June 2, 1993

All Simulator Instructors

Subject: List of "NOT AUTHORIZED FOR USE" Malfunctions & Snaps

As of June 2, 1993, this list designates the malfunctions and snaps that are currently "NOT AUTHORIZED FOR USE". This list supersedes any other source as to which malfunctions and snaps are currently unavailable for training.

The "NOT AUTHORIZED FOR USE" malfunctions are:

MALFUNCTION	DESCRIPTION
SI06	SI accum check vlv leak
TH19	Rx vessel bottom crack

SPECIAL NOTES:

NONE

Respectfully,

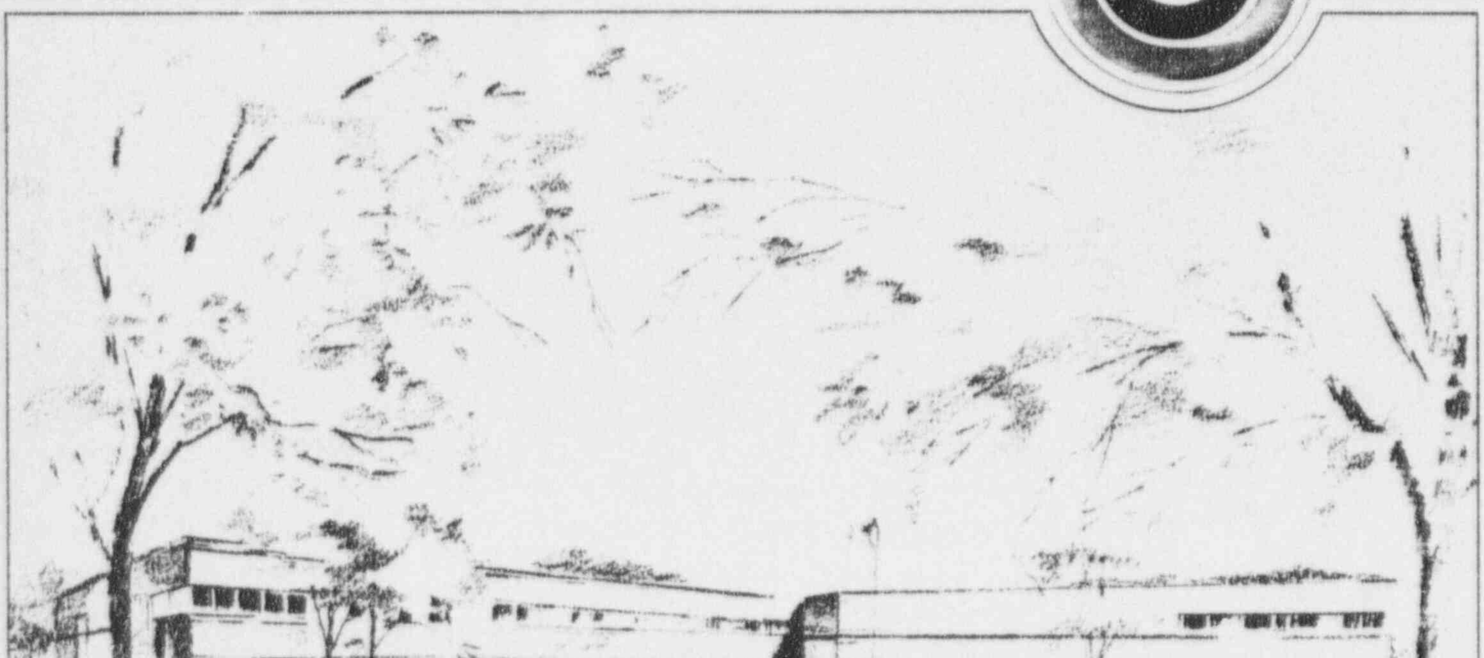
Thomas M Chasensky

Thomas M. Chasensky
Braidwood Simulator
Training Supervisor

cc: File: SIM-BW-1C3-93

COMMONWEALTH EDISON BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS



SIMULATOR vs. BRAIDWOOD UNIT 1 DIFFERENCES

BRWD. UNIT 1	SIMULATOR
1. Status lights are push-to-test.	Some status lights are not push-to-test.
2. Rod step counters are mechanical.	Rod step counters are digital.
3. Meter needles are black.	Meter needles are orange.
4. 0PM02J (HVAC panel) has a clock.	0PM02J (HVAC panel) does not have a clock.
5. Boric acid and primary water totalizers are mechanical counters.	Boric acid and primary water totalizers are digital counters.
6. 0PM02J (HVAC panel) is a full scale panel.	0PM02J U-2 fans are located on a mini- panel.
7. Sound powered phone jacks are installed on the control boards.	Sound powered phone jacks are not installed on the control boards.
8. Color banding is located on meter faces.	Color banding is located on meter scales.
9. Screws are used on nametags and mimics.	Screws are not used on nametags and mimics.
10. Calibration stickers are used on some meter faces.	Calibration stickers are not used on meter faces.
11. Manufacturer's name is on control board components.	Manufacturer's name may be missing on control board components.

SIMULATOR vs. BRAIDWOOD UNIT 1 DIFFERENCES

BRWD. UNIT 1	SIMULATOR
12. Following components are functional:	Following components are not modeled:
-TGTMS -RM-23's -Fire Detection Panel	-TGTMS -RM-23's -Fire Detection Panel

13. The following problems are awaiting correction:

    -IPC response is slow.

    -BA flow is not accurately recorded on the BA totalizer if CV-110A & B are open together. This will cause more boric acid to reach the core than is seen on the totalizer (approx. 2-1 ratio).

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

AN01 LOSS OF ANNUNCIATOR HORN

AN01 LOSS OF ANNUNCIATOR HORN

TYPE: GENERIC, RB

A) ANNUNCIATOR HORN 1AN02J (0PM01J) (Box 37/38)
B) ANNUNCIATOR HORN 1AN04J (0PM02J) (Box 31/33/34)
C) ANNUNCIATOR HORN 1AN05J (0PM03J) (Box 35)
D) ANNUNCIATOR HORN 1AN06J (1PM01J)
E) ANNUNCIATOR HORN 1AN07J (1PM02J/03J)
F) ANNUNCIATOR HORN 1AN08J (1PM05J)
G) ANNUNCIATOR HORN 1AN09J (1PM06J)
H) ANNUNCIATOR HORN 1AN17J (1PM04J) (Includes Box 14)
I) ANNUNCIATOR HORN RM11 (1PM14J; RM-11)
J) NOT USED
K) NOT USED
L) ANNUNCIATOR HORN 2AN05J (0PM03J) (Box 36)
M) ANNUNCIATOR HORN 2AN08J (1PM05J) (Bypass Perm. Panel)

CAUSE: LOOSE WIRE AT HORN

REF: 20E-1-4030 AN116
20E-1-4030 AN117

PLT STA: AS APPROPRIATE

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED ANNUNCIATOR HORN WILL NOT ACTIVATE AS REQUIRED FROM ANY ANNUNCIATOR SIGNAL.

MALFUNCTION REMOVAL WILL RESTORE THE ANNUNCIATOR HORN TO NORMAL.

EVENTS: NONE

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

CC01	CCW PUMP FAILS TO START/TRIP
CC02	CCW PUMP DISCH PRESS SWITCH FAILURE
CC03	CCW SURGE TANK LEVEL TRANSMITTER FAILURE
CC04	CCW FROM RHR HX LEAK
CC05	CCW TO CC HX PIPING BREAK
CC06	NON-ESSENTIAL CCW SYSTEM LEAK
CC07	RCP THERMAL BARRIER LEAK
CC08	CCW HX TUBE LEAK
CC09	THERMAL BARRIER CCW FLOW X-MITTER FAILURE

CC01 CCW PUMP FAILS TO START/TRIP

TYPE: GENERIC, RB

A) U-0 CC PUMP 0CC01P
B) 1A CC PUMP 1CC01PA
C) 1B CC PUMP 1CC01PB

CAUSE: FAULTY OVERCURRENT (450/451) RELAY

REF: 20E-0-4030 CC01
20E-0-4030 CC02
20E-1-4030 CC01
20E-1-4030 CC02
20E-1-4030 CC11
PLS

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THE SELECTED COMPONENT COOLING WATER PUMP BREAKER WILL OPEN. PUMP CURRENT INDICATION DECREASES TO ZERO, ANNUNCIATOR 2-A4 "CC PUMP TRIP" ACTUATES, AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. COMPONENT COOLING PUMP DISCHARGE PRESSURE DECREASES RESULTING IN THE AUTOMATIC START OF THE STANDBY COMPONENT COOLING WATER PUMP AT 85 PSIG DISCHARGE PRESSURE RESTORING SYSTEM PRESSURE. ANNUNCIATORS 2-B4 "CC PUMP AUTO START" AND 2-B5 "CC PUMP DSCH PRESS LOW" ACTUATE.

THE OPERATOR MAY RESET THE AMBER LAMP BY PLACING THE CONTROL SWITCH IN STOP. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL CLOSE, THEN IMMEDIATELY TRIP OPEN.

MALFUNCTION REMOVAL WILL RESTORE THE OVERCURRENT RELAY TO NORMAL OPERATION.

EVENTS: NONE

CC02 CCW PUMP DISCH PRESS SWITCH FAILURE

TYPE: GENERIC, RV 0-200 PSIG

A) 1PS-CC673A
B) 1PS-CC673B

CAUSE: PRESSURE SWITCH FAILURE

REF: 20E-0-4030 CC01, 02
20E-1-4030 CC01, 02, 11

PLT STA: CC SYSTEM IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED CCW DISCH PRESS TRANSMITTER TO FAIL TO A VALUE DEPENDENT UPON THE SELECTED SEVERITY. THIS MALFUNCTION WILL NOT AFFECT THE MCB INDICATION (1PI-CC107) EXCEPT AS STATED BELOW. IF THE SEVERITY LEVEL SELECTED IS GREATER THAN THE 85 PSIG SETPOINT, THE LOW PRESSURE AUTO START FOR THE ASSOCIATED CC PUMP WILL BE DISABLED AND THE PRESSURE SWITCH INPUT TO THE ANNUNCIATOR SYSTEM WILL BE DISABLED.

IF THE SEVERITY LEVEL SELECTED IS LESS THAN THE 85 PSIG SETPOINT, ANNUNCIATORS 2-B5 "CC PUMP DISCH PRESS LOW", AND 2-B4 "CC PUMP AUTO START" ACTUATE AND THE STANDBY CC PUMP WILL AUTO START. CC SYSTEM PRESSURE WILL INCREASE ABOVE INITIAL SYSTEM PRESSURE AS INDICATED ON 1PI-CC107.

MALFUNCTION REMOVAL WILL RESTORE THE CC PUMP DISCH PRESSURE SWITCH TO NORMAL.

EVENTS: 1) DVR 06-02-89-018

2C02

DEVIATION INVESTIGATION REPORT (DIR)

Facility Name	Form Rev 2.0
Byron Nuclear Power Station	PAGE 1 OF 0 1
Title: 2A COMPONENT COOLING PUMP AUTO-START	

EVENT DATE MONTH	DAY	YEAR	DIR NUMBER STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE MONTH	DAY	YEAR	OPERATING MODE	POWER LEVEL
0 2	0 1	8 9	0 6	0 2	8 9	0 1 8	0 0	0 3	1 5	8 9	6	0

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER
W. Walter, Assistant Tech Staff Supervisor Ext. 2244	AREA CODE 8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 02/01/89 / 0050

Unit 1 MODE 3 - Hot Standby Rx Power 0% RCS [AB] Temperature/Pressure Prep for Startup

Unit 2 MODE 6 - Refueling Rx Power 0% RCS [AB] Temperature/Pressure 85°F / 0.3 PSIG

B. DESCRIPTION OF EVENT:

On February 1, 1989, at 0050, the 2A Component Cooling (CC)[CC] Pump Auto-Started after receiving a "LOW DSCH PRESS" alarm. Upon investigation no indication of low discharge header pressure was found, and the 2B pump remained running. There was sufficient flow path to accommodate the increased flow.

It is common practice to have two pumps running in the refueling mode. The unit was running the 2B pump alone to support a CC modification test that required the 2A CC pump and the 0 CC pump to be racked-in to test. With the test complete, Operating returned the 2A Pump to service and was preparing to start it when the Auto-Start occurred. There was no instability as a result of this Auto-Start event. There were no ESF actuations as a result of, or during this event. Operator actions taken were prompt and correct.

C. CAUSE OF EVENT:

The cause of the event was a pressure leak-off between valve 2CC019 (U-2 CC PP DSCH HDR 2PS-673A ISOL VLV) and pressure switch 2PS-673A. Valve 2CC019 was taken Out of Service (OOS) at 0353, January 31, 1989, in support of a modification that would install additional pressure indicating instrumentation to the CC System. Construction had not cut into the line between the root valve and the pressure switch at the time of the event. The leak-off of approximately 50 PSIG (from 135 to 85 PSIG) therefore occurred during the 21 hour period from isolation of 2CC019 to the Auto-Start Event.

CC03 CCW SURGE TANK LEVEL TRANSMITTER FAILURE

TYPE: GENERIC, RV 0-100% OF TANK LEVEL

A) 1LT-CC670
B) 1LT-CC676

CAUSE: LEVEL TRANSMITTER FAILURE

REF: 20E-1-4030 CC01, 02, 11, 15
20E-1-4031 CC01, 02

PLT STA: CCW SYSTEM IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED CCW SURGE TANK LEVEL TRANSMITTER TO FAIL TO A VALUE DEPENDENT UPON THE SELECTED SEVERITY AND WILL BE INDICATED ON THE ASSOCIATED MCB LEVEL METER (1LI-CC670 OR 676). IF THE SEVERITY LEVEL IS GREATER THAN THE HIGH SETPOINT OF 65%, ANNUNCIATOR 2-A5 "CC SURGE TANK LEVEL HIGH LOW " WILL ACTUATE. IF THE SEVERITY LEVEL IS LESS THAN 35% , ANNUNCIATOR 2-A5 "CC SURGE TANK LEVEL HIGH LOW " WILL ACTUATE.

FOR LT-CC676: IF THE SEVERITY LEVEL IS GREATER THAN 55%, THE DEMIN WATER M/U VALVE 1CC-183 WILL AUTO CLOSE IF OPEN. IF THE SEVERITY LEVEL IS LESS THAN 50%, ANNUNCIATOR 2-E4 "CC SURGE TANK AUTO-M/U ON" ACTUATES AND THE DEMIN WATER M/U VALVE 1CC-183 AUTO OPENS. APPROXIMATELY 200 GPM DEMIN WATER M/U WILL CAUSE CC SURGE TANK ACTUAL LEVEL TO INCREASE AS INDICATED ON 1LI-CC670.

FOR LT-CC670: IF THE SEVERITY LEVEL IS GREATER THAN 55%, THE PRIMARY WATER M/U VALVE 1CC-182 WILL AUTO CLOSE IF OPEN. IF THE SEVERITY LEVEL IS LESS THAN 45%, ANNUNCIATOR 2-E4 "CC SURGE TANK AUTO M/U ON" ACTUATES AND THE PRIMARY WATER M/U VALVE 1CC-182 AUTO OPENS. APPROXIMATELY 200 GPM PRIMARY WATER M/U WILL CAUSE CC SURGE TANK ACTUAL LEVEL TO INCREASE AS INDICATED ON 1LI-CC676.

FOR BOTH LT-CC670 AND 676: DECREASING THE SEVERITY LEVEL TO LESS THAN THE 13% SETPOINT WILL HAVE NO EFFECT ON THE CC PUMP AUTO TRIP CIRCUITS.

MALFUNCTION REMOVAL WILL RESTORE THE CC SURGE TANK LEVEL TRANSMITTERS TO NORMAL.

EVENTS: NONE

CC04 CCW FROM RHR HX LEAK

TYPE: GENERIC, RV 0-10000 GPM @ 200 PSID

A) 1A RHR HX (1RH02AA)
B) 1B RHR HX (1RH02AB)

CAUSE: PIPE BREAK DOWNSTREAM OF 1CC9412A/B

REF: M-62
M-66 SHEET 2
PLS

PLT STA: RHR SYSTEM IN OPERATION

EFFECTS: THIS MALFUNCTION RESULTS IN INCREASED COMPONENT COOLING WATER FLOW THROUGH THE SELECTED RHR HEAT EXCHANGER. AS MALFUNCTION SEVERITY IS INCREASED, COMPONENT COOLING FLOW THROUGH THE RHR HEAT EXCHANGER INCREASES AS INDICATED ON 1FI-0689 (1FI-0688). RHR HEAT EXCHANGER OUTLET TEMPERATURE DECREASES AS INDICATED ON 1TR-612 (1TR-613). COMPONENT COOLING SYSTEM TEMPERATURES INCREASE AS THE RHR HEAT LOAD IS RAISED. ANNUNCIATOR 2-A6 "RH HX CC WTR FLOW HIGH LOW" ACTUATES. CC SURGE TANK LEVEL DECREASES AS MASS IS LOST.

MALFUNCTION REMOVAL WILL RESTORE THE PIPING TO NORMAL.

EVENTS: NONE

CC05 CCW TO CC HX PIPING BREAK

TYPE DISCRETE, RV 0-2000 GPM @ 100 PSID

CAUSE: PIPING BREAK IMMEDIATELY UPSTREAM OF 1CC9470B

REF: M-66 SHEET 3B
PLS

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THIS MALFUNCTION RESULTS IN A LOSS OF MASS FROM THE COMPONENT COOLING WATER SYSTEM. CCW SURGE TANK LEVEL WILL DECREASE AT A RATE DETERMINED BY THE SELECTED MALFUNCTION SEVERITY. AT 50% SURGE TANK LVL ANNUNCIATOR 2-E4 "CC SURGE TANK AUTO-M/U ON" ACTUATES AND THE DEMIN WATER M/U VALVE 1CC-183 AUTO OPENS. APPROXIMATELY 200 GPM OF DEMIN WATER M/U IS SUPPLIED TO THE SURGE TANK. AT 45% LVL THE PRIMARY WATER M/U VALVE 1CC-182 AUTO OPENS. APPROXIMATELY 200 ADDITIONAL GPM OF PRIMARY WATER M/U IS SUPPLIED TO THE SURGE TANK. ANNUNCIATOR 2-A5 "CC SURGE TANK LEVEL HIGH LOW" ACTUATES WHEN LEVEL DECREASES TO 35%.

AS MALFUNCTION SEVERITY IS INCREASED BEYOND M/U CAPACITY, COMPONENT COOLING WATER SURGE TANK LEVEL DECREASES, SYSTEM PRESSURE DECREASES, AND PUMP FLOW INCREASES. ANNUNCIATORS 2-B4 "CC PUMP AUTO START" AND 2-B5 "CC PUMP DSCH PRESS LOW" ACTUATE, AND THE STANDBY CCW PUMP AUTO STARTS AT 85 PSIG. AT SURGE TANK LEVEL OF 13% THE CCW PUMPS TRIP.

ALL COMPONENTS COOLED BY CCW WILL INCREASE IN TEMPERATURE.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE COMPONENT COOLING SYSTEM PIPING INTEGRITY.

EVENTS: 1) LER 06-01-87-012
2) LER 20-01-87-011
3) DVR 20-02-89-019

CC05

Facility Name (1) Byron, Unit 1

Docket Number (2) 0|5|0|0|0|4|5|4|0 Page (3) 1 of 3

Title (4) TWO TRAINS OF SAFETY RELATED COMPONENT COOLING INOPERABLE DUE TO LOSS OF WATER INVENTORY CAUSED BY PERSONNEL ERROR

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0|4	0|8	8|7	8|7	--- 0|1|2 ---	0|0	0|5	0|8	8|7	NONE	0|5|0|0|0| |
										0|5|0|0|0| |

OPERATING MODE (9): 6

POWER LEVEL (10): 0|0|0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	__ 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	X 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name: T. Schuster, Assistant Technical Staff Supervisor Ext. 2265

TELEPHONE NUMBER: AREA CODE 8|1|5 2|3|4|-|5|4|4|1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

__ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15) Month | Day | Year

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

On April 8, 1987, at approximately 1725, a contracted maintenance crew began work on the Limitorque motor operator of the "1A" Residual Heat Removal (RH) Heat Exchanger Component Cooling Water Outlet Isolation Valve, 1CC9412A. This valve was a point of isolation for work on the RH Heat Exchanger, which required it to be drained of Component Cooling Water (CC). Shift Operating personnel granted permission, with the understanding that if it became necessary for the crew to stroke the valve, they would obtain authorization. The maintenance crew stroked the valve in order to release torque on the motor gear set. It is unclear whether they actually received authorization or not. This allowed Component Cooling Water to back flow through 1CC9412A to the Heat Exchanger and out the drain. This caused the (CC) surge tank to reach the low level CC Pump Trip. The "1A" CC Pump tripped at 1726 on April 8, 1987. The surge tank is common to both trains, consequently, both trains of Component Cooling were inoperable. The leak was discovered and isolated. The system was then re-filled, and the "1A" CC Pump re-started. Total time both trains were inoperable was 17 minutes. The cause of the event was a communication breakdown between the maintenance crew and Shift Operating personnel. Corrective actions will require the contracted maintenance crew to obtain written authorization prior to manipulating a valve for work on the valve's operator. In addition, a modification has been initiated to provide automatic makeup water to the Component Cooling system in the event of a leak. The safety significance was minimal. RCS Temperature never exceeded 85°F. There was one similar previous occurrence reported in LER 455/86-01.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year		Sequential Number		Revision Number	PAGE (3)		
Byron, Unit 1	0 5 0 0 0 4 5 4	8 7	-	0 1 2	-	0 0	2	OF	4

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

Byron Unit 1 Event Date/Time 04/08/87 / 1725

MODE 6 - Refueling Rx Power 0% RCS [AB] Temperature/Pressure 85°F /de-pressurized

B. DESCRIPTION OF EVENT:

The "1A" Residual Heat Removal (RH)[BP] Heat Exchanger was Out of Service (OOS) for gasket replacement. The shell side, consisting of Component Cooling water (CC)[CC] was isolated and drained. One of the points of isolation of the OOS for CC was the RH Heat Exchanger CC Outlet Isolation Valve, 1CC9412A.

The grease in Limitorque valve motor operators was scheduled to be changed out during the refueling outage. This activity was being handled by contracted maintenance personnel, supervised by utility management. Since this involved numerous Limitorques, the Work Supervisor developed a plan with Operating Management that each valve would be only taken Out of Service electrically for personnel protection. If it became necessary to mechanically stroke the valve the maintenance crew foreman would ask the Shift Engineer for specific authorization. This plan was consistent with Station's work practices and programs.

At approximately 0842, on April 6, 1987, the Work Supervisor (utility non-licensed) for the RH Heat Exchanger gasket replacement requested a temporary lift of the mechanical portion of the Out of Service on 1CC9412A in order to perform the grease change and gear box flush on the motor operator of 1CC9412A. The Operating Shift Foreman (licensed) granted permission with the explicit agreement that work was only to be performed on the motor and that the valve was not to be stroked open for any reason.

At approximately 1725, on April 8, 1987, the contracted maintenance crew (non-licensed) began work on the valve. During the course of their work it became necessary to release the torque on the motor gear set which required stroking 1CC9412A approximately half open. They stroked the valve. This allowed Component Cooling water to back flow to the RH Heat Exchanger, fill the empty Heat Exchanger, and pass through the open drain valve. The CC surge tank level dropped to the low level CC Pump Trip setpoint. The "1A" CC Pump, which was running to support plant operations, tripped at 1726. The CC Surge Tank is common to both CC Trains, consequently both trains were inoperable at this time.

Shift Operations, in response to the "1A" CC Pump Trip and Low Surge Tank level, dispatched an operator to investigate. He quickly determined that CC was draining into and out of the RH Heat Exchanger. He then closed "1A" RH Heat Exchanger Component Cooling Outlet throttle valve, 1CC9507A, to isolate the leak. Water was then restored to the surge tank and the "1A CC" Pump re-started. The total time both trains of CC were inoperable was 17 minutes. There were no safety system actuations.

A Generating Station Emergency Plan Alert was declared and appropriate notifications made.

This report is required pursuant to 10CFR(a)(2)(vii).

(1370M/0164M)

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					PAGE (3)		
		Year		Sequential Number		Revision Number			
Byron, Unit 1	0 | 5 | 0 | 0 | 0 | 4 | 5 | 4	8|7	-	0|1|2	-	0|0	1|1	OF	1|1

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

C. CAUSE OF EVENT:

The root cause of this event was a communication breakdown between the contracted maintenance crew performing the work and Operating Shift personnel.

The contract maintenance personnel had been instructed to always receive permission from the Shift Engineer prior to manipulating the valve they are working on. The maintenance crew was interviewed and insist they did receive verbal permission to stroke 1CC9412A. However, they do not remember who they talked to. Shift operating personnel maintain that they never gave such permission. There was no requirement to document this permission in writing. Neither version could be corroborated.

D. SAFETY ANALYSIS:

Plant and Public safety were not affected. Loss of a heat sink for the Reactor Coolant System (RCS), without loss of circulation, has a negligible effect for the short period of time the loss occurred. Reactor coolant temperature was maintained at approximately 85 degrees Fahrenheit through-out the event and RCS forced re-circulation was maintained, via the operating RH train. The water level in the reactor cavity was greater than 23 feet, providing sufficient heat sink during the loss of Component Cooling. It would have provided sufficient heat sink and cooling for an extended period of time if RH flow had been lost.

E. CORRECTIVE ACTIONS:

Communications and proper work coordination between station maintenance personnel and Operating Shift personnel has been effective and does not warrant concern, consequently, corrective actions are focused on contracted maintenance personnel.

Contracted maintenance personnel have been re-informed of the requirement to obtain Shift Engineer authorization prior to stroking any valve they are working on during the Limitorque motor operator grease changeout. In addition, they are required to obtain this authorization in writing to document that Operating Management has given permission and is aware of the valve manipulation. This requirement will be extended to all similar work activities.

A modification to the Component Cooling System has been initiated to provide automatic makeup water to maintain surge tank level. This would attempt to maintain water inventory in the event of a leak. This is being tracked by a Action Item Record 6-87-113.

This report will be placed in the Licensed Operator required reading program. In addition, this report will be distributed to Station Departments to be disseminated to respective department personnel.

(1370M/0164M)

F. PREVIOUS OCCURRENCES:

LER NUMBER	TITLE
455-86-001	Both Trains of Component Cooling Inoperable Due to Personnel Error in a Relief Valve Setting.

G. COMPONENT FAILURE DATA:

a) MANUFACTURER NOMENCLATURE MODEL NUMBER MFG PART NUMBER

Not Applicable

b) RESULTS OF NPRDS SEARCH:

Not Applicable

LICENSEE EVENT REPORT (LER)

CCCS

Facility Name (1)	Docket Number (2)	Page (3)
Braidwood, Unit 1	0 5 0 0 0 4 5 6	1 of 0 2

Title (4) Loss of Residual Heat Removal due to loss of Component Cooling as result of a leaking Component Cooling Inlet Valve.

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0 1	2 1	8 7	8 7	0 1 1	0 0	0 2	2 0	8 7	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	X 50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	Other (Specify in Abstract below and in Text)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	Telephone Number	
	Area Code	
Howard James, Tech Staff Engineer, Ext. 2481	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

Cause	System	Component	Manufacturer	Reportable to NPRDS	Cause	System	Component	Manufacturer	Reportable to NPRDS
B	C C	I S V *	V 6 0 8	N					

SUPPLEMENTAL REPORT EXPECTED (14)

Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO	Expected Submission Date (15)	Month	Day	Year

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

The 1B Residual Heat Removal (RHR) Heat Exchanger (Hx) was out of service with the tube side drained. Preparations were made for draining the Component Cooling Water (CC) shellside of the Hx to allow replacement of a flange gasket. At 1745 draining of the shell side of the Hx was started. At 1802 the 1A CC pump tripped due to low level in the CC Surge Tank. The Low Level Alarm on the Main Control Board did not annunciate although the sequence of events recorder did indicate a low level. The draining was stopped, the Surge Tank refilled, and the Isolation Valves were checked. At 1816 the 1A CC pump was restarted and the system was restored to normal.

The cause was the CC Inlet Isolation Valve leaking and contributing was the failure of the CC Surge Tank High/Low Level Alarm to annunciate on the Main Control Board. Additionally, the CC Motor Operated outlet valve on the 1B RHR Hx was found 6 turns off its seat.

The leaking valve has been repaired, the limits for a Motor Operated Valve were adjusted, the Main Control Board alarm was troubleshot and the symptoms could not be duplicated. Work is in progress to check the calibration and scaling on the CC Surge Tank Instrument Loop.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Braidwood, Unit 1	0 5 0 0 0 4 5 6	8 7	- 0 1 1 -	0 0	0 2 OF 0 3

Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

Mode 5 - Cold Shutdown, Rx Power 0%, Reactor Coolant System [RB] Temperature/Pressure: 165°F/373 psig

B. Description of Event:

The 1B Residual Heat Removal (RHR) [BP] Heat Exchanger (Hx) was out of service with the tube side drained. Preparations were made for draining the Component Cooling Water (CC) [CC] shell side of the 1B RHR Hx. The A Train of RHR was in service and the 1C Reactor Coolant Pump (RCP) [AB] was running. The CC system was in its normal operating configuration with the 1A CC pump running and the 1B CC pump in standby.

At 1745 on January 21, 1987, draining of the shell side of the 1B RHR Hx was started by opening the shell side drain valve 1RH002B to allow replacement of the Hx flange gasket. The Unit 1 Nuclear Station Operator (NSO) verified that CC Surge Tank Level on the Main Control Board was not dropping. The Unit 1 NSO then went to the other side of the Control Room to perform an unrelated evolution.

At approximately 1751 the Sequence of Events Recorder (SER) indicated a low level on the A-side of the CC Surge Tank (setpoint 35%). The Main Control Board alarm, which receives the same signal that actuates the SER, did not annunciate.

At approximately 1755 the SER indicated a low level on the B side of the CC Surge Tank. Once again, the Main Control Board Alarm did not annunciate. The Unit 1 NSO had completed the unrelated evolution approximately one minute prior to this occurring.

At approximately 1802 the 1A CC pump tripped on Low Surge Tank Level (this comes from a separate level indicating switch, setpoint 13%). A low pressure signal was indicated on the SER and the Main Control Board as a result of the 1A CC pump tripping. This caused the 1B CC pump to auto start, however, the CC Surge Tank Level was less than 13% and tripped the pump. This occurred two more times before the pump was manually started at the direction of the Station Control Room Engineer (SCRE) who noted CC Surge Tank Level at 0% on the A-side and 20% on the B-side. The 1B CC pump tripped after 4 seconds. The SCRE had the control switches for both CC pumps placed in pull to lock and directed the NSO to stop the 1C RCP. Operating personnel immediately began refilling the Surge Tank and closed the 1B RHR Hx shell side drain valve. They also checked the CC Isolation Valves to the 1B RHR Hx and found the Inlet Manual Isolation Valve 1CC9504B valve fully closed and the Motor Operated Outlet Valve, MOV 1CC9412B valve 6 turns off its seat. The A-side and B-side CC Surge Tank Low Level Alarms were cleared by 1806.

At 1816 the 1A CC pump was restarted and the system was restored to normal operation thus ending the event.

This event is being reported under 10 CFR 50.73(A)(2)(V) - any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to remove residual heat.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Braidwood, Unit 1	0 5 0 0 0 4 5 6	8 7	- 0 1 1 -	0 0	0 3 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

C. Cause of Event:

The root cause of this event was leakage past the seat on 1CC9504B inlet valve to the 1B RHR Hx. The outlet motor operated valve was checked at the time of the event and was six turns off its fully closed position.

A contributing cause of this event can be attributed to the failure of the CC Surge Tank Level High/Low alarm to annunciate on the main control board. According to the Unit 1 NSO, SCRE, and an additional NSO on shift, there was no alarm indicating Low Surge Tank Level although the SER, which has no audible alarm, output showed a low level alarm condition was present. Had the alarm sounded, operators could have taken prompt action to restore CC Surge Tank Level before the 1A CC pump trip.

There were no unusual characteristics in the work location that contributed to this event.

D. Safety Analsyis:

Since the reactor has not yet been taken critical, there is no residual heat in the RCS and no spent fuel in the fuel pool. Therefore, no safety consequences resulted from the temporary loss of CC incident. Had the event occurred under more limiting conditions with residual heat in the RCS and the spent fuel pool full of spent fuel, plant safety would not have been compromised during the short term (14 minutes) while CC Surge Tank level and CC flow was being restored. The RCS would have a 15°F temperature rise (worst case) which would not result in a loss of sub-cooling. The fuel pool would take 4.5 hours for boiling to occur (worst case). Additionally, a minimum of two steam generators were available to remove heat as required by the Technical Specifications.

Corrective Action:

1. The valve body for 1CC9504B was repaired to allow 100% seating of the disc and it has been verified that the leakage past the disc and seat has been stopped.

2. The limits for Motor Operated valve 1CC9412B have been adjusted to ensure complete closure when the valve is actuated remotely.

3. The main control board annunciator was troubleshot and the symptoms could not be duplicated.

4. The calibration and scaling of the entire CC Surge Tank Level Instrumentation Loop is in progress. (Action Item 456-200-87-02901)

F. Previous Occurrences:

NONE

G. Component Failure Data:

Manufacturer	Nomenclature	Serial Number	Valve ID Number
Velan	12" Cast Bolted Bonnet Gate Valve	78G804 No Model Number	12G32

DEVIATION INVESTIGATION REPORT (DIR) CC05

Facility Name	Form Rev 2.0
Braidwood Unit 2	PAGE 1 OF 0 3
Title 2A and 2B Component Cooling Pump Trip While Performing Component Cooling System Lineup due to Deficient Planning	

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	OPERATING MODE	5
0 2	2 8	8 9	2 0	0 2	8 9	0 1 9	0 0	0 4	1 0	8 9	POWER LEVEL	0 0 0

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER AREA CODE	
Timothy K. Coomer, Tech Staff Engineer Ext. 2484	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 2; Event Date: February 28, 1989; Event Time: 2058;

Mode: 5 - Cold Shutdown; Rx Power: 0%;

RCS [AB] Temperature/Pressure: 98 degrees F/0 psig

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of the event which contributed to the severity of the event.

The Unit 2 Component Cooling (CC)[CC] system was lined up in its normal configuration. Both pumps supplying all portions of the Unit 2 System and isolated from the Unit 1 System.

At 1055 on February 28, 1989 the return to service for the CC loop serving the 2A Residual Heat Removal (RH) [BP] Heat Exchanger was authorized by the Shift Foreman (SF) Licensed Senior Reactor Operator (SRO). The Out-Of-Service was forwarded to the Control Room where the Return-To-Service positions were determined and verified by two Nuclear Station Operators (NSO) (Licensed Reactor operators). The Equipment Outage form was placed in the "to be done" bin at the Center Desk area of the Control Room along with numerous other items that were awaiting assignment for completion.

During the afternoon shift an Equipment Attendant (EA) (non-licensed operator) was assigned the task of performing the Return-To-Service along with several other tasks.

DEVIATION INVESTIGATION REPORT TEXT CONTINUATION

FACILITY NAME	DIR NUMBER							PAGE	Form Rev 2.0
	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER		
Braidwood Unit 2	2 0	0 2	8 9	—	0 1 9	—	0 0	2 OF 0 3	

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

B. DESCRIPTION OF EVENT: (Con't)

At 2057, the EA "cracked" open 2CC9504A, 2B RH Heat Exchanger CC Inlet Isolation Valve, which is a 12 inch gate valve. The EA was not aware that the CC side of the heat exchanger was empty. He observed that flow was passing through the valve and when it continued he shut the valve. In the Control Room the Unit 2 NSO received the CC Surge Tank Low Level Alarm and announced it over the radio. The Shift Engineer (SE) (Licensed SRO) dispatched a SF to the CC Surge Tank to initiate manual makeup.

At 2058, as the SF arrived at the CC Surge Tank the low-2 level setpoint was reached. This resulted in an automatic trip of the 2A and 2B CC Pumps at a CC Surge Tank level of 13% as designed. The SF immediately began refilling the CC Surge Tank by opening the manual makeup valve.

At 2059, the 2A and 2B CC pumps were restarted. The lowest level that the CC Surge Tank reached was 10% on one level indicator and 5% on the other. The duration that the tank was below the low-2 setpoint was less than 60 seconds.

The rapid response of operators to the Low CC Surge Tank Level Alarm decreased the severity of this event. Stable primary plant conditions were maintained throughout the event.

This report is being submitted pursuant to Section 1 Attachment G of the DVR Information Manual. Administrative/programmatic deficiencies regarding Technical Specification Equipment and Procedures.

CAUSE OF EVENT:

The root cause of this event was inadequate work planning. Manipulations that could affect inventory in the CC system should receive advance planning and discussion to insure that sufficient personnel are assigned to the task and that the Unit NSO is aware of the possible impact to his Unit. A second EA should have been provided to standby at the CC Surge Tank to monitor the level and initiate makeup as necessary.

A contributing cause to this event is a design deficiency. The existing design provides for only manual makeup. This caused a significant loss of inventory from the system until local manual operator action could rectify the situation.

D. SAFETY ANALYSIS:

This event had no effect on the safety of the plant or the public. All systems operated as designed. Although CC flow to the 2B RHR Heat Exchanger was interrupted for one minute, the RCS was at 98 degrees F and alternate methods were available to provide for long term cooling had restoration of the CC pumps been delayed.

Under the worst case condition, a loss of CC system inventory due to a piping rupture in one of the common headers, adequate valves are provided to isolate the break. Sufficient cooling would be provided. Cold water could be added to the Steam Generators [AB] using the Auxiliary Feed pumps [BA] or to the RCS using a Centrifugal Charging pump [CB] with RH letdown to the Holdup tanks. Both of these methods were available during this event. This is enveloped in Section 9.2 of the Updated Final Safety Analysis Report (UFSAR).

DEVIATION INVESTIGATION REPORT TEXT CONTINUATION

Form Rev 2.0

FACILITY NAME	DIR NUMBER							PAGE
	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
Braidwood Unit 2	2 0	0 2	8 9	—	0 1 1 9	—	0 0	3 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

E. CORRECTIVE ACTIONS:

The CC surge tank was immediately refilled and the pumps were restarted.

A tailgate session stressing the need for pre-job planning for evolutions that have significant impact on plant operations will be conducted with the appropriate Operating personnel. This will be tracked to completion by action item 457-200-89-01901.

An automatic makeup system to the CC Surge Tank is currently being installed per modification M20-2-88-031. This modification would have most likely prevented the occurrence of this event. This will be tracked to completion by action item 457-200-89-01902.

F. PREVIOUS OCCURRENCES:

There was an occurrence of loss of CC Surge Tank Level during RHR System evolutions. This occurred due to a defective valve. The corrective actions were implemented addressing both root and contributing causes. Previous corrective actions are not applicable to this event.

G. COMPONENT FAILURE DATA:

This event was not the result of component failure, nor did any components fail as a result of this event.

CC06 NON-ESSENTIAL CCW SYSTEM LEAK

TYPE: DISCRETE, RV 0-10000 GPM @ 100 PSID

CAUSE: PIPE BREAK IMMEDIATELY DOWNSTREAM 1CC9415 (SERVICE LOOP ISOL. VALVE)

REF: M-66 SHEET 4D
PLS

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THIS MALFUNCTION RESULTS IN A LOSS OF MASS FROM THE COMPONENT COOLING WATER SYSTEM. CCW SURGE TANK LEVEL WILL DECREASE AT A RATE DETERMINED BY THE SELECTED MALFUNCTION SEVERITY. AT 50% SURGE TANK LVL, ANNUNCIATOR 2-E4 "CC SURGE TANK AUTO-M/U ON" ACTUATES AND THE DEMIN WATER M/U VALVE 1CC-183 AUTO OPENS. APPROXIMATELY 200 GPM OF DEMIN WATER M/U IS SUPPLIED TO THE SURGE TANK. AT 45% LVL, THE PRIMARY WATER M/U VALVE 1CC-182 AUTO OPENS. APPROXIMATELY 200 ADDITIONAL GPM OF PRIMARY WATER M/U IS SUPPLIED TO THE SURGE TANK. ANNUNCIATOR 2-A5 "CC SURGE TANK LEVEL HIGH LOW" ACTUATES WHEN LEVEL DECREASES TO 35%.

AS MALFUNCTION SEVERITY IS INCREASED, COMPONENT COOLING WATER DISCHARGE PRESSURE DECREASES, AND PUMP FLOW INCREASES. WHEN DISCHARGE PRESSURE DECREASES TO 85 PSIG, ANNUNCIATORS 2-B4 "CC PUMP AUTO START" AND 2-B5 "CC PUMP DSCH PRESS LOW" ACTUATE AS THE STANDBY CCW PUMP AUTO STARTS RESTORING SYSTEM PRESSURE.

AT 13% SURGE TANK LEVEL THE CCW PUMPS WILL TRIP. ALL COMPONENTS COOLED BY CCW WILL INCREASE IN TEMPERATURE.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE COMPONENT COOLING SYSTEM PIPING INTEGRITY.

EVENTS: 1) LER 06-02-86-001

LICENSEE EVENT REPORT (LER)

CC06

Facility Name (1)	Docket Number (2)	Page (3)
Byron, Unit 2	0 5 0 0 0 4 5 5	1 of 0 3

Title (4): BOTH TRAINS OF COMPONENT COOLING INOPERABLE DUE TO A PERSONNEL ERROR IN A RELIEF VALVE SETTING

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
1 1	2 0	8 6	8 6	--- 0 0 1	--- 0 0	1 2	1 9	8 6	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): N A

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	__ 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	X 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER AREA CODE	
Don Brindle, U2 Operating Engineer Ext. 2218	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
A	C C	R V	C 7 1 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

__ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15): Month | Day | Year

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

On November 20, 1986 at 1026 Component Cooling (CC) Pumps 2A and 2B tripped. The Unit Operator had just shut down the 2A CC Pump because it was no longer needed to support plant operations. The shutdown caused a pressure spike which lifted a relief valve. The relief valve did not reseat and partially drained the CC System. The level in the CC Surge Tank fell below the low level interlock which tripped the 2B CC Pump. The 2A CC Pump started but also tripped on low level. The unit is in initial fuel load and precritical stage and therefore there is no decay heat load. The CC Pumps were not needed for any safety related loads because of this condition, therefore safety was not affected. The relief valve was isolated, CC Surge Tank Level restored, and the 2B CC Pump was restarted at 1038. The cause of the event was a personnel error in the initial setting of the relief valve. The relief valve was repaired and re-installed. Necessary procedures will be revised to caution operators of the possibility of this event. Other CC relief valves on both units will be bench tested. This is the first occurrence of this type.

(1135M)

CC07 RCP THERMAL BARRIER LEAK

TYPE: GENERIC, NRVI 0-300 GPM @ 2000 PSID

A)	1A RCP	1RC01PA
B)	1B RCP	1RC01PB
C)	1C RCP	1RC01PC
D)	1D RCP	1RC01PD

CAUSE: TUBE BREAK

REF: M-64 SHEET 1,2
M-66 SHEET 1A,1B
20E-1-4030 CC04,08,09
20E-0-4030 PR10

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THIS MALFUNCTION RESULTS IN THE LEAKAGE OF SEAL INJECTION/REACTOR COOLANT FROM THE SELECTED RCP INTO THE COMPONENT COOLING WATER SYSTEM.

AT LOW SEVERITY LEVELS, THE LEAKAGE INTO THE COMPONENT COOLING WATER SYSTEM WILL RESULT IN CCW SURGE TANK LEVEL INCREASING. CCW ACTIVITY LEVELS WILL INCREASE AS INDICATED ON 1RE-PR009 AND/OR 0RE-PR009, DEPENDENT UPON SYSTEM ALIGNMENT. WHEN EITHER DETECTOR REACHES ITS ALARM SETPOINT, CCW SURGE TANK VENT VALVE 1CC017 WILL AUTOMATICALLY CLOSE.

AS SEVERITY IS INCREASED, THE LEAKAGE INTO THE CCW SYSTEM WILL BEGIN TO ALSO COME DIRECTLY FROM THE REACTOR COOLANT SYSTEM. CCW TEMPERATURES WILL INCREASE SLIGHTLY. ANNUNCIATOR 7-E3 "RCP THERM BARR CC WTR TEMP HIGH" WILL ACTUATE. WHEN RCP THERMAL BARRIER CCW RETURN HEADER FLOW REACHES 192 GPM, ANNUNCIATOR 7-E4 "RCP THERM BARR CC WTR FLOW HIGH LOW" ACTUATES AND CCW FROM RCPs THERMAL BARRIER ISOLATION VALVE 1CC685 WILL AUTO CLOSE TO ISOLATE THE LEAK.

MALFUNCTION SEVERITY MAY ONLY BE INCREASED FOR THIS MALFUNCTION AND THE SIMULATOR MUST BE RESET TO REMOVE THIS MALFUNCTION.

EVENTS: NONE

CC08 CCW HX TUBE LEAK

TYPE: GENERIC, RV 0-1000 GPM @ 75 PSID

A) U-0 CC HX 0CC01A
B) U-1 CC HX 1CC01A

CAUSE: TUBE BREAK AT INLET TO HEAT EXCHANGER

REF: M-42 SHEET 2A
M-42 SHEET 2B
M-66 SHEET 3B
PLS

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THIS MALFUNCTION RESULTS IN A LOSS OF MASS FROM THE COMPONENT COOLING WATER SYSTEM TO THE ESSENTIAL SERVICE WATER SYSTEM. CCW SURGE TANK LEVEL WILL DECREASE AT A RATE DETERMINED BY THE SELECTED MALFUNCTION SEVERITY. M/U WILL INITIATE AUTOMATICALLY IN AN ATTEMPT TO MAINTAIN LEVEL.

IF THE SELECTED HEAT EXCHANGER IS ISOLATED, CCW SURGE TANK LEVEL WILL STOP DECREASING. ANNUNCIATOR 2-A5 "CC SURGE TANK LEVEL HIGH LOW" ACTUATES IF LEVEL DECREASES TO 35%. IF LEVEL DECREASES TO 13%, THE RUNNING CCW PUMP WILL TRIP

THE CONSEQUENCES OF THIS MALFUNCTION MAY BE LIMITED BY MAKEUP WATER BEING ADDED TO THE CCW SYSTEM.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE COMPONENT COOLING SYSTEM PIPING INTEGRITY.

EVENTS: NONE

CC09 THERMAL BARRIER CCW FLOW X-MITTER FAILURE

TYPE: DISCRETE, RV 0-200 GPM

CAUSE: FAULTY FLOW SWITCHES

REF: 20E-1-4030 CC04
C&ID M-2066 SHT 1

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE THERMAL BARRIER FLOW TRANSMITTER TO FAIL TO A VALUE DEPENDENT UPON SELECTED SEVERITY. IF THE SEVERITY LEVEL SELECTED IS >192 GPM (HIGH SETPOINT), ANNUNCIATOR 7-E4 "RCP THERM BARR CC WTR FLOW HIGH LOW" ACTUATES AND CLOSES 1CC685 . TAKING THE 1CC685 CONTROL SWITCH TO OPEN WILL RE-OPEN THE VALVE, HOWEVER AS SOON AS IT REACHES THE FULL OPEN POSITION IT WILL AUTO CLOSE IF THE MALFUNCTION IS STILL ACTIVE.

IF THE SEVERITY SELECTED IS <150 GPM (LOW SETPOINT) THEN ANNUNCIATOR 7-E4 "RCP THERM BARR CC WTR FLOW HIGH LOW" ACTUATES.

MALFUNCTION REMOVAL WILL RESTORE THE THERMAL BARRIER FLOW TRANSMITTER TO NORMAL.

EVENTS: NONE

H. OTHER RELATED DOCUMENTS:

None.

I. EFFECTIVENESS REVIEW:

None scheduled.

J. ADDITIONAL DATA:

a) Affected Technical Specification: 3/4.3.2 Engineered Safety Features Actuation System Instrumentation

b) Procedures: None.

c) Cause Code: CW4

d) Equipment Involved:

e) Other: LCOAR, Water Leak.

CH09 HYDROGEN MONITOR LINE LEAK

TYPE: GENERIC, RB

A)	TRAIN A	1PS47J
B)	TRAIN B	1PS48J

CAUSE: BREAK IN LINE NEAR HYDROGEN MONITOR PANEL

REF: M-68 SHEET 7

PLT STA: LARGE BREAK LOCA IN PROGRESS

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED HYDROGEN MONITOR LINE TO LEAK WHEN THE MONITOR IS PLACED IN SERVICE. DUE TO THE LOCA IN CONTAINMENT, GASEOUS ACTIVITY WILL BE RELEASED INTO THE AUX BUILDING. THIS WILL BE INDICATED BY AREA RAD MONITORS ON THE 401 LEVEL AND PROCESS MONITORS THROUGHOUT THE AUX BUILDING.

THE OPERATOR MAY CLOSE ONE OR MORE OF THE SUCTION VALVES TO ISOLATE THE LEAK. THIS WILL CAUSE AUX BUILDING RADIATION TO LOWER.

WITH EACH MALFUNCTION ACTIVE,THE RESPECTIVE HYDROGEN MONITOR WILL BE INCAPABLE OF DETECTING CONTAINMENT HYDROGEN.

MALFUNCTION REMOVAL WILL RESTORE THE HYDROGEN MONITOR LINE TO NORMAL.

EVENTS: NONE.

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

CH01	RCFC FAN FAILS TO START/TRIP, LOW SPEED
CH02	RCFC FAN FAILS TO START/TRIP, HIGH SPEED
CH03	CRDM FAN FAILS TO START/TRIP
CH04	REACTOR CAVITY BOOT FAILURE
CH06	BREAK IN CONTAINMENT INTEGRITY
CH08	CONTAINMENT PRESSURE TRANSMITTER FAILURE
CH09	HYDROGEN MONITOR LINE LEAK

CH01 RCFC FAN FAILS TO START/TRIP, LOW SPEED

TYPE: GENERIC, RB

A)	1A RCFC	1VP01CA
B)	1B RCFC	1VP01CB
C)	1C RCFC	1VP01CC
D)	1D RCFC	1VP01CD

CAUSE: FAULTY TRIP RELAY (SH/TR)

REF: 20E-1-4030 VP01
20E-1-4030 VP03
20E-1-4030 VP05
20E-1-4030 VP07
M-103 SHEET 2

PLT STA: RCFC FAN IN OPERATION AT LOW SPEED

EFFECTS: THE SELECTED REACTOR CONTAINMENT FAN COOLER FAN LOW SPEED BREAKER TRIPS ACTUATING ANNUNCIATOR 3-B5 "RCFC LOW SPEED BRKR TRIP". CURRENT INDICATION ON THE SELECTED FAN DECREASES TO ZERO AND THE TRIP LIGHT ON THE ASSOCIATED CONTROL SWITCH ILLUMINATES. CNMT TEMPERATURE RESPONDS ACCURATELY TO THE LOSS OF THE RCFC.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND TRIP LIGHT BY PLACING THE CONTROL SWITCH IN TRIP. IF THE OPERATOR ATTEMPTS TO RESTART THE FAN, THE BREAKER WILL CLOSE THEN IMMEDIATELY TRIP OPEN.

MALFUNCTION REMOVAL WILL RESTORE THE TRIP RELAY TO NORMAL OPERATION.

EVENTS: NONE.

CH02 RCFC FAN FAILS TO START/TRIP, HIGH SPEED

TYPE: GENERIC, RB

A)	1A RCFC	1VP01CA
B)	1B RCFC	1VP01CB
C)	1C RCFC	1VP01CC
D)	1D RCFC	1VP01CD

CAUSE: FAULTY TRIP REEAY (SH/TR)

REF: 20E-1-4030 VP02
20E-1-4030 VP04
20E-1-4030 VP06
20E-1-4030 VP08
M-103 SHEET 2

PLT STA: RCFC FAN IN OPERATION AT HIGH SPEED

EFFECTS: THE SELECTED REACTOR CONTAINMENT FAN COOLER FAN HIGH SPEED BREAKER TRIPS ACTUATING ANNUNCIATOR 3-A5 "RCFC HIGH SPEED BRKR TRIP". CURRENT INDICATION ON THE SELECTED FAN DECREASES TO ZERO AND THE TRIP LIGHT ON THE ASSOCIATED CONTROL SWITCH ILLUMINATES. CNMT TEMPERATURE RESPONDS ACCURATELY TO THE LOSS OF THE RCFC.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND TRIP LIGHT BY PLACING THE CONTROL SWITCH IN TRIP. IF THE OPERATOR ATTEMPTS TO RESTART THE FAN, THE BREAKER WILL CLOSE THEN IMMEDIATELY TRIP OPEN.

MALFUNCTION REMOVAL WILL RESTORE THE TRIP RELAY TO NORMAL OPERATION.

EVENTS: NONE.

CH03 CRDM FAN FAILS TO START/TRIP

TYPE: GENERIC, RB

A)	1A CRDM EXH. FAN	1VP03CA
B)	1B CRDM EXH. FAN	1VP03CB
C)	1C CRDM EXH. FAN	1VP03CC
D)	1D CRDM EXH. FAN	1VP03CD
E)	1A CRDM BOOSTER FAN	1VP04CA
F)	1B CRDM BOOSTER FAN	1VP04CB
G)	1C CRDM BOOSTER FAN	1VP04CC
H)	1D CRDM BOOSTER FAN	1VP04CD

CAUSE: FAILURE OF CR RELAY IN CONTROL CIRCUIT

REF: 20E-1-4030 VP10 - VP17

PLT STA: CRDM FANS IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED CRDM FAN TO TRIP OFF. THE TRIP LIGHT ON THE ASSOCIATED CONTROL SWITCH WILL ILLUMINATE AS WILL ANNUNCIATOR 33-A4, B4, C4, OR D4 "CRDM BSTR FAN TRIP", OR ANNUNCIATOR 33-A5 "CRDM EXHAUST FAN TRIP" FOR THE SELECTED CRDM FAN. WITH ONLY ONE CRDM EXHAUST FAN RUNNING, ANNUNCIATOR 33-B5 "CRDM EXH FLOW LOW" WILL ACTUATE.

THE OPERATOR MAY RESET THE TRIP LIGHT AND ANNUNCIATOR BY PLACING THE ASSOCIATED CONTROL SWITCH IN THE TRIP POSITION. THE AFFECTED FAN WILL NOT RESTART IF THE OPERATOR ATTEMPTS TO RESTART IT.

MALFUNCTION REMOVAL WILL RESTORE THE CR RELAY TO NORMAL.

EVENTS: NONE.

CH04 REACTOR CAVITY BOOT FAILURE

TYPE: DISCRETE, NRV 0-10,000 GPM

CAUSE: BOOT FAILURE

REF: M-152 SHEET 43

PLT STA: PLANT IS IN REFUELING MODE

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE REACTOR CAVITY TO LOSE MASS TO THE REACTOR CAVITY SUMP AT A RATE DETERMINED BY THE SELECTED SEVERITY. ANNUNCIATOR 6-C3 "REFUELING CAVITY LEVEL LOW ACTUATES". ANNUNCIATOR 12-A4 "PZR LEVEL LOW HTRS OFF LTDWN SECURED" ACTUATES AT 17% PZR LEVEL. ANNUNCIATOR 1-A2 "CNMT DRAIN LEAK DETECT FLOW HIGH" WILL ACTUATE AS WATER ACCUMULATES IN THE SUMPS. THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY MAKING UP TO THE REFUELING CAVITY FROM THE RWST.

THE REFUELING CAVITY LEVEL INDICATOR 1LI-RY046, 047, 048, AND 049 WILL INDICATE DECREASING LEVEL UNTIL THE INDICATORS ARE OFFSCALE LOW OR WHEN LEVEL HAS DECREASED TO APPROXIMATELY THE 400 FT LEVEL AT THE REACTOR VESSEL FLANGE.

CONTAINMENT AREA RADIATION MONITORS WILL SHOW INCREASING RADIATION LEVELS AS THE REFUELING CAVITY LEVEL DECREASES.

THE SIMULATOR MUST BE RESET TO RECOVER FROM THIS MALFUNCTION.

EVENTS: NONE.

CH06 BREAK IN CONTAINMENT INTEGRITY

TYPE: DISCRETE, RV 0-200 CFM

CAUSE: FAULTY PENETRATION (P95) ON PURGE EXHAUST

REF: M-105 SHT 1

PLT STA: PLANT IS IN REFUELING MODE

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A LEAK FROM THE CONTAINMENT TO THE AUXILIARY BLDG AT THE DESIRED SEVERITY. AREA RADIATION MONITORS IN THE AUXILIARY BLDG ACTUATE DEPENDENT UPON MALFUNCTION SEVERITY. CNMT PRESSURE DECREASES AT A RATE DEPENDENT UPON THE SEVERITY.

MALFUNCTION REMOVAL RESTORES THE CONTAINMENT INTEGRITY TO NORMAL.

EVENTS: NONE.

CH08 CONTAINMENT PRESSURE TRANSMITTER FAILURE

TYPE: GENERIC, RV 0-60 PSIG

A) PT-CS934
B) PT-CS935
C) PT-CS936
D) PT-CS937

CAUSE: FAULTY TRANSMITTER

REF: BwOA INST-2
M-2046 SHT 2

PLT STA: 100% POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED TRANSMITTER TO FAIL TO THE DESIRED SEVERITY. THE ASSOCIATED CNMT PRESSURE METERS, THE RECORDER ON THE 1PM06J AND STATUS LIGHTS (1PM06J),RESPOND PROPERLY TO MALFUNCTION SELECTION. THE FOLLOWING ESF FUNCTIONS ACTUATE WHEN THE PROPER CONDITIONS EXIST:

SAFETY INJECTION - ACTUATES WHEN 2/3 SIGNALS FROM PT-CS934, PT-CS935, OR PT-CS936 ARE RECIEVED AT 3.4 PSIG. SI THEN ACTUATES CNMT PHASE A ISOLATION, AND A CNMT VENT ISOLATION.

STEAM LINE ISOLATION - ACTUATES WHEN 2/3 SIGNALS FROM PT-CS934, PT-CS935, OR PT-CS936 ARE RECIEVED AT 8.2 PSIG.

CNMT SPRAY - ACTUATES WHEN 2/4 SIGNALS FROM PT-CS934, PT-CS935, PT-CS936, OR PT-CS937 ARE RECIEVED AT 20 PSIG.

CNMT PHASE B ISOLATION - ACTUATES WHEN 2/4 SIGNALS FROM PT-CS934, PT-CS935, PT-CS936, OR PT-CS937 ARE RECIEVED AT 20 PSIG.

MALFUNCTION REMOVAL WILL RESTORE THE PRESSURE TRANSMITTER TO NORMAL OPERATION.

EVENTS: 1) DVR 06-02-91-026

CH08

DEVIATION INVESTIGATION REPORT (DIR)

Form Rev 2.0

Facility Name	PAGE
Byron Nuclear Power Station	1 OF 0 3

Title

Containment Pressure Channel Transmitter Failed High due to Moisture Intrusion

EVENT DATE MONTH	DAY	YEAR	DIR NUMBER STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE MONTH	DAY	YEAR	OPERATING MODE	POWER LEVEL
1 1	1 8	9 1	0 6	0 2	9 1	0 2 6	0 0	1 2	2 3	9 1	1	0 9 0

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER AREA CODE	
T. Robinson, Technical Staff, Ext. 2250 J. VanLaere, Asst. Tech Staff Supervisor Ext. 2106	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X				Y					

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 11/18/91 / 0016

Unit 1 MODE 1 - Power Operation Rx Power 73.5% RCS [AB] Temperature/Pressure 572°F/2235 psig

Unit 2 MODE 1 - Power Operation Rx Power 96.0% RCS [AB] Temperature/Pressure 580°F/2235 psig

B. DESCRIPTION OF EVENT:

On November 18, 1991, at 0016, the 2PT-936 Containment Pressure (CS) [BE] Channel failed high. This event was discovered by the Unit 2 Operator (NSO) (RO, licensed). The associated containment pressure bistable was tripped per procedure 2BOA INST-2 at time 0026 and Limiting Condition for Operation Action Requirement (LCOAR) 3.2.1a was entered. This was done to ensure that acceptable conditions existed for continued operations. At 0028, the 2BOA INST-2 procedure was exited. No safety systems were activated. An investigation into the cause of this event was immediately undertaken by the Instrument Maintenance Department.

At the time of the failure, the plant was stable, with no other systems inoperable that may have contributed to the failure. No safety actuations occurred and the plant remained stable throughout the event. Operator actions did not affect the severity of this event.

C. CAUSE OF EVENT:

Investigation into this failure revealed that the termination box, supplied with the Containment Pressure transmitter (Barton 752) had water in it. This caused the transmitter output to fail high. The water entry into the termination box came from a roof leak (due to rain water) above the pressure transmitter. Hence, the cause of failure was rain water entering the transmitter's termination box.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

D. SAFETY ANALYSIS:

The failure had no effect on plant safety since all of the other Containment Pressure Channels were operational. The necessary procedure and LCOAR were entered in response to the failed channel. Partial trips occurred on Containment Pressure Hi-1, Hi-2, and Hi-3, but all of the other 7300 channels remained operable as well as other plant parameter indication systems.

The transmitter is Environmentally Qualified under EQDP ESE-4a, "Barton/Westinghouse, Differential Pressure Transmitter, Model 752" for temperature and radiation conditions. The transmitter was never qualified for LOCA/spray conditions. The 2PT-936 transmitter, located in the Auxiliary Building, would not normally be exposed to moisture. The transmitter was not designed to function under spray/moisture conditions. The transmitter termination box is not sealed to prevent moisture intrusion.

E. CORRECTIVE ACTIONS:

The Mechanical Maintenance Department repaired the leak in the roof and the termination box was uried out. At 0145, the Unit 2 operator restored the pressure channel 2PT-936 bystables to normal for an operability test. At 0417, LCOAR 3.2.1a was exited and operations resumed to normal.

F. RECURRING EVENTS SEARCH AND ANALYSIS:

a) EVENT SEARCH (DIR, LER)

No failures resulting from the intrusion of water were found.

b) INDUSTRY SEARCH (OPEX's NPRDS)

None.

c) NWR

None.

d) ANALYSIS

No trend identified.

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Barton	Differential Pressure Electronic Transmitter	752	

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

CS01 CONTAINMENT SPRAY PUMP FAILS TO START/TRIP

CS02 CONTAINMENT SPRAY PUMP SUCTION LINE BREAK

CS01 CONTAINMENT SPRAY PUMP FAILS TO START/TRIP

TYPE: GENERIC, RB

A) 1A CS PUMP 1CS01PA
B) 1B CS PUMP 1CS01PB

CAUSE: FAULTY OVERCURRENT (450/451) RELAY

REF: 20E-1-4030 CS01
20E-1-4030 CS02

PLT STA: CONTAINMENT SPRAY PUMP IN OPERATION

EFFECTS: THE SELECTED CONTAINMENT SPRAY PUMP BREAKER WILL OPEN. PUMP CURRENT INDICATION DECREASES TO ZERO, ANNUNCIATOR 3-A1 "CS PUMP TRIP" ACTUATES, AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. CS PUMP DISCHARGE FLOW, SUCTION AND EDUCTOR FLOW DECREASE TO ZERO.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND TRIP LIGHT BY PLACING THE CONTROL SWITCH IN AFTER-TRIP. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL CLOSE THEN IMMEDIATELY TRIP OPEN.

MALFUNCTION REMOVAL WILL RESTORE THE OVERCURRENT RELAY TO NORMAL.

EVENTS: NONE.

CS02 CONTAINMENT SPRAY PUMP SUCTION LINE BREAK

TYPE: GENERIC, NRV 0-500 GPM AT 35 PSID

A) TRAIN A CS
B) TRAIN B CS

CAUSE: PIPE BREAK DOWNSTREAM OF 1CS001A/B

REF: M-46 SHEET 1A
M-61 SHEET 4

PLT STA: REACTOR AT POWER, CS SYSTEM NOT IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A LOSS OF MASS FROM THE RWST AT A RATE DETERMINED BY THE SELECTED SEVERITY. AS THE RWST LEVEL DECREASES ANNUNCIATOR 6-C7 "RWST LEVEL LOW" ACTUATES. THE RWST LEVEL INDICATORS 1LI-930, 931, 932 AND 933 WILL INDICATE A DECREASING LEVEL BASED ON MALFUNCTION SEVERITY.

IF THE CS SYSTEM IS IN RECIRCULATION ON THE CONTAINMENT RECIRCULATION SUMPS, ACTIVITY LEVELS IN THE AUXILIARY BUILDING IN THE LOCAL AREA AND IN THE VENTILATION SYSTEM FLOW PATH WILL INCREASE.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY CLOSING THE APPROPRIATE CS SUCTION VALVE.

THE SIMULATOR MUST BE RESET TO RECOVER FROM THIS MALFUNCTION.

EVENTS: NONE.

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

CV01	CHARGING PUMP FAILS TO START/TRIP
CV02	PRI WATER MAKE-UP PUMP FAILS TO START/TRIP
CV03	BORIC ACID TRANSFER PUMP FAILS TO START/TRIP
CV04	VCT DIVERT VALVE FAILURE (112A)
CV05	PCV 131 AUTO CONTROLLER FAILURE
CV06	CLOGGED RCS FILTER (1CV3CF)
CV07	CLOGGED SEAL INJECTION FILTER
CV08	FAILURE OF PT-131 (LTDN PRESS)
CV09	FAILURE OF TE-130 (LTDN HX TEMP)
CV10	FLOW CONTROL VALVE 1CV121 FAILURE
CV11	CVCS UNBORATED MIXED BED DEMINERALIZER
CV12	LTDN RELIEF VALVE FAILS OPEN
CV13	CHARGING LINE LEAK OUTSIDE CONTAINMENT
CV14	REGENERATIVE HX TUBE LEAK
CV15	SEAL WATER HX TUBE LEAK
CV16	VCT LEVEL MALFUNCTION (LT-112)
CV17	VCT LEVEL MALFUNCTION (LT-185)
CV18	VCT PRESS MALFUNCTION (PT-115)
CV19	MAKE-UP CONTROL FAILURE
CV20	BORIC ACID FLOW TRANSMITTER (FT-110) FAILURE
CV21	CHARGING HEADER (1CV-182) CONTROL FAILURE
CV22	LTDN LINE LEAK INSIDE CONTAINMENT
CV23	LTDN HX TUBE LEAK
CV24	LTDN LINE LEAK OUTSIDE CONTAINMENT
CV25	CHARGING LINE LEAK INSIDE CONTAINMENT
CV26	SEAL INJECTION LINE LEAK
CV27	RCP #1 SEAL FAILURE
CV28	RCP #2 SEAL FAILURE
CV29	CHARGING PUMP DEGRADED IMPELLER

CV01 CHARGING PUMP FAILS TO START/TRIP

TYPE: GENERIC, RB

A) 1A CHARGING PUMP 1CV01PA
B) 1B CHARGING PUMP 1CV01PB
C) CHARGING PUMP 1CV02P (PDP)

CAUSE: FAULTY TRIP (TC/TR) DEVICE

REF: 20E-1-4030 CV01
20E-1-4030 CV02
20E-1-4030 CV03
M-64 SHEET 3A
M-64 SHEET 3B

PLT STA: CHARGING PUMP IN OPERATION

EFFECTS: THE SELECTED CHARGING PUMP BREAKER WILL OPEN. PUMP CURRENT INDICATION DECREASES TO ZERO, ANNUNCIATOR 9-A3 "CHG PUMP TRIP" ACTUATES, AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. PUMP DISCHARGE FLOW DECREASES TO ZERO AND DISCHARGE PRESSURE DECREASES.

DEPENDING ON THE INITIAL PLANT STATUS, IF THE SELECTED CHARGING PUMP WAS THE ONLY CHARGING PUMP IN OPERATION, CHARGING FLOW WILL BE LOST TO THE FOLLOWING FLOW PATHS:

- NORMAL CHARGING
- COLD LEG INJECTION
- AUX SPRAY
- SEAL INJECTION

THE SIMULATOR WILL RESPOND APPROPRIATELY FOR THE LOSS OF CHARGING FLOW PATH(S) FROM THE INITIAL PLANT CONDITION.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND TRIP LIGHT BY PLACING THE CONTROL SWITCH IN TRIP. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL CLOSE THEN IMMEDIATELY TRIP OPEN. THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY STARTING ONE OF THE OTHER CHARGING PUMPS.

MALFUNCTION REMOVAL WILL RESTORE THE CHARGING PUMP TRIP DEVICE TO NORMAL.

