E-01 > 1.0E+00	4.0E-02
Branch Models 1.0F.3+ser+op			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Serial Component Prob:	1.2E-03		
PORV.OPEN	1.0E-02	1.0E+00	
HPR/-HPI	1.5E-04	1.0E+00	4.0E-02
	1.0E-02	1.0E+00	
CSR	1.5E-04	3.4E-01	
LP1/HP1	6.7E-01	1.0E+00	
LPR/-HPI.HPR		1.0E+00	
LPR/HPI	1.5E-04	1.05.00	

e branch model file

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Austin 09-11-1987 11:42:53

FRECURSOR DESCRIPTION SHEET

LER No.: Date of Event: Plant:

280/86-031 Event Description: High-head injection system is unavailable October 30, 1986 Surry 1

EVENT DESCRIPTION

Sequence

At 0202 h with Unit 1 at 100% power and Unit 2 in a refueling shutdown, service-water flow was lost to the Unit 1 charging pump servicewater subsystem. Operation without service water to the charging pump service-water system is prohibited by Technical Specification 3.14 because the water provides cooling for the lube oil coolers.

An operator was exchanging the filter elements of an in-line duplex strainer upstream of the operating Unit 1 charging pump service-water pump (1-SW-P-10A) because high differential pressure had been indicated. By failing to fill and went the filter element, he inadvertently allowed air to enter the line. When the standby filter element was placed into service, the concharge pressure of the operating pump decreased to zero. The siging-pump service-water header's lowpressure alarm annunciated ar ! locked in the main control room, and the standby pump (1-SW P-108) au estarted but failed to clear the lowpressure alarm.

Corrective Action

The pumps were vented and returned to service.

Plant/Event Data

Systems Involved: High-head injection, service water

Components and Failure Modes Involved: Charging-pump service-water pumps - failed in operation HPI pumps - degraded

Component Unavailability Duration: 19 min Plant Operating Mode: 1 (100% power) Discovery Method: Maintenance Reactor Age: 14.2 years Plant Type: PWR

Event Identifier: 280/86-031

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Comments

Because of the increased high-head injection system pump heat load during a transient requiring HPI, the conservative assumption is failure of the high-head injection system. [See similar failure at Calvert Cliffs 1 in LER 317/80-027 in NUREG/CR-3591 (ORNL/NSIC-217) and at Surry 2, LER 281/86-010.]

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Postulated	transient	Base	case	nonrecovery
Postulated	LOOP	Base	case	nonrecovery
Postulated	LOCA	Base	case	nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

HPI

1.0

Given HPI demand in a transient, pump service water could not be recoverd in time

Plant Models Utilized

PWR plant Class A

CONDITIONAL CORE DAMASE PROBABILITY CALCULATIONS

Event Identifier: 280/86-031 Event Description: High Head Injection System is Unavailable Event Date: 10/30/86 Plants Surry 1

UNAVAILABILITY. DURATION= .3

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.4E-04
LOOP	5.45-07
LOCA	3.1E-07

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probabili	tv
CV		
TRANS	7.4E-07	
LOOP	2.7E-09	
LOCA	2.9E-07	
Total	1.05-06	
CD		
TRANS	3.4E-10	
LOOP	4.8E-11	
LOCA	2.7E-09	
Total	3.1E-09	
ATWS		
TRANS	0.0E+00	
LOOP	0.0E+00	
LOCA	· 0.0E+00	
Total	0.0E+00	
DOMINANT SEQUENCES		

Fnd State: CV Conditional Probability: 7.0E-07

109 TRANS -RT -45W -PORV.OR.SRV.CHALL SS.RELEAS.TERM HPI

End State: CD Conditional Probability: 1.1E-08

307 LOCA -RT -AFW HPI SS. DEPRESS

SEQUENCE CONDITIONAL PROBABILITIES

	Sequence	End State	Prob	N Rec**
101	TRANS -RT -AFW PORV. OR. SRV. CHALL -PORV. OR. SRV. RESEAT SS. RELE	CV	2.9E-08	3.4E-01
109 304 307	AS.TERM HPI TRANS -RT -AFW -PORV.OR.SRV.CHALL SS.RELEAS.TERM HPI LOCA -RT -AFW HPI -SS.DEPRESS -LPI/HPI -LPR/HPI LOCA -RT -AFW HPI SS.DEPRESS	CV CV CD	7.05-07 * 3.0E-07 1.1E-08 *	3.4E-01 4.3E-01 4.3E-01

dominant sequence for end state ++ 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:\asp\newmodel\pwratree.cmp
BRANCH MODEL:	c:\asp\newmodel\surry.txt
PROBABILITY FILE:	c:\asp\newmodel\pwr_b.pro

No Recovery Limit

BRINCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	4.82-04	1.0E+00	
LOOP	4.5E-06	3.9E-01	
LOCA	2.45-06	4.3E-01	
RT	2.88-04	1.2E-01	
RT/LOOP	0.02+00	1.0E+00	
EMERS. FOWER	5.4E-04	8.0E-01	
	3.8E-04	2.65-01	
AFW AFW/EMERG.POWER	5,0E-02	3.4E-01	
	1.9E-01	3.4E-01	
MFN COULCHALL	4.0E-02	1.0E+00	
PORV. OR. SRV. CHALL	2.0E-02	5.0E-02	
PORV. OR. SRV. RESEAT	2.0E-02	1.0E+00	
PORV.OR. SRV. RESEAT/EMERG. POWER		3.4E-01	
SS, RELEAS, TERM	1.58-02	3.4E-01	
SS.RELEAS. TERM/-MFW	1.5E-02	1.0E+00	
SS. DEPRESS	3.6E-02	3.4E-01	
COND/MFW	1.02+00	0.45-01	

HP1	1.5E-03 > 1.0E+00 ++	8.4E-01 > 1.0E+00	
Branch Model: 1.0F.3+ser			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Serial Component Prob:	1.2E-03		
HPJ(F/B)	1.5E-03 > 1.0E+00 ++	8.4E-01 > 1.0E+00	1 05-00
Branch Model: 1.0F.3+ser*o	pr	VITE VI / 1.VETUV	4.0E-02
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
Train 3 Cond Prob:	3.0E-01		
Serial Component Prob:	1.2E-03		
PORV. OPEN	1.0E-02	1.0E+00	
HPR/-HPI	1.5E-04	1.0E+00	4 45-44
CSR	1.0E-02	1.0E+00	4.0E-02
LP1/HP1	1.5E-04	3.4E-01	
LPR/-HP1.HPR	6.7E-01	1.0E+00	
LPR/HF1	1.5E-04	1.0E+00	
		1105.00	
* branch model file			

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LER No.: 281/86-010 Event Description: High-head injection system is unavailable Date of Event: July 11, 1986 Plant: Surry 2

EVENT DESCRIPTION

Sequence

With Unit 2 at 100% power and Unit 1 in refueling shutdown, emergency maintenance was being performed on 2-CC-P-2A, Unit 2's A chargingpump CCW pump. At that time the redundant pump, 2-CC-P-2B, was in operation supplying cooling water to the charging-pump seal coolers. At 1518 h, while operators were attempting to return pump A to service following the maintenance, the discharge pressure of pump B dropped to zero. Both charging-pump CCW pumps were inoperable.

The charging-pump cooling-water system is a closed system consisting of two redundant cooling-water pumps, two redundant intermediate coolers (cooled by service water), a head tank, and six seal coolers (two per charging pump). Shortly after the event began, a low level was noted to exist in the head tank. Therefore, a large leak was assumed to exist in the system, and efforts were directed toward finding the leak. However, no significant leaks could be found. Thus, it is speculated that air was introduced into the system during maintenance on pump A, which caused pump B to become vapor bound, resulting in zero discharge pressure.

Corrective Action

Pump A was properly vented, makeup water was added to the system, and pump A was returned to service at 1825 h — after operability of the pump was demonstrated. Subsequently, operability of pump B was demonstrated, and that pump was also returned to service.

Plant/Event Data

Systems Involved: High-head injection system, CCW system

Components and Failure Modes Involved: Charging-pump CCW pumps — One was out of service, failed in testing and one failed in operation

Component Unavailability Duration: 3 h Plant Operating Mode: 1 (100% power) Discovery Method: Testing Reactor Age: 13.3 years Plant Type: PWR

Comments

A similar event occurred at Calvert Cliffs 1 as reported in LER 317/80-027 and evaluated in NUREG/CR-3591 (ORNL/NSIC-217). (See also LER 280/86-031 for Surry 1 in this report.)

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Postulated		Base	case	nonrecovery
Postulated Postulated				nonrecovery
resturated	LOOP	Base	case	nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

HPI	1.0	Not	readily	recoverable	(assume	1088
				cooling fai		1000

Plant Models Utilized

PWR plant Class A

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 281/86-010 Event Description: High Head Injection System is Unavailable Event Date: 7/11/86 Surry 2 Plants

UNAVAILABILITY, DURATION= 3

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1,4E-03
LOOP	5.48-06
	3.1E-06
LOCA	

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator	Probability
CV	
TRAMS LOOP LOCA	7.4E-06 2.8E-08 2.9E-06
Total	1.0E-05

CD

Total

TRANS	3.4E-09
LOOP	4.8E-10
LOCA	2.7E-08
Total	3.1E-08

ATHS

TRANS	0.0E+00
LOOP	0.0E+00
LOCA	0.0E+00
Tota)	0.0E+00

DOMINANT SEQUENCES

End State: CV Conditional Probability: 7.0E-06	۴.
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109 TRANS -RT -AFW -PORV. DR. SRV. CHALL SS. RELEAS. TERM HPI

End State: CD

Conditional Probability: 1.1E-07

307 DCA -RT -AFM HPI SS. DEFRESS

SEQUENCE CONDITIONAL PROBABILITIES

Sequence		End State	Prob	N Rec++
101	TRANS -RT -AFW PORV.OR.SRV.CHALL -PORV.OR.SRV.RESEAT SS.RELE AS.TERM HP1	CV	2.98-07	3.4E-01
109 304 307	TRANS -RT -AFW -PORV.OR.SRV.CHALL SS.RELEAS.TERM HPI LOCA -RT -AFW HPI -SS.DEPRESS -LPI/HPI -LPR/HPI LOCA -RT -AFW HPI SS.DEPRESS	CV CV CD	7.0E-06 * 3.0E-06 1.1E-07 *	3,4E-01 4,3E-01 4,3E-01

dominant sequence for end state
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:\asp\newmodel\pwratree.cmp
BRANCH MODEL:	c:\asp\aewnodel\surry.txt
PROBABILITY FILE:	C:\asp\newsodel\pwr b.pro

No Recovery Limit

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BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	4.8E-04	1.0E+00	
LOCA	4.62-06	3.9E-01	
RT	2.4E-06 2.8E-04	4.3F-01 1.2E-01	
RT/LOOP	0.0E+00	1.0E+00	
EMERG. POWER AFW	5.4E-04	8.46-01	
AFW/EMERS. POWER	3.8E-04 5.0E-02	2.5E-01	
NFN	1.95-01	3.4E-01 3.4E-01	
PORV. OR, SRV, CHALL	4.0E-02	1.0€+00	
PORV.OR.SRV.RESEAT PORV.OR.SRV.RESEAT/EMERG.POWER	2.0E-02	5.0E-02	
SS. RELEAS. TERM	2.0E-02 1.5E-02	1.0E+00	
SS. RELEAS. TERM / - MFW	1.5E-02	3.4E-01 3.4E-01	
SS. DEPRESS	3.6E-02	1.0€+00	
COND/MFW	1.0E+00	3.4E-01	

Event Identifier: 281/86-010

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HPI	1.5E-03 > 1.0E+00 **	8.4E-01 > 1.0E+00	
Branch Models 1.0F.3+ser			
Train 1 Cond Prob:	1.0E-02		
	1.0F-01		
Train 2 Cond Prob:	3. 1 01		
Train 3 Cond Probs			
Serial Component Frobi	1.25-03	8.42-01 > 1.02+00	4.0E-02
H-1(F/B)	1.5E-03 > 1.0E+00 **	8.46-01 / 1.06400	TIVE VA
Branch Model: 1.0F.3+ser+opr			
Train 1 Cond Prob:	1.0E-02		
Train 2 Cond Prob:	1.0E-01		
	3.0E-01		
Train 3 Cond Probi	1.2E-03		
Serial Component Prob:		1.0E+00	
PORV. OPEN	1.0E-02		4.0E-02
HPR/-HPI	1.58-04		41.V6 VA
CSR	1.0E-02	1.0E+00	
LP1/HP!	1.5E-04	3.4E-01	
	6.7E-01	1.0E+00	
LPR/-Y	1.5E-04	1.0E+00	
LPR/HF1	1.02 .04		
branch model file			
** forced			

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Austin 09-11-1987 12:16:36 \$

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LER No.: 282/86-006 Event Description: Emergency power system is unavailable Date of Event: September 8, 1986 Plant: Prairie Island 1 and 2

EVENT DESCRIPTION

Sequence

Both units were at steady state power: Unit 1 at 100% and Unit 2 at 88% power. During an operability test, D-1 DG failed to start after cranking for 10 s. D-1 DG was declared inoperable. While the diesel cooling-water pump 22 was being run to prove operability of D-2 DG, a small oil line burst; the pump was shutdown and declared inoperable. With these two components inoperable, a power decrease w: begun on both units. The two station DGs serve both Units 1 and 2.

The cause of the inoperability of D-1 DG was not immediately apparent, but further investigation and testing revealed that leakage through the fuel-head pressure return orifice check valve allowed the fuel oil header to drain during idle periods.

Corrective Action

The brass cil line on diesel cooling-water pump 22 was replaced with a stainless steel line. The fuel-head pressure return orifice check valve was replaced on the D-1 DG.

Plant/Event Data

Systems Involved: Emergency power generation system and emergency-power cooling system

Components and Failure Modes Involved: DG — failed to start on demand Diesel cooling water pump — failed during operation

Component Unavailability Duration: 36 min Plant Operating Mode: 1 (100% power for Unit 1); 1 (88% power for Unit 2) Discovery Method: Testing Reactor Age: 12.8 years for Unit 1; 11.7 years for Unit 2 Plant Type: PWR

Comments

Because of the heavy heat loads imposed on the DGs during full load conditions, the conservative assumption is that the DGs are failed in a LOOP. The DGs serve both Units 1 and 2.

Both units were affected.

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Postulated LOOP Base case nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

1.0

EPS

Not readily recoverable

Plant Models Utilized

PWR plant Class B

Event Identilier: 282/86-006

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CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 282/86-006 Event Description: Emergency Power is Unavailab. Event Date: 9/8/86 Plant: Prairie Island 1 UNAVAILABILITY, DURATION= . 6 NON-RECOVERABLE INITIATING EVENT PROPABILITIES LOOP 1.1E-06 SEQUENCE CONDITIONAL PROBABILITY SUMS End State/Initiator Probability CV LOOP 5.3E-09 Total 5.3E-09 CD LOOP 1.9E-08 Total 1.92-08 ATHS LOOP 0.0E+00 Total 0.0E+00 DOMINANT SEQUENCES End State: CV Conditional Probability: 5.1E-09 217 LOOP -RT/LOOP EMERS. POWER -AF#/EMERS. POWER -PORV. OR. SRV. CHALL SS. RELEAS. TERM End State: CD Conditional Probability: 1.8E-08 218 LOOP -RT/LOOP EMERS. POWER AFW/EMERS. POWER SEQUENCE CONDITIONAL PROBABILITIES

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+ dominant sequence for end state

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:\asp\newsodel\pwrbtree.cmp
BRANCH HODEL:	c:\asp\newmodel\praislan.txt
PROBABILITY FILES	c:\asp\newmodel\pwr_b.pro

No Recovery Limit

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BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	4.8E-04	1.0E+00	
LOOP	4.68-06	3.9E-01	
LOCA	2.4E-06	4.3E-01	
RT	2.8E-04	1.2E-01	
RT/LOOP	0.0E+00	1.0E+00	
EMERS. POWER	2.9E-03 > 1.0E+00	8.0E-01 > 1.0E+00	
Branch Model: 1.0F.2			
Train 1 Cond Prob:	5.0E-02 > Unavailable		
Train 2 Cond Prob:	5.7E-02 > Unavailable		
AFW	1.3E-03	2.6E-01	
AFK/EMERG. POWER	5.0E-02	3.4E-01	
NFW	1.9E-01	3.4E-01	
PORV. OR. SRV. CHALL	4.0E-02	1.0E+00	
PORV. OR. SRV. RESEAT	2.0E-02	5.0E-02	
PORV. OR. SRV. RESEAT / EMERG. POWER	2.0E-02	1.0E+00	
SS. RELEAS. TERM	1.5E-02	3.4E-01	
SS. RELEAS. TERM / - MFW	1.5E-02	3.4E-01	
HP1	1.0E-03	8.4E-01	
HPI(F/B)	1.0E-03	8.4E-01	4.0E-02
HPR/-HPI	1.5E-04	1.0E+00	4.0E-02
PORV. OPEN	1.0E-02	1.0E+00	
SS. DEPRESS	3.6E-02	1.0E+00	
A ALTON MALES			

COND/MFW LPI/HPI LPR/-HPI.HPR LPR/HPI	1.0E+00 1.5E-04 6.7E-01 1.5E-04	3.4E-01 3.4E-01 1.0E+00
	1105 44	1.0E+00

* branch model file
** forced

Austin 09-11-1987 12:18:57

LER No.: 282/86-011 Event Description: Emergency power system is unavailable Date of Event: December 27, 1986 Plant: Prairie Island 1 and 2

EVENT DESCRIPTION

Sequence

While Units 1 and 2 were both at 100% power, DG 22 cooling-water pump (DCLP) was out of service for scheduled maintenance. At 0848 k during the daily test of DCLP 12, a water hose ruptured, rendering both DGs unavailable. A station shutdown was begun. At 1008 h, DG operability was restored, and the units returned to full power operation. The two DGs serve both Units 1 and 2.

Corrective Action

The hose was replaced.

Plant/Event Data

Systems Involved: Emergency power

Components and Failure Modes Involved: DG water hose — failed in test DGLP — was out for maintenance

Component Unavailability Duration: 1.25 h Plant Operating Mode: 1 (100% power) Discovery Method: Testing Reactor Age: 13.0 and 11.9 years, respectively Plant Type: PWR

Comments

Both units were affected because they share the same DG. Due to the heavy heat load imposed during full-load conditions, the conservative assumption is that the EPS would be failed in a LOOP.

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Postulated LOOP Base case nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

EPS 1.0 Not readily recoverable Planc Models Utilized

PWR plant Class B

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier:					
Event Description:		r System Unavailal	bility		
Event Date: Plant:	12/2//86 Prairie Island	1			
rient:					
UNAVAILABILITY, DUR	R110M= 1.20				
NON-RECOVERABLE INI	TIATING EVENT	PROBABILITIES			
LOOP				2.28-06	
SEQUENCE CONDITIONA	L PROBABILITY	SUMS			
End State/Init	iator			Probability	
CV					
LOOP				1.1E-08	
Total				1.12-08	
CD					
LOOP				4.0E-08	
Total				4.0E-08	
ATHS					
LOOP				0.0E+00	
Total				0.0E+00	
DOMINANT SEQUENCES					
End State: CV		Conditional	Probability:	1.1E-08	
217 LOOP -RT/LOOP	EMER6.POWER	-AFW/EMERG.POWER	-PORV.OR.SRV.CH	ALL SS.RELEAS.TERM	
End State: CD		Conditional	Probability:	3.8E-08	
218 LOOP -RT/LOOP	EMERS. POWER	AFW/EMERG. POWER			
SEQUENCE CONDITIONA	L PROBABILITIE	5			

Event Identifier: 282/86-011

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Sequence	End State	Prob	N Rec++
215 LODP -RT/LOOP EMERG.POWER -AFW/EMERS.POWER PORV.OR.SRV.CHALL -PORV.OR.SRV.RESEAT/EMERG.POWER SS.RELEAS.TERM	cv	4.4E-10	1.3E-01
216 LOOP -RT/LOOP EMERS.POWER -AFM/EMER8.POWER PORV.OR.SRV.CHALL PORV.OR.SRV.RESEAT/EMERS.POWER	CD	1.8E-09	3.92-01
217 LOOP -RT/LOOP EMERS.POWER -AFW/EMERS.POWER -PORV.OR.SRV.CHALL SS.RELEAS.TERM	CV	1,12-08 +	1.3E-01
218 LOOP -RT/LOOP EMERG. POWER AFW/EMERG. POWER	CD	3.8E-08 *	1.32-01

dominant sequence for end state ## 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:\asp\newmodel\pwrbtree.cmp
BRANCH MODEL:	c:\asp\newmodel\praislan.txt
PROBABILITY FILE:	c:\asp\newmodel\pwr_b.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	4.82-04	1.0E+00	
LOOP	4.65-06	3.9E-01	
LOCA	2.45-06	4.3E-01	
RT	2.8E-04	1.2E-01	
RT/LOOP	0.0E+00	1.0E+00	
EMERG, POWER	2.9E-03 > 1.0E+00	8.0E-01 > 1.0E+00	
Branch Model: 1.0F.2		**** ** / 1.VETVY	
Train 1 Cond Prob:	5.0E-02 > Unavailable		
Train 2 Cond Prob:	5.7E-02 > Unavailable		
AFN	1.3E-03	2.6E-01	
AFN/EMERS, POWER	5.0E-02	3.4E-01	
MFN	1.9E-01	3.4E-01	
PORV. OR. SRV. CHALL	4.0E-02	1.0E+00	
PORV. OR. SRV. RESEAT	2.0E-02	5.0E-02	
PORV. DR. SRV. RESEAT/EMERS. POWER	2.0E-02	1.0E+00	
SS. RELEAS. TERM	1.5E-02	3.4E-01	
SS. RELEAS. TERM / -MFW	1.5E-02	3.4E-01	
HPI	1.0E-03	8.4E-01	
HPI(F/B)	1.0E-03	8.4E-01	4.0E-02
HPR/-HPI	1.5E-04	1.0E+00	4.0E-02
PORV. OPEN	1.0E-02	1.0E+00	4.06-02
SS. DEPRESS	3.6E-02	1.0E+00	

COND/MFW	1.0E+00	3.4E-01
LP1/HP1	1.5E-04	3.4E-01
LPR/-HPI.HPR	6.7E-01	1.0E+00
LPR/HP1	1.5E-04	1.0E+00

* branch accel file
** forced

Austin 09-11-1987 12:21:19

LER No.:	285/86-001
Event Description:	Trip occurs, and automatic depressurization and turbine bypass sytem fails to open
Date of Event: Plant:	July 2, 1986 Ft. Calhoun

EVENT DESCRIPTION

Sequence

At 0534 h, during normal operation while the reactor was at 100% power, an instrument inverter trouble alarm was received in the control room. Control room operators quickly diagnosed a failed instrument inverter feeding bus AI-40A. They dispatched an equipment operator to the switchgear room to reenergize the bus manually by closing the breaker on a bypass transformer also feeding bus AI-40A. The inverter failure placed the RPS in a half-trip condition because the RPS operates on a two-out-of-four logic and the failed inverter was one of four feeding the independent channels of the RPS. About 10 s after the inverter failure, a reactor trip occurred when a second channel trip was received on the SG B low-level trip unit.

Several unusual transients were noted in the moments following the trip:

- RCS pressure increased to ~2400 psia for a short period of time. This caused PORVs to be actuated.
- SG pressure increased to the set point of the secondary safety valves, causing them to be actuated.
- 3. Overfeeding the SG resulted in abnormally high level and subsequent overcooling of the primary system. As a result, RCS pressure decreased to a low of ~1725 psia. The overfeeding occurred because the main feed regulating valves failed to ramp down; the failure was due to loss of power to a relay when the inverser failed.
- 4. Steam dump and bypass valves could not be opened because the inverter power was lost to their controllers as well as to a relay that causes the dump valve to open.
- 5. The operating charging pump stopped, and the two backup pumps could not be started because of loss of inverter power to the relay that controls the backup pump's operations. Although the operating pump should not have stopped, for an unknown reason it did.

Within 1 min of the reactor trip, the equipment operator had reenergized the lost instrument bus, and control room operators w re soon able to restore the plant to normal shutdown condition.

A diagnosis of the information revealed the following. The deenergized instrument bus AI-40A supplies power to electrohydraulic-control panel AI-50 with no alternate power. A turbine first-stage pressure transmitter that sends a signal to the electrohydraulic-control loadcontrol circuitry is powered from AI-50. Loss of power caused a loss of signal to the load-control unit, resulting in the turbine control valves closing witbout a reactor trip. This explains the high pressure seen in the primary system and the low SG level earlier in the transient.

Corrective Action

Modification was made to the bus and inverter power transfer controls to provide backup power.

Plant/Event Data

Systems Involved: Atmospheric steam dump, turbine bypass, charging, and electrical

Components and Failure Modes Involved: Inverter — failed in operation Automatic depressurization and turbine bypass system — failed on demand Charging — failed in operation

Component Unavailability Duration: Plant Operating Mode: 1 (100% power) Discovery Method: Operational event Reactor Age: 12.9 years Plant Type: PWR

Comments

None

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Transient 1.0 No recovery

Branches Impacted and Branch Nonrecovery Estimate

SS depressuri- 0.34 Recoverable locally at the valves zation

Plant Models Utilized

PWR plant Class G

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 285/86-001 Event Description: Trip and ADS/TBS/ Fails to Open Event Date: //2/86 Plant: Fort Calhoun

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

TRANS	1.0E+00
SEQUENCE CONDITIONAL PROBABILITY SUMS	

End State/Initiator Probability
CV
TRAWS 1.5E-06
Total 1.5E-06
CD
TRANS 4.1E-05
ATWS

TRANS 3.4E-05 Total 3.4E-05

DOMINANT SEQUENCES

End State: CV Conditional Probability: 1.3E-06 101 TRANS -RT -AFW PORV.OR.SRV.CHALL -PORV.OR.SRV.RESEAT SS.RELEAS.TERM HPI End State: CD Conditional Probability: 4.0E-08 102 TRANS -RT -AFW PORV.OR.SRV.CHALL PORV.OR.SRV.RESEAT -HPI HPR/-HPI End State: ATWS Conditional Probability: 3.4E-05

121 TRANS RT

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SEQUENCE CONDITIONAL PROBABILITIES

Sequence	End State	Prob	N Rec++
101 TRANS -RT -AFM PORV.OR.SRV.CHALL -PORV.OR.SRV.RESEAT SS.RELE	CV	1.3E-06 +	2.9E-01
AS, TERM HPI 102 TRANS -RT -AFW PORV. OR. SRV. CHALL PORV. OR. SRV. RESEAT -HPI HP	CD	4.02-05 +	5.0E-02
R/-HPI 115 TRANS -RT AFM MFM -HPI(F/B) HPR/-HPI -SS.DEPRESS -COND/MFM 118 TRANS -RT AFM MFM HPI(F/B) -SS.DEPRESS -COND/MFM	CV CV	1.1E-07 1.2E-07 3.4E-05 *	3.9E-02 3.2E-02 1.2E-01

dominant sequence for end state ## non-recovery credit for edited case

SEQUENCE MODEL:	c:\asp\newmodel\pwrqtree.cmp
BRANCH MODEL:	c:\asp\newsodel\calhoun.txt
PROBABILITY FILE:	c:\asp\newmodel\pwr_b.pro

No Recovery Limit

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BRANCH FREQUENCIES/PROBABILITIES

System	Non-Recov	Opr Fail
4.8E-04	1.0E+00	
4.65-06	3.9E-01	
2.4E-06	4.32-01	
2.85-04	1.2E-01	
0.0E+00	1.0E+00	
5.4E-04	8.0E-01	
3.8E-04	2.6E-01	
	3.4E-01	
	3.4E-01	
4.0E-02 > 1.0E+00 **	1.0E+00	
4. DE-02		
	5.0E-02	
ST12 12		
	57 128 W.C.	
3.6E-02 > Failed		
	3.4E-01	
		4.0E-02
	4.8E-04 4.6E-06 2.4E-06 2.8E-04 0.0E+00 5.4E-04 3.8E-04 5.0E-02 2.0E-01	4.8E-04 1.0E+00 4.6E-06 3.9E-01 2.4E-06 4.3E-01 2.8E-04 1.2E-01 0.0E+00 1.0E+00 5.4E-04 8.0E-01 3.8E-04 2.6E-01 5.0E-02 3.4E-01 2.0E-01 3.4E-01 4.0E-02 5.0E-02 2.0E-02 5.0E-02 2.0E-02 3.4E-01 1.5E-02 3.4E-01 1.5E-02 3.4E-01 3.6E-02 > 1.0E+00 1.0E+00 3.6E-02 > 1.0E+00 1.0E+00 3.6E-02 > 1.0E+00 1.0E+00 3.6E-02 > 1.0E+00 3.4E-01 3.6E-02 > 1.0E+00 3.4E-01 3.6E-02 > Failed 1.0E+00 1.0E+00 3.4E-01

Event Identifier: 285/86-001

19

PORV. OPEN	1.0E-02	1.0E+00
HPR/-HPI	1.58-04	1.0E+00
CSR	2.0E-03	3.4E-01

* branch model file ## forced

Austin 09-11-1987 12:25:00

LER No.: Date of Event: Plant:

293/86-027 Event Description: LOOP occurs lue to winter storm November 19, 1986 Pilgrim 1

EVENT DESCRIPTION

Sequence

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> While the reactor was shut down for refueling, an arcing of the high-voltage lines due to locally heavy ice and snow occurred a. 0819 h during a severe winter storm. A loss of offsite power occurred when both transmission lines tripped off due to near-simultaneously detected faults. All safety systems responded as designed. Offsite power was restored at 1015 h when the two lines were reenergized. By 1128 h the switchyard was restored to service.

Corrective Action

An inspection revealed no damage to any system.

Plant/Event Data

Systems Involved: Electrical

Components and Failure Modes Involved: Offsite power - failed in operation

Component Unavailability Duration: NA Plant Operating Mode: 6 (0% power) Discovery Method: Operational Reactor Age: 14.4 years Plant Type: BWR

Comments

Given the circumstances, this event could have occurred at full power and is analyzed on this basis.

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

LOOP 0.12 Backup power source available

Branches Impacted and Branch Nonrecovery Estimate

None

Plant Models Utilized

BWR plant Class C

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: Event Description: Event Date: Plant:	293/86-027 LOOP Due to Winter Sto 11/19/86 Pilgrim 1	78			
INITIATINE EVENT					
NON-RECOVERABLE IN	VITIATING EVENT PROBABI	LITIES			
LOOP				1.22-01	
SEQUENCE CONDITIO	NAL PROBABILITY SUMS				
End State/In	itiator			Probability	
CV					
LOOP				3.38-08	
Total				3,3E-08	
CD					
LOOP				7.76-06	
Total				7.7E-06	
ATWS				2.22-06	
LOOP					
Total				2.2E-06	
DOMINANT SEQUENC	ΈS				
End State: CV		Conditional	Probability:	3.2E-08	
226 LOOP -EME	RG.POWER SCRAM -SLC.OR	RODS MPCI	RCIC/TRANS.OR	LOOP -SRV.ADS -L	PCS -RHR(SDC)
End State: CD		Conditional	Probability:	5.6E-06	
201 LOOP -EME LPCI.RHR(RS.POWER -SCRAM SRV.C SDC1 C.I.AND.V/RHR(SDC	HALL/LOOPSO	RAM - SRV.CLOSE)	-HPC1 RHR(SDC)	RHR (SPC00L)/-
End State: ATM	6	Conditional	Probability:	2.1E-06	

Event Identifier: 293/86-027

240 LOOP -EMERG. FOWER SCRAM SLC.OR. RODS

SEQUENCE CONDITIONAL PROBABILITIES

	Sec	uence		End State	Prob	N Rec++
201	LOOP -EMERG.POWER -SCRAM CI RHR(SDC) RHR(SPCOOL C).RHR(SPCOOL)	SRV.CHALL/LOOPSCRAM -SR)/-LPCI.RHR(SDC) C.I.AND.V	V.CLOSE -HP (RHR(SD	CD	5.68-06 +	1.4E-02
208		SRV.CHALL/LOOPSCRAM -SRV CRD SRV.ADS	CLOSE HP	CD	2.1E-07	4.28-02
215		SRV.CHALL/LOOPSCRAM SRV	CLOSE IF	CD	1.4E-06	6.0E-02
226	P -SRV.ADS -LPCS -RHR (SD		AWS.OR.LOO	CV	3.2E-08 +	5.8E-02
240	LOOP -EMERS. POWER SCRAM	SLC.OR.RODS		ATWS	2.1E-06 +	1.2E-01
243	LOOP EMERG. POWER -SCRAM	SRV. CHALL/LOOP SCRAM -SRV	CLOSE HP	CD	2.3E-07	4.7E-02
	CI RCIC/TRAWS.OR.LOOP		100.000 18	~~~	2136-07	4.75-02
250	LOOP ENERG. POWER SCRAM			ATWS	9.62-08	9.68-02
* de	mainant sequence for end s					
	m-recovery credit for edi					
	ALLECOVERY CREDIT FOR EQ1	ceo case				
CEN IE	NCE MODEL: c:\asp\ne					
		woodel\bwrctree.cmp				
	H MODEL: c:\asp\new	wmodel\pilgrim.txt				
PRUBA	BILITY FILE: c:\asp\ne	wmodel/bwr_c.pro				
No Re	covery Limit					
BRANC	H FREQUENCIES/PROBABILITIE	15				
Branci						
pr anci	·	System	Non-Reci	DV	Opr Fail	
TRAKS						
		8.65-04	1.0E+00			
LOOP	and the first of the	1.7E-05 > 1.7E-05	3.2E-01	> 1.2E-01		
	ranch Model: INITOR					
	nitiator Freq:	1.7E-05				
LOCA		3.3E-06	5.0E-01			
SCRAM		3.5E-04	1.0E+00			
SL 08	R.RODS	1.0E-02	1.0E+00		4.0E-02	
PCS/TF	RANS	1.7E-01	1.0E+00		4.05-02	
PCS/LC	DCA	1.0E+00	1.0E+00			
SRY . CH	HALL / TRANS SCRAM	1.0E+00	1.0E+00			
	WILL/TRANS, SCRAM	1.0E+00				
	HLL /LOOP SCRAM	1.0E+00	1.0E+00			
	HALL/LOOP SCRAM	1.0E+00	1.0E+00			
SRV.CL			1.0E+00			
EKS		1.3E-02	1.0E+00			
ALL MARKED	a start	2.9E-03	8.0E-01			

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Exett 111 293/86-027

D-90

FW/PCS, TRANS	2.9E-01	3.4E-01	
FW/PCS.LOCA	4.0E-02	3.4E-01	
HFCI	2.9E-02	7.0E-01	
RCIC/TRANS.OR.LOOP	6.0E-02	7.0E-01	
RCIC/LOCA	1.0E+00	1.0E+00	
CRD	1.0E-02	1.0E+00	4.0E-02
SRV.ADS	3.7E-03	7.1E-01	4.0E-02
COND/FW.PCS	1.05+00	3.4E-01	
LPCS	3.0E-03	3.4E-01	
LPCI (RHR) /LPCS	1.0E-03	7.1E-01	
	5.0E-01	1.0€+00	4.0E-02
RHRSW/LPCS.LPCI.TRANS	5.0E-01	1,0€+00	4.0E-02
RHRSW/LPCS.LPCI.LOOP		1.0E+00	4.0E-02
RHRSW/LPCS.LPCI.LOCA	5.0E-01	3.4E-01	
RHR (SDC)	2.1E-02		
RHR (SDC) /-LPCI	2.0E-02	3.4E-01	
RHR (SDC) /LPCI	1.0E+00	1.0E+00	
RHR (SPCOOL) /-LPCI .RHR (SDC)	2.0E-02	1.0E+00	
RHR (SPCOOL) /LPCI .RHR (SDC)	5.2E-01	1.0E+00	
C.I.AND.V/RHR (SDC) .RHR (SPCOOL)	1.0E+00	3.4E-01	

* branch model file

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Minarick 02-24-1988 12:08:17

LER No.: 301/86-004 Event Description: MSIVs fail to close on demand Date of Event: September 28, 1986 Plant: Point Beach 2

EVENT DESCRIPTION

Sequence

The MSIVs had been in the open position since December 31, 1985. After that time, Unit 2 operated normally at full power with one runback and several power reductions requested by the power supply. The unit was shut down for refueling on September 27, 1986. On September 28, the reactor operator tried to shut the MSIVs from the control room. When the MSIVs did not shut, an operator was sent to the valves to close them manually. They were closed manually by the operator applying force to the operating arm of each valve. It should also be noted that the nonreturn valves also did not close under the no-flow conditions. The manual force needed to close the nonreturn valves was <7 ft-lb. This amount of force is minimal compared with the closing force that would have been applied to the valves if a reverse-steam-flow condition had occurred. Therefore, the conclusion is that the nonreturn valves would have closed under these circumstances.

Corrective Action

The immediate corrective action was to close the valves manually. The valves will be disassembled and inspected to determine the cause of the failure. Once the cause is determined, the valves will be repaired and tested.

Plant/Event Data

Systems Involved: Main steam isolation system

Components and Failure Modes Involved: MSIVs — failed to close on demand

Component Unavailability Duration: 4.5 months assumed Plant Operating Mode: 5 (cold shutdown going to refueling) Discovery Method: Operational event Reactor Age: 14.3 years Plant Type: PWR

Comments

None

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Postulated SLB Base case nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

SG isolation 1.0 Not considered recoverable in short term following the postulated SLB initiation

Plant Models Utilized

PWR plant Class E

Event Identifier: 301/86-004

2

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 301/86-004 Event Description: MSIVs Fail to Close on Demand Event Date: 9/29/86

UNAVAILABILITY, DURATION= 3240

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

SLB		3.6E-04
SEQUENCE CON	DITIONAL PROBABILITY SUNS	
End Sta	te/Initiator	Probability
CD .		

SLB	4.9E-07
Total	4.8E-07
ATWS	
SLB	0.0E+00
Total	0.0E+00

DOMINANT SEQUENCES

End State: CD Conditional Probability: 3.0E-07

106 SLB -RT REQ. SS. 150 -AFM -HP1 REQ. BA. ADDITION

SEQUENCE CONDITIONAL PROBABILITICS

	Sequence	End State	Prob	N Rec++
105	SLB -RT REQ.SS.ISO -AFM -HP1 -REQ.BA.ADDITION PORV.OPEN.DUE. TO.HP1 PORV.CLOSURE HPR/-HP1	CD	7.18-08	5.68-01
106 107	SLB -RT REQ.SG.ISO -AFM -HP1 REQ.BA.ADDITION SLB -RT REQ.SG.ISO -AFM HP1	CD CD	3.0E-07 + 1.9E-07	

dominant sequence for end state ## 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.

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c:\asp\newmodel\pwrbmslb.txt

MODEL: DATA:

6

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No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
SLB	1.1E-07	1.06+00	
RT	2.56-04	1.2E-01	
RE9.58.150	6.4E-04 > 1.0E+00	1.0E+00	
Branch Model: 1.0F.1			
Train 1 Cond Prob:	6.4E-04 > Failed		
AFM	1.0E-03	2.7E-01	
HP1	1.0E-03	5.2E-01	
HP1(F/B)	1.0E-03	5.2E-01	4.0E-02
HPR/-HP1	3.0E-03	5.6E-01	4.0E-02
PORV. OPEN	1.0E-02	1.0E+00	
REQ. BA. ADDITION	8.3E-04	1.0E+00	
PORV. OPEN. DUE. TO. HPI	8.0E-01	1.0E+00	
PORV. CLOSURE	6.0E-03	1.02+00	

+++ forced

Austin 09-11-1987 13:48:35

LER No.:	318/86-006
Event Description:	Trip occurs, and one atmospheric dump valve
Date of Event: Plant:	fails to close September 5, 1986 Calvert Cliffs 2

EVENT DESCRIPTION

Sequence

While the reactor was at 100% power, a surge capacitor in the 21A RCP failed and shorted to ground. The RCP tripped off, which led to an automatic reactor trip on low flow.

The cooldown rate was faster than expected because an atmospheric steam dump valve was stuck open. It was closed manually after 22 min.

Corrective Action

The solenoid valve associated with the atmospheric dump valve was found to be leaking high-pressure air by its seats. The solenoid valve materials were replaced.

Plant/Event Data

Systems Involved: RCS and atmospheric steam dump

Components and Failure Modes involved: RCP — failed in operation Atmospheric steam dump value — failed to close on demand

Component Unavailability Duration: NA Plant Operating Mode: 1 (100% power) Discovery Method: Operational event Reactor Age: 9.8 years Plant Type: PWR

Comments

None

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Transient 1.0 Nonrecoverable

Branches Impacted and Branch Nonrecovery Estimate

SS release 1.0 Atmospheric dump valve failed to close terminated from control room and is not closed locally for 22 min

Plant Models Utilized

PWR plant Class G

CONCITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

1.0E+00

Event Identifier: 316/86-006 Event Description: Trip and One ASD Valve Fails to Close Event Date: 9/5/86 Plant: Calvert Cliffs 2

INITIATING EVENT

TRANS

NOW-RECOVERABLE INITIATING EVENT PROBABILITIES

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator Probability CV TRANS 2.5E-04 Total 2.58-04 CD TRANS 1.8E-06 Total 1.88-06 ATWS TRANS 3.48-05 Total 3.4E-05

DOMINANT SEQUENCES

End State: CV Conditional Probability: 2.4E-04 104 TRANS -RT -AFW -PORV.OR.SRV.CHALL SS.RELEAS.TERM HPI End State: CD Cunditional Probability: 1.6E-06 102 TRANS -RT -AFW PORV.OR.SRV.CHALL PORV.OR.SRV.RESEAT -HPI HPR/-HPI End State: ATWS Conditional Probability: 3.4E-05

121 TRANS RT

SEQUENCE CONDITIONAL PROBABILITIES

	Sequence	End State	Prob	N Rec++
101 TRANS -RT -AFW	PORV. OR. SRV. CHALL -PORV. OR. SRV. RET AT SS. RELE	CV	1.0E-05	8.4E-01
AS, TERM HP1	FORV. OR. SRV. CHALL PORV. OR. SRV. RES" HPI HP	CD	1.6E-06 *	5.0E-02
R/-HPI 104 TRANS -RT -AFW 116 TRANS -RT AFW	-PORV.OR.SRV.CHALL SS.RELEAS.TERM HPI MFW -HPI(F/B) HPR/-HPI -SS.DEPRESS COND/MFW MFW HPI(F/B) -SS.DEPRESS COND/MFW	CV CD CD ATWS	2.4E-04 * 8.5E-00 8.9E-08 3.4E-05 *	8.4E-01 3.0E-02 2.5E-02 1.2E-01

dominant sequence for end state ## non-recovery credit for edited case

SEQUENCE MODEL:	c:\asp\newmodel\pwrgtree.cmp
BRANCH MODEL:	c:\asp\newmodel\calvert.txt
PROBABILITY FILE:	c:\asp\newsodel\pur_b.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	4.8E-04	1.0E+00	
	4.62-06	3.9E-1	
LOOP	2.4E-06	4.3E-01	
LOCA	2.8E-04	1,2E-01	
RT		1.0E+00	
RT/LOOP	0.0E+00	8.0E-01	
EMERB. POWER	5.4E-04		
AFW	3.8E-04	2.6E-01	
AFW/EMERS. POWER	5.0E-02	3.4E-01	
NFW	2.0E-01	3.4E-01	
PORV. DR. SRV. CHALL	4,0E-02	1.0E+00	
PORV. OR, SRV. RESEAT	2.0E-02	5.0E-02	
	2.0E-02	1.05+00	
PORV. OR. SAV. RESEAT / EMERG. POWER	1.5E-02 > 1.0E+00	3,4E-01 > 1.0E+00	
SS. RELEAS. TERA	1.55-02 / 1.02.00		
Branch Model: 1.DF.1			
Train 1 Cond Prob:	1.5E-02 > Failed	1 45-01	
SS. RELEAS. TERM / - MFW	1.5E-02	3.4E-01	
SS. DEPRESS	3.6E-02	1.0E+00	
COND/MFW	1.0E+00	3.4E-01	
HPI	3.0E-04	8.4E-01	
	3.0E-04	8.4E-01	4.0E-02
HP1(F/B)	1.05-02	1.0E+00	
PORV. OPEN	1108 04		

HPR/-HPI CSR	1.5E-04	1.0E+00
	2.0E-03	3.4E-01

* branch model file
** forced

Austin 09-11-1987 12:32:37

LER No.: 341/86-048 Event Description: RCIC and HPCI are unavailable Date of Event: December 24, 1986 Plant: Fermi 2

EVENT DESCRIPTION

Sequence

At 1015 h on December 26. 1986, Fermi 2 was operating at 920 psig, 530°F, and 8% reactor power. Setween 1248 h on December 24 and 1550 h on December 26, calibration activities for an RCIC system header flow instrument channel resulted in two occurrences of the RCIC system being declared inoperable. The second occurrence resulted in only slightly degraded flow capability. During these occurrences, the HPCI system was also inoperable. The HPCI system had been inoperable for a scheduled system outage since 1755 h on December 23, 1986.

Corrective Action

KCIC was recalibrated and restored to service

Plant/Event Data

Systems Involved: RCIC and HPCI

Components and Failure Modes Involved: RCIC — made unavailable HPCI — made unavailable

Component Unavailability Duration: 4.5 h Plant Operating Mode: 2 (5% power) Discovery Method: Testing Reactor Age: 1.5 years Plant Type: BWR

Comments

The reason for HPCI unavailability on December 26, 1986, is not given. The assumption is that it could not be readily recovered for duty.

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Postulated		Base	case	nonrecovery
Postulated		Base	case	nonrecovery
Postulated	LOCA	Base	case	nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

HPCI	1.0	No recovery assume	ed socials
RCIC		Recoverable in the	

Plant Models Utilized

BWR plant Class C

Event Identifier: 341/86-048 Event Description: HPC1/RCIC Are Unavail Event Date: 12/24/86 Plant: Fermi 2	able	
UNAVAILABILITY, DURATION= 4.5		
NON-RECOVERABLE INITIATING EVENT PROBAB	ILITIES	
TRANS LDOP LOCA		3.9E-03 2.4E-05 7.4E-06
SEQUENCE CONDITIONAL PROBABILITY SUMS		Developed a later
End State/Initiator		Probability
cv		
TRANS		7.2E-09
LOOP		9.3E-10
LOCA		7.8E-10
Total		8.9E-09
CD		
TRANS		4.9E-07
LOOP		5.85-08
LOCA		1.0E-07
Total		6.5E-07
ATWS		
*****		0.0E+00
TRANS LOOP		0.0E+00
LOCA		0.0E+00
Total		0,0E+00
DOMINANT SEQUENCES		
End State: CV	Conditional Probability:	2.4E-09

134 TRANS SCRAM -SLC.OR.RODS PCS/TRANS -SRV.CLOSE FW/PCS.TRANS HPCI RCIC/TRANS.OR.LOOP - SRV.ADS -COND/FW.PCS -RHR(SDC)

End State: CD

Conditional Probability: 4.6E-07

119 TRANS -SCRAM PCS/TRANS SRV.CHALL/TRANS.-SCRAM SRV.CLOSE FW/PCS.LOCA HPCI RCIC/LOCA SRV.ADS

SEQUENCE CONDITIONAL PROBABILITIES

	Sequence	End State	Prob	N Rec++
110	/PCS.TRANS HPCI RCIC/TRANS.OR.LOOP CRD SRV.ADS	CD	2.58-08	2.9E-02
119	TRANS -SCRAM PCS/TRANS SRV.CHALL/TRANSSCRAM SRV.CLOSE FW /PCS.LOCA HPCI RCIC/LOCA SRV.ADS	CD	4.6E-07 *	2.4E-01
134	TRANS SCRAM -SLC.OR.RODS FCS/TRANS -SRV.CLOSE FW/PCS.TRANS HPCI RCIC/TRANS.OR.LOOP -SRV.ADS -COND/FW.PCS -RHR(SDC)	CV	2.4E-09 *	2.7E-02
138	TRANS SCRAM -SLC.OR.RODS PCS/TRANS -SRV.CLOSE FW/PCS.TRANS HPCI RCIC/TRANS.OR.LOOP -SRV.ADS COND/FW.PCS -LPCS -RHR(SDC)	CV	1.2E-09	1.4E-02
155	TRANS SCRAM -SLC.OR.RODS PCS/TRANS SRV.CLOSE FW/PCS.LOCA HPCI RCIC/LOCA -SRV.ADS -COND/FW.PCS -RHR(SDC)	CV	2.3E-09	2.2E-01
159	TRANS SCRAM -SLC.OR.RODS PCS/TRANS SRV.CLOSE FW/PCS.LOCA HPCI RCIC/LOCA -SRV.ADS COND/FW.PCS -LPCS -RHR(SDC)	CV	1.2E-09	1.1E-01
215	LOOP -EMERG. POWER -SCRAM SRV. CHALL/LOOPSCRAM SRV. CLOSE HP CI RCIC/LOCA SRV. ADS	CD	5.1E-08	2.3E-01
226	LOOP -EMERG. POWER SCRAM -SLC.OR. RODS HPC1 RCIC/TRANS.OR.LOO P -SRV.ADS -LPCS -RHR (SDC)	CV	9.2E-10	3.8E-02
310	LOCA -SCRAM PCS/LOCA FW/PCS.LOCA HPCI RCIC/LOCA SRV.ADS	CD	1.0E-07	1.00.01
314	LOCA SCRAM -SLC.OR.RODS PCS/LOCA FW/PCS.LOCA HPCI -SRV.ADS -COND/FW.PCS -RHR(SDC)	CV	5.1E-10	1.2E-01 1.1E-01
318	LOCA SCRAM -SLC.OR.RODS PCS/LOCA FW/PCS.LOCA HPCI -SKV.ADS COND/FW.PCS -LPCS -RHR(SDC)	CV	2.6E-10	5.7E-02

* dominant sequence for end state

** 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. Farenthetical values indicate a reduction in risk compared to a similar period without the existing failures.

SEQUENCE MODEL :	c:\asp\newmodel\bwrctree.cmp
BRANCH MODEL :	c:\asp\newmodel\fermi.txt
PROBABILITY FILE:	c:\asp\newmodel\bwr c.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	8.6E-04	1.0E+00	
LOOP	1.7E-05	3.2E-01	
LOCA	3.3E-06	5.0E-01	
SCRAM	3.5E-04	1.0E+00	
SLC.OR.RODS	1.0E-02	1.0E+00	4.0E-02
PCS/TRANS	1.7E-01	1.0E+00	
PCS/LOCA	1.0E+00	1.0E+00	
SRV. CHALL / TRANS SCRAM	1.0E+00	1.0E+00	
SRV, CHALL/TRANS, SCRAM	1.0E+00	1.0E+00	
SRV. CHALL /LOOP SCRAM	1.0E+00	1.0E+00	
SRV.CHALL/LOOP.SCRAM	1.0E+00	1.0E+00	
SRV.CLOSE	5.0E-02	1.0E+00	
EMERG , POWER	3.0E-04	8.0E-01	
FW/PCS. TRANS	4.6E-01	3.4E-01	
FW/PCS.LOCA	1.0E+00	3.4E-01	
HPCI	2.9E-02 > 1.0E+00	7.0E-01 > 1.0E+00	
Branch Model: 1.0F.1			
	2.9E-02 > Unavailable		
RCIC/TRANS.OR.LOOP	6.0E-02 > 1.0E+00	7.0E-01 > 1.2E-01	
Branch Model: 1.0F.1			
Train 1 Cond Prob:	6.0E-02 > Unavailable		
RCIC/LOCA	1.0E+00	1.0E+00	1.00
CRD	1.0E-02	1.0E+00	4.0E-02
SRV.ADS	3.7E-03	7.1E-01	4.0€-02
COND/ -W . PCS	1.0E+00	3.4E-01	
LPCS	3.0E-03	3.4E-01	
LPCI (RHR) /LPCS	1.0E-03	7.1E-01	
RHRSW/LPCS.LPCI.TRANS	5.0E-01	1.0E+00	4.0E-02
RHRSW/LPCS.LPCI.LOOP	5.0E-01	1.0E+00	4.0E-02
RHRSW/LPCS.LPCI.LOCA	5.0E-01	1.0E+00	4.0E-02
RHR (SDC)	2.1E-02	3.4E-01	
RHR (SDC) /-LPCI	2.0E-02	3.4E-01	
RHR (SDC) /LPC1	1.0E+00	1.0E+00	
RHR (SPCOCL) /-LPC1.RHR (SDC)	2.0E-02	1.0E+00	
RHR (SPCOOL) /LPC1 .RHR (SDC)	5.2E-01	1.0E+00	
C. J. AND. V/RHR (SDC) . RHR (SPCODL)	1.0E+00	3.4E-01	

+ branch model file

** forced

Minarick 02-24-1988 12:11:30

PRECURSOR DESCRIPTION SHEET

LER No.: 362/86-011 Event Description: Saltwater and CCW systems are unavailable Date of Event: August 4, 1986 Plant: San Onofre 3

EVENT DESCRIPTION

Sequence

At 1550 h saltwater cooling flow through the train A CCW heat exchanger decreased as a result of fouling with marine growth. The fouling was below the postulated design-basis flow rate required for removal of CCW heat loads, and thus the exchanges was declared inoperable. At this time train B was operating with reverse saltwater cooling flow to remove similar fouling. Because there are only two saltwater cooling trains (two pumps per train), CCW water cooling flow was available. At 1605 h, operators commenced realignment of train B CCW heat-exchanger flow to the normal direction to return one train of CCW to its design configuration and thereby restore heat-removal capability of that train. Both trains of the saltwater cooling system were considered incperable until the realignment was complete.

Corrective Action

Train B saltwater cooling system was returned to operable status at 1635 h. Operating procedures will be revised to minimize the effect of marine fouling on the operability of the SWC system.

Plant/Event Data

Systems Involved: Essential raw cooling water system and CCW system

Components and Failure Modes Involved: CCW heat exchangers — degraded in operation

Component Unavailability Duration: 45 min Plant Operating Mode: 1 (100% power) Discovery Method: Operational event Reactor Age: 3.0 years Plant Type: PWR

Comments

The inoperable coding systems normally removed heat loads from the HPI, LPI, and CSR system pumps. These systems were assumed inoperable during unavailability of their cooling systems.

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Postulated	transient	Base	case	nonrecovery
Postulated	LOOP	Base	case	nonrecovery
Postulated	LOCA	Base	case	nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

HPI	0.34	One train of saltwater cooling re- coverable locally with realignment
LPI	0.34	One train of saltwater cooling re- coverable locally with realignment
CSR	0.34	One train of saltwater cooling re- coverable locally with realignment

Plant Models Utilized

PWR plant Class G

E	vent Identifier:	362/86-011	
		Salt and Component Cooling Water Syst	ess Are Unavailable
E	vent Date:	8/4/86	
P	lant:	San Onofre 3	
		DATION - 3F	
U	NAVAILABILITY, DU	KRIIUN= ./5	
N	ON-RECOVERABLE IN	ITIATING EVENT PROBABILITIES	
T	RANS		
	DOP		3.6E-04
	DCA		1.3E-06
			7.7E-07
SE	QUENCE CONDITION	AL PROBABILITY SUMS	
	End State/Ini	iator	Probability
1			
CV			
	TRANS		6.22-07
	LOOP		2.3E-09
	LOCA		(1.0E-12)
	Total		6.3E-07
CD			
LU			
	TRANS		4.7E-09
	LOOP		1.8E-11
	LOCA		2.5E-07
	Total		2.6E-07
ATM			
HIM	15		
	TRANS		
	LOOP		0.0E+00 0.0E+00
	LOCA		0.0E+00
	1122		0106400
	Total		0.0E+00
DOM	INANT SEQUENCES		

Event Identifier: 362/86-011

1

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24.10

104 TRANS -RT -AFM -PORV. DR. SRV. CHALL SS. RELEAS. TERM HPI

End State: CD Conditional Probability: 2.6E-07

302 LOCA -RT -AFN HPI

SEQUENCE CONDITIONAL PROBABILITIES

	Sequence		End State	Prob	N Rec**
		-PORV.OR.SRV.RESEAT SS.RELE	cv	2.58-08	1.2E-01
104 TRA	ERM HPI S -RT -AFM -PORV.OR.SRV.CHALL -RT -AFW HPI	SS.RELEAS.TERM HPI	CV CD	6.0E-07 * 2.6E-07 *	1.2E-01 1.5E-01

dominant sequence for end state ## 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:\asp\newmodel\PWRStree.cmp
BRANCH MODEL:	c:\asp\newsodel\SANONO2.txt
PROBABILITY FILE:	c:\asp\newmodel\pwr_b.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRAKS	4,82-04	1.0E+00	
LOOP	4.6E-06	3.9E-01	
LOCA	2.4E-05	4.3E-01	
RT	2.8E-04	1,2E-01	
RT/LOOP	0.0E+00	1.0E+00	
EMERG, POWER	2.9E-03	8,0E-01	
AFW	1.3E-03	2.6E-01	
AFW/EMERG. POWER	5,0E-02	3.4E-01	
MFW	2.0E-01	3,4E-01	
PORV. OR. SRV. CHALL	4.0E-02	1.0E+00	
	2.0E-02	5.0E-02	
PORV. OR. SRV. RESEAT	2.0E-02	1.0E+00	
PORV. OR. SRV. RESEAT / EMERG. POWER	1.5E-02	3.4E-01	
SS.RELEAS.TERM	1.5E-02	3.4E-01	
SS.RELEAS, TERM/-MFW		1.0E+00	
SS. DEPRESS	3.6E-02	3.4E-01	
COND/MFW	1.02+00	8,4E-01 > 3,4E-01	
HP1	3.0E-04 > 1.0E+00	0,45-01 / 0,45-01	

Branch Model: 1.0F.3			
Train 1 Cond Prob:	1.0E-02 > Unavailable		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Train 3 Cond Prob:	3.0E-01 > Unavailable		
HPI(F/B)	3.0E-04 > 1.0E+00	8.45-01 > 3.45-01	4.0E-02
Branch Model: 1.0F.3+opr			1106 02
Train 1 Cond Probs	1.0E-02 > Unavailable		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
Train 3 Cond Probs	3.0E-01 > Unavailable		
PORV, OPEN	1.0E+00	1.0E+00	
HPR/-HPI	1.5E-04	1.0E+00	
CSR	2.0E-03 > 1.0E+00	3.4E-01	
Branch Model: 1.0F.2		VITE VA	
Train 1 Cond Prob:	2.0E-02 > Unavailable		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
We want and a set of the set of the			
# branch andal file			

* branch model file
** forced

Austin 09-11-1987 17:35:49

PRECURSOR DESCRIPTION SHEET

LER No.: 3ú6/86-035 Event Description: LPCS system is unavailable Date of Event: November 13, 1986 Plant: Hatch 2

EVENT DESCRIPTION

Sequence

At 2319 h during testing of the LPCS system, personnel found the LPCS had been isolated. The pump electrical power had been removed, and the pump suction valves had been closed (both loops). The isolation was effected to carry out a different test [the Integrated Leak Rate Test (ILRT)], but the procedure for that test was in error.

Corrective Action

The procedure was revised.

Plant/Event Data

Systems Involved: LPCS

Components and Failure Modes Involved: LPCS pumps — made unavailable in testing LPCS valves — made unavailable in testing

Component Unavailability Duration: 12 h Plant Operating Mode: 5 (0% power) Discovery Method: Testing Reactor Age: 8.3 years Plant Type: BWR

Comments

None

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Postulated	transient	Base	case	nonrecovery
Postulated	LOCA	Base	case	nonrecovery
Postulated	LOOP	Base	case	nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

LPCS

.1

-

0.34 Recoverable locally at the equipment

Plant Models Utilized

BWR plant Class C

Event Identifier: 366/86-035

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Event Identifier: 366/86-035 Event Description: LPCS Is Unavailable Event Date: 11/13/86 Plant: Hatch 2		
UNAVAILABILITY, DURATION= 12		
NON-RECOVERABLE INITIATING EVENT PROBABIL	LITIES	
TRANS LOOP LOOP	6.5	E-02 E-05 E-05
SEQUENCE CONDITIONAL PROBABILITY SUMS		
End State/Initiator	Pro	bability
CV		
TRANS LOOP LOCA	(2.6	E-14) E-15) E-15)
Total	(1.5	5E-14)
CD		
TRANS LOOP LOCA	7.	5E-10 0E-12 4E-12
Total	з.	6E-10
ATWS		
TRANS LOOP LOCA	0.	0E+00 0E+00 0E+00
Total	0.	.0E+00
DOMINANT SEQUENCES		
End State: CV	Conditional Probability: 1	.6E-11

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Event Identifier: 366/86-035

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163 TRANS SCRAM -SLC.OR.RODS PCS/TRANS SRV.CLOSE FW/PCS.LOCA HPCI RCIC/LOCA -SRV.ADS C OND/FW.PCS LPCS -LPCI(RHR)/LPCS -RHR(SDC)/-LPCI

End State: CD Conditional Probability: 3.3E-10

116 TRANS -SCRAM PCS//RANS SRV.CHALL/TRANS.-SCRAM end up to the state of the sta

SEQUENCE CONDITIONAL PROBABILITIES

	Sequence	End State	Prob	N Rec**
116	TRANS -SCRAM PCS/TRANS SRV.CHALL/TRANSSCRAM SRV.CLOSE FW /PCS.LUCA HPCI RCIC/LOCA -SRV.ADS COND/FW.PCS LPCS -LP CI(RHR)/LPCS RHR(SDC)/LPCI RHR(SPCODL)/-LPCI.RHR(SDC) C .I.AND.V/RHR(SDC).RHR(SPCODL)	CD	3.3E-10 *	9.3E-03
118	TRANS -SCRAM PCS/TRANS SRV.CHALL/TRANSSCRAM SRV.CLOSE FW /PCS.LOCA HPCI RCIC/LOCA -SRV.ADS COND/FW.PCS LPCS LP CI(RHR)/LPCS RHRSW/LPCS.LPCI.TRANS	CD	1.8E-11	2.0E-02
142	TRANS SCRAM -SLC.OR.RDIG PCS/TRANS -SRV.CLOSE FW/PCS.TRANS HPCI RCIC/TRANS.OR.LOOP -SRV.ADS COND/FW.PCS LPCS -LPCI (RHR)/LPCS -RHR(SDC)/-LPCI	CV	8.2E-12	1.9E-02
163	TRANS SCRAM -SLC.OR.RODS PCS/TRANS SRV.CLOSE FW/PCS.LDCA HPCI RCIC/LOCA -SRV.ADS COND/FW.PCS LPCS -LPCI(RHR)/LPC S -RHR(SDC)/-LPCI	CV	1.6E-11 *	2.7E-02
230	LOOP -EMERG.POWER SCRAM -SLC.OR.RODS HPCI RCIC/TRANS.OR.LOO P -SRV.ADS LPCS -LPCI(RHR)/LPCS -RHR(SDC)/-LPCI	CV	6.0E-12	5.3E-02
322	LOCA SCRAM -SLC.OR.RODS PCS/LOCA FW/PCS.LOCA HPCI -SRV.ADS COND/FW.PCS LPCS -LPCI(RHR)/LPCS -RHR(SDC)/-LPCI	CV	5.0E-12	1.4E-02

* dominant sequence for end state

** 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:\asp\newmodel\bwrctree.cmp
BRANCH MODEL :	c:\asp\newmodel\hatch.txt
PROBABILITY FILE:	c:\asp\newmodel\bwr_c.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	8.6E-04	1.0E+00	

LOOP	1.7E-05	3.2E-01	
LOCA	3.3E-06	5.0E-01	
SCRAM	3.5E-04	1.0E+00	1.1.1.1
SLC.OR.RODS	1.0E-02	1.0E+00	4.0E-02
PCS/TRANS	1.7E-01	1.0E+00	
PCS/LOCA	1.0E+00	1.0E+00	
SRV.CHALL/TRANSSCRAM	1.0E+00	1.0E+00	
SRV.CHALL/TRANS.SCRAM	1.0E+00	1.0E+00	
SRV.CHALL/LOOPSCRAM	1.0E+00	1.0E+00	
SRV.CHALL/LOOP.SCRAM	1.0E+00	1.0E+00	
SRV.CLOSE	3.6E-02	1.0E+00	
EMERG . POWER	5.4E-04	8.0E-01	
FW/PCS.TRANS	4.6E-01	3.4E-01	
FW/PCS.LOCA	1.0E+00	3.4E-01	
HPCI	2.9E-02	7.0E-01	
RCIC/TRANS.OR.LOOP	6.0E-02	7.0E-01	
RCIC/LOCA	1.0E+00	1.0E+00	
CRD	1.0E-02	1.0E+00	4.0E-02
SRV ADS	3.7E-03	7.1E-01	4.0E-02
COND/FW.PCS	1.0E+00	3.4E-01	
LPCS	3.0E-03 > 1.0E+00	3.4E-01	
Branch Model: 1.0F.2	Dive ou / Tree ou		
Train 1 Cond Prob:	3.0E-02 > Unavailable		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
LPCI (RHR) /LPCS	1.0E-03	7.1E-01	
and the state of t	5.0E-01	1.0E+00	4.0E-02
RHRSW/LPCS.LPCI.TRANS	5.0E-01	1.0E+00	4,0E-02
RHRSW/LPCS.LPCI.LOOP	5.0E-01	1.0E+00	4.0E-02
RHRSW/LPCS.LPCI.LOCA	2.1E+02	3.4E-01	
RHR (SDC)		3.4E-01	
RHR (SDC) /-LPCI	2.0E-02	1.0E+00	
RHR (SDC) /LPCI	1.0E+00	1.0E+00	
RHR (SPCDOL) /-LPCI.RHR (SDC)	2.0E-02	1.0E+00	
RHR (SPCDOL) /LPCI.RHR (SDC)	5.2E-01	3.4E-01	
C.I.AND.V/RHR(SDC).RHR(SFCOOL)	1.0E+00	2.46-01	

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* branch model file

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Event Identifier: 366/86-035

PRECURSOR DESCRIPTION SHEET

LER No.:	370/86-006						
Event Description:	High-head injection is out of service	system	is	unavailable,	and	DG	A
Date of Event:	March 29, 1986						
Plant:	McGuire 2						

EVENT DESCRIPTION

Sequence

At 0315 h on March 19, 1986, DG 2-A was declared inoperable for maintenance repairs. Charging (SI) pump 2-A was also declared inoperable because DG 2-A could not provide emergency power to the pump. On March 28, 1986, at approximately 1100 h, a station engineer requested the responsible assistant shift supervisor to rack in and operate charging pump 2-A so a retest could be performed on charging pump 2-A. The engineer made the request without realizing that DG 2-A was inoperable. The assistant shift supervisor instructed station personnel to rack out charging pump 2-B and rack in charging pump 2-A. The assistant shift supervisor did not realize that DG 2-A was inoperable and that racking out charging pump 2-B would result in a loss of boration flow path. The assistant shift supervisor did not discuss the change with the designated control room senior reactor operator to ensure Technical Specifications requirements were mct.

At 1245 h, the retest on charging pump 2-A was completed.

On March 29, 1986, at 0700 h, during shift turnover, station personnel discovered that while charging pump 2-B had been racked out of service, charging pump 2-A had been racked in service without an emergency power supply. Immediately charging pump 2-A was racked out and charging pump 2-B was racked in, reestablishing compliance with Technical Specifications.

Corrective Action

Charging pump 2-B was placed in service.

Plant/Event Data

Systems Involved: High-head injection, emergency power

Components and Failure Modes Involved: DG 2-A - was out for maintenance Charging pump 2-B - was removed from service

Event Identifier: 370/86-006

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D-116

Component Unavailability Duration: 20 h Plant Operating Mode: 6 (refueling) Discovery Method: Testing Reactor Age: 2.9 years Plant Type: PWR

Comments

None

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Postulated LOOP Base case nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

HPI	Base case	Both high-head injection trains would be unavailable given a LOOP
EPS	Base case	DG A was out of service
AFW	Base case	One train was unavailable because DG A was unavailable
LPI	Base case	One train was unavailable because DG A was unavailable
LPR	Base case	One train was unavailable because DG A was unavailable
HPR	Base case	One train was unavailable since DG A was unavailable

Plant Models Utilized

PWR plant Class F

3.6E-05

Event Identifier: 370/86-006 Event Description: HHIS Is Unavailable and D6 'A' Is Out of Service Event Date: 3/29/86 Plant: McBuire 2

UNAVAILABILITY, DURATION= 20

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

LOOP

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator Probability CV LOOP 1.4E-08 Total 1.4E-08 00 LOOP 3.4E-08 Total 3.4E-08 ATWS LOOP 0.0E+00 Total 0.0E+00

DOMINANT SEQUENCES

End State: CV Conditional Probability: 6.5E-09 217 LOOP -RT/LOOP EMERS.POWER -AFW/EMERS.POWER -PORV.OR.SRV.CHALL SS.RELEAS.TERM End State: CD Conditional Probability: 2.3E-08 218 LOOP -RT/LOOP EMERS.POWER AFW/EMERS.POWER SEQUENCE CONDITIONAL PROBABILITIES

	1.1E-01
202 LUUP -KI/LUUP -EREKD.FUMER - AFM FURT. OK. SAT. UNALE FURT. SAT.	
V.RESEAT SS.RELEAS.TERM HPI SS.DEPRESS 206 LOOP -RT/LOOP -EMERG.POWER -AFW PORV.DR.SRV.CHALL PORV.DR.SR CV 1.6E-09	1.6E-02
V.RESEAT HPI -SS.DEPRESS -LPI/HPI -LPR/HPI 210 LOOP -RT/LOOP -EMERG.POWER -AFW -PORV.OR.SRV.CHALL SS.RELEAS. CV 4.6E-10	3.8E-02
TERM HP1 -SS. DEPRESS LP1/HP1	1.1E-01
TERM HP1 SS. DEPRESS	8.4E-02
215 LOOP -RT/LOOP EMERG. POWER -AFW/EMERG. POWER PORV. OR. SRV. CHALL CV 2.6P-10	1.0E-01
218 LUUP -KI/LUUP ENERG. FORCH - HER/ENERG. FORCE FORCE OF THE STORE	3.1E-01
PORV.OR.SRV.RESEAT/EMERG.POWER 217 LOOP -RT/LOOP EMERG.POWER -AFW/EMERG.POWER -PORV.OR.SRV.CHALL CV 6.5E-09 *	1.0E-01
SS.RELEAS.TERM 218 LOOP -RT/LOOP EMERG.POWER AFW/EMERG.POWER CD 2.3E-08 *	1.1E-01

+ dominant sequence for end state

** 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.

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SEQUENCE MODEL:	c:\asp\newmodel\pwrbtree.cmp
BRANCH MODEL:	c:\asp\newmodel\mcguire.txt
PROBABILITY FILE:	c:\asp\newmodel\pwr_b.pro

No Recovery Lisit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	
TRANS	4.8E-04	1.0E+00	
LOOP	4.6E-06	3,9E-01	
LOCA	2.4E-06	4.3E-01	
RT	2.85-04	1.2E-01	
RT/LOOP	0.0E+00	1.0E+00	
EMERB. POWER	2.9E-03 > 5.0E-02	8.0E-01	
Branch Model: 1.0F.2			
Train 1 Cond Prob:	5.0E-02		
Train 2 Cond Probs	5.7E-02 > Unavailable		
AFM	3,8E-04 > 1.3E-03	2.6E-01	
Branch Model: 1.0F.3+s	ser		
Train 1 Cond Prob:	2.02-02		
Train 2 Cond Prob:	1.0E-01 > Unavailable		

Train 3 Cond Prob:	5.0E-02		
Serial Component Prob:	2.8E-04		
AFW/EMERB. POWER		3.4E-01	
MFW	2.0E-01	3.4E-01	
PORV, OR, SRV, CHALL	4.0E-02	1.0E+00	
	3.0E-02	5.0E-02	
PORV. DR. SRV. RESEAT / EMERS. POWER		1.0E+00	
	1.5E-02	3.4E-01	
SS. RELEAS. TERM SS. RELEAS. TERM/-MFW	1.5E-02	3.48-01	
HP1	1.0E-03 > 1.0E+00	8,4E-01	
Branch Model: 1.0F.2		0,45-01	
Train 1 Cond Prob:	1.0E-02 > Unavailable		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
HP1(F/B)	1.0E-03 > 1.0E+00	8.4E-01	1 45 45
Branch Model: 1.0F.2+opr		0.46-01	4.0E-02
Train 1 Cond Prob:	1.0E-02) Unavailable		
Train 2 Cond Prob:	1.0E-01 > Unavailable		
HPR/-HP1	1.5E-04 > 1.0E-02	1.0E+00	1 05-00
Branch Model: 1.0F.2+opr	1100 04 7 1100 02	1.05700	4.0E-02
Train 1 Cond Prob:	1.0E-02		
	1.5E-02 > Unavailable		
PORV. OPEN	1.0E-02	1.0E+00	
SS. DEPRESS	3.6E-02	1.0E+00	
COND/MFW	1.0E+00	3.4E-01	
LP1/HP1	1.5E-04 > 1.0E-02	3.4E-01	
Branch Model: 1.0F.2	1100-04 / 1100-02	2145-01	
Train 1 Cond Prob:	1.0E-02		
	1.5E-02 > Unavailable		
LPR/-HP1.NPR	6.7E-01	1.0E+00	
LPR/HP1	1.5E-04 > 1.0E-02		
Branch Model: 1.0F.2	1.02-04 / 1.02-02	1.0E+00	
Train 1 Cond Prob:	1.05-12		
Train 2 Cond Probi			
	1.5E-02 > Unavailable		
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Austin 09-11-1987 12:45:01

PRECURSOR DESCRIPTION SHEET

LER No.: 389/86-011 Event Description: Emergency power system is unavailable Date of Event: July 9, 1986 Plant: St. Lucie 2

EVENT DESCRIPTION

Sequence

At 0854 h the 2-A emergency DG was started for a surveillance test conducted once every 7 d. The 2-A DG failed to meet the required generator voltage and frequency within 10 s after the start signal. One of the two engines in the 2-A DG set had failed to start. The 2-A DG was manually tripped by the operator at 0856 h.

At 0915 h the redundant 2-B DG was started. At 0917 h, 2-B DG was stopped because an operator observed 1 of the 12 cylinder cooling fan blades rubbing the cooling fan shroud.

The unit remained at 100% power throughout this event.

Corrective Action

2-B DG was repaired and returned to service at 1059 h. Troubleshooting of the 2-A DG revealed a problem in the mechanical portion of the Woodward governor. The problem was corrected, and 2-A DG was returned to service at 2010 h.

Plant/Event Data

Systems Involved: Emergency power generation system

Components and Failure Modes Involved: 2A DG — failed in testing 2B DG — failed in testing

Component Unavailability Duration: 84 h assumed (half of the 7-d surveillance period) Plant Operating Mode: 1 (100% power) Discovery Method: Surveillance test Reactor Age: 3.1 years Plant Type: PWR

Comments

LER 389/86-011 states that the fan rub associated with DG 2-B was minor. If the fan rub was minor in the sense that even with the fan rub, the DG would have operated for its mission time, then the emergency power system unavailability should be based on the maintenance unavailability period (~1 3/4 hours). However, if the fan rub was minor in the sense that it was easily repaired, but with the rub the DG would not have operated for its mission time, then an emergency power system unavailability based on the surveillance period is warranted. Since it is not possible to distinguish between these two cases based on information provided in the LER, a more conservative estimate based on surveillance internal was assumed.

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Postulated LOOP

Base case nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

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EPS

Given a LOOP, recovery assumed not possible in time to mitigate the transient (see comment above)

Plant Models Utilized

PWR plant Class G

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Event Identifier: 389/80 Event Description: Emerge	6-011 ency Power System Is Unavailable	
Event Date: 7/9/8	6	
Plant: St Lui	cie 2	
UNAVAILABILITY, DURATION	= 84	
NON-RECOVERABLE INITIATI	NG EVENT PROBABILITIES	
LOOP		1.5E-04
SEQUENCE CONDITIONAL PRO	BABILITY SUMS	
End State/Initiator		Probability
CV		
LOOP		7.5E-07
Total		7.5E-07
CD		
LOOP		2.68-06
Total		2.6E-06
ATWS		
LOOP		0.0E+00
Total		0.0E+00
DOMINANT SEQUENCES		
End State: CV	Conditional Probability:	7.4E-07
	ERG.POWER -AFW/EMERG.POWER -PORV.OR.SRV.CH	ALL SS.RELEAS.TERM
End State: CD	Conditional Probability:	2.68-06
212 LOOP -RT/LOOP EM	KERG.POWER AFW/EMERG.POWER	
SEQUENCE CONDITIONAL PR	ROBABILITIES	

Event Identifier: 389/86-011

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	Seq. ence	End State	Prob	N Rec++
211	LOOP -RT/LOOP EMERG.POWER -AFW/EMERG.POWER -PORV.OR.SRV.CHALL SS.RELEAS.TERM	CV	7.4E-07 +	1.3E-01
212	LOOP -RT/LOOP EMERG.POWER AFW/EMERG.POWER	CD	2.6E-06 +	1.3E-01

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dominant sequence for end state ## 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:\asp\newmodel\pwrgtree.cmp BRANCH MODEL: c:\asp\newmodel\lucie.txt PROBABILITY FILE: c:\asp\newmodel\pwr_b.pro

No Recovery Limit

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BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	4.85-04	1.0E+00	
LOOP	4,68-06	3.9E-01	
LOCA	2.4E-06	4.3E-01	
RT	2.8E-04	1.2E-01	
RT/LOOP	0.02+00	1.0E+00	
EMERG, POWER	2.9E-03 > 1.0E+00		
Branch Model: 1.0F.2		8.0E-01 > 1.0E+00	
Train 1 Cond Proti	5.0E-02 > Unavailable		
Train 2 Cond Prob:	5.7E-02 > Unavailable		
AFW	3.8E-04	5.15.41	
AFW/EMERB, POWER	5.0E-02	2.6E-01	
NFW	1.9E-01	3.4E-01	
PORV. OR. SRV. CHALL	2.0E-02	3.4E-01	
PORV. OR. SRV, RESEAT	1.08-02	1.0E-00	
PORV. OR . SRV. RESEAT / EMERS. POWER		5. 9E-02	
SS.RELEAS. TERM	1.0E-02	1.0E+00	
SS. RELEAS, TERM/-MFW	1.5E-02	3.4E-01	
SS.DEPRESS	1.5E-02	3.4E-01	
	3.68-02	1.0E+00	
COND/MFW	1.0E+00	3,4E-01	
HPI	3.0E-04	8.4E-01	
HP1(F/B)	3.0E-04	8.4E-01	4.0E-02
PORV. OPEN	1.0E+00	1.0E+00	
HPR/-HPI	1.5E-04	1.0E+00	
CSR	2.0E-03	3.4E-01	

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Event Identifier: 389/86-011

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PRECURSOR DESCRIPTION SHEET

LER No.: 409/86-023 Event Description: LOOP occurs due to lightning strike at coal-fired unit Date of Event: July 10, 1986 Flant: LaCrosse

EVENT DESCRIPTION

Sequence

With the plant in the cold shutdown condition, at 0630 h, the following opened: the 69-kV tie line breaker, 2NB11; the reverse transformer supply breaker, 2SNB4; and the 480-V main feed breakers, 452 MIA and 452 MIB. Both EDGs started and supplied the 1-A and 1-B 480-V essential buses. The containment building also isolated. The 1-A high-pressure service-water diesel pump started when service-water pressure cropped to the low-pressure set point.

A severe thunderstorm was in progress at the time. The operations center informed the plant that the adjacent coal plant's auxiliary switchyard had been hit by lightning. The operators reset the lockout relays. At 0642 h they closed breakers 2NB11 and 25NB4; at 0645 h they shut the 480-V main feed breakers and opened the EDG's output breakers. The 480-V essential bus breakers automatically closed, completing the electrical lineup restoration. At 0659 h the 1-A highpressure service-water DG was secured and returned to automatic. At 0702 h the EDGs were secured and returned to automatic.

An unusual event was declared due to the loss of offsite power. The plant was without offsite power for 12 min. The lightning had hit a static wire in the adjacent coal plant's auxiliary switchyard. The 69-kV tie line supplies power to LaCrosse's reserve transformer and the coal unit's reserve auxiliary transformer. When the lightning struck, the 69-kV tie line breaker and both transformers' supply breakers opened. The LaCrosse 480-V main feed breakers tripped on undervoltage. Undervoltage also started the EDGs and tripped the 480-V essential bus main feed breakers, which caused the emergency DGs' output breakers to close to supply the essential buses. If this incident had occurred while the reactor was at 100% power, a loss of load transient would have resulted. This is a design-basis transient, and the plant would still have been in a safe condition.

Corrective Action

The breakers were closed.

Plant/Event Data

Systems Involved: Electrical

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Components and Failure Modes Involved: Breakers — failed open in operation

Component Unavailability Duration: NA Plant Operating Mode: 4 (0% power) Discovery Method: Operational event Reactor Age: 17.6 years Plant Type: BWR

Comments

None

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

LOOP

Base case nonrecovery multiplied by 0.75 for occurring at power

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Branches Impacted and Branch Nonrecovery Estimate

None

Plant Models Utilized

Unique BWR class

Event Identifier: 409/86-023 Event Description: LOOP Due to Lightning Strike at Coal-Fired Unit Event Date: 7/10/86 Plant: LaCrosse

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES LOOP 2.7E-01 SEQUENCE CONDITIONAL PROBABILITY SUMS End State/Initiator Probability CD LOCP 2.0E-05 Total 2.0E-05 CV LOOP 0.0E+00 Total 0.0E+00 ATWS LOOP 5.3E-06 Total 5.3E-06

DOMINANT SEQUENCES

End State: CD Conditional Probability: 6.7E-06 205 LUOP -EMERG.POWER -SCRAM HPCS HPSN End State: ATWS Conditional Probability: 4.7E-06 206 LOOP -EMERG.POWER SCRAM -HPCS SLC.OR.RODS SEQUENCE CONDITIONAL PROBABILITIES

	Sequence	End State	Prob	N Rec++
202 LOOP -EMERG. POWER -SC	RAM HPUS -HPSW -SRV.CLUBE SHU	TOOWN.COND CD	1.6E-06	2.7E-01
FUSER MANUAL DEPRESS				
204 LOOP -EMERS. POWER -SC	RAM HPCS - HPSW SRV. CLOSE MAN	JAL. DEPRES CD	1.62-06	2.78-01
8		CD	6.7E-06 +	2.78-01
205 LOOP -EMERS. POWER -SC		ATWS	4.78-06 +	2.7E-01
206 LOOP -EMERS. POWER SC		ATWS	4.7E-07	2.7E-01
207 LOOP -EMERG. POWER SC	RAM HPCE		7.38-07	2.28-01
	RAM -HPSN -SRV.CLOSE SHUTDOWN.	LUNDENDEN UV		
MANUAL . DEPRESS		CD	6.1E-06	2.2E-01
210 LOOP EMERG. POWER -SC		CD	3.1E-06	2.2E-01
211 LOOP EMERG. POWER -SC		ATWS	2.2E-07	2.2E-01
212 LOOP EMERG, POWER SC	AHD			
· dominant sequence for en	d state			
** non-recovery credit for	edited case			
	\newmodel\lactree.cmp			
	\newmodel\lacrosse.txt			
PROBABILITY FILE: C:\asp	\newmodel\bwr_c.pro			
No Recovery Limit				
BRANCH FREQUENCIES/PROBABIL	ITIES			
Branch	System	Non-Recov	Opr Feil	
	e de l'architecture			
TRANS	8.62-04	1.0E+00 3.6E-01 > 2.7E-01		
LOOP	2.0E-05 > 2.0E-05	2.05-01 / 2./5-01		
Branch Model: INITOR				
Initiator Free:	2.0E-05	5.0E-01		
LOCA	3.3E-06	1.0E+00		
SCRAM	3.5E-04	1.0E+00	4.0E-02	
SLC. DR. RODS	1.0E-02	8,0E-01		
EMERG. POWER	2.9E-03			
FM	1.0E-01	3.4E-01		
HPSN	5.0E-03	1.0E+00		
SHUTDOWN, CONDENSER	1.0E-02	1.0E+00		
HPCS	5.0E-03	1.0E+00		
LPCS	1.0E-03	1.0E+00		
SRV. CLOSE	1.0E-02	1.0E+00		
DH	1.0E-02	1.0E+00		
MANUAL . DEPRESS	1.2E-01	1.0E+00		
· branch endel file				

+ branch model file

++ forced

PRECURSOR DESCRIPTION SHEET

LER No.: 413/86-031 Event Description: Small LOCA forces plant trip Date of Event: June 13, 1986 Plant: Catawba 1

EVENT DESCRIPTION

Sequence

The unit was at 48% power with the variable letdown orifice valve INV849 in service to reduce letdown flow to 30 gal/min. A leak (>1 gal/min) had been detected at the CCW/CVCS junction at the letdown heat exchanger. The fixed orifice flow paths were isolated. At 1100 h a leak of >1.5 gal/min was detected. At 1500 h an unusual event was declared. At 1542 h, alarms occurred indicating the loss of motor control center IMXD, which affected control power to valve INV849 and the generator hydrogen-cooler-temperature valve. The former failed open, and the latter failed closed.

Charging flow suddenly increased to 130 gal/min, and pressurizer level began to fall. The letdown line had suffered a guillotine rupture at valve INV849's downstream outlet flange as a result of vibrationinduced fatigue. At 1550 h the main-generator hydrogen temperature began to rise. At 1551 h the letdown orifice valve was closed, but pressurizer level continued to decrease. Hydrogen temperature continued to increase. Sump high-level alarms were actuated, and at 1610 h reactor power and turbine load were reduced. Maximum charging was maintaining pressurizer level, and additional letdown isolation valves were closed. The leak was contained by 1641 h. Hot standby was entered at 1700 h. Cold shutdown was entered at 0257 h the next day.

Corrective Action

Repairs were made to the letdown line. The MCC failure was due to a name plate that became unglued and caused a short circuit when it fell.

Plant/Event Data

Systems Involved: Electrical and chemical volume and control

Components and Failure Modes Involved: Motor control center transformer — failed in operation CVCS pipe — ruptured in operation

Component Unavailability Duration: NA Plant Operating Mode: 1 (48% power) Discovery Method: Operational event Reactor Age: 1.4 years Plant Type: PWR

Comments

We are assuming a 130 gpm SBLOCA since CP's were able to maintain pressurizer level with this flowrate

MODELING CONSIDERATIONS AND DECISIONS

Initiators M	Modeled and Initiator				
SBLOCA	0.12	Flow isolable valves	via	letdown	isolation
Branches Im	pacted and Branch Nor	recovery Estimat	e		

None

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Plant Models Utilized

PWR plant Class F

6

Event Identifier: 413/86-031

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Event Identifier: 413/86-031 Event Description: Small LOCA Forces Plant Trip Event Date: 6/13/86 Plant: Catamba 1

INITIATING EVENT

NON-RECOVERABLE INITIATING EVENT PROBABILITIES	
LOCA	1.2E-01
SEQUENCE CONDITIONAL PROBABILITY SUMS	
End State/Initiator	Probability
CV	
LOCA	1.6E-03
Total	1.6E-03
CD	
LOCA	3.3E-03
Total	3.3E-03
ATWS	
LDCA	4.0E-06
Totel	4.0E-06

DOMINANT SEQUENCES

10.1		-		
Eng	State:	CV	Conditional Probability:	1.5E-03
301	LOCA	-RT -AFM -HPI	HPR/-HPI -SS.DEPRESS -LPR/-HPI.HPR	
End	State:	CD	Conditional Probability:	3.1E-03
302	LOCA	-RT -AFM -HPI	HPR/-HPI -SS.DEPRESS LPR/-HPI.HPR	
End	State:	ATWS	Conditional Probability:	4.0E-06

326 LOCA RT

SEQUENCE CONDITIONAL PROBABILITIES

Sequence	End State	Prob	N Hectt
LOCA -RT -AFW -HPI HPR/-HPI -SS.DEPRESS -LPR/-HPI.HPR	CV	1.5E-03 *	1.2E-01
LOCA -RT -AFW -HPI HPR/-KPI -SS.DEPRESS LPR/-HPI.HPR	CD	3.1E-03 *	1.2E-01
LOCA -RT -AFW -HPI HPR/-HPI SS.DEPRESS	CD	1.7E-04	1.2E-01
LOCA -RT -AFW HPI -SS.DEPRESS -LPI/HPI -LPR/HPI	CV	9.7E-05	1.0E-01
LOCA RT	ATWS	4.0E-06 *	1.4E-02

dominant sequence for end state ** non-recovery credit for edited case

SEQUENCE MODEL:	c:\asp\news/del\pwrbtree.cmp
BRANCH MODEL:	c:\asp\newsodel\cataw.txt
PROBABILITY FILE:	c:\asp\newmodel\pwr_b.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	4.8E 04	1.0E+00	
LOOP	4.6E-06	3.9E-01	
LOCA	2.4E-06 > 2.4E-06	4.3E-01 > 1.2E-01	
Branch Model: INITOR			
Initiator Freq:	2.4E-06		
	2.8E-04	1,2E-01	
RT	0.0E+00	1.0E+00	
RT/LOOP	2.9E-03	8.0E-01	
EMERB. POWER	3.8E-04	2.6E-01	
AFW		3.4E-01	
SS. RELEAS. TERM			
SS. RELEAS. TERM / - MFW			
HP1			4 05-02
HP1(F/B)	1.0E-03		
HPR/-HP1	1.5E-04		4.05-02
	1.0E-02		
a state a second s	3.6E-02	1.0E+00	
	1.0E+00	3.42-01	
		3.4E-01	
LPR/-HP1.HPR	6.7E-01	1.0E+00	
HP1 HP1(F/B) HPR/-HP1 PORV.OPEN SS.DEPRESS COND/MFM LP1/HP1	1.0E-02 3.6E-02 1.0E+00 1.5E-04	3.4E-01 1.0E+00 5.0E-02 1.0E+00 3.4E-01 8.4E-01 8.4E-01 1.0E+00 1.0E+00 1.0E+00 3.4E-01 3.4E-01 3.4E-01 3.4E-01	4.0E-02 4.0E-02

LPR/HP1

1.0E+00

* branch model file
** forced

Austin 09-11-1987 12:58:13

PRECURSOR DESCRIPTION SHEET

LER No.: Event Description:	414/86-028 SG PORVs open inadvertently other failures occurs	in	test,	and	trip	with
	June 27, 1986 Catawba 2					

EVENT DESCRIPTION

Sequence

During a loss of control room function test from 24% power, an unexpected plant transient, SG depressurization, and reactor trip occurred as a result of test procedure errors. The procedure provides reactor guidelines to demonstrate, principally

- that the plant can be brought to hot standby-conditions from a moderate power level (10-25%) using only the auxiliary shutdown panel controls,
- that the plant can be maintained at hot standby conditions for 30 min from the auxiliary shutdown panels, and
- that the RCS can be cooled down at least 50°F from a steady state hot standby condition while being operated from the auxiliary shutdown panel controls.

In accordance with the test procedure, the reactor was manually tripped at the reactor trip switchgear at 0942 h. MFW isolation and the autostart of both motor-driven AFW pumps occurred 12 s later. Low-low levels subsequently occurred in all four SGs. The AFW pump turbine automatically started on low-low level in two out of four SGs. MFW pump 2-B later tripped at 0942:42 h on low suction flow.

Unit control was transferred from the control room to the auxiliary shutdown panel at 0942:49 h. The letdown pressure control valve, 2NV-148A, unexpectedly failed open when the transfer occurred. Letdown flow indication began to oscillate rapidly. Charging flow spiked to a maximum of 178 gal/min at approximately 0946:30 h. Letdown was manually isolated after pressurizer level dropped to <20%. Letdown flow dropped to ~15 gal/min by 0947:30 h.

At 0946:59 h, the SG PORV breakers at the AFW turbine control panel were closed in accordance with the procedure. When the breakers were energized, SG A, B, C, and D PORVs opened to 75%. This was a result of the SG PORV manual loaders being initially set to what was thought to be the 1125 psig opening set point. A design change had modified the SG PORV controls, but the modification had not been adequately understood.

The SG PORV opening caused a rapid depressurization of the secondary side with an accompanying cooldown of the primary side. Personnel observed the decreasing steam pressure and attempted to increase the set point for SG PORV opening, but they actually opened the PORVs further. Personnel in the control room observed the actual SG PORV positions go Open, but did not immediately communicate this to personnel at the auxiliary shutdown panel because of the nature of the test. SG levels responded to the SG PORV openings by first swelling and then dropping rapidly off the narrow range scale. The auxiliary shutdown panel operators were observing wide range indication. The AFW turbine had been secured at 0945:45 h. For ~4.5 min the SGs were. blowing down through the SG PORVs, with AFW flow being provided to SG D.

Pressurizer pressure dropped off scale (<1700 psig) ~2 min after the SG PORVs opened. SI condition on low pressurizer pressure (1845 psig) occurred at 0949:46 h. SI condition on low steam-line pressure loop D (725 psig) occurred at 0950:08 h. However, SI was the auxiliary shutdown panel operators. Several containment isolation valves closed automatically, and charging suction was automatically satisfied.

As pressurizer level continued to decrease, personnel at the auxiliary shutdown panel manually started centrifugal charging pump 2-B. However, because of valve controller labeling problems, operators as the auxiliary shutdown panel reduced charging flow rather than increasing it while adjusting the manual loader for 2NV-294, charging pumps flow control valve.

At approximately 0953:30 h, the decision was made to terminate the test and return control to the control room. At 0953:14 h, the senior reactor operator directed personnel to swap control back to the control room. When this was done, SI was immediately actuated due to the unblocking of the still-present actuation signal. Both DGs actuated on LOCA condition. The SG PORVs reclosed on transfer of controls. The SI signal started the RHR pumps, SI pumps, and the AFW turbine pump and opened volume-control-pump discharge to cold-leg isolation valves 2NI-9A and 2NI-10B and associated AFW valves. Valve 2NV-148A reclosed following the transfer.

Both DG load sequencers completed accelerated sequencing within ~ 21 s. SI flow restored pressurizer level to 33% and pressure to 1250 psig within ~ 5.5 min.

At 0958 h, SI was reset, the cold-leg-injection isolation values were closed, and the SI system and RHR pumps were secured. The SI had further reduced steam-line pressure to $\sim \!\!480$ psig and primary coolant temperature to $\sim \!\!468^\circ F$. The AFW was secured at 1000 h.

Corrective Action

A review of all design changes and construction department shutdown requests implemented after hot functional testing and before fuel load was performed prior to the unit reentering Mode 2, startup.

A review of both units' auxiliary shutdown panels and AFW pump turbine control panels was performed, and numerous unit differences and labeling problems were identified. Labeling problems were corrected. Revisions were made to operating and abnormal procedures. Also added were instructions to manually initiate SI, containment spray, and annulus ventilation if required following a loss of control room incident.

Plant/Event Data

Systems Involved: SG atmospheric dump system

Components and Failure Modes Involved: PORVs - failed open in test

Component Unavailability Duration: NA Plant Operating Mode: 1 (24% power) Discovery Method: Testing Reactor Age: 0.1 year Plant Type: PWR

Comments

None

MODELING CONSIDERATIONS AND DECISIONS

0.12

Initiators Modeled and Initiator Nonrecovery Estimate

SLB

No recovery assumed possible because of the test criteria; leak isolable from the control room when the test was terminated

Branches Impacted and Branch Nonrecovery Estimate

SS release terminated	1.0	Recoverable from control room but recovery was delayed due to numerous
HPI	1.0	procedure and operator errors Valve labeling errors resulted in the inability to provide sufficient HPI
		flow during the test

Plant Models Utilized

PWP plant Class F

Event Identifier: 414/86-028 Event Description: S6 PORVs Open with Plant Trip and Other Failures at Catamba Event Date: 6/27/86 INITIATING EVENT NON-RECOVERABLE INITIATING EVENT PROBABILITIES SLB 1.2E-01 SEQUENCE CONDITIONAL PROBABILITY SUMS End State/Initiator Probability CD SL B 1.1E-04 Total 1.1E-04 ATWS SLB 3.68-06 Total 3.6E-06 DOMINANT SEQUENCES End State: CD Conditional Probability: 7.7E-05 107 SLB -RT REQ. SS. ISO -AFM HP1 End State: ATWS Conditional Probability: 3.6E-06 112 SLB RT SEQUENCE CONDITIONAL PROBABILITIES Sequence End State Prob N Rec++ 104 SLB -RT -REQ. SB. ISO AFM HP1(F/B) CD 3.3E-05 3.2E-02 107 SLB -RT REP. SB. ISD -AFM HP1 CD 7.7E-05 + 1.2E-01 112 SLB RT ATWS 3.6E-06 + 1.4E-02

dominant sequence for end state ## non-recovery credit for edited case

MDDEL: c:\asp\newmodel\pwrbmslb.txt DATA:

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
SLB	1.1E-07 > 1.1E-07	1.0E+00 > 1.2E-01	
Branch Model: INITOR			
Initiator Freq:	1.1E-07		
RT	2.5E-04	1.2E-01	
REP. \$6.180	6.4E-04	1.0E+00	
AFM	1.0E-03	2.7E-01	
HPI	1.0E-03 > 1.0E+00	5.2E-01 > 1.0E+00	
Branch Model: 1.0F.2			
Train 1 Cond Probs	1.0E-02 > Failed		
Train 2 Cond Prob:	1.0E-01 > Failed		
HP1(F/B)	1.0E-03 > 1.0E+00	5.2E-01 > 1.0E+00	4.0E-02
Branch Hodel: 1.0F.2+opr			
Train 1 Cond Prob:	1.0E-02 > Failed		
Train 2 Cond Prob:	1.0E-01 > Failed		
HPR/-HPI	3.0E-03	5.6E-01	4.0E-02
PORV. CPEN	1.05-02	1.0E+00	
REQ. BA. ADDITION	8.3E-04	1.0E+00	
PORV. OPEN, DUE, TD. HPI	8.0E-01	1.05+00	
PORV. CLOSURE	6.0E-03	1.0E+00	

*** forced

Austin 09-11-1987 13:53:04

PRECURSOR DESCRIPTION SHEET

LER No.: 458/86-002 Event Description: Hand-held radio causes LOOP Date of Event: January 1, 1986 Plant: River Bend 1

EVENT DESCRIPTION

Sequence

At 0941 h with the unit in hot shutdown and cooling down from a eactor trip that had occurred ~6 h earlier (see LER 458/86-001) preferred station transformers A and C tripped. Recirculation pump A tripped, the operating condensate pump tripped, and the reactor water cleanup system isolated. RPS bus A deenergized, initiating a half scram and partial isolation of the nuclear steam supply shutoff system. The partial NSSSS isolation caused an instrument air isolation to the reactor building, which caused the scram valves to leak, thereby causing the scram discharge volume to fill. This filling subsequently resulted in an RPS actuation on high scram-discharge-volume level at 0957 h. Upon the preferred station transformer trips, division I and III DGs started, division I emergency ventilation systems autostarted, and standby service-water pumps A, B, C, and D load sequenced. Normal service-water pump B and circulating-water pump B were still running but without any bearing cooling water because bearing-cooling-water pump A had lost power. At 1001 h the MSIV automatically isolated due to decreasing condenser vacuum.

At 1003 h operators were dispatched, and they attempted to recover deenergized load centers. At 1031 h, RPS bus A was reset. Later, an electrical panel was discovered deenergized because of a blown fuse in a transformer. This loss had caused several control building HVAC and fuel building HVAC dampers to close, which then caused the division I control building chiller to trip. Subsequent attempts to restore operation of chillers B and D were also unsuccessful. The partial NSSSS isolation remained sealed in because of the deenergized electrical panel.

The RPS actuation was reset at 1042 h. At 1044 h, \sim 1 h after the initiating event, preferred station transformers B and D tripped. The station was now in a complete LOOP. The division II DG started and sequenced properly. An unusual event was immediately declared, and abnormal operating procedures were initiated. Reactor water level was +80 in. on the shutdown range, and pressure was at 240 psig.

At 1114 h the half RPS actuation was reset, and power to RPS bus B was restored. At 1124 h the preferred station transformers were energized, but the supply breakers to the plant could not be closed. It was determined that breaker closure was locked out by the tone-relaying transfer trip (fiber-optic) system, which could not be reset. At 1130 h this backup system was disabled and the breakers were closed. All inhouse loads were restored, and the unusual event ended after 1 h and 10 min.

An investigation of the protective relaying revealed that no protective relaying targets had been initiated. Further, the trip signals sent to the lockout relays could only have been initiated by a spurious signal in the backup pilot wire or tone-relaying transfer trip circuits. Functional and diagnostic testing of both the pilot wire and tonerelaying circuits showed that both systems were operating as designed at the time of testing. Two items were noted: (1) spurious trips could be generated on the tone-relaying system with hand-held radios in close proximity (within approximately a 10- to 12-ft radius) of the transmitters/receivers and (2) some of the tone-relaying keying and rack power were supplied from two separate battery sources. Although no spurious trips could be simulated by testing, this type of connection could result in transients within the tone-relaying equipment. It was decided to correct the wiring in the field such that keying power and rack power were supplied by the same battery source.

Two types of hand-held radios were tested. They are commonly used on site by security and operations personnel. Both of these radios were keyed to transmit iuside the control building of the Fancy Point switchyard, and both caused spurious trips on the tone-relaying system. Careful consideration led to the conclusion that it was highly probable that the LOOP was caused by radio frequency interference.

Also investigated was the difficulty in resetting the lockout relays. Because of the complexity of the tone-relaying and pilot-wiretripping circuitry, the resetting of the lockout relays must be performed in the proper sequence. Operations procedures were determined to have addressed the required sequence.

Corrective Action

As a result of this event, several corrective actions have been completed or are in progress. These corrective actions in part include

- installation of shielding on the tone-relaying equipment in the Fancy Point switchyard,
- rewiring the tone equipment such that both channels are required for tripping,

- changing dc power supplies to tone-relaying equipment such that the keying and rack power are both supplied from the same dc source,
- installation of sequence of event recorders in the switchyard and at the generator/transformers protective relaying panel,
- 5. installation of additional drainage reactors at the plant end of the pilot wire shielding, and
- 6. Installation of supervisory control and data acquisition (SCADA) system alarms to provide annunciation in the main control room and at the Government Street transmission and distribution control center for loss of channel signals on tone-relaying equipment.

Plant/Event Data

Systems Involved:

Power system (ac), pilot-wire relay system, tone-relaying transfer trip system, plant communications system

Components and Failure Modes Involved: Hand-held radios — gave false signals to tone-relaying transfer trip system Power relay (ac) — transferred open Main feedwater — failed in operation

Component Unavailability Duration: NA Plant Operating Mode: 3 (0% power) Discovery Method: Operational event Reactor Age: 0.2 year Plant Type: BWR

Comments

None

MODELING CONSIDERATIONS AND DECISIONS

1.0

Initiators Modeled and Initiator Nonrecovery Estimate

LOOP

Nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

None

Plant Models Utilized

BWR plant Class C

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: Event Description: Event Date: Plant:	Hand Held Radio Caus 1/1/86	ses LOOP				
INITIATING EVENT						
NON-RECOVERABLE IN	ITIATING EVENT PROBAL	BILITIES				
LOOP				1.0E+00		
SEQUENCE CONDITION	AL PROBABILITY SUMS					
End State/Ini	tiator			Probability		
CV						
LOOP				9.05-08		
Total				9.02-08		
CD						
LOOP				7.0E-05		
Total				7.08-05		
ATWS						
				1.95-05		
LOOP						
Total				1.98-05		
DOMINANT SEQUENCES						
End State: CV		Conditional	Probability:	8.96-08		
226 LOOP -EMERG.	POWER SCRAM -SLC.OR	RODS HPCI	RCIC/TRANS.OR.I	LOOP -SRV.ADS	-LPCS -RHR (S	SDC)
End State: CD			Probability:			
	POWER -SCRAM SRV.CH	HALL/LOOP SC	RAM -SRV.CLOSE	-HPCI RHR (SD	C) RHR (SPCOC	L)/-
	C.I.AND.V/RHR (SDC					
End State: ATWS		Conditional	Probability:	1.7E-05		

240 LOOP EMERG. POWER SCRAM SLC.OR. RODS

SEQUENCE CONDITIONAL PROBABILITIES

Sequence	End State	Prob	N Rec++
201 LOOP -EMERG.POWER -SCRAM SRV.CHALL/LOOPSCRAM -SRV.CLOSE -HP CI RHR(SDC) RHR(SPCOOL)/-LPCI.RHR(SDC) C.I.AND.V/RHR(SD C).RHR(SPCOOL)	CD	4.5E-05 +	1.1E-01
209 LOOP -EMERG.FOWER -SCRAM SRV.CHALL/LOOPSCRAM SRV.CLOSE -HP CI RHR(SDC) RHR(SPCOOL)/-LPCI.RHR(SDC) C.I.AND.V/RHR(SD C).RHR(SPCOOL)	CD	2.8E-06	1.1E-01
215 LOOP -EMERG. POWER -SCRAM SRV. CHALL/LOOPSCRAM SRV. CLOSE HP CI RCIC/LOCA SRV. ADS	CD	1.7E-05	2.48-01
226 LOOP -EMERG. POWER SCRAM -SLC.OR. RODS HPCI RCIC/TRANS.OR.LOO P -SRV.ADS -LPCS -RHR (SDC)	CV	8.9E-08 +	2.4E-01
240 LOOP -EMERG. POWER SCRAM SLC.OR. RODS	ATWS	1.7E-05 +	1.0E+00
243 LOOP EMERG.POWER -SCRAM SRV.CHALL/LOOPSCRAM -SRV.CLOSE HP CI RCIC/TRANS.OR.LOOP	CD	1.6E-06	1.9E-01
246 LOOP EMERG. POWER -SCRAM SRV. CHALL/LOOPSCRAM SRV. CLOSE HP CI RCIC/LOCA	CD	2.4E-06	2.7E-01
250 LOOP EMERG. POWER SCRAM	ATWS	2.1E-06	8.0E-01
* dominant sequence for end state			
** non-recovery credit for edited case			
SEQUENCE MODEL: c:\asp\newmodel\bwrctree.cmp BRANCH MODEL: c:\asp\newmodel\riverbnd.txt			
DDODADTI ITY ETLE.			