EVENTS: 1) SER 38-86

- CV-01

IS 633 FORSYTH (INPO) 26-NOV-86 11:17 PT
Subject: SER 38-86, POTENTIAL LOSS OF CHARGING FLOW

SUBJECT:	POTENTIAL LOSS OF CHARGING FLOW DUE TO LUBE OIL HEAT EXCHANGER PLUGGING
UNIT (TYPE):	FARLEY 1 (PWR)
DOC NO/LER NO:	50-348/(LATER)
EVENT DATE:	8/1/86
NSSS/AE:	WESTINGHOUSE/BECHTEL

SUMMARY:

SEDIMENT IN THE SERVICE WATER SYSTEM PLUGGED THE LUBE OIL HEAT EXCHANGERS FOR THE TWO AVAILABLE CENTRIFUGAL CHARGING PUMPS. THIS CAUSED THE LUBE OIL SYSTEMS FOR THE GEAR DRIVES OF THE PUMPS TO OVERHEAT. ONE CHARGING PUMP WAS MAINTAINED IN SERVICE BY USING ABNORMAL METHODS TO COOL ITS LUBE OIL HEAT EXCHANGER. THIS ALLOWED ENOUGH TIME TO CLEAN THE PLUGGED HEAT EXCHANGER ON THE OTHER CHARGING PUMP AND RETURN IT TO SERVICE.

THIS EVENT IS SIGNIFICANT BECAUSE IT INVOLVED A COMMON-MODE FAILURE MECHANISM THAT CAN DISABLE ALL CHARGING CAPABILITY. THE CHARGING SYSTEM AT THIS PLANT IS REQUIRED FOR EMERGENCY CORE COOLING DURING CERTAIN SMALL BREAK LOSS OF COOLANT ACCIDENTS.

DESCRIPTION:

WITH THE PLANT OPERATING AT 100% POWER, ONE CHARGING PUMP (1A) WAS OUT OF SERVICE FOR MAINTENANCE. A SECOND CHARGING PUMP (1B) WAS IN NORMAL SERVICE. A THIRD CHARGING PUMP (1C) IS A SWING PUMP. IT WAS LINED UP TO THE OUT-OF-SERVICE TRAIN AND WAS AVAILABLE FOR USE. ALL THREE ARE CENTRIFUGAL PUMPS.

AT 1230, INCREASING LUBE OIL TEMPERATURES INITIATED A CHARGING PUMP HIGH-TEMPERATURE ALARM. THE LUBE OIL TEMPERATURE FOR THE 1B CHARGING PUMP GEAR DRIVE WAS 145 DEGREES FAHRENHEIT. THIS INCREASED TO 155 DEGREES FAHRENHEIT BY 1238. THE 1C SWING CHARGING PUMP WAS STARTED, AND THE 1B PUMP WAS SHUT DOWN SO THAT MAINTENANCE PERSONNEL COULD CLEAN THE LUBE OIL HEAT EXCHANGER.

WHILE THIS MAINTENANCE WORK PROCEEDED, THE LUBE OIL TEMPERATURE FOR THE 1C PUMP WAS INCREASING AND INITIATED THE HIGH-TEMPERATURE ALARM (145 DEGREES FAHRENHEIT) AT 1310. IN ANTICIPATION OF LOSING THIS PUMP AND CONSISTENT WITH TECHNICAL SPECIFICATION RESTRICTIONS, REACTOR POWER WAS DECREASED. TO MAINTAIN THE 1C PUMP IN SERVICE AS THE LUBE OIL TEMPERATURE ROSE TO 150 DEGREES FAHRENHEIT, FANS, DEMINERALIZED WATER, AND ICE WERE USED TO COOL THE EXTERIOR SURFACE OF THE LUBE OIL HEAT EXCHANGER. THE CLEANING AND FLUSHING OF THE LUBE OIL HEAT EXCHANGER FOR THE 1B PUMP WAS COMPLETED, AND THE PUMP WAS RESTARTED BY 1523. THE 1C PUMP WAS SUBSEQUENTLY SHUT DOWN, AND ITS LUBE OIL HEAT EXCHANGER WAS CLEANED AND FLUSHED.

BOTH LUBE OIL HEAT EXCHANGERS HAD BEEN OBSTRUCTED BY ACCUMULATED SEDIMENT, I.E., MUD AND SILT. (A FEW CLAM SHELLS WERE ALSO FOUND BUT WERE MINOR CONTRIBUTORS TO THE OBSTRUCTION.) RECENT SURVEILLANCE TESTS OF THE SERVICE WATER SYSTEM, INVOLVING SOME UNUSUAL SYSTEM LINEUPS, HAD APPARENTLY CREATED FLOW AND PRESSURE TRANSIENTS THAT LOOSENED THE ACCUMULATED SEDIMENT IN THE SYSTEM. THIS SEDIMENT PREFERENTIALLY RESETTLED INTO THE CHARGING PUMP LUBE OIL HEAT EXCHANGERS BECAUSE THEY ARE LOCATED AT LOW POINTS IN THE SYSTEM.

BOTH TRAINS OF SERVICE WATER ARE SUPPLIED FROM A COMMON POND AND WET PIT AND HAVE SIMILAR PIPING, COMPONENTS, AND FLOW RATES. THEREFORE, THE PROBABILITY FOR INTRODUCING SEDIMENT FROM THE POND INTO SERVICE WATER COOLED COMPONENTS IS THE SAME FOR BOTH TRAINS. ALL THREE CHARGING PUMP LUBE OIL COOLERS HAVE CONTINUOUS SERVICE WATER COOLING FLOW EVEN WHEN THE PUMPS ARE OUT OF SERVICE. THE ONLY INDICATION TO THE OPERATOR OF CLOGGING IN A CHARGING PUMP LUBE OIL COOLER IS EXCESSIVE LUBE OIL TEMPERATURE RISE WHEN THE PUMP IS RUNNING. THEREFORE, IT IS POSSIBLE FOR CLOGGING TO OCCUR IN THE LUBE OIL COOLER OF AN OUT-OF-SERVICE PUMP WITHOUT ANY INDICATION TO THE OPERATOR.

SEDIMENT ACCUMULATION IN THE SERVICE WATER SYSTEM HAS BEEN AN ONGOING PROBLEM AT FARLEY. HOWEVER, THE PREVIOUS PROBLEMS HAVE BEEN MUCH LESS SERIOUS THAN THOSE IN THIS EVENT.

THE PLANT IS CONSIDERING USING COMPONENT COOLING WATER TO SUPPLY THESE COOLERS TO ELIMINATE THE POTENTIAL FOR SEDIMENTATION.

COMMENTS:

1. OPERATION OF THE SERVICE WATER SYSTEM WITH UNUSUAL FLOW CONDITIONS, SUCH AS SELDOM USED PUMP COMBINATIONS OR VALVE LINEUPS, CAN DISLODGE AND REDISTRIBUTE ACCUMULATED SEDIMENT. LOW POINTS AND LOW-FLOW AREAS IN THE SYSTEM ARE PARTICULARLY VULNERABLE TO RAPID ACCUMULATION OF SEDIMENT. THESE AREAS SHOULD BE CHECKED FOR SEDIMENT BUILDUP AFTER SYSTEM TESTS OR OTHER UNUSUAL FLOW CONDITIONS.

2. SOME HEAT EXCHANGERS AND PIPING CAN ACCUMULATE LARGE AMOUNTS OF SEDIMENT BEFORE THE CONDITION BECOMES APPARENT FROM PERFORMANCE DATA. PERIODIC VISUAL INSPECTIONS, PARTICULARLY OF SMALL HEAT EXCHANGERS, ARE NECESSARY TO IDENTIFY DEVELOPING SEDIMENT PROBLEMS BEFORE PERFORMANCE RAPIDLY DEGRADES. TRENDING OF SEDIMENT DEPOSITION, FLOW RATES, TEMPERATURE DIFFERENCES, AND OTHER PERTINENT DATA CAN BE A USEFUL TECHNIQUE FOR ANTICIPATING PROBLEMS THAT DEVELOP LESS RAPIDLY.

3. PERIODIC USE OF LOW-POINT DRAINS TO REMOVE SEDIMENT CAN HELP PREVENT PERFORMANCE DEGRADATION DUE TO CLOGGING.

IT IS RECOMMENDED THAT PLANT OPERATORS, THE OPERATIONS MANAGER, THE TECHNICAL SUPPORT MANAGER, AND THE MAINTENANCE MANAGER BE INCLUDED IN THE DISTRIBUTION OF THIS SER.

CV02 PRI WATER MAKE-UP PUMP FAILS TO START/TRIP

TYPE: GENERIC, RB

A) 0A PW MAKE-UP PUMP 0PW02PA
B) 0B PW MAKE-UP PUMP 0PW02PB

CAUSE: FAULTY CR CONTACT IN M RELAY CIRCUIT

REF: 20E-0-4030 PW01
20E-0-4030 PW02
M-74 SHEET 1

PLT STA: PRIMARY WATER MAKE-UP PUMP IN OPERATION

EFFECTS: THE SELECTED PRIMARY MAKE-UP PUMP WILL STOP. PUMP CURRENT INDICATION DECREASES TO ZERO, ANNUNCIATOR 38-A5 "PW PUMP TRIP OR AUTO START" ACTUATES, AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. PUMP DISCHARGE PRESSURE DECREASES. ANNUNCIATOR 38-C5 "PW PUMP DSCH PRESS LOW" ACTUATES.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND TRIP LIGHT BY PLACING THE CONTROL SWITCH TO STOP. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE PUMP WILL NOT RESTART AND THE TRIP LIGHT WILL ILLUMINATE IMMEDIATELY, AND THE ANNUNCIATOR WILL ACTUATE WHEN THE CONTROL SWITCH IS RETURNED TO AFTER START.

MALFUNCTION REMOVAL WILL RESTORE THE PRIMARY WATER MAKE-UP PUMP CR CONTACT TO NORMAL.

EVENTS: NONE

CV03 BORIC ACID TRANSFER PUMP FAILS TO START/TRIP

TYPE: DISCRETE, RB

CAUSE: FAULTY OVERLOAD CONTACT IN M RELAY CIRCUIT

REF: 20E-1-4030 AB01
M-65 SHEET 5A

PLT STA: 1AB03P OR 0AB03P IN MANUAL OPERATION

EFFECTS: BORIC ACID TRANSFER PUMP 1AB03P OR 0AB03P STOPS AS INDICATED BY ANNUNCIATOR 9-A4 "BA XFER PUMP TRIP" ACTUATING AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATING. ANNUNCIATOR 9-A6-"BA FLOW DEVIATION" ACTUATES AFTER 15 SEC TIME DELAY.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND TRIP LIGHT BY PLACING THE CONTROL SWITCH TO TRIP. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE PUMP WILL NOT RESTART, THE TRIP LIGHT WILL ILLUMINATE IMMEDIATELY, AND THE ANNUNCIATOR WILL ACTUATE WHEN THE CONTROL SWITCH IS RETURNED TO NORMAL AFTER CLOSE.

MALFUNCTION REMOVAL WILL RESTORE THE ASSOCIATED OVERLOAD CONTACT TO NORMAL.

EVENTS: 1) DVR 20-01-86-053

DEVIATION INVESTIGATION REPORT

CV03

TITLE: FAILURE OF "0" BORIC ACID TRANSFER PUMP SEAL DUE TO DEADHEADING

PAGE 1 OF 0 2

EVENT DATE			DIR NUMBER						REPORT DATE			OPERATING MODE	POWER LEVEL
MONTH	DAY	YEAR	STA	UNIT	YEAR		SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR		
1 1	2 2	8 6	2 0	0 1	8 6	—	0 5 3	0 0	0 1	3 0	8 7	5	0 0 0

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER
JENNY D. TOLAR, TECH STAFF ENGINEER Ext. 2484	AREA CODE 8 1 5 4 5 8 - 2 9 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
A	A B	P * *	G 2 0 0	N					

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

A. PLANT CONDITONS PRIOR TO EVENT:

Mode 5 Cold Shutdown RCS Temperature/Pressure 100°F/0 psig

B. DESCRIPTION OF EVENT:

During Shift III on 11-21-86, operating was transferring the contents of the Boric Acid Batching Tank [CB] to the Unit 1 Boric Acid Tank (per BwOP CV-25) using the Unit "0" Boric Acid Transfer Pump (0AB03P). After the transfer process was complete the system was lined up per BwOP AB-10 to recirculate the Boric Acid Tank (BAT) using 0AB03P so chemistry could obtain samples. After starting the pump on recirculation (at 2110), operating personnel verified a pump discharge pressure of 115 psig. Values of flow to the BAT or pressure around the filter were not noted.

During 11-22-86 Shift I, excessive leakage from the packing of 0AB03P was identified. At 0105, the pump was stopped, isolated (valves 1AB8465 and 1AB8468 were closed), and pump 1AB03P was lineup per BwOP AB-10 and started to continue the recirculation mode. During the process of verifying the lineup for 1AB03P, operating identified that valves 1AB8459 (BAT recirc. valve) and 1AB8446A (filter outlet valve) were closed. Thus, the Unit "0" Boric Acid Transfer Pump had been running without a recirculation path.

TITLE: FAILURE OF "0" BORIC ACID TRANSFER PUMP SEAL DUE TO DEADHEADING

DIR NUMBER					PAGE
STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
2 0	0 1	8 5	0 5 3	0 0	3 OF 3

TEXT

C. CAUSE OF EVENT:

The root cause of this incident is that valves 1AB8459 and 1AB8446A were closed causing pump 0AB03P to be deadhead and the seal failure. Per operating personnel, the valves were properly positioned prior to entering BwOP AB-10 to recirculate the BAT. If this were the case, unauthorized personnel repositioned (closed) the valves while the pump was running. Another possibility however, is that the valves were improperly positioned by operating prior to starting 0AB03P on recirculation (BwOP AB-10). Based upon a review of the pump curve for 0AB03P with the system conditions that existed at the time of the incident, this is possible. BwOP CV-25 requires valve 1AB8459 to be closed while transferring boric acid from the Batch Tank to the BAT. After completing BwOP CV-25 and prior to starting BwOP AB-10, valve 1AB8459 should have been opened per BwOP AB-M1 which is a prerequisite to BwOP AB 10. The valve (1AB8459) could have been missed or improperly repositioned by operating prior to entering BwOP AB-10. Valve 1AB8446A is to be open per BwAP AB-M1 and is not to be closed during BwOP CV-25 or BwAP AB-10. This valve (1AB8446A) could have been closed in error by operating prior to entering BwOP AB-10. In both cases, personnel error caused the improper positioning of the valves.

D. SAFETY ANALYSIS:

At the present plant conditions, deadheading 0AB03P while attempting to recirculate the BAT does not create any adverse safety consequences. Though this may damage the pump, the plant condition is not jeopordized. If the pump is damaged beyond use, the Unit "1" or Unit "2" pumps may be used to recirculate the BAT. Since the Boric Acid Transfer Pumps are used to borate the Reactor Coolant System (RCS) [AB] adverse safety conditions could result if this event (deadheading the pump) were to occur under the worst case conditions (using 0AB03P to borate the RCS at full power). The Technical Specifications (Tech Specs) require at least 2 of 3 boron injection flow paths to be operable. Thus, depending on the availability of other plant systems, the occurrence of this event could result in a Tech Spec violation.

E. CORRECTIVE ACTIONS:

The Shift Engineers will discuss with their personnel the importance of following procedures, especially and assuring proper valve lineups. This will be tracked by Item #456-200-86-05301

F. PREVIOUS OCCURRENCES:

None

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NO.	MFG PART NUMBER
Gould Pumps, Inc.	Boric Acid Transfer Pump	31965 T	N7388082-1-2-3

CV04 VCT DIVERT VALVE FAILURE (112A)

TYPE: DISCRETE, RV 0-100%

CAUSE: 1CV112A MECHANICAL BINDING

REF: M-64 SHEET 5
20E-1-4030 CV24

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: IF LETDOWN DIVERTS TO THE HOLD-UP TANK, 1CV112A WILL FAIL IN A POSITION DETERMINED BY THE SELECTED SEVERITY LEVEL AS THE VALVE TRAVELS THROUGH THAT POSITION. ONCE MECHANICALLY-BOUND, THE VALVE POSITION WILL NOT RESPOND TO OPERATION OF ITS ASSOCIATED CONTROL SWITCH OR TO MODULATING SIGNALS FROM ITS CONTROLLER.

MALFUNCTION REMOVAL WILL RESTORE 1CV112A TO NORMAL OPERATION.

EVENTS: NONE

CV05 PCV 131 AUTO CONTROLLER FAILURE

TYPE: DISCRETE, RV 0-600 PSIG

CAUSE: AUTO CONTROLLER INTERNAL PRESSURE SETPOINT FAILURE

REF: M-64 SHEET 5
PLS

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: LETDOWN LINE PRESSURE CONTROL VALVE 1CV131 WILL ATTEMPT TO MAINTAIN THE PRESSURE VALUE SELECTED AT MALFUNCTION INSERTION.

IF THE SEVERITY SELECTED RESULTS IN 1CV131 MODULATING CLOSED, LETDOWN LINE PRESSURE WILL INCREASE AND LETDOWN LINE FLOW WILL DECREASE. ANNUNCIATOR 8-B5 "LTDWN HX OUTLET PRESS HIGH" WILL ACTUATE IF PRESSURE INCREASES TO 490 PSIG. IF LETDOWN LINE PRESSURE INCREASES TO 600 PSIG, LETDOWN ORIFICE OUTLET RELIEF VALVE WILL LIFT.

IF THE SEVERITY SELECTED RESULTS IN 1CV131 MODULATING OPEN, LETDOWN LINE PRESSURE WILL DECREASE AND LETDOWN LINE FLOW WILL INCREASE. IF LETDOWN LINE PRESSURE DECREASES TO SATURATION PRESSURE FOR THE TEMPERATURE OF THE LETDOWN FLUID, FLASHING WILL OCCUR.

THE EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY PLACING THE CONTROLLER IN MANUAL.

MALFUNCTION REMOVAL WILL RESTORE 1CV131 AUTO CONTROLLER TO NORMAL.

EVENTS: NONE

CV06 CLOGGED RCS FILTER (1CV3CF)

TYPE: DISCRETE, RV 0-100% (TOTAL BLOCKAGE IS 100%)

CAUSE: FILTER DAMAGE

REF: M-64 SHEET 5

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THIS MALFUNCTION RESULTS IN THE INCREASE IN DELTA P ACROSS THE REACTOR COOLANT FILTER, 1CV3CF. AS MALFUNCTION SEVERITY IS INCREASED, LETDOWN LINE FLOW WILL DECREASE AND LETDOWN LINE PRESSURE WILL INCREASE. 1CV131 WILL MODULATE TO MAINTAIN UPSTREAM PRESSURE AT 370 PSIG.

AS SEVERITY IS INCREASED, PRESSURE DOWNSTREAM OF 1CV131 WILL INCREASE RESULTING IN LOW PRESSURE LETDOWN LINE TO VCT RELIEF VALVE,1CV8119, LIFTING AT 230 PSIG. AFTER THE RELIEF VALVE LIFTS, LETDOWN FLOW WILL BE DIVERTED TO THE VCT BYPASSING BTRS, BCMS, THE DEMINS, AND THE DIVERT VALVE.

MALFUNCTION REMOVAL WILL RESTORE REACTOR COOLANT FILTER 1CV3CF TO NORMAL.

EVENTS: NONE

CV07 CLOGGED SEAL INJECTION FILTER

TYPE: GENERIC, RV 0-100% (TOTAL BLOCKAGE IS 100%)

A) 1A FILTER 1CV01FA
B) 1B FILTER 1CV01FB

CAUSE: DAMAGED FILTER

REF: M-64 SHEET 3B

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE DELTA P ACROSS THE SELECTED FILTER TO INCREASE AS MALFUNCTION SEVERITY LEVEL IS INCREASED. AT 19 PSID ACROSS THE FILTER ANNUNCIATOR 7-A2 "RCP SEAL WTR INJ FLTR DP HIGH" ACTUATES. ANNUNCIATOR 7-B2 "RCP SEAL WTR INJ FLOW LOW" ACTUATES AT 6.6 GPM.

ALSO, AS MALFUNCTION SEVERITY LEVEL APPROACHES TOTAL BLOCKAGE, THE SEAL INJECTION FLOWS TO THE RCPs WILL DECREASE AS INDICATED ON MCB INDICATORS 1FI-142A, 143A, 144A AND 145A.

MALFUNCTION REMOVAL WILL RESTORE THE SEAL INJECTION FILTER TO NORMAL.

EVENTS: NONE

CV08 FAILURE OF PT-131 (LTDN PRESS)

TYPE: DISCRETE, RV 0-600 PSIG

CAUSE: TRANSMITTER FAILURE

REF: M-64 SHEET 5
M-2064 SHEET 8

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE OUTPUT OF THE PRESSURE TRANSMITTER TO INCREASE OR DECREASE TO A VALUE DETERMINED BY THE SELECTED SEVERITY LEVEL AND WILL BE INDICATED ON 1PI-131. LETDOWN LINE PRESSURE CONTROL VALVE 1CV131 WILL MODULATE IN AN ATTEMPT TO MAINTAIN LETDOWN LINE PRESSURE AT APPROXIMATELY 370 PSIG.

IF THE SEVERITY SELECTED IS > ACTUAL LETDOWN LINE PRESSURE THEN 1CV131 WILL MODULATE OPEN CAUSING ACTUAL LETDOWN LINE PRESSURE TO DECREASE AND LETDOWN LINE FLOW TO INCREASE SLIGHTLY AS INDICATED ON 1FI-132. IF LETDOWN LINE PRESSURE INCREASES > 490 PSIG THEN ANNUNCIATOR 8-B5 "LTDN HX OUTLET PRESS HIGH" ACTUATES. IF ACTUAL LETDOWN LINE PRESSURE DECREASES BELOW SATURATION THEN FLASHING WILL OCCUR IN THE LETDOWN LINE.

IF THE SEVERITY SELECTED IS < ACTUAL LETDOWN LINE PRESSURE THEN 1CV131 WILL MODULATE CLOSE CAUSING ACTUAL LETDOWN LINE PRESSURE TO INCREASE AND LINE FLOW TO DECREASE. IF LETDOWN LINE PRESSURE INCREASES TO > 600 PSIG THEN THE LETDOWN LINE RELIEF VALVE 1CV8117 WILL LIFT, ANNUNCIATOR 9-B1 "LP LTDWN RLF TEMP HIGH" ACTUATES AT > 140 °F AND RELIEF LINE TEMPERATURE WILL INCREASE AS INDICATED ON 1TI-125.

MALFUNCTION REMOVAL WILL RESTORE THE PRESSURE TRANSMITTER TO NORMAL.

EVENTS: NONE

CV09 FAILURE OF TE-130 (LTDN HX TEMP)

TYPE: DISCRETE, RV 50°F-150°F

CAUSE: DETECTOR FAILURE

REF: M-64 SHEET 5
M-2064 SHEET 8

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE OUTPUT OF THE TEMPERATURE TRANSMITTER TO INCREASE OR DECREASE TO A VALUE DETERMINED BY THE SELECTED SEVERITY LEVEL AND WILL BE INDICATED ON 1TI-130. LETDOWN HEAT EXCHANGER TEMPERATURE CONTROL VALVE 1CC-130A WILL MODULATE, ATTEMPTING TO MAINTAIN NORMAL LETDOWN HEAT EXCHANGER OUTLET TEMPERATURE.

IF THE SELECTED SEVERITY IS > THE ACTUAL TEMPERATURE, THEN 1CC-130A WILL MODULATE OPEN CAUSING A DECREASE IN ACTUAL LETDOWN LINE TEMPERATURE AND A DECREASE IN THE TEMPERATURE OF THE VOLUME CONTROL TANK AS INDICATED ON 1TI-116. ANNUNCIATOR 8-C5 "LTDWN HX OUTLET TEMP HIGH" WILL ACTUATE AT A SEVERITY LEVEL CORRESPONDING TO > 125°F.

IF THE SELECTED SEVERITY IS < THE ACTUAL TEMPERATURE, THEN 1CC-130A WILL MODULATE CLOSE CAUSING ACTUAL LETDOWN LINE TEMPERATURE TO INCREASE. A CORRESPONDING INCREASE IN VOLUME CONTROL TANK TEMPERATURE WILL OCCUR. WHEN ACTUAL LETDOWN TEMPERATURE INCREASES TO > 133 °F, ANNUNCIATOR 9-E2 "LTDWN TEMP HIGH" WILL ACTUATE AND DIVERT VALVE 1CV-129 WILL OPEN DIVERTING LETDOWN LINE FLOW TO THE VCT AND BYPASSING THE DEMINERALIZERS.

MANUAL OPERATION OF THE CONTROLLER WORKS PROPERLY.

MALFUNCTION REMOVAL WILL RESTORE THE TEMPERATURE TRANSMITTER TO NORMAL.

EVENTS: NONE

CV10 FLOW CONTROL VALVE 1CV121 FAILURE

TYPE: DISCRETE, RV 0-100% CONTROLLER OUTPUT

CAUSE: CONTROLLER FAILURE (AUTO ONLY)

REF: M-64 SHEET 3A

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE CHEMICAL AND VOLUME CONTROL (CVCS) CHARGING HEADER FLOW CONTROL VALVE 1FCV-121 TO MODULATE OPEN OR CLOSED DEPENDING ON THE SEVERITY LEVEL SELECTED.

IF THE SEVERITY LEVEL SELECTED CAUSES THE VALVE TO OPEN, THEN CHARGING HEADER FLOW WILL INCREASE AS INDICATED ON 1FI-121A. AT 150 GPM FLOW ANNUNCIATOR 9-D3 "CHG LINE FLOW HIGH LOW" ACTUATES. PRESSURIZER LEVEL WILL INCREASE AND VOLUME CONTROL TANK (VCT) LEVEL WILL DECREASE. ANNUNCIATOR 12-C3 "PZR LEVEL CONT DEV HIGH HTRS ON" ACTUATES AT +5% PROGRAM LEVEL AND ALL PRESSURIZER B/U HEATERS ENERGIZE. AUTO MAKE UP TO THE VCT IS INITIATED AT 37%. AS CHARGING HEADER FLOW INCREASES, THE SEAL INJECTION FLOWS TO THE REACTOR COOLANT PUMPS WILL SHOW A CORRESPONDING INCREASE.

IF THE SEVERITY SELECTED CAUSES THE FLOW CONTROL VALVE TO CLOSE, CHARGING HEADER FLOW WILL DECREASE AND SEAL INJECTION FLOW TO THE REACTOR COOLANT PUMPS WILL DECREASE. PRESSURIZER LEVEL WILL DECREASE AND VCT LEVEL WILL INCREASE. AT -5% PROGRAM PRESSURIZER LEVEL ANNUNCIATOR 12-B4 "PZR LEVEL CONT DEV LOW" ACTUATES. LETDOWN LINE FLOW BEGINS TO DIVERT TO THE HOLD UP TANKS AT 73% VCT LEVEL.

MANUAL OPERATION OF THE CONTROLLER WORKS PROPERLY.

MALFUNCTION REMOVAL WILL RESTORE FLOW CONTROL VALVE 1CV121 TO NORMAL.

EVENTS: NONE

CV11 CVCS UNBORATED MIXED BED DEMINERALIZER

TYPE: GENERIC, RV 0-100% DEBORATION

A) 1A MIXED BED 1CV01DA
B) 1B MIXED BED 1CV01DB

CAUSE: IMPROPERLY SATURATED DEMINERALIZER IS INADVERTANTLY VALVED INTO SERVICE

REF: CVCS SYSTEM DESCRIPTION
M-64 SHEET 6

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION RESULTS IN A DILUTION OF THE RCS UP TO A MAXIMUM OF 100 PPM BORON CONCENTRATION DEPENDING UPON THE SEVERITY LEVEL SELECTED. THE CVCS BORON CONCENTRATION MEASUREMENT SYSTEM WILL INDICATE THE DECREASING BORON CONCENTRATION. THE DILUTION WILL RESULT IN AN RCS TEMPERATURE (Tave) INCREASE. THE ROD CONTROL SYSTEM WILL COMPENSATE FOR THE TEMPERATURE RISE BY SLOWLY INSERTING THE CONTROL RODS. ANNUNCIATOR 10-B6 "ROD BANK LOW INSERTION LIMIT" WILL ACTUATE AT 10 STEPS ABOVE THE CALCULATED ROD INSERTION LIMIT. ANNUNCIATOR 10-A6 "ROD BANK LO-2 INSERTION LIMIT" WILL ACTUATE WHEN CONTROL RODS ARE AT THE CALCULATED INSERTION LIMIT.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY ISOLATING THE AFFECTED DEMINERALIZER OR BY BORATING THE RCS.

MALFUNCTION REMOVAL WILL RESTORE THE CVCS VALVE LINE UP TO NORMAL.

EVENTS: NONE

CV12 LTDN RELIEF VALVE FAILS OPEN

TYPE: DISCRETE, RV 0-300 GPM AT 600 PSID

CAUSE: RELIEF VALVE, 1CV-8117, SEAT LEAKAGE

REF: M-64 SHEET 5

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A MASS LOSS FROM THE CVCS SYSTEM TO THE PRESSURIZER RELIEF TANK. ANNUNCIATOR 9-B1 "LP LTDWN RLF TEMP HIGH" ACTUATES WHEN RELIEF LINE TEMPERATURE REACHES 140 °F. THE MASS LOSS FROM CVCS WILL RESULT IN A DECREASING VCT LEVEL. ANNUNCIATOR 9-A2 "VCT LEVEL HIGH LOW" ACTUATES AT 20% AND AUTO MAKE UP TO THE VCT IS INITIATED AT 37%. ACCUMULATION OF MASS IN THE PRESSURIZER RELIEF TANK WILL RESULT IN AN INCREASE IN PRT LEVEL AND PRESSURE. ANNUNCIATOR 12-B7 "PRT PRESS HIGH" WILL BE ACTUATED ON HIGH PRESSURE OF 6 PSIG. ANNUNCIATOR 12-C7 "PRT TEMP HIGH" WILL ACTUATE ON HIGH TEMPERATURE OF 125°F.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY ISOLATING NORMAL LETDOWN AND INITIATING EXCESS LETDOWN FOR PRESSURIZER LEVEL CONTROL.

MALFUNCTION REMOVAL WILL RESTORE RELIEF VALVE 1CV-8117 TO NORMAL.

EVENTS: NONE

CV13 CHARGING LINE LEAK OUTSIDE CONTAINMENT

TYPE: DISCRETE, RV 0-300 GPM AT 2300 PSID

CAUSE: PIPING FAILURE (DOWNSTREAM OF 1CV8105)

REF: M-61 SHEET 2

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THE MALFUNCTION BECOMES ACTIVE, A LOSS OF MASS FROM THE CVCS SYSTEM TO THE AUXILIARY BUILDING RESULTS. ACTIVITY LEVELS IN THE AUXILIARY BUILDING IN THE LOCAL AREA AND IN THE VENTILATION SYSTEM FLOW PATH WILL INCREASE.

IF NORMAL LETDOWN FLOW PLUS THE MALFUNCTION LEAKAGE EXCEED THE OPERATING CHARGING PUMP(S) CAPACITY, PRESSURIZER LEVEL WILL BEGIN TO DECREASE. THE DECREASE IN CHARGING LINE FLOW WILL RESULT IN INCREASED LETDOWN TEMPERATURE. THE INCREASED CHARGING FLOW THROUGH THE PIPE BREAK WILL RESULT IN DECREASED CHARGING HEADER PRESSURE AND A DECREASE IN SEAL INJECTION FLOW.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY STARTING ADDITIONAL CHARGING PUMPS.

MALFUNCTION REMOVAL WILL ONLY RESTORE CHARGING LINE INTEGRITY.

EVENTS:
1) DVR 06-02-90-015
2) DVR 06-01-89-031
3) DVR 06-02-88-028

DEVIATION INVESTIGATION REPORT (DIR)

CV/3

Form Rev 2.0

Facility Name	PAGE
Byron Nuclear Power Station	1 OF 0 1

Title

Reactor Coolant System Leakage Due To O-Ring Failure on 2B Seal Water Injection Filter

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	1
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	8 9
0 6	0 6	9 0	0 6	0 2	9 0	0 1 5	0 0	0 7	2 0	9 0		

CONTACT FOR THIS DIR

NAME		TELEPHONE NUMBER	
		AREA CODE	
T. Gierich, Operating Engineer	Ext. 2218	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	C B	F L T	P 0 5 0	Y					

SUPPLEMENTAL REPORT EXPECTED		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 6/6/90 / 1845

Unit 2 MODE 1 - Power Operation Rx Power 89% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On June 6, 1990 at 1845, with Unit 2 in power operation (Mode 1) at 89% reactor power, the Unit 2 Nuclear Station Operator (NSO, Reactor Operator) observed a decreasing Volume Control Tank (VCT) [CB] level trend on main control board panel 2PM05J. A Shift Foreman (SF, Senior Reactor Operator) and Equipment Operators (non-licensed) were dispatched to perform the leakrate checklist (Appendix A of 2BOS 4.6.2.1.d-1, "Leakage Sources Checklist") and to check the seal water injection filters and hold up tanks. Abnormal Operating Procedure, 2BOA PPI-1,Excessive Primary Plant Leakage, and Technical Specification Limiting Condition for Operation Action Requirement (LCOAR) 2BOS 4.6.2-1a were entered at 1905.

At 1906, operating personnel discovered a leak at the top o-ring of the 2B seal water injection filter. The 2B filter was isolated and the VCT level trend was verified stable at 1930. 2BOA PRI-1 and LCOAR 2BOS 4.6.2-1a were exited at 2030. Nuclear Work Request B77591 was written to repair the o-ring on the 2B seal water injection filter.

No plant systems or components were inoperable at the beginning of this event which contributed to the event. There were no manual or automatic safety system actuations as a result of this event. The plant was maintained in a stable condition during this event. Operator actions (finding and isolating the reactor coolant system leak) decreased the severity of this event.

C. CAUSE OF EVENT:

A tear in the top O-ring caused the filter leak. The root cause of the failure is indeterminate. It is believed the filter installation was correct, based on discussions with the Mechanical Maintenance foreman involved with the filter installation. Problem Analysis Data Sheet (PADS) 90-035 was written to investigate this event. There was no personnel error involved in this event.

D. SAFETY ANALYSIS:

There was no effect on plant and public safety. The leak was within the makeup capabilities of the Reactor Coolant Makeup System. The safety consequences would have been the same had this event occurred under a more severe set of initial conditions.

E. CORRECTIVE ACTIONS:

The 2B seal water injection filter was immediately isolated when it was determined to be the cause of the leak. The filter and o-ring were replaced by the Mechanical Maintenance Department under Nuclear Work Request B77591. The filter was tested on June 8, 1990. No further corrective actions will be taken.

F. RECURRING EVENTS SEARCH AND ANALYSIS:

a) EVENT SEARCH (DIR, LER)

DVR 6-2-88-028 was written for a seal injection filter o-ring leak. The o-ring failure was caused by a piece of plastic in the filter housing which cut the o-ring. DVR 6-1-89-031 was also written for a seal injection filter o-ring leak. The o-ring leak was caused by a damaged o-ring which occurred during a filter change with inadequate filter isolation.

b) INDUSTRY SEARCH (OPEX's NPRDS)

A NPRDS search indicated 2 filter o-ring failures industry wide but these were not Pall Trinity filters.

c) NWR

NWR B20309 replaced the 1A seal injection filter due to a crimped o-ring.
NWR B53226 and B53455 replaced the 2B seal injection filter (DVR 6-2-88-028).
NWR B65305 replaced the 1B seal injection filter (DVR 6-1-89-031).
NWR B77383 replaced the 2B seal injection filter due to o-ring leakage.

d) ANALYSIS

There is no adverse trend apparent due to the filter change frequency of all filters compared with the number of o-ring failures. The o-rings have not, themselves, had an unacceptable failure rate. The failures are attributable to various other unrelated causes.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

I. EFFECTIVENESS REVIEW:

None scheduled.

J. ADDITIONAL DATA:

a) Affected Technical Specification: 3/4.4.6.2

b) Procedures: 2BOA PRI-1 Excessive Primary Plant Leakage.

c) Cause Code: XPRMM1

d) Equipment Involved: 2B Seal Water Injection Filter (2CV01FB).

e) Other: None.

DEVIATION REPORT

CV13

DVR NO.

06 - 01 - 89 - 031
STA UNIT YEAR NO.

Form Rev 2.0

PART 1 | TITLE OF DEVIATION

EXCESSIVE PRIMARY PLANT LEAKAGE THROUGH FAILED SEAL INJECTION FILTER DUE TO PERSONNEL ERROR

OCCURRED: 02/28/89 (DATE) 0345 (TIME)

SYSTEM AFFECTED: CV / RC

PLANT STATUS AT TIME OF EVENT: MODE 1 POWER(%) 98%

WORK REQUEST NO.

TESTING: |__| YES |_X_| NO

DESCRIPTION OF EVENT

At 0345, the Unit One operator noticed excessive primary plant leakage due to frequent make-up to the VCT. Entered procedures 1BOA PRI-1 and 1BOA PRI-3 for reference to determine the source of leakage. Entered LCOAR 4.6.2-1a. The estimated leak rate was 6.5 GPM. The 1B Seal Injection Filter was determined to be the source of the leakage. Letdown was isolated and charging was reduced to 10 GPM to facilitate replacement of the damaged 1B seal injection filter. At 0645 leakage was verified to have stopped and LCOAR 4.6.2-1a was exited.

POTENTIALLY SIGNIFICANT EVENT PER NSD DIRECTIVE A-07 |__| YES |_X_| NO

10CFR50.72 NRC RED PHONE |__| 1 HOUR

NOTIFICATION MADE |__| 4 HOUR ______ TIME |_X_| NO

Bradley B. Milner — RESPONSIBLE SUPERVISOR — 02/28/89 DATE

PART 2 | OPERATING ENGINEER'S COMMENTS

None.

|_X_| NON REPORTABLE EVENT

|__| 30 DAY REPORTABLE/10CFR______

|__| 5 DAY REPORT PER 10CFR21

|__| ANNUAL/SPECIAL REPORT REQUIRED

A.I.R. #______

L.E.R. #______

NOTIFICATION ______ REGION III ______ DATE ______ TIME

Office of T. Maiman — NSD — 03/02/89 DATE — 1511 TIME

|__| CECO CORPORATE NOTIFICATION MADE IF ABOVE NOTIFICATION IS PER 10CFR21

TELECOPY ______ CECO CORPORATE OFFICER ______ DATE ______ TIME

PRELIMINARY REPORT COMPLETED AND REVIEWED: J. W. Schrock — OPERATING ENGINEER — 02/28/89 DATE

INVESTIGATION REPORT & RESOLUTION ACCEPTED BY STATION REVIEW: D Brunet 4/11/89; R Javant 4-13-89

RESOLUTION APPROVED AND AUTHORIZED FOR DISTRIBUTION: [signature] STATION MANAGER — 4/11/89 DATE

.-5176 (Form 15-52-1) 11-20-85

DOCUMENT ID

(0288R/0035R)

DEVIATION INVESTIGATION REPORT

C 1-13

TITLE

2B SEAL INJECTION FILTER LEAK DUE TO FAILED O-RING INDUCED BY FOREIGN MATERIAL

PAGE 1 OF 0 1

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	1
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	1 6
0 2	2 7	8 8	0 6	0 2	8 8	0 2 8	0 0	0 4	1 2	8 8		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
Dale St.Clair, Asst. Supt. Work Planning Ext. 2888	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 2/26/88 / 2000

Unit 1 MODE 1 - Power Operations Rx Power 98 RCS [AB] Temperature/Pressure Normal Operating

Unit 2 MODE 1 - Power Operations Rx Power 16 RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On 2/26/88, at 2000, the 2A Seal Injection Filter (2CV01FA) was taken off-line due to a high DP of 23# during a power decrease from approximately 95% power to 16% power on Unit 2. The 2B Seal Injection Filter (2CV01FB) was valved in to maintain seal injection flow. At 2140 the 120 gpm letdown flow was isolated due to low pressurizer level. An Equipment Attendant (EA) was sent to look for possible leakage in the vicinity of the 2B filter. The EA summoned a Shift Foreman (SF) and both of them identified a spraying sound beneath the 2B filter plug (401') and leakage was observed coming through a plugged vent/drain pipe penetration for the 2B filter (383'). At 2230 the 2B filter was taken off-line and administratively removed from service. The 2A filter had not yet been changed and was put back on-line with a measured 30# DP. Mechanical Maintenance personnel attributed the 2B filter leakage to a split O-ring on the filter cartridge. A new filter cartridge containing a new O-ring was installed under NWR B53455. No DVR was written for this initial O-ring failure.

At 0143, on 2/27/88, the 2B filter out-of-service was temporarily lifted and the 2B filter put back on-line while the 2A filter was taken off-line for filter cartridge replacement. At 2039 on 2/27/88 alarms annunciated for seal injection low flow and pressurizer level was again observed to be decreasing. The 2B seal injection filter was suspected and EA's were dispatched to the vicinity of the 2B filter and valves. Leakage was again observed coming through the same plugged vent/drain pipe penetration for the 2B filter (383'). At 2102 the 2B filter was isolated and the 2A filter was placed back on-line. Seal injection low flow alarms cleared as seal injection flow returned to normal and stable plant conditions were resumed.

TEXT

B. DESCRIPTION OF EVENT: (Continued)

Mechanical Maintenance personnel inspected the 2B filter housing sealing surfaces per directives specified in the filter cartridge replacement procedure BMP 3100-10 Rev. 3, 10/5/87, which had been revised to add prerequisites and steps for verifying that dirt problems and seal area damage both in the filter and the general vicinity are identified and corrected. Another split O-ring was found and the licensed Senior Reactor Operator (SRO) initiated this DVR.

The filter housing inspection did not identify any obvious mechanism for O-ring failure; however, in the more inaccessible lower portion of the filter housing, a piece of hard plastic (similar to model plastic) approximately 1/16" x 1" x 2" was found inside the housing. Personnel involved with the removal of the filter cartridge and plastic did not know the origin of the plastic or how it got inside the filter housing. The plastic and cartridge were disposed of as radwaste. A new 2B filter cartridge was installed. On 3/1/88 the 2B filter was pressurized and Tech Staff personnel performed a visual leakage examination through the 2B filter boroscopic floor plug opening and no leakage was observed. This exam completed testing for 2B filter NWR's B54355 and B53226. The 2B filter had been replaced on 2/20/88 due to high DP and had been restored with no apparent leakage problems. No systems were made inoperable during the sequence of 2A and 2B filter changeouts and stable plant conditions had been obtained at 2102 on 2/27/88.

C. CAUSE OF EVENT:

The root cause of the failed 2B O-ring is postulated to be from the piece of plastic cutting the O-ring. The most probable source of the plastic is that it entered the filter housing during the previous 2B filter changeout on 2/20/88 (NWR B53226). This is considered an isolated event.

D. SAFETY ANALYSIS:

There were no adverse safety consequences associated with this event because it was an isolated event in a dual train system. Under a more severe set of initial conditions where the plastic left the filter housing and entered the RCP or where both filter O-rings failed simultaneously, abnormal operating procedures 2BOA RCP-1, RCP Seal Failure and/or 2BOA RCP-2, Loss of Seal Injection, would be implemented as required.

E. CORRECTIVE ACTIONS:

Between July 1984 and February 1988, the four CV seal injection filters at Byron Units 1 and 2 have been changed a total of 88 times. Eighty seven (87) of these replacements were due to normal filter changeout in response to high DP. One of these was replaced due to a crimped O-ring (7/10/85) and no DVR had been written at that time. Since this is the first occurrence of a split seal injection filter O-ring at Byron, and existing procedures contain steps to preclude such failure, no further corrective action is deemed necessary at this time.

F. PREVIOUS OCCURRENCES:

There have been no previous occurrences documented by DVR's.

DVR NUMBER	TITLE
NONE	

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Pall-Trinity (P050)	Filter	SEH010602-22SEC32	SESC10670-_____-052

b) RESULTS OF NPRDS SEARCH:

Three previous similar occurrences of failed O-rings were found at the following locations:

2/25/85	Davis-Beese 1
7/10/85	Byron 1
5/23/86	V.C. Summer 1

The broken or split O-ring failures at Davis-Besse 1 and V.C. Summer 1 were attributed to wear and aging of the O-ring. The crimped O-ring failure at Byron could not be specifically identified and was attributed to either plant operation or filter cartridge installation. The current filter cartridge procedure BMP 3100-10, Rev. 2 6/5/83, in use at the time, did not contain any precautions or steps to assure a clean area and adequate sealing surface.

c) RESULTS OF MWR SEARCH:

See explanation in Corrective Action, Section E.

CV14 REGENERATIVE HX TUBE LEAK

TYPE: GENERIC, RV 0-150 GPM AT 100 PSID

A) 1A REGEN HX
B) 1B REGEN HX

CAUSE: TUBE FAILURE AT INLET TUBE SHEET

REF: M-64 SHEET 5

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, CHARGING SYSTEM WATER WILL LEAK INTO THE LETDOWN SYSTEM AT A RATE DETERMINED BY THE SELECTED SEVERITY. THE MASS OF CHARGING WATER LEAKING INTO THE LETDOWN SYSTEM WILL INCREASE AS MALFUNCTION SEVERITY IS INCREASED.

THE OVERALL EFFECT OF THIS MALFUNCTION IS A REDUCTION IN ACTUAL CHARGING FLOW INTO THE RCS. THIS ACTUAL FLOW REDUCTION INTO AND OUT OF THE RCS WILL RESULT IN EXTENDED TIME FOR REACTIVITY (BORATION/DILUTION) CHANGES TO TAKE AFFECT. OTHER INDICATIONS WILL BE A CHANGE IN THE REGEN HX CHARGING AND LETDOWN TEMPERATURES AS INDICATED ON 1TI-126 AND 1TI-127 RESPECTIVELY.

AS THE LEAK SEVERITY IS INCREASED, VCT LEVEL DECREASES DUE TO FLOW REVERSAL IN THE LETDOWN LINE.

MALFUNCTION REMOVAL WILL RESTORE THE REGENERATIVE HEAT EXCHANGER TUBING INTEGRITY AND CVCS TO NORMAL OPERATIONS.

EVENTS: NONE

CV15 SEAL WATER HX TUBE LEAK

TYPE: DISCRETE, RV 0-100 GPM @ 50 PSID

CAUSE: TUBE FAILURE AT SEAL RETURN INLET TO HX

REF: M-64 SHEET 4
M-66 SHEET 2

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THIS MALFUNCTION RESULTS IN THE LEAKAGE OF COMPONENT COOLING WATER INTO THE CV SYSTEM. VOLUME CONTROL TANK LEVEL WILL INCREASE AND THE CCW SURGE TANK WILL DECREASE AT A-RATE DETERMINED BY THE SELECTED MALFUNCTION SEVERITY.

AS SEVERITY LEVEL INCREASES, THE RCP SEAL #1 LEAKOFF FLOWS WILL DECREASE. THE ADDITION OF CCW WATER TO THE CVCS SYSTEM WILL RESULT IN A SLOW DILUTION OF THE RCS.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE SEAL WATER HEAT EXCHANGER INTEGRITY.

EVENTS: NONE

CV16 VCT LEVEL MALFUNCTION (1LT-112)

TYPE: DISCRETE, RV 0-100% LEVEL

CAUSE: 1LT-112 FAILURE

REF: M-2064 SHEET 6
20E-1-4030 CV32
20E-4031 CV13/CV24
PLS

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: VCT LEVEL TRANSMITTER 1LT112 FAILS TO THE VALUE SELECTED BY MALFUNCTION SEVERITY AS INDICATED ON 1LI-112. DEPENDING ON THE SEVERITY SELECTED, THE FOLLOWING ANNUNCIATORS/AUTO ACTIONS MAY OCCUR:

1CV112A FULLY DIVERTS TO THE HOLDUP TANK	@ 95%,
ANNUNCIATOR 9-A2 "VCT LEVEL HIGH LOW"	@ 95%
1CV112A BEGINS TO DIVERT TO HOLDUP TANK	@ 73%
AUTO MAKEUP STOPS	@ 55%
AUTO MAKEUP STARTS	@ 37%
ANNUNCIATOR 9-A2 "VCT LEVEL HIGH LOW"	@ 20%
AUTO SUCTION SWITCHOVER TO THE RWST	@ 5%
(MF CV17 @ <5% SEVERITY ALSO)	

MALFUNCTION REMOVAL WILL RESTORE VCT LEVEL TRANSMITTER 1LT-112 TO NORMAL.

EVENTS: 1) DVR 06-02-89-002

DEVIATION REPORT

CV-16

DVR NO.

06 - 02 - 89 - 002
STA UNIT YEAR NO.

Form Rev 2.0

PART 1 | TITLE OF DEVIATION VCT LOW LEVEL SUCTION SWITCHOVER TO RWST DUE TO FAILED CIRCUIT CARD AND INADEQUATE PROCEDURE

OCCURRED 01-05-89 DATE 1007 TIME

SYSTEM AFFECTED: CV

PLANT STATUS AT TIME OF EVENT: MODE 1 POWER(%) 31%

863707 WORK REQUEST NO.

TESTING |X| YES | | NO

DESCRIPTION OF EVENT

Investigations into a Rx makeup controller NWR were in progress that disconnected the output of VCT level transmitter 112. When these leads were lifted the centrifugal charging pump suction valve realigned to the RWST which should only happen if both VCT level channels 112 and 185 indicated a low level. Further investigations revealed that the 185 channel was already in the low level tripped condition due to a defective circuit card. All indications on the 185 channel read normal VCT level.

POTENTIALLY SIGNIFICANT EVENT PER NSD DIRECTIVE A-07 | | YES |X| NO

10CFR50.72 NRC RED PHONE | | 1 HOUR
NOTIFICATION MADE | | 4 HOUR ______ TIME |X| NO

M. Rasmussen RESPONSIBLE SUPERVISOR 01-05-89 DATE

PART 2 | OPERATING ENGINEER'S COMMENTS

There is no indication of coincidence for this function. Valves were immediately realigned.

|X| NON REPORTABLE EVENT

| | 30 DAY REPORTABLE/10CFR________

| | 5 DAY REPORT PER 10CFR21

| | ANNUAL/SPECIAL REPORT REQUIRED

A.I.R. #________

L.E.R. #________

NOTIFICATION ________ REGION III ____ DATE ____ TIME

T. Maiman NSD 01-11-89 DATE 1130 TIME

| | CECO CORPORATE NOTIFICATION MADE IF ABOVE NOTIFICATION IS PER 10CFR21

TELECOPY ________ CECO CORPORATE OFFICER ____ DATE ____ TIME

PRELIMINARY REPORT COMPLETED AND REVIEWED D. Brindle OPERATING ENGINEER 01-06-89 DATE

INVESTIGATION REPORT & RESOLUTION ACCEPTED BY STATION REVIEW W Kowsa 2-16-89
LE Didier 2/16/89

RESOLUTION APPROVED AND AUTHORIZED FOR DISTRIBUTION [signature] STATION MANAGER 2/16/89 DATE

86-5176 (Form 15-52-1) 11-20-85

DOCUMENT ID

(0237R/0029R)

CV17 VCT LEVEL MALFUNCTION (1LT-185)

TYPE: DISCRETE, RV 0-100% LEVEL

CAUSE: 1LT-185 FAILURE

REF: M-2064 SHEET 6
20E-1-4030 CV32
20E-4031 CV13/24
PLS

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: VCT LEVEL TRANSMITTER 1LT185 FAILS TO THE VALUE SELECTED BY MALFUNCTION SEVERITY AS INDICATED ON 1LR-185. DEPENDING ON THE SEVERITY SELECTED, THE FOLLOWING ANNUNCIATORS/AUTO ACTIONS MAY OCCUR:

ANNUNCIATOR 9-A2 "VCT LEVEL HIGH LOW"	@ 95%
ANNUNCIATOR 9-D2 "LTDWN FLOW DIVERTED TO HUT"	@ 95%
ANNUNCIATOR 9-A2 "VCT LEVEL HIGH LOW"	@ 20%
AUTO SUCTION SWITCHOVER TO THE RWST	@ 5%
(MF CV16 @ <5% SEVERITY ALSO)	

MALFUNCTION REMOVAL WILL RESTORE VCT LEVEL TRANSMITTER 1LT-185 TO NORMAL.

EVENTS: NONE

CV18 VCT PRESSURE MALFUNCTION

TYPE: DISCRETE, RV 2.4 - 89.7 PSIA

CAUSE: 1PT-115 FAILURE

REF: M-2064 SHEET 6
PLS

PLT STA: HOT SHUTDOWN

EFFECTS: VCT PRESSURE TRANSMITTER 1PT-115 FAILS TO THE VALUE SELECTED BY MALFUNCTION SEVERITY AS INDICATED ON 1PI-115 AND 1LR-185. ANNUNCIATOR 9-B2 "VCT PRESS HIGH LOW" ACTUATES IF THE SEVERITY SELECTED REACHES 67 PSIG (HIGH) OR 13 PSIG (LOW).

MALFUNCTION REMOVAL WILL RESTORE VCT PRESSURE TRANSMITTER 1PT-115 TO NORMAL.

EVENTS: NONE

CVI9 MAKE-UP CONTROL FAILURE

TYPE: DISCRETE, RB

CAUSE: MUX4 RELAY FAILURE

REF: 20E-1-4030 AB01
20E-1-4030 CV07
20E-1-4030 CV08
20E-1-4030 CV09
20E-0-4030 PW01

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: WITH MODE CONTROL SWITCH IN AUTO AND MAKEUP CONTROL SWITCH IN NORMAL AFTER START, WHEN VCT LEVEL DECREASES TO 37%, AUTO MAKEUP WILL NOT OCCUR. 0PW02PA WILL NOT AUTO START, 1CV111A DOES NOT OPEN, 1CV110A DOES NOT MODULATE, 1CV110B DOES NOT OPEN, AND 1AB03P DOES NOT AUTO START. ALL OTHER MODES OF VCT MAKEUP WILL OPERATE PROPERLY.

MALFUNCTION REMOVAL WILL RESTORE THE MAKEUP CONTROL MUX4 RELAY TO NORMAL.

EVENTS: NONE

CV20 BORIC ACID FLOW TRANSMITTER (1FT-110) FAILURE

TYPE: DISCRETE, RV 0-40 GPM

CAUSE: 1FT-110 FAILURE

REF: M-2064 SHEET 7
20E-1-4030 CV07
20E-1-4030 CV08

PLT STA: HOT SHUTDOWN

EFFECTS: BORIC ACID TRANSMITTER 1FT-110 FAILS TO THE VALUE SELECTED BY MALFUNCTION SEVERITY AS INDICATED ON 1FR-110. ANNUNCIATOR 9-A6 "BA FLOW DEVIATION" ACTUATES, 1CV110B CLOSES, AND 1CV111B CLOSES IF FLOW DEVIATES ± 0.8 GPM FOR GREATER THAN 15 SECONDS DURING AN AUTO MAKEUP, OR BORATION.

MALFUNCTION REMOVAL WILL RESTORE BORIC ACID FLOW TRANSMITTER 1FT-110 TO NORMAL.

EVENTS: NONE

CV21 CHARGING HEADER (1CV-182) CONTROL FAILURE

TYPE: DISCRETE, RV 0-100% CONTROLLER OUTPUT

CAUSE: 1HFK182 OUTPUT FAILURE

REF: M-2064 SHEET 5

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: CHARGING HEADER BACK PRESSURE CONTROL VALVE 1CV182 FAILS TO THE VALUE SELECTED BY MALFUNCTION SEVERITY. OPERATION OF 1CV182 CONTROLLER WILL HAVE NO EFFECT ON VALVE POSITION.

CLOSURE OF 1CV182 WILL RESULT IN INCREASED FLOW TO THE RCP SEAL INJECTION LINES AND HIGH FLOW TO THE RCP SEALS WILL RESULT IN ANNUNCIATOR 7-A2 "RCP SEAL WTR INJ FLTR DP HIGH" ACTUATING AT 19 PSID. THE DECREASE IN FLOW THROUGH THE REGENERATIVE HEAT EXCHANGER WILL RESULT IN INCREASED LETDOWN LINE TEMPERATURES. DUE TO THE THROTTLING ACTION OF 1CV8369, THE DECREASE IN NORMAL CHARGING LINE FLOW MAY NOT BE OFFSET BY THE INCREASE IN SEAL INJECTION LINE FLOW RESULTING IN PRESSURIZER LEVEL DECREASING.

FAILING 1CV182 OPEN MAY CAUSE A LOW RCP SEAL INJECTION FLOW CONDITION AND ANNUNCIATOR 7-B2 "RCP SEAL INJ FLOW LOW" MAY ACTUATE.

MALFUNCTION REMOVAL WILL RESTORE CHARGING HEADER BACK PRESSURE CONTROLLER 1HFK182 OUTPUT TO NORMAL.

EVENTS: NONE

CV22 LETDOWN LINE LEAK INSIDE CONTAINMENT

TYPE: DISCRETE, RV 0-120 GPM @ 600 PSID

CAUSE: PIPE BREAK IMMEDIATELY UPSTREAM 1CV8160

REF: M-64 SHEET 5

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THIS MALFUNCTION RESULTS IN THE LEAKAGE OF REACTOR COOLANT INTO THE CONTAINMENT ATMOSPHERE. CONTAINMENT AIRBORNE ACTIVITY LEVELS WILL INCREASE. CONTAINMENT TEMPERATURE, ACTIVITY LEVELS, AREA RADIATION LEVELS, AND SUMP LEVELS MAY INCREASE DEPENDENT UPON THE SELECTED MALFUNCTION SEVERITY.

AS MALFUNCTION SEVERITY IS INCREASED, NORMAL LETDOWN FLOW WILL DECREASE. 1CV131 WILL MODULATE CLOSED TO ATTEMPT TO MAINTAIN NORMAL LETDOWN PRESSURE. VCT LEVEL WILL DECREASE AT THE SELECTED MALFUNCTION SEVERITY.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY ISOLATING NORMAL LETDOWN AND INITIATING EXCESS LETDOWN FOR PRESSURIZER LEVEL CONTROL.

MALFUNCTION REMOVAL WILL ONLY RESTORE LETDOWN LINE INTEGRITY.

EVENTS: NONE

CV23 LTDN HX TUBE LEAK

TYPE: GENERIC, RV 0-100 GPM @ 450 PSID

A) 1A LETDOWN HX 1CV04AA
B) 1B LETDOWN HX 1CV04AB

CAUSE: TUBE BREAK AT LETDOWN INLET TO HEAT EXCHANGER

REF: M-64 SHEET 5
M-66 SHEET 2
M-66 SHEET 4D
20E-1-4030 CC09
20E-0-4030 PR10

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THIS MALFUNCTION RESULTS IN THE LEAKAGE OF REACTOR COOLANT INTO THE COMPONENT COOLING WATER SYSTEM. INDICATED LETDOWN FLOW WILL DECREASE AND 1CV131 WILL MODULATE TO MAINTAIN NORMAL LETDOWN PRESSURE. VCT LEVEL WILL DECREASE AT THE SELECTED MALFUNCTION SEVERITY RATE.

THE LEAKAGE INTO THE COMPONENT COOLING WATER SYSTEM WILL RESULT IN CCW SURGE TANK LEVEL INCREASING. CCW ACTIVITY LEVELS WILL INCREASE AS INDICATED ON 1RE-PR009 AND/OR 0RE-PR009, DEPENDENT UPON SYSTEM ALIGNMENT. WHEN EITHER DETECTOR REACHES ITS ALARM SETPOINT, CCW SURGE TANK VENT VALVE 1CC017 WILL AUTOMATICALLY CLOSE.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY ISOLATING NORMAL LETDOWN AND INITIATING EXCESS LETDOWN FOR PRESSURIZER LEVEL CONTROL.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE LETDOWN HEAT EXCHANGER INTEGRITY.

EVENTS: NONE

CV24 LTDN LINE LEAK OUTSIDE CONTAINMENT

TYPE: DISCRETE, RV 0-120 GPM @ 400 PSID

CAUSE: PIPE BREAK IMMEDIATELY DOWNSTREAM 1CV131

REF: M-64 SHEET 5

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THIS MALFUNCTION RESULTS IN THE LOSS OF MASS FROM THE REACTOR COOLANT SYSTEM TO THE AUXILIARY BUILDING VIA THE LETDOWN LINE. VOLUME CONTROL TANK LEVEL WILL BEGIN TO DECREASE AT A RATE DETERMINED BY THE SELECTED MALFUNCTION SEVERITY. 1CV131 WILL MODULATE TO MAINTAIN NORMAL LETDOWN LINE PRESSURE. ACTIVITY LEVELS IN THE AUXILIARY BUILDING IN THE LOCAL AREA AND IN THE VENTILATION FLOW PATH WILL INCREASE.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY ISOLATING NORMAL LETDOWN AND INITIATING EXCESS LETDOWN FOR PRESSURIZER LEVEL CONTROL.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE LETDOWN LINE INTEGRITY.

EVENTS: NONE

CV25 CHARGING LINE LEAK INSIDE CONTAINMENT

TYPE: DISCRETE, RV 0-120 GPM @ 2500 PSID

CAUSE: PIPE BREAK IMMEDIATELY UPSTREAM 1CV8324 A/B

REF: M-64 SHEET 5

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THIS MALFUNCTION RESULTS IN THE LOSS OF MASS FROM THE CHEMICAL AND VOLUME CONTROL SYSTEM TO THE CONTAINMENT. CONTAINMENT ACTIVITY LEVELS, AREA RADIATION LEVELS, AND SUMP LEVELS WILL INCREASE DEPENDENT UPON THE SELECTED MALFUNCTION SEVERITY. THE EFFECTS ON CONTAINMENT TEMPERATURE AND PRESSURE WILL BE MINIMAL DUE TO THE LOW TEMPERATURE OF THE CHARGING WATER.

AS MALFUNCTION SEVERITY IS INCREASED TO THE POINT WHERE NORMAL LETDOWN FLOW PLUS THE MALFUNCTION LEAKAGE EXCEED THE OPERATING CHARGING PUMP(S) CAPACITY, PRESSURIZER LEVEL WILL BEGIN TO DECREASE. THE DECREASE IN CHARGING LINE FLOW THROUGH THE REGENERATIVE HEAT EXCHANGER WILL RESULT IN INCREASED LETDOWN TEMPERATURE. THE INCREASED CHARGING FLOW THROUGH THE PIPE BREAK WILL RESULT IN DECREASED CHARGING HEADER PRESSURE AND A DECREASE IN SEAL INJECTION FLOW.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY ISOLATING NORMAL LETDOWN AND INITIATING EXCESS LETDOWN FOR PRESSURIZER LEVEL CONTROL.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE CHARGING LINE INTEGRITY.

EVENTS: NONE

CV26 SEAL INJECTION I E LEAK

TYPE: DISCRETE, RV 0-80 GPM @ 2500 PSID

CAUSE: PIPE BREAK IMMEDIATELY DOWNSTREAM SEAL INJECTION FILTERS

REF: M-64 SHEET 3B
PLS

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THIS MALFUNCTION RESULTS IN THE LOSS OF MASS FROM THE CHEMICAL AND VOLUME CONTROL SYSTEM TO THE AUXILIARY BUILDING. AUXILIARY BUILDING ACTI'ITY LEVELS, AREA RADIATION LEVELS, AND SUMP LEVELS MAY INCR ASE DEPENDENT UPON THE SELECTED MALFUNCTION SEVERITY.

AS MALFUNCTION SEVERITY IS INCREASED, CHARGING LINE FLOW AND PRESSURE WILL DECREASE. VCT LEVEL AND PRESSURIZER LEVELS WILL BEGIN TO DECREASE AT A RATE DETERMINED BY THE SYSTEM IMBALANCE.

SEAL INJECTION FLOWS WILL BEGIN TO DECREASE. ANNUNCIATOR 7-B2 "RCP SEAL WTR INJ FLOW LOW" WILL ACTUATE WHEN THE FIRST SEAL FLOW DECREASES TO 6.6 GPM. ANNUNCIATOR 7-A2 "RCP SEAL WTR INJ FLTR DP HIGH" WILL ACTUATE IF FILTER DIFFERENTIAL PRESSURE REACHES 19 PSID.

THE LOSS OF SEAL INJECTION TO THE REACTOR COOLANT PUMPS RESULTS IN A REVERSAL OF FLOW THROUGH THE LABYRINTH SEAL. COMPONENT COOLING WATER SYSTEM TEMPERATURES INCREASE, RCP LOWER BEARING TEMPERATURES INCREASE, AND RCP #1 SEAL OUTLET TEMPERATURES INCREASE.

MALFUNCTION REMOVAL WILL RESTORE THE SEAL INJECTION LINE INTEGRITY.

EVENTS: NONE

CV27 RCP #1 SEAL FAILURE

TYPE: GENERIC, NRVI 0-300 GPM @ 2200 PSID (ADDITIVE TO EXISTING FLOW)

A)	1A RCP	1RC01PA
B)	1B RCP	1RC01PB
C)	1C RCP	1RC01PC
D)	1D RCP	1RC01PD

CAUSE: EXCESSIVE WEAR

REF: M-64 SHEET 1,2

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: LEAK RATES LESS THAN 10 GPM WILL PREVENT EXCESSIVE RCP LOWER BEARING OR SEAL OUTLET TEMPERATURE RISE. LEAK RATES ABOVE 10 GPM WILL CAUSE AN EXCESSIVE TEMPERATURE RISE REQUIRING THE REACTOR AND AFFECTED RCP TO BE TRIPPED.

THE SELECTED REACTOR COOLANT PUMP #1 SEAL LEAKOFF FLOW WILL INCREASE BY THE SELECTED MALFUNCTION SEVERITY. THE RCP #1 SEAL DIFFERENTIAL PRESSURE WILL DECREASE. AT THE HIGHER SEVERITIES, FLOW REVERSES THROUGH THE LABYRINTH SEAL, COMPONENT COOLING WATER SYSTEM TEMPERATURES INCREASE, RCP LOWER BEARING TEMPERATURE INCREASES, RCP #1 SEAL OUTLET TEMPERATURE INCREASES, AND RCP #2 SEAL LEAKOFF FLOW WILL INCREASE. ANNUNCIATORS 7-C2 "RCP LOWER BRNG TEMP HIGH" AND 7-D3 "RCP SEAL OUTLET TEMP HIGH" WILL ACTUATE ON HIGH TEMP.

THE LOSS OF MASS FROM THE REACTOR COOLANT SYSTEM WILL CAUSE A DECREASE IN PRESSURIZER LEVEL. CHARGING FLOW WILL INCREASE TO MAINTAIN LEVEL. AT THE HIGHER SEVERITIES, VOLUME CONTROL TANK TEMPERATURE AND LEVEL WILL INCREASE, CHARGING FLOW WILL INCREASE, AND PRESSURIZER LEVEL WILL DECREASE.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY ISOLATING THE AFFECTED REACTOR COOLANT PUMPS #1 SEAL LEAKOFF.

MALFUNCTION SEVERITY MAY ONLY BE INCREASED FOR THIS MALFUNCTION AND THE SIMULATOR MUST BE RESET TO REMOVE THIS MALFUNCTION.

EVENTS: NONE

CV28 RCP #2 SEAL FAILURE

TYPE: GENERIC, NRB

A)	1A RCP	1RC01PA
B)	1B RCP	1RC01PB
C)	1C RCP	1RC01PC
D)	1D RCP	1RC01PD

CAUSE: EXCESSIVE WEAR

REF: M-64 SHEET 1
M-64 SHEET 2

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THE SELECTED REACTOR COOLANT PUMP #1 SEAL LEAKOFF FLOW WILL SLOWLY DECREASE TO ZERO, AND ANNUNCIATOR 7-C3 "RCP SEAL LEAKOFF FLOW LOW" ACTUATES. RCP #1 SEAL DIFFERENTIAL PRESSURE WILL INCREASE SLIGHTLY. RCP #2 SEAL LEAKOFF FLOW WILL INCREASE AND ANNUNCIATOR 7-B3 "RCP SEAL LEAKOFF FLOW HIGH" ACTUATES. CHARGING FLOW WILL INCREASE TO MAINTAIN PRESSURIZER LEVEL AND VCT LEVEL WILL DECREASE AT A RATE DETERMINED BY THE INCREASED RCP #2 SEAL LEAKOFF FLOW.

IF MALFUNCTION CV27 IS ACTIVE AT 100% ON THE SELECTED PUMP WHEN THIS MALFUNCTION IS INSERTED, RCP #3 SEAL WILL ALSO FAIL. THE LOSS OF ALL 3 RCP SEALS WILL RESULT IN A LOSS OF MASS FROM THE REACTOR COOLANT SYSTEM.

THE SIMULATOR MUST BE RESET TO REMOVE THIS MALFUNCTION.

EVENTS: NONE

CV29 CHARGING PUMP DEGRADED IMPELLER

TYPE: GENERIC, RV 0-100% TOTAL DEGRADATION

A) 1A CHARGING PUMP 1CV01PA
B) 1B CHARGING PUMP 1CV01PB

CAUSE: IMPELLER DISINTEGRATION

REF: M-64 SHTS 3A THRU 4

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED PUMP IMPELLER TO DISINTEGRATE TO THE DESIRED SEVERITY. AS MALFUNCTION SEVERITY IS INCREASED, PUMP DISCHARGE PRESSURE AND FLOW DECREASE, AND THE MOTOR AMPS ALSO DECREASE TO A NO LOAD VALUE. THE DECREASE IN FLOW THROUGH THE REGENERATIVE HEAT EXCHANGER WILL RESULT IN INCREASED LETDOWN LINE TEMPERATURES. PRESSURIZER LEVEL WILL DECREASE, AND FLOW TO THE RCP SEALS ALSO DECREASES. ANN 9-D3 "CHG LINE FLOW HIGH LOW" ACTUATES.

THE OPERATOR CAN MITIGATE THE EFFECTS OF THIS MALFUNCTION BY STARTING AN ADDITIONAL CHARGING PUMP.

MALFUNCTION REMOVAL WILL RESTORE THE PUMP IMPELLER TO NORMAL.

EVENTS: NONE

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

CW01 CIRC WATER PUMP FAILS TO START/TRIP

TYPE: GENERIC, RB

A)	1A CW PUMP	1CW01PA
B)	1B CW PUMP	1CW01PB
C)	1C CW PUMP	1CW01PC

CAUSE: FAULTY OVERCURRENT (450/451) RELAY

REF: 20E-1-4030 CW01
20E-1-4030 CW02
20E-1-4030 CW03
20E-1-4030 CW13
20E-1-4030 CW14

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THE SELECTED CIRCULATING WATER PUMP BREAKER WILL OPEN. PUMP CURRENT INDICATION DECREASES TO ZERO, ANNUNCIATOR 17-A13 "CW PUMP TRIP" ACTUATES, AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. THE ASSOCIATED CIRCULATING WATER PUMP DISCHARGE VALVE, 1CW001, AUTOMATICALLY CLOSES.

THE SELECTED PUMP'S DIFFERENTIAL PRESSURE DECREASES TO ZERO CAUSING ANNUNCIATOR 17-B13 "CW PUMP DP LOW" TO ACTUATE WHEN DIFFERENTIAL PRESSURE DECREASES TO 8 PSID. THE ANNUNCIATOR WILL RESET WHEN THE AFFECTED CIRCULATING WATER PUMP DISCHARGE VALVE REACHES FULL CLOSED.

THE CIRCULATING WATER PUMP TRIP WILL RESULT IN A DECREASE IN FLOW THROUGH THE MAIN CONDENSER. AS A NEW HEAT BALANCE IS ESTABLISHED AT THIS NEW FLOW RATE, CONDENSER VACUUM AND GENERATOR OUTPUT (IF MW OUT IS SELECTED) WILL DECREASE. THE MAGNITUDE OF THIS EFFECT IS DEPENDENT UPON THE INITIAL PLANT POWER.

MALFUNCTION REMOVAL WILL RESTORE THE CIRCULATING WATER PUMP OVERCURRENT RELAY TO NORMAL.

EVENTS: 1) DVR 06-01-87-172
2) DVR 06-02-88-114

DEVIATION INVESTIGATION REPORT

TITLE: 1A CW PUMP TRIP DUE TO SHORTED WIRE IN THE PUMP EXCITER TRANSFORMER

PAGE 1 OF 0 2

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	1
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	8 0
1 2	2 3	8 7	0 6	0 1	8 7	— 1 7 2	— 0 0	0 2	0 1	8 8		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
Don Brindle, U-2 Operating Engineer Ext. 2218	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	N N	E X C	E 1 2 0	N					

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
_ YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 12/23/87 / 0429

Unit 1 MODE 1 - Power Operation Rx Power 80 RCS [AB] Temperature/Pressure Normal Operating

Unit 2 MODE 1 - Power Operation Rx Power 29 RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On 12/23/87, at 0429, with Unit 1 at 80% reactor power, the 1A Circulating Water Pump Tripped from he Field Excitation Trip Relay.

As a result of the loss of 1A Circulating Water Pump an increase in condenser back pressure occurred. The U1 hogger vacuum pump was placed into service and the unit was ramped down approximately 10 megawatts. Proper condenser vacuum was subsequently reestablished.

Primary plant conditions followed the transient in a normal manner throughout the event. There were no other systems or components inoperable at the beginning of the event which contributed to the event.

C. CAUSE OF EVENT:

Investigation by station Electrical Maintenance revealed that the center tap on the automatic transformer for the motor exciter field had shorted causing a loss of excitation, thus, tripping 1A Circulating Water Pump. The damage to the center tap connection and associated wire was severe enough that the exact cause of the short could not be determined.

DEVIATION INVESTIGATION REPORT TEXT CONTINUATION

TITLE	DIR NUMBER						PAGE	
	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
1A CW PUMP TRIP DUE TO SHORTED WIRE IN THE PUMP EXCITER TRANSFORMER	0 6	0 1	8 7	—	1 7 2	—	0 0	2 OF 0 2

TEXT

D. SAFETY ANALYSIS:

The loss of a circulating water pump is not considered a safety failure. A controlled reduction of power is all that is required to prevent a loss of vacuum in the main condenser. No safety systems were affected by this event. The reactor protection system and S/G PORV's were operable had the turbine tripped on loss of condenser vacuum. There was no effect on the health and safety of the public.

E. CORRECTIVE ACTIONS:

The automatic transformer and approximately 3 ft of wire from the transformer to the local terminal block were replaced. The automatic transformers for 1B, 1C and 2A, 2B, 2C Circulating Water Pumps were inspected and all connections were intact.

The 1A motor was meggered and a polarization test performed per BHP 4200-52. The 1A Circulating Water Pump was restarted at 1419 on 12/23/87 and performed satisfactorily.

F. PREVIOUS OCCURRENCES:

There have been no previous occurrences of this type of failure on the transformer for the motor exciter field on the CW Pumps. A trend review was initiated based on two previous occurrences (DVR 6-86-009 and DVR 6-1-87-80). It was determined that though each instance resulted in loss of excitation to CW pumps the causes were separate and unrelated. (Trend 87-36)

DVR NUMBER	TITLE
NONE	

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Electric Machinery	Automatic Transformer	Synchro Pac 11	SI #789044

b) RESULTS OF NPRDS SEARCH:

N/A

c) RESULTS OF NUCLEAR WORK REQUEST (NWR) SEARCH

There are no previous NWRs written for this particular problem.

DEVIATION INVESTIGATION REPORT (DIR)

Facility Name	Form Rev 2.0
Byron Nuclear Power Station	PAGE
Title	1 OF 0
2A Circulating Water Pump Trip	

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	1
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	
1 1	1 4	8 8	0 6	0 2	8 8	— 1 1 4 —	0 0	1 2	2 3	8 8		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
T. Tulon, Assistant Superintendent Operating Ext. 2213	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	N N	X F M R	S 2 4 5	YES						

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 11-14-88 /1133

Unit 1 MODE 1 - Power Operation Rx Power 50% RCS [AB] Temperature/Pressure Normal Operating

Unit 2 MODE 2 - Power Operation Rx Power 53% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

At 1133 on November 14, 1988 with Unit 2 at 53% Reactor power, the 2A Circulating Water pump tripped causing a decrease in condenser vacuum (due to an increase in hotwell temperature). It was not necessary to ramp down the Unit during this event due to the quick response of the Unit Nuclear Station Operator (NSO-Licensed) in starting the standby Circulating Water pump and one of the hoggers (mechanical vacuum pump).

Upon investigation of the 2A CW pump trip by a Equipment Operator (EO) the exciter field relay and feed breaker for the exciter were found to be tripped.

Nuclear Work Request B62317 was written to allow for further investigation of the trip by the Electrical Maintenance Department.

All operator actions were correct and served to mitigate potential consequences of the event. No safety system actuations occurred during the event.

CW02 CIRC WATER PUMP DISCHARGE VALVE FAILURE

TYPE: GENERIC, RB

A)	1A CW PUMP	1CW01PA
B)	1B CW PUMP	1CW01PB
C)	1C CW PUMP	1CW01PC

CAUSE: FAULTY CRO CONTACT IN O RELAY CIRCUIT

REF: 20E-1-4030 CW13
20E-1-4030 CW14

PLT STA: HOT SHUTDOWN

EFFECTS: THE SELECTED CIRCULATING WATER PUMP DISCHARGE VALVE WILL NOT OPEN AS REQUIRED FROM AN OPEN SIGNAL GENERATED BY THE STARTING OF THE ASSOCIATED CIRCULATING WATER PUMP AS INDICATED BY ITS VALVE POSITION STATUS LIGHTS. IF THE ASSOCIATED CIRCULATING WATER PUMP IS STARTED, ANNUNCIATOR 17-C13 "CW PUMP RUNNING WITH DISCH VLV CLOSED" WILL ACTUATE 145 SECONDS AFTER THE PUMP BREAKER CLOSES.

MALFUNCTION REMOVAL WILL RESTORE THE CIRCULATING WATER PUMP DISCHARGE VALVE CRO CONTACT TO NORMAL.

EVENTS: NONE.

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

ED01	345 KV SWITCHYARD BREAKER FAILS TO TRIP
ED02	345 KV SWITCHYARD BREAKER TRIP
ED03	FAILURE OF UNIT AUX TRANSFORMER (UAT)
ED04	FAILURE OF SYSTEM AUX TRANSFORMER (SAT)
ED05	LOSS OF 6.9 KV BUS
ED06	FAILURE OF 6.9 KV ABT
ED07	LOSS OF 4160V BUS
ED08	LOSS OF FEED TO 480V NON-ESF BUS OR MCC
ED09	LOSS OF FEED TO 480V ESF BUS OR MCC
ED10	LOSS OF 120 VAC ESF CONSTANT VOLTAGE XFMR
ED11	120 VAC INSTRUMENT BUS INVERTER FAILURE
ED12	LOSS OF DC DISTRIBUTION BUS
ED13	DC CONTROL POWER FAILURE (4160V)
ED14	DC CONTROL POWER FAILURE (480V)
ED15	345 KV BUS FAULT
ED16	LOSS OF FEED TO 120V NON-ESF PANEL
ED17	LOSS OF FEED TO 120V ESF PANEL

ED01 345 KV SWITCHYARD BREAKER FAILS TO TRIP

TYPE: GENERIC, RB

A)	BT 1-3	G)	BT 9-10
B)	BT 1-8	H)	BT 9-15
C)	BT 3-4	I)	BT 10-11
D)	BT 4-7	J)	BT 11-14
E)	BT 7-11	K)	BT 14-15
F)	BT 7-8		

CAUSE: FAULTY TRIP COIL

REF: 20E-0-4001
20E-0-4146
20E-0-4149
20E-0-4154
20E-0-4157
20E-0-4159
20E-0-4160
20E-0-4161
20E-0-4163
20E-0-4166
20E-0-4171
AC DISTRIBUTION SYSTEM DESCRIPTION

PLT STA: SWITCHYARD RING BUS CLOSED

EFFECTS: INSERTING THIS MALFUNCTION PREVENTS THE SELECTED BREAKER(S) FROM TRIPPING OPEN UPON RECEIPT OF A VALID BREAKER TRIP SIGNAL. FAILURE OF THE AFFECTED BREAKER(S) TO TRIP WILL RESULT IN A LOCAL BREAKER BACKUP (LBB) TRIP SIGNAL BEING SENT TO THE BREAKERS IMMEDIATELY ADJACENT TO THE AFFECTED BREAKER(S). FAILURE OF THE SELECTED BREAKER(S) TO TRIP WILL CAUSE ANNUNCIATORS 35-D5 "BLUE SYSTEM LBB TRIP" AND/OR 36-D5 "RED SYSTEM LBB TRIP" TO ACTUATE. THE BREAKER RECEIVING THE LBB TRIP SIGNAL WILL TRIP OPEN. THIS IS INDICATED BY THE ASSOCIATED TRIP LIGHT AND THE OPEN LIGHT COMING ON, THE CLOSED LIGHTS GOING OUT, AND BREAKER TRIP ANNUNCIATOR ACTUATING ON 0PM03J.

MALFUNCTION REMOVAL WILL RESTORE THE FAULTY TRIP COIL TO NORMAL.

EVENTS: NONE.

ED02 345 KV SWITCHYARD BREAKER TRIP

TYPE: GENERIC, RB

A)	BT 1-3	G)	BT 9-10
B)	BT 1-8	H)	BT 9-15
C)	BT 3-4	I)	BT 10-11
D)	BT 4-7	J)	BT 11-14
E)	BT 7-11	K)	BT 14-15
F)	BT 7-8		

CAUSE: FAULTY TRIP COIL ACTUATION

REF: 20E-0-4001
20E-0-4146
20E-0-4149
20E-0-4154
20E-0-4157
20E-0-4159
20E-0-4160
20E-0-4161
20E-0-4163
20E-0-4166
20E-0-4171
AC DISTRIBUTION SYSTEM DESCRIPTION

PLT STA: HOT SHUTDOWN

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED 345 KV SWITCHYARD BREAKER TO TRIP OPEN. A BREAKER BEING TRIPPED OPEN IS INDICATED BY THE ASSOCIATED TRIP LIGHT COMING ON, THE CLOSED LIGHTS GOING OUT, AND THE BREAKER TRIP ANNUNCIATOR ACTUATING ON 0PM03J.

A COMBINATION OF SEVERAL BREAKER TRIPS INITIATED AT THE SAME TIME WILL CAUSE EITHER A LOSS OF LINES L-0104, L-0103, L-2001, L-2002, L-2003, L-2004, THE ASSOCIATED MAIN GENERATOR, AND/OR THE SYSTEM AUXILIARY TRANSFORMERS. THE PLANT WILL RESPOND ACCURATELY TO THE EFFECTS OF THESE MALFUNCTIONS. ANY ATTEMPT TO RECLOSE THE BREAKER WILL RESULT IN IT IMMEDIATELY TRIPPING OPEN.

MALFUNCTION REMOVAL RESTORES THE SELECTED 345 KV SWITCHYARD BREAKER(S) TO NORMAL.

EVENTS: NONE.

ED03 FAILURE OF UNIT AUX TRANSFORMER (UAT)

TYPE: GENERIC, RB

A) UAT 141-1
B) UAT 141-2

CAUSE: FAULTY DIFFERENTIAL (687-UT11/12) RELAY

REF: 20E-1-4030 MP01
20E-1-4030 MP02

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED UNIT AUX TRANSFORMER TO TRIP MAIN THE GENERATOR ON DIFFERENTIAL CURRENT. ANNUNCIATORS 19-A5 "UAT DIFF GEN TRIP" AND 19-E2 "GENERATOR LOCKOUT RELAY TRIP" ACTUATE AS THE MAIN GENERATOR TRIPS. A TURBINE TRIP AND REACTOR TRIP OCCUR. ALL UNIT AUXILIARY TRANSFORMER CURRENT AND WATT INDICATIONS DECREASE TO ZERO. ALL UNIT AUXILIARY TRANSFORMER BUS ALIVE INDICATORS DEENERGIZE. THE ASSOCIATED UAT DELUGE SYSTEM ACTUATES UPON TRIP SIGNAL. THE FOLLOWING BREAKERS RECEIVE TRIP SIGNALS TO COMPLETELY DEENERGIZE THE SELECTED UAT:

ACB 1571
ACB 1591
ACB 1431
ACB 1-8
OCB 7-8
ACB 1441
ACB 1561
ACB 1581
41 BREAKER

THE 6.9 & NON-ESF 4 KV BUSSES ABT TO MAINTAIN POWER TO BUSSES 157 & 143 (UAT 141-1) AND 156 & 144 (UAT 141-2).

MALFUNCTION REMOVAL RESTORES THE UAT DIFFERENTIAL RELAY TO NORMAL.

EVENTS: NONE.

ED04 FAILURE OF SYSTEM AUX TRANSFORMER (SAT)

TYPE: GENERIC, RB

A) 142-1
B) 142-2

CAUSE: FAULTY SUDDEN PRESSURE RELAY

REF: 20E-1-4030 AP01
20E-1-4030 AP02

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED SYSTEM AUX TRANSFORMER TO TRIP ON SUDDEN PRESSURE. ANNUNCIATORS 20-C3 "SAT 142-1 SUDDEN PRESS", or 20-C4 "SAT 142-2 SUDDEN PRESSURE", 20-A3 "SAT 142-1 LOCKOUT RELAY TRIP" AND 20-A4 "SAT 142-2 LOCKOUT RELAY TRIP" ACTUATE. ALL SYSTEM AUXILIARY TRANSFORMER CURRENT AND WATT INDICATIONS DECREASE TO ZERO. ALL SYSTEM AUXILIARY TRANSFORMER BUS ALIVE INDICATORS DEENERGIZE. THE SAT DELUGE SYSTEM ACTUATES UPON TRIP SIGNAL. THE FOLLOWING BREAKERS RECEIVE TRIP SIGNALS TO COMPLETELY DEENERGIZE THE SAT 142-1:

ACB 1572	ACB 1432
ACB 1592	ACB 3-4
ACB 1412	OCB 4-7

SAT 142-2 LOCKOUT RELAY (86ST12A) ACTUATES DUE TO THE SAT 142-1 LOCKOUT RELAY ACTUATION AND TRIPS THE FOLLOWING BREAKERS:

ACB 1562	ACB 1442
ACB 1582	ACB 3-4
ACB 1422	OCB 4-7

THE DIESEL GENERATORS AUTO START ON ESF BUS LOW VOLTAGE AND AUTO SEQUENCE ON THE ESF LOADS. THE 6.9 KV BUS ABTs TRANSFER TO MAINTAIN POWER TO THE BUSSES 159 (SAT 142-1) & 158 (SAT 142-2).

MALFUNCTION REMOVAL RESTORES THE SELECTED DIFFERENTIAL RELAY TO NORMAL.

EVENTS: 1) LER 20-1-88-022

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Braidwood 1	0 5 0 0 0 4 5 6	1 of 0 6

Title (4) Reactor Trip Due to 1C Reactor Coolant Pump Trip as a Result of Breaker Malfunction Due to Programmatic Deficiency

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
1 0	1 6	8 8	8 8	0 2 2	0 0	1 1	1 0	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 6

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	X 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name		TELEPHONE NUMBER	
		AREA CODE	
Freddie Ramos, Technical Staff Engineer	Ext. 2487	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
__ Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

At 2021 on October 16, 1988, a potential transformer failure on an off site 138 Kv line caused a line trip followed by a pole disagreement on a 345 Kv yard breaker in the Braidwood switchyard and resulted in a loss of off site AC power feed capability to Unit One. A blocked relay contact associated with the 1C Reactor Coolant Pump (RCP) allowed it to trip on instantaneous overcurrent during the bus transfer. This resulted in a reactor trip on RCP Low Flow Above 30% power and was followed by a turbine/generator trip as designed. Off site AC power was restored and normal hot standby conditions were established. The 345 Kv yard breaker was retimed to within acceptable tolerance. The potential transformer on the off site 138 Kv line has been replaced and the line returned to service. The relay block associated with the 1C RCP has been removed. Unit One 6.9 Kv and 4 Kv busses have been visually inspected for blocks. No blocks were found in any of the inspected relays. Additional administrative controls on the use and removal of blocks and/or jumpers on relays during periodic protective relay calibration have been issued. Additional emphasis and guidance on timely restoration of off site power has been given. There have been no previous occurrences of a loss of off site power due to a line perturbation.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Braidwood 1	0 5 0 0 0 4 5 6	8 8	- 0 2 2	- 0 0	0 2 OF 0 6

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 1; Event Date: October 16, 1988; Event Time: 2021 hrs;

Mode: 1 - Power Operation; Rx Power: 96%;

RCS [AB] Temperature/Pressure: 584 degrees F/2235 psig

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of the event which contributed to the severity of the event.

345 Kv line 2002 connects Braidwood to step down transformer 83 at Davis Creek Substation (Davis Creek).

October 16, 1988

At 2021, the phase 'A' potential transformer for 138 Kv line 8604 failed which caused a current surge on the low side of transformer 83. This caused the sudden pressure relay for transformer 83 to actuate, which resulted in a Transfer Trip signal to be sent to the 345 Kv breakers associated with line 2002 at both Braidwood Station and Davis Creek. Braidwood 345 Kv oil circuit breaker (OCB) 7-8 and air circuit breaker (ACB) 7-11 opened as designed. However, 345 Kv OCB 4-7 took longer to open and a Pole Disagreement actuation resulted. This caused the Local Breaker Backup (LBB) system to open 345 Kv ACB 3-4, which resulted in 345 Kv power being removed from the high side of Station Auxiliary Transformers (SATs) [EA] 142-1 and 142-2. The automatic bus transfer for 6.9 Kv busses 158 and 159 occurred as designed following closure of ACBs 1581 and 1591. The 1A and 1B Emergency Diesel Generators [EK] started and sequentially loaded upon loss of 4 Kv busses 141 and 142 as designed. The 1C reactor coolant pump (RCP) [AB] supply breaker, on bus 158, tripped on instantaneous overcurrent upon closure of ACB 1581. This caused the 2 out of 3 coincident logic signal for RCP 1C Flow Low Alert to be sent to Solid State Protection System (SSPS) [JG] and resulted in a reactor trip on RC Pump Low Flow Above 30% power. The reactor trip was followed by a turbine/generator trip (TG) [TB]. When 345 Kv OCB 7-8 and ACB 1-8 opened, the voltage on the Unit Auxiliary Transformers (UATs) 141-1 and 141-2 began to decay. This caused a loss of power to 6.9 Kv busses 156, 157, 158, and 159, as well as 4 Kv busses 143 and 144. As a direct result of this event on Unit One, the station air compressors (IA) [LD] tripped and instrument air header pressure started to decrease.

At 2029, the Commonwealth Edison Southern Division Load Dispatcher (LD) stated that an unknown problem existed at Davis Creek. The Shift Engineer (SE) informed the LD that 345 Kv ACB 3-4 opened due to a pole disagreement on 345 Kv OCB 4-7 and that it was open. The SE requested permission from the LD to close 345 Kv ACB 3-4 for the purpose of reenergizing the Unit One SATs. The LD denied the request because the status of the grid had not been verified.

At 2037, 4 Kv bus 143 was energized from bus 141 to restore two banks of pressurizer heaters to allow recovery of RCS pressure to its normal operating value.

At 2054, 4 Kv bus 144 was energized from bus 142.

At 2058, the SE classified the event in accordance with the Generating Station Emergency Plan (GSEP) as an Unusual Event.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

At 2110, appropriate notification of the Unusual Event was made to the Illinois Emergency Services Disaster Agency (ESDA), via the Nuclear Accident Reporting System (NARS), Pursuant to Emergency Action Level 10 - Loss of all offsite AC power required for a unit.

At 2112, line 2002 and unit main power transformer disconnects were opened at the request of the LD. The SE requested permission from the LD to close 345Kv ACB 3-4 for the purpose of reenergizing the Unit One SATs. The LD denied the request a second time because he still did not know the cause of the line trip.

The appropriate NRC notification via the ENS phone system was made at 2118 pursuant to 10CFR50.72(b)(2)(ii) and the GSEP Unusual Event.

At 2156, the LD gave permission to close 345 Kv ACB 3-4. The Unit One SATs were energized restoring offsite AC power.

At 2215, 345 Kv ACB 1-8 was closed as requested by the LD to start restoration of the 345 Kv ring bus on Unit One.

At 2216 an attempt to close 345 Kv OCB 7-8 was made. The attempt was unsuccessful as a result of the pole disagreement.

At 2217, following reset of the pole disagreement alarm, 345 Kv OCB 7-8 was closed to continue restoration of the 345 Kv ring bus on Unit One. Also, an attempt to make the Illinois ESDA one hour update call was placed on HOLD and subsequently disconnected at 2221.

At 2219, 4 Kv busses 143 and 144 were re-energized from the SATs to restore the normal offsite power lineup to Unit One.

At 2220, an attempt to close 345 Kv OCB 4-7 was made. The attempt was unsuccessful as a result of the pole disagreement.

At 2221, Commenced Illinois ESDA notification via outside phone lines for one hour update and to reclassify Unusual Event to Terminate Conditions. Notification completed at 2247.

At 2226, 6.9 Kv bus 159 was energized from the SATs in preparation for establishing RCS flow using 1D RCP.

At 2232, busses 156, 157, and 158 were energized from the SATs to place the Start-up Feedwater Pump on line and start the other RCPs.

At 2244, the 1D RCP was started to establish forced flow through the reactor core in preparation for establishing normal hot standby conditions.

The appropriate NRC notification via the ENS phone system was made at 2308 to provide followup notification pursuant to 10CFR50.72(c)(1)(iii) - A termination of the Emergency Classes. Also, to provide notification of the loss of the Unit Two Process computer pursuant to 10CFR50.72(b)(i)(v) - Any event that results in a major loss of emergency assessment capability, off site response capability, or communications capability.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

At 2354, SAT 142-1 paralleled to 4 Kv bus 142 to continue restoration of normal off site AC power.

At 2357, the 1B Diesel Generator was stopped and placed in STANDBY.

At 0001, SAT 142-1 paralleled to 4 Kv bus 141 to continue restoration of normal off site AC power.

At 0004, the 1A Diesel Generator was stopped and placed in STANDBY.

At 0008, 1C RCP started for establishing normal hot standby conditions.

At 0132, 1B RCP started for establishing normal hot standby conditions.

At 0149, 1A RCP started which established normal hot standby conditions.

At 0155, 345 Kv OCB 7-8 and ACB 1-8 were opened, as directed by the LD, in preparation for restoration on line 2002.

At 0201, line 2002 disconnect was closed.

At 0210, 345 Kv OCB 7-8 and ACB 1-8 were closed, establishing normal breaker lineup. Additionally, another attempt to cloo 345 Kv OCB 4-7 was made. The attempt was unsuccessful as a result of the pole disagreement.

Operator actions decreased the severity of the event by restoring the instrument air system and minimizing the effects of the Unit Two operations.

This event is being reported pursuant to 10CFR50.73(A)(2)(iv) - Any event or condition that resulted in manual or automatic actuation of any engineered safety feature, including the reactor protection system.

C. CAUSE OF EVENT:

The cause of the loss of line 2002 was due to a failure of the phase 'A' potential transformer for 138 Kv line 8604 at Davis Creek. This caused a current surge on the low side of transformer 83, which resulted in its sudden pressure relay to actuate. This caused a transfer trip signal to be sent to the 345 Kv breakers associated with line 2002 at both Braidwood Station and Davis Creek.

The cause of the loss of power to the SATs was improper time between opening for the different phase poles for 345 Kv OCB's 4-7 adn 7-8, which resulted in a Pole Disagreement actuation. This caused the LBB system to open 345 Kv ACB 3-4, which resulted in 345 Kv power removed from the high side of SATs 142-1 and 142-2.

The cause of the reactor trip was the result of the 1C RCP supply breaker, on 6.9 Kv bus 158, tripping on instantaneous overcurrent. This was caused by a piece of cardboard inserted in the 1C RCP breaker instantaneous overcurrent relay bypassing the 5-6 cycle time delay. It is suspected that the cardboard was inserted during the last maintenance on the relay as a relay block. This is considered to be a programmatic deficiency in that no mechanisms existed to ensure that relay blocks were removed following maintenance activities.

D. SAFETY ANALYSIS:

There was no effect on plant or public safety. All systems operated as designed in response to the loss of off site AC power to Unit One. Restoration of off site power was done in a controlled manner following verification of the cause of the initiating event at Davis Creek.

Under worst case conditions of a Loss of Coolant Accident coincident with the loss of off site AC power, there would have been no effect on plant or public safety as this event is enveloped in the Final Safety Analysis Report (FSAR).

Off site AC power remained available to Unit 2 throughout the event. The Unit 1 emergency diesel generators started and supplied AC power as designed and the Unit Two emergency diesel generators were operable throughout the event.

E. CORRECTIVE ACTIONS:

Off site AC power was restored to Unit One as directed by the LD.

Unit One was placed in a Safe Shutdown condition.

The phase 'A' potential transformer for 138 Kv line 8604 at Davis Creek has been replaced and the line returned to service.

The sudden pressure relay actuation on transformer 83 at Davis Creek was reset.

Davis Creek to Braidwood Station 345 Kv line 2002 has been restored.

The piece of cardboard which was used as a block in the instantaneous overcurrent relay for the 1C RCP was removed.

Unit One relays for 6.9 Kv busses 156, 157, 158, and 159 have been visually inspected for blocks. No blocks were found in any of the inspected relays.

Unit One relays for 4 Kv busses 141, 142, 143, and 144 have been visually inspected for blocks. No blocks were found in any of the inspected relays.

The above two inspections will be conducted for the Unit 2 counterpart relays at the next opportunity. This will be tracked by Action Item 456-200-88-23701.

345 Kv OCB 4-7 and OCB 7-8 were tested for proper timing of opening of the phase breakers. Pole disagreement, time between the opening of the different phase poles, was found to be out of tolerance. They were recalibrated to bring the times within acceptable tolerance.

Additional administrative controls on the use and removal of blocks and/or jumpers on relays during periodic protective relay calibration have been issued to the Division Operational Analysis Department personnel.

Additional emphasis and guidance has been given to the Division LDs to ensure off site power is restored in a timely manner and the restoration of power has the highest priority.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

F. PREVIOUS OCCURRENCES:

There have been previous occurrences of a loss of off site power which resulted in a reactor trip. The corrective actions were implemented addressing both root and contributing causes. Previous corrective actions are not applicable to this event.

G. COMPONENT FAILURE DATA:

This event was not caused by component failure at Braidwood Station nor did any components fail as a result of this event.

ED05 LOSS OF 6.9KV BUS

TYPE: GENERIC, RB

A) BUS 156
B) BUS 157
C) BUS 158
D) BUS 159

CAUSE: BUS GROUND OVERCURRENT CONDITION

REF: 20E-1-4030 AP07,08 20E-1-4030 AP15,16
20E-1-4030 AP11,12 20E-1-4030 AP19,20

PLT STA: REACTOR AT FULL POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED 6.9 KV BUS FEEDER BREAKER TO OPEN. THE ABT IS DEFEATED BY THE FEEDER BREAKER LOCKOUT. THE ALTERNATE POWER SUPPLY WILL ABT IF THE LOCKOUT IS RESET. THE ALTERNATE FEEDER BREAKER WOULD THEN TRIP ON OVERCURRENT, DEENERGIZING THE SELECTED BUS. THE SELECTED BUS CURRENT, VOLTAGE, AND WATTAGE INDICATIONS DECREASE TO ZERO. FOR EACH OF THE SEPARATE MALFUNCTIONS; THE FOLLOWING LOADS ARE LOST, AND ANNUNCIATORS ACTUATE:

BUS 156: 1A FEEDWATER PUMP, HEATER DRAIN PUMP 1B, AND RCP 1B, ANNUNCIATORS 20-B7 "BUS 156 FD BKR 1561 TRIP" & 20-D7 "BUS 156 VOLT LOW"

BUS 157: HEATER DRAIN PUMP 1A, HEATER DRAIN PUMP 1C, AND RCP 1A, ANNUNCIATORS 20-B5 "BUS 157 FD BKR 1571 TRIP" & 20-D5 "BUS 157 VOLT LOW"

BUS 158: RCP 1C, CONDENSATE AND CONDENSATE BOOSTER PUMPS 1B AND 1D, ANNUNCIATORS 20-A8 "BUS 158 FD BKR 1582 TRIP" & 20-D8 "BUS 158 VOLT LOW"

BUS 159: START-UP FEEDWATER PUMP, RCP 1D, CONDENSATE AND CONDENSATE BOOSTER PUMPS 1A AND 1C, ANNUNCIATORS 20-A6 "BUS 159 FD BKR 1592 TRIP" & 20-D6 "BUS 159 VOLT LOW".

A REACTOR TRIP MAY OCCUR DEPENDENT ON THE POWER LEVEL, AND/OR THE NUMBER OF BUSSES DEENERGIZED.

MALFUNCTION REMOVAL RESTORES THE SELECTED BUS TO A NON-GROUNDED CONDITION.

EVENTS: NONE.

ED06 FAILURE OF 6.9KV ABT

TYPE: GENERIC, RB

A)	ACB 1562	C)	ACB 1582
B)	ACB 1572	D)	ACB 1592

CAUSE: FAILURE OF 52/B CONTACT IN SAT FEEDER BRKR CLOSE CIRCUIT

REF: 20E-1-4030 AP08,12
20E-1-4030 AP16,20

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THE SELECTED MALFUNCTION CAUSES THE SAT FEEDER BREAKER TO FAIL TO AUTOMATICALLY CLOSE ON THE ABT ACTUATION AFTER THE UAT's LOSE POWER. THE AFFECTED BUS CURRENT, VOLTAGE, AND WATTAGE INDICATIONS DECREASE TO ZERO. FOR EACH OF THE MALFUNCTIONS; THE FOLLOWING LOADS ARE LOST AND ANNUNCIATORS ACTUATE:

BUS 156: 1A FEEDWATER PUMP, HEATER DRAIN PUMP 1B, AND RCP 1B, ANNUNCIATORS 20-B7 "BUS 156 FD BKR 1561 TRIP", AND 20-D7 "BUS 156 VOLT LOW".

BUS 157: HEATER DRAIN PUMP 1A, HEATER DRAIN PUMP 1C, AND RCP 1A, ANNUNCIATORS 20-B5 "BUS 157 FD BKR 1571 TRIP", AND 20-D5 "BUS 157 VOLT LOW".

BUS 158: RCP 1C, CONDENSATE AND CONDENSATE BOOSTER PUMPS 1B AND 1D, ANNUNCIATORS 20-A8 "BUS 158 FD BKR 1582 TRIP", AND 20-D8 "BUS 158 VOLT LOW".

BUS 159: START-UP FEEDWATER PUMP, RCP 1D, CONDENSATE AND CONDENSATE BOOSTER PUMPS 1A AND 1C, ANNUNCIATORS 20-A6 "BUS 159 FD BKR 1592 TRIP", AND 20-D6 "BUS 159 VOLT LOW".

THE REACTOR WILL TRIP IF > P-8.

EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY THE OPERATOR MANUALLY CLOSING THE SAT FEEDER BREAKER.

MALFUNCTION REMOVAL RESTORES THE SELECTED BREAKER OPERATION TO NORMAL.

EVENTS: NONE.

ED07 LOSS OF 4160V BUS

TYPE: GENERIC, RB

A) 4160V BUS 141 (BRKR 1412)
B) 4160V BUS 142 (BRKR 1422)
C) 4160V BUS 143 (BRKR 1431)
D) 4160V BUS 144 (BRKR 1441)

CAUSE: FAULTY GROUND OVERCURRENT RELAY ACTUATION (UAT FEED BREAKER ON ED07C/D)