PROBABILITY FILE: c:\asp\newmodel\bwr_c.pro

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS LOOP	8.6E-04	1.0E+00	
Branch Model: INITOR	1.7E-05 > 1.7E-05	3.2E-01 > 1.0E+00	
Initiator Freq:	1.7E-05		
LOCA	3.3E-06	5.0E-01	
SCRAM	3.5E-04	1.0E+00	
SLC.DR.RODS	1.0E-02	1.0E+00	4.0E-02
PCS/TRANS	1.7E-01	1.0E+00	
PCS/LOCA	1.0E+00	1.0E+00	
SRV. CHALL / TRANS SCRAM	1.0E+00	1.0E+00	
SRV. CHALL/TRANS. SCRAM	1.0E+00	1.0E+00	
SRV.CHALL/LOOPSCRAM	1.0E+00	1.0E+00	

SRV.CHALL/LOOP.SCRAM	1.0E+00	1.0E+00
SRV.CLOSE	5.9E-02	1.0E+00
EMERG . POWER	7.5E-03	8.0E-01
FW/PCS.TRANS	4.68-01	3.4E-01
FW/PCS.LOCA	1.0E+00	3.4E-01
HPCI	2.0E-02	3.4E-01
RCIC/TRANS.OR.LOOP	6.0E-02	7.0E-01
RCIC/LOCA	1.0E+00	1.0€+00
CRD	1.0E-02	1.0E+00
SRV.ADS	3.7E-03	7.1E-01
and the second s		3.4E-01
COND/FW.PCS LPCS	2.0E-02	3.4E-01
LPCI (RHR) /LPCS	6.0E-04	7.1E-01
	1.0E+00	1.0E+00
RHRSW/LPCS.LPC1.LOOP	1.0E+00	1.0E+00
RHRSW/LPCS.LPCI.LOCA	1.0E+00	1.0E+00
RHR (SDC)	2.1E-02	3.4E-01
RHR (SDC) /-LPCI	2.0E-02	3.4E-01
RHR (SDC) /LPCI	1.0E+00	1.0E+00
RHR (SPCOOL) /-LPCI .RHR (SDC)		1.0E+00
RHR (SPCOOL) /LPCI.RHR (SDC)		1.0E+00
C.I.AND.V/RHR(SDC).RHR(SPCDDL)		3.4E-01
C.1. MIND Y/ NAM (SPC) MAN (SPC00C)	A KYNE YYY	6176 VI

4.0E-02 4.0E-02

* branch model file

** forced

Minarick 02-24-1988 12:17:28

PRECURSOR DESCRIPTION SHEET

LER No.: 458/86-047 Event Description: Emergency power, LPCS, RHR train A, and RCIC systems are degraded twice Date of Event: July 31, 1986 Plant: River Bend 1

EVENT DESCRIPTION

Sequence

At 0243 h on July 31, 1986, and at 0637 h on August 2, 1986, containment unit cooler 1-A feeder breaker (1EJS*ACB36) and the switchgear tripped at the same time. The loss of the switchgear resulted in the automatic start of both trains of the following systems: (1) annulus mixing, (2) standby gas treatment, and (3) fuel building filtration. Additionally, power was lost to the division 1 DG fuel-oil transfer pump along with power to several valves on the LPCS, RHR train A, RCIC systems, and various drywell and unit coolers. Unit cooler operation was checked and found to be normal.

Feeder breaker lEJS*ACB36 is equipped with an overcurrent timer relay, which initiates a trip of breaker lEJS*ACB38 in the event ACB36 fails to clear a fault in sufficient time. The manufacturer has determined that the output transistor on the overcurrent timer relay was defective and caused the trip.

In neither event were redundant trains affected.

Corrective Action

The overcurrent timer relay was replaced. The manufacturer has recommended that the output transistor be replaced with a different type transistor, one that has higher voltage and lower leakage characteristics. The output transistors for all affected relays are being replaced.

Plant/Event Data

Systems Involved:

Containment heating and ventilation, emergency power, low-pressure core spray, RHR, and RCIC systems.

Components and Failure Modes Involved: Containment unit 1-A cooler feeder breaker - tripped in operation Switchgear feeder breaker - tripped in operation Component Unavailability Duration: 2 h assumed (1 h per breaker trip) Plant Operating Mode: 1 (94%/99% power) Discovery Method: Operational event Reactor Age: 0.75 year

Plant Type: BWR

Comments

None

MODELING CONSIDERATIONS AND DECISIONS

Initiators Modeled and Initiator Nonrecovery Estimate

Postulated	transient	Base	case	nonrecovery
Postulated	LOOP	Base	case	nonrecovery
Postulated	LOCA	Base	case	nonrecovery

Branches Impacted and Branch Nonrecovery Estimate

RCIC	Base case	Unavailable due to loss of	switch gear
LPCS	Base case	One train unavailable due	to loss of
		switch gear	

Plant Models Utilized

BWR plant Class C

CONDITIONAL CORE DAMAGE PROBABILITY CALCULATIONS

Event Identifier: 458/86-047 Event Description: One Train EPS, RHR, RCIC and LPCS Are Unavailable Event Date: 7/31/86 Plant: River Bend 1

UNAVAILABILITY, DURATION= 2

NON-RECOVERABLE INITIATING EVENT PROBABILITIES

-				
TRANS				1.7E-03
LOOP				
LOCA				1.1E-05
LUUM				3.32-06

SEQUENCE CONDITIONAL PROBABILITY SUMS

End State/Initiator		Probability
CV		
TRANS LOOP LOCA		6.1E-11 1.5E-11 (1.6E-16)
Total		7.7E-11
CD		
TRANS LOOP LOCA		6.4E-09 5.9E-10 6.8E-11
Total		7.1E-09
ATWS		
TRANS LOOP LOCA		0.0E+00 0.0E+00 0.0E+00
Total		0.05+00
DOMINANT SEQUENCES		
End State: CV	Conditional Probability:	4.0E-11

134 TRANS SCRAM -SLC.OR.RODS PCS/TRANS -SRV.CLCSE FW/PCS.TRANS HPCI RCIC/TRANS.OR.LOOP -SRV.ADS -COND/FW.PCS -RHR(SDC)

End State: CD Conditional Probability: 4.7E-09

101 TRANS -SCRAM PCS/TRANS SRV.CHALL/TRANS.-SCRAM -SRV.CLOSE -FW/PCS.TRANS RHR(SDC) RHR(S PCODL)/-LPCI.RHR(SDC) C.I.AND.V/RHR(SDC).RHR(SPCODL)

SEQUENCE CONDITIONAL PROBABILITIES

	Sequence	End State	Prob	N Rec**
101	TRANS -SCRAM PCS/TRANS SRV.CHALL/TRANSSCRAM -SRV.CLOSE -FW /PCS.TRANS RHR(SDC) RHR(SPCOOL)/-LPCI.RHR(SDC) C.I.AND. V/RHR(SDC).RHR(SPCOOL)	CD	4.7E-09 *	1.0E-01
102	TRANS -SCRAM PCS/TRANS SRV.CHALL/TRANSSCRAM -SRV.CLOSE FW /PCS.TRANS -HPCI RHR(SDC) RHR(SPCDDL)/-LPCI.RHR(SDC) C. I.AND.V/RHR(SDC).RHR(SPCDDL)	CD	8.7E-10	3,96-02
110	TRANS -SCRAM PCS/TRANS SRV.CHALL/TRANSSCRAM -SRV.CLOSE FW /PCS.TRANS HPCI RCIC/TRANS.OR.LOOP CRD SRV.ADS	CD	4.1E-10	5.7E-02
111	TRANS -SCRAM PCS/TRANS SRV.CHALL/TRANSSCRAM SRV.CLOSE -FW /PCS.LOCA RHR(SDC) RHR(SPCOOL)/-LPCI.RHR(SDC) C.I.AND.V /RHR(SDC), RHR(SPCOOL)	CD	2.3E-10	7.68-02
134	TRANS SCRAM -SLC.OR.RDDS PCS/TRANS -SRV.CLOSE FW/PCS.TRANS HPCI RCIC/TRANS.OR.LDDP -SRV.ADS -CDND/FW.PCS -RHR(SDC)	CV	4.0E-11 *	5.3E-02
138	TRANS SCRAM -SLC.OR.RODG PCS/TRANS -SRV.CLOSE FW/PCS.TRANS HPCI RCIC/TRANS.OR.LOOP -SRV.ADS COND/FW.PCS -LPCS -RHR(SDC)	CV	1.3E-11	1.8E-02
142		CV	7.5E-12	9.3E-03
163	TRANS SCRAM -SLC.OR.RODS PCS/TRANS SRV.CLOSE FW/PCS.LOCA HPCI RCIC/LOCA -SRV.ADS COND/FW.PCS LPCS -LPCI(RHR)/LPC S -RHR(SDC)/-LPCI	CV	1.4E-12	1.3E-02
201	LOOP -EMERS.POWER -SCRAM SRV.CHALL/LOOPSCRAM -SRV.CLOSE -HP CI RHR(SDC) RHR(SPCOOL)/-LPCI.RHR(SDC) C.I.AND.V/RHR(SD C).RHR(SPCOOL)	CD	2.1E-10	3.7E-02
226	LOOP -EMERS.POWER SCRAM -SLC.OR.RODS HPCI RCIC/TRANS.OR.LOO P -SRV.ADS -LPCS -RHR(SDC)	CV	9.78-12	5.0E-02
230	LOOP -EMERS.POWER SCRAM -SLC.OR.RODS HPC1 RCIC/TRANS.OR.LOO P -SRV.ADS LPCS -LPC1 (RHR) /LPCS -RHR (SDC) /-LPC1	QQ	5.5E-12	2.58-02
243	LOOP EMERS. POWER -SCRAM SRV. CHALL/LOOPSCRAM -SRV. CLOSE HP CI RCIC/TRANS.OR. LOOP	CD	2.7E-10	6.1E-02

* dominant sequence for end state

** 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:\asp\newmodel\bwrctree.cmp
BRANCH MODEL :	c:\asp\newmodel\riverbnd.txt
FROBABILITY FILE:	c:\asp\newmodel\bwr c.org

No Recovery Limit

BRANCH FREQUENCIES/PROBABILITIES

Branch	System	Non-Recov	Opr Fail
TRANS	8.6E-04	1.0E+00	
LOOP	1.7E-05	3.2E-01	
LOCA	3.3E-06	5.0E-01	
SCRAM	3.5E-04	1.0E+00	
SLC.OR.RODS	1.0E-02	1.0E+00	1 15-15
PCS/TRANS	1.7E-01	1.0E+00	4.0E-02
PCS/LOCA	1.0E+00	1.0E+00	
SRV. CHALL/TRANS SCRAM	1.0E+00	1.0E+00	
SRV.CHALL/TRANS.SCRAM	1.0E+00	1.0E+00	
SRV.CHALL/LOOPSCRAM	1.0E+00	1.0E+00	
SRV.CHALL/LOOP.SCRAM	1.0E+00	1.0E+00	
SRV.CLOSE	5.92-02	1.0E+00	
EMERG . POWER	7.5E-03		
FW/PCS, TRANS	4.68-01	8.0E-01	
FW/PCS.LOCA	1.0E+00	3.4E-01	
HPCI		3.48-01	
RCIC/TRANS.OR.LOOP	2.0E-02	3.4E-01	
Branch Model: 1.0F.1	6.0E-02 > 1.0E+00	7.0E-01	
Train 1 Cund Prob:	1 AP 33 1 10 10 10 10 10		
RCIC/LOCA	6.0E-02 > Unavailable		
CRD	1.0E+00	1.0E+00	
SRV , ADS	1.0E-02	1.0E+00	4.0E-02
COND/FW.PCS	3.7E-03	7.1E-01	4.0E-02
LPCS	1,0E+00	3.4E-01	
Branch Model: 1.0F.1	2.0E-02 > 1.0E+00	3.4E-01	
Train 1 Cond Prob:	5 AF 46 5 15		
LPCI (RHR) /LPCS			
RHRSW/LPCS.LPC1.TRANS	6.0E-04	7.1E-01	
RHRSW/LPCS.LPC1.LOOP	1.0E+00	1.0E+00	
RHRSW/LPCS.LPC1.LOCA	1.0E+00	1.0E+00	
	1.0E+00	1.0E+00	
RHR (SDC)	2.1E-02 > 3.0E-02	3.4E-01	
Branch Model: 1.0F.2+ser			
Train 1 Cond Prob:			
	1.0E-01 > Unavailable		
Serial Component Prob:	2.0E-02		
RHR (SDC) /-LPCI	2.0E-02	3.4E-01	
RHR (SDC) /LPCI	1,0E+00	1.0E+00	

RHR (SPCODL) /-LPC1.RHR (SDC)	2.0E-02	1.0E+00
RHR (SPCOOL) /LPCI .RHR (GDC)	5.2E-01	1.0E+00
C.I.AND.V/RHR(SDC).RHR(SPCOOL)	1,0E+00	3.4E-01

+ branch model file

++ forced

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APPENDIX E

LICENSEE EVENT REPORTS ASSOCIATED WITH PRECURSORS

APPENDIX E

LICENSEE EVENT REPORTS ASSOCIATED WITH PRECURSORS

This appendix contains copies of licensee event renorts (LERs) associated with precursors documented in Appendix D. A table of contents, Table E.l, is also provided.

Note that copies of LERs utilized in the Accident Sequence Precursor Program are also used in other Oak Ridge National Laboratory programs and may contain markings made during abstracting and coding in these programs.

LER No.	Event title	Plant name	Page number
247/86-017	Open condenser dump valves cause trip, and one safeguards train fails to start	Indian Point 2	E-6
247/86-035	Trip, LOFW, and two AFW train failures occur	Indian Point 2	E-10
249/86-013	HPCI and one train of the core spray and LPCI systems are inoperable	Dresden 3	E-14
250/86-036	Unavailability of DGs	Turkey Point Units 3 and 4	E=17
250/86-038	AFW system is unavailable	Turkey Point Units 3 and 4	E-20
250/86-039	Trip occurs with stuck-open PORV	Turkey Point 3	E-24
261/86-005	Bus failure causes a trip followed by a LOOP with a DG unavailability	Robinson 2	E-28
869/86-001	TRIP, LOFW, and a stuck-open MSRV occur	Oconee 1	E-38
89/86-011	Emergency condensor cooling system is unavailable	Oconee Station Units 1,2, and 3	E=43
77/86-003	DG trip in test causes scram	Peach Bottom 2	E-49
80/86-029	Charging pump service-water pumps are unavailable	Surry 1	E-53
80/86-031	High-head injection system is unavailable	Surry 1	E-56
81/86-010	High-head injection system is unavailable	Surry 2	E-59
82/86-006	Emergency power system is unavailabile	Prairie Island Units 1 and 2	E=62
82/86-011	Emergency power system is unavailabil	Prairie Island Units 1 and 2	E-66
85/86-001	Trip occurs, and automatic depressurization and turbine bypass system fails to open	Ft. Calhoun	E-68
93/86-027	LOOP occurs due to winter storm	Pilgrim 1	E-71
01/86-004	MSIVs fail to close on demand	Point Beach 2	E-75
118/86-006	Trip occurs and one atmospheric dump valve fails to close	Calvert Cliffs Unit 2	E-90
41/86-045	Condensate storage tank is lost	Fermi 2	E-95
41/86-048	RCIC and HPCI are unavailable	Fermi 2	E-106
62/86-011	Saltwater and CCW systems are unavailable	San Onofre 3	8-112
66/86-035	LPCS system is unavailable	Hatch 2	E-115

Table E.1. Index of precursor licensee event reports

LER No.	Event title	Plant name	Page number
370/86-006	High-head injection system is unavailable and DG A is out of	McGuire 2	E-121
389/86-011	service Emergency power system is	St. Lucie 2	E-125
409/86-023	unavailable LOOP occurs due to lightning strike at coal-fired unit	LaCrosse	E-131
413/86-031	Small LOCA forces plant trip	Catawba 1	E-133 E-137
414/86-028	SG PORVS open inadvertently in test, and trip when other failures occur	Catawba 2	E-13/
458/86-002	Hand-held ratio causes LOOP	River Bend 1	E-143
458/86-047	Emergency power, LPCS, RHR train A, and RCIC systems are degraded twice	River Bend 1	E-151

Table E.1 (continued)

Indian Point Unit #2		LILLINDLL EV	ENT REPORT	uch,	**	CARE RECORD PROVIDE ONLINE CARE LE R	Prince (a
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Cn May 26, 1966 at 1556, while operating at 30% power, the condenser steam dim values for the state of the selection trained to the second trained trained to the second trained trained to the second trained trained to the second trained to the second trained trained to the second trained trained to the second trained trained trained trained to the second trained trained trained trained trained to the second trained trai			· · · +	1	1000	-	90'P
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CD Nay 28, 1986 at 1556, while operating at 308 power, the condenser steam during values opened, creating a high steam flow condition coincident with a part of the public ware not affected. Cn Nay 28, 1986 at 1556, while operating at 308 power, the condenser steam during values opened, creating a high steam flow condition coincident with a safety injection signal as part of the recovery procedure. The health and mathematical interaction and a second ratio of the recovery procedure. The health and the safety of the public ware not affected. Beorrogoods Beorol Second Condenser steam Beorrogoods Beorol Second Second	Ray Sutton, Failure Analysis	Engineer			ANIA 0051 1	ILEPHONI NUM	11.*
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Ch Kay 28, 1986 at 1556, while operating at 308 power, the condenser steam during values opened, creating a high steam flow condition coincident with resulted in a safety injection actuation and a reactor training a subsequently actuated when an operator manually looked-in the inselection signal as part of the recovery procedure. The health and safety of the public were not affected.	TURE. TO MAR	54	CAURI POPPER	COMPOSE IT	TURES	TO MAR DE	
Ch May 26, 1986 at 1556, while operating at 30% power, the condenser steam durp values operad, creating a high steam flow condition coincident with a low reactor coolant system average temperature fT-NOC 541 57. This injection Train A actuated, however train B did not initially actuate. Train B was subsequently actuated there an operator manually looked-in the safety injection signal as part of the recovery procedure. The health and safety of the public were not affected.	B SIB PULL N FILIBIO Y						
Cn May 26, 1986 at 1556, while operating at 30% power, the condenser steam durp valves opened, creating a high steam flow condition coincident with a low reactor coolant System average temperature (T-AVG 541 S). This injection Train A actuated, however Train B did not initially actuate. Train B was subsequently actuated when an operator manually looked-in the safety of the public were not affected.	ALE BILLY VINTO Y	150312				1	
Cn May 26, 1986 at 1556, while operating at 30% power, the condenser steam durp valves opened, creating a high steam flow condition coincident with a low reactor coolant system average temperature (T-AVG 5415). This injection Train F actuated, however Train B di areactor trip. Gafety train B was subsequently actuated when an operator manually looked-in the safety injection signal as part of the recovery procedure. The health and safety of the public were not affected.	S. I						BAY TYL
On May 28, 1986 at 1556, while operating at 30% power, the condenser steam dump values opened, creating a high steam flow condition coincident with a low reactor coolant system everage temperature (T-NVC -541 F). This injection Train A actuated, however Train B did not initially actuate. Train B was subsequently actuated when an operator warually looked-in the safety injection signal as part of the recovery procedure. The heilth and selety of the public were not affected.							
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A CONTRACTOR OF A CONTRACT OF	Un May 28, 1986, Unit dump control system w "pressure" mode at 14 controller TC-412J. received an open sign (Foxboro Model 62R5E) duction in reactor co actuation. Safety injection Trai safeguards actuation, Train A signal result	2 was operating at 30 ras switched from the "1 55 due to erratic beha At 1556, all twelve (1 al due to a faulty stee . This resulted in an colant temperature, and . In A actuated resulting , however, Train B did ted in closure of the m ords after the reactor	power. The temperature m vior observed 2) condenser 6 am dump contro increased ste a subsection in a reactor not actuate. ain steam isol	ode to the on temperature team dump valves ller PC-404 am flow and a re- safety injection trip and Safety injection ation valves
1				

Indian Point Unit #2 Description of the second signal and the second signal indicates of the required for addet in the second signal and depressing safety injection react buttons. Safety injection Train B was successfully actuate due to Train B not function of the required reachant valving required for adjection. Safety injection Train B was successfully actuated at 1607 when the control from operators reset asfety injection per Optimal Recovery Procedure using a static injection react buttons. This actuates parallel contacts in both trains of the safety injection signal and depressing safety injection reset buttons. This actuates actuates parallel contacts in both trains of the safety injection signal, the introduction of the second signal indicates a stripped and autoratically restarted and all required redundant valves (Train A and Train B) operated normally. Analysis of Occurrence: This occurrence is a reportable event because it resulted in completion of Engineeries Safety injection sufficient sufficient of the steam of the use of the second signal indicates of the context. Cause of Occurrence: This occurrence: Investigation of the steam dum containent isolation is sufficient to mitigate the consequence of an accident. Safety injection was not required, and no water was injected during this event. There was no effect on the health and safety of the public as a result of this event. Cause of Occurrence: Investigation of the steam dum control circuit revealed that steam dum control is for the introl information was not required in corporatic was control. A high output i majoring high officiants are able to dimension to premetic control information with ano the acting and thealth and safety of the public as a result of		LICENSEE EVENT REPORT ILERI TEXT CONTINUATION	
Indian Point Unit 12 clicicle(2)247 2816 01172 010 01309 The required functions which did not fully actuate due to Train B not func- of the required redundant valving required for safety injection. Safety injection Train B was successfully actuated at 1607 when the control from operators reset safety injection per Optimal Recovery Procedure ES-11. Resetting safety injection consists of manally locking in another safety injection logic. Because Train B had not been actuated by the first safety injection logic. Because Train B had not been actuated by the first safety injection logic. Because Train B had not been actuated by the first safety injection logic. Because Train B had not been actuated by the first safety injection logic. Because Train B had not been actuated by the first safety injection signal, the introduction of the second signal initiated a separate safety injection expense. Safety injection equipment was first and Train B) operated normally. Analysis of Cocurrence: This occurrence is a reportable event because it resulted in completion of Engineered Safeguards logic and actuated the Reactor Protection System. One of two trains of safety injection and containment isolation is suffi- cient to mitigate the consequences of an accident. Safety injection was not required, and no water was injected during this event. There was no effect on the health and safety of the public as a result of this event. Cause of Occurrence: The safety injection of the stars dump control clicuit revealed that stars dump controller R0-404 output was erratic, with output : rail going high off- causes current to preventie controller SH-404 to increase its cutput signal to the stars dum valves, causing then to open. The safety injection circuit was examined to determine the cause of Train B actuates relays SLI and SL2, which actuate safety injection relays SLI - all and SLI-2 for Train B and SL2, which actuate safety injection relays SLI - all the dum actuates relays SLI and SL2, which actuates saf		GOC LL1 IN-AND . N U	_
Indian Point Unit #2 c [1] [c] [c] [2] [1] 7 [c] (c) [1] 7 - old old or fully The required functions which did not fully actuate due to Train B not func- col the required redundant valving required for safety injection. Safety injection Train B was successfully actuated at 1600 when the control room operators reset safety injection per Optimal Recovery Procedure ES-1.1. Resetting safety injection consists of neurally looking in another safety injection signal and depressing safety injection required by the first safety injection logic. Because Train B had not been actuated by the first safety injection signal, the introduction of the second signal initiated a stripped and autoratically restarted and all required redundant valves (Train A and Train B) operated normally. Analysis of Occurrence: This cocurrence is a reportable event because it resulted in completion of Effect on thights the consequences of an accident. Safety injection system. One of two trains of safety injection and containment isolation is suffi- cient to mitigate the consequences of an accident. Safety injection was not required, and no water was injected during this event. There was no effect on the health and safety of the public as a result of this event. Cause of Occurrence: Investigation of the stars durp control circuit revealed that steam durp controller PC-604 output was erratic, with output : mal going high off- causes current to preparate controller 6H-604 to during the signal from PC-604 causes current to preparate controller 6H-604 to during the string cause scale with nimer variations in input. A high output signal from PC-604 causes current to preparate controller 6H-604 to during a first off- cause scale with nimer variations in input. A high output signal function cause of the initial s		TALE STATES	2
 The required functions which did not fully actuate due to Train E not func- tioning were containment isolation place A ("" signal) Train E, and some of the required redundant valving required for safety injection. Safety injection Train E was successfully actuated at 1607 when the control room operators reset safety injection per Optimal Fecovery Procedure ES-1.1. Resetting safety injection consists of manually looking in another safety injection signal and depressing safety injection reset buttons. This action actuates parallel contacts in both trains of the safety injection signal, the introduction of the second signal initiated a separate safety injection sequence. Safety injection equipment was stripped and autoratically restarted and all required redundant valves (Train A and Train E) operated normally. Analysis of Cocurrence: This occurrence is a reportable event because it resulted in completion of Engineered Safeguards logic and actuated the Reactor Protection System. Cause of Decurrence: Investigation of the stam dum control direction that steam dum effect on the health and safety of the public as a result of this event. Cause of Occurrence: Investigation of the stam dum control clicuit revealed that steam dum controller PC-404 cuput was erratic, with output signal from PC-404 causes current to premetic controller GM-404 to increase its cuput signal is to the train of curval two serves in the open. The safety injection circuit was examined to determine the cause of Train B and SiA-2 for Train B. at the output signal from PC-404 causes current to premetic controller GM-404 to increase its cuput signal and SiA-2 for Train B at the set and serves and the open. 	Indian Poi	nt Unit /2	
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	Corrective Action: The steam dump control system repairs were as follows:	
1	 All electrolytic capacitors and the auto/manual relu- condenser steam dump controller PC-404 and the contribution recalibrated. The controller functioned normally for 	
1 442 ·	repair.	The state of the second state of the
2	 Ourrent to pneumatic converter PM-404 was replaced valibrated. 	with a new unit and
. 3	 Temperature controller TC-412J was replaced with a calibrated. 	new unit and
	4. Pressure transmitter PT-404 was calibrated.	엄마도 안 가 안 감기가 !
		ion circuit was as
1	Corrective action for the Safety Injection System Actuat follows:	
	 Felays SIA-2, SL-2, Ti 2 and TR-2X were replace Injection Actuation Log in was tested and very 	d. The Safety fied operational.
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	nea canque can a servi								
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	Identification of Oc Reactor trip breaker	r B opened when an	associ	ated	02.2	p res	a wiri	R.C.	
	The most probable ca	use of the relay de	energ;	zing	*0.5	1000	S		
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	The opening of rea	ctor trip breaker	"B" tr	Tober	s tri	*^ 3	E RISTR	this	happen
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Cause of Occurrence:

The opening of the reactor trip breaker is attributed to loose wires in the associated relay racks. Reactor trip relays RT3 and RT4 were found deenergized, as was relay SIAM 1-X. Several loose connections were subsequently discovered between relay SIAM 1-X and SI-13X which is the circuit which controls SIAM 1-X. A safety injection logic test was being performed on the other side of the relay cabinet at the time of the trip, and this activity apparently distured the relay circuitry. It was noted that the above listed relays could be actuated by moving the wiring burdles in the same cabinet during the subsequent troubleshooting operations.

The cause of the #21 Auxiliary Boiler Feed Pump circuit breaker tripping following initial pump start could not be positively determined. The circuit breaker is a Westinghouse DB-50 with amptector tripping devices. Both the instantaneous and time delay trip indicators were found actuated at the circuit breaker, however the indicators cannot be considered to be a reliable indication of circuit breaker actuation. The time delay trip actuation indicator was discovered to be in the "operated" position on the day following the circuit breaker trip as well. This indication was known to be false since the indicator had been reset previously, the circuit breaker was closed and the motor was operating at the time of discovery⁶ and the breaker was known to have not tripped during this time interval.

The auxiliary feed pumps were retested using an automatic start signal in order to simulate the actual pump conditions and challenge the equipment to the extent practical. Both motor driven auxiliary feed pumps were started simultaneously with reduced bus voltage to determine if system transients could be duplicated and pinpoint the cause of the trip. During the test, higher than expected motor current was measured. The higher current was due to higher than expected flow from the pump because of an improperly set auxiliary feedwater control valve to steam generator \$21. Although the circuit breaker did not trip during subsequent testing, the effect of increased current due to the motor characteristics and the high auxiliary feedwater flow was close enough to the circuit breaker minimum setpoint to represent a challenge to the amptector setting. Since the low suction pressure and the instantaneous overcurrent trip signal did not reach a value near their setpoints, the time/overcurrent setting must be considered the most probable cause of tripping.

The relief valve in the steam line to the turbine driven auxiliary feed pump was tested and found to open at 665 psi vs a nominal set point of 700 psi. The actuation of the relief valve was probably due to pressure hunting down stream of the control valve because of controller low proportional band setting problems and the high differential pressure across the valve.

	LICENSEE EVENT REPO	ORT (LER) TEXT CONTINU	ATION APPROVED OVER OF STREAM	
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1993 - Alberton (* 1995) 1995 - Alberton (* 1995)	Point Station Unit 2	0 15 10 0 0 0 2 4 7	8 6 - 13 5 - 010 014 01	V I
Indian	sport as service and age that south			
	Corrective Action: Since several loose co program was instituted the racks. The termini drawing to verify which Over 10,000 screws wer racks. Approximately half-turn to tighten th Although the relays wi normally, those which slightly, were replace Westinghouse BFD delay relays will be examine	to check and checked als are being checked in terminals were check re checked for tightn 1/4 of 1% of the term he screws. hich were deenergized could have been a con d. These relays, SIM s and are being replace d further for evidence	ess in the associated relay minals required greater than appeared to be functioning tributing factor, even if on A 1-X, SIAM 1-Y, and SI-13X, red with like kind. The remo e of any malfunction.	a ly are ved
	automatic start signal conditions and challe operability. Suction that the low pressure inadvertently trip. A delay trip setpoint we amperes. In addition to the steam generator value and keep total a In an attempt to prev setting on the turbir position (from 0%) to automatic auxiliary f signal several times relief valve would no	I with reduced bus to nge the equipment as header conditions we suction switches wi as a result of this te as raised from a nomin , the auxiliary feedwar rs were re-established auxiliary feedwater fl ent subsequent relief he driven auxiliary f provide an additional feed system actuation following this adjus	ker was retested by means of ltage to simulate actual sta- much as possible to ensure re also monitored to reveri 11 not cause either pump to ssting, the circuit breaker to hal value of 600 amperes to be ater flow control valve sett 1 to limit the flow to the pro- low within the proper limits. valve lifting, the speed cha eed pump was left at the 20 1 steam demand to the turbing. The pump was given a sta- tment to demonstrate that to control valve receives its e conditions during peveral- ief valve.	fy ime 660 ing oper nger s c nger he oper
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On August 2 power, with		area matery Artean	suger-game house	mirger analy (18)						1111
power, with core spray valve was de initisted an inoperable. 33-1. The of 1500-1 the that the val Subsequent is cracked. The The valve wa installation terminal blo the diesel of bandwheel re were reset. operator fro avoid this e performed on 9/8/85.	sur amaged ad an It w diesel 1501-1 10ve wa Invest bis wa as rep. h erro ock sc operation It in m electron	veillance so that Unucual H as discov closed c 3A valve s in mid- igation c s a resul aired and r. A rev rew was l ed proper ng ring b s not bel ttrically	the values the values (vent was vered that position of the 2- t of hig declare view of t oose in ly. A r ad disen ieved th cycling	red per T. ve would m a declared at the 2/3 3-1 without a double p b. The un 1402-48 w th torque ad operabl the 2/3 di junction eview of gaged. I at the di . The di	sole, S. 4. hot cl I. Th I dies this the solution the solution the sel box 3 the senga rect	whil 5.C.2 ose. e "B" el ge ident ident utdown revea ated his f. gener. TB-18 -1501- repla ged re tause	e performs it was fo A normal core spra nerator fa dication. n was achi- led that t by an inop ailure was ator failu 7. The so -13A valve aced and t etaining r can sot b	ing ound that unit shu ay system ailed to hile perf It was ieved at the motor perable t s possibl ure revea rev was a reveale the open/ ing prev	the 3-li the 3-li the was dec close in forming Du discovery 2007 hou housing orque swi y due to led that tightened d that the close line ented the ined. To	402-48 s lared to bus OS ed trs. Was ltch. an a d and be site
This event w condenser an mitigate the sutomatic in	conse	quences	of a LOC	A. Also	the 2	vere 3 die	operable	and avai	lable to	olation the
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LICENSEE	EVENT REPOR	T (LER) TEXT	01.11	ATIO				U AN P		
		DOCKET MUNBER G		T					PAGE	(3)
ACILITY MAME (1)		1.			-	BURNTIAL	1 11	V BION MQCA		
		1 Sec. 6. 11 St							1.	
Dresden Nuclear Power Sta	tion, Unit 3	0 5 0 0 0	12019	8 6	-	013		a 010	120	101
DT Of more space a required, one subfloored RAC from 3	64 (v) (17)									
On August 27, 1986 at	0030 hours,	with Unit 3	in the	run .	ode	at 19	Perc	ent r	evere	-
								201	.,	
declared inoperable (see DVB 12-3-	-80-527, Wui	rion 4	5.0.2	10	wa+ d:	scov	ered	that	
surveillances require	d per Technic	S Svatem Code	BH) VI	a not	pre	ssuri	ted.	Furt	ther	
was declared. Upon	discovering	the "B" core	spray !	subsys	cea	Techn	ical	Spec	ificat	tion
During the performan	ce Test", it	was discover	ed chat	the 2/	3 d	iesel	gene	rator	(EIIS	S
										Vas
									0-11	
"Low Pressure Coolan	t Injection (LPCI) (EIIS	System	Code I	ehau	system	loubl	e pos	ition	
Operability Test" th	e 1501-13A, L	PUI minimum	unaled.	the vi	alve	was 1	n mi	d-pos	at rated o 0 (2 of c o 0 (2 of c o 0 (2 of c at rated b) system red that Purther pow test l unit Event all pecification e spray. esel tor (EIIS enerator without position. mit , 1986. under ed. This rque switch attaches tor did a. As a until operator 00-2, "Core B" core about this nstallation torque was d thereford y caused) for ced exactly 1 be isize the on to atta box was	
indication. An inve	stigation of	successfull	y cycle	d thre	se c	imes.	The	unit		
The valve was manual	nd cold shutd	iown complete	d by 20	07 hor	17.P	on Aug	USC	27, 1		
shutdown continues .										
Subsequent investigs	tion of the 3	3-1402-48, "1	" pump	full	110%	test	valv	e, ur	This	
work request \$57254	revealed that	c che botor	peracor	-	evit	ch.	The t	orque	e swit	ch
			<pre>ch Unit 5 in the lon unit of system Code BJ) -52), while performing core spray Specification 4.5.C.2 it was discovered ystem Code BM) was not pressurized. Furt -45 walve core spray "B" pump full flow to would not close adequicely. A normal unit specification 3.5.C.3 and an Unusual Even "B" core spray subsystem inoperable, all surveillances required per Tachnical Specification of the repair of HPCI or core s discovered chatthe 2/3 diesel generator o bus 33-1. However, the 2/3 diesel generator o bus 33-1. However, the 2/3 diesel gene cident. Also, while performing DOS 150 CI) (EIIS System Code BM) System Valve I minimum flow valve, showed a double pos e event revealed the valve was in mid-pos uccessfully cycled three times. The unit n completed by 2007 hours on August 27, 1 402-48, "B" pump full flow test valve, un he motor operator housing was fractured. ble Limitorque torque switch. The torque hat the pinion gear located on the torque of a sheared roll pin. The roll pin att the torque switch inoperable, the motor of a sheared roll pin. The roll pin att the valve reached the closed position. the valve disc into the valve seat unti- tor operator housing. The limitorque op- et, and installed on the valve. DOS 1400- rmed at 2112 hourson 8/28/86 and the "B" i Limitorque Corporation was contacted abo- tor a design deficiency but rather an inst.</pre>							
							ull flow test normal unit nusual Event rable, all ical Specificati or core spray.)-1 "Diesel generator (EIIS essel generator v ng DOS 1500-1, n Valve double position. The unit gust 27, 1986. valve, under ractured. This The torque switc the torque switc ll pin attaches the motor did osition. As a seat until torque operator	i		
the forces created	fractured the	motor opera	alled a	on the	VA	lve.	DOS	1400-	2, "Co	ore
Spray Valve Operadi	operable. T	he Limitorqu	e Corpo	ration	va va	s cont	acte	d abo	ut thi	15
failure and replied error. If the corq	ue switch was	installed i	n the r	everse	e di	rectio		and t	heref	ore
be subjected to hig	h torsional I	orces, ine	Waines		Pro	cedure	s (D	MP) (or	
										ely.
seconded to the El	ACCTICAL WOLK	group, int			11 1	urther	emp	nasiz	te che	
importance is insta	lling the tor	que switch o	orrectl	у,						
A subsequent invest		a 2/3 diese	genera	tor f	ailt	tre to	clos	e on	to	
A subsequent invest bus 33-1 was conduc	igation of th	rk request #	7277 at	id it	Va 6	revea	led t	hat a	â	
terminal block scr located above a cal	le pan on the	e 517' eleva	tion nea	ir the	Vo	it 3 r	acto	T Let	ull in	=P
located above a cal room. The screw of	terminal 5 c	of TBJ was a	proxima	ately	a b	issive	0001	ace	which	
This wire is conner allows the diesel										
allows the diesel	generator orea	aver co cros								

#43

LICENSEE EVENT REPOR	T (LER) TEXT CONTINU	UATION	N						
ACUTY RANG ()	DOCKET NUMBER (2)	1		-	-	-			
		-	the second s	SUENTIAL		NIVERN		ADE D	2
Dresden Nuclear Baue Court	a series and the			Xall's	-	NUMBER			
Dresden Nuclear Power Station, Unit 3	0 6 0 0 0 2 4 9	B 6	- 0	1 1 3	-	00	03	OF	01
The acrew was tightened and the did to bus 33-1. A review revealed tha 3TB-187 during the Unit 3 maintenant required. DOS 6600-1 is performed. Nollowing the declaration of operab- times on August 27, 1986 at 1800 ho investigate and repair as necessary ring was disengaged and resting ring motor-operator. The retaining ring react. It is not believed that the from electrically cycling. The dir voltage signature was taken to veri 1500-1, "LPCI System Valve Operabil the valve operable after maintenant. In order to avoid the recurrence of valve operability surveillances wil weekly for one month after Unit 3 a This event was of minimal safety si on coolent accident (LOCA) the "A" is condenser and sutomatic depressuria; on to bus 33-1 and this part of the shutdown, was reported by DVR #12-3.	the maintenance was ce outage. No furth monthly and would is monthly and would is fility for the 3-150 burs, work request # . It was discovered to the handwheel beat was reinstalled and disengaged retaining et cause can not buy fy that the limits with ity Test" was perfor e. this event, all emul be performed on Or tartup. The surveil gnificance since in core spray system, " ation system were any event of a loss of us 33-1 regardless of logic is bypassed in	a perf her co dentif 1-13A 57278 d that ring o d the ag rin e dete were s rmed o ergenc nit 3 llance the un "B" LP vailab off-s of the in an a	valv valv vas the open s the open s pr rmin et pp n s y co prio s be n lik the tite loo 2/3 auto	d on tive y mal re by writt hand e SMU /close ed. roper 16/86 re cd r to gan a ely e ystem o mit power se te dies matic	jun act fun cyc ien ien ien ien ien ien ien ien ien ien	nction tion i nction cling to fu eel re 00 Lim limits the of curren . DOS to decl ing sy strup of 9/8 to f isolat isolat isolat ser inal sy	a box s s s s three trine tri tri tri tri tri tri tri tri tri tri	e r ng que e tor sss	

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	REPORT (LER) TEXT CONTIN	UATIO	N		**		-			
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		*8.4.8	8.6		AL P		T	T	-	
Turkey Point Unit 3	0 15 10 10 10 1 2 5 0	8,6	0	. 3.6				2 00	. 0	5.1
TAT IF many appears a required, while additional falls, farms Miles at 177		111	-	11			1	- 0	5	1
(EDGs) were declared out of serv for its operability run as per Oper	rating Procedure (OP) 4304.1 This was being done becaus	Eme	TOPO	cy D	iese	Cen		4.0.0		

CAUSE OF EVENT:

3.0.1.

An investigation into the cause of the event determined that the governor solenoid was out of odjustment.

its operability test at 0447 and was declared back in service. This took the units out of TS

.

ANALYSIS OF EVENT:

The B EDG had been taken out of service for instrument calibration by valving out the air start supply to the B EDG. The instrument calibrations would not have prevented the B EDG from starting. The B EDG could have been manually started by valving the air start supply if it had been necessary within a few minutes. In addition, the A EDG had just satisfactorily completed its daily run and during the shutdown sequence of the test, the EDG could not be stopped in its normal fashion. Based on the above, the health and safety of the public were not affected.

CORRECTIVE ACTIONS: 1)

An Event Response Team was formed to address the failure of the A EDG to stop after its daily operability test. The team utilized Administrative Procedure 0-ADM-011, Short Notice Outage Work (SNOW) Response Organization. They determined that the governor solenoid was out of adjustment.

2) The governor solenoid was properly adjusted and the A EDG was satisfactorily tested at 0045 on November 7, 1986, and declared back in service.

18.8.5 F.O.R.M. 386.4 19.43

and the second se	PORT (LER) TEXT CONTINU				620.00	15 8/2		1 80-0	-
ACILITY NAME (S)	DOCKET NUMBER DI	1				1	,	-	
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Turkey Point Unit 3	0 5 0 0 0 2 5 0	016	-0	1316	-0	10	013	OF	0 1
XT IF must guote a required, use additional Add, Amma 2004 3/ (17)		to and the second							
 Preventative Maintenance Instrumentation Calibrations adjustment. ADDITIONAL DETAILS: OP 4304.1 has been cancelled an included in Operating Surveillance Test. The two (2) EDGs used at Tu Inc. The engine was manufactured 20-645E4. The governor was suppli The solenoid type is de-energize to 85-043, 250-86-014, and 250-86-022 	, will be revised to d the information conta e Procedure 0-08P-023.1 rkey Point were supplied by General Motors, Elec ed by Woodward Governo shutdown. Similar Occu	includ ined by A. ctro N	in the sel (G. Sel (hat p Gener choon e Div	roced ator make ision, del n	or i Ope r Co Mo	is r trabi ompo xdel l er U	now lity No. G8.	

485 / pr = 386 (11)			LIC	ENSEE EVE	NTRE	PORT	(LER)		CLEAR RESUL	80 2:50 -	
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EVENT

On December 4, 1986, during the Operability Verification of Auxiliary Feedwater Train 2, the valve position indication for MOV-3-1403 was lost and the valve was declared out of service. At this time the B auxiliary feedwater (AFW) pump was out of service because of overspeed problems. This condition did not meet the definition of an operable AFW train, as described in the AFW System operating procedure, so the requirements of Technical Specification (TS) 3.8.5 were exceeded and Unit 3 entered TS 3.0.1 requiring the unit to be in hot standby within 7 hours. Evaluations were begun to determine the acceptability of aligning AFW pump C and MOV-4-1404 to AFW train 2. An evaluation by our Engineering Department determining the alignment of the C AFW pump and MOV-4-1404 to AFW train 2 to be acceptable was completed. The evaluation was then reviewed and concurred with by the Plant Nuclear Safety Committee (PNSC). Unit 3 was taken out of TS 3.0.1 and back into TS 3.8.5. The Cause of MOV-3-1403 being out of service was burned motor leads on the valve actuator. Similar valve octuators were inspected and no similar conditions were found. The cause of the B AFW pump being out of service was drift on the electronic overspeed trip setpoint. The setpoint was readjusted and the pump was satisfactorily tested.

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LICENSEE EVEN	T REPORT (LER) TEXT CONT	INUATION		ULATORY COMMISSION VE NO 3150-0104 BE
FACILITY NAME IT	DOCKET NUMBER 12	LER NUMB	£ # . #	PAGE 12
		YEAR MOUL	AL. AL. ACN N. WELK	
Turkey Point Unit 3	0 15 0 0 0 2 5	10 86-013	18 -010	0201014

EVENT:

On December 4, 1986, at 0550, while Unit 3 and Unit 4 were at 100% power, Operating Surveillance Procedure (OSP) 3-OSP-075.2, Auxiliary Feedwater Train 2 Operability Verification, was initiated to verify the operability of steam supply motor operated volve, MOV-3-1403, and flow transmitter (FT) FT-3-1458B in auxiliary feedwater (AFW) train 2. Procedure 3-OSP-075.2 was satisfactorily completed at 0620, however, during the shutdown sequence of the B AFW pump, the pump tripped on electronic overspeed. The B AFW pump was declared out of service and both units were placed in a 12 hour limiting condition for operation (LCO) as per Technical Specification (TS) 3.8.5.d. At 0730 the B AFW pump was tested to try and repeat the previous electronic overspeed trip but the trip could not be duplicated. The pump was kept out of service to continue troubleshoating the problem.

The AFW system at Turk+y Point consists of 3 turbine driven pumps of which two pumps are normally aligned to train 1 and one pump is aligned to train 2. Steam can be supplied to the pump turbines from either or both units through redundant steam headers. Two D. C. MOVs (MOV-*-1403 and MOV-*-1405) and one A.C. MOV (MOV-*-1404) on each unit isolate the 3 main steam lines from these headers

At 0745, Operations begon uligning the C AFW pump to train 2, as a result of the overspeed problem. At 0930, 3-OSP-075.2 was commenced to test the C AFW pump to train 2. At 0953, during the performance of 3-OSP-075-2, the valve position indication for MOV-3-1403 was lost. The valve was declared out of service and because operating procedure (OP) 3-OP-075, Auxiliary Feedwater System, describes AFW train 2 as consisting of MOV-3-1403 and AFW pump B during dual unit operation, Unit 3 exceeded the requirements of TS 3.8.5 and entered TS 3.0.1 requiring the unit to be in hot standby within (7) hours. Evaluations were begun to determine the acceptability of aligning AFW pump C and MOV-4-1404 to AFW train 2. At 1300, 3-OSP-075.2 was completed with the C AFW pump and MOV-4-1404 aligned to train 2. At 1345, 4-OSP-075.2 was completed with the C AFW pump aligned to train 2 on Unit 4. At 1435, an evaluation by our Engineering Department of the C AFW pump and MOV-4-1404 aligned to AFW train 2 was reviewed by the Plant Nuclear Safety Committee (PNSC) along with an on-the-spot-change (OTSC) to 3-OP-075 and the PNSC concurred with the evaluation. Unit 3 was taken out of TS 3.0.1 and back into TS 3.8.5.

An investigation into the overspeed trip of the B AFW pump found that the electronic overspeed trip setpoint had drifted low. The setpoint was readjusted and satisfactorily tested as per 0-05P-075.9, AFW Overspeed Test, at 0340 on December 5, 1986. 3-05P-075.2 was satisfactorily completed at 0715 and the B AFW pump was declared back in service and both units were out of TS 3.8.5.

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An inspection of the motor for MOV-3-1403 revealed burnt motor leads. Plant management decided to inspect similar motors to see if any other problems existed. At the time the inspections were begun Unit 4 was in mode 2 (start- up) for a scheduled mainter ace outage. Beginning at 1627 on December 6, 1986, MOV-4-1403, MOV-4-1405, MOV-4459A, MOV-6459B, and MOV-6459C were taken out of service one valve at a time for inspections. MOVs 6459A, 6459B, and 6459C are the throttle and trip valves for the AFW pumps. These valves automatically trip closed to protect the turbine from an overspeed condition. TS 3.8.4.b requires two independent AFW trains and a third AFW pump to be operable whenever both units are above mode 4 (hot shutdown). While these valves were out of service, the requirements of TS 3.8.4.b were exceeded which placed Unit 4 under the requirements of TS 3.0.1.

At 0325 on December 7, 1986, MOV-6459C was taken out of service for inspections. At 0345, AFW flow indicator HIC-3-1457B was declared out of service due to indicating flow when a no flow condition was present. This resulted in declaring AFW train 2 out of service to Unit 3. Since the C AFW pump was out of service, this placed AFW train 1 out of service which exceeded the requirements of TS 3.8.5 and placed Unit 3 in TS 3.0.1. The C AFW pump was declared back in service at 0346 which placed Unit 3 back in TS 3.8.4.5.

CAUSE OF EVENT:

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The cause of the electronic overspeed trip of the B AFW pump was due to setpoint drift. The cause of the problems with MOV-3-1403 was burnt motor leads. The cause of the problems with the flow transmitters was air in the sense lines.

ANALYSIS OF EVENT:

During the event, on December 4, 1986, both the A and C AFW pumps were operable and capable of supplying feedwater to the Unit 3 and Unit 4 steam generators. Also at least two steam supply MOVs on Unit 3 and 3 steam supply MOVs on Unit 4 were operable and capable of supplying steam to the AFW pumps. During the inspections of the MOVs only one valve was out of service at a time, therefore, at least one train of AFW was available to each unit. During the time that HIC+3-1457B was out of service, train 2 was technically declared out of service, however, the train would have been able to deliver feedwater to the steam generators if the need had arisen. Based on the above the health and safety of the public was not affected.

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CORRECT	TIVE ACTIONS:																		

- An evaluation was done by our Engineering Department to assess the acceptability of two unit power operation with MOV-3-1403 and AFW pump B out of service simultaneously. Engineering concluded that dual unit operation may continue for 72 hours (TS 3.8.5.b) provided that AFW pump C is aligned to train 2. This was done and an OTSC was written to 3-OP-075 to reflect this new alignment. The evaluation and OTSC were reviewed and concurred with by the PNSC.
- An inspection of MOV-3-1403 revealed burned motor leads. The motor was replaced, and the defective motor was sent off-site for an external root-cause analysis.
- Due to concerns over Limitorque DC operators, as indentified at another plant, MOV's with similar operators were inspected. The inspections did not reveal any similar problems.
- 4) The B AFW pump electronic overspeed trip setpoint was found to have drifted low. The setpoint was readjusted and the electronic overspeed trip setpoint satisfactorily tested.
- The flow transmitter for HIC-3-14578 was vented and satisfactorily tested for operability as per 3-OSP-075.2 at 2000 on December 7, 1986.

ADDITIONAL DETAILS:

The AFW pumps at Turkey Point are steam driven turbine pumps, type Terry 254, manufactured by the Terry Corporation, which is subsidiary of Ingersoll Rand. MOV-3-1403 and MOV-3-1405 are 3 inch Walworth gate valves with Limitarque SMB-00 actuators. MOV-4-1403 and MOV-4-1405 are 4 inch Velan Globe valves with Limitarque SMB-00 actuators. MOV-6459A, MOV-6459B, and MOV-6459C are 3 inch Gimpel Corporation Globe valves with Limitarque SMB-000 actuators. Similar Occurrences: LERs 250-85-037 and 250-86-016

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system pressure and a subsequent rapid electrical load decrease from 730 MWe to 0 MWe. No automatic control rod insertion occurred. The Reactor Control Operator (RCO), noting that the coolant temperature was increasing above the reference temperature, placed the rods under manual control, and initiated rod insertion. Concurrently, a second RCC attempted to raise the oil pressure, unsuccessfully. At this time, (about 24 seconds into the transient) It became clear that the unit could not be recovered, and it was manually tripped. During the transient, a PORV opened, then would not fully close, necessitating closure of the associated block valve. The unit was then stabilized, in less than 5 minutes. The most probable cause of the drop in oil pressure was the clearing of blockage of the governor impeller arifice, resulting in the auxiliary governor dumping control ail. The control rods failed to automatically insert due to two cold solder joints in the final variable gain summator of the power mismatch circuit. The cause of the PORV failure to close is still under investigation. The PORV, and turbine governor impeller and associated components were inspected, and no problems were found. The cold solder joints were repaired. The control, lube, and sed ail piping will be cleaned.

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EVENT

On December 27, 1987, at 0941, Unit 3 was tripped monually following a loss of turbine governor oil system pressure and a subsequent rapid electrical load decrease from 730 MWe to 0 MWe. The initial indications of a transient were several steam generator (SG) related alarms annunciating. The SG feedwater flow control valves were verified to be operating properly, in the "auto" mode, and were fully open. At 0940, the indicated turbine electrical load and turbine governor oil pressure began to decrease rapidly, without any decrease in reactor power. Automatic control rod insertion did not occur. The Unit 3 Reactor Control Operator (RCO), noting that the reactor coolant average temperature (Tave) was increasing above the coolant reference temperature (Tref), placed the control rods under manual control, and initiated manual insertion. Concurrently a second RCO attempted to raise the governor oil pressure, without success. Starting from the fully withdrawn position, (step 228), the manual rod insertion continued until step 214 was reached. At this time, (approximately 24 seconds into the transient) it became clear that the unit could not be recovered, as:

- the reactor was still generating significant power
- the turbine electrical load was at 0 MWe, and the turbine governor oil pressure was not responding to the RCO's actions (a) b)
- the SG safety valves were lifting, and c)
- the steam dumps were fully open. d)

With the unit in the above conditions, at the request of the Assistant Plant Supervisor-Nuclear, a second RCO manually tripped the reactor at 0941.