REF:
20E-1-4030 AP23
20E-1-4030 AP32
20E-1-4030 AP41
20E-1-4050 AP47
20E-1-4006 SERIES

```
*********************************
*             NOTE              *
*   IF SAT BREAKER IS FEEDING   *
*   BUS 143 OR 144, THIS        *
*   MALFUNCTION WILL NOT        *
*   DE-ENERGIZE THE BUS         *
*********************************
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PLT STA: REACTOR AT FULL POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, A FAULTY GROUND OVERCURRENT CONDITION WILL DEVELOP ON THE FEEDER BREAKER TO THE AFFECTED 4160 VOLT BUS. THIS RESULTS IN THE FEEDER BREAKER TRIPPING OPEN WITH THE SELECTED BUS GOING DEAD. LOADS NORMALLY POWERED BY THE AFFECTED BUS WILL BE DEENERGIZED WITH THE EFFECTS ON PLANT OPERATION BEING DIFFERENT DEPENDING ON WHICH BUS IS SELECTED. THE LOADS EFFECTED ARE LISTED IN THE 20E-1-4006 SERIES KEY DIAGRAMS.

BUS 141: ANNUNCIATORS 21-A7 "BUS 141 FD BKR 1412 TRIP", AND 21-C7 "BUS 141 OVERLOAD OR VOLT LOW" ACTUATE STRIPPING THE 141 ESF BUS OF IT'S ASSOCIATED LOADS. THE 1A DIESEL GENERATOR AUTO STARTS, BUT IT'S SUPPLY BKR WILL NOT CLOSE DUE TO THE ACB 1412 LOCKOUT.

BUS 142: ANNUNCIATORS 22-A7 "BUS 142 FD BKR 1422 TRIP", AND 22-C7 "BUS 142 OVERLOAD OR VOLT LOW" ACTUATE STRIPPING THE 142 ESF BUS OF IT'S ASSOCIATED LOADS. THE 1B DIESEL GENERATOR AUTO STARTS, BUT IT'S SUPPLY BKR WI LL NOT CLOSE DUE TO THE ACB 1422 LOCKOUT.

BUS 143: ANNUNCIATORS 21-A1 "BUS 143 FD BKR 1431 TRIP", AND 21-C1 "BUS 143 VOLT LOW" ACTUATE STRIPPING THE 143 BUS OF IT'S ASSOCIATED LOADS. THE SAT FEEDER BKR WILL NOT CLOSE DUE TO THE ACB 1431 LOCKOUT.

BUS 144: ANNUNCIATORS 22-A1 "BUS 144 FD BKR 1441 TRIP", AND 22-C1 "BUS 144 VOLT LOW" ACTUATE STRIPPING THE 144 BUS OF IT'S ASSOCIATED LOADS. THE SAT FEEDER BKR WILL NOT CLOSE DUE TO THE ACB 1441 LOCKOUT.

MALFUNCTION REMOVAL RESTORES THE SELECTED FEEDER GROUND OVERCURRENT RELAY TO NORMAL.

EVENTS: NONE.

ED08 LOSS OF FEED TO 480V NON-ESF BUS OR MCC

TYPE: GENERIC, RB

A) BUS 133V
B) BUS 133X
C) BUS 133Y
D) BUS 133Z
E) BUS 134V
F) BUS 134X
G) BUS 134Y
H) BUS 134Z
I) MCC 133U1
J) MCC 133V1
K) MCC 133V2
L) MCC 133V3
M) MCC 133V4
N) MCC 133V5
O) MCC 133X1A
P) MCC 133X1B
Q) MCC 133X3
R) MCC 133X4
S) MCC 133X6
T) MCC 133Y1
U) MCC 133Z2
V) MCC 134U1
W) MCC 134V1
X) MCC 134V2
Y) MCC 134V3
Z) MCC 134V4
AA) MCC 134V5
AB) MCC 134V6
AC) MCC 134X5
AD) MCC 134X7
AE) MCC 134Y1
AF) MCC 134Y2
AG) MCC 134Y3
AH) MCC 134Z2
AI) MCC 134Z4
AJ) MCC 033W3
AK) BUS 035

CAUSE: SUPPLY BREAKER(S) INADVERTENTLY OPENED

REF: 20E-1-4001A
20E-1-4007 SERIES
20E-1-4008 SERIES

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED BUS OR MCC FEEDER BREAKER TO TRIP DEENERGIZING THAT BUS OR MCC. LOADS NORMALLY POWERED BY THE AFFECTED BUS OR MCC WILL BE DEENERGIZED WITH THE EFFECTS ON PLANT OPERATION BEING DIFFERENT DEPENDING ON WHICH BUS IS SELECTED. THE LOADS AFFECTED ARE LISTED IN THE 20E-1-4007 AND 20E-1-4008 SERIES KEY DIAGRAMS. ANNUNCIATORS WILL RESPOND ACCURATELY TO PLANT EFFECTS.

MALFUNCTION REMOVAL RESTORES THE SELECTED SUPPLY BKR TO NORMAL RESTORING BUS OR MCC POWER.

EVENTS: NONE.

ED09 LOSS OF FEED TO 480 VOLT ESF BUS OR MCC

TYPE: GENERIC, RB

A) BUS 131X
C) BUS 132X
E) MCC 131X1
F) MCC 131X1A
G) MCC 131X2
H) MCC 131X2A
I) MCC 131X3
J) MCC 131X4
K) MCC 131X5
M) MCC 132X1
N) MCC 132X2
O) MCC 132X2A
P) MCC 132X3/132X5
Q) MCC 132X4
R) MCC 132X4A

CAUSE: SUPPLY BREAKER(S) INADVERTENTLY OPENED

REF: 20E-1-4001A
20E-1-4007 SERIES
20E-1-4008 SERIES

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED BUS OR MCC FEEDER BREAKER TO TRIP DEENERGIZING THAT BUS OR MCC. LOADS NORMALLY POWERED BY THE AFFECTED BUS OR MCC WILL BE DEENERGIZED WITH THE EFFECTS ON PLANT OPERATION BEING DIFFERENT DEPENDING ON WHICH BUS IS SELECTED. THE LOADS EFFECTED ARE LISTED IN THE 20E-1-4007 AND 20E-1-4008 SERIES KEY DIAGRAMS. ANNUNCIATORS WILL RESPOND ACCURATELY TO PLANT EFFECTS.

MALFUNCTION REMOVAL RESTORES THE SELECTED SUPPLY BKR TO NORMAL RESTORING BUS OR MCC POWER.

EVENTS: NONE.

ED10 LOSS OF 120 VAC ESF CONSTANT VOLTAGE XFMR

TYPE: GENERIC, RB

A) BUS 111 CVT
B) BUS 112 CVT
C) BUS 113 CVT
D) BUS 114 CVT

CAUSE: SUPPLY BREAKER INADVERTENTLY OPENED

REF: 20E-1-4001A
20E-1-4012 SERIES

PLT STA: REACTOR AT POWER (AFFECTED CVT IN-SERVICE)

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE AFFECTED INSTRUMENT BUS CVT BREAKER TO TRIP, DEENERGIZING THE BUS IF IT WAS SUPPLYING THE BUS. THE LOADS EFFECTED WILL BE IDENTIFIED BY THE 120 VAC INSTRUMENT BUS 20E-1-4012 SERIES KEY DIAGRAMS. THE MAJOR LOADS BEING REACTOR PROTECTION SYSTEM CHANNELS, SAFEGUARDS FEATURES, NUCLEAR INSTRUMENTATION POWER, AND MAIN CONTROL BOARD RECORDER POWER SUPPLIES. SEE MALFUNCTION ED11 FOR PARTIAL LISTING OF LOADS LOST.

THERE ARE NO ANNUNCIATORS DIRECTLY ASSOCIATED WITH THE LOSS OF A 120 VAC INSTRUMENT BUS, BUT THE LOADS AFFECTED BY THE LOSS WILL GENERATE ALARMS WHEN POWER IS LOST.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED SUPPLY BREAKER TO NORMAL.

EVENTS: NONE.

ED11 120 VAC INSTRUMENT BUS INVERTER FAILURE

TYPE: GENERIC, RB

A) INVERTER 111
B) INVERTER 112
C) INVERTER 113
D) INVERTER 114

CAUSE: FAULTY SHUNT TRIP DEVICE ON OUTPUT BREAKER (4CB)

REF: 20E-1-4002E
20E-1-4002F
20E-1-4030 IP01
20E-1-4030 IP02
20E-1-4030 IP03
20E-1-4030 IP04

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED INVERTER SHUNT TRIP BREAKER TO OPEN DUE TO A FAULTY OVERCURRENT SIGNAL. THE ASSOCIATED BUS ANNUNCIATOR 4-A5/B5/C5/D5 "BUS 111/112/113/114 INVERTER TROUBLE" ACTUATES ON LOSS OF AC OUTPUT FROM THE INVERTER.

THE LOADS EFFECTED WILL BE IDENTIFIED BY THE 120 VAC INSTRUMENT BUS 20E-1-4012 SERIES KEY DIAGRAMS. THE MAJOR LOADS BEING REACTOR PROTECTION SYSTEM CHANNELS, SAFEGUARDS FEATURES, NUCLEAR INSTRUMENTATION POWER, AND MAIN CONTROL BOARD RECORDER POWER SUPPLIES. IN ADDITION, POWER TO 1PA01J-8J AC CONTROLLERS IS LOST. SUMMARY OF AC CONTROLLERS LOSING POWER:

1PA01J (INST BUS 111) - WHPS BISTABLES TRIP (LEVEL AND PRESS), OPΔT/OTΔT RUNBACK BISTABLES TRIP.

1PA02J (INST BUS 112) - WHPS BISTABLES TRIP (LEVEL AND PRESS), OPΔT/OTΔT RUNBACK BISTABLES TRIP.

1PA03J (INST BUS 113) - WHPS BISTABLES TRIP (LEVEL AND PRESS), OPΔT/OTΔT RUNBACK BISTABLES TRIP.

1PA04J (INST BUS 114) - WHPS BISTABLES TRIP (LEVEL AND PRESS), OPΔT/OTΔT RUNBACK BISTABLES TRIP.

1PA05J (INST BUS 111) - CC-182 WILL NOT AUTO OPEN, BA/PW DEV ALARM (IF FLOW EXISTS) , LT-185 WILL NOT AUTO DIVERT CV-112A, LT-185 <5% BISTABLE TRIPS, STEAM DUMPS WILL NOT ARM ON C-7, PORV 455A WILL NOT AUTO OPEN, PRT PRESS HIGH BISTABLE TRIPS, LT-460 FAILING LOW HAS NO EFFECT ON LETDOWN OR HEATERS.

1PA06J (INST BUS 112) - CC-183 WILL NOT AUTO OPEN, C-5 ROD STOP OCCURS, PORV 456 WILL NOT AUTO OPEN, LT-459 FAILING LOW HAS NO EFFECT ON LETDOWN OR HEATERS, ON A 5% LEVEL DEV PZR HEATERS WILL NOT ENERGIZE.

1PA07J (INST BUS 113) - PORV 456 WILL NOT AUTO OPEN(< 2185#).

1PA08J (INST BUS 114) - LT-112 <5% BISTABLE TRIPS, NO AUTO M/U WILL OCCUR, C-16 ACTUATES, C-11 ACTUATES, PORV 456 WILL NOT AUTO OPEN IN ARM LOW TEMP, PORV 455A WILL NOT AUTO OPEN (< 2185#).

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY MANUALLY TRANSFERRING THE INSTRUMENT BUS TO THE CVT TO REENERGIZE THE BUS.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED INVERTER OVERCURRENT RELAY TO NORMAL.

EVENTS:
1) SER 32-87
2) LER 20-1-89-001
3) LER 20-1-89-005
4) LER 20-2-88-008

ED11

IS 721 I FORSYTH (INPO) 10-NOV-87 10:17 EST
Subject: SER 32-87, INSUFFICIENT VENTILATION CAPACITY FOR DC EQUIPMENT

UNIT:	TURKEY POINT 3 AND 4
DOC NO/LER NO:	50-250/87015
EVENT DATE:	5/22/87
NSSS/AE:	WESTINGHOUSE/BECHTEL
REFERENCES:	SOER 83-03, "INVERTER FAILURES"
	O&MR 0071, "INVERTER CAPACITOR EXCESSIVE OPERATING TEMPERATURES"
	O&MR 0243, "TEMPERATURE INDUCED INVERTER FAILURES"

SUMMARY:

AN ENGINEERING EVALUATION REVEALED THAT A COMPLETE LOSS OF THE HEATING, VENTILATING, AND AIR CONDITIONING SYSTEM (HVAC) FOR THE DC EQUIPMENT/INVERTER ROOMS COULD DISABLE THE DC SYSTEMS BECAUSE OF HIGH TEMPERATURES.

THIS EVENT IS SIGNIFICANT BECAUSE DESIGN CHANGES THAT ADDED FIRE BARRIERS IN COMPLIANCE WITH 10CFR50 APPENDIX R REQUIREMENTS AND INSTALLED ADDITIONAL ELECTRICAL EQUIPMENT INCREASED HEAT LOADS TO LEVELS THAT COULD DISABLE DC SYSTEMS. THIS PROBLEM WAS NOT IDENTIFIED UNTIL AFTER THE DESIGN CHANGES WERE INSTALLED.

DESCRIPTION:

AN ENGINEERING EVALUATION DURING DESIGN BASIS RECONSTITUTION IDENTIFIED THREE SCENARIOS THAT COULD RESULT IN A TOTAL LOSS OF VENTILATION SYSTEMS TO THE DC EQUIPMENT/INVERTER ROOMS. THESE SCENARIOS INCLUDED A LOSS OF OFFSITE POWER WITH A SINGLE ACTIVE FAILURE, LOSS OF AN AIR CONDITIONING UNIT DURING NORMAL OPERATIONS, AND A FIRE RESULTING IN A LOSS OF VENTILATION TO THE DC EQUIPMENT/INVERTER ROOMS. SUBSEQUENT TO THE LOSS OF VENTILATION, TEMPERATURES IN THE DC EQUIPMENT/INVERTER ROOMS MIGHT REACH 172 DEGREES F UNLESS OPERATOR ACTION WAS TAKEN. TEMPERATURES EXCEEDING 135 DEGREES F COULD ADVERSELY AFFECT THE OPERATION OF EQUIPMENT IN THESE AREAS INCLUDING THE 125V DC BATTERIES, BATTERY CHARGERS, INVERTERS, CONSTANT VOLTAGE TRANSFORMERS, DC MOTOR CONTROL CENTERS, AND TRANSFER SWITCHES. THE AIR CONDITIONING UNITS CURRENTLY SUPPLYING COOLING TO THESE ROOMS ARE NOT SAFETY-RELATED.

THE TURKEY POINT FSAR DID NOT SPECIFICALLY ADDRESS THE DESIGN BASIS FOR THE DC EQUIPMENT/INVERTER ROOM, AND THE ORIGINAL PLANT DESIGN MAY NOT HAVE CONSIDERED THIS VENTILATION SYSTEM AS AN ESSENTIAL SUPPORT SYSTEM FOR THE DC SYSTEMS. VARIOUS CHANGES TO THE PLANT DESIGN HAD INCREASED

EQUIPMENT HEAT LOADS AND INTERFERED WITH VENTILATION FLOW, BUT THE IMPACT OF THESE CHANGES ON NORMAL AND MAXIMUM ROOM TEMPERATURES HAD NOT BEEN RECOGNIZED IN THE MODIFICATION ANALYSIS. AS A RESULT OF DESIGN BASIS RECONSTITUTION REVIEWS, THE ADDITIONAL HEAT LOADS AND VENTILATION INTERFERENCES WERE IDENTIFIED TO BE SIGNIFICANT CHANGES TO PLANT DESIGN.

A STANDARD "ENGINEERING PACKAGE" CHECKLIST IS NOW BEING UTILIZED ON ALL PLANT MODIFICATIONS TO ENSURE THAT ALL DESIGN CONCERNS ARE ADDRESSED.

IN ADDITION, THE PLANT HAS IMPLEMENTED PROCEDURE CHANGES TO INSURE PROPER OPERATOR ACTION FOR A LOSS OF VENTILATION, INSTALLED PORTABLE FANS TO IMPROVE THE AIR FLOW DISTRIBUTION, INCREASED MONITORING OF ROOM TEMPERATURES, AND ENHANCED PERIODIC MAINTENANCE ACTIVITIES TO INCREASE THE RELIABILITY AND PERFORMANCE OF THE COOLING EQUIPMENT.

COMMENTS:

1. ASSESSMENTS OF DESIGN CHANGES THAT OBSTRUCT OR REDIRECT AIR FLOW OR INCREASE THE HEAT LOADS SHOULD CONSIDER THE ADEQUACY OF THE HEATING, VENTILATING, AND AIR CONDITIONING SYSTEMS. COOLING TO SAFETY RELATED EQUIPMENT SHOULD BE DESIGNED TO SATISFY THE SINGLE FAILURE CRITERIA.

2. AS PREVIOUSLY DISCUSSED IN THE REFERENCES, DESIGN EVALUATIONS SHOULD CONSIDER THE EFFECTS OF INTERNAL CABINET TEMPERATURE IN ADDITION TO ROOM AMBIENT EFFECTS. COMPONENT TEMPERATURES CAN EXCEED THEIR DESIGN CAPABILITIES DUE TO THE INTERNAL CABINET ENVIRONMENT.

3. FSAR DISCUSSIONS ARE SUMMARIES THAT DO NOT ALWAYS STATE EACH OF THE DESIGN REQUIREMENTS FOR EACH PLANT SYSTEM. ALL CHANGES TO PLANT CONFIGURATIONS SHOULD BE EVALUATED FOR BOTH THE CONCERNS DISCUSSED IN THE FSAR AND OTHER PERTINENT DESIGN CONSIDERATIONS (E.G., SYSTEM AND COMPONENT DESIGN SPECIFICATIONS, APPLICABLE INDUSTRY CODES AND STANDARDS, DESIGN ASSUMPTIONS, PREVIOUS REGULATORY COMMITMENTS, ETC.) TO ENSURE THAT THE COMPLETE IMPACT ON PLANT SAFETY AND RELIABILITY IS IDENTIFIED AND CONSIDERED.

DISTRIBUTION OF THIS SER SHOULD INCLUDE THE DESIGN ENGINEERING, TECHNICAL SUPPORT, AND OPERATIONS MANAGERS.

UTILITIES AND MEMBERS ARE REQUESTED TO PROVIDE FEEDBACK ON SIMILAR OCCURRENCES AND SOLUTIONS AT THEIR PLANTS TO THE INFORMATION CONTACT LISTED BELOW.

LICENSEE EVENT REPORT (LER)

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Title (4)

Reactor Trip Due To Spurious Loss of Output Voltage on Instrument Inverter 112

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0 2	0 6	8 9	8 9	0 0 1	0 1	0 3	0 7	8 9	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 3

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__	20.402(b)	__	20.405(c)	X	50.73(a)(2)(iv)	__	73.71(b)
__	20.405(a)(1)(i)	__	50.36(c)(1)	__	50.73(a)(2)(v)	__	73.71(c)
__	20.405(a)(1)(ii)	__	50.36(c)(2)	__	50.73(a)(2)(vii)	__	Other (Specify in Abstract below and in Text)
__	20.405(a)(1)(iii)	__	50.73(a)(2)(i)	__	50.73(a)(2)(viii)(A)		
__	20.405(a)(1)(iv)	__	50.73(a)(2)(ii)	__	50.73(a)(2)(viii)(B)		
__	20.405(a)(1)(v)	__	50.73(a)(2)(iii)	__	50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER: AREA CODE	
Joe Doyle, Technical Staff Engineer, Ext. 2660	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
__ Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

At 1322 on February 16, 1989 Unit 1 was in Mode 3 with all the control rods inserted and the Reactor Trip Breakers closed. A momentary loss of output voltage on Instrument Inverter 112 caused a Reactor Trip Signal due to Intermediate Range High Flux Bistable from Channel N36 reverting to its ESF safe configuration. The opening of the Reactor Trip Breakers coincident with RCS Average Temperature less than 564 degrees fahrenheit caused a Feedwater Isolation Signal. At 1323 the Feedwater Isolation Signal was reset and normal feedwater flow was reestablished. Personnel in the area at the time of the event were independently interviewed, their activities did not place them in contact with Instrument Inverter 112 physically or electrically. The momentary Loss of Instrument Inverter output voltage is still under investigation. The Unit 1 instrument inverters are scheduled for an inspection during the next outage of opportunity. This report will be supplemented should the root cause be determined. There have been previous occurrences of reactor trips involving instrument inverters, however the previous events were not the result of spurious perturbations on the inverter. The corrective actions for those events were implemented addressing both root and contributing causes. Previous corrective actions are not applicable to this event.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year		Sequential Number		Revision Number	Page (3)
Braidwood Unit 1	0 5 0 0 0 4 5 6	8 9	-	0 0 1	-	0 1	0 2 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 1; Event Date: February 6, 1989; Event Time: 1322;

Mode: 3 - Hot Standby; Rx Power: 0%;

RCS [AB] Temperature/Pressure: 553 degrees F/2235 psig

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of the event that contributed to the severity of the event.

At 1322 on February 16, 1989 Unit 1 was in Mode 3 with all the control rods inserted and the Reactor Trip Breakers closed. A momentary loss of output voltage on Instrument Inverter 112 caused a Reactor Trip Signal due to Intermediate Range High Flux bistable from Channel N36 reverting to its ESF safe configuration on the loss of power. The opening of the Reactor Trip Breakers coincident with RCS Average Temperature Less than 564 degrees fahrenheit caused a Feedwater Isolation Signal. At 1323 the Feedwater Isolation Signal was reset and normal feedwater flow was reestablished.

The appropriate NRC notification via the ENS phone system was made at 1415 pursuant to 10CFR50.72(b)(2)(ii).

This event is being reported pursuant to 10CFR50.73(a)(2)(iv) - any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature, including the Reactor Protection System.

C. CAUSE OF EVENT:

Prior to the event a ventilation damper that controls room temperature had failed. This failure resulted in the electrolyte temperature on 125 vdc batter 112 decreasing below its required minimum temperature. Electrical Maintenance installed a temporary electric heater as a compensatory measure to recover temperature until the damper was declared operable.

The cause of the reactor trip signal being generated was the Nuclear Instrumentation System (NIS) (IG) Intermediate Range High Flux bistable for channel N36 reverting to its ESF safe configuration on a momentary loss of power.

Operating, Electrical Maintenance, and Technical Staff personnel were in the area of Instrument Inverter 112 at the time of the occurrence. Operating was monitoring 125 vdc battery 112 electrolyte temperature. Electrical Maintenance and Technical Staff personnel were working on the failed ventilation damper.

Personnel in the area at the time of the event were independently interviewed. The results of the interviews concluded that the activities in progress at the time did not place them in contact with Instrument Inverter 112 physically or electrically.

Based on a review of the sequence of events recorder, the duration of the inverter loss of output voltage was 0.211 seconds. This short time frame precludes the possibility of personnel error relative to switch operation or mispositioning.

The momentary loss of output voltage has not [illegible] peated nor is there any history or voltage perturbations on the Instrument Inverters at Braidwood.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

D. SAFETY ANALYSIS:

This event had no effect on the safety of the plant or public as the unit was in Hot Standby at the time of the occurrence, all systems operated as designed and the plant remained in a stable condition.

The worst case condition is unit operation prior to the manual block of the intermediate range high flux trip. This is procedurally directed at approximately 16% reactor power. A reactor trip would have occurred as it did in this occurrence.

Three other instrument busses connected to their associated inverters were operable and available to provide redundant instrumentation. The instruments and controls powered from Inverter 112 have redundant power supplies, multiple coincidence logics, or otherwise fail to their ESF safe configurations, as did the Intermediate Range High Flux Trip in this event.

E. CORRECTIVE ACTIONS:

Immediate corrective action were to reset the feedwater isolation, reestablish normal feedwater flow to the steam generators, and initiate an investigation into the cause of the event.,

The root cause of the momentary loss of instrument inverter output voltage is still under investigation. The Unit 1 instrument inverters are scheduled for an inspection during the next outage of opportunity. The results of this inspection will be tracked to completion by action item 456-200-89-02501. This report will be supplemented should the root cause be determined.

F. PREVIOUS OCCURRENCES:

There have been previous occurrences of reactor trips involving instrument inverters, however the previous events were not the result of spurious perturbations on the inverter. The corrective actions for those events were implemented addressing both root and contributing causes. Previous corrective actions are not applicable to this event.

G. COMPONENT FAILURE DATA:

This event was not the result of component failure, nor did any components fail as a result of this event.

EC 11

LICENSEE EVENT REPORT (LER)

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Facility Name (1)	Docket Number (2)	Page (3)
Braidwood Unit 1	0 5 0 0 0 4 5 6	1 of 0 4

Title (4)

Reactor Shutdown Due to Failed Instrument Inverter

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0 4	2 3	8 9	8 9	0 0 5	0 0	0 5	1 8	8 9	None	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 8 8

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	Other (Specify in Abstract below and in Text)
20.405(a)(1)(iii)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER	
	AREA CODE	
Steve Eich, Technical Staff Engineer Ext. 2333	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	E F	I N V T	W 1 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

At 1349 on April 22, 1989 Instrument Inverter 111 tripped. At 1409 Instrument Bus 111 was re-energized from its associated Constant Voltage Transformer (CVT). At 1800 Maintenance began troubleshooting the inverter. The Inverter was not repaired within the twenty four hours provided for in the Technical Specifications. A plant shutdown was required. At 1248 on April 23 a reactor shutdown was initiated. An Unusual Event was declared. At 1259 the appropriate NRC notification was made. At 1820 Unit 1 entered Mode 3, Hot Standby. At 1500 on April 24 a shorted capacitor was found. The capacitor was replaced and the inverter was successfully started. At 0221 on April 25 the Instrument Bus 111 was transferred from the CVT to the inverter. At 1030 the inverter was declared operable. The cause of this event was the shorted capacitor. This was attributed to normal wear. The corrective actions were to re-energize instrument bus 111 from the CVT and repair the inverter. A cleaning and inspection program with a frequency of 18 months will be implemented. Capacitors will be replaced with a frequency of 3 years starting with the September 1989 refueling outage. There have been previous occurrences of loss of inverter output voltage. Previous corrective actions are not applicable to this event.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood Unit 1; Event Date: April 22, 1989; Event Time: 1349;

Mode: 1 - Power Operation; Rx Power: 88%;

RCS [AB] Temperature/Pressure: 579 degrees F/2225 psig

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of the event which contributed to the severity of the event.

At 1349 on April 22, 1989 the Bus 111 Inverter Trouble alarm annunciated in the Main Control Room. The Nuclear Instrumentation (NI) [IG] trip signals associated with Power Range NI Channel N41 which is powered from Instrument Bus 111 [EF], reverted to their loss of power, 'tripped' configuration. An Equipment Operator (non Licensed Operator) (EO) was dispatched to Instrument Inverter 111. The EO identified that the Inverter 111 AC input breaker 1CB had tripped on high DC input voltage. 1BwOA INST1, Nuclear Instrumentation Malfunction, was entered for failed power range N41. 1BwOA ELEC-2, Loss of Instrument Bus, was entered for Loss of Instrument Bus 111.

The following Limiting condition for Operations Action Requirements (LCOAR) were entered:

1BwOS 8.3.1-1a, Onsite Power Distribution, for loss of Instrument Bus 111.

1BwOS 3.1-1a, Reactor Trip System Instrumentation, for loss of power range channel N41, and

1BwOS 3.2-1a, Engineered Safety Features Actuation System Instrumentation, for Solid State Protection System, and Reactor Trip instrumentation respectively being inoperable.

At 1409 Instrument Bus 111 was re-energized from its associated Constant Voltage Transformer (CVT). Emergency procedures 1BwOA INST-1 and ELECT-2 were exited. LCOAR's BwOS 3.1-1a and BwOS 3.2-1a were exited. All reactor trip alerts were cleared. Nuclear Work Request (NWR) A30622 was written to repair the inverter.

At 1800 Electrical Maintenance Department (EMD) began troubleshooting the inverter. They found fuse 1FU blow

From 1801 April 22, 1989 to 1247 on April 23, 1989: Troubleshooting efforts continued to determine the cause of the Inverter failure. Initially the fuse 1FU, two silicon controlled rectifiers (SCR) and a gating board were replaced. A restart of the Inverter was attempted. Fuse 1FU 'blew' again. Next the gating board was replaced with another new gating board, numerous diodes and capacitors were checked and found to be in good working order. The fuse 1FU was replaced again. A restart of the inverter was attempted again. Once again fuse 1FU blew. Before both restart attempts the Inverter vendor, Westinghouse, had been contacted for assistance. After the failure of the second restart attempt it was determined that a vendor representative was required to investigate the cause of the inverter failure. A representative of the vendor was directed to report to the station.

At 1247 it was concluded that Inverter 111 would not be repaired within the twenty four hours provided for in Action Statement b. of Technical Specification 3.8.3.1. Per the additional requirements of the Action Statement, a plant shutdown would be required, while troubleshooting activities continued.

B. DESCRIPTION OF EVENT: (CONT'D)

At 1248 on April 23, 1989 a power decrease for reactor shutdown was initiated. A Generating Station Emergency Plan (GSEP) Unusual Event was declared per Emergency/Implementing Procedure, BwZP 200-1A1 Emergency Action Levels 3a. and 6a.; Equipment described in the Technical Specifications is degraded such that a Limiting Condition for Operation requires a shutdown and power decrease for reactor shutdown has commenced.

At 1252 the Nuclear Accident Reporting System (NARS) notification was made to the State of Illinois, to declare the Unusual Event.

The appropriate NRC notification via the ENS phone system was made at 1259 pursuant to:

10CFR50.72(b)(1)(i)(A) - The initiation of any nuclear plant shutdown required by the plant's Technical Specifications

10CFR50.72(a)(1)(i) - The declaration of any of the Emergency Classes specified in the Licensee's approved Emergency Plan.

At 1820 Unit 1 entered Mode 3, Hot Standby.

At 0515 on April 24, 1989 a cooldown was initiated to place Unit 1 in Mode 5, Cold Shutdown.

At 1500 EMD, with the assistance of the vendor representative, found commutating capacitor 2C shorted. Capacitor 2C is used to shut off the silicon controlled rectifiers (SCR). With the capacitor shorted, the SCRs conduct at all times. As a result of conducting at all times they draw excessive current and blow fuse 1FU. Capacitor 2C was replaced and the inverter was successfully started.

At 0102 on April 25, 1989 Unit 1 entered Mode 5 and the GSEP Unusual Event was terminated.

At 0106 the NARS notification to terminate the event was made.

At 0144 the appropriate NRC notification via the ENS phone system was made pursuant to 10CFR50.72(c)(1)(iii) - a termination of the Emergency Class.

At 0221 Instrument Bus 111 was transferred from the CVT to the inverter.

At 1030 the inverter had been connected to Instrument Bus 111 for greater than eight hours. The inverter performed satisfactorily indicating that the repair efforts had been successful. The inverter was declared operable. LCOAR 1BwOS 8.3.1-1a was exited.

There were no manual or automatic safety actuations. The rapid response of Operating Personnel transferring the instrument bus to its associated CVT decreased the severity of this event.

This event is being reported pursuant to 10CFR50.73(a)(2)(i) - The completion of any nuclear plant shutdown required by the plant's Technical Specifications.

C. CAUSE OF EVENT:

The root cause of this event was a component failure. The short in capacitor 2C caused the SCRs to draw current continually. This continual current draw caused fuse 1FU to blow. The failure of the capacitor is being attributed to normal wear.

D. SAFETY ANALYSIS:

This event had no effect on the safety of the plant or the public. All systems operated as designed.

Instrument Bus 111 was de-energized for 20 minutes. The Train A Solid State Protection System (SSPS) slave relays are powered from bus 111. During the time the bus was de-energized, Train A Engineered Safety Features (ESF) [JE] equipment could not have actuated automatically. Manual operation of all train A components was available. All Train B ESF equipment was operable and available. They would have automatically actuated upon demand. Both Trains of reactor trip capability were available during the event.

Under the worst case condition an accident occurring without power available to the automatic actuation relays of an ESF train, the operation of one of the redundant ESF trains is adequate to meet the assumptions of all accident analysis in the Updated Final Safety Analysis Report (UFSAR). This is enveloped in Section 15 of the UFSAR.

CORRECTIVE ACTIONS:

The immediate corrective actions were to re-energize Instrument Bus 111 from the CVT and repair the inverter.

Long term corrective actions will be to implement a cleaning and inspection program with a frequency of 18 months. This program will include checking terminal tightness and physical condition of the components. The establishment of this program will be tracked to completion by action items 456-200-89-06401.

All electrolytic and oil filled capacitors will be replaced with a frequency of 3 years starting with the September 1989 refueling outage. This will be tracked to completion by action item 456-200-89-06402.

F. PREVIOUS OCCURRENCES:

DVR/LER Number	Title
20-1-87-043/87-010	Inadvertent Loss of Power to Instrument 111 Resulting in a Reactor Trip Due to Personnel Error - Contractor
20-1-89-025/89-001	Reactor Trip Due to Spurious Loss of Output Voltage on Instrument Inverter 112

The corrective action were implemented addressing both root and contributing causes. Previous corrective actions are not applicable to this event.

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	MFG Part Number
Westinghouse	Capacitor	20 MFD, 600V	1589A93H23

LICENSEE EVENT REPORT (LER)

ED11

Facility Name (1): Braidwood, Unit 2

Docket Number (2): 0| 5| 0| 0| 0| 4| 5| 7

Page (3): 1 of 0 3

Title (4) Inadequate Capacitor Connection Results in Degraded Instrument Bus Voltage and Subsequent Reactor Trip

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0| 2	2| 0	8| 8	8| 8	0 0 8 / 0 | 2 | 4	0 | 0	0 | 3	1| 7	8| 8	NONE	0| 5| 0| 0| 0| | |
										0| 5| 0| 0| 0| | |

OPERATING MODE (9): 3

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	X 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name: Harold Hill, Technical Staff Engineer Ext. 2486

TELEPHONE NUMBER: AREA CODE 8| 1| 5 | 4| 5| 8| -| 2| 8| 0| 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	A| A	C| A| P| *	G| 0| 8| 0	NO					

SUPPLEMENTAL REPORT EXPECTED (14)

__ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15): Month | Day | Year

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

At 0626 on February 20, 1988, during the performance of startup test BwSU RD-70, there was a loss of power to Instrument Bus 212. This resulted in a reactor trip signal being generated, and caused the reactor trip breakers to open. This loss of power also caused a boron dilution protection system actuation. An equipment operator was sent to the bus and he re-energized it from its constant voltage transformer. Action to prevent recurrence will be to conduct an inspection of all "Fast-on-Connectors" for heat damage to the same connections for each inverter on both units.

There have been no previous occurrences.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Braidwood, Unit 2	0 5 0 0 0 4 5 7	8 8	- 0 0 8	- 0 0	0 2 OF 0 3

T Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 2 ; Event Date: February 20, 1988 ; Event Time: 0626
MODE: 3 - Hot Standby ; Rx Power: 0% ; RCS [AB] Temperature/Pressure: 557°F/2235 psig

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of the event which contributed to the severity of the event.

At 0626 on February 20, 1988, during the performance of BwSU RD-70, Control Rod Drive Mechanism Operational Test, there was a loss of power to Instrument Bus [EF] 212. This caused a Bus 212 Trouble Alarm [IB] in the control room. The loss of Instrument Bus 212 also caused the Reactor Trip Breakers [JG] to open due to the loss of control power to source range N-32 and intermediate range N-36 [IG]. Loss of control power to the source range also resulted in a Boron Dilution Protection System (BDPS) signal.

An equipment operator was dispatched to inverter 212 and found the inverter output voltage had degraded to 50 volts. Inverter 212 is the normal feed to Instrument Bus 212. The input AC and DC voltages were within their specified ranges. The inverter was shut down by the operator.

The equipment operator attempted to re-energize the bus from the Constant Voltage Transformer (CVT) [EA] but was unsuccessful as its output breaker tripped. The startup procedure for the CVT was repeated by the operator and the bus was energized. At 0652 on February 20, 1988, the plant was returned to a stable condition.

Operator actions neither increased nor decreased the severity of the event.

The appropriate NRC notification via the ENS phone system was made at 0712 on February 20, 1988, pursuant to 10CFR50.72(b)(2)(ii).

This event is being reported pursuant to 10CFR50.73(a)(2)(iv) - Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature, including the Reactor Protection System.

C. CAUSE OF EVENT:

The root cause of this event was a pre-service installation error by the vendor which resulted in a bad connection on a capacitor for the auto transformer in 2IP06E. The improper connection yielded excessive resistance which produced heat and caused it to burn off. This produced an imbalance in one phase of the transformer and a degraded output voltage condition.

The cause of the constant voltage transformer output breaker tripping is indeterminate as the symptoms would not repeat. Should this recur, then it will be addressed in a new report.

D. SAFETY ANALYSIS:

This event had no effect on the safety of the plant or public since the unit had not yet been taken critical and no radioactive effluent had been produced. All Engineered Safety Feature equipment functioned as designed. Under worst case conditions of the unit being at full power, the unit would have responded in the same manner as in this event. The CVT and emergency 125 VDC batteries used for instrument bus backup power supplies were available throughout the event.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

E. CORRECTIVE ACTIONS:

The immediate corrective action was to restore the instrument bus to the CVT. The inverter was repaired by replacing the capacitor and connecting leads.

Action to prevent recurrence includes a full inspection of all fast-on connections and an inspection for heat damage to the same connections for each inverter on both units. This will be tracked to completion by Action Item 457-200-88-02401.

There are no corrective actions proposed for the output breaker of the CVT since the symptoms could not be repeated. Should this recur, then it will be investigated and a new report will be submitted.

F. PREVIOUS OCCURRENCES:

There have been no previous occurrences of inverter capacitor or lead failures regardless of cause.

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
General Electric	13ufd Trimming Capacitor	770836	23L6066

ED12 LOSS OF DC DISTRIBUTION BUS

TYPE: GENERIC, RB

A) DIST CENTER 111
B) DIST CENTER 112

CAUSE: FAULTY OVERCURRENT TRIP OF AKR-50 BATTERY BREAKER

REF: 20E-1-4030 DC05
20E-1-4030 DC08
20E-1-4010 SERIES

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE BATTERY INPUT BREAKER (AKR-50) TO DIST CENTER 111/112 TO TRIP OPEN. ANNUNCIATORS 21/22-E6 "125V DC BATT 111/112 MAIN BRKR TRIP", AND 21/22-E10 "125V DC DIST PNL 111/113 (112/114) VOLT LOW" ACTUATE. THE REACTOR TRIPS DUE TO THE FEEDWATER REG. VALVES FAILING CLOSE (S/G LOW-LOW LEVEL). VARIOUS MAIN CONTROL BOARD, SWITCHGEAR, AND ESF EOUIPMENT THROUGHOUT THE PLANT LOSE THFIR 125V DC CONTROL POWER AS INDICATED ON THE CONTROL BOARDS.

THE EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY REENERGIZING THE DISTRIBUTION CENTER FROM THE UNIT TWO CROSS-TIE BREAKER.

MALFUNCTION REMOVAL RESTORES THE FAULTY OVERCURRENT SIGNAL TO NORMAL.

EVENTS: NONE.

ED13 DC CONTROL POWER FAILURE (4160V)

TYPE: GENERIC, RB

A) BUS 141 DC CONTROL POWER
B) BUS 142 DC CONTROL POWER
C) BUS 143 DC CONTROL POWER
D) BUS 144 DC CONTROL POWER

CAUSE: SUPPLY BREAKERS INADVERTENTLY OPENED

REF: 20E-1-4030 DC05
20E-1-4030 DC07
20E-1-4030 DC08
20E-1-4030 DC10

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED 4160V AC BUS TO LOSE DC CONTROL POWER. ANNUNCIATORS 21-B7/B1 "BUS 141/143 CONT PWR FAILURE" AND 22-B7/B1 "BUS 142/144 CONT PWR FAILURE" ACTUATE. LOSS OF CONTROL POWER PREVENTS THE OPERATOR FROM OPERATING ANY OF THE EQUIPMENT BREAKERS ASSOCIATED WITH THAT 125V DC CONTROL PANEL. BREAKER INDICATION LIGHTS ON THE MAIN CONTROL BOARD ALSO EXTINGUISH.

MALFUNCTION REMOVAL RESTORES THE SELECTED 125V DC CONTROL POWER SUPPLY BREAKERS TO NORMAL.

EVENTS: NONE.

ED14 DC CONTROL POWER FAILURE (480V)

TYPE: GENERIC, RB

A) BUS 131X	B) NOT USED
C) BUS 132X	D) NOT USED

CAUSE: SUPPLY BREAKERS INADVERTENTLY OPENED

REF: 20E-1-4030 DC05
20E-1-4030 DC07
20E-1-4030 DC08
20E-1-4030 DC10

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED 480V AC BUS TO LOSE DC CONTROL POWER. ANNUNCIATOR 21-B10 "BUS 131X CONT PWR FAILURE" OR 22-B10 "BUS 132X CONT PWR FAILURE" ACTUATES. LOSS OF CONTROL POWER PREVENTS THE OPERATOR FROM OPERATING THE BREAKER ASSOCIATED WITH THAT 125V DC CONTROL PANEL. BREAKER INDICATION LIGHTS ON THE MAIN CONTROL BOARD ALSO EXTINGUISH.

MALFUNCTION REMOVAL RESTORES THE SELECTED 480V AC 125V DC CONTROL POWER SUPPLY BREAKERS TO NORMAL.

EVENTS: NONE.

ED15 345 KV BUS FAULT

TYPE: GENERIC, RB

A)	BUS 1	F)	BUS 11
B)	BUS 3	G)	BUS 14
C)	BUS 4	H)	BUS 15
D)	BUS 7	I)	BUS 9
E)	BUS 8	J)	BUS 10

CAUSE: GROUND FAULT

REF: 20E-0-4001
20E-0-4102D
20E-0-4102E
20E-0-4104B
20E-0-4104B1
20E-0-4104C
AC ELECTRICAL DISTRIBUTION SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A SELECTED 345 KV BUS TO DEVELOP A GROUND FAULT WHICH RESULTS IN TRIPPING OPEN IT'S RESPECTIVE BUS TIE BREAKERS.

A COMBINATION OF SEVERAL BREAKER TRIPS INITIATED AT THE SAME TIME WILL CAUSE EITHER A LOSS OF LINES L-0104, L-2003, L-2004, L-2001, L-2002, L-0103, THE ASSOCIATED MAIN GENERATOR, AND/OR THE SYSTEM AUXILIARY TRANSFORMERS. THE PLANT WILL RESPOND ACCURATELY TO THE EFFECTS OF ANY OF THESE MALFUNCTIONS. ANY ATTEMPT TO RECLOSE THE BREAKER WILL RESULT IN IT TRIPPING IMMEDIATELY. ANNUNCIATORS WILL RESPOND ACCURATELY TO PLANT EFFECTS.

MALFUNCTION REMOVAL RESTORES THE SELECTED 345 KV SWITCHYARD BUS TO NORMAL.

EVENTS: NONE.

ED16 LOSS OF FEED TO 120V NON-ESF PANEL

TYPE: GENERIC, RB

A)	MCC	033W3	120V
B)	MCC	133U1	120V
C)	MCC	133V2	120V
D)	MCC	133V4	120V
E)	MCC	133V5	120V
F)	MCC	133X1A	120V
G)	MCC	133X1B	120V
H)	MCC	133X3	120V #1
I)	MCC	133X3	120V #2
J)	MCC	133X4	120V
K)	MCC	133Y1	120V
L)	MCC	133Z2	120V
M)	MCC	134U1	120V
N)	MCC	134V1	120V
O)	MCC	134V2	120V
P)	MCC	134V3	120V
Q)	MCC	134V4	120V
R)	MCC	134V6	120V
S)	MCC	134X5	120V
T)	MCC	134Y2	120V

CAUSE: SUPPLY BREAKER(S) INADVERTENTLY OPENED

REF: 20E-1-4007 SERIES
20E-1-4008 SERIES

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED MCC FEEDER BREAKER TO TRIP DEENERGIZING THAT 120V NON-ESF PANEL. LOADS NORMALLY POWERED BY THE AFFECTED 120V NON-ESF PANEL WILL BE DEENERGIZED WITH THE EFFECTS ON PLANT OPERATION BEING DIFFERENT DEPENDING ON WHICH 120V NON-ESF PANEL IS SELECTED. THE LOADS EFFECTED ARE LISTED IN THE 20E-1-4007 AND 20E-1-4008 SERIES KEY DIAGRAMS. ANNUNCIATORS WILL RESPOND ACCURATELY TO PLANT EFFECTS.

MALFUNCTION REMOVAL RESTORES THE SELECTED SUPPLY BKR TO NORMAL RESTORING BUS OR MCC POWER.

EVENTS: NONE.

ED17 LOSS OF FEED TO 120V ESF PANEL

TYPE: GENERIC, RB

A) MCC 131X1 120V
B) MCC 131X2 120V
C) MCC 131X3 120V
D) MCC 132X1 120V
E) MCC 132X2 120V
F) MCC 132X3 120V #1
G) MCC 132X3 120V #2

CAUSE: SUPPLY BREAKERS(S) INADVERTANTLY OPENED

REF: 20E-1-4007 SERIES
20E-1-4008 SERIES

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED MCC FEEDER BREAKER TO TRIP OPEN DEENERGIZING THAT 120V ESF PANEL. LOADS NORMALLY POWERED BY THE AFFECTED 120V ESF PANEL WILL BE DEENERGIZED WITH THE EFFECTS ON PLANT OPERATION BEING DIFFERENT DEPENDING ON WHICH 120V ESF PANEL IS SELECTED. THE LOADS EFFECTED ARE LISTED IN THE 20E-1-4007 AND 20E-1-4008 SERIES KEY DIAGRAMS. ANNUNCIATORS WILL RESPOND ACCURATELY TO PLANT EFFECTS.

MALFUNCTION REMOVAL RESTORES THE SELECTED SUPPLY BKR TO NORMAL RESTORING BUS OR MCC POWER.

EVENTS: NONE.

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

EG01	MAIN GENERATOR AUTO VOLTAGE REGULATOR FAILURE
EG02	MAIN GENERATOR EXCITER FAILURE
EG03	MAIN GENERATOR FIELD FORCING (VOLTAGE REGULATOR)
EG04	BASE FOLLOWER UNIT FAILS TO TRACK
EG05	MAIN POWER TRANSFORMER TRIP
EG06	D/G FAILURE TO FLASH GENERATOR FIELD
EG07	D/G ELECTRIC SPEED CONTROL FAILURE
EG08	D/G SEIZURE
EG09	D/G DIFFERENTIAL OVERCURRENT TRIP

EG01 MAIN GENERATOR AUTO VOLTAGE REGULATOR FAILURE

TYPE: DISCRETE, RB

CAUSE: FAULTY GENERATOR VOLTAGE BALANCE RELAY (660G1X)

REF: 20E-1-4030 MP02, 07

PLT STA: MAIN GENERATOR ON LINE

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE MAIN GENERATOR AUTO VOLTAGE REGULATOR WILL TRIP, TRANSFERRING VOLTAGE CONTROL TO THE BASE ADJUSER. ANNUNCIATOR 19-B8 "GENERATOR VOLT REG TRIP" IS ACTUATED. THE TRANSFER OF CONTROL WILL BE SMOOTH. THE SEVERITY OF THE TRANSIENT WILL DEPEND ON THE DIFFERENCE BETWEEN THE TWO SETTINGS.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY MANUALLY ADJUSTING THE BASE ADJUSTER TO KEEP THE AUTO AND MANUAL SETTINGS NULLED.

MALFUNCTION REMOVAL WILL RESTORE THE FAILED VOLTAGE REGULATOR TO NORMAL.

EVENTS: NONE

EG02 MAIN GENERATOR EXCITER FAILURE

TYPE: DISCRETE, RB

CAUSE: FAULTY 41T RELAY ACTUATION

REF: 20E-1-4030 MP06

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE PERMANENT MAGNET GENERATOR (PMG) OUTPUT BREAKER TO TRIP OPEN DUE TO A FAULTY 41T RELAY ACTUATION.

TURBINE/GENERATOR TRIPS ARE ACTUATED RESULTING IN A REACTOR TRIP. ALL MAIN GENERATOR MAIN CONTROL BOARD INDICATIONS DECREASE SHARPLY TO THEIR DEENERGIZED POSITION. ANNUNCIATOR 19-A8 "PMG OUTPUT BRKR TRIP" ACTUATES WHEN THE PMG 41 BREAKER OPENS.

MALFUNCTION REMOVAL RESTORES THE 41T RELAY TO NORMAL ALLOWING THE PMG OUTPUT BKR TO BE CLOSED AGAIN.

EVENTS: NONE

EG03 MAIN GENERATOR FIELD FORCING (VOLTAGE REGULATOR)

TYPE: DISCRETE, RV 0-100%

CAUSE: WTA VOLTAGE REGULATOR SIGNAL MIXER FAILURE

REF: 20E-1-4030 MP18
VENDOR MANUAL; WTA REG PROT DRAWER

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION RESULTS IN THE BASE ADJUSTER MOTOR DRIVEN POT BEING DRIVEN TO THE POSITION SELECTED BY THE MALFUNCTION SEVERITY.

AT SEVERITY LEVELS LESS THAN THE ACTUAL (CORRECT) AMOUNT OF EXCITATION, THE EXCITER OUTPUT AMPS WILL DECREASE, AND OUTPUT VARS WILL DECREASE (WILL GO "IN"). THE MAIN GENERATOR MAY TRIP ON LOSS OF GENERATOR FIELD.

AT SEVERITY LEVELS GREATER THAN THE ACTUAL (CORRECT) AMOUNT OF EXCITATION, THE EXCITER FIELD AMPS WILL INCREASE, AND OUTPUT VARS WILL INCREASE. IF MALF SEVERITY IS HIGH ENOUGH, THE MAXIMUM EXCITATION LIMITER (MXL) WILL NOT PREVENT EXCESSIVE EXCITATION CURRENT. ANNUNCIATOR 19-B6 "GENERATOR FIELD FORCING" ACTUATES AT 100 AMPS EXCITER FIELD CURRENT. ANNUNCIATOR 19-C8 "GENERATOR VOLT REG TROUBLE" ACTUATES AT 102 AMPS EXCITER FIELD CURRENT. AT 109 AMPS THE GEN EXCITATION SYSTEM PROT DRAWER WILL TRIP THE AUTO VOLT REG OFF (TD OF 120 SEC AT 109 AMPS, AND 0.3 SEC AT 147 AMPS). AFTER THE AUTO VOLT REG TRIPS OFF, IF THE EXCITATION AMPS ARE NOT LOWERED BELOW 109 AMPS USING THE BASE ADJUSTER, THE GENERATOR WILL TRIP.

THE OPERATOR CAN MITIGATE THE EFFECTS BY PLACING THE AUTO VOLT REG IN OFF AND USING THE BASE ADJUSTER.

MALFUNCTION REMOVAL RESTORES THE WTA VOLTAGE REGULATOR SIGNAL MIXER TO NORMAL.

EVENTS: 1) LER 20-1-87-052

EG03

OTHER UNIT NO

SCHEDULE CAT: TEST CONDITION 2
CONDITION TO:

PERSON: CENTRAL FILES DATE SENT: 11/17/87

1-87-336: REACTOR TRIP ON
MAIN POWER TRANSFORMER
PROTECTION RELAY ACTUATION FOR
UNKNOWN REASON

SEVERITY LEVEL: LER NO: 87-052 CRITERION

RESPONSE DUE DATE INSPECTOR HRS:
WALRATH TO BY SET BY SYSTEM : 77 IP
TSLL/BORTCH CORRECTIVE ACT :
R/F OUTAGE
PRIORITY
IR RESPONSE
BY/NN PROCEDURE:

D: L: Q: Z: NUD: DNS ESS. PIC: NFS

ORIGINAL DUE DATE: 10/24/87

STATUS: COMPLETE
RDY FOR CLOSURE: 10/07/87
ORIG EXIT DATE : 10/19/87
ORIG CLOSED : 10/20/87
DATE COMPLETED : 11/03/87

10CFR50.73(A)(2)(IV)
IR 336/87-035
FITZPATRICK

20-1-87-336

SYNCHRONIZING TO THE GRID, RECEIVED A MAIN TRANSFORMER
PROTECTION TRIP CAUSING A GENERATOR TRIP -> TURBINE TRIP ->
REACTOR TRIP. PERFORMED BWEP-0. ALL SYSTEMS RESPONDED NORMALLY.

NRC RED PHONE NOTIFICATION MADE, 09/24/87.

ENGINEER'S COMMENTS

S. P. MASTERS 09/25/87

RESIDENT INSPECTOR, NRC REGION III, 09/28/87, 1200
D. P. GALLE/T. J. MAIMAN, NSD/VP, 09/28/87, 0900

REPORTABLE 10CFR50.73(A)(2)(IV)

[illegible] 33600 (CONT)

[illegible]

[illegible]

[illegible]

[illegible] CONDITIONS PRIOR TO EVENT:
*
[illegible] 1; EVENT DATE: SEPTEMBER 2[illegible], 1987; EVENT TIME: 2219
MODE: 1 - POWER OPERATION, RX POWER: 13%; RCS (AB)
TEMPERATURE/PRESSURE: 557 DEGREES F/2235 PSIG
*

B. DESCRIPTION OF EVENT:
*
THERE WERE NO SYSTEMS OR COMPONENTS INOPERABLE AT THE BEGINNING
OF THE EVENT WHICH CONTRIBUTED TO THE SEVERITY OF THE EVENT.
*
AT 2219 ON SEPTEMBER 24, 1987, [illegible] POWER ASCENSION [illegible], WAS IN PROGRESS. DURING THE PROCESS OF SYNCHRONIZING THE MAIN GENERATOR (TB) TO THE SYSTEM GRID, THE MAIN TRANSFORMER OVEREXCITATION RELAY ACTUATED. THIS ACTIVATED THE EIGHTY-SIX LOCKOUT RELAY WHICH CAUSED THE MAIN GENERATOR, MAIN POWER TRANSFORMER (EL) AND UNIT AUXILIARY TRANSFORMER (EA) TO BE ELECTRICALLY ISOLATED FROM THE SYSTEM GRID. ACTIVATION OF THE EIGHTY-SIX LOCKOUT RELAY ALSO INITIATES A GENERATOR TRIP, TURBINE TRIP (JJ), AND REACTOR TRIP (JG) TO PROTECT THE EQUIPMENT FROM POSSIBLE ELECTRICAL FAULT DAMAGE. ALL SAFETY SYSTEMS OPERATED AS DESIGNED. STABLE PLANT CONDITIONS WERE ESTABLISHED AT 0100 ON SEPTEMBER 25, 1987.
*
OPERATING PERSONNEL RESPONDED APPROPRIATELY AND ENTERED PROCEDURE [illegible]-0, REACTOR TRIP OR SAFETY INJECTION.
*
[illegible] NRC NOTIFICATION VIA THE ENS PHONE SYSTEM WAS MADE AT 2302 ON SEPTEMBER 24, 1987 PURSUANT TO [illegible]50.72(B)(2)(II).
*
[illegible] BEING REPORTED PURSUANT TO 10CFR50.73(A)(2)(IV) [illegible] EVENT OR CONDITION THAT RESULTED IN MANUAL OR AUTOMATIC ACTUATION OF ANY ENGINEERED SAFETY FEATURE, INCLUDING THE REACTOR PROTECTION SYSTEM.
*

[illegible] CAUSE OF EVENT:
*
[illegible] CAUSE OF THE EVENT IS UNKNOWN. OPERATIONAL ANALYSIS

[illegible] (CONT):

[illegible] (OAD) PERSONNEL WERE CALLED [illegible] TO INVESTIGATE THE [illegible]. THE OAD INVESTIGATION ONLY YIELDED A LOOSE CONNECTION [illegible] CONTACT BLOCK LOCATED ON THE SECONDARY SIDE OF THE [illegible] PHASE 25 KV REGULATING POTENTIAL TRANSFORMER. THIS FAILURE [illegible] TO THE EVENT. HOWEVER, IT WAS NOT THE ROOT CAUSE. DURING THE SUBSEQUENT PLANT STARTUP AND POWER ASCENSION, OAD [illegible] CONTINUED TO MONITOR THE [illegible] AND [illegible] PHASE POTENTIAL TRANSFORMERS. INCLUDED IN THIS COMPREHENSIVE MONITORING PROGRAM WERE THE OVEREXCITATION RELAY, WHICH ACTIVATED DURING THIS EVENT, AND THE VOLTAGE REGULATOR. THIS TESTING REVEALED THAT [illegible] READINGS WERE NORMAL.

THE POTENTIAL THAT THIS WAS OPERATOR ERROR IS RULED OUT, AS THE OPERATORS WERE OBSERVED EXECUTING THE EVOLUTION IN ACCORDANCE [illegible] PROCEDURE RWGP 100-3. IT MUST ALSO BE POINTED OUT THAT THIS TASK DOES NOT REQUIRE SPECIAL TRAINING OR PRACTICE.

SHOULD ANY ADDITIONAL INFORMATION CONCERNING THE CAUSE OF THIS EVENT BECOME AVAILABLE, THIS REPORT WILL BE SUPPLEMENTED. SHOULD THIS EVENT RECUR, A NEW REPORT WILL BE ISSUED.

D. SAFETY ANALYSIS:

THERE WERE NO SAFETY CONSEQUENCES ASSOCIATED WITH THIS EVENT AS ALL SYSTEMS OPERATED AS DESIGNED. UNDER WORSE CASE CONDITIONS OF A HIGHER INITIAL POWER LEVEL, THE SYSTEMS WOULD HAVE RESPONDED THE SAME AS THIS EVENT AND THEREFORE, THERE WOULD BE NO SAFETY CONSEQUENCES. THE TIE CAPABILITY FOR NON-SAFETY RELATED BUSES TO SAFETY RELATED BUSES ON UNIT [illegible] WERE AVAILABLE THROUGHOUT THE EVENT.

E. CORRECTIVE ACTIONS:

[illegible] CORRECTIVE ACTIONS WAS TO RESTORE THE NORMAL [illegible] LINEUP AND PLACE THE UNIT IN A STABLE CONFIGURATION.

[illegible] CONTACT BLOCK WAS REPLACED.

[illegible] POTENTIAL TRANSFORMERS, OVEREXCITATION RELAY, AND THE VOLTAGE REGULATOR WERE MONITORED AS PART OF COMPREHENSIVE [illegible] FOR THE RETURN TO SERVICE OF UNIT [illegible]. THIS MONITORING [illegible] STARTUP AND POWER ASCENSION REVEALED NO ADDITIONAL [illegible] WITH RESPECT TO THE CAUSE. THEREFORE, NO ADDITIONAL ACTIONS ARE CONTEMPLATED AT THIS TIME.

[illegible] 33509 (CONT)

[illegible] SUMMARY (CONT):

*

[illegible] PREVIOUS OCCURRENCES:

*

[illegible] OCCURRENCES OF OVEREXCITATION RELAY ACTUATION.

*

*

[illegible] COMPONENT FAILURE DATA:

*

[illegible]	NOMENCLATURE	MODEL #	MFG PART #
[illegible]	CONTACT, SECONDARY, AND RECEPTACLE	25 KV REGULATING TRANSFORMER	2325083

[illegible]

EG04 BASE FOLLOWER UNIT FAILS TO TRACK

TYPE: DISCRETE, RB

CAUSE: LOSS OF INPUT SIGNAL FROM WTA VOLTAGE REGULATOR

REF: 20E-1-4030 MP08

PLT STA: MAIN GENERATOR ON LINE

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE BASE ADJUSTER FOLLOWER UNIT WILL FAIL AS IS. AS THE DIFFERENCE BETWEEN THE ACTUAL VOLTAGE AND THE BASE ADJUST VOLTAGE SETTING INCREASES, THE INDICATOR ON THE EXCITER VOLTAGE REGULATOR VOLTS METER (NULL METER),1EI-MP022, WILL MOVE FURTHER AWAY FROM ZERO.

MALFUNCTION REMOVAL WILL RESTORE THE FAILED BASE ADJUST FOLLOWER UNIT TO NORMAL.

EVENTS: NONE

EG05 MAIN POWER TRANSFORMER TRIP

TYPE: GENERIC, NRB

A) 1E MAIN POWER TRANSFORMER
B) 1W MAIN POWER TRANSFORMER

CAUSE: LIGHTNING STRIKE

REF: AC ELECTRICAL POWER SYSTEM DESCRIPTION
MAIN GENERATOR SYSTEM DESCRIPTION
20E-1-4030 MP01, MP04

PLT STA: MAIN POWER TRANSFORMERS ENERGIZED AND CONNECTED TO GRID

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED MAIN POWER TRANSFORMER WILL DEVELOP A GROUND ON THE LOW VOLTAGE SIDE FROM A LIGHTNING STRIKE. THE GENERATOR WILL TRIP WHEN RELAY 86G1B IS ACTUATED BY A MAIN TRANSFORMER SUDDEN PRESSURE CONDITION. ANNUNCIATOR 19-B4 "MAIN XFMR SUDDEN PRESS GEN TRIP" IS ACTUATED.

THE TRANSFORMER TRIP WILL RESULT IN A GENERATOR TRIP WHICH CAUSES ANNUNCIATOR 19-E2 "GENERATOR LOCKOUT RELAY TRIP" TO ACTUATE. THE GENERATOR TRIP WILL RESULT IN A TURBINE TRIP WHICH IN TURN WILL TRIP THE REACTOR IF REACTOR POWER IS GREATER THAN 30%. THE TRANSFORMER TRIP ALSO RESULTS IN ACTUATION OF THE TRANSFORMER DELUGE. THE MAIN TRANSFORMER COOLING SYSTEM SUPPLY BREAKERS WILL ALSO TRIP OPEN AND ACTUATE ANNUNCIATOR 19-A10 "MAIN XFMR 1E/1W FD BRKR TRIP". OTHER ANNUNCIATORS WILL RESPOND ACCURATELY TO THE TRANSIENT.

THE SIMULATOR MUST BE RESET TO RECOVER FROM THIS MALFUNCTION.

EVENTS: NONE

EG06 D/G FAILURE TO FLASH GENERATOR FIELD

TYPE: GENERIC, RB

A) 1A D/G
B) 1B D/G

CAUSE: FAILURE OF 14FX & 14FRX CONTACTS IN FIELD CIRCUIT

REF: 20E-1-4020B
20E-1-4021B
20E-1-4030 DG31
20E-1-4030 DG32
20E-1-4030 DG51
20E-1-4030 DG52

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION HAS NO AFFECT ON THE D/G INITIALLY. DURING AN AUTO/MANUAL STARTUP OF THE SELECTED DIESEL, THE FAILED 14FX & 14FRX CONTACTS DON'T CLOSE WHEN THE SPEED IS AT 250 RPM. THIS IS INDICATED BY THE LACK OF OUTPUT VOLTAGE, FREQUENCY INDICATION, AND THE BUS ALIVE LIGHT INDICATION. ANNUNCIATOR 21/22-C8 "DG 1A/1B TROUBLE/FAIL TO START" ACTUATES. INSERTING THIS MALFUNCTION WHILE THE DIESEL IS ALREADY RUNNING HAS NO AFFECT ON OPERATION.

MALFUNCTION REMOVAL RESTORES PROPER OPERATION TO THE FIELD FLASHING CIRCUIT.

EVENTS: NONE

EG07 D/G ELECTRIC SPEED CONTROL FAILURE

TYPE: GENERIC, RB

A) 1A D/G
B) 1B D/G

CAUSE: FAILURE OF ELECTRIC SPEED CONTROL CIRCUIT

REF: 20E-1-4020B
20E-1-4021B

PLT STA: SELECTED DIESEL IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION WHILE THE DIESEL GENERATOR IS OPERATING LOADED/UNLOADED CAUSES THE FREQ TO INCREASE TO 62.0 Hz, AND WILL NOT ALLOW THE OPERATOR ANY SPEED CONTROL. LOADING, UNLOADING, PARALLEL TRANSFERS, OR SPEED REDUCTION FOR COOLDOWNS FROM THE CONTROL ROOM CANNOT BE PERFORMED. INSERTING THE MALFUNCTION WHILE THE D/G IS PARALLELED WITH OFFSITE POWER CAUSES THE SELECTED DIESEL TO OVERLOAD AND TRIP. ANNUNCIATOR 21/22-C8 "DG 1A/1B TROUBLE/FAIL TO START" ACTUATES.

MALFUNCTION REMOVAL RESTORES THE ELECTRIC SPEED CONTROL CIRCUIT TO NORMAL.

EVENTS: 1) DVR 06-02-88-107

EG07

DEVIATION INVESTIGATION REPORT (DIR)

Facility Name: Byron Nuclear Power Station

Form Rev 2.0

PAGE 1 OF 0 2

Title: 2A DIESEL GENERATOR LOAD SWINGS IN TEST

EVENT DATE MONTH	DAY	YEAR	DIR NUMBER STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	REPORT DATE MONTH	DAY	YEAR	OPERATING MODE	POWER LEVEL
1 0	0 5	8 8	0 6	0 2	8 8	—	1 0 7	—	0 0	1 1	1 8	8 8	1	4 1

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER AREA CODE	
W. Walter, Assistant Tech Staff Supervisor Ext. 2244	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 10/05/88 / 0141

Unit 1 MODE 1 - Refueling Rx Power 0% RCS [AB] Temperature/Pressure 90°F/ Atmospheric

Unit 2 MODE 2 - Power Operation Rx Power 41% RCS [AB] Temperature/Pressure 568F / 2235 PSI

B. DESCRIPTION OF EVENT:

On 10/05/88, at 0141 the 2A Diesel Generator [EK] (DG) was started per Byron Monthly Operability Surveillance 2BOS 8.1.1.2.a-1. The 2A DG failed to reach rated speed and subsequently tripped on "Incomplete Sequence". Limiting Condition for OPeration Action Requirement (LCOAR) 2BOS 8.1.1-1a was immediately initiated for one Diesel Generator inoperable. At 0213 the 2A DG was started for troubleshooting. Following a successful start and warmup period, the DG output breaker was closed. During the loading sequence, at approximately 4500 Kilowatts (KW), load control became unstable. After several load swings of 400-700 KW, the 2A DG was unloaded and shutdown. Subsequent troubleshooting indicated intermittent operation of the 4EX3 relay. At 1655, on 10/05/88, the 2A was restarted per the monthly operability surveillance. No abnormal conditions were observed. At 1855, on 10/05/88, the operability surveillance was completed and LCOAR 2BOS 8.1.1-1a was exited. The 2A Diesel Generator was inoperable for 17 hours and 14 minutes during this event. There were no systems or components inoperable prior to the event that contributed to the event.

C. CAUSE OF EVENT:

The Root Cause for this event was intermittent operation of the 4EX3 relay. The 4EX3 relay aligns the electronic governor with a fixed resistance in the emergency mode for a constant speed reference. In the test mode, the relay lines up a motorized potentiometer for variable speed reference. Bench testing of the 4EX3 relay revealed a slight resistance across the test mode contacts (relay deenergized), and significant resistance across the emergency mode contacts (relay energized). The intermittent operation of the relay caused the 2A DG to loose its speed reference, which in turn caused the incomplete start, and the load swings.

D. SAFETY ANALYSIS:

There were no safety consequences as a result of this event. the 2B Diesel Generator was fully operable during the event, and could have supplied emergency AC power if required.

E. CORRECTIVE ACTIONS:

The 4EX3 relay was removed and replaced with a new relay. Long term corrective actions include replacement of the existing 4EX3 relays in all four Byron Diesel Generators with more reliable relays. This replacement is being done under Byron Modifications M6-1-87-166, and M6-2-87-166 for Units One and Two respectively. The modification has been completed on Unit One, and has thus far proven to be successful. The modification will be installed on the Unit Two engines during the 1/89 refueling outage. Action Item Record (AIR) 455-512-87-0257 will track completion of the modification.

F. PREVIOUS OCCURRENCES:

DVR NUMBER	TITLE
6-1-88-018	1B DG Load Sensor and 4EX3 Relay Failure.

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Agastat	Relay	GPDR	

b) RESULTS OF NPRDS SEARCH:

No other similar failures found in search

c) RESULTS OF NWR SEARCH:

No Additional NWR History

EG08 D/G SEIZURE

TYPE: GENERIC, RB

A) 1A D/G
B) 1B D/G

CAUSE: SHAFT SEIZURE

REF: 20E-1-4030 DG38
20E-1-4030 DG58

PLT STA: SELECTED DIESEL GENERATOR IN OPERATION

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED DIESEL GENERATOR WILL EXPERIENCE A SHAFT SEIZURE. THIS RESULTS IN THE DIESEL GENERATOR COMING TO AN IMMEDIATE HALT. THE DIESEL GENERATOR OUTPUT BREAKER TRIPS WITH ANNUNCIATOR 21/22-C8 "DG 1A/1B TROUBLE/FAIL TO START" ACTUATING. IF THIS MALFUNCTION IS ACTIVATED PRIOR TO STARTING THE SELECTED DIESEL GENERATOR, THAT DIESEL WILL FAIL TO START.

MALFUNCTION REMOVAL WILL RESTORE THE FAILED DIESEL GENERATOR SHAFT TO NORMAL.

EVENTS: 1) LER 06-02-88-003

EG08

LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
Byron, Unit 2	0 5 0 0 0 4 5 5	1 of 0 4

Title (4) TECHNICAL SPECIFICATION ACTION STATEMENT NOT SATISFIED DURING UNINTENDED 2B DIESEL GENERATOR INOPERABILITY DUE TO INCORRECT DRAWING

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0 3	2 9	8 8	8 8	— 0 0 3 —	0 0	0 4	2 6	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 3

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	Other (Specify in Abstract below and in Text)
20.405(a)(1)(iii)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	Telephone Number (Area Code)
D. Brindle, Operating Engineer, Extension 2218	8 1 5 2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

On March 29, 1988, the 2B Diesel Generator (DG) left bank starting air system was removed from service to repair a leaking valve. Upstream and downstream isolation points were chosen from a Piping and Instrumentation Drawing (P&ID) to permit the maintenance. The P&ID incorrectly represented the actual piping arrangement in the plant, therefore, the right bank of the starting air system was also isolated during the intended isolation of the left bank. This condition resulted in the isolation of all starting air from the 2B DG, thus, making it inoperable. The inoperability was identified on March 31, 1988, when an Equipment Operator noticed that the "Unit Available for Emergency" indicating light was extinguished, and the Technical Specification Limiting Condition for Operation Action Requirement was implemented.

The following corrective actions have been or are being taken:

1. Caution cards and labels explaining the piping discrepancies have been hung locally and in the main control room.

2. The P & ID will be corrected to represent actual plant conditions.

3. DG auxiliary equipment labels will have Byron part numbers.

4. The operating rounds procedure will require periodic checks of the "Unit Available for Emergency" indicating light.

5. Training programs will address the indicating light.

Similar events have not occurred previously.

(2015M/0206M)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Byron, Unit 2	0 | 5 | 0 | 0 | 0 | 4 | 5 | 5	8 | 8	- 0 | 0 | 3 -	0 | 0	0 | 2 OF 0 |

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 3/29/88 / 0815

Unit 2 MODE 1 - Power Operations Rx Power 93% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On March 29, 1988, preparations were in progress to remove the 2B Diesel Generator (DG)[EK] left bank starting air system from service in order to repair the left bank moisture separator drain valve (2SA141D) which had been leaking by its seat. A licensed reactor operator reviewed the applicable Piping and Instrumentation Drawing (P & ID) M-54 sheet 4B to determine upstream and downstream isolation points to safely permit the maintenance. By closing the left bank air receiver outlet valve 2SA140D, the left bank air receiver (2DG01SB-TD) would be isolated from the work area. By closing the left bank starting air valve, 2DG5182B, the left bank starting air system would be isolated from the work area. Diesel Generator operability requires only one of the two starting air banks to be in service, therefore, no Technical Specification Limiting Condition for Operation Action Requirement (LCOAR) was implemented prior to . isolating the left bank. At 2236 on March 29, 1988, a non-licensed Equipment Operator (EO) closed valves 2SA140D and 2DG5182B.

On March 31, 1988 at approximately 0400, an EO informed the licensed Senior Reactor Operator Shift Control Room Engineer (SCRE) that the "Unit Available for Emergency" indicating light at the 2B DG local control panel was not illuminated. There was no procedural requirement to periodically check the status of this light, but the EO noted this abnormal condition during a general inspection of the DG local control panel. This indicating light is illuminated if one of two starting air banks is pressurized, and one of two direct current (DC) power supplies is energized. The SCRE immediately initiated a nuclear work request to have the problem investigated. The initial investigation was conducted by Electrical Maintenance technicians and indicated that the right starting air bank pressure switch, which provides an input to the "Unit Available for Emergency" indicating light, was defective. The left starting air bank was known to be out of service and depressurized for maintenance. The combination of a defective right bank pressure switch and a left bank out of service condition explained the deenergized indicating light. Therefore, the LCOAR was not entered, since the problem appeared to be an indication failure unrelated to DG operability. Subsequent investigation determined the pressure switch to be in satisfactory working condition.

On March 31, 1988 at 0730, a non-licensed Technical Staff Engineer joined the investigation. Following a pressure check of both starting air banks on the 2B DG, the Technical Staff Engineer concluded that both starting air banks were depressurized to atmospheric pressure. At 0815 the engineer notified the SCRE of this condition. The SCRE immediately initiated "LCOAR Electrical Power Systems AC Sources Tech Spec LCO 3.8.1.1 Operating Procedure" (LCOAR 2BOS 8.1.1-1a) for the 2B DG inoperability.

Investigation continued to determine the cause of the depressurization of both starting air banks. The investigation revealed that the actual air bank piping arrangement did not agree with the P & ID piping arrangement. Specifically, the P & ID shows valve 2DG5182B isolating air from the left bank starting air system (2DG01SB-TD) at the engine. In actuality, valve 2DG5182B isolates the right bank starting air system at the engine. Therefore, when the EO closed 2DG5182B on March 29, he actually isolated right bank starting air from the DG. When he closed 2SA140D, he isolated left bank starting air from the DG. At this point the 2B DG became inoperable, since neither bank of starting air was available to start the DG.

B. DESCRIPTION OF EVENT: (Continued)

At 1315 on March 31, 1988, the actual air start piping arrangement had been verified, the right bank had been restored to service, the 2B DG had been declared operable and LCOAR 2BOS 8.1.1-1a was exited. There were no other systems or components inoperable prior to this event that contributed to the event.

From 2236 on March 29, 1988, until 0815 on March 31, 1988, the 2B DG was inoperable and the appropriate Technical Specification LCOAR was not satisfied. This event is reportable in accordance with 10CFR50.73(a)(2)(i)(B) as a violation of the plant's Technical Specifications.

C. CAUSE OF EVENT:

The intermediate cause of this event was an incorrect representation of the starting air piping on P & ID M-54, sheet 4B. The root cause of the errors in the drawing is indeterminate. The subject P & ID shows the left bank starting air receiver (2DG01SB-TD) feeding starting air valve 2DG5182B, and the right bank starting air receiver (2DG01SB-TC) feeding starting air valve 2DG5183B. In the actual piping arrangement, 2DG01SB-TD feeds valve 2DG5183B, and 2DG01SB-TC feeds valve 2DG5182B. Sections of both the left and right banks of air start piping are buried in concrete, therefore, the ED had no way to verify the connections between the receivers and the engine. The P & ID was used by the licensed reactor operator to determine maintenance isolation points that resulted in inoperability of the 2B DG.

D. SAFETY ANALYSIS:

There were no safety consequences as a result of this event. The 2A diesel generator was fully operable during this event and could have supplied emergency electrical power to Unit 2 if required. In addition, both Unit 1 diesel generators were fully operable and could have been electrically crosstied to Unit 2 if required.

Maintenance personnel were not endangered during this event, because the work area was completely isolated from sources of pressurized air.

E. CORRECTIVE ACTIONS:

The following corrective actions were taken or are planned to prevent reoccurrence of this event.

1. The 1A, 1B, and 2A diesel generator starting air systems were reviewed for similar discrepancies. No problems were identified on the Unit 1 diesel generators. The 2A diesel generator starting air piping was found to be similarly misrepresented on P & ID M-54 sheet 4B.

2. Caution cards explaining the discrepancy between the actual starting air piping, and the P & ID were hung on the DG local control panel switches, the main control board switches, and the air receivers for the 2A and 2B diesel generators.

3. Labels were placed on all Unit 1 and Unit 2 diesel generator air receivers with the correct Equipment Part Numbers (EPNs), and a statement explaining which side of the engine each receiver feeds.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

E. CORRECTIVE ACTIONS: (Continued)

4. A design change request has been submitted to change the P & ID to match the actual routing of the starting air piping. Action Item Record (AIR) #454-225-88-0078 will track completion of the changes.

5. Control switch labels for all auxiliary equipment on the DG local control panels will be changed to indicate the Byron EPNs. Currently, the labels have numbers assigned by the equipment supplier, while the P & IDs for the auxiliary equipment show Byron EPNs. This corrective action will prevent any equipment confusion when future out of services are implemented on DG auxiliary equipment. AIR #454-225-88-0080 will track completion of this item.

6. A temporary procedure change to the EO rounds procedure has been implemented to direct the EOs to check the status of the "Unit Available for Emergency" light on a shiftly basis. A permanent procedure change, which includes this addition, has been submitted. AIR #454-225-88-0079 will ensure the permanent procedure change is completed.

7. A training revision request will be submitted to include an explanation of the "Unit Available for Emergency" light in the EO and the license training programs to ensure operator understanding of the meaning of the status light. A required reading package will be issued to all licensed and Equipment Operator personnel to ensure an understanding of the indication on a short term basis. AIR #454-225-88-0077 will track the completion of this item.

F. PREVIOUS OCCURRENCES:

LER NUMBER	TITLE
NONE	

G. COMPONENT FAILURE DATA:

a) MANUFACTURER | NOMENCLATURE | MODEL NUMBER | MFG PART NUMBER

Not Applicable

b) RESULTS OF NPRDS SEARCH:

Not Applicable

c) RESULTS OF NWR SEARCH:

Not Applicable

EG09 D/G DIFFERENTIAL OVERCURRENT TRIP

TYPE: GENERIC, NRB

A) 1A D/G
B) 1B D/G

CAUSE: GROUND BETWEEN DG AND OUTPUT BREAKER ACB 1413

REF: AC ELECTRICAL POWER SYSTEM DESCRIPTION
20E-1-4030 DG01, DG02

PLT STA: DIESEL GENERATOR IN OPERATION

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED DIESEL GENERATOR WILL DEVELOP A GROUND BETWEEN THE GENERATOR AND THE OUTPUT BREAKER. ANNUNCIATOR 21-C9 (22-C9) "DG 1A (1B) GROUND" WILL ACTUATE. THE GROUND WILL RESULT IN A DIFFERENTIAL CURRENT CONDITION. THIS WILL TRIP THE DIESEL GENERATOR AND ITS ASSOCIATED OUTPUT BREAKER (ACB 1413 OR 1423). ANNUNCIATORS 21-A9 (22-A9) "BUS 141 (142) DG 1A (1B) FD BRKR 1413 (1423) TRIP, 21-D8 (22-D8) "DG 1A (1B) DIFF LOCKOUT/OVERSPEED" WILL ACTUATE. OTHER ANNUNCIATORS WILL RESPOND ACCURATELY TO THE TRANSIENT.

IF THE DIESEL GENERATOR IS POWERING THE ASSOCIATED BUS THEN ALL BUS LOADS WILL DE-ENERGIZE.

THE SIMULATOR MUST BE RESET TO RECOVER FROM THIS MALFUNCTION.

EVENTS: NONE

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

FW01	MAIN FW PUMP FAILS TO START/TRIP (MOTOR)
FW02	MAIN FW PUMP FAILS TO START/TRIP (TURBINE)
FW03	START-UP FEED PUMP FAILS TO START/TRIP
FW04	MAIN FW OIL PUMP FAILS TO START/TRIP
FW05	TURBINE DRIVEN MFP CONTROL VALVE FAILURE
FW06	TURBINE DRIVEN FW PUMP SPEED CONTROL FAILURE
FW07	FW PUMP SPEED CONTROL OSCILLATES
FW08	LOSS OF FW PUMP SPEED CONTROL
FW09	S/G FW CONTROL VALVE FAILURE
FW10	FW REGULATION BYPASS VALVE FAILURE
FW11	FW TEMPERING LINE ISOLATION VALVE FAILURE
FW12	FW PREHEATER BYPASS VALVE FAILURE
FW13	FW ISOLATION VALVE FAILURE
FW14	FEED LINE BREAK BETWEEN FW009 & CONTAINMENT
FW15	MAIN FW PUMP SHAFT BREAK
FW16	FW HEADER PRESS TRANSMITTER FAILURE
FW17	HEATER DRAIN TANK LEVEL CONTROLLER FAILURE
FW18	FW HEATER TUBE LEAK (17)
FW19	FW LINE BREAK INSIDE CONTAINMENT
FW20	FW LINE BREAK OUTSIDE CONTAINMENT
FW21	S/G TEMPERING LINE RUPTURE
FW22	CONDENSATE PUMP FAILS TO START/TRIP
FW23	FW HEATER BYPASS VALVE FAILURE (1CB025)
FW24	FAILURE OF AF SUCTION PRESSURE TRANSMITTER
FW25	GLAND STEAM CONDENSER MALFUNCTION
FW26	MAIN FW REG VALVE SEAT LEAKAGE
FW27	FW HEATER TUBE LEAK (11 DC)
FW28	FW HEATER TUBE LEAK (11)
FW29	FW HEATER TUBE LEAK (12)
FW30	FW HEATER TUBE LEAK (13)
FW31	FW HEATER TUBE LEAK (14)
FW32	FW HEATER TUBE LEAK (15 DC)
FW33	FW HEATER TUBE LEAK (15)
FW34	FW HEATER TUBE LEAK (16)
FW35	HEATER DRAIN PUMP FAILS TO START/TRIP
FW36	LOSS OF CONDENSER VACUUM
FW37	HOTWELL LEVEL CONTROLLER FAILURE (CD037)
FW38	HOTWELL LEVEL CONTROLLER FAILURE (CD038)
FW39	HOTWELL LEVEL CONTROLLER FAILURE (CD039)
FW40	HOTWELL LEVEL CONTROLLER FAILURE (CD040)
FW41	FW ISOL AUX RELAY FAILURE (TRAIN A)
FW42	FW ISOL AUX RELAY FAILURE (TRAIN B)
FW43	AUX FW PUMP FAILS TO START/TRIP (MOTOR)
FW44	AUX FW PUMP FAILS TO START/TRIP (DIESEL)
FW45	AUX FW VALVE FAILURE
FW46	AUX FW LINE RUPTURE
FW47	FW PUMP SUCTION HEADER BREAK

FW01 MAIN FW PUMP FAILS TO START/TRIP (MOTOR)

TYPE: DISCRETE, RB

CAUSE: FAULTY OVERCURRENT (550/551) RELAY ACTUATION

REF: 20E-1-4030 FW01
20E-1-4030 FW36

PLT STA: MOTOR DRIVEN FEEDWATER PUMP IN OPERATION

EFFECTS: MAIN FEEDWATER PUMP 1FW01PA BREAKER WILL OPEN. PUMP CURRENT INDICATION DECREASES TO ZERO, ANNUNCIATOR 16-A1 "FW PUMP 1A TRIP" ACTUATES, AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. MOTOR DRIVEN FEEDWATER PUMP 1A LUBE OIL PUMP 1FW01PA-B AUTO STARTS (IF C/S IS IN AFTER-START) AS LUBE OIL PRESSURE DECREASES.

FEEDWATER PUMP 1A DISCHARGE FLOW INDICATION DECREASES TO ZERO AND FEEDWATER PUMP DISCHARGE HEADER PRESSURE DECREASES. DEPENDING ON THE INITIAL PLANT CONDITION, NORMAL FEEDWATER FLOW TO THE STEAM GENERATORS MAY BE PARTIALLY OR TOTALLY LOST.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND TRIP LIGHT BY PLACING THE CONTROL SWITCH TO AFTER-TRIP. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL CLOSE THEN IMMEDIATELY TRIP OPEN.

MALFUNCTION REMOVAL WILL RESTORE THE FEEDWATER PUMP OVERCURRENT RELAY TO NORMAL.

EVENTS: NONE

FW02 MAIN FW PUMP FAILS TO START/TRIP (TURBINE)

TYPE: GENERIC, RB

A) 1B MFP 1FW01PB
B) 1C MFP 1FW01PC

CAUSE: FAULTY DEVICE 1PS-FW191/192 (LOW LUBE OIL PRESS. SW.)

REF: 20E-1-4030 FW25
20E-1-4030 FW26
20E-1-4030 FW33
20E-1-4030 FW34
20E-1-4030 FW54
20E-1-4030 FW55

PLT STA: SELECTED MAIN FEEDWATER PUMP IN OPERATION

EFFECTS: THE SELECTED FEEDWATER PUMP WILL TRIP AS INDICATED ON ITS CONTROL PANEL. ANNUNCIATOR 16-B1(C1) "FW PUMP 1B(1C) TRIP" ACTUATES. FEEDWATER PUMP TURBINE HP AND LP STOP VALVES CLOSE AND TURBINE SPEED DECREASES.

FEEDWATER PUMP 1B(C) DISCHARGE FLOW INDICATION DECREASES TO ZERO AND FEEDWATER PUMP DISCHARGE HEADER PRESSURE DECREASES. DEPENDING ON THE INITIAL PLANT CONDITION, NORMAL FEEDWATER FLOW TO THE STEAM GENERATORS MAY BE PARTIALLY OR TOTALLY LOST. THE OPERATOR WILL BE UNABLE TO RESET THE TURBINE WITH THIS MALFUNCTION ACTIVE.

MALFUNCTION REMOVAL WILL RESTORE THE MAIN FEEDWATER PUMP DEVICE 1PS-FW191/192 TO NORMAL.

EVENTS: 1) LER 06-02-88-001
2) LER 06-02-88-004
3) LER 06-01-88-004

FW02

LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
Byron, Unit 1	0 5 0 0 0 4 5 4	1 of 0 3

Title (4) Tachometer Failure Caused Overspeed Trip of Main Feed Pump Resulting in Reactor Trip

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
0 7	1 6	8 8	8 8	0 0 4	0 0	0 8	1 0	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 8

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	X 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER AREA CODE	
T. Tulon, Asst Superintendent Operating Extension 2213	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	S J	T A C	A 1 2 3	Y					

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
__ Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

Unit 1 was at 98 percent reactor power at 0431 on July 16, 1988, when the 1B Main Feedwater Pump (MFP) tripped. Steam Generator (S/G) levels decreased due to the feedwater flow-steam flow mismatch. In spite of licensed operator actions to reduce steam flow and increase feed flow, 1D S/G level decreased to the low-low reactor trip setpoint at 0434. An automatic reactor trip occurred and both Auxiliary Feedwater Pumps automatically started. The licensed operators complied with emergency operating procedures and brought the plant to a stable condition in Hot Standby at 0530. This report is submitted in accordance with 10CFR50.73 (a)(2)(iv) due to the automatic safety system actuations.

The 1B MFP's precision tachometer failed. The tachometer transmitted a constant increase speed output signal to the turbine's automatic speed control circuitry. Turbine speed increased until it reached the overspeed turbine trip setpoint and tripped.

The tachometer was repaired and 1B MFP operation was monitored during the subsequent Unit startup. The pump was returned to service without incident.

A similar previous occurrence was reported in Unit 2 Licensee Event Report 87-009.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 7/16/88 / 0434

Unit 1 MODE 1 - Power Operation Rx Power 98% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of this event that contributed to the event. Unit 1 was at 98 percent reactor power at 0431 on July 16, 1988, when the 1B Main Feedwater Pump (MFP) [SJ] turbine thrust bearing wear and the 1B MFP high discharge flow annunciators actuated in the main control room. The 1B MFP tripped and steam generator (S/G) levels decreased due to the feedwater flow-steam flow mismatch. The Nuclear Station Operator (NSO) (licensed reactor operator) initiated a Turbine Generator [TB] runback to 599 Megawatts-electric (MWe) at a rate of 175 MWe per minute and maximized feedwater flow rate by increasing 1C MFP speed and starting an additional Condensate/Condensate Booster Pump [SD]. In spite of these actions, S/G levels continued to decrease slowly and at 0434 1D S/G level dropped to the low-low level reactor trip setpoint (40.8%). An automatic reactor trip occurred and the 1A and 1B Auxiliary Feedwater Pumps (AFP) [BA] automatically started. A normal post reactor trip Feedwater Isolation occurred when average reactor coolant temperature (T_{avg}) decreased below 564°F with the reactor trip breakers open. The licensed operators entered and complied with "Reactor Trip or Safety Injection - Unit 1 Emergency Operating Procedure" (1BEP-0) and "Reactor Trip Response - Unit 1 Emergency Operating Procedure" (1BEP ES-0.1). At 0436 the NSO manually isolated chemical and Volume Control System [CB] letdown flow due to T_{avg} decreasing below the no load value and the corresponding decrease in pressurizer level. Auxiliary feedwater flow rate was reduced and the T_{avg} reduction was stopped at approximately 550°F. By 0450 T_{avg} returned to its no load value and letdown flow was established.

At 0451 the Feedwater Isolation signal was reset and the Startup Feedwater Pump was started and aligned to supply feedwater flow to the S/G's. At 0523 the 1B AFP was stopped and at 0527 the 1A AFP was stopped, since the pumps were no longer needed to maintain S/G levels. Stable plant conditions were achieved in Hot Standby at 0530.

This Licensee Event Report (LER) is submitted in accordance with 10CFR50.73 (a)(2)(iv) due to the automatic Reactor Protection System and Engineered Safety Features Systems actuations.

C. CAUSE OF EVENT:

The cause of the event was the loss of one Turbine Driven Feedwater Pump. The 1B Feedwater Turbine tripped due to an overspeed condition. The Feedwater Turbine's Tach-PAK series 600 Precision Tachometer was found to be defective. The tachometer transmitted a constant increase speed signal to the turbine's speed control circuitry. Turbine speed increased until it reached the overspeed turbine trip setpoint at which time the turbine tripped. The tachometer failure was caused by the electrical shorting of a diode.

D. SAFETY ANALYSIS:

Neither plant nor public safety were affected by the event. All safety systems actuated as designed. The AFP's actuated and provided feedwater flow to the Steam Generators as designed. The plant was stabilized in Hot Standby for investigation of the MFP trip.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION							
FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					Page (3)
		Year		Sequential Number		Revision Number	
Byron, Unit 1	0 5 0 0 0 4 5 4	8 8	-	0 0 4	-	0 0	0 3 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

E. CORRECTIVE ACTIONS:

The tachometer was repaired by replacing two failed diodes and a resistor and monitored for proper operation. The 1A Motor Driven Main Feedwater Pump was operated to conduct a Unit startup while allowing the 1B MFP to be monitored. The monitoring indicated proper operation of the 1B MFP and it was returned to service without incident.

No further corrective action is planned at this time.

F. PREVIOUS OCCURRENCES:

LER Number	LER Title
87-009 (Unit 2)	Manual Reactor Trip in Response to Decreasing Steam Generator Levels Resulting from a Feedwater Pump Trip Due to a Defective Speed Control Feedback Loop

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
AirPax Electronic Controls Division	Tack Pac Precision Tachometer	Series 600	990-000-815

FW02

LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
Byron, Unit 2	0 5 0 0 0 4 5 5	1 of 0 3

Title (4) REACTOR TRIP ON 2C STEAM GENERATOR LOW LEVEL DUE TO A FEEDWATER PUMP TRIP AND FAILURE OF DIGITAL ELECTROHYDRAULIC CONTROL SYSTEM TO RUNBACK TURBINE

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0 2	1 2	8 8	8 8	--- 0 0 1	--- 0 0				NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 4

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	Other (Specify in Abstract below and in Text)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	Telephone Number
T. Joyce, Assistant Superintendent Operating Ext. 2213	Area Code 8 1 5 — 2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	J J	V	M 4 2 3	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
X Yes (If yes, complete EXPECTED SUBMISSION DATE)	NO		1 0	3 1	8 8

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

On February 12, 1988, at 1804 hours with Byron Unit 2 in power operation (Mode 1) at 94% power, the 2C feedwater pump tripped on overspeed. Efforts to shed load were unsuccessful due to failure of the digital electrohydraulic control system to respond properly in the automatic mode. This resulted in loss of inventory in the steam generators and a reactor trip on low steam generator level. All safeguard actuation features functioned as designed. The feedwater pump trip was due to a failed servovalve which allowed the feedwater pump turbine high pressure governor valve to fail open. This caused the pump overspeed and trip. The defective servovalve was replaced. There have been previous reactor trips due to feedwater pump trips.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 2-12-88 / 1804

Unit 2 MODE 1 - Power Operations Rx Power 94% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On February 12, 1988, at 1804 hours. Byron Unit 2 was in power operation (Mode 1) at 94 percent power. The Digital Electrohydraulic (DEH)[TG] control system was maintaining the turbine generator at 1070 MWe in Auto with Impulse and Speed feedback loops "IN" and Megawatt Feedback loops "OUT". At this time the 2C Turbine Driven Feedwater pump (FW)[SJ] tripped due to overspeed. The Nuclear Station Operator (NSO, Licensed) correctly initiated a runback of the Main Turbine Generator (TG)[TB]. The ramp was programmed for 2000 MWe/min to 559 MWe per Byron Operating Procedures. The DEH Computer did not execute the runback properly and load only dropped 60 MWe and held at 1014 MWe. The operator depressed the "HOLD" button and the ramp was re-initiated in the manual control mode. The Turbine Generator runback was not sufficient and a "low low" level in the 2C Steam Generator caused a reactor trip. The Unit was maintained in Mode 3, Hot Standby, until initial investigations of the turbine runback failure and feedwater pump trip were conducted. The NRC was notified at 1853 hours on 2/12/88.

Unit 2 was brought back on line using the 2B Turbine Driven Feedwater pump in place of the 2C Feedwater pump. The high pressure governor on the 2C Feedwater pump was manually isolated upstream to allow monitoring of the 2C Feedwater pump governor valve without affecting pump operation. The governor valve's servo-actuator valve was replaced and was being monitored at seven different points by a strip chart recorder.

All safety systems responded as required. No other systems or components were inoperable prior to this event which contributed to this event. All operator actions were correct. This event is reportable pursuant 10CFR50.73 (a)(2)(iv).

On 2/22/88, at 0130 hours, Unit 2 was in Mode 1 at 86 percent power when the High Pressure Governor valve on the 2C Turbine Driven Feedwater Pump opened. The valve was being monitored following the replacement of the servo-actuator valve, and the steam supply was manually isolated. The Unit experienced no adverse affects from this occurrence.

C. CAUSE OF EVENT:

The cause of the 2C Feedwater pump trip on February 12, 1988, was found to be a failure of the servovalve on the High Pressure Governor Valve. The High Pressure Governor Valve on the Feedwater Pumps are only used at startup and shutdown. When the servovalve failed the High Pressure Governor Valves failed open causing the Feedwater Turbine to overspeed and trip. The servovalve has been sent to the manufacturer for a failure analysis to determine the root cause of the failure. A supplemental report will be issued when the results of this analysis are known.

C. CAUSE OF EVENT: (Continued)

The Feedwater pump trip required the Unit 2 NSO (licensed) to initiate a turbine runback which did not occur as planned. Previously, changes were made per Westinghouse instructions and in accordance with the Station's software control program to the DEH computer, in order to fine tune the main turbine governor valve operations. These changes were made to minimize load swings during governor valve testing. One of the changes made to the DEH computer involved reducing the deadband for the impulse pressure Feedback loop by adjusting some of the computer gains. Basically, this loop looks at a calculated impulse pressure. If the difference between the calculated impulse pressure and the actual impulse pressure is too great the DEH computer rejects the impulse pressure loop and stops the unit at that power level. The turbine runback, initiated after the 2C Feedwater Pump trip, dropped electrical output approximately 60 MWe when the gain value between calculated and actual impulse pressure was exceeded, halting the runback. The runback was completed manually.

On 2/22/88 the failure of the replacement servovalve was determined to be the cause of the High Pressure Governor Valve opening. In this event there were no adverse affects to the Unit due to the fact that steam to the High Pressure Governor Valve was isolated. The servovalve was found to have a defective coil with high internal resistance. In the first event the cause of the servovalve failure was not apparent.

D. SAFETY ANALYSIS:

All plant safety systems actuated and performed as designed. The reactor tripped on Low-2 level on the 2C Steam Generator. The Manual Turbine Generator runback was still available to runback the Turbine.

E. CORRECTIVE ACTIONS:

The servovalve was replaced on the 2C Feedwater Pump High Pressure Governor Valve after the 2/12/88 occurrence. The second High Pressure Governor Valve failure was being monitored and examination of the strip chart recordings showed that the servovalve was again malfunctioning but with a different mode of failure. The servovalve was again replaced after the 2C Feedwater Pump was taken off line, and the 2A Motor Driven pump was put into service. The 2C pump was monitored following component replacement, and will continue to be monitored when it is returned to operation to ensure proper operation. The DEH problem was resubmitted to Westinghouse Corporation for reevaluation. In the interim the gains in the DEH computer will be returned to their previous values. Subsequent valve tests have been conducted with satisfactory results.

F. PREVIOUS OCCURRENCES:
Previous reactor trips due to Feedwater pump trips were reported in the following LER's.

LER NUMBER	TITLE
454/85-061-01	
454/87-018-00	
455/87-009-00	

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Moog	Servovalve	A076-185 Moog Model 760	1161

b) RESULTS OF NPRDS SEARCH:
No pertinent information found during NPRDS search.

LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
Byron, Unit 2	0 5 0 0 0 4 5 5	1 of 0 4

Title (4) Main Feedwater Pump Trip Due to Improper Isolation of Electrohydraulic Control Fluid Supply Resulting in Reactor Trip

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
0 5	0 6	8 8	8 8	0 0 4	0 0	0 6	0 3	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 4

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	Other (Specify in Abstract below and in Text)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER AREA CODE	
M. Snow, Regulatory Assurance Supervisor Extension 2280	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

On April 27, 1988, an out-of-service condition on the 2C Main Feedwater Pump (MFP) was temporarily lifted to permit operation of the MFP, and the pump was started and placed in service. On May 6 a licensed reactor operator (NSO) noted that the temporary lift was due to expire on that day and requested a disposition from a licensed senior reactor operator (SCRE). The SCRE directed the NSO to terminate the temporary lift by returning the equipment to its original out-of-service condition. Both the SCRE and the NSO incorrectly believed that returning the valve listed on the temporary lift paperwork to its out-of-service closed position would not affect the operation of the 2C MFP. At 1214 on May 6 with Unit 2 at 94 percent power an Equipment Operator closed the valve, which isolated electrohydraulic (EH) fluid supply to the 2C MFP. At 1215 the 2C MFP tripped due to low EH fluid pressure. Steam generator levels lowered rapidly and the NSO manually tripped the reactor in anticipation of an automatic trip. Operator actions taken following the reactor trip were correct, and stable plant conditions were achieved in Hot Standby at 1330.

Several causes contributed to the improper closure of the EH valve. The NSO and the SCRE committed cognitive personnel errors by failing to recognize the consequences of the return to out-of-service condition. Both individuals made incorrect assumptions regarding system design without reference to system drawings. The administrative procedure for control of temporary lifts contributed to the personnel errors.

The Operating Department personnel involved in the event have been interviewed and specific performance weaknesses have been discussed. Administrative procedures will be revised appropriately to minimize recurrence.

There have been no previous similar occurrences of this event.

(0011R/0002R)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year		Sequential Number		Revision Number	Page (3)
Byron, Unit 2	0 5 0 0 0 4 5 5	8 8	-	0 0 4	-	0 0	0 2 OF 0 4

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 5/6/88 / 1216

Unit 2 MODE 1 - Power Operation Rx Power 94% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On April 9, 1988, at 1152 the 2C turbine driven Main Feedwater Pump (MFP) [SJ] was removed from service in order to replace the servo valve on the Low Pressure Governor Valve (2EH-5049B). The position of 2EH-5049B had been oscillating abnormally, and its servo valve was believed to be the cause. Establishment of the maintenance out-of-service boundary was accomplished by closing the Electrohydraulic Control Fluid Supply Valve (2EH-5058B) [JJ]. The servo valve was replaced but could not be tested to verify satisfactory operation because the internal components of 2EH-5049B had been damaged during the valve's oscillations. Therefore, the out-of-service condition could not be administratively cleared by completing the required valve testing. Complete repair of 2EH-5049B will require that the main condenser be at atmospheric pressure.

Due to feedwater piping vibration and flow control problems associated with the operation of the 2A motor driven MFP, the starting of the 2C turbine driven MFP was pursued. At 1730 on April 27, a Temporary Lift on the out-of-service was authorized by the Unit 2 Shift Foreman (licensed Senior Reactor Operator) to permit operation of the 2C MFP using steam supplied from the main steam header [SB] via the High Pressure Governor Valve. Valve 2EH-5058B was opened and the 2C MFP was started and aligned to supply feedwater to the steam generators. The 2A motor driven MFP, which had been supplying feedwater, was stopped. The 2B turbine driven MFP continued to operate with steam supplied to its turbine from a Moisture Separator Reheater via its Low Pressure Governor Valve, which is the normal steam supply for the MFP turbines at high power levels.

On May 6, 1988, a Unit 2 licensed reactor operator Nuclear Station Operator (NSO) noted that the Temporary Lift on the out-of-service for valve 2EH-5049B was due to expire on that day. The NSO delivered the Temporary Lift paperwork to a licensed senior reactor operator Shift Control Room Engineer (SCRE) to obtain a decision as to whether the Temporary Lift should be extended or terminated by restoring the out-of-service. The SCRE returned the Temporary Lift paperwork to the NSO and directed him to terminate the Temporary Lift by returning the equipment to an out-of-service condition. Both the SCRE and the NSO believed that the out-of-service would only affect 2EH-5049B by closing 2EH-5058B, however, in actuality the closing of 2EH-5058B also isolates fluid to the High Pressure Governor Valve which was supplying steam to the 2C MFP turbine at the time. The NSO directed an equipment operator (EO) to restore the out-of-service.

At 1214 on May 6, 1988, with Unit 2 at 94 percent power, the EO closed 2EH-5058B as instructed. Shortly thereafter the "Feedwater Pump Turbine Oil Pressure Low" annunciator actuated in the Control Room. This annunciator alarms due to either low lubricating oil pressure or low electrohydraulic (EH) fluid pressure. The NSO contacted the EO by radio immediately to notify him of the alarming condition. Simultaneously the EO heard the 2C MFP turbine speed decreasing and tried to open 2EH-5058B.

B. DESCRIPTION OF EVENT: (Continued)

At 1215 the "Feedwater Pump 2C Trip" annunciator actuated. The NSO initiated a preset main turbine [TB] runback to 559 Megawatts-electric (MWe) at 175 MWe/minute. At 1216 narrow range levels in all steam generators (S/G's) had decreased to the low alarm setpoint. The SCRE selected manual governor valve fast action to more rapidly runback the main turbine because the preset runback did not seem to be operating as expected. Narrow range levels in all S/G's continued to decrease. The SCRE directed the NSO to manually trip the reactor when narrow range S/G levels dropped to 20 percent. At 1216 the NSO manually tripped the reactor and an automatic turbine trip followed. The Control Room operators entered and complied with "Reactor Trip or Safety Injection Unit 2 Emergency Procedure" (2BEP-0). The 2A and 2B Auxiliary Feedwater Pumps (AFP's) [BA] automatically started due to the low low S/G levels resulting from the feedwater-steam flow mismatch and indicated level shrink on the trip. At 1217 a Feedwater Isolation occurred due to the expected decrease in Average Reactor Coolant Temperature (T_{avg}) to its low setpoint coincident with the reactor trip. The 2C S/G Power Operated Relief Valve (PORV) opened fully and remained open until the NSO placed its controller in manual and fully closed it at 1219.

At 1245 the Feedwater Isolation signal was reset, the Startup Feedwater pump was started, and a flow path from the Startup Feedwater pump to the S/G's was established. At 1308 the 2B AFP was stopped and at 1323 the 2A AFP was stopped. Stable plant conditions were achieved at 1330 with Unit 2 in Hot Standby (Mode 3). Operator actions taken following the reactor trip were correct and contributed to the safe conclusion of the event.

This event is reportable in accordance with 10CFR50.73(a)(2)(iv) due to the manual actuation of the Reactor Protection System.

C. CAUSE OF EVENT:

The cause of the reactor trip was the manual actuation of the reactor trip switch on the main control board by the NSO. The NSO manually actuated the trip due to downward trending low narrow range S/G levels in anticipation of an automatic reactor trip. The low narrow range S/G levels were caused by the tripping of the 2C MFP while the plant was at 94 percent power, which resulted in a steam flow-feed flow mismatch. Contributing to the low levels was indicated level shrink caused by the operator initiated main turbine runback. The 2C MFP trip was caused by the EO when he closed 2EH-5058B, which isolated the EH control fluid supply from the High Pressure Governor Valve. This caused the governor valve to close and block all steam flow to the 2C MFP turbine. The cause for the improper opening of the 2C PORV was not determined after extensive troubleshooting by Maintenance department technicians, and the valve was declared operable on May 11, 1988. The main turbine runback was determined to have been responding properly to the event.

Several causes contributed to the improper closure of 2EH-5058B. The licensed NSO and SCRE committed cognitive personnel errors by failing to recognize the consequences of the return to out-of-service condition. The NSO and SCRE directed the operation of plant equipment without fully understanding the impact of that operation. Both individuals believed that the closing of valve 2EH-5058B would only isolate the EH fluid supply from the Low Pressure Governor Valve, and that the High Pressure Governor Valve would remain unaffected and permit uninterrupted operation of the 2C MFP. Their belief was based on an incorrect assumption that the affected portion of the MFP EH system is designed similarly to the Main Turbine EH system where individual EH isolation valves are provided for each governor valve. Neither operator consulted piping system drawings to verify that the return to out-of-service could be performed without seriously impacting plant operation. There were no unusual characteristics of the plant environment that contributed to the personnel errors.

C. CAUSE OF EVENT: (Continued)

Byron Administrative Procedure "Administrative Requirements for Temporarily Lifting OOS Cards and/or Placing Equipment in Test" (BAP 331-1) provides direction and responsibility for the initiation and termination of Temporary Lifts. It does not require licensed senior reactor operator (SRO) involvement for the termination of a Temporary Lift. Although the NSO involved the SCRE during this event, the SCRE still was not required to document his approval of the proposed action. Additionally BAP 331-1 does not require a thorough review of the effects that a return to out-of-service condition may have on operating plant equipment, as would be required for the initial out-of-service per "Station Equipment Out-of-Service Procedure" (BAP 330-1).

D. SAFETY ANALYSIS:

Neither plant safety nor public safety were affected by this event. All Engineered Safety Feature (ESF) systems operated properly to minimize the consequences of the plant trip. Although the 2C S/G PORV opened and remained open until placed in manual and closed, this caused only a minor Reactor Coolant System cooldown to approximately 551°F and briefly delayed the achievement of stable plant conditions. The more severe condition of a MFP trip at 100% reactor power would only have accelerated the pace of events, and plant/public safety would have remained unaffected.

E. CORRECTIVE ACTIONS:

In order to permit continued operation of the 2C MFP using the High Pressure Governor Valve, an Onsite Review was completed to clear the out-of-service that required the closing of 2EH-5058B. This action eliminated the need to operate the 2C MFP with a temporary lift condition in effect.

The 2C S/G PORV was initially isolated by closing its manual isolation valve, and the associated Technical Specification Limiting Condition for Operation Action Requirement was satisfied. When troubleshooting efforts failed to identify any component failures, valve operability was verified and the valve was declared operable at 1212 on May 11, 1988.

The Operating Department personnel involved in the event have been interviewed and specific performance weaknesses have been discussed.

The BAP 331-1 will be revised to include:

1. SRO responsibility for the termination of Temporary Lifts.
2. A caution statement to ensure the conduct of a thorough technical review prior to returning temporarily lifted equipment to an out-of-service condition.
3. A requirement for Onsite Review of Temporary Lifts whose duration exceeds five working days.

The "Operating Shift Turnover and Relief Administrative Procedure" (BAP 335-1) will be revised to require Shift Engineer (licensed SRO) review of Temporary Lift packages each shift. Completion of the procedure revisions is tracked by Action Item Record 454-225-88-0117.

F. PREVIOUS OCCURRENCES:

NONE

G. COMPONENT FAILURE DATA:

Not Applicable

FW03 START-UP FEED PUMP FAILS TO START/TRIP

TYPE: DISCRETE, RB

CAUSE: FAULTY OVERCURRENT (550/551) RELAY ACTUATION

REF: 20E-1-4030 FW57
20E-1-4030 FW58

PLT STA: START-UP FEED PUMP IN OPERATION

EFFECTS: START-UP FEEDWATER PUMP 1FW02P BREAKER WILL OPEN. PUMP CURRENT INDICATION DECREASES TO ZERO, ANNUNCIATOR 16-A6 "START-UP FW PUMP TRIP" ACTUATES, AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. START-UP FEEDWATER PUMP AUX OIL PUMP 1FW02P-B AUTOMATICALLY STARTS.

START-UP FEEDWATER PUMP 1FW02P DISCHARGE FLOW INDICATION DECREASES TO ZERO AND FEEDWATER PUMP DISCHARGE HEADER PRESSURE DECREASES. DEPENDING ON THE INITIAL PLANT CONDITION, NORMAL FEEDWATER FLOW TO THE STEAM GENERATORS MAY BE PARTIALLY OR TOTALLY LOST.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND TRIP LIGHT BY PLACING THE CONTROL SWITCH TO AFTER-TRIP. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL CLOSE THEN IMMEDIATELY TRIP OPEN.

MALFUNCTION REMOVAL WILL RESTORE THE START-UP FEEDWATER PUMP OVERCURRENT RELAY TO NORMAL.

EVENTS: NONE

FW04 MAIN FW OIL PUMP FAILS TO START/TRIP

TYPE: GENERIC, RB

A)	1B MFP: 1A MAIN LUBE OIL PUMP	1FW01PB-A
B)	1B MFP: 1B MAIN LUBE OIL PUMP	1FW01PB-B
C)	1C MFP: 1A MAIN LUBE OIL PUMP	1FW01PC-A
D)	1C MFP: 1B MAIN LUBE OIL PUMP	1FW01PC-B

CAUSE: M RELAY FAILURE

REF: 20E-1-4030 FW22
20E-1-4030 FW30

PLT STA: SELECTED MAIN FEEDWATER OIL PUMP IN OPERATION

EFFECTS: THE SELECTED MAIN FEEDWATER OIL PUMP WILL STOP. ANNUNCIATOR 16-C2 "FW PUMP TURB LUBE OIL PUMP TRIP" ACTUATES, THE ASSOCIATED "OIL PRESS UP" LAMP DEENERGIZES, AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. THE STANDBY LUBE OIL PUMP WILL IMMEDIATELY AUTOMATICALLY START ILLUMINATING ITS "OIL PRESS UP" LAMP.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND TRIP LIGHT BY OPERATING THE CONTROL SWITCH TO STOP. IF THE OPERATOR ATTEMPTS TO RESTART THE SELECTED PUMP, THE PUMP WILL NOT RESTART, THE TRIP LIGHT WILL ILLUMINATE IMMEDIATELY, AND THE ANNUNCIATOR WILL ACTUATE WHEN THE CONTROL SWITCH IS RETURNED TO AFTER-START.

MALFUNCTION REMOVAL WILL RESTORE THE MAIN FEEDWATER OIL PUMP M RELAY TO NORMAL.

EVENTS: NONE

FW05 TURBINE DRIVEN MFP CONTROL VALVE FAILURE

TYPE: GENERIC, RV 0-100% OF VALVE TRAVEL

A) 1B MFP HP VLV 1FW01PB
B) 1B MFP HP VLV 1FW01PC
C) 1C MFP LP VLV 1FW01PB
D) 1C MFP LP VLV 1FW01PC

CAUSE: SERVO VALVE FAILURE

REF: SYSTEM DESCRIPTION

PLT STA: TURBINE DRIVEN MAIN FEEDWATER PUMP IN OPERATION

EFFECTS: WITH THIS MALFUNCTION ACTIVE, THE SELECTED MAIN FEEDWATER PUMP HP AND/OR LP GOVERNOR VALVE(S) WILL FAIL TO THE SELECTED SEVERITY. IF SEVERITY SELECTED IS LESS THAN THE ACTUAL VALVE POSITION THE SPEED OF THE TURBINE WILL DECREASE, DECEASING PUMP SPEED, DISCHARGE PRESSURE, AND FLOW. FAILING THE LP VALVE TO 0% CAUSES THE HP VALVE TO OPEN RETURNING THE PUMP TO THE DESIRED SPEED. IF SEVERITY SELECTED IS GREATER THAN ACTUAL VALVE POSITION DISCHARGE HEADER PRESSURE AND FLOW MAY INCREASE.

AN IMMEDIATE FAILURE IN THE OPEN DIRECTION OR CLEARING THE MALFUNCTION MAY CAUSE A MFP OVERSPEED TRIP.

MALFUNCTION REMOVAL WILL RESTORE THE TURBINE DRIVEN MAIN FEEDWATER PUMP GOVERNOR VALVE SERVO VALVES TO NORMAL.

EVENTS: 1) DVR 20-01-89-079
2) DVR 06-01-85-120

/

FWD5

DEVIATION INVESTIGATION REPORT

TITLE

1B FEEDWATER PUMP HIGH PRESSURE STOP VALVE CLOSURE

PAGE 1 OF 0 1

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	1
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	0 2 0
0 4	2 2	8 5	0 6	0 1	8 5	— 1 2 0	— 0 1	0 7	1 8	8 5		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
Richard M. Williams Ext. 2385	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	S J	R V	X 9 9 9	Y					

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

WHAT HAPPENED?

...le operating the 1B Turbine Driven Feedwater Pump, the High Pressure Stop Valve drifted closed.

WHAT WAS THE ROOT CAUSE?

The EHC Solenoid valve associated with the 1B Feedwater Pump High Pressure Stop Valve was found to be defective and improperly oriented.

HOW DID IT AFFECT PLANT AND/OR PUBLIC SAFETY?

The closure of this stop valve effectively eliminated the pumping capability of this pump. Another Feedwater Pump was running and prevented a loss of feedwater reactor trip. Thus, there was no impact on plant or public safety.

HAS IT HAPPENED BEFORE?

Yes, on two previous occasions, the same stop valve has drifted closed. (LER #85-039-00, DIR #85-098-00)

WHAT WAS DONE TO CORRECT THE CONDITION AND HOW ARE WE GOING TO PREVENT RECURRENCE?

The defective solenoid was not re-oriented. It was determined that the solenoid spring did not supply enough force to maintain valve position. A new solenoid valve with a stiffer spring was installed in the original position. This modification has eliminated the valve drift and the pump has been run tested without incident.

(0638M)

DEVIATION REPORT

DVR NO.

20 - 1 - 89 - 079

STA UNIT YEAR NO.

FW05

Form Rev 2.0

ART 1 | TITLE OF DEVIATION

.C FW Pp Trip

OCCURRED 05-04-89 (DATE) 1635 (TIME)

SYSTEM AFFECTED: FW

PLANT STATUS AT TIME OF EVENT: MODE 1 POWER(%) 88

WORK REQUEST NO.

TESTING: | | YES | x | NO

DESCRIPTION OF EVENT

1C FW Pp Stop Vlv failed closed (High press). LP Gov VCV MAN Isolation was isolated earlier due to LP Gov VLV problems. Therefore 1C FW Pp lost all Steam Flow and essentially tripped. Attempted 2000 MW/Min Ramp to 580 MW's as per BwOA SEC-1. Although DEHC did not operate properly, power was quickly reduced to 580MW's. No auto Rx trip setpoints were reached and no MANUAL Rx trip was initiated.

POTENTIALLY SIGNIFICANT EVENT PER NSD DIRECTIVE A-07 | | YES | X | NO

10CFR50.72 NRC RED PHONE NOTIFICATION MADE | | 1 HOUR | | 4 HOUR ____ TIME | X | NO

K. Eckert
RESPONSIBLE SUPERVISOR

5-24-89
DATE

PART 2 | OPERATING ENGINEER'S COMMENTS

T.S. and Procedure Group evaluating BwOR Sec. 1.

| X | NON REPORTABLE EVENT

| | 30 DAY REPORTABLE/10CFR____

| | 5 DAY REPORT PER 10CFR21

| | ANNUAL/SPECIAL REPORT REQUIRED

A.I.R. #____

L.E.R. #____

NOTIFICATION N/A
REGION III ____ DATE ____ TIME

OFC of VP PWR OPS 5-25-89 1300
NSD DATE TIME

| | CECO CORPORATE NOTIFICATION MADE IF ABOVE NOTIFICATION IS PER 10CFR21

TELECOPY N/A
CECO CORPORATE OFFICER ____ DATE ____ TIME

PRELIMINARY REPORT COMPLETED AND REVIEWED R. Schorie 5/25/89
OPERATING ENGINEER DATE

INVESTIGATION REPORT & RESOLUTION ACCEPTED BY STATION REVIEW DRPiano 6/15/89 Carl Smith 6-14-89

RESOLUTION APPROVED AND AUTHORIZED FOR DISTRIBUTION K C Kofer 6/15/89
STATION MANAGER DATE

86-5176 (Form 15-52-1) 11-20-85

2747m(061289)/1

FW06 TURBINE DRIVEN FW PUMP SPEED CONTROL FAILURE

TYPE: GENERIC, RB

A) 1B MFP
B) 1C MFP

CAUSE: SPEED FEEDBACK FAILS TO MIN SPEED OVER 2 MINUTES

REF: SYSTEM DESCRIPTION

PLT STA: SELECTED FEEDWATER PUMP IN AUTOMATIC OR MANUAL OPERATION

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED MAIN FEEDWATER PUMP LP/HP GOVERNOR VALVES WILL BEGIN TO OPEN. ACTUAL MAIN FEEDWATER PUMP SPEED WILL INCREASE. INDICATED MAIN FW PUMP SPEED WILL DECREASE. THE SELECTED MAIN FEEDWATER PUMP DISCHARGE FLOW WILL INCREASE AS SPEED INCREASES. WHEN PUMP SPEED REACHES 5720 RPM, THE FEEDWATER PUMP WILL TRIP ACTUATING ANNUNCIATOR 16-B1(C1) "FW PUMP 1B(1C) TRIP".

MALFUNCTION REMOVAL WILL RESTORE THE FEEDWATER PUMP TACHOMETER OUTPUT TO NORMAL.

EVENTS: NONE

FW07 FW PUMP SPEED CONTROL OSCILLATES

TYPE: GENERIC, RB

A) 1B MFP
B) 1C MFP

CAUSE: TACHOMETER ERRATIC OUTPUT

REF: SYSTEM DESCRIPTION

PLT STA: SELECTED FEEDWATER PUMP IN AUTOMATIC OPERATION

EFFECTS: THE SELECTED FEEDWATER PUMP TURBINE SPEED WILL BEGIN TO OSCILLATE ERRATICALLY WITH A MAXIMUM AMPLITUDE OF 100 RPM AS INDICATED BY TURBINE SPEED INDICATION. FEEDWATER PUMP DISCHARGE PRESSURE, FLOW, AND HP/LP VALVE POSITION(S) WILL RESPOND APPROPRIATELY TO THE VARIATION IN TURBINE SPEED.

NO CONTROL ROOM ACTION CAN STOP THE OSCILLATIONS.

MALFUNCTION REMOVAL WILL RESTORE THE FEEDWATER PUMP TACHOMETER OUTPUT TO NORMAL.

EVENTS: 1) DVR 06-02-88-044

DEVIATION INVESTIGATION REPORT

FW07

TITLE	PAGE
U-2 LOAD REDUCED DUE TO 2B AND 2C FEEDWATER PUMP LOW PRESSURE GOVERNOR VALVE OSCILLATIONS	1 OF 0 3

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	1
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	9 3
0 4	0 8	8 8	0 6	0 2	8 8	0 4 4	0 0	0 5	2 3	8 8		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
Don Brindle, Operating Engineer Ext. 2218	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	J J	V	M 4 2 3	Y					

SUPPLEMENTAL REPORT EXPECTED		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
X YES (if yes, complete EXPECTED SUBMISSION DATE)	NO		0 4	0 1	8 9

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Event 1 Date/Time 4/8/88 / 1645

Event 2 Date/Time 4/9/88 / 0630

Unit 2 MODE (Prior to Event 1) 1 - Power Operations Rx Power 94% RCS [AB]
Temperature/Pressure Normal Operating

Unit 2 MODE (Prior to Event 2) 1 - Power Operations Rx Power 80% RCS [AB]
Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

Event 1 - At 1645 on 4/8/88 the 2C Feedwater Pump (FW) [SJ] began experiencing low pressure governor valve oscillations. During this event these oscillations incresed in intensity, reaching a peak at a frequency of approximately 2 oscillations per second. Each oscillation cycled the valve about 3 inches open and closed causing speed and flow control problems.

At 1703 a load reduction was commenced at 2 MW/Min in the event that a Feedwater Pump trip became necessary. At 1806 with conditions worsening, the load reduction was increased to 4 MW/Min.

At 1810 conditions had deteriorated to the point where the 2C FW Pump was unable to maintain speed or load. The load reduction was again increased, this time to 10 MW/Min. The manual steam isolation was opened for the high pressure governor valve and at 1817 steam was admitted to the 2C FW pump high pressure turbine. The 2C FW pump became stable and began to pick up load. The load reduction was reduced to 3 MW/Min. The valve oscillations had damaged a control linkage on the low pressure governor valve, so, as a precaution, the steam supply to the low pressure governor valve was manually isolated.

B. DESCRIPTION OF EVENT: (Continued)

At 1831 with plant conditions now stabilized, the load reduction was stopped. Preparations were also made for starting the 2A Motor Driven FW Pump. At 1840 the 2A FW pump was started and the speed was reduced on the 2C FW pump. At 1842 with the 2A FW pump picking up the load, the 2C FW pump was tripped and placed on its turning gear.

Event 2 - Approximately 15 hours after the 2C FW pump failed, the 2B FW experienced oscillations on its low pressure governor valve. At the time of Event 2, the unit was being ramped down at 2 MW/Min to reach a target value of 585 MWE for Main Turbine repairs and to perform a required valve test.

At 0645 on 4/9/88 the 2B FW pump low pressure governor valve began oscillating violently. At 0648 the load reduction was increased to 10 MW/Min. Attempts were made to stabilize the pump using the high pressure governor valve with little success.

At 0652 the load reduction was increased to 15 MW/Min to bring the unit down to approximately 60% power. At 0657 the 2B FW pump was tripped and feed flow conditions stabilized. Steam supplies were manually isolated to both governor valves and the 2B FW pump was placed on turning gear.

C. CAUSE OF EVENT:

The cause of events 1 and 2 appear to be failed servo valves on the low pressure governor valves. The 2C FW pump low pressure governor valve was damaged by the intense valve oscillations. A control linkage was broken and the low pressure governor valve was jammed partially open. The extent of the damage to the governor valve itself has yet to be determined.

The 2B FW pump had only minor damage consisting of two stripped setscrews on the governor valve linkage. The servo valve was replaced the next day as were the stripped setscrews. Operational Analysis Department (OAD) personnel examined the electrical signals from the DEH controller to the servo valve and found them to be satisfactory. Westinghouse personnel were present for linkage rod adjustments as well as pump startup. At 1424 on 4/10/88 the 2B FW pump was placed in service. The pump start was normal and the Unit was ramped up during the remainder of the day, without incident.

D. SAFETY ANALYSIS:

There were no safety consequences as a result of this event. No safety systems were initiated and the controlled runbacks precluded reactor trips. All systems performed as designed during these events.

E. CORRECTIVE ACTIONS:

The failed servo valves have been sent out for analysis by the manufacturer. At the time of this report results of this analysis have not yet been finalized. Action Item Record (AIR) 88-069 is tracking this analysis.

A supplemental report to this DVR will be written to document cause, if determined, and component failures after analysis.

TEXT

F. PREVIOUS OCCURRENCES:

Previous Servo Valve failures have been documented in the following DVR's.

DVR NUMBER	TITLE
6-1-87-106 (LER 87-019)	Safety Injection and Reactor Trip from Low Steam Line Pressure due to Failed Main Turbine Throttle Valve During the Throttle Valve to Governor Valve Transfer.
6-2-88-026 (LER 88-001)	Reactor Trip on 2C Steam Generator Low Level Due to a Feedwater Pump Trip and Failure of Digital Electrohydraulic Control System to Runback Turbine
6-2-88-030	Unit 2 Derating Due to Electro Hydraulic (EH) System Pressure Problems

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Moog	Servovalve	Moog Model 76	1161

b) RESULTS OF NPRDS SEARCH:

No Moog valve failures were found as they are not usually reportable to NPRDS.

c) RESULTS OF NWR SEARCH:

Several Moog Model #76 Servovalve failures have occurred other than those mentioned in this report. These valves have been replaced under the following Work Requests.

7/15/87	NWR #B47189	3/27/88	NWR #54493
8/13/87	NWR #B48114	3/27/88	NWR #54495
9/16/87	NWR #B49029	3/29/88	NWR #54549
2/22/88	NWR #B53242	3/29/88	NWR #54551
2/29/88	NWR #B53566	4/09/88	NWR #54884

FW08 LOSS OF FW PUMP SPEED CONTROL

TYPE: GENERIC, RB

A) 1B MFP (1SK-509B)
B) 1C MFP (1SK-509C)

CAUSE: AUTO CONTROLLER OUTPUT FAILS

REF: SYSTEM DESCRIPTION
BwOP FW-1

PLT STA: SELECTED FEEDWATER PUMP IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE LOSS OF THE FW CONTROL SIGNAL. THE AFFECTED TDFW PUMP TRANSFERS FROM "BOILER CONTROL" TO "SPEED SETTER" MODE AT THE EXISTING PUMP SPEED SINCE THE SPEED SETTER AUTOMATICALLY TRACKS BOILER CONTROL. ANNUNCIATOR 16-E4 "FW PUMP SPEED CONT SIGNAL LOST LOCAL CONT" ACTUATES, AND THE "FEEDWATER CON. SIG. LOST" LAMP ILLUMINATES ON THE ASSOCIATED FW PUMP CONTROL PANEL. THE ASSOCIATED FW PUMP CONTROL PANEL "BOILER CONTROL" LAMP GOES OUT, AND THE "SPEED SETTER" LAMP ILLUMINATES. THE TDFW PUMP CAN ONLY BE CONTROLLED BY THE "INCREASE SPEED" AND "DECREASE SPEED" PUSHBUTTONS ON THE FW PUMP CONTROL PANEL.

MALFUNCTION REMOVAL RESTORES THE AUTO CONTROLLER OUTPUT TO NORMAL

EVENTS: NONE

FW09 S/G FW CONTROL VALVE FAILURE

TYPE: GENERIC, RV 0-100% OF VALVE TRAVEL

A) 1FW510
B) 1FW520
C) 1FW530
D) 1FW540

CAUSE: POSITIONER FAILURE (VALVE PROBLEM, NO MANUAL CONTROL AVAILABLE)

REF: 20E-1-4030 FW04
PLS

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THE SELECTED STEAM GENERATOR MAIN FEEDWATER CONTROL VALVE WILL FAIL TO THE POSITION SELECTED BY MALFUNCTION SEVERITY. IF THE SEVERITY SELECTED IS GREATER THAN THE ORIGINAL VALVE POSITION, FEEDWATER FLOW TO THE ASSOCIATED STEAM GENERATOR WILL INCREASE RESULTING IN STEAM GENERATOR LEVEL INCREASING AT A RATE DETERMINED BY THE FEED FLOW-STEAM FLOW MISMATCH. ANNUNCIATOR 15-A3(B3,C3,D3) "S/G 1A(B,C,D) FLOW MISMATCH STM FLOW LOW" WILL ACTUATE IF FEED FLOW INCREASES TO 750K LBS/HR GREATER THAN STEAM FLOW.

IF THE SEVERITY SELECTED IS LESS THAN THE ORIGINAL VALVE POSITION, FEEDWATER FLOW TO THE ASSOCIATED STEAM GENERATOR WILL DECREASE RESULTING IN STEAM GENERATOR LEVEL DECREASING AT A RATE DETERMINED BY THE FEED FLOW- STEAM FLOW MISMATCH. ANNUNCIATOR 15-A4(B4,C4,D4) "S/G 1A(B,C,D) FLOW MISMATCH FW FLOW LOW" WILL ACTUATE IF FEED FLOW DECREASES TO 750K LBS/HR LESS THAN STEAM FLOW.

THE CONTROL VALVE WILL STILL CLOSE ON RECEIPT OF ANY FEEDWATER ISOLATION SIGNAL.

MALFUNCTION REMOVAL WILL RESTORE THE MAIN FEEDWATER CONTROL VALVE POSITIONER TO NORMAL.

EVENTS: 1) DVR 06-02-89-055

DEVIATION REPORT

FW09

DVR NO. 06 - 02 - 89 - 055
STA UNIT YEAR NO.

Form Rev 2.0

PART 1 | TITLE OF DEVIATION

FEEDWATER FLOW OSCILLATIONS DUE TO CARD FAILURE

OCCURRED 04/22/89 2100
DATE TIME

SYSTEM AFFECTED	PLANT STATUS AT TIME OF EVENT	WORK REQUEST NO.	TESTING
FW	MODE 1 POWER(%) 58.5%	B66644	YES [ ] NO [X]

DESCRIPTION OF EVENT

The Unit experienced three separate events of Feedwater flow oscillations over an eight hour period. The 2FW540 valve initiated the oscillations each time. The failure of the 540 reg valve set up oscillations in the other three generators that required manual control to dampen out.

POTENTIALLY SIGNIFICANT EVENT PER NSD DIRECTIVE A-07 [ ] YES [X] NO

10CFR50.72 NRC RED PHONE [ ] 1 HOUR
NOTIFICATION MADE [ ] 4 HOUR ______ TIME [X] NO

D. J. Peterson 04/22/89
RESPONSIBLE SUPERVISOR DATE

PART 2 | OPERATING ENGINEER'S COMMENTS

Continue on going investigate.

[X] NON REPORTABLE EVENT

[ ] 30 DAY REPORTABLE/10CFR______

[ ] 5 DAY REPORT PER 10CFR21

[ ] ANNUAL/SPECIAL REPORT REQUIRED

A.I.R. #______

L.E.R. #______

NOTIFICATION ______ ______ ______
REGION III DATE TIME

Office of T. Maiman 04/28/89 0820
NSD DATE TIME

[ ] CECO CORPORATE NOTIFICATION MADE IF ABOVE NOTIFICATION IS PER 10CFR21

TELECOPY ______ ______ ______
CECO CORPORATE OFFICER DATE TIME

PRELIMINARY REPORT COMPLETED AND REVIEWED R. Hopkins 04/24/89
OPERATING ENGINEER DATE

INVESTIGATION REPORT & RESOLUTION ACCEPTED BY STATION REVIEW K Parsons 6/5/89

D Burrill 6/5/89

RESOLUTION APPROVED AND AUTHORIZED FOR DISTRIBUTION G K Schwartz 6/5/89
STATION MANAGER DATE

86-5176 (Form 15-52-1) 11-20-85

MENT ID

(0344R/0044R)

FW10 FW REGULATION BYPASS VALVE FAILURE

TYPE: GENERIC, RV 0-100% OF VALVE TRAVEL

A) 1FW510A
B) 1FW520A
C) 1FW530A
D) 1FW540A

CAUSE: POSITIONER FAILURE (VALVE PROBLEM, NO MANUAL CONTROL AVAILABLE)

REF: 20E-1-4030 FW05

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THE SELECTED STEAM GENERATOR BYPASS FEEDWATER CONTROL VALVE WILL FAIL TO THE POSITION SELECTED BY MALFUNCTION SEVERITY. IF THE SEVERITY SELECTED IS GREATER THAN THE ORIGINAL VALVE POSITION, FEEDWATER FLOW TO THE ASSOCIATED STEAM GENERATOR WILL INCREASE RESULTING IN STEAM GENERATOR LEVEL INCREASING AT A RATE DETERMINED BY THE FEED FLOW-STEAM FLOW MISMATCH. ANNUNCIATOR 15-A3(B3,C3,D3) "S/G 1A(B,C,D) FLOW MISMATCH STM FLOW LOW" WILL ACTUATE IF FEED FLOW INCREASES TO 750K LBS/HR GREATER THAN STEAM FLOW.

IF THE SEVERITY SELECTED IS LESS THAN THE ORIGINAL VALVE POSITION, FEEDWATER FLOW TO THE ASSOCIATED STEAM GENERATOR WILL DECREASE RESULTING IN STEAM GENERATOR LEVEL DECREASING AT A RATE DETERMINED BY THE FEED FLOW-STEAM FLOW MISMATCH. ANNUNCIATOR 15-A4(B4,C4,D4) "S/G 1A(B,C,D) FLOW MISMATCH FW FLOW LOW" WILL ACTUATE IF FEED FLOW DECREASES TO 750K LBS/HR LESS THAN STEAM FLOW.

THE CONTROL VALVE WILL STILL CLOSE ON RECEIPT OF ANY FEEDWATER ISOLATION SIGNAL.

MALFUNCTION REMOVAL WILL RESTORE THE MAIN FEEDWATER BYPASS CONTROL VALVE POSITIONER TO NORMAL.

EVENTS: 1) LER 20-02-88-021

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Braidwood 2	0 5 0 0 0 4 5 7	1 of 0 1

Title (4) Steam Generator Hi-Hi Level Due To Feedwater Regulating Bypass Valve Positioner Calibration Drift

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0 9	1 5	8 8	8 8	-- 0 2 1 --	0 1	1 2	2 8	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	2	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)			
POWER LEVEL (10)	0 0 2	__ 20.402(b)	__ 20.405(c)	X 50.73(a)(2)(iv)	__ 73.71(b)
		__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
		__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
		__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
		__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
		__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER	
	AREA CODE	
David W. Ibrahim, Technical Staff Engineer Ext. 2402	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

At 1313 on September 15, 1988 a high level alarm occurred on the 2C steam generator (S/G). The Nuclear Station Operator verified that the 2FW530A, feedwater regulating bypass valve, was in automatic and indicating zero demand. 2C S/G water level stayed constant and then started rising again. Feedwater was manually isolated. However, this did not result in the water level in the 2C S/G to drop before it reached the hi-hi trip setpoint causing turbine trip P-14. The S/G hi-hi alert and turbine trip P-14 were reset, due to a level drop, a few seconds after the trip. At 1415, the high level was reset. The root cause for this event was an out of calibration positioner for the 2FW530A. Nuclear Work Request A25549 was issued to repair or adjust the out of calibration positioner. Work was performed later and the valve positioner was recalibrated. There have been no previous occurrences due to this cause.

Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 2; Event Date: September 15, 1988; Event Time: 1356;

Reactor Mode: 2; Mode Name: Startup; Power Level: 2%

RCS [AB] Temperature/Pressure: 558 degrees F/2240 psig

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of the event which contributed to the severity of the event. The unit was coming down from full power during the previous shift as a result of a high cation conductivity and many of the activities as a result of that were still in progress.

Prior to the event, feedwater [SJ] was being routed to the 2C Steam Generator (S/G) [AB] through the upper nozzle. Flow was being controlled by the feedwater regulating bypass valve 2FW530A.

At 1313 on September 15, 1988 a high level alarm occurred on the 2C steam generator. The Nuclear Station Operator (NSO) made sure that the manual/automatic (M/A) station for the 2FW530A valve was in automatic and indicating zero demand (i.e. the valve was supposed to be closed). He waited for the water level in the 2C S/G to drop, but the S/G water level stayed almost constant for some time and then started rising again. The NSO manually closed the 2FW530A, Preheater Bypass Downstream Isolation Valve 2FW039C, Tempering Line Flow Control Valve 2FW034C, and the Tempering Line Isolation Valve 2FW035C valves. This action did not result in the water level in the 2C S/G dropping before it reached the hi-hi trip setpoint causing turbine trip P-14. Both the S/G hi-hi alert and turbine trip P-14 were reset, due to a level drop, a few seconds after the trip.

At 1415, the high level was reset. Operator actions neither increased or decreased the severity of the event.

The appropriate NRC notification via the ENS phone system was made at 1916 on September 19, 1988, pursuant to 10CFR50.72(b)(2)(ii).

This event is being reported pursuant to 10CFR50.73(a)(2)(iv) - Any event or condition that resulted in manual or automatic actuation of any engineered safety feature, including the reactor protection system.

C. CAUSE OF EVENT:

The root cause for this event was an out of calibration positioner for the feedwater regulating bypass valve 2FW530A. Maintenance work on the valve was performed after the event.

D. SAFETY ANALYSIS:

There was no affect on the plant or public safety as all engineered safety features operated as designed.

Under worst case conditions of the plant operating at 100% power with a design basis loss of feedwater, there would be no impact on the safety of the plant or public as this is enveloped by the Final Safety Analysis Report (FSAR). The Auxiliary Feedwater System was operable throughout the event.

E. CORRECTIVE ACTIONS:

Immediate corrective action was to restore the S/G water level to its normal operating band.

Nuclear Work Request A25549 was issued to repair or adjust the out of calibration positioner. Work was performed later and the valve positioner was recalibrated.

F. PREVIOUS OCCURRENCES:

There have been previous occurrence of a reactor trip involving steam generator level perturbations. The corrective actions were implemented addressing both root and contributing causes. However, the root cause of this event is different in that leaking through of the feedwater regulating bypass valve was involved. Previous corrective actions are not applicable to this event.

G. COMPONENT FAILURE DATA:

This event was not caused by component failure, nor did any components fail as a result of this event.

FW11 FW TEMPERING LINE ISOLATION VALVE FAILURE

TYPE: GENERIC, RV 0-100% OF VALVE TRAVEL

A) 1FW035A
B) 1FW035B
C) 1FW035C
D) 1FW035D

CAUSE: MECHANICAL BINDING

REF: 20E-1-4030 FW42
20E-1-4030 FW43
20E-1-4030 FW44
20E-1-4030 FW45
M-2036 SHEET 15

PLT STA: PLANT HEATUP

EFFECTS: THE SELECTED STEAM GENERATOR FEEDWATER TEMPERING ISOLATION VALVE WILL FAIL TO THE DESIRED POSITION. IF THE SEVERITY SELECTED IS LESS THAN ACTUAL POSITION THEN TEMPERING LINE FLOW DECREASES. THE ASSOCIATED STEAM GENERATOR LEVEL WILL DECREASE AT THE FEED FLOW-STEAM FLOW MISMATCH RATE. OPERATION OF THE ISOLATION VALVE CONTROL SWITCH OR FW ISOLATION SIGNAL WILL HAVE NO EFFECT ON THE VALVE POSITION. IF THE SEVERITY SELECTED IS GREATER THAN THE ACTUAL VALVE POSITION THEN FLOW THROUGH THE TEMPERING LINE WILL INCREASE. S/G LEVELS WILL INCREASE AND A SLIGHT COOLDOWN WILL BE NOTED.

MALFUNCTION REMOVAL WILL RESTORE THE STEAM GENERATOR FEEDWATER TEMPERING ISOLATION VALVE TO NORMAL.

EVENTS: NONE

FW12 FW PREHEATER BYPASS VALVE FAILURE

TYPE: GENERIC, RV 0-100% OF VALVE TRAVEL

A) 1FW039A
B) 1FW039B
C) 1FW039C
D) 1FW039D

CAUSE: MECHANICAL BINDING

REF: 20E-1-4030 FW46
20E-1-4030 FW47
20E-1-4030 FW48
20E-1-4030 FW49

PLT STA: UNIT STARTUP IN PROGRESS

EFFECTS: MALFUNCTION INSERTION CAUSES THE SELECTED STEAM GENERATOR FEEDWATER PREHEATER BYPASS VALVE TO FAIL AT THE SELECTED POSITION. THE BYPASS VALVE WILL NOT RESPOND TO CONTROL SWITCH DEMANDS OR FEEDWATER ISOLATION SIGNAL. EXCESSIVE OR INSUFFICIENT FW FLOW TO THE S/G MAY RESULT IN S/G LEVEL CONTROL TRANSIENTS. ALL FW AND S/G ANNUNCIATORS AND ASSOCIATED FLOW INDICATIONS WILL RESPOND APPROPRIATELY.

MALFUNCTION REMOVAL WILL RESTORE THE FEEDWATER PREHEATER BYPASS VALVE TO NORMAL.

EVENTS: 1) DVR 06-02-88-004
2) DVR 20-02-88-168
3) LER 06-02-88-009
4) LER 06-02-88-007

FW12

LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
Byron, Unit 2	0 5 0 0 0 4 5 5	1 of 0 3

Title (4) Thermal Binding of Steam Generator Preheater Bypass Valves Resulting in Low Steam Generator Level Reactor Trip

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
0 7	1 5	8 8	8 8	— 0 0 9 —	0 0	0 8	0 9	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 2

POWER LEVEL (10): 0 0 2

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	X 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	Telephone Number Area Code	
L. Sues, Assistant Superintendent Technical Services Ext. 2214	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	S J	V	A 3 9 1	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

__ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15) Month | Day | Year

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

On July 15, 1988, Byron Unit 2 reactor power was 2 percent. At 0436 a Nuclear Station Operator (NSO) attempted to open the Steam Generator Preheater Bypass Valves (2FW039A,B,C,D) to feed the Steam Generators (S/G's). The A and D valves opened properly, but the B and C valves failed to open as demanded. The levels in the 2B and 2C S/G's lowered to the low-low level reactor trip setpoint at which point the reactor automatically tripped. The licensed operators entered and complied with emergency operating procedures. Stable plant conditions were achieved in Hot Standby at 0500.

The event was caused by the failures of valves 2FW039B and 2FW039C to open on demand. These valves need to be open to provide sufficient feedwater flow to the S/G's at 1 to 2 percent reactor power. The level instabilities induced by the valve failures made level control difficult. Investigation revealed that the valves had become thermally bound following their automatic closure during a reactor trip event on July 14, 1988.

Both valves were opened using hydraulic lifts and 2FW039C operated properly, however, 2FW039B still would not open when demanded by the handswitch. A non-safety related air check valve was replaced and the 2FW039B valve operated properly.

A previous similar occurrence was reported in Unit 2 Licensee Event Report 88-007.

(0075R/0008R)

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Byron, Unit 2	0 5 0 0 0 4 5 5	8 8	- 0 0 9 -	0 0	0 2 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 7/15/88 / 0436

Unit 2 MODE 2 - Startup Rx Power 2% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of this event that contributed to the event. On July 15, 1988, Byron Unit 2 was in the Startup Operational Mode (Mode 2) with reactor power at 2 percent. At 0436 a Nuclear Station Operator (NSO) (licensed reactor operator) attempted to open the Steam Generator Preheater Bypass Valves [SJ] (2FW039A,B,C,D) to feed the Steam Generators (S/G's). The 2FW039A and 2FW039D valves opened properly, but the 2FW039B and 2FW039C valves failed to open as demanded. The NSO isolated the blowdown flow paths from the 2B and 2C S/G's in an effort to slow the rate of level decrease in the S/G's. The levels in the 2B and 2C S/G's continued to lower. As the low-low level reactor trip setpoint (17%) was approached and it was evident that levels could not be restored, a licensed Senior Reactor Operator directed the NSO to manually trip the reactor, however the Reactor Protection System initiated an automatic reactor trip due to low-low level in the 2C S/G before the manual trip was accomplished. The licensed operators entered and complied with the "Reactor Trip or Safety Injection-Unit 2 Emergency Operating Procedure" (2BEP-0) and the "Reactor Trip Response Unit 2 Emergency Operating Procedure" (2BEP ES-0.1). The 2A and 2B Auxiliary Feedwater Pumps (AFP) [BA] automatically started due to the low-low S/G level condition as expected. An expected Feedwater Isolation occurred due to the opening of the reactor trip breakers coincident with low average reactor coolant temperature (T_{avg}) of 564°F.

At 0446 the 2B AFP was stopped since its operation was not required to maintain S/G levels. At 0451 the Feedwater Isolation signal was reset. At 0454 the Startup Feedwater Pump was started and flow was established to the S/G's. At 0455 the 2A AFP was stopped. At approximately 0500 the stable plant conditions were achieved in Hot Standby (Mode 3). Valves 2FW039B and 2FW039C were declared inoperable and Technical Specification Limiting Condition for Operation Action Requirement (LCOAR) 3.6.3 for the two containment isolation valves was entered and satisfied.

This Licensee Event Report (LER) is submitted in accordance with 10CFR50.73 (a)(2)(iv) due to the automatic Reactor Protection System and Engineered Safety Features (ESF) actuations.

C. CAUSE OF EVENT:

The event was caused by the failures of valves 2FW039B and 2FW039C to open on demand. At 1 to 2 percent reactor power the Preheater Bypass Valves must be open to provide sufficient feedwater flow to maintain S/G levels. The Byron Unit 2 S/G's are Westinghouse Model D-5. The shrink/swell phenomena are most pronounced at low power, and Unit 2 was at 2 percent reactor power at the time of the event. The level instabilities induced by the failure of the 2FW039B and 2FW039C valves made control of the S/G levels difficult. The licensed operators' actions were in accordance with Station Operating Procedures and operating strategies for D-5 S/G level control.