During the transient, the Reactor Coolant System (RCS) pressure was increasing, until Power Operated Relief Valve (PORV) PCV-3-456 opened. The maximum RCS pressure reached was 2330 psig, just prior to the opening of the PORV. Approximately 26 seconds after the trip, the Low Pressurizer Pressure Safety Injection Block alarmed, indicating that pressurizer pressure had decreased below 2000 psig. Following the alarm, the RCO's noted that the pressure was continuing to decrease. The pressurizer spray valve controller demand signal was verified to be zero (the demand signal for closed spray valves) and a check of the PORVs revealed that PCV-3-456 had dual position indication, suggesting that it had not closed fully. The RCO then attempted to close it manually. After the PORV failed to close under manual control, at 0942 the RCO manually closed block valve MOV-3-535, which is the block valve associated with PCV-3-456, halting the pressurizer pressure decrease. PCV-3-456 also closed at about this time, without any additional operator intervention. The minimum RCS pressure reached was 1760 psig. No Safety Injection (SI) occurred, as the set point for SI is 1723 psig. The unit was stablized, with pressurizer pressure increasing, in less than 5 minutes.

Investigations were initiated into the causes of the PORV malfunction, automatic rod control failure to insert, and the governor control oil pressure loss. At the conclusion of the investigations, required maintenance, PNSC approval of the Past Trip Review, and ofter receiving Plant Manager approval, the reactor returned to criticality on January 4, 1987.

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CAUSE OF EVENT

The immediate cause of the manual reactor trip was loss of turbine electrical load, due to a loss of turbine governor control oil pressure. A gradual decrease in impeller oil pressure was noted in the days prior to the event. The pressure just prior to the event was 24 psig. The most likely cause of the pressure decrease was a blockage of the impeller orifice caused by a gradual accumulation of dirt. The most probable cause of the blockage. Upon the blockage clearing, impeller oil pressure would have suddenly increased about 12% to approximately its pre-blockage pressure (over 27 psig). The auxiliary governor interprets oil control pressure increases greater that 3% per second as excessively swift increase in turbine speed, and the result is that the auxiliary governor starts dumping control oil. The loss of control oil is followed by automatic closure of the governor and steam intercept valves.

The absence of any automatic control rod insertion was due to two reasons:

- 1) The Tave Tref circuits functioned as designed. Even an a sudden, complete load rejection, it takes approximately 15 seconds for this partian of the control circuit to initiate rod motion. This is consistent with the time constants in the circuit lead/lag units. During this event the load rejection was gradual and the time for total load rejection was approximately 15-20 seconds. This circuit would have eventually initiated inward rod motion, but with a time delay that would not have prevented this transient.
- 2) The power mismatch circuit should have initiated prompt rod motion on the loss of turbine load. Troubleshooting identified two cold solder joints, resulting in no output, on the final variable gain summator. This would have prevented the circuit from functioning. Testing performed after repair of the cold solder joints showed that the nonlinear gain unit would break down to zero output during rapid load changes. Bench testing of the unit revealed normal responses to static load conditions. A new function generator was installed and calibrated. Normal system response was obtained during subsequent testing.

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The cause of the PORV failure to fully close is under investigation. Extensive troubleshooting, including dissembly, bench testing, and operation with flow through the valve, failed to identify any malfunctions, or any reason for the earlier failure to fully close.

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POCKET NUMBER (2)	1	LA NUMBER IS		Face 3
1. S. S. S. F. F.	+84.4	SECURE A	No. 481.8	
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		+1.4*	DOCKEY NUMBER (2)	DOCK17 HUMBLE (2)

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EVENT DESCRIPTION

On January 28, 1986, the Plant was operating at approximately 80% power. Emergency Diesel Generator (EDG) "B" had just been taken out-of-service to install a solid state overcurrent trip device (amptector) on the EDG "B" output breaker. This breaker upgrade was being performed on all Westinghouse tie DB safety-related breakers and was completed on EDG "A" the week before. At 0917 hours, the EDG "B" output breaker had just been "racked out" when the Emergency Bus "E-2" was lost. This also resulted in the loss of Instrument Bus 4 (IB-4) which is supplied by Motor Control Center "MCC-6". Nuclear Instrumentation System Power Range Channel N-44 (fed from IB-4) was lost, which initiated a turbine runback (rod drop feature). The automatic rod control system (input from N-44) and the steam dump control system (powered from IB-4) could not function properly. As a result, a reactor trip was received on "Hi Pressuriker Pressure" approximately twenty-one seconds after "E-2" was lost.

One minute after the reactor trip, the main generator oil circuit breakers (OCBs) opened and the Plant Auxiliaries (those powered by the Auxiliary Transformer during operation) shifted to the Startup Transformer as part of the normal Turbine Generator Lockout feature. Approximately one second later, a West Bus Lockout occurred in the 115 KV switchyerd which demengized the Unit No. 2 Startup Transformer (See Figure 1). This resulted in a loss of offsite AC power. EDC "A" started automatically and loaded Emergency Bus "E-1". Approximately sixty=seven suconds after the West Bus Lockout was received, a Safety Injection (SI) and Main Steam Isolation Valve (MSIV) signal were received. These were caused by High Steam Line Flow coincident with Low Tave. The Low Tave signal was caused by the Plant cooldown as a result of the Reactor Trip. The High Steam Line Flow signal was present due to loss of IB-4. When the loss of offsite AC power occurred, all three Reactor Coolant Pumps coasted down and the Plant was cooled by natural circulation flow. The Plant was stabilized at Hot Shutdown conditions with Reactor Coolant System (RCS) temperature being controlled with the Steam Generator (S/G) PORVs. An Unusual Event was declared at 0935.

At 1027 hours, power was restored to "E-2" by manually starting and loading EDG "B".

At 1115 hours, after investigation revealed no faulted condition on the Startu-Transformer and its associated circuits, offsite AC power was restored to the Plant non-wital Electrical Distribution System.

At 1228 hours, a second SI signal was received. This was caused by "C" Steam Line Bigh Differential Pressure which resulted when frozen sensing lines caused "C" S/C PORV to stick open. The "C" PORV was closed by isolating the air supply to the PORV, thus, correcting the situation. The freezing condition resulted when power was lost to freeze protection circuits.

Throughout the event, RC3 pressure remained above the shutoff head of the SI pumps: therefore, no SI flow entered the RCS. At no time did RCS temperature and pressure approach saturation. Offsite AC power was restored to the entire Plant Electrical Distribution System at 1601 hours and the Unusual Event was terminated at 1634 hours.

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					46.355 5.8113		
	Robinson	0 8 0 0 0 216 1	8 6 -	01015	- 012	013	01 1
	e a resultation, une antificiaria Antific Fauna abbat y 1135						
During	the Loss of Offsite Power	, other conditions resu	ilted e	s follo	ws:		
1.	The Control Room Ventilat Emergency Recirculation M caused by the loss of pow corrected when the power	ode when the initial SI er to "MCC-6" (supplied was restored to "E-2" a	signa by "E	1 was r -2"). C-6".	eceived This even This por	. Th ent w	43
	failure mode, including c the Control Room Habitabi	orrective action, was a lity issue (NUREG-0737,	Item	ed in C No. III	.D.3.4)	spon	se to
2.	During the event (from 09 Containment Vessel (CV) w accompanying the initial FP-256 and FP-248. Techn station in the CV is out- one (1) hour. Although t reopening these valves (a the Operators to concentr power. If required, thes	as isolated when the Ph SI signal closed the Fi ical Specification 3.14 of-service, back-up pro he Phase "A" Isolation remote operation) was ate on the actions bein	ase "A re Wat .4.2 r tectio Signal intent g take	" Isola er CV I equires n must was re ionally n to re	tion Sig solation that if be provi set at (postpor	nal Val ded 0936 med c feix	ose withi hours o all
3.	During the initial attemp cycling on "E-2" was obse power fuses from breakers pulled all the control po pulled. SI Pump "B" is s breaker is physically loc stopped in accordance wit "E-1" was available, SI P "E-1".	rved. An operator was on "E-2" to prevent br wer fuses; however, SI upplied from the "E-1" ated on the "E-2" Bus. h End Path Procedure. E	direct eaker Pump " and "E The p PP-7.	ed to r damage. 8" fuse -2" Tie ump had SI Term	emove al The Op s were a Bus, bu already	l co berat ilso it it bee si	ntrol or s
	Since the control power f for automatic start or main not required at the time. closing the breaker or in fuses for the "E-2" Bus 1 "E-2" Undervoltage (UV) R affected components to the	nual start for approxim SI Pump "B" could hav stalling the control po oads (including SI Pump elay Fuge was replaced.	ately e been ver fu "B") This	40 minu starte ses. Th were re-	tes. Si d by loc ne contr placed a	Pum ally ol p	ower
suppli Diesel the Er of tim	this event, there was no ed needed power to one of was available if needed. ergency Operating Procedur without any AC power unt Emergency Diesels, or the	two redundant Emergency In addition, appropria es (EOPs) to concrol th ii some form of AC powe	Busse te pro e Plan r is r	s. The visions t for at	Dedicat are ava	ed Si ilab	hutdo le in eriod

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and environments a second of				
USE/CORRECTIVE ACTION				
tensive investigations have bee	n performed to identif	y the cause	of the ev	ent. As
sult of these investigations have bee ergency Bus "E-2" and the loss	k has been concluded t	DAC (WO MA)	or events	11033 01
ergency Bus "E-2" and the loss on one another.	of offsice Au power/ w	ere separat	C 804 104	
SS OF EMERGENCY BUS "E-2"				
e potential cause of the loss o	f Emergency Bus "E-2"	is attribut	ed to a bl	lown fuse
e secondary side of the Potenti lay. This would have caused a	al Transformer (PT) 51	0019108 008	L-2 001	5 4 7
second affected from the book compain	Pha fues Use FPD acpd.	108 5-4	BUS UV IN	elays.
initian the bus loss of nover as	sugare which sheds t	he motor lo	ads off th	ne pus py
the she land basebase asses	Pha normal (offelts bo	WALL SUDDLY	Dreaker	308703 00
" and closes EDG "B" output bre uld not start nor could its out	aker. However, since	the blown	Euse was	replaced
-ing the sugar An extendive	NUPSTIGSTION OF THE LL	G B OULDU	IC Dreaker	, 5-2 00
send ashings associated citer	its, and wiring was be	ertormed, b	io unusual	COUGT (700
to and they would have caused	the blown fuse. The	blown tuse	was a cyp.	1081
re found that would have caused nrenewable type 6 amp fuse. In ndition. The blown fuse could	spection of the blown have been a random fai	lure that c	annot be	related to
e removal of EDG "B" from servi	ce.			
e second potential cause of the		crovered on	March 6.	1986. whi
· Bloos was shutdown for refuel	ine. The Station Seri	vice Transfo	ormers (55	18/ 20 800
the second problem and an appress	A FAR MAINTARADER OF	the common i	UDDLY DIE	38.01 101
ese SSTs. While in process of -2"), the "E-2" normal supply b	energizing "E=2" via	e to degrad	ied voltag	e relay
-2"), the "2-2" normal supply to	preaker crapped open of			
				d high
e degraded voltage relays prote rrent conditions by opening the	a normal (offeite bove	r) subbly br	reaker to	rue witer
ergency Bus should bus voltage	become less than 415	volts for an	pproximate	1y 10
conds.				
bsequent reviews and inspection	is revealed that the c	ause of the	degraded	voltage
Your appropriate time a longer fille	holder on the degrade	d voltage L	mergency o	103 5 6 1
parently caused the fuse holder	tie breaker was close	d during ch	e occurren	
			. in the	di mari a
e fuses and fuse holders on But	ses "E-1" and "E-2" we	re inspecte	d and elev	R on "E-2
ere found to be loose to some de	seves and were replace	di AD EVAL	ngerou ra	Attracted
termine a better method to fus	e these circuits on a	long-term b	asis to pr	revent the
ose fuse holder problem.				

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NCIUTY RANG (1)	DOCK 67 HUMBER (2)	117 #11 1 1 1 1
	A second s	HEAR MARTIN HALLS
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DY if nere good a repartit, an internet sall, fam aller 2 137		1 8 6 - 0 0 5 - 0 2 0 5 0 1 0
Subsequent investigation aft fuse holder could have cause operator racking out the EDG sufficient vibration to open If the loose fuse holder did addressed earlier could have events that had occurred, it actual initiating event. Both the UV and degraded vol fuse failures. A modificati buses to increase the rating protection of the PT seconda	d the loss of "E-2" on Ja "B" output breaker for a the connection in the fu- initiate the loss of "E- blown when the "E-2" Bus would have not been easi tage relays were determined on has been performed to of the PT primary side for try side to the relays.	-2", then the UV relay fuse s was reenergized. Because of the ily discernable [®] which was the ned to be susceptible to random PT these circuits for both emergency fuses and to eliminate fuse This modification provides
sufficient circuit protectio failure. Should a PT fail, from the Emergency Bus. The completed prior to startup f	on while minimizing the vo the higher rated primary s modification was perform	ulnerability to a random fuse fuses would still clear a fault med to both Emergency Buses and was
LOSS OF OFFSITE POWER		
power. The Plant trip cause the Startup Transformer. Co phase Differential Relay ope 115 kV West Bus Lockout Rela	ed A normal transfer of the bindident with the transf- erated on the Startup Transfer ay, which in turn opened	he cause of the los: of the offsit he Auxiliary Transformer load to er of the load, however, a "C" nsformer. This relay tripped the all source and load supply circuit Unit No. 2 Startup Transformer
a tripping of the Startup Tr were systematically examined included oil samples from th the transformers relays Onsi breakers and related buses, system. Westinghouse was co During the course of the inv actual conditions present we	ransformer due to a "C" p d. Some of the tests and he Startup Transformer, s ite and at Westinghouse, and breaker coordination onsulted for independent vestigation, all aspects ere anglyzed. None of th component failures which c cted during breaker coord	itions that could have resulted in hase Differential Relay operation inspections that were performed etpoint checks and bench testing o inspection of associated circuit testing of the auxiliary transfer analysis and recommendations. of equipment, equipment inputs, an e inspections or tests identified could have contributed to the lination testing revealed a
the the three (3) current to Transformer are susceptible	ransformers (CT) on the p to DC saturation. These	nstalled equipment revealed that primary side of the Startup + CTs provide the primary (115 KV) + a primary CT were in saturation,

TY BANK (1)		DOCK #7 NUMBER (3)	(cf+Q-12-0
		DOCKET NOMEN OF	-14.4 (112-14) AL (112-2) -14.4 (112-14) AL (112-2) -14.4 (112-14) AL (112-2) -14.4 (112-14) AL (112-2)
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17	a managerial, and antideness sails frame Millel to 117		
the Sta Lockout The cord during transfi the CT circuit condit curren West B Condit	Auxiliary load on Unit i CP&L system voltage pro- CP&L system voltage pro- current effects more pro- current freeze pro- current freeze pro- current effects more pro- estimation of the pro- current effects more pro- current effe	rate, leading to a prof Ts are not susceptible esults from a DC compor ransformer at the time y to this saturation i er the in-rush conditi. .). Although this sus n present during the o ly had not caused CT s that operated a Differ ed during the operatin 18, 1986, that apparent (ile had been increased twas operating at full lower system impedance) adominant. 2 was slightly higher d running of other equip ircuits creating a high ductive loads) more dur of auxiliary power use e (approximately) dead the phase angle increased his will further increased	nent of the AC in-rush current
	Auxiliary load at Rob.s	and has slowly increase	ed over the years due to backfit
4.	modifications.		ons on that morning, each

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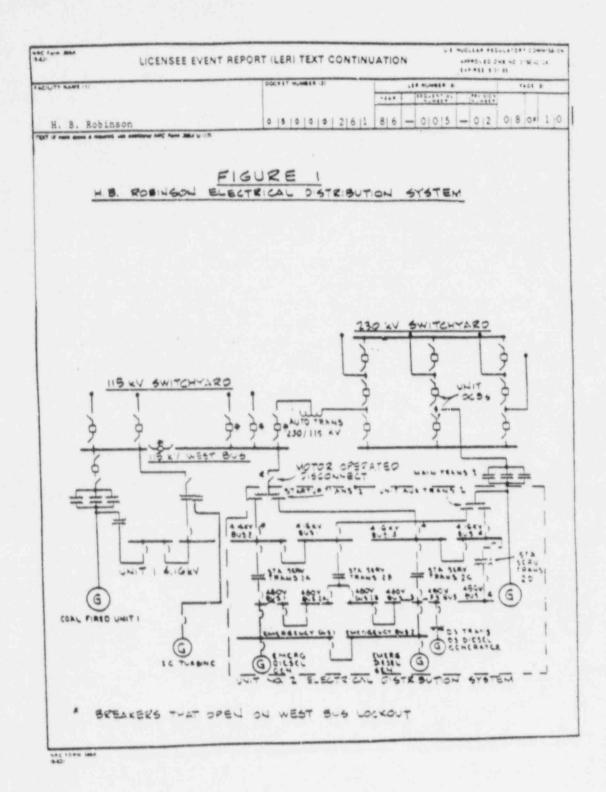
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	DOCKET NUMBER ()	LER NUR		***** 3
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Finder scores a resumera, we analysing with fame applies or 1171	the second se	and and and and and a		
The susceptibility of the DC satu assentially increases the rating second GT in parallel. This modi primary side of the Startup Trans refueling outage. This modificat PS4L believes that its systematic Plant's Electrical Distribution S preclude this event from occurrin ADDENDUM	of each CT (increased fication was perform former and was completion ion is based on recor- and detailed review ystem and the identiti	d turn ratio ed on all th eted prior to mmendations of the affe	 and conn aree phases startup from Westing 	ects a of the from the nghouse.
During the week of February 17, 1 Input CTs were evaluated for oper transfer of unit auxiliary load f data available at the time, the d determined not to be susceptible. reveal susceptibility and subsequ	ational susceptibilit rom the Auxiliary Tra ifferential input 400 Calculations perfor	ty to DC sat ansformer, DO/5 amp CTs rmed for the	uration du Based on b on the 4 115 kV si	ring est-estima kV side us
Follow-up review of all data de refinement of original calco scenario. CT nameplate data specific CT characteristic curve	<pre>#4 during the init account for a b provided to the the curve was obtain account was obtain account of the second second account of the second second account for a basis account for account for account for a basis account for account for account</pre>	manufacture	worst cas	e equest for
Mey parameters determined from the may have been in error. Specific, were also susceptible to DC satur- the 4 kV side CTs were tested using confirmed the accuracy of the CT wide CTs to DC saturation was con-	ally, the new calcula ation. To assure con ng equipment not avai characteristic curve.	ations indic rrect input ilable in Fe	data was b	kV side (eing used, here rest
It was recognized at that time the of improved characteristic CTs. ionstruction, involving a lead the rolution, a 10-cycle time delay in relay output contacts to the West moure that if DC saturation were rould be allowed for saturation de ion-nuclear safety-related function ogic creates no safety concern.	These CTs, however, we me of several months, n the trip path from 115 kV Buss Lockout to occur, sufficient ecay. Since the rela	would requir In order the S/U Tra Relay was i time (appr w different	e special to provide nsformer d nstalled. oximately ial scheme	design and an interi ifferentia This woul 5 cycles) provides
The installation of the 10-cycle sturation was accomplished August the potential for DC saturation is	t 8. 1986. Replaceme	ant of the 4	bility to 000/5 CTs	DC co elimina

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JANUARY 28, 1986, 5	EQUENCI	OF EVENTS							
and the second s									
APPROX. TIME (HR/MIN/SEC)			EVENT DESCRIPTIO	N					
Initial Condition	-	Plant at	80% power						
		Plant coa	sting down for refu	elin	g.				
0915:00		EDG "B" o	utput breaker racke	d ou	t for bre	aker up	grade		
0917:15	-		mergency Bus E-2, 8 t Bus IB-4, and all						s .
	-	Initiacio	n of Turbine Runbac	:k.					
0917:36	-	Reactor T	rip - "High Pressur	rizer	Pressure	×**			
0918:36	-	Turbine G Transform	enerator Lockout () er loads shifted to	sta	OCBs open rtup Tran	ed and sformer	Auxi).	liary	
0918:37		Loss of o West Bus	ffsite power (Star: Lockout.	tup T	ransforme	r) due	to 1	15 kV	ſ
	-	ED: "A" A	uto Start.						
	~		Reactor Goolant Po on initiated).	umps	coasted d	own (Na	cura	1	
0919:44	-	SI and MS with low	IV closure signal - Tave".	- "Hi	gh Steaml	ine Flo	v co	incid	iet
0935:00		Declared	Unusual Event.						
0936:00	-	Reset SI,	Phase "A", and Fe	edwat	er Isolat	ion.			
0941:00	. •		ocal Start - Attem out prevented due t					ergei	ne.
		signal.	Investigation foun- he Potential Trans	d bla	wn fuse i	n the s	econ	dary UV	
1027100		Bus E-2 1	oaded.						
1058:00		Energized	Startup Transform	er.					
1115:00	1		offsite power from and 2 (Non-vital 1			former	to 4	kV	

UC UC	ENSEE EN	ENT REPOR	T (LER) TEXT CONTIN	UATION	44480-60 246 40 1 5244 5 44480-60 246 40 1 524 5 644865 8 21 55
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				(Let] [110-15" +	HULLS STATE
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T if many spaces a negative, was addressing		1175			
JANUARY 28, 1986.	SEQUENC	E OF EVENT	5		
APPROX. TIME (HR/MIN/SEC)			EVENT DESCRIPT	ON	
1226:00		"C" \$/G	PORV stuck open (du	le to frozen s	ensing lines).
1228:00			I signal - "Steam 1	Line "C" High	Differential
		Pressure			
1230:00	-	"C" \$/G	PORV closed by iso	lating air sup	ply to the PORV.
1237:00		Reset se	cond SI, Phase "A"	, and Feedwate	r Isolation.
1255:00	•	E-1 plac	ed on offsite power	r, secured EDG	."A" .
1304:00	-	Restored Busses 3	offsite power from and 4.	m Startup Tran	sformer to 4 kV
1601:00		E-2 plac	ed on offsite powe	r, secured EDC	"B".
		Plant Po	wer Distribution r	estoration to	normal complete.
1634:00		Terminat	ed Unusual Event.		
					Contraction of the second second second

did).			Lic	ENSEE EVE	NT RE	PORT	(LER)		UCLEAN REQUES	7081 CONNENSION NO 31 M -0104
FACILITY NAME	(1)							DOCT OF HUMBER	1.01	1011
Ocon	ee Nuclea	r Station,	Unit 1							1 01 013
Gene EVENT BAT	rator/Rea	Les montes .	Due to	Failure of	PCB	-20 11	n 230 kv	Switchyar	d	
MONTH DAY		and the second se		NGRONT BAT	VEAR		PADUTY NA	FACILITIES HEY I	00-1617 10-0010	
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GREALTING MIL	7468.8	87087 & 8,480×778	D PURBULART I		ATS OF 1	CP# \$ 0		e' die Nerssang (1	15	
ACRIME &	and the second se	Allenating	-	28.495-s		X	18.78ardices		12,118	
LEVEL 1	Research Co.	Call True 100		N She D		-	M. That Bress		Y STREE IS	
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A. 100				CENINE I CONTINCT	ADA THEA	1.88 112			-	
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M. A	. Heghi.							0 1014	3,713 -	1410 1610
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CAUSE FYFTEM	COMPONEN?	NAME A	PEPCS ALL		CAUSE	1.414	COMPOSEN?	NAMUTAL TURES	TO MEDI	-
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a province of the section of	ter den de enderen		and the second se	1XH10710 18				- I dad	with 1	N 047 1184.8
_		and the second se						8,84 34	80	
125 17 VB 1		Rubersbolw da 19		*		_		DATE -		1 1 1 1
when t breaks breaks explod failur	the yellow the yellow tr (PCB-2) r (PCB-2) de causir e of PCB-	<pre>>> Generat urbine/Rea >reaker (P >> bus lock () opened.)), which ag the Tur -20 was ap;</pre>	or/Turb ctor tr CB-20) out occu Immed: was sub; bine/Ger parentl;	urs. Unic ine Antici ip was ini failed fol urred the is aly aft jected to herator to y caused b	pator tiate lowin Unit erwar the e trip y deg	y Rea d whe g a 2 l gen ds th ntire on u raded	etor tri n a 230 30 kv ye. erator y e parall. generat. mder-vol breaker	p signal. KV switch llow bus ellow bus el genera or output tage. Th contacts	The yard tie tor , e	
The im The im Dhit 1	ter pumpe s. A pre- t run bac mediate c pplementa was allo	pped on h starred : liminary : k as expect orrective l correct; wed to res	igh disc immediat investig tted. action ive action ive action	h Main Fee tharge pre- tely and r gation has was to at.	ssure emain reve sbili d tha	. Al ed in aled ze th t PCB	1 three wervice the MPWP e unit at -20 was :	mergency for abou speed de bot shu solated	t 38 mænds tdown. and	
104 18				lic was n	ot af	fecte	d by this	inciden	£	16 27
2 Parm 386	660	3170028 ADOCK	05000	269 -				construction operation		1 11

LICEN	SEE EVENT REPO	RT (LER) TEXT CONTIN	NUATION APPROVE	5 0with NO 3150-2104 8/21/83
C/74 NAME (11		DOGEET NUMBER OF	LER NUMBER (6	Page 3
			YEAR BEQUENTIAL MAILE	11
And the second second	and a start of the	0 15 10 10 10 15 16 10	9 816 - d d 1 - al	0 0 2 0 0
Oconee Nuclear Sta			Carta a se a contration	
BACKGROUND				
transformer, pow fed from either Power up the sta station from any transmission cir hydroelectric um	er is supplied or both of the irtup transforme one of fourtee cuits, two much hits and the 500	buses in the 230 kv sr can flow through an supplies. These lear generating unit 0 kv switching stati	tor through the auxili its startup transform writching station. the 230 kv switching include eight 230 kv s if operating, two lon. When the 230 kv circuit breaker (PCB-1 o the 230 kv Switchyard	20)
and the Duke Sys	item.			
DESCRIPTION OF	COURRENCE			
problems experi- from both the r were checked an constant resist	ed and yell w 2 unbalance cond ance on PCE-22.	30 kv switchyard bu ition was found on	shooting for microwave were opened and isolat ses. When PCB-22 and FCE-23 as a result of	-23 high
reset of the ge relay logic was the 230 kv yell	satisfied to c ow hus tie brea	tause a yellow bus 1 akers opened.	ually closed without t breaker was closed th ockout. Consequently	11
Onit 1 gener 2 for current flo PCB-20 faulted, PCB-21 opened. anticipatory r: ITE Model 230-0 NPRDS. There 5	or to yellow but ow from Unit 1 G , undergoing an At 1547 hours eactor trip from GA-20-30 power : were no personn	Generator to the Swi explosion, approxim the turbine/generat m 100% stable power circuit breaker and el injuries as resu	a lockout was PCB-21, t it opened the only pa- tichyard was wis PCB-20 mately 17 seconds after for tripped initiating conditions. PCB-20 is is not reportable to lt of this explosion.	o, r an s an
high discharge speed demand d pumps started Emergency feed	pressure. A p id not run back immediately to water remained	as expected. All supply feedwater fl in service for about	Pumps (MPWP) tripped o ation has shown the MP three emergency feedwa ow for decay heat remo t 38 minutes.	val.
unit. On the was .016 gallo steam relief v	day of the incl m per minutes. alves (1MS-8) c	Following the reac opened for 11 minute		in
Unit 1 was sta	bilized at bot	shutdown conditions	with no actuations of relief valves and no	:

	ORT (LER) TEXT CONTIN	UATION AMBOVED D	
TY NAME (1)	DOCK IT MANDER ID	LEA NUMBER IN	- 25
		THER BIOLENTIAL MILECO	****
Oconee Nuclear Station. Unit 1	0 18 10 10 10 12 16 19		1.4.4
man anno e mourad, an ananana latt, April 2017	1.1.1.1.1.12 1619	1816 - 0 0 1 - 0 0	0305
CAUSE OF OCCURRENCE			
A discussion with the personnel procedure they were using to tr however a Maintenance Procedure breaker maintenance work in the maintenance procedure has in it the failure to notify the Shift in a switchyard isolation. How clear definition of responsibil lockout reset, PCB-24 was close set up a relaying fault conditi bus. The failure of PCB-20, however, this incident. The cause for t of the breaker contacts. Under is not expected to cause a unit PCB-21 opened, causing a turbin	ouble shoot is very a that gives more spec- 230 kv Switchyard. a safety considerati Supervisor to reset ever, due to misunder ity for verifying the d without the generat on and consequently 1 has been evaluated a be PCS-20 fault was a normal circumstances trip. In this case s/generator trip. PC	reneral. There is if is details concerning This breaker on statement indicating the lockout may result standing and lack of a i Dnit 2 generator or lockout reset. This locking out the yellow is the root cause of ittributed to failures , a yellow bus lockout PCB-20 failed when B-20 is rated to carry	8
the full output from the genera in which following a yellow bus faulted; therefore, this failur The failure of the MFWP to run	lockout, the alterna e is considered an is back, as expected, wa	ite generator breaker colated case.	
problem on one of the MFWFs. D when the integrated control sys mechanical linkage on the LA MF appropriate valve position did speed was below the normal spee feedwathr valve was below the s setpoint error built in the ICS saturate high. When the reacto expected because the saturated the pump speed to increase. Th consequently feedwater discharg approximately 1155 peig at which	tem (ICS) demanded a WP that converts an e not operate properly. d. Conrequently, the etpoint of 35 psi. O lA MFWP speed integr r tripped, the MFWP d integral was still at e feedwater valves ha e pressure increased h point both main fee	certain speed, the lectrical signal to an As a result, the pump AP across the wer a period of time a al which caused it to id not run back as a position calling for d gone closed, until it reached dwater pumps tripped.	
There have been four incidents main feedwater pumps, however, during this event; therefore, t isolated case.	the causes were unrel	ated to what occurred	
CORRECTIVE ACTION			
The immediate corrective action conditions. Supplementary corr PCB-20 as the cause of the reac	ective actions identi	fied the failure of	
N44			

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UCENSES EVENT REI	PORT (LER) TEXT CONTIN	UATION AMPROVED SHE NO 2 18 -015
Y RANG 11	DOCLET MAREER (2)	LEA NUMBER 18 PAGE 13
		TELE BEOLEY AL MURCH
		816 -0 1011 -01001400
Oconee Nuclear Station, Unit 1	0 0 0 0 0 0 2 0 9	
remainder of the switchyard, Actions ware also taken to in 1A and 1B feedwater pumps and Steam reliaf IMS-8. Planned corrective actions wi -A Safety Consideration state Maintenance Procedure will 1 -Signs will be posted on equ personnel working on a break specific relays are not pro -Implement changes as necessa during future equipment iso -Safety related equipment in identified as safety relate -The main feedwater pumps will in the mechanical linkage.	it o investigate and re ill include the following ement similar to the or be added to the Trouble ipment in the 2.30 kv So ker that a yellow bus perly set. hary to ensure relaying blations. In the switchyard will be ed. ill be worked on during the made as necessary	pair as necessary Main ng: te found in the shooting Procedure. witchyard to inform lockout may occur if requirements are met e appropriately the uncoming refueling
ANALYSIS OF OCCURRENCE This event was a loss of lo feedwater. The criteris fo the minimum DNBR will not b not exceed code limits (see	re less than 1.3, and to FSAR Chapter 15.8).	he system pressure will
After the lockout of the ye transformers was supplied f Emergency power situation of	did not occur and was t	ot required.
Ead the red bus been lost of have been actuated and pow- safeguard switchgear would underground line. The yel as required by the Technic	during this period of t er to the unit 1 and 2 have been supplied fro low bus was returned to al Specification.	time emergency power would 4160 wolt engineered on Ecowee bydro via the o operation within 72 hours
The reactor tripped on 2 o Turbine/Generator trip. A specified time, with a max margin was approximately 4	rimm delay time of 51	ignals following CED) breakers opened within msec. Minimum subcooling

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LICENSEE EVENT R	EPORT (LER) TEXT CONTIN	UATION	AMEOUED DUE NO 1180-0
TY NAME (1)	DOCK IT NUMBER (2)		1100011 1/2: 15
		TEAR STREET	
0000000	1.1	TEAN BEQUESTIN	N N N N N N N N N N N N N N N N N N N
Oconee Nuclear Station, Unit 1	0 18 10 10 10 12 16 15	ele - del	- and in the
100 100 100 100 100 100 100 10 117.	the two		-01001500
Four seconds after the reactu- high discharge pressure. All designed and supplied feedwar 38 minutes. The minimum leve inches. The consequences of limits.	three emergency feedwa er to both steam genera 1 maintained in both at	ter pumps start tors for approx	ied as rimately
Reactor coolant temperature a Maximum reactor coolant syste trip. Minimum RCS temperatur maximum cooldown rate of 100 pressure was 2200 psig, well pressure was approximately 17 core cooling system (ECCS) ac	m temperature was 504 d e was approximetely 548 degrees F was not viola below the code limits.	egrees F prior degrees F. Th ted. Maximum B	to the os
Assotor coolant inventory was minimum pressurizer level of	control led within nor 40 inches and a maximum	mal limits, bet of 260 inches.	veen a
Minimum steam generator pressu trip. The steam pressure peak turbine trip. One main steam approximately 11 minutes after steam pressure to 975 peig.	ure was approximately 9 ked at 1125 paig immedia raliaf value (MCPU) 10	00 paig, prior ately following	to the the
When Dnit 1 tripped, a steam ; 1533, on the day of the incide GPM. Conservative estimate of the dose rates were well below	Games and Bass docude	e leak size was	.016
In conclusion, all safety crit Emergency Power systems were n public were not affected.	eria were met during the or actuated. The healt	is event. ECC: th and safety of	and the
	the second s		and the second

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9-83)				1.100	ENSEE EVEN	TREP	ORT	LER)		IXPARS &	21.000		
				LIC	ENGES SYEN	er mar							
								00	CKET NUMBER	121		9401	
ACILITY NAME	Nucle	ar Sr	ation, Un	it 1				0	151010	10121	6 19	1 OF	16
Inopera	bilit	y of	the Emerg	ency Cor	denser Cir	culat	ing k	ater syste	CILITIES INVO			-	
EVENT DAT		1	LER HUMBER I	67	REPORT DATE	17		FACILITY NAME			-	1	
MONTH DAY	TEAR	FEAR	BROURST AL	ALL BACK	WONTH DAY	TEAR	Ocone	e, Unit 2		0 151	0 0	012	710
		1				-				1		Anna an Albana a	
b oli	1016	86	- otili	-010	1 2 1 2			e, Unit 3		- Andrewski and a state of the	0 0	0 2	817
1 Partie	1 a u		ORT IS BURNITTE	D PURPLANT T	TO THE REQUIREMENT	NTS OF 16	CPA \$ 10	best are a mare of	the home-sning (12)			
MODE IN	N	30.4	484(h)		20.406(a)		-	\$5.77e1021(w)		and the second	75.60		
POWER		Manager 1	666 (4.111 10)		88.385s2(1)		X	8.796102(v)		the second secon			TWO I
HOWER LEVEL	1010	and the second	ette and the second	-	95		-	M Theithicki			the prod is the	teri NAG	7 gen
		(country)	ADD(a)(1)(0)	X	ES.7861021(9)		-	80.734+(2)2+(8)181				1)(11	
		paramet.	406 (a) 23 (in)		90.736a1021081			80.72ia/SELAI		50.7	2(b)(2)(11	2)
					CENSES CONTACT	FOR THIS	LER (12)			73. 57.	INE NUM		
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Paul F.	Gui	hi hi	censine	COLL LINE FOR	LACH COMPONENT	PAILURE	DESCRIBI	O IN THIS REPORT					
			MANUFAC	WEADSTARLE	T	1		COMPONENT	MANUFAC	MEMO	TABLE		
CAUSE EVETEN	0044	ONEN?	TURER	TO NPROS	1996	CAUSA			TURER	101			
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	<u></u>	Jul-		INTAL REPORT	-		daman dama m	L de artes e de res	1.94		MONTH	DAY	YEAR
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VES 117 +1	-	*******	SUBMISSION DATE	0	X NO						L.L.	L.L.	
shut Loss CCW) Circo Main Blac The the the sctic basi	down Press of p syst ulati Cond kout) immed Load ons i s of root em.	for re ure Se rime : en war ng Coo enser iate o Shed ? nciud the Eo cause This	efueling, ervice Was in the Cou s the caus oling Wats for decay corrective Test, and ed redesi CCW system of this s led to a S	s Load ter (LPS ndenser se for t er (ECCW y heat r shut do gu of th s. event is failure	with Units Shed Test Shed Test Ship Pump wa Circulatin the loss of () System i emoval dur was to an own Oconce the CCW pump to the inade of the ECC by Analysis	on Un s los g Wat the s req ing a alyze Units flan quate W sys	it 2 t abo er (C LPSW uired Loss the 1 an ges a desi tem t	<pre>was perfor ut one hou CW) Siphon pumps. Th to provid of All AC failures t d 3. Subs nd determing gn and tes o perform</pre>	med. S ir into i Flow (e Emerg e water Power that occ equent instion	uction the to or Eme ency (throw event urred correct of the the 1	h to set. srgen Conde ugh t (Sta duri ctive e des ECCW	the The cy nser he tion ng	
syste	escri		C FILE LTIN		1			0.00.71					

	LICENSEE E	VENI MEPG	ORT (LER) TEXT	CONTINU	OITAL	N			ROVED ON	£ 10 3	11 60 -01 64
CILITY NAME ITS		Sector to be an other	DOCKET NUMBER (2)				-	-		-	401 3
					*14.8		NUMBER		AFV BOX		11
Oconee Nuc	lear Station, Un	ic l	0 5 0 0 0	2 6 9	8 6	-	0111	1	0100	1.5	05
T IF more spece is require	as its separation with, take \$64 1.	(12)			And its shad		-X+-1		× 1 × 1 ×	i nati	
Backgrou The desi provide AC Power that Cor to remov Turbine For the 1) 2) 3) 4) 5) 6) The Initis performed on Fetrua 1. 2. DESCRIPTI On Wednes load shed initiated	und: ign basis of the Water to the con r event (Station idenser Cooling W re decay heat. D Bypass Valves. Driven Emergency Station Blackout	Emorgency denser fo Blackout) ater (CCW ecay heat Feedwater Feedwater event, t) uus Primin ency Air I um (V) Sys fin MS and enser Cool ished and nvel for a water (EFW tion while e Elevated Steam (M valves (T laim the liesel air Storage T TDEFWP a HPSW). e CCW Sys nuary 10, acceptan ssary to the Keowu : 1986, whil	<pre>r the removal . The Station) siphon flow is transferre is delivered ? Pumps (TDEFW he following s mg (CVP) Syste Ejector. stem - The Com maintair cond ling Water (EC function with all three unit ?) System - The ibeing cooled Water Storage (S). TEV) - The TBV condensate for compressor as ank - The EWS nd sealing for tem Gravity ar 1972, on Unit ce criteris for perform this f ee tailrace was le Unit 2 was Doonee, a load</pre>	ling Wa of deca; Blacko through d to the to the P). ystems : m - Usir densate enser va CW) Syst the app s. e turbin by High e Tank (must re r EFW. s a back f must b the CC d Recir 2 on M or the 1 unction s visual in a rei shed of	ter () y heat ut so the r e main stear nust o ng the Stear court Pres EWST) lieve These up. e ava V pum culat arch nitia were lly vi fuelin	ECCC t d ena get get t d ena st tope: t A: tope: t A: tope: t A: tope: t at t at t at t at t at t at t at t a	Olli W) Sys uring rio is n cond ondens nerato rate: sin St is grav. desin t Eje t grav. desin t Eje t grav. desin t Eje t grav. desin t Eje t grav. desin t Eje t grav. desin t Eje t grav. t f est t f est t f est t f est t e	tem a Long ense er v rs v eam ctor ity grn f genc; vice ity ity senc; vice ity ity ite ite ite ite ity ite ite ite ite ite ite ite ite ite ite	o o o o is to ss of iting, r is to ia the ia the ia the (MS) (CSAF flow so low fo y feed water P is e cond sir svide Press tion to o o to n to sr e Unit ads is	all indised iph iph wat wat ure was nit	on er er

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			VEAR EXCLUSION AL MAILEON
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Oconee Nuclear St.	ation, Unit 1	0 0 0 0 0 0 0	
x1 of many space of separations and approximations	CARL CARE AND CONTRACT		
deenergized. B and to allow th discharging to and a siphon ef to flow. For t blocked because After about an and stop pumpir second LPSW pum Various high to the Control Roo restored to its Prior to the oi gpe/pump. The being supplied In the evening drain feature to flow. The NRC. concurred that fully evaluate evaluation of Siphon flow, wo inoperability the two operat Both units reas The subsectent flange on the be leaking wat siphon flow ap maintain flow. On October 3, flange. This meintained T 4, 1986, wher design of the	ormally, this cau e flow of water fi the Keewee tailras fect are used to his test, the con- this test, the con- this was not par hour, the Low Pre- us. One LFSW pump to was observed to imperature alarms om. CCW flow was in ormal powered c courrence, two LPS LPSW pumps are su from Unit 2 at th of October 1, 198 war slow tested. (Ragior JI was adv units 1 and 3 cou- the tests revealed is questionable for of the LFSW system ing units was beging the tests revealed is questionable for of the LFSW system investigation of CCW Pumps, which was er when the pumps uired air was such uspecion, a tempol st of the gravity peared to improve was postulated to and collect in the 1986, plans were the would prevent air wo pumps and thre it was discovered pump and casing r	ses the gravity flow rom the intake struct ce into Lake Kartwell cause the condenser of denser gravity drain t of the test. ssure Service Water was stopped by the of have low discharge i for the components or restored by restartin ordition without any W pumps were operatin ordition without any W pumps were operatin ordition without any W pumps were operatin ordition the CCW at time. He results were the rised of these result is the test was repe tised of these result is dontinue to opera were at 100% power. I that the operation of Units 1 and 3, and as for Oconee. As a d as required by Tec conditions by Octob this incident by Duk was exposed due to th were running. Initi ked into the pumps, d rary repair was made drain feature was ro , and, the cause for be the leaking flang he high point of the made to pull all of the in-leakage and would e motors had been pu-	the Personnel indicated that the he low lake level, was noted to lal thoughts were that when the destroying the siphon. To to a pump with a polyurethane un overnight. In this case, the failure of the ECCW system to ge which allowed air to enter piping, thus terminating CCW the CCW pumps and repair the d enable the siphon to by lled by the morning of October ix would probably not work. The lip to support the pump. The is joint inasmuch as it was not

	LICENSEE EVENT REP	PORT (LER) TEXT CONTIN	L'ATION APPROVED DIE NO	
ILITY NAME IN			Extract to as	er beingrigen.
		DOCKET NUMBER (2)		4.01
			TEAR REDUCT AL MELSION NUMBER NUMBER	11
Oconee Nu	clear Station, Unit 1	0 15 10 10 10 12 16 10	816 - 011 11 - 010 01	and a
17 more apace 4 mp	with all address (C. fam Jack (191	and a second sec	19191 1911 141 1910 101 1	<u>electo i</u>
barrel allow a of the propose saintai system. 11, 198 In addj intake establight blanked adequate pumps. original establight Successive the sight CAUSE OF The root Freegenc System to conditio	d fix did resolve the pro- ned for an extended period Repairs were begun on th 6. tion to the above, each Co was inspected for blockage shed on Unit 3 intake high off with blind flanges whe vacuum on the CCW intake These flanges had apparen base flanges had apparen system hydro testing. T shed on Units 1 and 2 intake ful testing was also perform on effect. By October 23 <u>COCCERRENCE:</u> Cause of this incident i by Condenser Cooling Water	a tubber boot about t er water to allow wate case a pair of CCW pu . The results of the blem and that flow thr d of time (greater tha he other CCW pumps and ontinuous Vacuum Primi). 	he pump casing and pump r to flow out but not to mps were modified and a tes test indicated that the ough the COW piping could b a 8 hours) by the ECCW were completed by October ng (CVF) line at the COW clear and vacuup was and Dnit 2 lines were found pumps from developing overcome air inleakage at th at the completion of the ed and a vacuum was Turbine Bypass Valves, and s were returned to service.	e d he
incident	A	are contributing facto	rs to the cause of this	
CORRECTI	VE_ACTION :			
The follo this inc	owing is a listing of the ident:	corrective actions th	at were taken as a result o	£
1) 2) 3) 4) 5)	Evaluation of the test of determine reasons for fa Bringing Dhits 1 and 3 t Specification 3.3.7 Modification of the leak Determination of the des Removal of blank flanges lines and the inspection	to cold shutdown in act ting flange on the CCW ting basis of the gravi- from Units 1 and	s. cordance with Technical pump casings ity flow function	

840 April 3664 9-72		LICENSEE EVENT REPO	RT (LER) TEXT CONTIN	UATION AND AND AND AND AND AND AND AND AND AN
			DOCKET NUMBER (2)	LEA NUMBER & PAGE 2
				-EAR SEQUENT AL REVISION NUMBER NUMBER
	Nuclear	Station, Unit 1	0 5 0 0 0 2 6 9	9010 -011 1- 000 SOF 0
DC DD B B	10001001	mone hat's farm abled to 1171		
			and sir compressor	and hard piping into the
	6) Se	rvice air header	teser art compressor	
	7) Th	e procedure for loss	of power was changed	to include:
	// 10 8.	Verification of fi	ow through CCW-8.	
	ь.	Mandearing of Eles	sted Water Storage T	ank (ENST) level.
	с.	Testation stamm fr	all first stage Con	densate Steam Air Ejector
		(PELEA) sharahy a	llowing the system t	o maintain condenser vacuum.
	d.	Aligning the Conde	ensate side of the CS	AEs to provide an emergency
		cooling flow path	from the UST to the	BOTVELL.
	۰.	Prompt alignment of	of the Turbine Driven nore UST water for co	EFW pump suction to the
	1.1.1	Fotwell to allow I	Acte usi water for co	supply LPSW suction from any
	£ -	unit with operation	e CCW himps.	support and a second
		A	SW jockey pump onto C	T-4 following a Load Shed
	8.	Pacat		
	8) Ch	sector the exception	for the compressed a	ir system to reflect the
	20, 20,	diffication which had	piped the diesel air	compressor into the Service
	14	+ Custan		
	15 15	endens the procedure	for the instrument a	ir system to instruct the
	0.00	erators to start the	diesel air corpresso	ors and open the service air to
	64	strument air cross-co	onnect, upon loss of	instrument all
	10) Sa	tisfactory performant	ce of the following t	tests:
	. B.	Siphon Flow Test	without continuous pr	TIDITE ON ALL UNICS.
	h.	Diesel air compre	shor test to hold bot	th the Service Air and
		Tratrument Air he	acer at a pressure gi	rester than 75 psig with the
		Instrument and be	rvice Air Compressors	after Condensate flow was
	6.4	repoved from the	CAF.	arter concentrate and art
		Tercved from the	without RPSE flow to	CCW bearing injection
	ć.	connertions.		
	11) Re	nate the CCV dischart	ge high point vent ve	alves
	101 Pa	ANTREPARTIN TO STORE !	CCW nume motor cable	and replacement breakel 107
	da	mage control measure	s for use after an Aj	ppendix R fire has been added
	to	the Appendix R requ	irement list.	
				ations and as follows:
The	Planned	Corrective Actions 1	n response to this if	ncident are as follows:
			the seismicity of th	he CCW system.
	1) Fe	view and analysis of	rogram to include CO	W piping in a routine
		spection	togram to increase the	
1	3) A	review of the validi	ty of the testing pro	ogram to ensure that systems
		d commonants are tes	ted adequately.	
	4) A	review of Technical	Specifications will	be performed to determine if
	an	y revisions are nece	ssary.	
AXA1	YSIS OF	OCCURRENCE:		
Carr		1 of the Coonee Fin	al Safety Analysis R	eport (FSAR) assumes
	148734 P. 10.	of the Trevency Con	denser Circulating W	STET (ECCH) SYSTER ID FOR EXENT
1 16 1	ines of	all AC nower to the	station (blackout).	included in this is the
absi	ity to m	sintain siphon flow	of condenser circula	ting water through the

LICENSEE EVENT REP	ORT (LER) TEXT CONTINU	JATION APPROVED ONE NO STALLOW
ACTLIFY MANE IN		E19 AES \$21 85
	DOCKEY NUMBER (2)	
		VEAR HEQUERTAL MELTICS
Oconee Nuclear Station, Unit 1	0 5 0 0 0 2/6 9	816 -0 111 - 0 0 0 600 0
17 if mans space a securite use additional Add, form addit (111)		
Condenser of each operating unit Sypass Valves and the Turbine Dr can be recirculated through the operation of the ECCW System and nature following a station black maintained and the ability to re atmosphere through the Main Steam With the TDEFVT available but the event, approximately 6.7 hours as gallon supply of stored condenses normal amount of condense e avail blackout is approximately twice of decay heat removal would be possi Conce a C power distribution syst restored within 4 hours of a stat texture of the testored to supply needed to remove decay heat Filewated Water Storage Tank will hours. The Loss of Coolant Accident (hour case single failure. In Section failure of one bus of emergency p pressure Injection and one train in this situation the overhead em svailable; accordingly, the CCW p flow through the CCV system durin provided by the ECCW system is no the overhead life is lost as the p prise of the second tri CT-4, which would constitute a lost load shed. Operators would then a immediately following the load she the initial load shed to provide v and gravity flow would not be requ of the FSAR.) The diverse and reliable sources of very low. Sufficient condensate is following blackout without the oper- tectreulation of condensate. A CC following blackout without is oper- precision of condensate. A CC following blackout without je the and safety of the public.	<pre>Iven Emergency Feedwat: Condenser to remove det the Turbine Bypass Va. out event based on the lease decay heat by ste m Relief Valves. e FCCW System unavailable to available before exists to required by Technical lable to the emergency the Technical Specifica lable. Based on the red tem, it can be assumed tion blackout event; th service well before de it. During the Station supply cooling water t (A) assumes a loss of c 15.14 of the FSAR the over which results in of Righ Pressure Injec ergency power supply p upps would still have j at the event. As such, t required . Rowever, single failure and the ion loads would then be ain of emergency AC pow ad shed situation in with manually load the CCW pump water for the Low Press uired. (Peference Sect of AC power we a load she is available to provide tration of the ECCW Sys W pump can be restarte event at the sector of the top of the top of the top of the sector of the ECCW Sys W pump can be restarted.</pre>	er Fumps such that feedwater cay heat. However, immediate lves is not of a critical inventory of feedwater eaming directly to the ble following a blackout hausting the minimum 72,000 al Specifications. The feedwater pumps following ation minimum, so extended hundancy designed into the that AC power would be merefore, the normal CCW and upleting the condensate h blackout scenario, the to the TDEFWP for 5 to 13 ffsite power and a worst worst single failure is the the loss of one train of Low tion. ath from Keowee is power to operate and provide the gravity flow capability if it is also assumed that FCCW system was unable to e supplied by the wer) through transformer hich the CCW pumps would be pumps onto the emergency bus ps could be restarted after sure Service Water System tion 8.3.1.2 and Table 8.1-1 to Oconee make the ed coincident with a LOCA a decay heat removal stem, which would allow ed to supply LPSW suction

E-48

alt has Be				LIC	ENSEE EVE		PORT	(LER)	8.2 8.6		941 60mm aft 40 1 mil 21 86 8 19
	137							14			7417-12
Peach	Bott	om A	tomic 1	ower :	Station	- Un	it 2			0121717	1 01 01
Reactor	Scra	m Due	to Fail	ure of	E-2 Diese	Ger	erato	r			
antal Ba	Fa cas	[A67941 8411				-		
udela dut	8148	YEAR	ALBRIT (VI	ALL LE COM	udula Bal	1164		fell will have	-		
Q 1/21.	8 16	10 K L		- 010		8 6					101 1.1
	9,5	- 18.9 8.4 8.4	UNT IS Burber 1775 Annual Baudhach Barlandh Scaeth Annual Baudh Annu	Politicas I	10 141 11 04/14 (101 30 405 ml 31 35 405 ml 34 35 407 (1) 34 35 407 (2) 34 1 34 407 (2) 34 1 34 407 (2) 34 1 34 40 (2) 34 1 4 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1	rn er 1	A L A	Nation In State 1 M.J. State State M.J. State		Ph21et Ph21et	- 7
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X 1 81 3	Un IE	2.1	A1611 E		LAFETTER INF			- Andrews	6445071	-	8.47 75
A PROVIDE NO. 19. CO.					- 61				Ball it		1,1,
orrest of the shoon an	ran estate estat	uary tion of ten) of to VS 0 f to VS 0 f to VS 0 to rten) to rten) of to rten) of to to to to to to to to to to to to to	system rred as treloss =2-2-86 loss of two resolutions to stay wer to to stay wer to to stay wer to to stay to stay	86, wi (RPS) a res re of B and AC po dundar odundar E=22 c . The dissel dissel dissel eling, Unit via th nerato	ith Unit initia sult of 1 outboar AO-2-2-1 ower to 1 t DC so during 1 occurred to trip ations of as a re 3 outboa e E-23 b r was sa	ted h i Ma. Sous oh sous oh sous oh sous of sous of so	a ful in S E-22 ids lo a res erato Addit rred t of sola	<pre>ll reac e flux team Is e MSIVs in con which a oss of sult of or air tionall; on Uni loss of stion lo</pre>	tor scr caused olation failed junction AC powe E-2 di. intake 1 y, Group t 3, wh f power ogic rei	am. The by the Valves closed n with gned to r condi- esel plower p II an ich is to E-2 lays an	i as a the allow ition. failed d 3 bus e
ti (gis 201					3, 1986.	-					

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	PORT ILERI TEXT CONTIN	UATION	A MACLAR RECOLATORY COmments on GATADAL COMMEND, 11 March 60 Extents 60:00
Peach Bottom Atomic Power Station - Unit 2	BOCALI BURNELA U:	LES BURGES	-
1111 cm dates generic a requirement and reducionar #25 Faret 26day (0.2)	0 5 0 0 2 7	7 816 -0101	3-01002 01016
Unit Conditions Prior to			
Unit 2 was operating at 9 supplying buses E-22 and test on Unit 3.	5% power level w: E-23 in preparati	ith E-2 die ion for a lo	sel generator DSS of power
Description of the Event:			
On January 24, 1986, at O automatically tripped the buses. Main steam isolat 2-86D inadvertently close these valves resulted in sufficient to initiate a following the scram, react Group II and Group III iso inches water level. The s automatically increased to inches the reactor feedpun reactor water level trip s tripped properly. Both re properly during the 13.2 % 'C' reactor feedpump was r placed in service to contr recirculation pumps were r Additionally, Group II and on Unit 3 as a result of t	ion valves (MSIVs d following the d a high core flux full reactor scra tor water level d olations occurred speeds of all thr o recover reactor mps and the main signals. The fee eactor recirculat KV bus fast trans reset from the hig rol reactor water returned to service	power to E) A0-2-2-86 iesel trip. condition w m. Immedia ecreased to properly a ee reactor water leve turbine rec dpumps and ion pumps t fer. At 06 gh water le level. Bo ce by 0645	-22 and E-23 B and AO-2- Closure of hich was tely +32 inches. t zero feedpumps 1. At +45 eived high main turbine ripped 34 hours the vel trip and th hours.
Cause of the Event:			
Prior to the event, E-2 di E-22 and E-23 emergency bu test on Unit 3. At 0612 h thereby removing all power power from E-22 bus de-ene outboard MSIVs. By design energized to allow the MST Subsequent to the event, th A0-86D were found to be fam	to E-22 diesel g to E-22 and E-23 rgized the AC sol , a redundant DC V to stay open du he DC solenide f	enerator tr buses. Re enoids of a solenoid re ring such a	is of power Tipped, moval of The four Mains Condition.