The 2FW039B and 2FW039C valves were found to be thermally bound. The valves had automatically closed when Unit 2 tripped on July 14, 1988 (see Unit 2 LER 88-008). Normally, the valves are manually closed during a controlled shutdown of the plant. A controlled shutdown does not close the Preheater Bypass Valves at such high feedwater temperatures.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year		Sequential Number		Revision Number	Page (3)
Byron, Unit 2	0 5 0 0 0 4 5 5	8 8	-	0 0 9	-	0 0	0 3 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

D. SAFETY ANALYSIS:

The event occurred when the majority of Feedwater Isolation Valves were already closed. All ESF systems actuated and functioned as designed. The S/G levels were reestablished and the Unit stabilized operations in Mode 3. The Failed Preheater Bypass Valves failed in their safe position. Neither plant nor public safety were affected by this event.

E. CORRECTIVE ACTIONS:

The 2FW039B valve was uncoupled from its actuator. The actuator was found to be operating correctly. The 2FW039B and 2FW039C were then opened using hydraulic lifts. The 2FW039C stroked properly. The 2FW039B did not operate properly from the control switch. The problem was isolated to a non-safety related air check valve for the "C" solenoid. The air check valve was replaced and the 2FW039B stroked properly when demanded from the handswitch. The valves were successfully tested and returned to operable status. Technical Specification LCOAR 3.6.3 for the failed valves was exited at 1838 on July 15, 1988. In addition, applicable operating procedures will be reviewed for the possible inclusion of a step to stroke the Preheater Bypass Valves subsequent to a reactor trip. This may prevent thermal binding of the type that was experienced during this event. This proposed corrective action is tracked by Action Item Record 454-225-88-0158.

The Preheater Bypass Valves have not thermally bound following a normal unit shutdown. No further corrective action is planned at this time.

F. PREVIOUS OCCURRENCES:

LER NUMBER	TITLE
88-007	Feedwater Isolation Actuation due to S/G Preheater Bypass Valve Failure to Open

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Stratafio Products	1/2-inch NPT Check Valve		

LICENSEE EVENT REPORT (LER)

FW12

Facility Name (1)	Docket Number (2)	Page (3)
Byron, Unit 2	0 5 0 0 0 4 5 5	1 of 0 3

Title (4) Feedwater Isolation Actuations Due to Steam Generator Preheater Bypass Valve Failure to Open

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0 6	0 3	8 8	8 8	0 0 7	0 0	0 6	3 0	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 2

POWER LEVEL (10): 0 0 2

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	X 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER	
	AREA CODE	
D. Brindle, Operating Engineer Ext. 2218	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
__ Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

On June 3, 1988, Unit 2 was in the Startup operational mode at 2% reactor power. A Nuclear Station Operator (NSO) had established Steam Generator Preheater Bypass feedwater flow to the A, B and D Steam Generators, but could not open the Preheater Bypass Valve (2FW039C) for the C Steam Generator. Levels in the A, B and D Steam Generators increased while C Steam Generator level decreased. Valve 2FW039C was opened by a non-licensed operator locally, but D Steam Generator level reached the high-2 level setpoint of 78.1% which actuated the P-14 permissive and caused an automatic feedwater isolation. While recovering from this feedwater transient another P-14 permissive actuation occurred due to high-2 level in the A Steam Generator at 1338. Levels were restored, the Feedwater Isolation signal was reset, and the plant startup continued without further incident.

The root cause of the event was the unexpected failure of the 2FW039C valve to open when demanded remotely by the NSO in the Main Control Room. The valve failure initiated a feedwater flow control disturbance that resulted in the high-2 steam generator level condition. Subsequent stroking of the 2FW039C valve remotely was successful in all attempts.

NSO actions during the event were in accordance with current operating strategies for dealing with D-5 Steam Generator level control problems. The strategies have been successful during normal plant startups, but were not able to compensate for the level effects caused by the 2FW039C valve failure. All licensed operators will be required to read this Licensee Event Report (LER) to reinforce their understanding of D-5 Steam Generator level control during low power operations.

Previous similar occurrences were reported in Unit 2 LER 87-002.

(0037R/0002R)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					Page (3)		
		Year	////	Sequential Number	////	Revision Number			
Byron, Unit 2	0 5 0 0 0 4 5 5	8 8	-	0 0 7	-	0 0	0 2	OF	0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 6/3/88 / 1327

Unit 2 MODE 2 - Startup Rx Power 2% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On June 3, 1988, Unit 2 was in the Startup operational mode at 2% reactor power. There were three Nuclear Station Operators (NSO's) (licensed reactor operators) assigned to the Startup. One had assumed the reactor operations board, the second had the feedwater board, and the third was preparing for a Main Turbine [TB] roll. The NSO at the feedwater board was attempting to initiate feedwater (FW) [SJ] flow through the Steam Generator Preheater Bypass Flow valves, 2FW039A, B, C, D. At 1327 the NSO had established flow through 2FW039A, B and D and was attempting to open the final Preheater Bypass valve, 2FW039C, when it failed to open. The feedwater board NSO's attention was diverted from the A, B and D Steam Generator levels to the opening of the 2FW039C valve. The NSO at the Main Turbine control board ceased his activities there and immediately assumed a position at the feedwater control panel. Levels in the A, B and D Steam Generators began to rise and C Steam Generator level lowered. Feedwater flow to the A, B and D Steam Generators was manually isolated in an attempt to control the level increases while attempts were made to establish preheater flow to the C Steam Generator as its level continued to lower. The 2FW039C valve was opened by a non-licensed operator locally. At 1327 the D Steam Generator level reached a high-2 level setpoint of 78.1% which actuated the P-14 permissive and caused an automatic Feedwater Isolation and a Main Turbine Trip. The NSO's restored all Steam Generator levels to normal and reset the Feedwater Isolation signal. At 1338 while attempting to stabilize Steam Generator levels the A Steam Generator level swelled causing another P-14 actuation and Feedwater Isolation. Levels were restored to normal, the Feedwater Isolation signal was reset and the startup continued without further incident.

This Licensee Event Report is submitted in accordance with 10CFR50.73 (a)(2)(iv) due to the two Engineered Safety Feature (ESF) actuations.

C. CAUSE OF EVENT:

The intermediate cause of the event is attributed to the atypical response of D5 Steam Generator level indications at low power levels. The failure of the 2FW039C valve to open upset the controlled addition of feedwater through the 2FW039A, B, D Preheater Bypass Valves which in turn initiated a swell in the Steam Generators. The shrink/swell phenomena are most pronounced at low power, and Unit 2 was at 2 percent reactor power at the time of the event. One NSO's attention was diverted from the remaining Steam Generator levels but the duty was immediately assumed by the turbine control operator, therefore, inattention was not a contributor to the event.

The second level excursion occurred at 1338 hours when NSO's were recovering from the first feedwater transient and were attempting to stabilize Steam Generator levels. The A Steam Generator level swelled and initiated a Feedwater Isolation. The level was restored and the unit startup continued.

C. CAUSE OF EVENT: (Continued)

The cause of the event is attributed to the unexpected failure of the 2FW039C to open when demanded remotely by the NSO. This occurrence initiated a disturbance in the controlled feed to all four Steam Generators and resulted in Steam Generator levels shrinking on C and swelling on A, B, D. The 2FW039C was opened by a non-licensed operator who momentarily interrupted air flow from the air operated valve's diaphragm. The valve worked properly during several subsequent strokes. It is speculated that a check valve in the air flow path to 2FW039C was not properly seated, and this was the root cause of the failure of the valve to open. The non-licensed operator interrupted the air flow, thus permitting the check valve to seat and permit subsequent satisfactory operation of 2FW039C.

D. SAFETY ANALYSIS:

The Feedwater Isolations occurred when the majority of Feedwater Isolation Valves were already closed due to the low power operating condition. All safety systems actuated as designed. The Main Turbine had been latched in preparation to roll, therefore, there were no significant effects on the turbine. Steam Generator levels were quickly lowered from the P-14 Permissive setpoint and the Feedwater Isolation signals were reset. All ESF equipment functioned as designed. At no time was plant or public safety threatened.

E. CORRECTIVE ACTIONS:

All operator actions were correct in responding to indicated process changes. The operator followed the four main strategies for controlling D-5 Steam Generator levels:

1. Give 100% dedicated attention to S/G level control during low power transients.

2. Prior to inducing a planned transient, all plant parameters shall be stable at their nominal valve.

3. Only one parameter and one Steam Generator at a time will be altered unless inaction would result in a protective feature actuation.

4. All changes are made in small increments with time for stabilization between steps.

The level instabilities induced by the failure of the 2FW039C valve made control of the Steam Generator levels difficult. The strategies have been successful during normal plant startups, but were not able to compensate for the level effects caused by the 2FW039C valve failure. All licensed reactor operators and senior reactor operators will be required to read this Licensee Event Report to reinforce their understanding of the care that must be exercised in controlling D-5 Steam Generator levels at low power. The 2FW039C valve was successfully stroked several times following its initial local opening by the non-licensed operator. The valve remained in service and functioned normally while the power escalation continued.

F. PREVIOUS OCCURRENCES:

LER NUMBER	TITLE
87-002	Reactor Trips and Feedwater Isolations Due to Operator Difficulty in Controlling Steam Generator Level Transients at Low Power

G. COMPONENT FAILURE DATA:

	MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
a)	Not Applicable			

FW12

DEVIATION INVESTIGATION REPORT (DIR)

Form Rev 2.0

Facility Name	PAGE
Braidwood Unit 2	1 OF 0 2

Title: Failure of Feedwater Bypass Isolation Valve to Open due to Mechanical Binding

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	1
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	0 9 0
1 0	0 7	8 8	2 0	0 2	8 8	1 6 8	0 0	1 1	1 5	8 8		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
David W. Ibrahim, Technical Staff Engineer Ext. 2402	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
X YES (if yes, complete EXPECTED SUBMISSION DATE)	NO		0 6	0 1	8 9

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 2, Event Date: October 7, 1988 Event Time: 0607;

Mode: 1 - Power Generation Rx Power: 90%;

RCS [AB] Temperature/Pressure: N.O.T./N.O.P.

B. DESCRIPTION OF EVENT:

On 10/7/88 at 0607, during power ascension according to procedure 2BwGP 100-3, Power Ascension 5% to 100% the Nuclear Station Operator (NSO) performed step F.59 to open the feedwater isolation valves 2FW039A, 2FW039B, 2FW039C and 2FW039D. He noticed that 2FW039A valve failed to open. Nuclear Work Request number (NWR #) A26043 was issued and Limiting Condition for Operating Action Requi-ement (LCOAR) 6.3-1A, Containment Isolation Valves Tech Spec LCO 3.6.3, was entered. There were no inoperable systems or components that contributed to the event.

C. CAUSE OF EVENT:

The cause of this event is partially due to tight packing resulting in high friction between the valve stem and the packing material. In addition it is also suspected that mechanical or thermal binding may have occurred within the valve internals.

D. SAFETY ANALYSIS:

The feedwater bypass line is a means to deliver feedwater to the Steam Generator (S/G) through the upper nozzle, instead of the main, lower, nozzle, at low loads to prevent the formation of water hammer in the S/G preheater section. The feedwater isolation bypass valve (2FW039A) safety function is to isolate between the S/G, in the Containment Building, and the rest of the feedwater system if there is an Engineering Safeguard Feature (ESF) actuation. The valve was already closed, functioning as an isolation valve, and at no time were the public, personnel or the equipment in danger.

E. CORRECTIVE ACTIONS:

i) Immediate Actions:

NWR #A26043 was issued and LCOAR 63-1A was entered. Packing was adjusted and the stem was lubricated. The valve was operated and passed Technical Staff surveillance BwVS 6.3.3-20.

ii) Long Term Actions:

1) In response to the possible thermal/mechanical binding problem, BwGP 100-3 will be changed to instruct the NSO's, at step F.59, to open the feedwater bypass isolation at 70% power. This will reduce the pressure differential across the valve and, consequently, will reduce mechanical binding of the valve internals.

2) This event may be similar to Byron's events reported in Byron's Licensee Event Reports number 88-007 and 88-009. The applicable procedures will be reviewed for possible inclusion of a step to stroke the subject feedwater bypass valves subsequent to a reactor trip. This is being done as a result of both the event at Braidwood and in response to INPO SER 8-88, Pressure Locking of Residual Heat Removal Gate Valves.

All procedures review will be tracked by action item 457-200-88-16801.

3) NWR #A26308 has been issued to inspect the valve internals. Also removing the check valves 2FW078A, 2FW078B, 2FW078C and 2FW078D, per modification number M20-2-88-029, will reduce the pressure differential across each of the feedwater bypass isolation valves and will further reduce the internals binding. Results will be tracked by action item 457-200-88-16802.

F. PREVIOUS OCCURRENCES:

There have been no previous occurrences of mechanical binding recorded in a Deviation Report.

G. COMPONENT FAILURE DATA:

None.

DEVIATION INVESTIGATION REPORT

FW12

TITLE

FAILURE OF FEEDWATER PREHEATER BYPASS VALVE 2FW039B TO CLOSE FROM HANDSWITCH

PAGE 1 OF 0 2

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	POWER LEVEL
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR		
0 1	0 4	8 8	0 6	0 2	8 8	0 0 4	0 1	1 2	3 0	8 8	1	0 8 0

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER: AREA CODE	
W. Walters, Asst Tech Staff Supervisor Ext. 2240	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED

YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO	EXPECTED SUBMISSION DATE: MONTH	DAY	YEAR

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 1/4/88 / 1118 hrs

Unit 2 MODE 1 - Power Operations Rx Power 80% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On 1/4/88, at 1118, Byron Unit 2 was in Mode 1 operating at 80% power when the Feedwater (FW) [SJ] Preheater Bypass Valve 2FW039B failed to close on demand from the Main Control Board, 1PM04J, control switch. Subsequently, the air supply to the valve actuator was closed and the valve failed closed. A LCOAR, Limiting Condition For Operation Action Requirement, was entered.

C. CAUSE OF EVENT:

The cause of the failure of the 2FW039B valve to close has been determined to be the inability of the "C" solenoid to vent the air from the valve diaphragm/accumulator. The valve opons properly and would close using the two remaining safety related solenoids.

D. SAFETY ANALYSIS:

The plant or public safety was not affected by the failure of the 2FW039B valve to close from the Main Control Board switch signal. The valve would have closed on a Feedwater Isolation Signal due to the two remaining safety related solenoids. The affected solenoid does not receive a safety signal and is used only for manual opening and closing functions on the 2FW039B valve. The last known operable time of the "C" solenoid was on 12/12/87 when 2FW039 was tested under 2BVS 6.3.3-20.

DEVIATION INVESTIGATION REPORT TEXT CONTINUATION

TITLE	DIR NUMBER							PAGE
	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
FAILURE OF FEEDWATER PREHEATER BYPASS VALVE 2FW039B TO CLOSE FROM HANDSWITCH	0 6	0 2	8 8	—	0 0 4	—	0 1	2 OF 0 2

TEXT

E. CORRECTIVE ACTIONS:

The Preheater Bypass valve was verified to close by using one of the safety related solenoids. The "C" solenoid, which is non-safety related, was replaced under Nuclear Work Request B51747, and it was verified that the solenoid was receiving the signal from the control switch. The problem has been determined to be the failure of the air to vent from the "C" solenoid. A nuclear work request was written to investigate and repair the venting problem. The valve was declared operable when the valve was verified by temporary procedure 88-2-014, to close in less than 6 seconds on 1/6/88, using one of the safety related solenoids to close the valve. On 7/22/88, the instrument air check valve was replaced. The check valve disc had become disconnected from its stem and stem nut. The valve was successfully tested and returned to operable status.

F. PREVIOUS OCCURRENCES:

DVR NUMBER	TITLE
NONE	

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Parker Hannifin	Check valve	3/4"	

b) RESULTS OF NPRDS SEARCH:

Not Applicable

c) NWR

None

FW13 FW ISOLATION VALVE FAILURE

TYPE: GENERIC, RV 0-100% OF VALVE TRAVEL

A) 1FW009A
B) 1FW009B
C) 1FW009C
D) 1FW009D

CAUSE: MECHANICAL BINDING

REF: 20E-1-4030 FW18
20E-1-4030 FW19
20E-1-4030 FW20
20E-1-4030 FW21

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: MALFUNCTION INSERTION CAUSES THE SELECTED STEAM GENERATOR FEEDWATER ISOLATION VALVE TO FAIL AT THE SELECTED SEVERITY WHEN THE VALVE PASSES THROUGH THE SELECTED POSITION. MAIN FEEDWATER FLOW TO THE ASSOCIATED STEAM GENERATOR WILL DECREASE TO ZERO WITH STEAM GENERATOR LEVEL DECREASING AT A RATE DETERMINED BY THE FEED FLOW-STEAM FLOW MISMATCH. ANNUNCIATOR 15-A4(B4,C4,D4) "S/G 1A(B,C,D) FLOW MISMATCH FW FLOW LOW" WILL ACTUATE IF FEED FLOW DECREASES TO 750K LBS/HR LESS THAN STEAM FLOW. OPERATION OF THE ISOLATION VALVE CONTROL SWITCH OR FW ISOLATION SIGNAL WILL HAVE NO EFFECT ON THE VALVE POSITION.

MALFUNCTION REMOVAL WILL RESTORE THE FEEDWATER ISOLATION VALVE TO NORMAL.

EVENTS: NONE

FW14 FEED LINE BREAK BETWEEN FW009 & CONTAINMENT

TYPE: GENERIC, NRVI 0-3.5 MLBM/HR AT 900 PSID

A)	1A FW LINE	C)	1C FW LINE
B)	1B FW LINE	D)	1D FW LINE

CAUSE: PIPE BREAK BETWEEN FW009 AND CONTAINMENT WALL

REF: M-36 SHEETS 1A - 1D

PLT STA: 100% REACTOR POWER

EFFECTS: THIS MALFUNCTION RESULTS IN THE LEAKAGE OF WATER FROM THE FEEDWATER SYSTEM INTO THE STEAM TUNNEL ENVIRONMENT. ACTUAL FEEDWATER FLOW TO THE STEAM GENERATOR ASSOCIATED WITH THE SELECTED LINE WILL DECREASE. STEAM GENERATOR LEVEL WILL BEGIN TO DECREASE DUE TO THE MISMATCH BETWEEN STEAM FLOW AND FEED FLOW. THE STEAM GENERATOR WATER LEVEL CONTROL SYSTEM WILL RESPOND TO PROVIDE ADDITIONAL FLOW TO THE AFFECTED LINE. AS THE MALFUNCTION SEVERITY IS INCREASED, THE RATE OF DECREASE IN STEAM GENERATOR LEVEL WILL INCREASE. AS STEAM GENERATOR LEVEL DECREASES, A STEAM GENERATOR LO-LO WATER LEVEL REACTOR TRIP WILL OCCUR.

THE INITIAL REDUCED FEEDWATER FLOW WILL CAUSE AN INCREASE IN REACTOR COOLANT SYSTEM T_{AVE} DUE TO THE REDUCED FLOW OF FEEDWATER INTO THE STEAM GENERATOR. NORMAL REACTOR COOLANT AND REACTOR POWER CONTROL SYSTEMS WILL RESPOND TO THE CHANGES IN SYSTEM TEMPERATURE.

THE STEAM GENERATOR WILL CONTINUE TO BLOWDOWN UNTIL DRY. AS THE BLOWDOWN CONTINUES THE RCS TEMPERATURES WILL DECREASE. THE OPERATOR MAY SLOW/STOP THE RCS COOLDOWN BY ISOLATING AUX FEEDWATER TO THE FAULTED STEAM GENERATOR.

MALFUNCTION SEVERITY MAY ONLY BE INCREASED FOR THIS MALFUNCTION AND THE SIMULATOR MUST BE RESET TO REMOVE THIS MALFUNCTION.

EVENTS: NONE

FW15 MAIN FW PUMP SHAFT BREAK

TYPE: GENERIC, NRB

A) 1A FW PUMP
B) 1B FW PUMP
C) 1C FW PUMP

CAUSE: MECHANICAL FAILURE OF FW PUMP SHAFT

REF: M-?6 SHE?T 2

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SHAFT OF THE SELECTED FEEDWATER PUMP TO BREAK. AT LOWER POWER LEVELS, THE SELECTED FEEDWATER PUMP'S TURBINE HP AND LP STOP VALVES OPEN ATTEMPTING TO INCREASE TURBINE SPEED. AT 100% POWER THE THE SELECTED TURBINE MAY OVERSPEED. THE MOTOR DRIVEN PUMP, IF SELECTED, DICHARGE FLOW CONTROL VALVE (1FW016) WILL MODULATE OPEN ATTEMPTING TO RESTORE FLOW AT ANY POWER LEVEL.

THE SELECTED FEEDWATER PUMP DISCHARGE FLOW INDICATION DECREASES TO ZERO AND FEEDWATER PUMP DISCHARGE HEADER PRESSURE DECREASES. DEPENDING ON THE INITIAL PLANT CONDITION, NORMAL FEEDWATER FLOW TO THE STEAM GENERATORS MAY BE PARTIALLY LOST. AS THE PUMP DISCHARGE FLOW DECREASES, THE ASSOCIATED PUMP'S RECIRC VALVE (1FW012A, B, C) WILL AUTO OPEN. THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY RAMPING BACK THE MAIN TURBINE AND/OR STARTING THE STANDBY FEEDWATER PUMP.

THE SIMULATOR MUST BE RESET TO RESTORE THE MAIN FEEDWATER PUMP SHAFT TO NORMAL.

EVENTS: NONE

FW16 FW HEADER PRESSURE TRANSMITTER FAILURE

TYPE: DISCRETE, RV 0-1500 PSIG

CAUSE: PT-508 FAILURE

REF: M-2036 SHEET 3

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: MALFUNCTION INSERTION CAUSES PT-508 TO FAIL TO THE SELECTED SEVERITY. FEEDWATER HEADER DISCHARGE PRESSURE INDICATION (1PI-508) INDICATES THE SELECTED SEVERITY HEADER PRESSURE.

IF THF MALFUNCTION SEVERITY SELECTED IS GREATER THAN ACTUAL HEAL ER PRESSURE, THE MASTER FEEDWATER PUMP SPEED CONTROLLER OUTPUT WILL DECREASE. MAIN FEEDWATER PUMPS 1B AND 1C TURBINE SPEEDS WILL DECREASE WITH RESULTING DECREASES IN MAIN FEEDWATER DISCHARGE HEADER PRESSURE AND FLOW.

IF THE MALFUNCTION SEVERITY SELECTED IS LESS THAN ACTUAL HEADER PRESSURE, THE MASTER FEEDWATER PUMP SPEED CONTROLLER OUTPUT WILL INCREASE. MAIN FEEDWATER PUMPS 1B AND 1C TURBINE SPEEDS WILL INCREASE WITH RESULTING INCREASES IN MAIN FEEDWATER DISCHARGE HEADER PRESSURE AND FLOW.

MALFUNCTION REMOVAL WILL RESTORE PT-508 TO NORMAL.

EVENTS: NONE

FW17 HEATER DRAIN TANK LEVEL CONTROLLER FAILURE

TYPE: DISCRETE, RV 0-100%

CAUSE: FAULTY LEVEL CONTROLLER 1LK-HD009A (AUTOMATIC ONLY)

REF: 20E-1-4031 HD07
20E-1-4030 HD24
M-2041 SHEET 33

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE OUTPUT OF HEATER DRAIN (HD) TANK LEVEL CONTROLLER 1LK-HD009A WILL FAIL TO THE VALUE DETERMINED BY THE SELECTED SEVERITY.

IF THE SELECTED SEVERITY IS GREATER THAN THE CONTROLLER OUTPUT, THEN THE HEATER DRAIN PUMP DISCHARGE VALVES TO THE CONDENSATE HEADER, 1HD046A/B,WILL MODULATE OPEN. THIS WILL RESULT IN AN INCREASE IN HEATER DRAIN PUMP FLOW AND A DECREASE IN CONDENSATE PUMP FLOW. AS HEATER DRAIN PUMP FLOW INCREASES, ANNUNCIATOR 17-D3 "HD PUMP DSCH FLOW HIGH" ACTUATES. THE INCREASE IN HD PUMP FLOW RESULTS IN A DECREASE IN HD TANK LEVEL AS INDICATED ON 1LI-HD009. THE HD TANK MAKEUP VALVE 1HD122 MODULATES OPEN TO MAINTAIN TANK LEVEL, BUT MAKEUP WILL OCCUR ONLY IF IT IS UNISOLATED.

IF THE SELECTED SEVERITY IS LESS THAN THE CONTROLLER OUTPUT, THE HEATER DRAIN PUMP DISCHARGE VALVES TO THE CONDENSATE HEADER, 1HD046A/B, WILL MODULATE CLOSE. THIS WILL RESULT IN AN DECREASE IN HEATER DRAIN PUMP FLOW AND A INCREASE IN CONDENSATE PUMP FLOW. HEATER DRAIN TANK LEVEL INCREASES AS INDICATED ON 1LI-HD009. AS HEATER DRAIN TANK LEVEL INCREASES ANNUNCIATOR 17-E4 "HD TANK LEVEL HIGH LOW" IS ACTUATED. HD TANK OVERFLOW VALVE (1HD117) OPENS AND ANNUNCIATOR 17-E5 "HD TANK OVERFLOW VALVE OPEN" ACTUATES.

MALFUNCTION REMOVAL WILL RESTORE THE CONTROLLER TO NORMAL.

EVENTS: NONE

FW18 FW HEATER TUBE LEAK (17)

TYPE: GENERIC, RV 0-3 MLBM/HR @ 700 PSID

A) 17A HP HEATER
B) 17B HP HEATER

CAUSE: TUBE FAILURE AT FW INLET TO HEATER

REF: 20E-1-4030 HD18
20E-1-4030 HD28
M-36 SHEET 2
M-41 SHEET 2

PLT STA: 100% REACTOR POWER

EFFECTS: THIS MALFUNCTION RESULTS IN THE LEAKAGE OF FEEDWATER FROM THE MAIN FEEDWATER HEADER INTO THE SELECTED HIGH PRESSURE HEATER. AT LOW SEVERITY LEVELS, THE NORMAL HEATER DRAIN OUTLET VALVE 1HD008A(B) MODULATES TO MAINTAIN NORMAL LEVEL. AS MALFUNCTION SEVERITY IS INCREASED, EMERGENCY DRAIN OUTLET VALVE 1HD038A(B) WILL OPEN IN AN ATTEMPT TO MAINTAIN HEATER LEVEL AND ANNUNCIATOR 17-C1 "HTR 17 EMERGENCY DRAIN VLV OPEN" WILL ACTUATE. ANNUNCIATOR 17-B1 "HTR 17 LEVEL HIGH LOW" WILL ALSO ACTUATE.

IF HEATER LEVEL INCREASES FURTHER, ANNUNCIATOR 17-A1 "HTR 17 LEVEL HI-2" WILL ACTUATE, THE ASSOCIATED SECOND STAGE REHEATER DRAIN TANK OUTLET VALVES TO HP HEATER 17A(B) 1HD005A/1HD005C (1HD005B/1HD005D) WILL CLOSE, HP HEATERS 17A/17B EXTRACTION ISOLATION VALVE 1ES004 WILL CLOSE, HP HEATER 17A/17B EXTRACTION CHECK VALVE 1ES005 WILL CLOSE, AND HP HEATERS 17A/17B EXTRACTION STEAM SPILL VALVE 1ES022 WILL OPEN.

THE MAIN FEEDWATER SYSTEM WILL REFLECT THE LEAKAGE AS A DECREASE IN MAIN FEEDWATER PUMP DISCHARGE PRESSURE AND INCREASED FEEDWATER PUMP DISCHARGE FLOW. THE STEAM GENERATOR WATER LEVEL CONTROL SYSTEM WILL INCREASE FW FLOW UNTIL MFP CAPACITY IS REACHED, THEN S/G LEVELS WILL DECREASE. MAXIMUM FAILURE MAY CAUSE AN OPDT RUNBACK FROM 100% POWER.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE INTEGRITY OF THE HP HEATERS TUBES.

EVENTS: NONE

FW19 FW LINE BREAK INSIDE CONTAINMENT

TYPE: GENERIC, NRVI 0-3.5 MLBM/HR @ 900 PSID

A) FW LINE A
B) FW LINE B
C) FW LINE C
D) FW LINE D

CAUSE: PIPE RUPTURE IMMEDIATELY DOWNSTREAM OF CNMT PENETRATION

REF: M-36 SHEET 1A
M-36 SHEET 1B
M-36 SHEET 1C
M-36 SHEET 1D

PLT STA: 100% REACTOR POWER

EFFECTS: THIS MALFUNCTION RESULTS IN THE LEAKAGE OF WATER FROM THE FEEDWATER SYSTEM INTO THE CONTAINMENT ENVIRONMENT. ACTUAL FEEDWATER FLOW TO THE STEAM GENERATOR ASSOCIATED WITH THE SELECTED LINE WILL DECREASE. STEAM GENERATOR LEVEL WILL BEGIN TO DECREASE DUE TO THE MISMATCH BETWEEN STEAM FLOW AND FEED FLOW. THE STEAM GENERATOR WATER LEVEL CONTROL SYSTEM WILL RESPOND TO PROVIDE ADDITIONAL FLOW TO THE AFFECTED LINE. AS THE MALFUNCTION SEVERITY IS INCREASED, THE RATE OF DECREASE IN STEAM GENERATOR LEVEL WILL INCREASE. AS STEAM GENERATOR LEVEL DECREASES, A STEAM GENERATOR LO-LO WATER LEVEL REACTOR TRIP-TURBINE TRIP WILL OCCUR.

THE INITIAL REDUCED FEEDWATER FLOW WILL CAUSE AN INCREASE IN REACTOR COOLANT SYSTEM T_{AVE} DUE TO THE REDUCED FLOW OF FEEDWATER INTO THE STEAM GENERATOR. NORMAL REACTOR COOLANT AND REACTOR POWER CONTROL SYSTEMS WILL RESPOND TO THE CHANGES IN SYSTEM TEMPERATURE.

CONTAINMENT TEMPERATURE, HUMIDITY, PRESSURE, AND SUMP LEVEL(S) WILL INCREASE DUE TO THE HIGH TEMPERATURE FEEDWATER ENTERING THE CONTAINMENT ENVIRONMENT. CONTAINMENT TEMPERATURE AND PRESSURE WILL INCREASE UNTIL THE AFFECTED STEAM GENERATOR HAS COMPLETELY BLOWNDOWN THROUGH THE PIPING RUPTURE.

MALFUNCTION SEVERITY MAY ONLY BE INCREASED FOR THIS MALFUNCTION AND THE SIMULATOR MUST BE RESET TO REMOVE THIS MALFUNCTION.

EVENTS: NONE

FW20 FW LINE BREAK OUTSIDE CONTAINMENT

TYPE: GENERIC, NRVI 0-3.5 MLBM/HR @ 900 PSID

A) FW LINE A
B) FW LINE B
C) FW LINE C
D) FW LINE D

CAUSE: PIPE RUPTURE BETWEEN 1FW079 & 1FW009

REF: M-36 SHEET 1A
M-36 SHEET 1B
M-36 SHEET 1C
M-36 SHEET 1D

PLT STA: 100% REACTOR POWER

EFFECTS: THIS MALFUNCTION RESULTS IN THE LOSS OF MASS FROM THE FEEDWATER SYSTEM TO THE STEAM TUNNEL. ACTUAL FEEDWATER FLOW TO THE STEAM GENERATOR ASSOCIATED WITH THE SELECTED LINE WILL DECREASE. STEAM GENERATOR LEVEL WILL BEGIN TO DECRFASE DUE TO THE MISMATCH BETWEEN STEAM FLOW AND FEED FLOW. THE STEAM GENERATOR WATER LEVEL CONTROL SYSTEM WILL RESPOND TO PROVIDE ADDITIONAL FLOW TO THE AFFECTED LINE. AS THE MALFUNCTION SEVERITY IS INCREASED, THE RATE OF DECREASE IN STEAM GENERATOR LEVEL WILL INCREASE. AS STEAM GENERATOR LEVEL DECREASES, A STEAM GENERATOR LO-LO WATER LEVEL REACTOR TRIP-TURBINE TRIP WILL OCCUR.

THE INITIAL REDUCED FEEDWATER FLOW WILL CAUSE AN INCREASE IN REACTOR COOLANT SYSTEM T_{AVE} DUE TO THE REDUCED FLOW OF FEEDWATER INTO THE STEAM GENERATOR. NORMAL REACTOR COOLANT AND REACTOR POWER CONTROL SYSTEMS WILL RESPOND TO THE CHANGES IN SYSTEM TEMPERATURE.

THE STEAM GENERATOR WILL CONTINUE TO BLOWDOWN UNTIL A FEEDWATER ISOLATION OCCURS. FOLLOWING THE FEEDWATER ISOLATION, THE STEAM GENERATOR LEVEL MAY BE INCREASED USING AUXILIARY FEEDWATER.

MALFUNCTION SEVERITY MAY ONLY BE INCREASED FOR THIS MALFUNCTION AND THE SIMULATOR MUST BE RESET TO REMOVE THIS MALFUNCTION.

EVENTS: 1) DVR 06-02-88-115

DEVIATION INVESTIGATION REPORT (DIR)

FW 2.0

Form Rev 2.0

Facility Name	PAGE
Byron Nuclear Power Station	1 OF 0 3
Title: 2B Main Feedwater Pump Recirculation Line Leak	

EVENT DATE MONTH	DAY	YEAR	DIR NUMBER STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE MONTH	DAY	YEAR	OPERATING MODE	POWER LEVEL
1 1	2 4	8 8	0 6	0 2	8 8	1 1 5	0 0				1	0 4 8

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER AREA CODE	
D. Brindle, Operating Engineer Ext. 2218	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
X YES (if yes, complete EXPECTED SUBMISSION DATE)	NO		0 4	1 5	8 9

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 11-24-88 / 2000

Unit 2 MODE 1 - Power Operation Rx Power 48 RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On 11-24-88 at 2000 hours Unit 2 was at 48 percent power when shift personnel received a call concerning a leak on 401 elevation of the turbine building. Subsequent investigation revealed the source of the leak to be a hole in the 2B Main Feedwater (FW) [SJ] Pump recirculation line at the elbow downstream of valve 2FW012B. The pump was taken out-of-service. Only one pump is required at 48 percent power, so a power reduction was not required.

All operator actions were correct. No other system or components were inoperable at the beginning of this event that contributed to the event. No safety system actuations occurred.

DEVIATION INVESTIGATION REPORT TEXT CONTINUATION

FACILITY NAME	DIR NUMBER							PAGE	Form Rev 2.0
	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER		
Byron Nuclear Power Station	0 6	0 2	8 8	—	1 1 1 5	—	0 0	2 OF 0 3	

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

C. CAUSE OF EVENT:

The cause of the event was erosion of the pipe elbow due to leakage through the upstream 2B Feedwater Pump recirculation valve 2FW012B. The problem of elbow erosion downstream of the Main Feedwater recirculation valve had also been seen on Unit 1. The problem with Unit 1 was found to be the mis-application of a flow directional cone on the downstream side of the valve plug. The nose cone directed any valve leakby directly into the elbow immediately downstream. The nose cones on all of the 2FW012 valves were removed prior to initial startup in response to the Unit 1 problem. It is possible that the erosion damage on 2FW012B was done prior to the cone removal, but it seems more likely that the valve may have been excessively leaking by the seat. Unit 1 elbows are being monitored for pipe thinning and the indications are showing no thinning in the general area of the elbow. Unit 2 elbows downstream of 2FW012A and 2FW012C were tested and the results showed that there has been no thinning. The indications showed an average elbow wall thickness of 0.82 inches while a new elbow has a nominal wall thickness of 0.75 inches.

Further detailed investigations will be conducted during the upcoming refueling outage. The final determinations of the causes will be reported in a supplement to this report.

D. SAFETY ANALYSIS:

The plant or public safety was not compromised by the failure of the elbow downstream of the 2FW012B. The leak found on the pipe elbow was from a small hole in one location. The leak was on the condenser vacuum side of 1FW012B, therefore most of the water/steam was drawn into the main condenser steam space. The recirculation line on the main feedwater pump can be easily isolated. The operation of the pump was not immediately affected by the leak. The pump was taken out of service in order to repair the elbow.

E. CORRECTIVE ACTIONS:

The pump was taken out of service and the manual isolation valve was closed. The remaining feedwater pump recirculation elbows were examined and showed no erosion on the elbow. The damaged pipe elbow will be replaced in the next refueling outage under Nuclear Work Request B62633. The valve internals on 2FW012B will be inspected to verify proper valve closure under Nuclear Work Request B62836. Both Nuclear Work Requests will be tracked under Action Item Record 88-0294. Pipe wall thickness is being monitored under the station's erosion/corrosion program. Relocation of the pipe elbows to reduce erosion potential was also considered but determined to not be a cost effective measure. Additional actions will be reported in a supplement to this report.

F. RECURRING EVENTS SEARCH AND ANALYSIS:

a) EVENT SEARCH (DIR, LER)

There was one previous occurrence on Byron Unit 1 in early 1986.

b) INDUSTRY SEARCH (OPEX's NPRDS)

Problem known; vendor contacted and substantiated problem.

SOER 87-03 and 82-11

DEVIATION INVESTIGATION REPORT TEXT CONTINUATION

FACILITY NAME	DIR NUMBER							PAGE	Form Rev 2.0
	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER		
Byron Nuclear Power Station	0 6	0 2	8 8	—	1 1 5	—	0 0	3 OF 0 3	

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

c) NWR

Elbows:

B27560	1C	1FW079AC 1C Recirculation Line Elbow (4-2-86)
B27668	1B	1FW079AB 1B FW PR Recirc Line Elbow (4-11-86)
B62559	2B	2FW079AB 2B FW PR Recirc Line Elbow (11-24-88)

Recirc Valves:

B27552	1FW12A verify proper closure
B27554	1FW12B verify proper closure

d) ANALYSIS

There has been several instances of pipe erosion problems in the industry.

This event, while significant in that sense, appears to have minimal operating or safety significance itself. Corrective actions appear adequate.

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Control Components Inc.	dragvalve	922101036	

H. OTHER RELATED DOCUMENTS:

NRC IEB 87-11

I. EFFECTIVENESS REVIEW:

Scheduled 03-01-90

J. ADDITIONAL DATA:

a) Affected Technical Specification: None

b) Procedures: None

c) Cause Code: XPRMM2MM

d) Equipment Involved: FW Pump Recirculation Line

e) Other: Valve leakby, Pipe erosion

FW21 S/G TEMPERING LINE RUPTURE

TYPE: GENERIC, RV 0-200 GPM @ 900 PSID

A) 1A TEMPERING LINE
B) 1B TEMPERING LINE
C) 1C TEMPERING LINE
D) 1D TEMPERING LINE

CAUSE: PIPING BREAK IMMEDIATELY DOWNSTREAM 1FW034

REF: M-36 SHEET 1A
M-36 SHEET 1B
M-36 SIIEET 1C
M-36 SHEET 1D

PLT STA: 100% REACTOR POWER

EFFECTS: TEMPERING LINE FLOW WILL DECREASE TO THE AFFECTED STEAM GENERATOR FROM THE SELECTED TEMPERING LINE. THE TEMPERING FLOW CONTROLLER WILL MODULATE 1FW034 OPEN IN AN ATTEMPT TO MAINTAIN NORMAL TEMPERING LINE FLOW TO THE STEAM GENERATOR.

IF THE MALFUNCTION SEVERITY RESULTS IN A DECREASE IN TEMPERING LINE FLOW TO THE STEAM GENERATOR, THE ASSOCIATED AUXILIARY NOZZLE TEMPERATURE MAY INCREASE.

THE OPERATOR MAY MITIGATE THE EFFECTS BY CLOSING THE ASSOCIATED 1FW034 AND 1FW035 VALVES.

MALFUNCTION REMOVAL WILL RESTORE THE FEEDWATER TEMPERING LINE PIPING INTEGRITY.

EVENTS: NONE

FW22 CONDENSATE PUMP FAILS TO START/TRIP

TYPE: GENERIC, RB

A)	1A CD/CB PUMP	1CD05PA & 1CB01PA
B)	1B CD/CB PUMP	1CD05PB & 1CB01PB
C)	1C CD/CB PUMP	1CD05PC & 1CB01PC
D)	1D CD/CB PUMP	1CD05PD & 1CB01PD

CAUSE: FAULTY OVERCURRENT (550/551) RELAY ACTUATION

REF:

20E-1-4030 CB06	20E-1-4030 CB13
20E-1-4030 CB14	20E-1-4030 CD01
20E-1-4030 CD02	20E-1-4030 CD03
20E-1-4030 CD04	20E-1-4030 CD08
20E-1-4030 CD13	20E-1-4030 CD14

PLT STA: SELECTED CONDENSATE PUMP IN OPERATION

EFFECTS: THE SELECTED CONDENSATE/CONDENSATE BOOSTER PUMP BREAKER WILL OPEN. MOTOR CURRENT INDICATION DECREASES TO ZERO AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. ANNUNCIATOR 17-A9 "CD/CB PUMP TRIP" ACTUATES AND ANNUNCIATOR 17-A7 "HTR 11 LEVEL HI-2" ACTUATES. THE 11C HEATER STRING WILL ISOLATE WITH THE HEATER STRING BYPASS VALVE OPENING.

THE SELECTED BOOSTER PUMP FLOW INDICATION DECREASES TO ZERO. IF THE LOW FEEDWATER PUMP NPSH PRESSURE SETPOINT IS REACHED, THE STANDBY CONDENSATE/CONDENSATE BOOSTER PUMP WILL AUTOMATICALLY START TO RESTORE FEEDWATER PUMP SUCTION PRESSURE.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND TRIP LIGHT BY PLACING THE CONTROL SWITCH TO AFTER-TRIP. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL CLOSE THEN IMMEDIATELY REOPEN.

MALFUNCTION REMOVAL WILL RESTORE THE CONDENSATE PUMP OVERCURRENT RELAY TO NORMAL.

EVENTS: 1) DVR 06-01-89-155

DEVIATION INVESTIGATION REPORT (DIR)

Form Rev 2.0

Facility Name	PAGE
Byron Nuclear Power Station	1 OF 0 4

Title
CONDENSATE/CONDENSATE BOOSTER PUMP MOTOR FIRE DUE TO LOOSE BOLTS ON FAN

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	1
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	9 4
1 2	1 2	8 9	0 6	0 1	8 9	— 1 5 5 —	0 1	0 8	0 6	9 0		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
Kevin Passmore, Asst. Tech Staff Supervisor Ext. 2380	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	S D	M O	W 1 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 12/12/89/ 1142

Unit 1 MODE 1 - Power Operations Rx Power 94% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

At 1142 on 12/12/89, the 1B Condensate (CD)/Condensate Booster (CB) [SD] Pump was being started after a return to service following routine 6 month preventive maintenance to vacuum the motor casing vents. As the pump was started, a fire occurred in the pump motor and the pump tripped. A ground overcurrent flag was up at the pump breaker. An equipment attendant at the pump notified the control room, and the fire brigade was dispatched to the condensate pit. The NSO printed out the standard equipment out of service for the pump and the power supply for the motor was racked out.

The fire brigade was attempting to extinguish the fire when the fire chief saw sparks coming off the discharge of the carbon dioxide extinguisher. To aid in extinguishing the fire, parts of the shroud were removed. The fire chief consulted the NSO, concerned that not all equipment had been de-energized. The NSO compared the standard out of service printout with an electrical distribution manual and noticed a power feed to a lube oil pump pressure switch. The equipment identification for the booster pump's lube oil pump is 1CB01PB-A whereas the condensate pump's lube oil pump is identified as 1CD05PB-B. The NSO accidentally misinterpreted 1CD05PA-B (the lube oil pump for the 1A Condensate Pump) as the lube oil pump for the 1B Condensate Pump. Due to the urgency of the situation, the NSO misinterpreted the "-B" as being associated with the 1B Condensate Pump. He wrote the power feed breaker for that lube oil pump on the standard out of service printout and dispatched the Equipment Attendant to de-energize the breaker. No review of the revised OOS was performed by a Senior Reactor Operator (SRO). All of the SRO personnel were busy making the necessary notifications. A proper review would have shown the pressure switch power feed was not a factor creating the sparks.

DEVIATION INVESTIGATION REPORT TEXT CONTINUATION

FACILITY NAME	DIR NUMBER							PAGE			Form Rev 2.0
	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER				
Byron Nuclear Power Station	0 6	0 1	8 9	—	1 5 5	—	0 1	2	OF	0 4	

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

B. DESCRIPTION OF EVENT: (Continued)

The breaker was de-energized and shortly thereafter, the 1A Condensate/Condensate Booster Pumps tripped on low oil pressure. With Unit 1 operating at 94% power, 3 condensate pump trains are needed. When the 1A pumps tripped, only 2 trains were running. The NSO quickly alerted the Equipment Attendant to energize the breaker and the 1A pumps were restarted. Although there were no consequences, this was a near-miss condition.

A GSEP unusual event was declared at 1158 due to a fire lasting longer than 10 minutes. At 1235 hours the fire was extinguished, and the "Unusual Event" was terminated at 1241 hours. There were no systems or components inoperable at the start of this event that contributed to this event. Normal plant operations continued during this event.

C. CAUSE OF EVENT:

The motor was initially inspected while still in place on the pump pedestal. The upper half end bell on the burned connection end was removed to permit examination of the burned areas of the winding. Additionally a side cover and the opposite end upper half end bell were removed to enable further examination of the motor.

On the burned end of the motor a nick in the coil insulation was noticed at the 8 o'clock position. The nick, which was completely through the insulation to the copper winding was located directly under the fan ring. The fan ring was found to be loose, with five of the six bolts missing. The fan is designed to be held in place by six 3/8" diameter x 1-1/4" long bolts. Four bolts were found in the bottom of the motor. One bolt could not be found.

A lower side cover was removed so that the core could be inspected. Oil drips could be seen on the bottom of the core nearest the end where the burning occurred. The burned end of the motor apparently had a heavy coating of oil, possibly caused by excessive bearing inboard seal clearance.

The motor apparently caught on fire when one of the missing fan ring bolts became lodged between the coils and the fan ring. Upon starting the motor, the bolt was driven into the coil and caused a flashover to ground through the motor rotor. The fire was a result of the flashover occurring in the presence of the oil soaked end turns.

The fan ring bolts were secured in place by tack welding them to the fan ring. The fan ring is stainless steel and the bolts are everdur copper.

The sparks from the extinguisher were static sparks caused by the carbon dioxide discharge through a defective extinguisher discharge hose.

The confusion in equipment identification numbers is considered isolated. The Operator involved in the event did know the "-B" was the designator for lube oil pumps, but a quick inspection of the electrical prints placed the B in the pump designator space. The equipment identification program underwent extensive human factors review prior to operation, and is considered acceptable.

D. SAFETY ANALYSIS:

There was no effect on the safety of the plant or the public as a result of this event. The fire was contained to the motor housing and was unable to spread outside this area. Three of four pumps are required to maintain 100% power operation and were in operation at the time of the fire. Under a more severe set of initial conditions such as one CD/CB pump out of service, there would still be no impact on the safety of the plant or public since this would have caused a reduction in condensate flow, a reduction in feedwater flow and possible reactor trip due to low-low steam generator water level. This would be bounded by a loss of normal feedwater event and would not interfere with the safe shutdown of the plant.

E. CORRECTIVE ACTIONS:

The motor was sent to Koontz-Wagner service shop (job #27989) for repairs and rewind. While at the shop, further inspection of the rotor revealed cracking of the welds on the fan ring bolts on the other end of the rotor. The cracked welds were ground off and the fan rings and bolts were reinstalled with pantleg washers underneath the bolts. These carbon steel pantleg washers were welded to the stainless steel rings using ER309 rod. The fiberglass banding on each end of the rotor was removed and the rotor bars and shorting rings were visually checked and growler checked for cracks. Nothing unusual was found. The stator was rewound and VPI treated. The rotor was rebanded, balanced and the motor was reassembled, test run and returned to Byron Station.

Byron Station set the motor down hard during the course of reinstalling it and when the motor was mounted in place and run, high axial vibration was measured. The motor was removed and sent back to Koontz-Wagner for inspection. Koontz-Wagner rebalanced the rotor, adding approximately 130 grams of weight to the rotor. The bearing to housing clearance on both ends was found to be .007" - .008" loose. The shop built up the bearings with Belzona to reduce the clearance to .001" - .002" loose. The motor base required shimming opposite corners .025" to level the motor. The motor was test run at rated voltage on 3/07/90. Vibration and temperature data were acceptable and the motor was shipped back to Byron Station.

Also at this time, the 1D Condensate-Condensate Boost Pump Motor was removed and sent to National Electric Coil's Joilet shop (job #9502) for clean-up and inspection. During the course of the inspection, the bolts securing the fan ring were discovered to have cracked tack welds. The welds were ground off and new bolts were installed over pantleg washers following the same method used on motor 1B.

Having inspected two motors and finding both to have broken tack welds, the fan ring bolts on the remaining Unit 1 and Unit 2 CD/CB pumps were inspected under NWR #B99621 (Release #'s 042, 043, 044, 045, 037 and 038) per S.O.A.D recommendation. All were found to be in good condition. Also, all fan-ring bolt assemblies will be inspected annually during the air intake inspection which is tracked on the General Surveillance Program (GSRV). If any tack welds are found to be suspect, pantleg washers will be used to secure the bolts.

F. RECURRING EVENTS SEARCH AND ANALYSIS:

a) EVENT SEARCH (DIR, LER)
None.

b) INDUSTRY SEARCH (OPEX's NPRDS)
Several motor fires were found. Two events were related to the condensate system. However, the root causes were due to electrical failures.

c) NWR
None.

d) ANALYSIS
No adverse trend indicated.

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Westinghouse	Motor	8011	- - -

H. OTHER RELATED DOCUMENTS:

None.

I. EFFECTIVENESS REVIEW:

Not required.

J. ADDITIONAL DATA:

a) Affected Technical Specification: None

b) Procedures: None

c) Cause Code: XPMM4

d) Equipment Involved: Condensate/Condensate Booster Pump Motor

e) Other: None

FW23 FW HEATER BYPASS VALVE FAILURE (1CB025)

TYPE: DISCRETE, RB

CAUSE: FAULTY LS-HD293X CONTACT (FAILS OPEN)

REF: 20E-1-4030 CB06
M-40 SHEET 1
M-40 SHEET 2

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: LOW PRESSURE HEATERS 11-14 BYPASS VALVE 1CB025 FAILS OPEN. THE VALVE WILL NOT RESPOND TO OPERATION OF IT'S CONTROL SWITCH.

CONDENSATE BOOSTER PUMP DISCHARGE HEADER PRESSURE DECREASES AND FEEDWATER PUMP SUCTION HEADER PRESSURE INCREASES. THE DECREASED FLOW THROUGH THE LOW PRESSURE HEATER STRINGS WILL RESULT IN A DECREASING TEMPERATURE AT THE FEEDWATER PUMP SUCTION.

MALFUNCTION REMOVAL WILL RESTORE THE LOW PRESSURE HEATERS 11-14 BYPASS VALVE 1CB025 LS-HD293X CONTACT TO NORMAL.

EVENTS: 1) DVR 06-01-85-284

DEVIATION INVESTIGATION REPORT

FW23

TITLE

FAILURE OF 1CB025 TO CLOSE

PAGE 1 OF 0 1

EVENT DATE MONTH	DAY	YEAR	DIR NUMBER STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE MONTH	DAY	YEAR	OPERATING MODE	POWER LEVEL
0 9	0 7	8 5	0 6	0 1	8 5	- 2 8 4 -	0 0	1 0	2 2	8 5	1	0 5 0

CONTACT FOR THIS DIR

NAME		TELEPHONE NUMBER AREA CODE	
Leo Wehner	Ext. 2384	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	S D	/ / / V	M 1 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

WHAT HAPPENED?

While operating at 50% power on 9-7-85, the Control Room operators shutdown the 1B feedwater pump. Opening of the FW pump recirculation valve caused a high level in the #11C FW heater. This actuated the heater bypass valv 1CB025 open. After the high heater level condition was reset, the bypass valve would not reclose.

WHAT WAS THE ROOT CAUSE?

The root cause for this deviation is indeterminate. The valve has been stroked several times since the deviatio occurred. No abnormalities were found. The Electrical Maintenance department checked the solenoid valve which controls the position of the bypass valve. The solenoid and control circuit were found to be operating as designed.

HOW DID IT AFFECT PLANT AND/OR PUBLIC SAFETY?

This deviation did not affect plant or public safety. Turbine load was decreased manually because of the lowering of Tavg. The plant is designed to mitigate the consequences of excessive heat removal by the secondary plant. Continued low feedwater temperatures would eventually cause a reactor trip on High Neutron Flux Power Range, overpower ΔT, or Overtemperature ΔT.

HAS IT HAPPENED BEFORE?

This deviation has not happened before.

WHAT WAS DONE TO CORRECT THE CONDITION AND HOW ARE WE GOING TO PREVENT RECURRENCE?

After the high level alarm was cleared, the Control Room operator could not close the heater bypass valve. Operators were dispatched to the valve and failed the instrument air to the valve actuator. The valve then stroked closed. AIR 6-85-372 has been generated to further investigate into the operation of the electrical solenoid valve.

(0638M)

FW24 FAILURE OF AF SUCTION PRESS TRANSMITTER

TYPE: GENERIC, RV 0-40 PSIA

A)	1A AF PUMP	1AF01PA
B)	1B AF PUMP	1AF01PB

CAUSE: TRANSMITTER PT-AF051/55 FAILURE

REF: M-2037 SHT 2
20E-1-4031 AF13 & AF14

PLT STA: AF PUMP IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED AF PUMP SUCTION TRANSMITTER TO FAIL AT THE DESIRED SEVERITY. IF FAILED LOW, ANNUNCIATOR 3-A7 "AF PUMP SUCT PRESS LOW" ACTUATES. ANNUNCIATOR 3-E7 "AF PUMP SX SUCT VLVS ARMED" ACTUATES (FOR 1A AF PP ONLY IF PUMP IS RUNNING) AND WILL ALLOW THE SX SUCTION VALVE TO THE AF PUMPS TO OPEN COINCIDENT WITH AN AUTO START SIGNAL. ANNUNCIATOR 3-A6 "AF PUMP TRIP" ACTUATES WHEN THE PUMP TRIPS ON LOW SUCTION PRESSURE.

MALFUNCTION REMOVAL RESTORES THE PRESSURE TRANSMITTER TO NORMAL OPERATION.

EVENTS: NONE

FW25 GLAND STEAM CONDENSER MALFUNCTION

TYPE: GENERIC, RB

A) 1CD157A
B) 1CD157B

CAUSE: BLOCKED AIR SUPPLY LINE (FAILS OPEN)

REF: 20E-1-4030 CD11
M-39 SHEET 3

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: THE SELECTED GLAND STEAM CONDENSER BYPASS VALVE, 1CD157A(B), WILL FAIL OPEN. CONDENSATE PUMP DISCHARGE HEADER PRESSURE WILL DECREASE SLIGHTLY AND CONDENSATE BOOSTER PUMP HEADER PRESSURE WILL INCREASE SLIGHTLY. THE ASSOCIATED GLAND STEAM CONDENSER FLOW WILL DECREASE. ANNUNCIATOR 17-D1 "GLND STM CNDSR FLOW HIGH LOW" MAY ACTUATE WHEN FLOW DECREASES.

MALFUNCTION REMOVAL WILL RESTORE THE GLAND STEAM CONDENSER BYPASS VALVE, 1CD157A(B), AIR SUPPLY LINE TO NORMAL.

EVENTS: NONE

FW26 MAIN FW REG VALVE SEAT LEAKAGE

TYPE: GENERIC, RV 0-2000 GPM AT 250 PSID

A)	1A FRV	1FW510
B)	1B FRV	1FW520
C)	1C FRV	1FW530
D)	1D FRV	1FW540

CAUSE: WORN VALVE SEAT

REF: M-36 SHEET 1A
M-36 SHEET 1B
M-36 SHEET 1C
M-36 SHEET 1D

PLT STA: REACTOR AT LOW POWER (S/U)

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED MAIN FEED REG VALVE WILL LEAK BY. THE RATE OF LEAKAGE WILL BE DETERMINED BY THE SELECTED SEVERITY. THE INCREASE IN FEEDWATER FLOW TO THE ASSOCIATED STEAM GENERATOR WILL CAUSE LEVEL TO INCREASE. ANNUNCIATOR 15-A9,B9,C9 AND D9 "STEAM GENERATOR A/B/C/D LEVEL DEVIATION HIGH LOW" WILL ACTUATE AT 5% LEVEL DEVIATION FROM PROGRAM LEVEL. THE AFFECTED FEED REG VALVE, IF IN AUTO, WILL CLOSE DOWN IN AN ATTEMPT TO RETURN STEAM GENERATOR LEVEL TO THE PROGRAM LEVEL.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED FEED REG VALVE TO NORMAL.

EVENTS: NONE

FW27 FW HEATER TUBE LEAK (11 DC)

TYPE: GENERIC, RV 0-3 MLBM/HR @ 800 PSID

A) 11A DRAIN COOLER 1CB01AA
B) 11B DRAIN COOLER 1CB01AB
C) 11C DRAIN COOLER 1CB01AC

CAUSE: TUBE FAILURE AT FW INLET TO HEATER

REF: 20E-1-4030 HD20
20E-1-4030 HD28
20E-1-4030 HD29
M-40 SHEET 2
M-41 SHEET 4
M-41 SHEET 5

PLT STA: 100% REACTOR POWER

EFFECTS: THIS MALFUNCTION RESULTS IN THE LEAKAGE OF CONDENSATE FROM THE MAIN CONDENSATE HEADER INTO THE SELECTED DRAIN COOLER. AT LOW SEVERITY LEVELS, THE NORMAL FLASH TANK DRAIN OUTLET VALVE 1HD029A/B/C MODULATES TO MAINTAIN NORMAL LEVEL. AS MALFUNCTION SEVERITY IS INCREASED, FLASH TANK EMERGENCY DRAIN VALVE 1HD094A/B/C WILL OPEN IN AN ATTEMPT TO MAINTAIN FLASH TANK LVL AND ANNUNCIATOR 17-C8 "FLASH TANK EMERGENCY DRAIN VLV OPEN" AND 17-B8 "FLASH TANK LEVEL HIGH LOW" WILL ACTUATE.

THE CONDENSATE SYSTEM WILL REFLECT THE LEAKAGE AS A DECREASE IN CONDENSATE BOOSTER PUMP DISCHARGE PRESSURE, A DECREASE IN THE FEEDWATER PUMP SUCTION PRESSURE, AND INCREASED BOOSTER PUMP DISCHARGE FLOW.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE INTEGRITY OF THE LP HEATER TUBES.

EVENTS: NONE

FW28 FW HEATER TUBE LEAK (11)

TYPE: GENERIC, RV 0-3 MLBM/HR @ 800 PSID

A) 11A LP HEATER 1CB02AA
B) 11B LP HEATER 1CB02AB
C) 11C LP HEATER 1CB02AC

CAUSE: TUBE FAILURE AT FW INLET TO HEATER

REF: 20E-1-4030 HD20
20E-1-4030 HD28
20E-1-4030 HD29
M-40 SHEET 2
M-41 SHEET 4
M-41 SHEET 5

PLT STA: 100% REACTOR POWER

EFFECTS: THIS MALFUNCTION RESULTS IN THE LEAKAGE OF CONDENSATE FROM THE MAIN CONDENSATE HEADER INTO THE SELECTED LOW PRESSURE HEATER, THEN DIRECTLY INTO THE ASSOCIATED FLASH TANK. AT LOW SEVERITY LEVELS, THE NORMAL FLASH TANK DRAIN OUTLET VALVE 1HD029A/B/C MODULATES TO MAINTAIN NORMAL LEVEL. AS MALFUNCTION SEVERITY IS INCREASED, ANNUNCIATOR 17-B7 "HTR 11 LEVEL HIGH" WILL ACTUATE. FLASH TANK EMERGENCY DRAIN VALVE 1HD094A/B/C WILL OPEN IN AN ATTEMPT TO MAINTAIN FLASH TANK LVL AND ANNUNCIATOR 17-C8 "FLASH TANK EMERGENCY DRAIN VLV OPEN" WILL ACTUATE.

IF HEATER LEVEL INCREASES FURTHER, ANNUNCIATOR 17-A7 "HTR 11 LEVEL HI-2" WILL ACTUATE, LP HTRS 11/12/13/14 BYP VALVE 1CB025 OPENS, THE ASSOCIATED LP HTRS 11/12/13/14 ISOL VALVES 1CB003A/B/C/ AND 1CB029A/B/C CLOSE, AND THE FLASH TANK 1A/B/C AND DRAIN COOLER 11A/B/C NORM LVL CONTROL VALVE 1HD029A/B/C WILL OPEN.

THE CONDENSATE SYSTEM WILL REFLECT THE LEAKAGE AS A DECREASE IN CONDENSATE BOOSTER PUMP DISCHARGE PRESSURE, A DECREASE IN THE FEEDWATER PUMP SUCTION PRESSURE, AND INCREASED BOOSTER PUMP DISCHARGE FLOW.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE INTEGRITY OF THE LP HEATERS TUBES.

EVENTS: NONE

FW29 FW HEATER TUBE LEAK (12)

TYPE: GENERIC, RV 0-3 MLBM/HR @ 800 PSID

A) 12A LP HEATER 1CB03AA
B) 12B LP HEATER 1CB03AB
C) 12C LP HEATER 1CB03AC

CAUSE: TUBE FAILURE AT FW INLET TO HEATER

REF: 20E-1-4030 HD20
20E-1-4030 HD28
20E-1-4030 HD29
M-40 SHEET 2
M-41 SHEET 4
M-41 SHEET 5

PLT STA: 100% REACTOR POWER

EFFECTS: THIS MALFUNCTION RESULTS IN THE LEAKAGE OF CONDENSATE FROM THE MAIN CONDENSATE HEADER INTO THE SELECTED LOW PRESSURE HEATER. AT LOW SEVERITY LEVELS, THE NORMAL HEATER DRAIN OUTLET VALVE 1HD026A/B/C MODULATES TO MAINTAIN NORMAL LEVEL. AS MALFUNCTION SEVERITY IS INCREASED, EMERGENCY DRAIN OUTLET VALVE 1HD054A/B/C WILL OPEN IN AN ATTEMPT TO MAINTAIN HEATER LEVEL AND ANNUNCIATOR 17-C6 "HTR 12 EMERGENCY DRAIN VLV OPEN" WILL ACTUATE. ANNUNCIATOR 17-B6 "HTR 12 LEVEL HIGH LOW" WILL ALSO ACTUATE. THE ASSOCIATED FLASH TANK 1A/B/C LEVEL INCREASES.

IF HEATER LEVEL INCREASES FURTHER, ANNUNCIATOR 17-A6 "HTR 12 LEVEL HI-2" WILL ACTUATE, THE HP HTR 12A/B/C NORM LVL CONTROL VALVES 1HD026A/B/C AND EMERGENCY DRAIN VALVES 1HD054A/B/C WILL OPEN, LP HEATER 13A/B/C NORM LVL CONTROL VALVE 1HD023A/B/C WILL CLOSE, LP HEATERS 12A/12B EXTRACTION ISOLATION VALVE 1ES010A/B/C WILL CLOSE, LP HEATER 12A/12B EXTRACTION CHECK VALVE 1ES011A/B/C WILL CLOSE, AND LP HEATERS 12A/12B EXTRACTION STEAM SPILL VALVE 1ES028A/B/C WILL OPEN.

THE CONDENSATE SYSTEM WILL REFLECT THE LEAKAGE AS A DECREASE IN CONDENSATE BOOSTER PUMP DISCHARGE PRESSURE, A DECREASE IN THE FEEDWATER PUMP SUCTION PRESSURE, AND INCREASED BOOSTER PUMP DISCHARGE FLOW.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE INTEGRITY OF THE LP HEATERS TUBES.

EVENTS: NONE

FW30 FW HEATER TUBE LEAK (13)

TYPE: GENERIC, RV 0-3 MLBM/HR @ 800 PSID

A) 13A LP HEATER 1CB04AA
B) 13B LP HEATER 1CB04AB
C) 13C LP HEATER 1CB04AC

CAUSE: TUBE FAILURE AT FW INLET TO HEATER

REF: 20E-1-4030 HD20
20E-1-4030 HD28 & HD29
M-40 SHEET 2
M-41 SHEET 4 &5

PLT STA: 100% REACTOR POWER

EFFECTS: THIS MALFUNCTION RESULTS IN THE LEAKAGE OF CONDENSATE FROM THE MAIN CONDENSATE HEADER INTO THE SELECTED LOW PRESSURE HEATER. AT LOW SEVERITY LEVELS, THE NORMAL HEATER DRAIN OUTLET VALVE 1HD023A/B/C MODULATES TO MAINTAIN NORMAL LEVEL. AS MALFUNCTION SEVERITY IS INCREASED, EMERGENCY DRAIN OUTLET VALVE 1HD051A/B/C WILL OPEN IN AN ATTEMPT TO MAINTAIN HEATER LEVEL AND ANNUNCIATOR 17-C5 "HTR 13 EMERGENCY DRAIN VLV OPEN" WILL ACTUATE. ANNUNCIATOR 17-B5 "HTR 13 LEVEL HIGH LOW" WILL ALSO ACTUATE.

IF HEATER LEVEL INCREASES FURTHER, ANNUNCIATOR 17-A5 "HTR 13 LEVEL HI-2" WILL ACTUATE, THE HP HTR 13A/B/C NORM LVL CONTROL VALVES 1HD023A/B/C AND EMERGENCY DRAIN VALVES 1HD051A/B/C WILL OPEN, LP HEATERS 14A/14B NORM LVL CONTROL VALVES 1 HD020A/B/C WILL CLOSE, LP HEATER 13A/13B EXTRACTION ISOLATION VALVE 1ES013A/B/C WILL CLOSE, LP HEATER 13A/13B EXTRACTION CHECK VALVE 1ES015A/B/C WILL CLOSE, AND LP HEATERS 13A/13B EXTRACTION STEAM SPILL VALVE 1ES030A/B/C WILL OPEN.

THE CONDENSATE SYSTEM WILL REFLECT THE LEAKAGE AS A DECREASE IN CONDENSATE BOOSTER PUMP DISCHARGE PRESSURE, A DECREASE IN THE FEEDWATER PUMP SUCTION PRESSURE, AND INCREASED BOOSTER PUMP DISCHARGE FLOW.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE INTEGRITY OF THE LP HEATERS TUBES.

EVENTS: NONE

FW31 FW HEATER TUBE LEAK (14)

TYPE: GENERIC, RV 0-3 MLBM/HR @ 700 PSID

A)	14A LP HEATER	1CB05AA
B)	14B LP HEATER	1CB05AB
C)	14C LP HEATER	1CB05AC

CAUSE: TUBE FAILURE AT FW INLET TO HEATER

REF: 20E-1-4030 HD20
20E-1-4030 HD28
20E-1-4030 HD29
M-40 SHEET 2
M-41 SHEET 4
M-41 SHEET 5

PLT STA: 100% REACTOR POWER

EFFECTS: THIS MALFUNCTION RESULTS IN THE LEAKAGE OF CONDENSATE FROM THE MAIN CONDENSATE HEADER INTO THE SELECTED LOW PRESSURE HEATER. AT LOW SEVERITY LEVELS, THE NORMAL HEATER DRAIN OUTLET VALVE 1HD020A/B/C MODULATES TO MAINTAIN NORMAL LEVEL. AS MALFUNCTION SEVERITY IS INCREASED, EMERGENCY DRAIN OUTLET VALVE 1HD048A/B/C WILL OPEN IN AN ATTEMPT TO MAINTAIN HEATER LEVEL AND ANNUNCIATOR 17-C4 "HTR 14 EMERGENCY DRAIN VLV OPEN" WILL ACTUATE. ANNUNCIATOR 17-B4 "HTR 14 LEVEL HIGH LOW" WILL ALSO ACTUATE.

IF HEATER LEVEL INCREASES FURTHER, ANNUNCIATOR 17-A4 "HTR 14 LEVEL HI-2" WILL ACTUATE, THE HP HTR 14A/B/C NORM LVL CONTROL VALVES 1HD020A/B/C AND EMERGENCY DRAIN VALVES 1HD048A/B/C WILL OPEN, LP HEATERS 14A/B/C EXTRACTION ISOLATION VALVE 1ES016A/B/C WILL CLOSE, LP HEATER 14A/B/C EXTRACTION CHECK VALVE 1ES017A/B/C WILL CLOSE, AND LP HEATERS 14A/B/C EXTRACTION STEAM SPILL VALVE 1ES032A/B/C WILL OPEN.

THE CONDENSATE SYSTEM WILL REFLECT THE LEAKAGE AS A DECREASE IN CONDENSATE BOOSTER PUMP DISCHARGE PRESSURE, A DECREASE IN THE FEEDWATER PUMP SUCTION PRESSURE, AND INCREASED BOOSTER PUMP DISCHARGE FLOW.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE INTEGRITY OF THE LP HEATERS TUBES.

EVENTS: NONE

FW32 FW HEATER TUBE LEAK (15 DC)

TYPE: GENERIC, RV 0-5 MLBM/HR @ 600 PSID

A) 15A DRAIN COOLER 1CB06AA
B) 15B DRAIN COOLER 1CB06AB

CAUSE: TUBE FAILURE AT FW INLET TO HEATER

REF: 20E-1-4030 HD19
20E-1-4030 HD24
M-40 SHEET 2
M-41 SHEET 2
M-41 SHEET 3

PLT STA: 100% REACTOR POWER

EFFECTS: THIS MALFUNCTION RESULTS IN THE LEAKAGE OF CONDENSATE FROM THE MAIN CONDENSATE HEADER INTO THE SELECTED DRAIN COOLER. AT LOW SEVERITY LEVELS, THE HEATER DRAIN PUMP DISCH VALVES 1HD046A/B MODULATE TO MAINTAIN LEVEL. AS MALFUNCTION SEVERITY IS INCREASED, THE HEATER DRAIN TANK LVL INCREASES. THE HEATER DRAIN TANK OVERFLOW VALVE 1HD117 STARTS TO OPEN IN AN ATTEMPT TO MAINTAIN LEVEL AND ACTUATES ANNUNCIATOR 17-E5 "HD TANK OVERFLOW VALVE OPEN. ANNUNCIATOR 17-E4 "HD TANK LEVEL HIGH" WILL ACTUATE AS LEVEL INCREASES.

IF HEATER DRAIN TANK LEVEL CONTINUES TO INCREASE, ANNUNCIATOR 17-D4 "HD TANK LEVEL HI-2" WILL ACTUATE, THE MSR SHELL DRAIN TANK 1A/B/C/D OUTLET VALVES 1HD009A/B/C/D WILL CLOSE.

THE CONDENSATE SYSTEM WILL REFLECT THE LEAKAGE AS A DECREASE IN CONDENSATE BOOSTER PUMP DISCHARGE PRESSURE, A DECREASE IN THE FEEDWATER PUMP SUCTION PRESSURE, AND INCREASED BOOSTER PUMP DISCHARGE FLOW.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE INTEGRITY OF THE DRAIN COOLER TUBES.

EVENTS: NONE

FW33 FW HEATER TUBE LEAK (15)

TYPE: GENERIC, RV 0-3 MLBM/HR @ 600 PSID

A) 15A LP HEATER 1CB07AA
B) 15B LP HEATER 1CB07AB

CAUSE: TUBE FAILURE AT COND INLET TO HEATER

REF: 20E-1-4030 HD19 & HD28
M-40 SHEET 2
M-41 SHEET 2

PLT STA: 100% REACTOR POWER

EFFECTS: THIS MALFUNCTION RESULTS IN THE LEAKAGE OF CONDENSATE FROM THE MAIN CONDENSATE HEADER INTO THE SELECTED LOW PRESSURE HEATER. AT LOW SEVERITY LEVELS, THE EMERGENCY DRAIN OUTLET VALVE 1HD062A(B) MODULATES TO MAINTAIN NORMAL LEVEL AND ANNUNCIATOR 17-C3 "HTR 15 EMERGENCY DRAIN VLV OPEN" WILL ACTUATE. AS MALFUNCTION SEVERITY IS INCREASED, EMERGENCY DRAIN OUTLET VALVE 1HD062A(B) WILL CONTINUE TO OPEN IN AN ATTEMPT TO MAINTAIN HEATER LEVEL. ANNUNCIATOR 17-B3 "HTR 15 LEVEL HIGH" WILL ACTUATE AS LEVEL INCREASES.

IF HEATER LEVEL INCREASES FURTHER, ANNUNCIATOR 17-A3 "HTR 15 LEVEL HI-2" WILL ACTUATE, THE EMERGENCY DRAIN OUTLET VALVE 1HD062A(B) OPENS, THE LP HTR 16A(B) NORM LVL CONTROL VALVES 1HD011A(B) TO LP HEATER 15A(B) WILL CLOSE, LP HEATERS 15A/15B EXTRACTION ISOLATION VALVE 1ES007 WILL CLOSE, LP HEATER 15A/15B EXTRACTION CHECK VALVE 1ES008 WILL CLOSE, LP HEATERS 15A/15B EXTRACTION STEAM SPILL VALVE 1ES024 WILL OPEN, AND FIRST STAGE REHEATER DRAIN TANK 1A/C OR B/D TO LP HEATERS 15A/15B ISOLATION VALVES 1HD002A&C OR 1HD002B&D WILL CLOSE.

THE CONDENSATE SYSTEM WILL REFLECT THE LEAKAGE AS A DECREASE IN CONDENSATE BOOSTER PUMP DISCHARGE PRESSURE, A DECREASE IN THE FEEDWATER PUMP SUCTION PRESSURE, AND INCREASED BOOSTER PUMP DISCHARGE FLOW.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE INTEGRITY OF THE LP HEATERS TUBES.

EVENTS: NONE

FW34 FW HEATER TUBE LEAK (16)

TYPE: GENERIC, RV 0-5 MLBM/HR @ 400 PSID

A) 1CB08AA 16A LP HEATER
B) 1CB08AB 16B LP HEATER

CAUSE: TUBE FAILURE AT COND INLET TO HEATER

REF: 20E-1-4030 HD19
20E-1-4030 HD29
M-40 SHEET
M-41 SHEET

PLT STA: 100% REACTOR POWER

EFFECTS: THIS MALFUNCTION RESULTS IN THE LEAKAGE OF CONDENSATE FROM THE MAIN CONDENSATE HEADER INTO THE SELECTED LOW PRESSURE HEATER. AT LOW SEVERITY LEVELS, THE NORMAL HEATER DRAIN OUTLET VALVE 1HD011A(B) MODULATES TO MAINTAIN NORMAL LEVEL. AS MALFUNCTION SEVERITY IS INCREASED, ANNUNCIATOR 17-B2 "HTR 16 LEVEL HIGH LOW" WILL ACTUATE. EMERGENCY DRAIN VALVE 1HD041A(B) WILL OPEN IN AN ATTEMPT TO MAINTAIN HEATER LEVEL AND ANNUNCIATOR 17-C2 "HTR 16 EMERGENCY DRAIN VLV OPEN" WILL ACTUATE.

IF HEATER LEVEL INCREASES FURTHER, ANNUNCIATOR 17-A2 "HTR 16 LEVEL HI-2" WILL ACTUATE, THE HP HTR 17A/B NORM LVL CONTROL VALVES TO LP HEATER 16A(B) 1HD008A(B) WILL CLOSE, LP HEATERS 16A/16B EXTRACTION ISOLATION VALVE 1ES001 WILL CLOSE, LP HEATER 16A/16B EXTRACTION CHECK VALVE 1ES002 WILL CLOSE, AND LP HEATERS 16A/16B EXTRACTION STEAM SPILL VALVE 1ES019 WILL OPEN.

THE CONDENSATE SYSTEM WILL REFLECT THE LEAKAGE AS A DECREASE IN CONDENSATE BOOSTER PUMP DISCHARGE PRESSURE, A DECREASE IN THE FEEDWATER PUMP SUCTION PRESSURE, AND INCREASED BOOSTER PUMP DISCHARGE FLOW.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE INTEGRITY OF THE LP HEATERS TUBES.

EVENTS: NONE

FW35 HEATER DRAIN PUMP FAILS TO START/TRIP

TYPE: GENERIC, RB

A)	1A HD PUMP	1HD01PA
B)	1B HD PUMP	1HD01PB
C)	1C HD PUMP	1HD01PC

CAUSE: FAULTY OVERCURRENT (550/551) RELAY ACTUATION

REF: 20E-1-4030 HD01
20E-1-4030 HD02
20E-1-4030 HD03

PLT STA: 100% REACTOR POWER

EFFECTS: THIS MALFUNCTION RESULTS IN THE TRIPPING OF THE SELECTED HEATER DRAIN PUMP BREAKER. PUMP CURRENT INDICATION DECREASES TO ZERO, ANNUNCIATOR 17-D2 "HD PUMP TRIP" ACTUATES, AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. ASSOCIATED HEATER DRAIN PUMP DISCHARGE FLOW DECREASES TO ZERO. THE ASSOCIATED RECIRC VALVE OPENS.

ANNUNCIATOR 16-E1 "FW PUMP NPSH LOW" MAY ACTUATE AT A VARIABLE FEED PUMP SUCTION PRESSURE. 6.3% BELOW THE SETPOINT THE STANDBY CONDENSATE/CONDENSATE BOOSTER PUMP AUTO STARTS, GLAND CONDENSER BYPASS VALVES 1CD157A/B OPEN, CONDENSATE PUMPS RECIRC VALVE 1CD152 CLOSES, AND THE HD PUMP COMBINED DISCHARGE VALVES 1HD046A/B OPEN TO MAINTAIN FEED PUMP SUCTION PRESSURE.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND TRIP LIGHT BY OPERATING THE CONTROL SWITCH TO TRIP. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL CLOSE THEN IMMEDIATELY TRIP OPEN.

MALFUNCTION REMOVAL WILL RESTORE THE HEATER DRAIN PUMP OVERCURRENT RELAY TO NORMAL.

EVENTS: 1) DVR 20-01-89-078

DEVIATION INVESTIGATION REPORT (DIR)

FW35

Facility Name: Braidwood 1

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Title: 1B Heater Drain Pump Trip Due to Ground Overcurrent Relay Actuation as a Result of Door-induced Vibration

EVENT DATE			DIR NUMBER							REPORT DATE			OPERATING MODE	1
MONTH	DAY	YEAR	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	0 9 6
0 5	2 3	8 9	2 0	0 1	8 9	—	0 7 8	—	0 0	0 6	1 3	8 9		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
R. Dortch, Technical Staff Engineer Ext. 2411	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
_ YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. DESCRIPTION OF EVENT:

Heater Drain Pump 1C was out of service for preventive maintenance during the time of the event. This component contributed to the severity of the event.

On May 23, 1989 at 0800 hours, the Unit 1 Reactor was operating at 96% power with Control Bank D at 211 steps, RCS Temperature at 583 degrees F and Boron Concentration at 95ppm. At 0804 hours, Annunciator Window 1-17-D2 (HD Pump Trip) alarmed and the sequence recorder verified that the 1B Heater Drain Pump tripped. At that point, the unit was ramped down to 700 MWe at 350MWe/Min. At 0807 hours, the unit output was reduced further to 559MWe at 350MWe/Min. At 0808 hours, 200 gallons of Boric Acid was put into the Reactor Coolant System to compensate for the power reduction and the Rod LO-2 alarm. The Ground Overcurrent Relay for the 1B Heater Drain Pump was found actuated and the Relay Target was reset at 0814 hours. At 0818 hours, the unit was then ramped up to 600MWe at a rate of 3MWe/Min for Delta I and QPTR (Quadrant Power Tilt Ratio) concerns. At 0832 hours, a dilution was started, due to a decreasing Tave. At 0843 hours, the Power Range Tilt Alarm annunciated, due to QPTR exceeding its limits. At 0906 hours, (due to QTPR) a ramp down was started to 559MWe at 2MWe/Min. An additional ramp down to 520MWe at .25MWe/Min. was started at 1136 hours, due to decreasing temperature. At 1400 hours, LCOAR (Limiting Condition for Operation Action Requirements) 2.2-1A was entered because the Delta I limit was exceeded. At this time, the unit experienced the "Pop-Up" effect. The Pop-Up effect is when rod motion has essentially no effect on controlling Delta I. The Shift Engineer with the concurrence from the Technical Staff Nuclear Group, decided to let Xenon peak. At 1700 hours, QPTR reached it's peak at a value of 1.05. Delta I peaked at 2200 hours, with a value of 21% and QPTR was within its limit at 2300 hours. Stable plant conditions were established on May 24 at 0400 hours. Delta I was on target and was held there by withdrawing control rods and controlling temperature.

The reason for this DVR is due to an unplanned Technical Specification Action Statement entry.

B. CAUSE OF EVENT:

On May 23, 1989, two Electrical Maintenance personnel were placing label tags on various 4KV and 6.9KV Circuit Breakers. The task involved opening each Breaker Cubicle Door, attaching the plastic identification tag in front of the breaker and then closing the door. While closing the 1B Heater Drain Pump Breaker Cubicle Door, the door scraped against the concrete floor and shook. This vibration caused the Ground Overcurrent Relay, which is mounted on the door, to actuate which in turn tripped the running 1B Heater Drain Pump.

C. CORRECTIVE ACTIONS:

Maintenance personnel will be informed of the Heater Drain incident highlighting the importance of being careful and being aware of their surroundings. Maintenance personnel should know the consequences of their actions, when working with 4KV and 6.9KV breakers. This event will be discussed at a safety meeting for maintenance personnel (EMD). This item will be tracked by action item 456-200-89-07801.

A Nuclear Work Request (A32077), has been generated to sand down the floor and help prevent a reoccurrence of the incident mentioned above in Section B. This item will be tracked by action item 456-200-89-07802.

Operating and the Electrical Maintenance Department will verify the operability of all 6.9KV and 4KV Breaker Cubicle doors and identify those that have the potential of repeating the incident mentioned in Section C. This item will be tracked by action items 456-200-89-07803 and 456-200-89-07804 for Units 1 and 2, respectively.

FW36 LOSS OF CONDENSER VACUUM

TYPE: DISCRETE, RV 0-1000 CFM @ 30" HG

CAUSE: CONDENSER EXPANSION BOOT FAILURE

REF: M-39 SHEET 2
20E-1-4030 TO09

PLT STA: 100% REACTOR POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, CONDENSER VACUUM WILL BE LOST TO THE ATMOSPHERE AT A RATE DETERMINED BY THE SELECTED SEVERITY. 1PI-ES043,46, AND 49 SHOW THE DEGRADING VACUUM CONDITION. ANNUNCIATOR 18-D4 "CNDSR VACUUM LOW" WILL ACTUATE.

CONDENSER TEMPERATURES WILL INCREASE AND MEGAWATT OUTPUT WILL DECREASE IN PROPORTION TO THE LOSS OF MAIN CONDENSER VACUUM (IF THE MW FEEDBACK LOOP IS NOT IN SERVICE). THE OPERATOR MAY TRY TO LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY PLACING ADDITIONAL AIR REMOVAL EQUIPMENT IN OPERATION, HOWEVER THE MAXIMUM SEVERITY OF THIS MALFUNCTION IS IN EXCESS OF THE TOTAL AIR REMOVAL CAPABILITIES.

CONTINUED VACUUM LOSS WILL RESULT IN A MAIN TURBINE AND SUBSEQUENT REACTOR TRIP WITH ANNUNCIATOR 18-E2 "CNDSR VACUUM LOW TURB TRIP" ACTUATING FOLLOWED BY BOTH TURBINE-DRIVEN FW PUMPS TRIPPING. ANNUNCIATOR 11-A9 "TURB TRIP ABOVE P8 RX TRIP" IS ACTUATED. IN ADDITION, LOSS OF STEAM DUMP CAPABILITY TO THE MAIN CONDENSER RESULTS. THE EXCESS DECAY HEAT WILL BE RELEASED TO THE ATMOSPHERE BY THE STEAM GENERATOR POWER OPERATED RELIEF VALVES.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE MAIN CONDENSER EXPANSION BOOT TO NORMAL.

EVENTS: 1) DVR 06-02-87-092
2) DVR 20-01-89-090

FW36

DELTA T RUNBACK DUE TO LOSS OF VACUUM AT 2C FEEDWATER PUMP

EVENT DATE			DIR NUMBER							REPORT DATE			OPERATING MODE	
MONTH	DAY	YEAR	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	MONTH	DAY	YEAR		
0 9	1 5	8 7	0 6	0 2	8 7	—	0 9 2	—	0 0	1 0	3 0	8 7	POWER LEVEL	0 9 4

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
T. Didier, Operating Engineer Ext. 2217	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 9/15/87 / 1115 Hrs

Unit 1 MODE 1 - Power Operation Rx Power 94% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On 9/15/87, at 1115 hours, Byron Unit 2 was operating in Mode 1 at 94 percent power. The 2C Turbine Driven Feedwater Pump (FW)[SJ] was out-of-service at this time for maintenance. The pump draining incorporated the use of condenser vacuum through the pumps recirculation line. The pumps casing drain to the floor drains was opened slightly to aid draining. In addition, Mechanical Maintenance personnel removed a leaking section of line, opening the pump casing to atmosphere, thus, opening the condenser to additional air inleakage. This additional opening to the main condenser through the pumps recirculation lines caused a rapid loss of vacuum. The rapidly decreasing condanser vacuum caused an increase in Nuclear Power (5%) due to the decreased efficiency. An Over Power Delta Temperature (OPΔT) runback of 70 megawatts and a one minute delta I penalty resulted. The Unit 2 Hogging Vacuum Pump was started and the recirculation line on the pump was isolated.

C. CAUSE OF EVENT:

The loss of vacuum was due to the manual recirculation valve being slightly opened. The pump casing drain to the floor drains was throttled open causing air inleakage. In addition, Mechanical Maintenance was repairing a leak and removed a section of line that was connected to the pump casing causing another source of air inleakage. The volume of air inleakage caused a loss of vacuum. The loss of vacuum caused a decrease in efficiency, but the load remained the same so Nuclear Power increased 5% in order to maintain load. An OP ΔT runback of 70 megawatts occurred and a one minute delta I penalty resulted. There was a lack of communication and awareness between shift personnel and Mechanical Maintenance at the point when the work actually started on the pump.

D. SAFETY ANALYSIS:

The plant or public safety was not affected by the incident. Safety functions operated per design on the Reactor Protection System. Starting the Unit 2 Hogging Vacuum pump stopped the loss of vacuum on the main condenser.

E. CORRECTIVE ACTIONS:

The Operating staff had communications established with the Maintenance people during the pumps draining. The Operating staff had started the scenario by creating a minor leak to the main condenser through the pumps recirculation line. Maintenance people removed a leaking section of line on the pump causing additional air inleakage. This Deviation Report will be required reading for both departments foreman in order to stress the need for attention to condenser vacuum not only on the Feedwater Pumps but on all possible air inleakage Components.

F. PREVIOUS OCCURRENCES:

LER NUMBER	TITLE
NONE	

G MPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Not Applicable			

b) RESULTS OF NPRDS SEARCH:

Not Applicable

FW 36

DEVIATION INVESTIGATION REPORT (DIR)

Facility Name: Braidwood Station

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Title: Loss of Condenser Vacuum As a Result of Missed Isolation Point Due to Personnel Error

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	1
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	0 8 9
0 6	2 0	8 9	2 0	0 1	8 9	— 0 9 0	— 0 0	0 7	2 7	8 9		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
Jerald Wagner, Regulatory Assurance Ext. 2497	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. DESCRIPTION OF EVENT:

Unit 1 was operating at 89% power and BwVS 3.1.1-5 Incore-Excore Axial Flux Quarterly Calibration was in progress. MW/IN was selected on the turbine digital electro-hydraulic DEH control panel.

There were no systems or components inoperable at the beginning of the event which contributed to the severity of the event.

The event started on June 14, 1989 when Out Of Service (OOS) 89-1-1139 was written to isolate 1HD032B (First State Reheater Drain Tank 1B Emergency Drain Valve) (HD) [SI] for maintenance. OOS 89-1-1139 did not include Condenser Isolation Valve 1HD033B (the manual isolation valve between the Emergency Drain Valve and the Main Condenser). The bonnet for valve 1HD032B was loosened at approximately 0845 on 06-20-89 causing a large air leak and the resulting loss of condenser vacuum. Maintenance personnel (MMD) noting the rush of air at 1HD032B realized 1HD033B was not isolated and closed 1HD033B.

The loss of condenser vacuum, approximately 0.5 inch Hg. caused the following events: Reactor Coolant system average temperature decreased (Tavg), steam flow and feed flow increased approximately 1.2 M lbm/hr total, reactor power increased to 92%, and electrical power increased from 1069 to 1080 MWe.

Plant stability was attained at approximately 0915. Stability was achieved because of two separate actions; MMD closing the isolation valve 1HD033B and Operations ramping the Unit less than 90% then switching Main Turbine DEH control to MW/OUT.

DEVIATION INVESTIGATION REPORT TEXT CONTINUATION

FACILITY NAME	DIR NUMBER								PAGE			Form Rev 2.
	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER					
Braidwood Station	2 0	0 1	8 9	—	0 9 0	—	0 0		2	OF	0	

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

At 0905, Technical Specification 3.2.1 Action A.2 was entered when Reactor power exceeded 90% and axial flux difference was outside of the required band. At 0915, the action was exited when reactor power was reduced below 90%.

This event is being reported in accordance with Secion 1 attachment G, of the DVR Information Manual - events that result in unplanned entry into a Technical Specification Action Statement where a personnel error is indicated.

B. CAUSE OF EVENT:

The root cause of this event was personnel error in identifying the isolation boundaries required to perform maintenance work on 1HD032B. This is the responsibility of the Center Desk Operator per BwAP 330-1A1 TPC 4094 C.6. The error was also missed by the MMD Foreman who is charged with verifying the OOS is placed correctly and the equipment is safe to work on per BwAP 330-1A1 I.1.

C. CORRECTIVE ACTIONS:

Operations reviewed a similar OOS performed at the same time and prevented a repeat of this problem by adding the Condenser Isolation valve 1HD036D to OOS 89-1-1140 on 06-20-89.

Based on the initial information associated with this event a "Braidwood Station Error Evaluation" will be conducted to review this event with the personnel directly involved and their supervisor. The corrective actions addressing both root and contributing causes will be determined at this presentation and will be tracked by action item 456-200-89-09001.

FW37 HOTWELL LEVEL CONTROLLER FAILURE (CD037)

TYPE: DISCRETE, RV 0-48"

CAUSE: CONTROLLER 1LC-CD037 FAILURE

REF: M-2039 SHEET 4
M-2039 SHEET 6

PLT STA: 100% REACTOR POWER

EFFECTS: WHEN THIS MALFUNCTION IS INSERTED AT A SEVERITY HIGHER THAN ACTUAL 1LC-CD037 HOTWELL LEVEL, THE EMERGENCY OVERFLOW VALVE 1CD141 WILL OPEN. ANNUNCIATOR 17-C12 "CNDSR EMER OVER FLOW VALVE OPEN" ACTUATES. HOTWELL LEVEL DECREASES, AS INDICATED ON 1LI-CD042 AND 1LI-CD089. THE NORMAL M/U VALVE 1CD032 WILL OPEN TO MAINTAIN HOTWELL LEVEL.

INCREASING THE SEVERITY COUPLED WITH THE SAME SEVERITY MALFUNCTION ON THE NORMAL OVERFLOW CONTROLLER (MF FW39) WILL CAUSE THE STANDBY COND M/U PUMP TO AUTO START AND THE EMERGENCY M/U VALVE TO OPEN . ANNUNCIATOR 17-A12 "CNDSR HOTWELL LEVEL HIGH LOW" ACTUATES.

EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY CLOSING THE MANUAL ISOLATION VALVES ON THE NORMAL AND EMERGENCY OVERFLOW VALVES.

FAILING THIS LEVEL CONTROLLER AT A LOWER SEVERITY THAN NORMAL WILL PREVENT THE OPENING OF 1CD141.

MALFUNCTION REMOVAL WILL RETURN THE LEVEL CONTROLLER TO NORMAL OPERATION.

EVENTS: NONE

FW38 HOTWELL LEVEL CONTROLLER FAILURE (CD038)

TYPE: DISCRETE, RV 0-48"

CAUSE: CONTROLLER 1LC-CD038 FAILURE

REF: M-2039 SHEET 4
M-2039 SHEET 6

PLT STA: 100% REACTOR POWER

EFFECTS: WHEN THIS MALFUNCTION IS INSERTED AT A SEVERITY HIGHER THAN ACTUAL 1LC-CD038 HOTWELL LEVEL, THE NORMAL OVERFLOW VALVE 1CD144 WILL OPEN. HOTWELL LEVEL DECREASES, INDICATED ON 1LI-CD042 AND 1LI-CD089. THE NORMAL M/U VALVE 1CD032 WILL OPEN TO MAINTAIN HOTWELL LEVEL.

INCREASING THE SEVERITY COUPLED WITH THE SAME SEVERITY MALFUNCTION ON THE EMERGENCY OVERFLOW CONTROLLER (MF FW37) WILL CAUSE THE STANDBY COND M/U PUMP TO AUTO START, AND THE EMERGENCY M/U VALVE TO OPEN. THE ANNUNCIATOR 17-A12 "CNDSR HOTWELL LEVEL HIGH LOW" ACTUATES.

EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY CLOSING THE MANUAL ISOLATION VALVES ON THE NORMAL AND EMERGENCY OVERFLOW VALVES.

FAILING THIS LEVEL CONTROLLER AT A LOWER SEVERITY THAN NORMAL WILL PREVENT THE OPENING OF 1CD144.

MALFUNCTION REMOVAL WILL RETURN THE LEVEL CONTROLLER TO NORMAL OPERATION.

EVENTS: NONE

FW39 HOTWELL LEVEL CONTROLLER FAILURE (CD039)

TYPE: DISCRETE, RV 0-48"

CAUSE: CONTROLLER 1LC-CD039 FAILURE

REF: M-2039 SHEET 4
M-2039 SHEET 6

PLT STA: 100% REACTOR POWER

EFFECTS: WHEN THIS MALFUNCTION IS INSERTED AT A SEVERITY LOWER THAN ACTUAL 1LC-CD039 HOTWELL LEVEL, THE NORMAL M/U VALVE 1CD032 WILL OPEN. HOTWELL LEVEL INCREASES, INDICATED ON 1LI-CD042 AND 1LI-CD089. THE NORMAL OVERFLOW VALVE 1CD144 WILL OPEN TO MAINTAIN HOTWELL LEVEL.

INCREASING THE SEVERITY TO 0% COUPLED WITH THE SAME SEVERITY MALFUNCTION ON THE EMERGENCY M/U CONTROLLER (MF FW40) WILL CAUSE THE EMERGENCY OVERFLOW VALVE TO OPEN WHEN ACTUAL LEVEL INCREASES TO THE EMERGENCY OVERFLOW SETPOINT. ANNUNCIATOR 17-A12 "CNDSR HOTWELL LEVEL HIGH LOW" ACTUATES.

EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY CLOSING THE MANUAL ISOLATION VALVES ON THE NORMAL AND EMERGENCY M/U VALVES.

FAILING THIS LEVEL CONTROLLER AT A LOWER SEVERITY THAN NORMAL WILL PREVENT THE OPENING OF 1CD032.

MALFUNCTION REMOVAL WILL RETURN THE LEVEL CONTROLLER TO NORMAL OPERATION.

EVENTS: NONE

FW40 HOTWELL LEVEL CONTROLLER FAILURE (CD040)

TYPE: DISCRETE, RV 0-48"

CAUSE: CONTROLLER 1LC-CD040 FAILURE

PLT STA: 100% REACTOR POWER

REF: M-2039 SHEET 4
M-2039 SHEET 6

EFFECTS: WHEN THIS MALFUNCTION IS INSERTED AT A SEVERITY LOWER THAN ACTUAL 1LC-CD040 HOTWELL LEVEL, THE EMERGENCY M/U VALVE 1CD029 WILL OPEN. HOTWELL LEVEL INCREASES, INDICATED ON 1LI-CD042 AND 1LI-CD089. THE NORMAL OVERFLOW VALVE 1CD144 WILL OPEN TO MAINTAIN HOTWELL LEVEL.

INCREASING THE SEVERITY TO 0% COUPLED WITH THE SAME SEVERITY MALFUNCTION ON THE NORMAL M/U CONTROLLER (MF FW39) WILL CAUSE THE EMERGENCY OVERFLOW VALVE TO OPEN WHEN ACTUAL LEVEL INCREASES TO THE EMERGENCY OVERFLOW SETPOINT. ANNUNCIATOR 17-A12 "CNDSR HOTWELL LEVEL HIGH LOW" ACTUATES.

EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY CLOSING THE MANUAL ISOLATION VALVES ON THE NORMAL AND EMERGENCY M/U VALVES.

FAILING THIS LEVEL CONTROLLER AT A LOWER SEVERITY THAN NORMAL WILL PREVENT THE OPENING OF 1CD029.

MALFUNCTION REMOVAL WILL RETURN THE LEVEL CONTROLLER TO NORMAL OPERATION.

EVENTS: NONE

FW41 FW ISOL AUX RELAY FAILURE (TRAIN A)

TYPE: GENERIC, RB

A) AUX RELAY FWI 1A
B) AUX RELAY FWI 2A
C) AUX RELAY FWI 3A
D) AUX RELAY FWI 4A

CAUSE: FAULTY AUX RELAY

REF: 20E-1-4030FW56

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED AUX RELAY TO FAULT SUCH THAT WHEN A FEEDWATER ISOLATION SIGNAL IS GENERATED THE AUX RELAY WILL FAIL TO ENERGIZE. THIS PREVENTS THE AUX RELAY FROM SIGNALING ITS ASSOCIATED COMPONENTS TO ISOLATE. FAILURE OF THE ASSOCIATED COMPONENTS TO ISOLATE WILL OCCUR ONLY IF BOTH TRAIN A AND TRAIN B AUX RELAYS ARE FAILED (ie: TRAIN A FWI 1A & TRAIN B FWI 1B). FAILING ONLY ONE TRAIN WILL NOT PREVENT THE ASSOCIATED COMPONENTS FROM ISOLATING. THIS MALFUNCTION MUST BE INSERTED WITH MALFUNCTION FW42 "FW ISOL AUX RELAY FAILURE (TRAIN B)" TO PREVENT COMPONENT ISOLATION.

FAILURE OF RELAY 3A ALSO REMOVES THE FEEDWATER ISOLATION SEAL IN SIGNAL. THIS MEANS THAT IF A P-14 "S/G LEVEL HI-2" CAUSED THE FEEDWATER ISOLATION THEN ONLY THE P-14 FWI SIGNAL HAS TO BE CLEARED TO RE-OPEN THE FEEDWATER VALVES. THE AUX RELAYS WOULD NOT HAVE TO BE RESET.

MALFUNCTION REMOVAL WILL RESTORE THE AUX RELAYS TO NORMAL.

EVENTS: NONE

FW42 FW ISOL AUX RELAY FAILURE (TRAIN B)

TYPE: GENERIC, RB

A) AUX RELAY FWI 1B
B) AUX RELAY FWI 2B
C) AUX RELAY FWI 3B
D) AUX RELAY FWI 4B

CAUSE: FAULTY AUX RELAY

REF: 20E-1-4030FW56

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED AUX RELAY TO FAULT SUCH THAT WHEN A FEEDWATER ISOLATION SIGNAL IS GENERATED THE AUX RELAY WILL FAIL TO ENERGIZE. THIS PREVENTS THE AUX RELAY FROM SIGNALING ITS ASSOCIATED COMPONENTS TO ISOLATE. FAILURE OF THE ASSOCIATED COMPONENTS TO ISOLATE WILL OCCUR ONLY IF BOTH TRAIN A AND TRAIN B AUX RELAYS ARE FAILED (ie: TRAIN A FWI 1A & TRAIN B FWI 1B). FAILING ONLY ONE TRAIN WILL NOT PREVENT THE ASSOCIATED COMPONENTS FROM ISOLATING. THIS MALFUNCTION MUST BE INSERTED WITH MALFUNCTION FW41 "FW ISOL AUX RELAY FAILURE (TRAIN A)" TO PREVENT COMPONENT ISOLATION.

FAILURE OF RELAY 3B ALSO REMOVES THE FEEDWATER ISOLATION SEAL IN SIGNAL. THIS MEANS THAT IF A P-14 "S/G LEVEL HI-2" CAUSED THE FEEDWATER ISOLATION THEN ONLY THE P-14 FWI SIGNAL HAS TO BE CLEARED TO RE-OPEN THE FEEDWATER VALVES. THE AUX RELAYS WOULD NOT HAVE TO BE RESET.

MALFUNCTION REMOVAL WILL RESTORE THE AUX RELAYS TO NORMAL.

EVENTS: NONE

FW43 AUX FW PUMP FAILS TO START/TRIP (MOTOR)

TYPE: DISCRETE, RB

CAUSE: FAULTY OVERCURRENT (450/451) RELAY ACTUATION

REF: 20E-1-4030 AF01
M-37 SHEET 1

PLT STA: 1AF01PA PUMP IN OPERATION

EFFECTS: AUXILIARY FEEDWATER PUMP 1AF01PA BREAKER WILL OPEN. PUMP CURRENT INDICATION DECREASES TO ZERO, ANNUNCIATOR 3-A6 "AF PUMP TRIP" ACTUATES, AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. 1A AF PUMP DISCHARGE PRESSURE INDICATION AND THE ASSOCIATED S/G FLOW INDICATIONS DECREASE TO ZERO.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND TRIP LIGHT BY PLACING THE CONTROL SWITCH TO TRIP. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL CLOSE THEN IMMEDIATELY TRIP OPEN. AUTOMATIC START SIGNALS WILL HAVE THE SAME AFFECT AS A MANUAL START.

MALFUNCTION REMOVAL WILL RESTORE THE AUXILIARY FEEDWATER PUMP OVERCURRENT RELAY TO NORMAL.

EVENTS: NONE

FW44 AUX FW PUMP FAILS TO START/TRIP (DIESEL)

TYPE: DISCRETE, RB

CAUSE: FAULTY LUBE OIL PRESS SWITCH 1PS-AF143

REF: 20E-1-4030 AF02
20E-1-4030 AF12
M-37 SHEET 1

PLT STA: 1AF01PB IN OPERATION

EFFECTS: AUXILIARY FEEDWATER PUMP DIESEL WILL TRIP ON LOW LUBE OIL PRESSURE DUE TO PRESSURE SWITCH 1PS-AF143 FAILING LOW. ANNUNCIATOR 3-A6 "AF PUMP TRIP" ACTUATES, AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. 1B AF PUMP DISCHARGE PRESSURE INDICATION AND THE ASSOCIATED S/G FLOW INDICATIONS DECREASE TO ZERO.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND TRIP LIGHT BY PLACING THE CONTROL SWITCH TO STOP. THE OPERATOR CANNOT RESTART THE PUMP UNTIL THE MALFUNCTION IS REMOVED. AUTOMATIC START SIGNALS WILL HAVE NO EFFECT WHILE THE MALFUNCTION IS ACTIVE.

MALFUNCTION REMOVAL WILL RESTORE THE AUXILIARY FEEDWATER PUMP PRESSURE SWITCH TO NORMAL.

EVENTS: 1) DVR 06-01-89-057

DEVIATION REPORT

FW4 ½

DVR NO.			
06	01	89	057
STA	UNIT	YEAR	NO.

Form Rev 2.0

PART 1 | TITLE OF DEVIATION

1B AUXILIARY FEEDWATER PUMP TRIP ON OVERTEMPERATURE

OCCURRED 04/17/89 (DATE) 1118 (TIME)

SYSTEM AFFECTED: AF/SX

PLANT STATUS AT TIME OF EVENT: MODE 1 POWER(%) 100%

WORK REQUEST NO.: B66658

TESTING: [X] YES [ ] NO

DESCRIPTION OF EVENT

During performance of 1BVS 7.1.2.1.a-2, the Diesel Driven Auxiliary Feedwater (AF) Pump Monthly Surveillance, the 1B AF Pump was started successfully and ran for approximately 15 minutes. At 1118 hrs the Diesel Trouble alarm was received at 1PM06J, the operator just outside the pump room was requested to check what alarm condition was present at the local panel. As he entered the room, the pump tripped on overtemperature. The operator (EA) and STE in the room checked the Engine Jacket Water Heat Exchanger for overheating and found it hot to the touch. The 1SX173 and 1SX178 valves, SX supply and discharge, respectively, for the 1B AF Pump Diesel Jacket Water and pump accessory cooling were suspect. The Cubicle cooler was started to determine if the valves were receiving an open signal. The 1SX173 valve opened, but 1SX178 failed to open. The instrument air to the valve was then manually isolated and the valve failed open. Subsequent restoration of IA to the valve and start of the cubicle cooler resulted in successful opening of both 1SX173 and 1SX178. NWR B66658 was written to determine cause of failure. LCOAR 1BOS 7.1.2-1a had been entered for performance of the surveillance. The "B" Train of AF was left in LCOAR pending NWR resolution.

POTENTIALLY SIGNIFICANT EVENT PER NSD DIRECTIVE A-07 [ ] YES [X] NO

10CFR50.72 NRC RED PHONE NOTIFICATION MADE [ ] 1 HOUR [ ] 4 HOUR ______ TIME [X] NO

Mark E. Orth — RESPONSIBLE SUPERVISOR — 04/17/89 DATE

PART 2 | OPERATING ENGINEER'S COMMENTS

None.

[X] NON REPORTABLE EVENT

[ ] 30 DAY REPORTABLE/10CFR______

[ ] 5 DAY REPORT PER 10CFR21

[ ] ANNUAL/SPECIAL REPORT REQUIRED

A.I.R. #______

L.E.R. #______

NOTIFICATION

	DATE	TIME
REGION III		
Office of T. Maiman (NSD)	04/19/89	1030

[ ] CECO CORPORATE NOTIFICATION MADE IF ABOVE NOTIFICATION IS PER 10CFR21

TELECOPY ______ CECO CORPORATE OFFICER ______ DATE ______ TIME

PRELIMINARY REPORT COMPLETED AND REVIEWED: J. W. Schrock — OPERATING ENGINEER — 04/18/89 DATE

INVESTIGATION REPORT & RESOLUTION ACCEPTED BY STATION REVIEW: W. Walts 5/31/89 T. Tierinck 5/31/89

RESOLUTION APPROVED AND AUTHORIZED FOR DISTRIBUTION: S. Lommen — STATION MANAGER — 6/1/89 DATE

86-5176 (Form 15-52-1) 11-20-85

DOCUMENT ID (0343R/0044R)

FW45 AUX FW VALVE FAILURE

TYPE: GENERIC, RV 0-100% = VALVE TRAVEL

A)	1AF005A	E)	1AF005E
B)	1AF005B	F)	1AF005F
C)	1AF005C	G)	1AF005G
D)	1AF005D	H)	1AF005H

****
* NOTE *
* USE THESE MALFUNCTIONS TO *
* FAIL OPEN THE AF005 VALVES *
* ON A LOSS OF INSTRUMENT *
* BUS 111 OR 114 IF REQUESTED *
* TO LOCALLY FAIL AF005's *
* OPEN. *
****

CAUSE: POSITIONER FAILURE

REF: M-37
20E-1-4030 AF05
20E-1-4030 AF06

PLT STA: BOTH AUX FW TRAINS IN OPERATION

EFFECTS: MALFUNCTION INSERTION CAUSES THE SELECTED AF FLOW CONTROL VALVE TO FAIL IN THE SELECTED POSITION. THERE IS NO AFFECT INITIALLY WITHOUT THE ASSOCIATED AF PUMP RUNNING.

IF THE SEVERITY IS 100%, THE SELECTED CONTROL VALVE IS FAILED OPEN OVERFILLING THE ASSOCIATED S/G. THE AFFECTED S/G FLOW WILL BE GREATER THAN THE OTHER S/G FLOW.

FAILING THE AF FLOW CONTROL VALVES IN THE CLOSED POSITION WILL RESTRICT FLOW TO THE ASSOCIATED S/G. THE AFFECTED AF PUMP FLOW WILL ALSO BE LOWER.

MALFUNCTION REMOVAL WILL RESTORE THE SELECTED AF FLOW CONTROL VALVE TO NORMAL OPERATION.

EVENTS: 1) DVR 20-02-88-103
2) DVR 06-02-88-102

DEVIATION INVESTIGATION REPORT (DIR)

FW45

Form Rev 2.0

Facility Name	PAGE
Byron Nuclear Power Station	1 OF 0 2
Title	
POWER SUPPLY SURGE IN CABINET 2PA33J	

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	POWER LEVEL
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR		
0 9	2 0	8 8	0 6	0 2	8 8	— 1 0 2 —	0 0	1 1	0 3	8 8	1	0 6 5

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
Tim Tulon, Asst. Superintendent Operating Ext. 2213	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	B A	C N V	W 1 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 09/20/88 / 0346

Unit 2 MODE 1 - Power Operation Rx Power 65% RCS [AB] Temperature/Pressure Normal Operation

B. DESCRIPTION OF EVENT:

At 0346 hrs. on 09/20/88, a "CONT CAB PWR TROUBLE" alarm was received by the Unit 2 Operator for Panel 2PA33J, Train "A", failure. The flow setpoint signal for flow control to Steam Generator "B", from the "A" Train of the Auxiliary Feedwater (AF) [BA] System failed low due to loss of power to signal converter 2FY-AF033C. The Train "A", SG "B", flow Control Valve, 2AF005B, logic interpreted the zero setpoint as a requirement for zero flow and subsequently closed the valve. LCOAR 2BOS 7.1.2-1a was entered based on inoperability of the 2AF005B valve affecting operability of the "A" Train of the AF system. Concurrently, flow indication at 2FI-CS015 (Not Required by Technical Specifications), CS Eductor 2A flow, was lost due to failure of the Loop Power Supply, 2FY-CS015A. This card regulates the 26VDC cabinet power for use by its loop circuitry.

Nuclear Work Requests B59824 and B59825 were written to troubleshoot and repair the 2AF005B control circuit and loop 2CS015. The 2FY-AF033C card was replaced, the 2AF013 loop was calibrated satisfactorily, and the 2AF005B valve declared operable. The LCOAR for train "A" of the AF System was exited at 0710 hrs. on 09/21/88. A fuse was replaced on the 2FY-CS015A power supply and indication restored. There were no known components inoperable prior to the occurrence of this event which contributed to the event. All operator actions were correct.

C. CAUSE OF EVENT:

The cause of the event was surge in the 26VDC power supply to the card files in panel 2PA33J. The root cause for the surge in the power supply is indeterminate. The power surge at panel 2PA33J caused the fuse on the 2FY-CS015A to blow and damaged the 2FY-AF033C Signal Card. The power supply responsible for supplying the two failed cards is also responsible for supplying 26VDC power, where required, to all of the card racks in 2PA33J. No additional failures were seen in other cards in these racks due to this surge. Maintenance history indicates that this is not a reccuring problem and therefore should be considered an isolated incident. No further corrective action is planned.

D. SAFETY ANALYSIS:

The "B" train of Auxiliary Feedwater was available for the duration of the event per the requirements of LCOAR 2BOS 7.1.2-1a. The "B" Train of Auxiliary Feedwater is capable of supplying the flow and head required in the basis for Technical Specification 3/4.7.1.2 for the duration of the event, valve 2AF005B was capable of being manually manipulated using its pneumatic control circuit at the Remote Shutdown Panel to override the erroneous demand signal from the flow control loop.

E. CORRECTIVE ACTIONS:

The damaged signal converter card, 2FY-AF033C, was replaced and calibrated per NWR B59825. Replacement of the fuse of power supply 2FY-CS015A per Nuclear Work Request B59824 resulted in restoration of loop 2CS015 operability. No further corrective action is deemed necessary at this time.

F. PREVIOUS OCCURRENCES:

No past occurrences of this or similar events is documented in the Nuclear Work Request history file for 1/2PA33J or any of the 2AF-013 loop components.

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Westinghouse	Signal Converter Card	NSC Card	2837A10G08

b) RESULTS OF NPRDS SEARCH:

Not Applicable

c) RESULTS OF NWR SEARCH:

See "F" above.

DEVIATION INVESTIGATION REPORT

FW45

TITLE Failure of Auxiliary Feedwater Throttle Valve Due to Mispositioning of Handwheel by Person Unknown

PAGE 1 OF 0 2

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR		1
0 6	2 0	8 8	2 0	0 2	8 8	1 0 3	0 0	0 8	0 1	8 8	POWER LEVEL	0 4 9

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
Cheryl A. Melone, Technical Staff Engineer Ext. 2400	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 2; Event Date: June 20, 1988; Event Time: 1403
MODE: 1 - Power Operations; Rx Power: 49%; RCS [AB] Temperature/Pressure: 555 Degrees F/2235 psig

B. DESCRIPTION OF EVENT:

During an Auxiliary Feedwater (AF) [BA] actuation due to Lo-Lo Steam Generator levels, the Nuclear Station Operator (NSO) reduced flows to the steam generators to maintain proper levels. The 2C steam generator level continued to increase after the potentiometer for valve 2AF005G AF flow control valve, was set to its minimum setting. With the potentiometer set to its minimum setting, the flow to the generator was still greater than 200 gallons per minute (gpm) when the flow required to cool the steam generators is 160 gpm. The NSO maintained proper steam generator level by throttling the AF steam generator isolation valve, 2AF013G. Plant stability, given the in progress recovery from Lo-Lo 2C Steam Generator level condition, did not degrade as a result of this event.

C. CAUSE OF EVENT:

Upon investigation it was revealed that the valve handwheel for throttle valve 2AF005G was not in the neutral position. Personnel on shift were unaware of any prior repositioning of the handwheel. A review of shift logs and of the Unit 2 Component Abnormal Position Log yielded no documentation with regard to its repositioning. Thus the root cause of this event is valve mispositioning by person or persons unknown. This did not allow positioning of the throttle valve from the main control board. The purpose of this handwheel is to allow throttling of the flow control valves, on B train AF only, without the use of air or electricity.

TITLE Failure of Auxiliary Feedwater Throttle Valve Due to Mispositioning of Handwheel By Person Unknown

DIR NUMBER						PAGE
STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
2 0	0 2	8 8	—	1 0 3	—	0 0	2 OF 0 2

TEXT

D. SAFETY ANALYSIS:

Since the AF Steam Generator Isolation Valve 2AF013G was throttled to maintain steam generator level, no safety concerns were raised.

In a worst case scenario with the NSO unable to throttle the isolation valve with an increasing level in the affected steam generator due to excessive AF flow, two actions could restore steam generator level to normal:

1. Trip the affected train pump, B-train in this case, since both trains of AF were running.

2. Manually throttle the B train AF flow control valve by use of the manual handwheel.

E. CORRECTIVE ACTIONS:

The NSO took prompt action to maintain steam generator level by throttling the AF steam generator isolation valve, 2AF013G.

Procedure 1BwOS 7.1.2.1.a-2 will be revised to verify that the valves are in "Neutral" as well as open. This will be tracked to completion by action item 456-200-88-10301.

F. PREVIOUS OCCURRENCES:

There have been several occurrences of mispositioning events by person or persons unkonwn.

G. COMPONENT FAILURE DATA:

NONE

FW46 AUX FW LINE RUPTURE

TYPE: GENERIC, RV 0-225 GPM @ 1000 PSID

A) DOWNSTREAM 1AF005A (1A S/G)
B) DOWNSTREAM 1AF005E (1A S/G)
C) DOWNSTREAM 1AF005B (1B S/G)
D) DOWNSTREAM 1AF005F (1B S/G)
E) DOWNSTREAM 1AF005C (1C S/G)
F) DOWNSTREAM 1AF005G (1C S/G)
G) DOWNSTREAM 1AF005D (1D S/G)
H) DOWNSTREAM 1AF005H (1D S/G)

CAUSE: PIPE BREAK IMMEDIATELY DOWNSTREAM OF 1AF005

REF: M-37

PLT STA: BOTH AUX FW PUMPS IN OPERATION

EFFECTS: INSERTION OF THIS MALFUNCTION CAUSES A LOSS OF AUX FEEDWATER FLOW TO THE AFFECTED S/G. THE AFFECTED AF005 VALVE WILL MODULATE SHUT IN AN ATTEMPT TO MEET DEMANDED FLOW. THE AFFECTED S/G WATER LEVEL WILL DECREASE DUE TO THE LOSS OF FLOW.

EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY ISOLATING THE ENTIRE TRAIN, OR BY SHUTTING THE ASSOCIATED AF005 VALVE.

MALFUNCTION REMOVAL ONLY RESTORES THE PIPING INTEGRITY.

EVENTS: NONE

SSINS No.: 6835
IN 86-106

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

December 16, 1986

IE INFORMA ION NOTICE NO. 86-106: FEEDWATER LINE BREAK

Addressees:

All nuclear power reactor facilities holding an operating license or a construction permit.

Purpose:

This information notice is to alert addressees of a potentially generic problem with feedwater pipe thinning and other problems related to this event. Recipients are expected to review the information for applicability to their facilities and consider actions, if appropriate, to preclude similar problems occurring at their facilities. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On Tuesday, December 9, 1986, at 2:20 p.m., both units at the Surry Power Station were operating at full power when the 18-inch suction line to the main feedwater pump A for Unit 2 failed catastrophically. Eight workers who were replacing thermal insulation on a nearby line were burned by flashing feedwater. All were transported to area hospitals. Two workers were treated and released. Four other workers subsequently died.

Units 1 and 2 are identical. In each unit, feedwater flows from a 24-inch header to two 18-inch suction lines that each supply one of two main feedwater pumps. At maximum load under normal conditions, feedwater flow through each pump is 5 million lb/hr. Feedwater temperature, pressure, and enthalpy are 370°F, 450 psig, and 346 Btu/lb, respectively. At these conditions the fluid is in the single phase, liquid only regime. That is, the piping does not see a mixture of liquid and vapor.

The event was initiated by the main steam isolation valve on steam generator C failing closed. Because of the increased pressure in steam generator C that collapsed the voids in the water, the reactor tripped on low-low level in that steam generator. A 2-by-4 foot section of the wall of the suction line to the A main feedwater pump was blown out and came to rest in an overhead cable tray. The break was located in an elbow in the 18 inch line about one foot from the 24-inch header. The lateral reactive force generated by escaping

8612160250

feedwater completely severed the suction line. The free end whipped and came to rest against the discharge line for the other pump.

Steam flashing from the break and condensing in control cabinets and in open conduit piping apparently caused the fire suppression system to actuate, resulting in release of halon and carbon dioxide in the emergency switchgear room and in various cable tunnels and vaults and in the cable spreading room. Because of the volume of water and steam being released, operators isolated lines carrying high energy fluids to areas inundated by steam. Steam generator water levels were maintained with the auxiliary feedwater system, and system cooling was provided by actuating atmospheric dump valves as necessary.

The primary system responded normally to the loss of load transient with a partial loss of main feedwater. Primary coolant temperature was stabilized at 520°F and pressurizer level was recovered as it reached the low level set point. Primary pressure decreased from 2235 to 2015 psig following the reactor trip. By 2 a.m. on the following day, reactor temperature had been reduced to the point where the residual heat removal system could be put on line. The unit reached cold shutdown that morning. During the recovery effort, the operators and the plant performed as expected.

Discussion:

The pipe material is A-106B carbon steel and the elbow is 18-inch, extra strong A-234 grade WPB carbon steel. Nominal wall thickness of the suction piping is 0.500 inch. Measurements of the wall fragment demonstrated that the wall had been generally eroded to about 0.25 inch and was one of the causes of the failure. Preliminary examination of the 2-by-4 foot section of pipe blown out during the event shows the thinning to be relatively uniform except for some small localized areas. The thinnest areas are localized and appear to be about 1/16 inch thick. Some corrosion pitting is present. A preliminary micro-examination indicated that the pipe surface near the fracture had not been highly strained as with a high stress event, such as a high pressure spike in the system.

It has not been determined at this time whether a pressure spike in the system was a contributor to this event. There was no damage evident in the hanger supports to the condensate system.

Inspection revealed a disabled check valve in the discharge piping of the A main feedwater pump. This check valve was found with its seat displaced and a hinge pin missing.

On December 10, the licensee shut down Unit 1 for examination of the condition of feedwater piping. Inspection of the Unit 1 feedwater piping shows wall thinning similar to but not as severe as that in Unit 2.

The NRC dispatched an augmented investigation team (AIT) to the site. The AIT includes a metallurgist and a water hammer analyst.

The NRC will issue additional information as more inspection and analysis is completed.

No specific action or written response is required by this information notice. If you have questions about this matter, please contact the Regional Administrator of the appropriate NRC regional office or this office.

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contact: Roger Woodruff, IE
(301) 492-7205

Vincent Panciera, Region II
(404) 331-5540

Attachment:
List of Recently Issued IE Information Notices

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

HV01 CONTROL ROOM M/U FAN FAILS TO START/TRIP

TYPE: GENERIC, RB

A) 0A M/U FAN 0VC03CA
B) 0B M/U FAN 0VC03CB

CAUSE: FAULTY FLOW SWITCH 0FS-VC236 FOR A TRAIN AND 0FS-VC235 FOR B TRAIN

REF: 20E-0-4030 VC05, VC06, VC07, VC09

PLT STA: CONTROL ROOM HVAC IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES AN AUTOMATIC TRIP OF THE SELECTED CONTROL ROOM MAKEUP FAN (0VC03CA/CB) (IF IN OPERATION) AND WILL BE INDICATED BY ILLUMINATION OF THE TRIP LIGHT AND BY LOSS OF THE RUN LIGHT. ANNUNCIATORS 33- A10 "MCR M/U AIR FAN 0A TRIP FLOW HI/LO" AND 33-B10 "MCR M/U AIR FAN 0B TRIP FLOW HI/LO" WILL ACTUATE ACCORDINGLY.

MALFUNCTION REMOVAL WILL RESTORE THE PRESSURE SWITCH TO NORMAL

EVENTS: NONE

HV02 AUX BLDG CHARCOAL BOOSTER FAN FAILS TO START/TRIP

TYPE: GENERIC, RB

A) 0A FAN
B) 0B FAN
C) 0C FAN
D) 0D FAN
E) 0E FAN
F) 0F FAN

CAUSE: FAULTY LIMIT SWITCH ON DELUGE VALVE

REF: 20E-0-4030 VA13 - VA18
M-2095 SHEETS 7 & 8

PLT STA: AUX BLDG CHARCOAL BOOSTER FAN IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED AUX BUILDING CHARCOAL BOOSTER FAN BREAKER TO TRIP. FAN TRIP ANNUNCIATORS WILL RESPOND ACCORDINGLY. THE TRIP LIGHT ON THE ASSOCIATED CONTROL SWITCH ILLUMINATES AND THE RUN LIGHT GOES OUT. IF THE TRIPPED FAN HAD AUTO STARTED DUE TO A SAFETY INJECTION,THE STANDBY FAN WILL AUTO-START WHEN THE TRIPPED FANS DISCHARGE DAMPER IS FULLY CLOSED.

MALFUNCTION REMOVAL WILL RESTORE THE FAULTY LIMIT SWITCH TO NORMAL.

EVENTS: NONE

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

IA01	LOSS OF INSTRUMENT AIR
IA02	LOSS OF SERVICE AIR
IA03	IA LEAK INSIDE CONTAINMENT
IA04	IA LEAK ON TURBINE BLDG HEADER
IA05	SERVICE AIR COMPRESSOR FAILS TO START/TRIP
IA06	MSIV ROOM HEADER LEAK
IA07	STEAM DUMP HEADER LEAK
IA08	AUX FEED VALVES HEADER LEAK
IA09	AUX BUILDING IA LEAK

IA01 LOSS OF INSTRUMENT AIR

TYPE: GENERIC, RV 0-2600 CFM @ 110 PSID

A)	U-1 IA RECEIVER	1IA01T
B)	U-0 IA RECEIVER	0IA01T

CAUSE: RUPTURED IA RECEIVER

REF: M-55 SHEET 1

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION AT A LOW SEVERITY CAUSES AIR - LEAKAGE FROM THE SELECTED AIR RECIEVER TO THE ATMOSPHERE. THE RUNNING AIR COMPRESSOR(S) UNLOADS LESS FREQUENTLY TO MAINTAIN SYSTEM AIR PRESSURE AT 115 PSIG. INCREASING THE SEVERITY BEYOND THE CAPABILITY OF THE RUNNING AIR COMPRESSOR CAUSES THE INSTRUMENT AIR AND SERVICE AIR PRESSURE TO DECREASE (0PI-IA007 & 0PI-SA006). WHEN AIR PRESSURE IN ANY STATION AIR RECEIVER DROPS TO 105 PSIG, THE ASSOCIATED AIR COMPRESSOR AUTO STARTS. IF STATION AIR RECEIVER PRESSURE DROPS TO 95 PSIG, ANNUNCIATOR 37-C2 "SAC 1 RCVR PRESS LOW" AND/OR 38-C2 "SAC 0 RCVR PRESS LOW" ACTUATE. ANNUNCIATORS 37-C3 "IA RCVR 1 PRESS LOW" & 38-C3 "IA RCVR 0 PRESS LOW" ACTUATE AT 72 PSIG. AS PRESSURE CONTINUES TO DECREASE, THE AIR OPERATED VALVES IN THE PLANT BEGIN TO MOVE TO THEIR FAIL POSITIONS DEPENDENT ON HEADER PRESSURE. THE PLANT SYSTEMS RESPOND ACCURATELY TO THE RECEIVER FAILURE.

THE EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY ISOLATING THE AFFECTED RECEIVER USING REMOTE FUNCTIONS.

MALFUNCTION REMOVAL RESTORES THE RUPTURED AIR RECEIVER TO NORMAL ALLOWING THE SYSTEM TO REPRESSURIZE.

EVENTS: NONE

IA02 LOSS OF SERVICE AIR

TYPE: GENERIC, RV 0-2600 SCFM @ 115 PSID

A) U-1 SA RECEIVER 1SA01T
B) U-0 SA RECEIVER 0SA01T

CAUSE: RUPTURED SA RECEIVER

REF: M-54 SHEET 1A

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION AT A LOW SEVERITY CAUSES AIR LEAKAGE FROM THE SELECTED AIR RECIEVER TO THE ATMOSPHERE. THE RUNNING AIR COMPRESSOR(S) UNLOADS LESS FREQUENTLY TO MAINTAIN SYSTEM AIR PRESSURE AT 115 PSIG. INCREASING THE SEVERITY BEYOND THE CAPABILITY OF THE RUNNING AIR COMPRESSOR CAUSES THE INSTRUMENT AIR AND SERVICE AIR PRESSURE TO DECREASE (0PI-IA007 & 0PI-SA006). WHEN AIR PRESSURE IN ANY STATION AIR RECEIVER DROPS TO 105 PSIG, THE ASSOCIATED AIR COMPRESSOR AUTO STARTS. IF STATION AIR RECEIVER PRESSURE DROPS TO 95 PSIG, ANNUNCIATOR 37-C2 "SAC 1 RCVR PRESS LOW" AND/OR 38-C2 "SAC 0 RCVR PRESS LOW" ACTUATE. ANNUNCIATORS 37-C3 "IA RCVR 1 PRESS LOW" & 38-C3 "IA RCVR 0 PRESS LOW" ACTUATE AT 72 PSIG. AS PRESSURE CONTINUES TO DECREASE, THE AIR OPERATED VALVES IN THE PLANT BEGIN TO MOVE TO THEIR FAIL POSITIONS DEPENDENT ON HEADER PRESSURE. THE PLANT SYSTEMS RESPOND ACCURATELY TO THE RECEIVER FAILURE.

MALFUNCTION REMOVAL RESTORES THE RUPTURED AIR RECEIVER TO NORMAL ALLOWING THE SYSTEM TO REPRESSURIZE.

EVENTS: 1) DVR 06-01-88-239

IA02

DEVIATION INVESTIGATION REPORT (DIR)

Facility Name	Form Rev 2.0
Byron Nuclear Power Station	PAGE 1 OF 0 3

Title

Moisture Separator Rupture Resulting in Air Pressure Drop and Auto Start of the Standby Compressor

EVENT DATE			DIR NUMBER							REPORT DATE			OPERATING MODE	1
MONTH	DAY	YEAR	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	1 0 0
1 2	2 0	8 8	0 6	0 1	8 8	—	2 3 9	—	0 0	0 2	0 2	8 9		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
D. Brindle, Operating Engineer Ext. 2218	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	L F	S E P	I 2 1 1	N						

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 12-20-88 / 1140

Unit 1 MODE 1 - Power Operation Rx Power 99.9 RCS [AB] Temperature/Pressure Normal Operating

Unit 2 MODE 1 - Power Operation Rx Power 40 RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of this event which contributed to its severity. At 1140 hours on December 20, 1988, the Unit 2 Station Air Compressor (SAC) [LF] auto started on low header pressure. Investigation into the low header pressure revealed a rupture in the moisture separator on the Unit 1 SAC. The Unit 1 SAC was shutdown and the moisture separator was isolated from the system. Nuclear Work Request B63350 was written to replace the Unit 1 SAC moisture separator. The moisture separator was replaced and the Unit 1 SAC was made available on December 23, 1988. The Unit 1 SAC is Non-Safety Related and as such was not declared inoperable following the event; however, the Unit 1 SAC was made unavailable due to the moisture separator failure. On January 9, 1989 the Unit 2 SAC moisture separator developed a pin hole leak. The Unit 2 SAC was shutdown and the Unit 1 SAC was manually started after the leak was identified by an equipment attendant (Non-licensed). The Unit 2 SAC was made unavailable due to the leak. The moisture separator will be replaced under Nuclear Work Request B63631. There were no manual or automatic safety system actuations due to this event and plant conditions were stable at all times. All operator actions taken during this event were correct.

C. CAUSE OF EVENT:

The cause of the events is due to a combination of factors. The drain line which automatically drains the moisture separator on the Unit 1 SAC was found plugged by the Mechanics replacing the moisture separator. The drain line for the Unit 2 SAC moisture separator was found plugged on December 30, 1988 as part of the investigation into the Unit 1 SAC moisture separator failure. The accumulation of water caused corrosion of the interior of the moisture separators. The corrosion eventually caused weakening of the moisture separators which finally resulted in rupture. Corrosion of the moisture separator interiors cannot be prevented, but could be kept to a minimum by keeping the moisture separators drained.

It is unknown how long either drain was plugged since there were no work requests written prior to the events which described the condition. The operator rounds currently do not require the operator to manually drain the moisture separators.

D. SAFETY ANALYSIS:

There were no safety consequences due to this event. The Unit 2 SAC was available and Auto-Started to maintain air system pressure during the Unit 1 rupture. If the Unit 2 SAC had not been available, then only the Unit 0 SAC would have been in operation to maintain system pressure. During the Unit 2 moisture separator leak, the Unit 1 SAC was manually started to maintain air system pressure. If the Unit 1 SAC had not been available, then only the Unit 0 SAC would have been in operation to maintain system pressure. Braidwood station normally operates only one SAC and therefore it could be expected that the Unit 0 SAC at Byron could maintain air system pressure by itself. However, the capability of the Unit 0 SAC to maintain system pressure is highly dependant on air usage and air system leakage which is unique to Byron. If the Unit 0 SAC could not maintain sufficient pressure, then a plant transient and possibly a Unit trip may have resulted.

E. CORRECTIVE ACTIONS:

The moisture separator for the Unit 1 SAC was replaced and the drain line was cleared. The manual drain lines for both the Unit 0 and the Unit 2 SAC moisture separator were checked for flow. The Unit 2 SAC moisture separator manual drain was found plugged. Nuclear Work Request B63631 was initiated to clear the drain line and replace the Unit 2 SAC moisture separator which developed a leak on January 9, 1989. Procedure revisions were initiated to the operator rounds procedures (BOP 199-A28 and BOP 199-A47) to require manual draining of the moisture separators. Implementation of the procedure revisions is tracked by Action Item Record (AIR) 454-225-89-0025. A preventive maintenance request (BMP 3200-T15) was initiated to place the moisture separators on a 5 year inspection interval. The update to the preventive maintenance program is tracked by AIR 454-225-89-0024.

An Ultrasonic (UT) inspection was performed on the Unit 0 SAC moisture separator to determine if its walls were thinning. The UT inspection of the Unit 0 moisture separator did not show thinning to the extent that was found on the Unit 1 and Unit 2 SAC's.

DEVIATION INVESTIGATION REPORT TEXT CONTINUATION

FACILITY NAME	DIR NUMBER							PAGE	Form Rev 2.0
	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER		
Byron Nuclear Power Station	0 6	0 1	8 8	—	2 3 9	—	0 0	3 OF 0 3	

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

F. RECURRING EVENTS SEARCH AND ANALYSIS:

a) EVENT SEARCH (DIR, LER)

No similar previous events found at the station.

b) INDUSTRY SEARCH (OPEX's NPRDS)

No relevant information found

c) NWR

The drain line for the Unit 2 SAC moisture separtor was previously found plugged on September 30, 1987. (B49467)

d) ANALYSIS

Drain line plugging will be more quickly identified once manual draining of the moisture separator is added to the operator rounds.

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
U-1 ITT Fluid Handling Div.	Moisture Separator	Cat. No. ME71008	Manufactured 1977
U-2 ITT Fluid Handling Div.	Moisture Separator	Cat. No. ME72500	Manufactured 1979

H. OTHER RELATED DOCUMENTS:

NUREG 1275,Living SOER 81-09

I. EFFECTIVENESS REVIEW:

Scheduled for completion 2/1/90

J. ADDITIONAL DATA:

a) Affected Technical Specification: None

b) Procedures: BOP 199-A28 and BOP 199-A47

c) Cause Code: XPRMM1MM

d) Equipment Involved: 1SA02A and 2SA02A Moisture Separators

e) Other: Corrosion

(0220R/0025R/012089)

IA03 INSTRUMENT AIR LEAK INSIDE CONTAINMENT

TYPE: DISCRETE, RV 0-2000 SCFM

CAUSE: RUPTURE OF LINE 1IA65A (DOWNSTREAM OF 1IA066)

REF: M-55 SHEET 10

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION IS INSERTED, AN AIR LEAK AT THE SELECTED SEVERITY OCCURS JUST DOWNSTREAM OF 1IA066 IN THE CONTAINMENT. AT LOW SEVERITIES, THE OPERATING COMPRESSOR(S) UNLOADS LESS OFTEN IN ORDER TO MAINTAIN SYSTEM PRESSURE AT 115 PSIG. AT-HIGH SEVERITIES, 1IA066 WILL AUTO CLOSE RESULTING IN A LOSS OF PRESSURE TO THE AIR OPERATED VALVES IN THE CONTAINMENT. THE FOLLOWING ACTUATIONS OCCUR IN THE CONTAINMENT:

1) LETDOWN ISOLATION
2) PZR SPRAY VALVES CLOSE
3) PZR PORV'S LOSE AIR BUT HAVE AN ACCUMULATOR
4) REGEN HX ISOLATES (LOSS OF NORMAL CHARGING)

THE EFFECTS OF THIS MALFUNCTION ON THE REST OF THE INSTRUMENT AIR SYSTEM CAN BE MITIGATED BY CLOSING 1IA065 & 1IA066 ISOLATING THE CONTAINMENT.

MALFUNCTION REMOVAL RESTORES THE PIPE INTEGRITY ALLOWING REPRESSURIZATION OF THE CONTAINMENT INSTRUMENT AIR SYSTEM.

EVENTS: NONE

IA04 IA LEAK ON TURBINE BLDG RING HEADER

TYPE: DISCRETE, RV 0-1000 SCFM AT 115 PSID

CAUSE: PIPING FAILURE ON LINE 1IA56A DOWNSTREAM OF 1IA073 HEADER ISOLATION VALVE

REF: M-55 SHEET 2M

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION AT A LOW SEVERITY CAUSES AIR LEAKAGE FROM THE TURBINE BUILDING HEADER TO THE ATMOSPHERE. THE RUNNING AIR COMPRESSOR(S) UNLOADS LESS FREQUENTLY TO MAINTAIN SYSTEM AIR PRESSURE AT 115 PSIG. INCREASING THE SEVERITY BEYOND THE CAPABILITY OF THE RUNNING AIR COMPRESSOR CAUSES THE INSTRUMENT AIR AND SERVICE AIR PRESSURE TO DECREASE (0PI-IA007 & 0PI-SA006). WHEN AIR PRESSURE IN ANY STATION AIR RECEIVER DROPS TO 105 PSIG, THE ASSOCIATED AIR COMPRESSOR AUTO STARTS. WHEN STATION AIR RECEIVER PRESSURE DROPS TO 95 PSIG, ANNUNCIATOR 37-C2 "SAC 1 RCVR PRESS LOW" AND/OR 38-C2 "SAC 0 RCVR PRESS LOW" ACTUATE. ANNUNCIATORS 37-C3 "IA RCVR 1 PRESS LOW" & 38-C3 "IA RCVR 0 PRESS LOW" ACTUATE AT 72 PSIG. AS PRESSURE CONTINUES TO DECREASE, THE AIR OPERATED VALVES ON THE ASSOCIATED HEADER BEGIN TO MOVE TO THEIR FAIL POSITIONS DEPENDENT ON HEADER PRESSURE. THE PLANT SYSTEMS RESPOND ACCURATELY TO THE PIPING FAILURE.

MAJOR LOADS LOST ON THIS FAILURE INCLUDE: CB RECIRC VALVES, CD210A/B, 1FW016, HOTWELL LEVEL CONTROLLERS AND VARIOUS WG, GS AND HD VALVES.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY CLOSING THE HEADER ISOLATION VALVE 1IA073.

MALFUNCTION REMOVAL WILL RESTORE PIPING INTEGRITY TO NORMAL ALLOWING THE SYSTEM TO REPRESSURIZE.

EVENTS: 1) LER 20-01-88-025

IA04

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Braidwood Unit 1	0 5 0 0 0 4 5 6	1 of 0 4

Title (4) Manual Reactor Trips due to approaching Low Low Steam Generator Levels as a result of Loss of Instrument Air

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
1 1	1 5	8 8	8 8	0 2 5	0 0	1 2	1 3	8 8	Braidwood Unit 2	0 5 0 0 0 4 5 7
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 6

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	X 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER	
	AREA CODE	
Paul Nykaza, Technical Staff Engineer Ext. 2477	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
B	L D	P S F *	M 4 7 6	N					

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
__ Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

At 0904 on November 15, 1988 low instrument air pressure was observed. The rapid decrease in the instrument air header pressure caused the feedwater regulating valves to go closed. This decreased flow to the steam generators on both units. At 0908 both Units were manually tripped due to decreasing steam generator levels. The cause of this event was inadequate installation of a coupling in the instrument air header, line 0IA05B during construction. The inadequate solder joint was stressed by contract personnel standing on the line. The instrument air header was isolated and the line repaired by replacing the joint. The line was inspected upstream and downstream of the break for other possible leaks that may have occurred as a result of the break. Two other joints were repaired for pinhole leaks. Additional pipe supports will be added to the header. A letter was issued on November 16, 1988 to all site personnel reemphasizing the need for all personnel to exercise care in working around all plant equipment. There have been no previous occurrences of a reactor trip as a result of a loss of instrument air.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION								Form Rev 2.0		
FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)						Page (3)		
		Year	///	Sequential Number	///	Revision Number				
Braidwood Unit 2	0 5 0 0 0 4 5 6	8 8	-	0 2 5	-	0 0		0 2	OF	0 4

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 1; Event Date: November 15, 1988; Event Time: 0908;

Mode: 1 - Power Operation; Rx Power: 96%;

RCS [AB] Temperature/Pressure: 585 degrees F/2235 psig

Unit: Braidwood 2; Event Date: November 15, 1988; Event Time: 0908;

Mode: 1 - Power Operation; Rx Power: 79%;

RCS [AB] Temperature/Pressure: 578 degrees F/2240 psig

B. DESCRIPTION OF EVENT:

At 0904 on November 15, 1988 low instrument air (IA) [LD] receiver pressure was observed by control room personnel. Operators were dispatched to check for instrument air leaks. The rapid decrease in the instrument air header pressure caused the feedwater (FW) [SJ] regulating valves, (2)1FW510, (2)1FW520, (2)1FW530, and (2)1FW540, to go closed. This decreased flow to the steam generators (SG) [JB] on both units.

At 0908, both Units were manually tripped due to decreasing steam generator levels.

Operator actions decreased the severity of this event since the reactors were manually tripped prior to any Engineered Safety Feature (EF) [JE] actuation.

The Auxiliary Feedwater (AF) [BA] pumps automatically started to maintain steam generator levels as designed.

The appropriate NRC notification via the ENS phone system was made at 1000 pursuant to 10CFR50.72(a)(1)(i), and 10CFR50.72(b)(2)(ii).

Braidwood Station met with NRC Region III personnel on December 6, 1988, to discuss this event and proposed corrective actions.

This event is being reported pursuant to 10CFR50.73(a)(2)(iv) - any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature, including the Reactor Protection System.

C. CAUSE OF EVENT:

The root cause of this event was inadequate installation of a copper coupling in the instrument air header, line 0IA05B during construction. The inadequate solder joint was stressed by contract personnel standing on the line.

When operators arrived at the break location, there was evidence that the line had been used to stand on while painting another line above it. The painting of the line above had abruptly ended directly above the break and a wet paint roller was found on the floor below the broken line. It was verified that a painter had been standing on the line.

C. CAUSE OF EVENT: (continued)

The resulting loss of instrument air caused the feedwater regulating valves to go closed. This resulted in a reduction of feedwater flow to the steam generators leading to the manual reactor trips.

D. SAFETY ANALYSIS:

This event had no effect on the safety of the plant or the public. All engineered safety systems operated as designed.

Under the worst case conditions of a plant operating at 100% power with a loss of instrument air and no operator action, there would be no additional adverse impact on the safety of the plant or public as this is enveloped by the Final Safety Analysis Report (FSAR), Process Auxiliaries.

E. CORRECTIVE ACTIONS:

The immediate corrective action was to recover steam generator levels and establish stable conditions.

The instrument air header was isolated and the line repaired by replacing the joint. The line was also inspected upstream and downstream of the break for other possible leaks which may have occurred as a result of the break. Two other joints were repaired for pinhole leaks. Additional pipe supports will be added to the header. This will be tracked to completion by action item 456-200-88-26701.

PWR Engineering will evaluate the solder quality for portions of the IA System by sampling a few of these joints during the Unit 2 surveillance outage. Any additional actions will be based on the results of this sample. This will be tracked to completion by Action Item 456-200-88-26702.

Braidwood letter 88-1439 was issued on November 16, 1988 to all site personnel reemphasizing the need for all personnel to exercise care in working around all plant equipment.

F. PREVIOUS OCCURRENCES:

There have been no previous occurrences of a reactor trip as a result of a loss of instrument air.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	MFG Part Number
Mueller Brass	Elbow, tubine; 45 deg 4 inch copper	N/A	N/A
Mueller Brass	Elbow tubing; 90 deg 4 inch copper	N/A	N/A
Mueller Brass	Coupling; tubing; 4 inch copper	N/A	N/A
Mueller Brass	Turbine; copper; 4 inch x 20 ft. Type K Hard temper ASTM B888	N/A	N/A

IA05 SERVICE AIR COMPRESSOR FAILS TO START/TRIP

TYPE: GENERIC, RB

A) U-1 SA COMPRESSOR
B) U-0 SA COMPRESSOR
C) U-2 SA COMPRESSOR

CAUSE: FAULTY HOT AIR TEMPERATURE SWITCH: 1TS-SA046, 0TS-SA046, 2TS-SA046 RESPECTIVELY

REF: 20E-1-4030 SA01, SA02
20E-0-4030 SA01, SA02
20E-2-4030 SA01, SA02

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED SERVICE AIR COMPRESSOR TO TRIP ON HIGH AIR TEMPERATURE. ANNUNCIATORS 37-A2 "SAC 1 TRIP" AND/OR 38-A2 "SAC 0 TRIP" ACTUATE AS DO THE ASSOCIATED SAC TROUBLE ANNUNCIATORS 37-B2 AND 38-B2. THE INSTRUMENT AIR AND SERVICE AIR HEADER PRESSURE WILL DECREASE AS INDICATED ON 0PI-IA007 & 0PI-SA006. WHEN AIR PRESSURE IN ANY STATION AIR RECEIVER DROPS TO 105 PSIG, THE ASSOCIATED STANDBY AIR COMPRESSOR AUTO STARTS. WHEN STATION AIR RECEIVER PRESSURE DROPS TO 95 PSIG, ANNUNCIATOR 37-C2 "SAC 1 RCVR PRESS LOW" AND/OR 38-C2 "SAC 0 RCVR PRESS LOW" ACTUATE. ANNUNCIATORS 37-C3 "IA RCVR 1 PRESS LOW" & 38-C3 "IA RCVR 0 PRESS LOW" ACTUATE IF PRESSURE DROPS TO 72 PSIG. ANY ATTEMPT TO RESTART A TRIPPED COMPRESSOR WILL BE UNSUCCESSFUL.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY ENSURING THE STANDBY COMPRESSOR AUTO STARTS.

MALFUNCTION REMOVAL WILL RESTORE THE FAULTY TEMPERATURE SWITCH TO NORMAL.

EVENTS: NONE

IA06 MSIV ROOM HEADER LEAK

TYPE: GENERIC, RV 0-2000 SCFM AT 115 PSID

A) EAST MSIV ROOM
B) WEST MSIV ROOM

CAUSE: PIPING FAILURE DOWNSTREAM OF 1IA127 ISOLATION VALVE FOR THE EAST MSIV ROOM AND DOWNSTREAM OF 1IA124 ISOLATION VALVE FOR THE WEST MSIV ROOM.

REF: M-55 SHEET 11

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION AT A LOW SEVERITY CAUSES AIR LEAKAGE FROM THE MSIV ROOM TO THE ATMOSPHERE. THE RUNNING AIR COMPRESSOR(S) UNLOADS LESS FREQUENTLY TO MAINTAIN SYSTEM AIR PRESSURE AT 115 PSIG. INCREASING THE SEVERITY BEYOND THE CAPABILITY OF THE RUNNING AIR COMPRESSOR CAUSES THE INSTRUMENT AIR AND SERVICE AIR PRESSURE TO DECREASE (0PI-IA007 & 0PI-SA006). WHEN AIR PRESSURE IN ANY STATION AIR RECEIVER DROPS TO 105 PSIG, THE ASSOCIATED AIR COMPRESSOR AUTO STARTS. WHEN STATION AIR RECEIVER PRESSURE DROPS TO 95 PSIG, ANNUNCIATOR 37-C2 "SAC 1 RCVR PRESS LOW" AND/OR 38-C2 "SAC 0 RCVR PRESS LOW" ACTUATE. ANNUNCIATORS 37-C3 "IA RCVR 1 PRESS LOW" & 38-C3 "IA RCVR 0 PRESS LOW" ACTUATE AT 72 PSIG. AS PRESSURE CONTINUES TO DECREASE, THE AIR OPERATED VALVES ON THE ASSOCIATED HEADER BEGIN TO MOVE TO THEIR FAIL POSITIONS DEPENDENT ON HEADER PRESSURE. THE PLANT SYSTEMS RESPOND ACCURATELY TO THE PIPING FAILURE.

MAJOR LOADS LOST:

<u>IA06A:</u> B/C S/G:FW VALVES, MSIV BYPASS VALVES, S/G BLOWDOWN/SAMPLE VALVES.

<u>IA06B:</u> A/D S/G:FW VALVES, MSIV BYPASS VALVES, S/G BLOWDOWN/SAMPLE VALVES.

THE 0PERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY CLOSING THE HEADER ISOLATION VALVE 1IA127 FOR THE EAST MSIV ROOM AND 1IA124 FOR THE EAST MSIV ROOM.

MALFUNCTION REMOVAL WILL RESTORE PIPING INTEGRITY TO NORMAL ALLOWING THE SYSTEM TO REPRESSURIZE.

EVENTS: NONE

IA07 STEAM DUMP HEADER LEAK

TYPE: GENERIC, RV 0-2000 SCFM AT 115 PSID

A) STEAM DUMPS A-D HEADER
B) STEAM DUMPS E-H HEADER
C) STEAM DUMPS J-M HEADER

CAUSE: PIPING FAILURE DOWNSTREAM OF ISOLATION VALVES 1IA274A, 1IA274B, AND 1IA274C RESPECTIVELY

REF: M-55 SHEET 2H

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION AT A LOW SEVERITY CAUSES AIR LEAKAGE FROM THE TURBINE RING HEADER TO THE ATMOSPHERE. THE RUNNING AIR COMPRESSOR(S) UNLOADS LESS FREQUENTLY TO MAINTAIN SYSTEM AIR PRESSURE AT 115 PSIG. INCREASING THE SEVERITY BEYOND THE CAPABILITY OF THE RUNNING AIR COMPRESSOR CAUSES THE INSTRUMENT AIR AND SERVICE AIR PRESSURE TO DECREASE (0PI-IA007 & 0PI-SA006). WHEN AIR PRESSURE IN ANY STATION AIR RECEIVER DROPS TO 105 PSIG, THE ASSOCIATED AIR COMPRESSOR AUTO STARTS. WHEN STATION AIR RECEIVER PRESSURE DROPS TO 95 PSIG, ANNUNCIATOR 37-C2 "SAC 1 RCVR PRESS LOW" AND/OR 38-C2 "SAC 0 RCVR PRESS LOW" ACTUATE. ANNUNCIATORS 37-C3 "IA RCVR 1 PRESS LOW" & 38-C3 "IA RCVR 0 PRESS LOW" ACTUATE AT 72 PSIG. AS PRESSURE CONTINUES TO DECREASE, THE AIR OPERATED VALVES ON THE ASSOCIATED HEADER BEGIN TO MOVE TO THEIR FAIL POSITIONS DEPENDENT ON HEADER PRESSURE. THE PLANT SYSTEMS RESPOND ACCURATELY TO THE PIPING FAILURE.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY CLOSING THE HEADER ISOLATION VALVE 1IA274A FOR FOR STEAM DUMP HEADER A-D, 1IA274B FOR STEAM DUMP HEADER E-H, AND 1IA274C FOR STEAM DUMP HEADER J-M.

MALFUNCTION REMOVAL WILL RESTORE PIPING INTEGRITY TO NORMAL ALLOWING THE SYSTEM TO REPRESSURIZE.

EVENTS: NONE

IA08 AUX FEED VALVES HEADER LEAK

TYPE: DISCRETE, RV 0-2000 SCFM AT 115 PSID

CAUSE: PIPING FAILURE DOWNSTREAM OF ISOLATION VALVE OIA106

REF: M-55 SHEET 8

PLT STA: AF SYSTEM IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION AT A LOW SEVERITY CAUSES AIR LEAKAGE FROM THE AUX. BLDG RING HEADER TO THE ATMOSPHERE. THE RUNNING AIR COMPRESSOR(S) UNLOADS LESS FREQUENTLY TO MAINTAIN SYSTEM AIR PRESSURE AT 115 PSIG. INCREASING THE SEVERITY BEYOND THE CAPABILITY OF THE RUNNING AIR COMPRESSOR CAUSES THE INSTRUMENT AIR AND SERVICE AIR PRESSURE TO DECREASE (0PI-IA007 & 0PI-SA006). WHEN AIR PRESSURE IN ANY STATION AIR RECEIVER DROPS TO 105 PSIG, THE ASSOCIATED AIR COMPRESSOR AUTO STARTS. WHEN STATION AIR RECEIVER PRESSURE DROPS TO 95 PSIG, ANNUNCIATOR 37-C2 "SAC 1 RCVR PRESS LOW" AND/OR 38-C2 "SAC 0 RCVR PRESS LOW" ACTUATE. ANNUNCIATORS 37-C3 "IA RCVR 1 PRESS LOW" & 38-C3 "IA RCVR 0 PRESS LOW" ACTUATE AT 72 PSIG. AS PRESSURE CONTINUES TO DECREASE, THE AF005's BEGIN TO MOVE TO THE OPEN POSITION DEPENDENT ON HEADER PRESSURE.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY CLOSING THE HEADER ISOLATION VALVE 0IA106.

MALFUNCTION REMOVAL WILL RESTORE PIPING INTEGRITY TO NORMAL ALLOWING THE SYSTEM TO REPRESSURIZE.

EVENTS: NONE

IA09 AUX BUILDING IA LEAK

TYPE: GENERIC, RV 0-1000 SCFM AT 115 PSID

A) SX/CV VALVES
B) CV VALVES

CAUSE: PIPING FAILURE DOWNSTREAM OF VALVE 0IA100 FOR IA09A, AND DOWNSTREAM OF 0IA661 FOR IA09B.

REF: M-55 SHEET 7

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION AT A LOW SEVERITY CAUSES AIR LEAKAGE FROM THE AUX BLDG HEADER TO THE ATMOSPHERE. THE RUNNING AIR COMPRESSOR(S) UNLOADS LESS FREQUENTLY TO MAINTAIN SYSTEM AIR PRESSURE AT 115 PSIG. INCREASING THE SEVERITY BEYOND THE CAPABILITY OF THE RUNNING AIR COMPRESSOR CAUSES THE INSTRUMENT AIR AND SERVICE AIR PRESSURE TO DECREASE (0PI-IA007 & 0PI-SA006). WHEN AIR PRESSURE IN ANY STATION AIR RECEIVER DROPS TO 105 PSIG, THE ASSOCIATED AIR COMPRESSOR AUTO STARTS. WHEN STATION AIR RECEIVER PRESSURE DROPS TO 95 PSIG, ANNUNCIATOR 37-C2 "SAC 1 RCVR PRESS LOW" AND/OR 38-C2 "SAC 0 RCVR PRESS LOW" ACTUATE. ANNUNCIATORS 37-C3 "IA RCVR 1 PRESS LOW" & 38-C3 "IA RCVR 0 PRESS LOW" ACTUATE AT 72 PSIG. AS PRESSURE CONTINUES TO DECREASE, THE AIR OPERATED VALVES ON THE ASSOCIATED HEADER BEGIN TO MOVE TO THEIR FAIL POSITIONS DEPENDENT ON HEADER PRESSURE. THE PLANT SYSTEMS RESPOND ACCURATELY TO THE PIPING FAILURES.