MSIVs AO-86B and A the four main stea 13 PSI pressure sp	ne loss of power to	21717 816 - 01013 - 010 013 c	T
each Bottom Atomic Power tation - Unit 2 conjunction with the MSIVS A0-86B and A the four main stea 13 PSI pressure sp	ne loss of power to	21717 816 - 01013 - 010 013 c	.0
conjunction with t MSIVs A0-86B and A the four main stea 13 PSI pressure sp	ne loss of power to	o the AC solenoids, caused	10
conjunction with t MSIVs A0-86B and A the four main atea 13 PSI pressure sp	ne loss of power to	o the AC solenoids, caused	-
conjunction with t MSIVs A0-86B and A the four main atea 13 PSI pressure sp	ne loss of power to	o the AC solenoids, caused	
MSIVs AO-86B and A the four main stea 13 PSI pressure sp		o the AC solenoids, caused	
flux spike was suf	ike in the reactor detected by the in- ficient for the RPS	of these lines produced a which, turn, produced a -core flux monitors. The s to initiate the full scra outboard isolations occurr	в.
on Unit 3 as a res outboard isolation center.	logic relays are j	powered via the E=23 bus lo	ad
51 hours prior to low loads during t 2600 KW). At low	hat period (nomina	operation for approximatel esel was run at relatively lly 550 KW, although rated ion air to the diesel is blower. When the air blowe d and tripped.	at
Consequences of D	e Event:		
And the second s			
System initiation did any occur) du	s were necessary as	No Emergency Core Cooling s a result of this event (no operation of the reactor -2 diesel, all systems were ne event.	
Corrective Action	<u>8</u> 1		
performed on a mo solenoid coil con completed RT-15.6 satifactory opera just two days pri	nthly basis for the tinuity. A review indicated that all ting currents when or to the event. placed on January n sent to PECo Ele	lenoid Continuity Test" is e purpose of verifying MSIV of the most recently 1 MSIV AC and DC coils had tested on January 22, 1986 The AO-86B and AO-86D DC 25, 1986. One of the faile ctrical Engineering Divisio	, d

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E-51

each Bottom Atomic Power		ttes billingenets at	
tation - Unit 2			
many grants & manural, our techniques Bill Farm allate D 11	· [*]*]*]*]0] 2]7]7	10101-1010 B	1-1010101-1010
The E-2 Diesel air blowe satisfactorily tested an	r was replaced -	nd R. 0 41	
satisfactorily tested an 1986. The failed diesel	d returned to se	IVICE by Pa	el was
1986. The failed diesel Colt/Fairbanks Morea for	air blower has	been sent t	O CLUBERT 21
Colt/Fairbanks Morse for	failure analysi	s and rebui	lding.
Manufactor in a construction of the			
Previous Similar Occurre	nces:		
None.			

POW 28-06-01 APPROVED DATE NO THE FOR 8.82 hare 384 -----LICENSEE EVENT REPORT (LER) PACE 3 -----PACILITY RANG IT 0 | 5 | 0 | 0 | 0 | 2 | 8 | 0 ' | 0 * 3 | 3 Surry Power Station, Unit 1 TITLE IM Loss of Charging Pump Service Water Pumps Teles date in anter a anter a ------WONTYN GAT TEAR TEAR BEGUNTTAN MONTA DAT TEAR -----0 15 10 10 101 8 6 0 2 9 -0 15 10 10 10 0 010 2 8 0 9 2 9 8 6 8 ITS REPORT IS BUSIN TTED PURBLANT TO THE REQUIREMENTS OF IS CFR \$ Chast ans ar mans at the fair 12.15 (8) -----NF 90.73mm(2)14 16. emiliar -75,710 B.TRACED 8.486ar115 17.16.16 M --LEVEL OTHER Gamerty is Address -11010 8.364.07 --M. TRecOmminA -----X H.TBerillis _ 8.78a-21-60-8 41.73a-(2111) --------M.Theidrai M. That do m ------LICENSES CONTACT FOR THIS LER ITP ------------80 4 3 5 7 - 311 84 R. F. Saunders, Station Manager COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBES IN THIS REMONT (3) REPORTAR.E NAM PAC PERORTAN, 8 GAUES SYSTEM COMPONENT NAMUTAG. TURER CAUES FYTTEN COMPONEN? MONTA DAY YEAR BURN, BARNTAL BENGRT EXPECTED IN 8-04-55-04 and the second TES IT HE ANTHING EXTERNED SUBMITSION ON TE X 12 ABSTRACT Loss & 140 areas & secondary their argent On September 29, 1986 with Surry Units 1 and Unit 2 at 100% power, all service water flow to the Unit 1 Charging Pump Service Water Subsystem was lost due to the pump becoming air bound. This abnormal condition affected the heat sink for the charging pump lubricating oil coolers and the intermediate heat sink for the charging pump mechanical seals. Immediate attention was provided to return a flowpath to service. The affected Unit 1 pump was subsequently vented. Following verification of proper operation, the Unit 1 subsystem was returned to normal status. JE22 8411040010 841000880 ADOCX 05000880 1/1

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POW 28-06-01

1.0 Description of the Event

On September 29, 1986 with Surry Units 1 and 2 at 100% power, Service Water (SW) flow to the Unit 1 Charging Pump Service Water (ChgPSW) subsystem was lost. The approximate time of the event was 1445 hours.

Earlier in the day, one of the redundant Unit 1 ChgPSW pumps (EIIS-P), 1-SW-P-10A, had been removed from service for replacement. Maintenance activities in the 1-SW-P-10A room required that grinding be performed on a pump support prior to pump replacement. The grinding activity resulted in actuation of a smoke detector which automatically closed a SW fire isolation valve. Due to a leck on a strainer blow down line in the SW supply line, closure of the SW fire isolation valve allowed air in-leakage which caused 1-SW-P-10B to become air bound.

2.0 Safety Consequences and Implications

During normal operation, the charging pumps are used as part of the Reactor Coolant Chemical and Volume Control System (CVCS) and take suction from the Volume Control Tank (VCT) (EIIS-CB). During accident conditions, with Safety Injection (SI) actuated, the charging pumps (EIIS-P) are used as High Head Safety Injection Pumps and take suction from the Refueling Water Storage Tank (RWST) (EIIS-BQ).

The ChgPSW pumps provide a heat sink for the charging pump lube oil coolers and the component cooling water subsystem (which is the heat sink for the charging pump mechanical seal coolers) (EIIS-CLR). Recent analyses and communications with the vendor indicate that no heat sink for the mechanical seal coolers is required. The effect of loss of heat sink to the charging pump lubricating oil coolers was monitored by the reactor operator on the plant computer. The highest bearing temperature observed during the event was approximately 160°F. Subsequent operating evaluation has revealed no pump degradation due to the short term lube oil temperature increase. Public health and safety were not affected during the event.

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POW 28-06-01 U.L. NUCLEAR REQULATORY COMMISSION RRC Farm 3864 APPROT ED DWE NO 2180-0-04 LICENSEE EVENT REPORT (LER) TEXT CONTINUATION -----DOCKET MUMBER (2) --------------TEAS REQUEST AL DEVELS Surry Power Station, Unit 1 0 15 0 0 0 2 80 8 6 -0 2 9 -0 0 0 3 0 0 0 3 TEXT of many amount & measured, use antideness raft, Apres 388-4 () 3.0 Cause The cause of the event was introduction of air into the Charging Pump SW supply line. Grinding activities in the cubicle that houses ChgPSW pumps 1-SW-P-10A and 2-SW-P-10A actuated the smoke detector and closed the SW isolation valve. A leak on the blowdown line for the in-line strainer (EIIS-STR) in the SW supply line provided an air in-leakage path into the system. 4.0 Immediate Corrective Action Operators were dispatched to locate the source of the problem and return the pumps to normal. The temperatures of the operating charging pump were monitored. 5.0 Subsequent Corrective Action The leak at the in-line strainer was repaired. The Maintenance Predictive Analysis Group subsequently monitored the Unit 1 charging pump that had been in operation during the event and found its operating parameters to be normal. 6.0 Actions Taken to Prevent Recurrence This is considered an isolated event, therefore, no additional actions were taken. 7.0 Generic Implications None.

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	LICENSEE EVEN	NT REPO	TRC	LER)		ANNOVED ONNE NO STURES ECTOR	2180-0104
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On October 30, 1 2 at refueling s water flow to th Subsystem was lo a service water : being vented. Th for the charging intermediate hear seals. Operation service water sy Specification 3.	e Unit 1 Char st due to the strainer was p is abnormal or pump lubricat t sink for the without servi- stem is prohib	pumps placed onditi ting of char	be i be i n in iii	Service coming a service affected coolers g pump m	Water ir bou witho the h and the echani	rvice nd when ut eat sink e	
Immediate attent: flowpath to serv: subsequently vent operation, the Ur status. The syste minutes. This re lOCFR50.73(a)(2)(ted. Followin hit 1 subsyste am was out of	ig ver	ifi re	cation o turned t	s were f prope o norma	er	
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	-			POW 28-06-01			
edit Form 2004 9-67		LICENSEE EVENT REPO	ORT (LER) TEXT CONTINUA	TION .	CLEAR RECULATORY COMMITME PRACYED OWE NO. 3150-0104 KRIES & 21-18		
ADUTY HANG OF			BOOKET NUMBER (D)	LER NUMBER (6)	Page 15		
	A	wer Station, Unit 1	0 5 0 0 0 2 8 0 8	16 - 0 3 1-	- 010 012 0F 01		
	1.0	Description of t	he Event				
		Unit 1 at 100% p with the reactor the Unit 1 Charg (EIIS-BI) was lo	30, 1986 at 0202 how ower and Unit 2 at 1 defueled, service w ing Pump Service wat st. Operation with pump service water s cification 3.14.	efueling shu Mater (SW) f: er subsyster Sut service (utdown, low to m water		
		an in-line duple: operating Unit 1 to an indicated is standby filter e discharge pressu zero. The ChgPs and locked-in in	was swapping the fi x strainer (EIIS-STF ChgPsw pump (1-SW-1 high differential pr lement was placed in re of the operating w header low pressur the main control ro) auto-started but f rm.	() upstream () P=10A) (EIIS) ressure. Whito service, pump decrea re alarm anni com and the s	of the -P) due en the the sed to unciated standby		
	2.0	Safety Consequen	ces and Implications				
		charging pump lu cooling water su charging pump me Recent analyses indicate that no coolers is requi to the charging p monitored by the computer. The during the event fahrenheit. This degrade the equi	pumps provide a heat be oil coolers and to bsystem (which is the chanical seal cooler and communications we heat sink for the me red. The effect of pump lubricating oil reactor operator of ighest bearing tempe was approximately is is an acceptable we pment. The public he is during the event.	the component the heat sink (EIIS-CL with the vent the the vent the the vent coolers wat the plant that use obset 43 degrees thus and wil	t for the R). dor eal t sink s rved l not		
	3.0	Cause					
		into the ChgPSW ; filter element w filled and venter pump becoming air	f the event was intro- pump supply line. We as placed into serve d which resulted in r bound. It has not pump failed to clear on.	When the sta loe, it was the operati t been deter	ndby not ng mined		
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ас тойн Хееа 1407							

ART from Mile		POW 28-06-0	1			
LICENSEE EVENT	REPORT (LER) TEXT CONTINU	JATION APPROVED	U.S. NUCLEAR REDULATORY COMMISSION APPROVED DWS NO 2150-0104 EXPRACE 6-21-88			
	DORCK & Y MANUELA (2)	LER NUMBER IE	FADE (3			
Surry Power Station, Unit	1 0 16 10 10 1 2 8 1 0	8 6 0 3 1 0 0				
TEXT IF many goods a required, was additional fully form JABATS/1171	and a standard by a decide with the second		10/2/04/013			

Subsequent investigation of the event revealed an accumulation of marine growth in the Unit 2 Service Water supply line to the ChgPSW pump and control/relay room area chiller subsystems. This growth reduced the interior pipe size of the supply line and increased system losses. The increase in suction side losses is believed to be a contributor to the loss of SW events.

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4 0 Immediate Corrective Action

Operations personnel were dispatched to identify and correct the problem and return the pumps to service. The control room operator monitored the temperatures of the operating charging pump on the plant computer.

5.0 Additional Corrective Actions

The affected Unit 1 pumps were successfully vented and were returned to service at approximately 0224 hours. The system was out of service for approximately 19 minutes.

6.0 Actions Taken to Prevent Recurrence

Operators have been instructed to ensure in-line strainers are filled and vented prior to placing them into service. The marine growth was cleaned from the Unit 1 and 2 service water supply lines.

7.0 Similar Events

Unit 1 LER 86-024. Unit 1 LER 86-030.

8.0 Manufacturer/Model Number

Not required.

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LICENSEE EVENT REPORT (LER)	U & BUCITAT RESULTOR - COMMISSION American Commission NO
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On 7/11/86 with Unit 1 in refueling shutdown and Un operators were attempting to return the 'A' chargin cooling water pump to service following emergency m 1518 hours, the redundant 'B' pump, which had been water to the charging pump seal coolers, lost disc This resulted in both pumps being inoperable. It i introduced into the system during maintenance on t the 'B' pump to become vapor bound. The 'A' pump w added to the system, and the pump was returned to i Subsequently, operability of the 'B' pump was demon also returned to service.	maintenance. At supplying cooling harge pressure. Is assumed that air he 'A' pump caused was vented, water was service at 1825 hours.
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		UCENSEE E	VENT REPO	RT (LER) TEXT CONTIN	UATION		APPROVED 0	-		n þr
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		Incorrable Cha	raina Pira	Component Cooling)	Water P	TES				
	1)	Description of	Event							
		shutdown, emer- the Unit 2 'A' At that time ti ing cooling wa while attempti maintenance, t This resulted	gency mair charging he redunds ter to the ng to retu he 'B' pur in both d	hit 2 at 100% power itenance was being pump component cool int pump, 2-000-P-28, charging pump seal int the 'A' pump to s p's discharge press arging pump componen event is contrary to	erforme ing wat was in cooler service ure dro nt cool	d on 2-4 er pump operat: s. At 1 follown pped to ing wate	XC-P-2A, (EIIS C lon supp (518 hou ing the zero. er pumps	XC). 21 y- 47 \$,		
	2)	Safety Conserv	ences and	Implications						
		Reactor Coolar: suction from ti accident condi- charging pumps from the Refue ing pump compor- the charging p temperature of the RMST is is required for below 115 F. water pumps di-	t Chemical he Volume tions, wit are used ling Water ment cooli ump seals the VCT i 45 F. The the chai Therefore, d not created	the charging pumps and Volume Control Control Tank (VCT) h Safety Injection as the Hi Heed SI p Storage Tank (RMST ng water pumps prov through a seal cool is 115 F and the max we vendor has indication ging pumps as long the loss of the ch ite an unreviewed sa public were not af	System (EIIS C (SI) ac umps an) (EIIS ide coo er. Th imum al ted tha as the arging fety qu	(CVCS) B). Dui tuated, d take : BQ). ling we e norma lowed to t no set pumped : pump co estion.	and tak ring the suction The char ter for I maximum success for fluid in coling	in in in i		
	3)	Cause								
		of two redunda coolers (coole coolers (2 pr of the event, tank. It was and efforts we significant le air was introd	nt cooling d by serva narging , was not therefore re directe aks could used into sed the 'H	ng water system is a) water pumpe, two is (ce water (EIIS BI)) pump). Shortly, fo (ced that a low level assumed a large lea id toward finding th be found. Therefor the system during m i pump to so de va	edundan , e hea llowing existe k exist e leak. e. it i aintena	t inter d tank d tank d in th Howev s specy nce on	mediate and 6 se itiation s head he syste er, no lated ti the 'A'	eal o em		
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Surry Power S	Station, Unit 2	0 16 10 10 10 12 1811	86-0110-01	
T IF saws game & myslest.	aar Adabberar ABAC Awax 2004 'n (17)			
4)	Immediate Corrective	Action		
	return the pumps to a charging pump were m ing water pumps were were observed. Also	tched to locate the so operation. The temper onitored and trended of inoperable and no in , a Technical Specific ared and preparations d.	ratures of the opera during the period th crease in temperatur cation Limiting Cond	iting ne cool- res lition
5)	A tional Corrective	e Action		
	the system and the 'A after operability of	perly vented and make A' pump was returned the pump was demonst B' pump was demonstra Vice.	to service at 1825 h rated. Subsequently	NOULTS
6)	Actions Taken to Pre-	vent Recurrence		
	This is considered to actions were taken.	o be an isolated even	t, therefore, no add	litional
7)	Generic Implications			
	Noné			

40.			LICE	NSEE EVEN	TREP	ORT (LER)	U.S. RUCLEAN REQULATORY COMMINISION APPROVED DWE NO 3150-0104 EXPIRES SOLARS					
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acture.									
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At 0138 on September 8, the 86 Lock	cout was re	set so i	chat :	it c	could 1	he d	leter	rmine	d
if the envernor shutdown solenoid p	lunger was	stuck :	in the	e sh	illedow	n po	08101	101.	
The plugger use found to be in the	STODET DOS	ition.	AE U	150,	. DI ¥	85 5	start	ced	
from the control ross. During this	s start, it	Vas no	ted ti	hat	Dito	OK a		bnors	ally
long time to start. It was suspect start was due to a fuel system prob	ted at this	time th	hac ci	ne :	allur	e 01	2 × #3	bie .	
start was due to a fuel system proc time no problems were observed.	orem. we .	720, 01	*65	****	in ova		.,		
At approximately 0810, work was beg	gun to inve	stigate	pote	ntia	al pro	blet	15 V	ith t	the
fuel everem laskage tests were be	erformed or	the fu	el oí	1 90	ib dau	scha	arge	chec	2 K
valve (V) and the fuel oil pump suc	ction pipit	ig back	to th	e to	sot va	live	(V)	in t	ine
day tank. These tests revealed no	problems.	the pa	CKIES	10	the r	Val	1011	(RV)	ř
priming pump (P) was replaced. The was removed and tested. The invest	rigative w	pump ur	vered	00	probl	ems	whi	ch	
could have caused the failure of D	1 to start.								
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Products Valued Barry			
	0 5 0 0 0 2 8	2 8 6 -0 01	6 - 0 0 0 3 0 0 0
Prairie Island Unit 1 On the morning of September 9, DI diesel generator room, while in 1 starts, the air supply to the gow to determine if this would advers It was found that without the oil the diesel still started within t normal surveillance procedure to operable. A schedule was caveloped for incr to assure its reliability since to determine if the fuel oil header The fuel header pressure was month the fuel oil hand priming pump, dependent on the amount of time to On the morning of September 19, D pressure return orifice check walve nick in the rubber O-ring seat. scarted again and at 1224 DI was The apparent cause of the DI inopera- toring idle periods. The purpose of fuel oil to return to the day tank (prevent the fuel header from drainin hlowed to enter the fuel header from ordies of the injection pumps. This raive, caused the fuel header to drainin taive, caused the fuel header fuel header to drainin taiv	I was successfully s local control. Frice vernor oil booster w sely affect the abil boost, the governo can seconds. At 120 prove operability. reased testing and f the cause of its fai essfully on the foll tember 18. A speci was draining down a tored immediately f This revealed that the diesel was idle. Of was removed from twe (V) was removed a revealed seat leas The valve was repla declared operable. Which allowed the i this spring check (TK) while the enging down while the enging the clean fuel re-	2 8 6 - 0 0 1 0 1 tarted three t to the secon as disconnecte ity of the die tr vas shower t 3, DI was star At 1501, DI w urther investi lure to start owing days: Se al test was de at test was de the pressure d service and th for inspection the pressure d service and at 11 through the fu fuel oil heade valve is to al e is operating gine is idlo.	6 - 0 0 0 3 of 0 imes from the d of these d and capped sel to start. o respond, but ted per its as declared garion of D1 had not been ptember 9 veloped to ween starts. peration of ecay rate was e fuel header . The cesting caused by a 15 D1 was el header r to drain low excess and to Air is s through the
ime between diesel starts was a fac nwo to three wueks; on this occasion threwed draining to continue to the	the interval was	5 days. This	apparently
AFETY CONSIDERATIONS			
This event is being reported voluntu afety of the public were not affect throughout the event. Operator acti	ad since offsite po	Wer sources we	ealth and re intact
(rotw Wes)			

LICENSEE EVEN	T REPORT (LER) TEXT CONTIN	UATIO	N							0104
ACUST RAME ID	DOCX17 H.4464 # (2)			-		-	1 640 4		-	3
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Prairie Island Unit 1	0 5 0 0 2 8 2	8 6	-	0101	6	-	010	01	4 OF	0
"EV? IP more spece a required, ups accessor acts" form JBEA's: (17)										
CORRECTIVE ACTIONS										
A temporary repair to the ruptu was made quickly. The oil line tubing with stainless steel tub Cooling Water Pump.	bing. This repair will a	lso be	do	ne o	N	0.	12	Dies	el	
The leaky check valve in the further similar check valve install next scheduled preventive main being operated daily to assure	led on Di Diesel Generato	the h	and	pri	min	8 1				
FAILED COMPONENT IDENTIFICATION										
The leaking valve is a spring generator made by Fairbanks-Mo	check valve supplied with rse.	the 1	fode	1 38	TDS	8-1	/8 d	ies	el	
SIMILAR PREVIOUS EVENTS										
One previous instance of inope cooling water pump was reporte	rability of one diesel ge d as RO 80-06.	nerat	or a	and o	ne	di	ese	1		
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1000	r)	(1)	W-2	IS OU	1 0	1 Servic	e for p	lanne	d mai	ntena	DCP.	During +	he dollo	anarahit	1000
run	0I	NO	a . 4	12 DC	LP	(require	d by Te	C. D. C	al Sp	ecifi	catio	ns 1 .n n	no Mereo	1 Cooling	Unter
rump	- 2	8 C	3.01	OI S	erv	100), 8	1acket	Water	hose	TURF	urad.	The No.	12 507 5	then immediately	110000
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Vas	87	000	81	Th	4U	UT use a	0.000	che p	ump w	as de	clare	d operabl	e and ch	e load de	ecrease
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che .	ev	ent								Poses .	0001	ere sere	everteor	e chroug	1045
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19211	ng	×	HOS	es or	n N	0. 22 DC	LP were	repl	aced	with	hoses	of highe	r raring	last ver	in a sole
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LICENSEE EVENT REPOR	T (LER) TEXT CONTINU	OITA	N			1.50		-	104+ COMMISS- 0 3150-0104
ADILITY NAME IN	DOCKET NUMBER (2)			-					PAGE 3
		-14.8		10011	3	_	A1. 50-	-	
PRAIRIE ISLAND NUCLEAR GENERATING PLT	0 15 10 10 10 1 218 12	816	_	0 11	11	-	010	0	12 05 0
EXT of many same of many set and and the set of the set	12 12 12 12 12 12 12 12 12 12							-	
EVENT DESCRIPTION									
On December 27, 1986, both units were (DCLP)(P) was out of service for plant run of No. 12 DCLP (required by Techni Pump is out of service), a jacket wat shutdown and declared inoperable at OU units sills both diesel cooling water Unusual Event (NUE) was declared. Re was accomplished quickly; at 1008 the was stopped. The NUE was terminated	ical Specifications er hose ruptured. 848. At 0947, a lo pumps were inopera placement of the ja	when The N ad de ble. cket	on o. cre A wat	e Di 12 D ase Noti er h	wa lfi hos	el P s t c at e t e	Cool egut ion load	imme of o. d d	g Water ediately n both 12 DCLP ecrease
CAUSE									
Cause of the event was failure of a f investigation indicates that expected hose had been installed in early 1980	Selvice tite of e.	y Aen his ho	ogu	is is	Cor abo	p. ot	Pre 5 y	lim ear	inary s; the
SAFETY CONSIDERATIONS									
This event is being reported voluntar of the public were unaffected since o the event.	ily for informatio ffsite power sourc	n pur es we	pos re	es. avai	Helat	al	th a thr	oug	safety ;hout
CORRECTIVE ACTIONS									
All the flexible hoses on No. 12 DCLF rating. Hoses on No. 22 DCLP were re The hose replacement program is being	SDIACED MI'N HORES	W.B. 11.6	8	of a r ra	h	igh	last	em; y	perature ear.
PREVIOUS SIMILAR EVENTS									
There have been no previous similar (events.								

			ы	ENSEE E	ENT RE	PORT	(LER)		CLEAR REDULAT	
OFT CA							0	OC.1.17 HL.MAL	(2)	FAST 3
ful in	inoun stat	tion, Uni	t No. 1		-		0	151010	19121815	1 01 01
leactor	Trip Caus	sed by Ins	trument	Inverter	Failu					
EVENT DA	1	LER HUMBER	10	REPORT		e	OTHER P			
ONTH DAY	YEAR YEAR	Our Station , Unit No. 1 rip Caused by Instrument Inv		MOR TH DA	TELA		FADUTT KAN		DOCKET NUMBER	4
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and lohe	C Adams	Number		ICENSEE CONT.		S.E.R. (12)				
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evel a nitiati f 120 v onseque ypass v ailures eactor	fter the ed by a c VAC instru- ences of valves and combined coolant s ent bus ar	failure of losure of ument powe the loss of failure d to cause system. A nd. within	a safe the tur r to th f power of the an abn	ty relat bine con e electr include feedwate ormal po	ed inst trol va ohyorau d inope r valve st-trip	rumer lives lic c rabil ramp	which was ontrol un ity of th down circ sure tran	w steam r. The caused it. Oth e steam uitry. sient in	generator trip was by the lo er dump and These the	55
nstrume hutdowr	condicit	/n .			ene pr	ant w	as placed	in a no	rmal hot	

LICENSEE EVENT REP	ORT (LER) TEXT CONTINU	ATION					-		
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CILITY NAME IT		-14.4	181		•1	NUMBER			
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ort Calhoun Station, Unit No. 1	0 5 0 0 0 2 8 5	1810		0101		10 10	LATE		
IT if many speek a required, and addressed totic farm 3864 (c) 1171									
During normal operation at 100% preceived in the control room at 0 quickly diagnosed a failed instrum an equipment operator to the switc closing the breaker on a bypass t inverter failure placed the React condition since the RPS operates inverter was one of four feeding seconds after the inverter failur trip was received in the steam get Several unusual transients were no 1. Reactor Coolant System pressus short per od of time. This of 2. Steam generator pressure incre valves causing them to be act 3. Overfeeding the steam general subsequent overcooling of the decreased to a low of about 4. Steam dump and bypass valves 5. The operating charging pump started. Within a minute of the reactor t lost instrument bus and control plant to normal shutdown conditi A diagnosis of computer and reco explanation for transients seen instrument bus AI-40A supplies A turbine first stage pressure control circuitry is powered fro to the load control unit. This going closed without a reactor explanation for transients seen instrument bus AI-40A supplies to A turbine first stage pressure fro to the load control unit. This going closed without a reactor explains the high pressure seen level earlier in tho transient. loss of a heat removal path tha steam dump and bypass controlle steam dump valves on a turbine again became operable when powe	ment inverter feeding chgear room to manuall ransformer also feeding or Protective System (on a two-out-of-four) the independent channe e, a reactor trip occi- inerator B low level tr noted in the moments for reased to the moments for reased to the setpoint tuated. tors resulted in abnor e primary system. As 1725 psia could not be opened. stopped and the two ba trip, the equipment oper room operators were so ion. Drder information reve after the inverter fa power to EHC panel Al- transmitter which send om Al-50. Loss of pow ultimately resulted i trip. The load reject in the primary system The high steam gener t occurred when invert rs. Also, a relay wh	bus A y ree g bus RPS) ogic ls of rip un ollowi ximate tuated of th mally a resi ackup erator soon at sled f ilure 50 wif s a s s a s care n the ion b and rator crep points are from	I-4rI anthe ing al. he hit put he the prevent	OA an gize -40A. a hal the reside RPS in a since the reside RPS in a since the reside reside reside reside to reside reside reside to reside reside reside reside to reside reside reside reside reside to reside reside reside reside reside reside reside reside to reside	d det f-ti fai secco tri; ps dar pr ould eest oscotes was los	ispatible bus he rip led About and cl bi is fo y saf and cl saf y saf and ressur i not rgize ore t ng gize ore t ng saf essur i not s du s du s du s du s du s du s du s du	ched by t ter hanne r a ety be d the he ed the ignal val ignal the	e r.dl ves itore	

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Fort Calhoun St			TEAN BEQUENTIAL	the second se
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which energi therefore di was required discovered. The loss of i operation of the backu, ch operating pun was not in it regained oper The loss of a condition. To of the Reacto Loss of a sin functions. I several undes probability o An emergency from bus AI-4 automatic tra initiating sc modification Engineering E desirability o	zes after a trip to d not perform its fu to restore steam ge the inverter also ca backup charging pum harging pumps if the mp should have kept ts normal position. Tability. Investiga a safety related invol- here are four such or Protective System rigle inverter cannot t is apparent, however irable consequences. If this incident bein modification was per OA to bus AI-428 whi nsfer to alternate p enario will be repea was completed before valuation Assistance of redistributing th d not have all the c	nerators after the trip not rampdown as they n initiate the rampdown unction when the reacto enerator level to norma used deenerization of n nps. The loss of these y had been operating (1 running unless the char when AI-40A was reener tion of this problem is erter does not put the inverters which supply and to the Engineered prevent these systems ver, that the failure o . Several steps were t ng repeated. These ste rformed to transfer the ich is supplied by an i power. This reduces th ated on the loss of an e the plant was returned a Request has been init tonsequences as describe	ormally would. is powered fro r tripped. Op l once the ove relays which co relays would i they were not) ging pump sele rgized, all cha continuing. plant in an un power to the f Safety Feature from performin f inverter A l aken to reduce ps are summari power supply nverter which e probability inverter. Thi d to power. A iated to study hat the loss o ed above.	A relay m AI-40A and erator action rfeeding was ontrol the have shutdown . The ector switch erging pumps rsafe our channels s logic. g their ed to the zed below. for AI-50 has that the s n the f a single
first time cha adverse consec determine how	at such a failure ha quance. Consultatio to improve the reli	a problem on the DC-DC ers at Fort Calhoun Sta is caused a reactor trip in with the vendor has b ability of the inverter	o or any other been taking pla s.	s is the kind of ace to
the loss of lo in the Updated during the tra in the ULAR.	bad aspect of this e d Safety Analysis Re ansient were, in all	vent wus compared to th port (USAR). The value cases, more conservati	te loss of load s of critical ve than those	analysis parameters predicted
A similar loss control valves That event was	of load trip occurs partially closed wi not caused by a los	red on May 28, 1976, in hile the plant was oper ss of inverter power.	which the tur ating at 100%	tine power.

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Internet Loss of Offsite Power Due to Sev RVENT DATE 10 LEE TLANDER 10 N RTT DAT TELX TILL SETURATION OF SEVERATION OF SETURATION OF SET	DECENSE EVENT REPORT ILLE	0 2 9 3 re0 18 0 5 0 0 15 0 0 1727180 107180 100					
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cold condition/refuel mode mode during a severe winter storm. To supplying power to the emergency also was available during this e minutes. The safety systems res configuration. Power was restored to the plant and inspection determined the mod have been the ice and snow associ corrective actions are planned.	he emerg buses. vent exc ponded systems ost prob	by a able	dies seco for a pecte	el generi ndary of period d for th imately e for the	ators st fsite 23 of appro e existi 1238 hou loss of	arted and KV power ximately ng plant urs. Inve offsite	supply three stigatic power to
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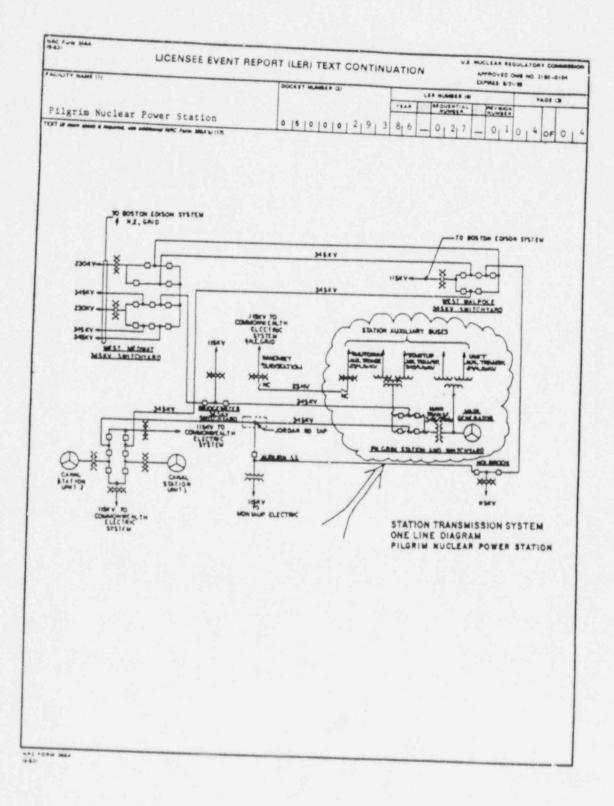
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ilgrim Nuclear Power Station	0 5 0 0 0 2 9	3816 -01217 -0110	2 OF 0
At approximately 0800 hours on No Station (PNPS) was in a cold conc position, when a severe winter st (LOP). In response to the LOP, t EK) automatically started and loa The other engineered safety featu existing plant configuration incl Secondary Containment Isolation S Protection System (EIIS Code JC).	dition with the mode form caused a loss of both emergency diesel ided, supplying power ures also responded as uding appropriate por systems (FIIS Code JW)	switch in the "Refuel" preferred offsite power generators (EIIS Code to the emergency buses. s expected for the	
The operations staff responded as maintaining the reactor in a safe Center was notified of the event	required by plant pr	Empraney Anarshiers	
The Pilgrim Station preferred off onsite 345 KV ring bus which conn transformers to two 345 KV transm transmission lines tripped as des faults. The plant can also be su This backup offsite power supply period of approximately three min	site power distributi ects the main and sta dission lines. At the igned due to near sim pplied with power fro was available during	on system consists of an artup auxiliary time of the LOP, both hultaneous, detected xm the local 23 KV grid.	1
Investigative efforts were coordi control center. Onsite efforts f inspections conducted to verify p to locate the source of the fault switchyard area verified that no were observed. Simultaneously th check the distribution lines.	ocused on the switchy roper operation durin . Interviews with pe unusual or unexclaime	ard, with interviews and g the LOP and to attempt rsonnel in the	
In order to attempt to locate the offsite transmission line was re- other 345 KV line was re-energize switches open in the plant switch fault, and observers in the switch 1106, the plant switchyard was pai operate. Again no faults or prob balance of the switchyard and remu At about 1133 normal power restors completed at approximately 1238.	energized, and at app d, each with the resp yard. Neither line i hyard witnessed no pr rtially re-energized lems were indicated, abing transmission	roximately 1015 the ective disconnect ndicated a continuing oblem. At approximately to verify its ability to and at about 1128 the	
A follow up inspection of the tran helicopter as soon as the local we or failure was observed.	nsmission lines was pr eather would permit.	erformed with a No physical decradation	

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E-72

DICENSEE EVENT R	REPORT (LER) TEXT CONTIN	UL NUCLEAR REQULATORY COM	
ACILITY RANGE (1)	DOCKET MUMBER (2)	LER MARRER IN PAGE CS	-
LETTY RADIE (1)		YEAR BEQUENTIAL D PRYMERA	
		01305	0 1
Pilgrim Nuclear Power Station	0 5 0 0 0 2 9	3816 -01217 -011 01305	-
XT (# mans apace a repury), saw additional ARC Form 3864 (17)			
Based on the inspections and i storm, it was concluded that t voltage lines due to locally h	eavy ice and snow.	to areing or the try	l
In response to this event, the automatically, supplying power safety features responded as e Though not used, the backup o margin of safety. This event public. No further corrective	expected for the existin offsite power supply pro had no effect on the he	g plant conditions.	
A somewhat similar event was r	reported in LER's 77-021	T and 78-003X.	



AC Parm INA AC			LICE	INSEE EVEN	TREP	ORT	LER)		-	A REQULATOR VEO DIME NO ES 872185	1180 -01.04	
1							Text		A (2)		7151	T
								161010		13 10 11	1 05	11
		Nuclear						- Annual and a star			-	
Failur.	A OF I	nit 2 Ma	in Ste	am Isola	tion	Valv	res to C	lose u	por	Deman	d	
EVENT DATE OF	e or u			REPORT DATE	in		OTHER P	PERFORMENT INTO		and the second second		
		Biouth TiAs	N vision		YEAR		FACILITY RAM		1	CKET NUMBER		
ONTH BAY YE		RUNNETS	14,4611						0	1010	011	1
9 2 8 8	6 8 6	- 0 10 14	-011	0 3 1 0	8 7		None		_	151010	011	1
Contraction and the statement of the statement of	7908 897		PURBUART ?		NTS OF 18	CPR \$ 10		Pa Sciparry	T	71.718		
014 8.1 THE				98.489Lel			68.7360(210)		-	73.7114		
POINTS #		STREAM THE	-	88.385at(1)		- A	68 726-C211481		-	01148 A 1344		-
	0 .	100 to 101 5 105	-	10.2040 (T)		-	68.73621(21+48)1A	i i	-	Manager and in July 1	Tear MAC	1.000
N. Same	and and a second	100 Mar 111 (100)	-	00.73ms+02H(1)		-			1			
and the second	-	1464 F31 at 686	-	00.73601001001		H	10 72mm1214a1					
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1 1		MANUFAC			-		COMPONENT	MANUFAC TURER		NEPORTABLE TO MPROS	1.1	
LAUBE EVETEN C	T#BROMMON	TURIA	10 000.04									-
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TEL IN pas some		BUBHIESHON DATI		X NO				+				****
shutdo 2CV-20 the co applie disass failur valves Once t reasse Specif	wn for 18 fai ntrol d for embled with he fai mbled. icatio	refuel: led to d room. 2 e to the l and ini- addition lure mod Both ons for 7, 1987 stuck	ing the close i The value spected have by nal fr de was valves a clos , duri open.	r 28, 198 e main si upon demu lves did ating arm d to detu een due iction i identif were te ure time ng testi The dis	team and f clos m of ermin to ex n one ied t sted of 1 ng 20 c sto	rom e wh each e th cess val he v as r ess	ation vi the manu- en opera- valve. e failu: packind ve's op- alves wi equired than 5	alves lal pul The The or mod fric eratin ere re by Te second disc	shb pe val tic pai sto	ves we The on in b ylinde red an hical	1 re oth r. id	
and th	V-201	Was re	ratued	CO BGIA	a serve							
and th	V-201	7 was re	curnea	CO Berv								
and th	V-201	7 was re	curned	to Berv								

LICENSEE EVENT REP	ORT (LER) TEXT CONTIN	UATION MALLAN ALLAN	14 NO 2180-0104
	DOCK IT HUMBER (2)	LER NUMBER IS	PAGE 13
		VEAN REQUENTIAL MERSION	T
Point Beach Nuclear Plant	0 5 0 0 3 0 1	8 6 - 0 0 4 - 0 1	
EVENT DESCRIPTION The main steam isolation val	VAC (MCTUR) had h		
1986, the Unit 2 reactor one	, 1986. At 0250]	hours on September :	
			he h."
The secondary side steam gen	erator pressure a	t the time was	

approximately 300 psig with essentially no steam flow in the main steam lines. When the MSIVs did not shut (i.e., position indication lights did not charge), an operator was sent to the valves to manually close the valves. The valves were manually closed by the operator applying force to the operating arm of each valve. It should also be noted that the non-return valves also did not close under the no flow conditions. The manual force needed to close the non-return valves was less than 7 ft-lbs. This amount of force is minimal compared to the large amount of force which would have been it is concluded that the non-return valves would have closed under these circumstances.

PLANT AND SYSTEM RESPONSES

Other systems operated as expected during the September 28 shutdown sequence.

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SYSTEM DESCRIPTIONS

Point Beach is a two-loop pressurized water reactor with a main steam isolation valve (MSIV) located in each main steam line just outside containment. A non-return check valve is installed in the turbine hall down stream of each MSIV. MSIV 2CV-2017 and non-return valve 2017A are associated with the "B" steam generator. MSIV 2CV-2018 and non-return valve 20.8A are associated with the "A" steam generator. The non-return check valves are installed to prevent the blowdown of the non-faulted steam generator in the event if a steam line break. For example, referencing the attached print, MSIV 2CV-2017 did not close, the "B" steam generator would blow down through the break until 2CV-2017 was closed. If 2CV-2017 failed to the break until dry and 2017A would prevent the "A" steam generator from blowing down through the break. CV-2018 can also be closed manually from the control room to prevent blowdown of both steam generators. It should also be noted that each steam generator has a

NAC FORM MAL

	1	1	16 PAGE 2
TY MARE (1)	DOCKET NUMBER ()	VEAN BEOUTING	
Point Beach Nuclear Plant	0 15 10 0 0 3 10 11	8 6 - 0 01	4 - 0 1 0 3 0 1
Finant group & repursed and selectioner Mile Auron Mills 10 (17)	and a deader of a second s		
flow limiting orifice immedi outlet. As can be seen from of both MSIVs in conjunction and a non-return valve the c is limited by the non-return orifices.	with a steam lin	e break bet	ween an MSIV ant system
The MSIVs at Point Beach are inclined seat, check valves Salem, Massachusetts. The of The valves are straight-thro on a heavy shaft with bushed external air operated cylind the valve open. Two series which maintains the valves is operation. These series so a main steam isolation sign pressure in the cylinder if closed, energize to latch of venting the air upon receip vented, the MSIV closes assister steam flow impinging on the	manufactured pumber for brawing number for bugh type having a di bearings. Each der which requires solenoid valves of in an open config lenoid valves are al. An accumulat the supply air i pen, solenoid val t of an isolation isted initially b back of the disc	r the valve a swinging d of the valve s 80-100 psi control the uration duri closed upor or maintains s lost. Two ves are prov signal. Wh y a spring a	is 20735-H. lise rotating ves has an .g to hold supply air .ng h receipt of s air o normally vided for hen the air is and then by
The energy industry identif identifier and system name	ication system co of MSIVs are as f	mponent fun ollows:	ction
Component function ide System name identifier	ntifier: ISV : SB		
GENERIC IMPLICATIONS			
The Unit 1 MSIVs are the sa It is possible that the Uni packing adjustment as discu be inspected, and, if necess service during the Unit 1 s	ssed in this LER.	The Unit	1 MSIVs will turned to
REPORTABILITY			
This event is reported purs or condition that alone cou safety function of structur the consequences of an acci	res or systems the		
the second se			

CAUSE Point Beach Nuclear Plant Order events Point Beach Nuclear Plant Order to determine the cause of the valve failing to close and provide for appropriate corrective action, a systematic approach for diagnosing the problem was devolved. CAUSE In order to determine the cause of the valve failing to close and provide for appropriate corrective action, a systematic approach for diagnosing the problem was devolved. The second state of the valve failing to close and provide for appropriate corrective action, a systematic approach for diagnosing the problem was devolved. The second state of the valve failing to close and provide for appropriate corrective action, a systematic approach for diagnosing the problem was devolved. We was the second state of the valve shuft of the work activity and document the data and results obtained. In addition, a representative from the manufacturer witnessed the disassembly and assisted in problem assessment. Parameters measured included presure in the operating cylinder required to start the valve opening, pressure needed to start the valve shufting, and pressure the testing activity: a. lequate clearance was observed or demonstrated between the stuffing box bushings and the disc arms in both valves. b. The shaft-to-stuffing box bushing clearances were within specified values for both valves. C. There were no signs of galling on the shaft in the gland pusher area and in the gland pusher. This galling was determined to not be recent, and was not considered to be a potential contributor to the current problem. e. A slight misalignment was noted between the vertical planes of the air cylinder shaft and the valve operating arm. There appeared to be sufficient margin to accommodate the mis- alignment. f. Each stuffing box had about 12-13 rings of packing and the packing was tightly compressed to the bottom of the stuffing box.			EPORT (LER) TEXT CONTIN	
Point Beach Nuclear Plant Discipicing of 1 Discipicing 1 Discipicing 1 Discipicing 1 Discipicing 1 <thdiscipicing 1<="" th=""> Discipicing 1 Discipici</thdiscipicing>	ACILITY N	LARE (3)	DOCLET MOMBER D	117 ALS \$7: 11
 Point Beach Nuclear Plant Plant Plant Plant Plan				TEAR BIQUEST AL MELBON
 CAUSE In order to determine the Cause of the valve failing to close and provide for appropriate corrective action, a systematic approach for diagnosing the problem was developed. This approach consisted of step-by-step disassembly of each valve, while observing any signs that might have contributed to the valve hanging up. Special maintenance procedure (SMP) #754 was prepared to control the work activity and document the data and results obtained. In addition, a representative from the manufacturer witnessed the disassembly and assisted in problem assessment. Parameters measured included pressure in the operating cylinder required to start the valve opening, pressure needed to start the valve shutting, and pressure required to fully shut the valve. The following was found during the testing activity: a. lequate clearance was observed or demonstrated between the stuffing box bushings and the disc arms in both valves. b. The shaft-to-stuffing box bushing clearances were within specified values for both valves. c. There were no signs of galling on the shaft in bushing area, or in the bushings of either valve. This galling was determined to not be recent, and was not considered to be a potential contributor to the current problem. e. A slight misalignment was noted between the valve air cylinder shaft and the valve operating arm. There were any signs of galling in the linkage and there appeared to be sufficient margin to accommodate the misalignment. 	Poi	nt Beach Nuclear Plant	0 15 10 10 10 10 10	
 In order to determine the cause of the valve failing to close and provide for appropriate corrective action, a systematic approach for diagnosing the problem was developed. This approach consisted of step-by-step disassembly of each valve, while observing any signs that might have contributed to the valve hanging up. Special maintenance procedure (SMP) #754 was prepared to control the work activity and document the data and results obtained. In addition, a assisted in problem assessment. Parameters measured included pressure in the operating cylinder required to start the valve shutting, and pressure required to fully shut the valve. The following was found during the testing activity: a. lequate clearance was observed or demonstrated between the stuffing box bushings and the disc arms in both valves. b. The shaft-to-stuffing box bushing clearances were within specified values for both valve. d. There were no signs of galling on the shaft in bushing area, or in the bushings of either valve. d. There were some signs of galling on the shaft in the gland pusher area and in the gland pusher. This galling was determined to not be recent, and was not considered to be a potential contributor to the current problem. e. A slight misalignment was noted between the waive operating arm. There were appeared to be sufficient margin to accommodate the misalignment. 		amout a required, and addression tolly facts \$85.4 to (17)	1.	
 diagnosing the problem was developed. This approach consisted of step-by-step disassembly of each valve, while observing any signs that might have contributed to the valve hanging up. Special maintenance procedure (SMP) #754 was prepared to control the work activity and document the data and results obtained. In addition, a representative from the manufacturer witnessed the disassembly and assisted in problem assessment. Parameters measured included pressure in the operating cylinder required to start the valve opening, pressure needed to start the valve shutting, and pressure required to fully shut the valve. The following was found during the testing activity: a. lequate clearance was observed or demonstrated between the stuffing box bushings and the disc arms in both valves. b. The shaft-to-stuffing box bushing clearances were within specified values for both valves. c. There were no signs of galling on the shaft in bushing area, or in the bushings of either valve. d. There were some signs of galling on the shaft in the gland pusher area and in the gland pusher. This galling was determined to not be recent, and was not considered to be a potential contributor to the current problem. e. A slight misalignment was noted between the vertical planes of the air cylinder shaft and the valve operating arm. There were appeared to be sufficient margin to accommodate the misalignment. 	CAU	ISE		
 a. lequate clearance was observed or demonstrated between the stuffing box bushings and the disc arms in both valves. b. The shaft-to-stuffing box bushing clearances were within specified values for both valves. c. There were no signs of galling on the shaft in bushing area, or in the bushings of either valve. d. There were some signs of galling on the shaft in the gland pusher area and in the gland pusher. This galling was determined to not be recent, and was not considered to be a potential contributor to the current problem. e. A slight misalignment was noted between the vertical planes of the air cylinder shaft and the valve operating arm. There were appeared to be sufficient margin to accommodate the misalignment. 	dia ste tha act rep act rep rep rep	gnosing the problem was p-by-step disassembly of t might have contributed ntenance procedure (SMP) ivity and document the d resentative from the man isted in problem assess ssure in the operating of ning, pressure needed to uired to fully shut the	developed. This a f each valve, while to the valve hang #754 was prepared data and results ob hufacturer witnesse ment. Parameters m sylinder required t	systematic approach for approach consisted of e observing any signs ging up. Special i to control the work tained. In addition, a d the disassembly and measured included to start the valve
 b. The shaft-to-stuffing box bushing clearances were within specified values for both valves. c. There were no signs of galling on the shaft in bushing area, or in the bushings of either valve. d. There were some signs of galling on the shaft in the gland pusher area and in the gland pusher. This galling was determined to not be recent, and was not considered to be a potential contributor to the current problem. e. A slight misalignment was noted between the vertical planes of the air cylinder shaft and the valve operating arm. There were no signs of galling or rubbing in the linkage and there appeared to be sufficient margin to accommodate the misalignment. 		iequate clearance was	observed as a	
 d. There were some signs of galling on the shaft in the gland pusher area and in the gland pusher. This galling was determined to not be recent, and was not considered to be a potential contributor to the current problem. e. A slight misalignment was noted between the vertical planes of the air cylinder shaft and the valve operating arm. There were no signs of galling or rubbing in the linkage and there appeared to be sufficient margin to accommodate the misalignment. f. Each stuffing how had about 12.12 minutes. 	b.	The shaft-to-stuffing	how turbing since	
 determined to not be recent, and was not considered to be a potential contributor to the current problem. e. A slight misalignment was noted between the vertical planes of the air cylinder shaft and the valve operating arm. There were appeared to be sufficient margin to accommodate the misalignment. f. Each stuffing how had about 12.12 minutes. 	c.	There were no signs of in the bushings of eit	galling on the sh	aft in bushing area, or
 A slight misalignment was noted between the vertical planes of the air cylinder shaft and the valve operating arm. There were no signs of galling or rubbing in the linkage and there appeared to be sufficient margin to accommodate the mis- alignment. Each stuffing box had about 12.12 minor. 	d.	determined to not be r	ecent and use not	is galling was
f. Each stuffing box had about 12-13 rings of packing and the packing was tightly compressed to the bottom of the stuffing box.	e.	A slight misalignment the air cylinder shaft no signs of galling or appeared to be suffici.	was noted between and the valve oper	the vertical planes of rating arm. There were
	f.	Each stuffing box had packing was tightly con box.	about 12-13 rings of mpressed to the bot	of packing and the toom of the stuffing

	LICENSEE EVENT REI	PORT (LER) TEXT CONTIN	EXPINES & STAS	04
CILITY MANS	13)	DOCE ET HUNNETA (2)	LER NUMBER IS FASE IS	÷
			VEAR BEQUENTIAL PLUEROS	
Poin	t Beach Nuclear Plant	0 5 0 0 0 3 0 1	1 8 6 - 0 0 4 - 0 1 0 5 0F	*
CT (# mars apon	a supported, was addressive Auto: Auron 38864 to/ (17)			
g.	about 1 psi pressure of 2CV-2018 operated in a	a jerky motion. The second	smoothly, requiring only it. The cylinder for this was considered to be soment between the lower al in the dashpot.	
	removed under an appro reducing all potential and operating system. can for the cylinder s downward travel of the	no longer used, the oved modification is contributors to i The dashpot remains shaft outside the of e cylinder when not	in the interest of friction in the valve ains only as a protective cylinder and limits ot attached to the valve.	
h.	slightly when air pre- the valve disc. This operator who initiall 5-10% and it was conc. moved at all during i	ssure is released v issue was discuss y reported the val luded that the val ts test.	without actually moving sed with the auxiliary lve had closed about lve had not, in fact,	
i.	been axially-laminate better job of sealing present'y-recommended	d graphite. The w than the original packing; however,	wound graphite does a 1 material and is the	
CONC	LUSIONS			
The		were reached as a	result of the testing	
a.	The reason that 2CV-2 packing friction.	017 failed to clos	ose was solely excessive	
b.	not close 5-10% as pr	eviously thought.		
c.	and for the strong and	excessive friction	ODELECTING CLARINGER HER	
d.	a second and the manual of The	not be explained.	load after the valves had ecification required test . A number of theories	

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CIUTY HARE IN	PORT (LER) TEXT CONTIN	UATION			- 04 840	1 COMMIS	54.
	DOCKE / NUMBER (2)	1		And in case of the local division of the loc	-		_
		*8.4.8	BEQUENTIAL BUMPER	Biven		4.61 3	
Point Beach Nuclear Plant			1				
T IT IT THE BOARD & TOPHING AND PRODUCT AND AND AND AND AND	o le lo lo lo B P P	B 16 -	0014	-01	016	OF 1	1
 One theory is that the the load required to proto the test and as the value or break-away friction packing into the value of friction is not overcome reduction and value mome Both of these would affed performance. A definitive statement the steam line break accide However, since we have the value in the full or line, this would indicat 	friction load may event initial mot ve sat in the ope increased slight! shaft irregularit e, the static-to- entum forces woul ect valve closure that the valve would dent with steam fin had problems in the	have h ion of n posit y due t ies. I dynamic d not b during uld hav low ca he past	een ju the va ion th o flow f the frict e real test e clos nnot b with	est bel live du static ion for ized. ed du e made holdin	ow ring ic e ad rin,		
line, this would indicat to pull the disc down in CORRECTIVE ACTIONS	the flow stream.	a force	which	attem	pts		
The following corrective acti	ons have been imp	lement	ed:				
a. Spacers have been instal reduce the number of pac task was accomplished th and in accordance with d	rough an approved iscussions with t	modif: the manual	o 8-9. ication ifactum	This n reque rer.	est		
Prior to the Unit 1 refu modifications will be ma facilitate testing of th modified to collect more Technical Specification modifications will be do fall refueling/maintenance	eling/maintenance de to change the e valves. The in data than a simp limit for the clo	outage air sys service le pass	tem to tem to test	will h	e		
Even though we have not e MSIVs, they will be modified rings similar to Unit 2 c	experienced a pro fied to reduce th during the spring	blem wi e numbe 1987 o	th the r of p utage.	Unit acking	1		
The details of this event and maintenance personnel upon the importance of no subsequent post maintenan operability. Post-mainte FBNF 3.1.3, "Maintenance	t will be dissemin with special emp of tightening val- nce testing to ver	nated t phasis ve pack	o oper being ing wi	ating			

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LICENSEE EVENT RI	EPORT (LER) TEXT CONTIN	NUATION APPROVED DAT	e no 3180-0104 85
ACILITY MARIE (1)	DOCKET NUMBER 12	LEA WUMBER IS	PAGE (3
		VEAN MOUTATIAL MEVERS	
Point Beach Nuclear Plant	0 5 0 0 3 0	186-0104-011	0 7 0 1 1
EXT IF was appeared a supported, was additional helds from Allian to (17)			
e. A modified IT-285 val completion of the mai indicated that 2CV-20 closed in 2.5 seconds Technical Specificati	17 closed within 1 These results a	.5 seconds and 2CV-2 re well within the	ts 018
SAFETY ASSESSMENT			
	hut upon demand co	ould have an effect o	n a
Failure of both MSIVs to s steam line break event. F	ailure of the MSIV	to close would resu	
steam line break event. F in the faulted steam gener	ator blowing down	completely. This re	Team
in the faulted steam gener is the same as the result	of the steam line	break nappening upst	ar a
of the MSIV. It should be	noted that Forne	ne steam generator.	As
steamline break which invo	line break upstream	of the non-return V	valve
in one steam line will not	result in the blo	owdown of the	
in one steam line will not non-faulted steam generato	or due to the funct	tion of the non-retur	rn
valve.			
		when with the	
The worst case steam line failure of both MSIVs to s stream of the non-return v rapid cooldown of the read generators were dry or the Action Procedure ECA 2.1 a circumstances. Note that located in the turbine ha	shut would be a sci valves. This accid tor coolant system e MSIVs were shut. addresses operator 2017A and 2018A a	dent would result in m until both steam Emergency Continge actions in these nd downstream piping ore. possible to rea	are ch
located in the turbine has and manually close these to hot steam from a break in under these conditions wo reactor coolant system cou the flow orifices at the limit the initial cooldow	this area. Manua uld result in rega oldown rate. The	l closing of the MSI ining control of the flow limiting featur the steam generators	Vs e of
			ally
The overall conclusion is resulting in the ultimate pressure and decay heat r dumping steam through atm	ability to contro	iary feedwater flow	and

DUTY NAME (1)	PORT (LER) TEXT CONTIN	UATION	APPROVED ON	8 NO 2150-0104
			EXPIRES \$121	18 NO 2180-0104 185
	DOCKET WUNDER (2)	LER NUPRE		PAQE (3
		TEAR BEDUEN	A. MIVEON	
Point Beach Nuclear Plant	0 5 0 0 3 0 1	816 - 010		
I of many append is required, use additional table form 2004 (a) (17)		10101-1010	4 - 0 1	0 8 0 1
SIMILAR OCCURRENCES				
December 31, 1985, after a when the operator in the commanually. During a post-tradjusted and the valve pass. Technical Specifications. all three test times being 1 LER 65-005-00, "Reactor Trip details on this event.	ip investigation a ed the stroke test The test was perfor	valve pack required f red three	valve ing was y times wit	
On January 17, 1987 during a failed to close completely v required time of 5 seconds. relatively the same results. easily to within 85 to 90% of The valve was tested a fifth which normally holds the val to 0 psig. The valve again close the valve completely w valve was again reopened and 90% closed position. Since been a different pressure in atmospheric dump valve was o easily from the 90% closed p	The valve was tes In each instance closed within appro- time and the oper lve open during ope closed to 85% clos was measured at les tripped closed. it was suspected to the "B" steam gen	Il Specific ted 4 time the valv ximately 2 ating air ration was ed. The t s than 50 The valve hat there i	ation s, each we e closed -3 second cylinder bled dow orque to ft-lbs. T closed to may have	n he a
The packing of the valve was determine if the packing was difficulties. The valve was Each closure required less t closure, one more test was t closures were due to a diffe generators because of openin when the valve operating air remained in the fully open	then closed compl han 2.5 seconds. o be run to determ rential pressure b	ctor to the etely three After the l ine if the etween the pheric dump completely	e closing times. ast three goo two stear valve.	od

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	ORT (LER) T	EXTC	ONTINU	ATION			
514/74 RAME (1)	DOCK IT NUM	11 Q					PAGE (3
				*844	BED-LATA		
Point Beach Nuclear Plant	0 5 0	0 0 1	3 10 11	8 6 -	-01014	-0110	19 OF
T IF many spaces a response, use independent MMC Agent \$86.4 \$1 1371							
	TTACHMEN	A" T	u				
MSIV 2CV-2017 FAILURE TO CLC						í na	
Unit 2 was placed in a hot s to repair a packing leak on transmitter root isolation v transmitter, a test of the L (MSIVs) was performed to ver these MSIVs which had stuck temporary change to IT-285 v shutdown and to include meas air-operating cylinders requ	valve. F Unit 2 ma rify the open dur was initi surement pired to	ollo in s cont ing ated of t stro	the late the pre-	operion operion ast or est ti essur- e val	ability tion vi ability tage. he MSI es in ves.	of the alves y of A Vs at hot the	
The testing was performed or All steam loads were minimiz Unit 1 for part of the Unit "A" valve was performed first tests of 1.80, 1.47, and 1.5 stroking of the valve had or trip.	2 gland st with v 35 second courred p	stea valve is. prior	am syst e closu It sho r to th	tem. ure t ould he fi	Testi imes i be not rst IT	ng of the n the thi ed that i -285 tes	on
Part of the changes to IT-2 pressure in the air operation of the valve after valve cl	osure ter	367 V	any rula	defi	ment c nable	f the position	s
The test results along with 1986 refueling outage are a	referen s follow	ce a s:	s left	data	from	the fall	
The test results along with	E IOIIOW		MSIV	(2CV-	2018)		
The test results along with 1986 refueling outage are a Valve position	As f		MSIV	(2CV-	2018)	the fall oft press 1986	
The test results along with 1986 refueling outage are a Valve position Test #:	As f	"A" ound 2	MSIV press 3	(2CV-	2018)	eft press	
The test results along with 1986 refueling outage are a Valve position Test #: Starts to close	As f 1 33	ound 2 34	MSIV press 3 33	(2CV-	2018)	eft press 1986 34	
The test results along with 1986 refueling outage are a Valve position Test #:	As f 1 33	"A" ound 2	MSIV press 3 33	(2CV-	2018)	eft press 1986 34 19	
The test results along with 1986 refueling outage are a Valve position Test #: Starts to close	As f 1 33	ound 2 34	MSIV press 3 33	(2CV-	2018)	eft press 1986 34	
The test results along with 1986 refueling outage are a Valve position Test #: Starts to close Reaches 80% closed Starts open from 20%	As f 33 17	"A" ound 2 34 16 59	MSIV press 3 33 16	(2CV-	2018)	eft press 1986 34 19	
The test results along with 1986 refueling outage are a Valve position Test #: Starts to close Reaches 80% closed	As f 1 33 17 59 83 sentiall the outa	"A" ound 2 34 16 59 84 y nc	MSIV press 3 33 16 59 84 chang n the	(2CV-	2018) As le	eft press 1986 34 19 57 81 performa 86. The the valve	ance
The test results along with 1986 refueling outage are a Valve position Test #: Starts to close Reaches 80% closed Starts open from 20% Just reaches 100% open These pressures indicate es of the valve at the end of slight increase (approximat could indicate an increase	As f 1 33 17 59 83 sentiall the outa	"A" ound 2 34 16 59 84 y nc	MSIV press 3 33 16 59 84 chang n the	(2CV-	2018) As le	eft press 1986 34 19 57 81 performa 86. The the valve	ance

E-83

NAC Farm MAA US NUCLEAR REQULATORY COMMISSION LICENSEE EVENT REPORT (LER) TEXT CONTINUATION APPROVED ONE NO 2150-0104 117.845 \$21.8 FACILITY MANS (1) DOCK! MARKEN -----FAGE 3 TEAR SEDUCTION Taywest . Point Beach Nuclear Plant 0 |5 |0 |0 |3 |0 |1 8 |6 - 0 |0 |4 - 0 |1 1 10 0F 1 |4 The "B" MSIV (2CV-2017) was then trip tested with the following results. Again it should be noted that no prior stroke had been done. Valve stroked to 85% closed position in 3.07 sec. Valve stroked to 85% closed position in 2.27 sec. Valve stroked to 90% closed position in 3.03 sec. Valve stroked to 90% closed position in 2.70 sec. In each case the valve disc appeared to drop to the 85-90% closed position rather rapidly then hit a cushion. The control room was contacted to see if there was any difference in steam generator pressures and that both reactor coolant pumps were operating. Both pumps were running and there was no discernable difference in pressure between the steam generators. A plant process computer printout of the steam generator pressures obtained after the event confirmed the lack of an indicated pressure difference. Instrument & Control personnel however believe that a small pressure differential could exist even without indication in the control room due to the allowable inaccuracies in the instrumentation. Following is the pressure data similar to the "A" MSIV for the air operating cylinder: "B" MSIV (2CV-2017) Valve position As found press. As left press. Test #: 2 2 1986 Starts to close 13 14 13 16 Reaches 80% closed 3 3 3 6 Starts open from 20% 29 30 29 31 Just reaches 100% open 41 43 43 43 Essentially all the pressures are slightly lower than the as left pressures and would tend to indicate a difference in the pressure gauges rather than a difference in the valve. There was a slight increase in the pressure difference required to change the direction of the valve; about 5% on the average. This could be caused by a slight increase in the friction of the valve. In general, the valve performed as expected and the pressure results appeared to be essentially the same as left after the refueling outage. -----

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TY NAME (1)	DOCKET NUMBER IS	LER NUMB		1851 3
		**** SLO-1		
	0 16 10 10 10 13 10	11 916 -010	14-011	111 0+ 1 4
Point Beach Nuclear Plant	0 0 0 0 0 0 3 1	19 10 10 1 12.04		
F HEAR'S ADDIES & FORMETER, LAN PROPERTY ADDI. TOTAL ADDIES				
The pressure in the air oper the final pressure test and torque required to fully clo The valve was reopened and about 90% closed. The "A" is was opened to drop that gem- it was less than that in thi- that there was no pressure steam generator holding the atmospheric dump valve was Therefore, it is felt that shut if a small reverse pre existed during the initial To evaluate the packing tor resistance of the valve, th and retorqued to 65 ft-lbs A reduction in the packing because a slight amount of torquing. The valve was th completely with times of 2 the valve did not complete from the packing or a smal generators. The MSIV was started to take pressure r during operation of the va the valve remained in the the valve appeared to be f theorized that the valve did and open the valve body for ASME Code Section VIII was would be required if weld In addition, the valve man guidance. It was determin <u>CAUSE</u> When the valve was opened valve disc and the disc h	ose the valve w tripped shut w steam generator erator's pressu e "B" steam gen differential he "B" MSIV open. opened, the "B" the valve would ssure different tests. rque contribution tests. rque contribution tests. rque contribution torque was not steam leakage hen tripped and 09, 2.25, and ly close instan l differential opened one more eadings on the fully open post pue to the valve further open the disc stop may h the the unit in or inspection. s reviewed for ing would be re nufacturer was ned the welding	as less than ith the valve atmospheric re slightly erator. Thi tween the "A As soon as MSIV closed i have gone of tial condition on to the clo he "B" MSIV v eft in Novemu considered of existed with the valve of 2.21 seconds thy was due pressure bet the time and th operating al air was bein tion. Attem shaft were an normal and ave broken. the cold shut guidance to guired to re contacted fo would be ac	e stoppin dump val to make s s would e " and "B" the "A" I fully. completely on had not osing was loose ber 1986) desirable the curr losed the curr losed the curr losed the ricti ween stea e work was r cylinde of bled of opts to cl unsuccess i it was The decis tdown cond determine pair the r any add ceptable.	g at ve ure msure ned ent ason on ms ff. lose sful. sion dition what valve. litional
when the valve disc and the disc h area and hung up on some held the valve stop to th the valve disc at the poi not appear that a signifi shaft by the as-found cor	of the remaining the valve body. Int of hang-up i icant loading h	one rap of a	a large ha	ammer on does
				And an other data and the second data and the

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	DOCKET NUMBER (2)		Name and Address of the Owner, or other Designation of the Owner, where th
		TEAR STOUTUNAL MANAGE	PAGE (3)
Date: Date: No. 1			4
Poir, Beach Nuclear Plant	0 5 0 0 0 3 0 3	1 8 6 - 0 0 4 - 0 1	112 05 1
A good portion of the weld f It appears to have initiated stop had been cut back sever disc travel. The original h by a grinding wheel and the the piece left in the valve. likely location for the init should also be noted that the indication of areas of poor <u>GENERIC IMPLICATIONS</u> The design of the disc stop of 1CV-2017, and 2CV-2018) of th However, we do not believe th to cracking of the fillet we 2CV-2017 has a larger valve of three valves. Often, when th differential steam pressure of larger operating air cylinder closed seat. At times the st to prevent the opening of the the atmospheric dump valve to generator. When the atmosphe opens very quickly and in the against its disc stop. Becat operating cylinder, the press 40 psig Tange, whereas the ot require at least 80 psig. Th for 2CV-2017 builds up to ness before the valve opens, there take the valve all the way op vent hole on the nonpressure 2CV-2017 is 1 inch line where line is 1/8 inch in diameter. prevent the buildup of air pr operating piston during fast valves, the small vent line a the piston during quick movem novement of the valve into th	on the corner of al years ago to a orizontal cut on arc from the cutt This appears to iation of the poi e old weld was a fusion. on three other val fusion. on three other val that the other val lds as experience operator air cyli he valves are ope across the valve of pressure to mov team pressure difference of cuce pressure eric dump valve i e case of 2CV-201 use 2CV-2017 has sure to open the ther valves at Po hus, if the press ar instrument air e can be enough s ben to the disc s ted side of the a best for the remai The 1 inch ven ressure on the bai movement. In the illows the buildu	the stop where the allow for additional the stop had been ting wheel extended o have been the mos- int of fracture. If fillet which had alves (1CV-2018, the beach is the same lives are as suscept ed in 2CV-2017. Inder than the other end, there is a which requires a more the valve from if ferential is adequired and the effected s is opened, the MSIV 7. the valve from if ferential is adequired and the effected s is opened, the MSIV 7. the valve slams a much larger air valve is usually fure in the air cyl fored energy to qui stored energy to qui stored energy to qui stored energy to qui top. In addition, air cylinder pistor ander of the valves is large enough ick side of the the case of the other	al made into it ible it ible er much its iate y of steam y in the inder up) lickly the i for i this to

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CILITY BANE (1)		DOCKET MUMBER (2)			PAGE (3
			VEAS MOUSELS	AL MEYECK NUMPLE	
Point Bea	ach Nuclear Plan	o [5]0]0[3]0	11 8 6 - 0101	4 - 011 11	3 08 1 1
CONCLUSIO					
considered that the accounts It is als the integ difference maintenat	ed operable at i differential p for the behavi so believed the grity of the di	ented here, we believ the time of the init: ressure between the for or of 2CV-2017 during re is no reason for sc stops in the other cylinders. We will, ts to inspect the di- nvenient.	two steam gen g the initial immediate con r MSIVs becau however, iss	erators testing. cern about se of the ue	
REPORTAB	and a statement of				
not repor	rtable, a court	luation determined t esy emergency notifi anuary 17, 1987.	he broken dis cation system	sc stop was n call was	
CORRECTI	VE ACTIONS				
informat propagat beveled, with a g	ional die penet ed into the val	ned up and the area rant test (PT) to ma ve body. The old di elded back in. The t weld and an inform	sc stop piec	e was installed	
the second in the second states and	reinstallation The pressures r	op was reinstalled t of the cover to ver equired to stroke th	LIV ILES DOV	CHICHT AT A	he as
Valve po	sition	As found pr Test 1	ress. (psig) Test 2		
Start va	lve open	22	22		
Valve fu	11 open	39	39		
Valve st	arts shut	19	19		
Valve fu		2	1		

UCENSEE EVENT REPO	ORT (LER) TEXT CONTINU	JATION		004 A104 + COMMISS
ACIUTY MANE (1)	DOC411 HUMB14 (2)	and the second se	A011 (6	Page 13
		*84.8 54.0 B	WHIN NUMBER	
Point Beach Nuclear Plant	0 5 0 0 0 3 0 1	816 - 0	014 - 011	1 4 0 1 1
These values are comparable taken during the testing per Full opth and full closed in the valve was tripped from a clest d.	formed during the dication was also	last re: verified	fueling ou	1200
Vien the valve was reassemble operating temperature, IT-28 this testing was acceptable. steam through the packing has	5 was again perfor It should be not	med. Th	a requit	of ge of
We are considering the insta port of the air cylinder for testing would be done to dete would have on the valve closs impact, the plug could be per	2CV-2017 when ope ermine the impact ing time. If they	this ins	valve.	Some
SAFETY ASSESSMENT				
The safety assessment for the the original report.	e supplemental rep	port is t	the same a	ε
SIMILAR OCCURRENCES				
This is the first case of a h	broken stop found	in an MS	IV.	

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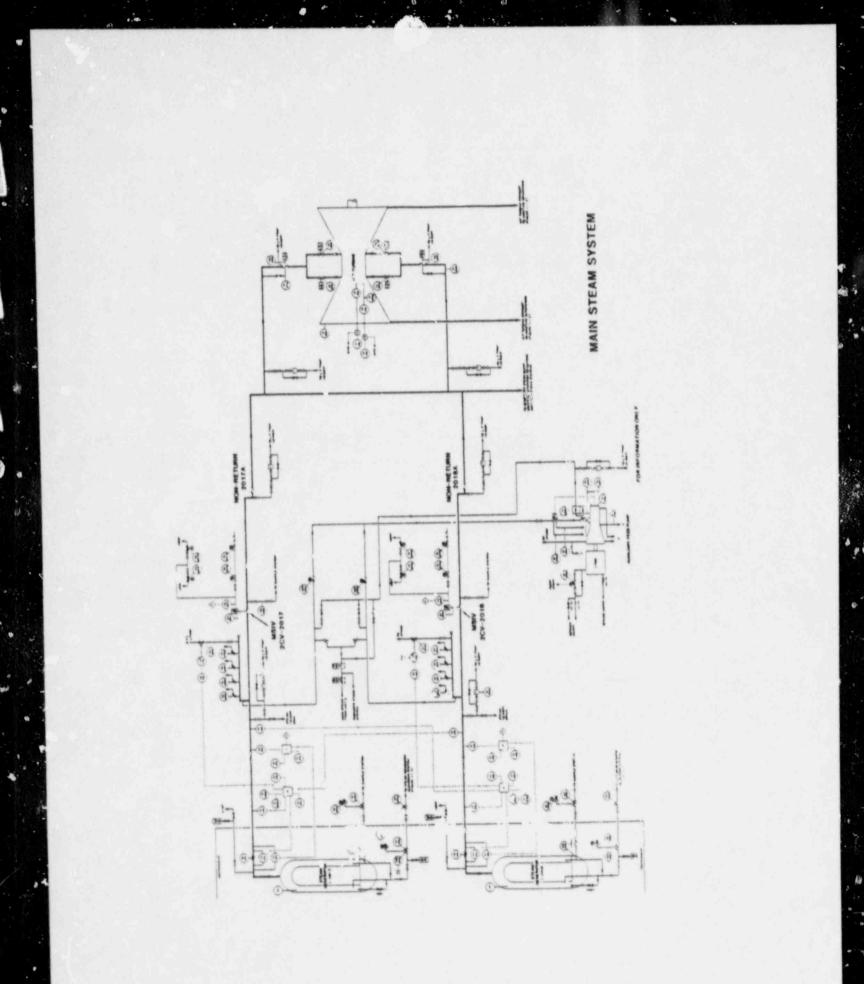
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		LIC	ENSEE EVI	ENT REP	ORT	(LER)	U.L. N	APPROVIO		
CILITY NAME (1)					-		DOCKET NUMBER	141		
Calvert Cliffs.	Unit 2						0 15 10 10		18.110	SF CI
ful is						and the second second	. 1. 1. 1.	1-1-11	151.10	me l
Reactor Trip Ca	used by Re	actor Pu	ump Surge	Capacit	or F	ailure				
ONTH DAY YEAR YEAR			REPORT DA			Concerning and the second s	FALILITIES INVO	and the second se		
	NUMBER	ME VIERON	MONTH DAY	YEAR		FACILITY NAS	*85	DOCKET N.		
이 같은 것 같아.	1.1.				-			0 15 0	10101	11
9058680	0 0 0 6		0 2 0 5	8 7				0 18 10	10101	11
	0.40216	U PORBURNI	20 4061a1	19 N 18 OF 10 C	1 9 10	Cherry one or more o	of the factoring) (1			
POWER	0 406 (all 1 1 1)		80 38-4113		A	50.73(a)(2/(w)		73 71		
11010 1	. 408 (411(a)		60.000002		-	\$0.73ia1(2)(va)		Surrouge .		
	. 406 is http://		60.73ia102111				6		A Specific in a second	sac Apre
present.	0.408163137/0x1	_	\$0.73ia1(2)(a)			60 72m1(211vist)				
	2.4081a211.21a1		\$4.7 \$x1(2) (ac)			\$0.73ia1(21)41				
wt			ICE NEEL CONTAC	FOR THE LE	# 112/					
							AREA CODE	TELEPHONE	NUMBER .	
L. S. Larragoit	, Licensi	ng Engin	eer				31011	2.6.0	1+141	6.6.
	COMPLETE	ONE LINE FOR	TACH COMPONEN	T PAILURE DE	RC #184		1 (13)	1-1-1-	1-141	7121
USE STEEN COMPONENT	MANUFAC	PERORTABLE		CAUSE S				mannar .		
	TURER	+0 N#805		CAVIT 1		COMPONENT	*1',454C	REFORTA TO NRE	01	
B AIB ICIAIP	W111210	Y		++	-		111		-	
E SB IFISIN	A161110	Y I		1 1	1	1.1.1	1 1 1 1	1	1.0	
	B'tes/ Lint	INTAL REPORT	EXPECTED INE				EXPLOY			
111			Y NO				BUBM SS	ON C		
At 2358 on Se Cliffs Urit 2 initiated by R steam dump v additional pri Auxiliary Fee 22. The low 1 cooldown rate surge capacits at 0825 on Sej associated ste the actuator. The correctivy R CP breakers technical man shimming proc	ptember 5, reactor aut eactor Coo alve for 22 mary coold dwater Act evel occurr to Troubles or. The sur otember 6, am dump so The soleno e action is t switchgear. ual was cha	1986, wh omaticall lant Pum Steam G own and v uation Si ed while hooting d ge capaci 1986. Th olenoid va id valve i to replace Additio inged (up	ile operation by tripped of p (RCP) 21 enerator () was manual gnal was go etermined itor was re- e atmosph alve leakin internals w a RCP sury nally, the s	on a low IA break SG) faile enerated controllin the RCI placed a eric stee g air by ere repl ge capaci at mosphe endation	reac er tr d to ed. due bre nd ti its s aced itors	tor coolan ipping ope reseat foll At 0010 of to a temp 5 levels to aker trip v he pump w ump failed eats and m with indus	t flow trip n. The at lowing the n Septemb orary low limit the was due to as returned to reseat aintairing ctors loca	p signal mosphe trip ca er 6, 1º level ir primany a faile d to set due to pressu ted at t	ric using 186 an 186 V d rvice its re on he	
				A						

Tailor Note: 10 Sector World St Sector World St Sector World St Calvert Cliffs, Unit 2 0 is [0 [0] 3] [18] \$10 - 0[0 [0 - 0]] 0[2 [0] 0] Sector World St Sector World St Sector World St On September 5, 1986, at 2358, while Calvert Cliffs Unit 2 was operating in MODE 1 at 1003, power, the reactor (EIIS A C-R CT) was automatically tripped on a Low Reactor Coolant FLow The ginal resulting from Reactor Coolant Pump (R CP) (EIIS A B-P) 21A Cooling in the trip, the primary cooldown rate was faster than expected and the atmospheric staan dump valve (EIIS SH-PCV) for 22 Staan Generator (SG) (EIIS SH-SG) was noted to still be open. The dump valve was manually isolated at 010 on Septem ber 6. 1986. While memory outwork (EIIS SH-PCV) for 22 Staan Generator (SG) (EIIS SH-SG) was noted to still be open. The dump valve was manually isolated at 010 on Septem ber 6. 1986. While memory outwork (EIIS SH-PCV) for 22 Staan Generator (SG) (EIIS SH-SG) was noted to still be open. The dump valve was manually isolated at 010 on Septem ber 6. 1986. While memory outwork (EIIS SH-PCV) for 21 Started automatically as expected and was secured when 5G level was promptly restored. Post trip review data showed the reactor protection system (EIIS-JC) functioned properly and no Technical Specification limits were exceeded. There are no safety consequences since this event was much lass severe than the Loss of Coolant Flow Analysis in the Final Safety Analysis in the SH-CPOS) was also replaced due to dign of the staan dump solend ware on the inkage. Upon completion of repairs the atmospheric staan dump solend ware on the staan dump solend ware on the inkage. Upon completind the resthater to allow qdik coparing. The valve inte	LICENSEE EVENT	REPORT (LER) TEXT (ONTIN	OITAL	N			HO-10	она но 1 1 ж	16-215	
Calvert Cliffs, Unit 2 0 s 0 9 1 8 0 0 0 0 0 0 0 0 0	ACTUTY NAME (A)	DOCKET NUMBER (2)		T						AGE 3	
We serve a serve a serve server of the server of the server of the server and the server of the s		and the second		YEAR	5.0	SUENT A.	T	ALL BOT	1	TT	
We serve a serve a serve server of the server of the server of the server and the server of the s	and the address date of					de la	1	1.10	a fa	OF	
On September 5, 1986, et 2358, while Calvert Cliffs Unit 2 was operating in MODE 1 at 100% power, the reactor (EIIS AC-P.CT) was automatically tripped on a Low Reactor Counter Flow This signal resulting from Reactor Counter Fung (RCP) (EIIS AB-P) 21A breaker (EIIS AB-BK 8) tripping other. Emergency Operating Procedure (EOP>O (Post Trip in netiate Actions) and EOP-1 (Reactor Trip) were properly carried out. Following the trip, the prim ary cooldown rate was faster than expected and the atmospheric stam dump valve (EIIS SB-PCV) for 22 Stam Genemator (SG) (EIIS SB-SG) was noted to still be open. The dump valve was manually isolated at D010 on September 6. How manually controlling SG level us in the primary cooldown rate, an Audilary Feedwater Actuation Signal (EIIS JB) was generated at D010 when 22 SG level manually ontrolling SG level was monable to a level of -175 in ches. The motor driven Audiliary Feedwater Pung (EIIS SJ-P) started automatically as expected and was secured when SG level was promptly restored. There are no safety consequences the sound to be any been more severe under alternative diruum stances.		0 5 0 0 0	31118	18 0		21010	1	1011	1015	1.1	013
	 On September 5, 1986, &: 2358, will 100% power, the reactor (EIIS ACCoolant Flow Trip signal resulting breaker (EIIS AB-BK P) tripping of Inmediate Actions) and EOP-1 (Following the trip, the primary of atmospheric steam dump valve (EW was noted to still be open. The diate Activation Sepased through the actuation sets The motor driven Auxiliary Feedwater Actuation Sepased through the actuation sets The motor driven Auxiliary Feedwater and was secured when SG level with the motor driven Auxiliary Feedwater Actuation is given this event was much less sets and no Technical Specification is since this event was much less sets Safety Analysis Report, Section is would not have been more severe. Investigation revealed hig), press solenoid valve (EIIS SB-FSV). The atmospheric steam dump valve as steam dump's positioner (FIIS SB-the linkage. Upon completion of the norm is and quick actuation motion sets are and and quick actuation the changed to include a shim ming publing wear was due to the linkage. Donor completion of the asymptotic to ground installed for each RCP motor. A runge capacitors were installed to include state by the motor and the set of the motor. A surge capacitors were installed to include the set of the motor of the set of the motor of the differential and ground breaker. Investigation determine (AP) internally shorted to ground installed for each RCP motor. A surge capacitors were installed to initial voltage surge seen by the motor of the motor of the motor. 	C-RCT) was automat g from Reactor Cool open. Emergency Op Reactor Trip) were p coldown rate was far EIIS SB-PCV) for 22 S hump valve was manu g SG level to limit th ignal (EIIS JB) was g point (+170 inches) ar water Pump (EIIS SJ- ias promptly restored reactor protection s mits were exceeded. evere than the Loss o 14.9. Also, the react s under alternative c ure air was leaking b is valve applies high chator to allow quid ted. Although not be -CPOS) was also repl repairs the at mosp nodes. wheric steam dump's (age being improperly nufacturer, the Tecl rocedure (using flat overcurrent relays w ed the root cause to d. There are three s lithough not needed to provide protection windings when the fu-	ically tr ant Pum imating i operly of ther than taam G ally isole e primais eneratek d reach P) start d reach P) start f Coolai or was i roumsti y the se pressure k openiu lieved t aced du eric sti vashers ere fou be a RC unge cai while of to the s reder br is distant	upped p (R C Procession carries in expe- eneration ated a sy coo lat O ed a lu- ed a u- ed a u- ed a u- ed a u- ed a u- inces. at of e air d ar sho ar sho a	on a a P)(I) ture of to cted to cted	Low I EIIS AI (EOP) and tiss (EOP) and tiss and tiss (EOP) and tiss and	dure the set of the se	ctor) 21 A Post SB+SC temb C levis iches. expense the ev prop quenc the ev prop quenc the ev spe prop quenc the ev spe se tes c levis to of tr is c levis to c levis to c c levis to c levis to c levis to c levis to c levis to c levis to c levis to c levis to c levis to c levis to c levis to c levis to c levis to c levis to c levis to c levis to c levis to levis levis to levis to levis levis to levis to levis levis to levi	Trip ()) er 6. el icted erly es inal ent f the heric r on ted in ve xox. was cage. (B- ese) xe e		

LICENSEE EVENT REPOI		EXPROVED ONE N EXPRES & D B	6 236-0204 -
COLOTT NAME IN	DOCKET NUMBER (2)	LER NUMBER IS	FAGE IS
		VERA BIOLENTIAL MELSON NUMBER NUMBER	11
Calvert Cliffs, Unit 2	the second second second		111
T if many space is required use addressing tub? Form 306.4 (11)	6 6 0 0 0 3 1 8	816-01016-011 01	3 01 0
The manufacturer has provided recommon surge capacitors which are: Las than 1 nameplate rated voltage, 149 degrees F 0.2 g. Of these, both temperature and Although RCP bays are approximately does not reach inside the capacitor end capacitor porcelain housings indicate the temperature was 180 degrees Fahren vibration on the RCP motor casing is la vibration. However, measurements tak RCPs (during the fall 1986 refueling out ranging from 0.28 to 2.40. As noted in LER 86-04 for the Unit 1 to effectiveness of surge capacitors in pro- alternate system which can provide the breaker to RCP motor has been modeled to stator windings without any protect located at the RCP breaker switchgear. with an equivalent length of cabling and compare to the computer model. The m show that surge capacitors do provide so inductors are a viable alterne twe to sur- breaker switchgear is outside the conta- invironment of the containment are re- greater reliability than capacitors.	(3) Fads/fr, 70 paig ext ahrenheit ambiant tem vibration appear to be 120 degrees Fahrenheit losures. Temperature vat at one time during a heit but 190 degrees F iss than the manufactur en on the surge capacit tage) showed maximum ip on July 20, 1986, BG viding protection to wi same protection. The d by computer to show ion, with surge capacit Additionally, our span is pulse generator to e iodel and experimental one reduction in the v ge capacitors. Addition int it, the potential p noved. Finally, inducto dor. The above test da ert concurs the induct stator insulation. The	emal pressure, 10% above perature and a vibration of exceeded during operation. , contain ment cooling air iots installed on RCP surge 115 month period, the ahrenheit. The measured ar's recommended maximum. for terminal boxes for Unit 1 peak g vibration levels & E has reviewed both the nding insulation and an electrical system from the voltage surge seen by ors, and with inductors e RCP motor has been used xperimentally obtain data to data compare favorably, oltage surge, and that nally, since the RCP roblems associated with the rs have an inherently ta and computer model were	

	EPORT (LER) TEXT CONTIN	EXPLASS & J. M
ACTUARY NAME (1)	DOCAET NUMBER (2)	LER NUMBER & PASE
	and the second	TEAR BEAUTY AL BELBON
Calvert Cliffs, Unit 2	0 15 10 10 10 1 211 13	8 8 6 - 0 0 6 - 0 1 0 4 04
CET IS more grace a recurred, use additional hills: April 386.4 (c) 1171	and the second	and the second
 Each surge capacitor consists of 5 in series. Each "packet" is made of dielectric, and wrapped in two motinsulating sheath (made of a glass helium filled porcelain container with a 3 % increase in measured capter in at least two of the fifty-four surge capacitors were also related continuity was present in these surge capacitors were also related continuity as present in these surge capacitors were also related to a 1 for the loose terminal several possible causes: wibration design deficiencies, or mishandling inches long and 8 inches in diamet when handling. Additionally, look capacitor. Calvert Cliffs Unit 1 has experier October 26, 1977; June 6, 1983; as September 7, 1979 and April 15, 1 resulted from a RCP breaker oper several deficiencies in the design made by the manufacturer in their the third modification to this styl previous failures were occurring a capacitor/insulating sheath junctim odifications to reduce the possition of an arc tracking along the strip on the other side of the myl cause, a series of high temperature environment tests on a good surg capacitors, ne moved from the September throug, the dielectric motion in t	of two thin metallic foll shi re sheets of mylar. These filled polyester material) a with a metal base plate. Were checked. Three surge pacitance (from baseline de and no degradation in capa replaced. hugs on the surge capacitors are ter) and the terminal lugs p twashers or equivalent devi ind July 20, 1986. Unit 2 he 984. In each case, a low R ming due to a shorted surge of surge capacitors and se r structural design. Surge is surge capacitor. Until the at the edge of the capacitor ion. This mode of failure w hillity of abrasion to the my sit 1 trip on July 20, 1986, is e mylar dielectric from on ar dielectric. Although mo are (up to 100 degrees Celsi e capacitor were in onclus ptember 5, 1986 Unit 2 trip ackets' exterior as well as	 weets, separated by a mylar are all enclosed in an analysis was done on the statut for the insulating boards a capacitors were replaced due data). This indicated a fullure in an associated decrease in gs (EIIS E-CON). Although acitance material was found, wers is unknown. There are ontraction, manufacturing or re heavy (70 pounds), bulky (26 provide an easy surface to grab ices are not used in the surge di 2, 1976; October 24, 1977; was had similar events on Reactor Coolant Flow Trip a capacitor. BG&E has noted everal improvements have been a capacitors presently used are the July 20, 1986 Unit 1 trip, all or "packets" at the was the basis for previous invariant for the insulating boards an analysis was done on the stailure appeared to be the ne foil strip to the other foil oisture could be one possible sing) and high humidity sive. Examination of the surge in, showed evidence of both

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ACTUIT AAME IN	LICENSEE EVENT I	REPORT (LER) TEXT CONTINU	ATION		APPROVED ONE EXPRES \$ 2 1	
Cilvert Cliffs, Unit 2 0 18 10 10 10 11 18 816 - 01016 - 011 015 FAILURE DATA: Surge Capacitor Westinghouse Electric Corporation Radiation Resistant Surge Capacitor (.05 uf) Style # 634A269A02 Steam Dump Solenoid Valve Automatic Switch Company (ASCO) Solenoid Valve Model #8300064 Atmospheric Steam Dump Positioner Moore Products Co. Mode # 726315		DOCKET NUMBER (2)				**01 (3
FAILURE DATA: Westinghouse Electric Corporation Surge Capacitor Westinghouse Electric Corporation Steam Dump Solenoid Valve Automatic Switch Company (ASCO) Solenoid Valve Model #8300064 Atmospheric Steam Dump Moore Products Co. Model # 72G315 Model # 72G315			*EAR	REQUENTAL AURILE	AL-BCS NUMBER	11.
FAILURE DATA: Surge Capacitor Westinghouse Electric Corporation Radiation Resistant Surge Capacitor (.05 uf) Style # 634A269A02 Steam Dump Solenoid Valve Automatic Switch Company (ASCO) Solenoid Valve Model #8300C64 Atmospheric Steam Dump Positioner Moore Products Co. Mode # 72G315	s, Unit 2	0 5 0 0 0 311 18	816 -	01016	- 0110	12 05 0
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Radiation Resistant Surge Capacitor (.05 uf) Steam Dump Solenoid Valve Automatic Switch Company (ASCO) Solenoid Valve Model #8300064 Atmospheric Steam Dump Positioner Model #72G315	ATA:					
Atmospheric Staam Dump Positioner Moore Products Co. Mode # 72G315	itor	Kadiation Resistant Su	Corpors inge Cap	ation pacitor (.0	5 af)	
Positioner Moore Products Co. Mode # 720315	Solenoid Valve	Solenoid Valve	n pany (A	A\$CO)		
	Steam Dump	Moore Products Co. Mode # 726315				

ukç fare 184 3-4)	LICENSEE EVE	NT REPORT (LER)	UE ROCLEAR RECULATORY COMMISSION APPROVED DWY NO DISCOM ELEMAGE & D' B
ACILITY NAME IT		ana dia dari manin menangkan		T NUMBER (2) FAST 3
Fermi J			and the second se	101010[314]1 1 0F 016
HPC1/RCIC Inop He	quires Entry into	Tech Spec 3	.0.3 Caused	by
Personnel Error I	Muring RCiC System	Troubleshoo	CIUE SUM 16	LITER IRADIAED IN
EVENT DATE IS LER HUNG		spinster, and the second particular statement of the	FACILITY NAMES	DOCKET NUMBER S
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channel causes event was inade and the persons restoration of corrective acti Specification 3 counseled, and verification was instrument surv	ARTUP), operating 8 percent react hannel, two occu and High Pressui- lted. Entry ind lowing each evo- caused by a per- that the calib RCIC to become quate communica performing the the RCIC pump forms, training w .0.3, the perso required readin is provide, to 1 reillance proced	tor power. prences of to Technic ent. sonnel err ration pro- tion between test, and low control inl be pro- nnel involog related icensed of lores will training	During During f Reactor Injectio al Specif for which bedure for c. Contri ben Operat d a persor pller sety ovided on lved in th to indepu perators. be revisu	calibration of 1 Core Isolation n (HPCI) being lication 3.0.3 resulted from or the RCIC flow buting to the tions personnel anel error during point. As Technical his event were endent Additionally. ed to clarify rovided regarding
event did not	llow control cir involve any fail	ed compon	ents or s	ystems.
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CILITY BANE (1)	DOCKET No MER ()		EXPLACE 8-21-18	
	Protect and and a	TEAR SECURITI	AL MEVBION	AG1 3
Fermi 2				
It if more apr's a reputric, use additional NRC from Addition (17)	0 5 0 0 0 3 4	186-048	3 - 0 h jo p	OF OF
At 1015 hours on December CONDITION 2 (STARTUP), ope Pahrenheit, and 8 percent December 24 and 1550 hours for a Reactor Core Isolati flow instrument channel re System being inoperable. Pressure Coolant Injection In OPERATIONAL CONDITIONS SHUTDOWN), operability of steam dome pressure is abo are required, but inoperat 3.0.3, and initiation of a required. The HPCI System system outage since 1755 f During a routine performan the Remote Shutdown Panel December 24 Operations per that the RCIC header flow Shutdown Panel was reading the RCIC System was not in flow indicator located in gpm. Subsequently, a work order discrepancy, and recalibra at the RSP if necessary. investigation, at approxim fnon-licensed, utility) we a calibration surveillance Remote Shutdown Panel. Prior to initiating work o procedure was reviewed by going to perform the surve presented to the Nuclear A (NASS) (licensed, utility) procedure was reviewed by to the field for execution	26, 1986 Fermi 2 rating at 920 psi reactor power. B on December 26, ion Cooling (RCIC) esulted in two occ During these occu n (HPCI) (BJ) System 1 (POWER OPERATIO both HPCI and RCIO both HPCI and RCIO both HPCI and RCIO both System 1 (POWER OPERATIO both HPCI and RCIO both System 1 (POWER OPERATIO both HPCI and RCIO both System 1 (POWER OPERATIO both HPCI and RCIO both HPCI and RCIO both System 1 (POWER OPERATIO both HPCI and RCIO both System 1 (POWER OPERATIO both HPCI and RCIO both HPCI and RCIO both HPCI and RCIO both HPCI and RCIO both System 1 (RSP) instrumenta sonnel (non-licent indicator (FI) loo 100 gpm. This will operation at the the main control was issued to invite the RCIC pump In order to facil ately 1215 hours for the Kaintenance per illance. The proc ssistant Shift Sug for review and woil the NASS without a	was in OPERA g, 530 degre etween 1248 calibration (BN) System urrences of rrences, the m was also i N), through C is require n both HPCI chnical Spec ithin one ho ble for a sc 23, 1986. CHECK proce tion, at 023 sed, utility cated at the as not expec time, and room was rea vestigate th flow indicat itate the Maintenance eir foreman indicator a l, the surve ersonnel who cedure was t pervisor rk authoriza	TIONAL es hours on activities header the RCIC High noperable. 3 (HOT d when and RCIC ification ur is heduled dure for 0 hours on) noted Remote ted since the RCIC ding 0 e or located personnel to perform t the illance were hen	

AC Form 386A	and other states in the state of the state o		THE REAL PROPERTY CONTIN		APPROVED ONE FO 3150-2104
40	LICE	NSEE EVENT R	EPORT (LER) TEXT CONTIN	IUATION	Exercise 1 31 86
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				VEAR STOVEN'S	NUMBER
	Fermi 2		0 5 0 0 3 4	1816-01418	-0110130F0 16
		C Auros Body to 1171			
	Surveillance at approxima surveillance were found t indicated th range, and t originating associated f The Maintena were only au recalibrate work scope a any further under a more exited. The configuratio The prerequi performance to be inoper the NASS he Maintenance ensure that inoperable. procedure re simultaneous Additionally surveillance (NSS) (liceni the flow tri procedure. potentially out-of-tole communicati respond to	activities tely 1250 h the output o be out-of at the flow hat the flow hat the out in either t low loop sq nce personn thorized to the RSP flo work on the specific w RCIC flow on at this t isites for t of the proc tative. How did not tak personnel r the NASS we As a resul esulted in t sly. y, when the e they did is sed, utility ansmitter w As a resul inoperable inoperformed in ons proper this event.	involving the flow ours. During perfo readings from the -olerance. Furthe indicator was with -of-tolerance flow he RCIC flow loop t uare root signal co mel who were perform by indicator. As a on, at 1720 hours it of low loop would have ork package and the loop was restored to the calibration produces the calibration produces the ever, when the product as also aware that I lt, performance of the both HPCI and RCIC to Maintenance personn not notify the Nucli y) that either the as not functioning t, it was not recog as a result of the each case, as a res actions were not ta	t 8 6 - 0 4 8 channel were rmance of the RSP flow ind: r investigat: in the calibi- readings were ransmitter (1) nverter. ing the calib- result of the was determi- we to be per- surveillance to the as-four- co the as-four- redure state the RCIC flow redure was re- the action. Al- ion they negle RCIC would be the calibrati- being inopera- nel exited the square root of properly, as nized that RC flow loop be ult of inaded ken to prever-	- o 1 0 3 or 0 6 r initiated icator loop ion ration PT) or the pration llance and eir limited hed that for sed e was nd that controller viewed by though the ected to on ble required by llower or required by llower or llower or required by llower or llower
	converter o This was th	* the flow	ift recognized that transmitter was not the that the problem ins staff.	Inuctioning.	hrakerel.

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IRC Form ABEA HAD	LICENSEE EVENT REPOR	T (LER) TEXT CONTINU	ATION	APPROVED OWS NO 3 EXPARS & 51 IB	
ACILITY BOSHE IS	manufacture and the second state of the second state	DOCKET NUMBER 2:	LER WUNDER IS		6. 10A
			TEAR SIDUENTIAL	JAL HON	1 1
Fermi 2			N- W81 *	Number .	1.1.
1 61 164		0 5 0 0 0 3 4 1	816-01418	-011010	OF AU
N7 # mere grane & required, use at	adaman Addi Anna 2004 2/ 117)	de wederender oder oder der der oderen		1.1.1.2.1.2	1.1.4
The NSS id delivered recalibrai 1015 hours realized cause RCI immediate was entered The calible approximat room flow respective the cover on the me Based on the indicator troublesh converter various s uniformly Subsequen flow loop RCIC Syste	Mentified the proble the revised work pro- tion of the RCIC flu- s on December 26, 1 that the execution of the inoperable. In declared inoperal ed. Shutdown of the ration surveillance tely 1015 hours. A indicators were st ly At approximation of the main contro- ter the control roo this observation and had been calibrate boting of the flow was initiated. Whi imulated levels was out-of-tolerance b thy, the flow trans surveillance test em was returned to Specification 3.0.	ackage for troubl ow loop for work 986. At this tim of the surveillan Upon discovery, ble and Technical e plant was initi was re-entered i t this time the F ill reading 100 g ely this time an 1 room flow indic m indicator also d the fact that t d only two days e transmitter and t en the as-found t measured it was y approximately 1 mitter was re-cal was performed sat	eshooting an authorizatio e it was als ce procedure RCIC was Specificati ated at 1103 mmediately a SP and main pm and 0 gpm operator tap ator. After read 100 gpm the RSP flow arlier, the square ro found to be millivolt D librated, and isfactorily.	n at would on 3.0.3 hours. t control ped on tapping ot utput at C. the The	
RCIC pump was incor setpoint immediate (licensed It is bel positione ontrolle performed entered i setpoint, The incor RCIC and that the pump flow was opera required	ours on December 26 flow controller lo rectly set to 505 g is 605 gpm. The fl ly reset to 605 gpm , utility). ieved that the flow d by an operator (1 r calibration and t on December 26. T mmediately upon dis and emited upon re rect RCIC pump flow RPCI being inoperab setpoint error was setpoint error was setpoint was not d ted for approximate entry into Technica on of the requireme	cated at the main pm. The correct ow controller set by a control row controller setp- icensed, utility roubleshooting a echnical Specific covery of the in- solution. controller setp de between 1310 discovered. Sin etected until 15 hly 2.5 hours in a Specification	n control roo flow control tpoint was om operator oint was inco during the ctivities whi cation 3.0.3 correct flow oint resulted hours and the ce the incor 50 hours the a condition	orrectly flow ich were was control d in both e time rect RCIC plant which	

C Fare 386A	LICENSEE EVENT P	REPORT (LER) TI XT CONTINU	JATION APPROVED DWE NO 2150-0 EXPRES & 21 00	1.04
ILITY BANK	105	DOCK ET NUMBER (2)	LER NUMBER (8) PAGE :	3
		5. B. S. S. S. S.	· CAR SEC. AL RESOL	
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	coar & respectively, was asked survey lastic Aware 3854 to (17)	and the second s		
	The flow transmitter when the RSP also provides in the main control room a Availability of the RC1 of the proper flow consistent of the proper flow consistent.	input to the RCIC flo and the automatic RCI IC pump flow controll trol setpoint are req	<pre>w indicator located in C flow controller. er and establishment uired by Technical</pre>	
	Evaluation of the out- transmitter has demons resulted in controlled Technical Specificatio condition for the flow inoperable, or require	trated that this cond RCIC flow below the ns. As a result the	600 gpm required by out-of-tolerance cause RCIC to be	
	The incorrect RCIC pum to be inoperable since automatically providin the incorrect flow con RCIC firm being able t from the main control	p flow controller set it would have prever g the minimum require troller setpoirt woul o provide flow, or be room.	tpoint did cause RCIC nted the system from ed flowrate. However, Id not have prevented e manually controlled	
	incorrect RCIC pump fl simultaneously inopers hours) on December 24, hours) on December 26,	ow setting resulted able between 1250 hour 1986, and 1015 hour 1986. The existence into Technical Specif opropriate actions we (3.0 hours) on Decemb	ication 3.0.3 was re taken, between 1015	1
	the event were: two in involving the Mainten calibration and the N restoration of the pur testing.	libration procedure f CIC to become inopera ive personnel error. nstances of inadequat ance personnel who we ASS, and an operator mp flow controller se	or the Kit How ble. The event was Contributing causes to e communications re performing the error involving etpoint following	
	during this event, en	uring the event. If try into Technical Sp d. This event did no components, structure	pecification 3.0.3 would	

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 As corrective action, the following actions will be or taken: 1) Licensed operators will receive supplements on the requirements of Technical Specificat 2) The Operations and Maintenance personnel in this event have been counseled, and require on performing independent verification has provided for licensed operators. 3) Instrument surveillance procedures (44.XXX, will be revised under a procedures improven program to include clearer statements regard operations impact. 4) Training will be provided regarding the sign of the RCIC pump flow control circuit interbetween HPCI and RCIC, and the control circuit effects on system operablity. Completion of corrective actions listed in items 1 and scheduled to be completed by June 1, 1987. The correct actions listed in item 3 are scheduled to be completed by June 1, 1987. The correct actions listed in item 3 are scheduled to be completed by June 1, 1987. The correct actions listed in item 3 are scheduled to be completed by June 1, 1987. The correct actions listed in item 3 are scheduled to be completed by June 1, 1987. The correct actions listed in item 3 are scheduled to be completed by June 1, 1987. The correct actions listed in item 3 are scheduled to be completed by June 1, 1987. The correct actions listed in item 3 are scheduled to be completed by June 1, 1987. The correct actions listed in item 3 are scheduled to be completed by June 1, 1987. The correct actions listed in item 3 are scheduled to be completed by June 1, 1987. The correct actions listed in item 3 are scheduled to be completed by June 1, 1987. The correct actions listed in item 3 are scheduled to be completed by June 1, 1987. The correct actions listed in item 3 are scheduled to be completed by June 1, 1987. The correct actions listed in item 3 are scheduled to be completed by June 1, 1987. 	18 0 1 0 6 0* 0
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function associated with these systems, the Standby Fee	dwater
System, the Automatic Depressurization System (ADS) and	the
ability to manually control the RCIC pump were svailabl continuously throughout this event. As a result, this	event did
not affect the safe operation of the plant, or the safe	
public.	
Events which required entry into Technical Specification	n 3.0.3
because of concurrent inoperability of RPCI and RCIC ha	
reported in Licensee Event Reports 86-037, and 85-038.	

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the Train B Component Cooling Water (CC (EIIS Component Code HX) decreased belo would support removal of the postulated CCW loop). This decrease in flow was d and debris. In accordance with operati CCWHX was reversed using recently insta such flow restrictions. With reverse S CCWHX is reduced, but remains sufficien is, therefore, considered operable purs time, there was minor fouling of the Tr sufficient to remove the critical CCW h	whe flow rate at design basis accidue to unusually highly ing procedures the s liled provisions to SWC flow, the heat r it for removal of cr uant to Technical s ain A CCWNY, as we	which the heat excha sent heat load (criti an deposits of marine SWC flow through the enable Operators to removal capability of ritical loop heat loa Specifications. At t	nger cal growth Train B remove the ds and his
operable. At 0201, on August 4, 1986, surveillanc (LPSI) valve was commenced, which resul inoperable status. As this surveillanc the Train A SWC flow was monitored by o marine fouling would not reduce the SWC operability prior to completion of the Train B SWC from reverse flow to its no valve test on Train A LPSI, thereby avo (LPSI) inoperable condition. However, than anticipated.	ted in placing the e progressed, durin perations and it wo flow below the rat LPSI valve test. rmal configuration idino a simultaneo	Train A LPS1 in an ng graveyard and day as anticipated that t te required for CCWHX It was intended to re after completion of us Train B (CCW) and	shift, he turn the the Train A
At 1550 on August 4, 1986, increased fo through the Train A CCWHX to a level wh incapable of removing post accident (cr therefore declared inoperable pursuant	ere the heat exchan- itical CCW loop) co	nger would have been omponent heat loads.	
At 1605, operators commenced realignmen direction in order to return one train increase heat removal capability of tha of the SWC system were considered to be Limiting Condition for Operation (LCO) restored to the Train B CCWHX, in the n LCO 3.0.3 was exited.	of CCW to its design t train. During the inoperable contra 3.7.4. and LCO 3.0	gn configuration and he realignment, both ry to Technical Speci 3 was entered Sur	thereby trains fication
Before proceeding to re-align Train B S Train A CCWHX capability by reversing S thermal transient in the CCW system, wh	WC flow, However,	this would have crea	ng ted a

Train A CCWHX capability by reversing SWC flow. However, this would have created a thermal transient in the CCW system, which, in turn, would have accelerated degradation of Reactor Coolant Pump (RCP) seals. Operators had also considered transferring the RCP seal and other non-critical loop CCW heat loads to the Train B CCWHX before reversing the flow in the Train A SWC system. This would have resulted in defouling the Train A CCWHX prior to realigning Train 8 SWC system. Operating procedures, however, did not provide for transferring of such heat loads to a CCWHX operating with reverse SWC flow.

NAC Form 3664

N.R.C. Form 348.A (9.83)	LICENSEE EVENT REPORT TEXT CONTINUATION	LER)		APER		NO. 21500104	
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SCE has recognized the significant impact marine fouling of the CCWHX can have on plant operation. As a result, the capability to reverse the SWC system flow was provided by a recently completed plant design change. This has yielded considerable operational flexibility resulting in a substantial increase in availability of the SWC system.

The following corrective actions will be taken:

- Operating proclaures will be revised to provide for transferring non-critical loads to a CCWHX with reverse flowing SWC;
- RCP seal design change, already completed on Unit 2, will be completed on Unit 3 during the next refueling outage. This new seal arrangement is less sensitive to thermal transients and permits reversal of SWC flow in CCWHXs without detrimental effect on the seals.

Neither the health and safety of plant personnel nor the health and safety of the public was affected by this event.

NRC Form 365A

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YEL IT OF THE PARTY IS DURATION OF Y Yel Yel IT on the presence is the event when the result of the the transmission of the thermal power and the mode switch was in the refuel position. At that time, plant personnel found, while performing a plant procedure, that both loops of core spray (CS) were incapable of performing their intended (unction). The foot cause of the event was personnel error in developing a clearance and the integrated Leak Rate Test (LERT) procedure. The plant is closed and power Demoved from the pumps. Corrective actions include: 1) returning the CS system to operable status, 2) performing a management critique of the defective clearance and procedure, 3) making personnel aware of the event and steps leading to it, and 4) incorporating the event into the operator training program. The last two actions will be performed in conjunction with the corrective actions for LER 50-321/1986-036.	BURGLEMENTAL REPORT EXPECTED IVA	1X752780
<pre>AAATAACT (where a new order and new own the on 11/13/86 at approximately 2319 CDT, Unit 2 was at 0 percent of rated thermal power and the mode switch was in the refuel position. At that time, plant personnel found, while performing a plant procedure, that both loops of core spray (CS) were incapable of performing their intended function. The root cause of the event was personnel error in developing a clearance and the Integrated Leak Rate Test (ILRT) procedure. The plant ILRT procedure called for both loops of CS to have the suction valves closed and power removed from the pumps. Corrective actions include: 1) returning the CS system to operable status, 2) performing a management critique of the defective clearance and procedure, 3) making personnel aware of the event and steps leading to it, and 4) incorporating the event into the operator training program. The last two actions will be performed in conjunction with the corrective actions for LER 50-321/1966-036.</pre>	THE IS A DESCRIPTION FOR THE SAME AND A DESCRIPTION OF THE SAME AN	But Stor
Belargoger Belals PDR ADOCK OSODOgee S Y	<pre>on 11/13/86 at approximately 2319 CpT, Unit 2 was at 0 thermal power and the mode switch was in the refuel po time, plant personnel found, while performing a plant both loops of core spray (CS) were incapable of perfor intended function. The root cause of the event was personnel error in dev clearance and the Integrated Leak Rate Test (1LRT) pro ILRT procedure called for both loops of CS to have the closed and power removed from the pumps. Corrective actions include: 1) returning the CS syste status, 2) performing a management critique of the def and procedure, 3) making personnel aware of the event to it, and 4) incorporating the event into the operato program. The last two actions will be performed in co </pre>	sition. At that procedure, that ming their eloping a wedure. The plant suction valves m to operable ective clearance and steps leading or training mjunction with the
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1.80 Farm 3864 (F-80)	LICENSEE EVENT REP	ORT (LER) TEXT CONTINU	ATIO	N		2.8			NULATON			1			
	and the state of the second state of a second state of the						LER BURNET IS				FAGE IS				
			-44.8		News	CHRIST PRIVACE			1						
	ATCH, UNIT 2	0 5 0 0 0 3 6 6	8, 6		0 1 3	15	_	010	012	OF	0	1			
	reparted, we address with Asm Milet at 157				-				1	-		here			
	REQUIREMENT FOR REPORT														
A+	the second se														
	This report is required p	er 10 CFR 50.73 (a)(2)	$(\mathbf{x})_{i}$	be	caus	e b	ot	1							
	trains of the Core Spray their safety function and	(CS) system were incas	00010	10	per	201	m'r.	19							
	condition caused 2 indepe	ndent trains to become	a ino	peri	able	in									
	single system designed to	remove residual heat	and	mai	ntai	n s	ne.								
	reactor in a safe shutdow	n condition.													
8.	UNIT(8) STATUS AT TIME OF	EV ENT													
	A REAL PROPERTY AND A REAL PROPERTY A REAL PROPERTY AND A REAL PRO														
	Unit 2 was at an approxim thermal power. The react	ate power level of 0 ;	perce	nt.	011	ate	10								
	position. One loop of Re	sidual Reat Removal ()	RHR)	SYS	tem	was									
	tagged out for maintenand	e and the other loop	was i	n L	ne s	hut	t do	with i							
	cooling (SDC) mode of ope	ration.													
с.	DESCRIPTION OF EVENT														
	on 11/13/86 at approximat	ely 2319 CST, during	the p	erf	orma	ner	8 0	£							
	*Emergency Core Cooling S	ystems operability St	atus.	che	CKS.										
	procedure 3400-OPS-033-25 loops of the Core Spray (on Unit 2, plant per	8000.04 	1 1	ound	1 - 21	950								
	inoperability occurred be	cause both the suctio	n val	ves	(01	ie i	tor								
	each subsystem) were clos	ed and the electrical	powe	1 1	07 1	oti	h q	£							
	the CS pumps (one pump in	each subsystem; was	remov	ęd.											
	An investigation of the e	went showed that plan	t per	son	nel	¥9	te.								
	aligning plant systems in	order to perform the	1054	1924	ted	1.0	āk-	Rate							
	Test (ILRT). The system	alignment occurred so	metin v ela	ie a	IT THE	-	445	CST							
	on 11/13/86. Plant perso accordance with a complet	annel were manipulation	e she	ins iet.	6401 (F)	i por	¢	***							
	clearance sheet was used	because plant personn	el wa	nte	d to	s p	làs	e							
	hold taos on selected pla	int equipment to preve	nt cl	ang	ing	27	e								
	condition of the equipment	it while the ILRT was	in p	1091	ess.	5									
	One of the documents used	to develop the clear	ance	*85	the	e 1	LRS								
	procedure. This procedut	e, however, contained	an s	1111	i ti	hāt									
	resulted in both subsyste	ms of the CS system b	eing	100	peri	abl	6 8	8							
	described above. At the	time of the event, CS	was	sec.	ULT:	e0 	20	De :							
	operable per Technical Sp vessel head was in place	and being tensioned.													
	and the second second second	and the second se													

1.121.11	LICENSEE EVI	ert ner ent teert ti	EXT CONTINU	UATION		APPROVED DI EXPIRES 8/01		S(-5)	64
ANDILITY NAME IT		DOCK ET MUM	HA 2-				+ 61 - 3	-	
				184.8	REDURNTAL	NEVERS		Π	
	HATCH, UNIT 2	0 5 0	0 0 3 6 6	8 6	-0 1315	- 010	013	OF	0.1
"EX" of many space of	t Televiter, and additioned Arth' Agent 2014 (1)	71						deres de	-
	The Unit 2 Technic with both CS subsy that at least one is operable and bo Otherwise, suspend draining the react subsystem is opera SDC mode of operat Out of the SDC mod were a LPCI initia	stems inoperable, Low Pressure Cools th LPCI subsystem all operations to or vessel and ver- ble within 4 hours ion, the RHS syste e to align itself	operation i ant Injection s are operal hat have a p ify that at s. With this em can not in to the LPC	may co on (LP ble wi potent least e RHR automa	ntinue pro CI) subsys thin 4 hou ial for one LPCI system in tically or	the			
	The event was disc documented on a de were no activities the potential for trains of CS were returning both sub statement of the L satisfied.	overed at 2319 cs ficiency report at in progress, or ; draining the react returned to servit systems of cs to ;	T on 11/13/0 t 0100 CST of planned in t tor pressur- ce on 11/14, operable st	on 11/ the pl e vess /86 at atus,	14/86. Th ant, that el. Both 0232 CST. the action	had By			
D.	CAUSE OF EVENT								
	The intermediate c and implementation contributing facto follows: length of restricted time so staffing insufficie	of clearance she ta leading to this f the clearance s hedule to draft, i	et number 2 a personnel heet (3,000 review and	-86-17 error to 3, implem	58, Major are as 500 items ent, and				
	The root cause was Unit 2 ILRT proced old procedure to 1 made to the new pro procedure (that red removal of supply 1 remained in the new a basis for the cle the clearance and	ure, Then when an te present format ocedure. However, quired the closur- power to both CS ; w procedure, when eatance, the techn	ngineering ; , numerous , the techn e of the su pumps) was n the new p nical error	person improv ical e ction not di rocedu was t	nel change ements wei rror in ti valves and scovered i re was use ransmitter	ed the re ne old i ind ed as			
	It is to be noted : Nowever, the proces Opgrade Program (P)	dure had not been	through the	its fo e full	rmat chang Procedure	red.			

			II CONT	NU.	ATIO	N		APPROVED EXPIRES 8		1.10-2	24	
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-	damon open b	0 (6 (0)0		14	8.6	- 0	1315	- 0.0	014	OF	5.1	
And the second second second second	HATCH, UNIT 2 a required and additional Auto Autor Autor at 1070	10 10 0 0	Is to to	- 1	9 -	-12	1-1-	I-I-I-	1-1-	10.1	- 1	
1.00	LUCIUMIA AN AVAUR											
Ε.	ANALYSIS OF EVENT											
γ.	The basis for requiring mode of operation is to event occur which could time of this event, ther had the potential for dr Additionally, there are in the refueling mode, w system could not automat the LPCI mode, plant ope system well within the 4 required. Other circums significance of this eve reactor was minimal beca new unirradiated fuel, a Based on the above infor no impact on safety. CORRECTIVE ACTIONS On identification of thi returned to service on 1 All of the LERT procedur until completion of a ma on 11/14/86 at approxima LERT procedure and the c CS system were reviewed Specifications. No addi Plant personnel performe LERT procedure which bro rechnical Specifications also corrected to reflec Engineering pelsonnel ar to it. Engineering mana involved in procedure re current Technical Specifi	provide inven drain the rea e were no on- aining the re no postulated hich could dr ically come o rators could hour LCO act tances which nt are: 1) use one-third nd 2) all con mation, it is s incident, b 1/14/86 at ap e and clearan nagement crit tely 0630 CST learance that against the r tional discre d a temporary upt the proc requirements t the newly c	tory (w ctor pr going p actor p actor p actor p actor p actor p is conclu- of the trol ro- conclu- oth Cor proxima ce acti- ique, . Duri caused edure i . The or ceet et conclu-	action hnee a d d et vingt e undd e	<pre>c) mi ure v sure v sure s sure s sure s sol v rame s the set the set the set the set the set the set the set the set the set of re n mod ecti cet i re v set the set set the set the set set the set set the set set the set set set set set set set set set se</pre>	skeuses versives vers	p shoup sl. n) ties i sel. i ned ties ned ties n	ald an ar the which 2 PSAR, the RHR I been the ed of serted. ent had the f the nical to the ith the nce was cading nnel				

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	a manurat are antifered tothe Auror Miles arith		
G.	to it. Operations be for personnel involve detailed reviews when components. These pe assistance if needed electrical circuits. completed in conjunct 50-321/1986-036 and 5 scheduled to be lompl This event and associ Training pepartment t initial and requalifi completed on approxim ADDITIONAL INFORMATIO 1. FAILED CO NO 2. PREVIOUS Pre- ing cor The and LES pid rei		emphasize the need ation process to do wirances on plant o to get further of isolating these is LER will be ction in LER's rective action is 9/87. provided to the ing experience into acheduled to be wents where occurred in tance of a clearance. 5-036 (dated 10/20/86) 11/5/86). ed an event where form an adequate . The inadequate ive clearance, that ations of ESFS. ed an event where form an adequate and an inadvertent

1 **** 200.4	LICENSEE I	VENT REPORT (LER) TEXT C	ONTINUATION			
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DWIN I. HATCH	UNIT 2	0 6 0 0 0	3 16 6 8 6 -	-b 1312		
NT of many special & reported	an and server tothe Autor Miller	w 1179				
		The corrective actions included: 1) securing counseling of personne personnel of th events the events, and 4) ind into the operator train numbers 3 and 4 were to actions to prevent rad events. These correct for completion on app 5/19/87 respectively.	affected by 1, 3) inform and the con- corporating t ning program the long term currence of t ve actions	ing other sequences he experis. Item correct hase typ are sche	of ences tve es of duled	

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*4)		U	CENSEE EVI	ENT RI	PORT	(LER)		MOLEAN REDUC	NO STRENDION
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TES IN UN COMPANY EXPECTS	Putering and the		-				8.448078 8.4940 8840 0.478 118	D BRORY	0.4V VE
On March 29, (NV) charging had been rack Technically, charging pump approximately Immediately u NV charging p Unit 2 was in This incident Assistant Shi have availab). There wer NV chargin,	1986, at 0 pump 25 b ed in serv this result as requir twenty ho pon discov ump 28 was Mode 6, P has ft cor cor co	rery, N' rery, N'	was disco recked on hout havin no operat echnir echnir G. M. . et . et . et		344	 while cy power low pate low pate<td>use the s ging pump</td><td>ing pump rable NV (1.2.3 f ervice, responsi p 2A did</td><td>2A or and ble not</td>	use the s ging pump	ing pump rable NV (1.2.3 f ervice, responsi p 2A did	2A or and ble not
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LICENSEE EVENT REPO	ORT (LER) TEXT CONTINU	UATIO	N			HOVED D		180-0104
	DOCKET NUMBER (2)							4.61 IF
ACILITY MANES (1)		16.64	1	REAL PLANE	•	NE-BON NUMPLE		
and the second sec	0 15 10 10 10 13 17 10	8.6		01016	-	90	0 12	01 0
McGuire Nuclear Station - Unit 2	10 12 10 10 10 1-1 1-	-			-			pakanan di samuni
On March '9, 1986, at 0700, it (NV) [EIIS:CB] charging pump 2A had been racked in ser- trachnically, this resulted in y charging pump as required by Ta approximately twenty hours. Immediately upon discovery, NV NV charging pump 2B vas racked Unit 2 was in Mode 6, Refuelin BACKGROUND: The NV system is designed to P (NC) [EIIS:CB] system. One se absorber (boric acid) concentr conditions, one NV charging pu (VCT) which uses an automatic In an event requiring emergence suction from the VCT or the Ke tank can also be aligned to su (VCT) which uses an automatic In an event requiring a loss operate automatically as part take suction from the RWST and T.S. 3.1.2.1 and 3.1.2.3 speci be operable and capable of bas in Modes 5 and 6. Without an from an emergency power sources positive resctivity changes bas in Modes 5 and 6. Without an insient in the NC system vitef valve. 	b had been racked out vice without having an no operable boration f echnical Specification charging pump 2A was in service. g, at the time of this provide several service revice is to control to ration and makeup. Du mp takes suction from makeup system to main ty boration and makeup efuling Water Storage upply borated water to s of coolant accident, of the Safety Injecti d injects borated wate ify a minimum of one b ing powered from an en operable VV charging e. ofer involvin d d. T. per and service is racked out. This 3.2. To restore the teat Removal (ND) -IT ND system during a	or see a emery flow pr flow pr as and racke s inci es to he sol tring to be sol trank be to the sol trank be to be to the sol trank be to be to the sol trank be to the sol trank	d on den the up	y pow or ogi i and ut of t. Reac e che al op me Co borat char ST). ion harg ISI: iC or '''''''''''''''''''''''''''''''''''	torication of a set o	Coola ing il Tar ater pump i Systi tem. pump systi tem. systi systi tem. systi syst	y, V for and int int int int int int int int int int	1. ike id od ered of ly ure t eat

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ACILITY NAME (1)	DOCKET NUMBER (2)	ExP.#.83 4-21 45	
		LER MUNABER (8)	FAGE IP
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Manager and the second s		NUMBER PLEASES	1.4
McGuire Nuclear Station - Unit	2 0 15 0 0 0 0 1 3 1	70 36-01016-000	i ne i
AT IR many space a required can addressing ball (April 2006 2117)	and the design of the design o	1101 a 61 1 01016 [-] a 010	31-101
		,	
DESCRIPTION OF EVENT:			
diesel generator 2A could 1986, at approximately 110 Assistant Shift Supervisor retest could be performed without realizing diesel g Supervisor instructed Stat in NV charging pump 2A. T generator 2A was inoperabl result in a loss of borati 3.1.2.1 and 3.1.2.3. The	not provide emergency p NO, a Station Engineer p to rack in and operate on NV charging pump 2A. enerator 2A was inopera- ion personnel to rack o he Assistant Shift Super e and that racking out on flowpath as defined Assistant Shift Supervi Logbook (TSAIL) entry o nior Reactor Operator (were met. harging pump 2A was com	t NV charging pump 2A so a The Engineer made the requible. The Assistant Shift out NV charging pump 2B and r. rvisor did not realize diese NV charging pump 2B would by Technical Specifications sor did not make a Technical r discuss the change with the SRO) to ensure Technical pleted.	28, est ack l
2A had been racked in servi charging pump 2A was racked	ice without an emergency	, station personnel discovere t of service, NV charging pum y power supply. Immediately ump 28 was racked in stions.	nd NV
2A had been racked in servi	ice without an emergency	y power supply. Immediately	ed ap NV
2A had been racked in servi charging pump 2A was racked	ice without an emergency	y power supply. Immediately	ed BP NV
2A had been racked in servi charging pump 2A was racked reestablishing compliance . CONCLUSION: According to Duke Power Man	ice without an emergenc; fout and NV charging pr with Technical Specifica	y power supply. Immediately ump 28 was racked in ations.	ed sp NV
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On July 9, w taken out of was started frequency w D/G was sto determinatio were comple Specification to vibration lossen. The drive beit fl failure of th immediate in Lucie diesel make every	chile Unit 2 service due to satisfy th ithin the record poped due to n was made ted and the time limit. within the root cause of apping. A construction topection of generators a attempt to	was at full power, to failure to meet e redundant D/G o quired start time. o observation of co to declare the 22 28 D/G set was re The intermediate cooling unit, ther f the vibration wit continuing eifort i was caused by ove torque values of to was made. D/G m contact the vendo iscussed in the tec	Its require operability. Upon comone of its b D/G out eturned to cause of the eby, causing thin the coxis anderwa ertightening the friction maintenance	d start time The 28 D/ opleting the cooling fan of service. operable st he 28 D/G of the set bling units i y to correct g of a fric clutch lock personnel iarly in the	esel Gene e. The rei G came up e surveilla is rubbing Repairs atus withi cooling far screws in is associat it the belt tion cluto muts in th	rator (D/G dundant 2B p to voltag nde run tr its shroud on the 2B n the Tech h event wa the fan hi ed with th t flapping. h locknut. e remainir) was D/G e and te 2B t. A D/G inical s due ub to e fan The An ig St.

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EVENT DESCRIPTION: At 0854 hours on July 9, 198	6, St. Lucie Uni	t 2 was oper	rating a	t 100%.	The U	nit rem	ained
at 100% throughout the even At 0855 hours on July 9, 15 started for a once par sever valid failures within the las generator voltage and freque the start signal. An alarm D/G set had failed to start. temperature between the power unit consists of two (645E4, driving one (1) Elec- forming a diesel-generator i 0.856 hours.	986, the ZA Emi (7) days surveil st 100 valid star wency of \$160.45 was recieved with The engine fail turbo charger e (2) " WD diesel e c Products g L mbly. The 2	ergency Die llance test l ts). The 2 to volts and uich indicate to start ala xhausts of mgines, a 1 enerator co A D/G was	A D/G A D/G 60±1.2 ed that the enj 2 cylind pupled a mauall	tailed t Hz wit one of t ctuated gines in er-645E with EM y trippe	(D/G) (rs based o meet hin 10 the engine by hig the D the D tand d by the	EILS:EX i on thr the re- seconds mes in t h differ /G set. a 16 cy em cou we operio	() was eee (3) quired i after the 2A rential . The linder- plings, ator at
At 0915 hours on July 9, Specification ACTION (a) performance of Surveilland within one (1) hour and at 1 performed since the 2A T therefore, meeting the Surv was stopped due to an oper rubbing the cooling fan shru	ce Requirement east once per ei D/G came up 1 veillance Requir ator observation oud.	5.3.1.1.2a.4 ght (\$) hour to voltage ement; 4.3. tof one (1)	(redun rs there frequen 1.1.2a.4 of the 1	dant D after. cy with . At 0 2-cylin	/G oper This su hin ten 917 hou der coo	rveillan (10) suirs the ling fan	check) ice was econds, 28 D/G i blades
A decision was made to ta rub. In accordance with A was verified and immedia Generators.	ate actions wer	e taken to	repair	both t	he 2A	and 28	Diesel
The 2B D/G rub was deter the 12-cylir '* engine coo met the required start tir With the 2B D/G back in so	me. The 2B D/ ervice ACTION	G was decl (a) of LCO	ared bi 3.8.1.1	was ma	untaine	at 105 d.	9 hours.
Trouble shooting of the Woodward governor. The at 2010 hours on July 9, 19	brotem and co	ed a proble prected and	m in the 2A	D/G w	hanical vas retu	portion arned to	service

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CAUSE OF EVENT:

28 Diesel Generator Set

The intermediate cause of the 2B D/G 12-cylinder engine cooling fan event was vibration within the 12-cylinder engine Vertical Cooler Unit (ES-165), thereby, causing the set screws in the fan hub to loosen. With the loosening of the fan hub set screws, the fan shifted and began rubbing the shroud. The root cause of the vibration within the Cooler Unit is a design problem associated with fan drive belt flapping. A continuing engineering effort is underway to correct the belt flapping and thus reduce the resulting vibration.

2A Diesel Generator Set

The intermediate cause of the 2A2 diesel failing to start was the failure of a roll pin (Ref. No. \$2340-44) in the mechanical section of the Woodward EGB-13P engine governor. This governor consists of an electrical section which operates at or near rated engine speed, and a mechanical section which is mainly used during engine startup and shutdown. During startup, a small speed setting motor is used to run up the mechanical governor to allow the engine to reach rated speed where the electrical governor assumes control of engine speed. This speed setting motor operates on the linkage of the mechanical governor by a friction clutch.

Investigations revealed a roll pin which holds the intermediate gear on the pinion shaft of the speed setting controls had broken. This gear arrangement drives the dial stop gear which actuates the upper and lower stops of the speed setting motor. With the failure of the roll pin the speed setting motor caused excessive wear on the friction clutch which, in turn, allowed excessive slippage of the friction drive shaft and prevented the mechanical governor from demanding sufficient fuel flow to pick up load on the 12-cylinder engine and allowing the electric portion of the governor from taking control at the designated engine speed.

The root cause of the roll pin failure was the result of friction clutch adjustments made on the 2A 12-cylinder D/G mechanical governor as described in LER 389-86-006 (see previous similar events section). The root cause of LER 389-86-006 was determined to be a loose locknut in the friction clutch. This allowed excessive slippage and prevented the mechanical governor from demanding sufficient fuel flow to pick up load on the 2A 12cylinder engine. The corrective action was to tighten the loose locknut on the clutch. The friction clutches are supplied as assembled units and are not required to be disassembled and inspected as part of the vendor's recommended preventative maintenance program.

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Previous to tightening the loose locknut, a review of the technical manual was completed to determine the torque value for the locknut. A torque value was not supplied in the technical manual. A self-determined adjustment was made and the engine was retested with positive results. Upon later conversations with the vendor it was learned that the locknut and the clutch of the 2A 12-cylinder D/G mechanical governor had been tightened beyond the vendor prescribed torque value. The overtightened locknut provided the stress necessary for the roll pin in the speed setting control to break. Thus, the root cause of this component failure was a cognitive personnel error by utility maintenance personnel.

EVENT ANALYSIS

The event is reportable under 10 CFR 50.73(aX2Xv) as neither diesel generator set was operable between the time the 2A D/G failed and the 2B D/G was returned to service. This condition is allowed for a period not to exceed two (2) hours by LCO 3.3.1.1, provided both offsite power sources are available. Both offsite power sources were operable throughout this event and the time both D/G sets were out of service was less than two (2) hours ' your & minutes). Also, as per Surveillance Requirement 4.3.1.1.3, all diesel (2) hours ' your & minutes). Also, shall be reported to the Commission pursuant to Specification 6.9.1.

The 2A D/G governor component failure was readily detected during routine surveillance testing. The event was determined to be a valid failure in accordance with Regulatory Guide 1.108.

The 28 D/G 12-cylinder cooling fan event was observed while satisfying ACTION (a) of LCO 3.3.1.1. The effect of the cooling fan rubbing the shroud did not inhibit the 28 D/G set from coming up to voltage and frequency within ten (10) seconds. During troubleshooting, it was determined that had it been necessary for the 28 D/G to perform its saftey function the 12-cylinder cooling fan would have worn the point of contact on the shroud to where no further rubbing would have occured. The 28 D/G was taken out of service strictly as a precautionary measure and based on the above observation it was determined that the event would not be considered a valid failure per Regulatory Guide 1.108.

In the unlikely event of a complete loss of AC power (onsite and offsite) for St. Lucie 2 and, for the benefit of a conservative analysis, the simultaneous loss of offsite power and one diesel generator at St. Lucie 1, the remaining diesel generator in St. Lucie 1 is able to operate the minimum safeguard loads such that both Units are maintained in a safe, hot stand-by condition. The present St. Lucie design does have the capability of electrically connecting the two units (Reference: St. Lucie 2 UFSAR, Updzted Final Safety Analysis Report, Section \$.3.1.1.2, Pg.8.3-19d).

This was the fourth valid failure in the last 100 valid tests. Thus, the current surveillance interval is once per (3) days. This surveillance interval is in conformance with the schedule of regulatory position c.2.d of Regulatory Guide 1.103.

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CORRECTIVE ACTIONS			
2B D/G SET			
The 2/B D/G cooling fan shroud to provide sufficient clearance Upon completion of the re-po tightened. Corrective actions re	sitioning the fac ba	e tips and the	ioned in order e fan shroud. were securely
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2A D/C SET			
The 2A2 D/G governor roll pin w total; this included both the roll stop cams had to be made and a Corrective actions resulting from	test the the triction of		
A. An immediate inspection friction clutch locknuts, in	of torque values of the remaining St. Luci	the like com e Plant D/G's v	ponent, i.e., was made.
 D/G maintenance personn contact the appropriate ve particulary, in the area we in the technical manual. 	el have been instructed	to make every	y attempt to
ADDITIONAL INFORMATION			
FALLED COMPONENT INFORMA	TION		
The failure of each D/G set was Model EGB-13P. The roll pin (t \$2340-44. The 2B D/G cooling f designed by the O&M Manufacture	an is part of an ES IC		

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PREVIOUS SIMILAR EVENTS LER 389-86-6 reported a pr simultaneously out of service f On March 10, 1986, the 28 E service to repair an idler pulle March 12, while performing a D/G, one of the two engines in the 2B D/G were completed ar time limit allowed by the app idler pulley is believed to be observed on the 12 Cylinder en caused by a loose locknut in mechnical governor used for both diesels and inspect the re failures. The friction clutche during the Unit 2 refueling out SUPPLEMENTAL REPORT Upon completion of the engin set cooling units, a supplemen submitted.	or the following related of mergency Diesel Genera ey wheel on the belt-driv required operability surve in the diesel genera tor set ind the unit was returned to dicable Technical Specifi related to the belt 'llapp ngines in the D/G set. This is the friction clutch as engine start-up. Correct emaining idler pulley whe is on the remaining engine tage, April 1986.	tor (D/G) was taken out of ren engine cooling fan. On sillance on the redundant 2A failed to start. Repairs on to operable status within the cation. The damage to the ing problem which has been e failure of the 2A D/G was sembly which operates the tive actions were to repair rels on the diesels for similar the governors were inspected

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RC Form 384 420		LICENSEE EVE	NT REPORT	(LER)		LEAR REGLICATORY COMMUNES MNOV 8 CMM NO. 2180-2104 MRE- 6/21.86
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Posses 38.4826		BMine	_	M.73N/CDOH	L	78,7184
THE 01418 B		Budato	-	M./Reiches	-	78.71 int
And and the same		X M./Mardbia	-	H./Reconstru	in	STREET Constitute in Adversed
H. Car		M.78withit		8.78+0+6-0		
R. alter	HT Her	BL736H2H08H	HOM THME LER ITED	B.7Ber(2Na)		
4		LALEMAN CONTACT	HOM THE LER (12)			TUPHONE NUMBER
					LABA 3000	
Roger H. Quelle	tte - Assoc	late Engineer	. Licensing	2	1710161	3 1 7 1 3 1 -1 7 1 51 3
		HE FOR LACK COMPONEN	T FAILURE DESCRIM	ED IN THIS REPORT	And in case of the local division of the	
UNI PYETEN COMPONENT IN		and the second	SAURE EVEREN	COMPONENT	BANK FAC	TO NUMBER
		100 - 17 Santa				
	11	C. Y. P. ST.		1.1.1	111	1988 A
111111		PAR 2-15				1 45.42.00
	BURN, BOORT AL			1111	111	105.853(191
					L L L	NONTH DAY Y
THE OF SEL AND AND A CONSCIENCE AND	NUMBER OF THE	NEPORT EXPECTED INA			L I I Expertence Date ing	NONTH DAY Y
On June 13, 1986, resulted in excess loss of power to a Variable Letdown of This incident is a Construction/Inst Letdown Orifice for	at approx sive Reacts a Motor Cor Drifice. 1 at, a 1 in owel wt the assigned Ca allation de	imately 1500 b or Coolant Sys atrol Canter of The orifice fa ch pipe weld o a time of this ause Code B, D aficiency. A	ten leakag sused a lou iled in the incident. weign, Man weld on the	<pre>e causing ss of cont e opened p harge of t ufacturing e outlet f</pre>	n Letdown a forced rol power caition. the orific	piping sbutdown. A to the During the s broke. The

RC Awa 3864	LICENSEE EVENT REP	ORT (LER) TEXT CONTIN		NUCLEAR REGULATORY COMMINS
ACTLITY NAME (1)		DOCKET NUMBER (2)		8 FAGE (3
		1.1	YEAR BEQUERT AN SURGES	
Ca	tawba Nuclear Station, Uni	LE 10 5 0 0 4 1	3 8 6 - 0 3 1	- 0 2 0 2 0 0
	united, use antideamer light: Agrie 3864 (2/17)			
BACKGRO	CND			
maintai in the chargil chosen combin pressu lines variab the no mainta during Techni to l g	the functions of the Chemins a programmed water level Reactor Coolant (NC) Syst and letdown, which is a sutcomstically controlled to suit various plant open ation of letdown orifices. The and control the flow of incorporate fixed letdown le orifice. An alternate traal letdown path is inopen in normal heatup rate of the heatup. The Specification (Tech Sp pen of unidentified leakage akage rate must be reduced andby within the next 6 ho	<pre>1 in the pressuries em (EIIS:AB). This i continuous feed and by Pressuriesr level rational requirements Three parallel line reactor coolant leav orifices. The third (excess) letdown path rable. The excess le he unit, by providing ec) 3.4.5.2 states th . With unidentified to within limits with</pre>	and a required by may bleed process. . The letdown in by selecting the same provided in ring the NC System line utilizes a is provided in tdown can also g additional let hat NC leakage s leakage greater thin 4 hours or	hans of The charging rate can be to reduce the em. Two of the valve as a the event that be used to down capability hall be limited than 1 gpm, be in at least
Contraction of the local distance of the loc	PTICH OF INCIDENT		en. Sidded	
was in to Com	e 13, 1986, the Dnit was a service and throttled to ponent Cooling System (EII orifices flowpaths were is	approximately 30 gpm (S:CC) leak at the Le	CO LEGNCE htess	101 C 010 F
uniden due to hours, This r Temper respec Pressu leak. temper Pressu also c alara	O hours, the Unit entered tified NC laskage of 1.486 unidentified NC laskage of alarms were received indi esulted in the loss of con ature control valve, causi tively. Charging flow su rizer lavel began decrease At 1550:11 hours, an alas ature on the Generator. rizer lavel continued to continued to increase. The was received at 1601:13 he	5 gpm. At 1500 hours of greater than 1.0 g icating the loss of M ntrol power to 187849 ing the valves to fai ing. This gave the i rm was received indic At 1551:02 hours, 187 decrease. The hydrog a Containment Floor a ours, confirming NC 1	, sh obustai iv pu. At approximi- lotor Control Cer and IRL138, By 1 open and close proximately 13 indication of a ating an increa- 849 was isolate gen temperature ind Equipment Su leakage.	nately 1542 acter (MCC) LMXD. drogen Gooler ed. 0 gpm and probable NC sing Eydrogen d, but the on the Generator mp B High Level
high (mainta trippe 1NV23	roximately 1610 hours. Re- enerator hydrogen tempera- ined by maximum charging. d and the Unit entered Mo NC Letdown to Regenerati in an attempt to isolate 8 hours. Preseurizer lev	ture and NC leakage. At 1638:34 hours, 1 de 2, Startup. At 1 on Beat Exchanger Is.	rressurizer is the Main Turbine 638:46 hours, va clation valves, a Latdown was as	was manually lives 1NV1A and were manually stablished at

LICENSEE EVENT REPOR	RT (LER) TEXT CONTINU	UATIO	N		v.8	-				
						-				
	TEAR DECENTION		E	MILES.						
Catavba Nuclear Station, Unit 1	avba Nuclear Station, Unit 1 0 18 10 10 10 10 11 11 016		0 13	11	-			OF	n 14	
and repair MCC 1MCD. At 2105 hours June 15, 1986, at 0257 hours, the D Lequest was completed on June 18, 1 CONCLUSION	nit entered Kode 5, 1 986.	Cold	Shu	Edo	wa.	1	be W	ork		
This incident is assigned Cause Cod Installation deficiency. Upon shut 360 degree circumferential weld fai preliminary conclusion is that the cavitation induced vibration of LNV	down of the unit, an lure on the outlet f	inve lange	of	gat 1N	100 784	re 9.	The	ed a		

approximately one month to reduce pressure due to the Latdown Heat Used for A formal failure analysis is being performed by Vestinghouse. The interim resolutions are to replace the failed weld and limit operation of 187849 to a short period of time, 5 to 10 minutes, at low flow when re-establishing latdown to minimize shock on the piping. Vibration in the vicinity of the latdown orifices will be monitored to verify acceptable operations. The weld has been repaired. Upon receipt of the Westinghouse report of the failure analysis, Duks will review the data and make recommendations for a permanent resolution.

A contributing cause to this incident is loss of power on NGC IMID. Investigation revealed that the nameplate on a control transformer in compartment RO4A of MCC IMID became unglued. The nameplate fell against a terminal strip and caused an overload. The overload tripped the normal feeder breaker to the MCC and prevented the alternate feeder breaker from closing in. The failure of NGC IMID caused a loss of control power to INV849 and IRL138. The breaker in IMID RO4A was replaced and the work completed on June 18, 1986. On May 15, 1986, a Work Requast was initiated to remove the nameplates from all control transformers in all Unit 1 and shared Melson 600V MCCs. The work request was completed on June 16, 1986. The removal of all control transformer nameplates from Duit 2 Melson 600V MCC was done in 1985 per Significant Deficiency No. 414/85-09. At that time, an inspection of Unit 1 revealed a low percentage of fallen nameplates. A decision was made to remove the Duit 1 nameplates during the first refueling outage.

An exact value of the leakrate was found to be difficult to determine. Personnel employed various calculation methods in an attempt to quantify the NC leakage. The method which was chosen as the most accurate was the Radwaste inventory method. This data covered the entire span of the incident, and the Radwaste input before and after the incident was constant. Therefore, using the Radwaste generation base as the most reliable indication, the average leakrate was determined to be 87 gpm.

A post shutdown inspection for NC laskage revealed a leak on the manway of the Pressurizer. The leak has been repaired.

Fost incident vibration monitoring of the associated Unit 2 piping revealed no umusually high vibrations on the variable latdown orifice flowpath.

	UCENSEE EVENT REPO	RT (LER) TEXT CONTIN	UATION SHAD	Devis NO 3180-2134
TY BAME	11	DOCKET NUMBER (2)		PAGE (3
		1	TEAR REQUESTION METER	
	tawba Nuclear Station, Unit 1	0 5 0 0 0 4 1 3	816 - 0 3 11 - 013	014 01 0
	a neurost an seasons and from 2004 to 177			
	view of MPRDS indicated there on MCC.	are no reported fails	ares of this type Lavo	lving
There	s were two previous incidents wha (see LER's 413/85-59 and 4	of a Unit Shutdown du 13/85-61).	as to unidentified les	kage at
CORR	ECTIVE ACTION			
(1)	Reactor power and Turbine los	d were decreased.		
(2)	Valves 1NV1A and 1NV2B were o	losed.		
(3)	A Work Request to repair MCC	LMED ROAA, was initi	ated and completed.	
(4)	A Work Request to remove all Unit 1 MCCs, was completed.	remaining control tr	ansformer nameplates	from
(3)	Procedure changes to OF/1/A/0 and OF/1/A/6200/01, Chamical add operational limitations	and Volume Control S	Procedure for Unit St. ystem, wore implement	artup ed to
(6)	The weld failure was repaire	d.		
(7)	A followup surveillance will stated in the final resolution	be performed based on of the weld failur	on the proposed action to cause.	
(8)	An accurate determination of completed.	the NC leakrate dur:	ing this incident has	been
SAFT	TT ANALTSIS *			
Equ: DTA: prov Love Stor	SC System leakage from the fa ipment Sump. The water was pu in Tank, and the Starm Generat cessed, sampled, and discharge antory was maintained by the C rage Tank. There were no unex tainment Building.	amped to the Waste Evi for Drain Tank. All v ed through the Liquid	aporator Feed TADE, Fi water was eventually Radwaste System. NC Pumps and the Fueling	System Vater
Any Pre deg Ste	ing the Unit Shutdown, NC Syst 1 hour period. Pressurizer ; ssurizer level did not decrease rees F post shutdown, but stal am Generator levels decreased bilized at approximately 50%;	pressure did not decr se below 191. Averag bilized at 551 degree to a minimum of 351	 temperature decrease F after 30 minutes. 	ed to 530 The
	health and safety of the publ	the many and addressed	he shis inclient.	

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4.80 FORM 3864

4472 Faun 2006 (9-43)	LICENSEE EVEN	T REPORT	(LER)			4108 Y COMBINE NO 3180-0104
FACILITY HAME IN				DOCK #? NUMBER	9	I MART S
Catawba Nuclear Station, Unit	2			0 5 0 0		Provenue and
Safety Injection During Loss		on Taxa D				ube be and a such that
LEA NUMBER IN	REPORT DATE	IN TEST IN	OTHER	ADULTIES HAVE	lency	
WONTH DAY YEAR YEAR REDUCET AL AR	WEEDA MONTH DAY	TEAR	PACILITY NAM		DOCKET MUN	
			N/A	in the second second	0 18101	01011
and a state of the second of t	0 0 0 7 2 5	8 6			0 16 10 1	01011
MODE BI 3 30.4026	BI. 408-La	The De He Cam & A	Charlet and an many in 96.736a1(2111)	of the Actionships (1		
POWER 8	40.30int(1)	100	# 7 Hald Not		71.716	
1140 01010 30 400 WHOM	E selector				V OTHER	
50-409-us1(11)(8) 30-409-us1(11)(8)	80.7361Q1(0		#17861031480G		386.4	e in Yart, NRC Fa
B. 400 million	80.73ka1.21(8) 80.73ka1.21(8)	-	8.786/001414		50.72(b	(1)(4v)
a mental second se	LICENSES FORTACT A		86.796s/CENaj			
uni					TELEPHONE N	LWBEA
Posses V. Confilments in the				AREA COOR		
Roger W. Ouellette, Associate	Engineer - Li	censing		71014	317 13 1	-17 1 513
AUSS EVETER COMPONENT MANUFAC REPORT	48.1			T	1	Internet in the
TURER TO HAM	RDI	CAUSE PUPTER	DOMPONEN?	TUMER	TO NERDE	P.C.S.H
	1.19.24			111		March .
	1 States		111			A State
B. NP. EMENTAL RE	PORT EXPECTED IN		ter te de de a	EXPRCT1	- MON	TH BAY YE
TES IN HE DERAME EXPECTED PUBLICAD DATE				8.6×187		
TRACT /LINK IS 1400 percent is approximately Proper proper seco	V typeneritter Brase (18)					
On June 27, 1986, at 0953:14 f Pressurizer and Main Steam Lin decreased during the Loss of C Steam Generator (S/G) Power Op in Pressurizer and Steam Line be actuated when unit control was in Mode 3, Hot Standby, at The incident is assigned Cause Installation Deficiency, and C controls were changed since in located on the Auxiliary Shutd that personnel could accurately manner in which to adjust the 1 implementation of the design ch change to S/G PORV controls on	Control Room T perated Relief pressure was was transferr the time of Code B, Desi ause Code D, 1 itial install own Panels (AS position the PORV controle	est due to Valves (F sufficient ed back to the Safety gn, Manufa Defective stion, but SPS) were PORVS.	er and Sc. inadver: 'ORVs). 1 t co cause t to cause the Cont 'Injectio cturing, Procedure PORV con Bot revis Procedure	tem Line ; tent open. The ensuin safety : trol Room. Construct . The S/ trol devi ed accord a which a	pressure ing of t ng decre Injectio . The u tion/ /G PORV ice lege lingly e	be ase n to nit nds o
This incident is reportable pur 50.73, Section (a)(2)(iv). B608060292 860725	revent to 10 C) and 10	
PDR ADOCK OBOODEDR					1.1.1	

LICENSEE EVENT R	EPORT (LER) TEXT CONT	INUATION ATTRAVES ONE NO 2160-2104
	DOCKET NUMBER (2)	LER NURMEER IS PAOL IS
CTTY MAME IN		-114 SEDUCITION MELTINGS
Catawba Nuclear Station, Unit 2	0 5 0 0 4 1	14816 - 0218 - 010 012 08 3
to apple particle a required lass additional hits; form 305.4 sci (1):		
BACKGROUND		
Procedure TP/2/A/2650/03, Los	s of Control Room Fun	ctional Test, provides guidelines
to demonstrate:		
level (10-25%) using Aux	ciliary Shutdown Fanel	conditions from a moderate power (ASP) controls and following mb).
(2) That the plant can be ma	aintained at Bot Stand	by conditions for 30 minutes from
(2) That the plant can be by	rought to Hot Standby	and maintained in that condition
 (3) That the minimum shift t (4) That the Reactor Coolant 	F Suprem (FIIS:AB) CAD	be cooled down at least by
degrees F from a steady the ASPs.	state Hot Standby con	dition while being operated from
being Pressurizer Pressure 1:	nal is initiated on se ess than 1845 psig, or	everal conditions, among those r Main Steam Line Pressure less
than 725 paig.		
when controls are transferred initiating conditions are pr	d to the Auxiliary Sh esent, these component rol Room. A S/I signs ase A Isolation isolat	tes all Containment penetrations
DESCRIPTION OF INCIDENT		
at 24% power. In accordance tripped at the Reactor Trip (EIIS:SJ) Isolation and the (EIIS:BA) Pumps occurred at	with the test proced Switchgear at 0942:20 autostart of both Moti 0942:32 hours. Low-L D. The CA Pump Turbi	onal Test was begun with the unit ure, the Reactor was manually :855 hours. Main Feedwater (CF) or Driven Auxiliary Feedwater (CA) ow levels subsequently occurred in ne (CAPT) automatically started on er tripped at 0942:42 hours on low
0942:49 hours. Letdown Pres when the transfer occurred. approximately 3 minutes. Ch interference and the second second	soure Control Valve, Z Letdown flow indicat harging flow during th gh to a maximum off 17 isolated after Press	7.8 gpm at approximately 0946:30 murizer level dropped to below 201.
Panal (AFUPT/P) ware closed	in accordance with th and D Power Operated	ary Feedwater Pump Turbine Control te procedure. When the breakers Relief Valves (PORVs) opened to the AVUPTOP being initially set

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	LICENSEE EVENT REPO	ORT (LER) TEXT CONTIN	114 91 10 81	PROVED ONE NO 3150-0104
**	10117 Bar	DOCAST NUMBER (2)		*.#E1 10:45
			VEAN BEOUEVE	PAGE (3
K	Catawba Nuclear Station, Unit 2	0 5 0 0 0 4 1 4	816 - 01218 -	010 :3 01 01
	per procedure to what was thought Design Change had modified the PK ASP PORV controls had not been ad Therefore, the changes had not be nor had manual loader legends/sca positions on the ASP. The S/G PO secondary side with an accompany! AFWPTCP observed the decreasing s setpoint for PORV opening, but ac Control Room observed actual PORV but did not immediately communica test. S/G levels responded to th dropping repidly off the narrow r range indication. The CAPT had b 4.5 minutes, the S/Gs were blowing provided to S/G D. Pressurizer Pressure dropped off of minutes after the S/G PORVs opened Pressurizer Pressure (1845 peig) of Low Steam Line Pressure Loop D (72 S/I was partially blocked at that ASPs. Several containment isolati suction was succastically aligned conditions were satisfied.	dequately communicate dequately communicate then incorporated into alles been changed to pRV opening caused a ing cooldown of the p team pressure and at tually opened the PO position go OPEN af te this to ASP person e PORV openings by fi ange scale. The ASP een secured at 0945:4 g down through the PO cocale (less than 1700 d. Safety Injection becurred at 0949:46:1 25 psig) occurred at time due to control lon valves closed aut to the Refueling Wat	impact on the funderstood. all applicable preserved. accurately indicar rapid depreserved. respied to increase RVs further. Peri- ter energizing the anel due to the pa- irst svelling and Operators were of 45 hours. For app DRVs with CA flow Desig) approximate (S/I) condition of 79 hours. S/I co 0950:08:107 hours being transferred comatically and Ch er Storage Tank w	nction of the rocedures, te the PORV stion of the sonnel at the sonnel in the e AFWPTCP, ature of the then berving wide proximately being rely 2 m Low mdition on . Bowever, to the arging then the S/I
	manually started Centrifugal Charg controller labeling problems, ASP increasing it while adjusting the Control Valve.	the this samp so. B	owever due to val	ve
	At approximately 0953:30 hours, the return control to the Control Room directed personnel to swap control S/I was immediately actuated due to signal. Both Diesel Generators (D. reclosed on transfer of controls to Residual Heat Removal (ND) Pumps, 1 2NI-9A, and 2NI-10B, NV Pumps Disch associated CA valves. 2NV-148A rec Room.	back to the Control o the unblocking of /G) actuated on LOCA o the Control Room. Safety Injection (NI) harge to Cold Leg Iso closed following the	the Senior React: Room. When this the still-present condition. The i The S/I signal a: Pumps, the CAPT, olation Valves, at transfer back to	or Operator was done, actuation S/G PORVs tarted the , and opened ad the Control
	Both D/C load sequencers completed seconds. Containment isolation val on the Phase A Isolation signal. S	accelerated sequence	ing within approxi	

10 Farm 2005.4	LICENSEE EVENT REPORT (LER) TEXT CONTINUATION
	TODOLET NUMBER IN LEA NUMBER IS FAOL IN
CILITY BANK (1)	TEAM SECURIT SUBER
Catavba Nucl	car Station, Unit 2 0 18 10 10 14 11 4 8 16 - 01 218 - 01 0 014 0F 0
17 If mans anoth a required	a adottana kiti fare abbit si titi
	and the television television Values were closed
	ours, S/I was reset, the Cold Leg Injection Isolation Valves were closed II and ND Pumps were secured. The S/I had further reduced Steam Line
and the	I and ND Fumps were accured. I and primary coolant temperature to approximately to approximately
468 degr	es F. The CAPT was secured at 1000 hours.
	NV Pumps suction was automatically swapped to the Refueling Water Storage
	mi materia Parta (V(T) lavel began increasing une to no romp
and the second sec	The DAT was realigned to the succide of the rise was
FECT Suc	ion Valves were closed at 1007 hours, completing the realignment of normal
charging	
CONCLUSI	
In James	y 1985, a Design Change Authorization (DCA) was originated and approved.
and a second of	in and the C/C DOPUS to asfety-related ber HURLO-USDA, Suppresent no
the shide of	is how the function of the ASP S/G PORY REDUKI LORGETS WAS classed to
that the	PORVs open in direct response to any position dialed-in on the loader.
Previous	y, the loader provided a setpoint selection so that the PORVs open fully setpoint was exceeded. This DCA was implemented after Bot Functional
when the	BFT) was completed in October 1985.
While ch	nging the Manual Loader functions for this DCA, the legends on the loaders
and the second s	cince the manual loaders appeared physically the seam as you and
an anna a b a	APP Answertows appeared to close the PUEVE OV IDCIESELIE to c sectored of
in actua	asy opened the PORVs further. Therefore, this incident is assigned Cause esign, Manufacturing, Construction/Installation Deficiency.
	esign, Maburacturing, construction, instantion
This incl	dent is also assigned Cause Code D, Defective Procedure. On several
	dent is also assigned Cause Code D, Defective Procedure. On several during December 1985, January 1986, and February 1986, responsible
occasion	during December 1985, January 1986, and February 1966, responsed discussed the DCA, but the effects on ASP operation were not understood.
occasion	dent is also assigned Cause Code D, Defective Procedure. On several during December 1985, January 1986, and February 1986, responsible discussed the DCA, but the effects on ASP operation were not understood, fore procedures and training were not modified.
occasion personne and there	during December 1985, January 1986, and February 1986, response of discussed the DCA, but the effects on ASP operation were not understood, fore procedures and training were not modified.
occasion personne and ther During th	during December 1985, January 1988, and February 1988, responsible discussed the DCA, but the effects on ASP operation were not understood, fore procedures and training were not modified. e transient, several deficiencies were identified:
occasion personne and ther During th (1) D/G	during December 1985, January 1986, and February 1966, tesponance discussed the DCA, but the effects on ASP operation were not understood, fore procedures and training were not modified. e transient, several deficiencies were identified: Load Sequencer & Load Group 7 was 0.1 second late. The timer has been
occasion personne and ther During the (1) D/G rec.	during December 1985, January 1986, and February 1966, tesponance discussed the DCA, but the effects on ASP operation were not understood, fore procedures and training were not modified. e transient, several deficiencies were identified: Load Sequencer & Load Group 7 was 0.1 second late. The timer has been Jibrated. ASP controller for the Charging Pumps Flow Control Valves, 2NV-294, was
occasion personne and ther During th (1) D/G rec (2) The	during December 1985, January 1986, and February 1966, tesponance discussed the DCA, but the effects on ASP operation were not understood, fore procedures and training were not modified. e transient, several deficiencies were identified: Load Sequencer & Load Group 7 was 0.1 second late. The timer has been Jibrated. ASP controller for the Charging Pumps Flow Control Valves, 2NV-294, was and incovractly. The label was corrected.
occasion personne and ther During th (1) D/G rec (2) The lab	during December 1985, January 1986, and February 1986, responsed discussed the DCA, but the effects on ASP operation were not understood, fore procedures and training were not modified. the transient, several deficiencies were identified: Load Sequencer & Load Group 7 was 0.1 second late. The timer has been Dibrated. ASP controller for the Charging Pumps Flow Control Valves, 2NV-294, was led incorrectly. The label was corrected. Pacedar (FP) rimes were 4 hours late. The timer was reset.
occasion personne and ther During the (1) D/G rec. (2) The lab (3) Even (4) Int	during December 1985, January 1986, and February 1966, responsed discussed the DCA, but the effects on ASP operation were not understood, fore procedures and training were not modified. e transient, several deficiencies were identified: Load Sequencer & Load Group 7 was 0.1 second late. The timer has been Jibrated. ASP controller for the Charging Pumps Flow Control Valves, 2NV-294, was lied incorrectly. The label was corrected. It Recorder (ER) times were 4 hours lats. The timer was reset. rmediate Range Channel N-35 was undercompensated. The voltage was
occasion personne and ther During the (1) D/G rec (2) The lab (3) Even (4) Inte adj	during December 1985, January 1986, and February 1966, responsed discussed the DCA, but the effects on ASP operation were not understood, fore procedures and training were not modified. te transient, several deficiencies were identified: Load Sequencer & Load Group 7 was 0.1 second late. The timer has been Jibrated. ASP controller for the Charging Pumps Flow Control Valves, 2NV-294, was lied incorrectly. The label was corrected. tt Recorder (ER) times were 4 hours lats. The timer was reset. rmediate Range Channel N-35 was undercompensated. The voltage was ated from -10V to -16V.
occasion personne and ther During the (1) D/G rec (2) The lab (3) Even (4) Inte adj (5) 287	during December 1985, January 1986, and February 1966, responsed discussed the DCA, but the effects on ASP operation were not understood, fore procedures and training were not modified. te transient, several deficiencies were identified: Load Sequencer & Load Group 7 was 0.1 second late. The timer has been librated. ASP controller for the Charging Pumps Flow Control Valves, 2NV-294, was led incorrectly. The label was corrected. It Recorder (ER) times were 4 hours lats. The timer was reset. rmediate Range Channel N-35 was undercompensated. The voltage was ated from -10V to -16V. 13A, Letdown Orifics 2A Outlet Containment Isolation valve, could not be
occasion personne and ther During the (1) D/G rec (2) The lab (3) Even (4) Inte adju (5) 2NV clo	during December 1985, January 1986, and February 1966, responsed discussed the DCA, but the effects on ASP operation were not understood, fore procedures and training were not modified. te transient, several deficiencies were identified: Load Sequencer & Load Group 7 was 0.1 second late. The timer has been librated. ASP controller for the Charging Pumps Flow Control Valves, 2NV-294, was led incorrectly. The label was corrected. t Recorder (ER) times were 4 hours lats. The timer was reset. rmediate Range Channel N-35 was undercompensated. The voltage was ated from -10V to -16V. 13A, Letdown Orifics 2A Outlet Containment Isolation valve, could not be ed during the transient. The valve was found to be stuck open. The valve
occasion personne and ther During the (1) D/G rec (2) The lab (3) Even (4) Inte adj (5) 287 clos	during December 1985, January 1986, and February 1966, Responsed discussed the DCA, but the effects on ASP operation were not understood, fore procedures and training were not modified. e transient, several deficiencies were identified: Load Sequencer & Load Group 7 was 0.1 second late. The timer has been librated. ASP controller for the Charging Pumps Flow Control Valves, 2NV-294, was lied incorrectly. The label was corrected. It Recorder (ER) times were 4 hours lats. The timer was reset. rmediate Range Channel N-35 was undercompensated. The voltage was ated from -10V to -16V. 13A, Letdown Orifics 2A Outlet Containment Isolation valve, could not be ed during the transient. The valve was found to be stuck open. The valve freed and successfully retested. 2NV-13A is a 2 inch Borg Warner Gate . Per NPEDS, there are no reported failures of 2 inch Borg Warner valves
occasion personne and ther During the (1) D/G rec (2) The lab (3) Even (4) Inte adju (5) ZNV clow was Val	during December 1985, January 1986, and February 1966, Responsed discussed the DCA, but the effects on ASP operation were not understood, fore procedures and training were not modified. e transient, several deficiencies were identified: Load Sequencer & Load Group 7 was 0.1 second late. The timer has been Jibrated. ASP controller for the Charging Pumps Flow Control Valves, 2NV-294, was led incorrectly. The label was corrected. It Recorder (ER) times were 4 hours lats. The timer was reset. rmediate Range Channel N-35 was undercompensated. The voltage was sted from -10V to -16V. 13A, Letdown Orifics 2A Outlet Containment Isolation valve, could not be ed during the transient. The valve was found to be stuck open. The valve freed and successfully retested. 2NV-13A is a 2 inch Borg Warner Gate e. Per NFRDS, there are no reported failures of 2 inch Borg Warner valves
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	LICENSEE EVENT REPORT (LER) TEXT CONTINUATION
ACUITY NAM	
	LAR NUMBER IS PAGE 13
	TEAS BEDUENTAL BELOOK
Carav	
3/7 18 James #10	ba Nuclear Station, Unit 2 0 9 0 0 0 4 11 4 8 6 0 2 8 0 0 0 0 4 0 0
(7) Several additional problems were identified with Operator Aid Computer indication. The problems are bains immediate with operator Aid Computer
	indication. The problems are being investigated.
(8	
	Additional labeling on the ASPs was added to clarify control requirements for 2NV-309. The appropriate operations procedures for
	2NV-309. The appropriate operations procedures were also revised to clarify use of the valve.
10	use of the valve.
(9	
	the ASP. A poor electrical connection in the control circuitry was found and corrected. The control circuit for the value was found and
	corrected. The control circuit for the valve had maintenance performed on it in January 1986. It is not certain if the valve had maintenance performed on it
	in January 1986. It is not certain if the poor connection was the result of this previous maintenance activity
(10)	this previous maintenance activity.
(10)	WIFFICULTY Was approximated in another
	Valves. The problem could not be recreated during investigation. The associated Monthly Surveillance Territory investigation.
	associated Monthly Surveillance Test was performed successfully.
None	of the discussion of the second s
0.004	of the identified Equipment Malfunctions are reportable to NPRDS.
This	fordant de la contraction de
Modif	incident is considered to be an isolated occurrence. The Nuclear Station
mode	fication program currently in use provides more stringent controls on
	fications than the Design Change program.
This	fo the first second a second
This	is the first accustion of Safety Injection on Unit 2.
	is the first accustion of Safety Injection on Unit 2.
	is the first accuation of Safety Injection on Unit 2. ECTIVE ACTION
	ECTIVE ACTION
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CORR	In light of degrading conditions, the Loss of Control Room Test was terminated and control was transforred back to the Loss of Control Room Test was terminated
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ra MAA	LICENSEE EVEN	T REPORT (LER) TEXT CONT	INUATION	U.E. BUCLTAR REQULATORY COMMI APPROVED DNB ND 3150-0104 EXPARTS \$11-85
UTY BANK (I)		DOCKET KOMBER ()	584 NUMB	
Catavba Nus	lear Station, Unit	2 0 6 0 0 0 0 0	14 816 - 012	18 - 010 016 0*
ch ch 1/ (8) Co	anges made to Opera anges made to the A Duit 2 control diff mpletion of ASP sur ntrol board cutting	to appropriate personnel ting and Abnormal Procedu SPs as a result of the po- erences. face enhancements not req , or a system shutdown at be implemented during the	pres, labeling ost incident an quiring hardwar te to be identi	and surface uslysis, and Unit e modifications. ified. The
SAFETY A	NALYSIS			
power im approxim Pumps and approxim approxim Reactor (mediately decreased ately 15% and were d the Auxiliary Fee ately 29%. Both ps Coolant (NC) Highes	r Trip in accordance with to zero. S/G narrow rad being restored by both Mc dwater Pump Turbine. Fre the time of the trip and rameters began slowly dec t average temperature was 1030 paig at the time of	ige levels decr otor Driven Aux sasurizer Press Pressurizer le creasing follow 550.9 degrees	eased to filiary Feedwater sure was at rvel was at ring the trip.
increased made to i opening of narrow ra range lev actual le at approx completel paig. Af occurred intriation	d to 177.8 gpm befor increase charging f of the S/G POR's re- ange level increase vel did not decrease vel in the S/Gs the cimately 59%. While by in approximately for Pressurizer lev- in the NC System. on of Safety Injects letect any significa-	ferred to the Auxiliary S re being manually isolate low, due to controller is sulted in a rapid decreas d and then rapidly decreas e below 54.4% indicated, roughout the incident. T e the PORVs were open, Pr 2 minutes, and NC wide r vel was lost, approximate This is based on the rat ion. Personael and the S ant drop in Reactor Vesse	ed. An unsucce agend problems. te in Main Stea used and droppe which is appro- the top of the reasurizer leve ange pressure ally 400-500 cub te of level rec- tafety Paramete	The inadvertent The inadvertent m pressure. S/G d off scale. Wide ximately 631 S/G tube bundle is the was lost decreased to 702 dic feet of voiding covery following r Display System
automatic closed au	ally as Pressurizer tomatically. Cold	ferred back to the Contro r Pressure and Steam Pres Leg Injection was initia ing Pumps. All S/I equip	sure were low. ted with the S	The S/G PORVs afery Injection
Pressuris was resto	er pressure was rea red to 332 after ap	S/I, NC pressure and lev stored to approximately 1 pproximately 5.5 minutes. ent was 95.5 degrees F/ho	250 psig and F The maximum	ressurizer level
	Health Physics samp ve releases during	oling of Main Feedwater a this incident.	ctivity, there	were no
The healt	h and safety of the	public were not affecte	d by this incl	dent.
				and the second design of the s

10, ton 10	UCENSEE EVENT	REPORT (LER)		UCLER # 850000 108 - COMBINE UMTROVES COMBING 3180-0180 EXTINGE 65:08
NOR/TY MADE IS			DOCK IT IS, SUPE	
River Bend Station			0 15 10 10	10141518 1 00
Hand Held Radio Causes	Loss of Adda	A Bawar		and the second
EVENT BATE IS LEA MUNICA IS	NOT OT TAT	Street or any other start and interest of the start of th		
DETA BAT TEAS YEAR BUD BUT ILL	MANA MONTH BAT TE			
				0 181010101 1
11 0 1 8 6 8k T d dz T	0 0 0 2 0 4 8	6		
	RAW	Y B. Thurde	Control Statement on a general second second statement of	1 127100
	B.Barri	B. Photos		Tanua.
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B. Martin	*	8.7840		MAN
a Auchas	X 8.78-00-08	a filerati		
	LICTIMES CONTINCT FOR	THE LEA IND		
			1.841. 3884	*\$1,8****
G. Alan Bysfield - Seni	or Systems En	gineer	15 10 14	613151-16101
	IN THE LAD CONTRACT IN	LUNE BRECHERS IS THE S	-	and and an and a state of the
well states compared with the well		Audit (******* 000*30%	NY NAME	
FI J TI M RC 6 10 18 10	N	1.111	1 111	1. 1. 1. 1.
and the second s	N		1 111	
Burn, heart's at				
"18 17	- T -		Da Ta T	1
entered into appropriat	line. One ho nsformers B as esulted in a s usual Event wo e Abnormal Ope	our later, a nd D also tr total loss o as declared erating Proc	t approx ipped pri f offsite at 1045 a edures.	imately 1044 or to A and C power (LOP) nd operations The plant was
shutdown at the time				cram that had
occurred approximately	six hours ear.	Lier. Upon	investiga	tion it was
determined that hand	held radios mo	ost likely c	aused spu	rious signals
in the tone relaying tr	ansfer trip re	eceivers of	the prefe	rred station
transformers. Correct	ive action	is being t	aken in	an effort to
minimize the probabilit	y of recurrent	ce. This ev	ent did n	ot affect the
public safety and welfa	re.			1/1

ac fun Ban	CENSEE EVENT REPO	AT (LER) TEXT CONTI	NUATION	* #80-0-410#1 committee v80 000# 140 2180-0-01 15 8 2 85
NUMER ALLONG (TO		00CX17 NuMB1# 0	128 MUNIELS 6	FA81 B
			Sa 9815 - No	MALL.
River Be	nd Station	0 0 0 0 0 0 4 5	8 8 16 -0 0 2 -0	10 0 2 0000
Sequence of Ev	ents:			
00 01/01/	86 at 0941 with	h the unit in op	erational conditi	on 3 (hot
	cooling down		1	nccurred
			ance LER 86-001).	preferred
approximately	and the second sec			
station transf	formers 'A' and	.C. errpped.	Recirculation pump	or Water
tripped, the o	operating conde	nsate pump trips	ped, and the React	UPPER hus
Cleanup (RWC)			Protection System	
'A' de-energi	zed initiating	a half scram a	nd partial Nucl	
Supply Shuto	ff System (NSSS	s) isolation.	The partial NSSSS	isolation
	trument air is		e Reactor Build	ing which
	scram valves	to leak filling	the Scram Discha	rge Volume
			s actuation on	
(SDV). This	subsequences :.	adverted at at ion	transformer trip	s Division
level at 095	7. Upon the pl	reserved acación	an I emergency V	entilation
I and III die	sel generators	started, Divisi	on I emergency v	1231 B. C.
systems auto	ostarted, and		water pumps 1SWP.	
and D load		Normal service		
circulating	water pump CWS	-P1B were still	running but with	but, pearing
cooling wate	r since bearing	cooling water	pump BCS-PlA had	Lost power.
At 1001 the	Main Steam Isol	ation Valves (M	SIVs) automatical	ly isolated
	asing condenses			
		are dispatched	and attempted	to recove
At 1003			S bus 'A' was res	et. Later
de-energized	load center			
panel ISCM*P	NLOIA was disc	OAeled de-eveld?	zed due to a blow	
and the second sec				

1.82 10.8% MMA

NAC Form Make	UCENSEE EVENT REPOR		ATION MICLEAR PR	How A TOP & COMMISSION
	10 C	DOCATT NUMBER ()	178 MUNICE IN	****
			TAN MERCENT AL MURER	
River Bei	nd Station	0 16 10 10 10 14 15 18	8 6 - 0 0 2 - 0 0	
Division 3 attempts (unsuccessfu	Fuel Building EVAC Control Building	(HVF) dampers t chilier (HVK) on of HVK chille SS isolation rem	to trip. Subi rs 'B' and 'D' we	sed the sequent re also

The RPS actuation was reset at 1042. At 1044, approximately one hour after the initiating event, preferred station transformers 'B' and 'D' tripped. The station was now in a complete loss of offsite power (LOP). The Division II diesel generator started and sequenced properly. An Unusual Event was immediately declared and Abnormal Operating Procedures (AOP) 004, 005, 0010 and 0042 were initiated. Reactor Water level was +80 inches on the shutdown range and pressure was at 240 psig.

At 1114 the half RPS actuation was reset and power to RPS bus 'B' was restored. At 1124 the preferred station transformers were energized, but the supply breakers to the plant could not be closed. It was determined that breaker closure was locked out by the tone relaying transfer trip (fiber optic) system which could not be reset. At 1130 this backup system was disabled and the breakers were closed. All in house loads were restored and the Unusual Event ended after an hour and ten minute duration. The plant wis stabilized.

44):	U	CENSE	E EVEN	T REPORT	T II	E	R) 1	0	(T	00	NT	INU	A1	nio	N					-	-	06.000 81.01.01	
TABILITY NAME IS					-		-	10.1	6				-	4.4		14 MA	 1.7	R		F	*	-64 2	
River	Bend	Stati	on			18	1.0			Ľ	4	5 8	8	6	_		 -		0	0	4	0.00	18

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Investigations,

In an effort to determine the cause of the transformer trips an investigation of the protective relaying was conducted and revealed that no protective relaying targets were initiated. It was further determined that the trip signals sent to the lockout relays could only have been initiated by a spurious signal in the backup pilot wire or tone relaying transfer trip circuits. Functional and diagnostic testing of both the pilot wire and tone relaying circuits showed that both systems were operating as designed at the time of testing.

As a result of this testing two items were noted. First, spurious trips could be generated on the tone relaying system with hand held radios in close proximity (within approximately a 10-12 foot radius) of the transmitters/receivers. Second, some of the tone relaying keying and rack power were supplied from two separate battery sources. Although no spurious trips could be simulated by testing, this type of connection could result in transients within the tone relaying equipment. It was decided to correct the wiring in the field such that keying and rack power were supplied by the same battery sources.

The two types of hand held radios tested were the 4 watt, 150 MHz Motorola and 5 watt, 450 MHz Motorola. Both are commonly used on site by security and operations personnel. Both of these radios were keyed to transmit inside the control building of the Fancy Point switchyard and both caused spurious trips on the tone relaying system. Also

1.81 10.8% MMA

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LICENSEE EVENT REPOR	T (LER) TEXT CONTINU	ATION ATION	terne fan i somme da se Swit wei stat di se Stat
	MOLT MORES 1		**** 3
		the Browley av Mr. 107	
River Bend Station	0 18 10 10 10 1 41 51 8	8 6 - 0 92 - 0 0	0.5 400
tested were 100 watt, 50 MHz just outside the switchyard con	ntrol building w	ith the doors open	n. The
mobile radios did not initiate	either a trip of	r a loss of guard	signal
in the tony relaying system.			
concluded with high probabil	lity that the	LOP was caused by	radio

Also investigated was the difficulty in resetting the lockout relays. Because of the complexity of the tone relaying and pilot wire tripping circuitry, the resetting of the lockout relays must be performed in the proper sequence. It was determined that operations procedures did not address the required sequence.

Corrective Actions:

frequency interference.

As a result of this event several corrective actions have been completed or are in progress. These corrective actions include:

 Installation of shielding on the tone relaying equipment in the Fancy Point switchyard. Shielding of the equipment in the plant is not required because the equipment is enclosed in a reinforced concrete room with locked doors and a sign restricting the use of radios on each door. This activity is presently scheduled for completion by 02/12/86.

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 Rewiring the tone equipment such that both channels are required for tripping. At the time of the event if one channel had a loss

		DOCK 17 NUMBER ()	148 MUNICE M 1455 3
-	- Ma		The BUSINESS BUSINESS
81	iver Bend Station	0 16 0 0 0 4 15 18	8 8 16 -0 0 2 - 0 0 0 6 0 0 0
-	of muand and the other	channel had a trip	signal the transfer trip
			change provides increased
	reliability to help		
			Ign change (MR 86=0081) is
	Alterations sere and	on prior to the pl	lanned 35 percent power
	scram at the conclusion		
			laying equipment such that
3.	changing by power o	war are both sup	plied from the same DC
	the Keying and leak po	earstions were inst	talled. The design change
			prior to the planned 35
	percent scram at the c		
			ders in the switchyard and
4.			e relaying panel. This
	at the generator/trans	ion of two design	modifications (MR 86-0027
	requires the complet	final installation	of these recorders is to
	and an ecover, the	an outage just af	ter the plannod 35 percent
	power scram.	ional drainage read	ctors at the plant end of
5.	installation of boats	elding. This des	sign change (MR 86-0093) is
	the pilot site that	ion prior to the pl	lanned 35 percent scram at
	the conclusion of tes		
			nd Data Acquisition (SCADA)
6.	sustan alarma to prov	ide annunciation i	n the main control room and
	elecen ergrup to biot	And and a sector of a	

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-	A 10 11	The second s	and the second se	488800182 294 40 1 40 40 5 844 882 5 1 10
	liver Bend Station	0 15 10 10 10 1 4 1 4	-tax HEreinit	
-1	anne a meuras, un antenna hall fant Aller a (13)			the property of the second sec
	at the Government Str	eet transmission	n and distribu	tion control
	center for loss of chann	el signals on	tone relaving	equipment.
		or completion a		
	capabilities for monitor	ing trip and loss	s of guard sig	nals will be
	added (MR 86-0094) up	on receipt of al	arm cards by a	pproximately
	04/18/86.			
7.	Training personnel on th	· restricted use	of radios	#1
				Signs have
	been posted in the Fan			a construction of the
	radios in the control bu	ilding. Signs he	ive also been	posted on
	the doors of the room	m in the turbine	building whic	h houses the
	tone relaying equipment.	Letters have	been sent t	o Security
	Operations, Maintenance			
			ission and	
	personnel informing them	of the radio	use restrict	ions. This
	action is complete.			
8.	Training operations per	rsonnel on the	resetting o	f lockouts,
	including necessary proce			
				re presently
	undergoing requalificatio	on and will be	trained on	protective
	relaying, tone relaying	g, and pilot wi	re relaying i	ncluding the
	proper resetting of the 1	lockout relays.	This process	is presently
	scheduled to be completed			
9.				1.67 2.4
<i>₹</i> 1	A procedure for the ;			ne relaying
	equipment and the proper	r operation of	the sequence	e of event

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1000 farm 8881 8-63	LICENSEE EVENT REPO	RT (LER) TEXT CONTINU	ATION STREET	10-54 10 1 56 40-54 10-54
FLOUTY ALMI (1)		BOCE 17 16-1882 8 12-	144 053014 0 1444 812,457 2, 80,42 5,4812 (5,511	Pa 64 - 5
River Be record		en and is schedul	ed to be complete	d prior
to the condit		ent scram at t	the conclusion o	f test

Safety Assessment:

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> There were no safety consequences to the public as a result of this event. The safety implications of a loss of offsite power are however, clearly recognized and it is for this reason that the above corrective actions are being taken.

UCENSEE EVENT REPORT	(LER)
RIVER BEND STATION	0 1 8 1 9 1 9 1 9 1 4 5 5 1 0 PT
Trip of EJS*S#G2A Due To & Faulty Jutput	the second s
EVENT BATE IS	STARS & Lack, Pale Javas, vise at
BUTS BAT YEAR TEAR BECANT AL BOOKER BORTS BAT YEAR	TADUTY NAME DOCLET NAMESHIE
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d 73 12 8 16 8 6 0 4 7 0 1 1 dz ic 8 6	
	And on a man a an denoming (1)
	B.Rosen
The strategy of Gammary Without and	BMandfelden Distantion of Annual Annu
	B./NetBurra
Licenses pourtant res two Las res	61.79%/520v
	TELEPHONE NUMBER
E.R. Grant - Director-Nuclear Licensing	5, 0, 4 6, 3 5- ;6, 9, 9
and an and a second sec	I THE REPORT TO
an arran consents and to make the total the set	Bar Summer to union
EIC 1513 X1919 Y	a start barre
1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1	
BURN, BARRY AL REPORT EVERATE INA	unter and and a
	BATERS
And? Autor & Last amon. 14. approximate Minor angle-anne baseries angl 718	bell 1
On 7/31/86 with the unit at 94% rated power al init at 99% rated power 1EJS*SWG2A feeder bre- sutomatic start of both divisions of Annulus + Treatment and Fuel Building Ventilation System Prywell and containment unit coolers as well a enerator fuel oil transfer pump were lost. If n the feeder breaker to the 1A containment Uf he cause of the trip. During the 7/31/86 eve pecification surveillance 4.8.1.1.1.a was not s required by Action 3.8.1.1.b. There were r ealth and safety of the public as a result of tart of these Engineered Safety Features plac onservative condition by filtering any radios o releasing it to the environment.	aker tripped causing the Mixing, Standby Gas ms. Additionally, various as the Division I diesel A defective protective rela hit Cooler was found to be ent the Technical t completed within one hour to adverse affects on the f these events since the
B610270287 861020 PDR ADOCK 05000458 S FDR	TEDD

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UCENSEE EVENT	REPORT (LER) TEXT CONTI	NUATION	*
		LES INVIRGES IS	
1174 Roome 111		The Black Mitte	
RIVER BEND STATION	0 18 10 0 0 4 5	18 8 6 - 0 4 7 - 0 1 0 200	0
and the second se	and should save the same the surface of the surface of the same		
and a station with the	e unit at 94% rated	power containment unit	
			1
switchgear 1EJS*SWG2A feed	er breaker 1EJS*ACE	138 tripping. the	. 1
resulted in the automatic in Engineered Safety Feature	start of both train	ulus Mixing, Standby Gas	
Engineered Safety Feature Treatment, and Fuel Buildin	EST Systems. Add	litionally, power was lost	
			-
power to several valves on	the Low Pressure C	Core Spray (LPCS),	
was immediately reduced to	ADDTOXIMATELY JJ4	FO BERTHERE	
temperatures, due to the lo	oss of area coolers	4.	
	10024 and 10020 H	a performed and no	
Testing of feeder breakers problems were identified.	ACBIG and ACBIG WO	ion was checked and found	
problems were identified. to be normal. Drawings, de	Unit Cooler operations	, and as built wiring	
to be normal. Drawings, de were reviewed for correctne	esign modificacion	ities impacting this	
problem were found. Feeder overcurrent timer relay (t)	vpe ITE 50D) which	initiates a trip of	
overcurrent timer relay (t) breaker ACB38 in the event	ACB36 fails to cle	ear a fault in sufficient	
breaker ACB38 in the event time. This circuitry and :	relay were tested a	ind found to function	
time. This circuitry and a correctly. Unit Cooler 1A	was placed into se	rvice and functioned	
correctly. Unit Cooler 1A properly. At the conclusion	on of these investi	gations the unit, was	
returned to service but Uni	it Cooler 1A was le	art in standby as a	
precaution.			
Technical Specification Act	tion 3.8.1.1.b reg	ired surveillance	
Technical Specification Act 4.8.1.1.1.a to be performed	d within one hour	due to the loss of the DG	
4.8.1.1.1.a to be performed fuel oil transfer pump).	This surveillance w	as satisfactorily	
fuel oil transfer pump). completed at 0900, beyond	the allowed time in	nterval. This is an	
inclated case of a missed	Lecunrear sheares	ation Action and is not	
indicative of a generic pro	oblem.		
	to an and resed	power, feeder breaker	
On 8/2/86 at 0637 with the ACB38 again tripped on app	ant overload of	Unit Cooler 1A with all	
ACB38 again tripped on appe events and actions occurrin	na as described in	the 7/31/86 event above,	
events and actions occurrin with the exception that Un	nit Cooler 1A was I	not inservice at the time.	
with the exception that .		1	
with the overload indication	on having been rep	orted on cooler octa,	
With the overload indication maintenance tested the ITE	50D relay in place	. The relay ceated	
maintenance tested the ITE normal except that it's pur	shbutton test circi	since the ambient	
would not work, indicating	BOME marringert	malification temperature	
temperature was very close	to the equipment	may be heat sensitive.	
of 114 degrees r 1t was ie.	it that that slace	a in an oven at 120	
The relay was removed from	Bervice and the	circuit monitored. At	
degrees with dc power conn	ecceu and the star	ailed with a continuous	
one hour and 10 minutes, t trip output. The malfunct	ion of this relay	caused the overload	
trip output. The malfunct indication and trip of fee	der breaker ACB38.	The relay was replaced	
with a new unit.			
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UCENSEE EVENT REPO		UATION MANOVES	
CILITY MANNE IN		(11 mundet ()	****
		tes plaufatur Menge	
RIVER BEND STATION	0 18 10 10 10 14 15 15		
I all mant space a manufact, say anteriored fully from Miles to (17)		and all the second	013000
The defective relay (ITE 50D) Bovari. GSU asked the manufac the relay. GSU suspected that manufacturer to determine if t close to a 10 watt resistor on circumstances, GSU initiated a replace the trip function of ti- relays with electro-mechanical Additionally, there are two IT on the recirculation pumps and only overload indication in use MR 86-1316 has subsequently been has satisfactorily determined the defective and caused the trip. manufacturer has determined the over time and is indicative of one. The manufacturer has reco 03 by the vandor on his circuit type transistor which has highe characteristics. The transistors in all t replacement of the transistors in all t replacement of the transistors in pected to be complete by 11/1 with only an indication functio This failure resulted in a part wither event were redundant tr utomatic and manual, resulted o time was the health and safe	the transistor the transistor he output transi the circuit boa Modification Re- his and two othe time overcurren E 46D relays tha 26 other ITE 50 e at River Bend en cancelled beck that the output The preliminary at the transistor a batch problem ommended this tra- board, be replay to that are bein or the relays wi 1/86 and by 12/3 n.	ne the failure mode had failed and aske stor is located too rd. Under the quest (MR 86-1316) r similar (ITE 50D) t trip devices. t provide trip func D relays that provi Station. ause the manufactur transistor only was y investigation by r developed high le rather than a gene insistor, identifie wer leakage og removed will be GSU is now replac ys via MR 86-1577. th a trip function 1/86 for the relay sion I power. In d all actions, bot	to to tions de er the akage ric d as nt sent ing The is s

APPENDIX F

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OPERATIONAL EVENT LISTINGS

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APPENDIX F

OPERATIONAL EVENT LISTINGS

This appendix contains listings of events that are of potential concern from a core damage standpoint but most of which were not, in general, selected as actual accident sequence precursors. Included are unavailabilities of RHR systems, mostly at zero-power conditions (Table F.1), and unavailabilities of single-train safety systems, such as HPCI, HPCS, RCIC, and soluble boron (Table F.2). These were not, in general, selected as accident sequence precursors.

Table F.3 through F.6 list events reviewed in this study by docket and LER number. Table F.3 provides a listing of the events involving a reactor trip from power or a trip that occurred during shutdown but with no additional failures or reported unevailabilities. Table F.4 lists the 1984 trip events that were more complicated, including events involving a core-damage initiator such as LOCA, SLB, LOOP, or LOFW or trips involving additional system failures. Several of the precursors were selected from this group.

Table F.5 provides a listing of the events involving a trip followed by an LOFW and no additional unavailabilities. Table F.6 lists LOFWs not requiring a trip and those that initiated a trip. Table F.7 lists events involving multiple failure or unavailabilities of various plant systems required in core-damage mitigation sequences.

Finally, Table F.6 lists all of the 1986 events selected for detailed review before selection of the 36 precursors.

Event identifier	Plant name	Event date	Reactor power (%)	Event description
275/86-012	Diablo Canyon l	09/08/86	0	A technician inadvertently grounded a power supply while performing main- tenance, isolating the RHRS
280/86-017	Surry 1	05/24/86	0	A spurious signal during maintenance activities
302/86-002	Crystal River 3	01/10/86	0	caused the RHRS to trip The RHRS was disabled when the service-water system was shut off to recover two divers who died while working in the sea pump systems
302/86-003	Crystal River 3	02/02/86	0	RHRS was unavailable for 24 min when the pump A shaft failed; pump B iso- lation valve failed
312/86-016	Rancho Seco	10/03/86	0	closed A maintenance error caused a short circuit, which
312/86-024	Rancho Seco	11/15/86	0	isolated RHR for 13 min The RHRS isolated when a false signal was gener- ated during inverter fuse
312/86-030	Rancho Seco	12/08/86	0	replacement The RHRS isolated due to a maintenance error com- mitted while an employee was working on a trans-
17/86-002	Calvert Cliffs 1	03/22/86	0	former During a test, a mainten- ance procedure error caused the RHRS to trip
23/86-002	Diablo Canyon 2	01/17/86	0	off An unlicensed operator inadvertently transferred power to the wrong panel, causing a valve to close
52/86-025	Limerick 1	05/13/86	0	and forcing an RHRS trip During a test, a personnel error caused the RHRS to isolate

Table F.I. Residual heat removal system unavailabilities reported in 1986

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Event identifier	Plant name	Event date	Reactor power (%)	Event description
354/86-093	Hope Creek 1	12/09/86	0	During instrument main- tenance, an error caused a false signal to isolate the RHRS
361/86-007	San Onofre 2	03/26/86	0	During shutdown for repairs, instrument error led to the RCS being drained too far, which caused the RHRS to iso- late
366/86-027	Hatch 2	09/21/86	0	A procedure error in test- ing caused the RHRS iso- lation valve to close
373/86-022	Las_lle l	06/12/86	0	An operator error during maintenance tripped the wrong switch and isolated RHRS for 2 min
373/86-022	Lasalle l	06/12/86	0	The RHRS was isolated for 2 min when an operator activated the wrong key switch during testing
373/86-039	Lasalle l	10/16/86	90	A personnel error in testing caused the high- pressure switch to be look-wired closed
388/86-015	Susquehanna 2	10/12/86	0	The outboard suction isolation valve closed or a spurious signal when switching "B" and "D" RHI pumps
388/86-015	Susqueharna 2	10/12/86	0	The RHRS isolated twice because of false signals a waterhammer occurred the second time.
397/86-021	WNPS 2	06/20/86	75	Leakage from the loop B SDC return-line isolatio valve was detected; the plant was shut down
413/86-044	Catawba l	08/15/86	0	The RHRS was isolated for 15 min when a fuse blew during maintenance, causing several valves t close

Table F.1 (continued)

F-5

Event identifier	Plant name	Event date	Reactor power (%)	Event description
416/86-034	Grand Gulf 1	10/14/86	0	A personnel error committed in maintenance caused an electrical ground, which closed the isolation valves
440/86-032	Perry 1	07/05/86	0	Several RHRS isolations (16 min each) occurred as a result of communication failures among personnel
440/86-048	Perry 1	08/19/86	0	Several RHRS isolations of several minutes' duration occurred because of procedural errors during testing
440/86-088	Perry 1	12/09/86	0	During a test, an operator error caused the RHRS to become isolated
458/86-024	River Bend 1	03/25/86	0	RPS bus B tripped off spuriously, causing operating RHR train A to isolate
458/86-025	River Bend 1	03/26/86	0	During testing, a tech- nician's error caused a short and an RHRS isola-
458/86-064	River Bend 1	10/31/86	0	tion An operator performing maintenance accidently grounded a power supply, isolating the RHRS

Table F.1 (continued)

Event identifier	Plant name	Event date	Reactor power (%)	Event description
249/86-014	Dresden 3	08/26/86	18	HPCI was found to be de- graded for 6.5 d because of a condenser hotwell drain failure
254/86-034	Quad-Cities 1	11/17/86	70	HPCI was removed from ser- vice to repair a leaking steam supply valve
263/86-002	Monticello	01/07/86	97	The pump flow controller failed in auto mode be- cause of bad contacts during routine testing
265/86-004	Quad-Cities 2	03/14/86	95	The HPCI turbine exhaust pressure switch failed, causing HPCI to isolate
277/86-016	Peach Bottom 2	07/09/86	100	During a test, the turbine control valve would not open beyond 75% of full open
298/86-017	Cooper	08/18/86	93	HPCI was made unavailable for 3 min due because of a personnel maintenance error in closing a valve
324/86-023	Brunswick 2	10/09/86	100	HPCI was inadvertently isolated as a result of electrical problems that occurr during reactor- water-cleanup system testing
331/86-022	Duane Arnold	10/22/86	87	In annual testing, the torus supply stop valve motor was found failed, making causing HPCI unavailable
331/86-024	Duane Arnold	11/21/86	94	A false signal caused by an electrical short circuit isolated HPCI
333/86-021	Fitzpatrick	12/23/86	82	A fire line valve cracked spraying water onto breakers and the battery motor control center, causing the HPCI unavailability
341/86-029	Fermi 2	08/23/86		While the reactor was at 4% power, HPCI was foun to have been essentiall unavailable for 7 d be- cause of sensor problem

Table F.2. Unavailabilities in BWR single-train safety systems reported in 1986

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Event identifier	Plant name	Event date	Reactor power (%)	Event description
341/86-041	Fermi 2	07/22/86	2	During maintenance activities, HPCI was made unavailable when an error caused lube oil cooling to be lost
352/86-050	Limerick 1	10/17/86	100	The steam supply outboard isolation valve was found with a packing failure,
352/86-052	Limerick l	11/10/86	100	making HPCI unavailable The inboard steam isola- tion valve closed on a false signal caused by an
354/86-051	Hope Creek 1	10/23/86	70	instrument technician HPCI had a degraded start- up time in testing caused by hydraulic system prob- lems
366/86-014	Hatch 2	07/17/86	84	The HPCI flow was found to have been degraded for an
388/86-002	Susquehanna 2	01/22/86	66	<pre>indefinite time (<30 d) HPCI was removed from ser- vice to repair a ruptured (5-gal/min) lube oil cooler water pressure-</pre>
245/86-010	Millstone 1	03/26/86	100	control valve During a test, the isola- tion condenser condensate return valve was found to be closed for 48 h
254/86-023	Quad-Cities 1	05/05/86	96	RCIC turbine tripped on overspeed as a result of mechanical problems with
254/86-027	Quad-Cities 1	09/05/86	1	the trip linkage RCIC was made unavailable for 10 min when the pump isolation valve failed to
254/86-028	Quad-Cities 1	10/03/86	79	open on demand RCIC turbine trip throttle valve spuriously closed for 20 min
278/86-024	Peach Bottom 3	11/08/86	45	A loose wire caused a false signal, resulting in RCIC isolation
331/86-007	Duane Arnold	03/15/86	0	A spurious sign l isolated RCIC for a very brief time

Table F.2 (continued)

Event identifier	Plant name	Event date	Reactor power (%)	Event description
331/86-023	Duane Arnold	11/04/86	98	During a test, the pump flow meter failed, thus requiring RCIC removal
				from service for repairs
333/86-005	Fitzpatrick	04/04/86	100	A false signal caused RCIC to momentarily isolate and its turbine to trip
333/86-015	Fitzpatrick	09/04/86	100	RCIC isolated on a spuri- ous high steam flow signal
341/86-048	Fermi -	12/26/86	5	Maintenance personnel failed to notify the control room when remov- ing RCIC from service
354/86-082	Hope Creek 1	10/28/86	70	RCIC was taken out of service for valve packing repairs
354/86-082	Hope Creek 1	10/28/86	70	RCIC was removed from ser- vice to repack a leaking valve
366/86-017	Hatch 2	09/15/86	87	The RCIC inboard primary isolation valve closed on a spurious signal
374/86-002	LaSalle 2	02/06/86	99	A loose torque switch caused the steam-line outboard isolation valve to behave erratically
397/86-031	WNPS 2	09/10/86	100	Technicians input a signal to the wrong circuit, causing the RCIC inboard steam valve to close
440/86-066	Perry 1	10/06/86	2	The RCIC isolated on a spurious signal for 25 min
458/86-016	River Bend 1	01/30/86	23	The steam-line isolation valve closed on a false signal
458/86-018	River Bend 1	02/07/86	34	A false signal caused RCIC to isolate
458/86-067	River Bend 1	12/10/86	, 100	During a test, an instru- ment error caused the RCIC/RHR steam-supply line to isolate
458/86-06	8 River Bend 1	12/23/86	5 100	The steam line high flow transmitter failed, caus ing RCIC isolation

Table F.2 (continued)

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Table F.3.	1986 accident sequence precursor uncomplicated trip events	
	reviewed (docket/LER)	

029/86-012	265/86-012	298/86-022	323/86-021	362/86-010
155/86-005 206/86-008	266/86-003	301/86-001	323/86-023	362/86-013 364/86-007
213/86-027	266/86-005	302/86-001	323/86-spr	366/86-009
213/86-041	269/86-008 270/86-001	304/86-011	324/86-017	366/86-010
213/86-044	270/86-002	304/86-016 305/86-007	324/86-020	366/86-022
219/86-004	270/86-004	305/86-008	325/86-009 325/86-010	366/86-023
237/86-001	270/86-005	306/86-001	325/86-021	368/86-001
237/86-009	272/86-001	306/86-002	325/86-024	368/86-004 368/86-007
237/86-017	272/86-003	306/86-003	325/86-029	368/86-011
237/86-019 244/86-004	272/86-012	309/86-001	333/86-010	369/86-002
244/86-005	272/86-013 272/86-016	309/86-002	333/86-013	369/86-007
244/86-008	275/86-010	309/86-003 309/86-005	334/86-001	370/86-001
244/86-011	276/86-006	309/86-006	334/86-012 335/86-002	370/86-016
244/86-018	277/86-001	309/86-007	335/86-002	370/86-017
244/86-SPR1	277/86-013	309-spr	335/86-009	370/86-021 373/86-043
244/86-SPR2	277/86-014	311/86-002	336/86-002	374/86-004
245/86-005	277/86-022	311/86-004	336/86-004	374/86-008
245/80-027 247/86-001	278/86-012	311/86-006	336/86-005	382/36-001
247/86-021	278/86-016 278/86-013	311/86-007	336/86-006	382/86-002
247/86-024	278/86-019	311/86-009 311/88-014	336/86-022	382/86-009
247/86-027	278/86-020	313/86-004	338/86-002 338/86-006	382/86-013
247/86-031	278/86-022	313/86-008	338/86-008	382/86-019 382/86-023
247/86-036	280/86-001	315/86-012	338/86-009	388/86-004
247/86-037	280/86-002	315/86-015	338/86-015	388/86-010
249/86-012 249/86-016	280/86-003	315/86-017	339/86-005	389/86-001
249/86-019	280/86-005 280/86-025	315/86-023	339/86-008	389/86-002
249/36-021	281/86-003	316/86-005 316/86-024	339/86-009	389/86-013
249/86-025	281/86-005	317/86-001	341/86-035 344/86-007	395/86-006
250/86-006	281/86-007	317/86-004	344/86-008	395/86-009 395/86-011
250/86-021	282/86-010	317/86-006	344/86-011	395/86-014
250/86-030	285/86-004	318/86-004	346/86-001	397/86-003
250/86-032 250/86-034	286/86-001	318/86-005	346/86-043	397/86-020
251/86-019	286/86-003	318/86-006	348/86-004	397/86-023
251/86-023	286/86-005 286/86-006	318/86-007	348/86-007	397/86-025
251/86-025	286/86-010	318-sor 321/86-018	348/86-008	397/86-026
254/86-026	286/86-011	321/86-023	352/86-011 354/86-065	397/86-030
255/86-015	286/86-012	321/86-030	354/86-080	409/86-001 409/86-006
261/86-002	287/86-001	321/86-043	354/86-085	409/86-018
261/86-003	289/86-002	323/86-001	361/86-015	409/86-019
261/86-004	289/86-006	323/86-005	361/86-018	409/86-020
261/86-011 261/86-012	289/86-011	323/86-007	361/86-019	409/86-021
261/86-013	293/86-002 293/86-009	323/86-008	361/86-022	409/86-029
261/86-014	295/86-012	323/86-011 323/86-012	361/86-027	409/86-036
263/86-025	298/86-006	323/86-012	361/86-029 362/86-001	413/86-006
265/86-001	298/86-016	323/86-020	362/86-001	413/86-022 413/86-025
				110/00-020

Table F.3. (continued)

$\begin{array}{rrrrrrrrrrrrrrrrrrrrrrrrrrrrrrrrrrrr$	458/86-037 458/86-039 458/86-041 458/86-044 458/86-045 458/86-055 458/86-056 458/86-069 458/86-SPR 458/86-SPR 482/86-018	483/86-018 483/86-019 483/86-022 483/86-027 483/86-029 528/86-003 528/86-018 528/86-018 528/86-024 528/86-033 528/86-042 528/86-044 528/86-045	528/86-047 528/86-053 528/86-056 528/86-063 528/86-017 529/86-026 529/86-027 529/86-033 529/86-034 529/86-047 529/86-049
--	--	--	--

Table F.4. 1986 accident sequence precursor trip events reviewed involving initiators or additional system failures (docket/LER)

> 247/86-017 247/86-035 250/86-039 261/86-005 269/86-001 277/86-003 285/86-001 293/86-027 318/86-006 409/86-023 413/86-031 414/86-028 458/86-002

Table F.5. 1986 accident sequence precursor trip events followed by an uncomplicated LOFW (docket/LER)

029/86-004 247/86-019 249/86-017 254/86-030 263/86-024 270/86-006 272/86-010 281/86-020 281/86-026 304/86-014 341/86-040 361/86-004 362/86-010 364/86-001 366/86-012 369/86-018* 370/86-020* 395/86-002 397/86-038 409/86-035 414/86-014 414/86-034 414/86-051 414/86-053 423/86-012* 423/86-015 458/86-001 482/86-038 482/86-069 482/86-070*

*LOFW without trip, generally from very low power.

Table F.6. 1986 accident sequence precursor uncomplicated LOFW* and LOFW events followed by a trip (docket/LER)

```
249/86-013
250/86-036
250/86-038
269/86-011
280/86-029
280/86-031
381/86-010
282/86-006
282/86-011
301/86-004
341/86-045
341/86-048
362/86-011
366/86-035
370/86-006
389/86-011
458/86-047
```

*LOFW without trip, generally from very low power.

Table F.7. 1986 accident sequence precursor unavailability events not involving a trip (docket/LER)

> 287/86-002 318/86-002 344/86-001 346/86-043 348/86-015 369/86-018 370/86-012 413/86-008 482/86-007 482/86-037 483/86-013 483/86-022 483/86-030 528/86-020 528/86-061 529/86-023

	for selection	as precursors	(docket/LER)	
029/86-004	251/86-025	000 100 000	1	
029/86-012	254/86-026	280/86-005	309/86-003	325/86-029
155/86-005	254/86-030	280/86-025	309/86-005	333/86-010
206/86-008	255/86-015	280/86-029	309/86-006	333/86-013
213/86-027	261/86-002	280/86-031	309/86.007	334/86-001
213/86-041	261/86-003	281/86-003	311/86-002	334/86-012
213/86-044	261/86-004	281/86-005	311/86-004	335/86-002
219/86-004	261/86-005	281/86-007 281/86-010	311/86-006	335/86-004
237/86-001	261/86-011	281/86-020	311/86-007	335/86-009
237/86-009	261/86-012	281/86-026	311/86-009	336/86-002
237/86-017	261/86-013	282/86-006	311/86-014	336/86-004
237/86-019	261/86-014	282/86-010	313/86-004	336/86-005
244/86-004	263/86-024	282/86-011	313/86-008 315/86-012	336/86-006
244/86-005	263/86-025	285/86-001	315/86-015	336/86-022
244/86-008	265/86-001	285/86-004	315/86-017	338/86-002
244/86-011	265/86-012	286/86-001	315/36-023	338/86-006
244/86-018	266/86-003	286/86-003	316/86-005	338/86-008
244/86-SPR1	266/86-005	286/86-005	316/86-024	338/86-009 338/86-015
244/86-SPR2	269/86-001	286/86-006	317/86-001	339/86-005
245/86-005	269/86-008	286/86-010	317/86-004	339/86-005
245/86-027	269/86-011	286/86-011	317/86-006	339/86-009
247/86-001	270/86-001	286/86-012	318-spr	341/86-035
247/86-017	270/86-002	287/86-001	318/86-002	341/86-040
247/86-019	270/86-004	287/86-002	318/86-004	341/86-045
247/86-021	270/86-005	289/86-002	318/86-005	341/86-048
247/86-024	270/86-006	289/86-006	318/86-006	344/86-001
247/86-027	272/86-001	289/86-011	318/86-007	344/86-007
247/86-031	272/86-003	293/86-002	321/86-018	344/86-008
247/86-035	272/86-010	293/86-009	321/86-023	344/86-011
247/86-036	272/86-012	293/86-027	321/86-030	346/86-061
247/86-037	272/86-013	295/86-012	321/86-043	346/86-043
249/86-012	272/86-016	298/86-006	323/86-001	348/86-004
249/86-013	275/86-010	298/86-016	323/86-005	348/86-007
249/86-016	276/86-006	298/86-022	323/86-007	348/86-008
249/86-017	277/86-001	301/86-001	323/86-008	348/86-015
249/86-019	277/86-003	301/86-004	323/86-011	352/86-011
249/86-021	277/86-013	302/86-001	323/86-012	354/86-065
249/86-025	277/86-014	304/86-011	323/86-016	354/86-080
250/86-006	277/86-022	304/86-014	323/86-020	354/86-085
250/86-021	278/86-012	304/86-016	323/86-021	361/86-004
250/86-030	278/86-016	305/86-007	323/86-023	361/86-015
250/86-032	278/86-018	305/86-008	323/86-spr	361/86-018
250/86-034	278/86-019	306/86-001	324/86-017	361/86-019
250/86-036	278/86-020	306/86-002	324/86-020	361/86-022
250/86-038	278/86-022	306/36-003	325/86-009	361/86-027
250/86-039	280/86-001	309-spr	315/86-010	361/86-029
251/86-019	280/86-002	309/86-001	325/86-021	362/86-001
251/86-023	280/86-003	309/86-002	325/88-024	362/86-005

Table F.8. All 1986 accident sequence precursor events reviewed* for selection as precursors (docket/LER)

*Detailed review.

362/86-010	397/86-023	458/86-001	529/86-026
362/86-011	397/86-025	458/86-002	529/86-027
362/86-013	397/86-026	458/86-007	529/86-033
364/86-001	397/86-030	458/86-019	529/86-034
364/86-007	397/86-038	458/86-022	529/86-047
366/86-009	409/86-001	458/86-032	529/86-049
366/86-010	409/86-006	458/86-033	
366/86-012	409/86-018	458/86-035 458/86-037	
366/86-022	409/86-019	458/86-039	
366/85-023	409/86-020 409/86-021	458/86-041	
366/86-035	409/86-023	458/86-044	
368/86-001 368/86-004	409/86-029	458/86-045	
368/86-007	409/86-035	458/86-047	
368/86-011	409/86-036	458/86-055	
369/86-002	413/86-006	458/86-056	
369/36-007	413/86-008	458/86-069	
369/86-018	413/86-022	458/86-SPR	
369/86-018	413/86-025	482/86-007	
370/86-001	413/83-026	482/86-018	
370/86-006	413/86-030	482/86-037	
370/86-012	413/86-031	482/86-038	
370/86-016	413/86-040	482/86-069	
370/86-017	413/86-042	482/86-070	
370/86-020	414/86-014	483/86-013	
370/86-021	414/86-015	483/86-018	
373/86-043	414/86-028	483/86-019	
374/86-004	414/86-030	483/86-022 483/86-022	
374/86-008	414/86-034 414/86-051	483/86-027	
382/86-001	414/86-053	483/86-029	
382/86-002	416/86-003	483/86-030	
382/86-009 382/86-013	416/86-004	528/86-003	
382/86-019	416/86-011	528/86-018	
382/86-023	416/86-025	528/86-020	
388/86-004	416/86-028	528/86-024	
388/86-010	423/86-012	528/86-033	
389/86-001	123/86-015	528/86-042	
389/86-002	423/86-035	528/86-044	
389/86-011	423/86-041	528/86-045	
389/86-013	423/86-048	528/86-047	
395/86-002	423/86-049	528/86-053	
395/86-006	423/86-051	528/86-056	
395/86-009	454/86-001	528/86-061	
395/86-011	454/86-003	528/86-063	
395/86-014	454/86-008	528/86-spr	
397/86-003	454/86-027	529/86-017 529/86-023	
397/86-020	456/86-004	020100-020	

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Table F.8. (continued)

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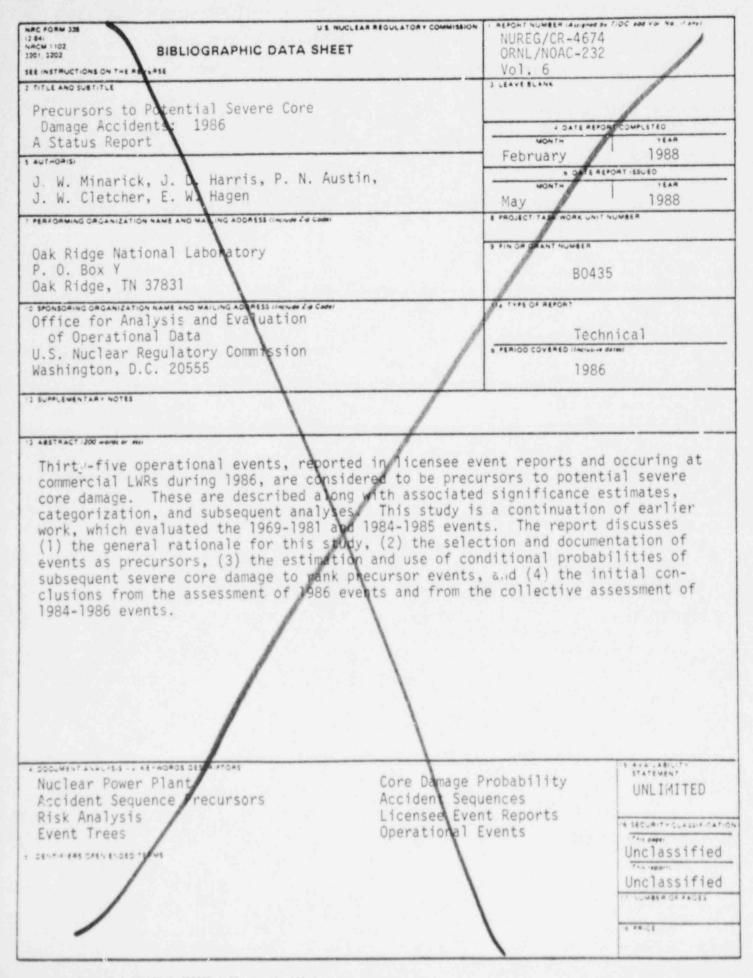
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