MAJOR LOADS LOST ON IA09A FAILURE INCLUDE: RMCS VALVES, SX TO CNMT CHILLERS, VQ DAMPERS AND VARIOUS BR VALVES.

MAJOR LOADS LOST ON IA09B FAILURE INCLUDE: CHARGING FLOW CONTROL VALVE (1CV121), PCV131, INLET VALVES TO BOTH LETDOWN HX'S AND 1CC130A/B.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY CLOSING THE APPROPRIATE ISOLATION VALVE (0IA100/0IA661).

MALFUNCTION REMOVAL WILL RESTORE PIPING INTEGRITY TO NORMAL ALLOWING THE SYSTEM TO REPRESSURIZE.

EVENTS: NONE

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

MS01	FAILURE OF MAIN STEAM ISOLATION VALVE(S)
MS02	MSIV BYPASS VALVE FAILURE
MS03	S/G SAFETY VALVE FAILURE
MS04	S/G PORV CONTROLLER FAILURE
MS05	STUCK STEAM DUMP
MS06	MSR FAILS TO ISOLATE
MS07	STEAMLINE BREAK INSIDE CONTAINMENT
MS08	STEAMLINE BREAK OUTSIDE CONTAINMENT
MS09	MAIN STEAM HEADER CROSS-TIE RUPTURE
MS10	HEATER 13 EXTRACTION STEAM LINE BREAK
MS11	LOW PRESSURE TURBINE INLET PRESS SWITCH FAILURE

MS01 FAILURE OF MAIN STEAM ISOLATION VALVE(S)

TYPE: GENERIC, RV 0-100%

A) 1MS001A
B) 1MS001B
C) 1MS001C
D) 1MS001D

CAUSE: MECHANICAL BINDING

REF: 20E-1-4030 MS01
20E-1-4030 MS02
20E-1-4030 MS03
20E-1-4030 MS04

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION AT THE DESIRED SEVERITY INITIALLY HAS NO AFFECT ON THE POSITION OF THE SELECTED MSIV. VALVE ACTUATION BY EITHER MANUAL OR AUTOMATIC ACTION, IN EITHER THE OPEN OR CLOSE DIRECTION, WILL CAUSE THE VALVE TO BECOME MECHANICALLY BOUND IN THE SELECTED POSITION.

FAILING THE VALVE AT 100% (OPEN), THEN CREATING A MAIN STEAM LINE ISOLATION SIGNAL CAUSES THE SELECTED VALVE TO BIND IN THE OPEN POSITION.

FAILING THE VALVE AT 0% (CLOSE), THEN ATTEMPTING TO OPEN THE SELECTED MSIV WILL BIND THE MSIV IN THE CLOSED POSITION.

MALFUNCTION REMOVAL RESTORES THE MSIV MECHANICAL OPERABILITY TO NORMAL.

EVENTS: 1) DVR 20-02-88-165
2) LER 20-01-88-024

MS01

DEVIATION INVESTIGATION REPORT (DIR)

Facility Name	Form Rev 2.0
Braidwood 2	PAGE 1 OF 0 2

Title Failure of the 2C MSIV to Close Due to Low Hydraulic Pump Pressure and Low Accumulator Precharge Pressure

EVENT DATE			DIR NUMBER					REPORT DATE				
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	OPERATING MODE	3
1 0	0 1	8 8	2 0	0 2	8 8	1 6 5	0 0	1 0	2 8	8 8	POWER LEVEL	0 0 0

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
Dan Stroh, Technical Staff Engineer Ext. 2477	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. Plant Conditions Prior to Event:

Unit: Braidwood 2; Event Date: October 1, 1988; Event Time: 1600

Mode: 3 - Hot Standby; Rx Power: 0%

RCS [AB] Temperature/Pressure: 558 degrees F/2238 psig

B. Description of Event:

On October 1, 1988 at 1600 hrs, the control switches for all four Main Steam Isolation Valves (MSIV) were taken to the CLOSE position in preparation to take the Unit 2 Condenser out of service. All four valves went full closed with the exception of the 2C valve which showed dual light indication. A B-man was dispatched to the field and he verified the valve was almost closed but still off of the lower limit switch. A manual Main Steam Isolation was initiated from the Control Room. This signal uses both the active and standby accumulator for closing. When initiated, the 2C MSIV went full closed. Nuclear Work Request (NWR) #A25932 was written to investigate and repair the non-closure problem. Upon investigation, the hydraulic pressure of the accumulators was found at 4500 psig, it should have been at 5000 psig. Mechanical Maintenance verified there were no problems with hydraulic or pneumatic pilot valves on the unit, and both hydraulic and pneumatic accumulators were properly recharged. The valve was returned to service on Tuesday 10-4-88. The plant remained stable throughout the event.

DEVIATION INVESTIGATION REPORT TEXT CONTINUATION

FACILITY NAME	DIR NUMBER					PAGE
	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Braidwood 2	2 0	0 2	8 8	— 1 6 6 —	0 0	2 OF 0 2

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TEXT

C. Cause of Event:

The cause of the event was two fold: low hydraulic pump pressure and low nitrogen precharge pressure on the accumulator. The cause of the low hydraulic pressure was low air pressure to the pump. The lock nut on the air regulator was loose and apparently the regulator had backed off. The regulator was properly adjusted and the lock nut tightened.

D. Safety Analysis:

Proper actions were initiated by the operator in a timely manner. No unusual safety concerns resulted from the equipment failure since the valve was brought to the Engineered Safety Position (closed) when the problem was discovered. The inactive train was charged and was used to fully close the valve within 5 seconds after receipt of a close signal.

E. Corrective Action:

NWR #25932 was generated to troubleshoot and recharge both the nitrogen and hydraulic systems. The air regulator to the hydraulic pump was adjusted and locked in place.

F. Previous Occurrence:

None

G. Component Failure Data:

None

vm 307

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Braidwood Unit 1	0 5 0 0 0 4 5 6	1 of 0 4

Title (4) Inoperable Main Steam Isolation Valve Due to Failure of M1 Four Way Solenoid Valve

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
1 0	3 1	8 8	8 8	— 0 2 4 —	0 0	1 1	0 7	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 7

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	__ 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	X 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER AREA CODE	
Dan Stroh Technical Staff Engineer Ext. 2477	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	S B	H C V	A 3 9 1	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

__ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15) Month | Day | Year

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

At 1500 on October 31, 1988, Limiting Condition for Operation Action Requirement (LCOAR) 3.2-1a was entered due to the 1A Main Steam Isolation Valve (MSIV) standby hydraulic accumulator pressure below 4800 psig. Shortly thereafter, the active hydraulic accumulator pressure decreased to 4800 psig and the hydraulic pump could not maintain hydraulic accumulator pressure. At 1645 the 1A MSIV was declared inoperable and preparations were made to go into mode 2, <5% power, per Technical Specification 3/4.7.1.5. Mechanical maintenance began work on the valve by preparing to remove the "N1" 4 way hydraulic valve which was thought to be internally leaking. At 1803 an Event Notification System phone call was made to the NRC per Braidwood Administrative Procedure 1250-6A3, I.15 example h.iii. At 2043 the source of the hydraulic leak had not been found and a power reduction was begun. Because of forced power reduction per technical specification, an unusual event was declared due to Emergency Action Level #14 at 2115. At 2117, a Nuclear Accident Reporting System phone call was made. At 0040 on November 1, 1988, Unit 1 entered mode 2, and at 0119 the 1A MSIV was closed using only the hydraulic pump. : 0225, a mechanical block was installed on the 1A MSIV and the event was terminated at 0230. Replacement of the defective "MI" 4 way hydraulic valve was completed by 1000 and the 1A MSIV was tested and returned to service at 1921.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 1: Event Date: October 31, 1988; Event Time: 1645;

Mode: 1 - Power Operation: Rx Power: 98%;

RCS [AB] Temperature/Pressure: 557 degrees F/2231 psig

B. DESCRIPTION OF EVENT:

There were no structures, systems or components inoperable or degraded at the beginning of the event that contributed to the event.

At 1400 on October 31, 1988 an operator noted that the hydraulic pump on the 1A Main Steam (MS) [SB] Isolation Valve (MSIV) was pumping continuously. Further investigation revealed that the standby hydraulic accumulator pressure was at 3700 psig. The active accumulator pressure was still at 5000 psig. Because a train is considered inoperable when the hydraulic pressure decreases below 4800 psig, Technical Specification 3/4.3.2 Action Statement number 23, which requires the inoperable train be restored within 48 hours, was entered at 1500. Mechanical Maintenance (MMD) began to prepare a work package to replace the "N1" four way hydraulic valve which was thought to have an internal leak. Shortly after 1630, the pressure on the active hydraulic accumulator began to decrease. At 1645 the 1A MSIV was declared inoperable after it became apparent that the hydraulic pump could not maintain the required hydraulic pressure on either accumulator. This was a result of pump discharge flow being diverted to the leak on the standby system. This prevented the active hydraulic accumulator from being repressurized and in fact allowed the active hydraulic accumulator to depressurize due to normal system losses. Per Technical Specification 3/4.7.1.5 Mode 1 Action Statement, the valve had to be repaired within 4 hours (2045), or the plant be placed in Hot Standby in 6 hours and in Hot Shutdown within the following 6 hours. Upon entry into Mode 2, this Technical Specification will allow indefinite operation as long as the inoperable MSIV is closed and maintained closed. A decision was made to attempt to repair the valve under the allotted 4 hour time clock and if repairs could not be completed, perform a power reduction to Mode 2 and complete the repairs.

At 1930 the 1A MSIV was taken out of service and MMD replaced the "N1" four way hydraulic valve by 2000. However, the leak was not stopped and hydraulic pressure could not be restored. Subsequent bench testing of the "N1" valve proved that it was functioning properly. Following replacement and testing of the suspect "N1" valve, the Anchor Darling vendor representative was consulted. It was then determined that the only other cause could be the second valve in the hydraulic circuit (i.e. "M1"). This valve is identical to the "N1" valve. The decision was made to exchange the good "N1" valve that was removed, with the installed "M1" valve. At 2043 a power reduction was begun to satisfy Technical Specification 3/4.7.1.5.

At 2115, because of a forced power reduction per Technical Specifications, an Unusual Event was declared per Emergency Action Level (EAL) #14 at 2115.

At 2121, the System Power Supply Office (SPSO) verified the Generating Stations Emergency Plan (GSEP) classification. The Illinois Emergency Services and Disaster Agency (IESDA) was also contacted.

At 2127, the Station Duty Officer (SDO) and the Nuclear Duty Officer (NDO) were notified of the GSEP event.

Hourly updates were made to IESDA and Illinois Department of Nuclear Safety (IDNS).

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION								Form Rev 2.0
FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)						Page (3)
		Year	///	Sequential Number	///	Revision Number		
Braidwood Unit 1	0 5 0 0 0 4 5 6	8 8	-	0 2 4	-	0 0		0 3 OF 0 4

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

B. DESCRIPTION OF EVENT, Continued:

At 0040 on November 1, 1988, Unit 1 entered Mode 2, and at 0119 the 1A MSIV was closed using the hydraulic pump.

At 0225, a mechanical restraint was installed on the 1A MSIV.

At 0230, the Unusual Event was terminated after Technical Specification 3/4.7.1.5 Action Statement for Mode 2 was fully complied with.

Repairs were made on the 1A MSIV, the valve was tested and returned to operable status by 1921 on October 21, 1988.

The appropriate NRC notification via the ENS phone system was made at 1803 pursuant to 10CFR50.72(b)(1)(ii) and Braidwood Administrative Procedure BwAP 1250 - 6A3, I.15 example h.iii. This was 18 minutes late due to the delay in recognizing the one hour time requirement.

This event is being reported pursuant to 10CFR50.73(a)(2)(ii) - Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded, or that resulted in the nuclear power plant being in an unanalyzed condition that significantly compromised plant safety.

C. CAUSE OF EVENT:

The MSIV has two redundant hydraulic accumulators available to close the valve. Each hydraulic accumulator has two identical four way hydraulic valves; an "M1" and an "N1" valve. The "M1" valve controls the opening and closing of the MSIV and the "N1" valve controls the charging of the hydraulic accumulator. Because the standby hydraulic accumulator lost pressure, the "N1" valve was first suspected. However, replacement of the "N1" valve did not stop the hydraulic leak. When the "M1" valve was replaced, the hydraulic system went solid and the hydraulic pump was able to fully pressurize both the active and standby accumulator. The cause of the event was a failure of the "M1" four way valve. When the "M1" four way valve was disassembled, two failed O-rings were found. The failure of these O-rings allowed hydraulic oil to pass through the valve internals back to the reservoir. This prevented the hydraulic system from building required pressure.

D. SAFETY ANALYSIS:

Although neither the active or standby accumulator were available to close the 1A MSIV within the required 5 seconds, the hydraulic pump was available and was used to close the valve. Even if the valve had totally failed in the open position, Technical Specification 3/4.7.1.5 could have been complied with and the plant would have been brought into Hot Shutdown within ten hours in a safe and controlled manner. The worst case scenario would have been a main steam line break during Hot Shutdown at zero power which has been analyzed in the FSAR as a uncontrolled cooldown of one steam generator.

E. CORRECTIVE ACTIONS:

The immediate corrective action involved reduction of power and isolation of the 1A MSIV. Nuclear work Request (NWR) A26531 was written to replace the defective four way hydraulic valve ("M1"). Subsequent testing restored both trains of the MSIV to an operable status.

Long term corrective action is being pursued by Pressurized Water Reactor Engineering (PWRE). They will be utilizing the services of a third party expert to review the operating history of the MSIV's as well as the identified root cause for this failure. Based on this review, PWRE will be providing recommendations for long term corrective action. This review will be tracked to completion by Action Item 456-200-88-25501.

F. PREVIOUS OCCURRENCES:

There have been no previous occurrences of MSIV failure due to O-ring failures.

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	MFG Part Number
Anchor-Darling	Four Way hydraulic directional slide valve	N/A	23304

MS02 MSIV BYPASS VALVE FAILURE

TYPE: GENERIC, RV 0-100%

A) 1MS101A
B) 1MS101B
C) 1MS101C
D) 1MS101D

CAUSE: MECHANICAL BINDING

REF: 20E-1-4030 MS06
20E-1-4030 MS07

PLT STA: MAIN STEAM LINE HEATUP

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED VALVE TO FAIL AT THE POSITION DESIRED. INITIALLY THERE ARE NO EFFECTS UNTIL AN AUTOMATIC SIGNAL IS RECEIVED OR AN ATTEMPT TO POSITION THE VALVE MANUALLY IS MADE. THE VALVE DOES NOT REPOSITION ON MALFUNCTION INSERTION. FAILING THE VALVE AT 0% PRIOR TO A STEAM LINE HEATUP WILL NOT ALLOW THAT LINE TO BE HEATED UP. FAILING THE VALVE OPEN (100%) AFTER THE WARMUP HAS BEGUN WILL CAUSE AN EXCESSIVE HEATUP OF THE PIPING, AND COOLING OF THE S/G.

IF THE DIFFERENTIAL PRESSURE ACROSS THE MSIV IS >50 PSID AND THE BYPASS VALVE IS FAILED SHUT THE MSIV CANNOT BE OPENED.

MALFUNCTION REMOVAL RESTORES THE AFFECTED BYPASS VALVE TO NORMAL.

EVENTS: NONE

MS03 STEAM GENERATOR SAFETY VALVE FAILURE

TYPE: GENERIC, RV 0-100%

A)	1MS013A	K)	1MS015C
B)	1MS013B	L)	1MS015D
C)	1MS013C	M)	1MS016A
D)	1MS013D	N)	1MS016B
E)	1MS014A	O)	1MS016C
F)	1MS014B	P)	1MS016D
G)	1MS014C	Q)	1MS017A
H)	1MS014D	R)	1MS017B
I)	1MS015A	S)	1MS017C
J)	1MS015B	T)	1MS017D

CAUSE: VALVE FAILURE

REF: M-35 SHEET 1
M-35 SHEET 2
M-2035 SHEET 3

PLT STA: 100% REACTOR POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED STEAM GENERATOR SAFETY VALVE WILL OPEN. THE LEAK RATE WILL BE DETERMINED BY THE SELECTED SEVERITY. THE MAIN STEAM LINE FLOWS WILL INCREASE BY THE AMOUNT OF STEAM BEING LEAKED TO THE ATMOSPHERE. ANNUNCIATOR 15-E2 "MS PRESS LOW" WILL ACTUATE IF STEAMLINE PRESSURE DECREASES SUFFICIENTLY. AFTER SWELLING SLIGHTLY, THE ASSOCIATED STEAM GENERATOR LEVEL WILL DECREASE SLIGHTLY, WITH A CORRESPONDING DECREASE IN THE COLD LEG TEMPERATURE, DUE TO THE ADDITIONAL STEAM REMOVAL, WHICH WILL ALSO CAUSE STEAM GENERATOR PRESSURES TO DECREASE.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED STEAM GENERATOR SAFETY VALVE TO NORMAL.

EVENTS: NONE

MS04 S/G PORV CONTROLLER FAILURE

TYPE: GENERIC, RV 0-100%

A) 1MS018A
B) 1MS018B
C) 1MS018C
D) 1MS018D

CAUSE: COMPARISON CIRCUIT FAILURE

REF: M-2035 SHEET 2
20E-1-4030 MS39
20E-1-4030 MS40
20E-1-4030 MS41
20E-1-4030 MS42

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE AFFECTED S/G PORV TO FAIL AT THE SELECTED SEVERITY. FAILING THE COMPARISON CIRCUIT AT 100% CAUSES THE ASSOCIATED VALVE TO OPEN FULLY INCREASING STEAM FLOW AND DECREASING S/G PRESSURE. FAILING THE COMPARISON CIRCUIT AT 0% CAUSES THE VALVE TO FAIL CLOSED.

THE CONTROLLER ON 1PM04J WILL NOT FUNCTION. THE S/G PORV MAY BE CLOSED BY PLACING ITS CONTROL SWITCH TO CLOSE. ACTUAL VALVE POSITION IS INDICATED BY THE VALVE POSITION LVDT METER AND INDICATING LIGHTS.

MALFUNCTION REMOVAL RESTORES THE AFFECTED S/G PORV COMPARISON CIRCUIT TO NORMAL.

EVENTS: 1) DVR 06-02-90-008
2) DVR 20-01-89-080
3) DVR 20-02-88-157
4) DVR 06-01-88-074
5) SER 29-86

DEVIATION INVESTIGATION REPORT (DIR)

Facility Name	Form Rev 2.0
Byron Nuclear Power Station	PAGE 1 OF 0 3

Title

Spurious Opening of 2A Steam Generator PORV Due to Failure of Linear Variable Differential Transformer

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	1
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	8 7
0 2	1 7	9 0	0 6	0 2	9 0	— 0 0 8 —	0 0	0 4	0 3	9 0		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
T. Tulon, Assistant Superintendent of Maintenance Ext. 2221	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	S B	X F M R	B 3 5 0	Y					

SUPPLEMENTAL REPORT EXPECTED		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 02/17/90 / 1442

Unit 2 MODE 1 - Power Operations Rx Power 87% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the start of this event that contributed to this event. At 1442, on February 17, 1990, the 2A Steam Generator Power Operated Relief Valve (PORV), 2MS018A, control switch was placed in the closed position after a spurious opening of the PORV (MS) [SB]. Limiting Condition for Operation Action Requirement (LCOAR) 1BOS 6.3-1a was entered and the manual isolation valve for the PORV, 2MS019A, was closed. Nuclear Work Request (NWR) B74034 was written to resolve the spurios opening.

No manual or safety system actuations occurred. Stable plant conditions were maintained throughout this event. All operator actions were correct.

C. CAUSE OF EVENT:

A strip chart recorder was connected to various loop components. The root cause of this event was determined to be the failure of the linear variable differential transformer (LVDT). The root cause of the LVDT failure is indeterminate, but is believed to be age related. After the LVDT replacement, the recorder continued to monitor the loop and no further abnormalities were noted.

D. SAFETY ANALYSIS:

There were no safety consequences of this event as the PORV was isolated when the spurious opening was noticed. There were no radiological consequences of this event as there is no evidence of any steam generator tube leakage which would result in potential radioactive release to the environment. Under a more severe set of circumstances, a radioactive release could have occurred.

E. CORRECTIVE ACTIONS:

The LVDT for the 2A PORV was replaced. No further corrective actions are necessary.

F. RECURRING EVENTS SEARCH AND ANALYSIS:

a) EVENT SEARCH (DIR, LER)

There are no previous DVRs due to LVDT failures on PORVs. There have been previous spurious PORV openings as documented in the following DVRs.

6-1-87-152 1MS018B Inoperable Due to Loop Power Supply Card Failure.

6-1-88-074 1MS018D Inoperable Due to Pressure Switch Failure.

6-2-89-071 2MSD018A Inoperable Due to Positioner Leakage and Setpoint Drift.

b) INDUSTRY SEARCH (OPEX's NPRDS)

Numerous PORV failures were identified by NPRDS. A common mode failure mechanism was not identified for LVDT failures.

c) NWR

NWRs have been written for various PORV problems including failures of an LVDT.

d) ANALYSIS

No adverse trend identified.

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Borg Warner	Relief Valve	86816	

H. OTHER RELATED DOCUMENTS:

Industry documentation did not apply to this type of failure.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

I. EFFECTIVENESS REVIEW:

None scheduled

J. ADDITIONAL DATA:

a) Affected Technical Specification: 3/4.6.3

b) Procedures: None

c) Cause Code: XPEUK

d) Equipment Involved: Steam generator power operated relief valve.

e) Other: Linear variable differential transformer.

(0556R/0068R)

M504

DEVIATION INVESTIGATION REPORT (DIR)

Facility Name: Braidwood 1

Form Rev 2.0

PAGE 1 OF 0 1

Title: 1MS018C, S/G PORV Erratic Operation

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	POWER LEVEL
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR		
0 5	2 5	8 9	2 0	0 1	8 9	— 0 8 0	— 0 0	0 5	3 1	8 9	1	0 9 7

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
Dan Stroh, Tech Staff Engineer Ext. 2477	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. DESCRIPTION OF EVENT:

There were no structures, systems, or components inoperable or degraded at the beginning of the event that contributed to the event.

At 1230 on May 25, 1989 an operator noted steam escaping from the vent stack discharge from Power Operated Relief Valve (PORV) 1MS018C. Main control board indication verified that the valve was oscillating off of it's seat. The operating staff declared the 1MS018C PORV inoperable, entered Limiting Condition for Operation Action Requirement (LCOAR) 6.3-1a. and isolated the PORV with the manual upstream isolation valve 1MS019C. Nuclear Work Request (NWR) A31586 was written to investigate, troubleshoot and repair as necessary.

B. CAUSE OF EVENT:

Air in the hydraulic system was suspected as the cause of the pump cycling. Mechanical Maintenance (MMD) executed the fill and bleed surveillance, however, the problem was not resolved. The instrument maintenance department then calibration checked the pressure switches and electronic control modules. All components were found within station tolerance. Presently, MMD is still troubleshooting the valve actuator and results of this investigation will be tracked and documented on Action Item Report (AIR) #456-200-89-08001.

C. CORRECTIVE ACTIONS:

Immediate corrective action consisted of isolating the PORV and entering LCOPR 6.3-1a. NWR #A31586 was written to troubleshoot and repair. MMD fill and bleed procedure was utilized to remove air from the hydraulic system.

MS04

DEVIATION INVESTIGATION REPORT

TITLE 1D STEAM GENERATOR PORV 1MS018D INOPERABLE

PAGE 1 OF 0 1

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	POWER LEVEL
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR		
0 5	1 1	8 8	0 6	0 1	8 8	0 7 4	0 0	0 6	2 1	8 8	1	0 9 8

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER (AREA CODE)
Alex Javorik, Assistant Tech Staff Supervisor Ext. 2106	8 1 5 2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	M S	P S	G 0 8 2	N					

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 5/11/88 / 1130

Unit 1 MODE 1 - Power Operation Rx Power 98% RCS [AB] Temperature/Pressure Normal Operating

Unit 2 MODE 2 - Power Operation Rx Power 89% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On May 11, 1988, while operating at 98% reactor power, a Byron Station Equipment Operator performing rounds noted steam relieving from the 1D Main Steam (MS) [SB] Power Operated Relief Valve (PORV). The Unit 1 Operator (licensed) was immediately informed of this anomaly and the 1D PORV Manual/Automatic Control Station was placed in the Manual Closed position. With the valve handswitch now in the Closed position, the steam flow coming from the PORV outlet was noted by visual observation to have stopped completely.

Limiting Condition for Operation Action Requirement (LCOAR) 1BOS 6.3-1A was entered by the Operating Department personnel and at 1155 hours on May 11, 1988, the 1MS018D was isolated by closing the upstream manual isolation valve.

Nuclear Work Request (NWR) number B55825 was generated to investigate and repair the valve's automatic controls and Instrument Maintenance Department personnel were dispatched to investigate and repair the system. The "drift open" problem being experienced on the PORV was determined to be the result of a newly installed, defective, "Close Pressure Switch, PS-1". The function of this component is to ensure sufficient system hydraulic pressure so as to maintain the PORV in its fully closed position. However, with the failure of this pressure switch, coupled with the slight inherent internal leakage of associated positioner components, the system pressure bled down, over time, and was never restored by the hydraulic pump, thus resulting in the valve drifting partially open.

DEVIATION INVESTIGATION REPORT (DIR) MS04 Form Rev 2.0

Facility Name	PAGE
Braidwood 2	1 OF 0 3

Title Spurias Lifting of 2MS018A (2A Steam Generator Power Operated Relief Valve) due to faulty pressure switch and leaky nitrogen fill valve.

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR		3
0 9	2 7	8 8	2 0	0 2	8 8	1 5 7	0 0	1 1	0 9	8 8	POWER LEVEL	0 0 0

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
Dan Stroh, Technical Staff Engineer Ext. 2477	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
C	S B	I M V	B 3 5 0	Y					

SUPPLEMENTAL REPORT EXPECTED		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. Plant Conditions Prior to Event:

Unit: Braidwood 2; Event Date: 9/27/88; Event Time: 0456

Mode: 1 - Hot Standby; Rx Power: 0%

RCS[AB] Temperature/Pressure: 557°C/2238 psig

B. Description of Event:

On September 27, 1988, the shift foreman and a B man were performing routine inspections in the main steamline isolation valve room (MSIV room). For no apparent reason the 2A Steam Generator Power Operated Relief Valve (PORV) lifted momentarily. The Unit 2 Steam Generator pressure was stable at 1085 psi and Reactor Coolant System Temperature and Pressure had been stable for several hours. The 2A PORV was isolated and declared inoperable and Limiting Condition for Operation Action Requirement (LCOAR) 6.3-1A was entered. Nuclear Work Request (NWR) A25840 was written to troubleshoot and repair the problem. Steam pressure remained stable througnout the event and the Manual Isolation Valve Upstream of the PORV was closed to isolate the inoperable PORV.

DEVIATION INVESTIGATION REPORT TEXT CONTINUATION

Form Rev 2.0

FACILITY NAME	DIR NUMBER					PAGE
	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Braidwood 2	2 0	0 2	8 8	— 1 5 7 —	0 0	2 OF 0 3

TEXT

C. Cause of Event:

The cause of the event can be traced back to Friday 9-23-88. Alarm 2-15-A10, "2A PORV TROUBLE" came up and LCOAR 6.3-1A was entered. NWR #A25809 was written to investigate the cause. Because the 2A PORV was needed for the 100% unit trip test Friday night, the Startup Test Engineer (STE) was asked to troubleshoot the valve. He found both nitrogen pressure and hudrailic oil level good. He concluded that the nitrogen low pressure switch was faulty and giving the alarm. A meeting was held between representatives of the quality control (QC) department, operating (ops) department and the startup group to determine a way to repair the valve before the scheduled trip on Friday night. It was agreed that an operability surveillance would be run and if successful, the valve would be placed back in service even though the alarm was still up. The surveillance was successfully performed, NWR #A25809 was closed out, and LCOAR 6.3-1A was exited at 1323 on 9-23-88.

During the 100% trip test, the 2A PORV did lift as required and alarm 2-15-A10 "PORV TROUBLE" reset and stayed clear. The 2A PORV was as a result, left in service and LCOAR 6.3-1A was not re-entered.

On Tuesday 9-27-88 at 0456, 2A PORV lifted, annunciator 2-15-A10 came in, and LCOAR 6.3-1A was entered. NWR #A25840 was written to investigate.

Upon troubleshooting, it was found that the pressure switch was faulty and there was a nitrogen leak at the pneumatic fill valve. Apparently between Saturday 9-24-88 and Tuesday 9-27-88, the system nitrogen had bled out and the pressure switch had not annunciated. On Tuesday, the PORV lifted momentarily or puffed because of this low nitrogen pressure condition, and was not recognized because of the faulty pressure switch. Root cause of event was a faulty pressure switch in combination with a leaking nitrogen fill valve.

D. Safety Analysis:

LCOAR 6.3-1A was entered in a timely manner after the trouble alarm came in on both Friday 9-23-88 and Tuesday 9-27-88. However, when the valve was declared operable on Friday, routine surveillances should have been performed on the valve to verify proper pressure while the pressure switch was inoperable. If the valve had been opened anytime between Saturday and Tuesday and had received an emergency closure signal, there would not have been enough nitrogen pressure in the accumulator to quick close the valve. Because the hydraulic system was still fully operable at all times, the valve could have been slow closed using the manual/auto station in the main control room. The additional time required to close the valve using the normal modulation controller would not have any serious impact on steam line pressure and no unusual safety concerns would have resulted from either the failed pressure switch or leaking fill valve. Even if the hydraulic system had failed, the PORV could have been manually positioned using a locally mounted hand pump which is connected to the PORV actuator. Also the 2A PORV could have been manually isolated at any time.

E. Corrective Action:

Immediate corrective action consisted of isolating the 2A PORV and entering LCOAR 6.3-1A. NWR #A25840 was written to troubleshoot and repair the leaking pneumatic fill valve. NWR #A25874 was written to recal/repair the nitrogen pressure switch.

F. Previous Occurrence:

None

G. Component Failure Data:

Manufacturer	Nomenclature	Model Number	Part Number
Borg Warner	Schrader Fill Valve	N/A	38938000

IS 621 FORSYTH (INPO) 12-AUG-86 08:24 PT
Subject: SER 29-86, RAPID COOLDOWN AND DEPRESSURIZATION

MS04

SUBJECT: INADVERTENT RAPID COOLDOWN AND DEPRESSURIZATION DURING A REMOTE SHUTDOWN TEST

UNIT (TYPE)] CATAWBA 2
DOC NO/LER NO: 50-414/86028
EVENT DATE: 6/27/86
NSSS/AE: WESTINGHOUSE/DUKE POWER COMPANY

SUMMARY:

DURING THE PERFORMANCE OF A REQUIRED POWER ASCENSION TEST (LOSS OF CONTROL ROOM), THE PRIMARY SYSTEM EXPERIENCED A RAPID COOLDOWN AND DEPRESSURIZATION. AS PART OF THE TEST, THE UNIT WAS TRIPPED AT 24 PERCENT POWER, AND CONTROL WAS TRANSFERRED TO THREE REMOTE SHUTDOWN PANELS. WHEN CONTROL WAS TRANSFERRED, ALL FOUR STEAM GENERATOR POWER-OPERATED RELIEF VALVES (PORVs) OPENED, CAUSING A RAPID DECREASE IN PRIMARY SYSTEM TEMPERATURE AND PRESSURE. AS A RESULT OF THE TEMPERATURE DECREASE, PRESSURIZER LEVEL INDICATION WENT OFFSCALE LOW. WHEN THE TEST WAS TERMINATED AND CONTROL WAS TRANSFERRED BACK TO THE MAIN CONTROL ROOM, AN AUTOMATIC SAFETY INJECTION OCCURRED (PER DESIGN) ON LOW STEAM LINE PRESSURE.

THE INITIATING CAUSE OF THIS EVENT WAS INADEQUATE IMPLEMENTATION OF A DESIGN MODIFICATION TO THE STEAM GENERATOR PORV CONTROLLERS. AN EQUIPMENT MALFUNCTION, IMPROPER LABELING ON THE REMOTE SHUTDOWN PANELS, AND LACK OF EXPLICIT TEST TERMINATION CRITERIA CONTRIBUTED TO THE EXTENT AND DURATION OF THE EVq -

THIS EVENT IS SIGNIFICANT BECAUSE INADEQUATE DESIGN REVIEW AND CONTROL CREATED A PROBLEM FOR OPERATORS AND IF THE SITUATION HAD REQUIRED AN ACTUAL CONTROL ROOM EVACUATION, EXISTING CONDITIONS COULD HAVE PRECLUDED A SAFE AND ORDERLY UNIT SHUTDOWN.

DESCRIPTION:

ON 6/27/86, CATAWBA UNIT 2 WAS OPERATING AT 24 PERCENT POWER, AND PREPARATIONS WERE UNDERWAY TO PERFORM A LOSS OF CONTROL ROOM TEST. THIS TEST WAS INTENDED TO VERIFY THE ABILITY TO SHUT DOWN THE PLANT FROM OUTSIDE THE CONTROL ROOM. THE OPERATING SHIFT CONDUCTED A PRETEST BRIEFING AND PROCEDURE WALK-DOWN ON THE PREVIOUS AFTERNOON. NO PROBLEMS WERE IDENTIFIED. THE TEST PROVIDED FOR THE NORMAL OPERATING SHIFT TO CONDUCT THE TEST. A MINIMUM NUMBER OF OBSERVERS WERE TO REMAIN IN THE CONTROL ROOM TO MONITOR THE OPERATION OF THE REACTOR COOLANT PUMPS. THESE PUMPS WERE TO REMAIN IN OPERATION TO SIMULATE DECAY HEAT. THE CONTROL ROOM OBSERVERS WERE A SENIOR REACTOR OPERATOR AND A LICENSED OPERATOR WHO WERE TO MAINTAIN COMMUNICATIONS WITH THE OPERATING SHIFT. HOWEVER, THEY WERE ONLY TO COMMUNICATE INFORMATION PERTAINING TO THE EQUIPMENT INTENTIONALLY LEFT OPERATING AFTER TEST INITIATION.

AN EQUIPMENT MODIFICATION HAD BEEN IMPLEMENTED BETWEEN HOT FUNCTIONAL TESTING, WHEN THIS TEST HAD PREVIOUSLY BEEN PERFORMED, AND UNIT 2 LICENSING. THIS MODIFICATION CHANGED THE FUNCTIONAL CHARACTERISTICS BUT NOT THE PHYSICAL APPEARANCE OF THE STEAM GENERATOR PORV CONTROLS ON THE AUXILIARY FEEDWATER PUMP TURBINE CONTROL PANEL. PRIOR TO THE MODIFICATION, THE CONTROLLERS FUNCTIONED AS STEAM GENERATOR PRESSURE SET POINT CONTROLLERS. IN THIS MODE, THE CONTROLLERS PROVIDED MANUAL ADJUSTMENT OF THE PRESSURE SET POINT AT WHICH THE PORVs WOULD OPEN. THE CONTROLLERS HAD A SINGLE SCALE READING IN UNITS OF PSIG (FULL SCALE BEING 1500 PSIG) AND DUAL POINTERS; ONE POINTER INDICATED STEAM GENERATOR PRESSURE AND THE SECOND INDICATED PRESSURE SET POINT. AFTER THE MODIFICATION, THE CONTROLLERS FUNCTIONED AS DIRECT MANUAL STEAM GENERATOR PORV POSITION DEMAND LOADERS, AND THE SECOND POINTER INDICATED VALVE POSITION DEMAND. HOWEVER, THE SCALE STILL READ IN PSIG UNITS RATHER THAN PERCENT OF VALVE DEMAND. ACCORDING TO THE TEST PROCEDURE, WHICH WAS IN ERROR, THE CONTROLLERS WERE SET AT WHAT WAS BELIEVED TO BE A STEAM GENERATOR PRESSURE SET POINT OF 1125 PSIG. IN REALITY, THIS SETTING PROVIDED A 75 PERCENT OPEN DEMAND SIGNAL TO jHE FOUR STEAM GENERATOR PORVs.

A NORMAL SHIFT TURNOVER OCCURRED AT 0700 ON 6/27/86, AND PREREQUISITES FOR THE TEST WERE COMPLETED BETWEEN 0800 AND 0900. THE TEST WAS INITIATED AT 0941 WHEN THE OPERATIONS PERSONNEL WERE DISPATCHED FROM THE CONTROL ROOM TO THEIR ASSIGNED STATIONS. AT THAT TIME, THE PRIMARY PRESSURE WAS 2238 PSIG, TEMPERATURE WAS 560 DEGREES FAHRENHEIT, AND PRESSURIZER LEVEL WAS 28 PERCENT. THE STEAM GENERATOR PRESSURE WAS 1030 PSIG. AT 0942, A LICENSED REACTOR OPERATOR TRIPPED THE REACTOR TRIP BREAKERS IN ACCORDANCE WITH THE TEST PROCEDURE. HE THEN PROCEEDED TO THE AUXILIARY FEEDWATER PUMP TURBINE CONTROL PANEL. THIS PANEL IS ONE OF THE THREE AUXILIARY SHUTDOWN PANELS FROM WHICH REMOTE SHUTDOWN IS PERFORMED. ALSO AT 0942, LOCAL CONTROL WAS TAKEN AT AUXILIARY SHUTDOWN PANELS A AND B. THESE TWO PANELS HAVE INDICATIONS AND CONTROLS FOR FUNCTIONS SUCH AS LETDOWN/CHARGING AND SEAL INJECTION. WHEN THIS TRANSFER OF CONTROL WAS PERFORMED, A LETDOWN/CHARGING FLOW MISMATCH OCCURRED. THIS LETDOWN/CHARGING MISMATCH, WHICH WAS GREATER THAN ANTICIPATED, RESULTED IN AN INCREASING VOLUME CONTROL TANK LEVEL AND A DECREASING PRESSURIZER LEVEL. BY 0947, THE PRESSURIZER LEVEL HAD DROPPED TO AN INDICATED 18 PERCENT.

AT 0943, THE LICENSED OPERATOR WHO HAD TRIPPED THE REACTOR TRIP BREAKERS ARRIVED AT THE AUXILIARY FEEDWATER PUMP TURBINE CONTROL PANEL, AND TOOK CONTROL. THIS PANEL HAS INDICATION AND CONTROLS ASSOCIATED WITH THE STEAM GENERATOR PORVs AND OTHER FUNCTIONS. WHEN THE LOCAL POWER FEEDER BREAKERS FOR THE STEAM GENERATOR PORVs WERE CLOSED AT APPROXIMATELY 0947, ALL FOUR STEAM GENERATOR PORVs OPENED TO 75 PERCENT FULL OPEN, THE PERCENTAGE OF FULL SCALE AT WHICH THE CONTROLLERS HAD BEEN SET.

THE OPERATOR AT THE AUXILIARY FEEDWATER PUMP TURBINE CONTROL PANEL IMMEDIATELY BECAME AWARE OF THE SUDDEN DECREASE IN STEAM GENERATOR PRESSURE AND, IN AN ATTEMPT TO ENSURE THE PORVs WERE CLOSED, MANUALLY ADJUSTED (WHAT HE THOUGHT TO BE) THE PRESSURE SET POINT UPWARD. THIS ACTION ACTUALLY CAUSED THE PORVs TO OPEN EVEN MORE. NO DIRECT INDICATION WAS AVAILABLE TO THE AUXILIARY FEEDWATER PUMP TURBINE CONTROL PANEL OPERATOR OF VALVE POSITION FOR THE STEAM GENERATOR PORVs. THE CONTROL ROOM OBSERVERS DID HAVE POSITION INDICATION IN THE FORM OF RED AND GREEN LIGHTS; HOWEVER, THEY WERE RELUCTANT TO COMMUNICATE SUCH INFORMATION TO OPERATORS ON THE REMOTE SHUTDOWN PANELS TO AVOeD INVALIDATING THE TEST. AT 0950, THE PRESSURIZER PRESSURE HAD DECREASED TO 1845 PSIG AND STEAM LINE PRESSURE TO 725 PSIG, THE SAFETY INJECTION SET POINT. HOWEVER, AUTOMATIC SAFETY INJECTION WAS PARTIALLY BLOCKED (PER DESIGN) BY TRANSFER TO THE REMOTE SHUTDOWN PANELS. AT 0952, THE SRO IN THE CONTROL ROOM ORDERED TEST TERMINATION AND TRANSFER OF CONTROL BACK TO THE MAIN CONTROL ROOM. UPON TRANSFER, AT 0953, AUTOMATIC SAFETY INJECTION ACTUATION WAS UNBLOCKED AND OCCURRED, THE STEAM GENERATOR PORVs CLOSED, AND THE PRESSURIZER PRESSURE AND LEVEL BEGAN RECOVERING. BY 0958, THE PRESSURIZER LEVEL AND PRESSURE HAD RETURNED TO APPROXIMATELY 30 PERCENT AND 1300 PSIG, RESPECTIVELY. AT THIS POINT, THE SAFETY INJECTION WAS RESET AND REACTOR COOLANT TEMPERATURE WAS STABILIZED AT 468 DEGREES FAHRENHEIT.

SUBSEQUENT INVESTIGATION REVEALED THE FOLLOWING FACTORS THAT CONTRIBUTED TO THE PROGRESSION OF THIS EVENT:

A. THE DESIGN MODIFICATION TO THE STEAM GENERATOR PORV CONTROL SCHEME DID NOT ADEQUATELY ADDRESS THE CHANGES NEEDED TO THE AUXILIARY FEEDWATER PUMP TURBINE CONTROL PANEL. FIGURE 2 SHOWS A SIMPLIFIED CONTROL DIAGRAM, BEFORE AND AFTER THE MODIFICATION. THE MODIFICATION REPLACED THE STEAM GENERATOR PORV PRESSURE SET POINT LOADER IN THE CONTROL ROOM WITH A VALVE POSITION DEMAND LOADER AND REMOVED THE PROPORTIONAL CONTROLLER THAT WAS COMMON TO CONTROL ROOM AND AUXILIARY FEEDWATER PUMP TURBINE CONTROL PANEL OPERATION. THIS RESULTED IN THE PRESSURE SET POINT LOADER, LOCATED ON THE AUXILIARY FEEDWATER PUMP TURBINE CONTROL PANEL, FUNCTIONING AS A VALVE POSITION DEMAND LOADER WHEN CONTROL WAS TRANSFERRED TO THIS PANEL. HOWEVER, THE AUXILIARY FEEDWATER PUMP TURBINE CONTROL PANEL CONTROLLER WAS NOT REPLACED OR RELABELLED. THE NEED FOR RELABELLING WAS NOT IDENTIFIED DURING THE DESIGN MODIFICATION PROCESS.

B. THE MODIFICATION WAS MADE KNOWN TO THE PLANT STAFF, AND INPUT FROM THE APPROPRIATE DESIGN PERSONNEL WAS SOLICITED BY THE PLANT TO BE USED FOR ANY REQUIRED PROCEDURAL AND TRAINING REVISIONS. HOWEVER, THERE WAS INADEQUATE TRANSFER OF INFORMATION CONCERNING THE CHANGE IN FUNCTION OF THE AUXILIARY FEEDWATER PUMP TURBINE CONTROL PANEL CONTROLLER. BECAUSE THE APPEARANCE OF THE AUXILIARY CONTROLLER HAD NOT BEEN ALTERED, THE PLANT HAD THE IMPRESSION THAT THE FUNCTION OF THE CONjROLLER HAD ALSO REMAINED UNCHANGED. THIS RESULTED IN PROCEDURAL REVISIONS AND OPERATOR TRAINING THAT INCORPORATED THE CONTROL ROOM CONTROLLER PORTION OF THE MODIFICATION IMPACT ON SYSTEM OPERATION, BUT IT DID NOT INCLUDE THE EFFECT ON THE AUXILIARY FEEDWATER PUMP TURBINE CONTROL PANEL CONTROLLER.

C. THE LETDOWN/CHARGING MISMATCH THAT OCCURRED AT THE BEGINNING OF THE EVENT WAS CAUSED WHEN LETDOWN PRESSURE CONTROL VALVE 148, DESIGNED TO PROVIDE BACKPRESSURE FOR THE FLOW ORIFICES IN THE LETDOWN LINE, FAILED OPEN DUE TO A FAULTY ELECTRICAL CONNECTION (SEE FIGURE 1). THE SITUATION WAS COMPLICATED BY THE FOLLOWING:

PRIOR TO TEST INITIATION, CHARGING FLOW CONTROL VALVE 294 HAD BEEN ADJUSTED TO 32 GPM, AND SEAL INJECTION BACKPRESSURE CONTROL VALVE 309 WAS CLOSED BY ADJUSTMENT OF AUXILIARY SHUTDOWN PANEL "A" CONTROLLERS TO LIMIT THE TRANSIENT EFFECT ON THE REACTOR COOLANT PUMP (RCP) SEALS DURING THE TRANSFER. THE CONTROLLER FOR VALVE 309 ON AUXILIARY SHUTDOWN PANEL "B" WAS LEFT AT ITS NORMAL OPEN SETTING. HOWEVER, UNKNOWN TO THE OPERATOR, VALVE 309 WOULD RESPOND TO THE OPEN

UPON TRANSFER. THIS RESULTED IN THE VALVE GOING TO AN OPEN POSITION, INSTEAD OF THE FULL-CLOSED POSITION THAT WAS DESIRED UPON TRANSFER TO THE AUXILIARY SHUTDOWN PANELS.

WHEN VALVE 309 OPENED, THE OPERATOR ATTEMPTED TO COMPENSATE FOR THE LACK OF FLOW TO SEAL INJECTION BY OPENING VALVE 294; HOWEVER, THE MANUAL VALVE POSITION DEMAND CONTROLLER FOR VALVE 294 ON THE AUXILIARY SHUTDOWN PANEL WAS LABELED BACKWARD (INCREASING AND DECREASING DESIGNATIONS WERE TRANSPOSED). THEREFORE, THE OPERATOR'S ATTEMPTS TO OPEN VALVE 294 RESULTED IN CLOSING IT.

THE PLANT MADE THE NECESSARY EQUIPMENT LABELLING AND PROCEDURAL CHANGES AND CONDUCTED TRAINING TO INCORPORATE THE IMPACT OF THIS MODIFICATION COMPLETELEY. THE LOSS-OF-CONTROL-ROOM TEST WAS SUCCESSFULLY RE-CONDUCTED ON 7/11/86. THE PLANT IS IN THE PROCESS OF REVIEWING ALL MODIFICATIONS PERFORMED BETWEEN COMPLETION OF HOT FUNCTIONAL TESTING AND LICENSING TO DETERMINE IF ANY SIMILAR SITUATIONS EXIST.

COMMENTS:

1. AN ESSENTIAL ELEMENT OF THE DESIGN MODIFICATION PROCESS IS THE DEVELOPMENT AND DOCUMENTATION OF A BASIC FUNCTION DESCRIPTION. THIS WOULD COMPLEMENT TECHNICAL AND HARDWARE DESCRIPTIONS OF HOW TOTAL SYSTEM OPERATION IS AFFECTED BY A MODIFICATION. ONLY WITH SUCH A FUNCTIONAL DESCRIPTION CAN THE SAFETY IMPACT OF A MODIFICATION BE COMPLETELY EVALUATED. THIS DESCRIPTION SHOULD IDENTIFY NECESSARY REVISIONS TO OPERATING PROCEDURES, MAINTENANCE PROCEDURES, TESTING REQUIREMENTS, AND TRAINING LESSON PLANS, AND SHOULD BE INCLUDED AS PART OF THE MODIFICATION PACKAGE DOCUMENTATION. PROCEDURE CHANGES, DRAWING UPDATES, AND APPROPRIATE TRAINING SHOULD BE COMPLETED PRIOR TO RETURNING A MODIFIED SYSTEM TO SERVICE.

2. FOLLOWING THE MODIFICATION TO THE STEAM GENERATOR PORV CONTROLS, INDIVIDUAL COMPONENT CHECKS WERE MADE, BUT COMPLETE LOOP/SYSTEM TESTING WAS NOT PERFORMED. TO BE COMPLETE, POST-MODIFICATION TESTING SHOULD NOT ONLY VERIFY INDIVIDUAL COMPONENT OPERATION, BUT WHERE APPROPRIATE, SHOULD ALSO VERIFY SYSTEM FUNCTIONAL OPERATION. ADDITIONALLY, APPROPRIATE COMPONENT/SYSTEM OPERABILITY SHOULD BE VERIFIED PRIOR TO CRITICAL TESTS.

3. WHEN APPLICABLE, CRITICAL TEST PROCEDURES SHOULD PROVIDE SPECIFIC CRITERIA FOR TEST TERMINATION AND SPECIFIC STEPS TO ENSURE TERMINATION IS CONDUCTED IN A SAFE AND ORDERLY MANNER. DURING THE CONDUCT OF THE LOSS OF CONTROL ROOM TEST, EXPLICIT TEST TERMINATION CRITERIA WERE NOT GIVEN TO THE SRO OBSERVER IN THE CONTROL ROOM. THIS MAY HAVE RESULTED IN INCREASING THE EXTENT AND DURATION OF THE TRANSIENT.

4. IT IS IMPORTANT THAT THE APPROPRIATE PLANT PERSONNEL ARE WELL-TRAINED, PRACTICED, AND HAVE A COMPLETE UNDERSTANDING OF THE PROCESS INVOLVED WITH THE EVACUATION OF THE MAIN CONTROL ROOM. AREAS OF IMPORTANCE INCLUDE INDICATIONS AND CONTROLS AVAILABLE AT THE REMOTE SHUTDOWN PANELS AND THE DIFFERENCES BETWEEN A CONTROL ROOM SHUTDOWN AND A REMOTE SHUTDOWN. THIS IS PARTICULARLY IMPORTANT WITH RESPECT TO CONTROLLING THE PLANT UNDER ABNORMAL CONDITIONS.

5. THE HUMAN PERFORMANCE PROBLEMS THAT OCCURRED DURING THIS EVENT HIGHLIGHT THE IMPORTANCE OF THE APPLICATION OF HUMAN FACTORS CONSIDERATIONS TO ALL PANELS IN THE PLANT, NOT JUST THOSE IN THE CONTROL ROOM. THIS IS PARTICULARLY IMPORTANT FOR PANELS USED DURING INFREQUENT OR OFF-NORMAL CONDITIONS SUCH AS REMOTE SHUTDOWN. APPLICATION OF HuspgPBACTORS SHOULD BE AN INTEGRAL PART OF THE DESIGN MODIFICATION PROCESS FOR ALL CONTROLS AND INDICATIONS.

AS A MINIMUM, THIS SER SHOULD BE REVIEWED BY PLANT ORGANIZATIONS RESPONSIBLE FOR OPERATIONS, TRAINING, INSTRUMENTATION AND CONTROLS, AND THE DESIGN MODIFICATION PROCESS.

ILLUSTRATIONS, WHICH MAY BE HELPFUL IN UNDERSTANDING THIS SER, ARE BEING TRANSMITTED BY TELECOPY TO THE UTILITY AND PARTICIPANT SEE-IN CONTACTS. RECIPIENTS WHO DO NOT HAVE TELECOPY RECEPTION CAPABILITIES AT THEIR LOCATION CAN OBTAIN A COPY OF THE ILLUSTRATIONS FROM THEIR SEE-IN CONTACT OR JEFF WHEELOCK, INPO, 404/951-4730. RECIPIENTS WITH TELECOPY RECEPTION CAPABILITIES WHO EXPERIENCE PROBLEMS IN RECEIVING ANY TRANSMISSION SHOULD CONTACT SKIP HEEKE, INPO, 404/953-7675.

INPO'S EVALUATION OF THIS EVENT IS COMPLETE.

Information Contact: RICHARD H. REYNOLDS, INPO, 404/953-5392

MS05 STUCK STEAM DUMP

TYPE: GENERIC, RV 0-100%

A) 1MS004A
B) 1MS004B
C) 1MS004C
D) 1MS004D
E) 1MS004E
F) 1MS004F
G) 1MS004G
H) 1MS004H
I) 1MS004J
J) 1MS004K
K) 1MS004L
L) 1MS004M

CAUSE: MECHANICAL BINDING

REF: M-35 SHEET 3
M-2035 SHEET 5

PLT STA: PLANT TRIP

EFFECTS: MALFUNCTION INSERTION CAUSES THE SELECTED STM DUMP VALVE TO FAIL TO THE SELECTED POSITION AS THE VALVE PASSES THROUGH THAT POSITION. THIS CONDITION WILL CAUSE THE REMAINING DUMP VALVES TO COMPENSATE FOR THE STUCK DUMP VALVE. AN UNCONTROLLED COOLDOWN OF THE PRIMARY COULD OCCUR RESULTING IN AN INCREASE IN MAIN STEAM FLOW, DECREASE IN TAVE, AND DECREASE IN PZR LEVEL AND TEMPERATURE.

EFFECTS OF THIS MALFUNCTION CAN BE INCREASED BY INSERTING SEVERAL FAILURES AT ONE TIME. THE EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY CLOSING THE AFFECTED DUMP VALVE ISOLATION VALVE.

MALFUNCTION REMOVAL RESTORES THE DUMP VALVE TO NORMAL OPERATION.

EVENTS: NONE

MS06 MSR FAILS TO ISOLATE

TYPE: GENERIC, RB

A) MSR 1A "A" VALVES
B) MSR 1B "B" VALVES
C) MSR 1A "C" VALVES
D) MSR 1B "D" VALVES

CAUSE: REHEAT TEMPERATURE CONTROLLER (RTC) FAILURE

REF: M-35 SHEET 4
MAIN TURBINE AND REHEATERS SYSTEM DESCRIPTION
C&ID M-2035 SHEET 9
20E-1-4030 MS23
20E-1-4030 MS24

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION PREVENTS THE REHEAT TEMPERATURE CONTROLLER FROM SIGNALING THE SELECTED MSR TUBE BUNDLE TO ISOLATE AFTER A TURBINE TRIP. THE MSR REHEAT CONTROL VALVES 1MS010 A-D AND 1MS147 A-D WILL FAIL AS IS AS WILL THE REHEATER STOP VALVES 1MS009 A-D. AN UNCONTROLLED COOLDOWN OF THE PRIMARY COULD OCCUR RESULTING IN AN INCREASE IN MAIN STEAM FLOW, DECREASE IN T_{ave}, AND DECREASE IN PZR LEVEL AND PRESSURE. THE STEAM DUMP VALVES WILL INDICATE A LOWER DEMAND AFTER THE TRIP. ANNUNCIATOR 14-E1 "T_{ave} CONT DEV LOW" WILL ACTUATE WHEN AUCT HIGH T_{ave} IS 3 °F BELOW Tref. AS T_{ave} DECREASES BELOW 550 °F, THE BYPASS PERMISSIVE LIGHT "L0-2 T_{ave} STM DUMP INTLK P12" IS ACTUATED. THIS RESULTS IN A CLOSE SIGNAL TO THE STEAM DUMPS 1MS004A-M. OTHER PLANT ANNUNCIATORS WILL RESPOND ACCURATELY TO THE MALFUNCTION.

EFFECTS OF THIS MALFUNCTION CAN BE INCREASED BY INSERTING SEVERAL FAILURES AT ONE TIME. THE EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY CLOSING THE AFFECTED MSR REHEATER STOP VALVES 1MS009A-D.

MALFUNCTION REMOVAL RESTORES THE REHEAT TEMPERATURE CONTROLLER TO NORMAL OPERATION.

EVENTS: NONE

MS07 STEAMLINE BREAK INSIDE CONTAINMENT

TYPE: GENERIC, NRVI 0-4 MLB/HR @ 975 PSID

A) 1A MS LINE
B) 1B MS LINE
C) 1C MS LINE
D) 1D MS LINE

CAUSE: PIPE BREAK DOWNSTREAM OF FLOW RESTRICTOR

REF: M-35 SHEET 1
M-35 SHEET 2
UFSAR SEC 15.1.3

PLT STA: 100% REACTOR POWER

EFFECTS: THIS MALFUNCTION RESULTS IN A STEAMLINE BREAK RELEASING STEAM TO THE CONTAINMENT ATMOSPHERE. MAIN STEAM FLOW ON ALL S/Gs WILL INCREASE, S/G PRESSURE WILL DECREASE, AND THE AFFECTED S/G LEVEL WILL INITIALLY SWELL THEN DECREASE. THIS CAUSES A DECREASE IN T_{ave}, AND AN INCREASE IN REACTOR POWER.

AS THE SEVERITY IS INCREASED PZR PRESS, PZR LEVEL, S/G PRESSURE AND LEVEL, AND T_{ave} WILL DECREASE AT A RAPID RATE. CONTAINMENT PRESSURE, TEMPERATURE, AND HUMIDITY WILL INCREASE.

THE REACTOR WILL TRIP ON ANY OF THE FOLLOWING PROTECTIVE FUNCTIONS: OTDT, OPDT, L0-2 S/G LEVEL, LOW PZR PRESSURE, OR ANY OF THE FOLLOWING SAFETY INJECTION SIGNALS: LOW PZR PRESSURE, LOW STEAMLINE PRESSURE, OR HIGH CONTAINMENT PRESSURE. MAIN STEAMLINE ISOLATION IS AUTOMATICALLY INITIATED BY A LOW STEAMLINE PRESSURE SI SIGNAL OR CNMT PRESSURE REACHING 8.2 PSIG. HI-3 CONTAINMENT PRESSURE CAUSES A CONTAINMENT SPRAY ACTUATION AND A CONTAINMENT PHASE B ISOLATION. THE AFFECTED S/G CONTINUES TO BLOWDOWN UNTIL EMPTY.

MALFUNCTION SEVERITY CAN ONLY BE INCREASED FOR THIS MALFUNCTION. TO RECOVER FROM THIS MALFUNCTION THE SIMULATOR MUST BE RESET.

EVENTS: NONE

MS08 STEAMLINE BREAK OUTSIDE CONTAINMENT

TYPE: GENERIC, NRVI 0-4 MLB/HR @ 975 PSID

A) 1A MS LINE
B) 1B MS LINE
C) 1C MS LINE
D) 1D MS LINE

CAUSE: PIPE BREAK IMMEDIATELY DOWNSTREAM OF MSIV

REF: M-35 SHEET 1
M-35 SHEET 2

PLT STA: 100% REACTOR POWER

EFFECTS: THIS MALFUNCTION RESULTS IN A STEAMLINE BREAK RELEASING STEAM TO THE STEAM TUNNEL ATMOSPHERE. MAIN STEAM FLOW ON ALL S/Gs WILL INCREASE, S/G PRESSURE WILL DECREASE, AND THE AFFECTED S/G LEVEL WILL SWELL INITIALLY THEN DECREASE. THIS CAUSES A DECREASE IN T_{ave}, AND AN INCREASE IN REACTOR POWER.

AS THE SEVERITY IS INCREASED PZR PRESS, PZR LEVEL, S/G PRESSURE AND LEVEL, AND T_{ave} WILL DECREASE AT A RAPID RATE.

THE REACTOR WILL TRIP ON ANY OF THE FOLLOWING PROTECTIVE FUNCTIONS: OTDT, OPDT, L0-2 S/G LEVEL, LOW PZR PRESSURE, OR ANY OF THE FOLLOWING SAFETY INJECTION SIGNALS: LOW PZR PRESSURE OR LOW STEAMLINE PRESSURE. MAIN STEAMLINE ISOLATION IS AUTOMATICALLY INITIATED BY A LOW STEAMLINE PRESSURE SI SIGNAL. AFTER THE MAIN STEAM LINE ISOLATION, THE S/G PRESSURE AND REACTOR COOLANT COOLDOWN WILL HAVE TO BE CONTROLLED USING THE S/G PORV'S.

MALFUNCTION SEVERITY CAN ONLY BE INCREASED FOR THIS MALFUNCTION. TO RECOVER FROM THIS MALFUNCTION THE SIMULATOR MUST BE RESET.

EVENTS: NONE

MS09 MAIN STEAM HEADER CROSS-TIE RUPTURE

TYPE: DISCRETE, NRVI 0-4 MLB/HR @ 900 PSID

CAUSE: PIPE FAILURE ON CROSSTIE HEADER

REF: M-35 SHEET 1
M-35 SHEET 2
M-35 SHEET 3

PLT STA: HOT STANDBY

EFFECTS: THIS MALFUNCTION RESULTS IN A STEAMLINE BREAK RELEASING STEAM TO THE TURBINE BLDG ATMOSPHERE. MAIN STEAM FLOW FROM ALL THE S/Gs WILL INCREASE, S/G PRESSURE WILL DECREASE, AND THE S/G LEVEL WILL SWELL INITIALLY THEN DECREASE. THIS CAUSES A DECREASE IN T_{ave}, AND AN UNCONTROLLED AND EXCESSIVE REACTOR COOLANT COOLDOWN.

AS THE SEVERITY IS INCREASED PZR PRESS, PZR LEVEL, ALL S/G PRESSURES AND LEVELS, AND T_{ave} WILL DECREASE AT A MORE RAPID RATE. SAFETY INJECTION AND MAIN STEAM LINE ISOLATION SIGNALS WILL ALSO BE GENERATED.

EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY ISOLATING THE MAIN STEAM SYSTEM AND CONTINUING THE PLANT COOLDOWN ON THE S/G PORV'S.

MALFUNCTION SEVERITY CAN ONLY BE INCREASED FOR THIS MALFUNCTION. TO RECOVER FROM THIS MALFUNCTION THE SIMULATOR MUST BE RESET.

EVENTS: NONE

MS10 HEATER 13 EXTRACTION STEAM LINE BREAK

TYPE: GENERIC, RV 0-.16 MLB/HR AT 50 PSID

A) HEATER 13A LINE BREAK
B) HEATER 13B LINE BREAK
C) HEATER 13C LINE BREAK

CAUSE: PIPING FAILURE IMMEDIATELY DOWNSTREAM OF NON-RETURN CHECK VALVES 1ES015A, 1ES015B, AND 1ES015C RESPECTIVELY.

REF: M-38 SHEET 1, 2A, & 2B

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED LP HEATER EXTRACTION LINE WILL LEAK STEAM TO THE TURBINE BUILDING ATMOSPHERE. THE RATE OF STEAM LEAKAGE WILL BE DETERMINED BY THE SELECTED SEVERITY. THE SLIGHT DECREASE IN FEEDWATER TEMPERATURE TO THE STEAM GENERATORS WILL CAUSE A SLIGHT INCREASE IN REACTOR POWER AS THE COOLER COLD LEG COOLANT RETURNS TO THE REACTOR.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY MANUALLY CLOSING THE ASSOCIATED EXTRACTION STEAM ISOLATION VALVE, 1ES013A/B/C.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE PIPING INTEGRITY.

EVENTS: NONE

MS11 LOW PRESS TURBINE INLET PRESS SWITCH FAILURE

TYPE: DISCRETE, RV 0-100% TURBINE POWER

CAUSE: FAULTY PRESSURE SWITCH (1PS-ES080)

REF: 20E-1-4030 ES21
20E-1-4030 MS18
MAIN TURBINE AND REHEATERS SYSTEM DESCRIPTION
C&ID M-2035 SHEET 9

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE OUTPUT OF PRESSURE TRANSMITTER 1ES080 TO RESPOND AS IF IT WERE AT THE SELECTED POWER LEVEL. (i.e. IF A 50% SEVERITY LEVEL IS SELECTED THEN THE TRANSMITTER OUTPUT WILL CORRESPOND TO 50% TURBINE LOAD). AS THE SEVERITY IS INCREASED ABOVE 20% POWER THE MAIN TURBINE DRAINS VALVES 1MS040 A-D AND 1MS045 WILL AUTO CLOSE AS WILL THE CROSSUNDER PIPE DRAIN VALVES TO THE MSRs (1ES034 A&B AND 1ES061A-D). AS SEVERITY IS DECREASED BELOW 20% POWER, THE MAIN TURBINE DRAINS VALVES 1MS040 A-D AND 1MS045 WILL AUTO OPEN AS WILL THE CROSSUNDER PIPE DRAIN VALVES TO THE MSRs (1ES034 A&B AND 1ES061A-D)(ASSUMING THEIR CONTROL SWITCH IS IN AUTO).

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THE MALFUNCTION THROUGH MANUAL OPERATION OF THE DRAIN VALVES.

MALFUNCTION REMOVAL WILL RESTORE THE PRESSURE SWITCH TO NORMAL.

EVENTS: NONE

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

NI01	SR CHANNEL FAILURE
NI02	NOISY SR CHANNEL
NI03	SR CHANNEL HIGH VOLTAGE FAILURE
NI04	FAILURE OF SR HIGH VOLTAGE TO DISCONNECT
NI05	SR DISCRIMINATOR FAILURE
NI06	IR CHANNEL FAILURE
NI07	IR CHANNEL GAMMA COMPENSATION FAILURE
NI08	PR DETECTOR FAILURE
NI09	PR CHANNEL FAILURE
NI10	INCORE MONITORING SYSTEM FAILURE
NI11	STUCK INCORE DETECTOR
NI12	LEAK INTO GUIDE TUBE FOR INCORE DETECTOR

NI01 SR CHANNEL FAILURE

TYPE: GENERIC, RV 0-6 DECADES (CPS)

A) N31
B) N32

CAUSE: DETECTOR FAILURE

REF: SYSTEM DESCRIPTION

PLT STA: PLANT START-UP

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED SOURCE RANGE CHANNEL TO FAIL. THE VALUE OF THE AFFECTED SOURCE RANGE CHANNEL WILL BE DETERMINED BY THE SELECTED SEVERITY AND WILL BE INDICATED ON THE FOLLOWING EQUIPMENT:

- -N31 NEUTRON LEVEL (CPS) ON 1PM07J
- -N32 NEUTRON LEVEL (CPS) ON 1PM07J
- -AUDIBLE COUNT RATE SPEAKER, SELECTED TO AFFECTED CHANNEL ON 1PM07J
- -RATE METER, SELECTED TO AFFECTED CHANNEL, ON 1PM07J
- -SR COUNT RATE, 1NI-31B/32B, ON 1PM05J
- -SR START-UP RATE, 1NI-31D/32D, ON 1PM05J
- -NUCLEAR POWER RECORDER, 1NR-45, ON 1PM05J

ANNUNCIATOR 10-A1 "SR S/D FLUX HIGH" ACTUATES WHEN THE MALFUNCTION SEVERITY IS INCREASED ABOVE THE ALARM SETPOINT. IF THE SELECTED SEVERITY IS INCREASED SO THAT THE APPARENT DETECTOR OUTPUT IS GREATER THAN 10^5 COUNTS, WITHOUT BLOCKING THE HIGH FLUX TRIP, THE REACTOR WILL TRIP. ANNUNCIATOR 11-A2 "SR HIGH FLUX RX TRIP" ACTUATES ON THE TRIP. BDPS WILL ACTUATE IF THE DETECTOR OUTPUT IS DOUBLED WITHIN A 10 MINUTE PERIOD AND NOT BLOCKED. ANNUNCIATORS 10-E3 "BDPS FLUX DOUBLED", AND 10-E5 "BDPS ACTUATED CHG SUCT SWITCHOVER" ACTUATE, 1CV112D & E OPEN, AND 1CV112B & C CLOSE.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED SOURCE RANGE CHANNEL TO NORMAL.

EVENTS: 1) LER 20-02-88-022

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Braidwood 2	0 5 0 0 0 4 5 7	1 of 0 3

Title (4) Rx Trip Due to Loose Connections in 2PM05J (Source Range Hi Flux)

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
0 9	1 9	8 8	8 8	0 2 2	0 0	1 0	1 1	8 8	Braidwood	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 2

POWER LEVEL (10): 0 0 3

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

___ 20.402(b)	___ 20.405(c)	X 50.73(a)(2)(iv)	___ 73.71(b)
___ 20.405(a)(1)(i)	___ 50.36(c)(1)	___ 50.73(a)(2)(v)	___ 73.71(c)
___ 20.405(a)(1)(ii)	___ 50.36(c)(2)	___ 50.73(a)(2)(vii)	X Other (Specify in Abstract below and in Text) 50.72
___ 20.405(a)(1)(iii)	___ 50.73(a)(2)(i)	___ 50.73(a)(2)(viii)(A)	
___ 20.405(a)(1)(iv)	___ 50.73(a)(2)(ii)	___ 50.73(a)(2)(viii)(B)	
___ 20.405(a)(1)(v)	___ 50.73(a)(2)(iii)	___ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER Area Code	Number
Freddie Ramos, Technical Staff Engineer Ext. 2487	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

___ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15): Month ___ Day ___ Year ___

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

At 1800 on September 19, 1988 a reactor trip occurred due to source range channel N31 exceeding its setpoint of 1.0xE5 counts per second (CPS). "A" reactor trip breaker opened automatically. The Nuclear Station Operator initiated a manual trip to open the "B" reactor trip breaker. The cause of this event was due to a loose connection in main control room panel 2PM05J, which allowed channel N-31 to re-energize. Since reactor power was approximately 3%, the reactor trip occurred. Subsequent investigation revealed that an actuation had only occurred on Train "A" and no failure of Train "B" actually occurred. The investigation revealed loose connections at the back of 2PM05J which were associated with the various Nuclear Instrumentation System blocking functions. These connections were tightened to prevent any further breaks in the blocking circuits. Additional terminal strips were checked for loose connections on both units. There have been no previous occurrences of loose connections in the source range resulting in a reactor trip.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

Form Rev 2.0

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					Page (3)		
		Year	///	Sequential Number	///	Revision Number			
Braidwood 2	0 | 5 | 0 | 0 | 0 | 4 | 5 | 7	8 | 8	-	0 | 2 | 2	-	0 | 0	0 | 2	OF	0 | 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 2; Event Date: September 19, 1988; Event Time: 1800;

Reactor Mode: 2; Mode Name: Startup; Power Level: 3%;

RCS [AB] Temperature/Pressure: 557 degrees F/2240 psig

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of the event which contributed to the severity of the event.

At 1800 on September 19, 1988 a reactor trip occurred on Unit 2. First out annunicator, "Sr High Flux Rx Trip", illuminated at the time of the event. Further investigation revealed that source range channel N31 (IG) exceeded its setpoint of 1.0xE5 counts per second (CPS). "A" reactor trip breaker opened automatically. "B" reactor trip breaker did not open automatically. Unit 2 Nuclear Station Operator (NSO) initiated a manual trip to open the "B" reactor trip breaker. Subsequent investigation revealed that an actuation had only occurred on Train "A" and no failure of Train "B" actually occurred.

Operator actions neither increased or decreased the severity of the event.

The appropriate NRC notification via the ENS phone system was made at 1916 on September 19, 1988, pursuant to 10CFR50.72(b)(2)(ii).

This event is being reported pursuant to 10CFR50.73(a)(2)(iv) - Any event or condition that resulted in manual or automatic actuation of any engineered safety feature, including the reactor protection system.

C. CAUSE OF EVENT:

The cause of this event was due to a loose connection in main control room panel 2PM05J, Section B2, Part 11, Riser A-2, Terminals 56-1 and 56-2. This loose connection caused a break in the Train "A" source range reset circuit which allowed the source range channel high flux reactor trip associated with channel N-31 to become unblocked. This allowed channel N-31 to re-energize. Since reactor power was approximately 3% the 1.0XE5 CPS setpoint was exceeded and the reactor trip occurred. The loose connection was disturbed when a Nuclear Station Operator (NSO), license reactor operator, was changing the paper on a nearby chart recorder associated with the volume control tank level, LR-185. This effect was duplicated during troubleshooting of the source range block circuit. N-32 did not energize because the block/reset and high voltage cutout remained funcional.

D. SAFETY ANALYSIS:

There was no effect on the plant or public safety. The plant responded per design which is to trip the unit on source range high flux (i.e. 1 out of two coincidence logic). "B" reactor trip breaker did not open automatically because only the Train "A" had its source range unblocked due to loose connections in 2PM05J. Under worst case conditions with the loose connections in 2PM05J being jarred and the plant at 100% power, source range high flux would cause a reactor trip to occur per design.

E. CORRECTIVE ACTIONS:

The immediate corrective action by the Unit 2 operator was to trip reactor trip breaker "B".

A partial surveillance on Train "A" solid state protection system, 2BwOS 3.1.1-20 was performed to determine if the universal cards associated with the source range block circuits were functional. The surveillance did not reveal any abnormalities with SSPS.

Nuclear Work Request A25642 was written to investigate cause of the source range channel N-31. The investigation revealed loose connections at the back of 2PM05J which were associated with the various Nuclear Instrumentation System blocking functions. These connections were tightened to prevent any further breaks in the blocking circuits.

Additional terminal strips were checked for loose connections on Unit 1 and Unit 2.

F. PREVIOUS OCCURRENCES:

There has been previous occurrence of a reactor tirp involving source range monitoring instrumentation. The corretive actions were implemented addressing both root and contributing cause. However, the root cause of this event is different in that loose terminal wiring for the source range instrumentation was involved. Previous corrective actions are not applicable to this event.

G. COMPONENT FAILURE DATA:

This event was not cause by component failure, nor did any components fail as a result of this event.

NI02 NOISY SR CHANNEL

TYPE: GENERIC, RB

A) N31
B) N32

CAUSE: IMPROPER CABLE SHIELDING

REF: SYSTEM DESCRIPTION

PLT STA: PLANT START-UP

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED SOURCE RANGE CHANNEL TO BECOME NOISY. THE NOISE SPIKES WILL BE INDICATED ON THE FOLLOWING EQUIPMENT:

-N31 NEUTRON LEVEL (CPS) ON 1PM07J
-N32 NEUTRON LEVEL (CPS) ON 1PM07J
-AUDIBLE COUNT RATE SPEAKER, SELECTED TO AFFECTED CHANNEL ON 1PM07J
-RATE METER, SELECTED TO AFFECTED CHANNEL, ON 1PM07J
-SR COUNT RATE, 1NI-31B/32B, ON 1PM05J
-SR START UP RATE, 1NI-31D/32D, ON 1PM05J
-NUCLEAR POWER RECORDER, 1NR-45, ON 1PM05J

IF THE INITIAL SOURCE RANGE COUNT RATE IS HIGH ENOUGH, THE ADDITION OF THE NOISE WILL BE SUFFICIENT TO ACTUATE ANNUNCIATOR 10-A1 "SR S/D FLUX HIGH". BDPS WILL ACTUATE IF THE DETECTOR OUTPUT IS DOUBLED WITHIN A 10 MINUTE PERIOD. ANNUNCIATORS 10-E3 "BDPS FLUX DOUBLED", AND 10-E5 "BDPS ACTUATED CHG SUCT SWITCHOVER" ACTUATE, 1CV112D & E OPEN, AND 1CV112B & C CLOSE.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED SOURCE RANGE CHANNEL TO NORMAL.

EVENTS: 1) DVR 20-01-86-038

DEVIATION INVESTIGATION REPORT

NI02

TITLE "Noise Spiking on Source Range Channel N31 Resulting in Containment Evacuation Alarm Actuation

PAGE 1 OF 0

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	6
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	0 0 0
1 1	1 1	8 6	2 0	0 1	8 6	— 0 1 8	— 0 0	1 2	0 3	8 6		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER AREA CODE	
Richard Schliessmann Ext. 2495	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
_ YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Mode 6 - Fuel load completed, vessel head off. Reactor Coolant System [AB] (RCS) temperature and pressure were at ambient conditions.

B. DESCRIPTION OF EVENT

On November 11, 1986 at 0855 [illegible] source range channel N31 [IG] began spiking occasionally over a two hour period
of time. The spiking obser[illegible] was high enough to actuate the High Flux at Shutdown alarm, and containment was
evacuated. Channel N32 did not show any spiking during this time frame. Operating walked down the area around
the preamplifier and cable conduit but found no contractor work that may have induced noise onto the system. At
1041 the N31 High Flux at Shutdown alarm was blocked to prevent the spiking from actuating the evacuation
alarm. At 1057, LCOAR 9.2-1a was entered because of the blocked alarm, and N31 was declared inoperable. At
both 1000 and 1115, Rad/Chem completed neutron surveys of Unit 1 containment, verifying no activity. No other
spiking was observed, and on November 13, 1986 at approximately 2130, LCOAR was exited and N31 was declared
operable.

C. Cause of Event:

Cause of spiking was due to noise induced onto the channel. Though no construction work in the area of the preamplifier or conduit was observed during the spiking, previous experience with such a problem, and the fact that the noise did not reoccur in the next two days, indicated that the spiking was due to construction activity

D. SAFETY ANALYSIS:

There were no safety consequences resulting from this event. N31 performed its required function by actuating the containment evacuation alarm. There was no component failure, and N32 was operable, available and not affected by this event.

E. CORRECTIVE ACTIONS:

No corrective actions are necessary at this time.

F. PREVIOUS OCCURRENCES:

DVR Number	Title
20-1-86-020	Containment Evacuation Horn Actuation Due to Spiking on N31

G. COMPONENT FAILURE DATA:

None.

NI03 SR CHANNEL HIGH VOLTAGE FAILURE

TYPE: GENERIC, RB

A) N31
B) N32

CAUSE: POWER SUPPLY FAILURE

REF: SYSTEM DESCRIPTION

PLT STA: PLANT START-UP

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED SOURCE RANGE CHANNEL HIGH VOLTAGE POWER SUPPLY TO FAIL. THE HIGH VOLTAGE WILL BE FAILED TO ZERO CAUSING THE INDICATED COUNT RATE FOR THE AFFECTED CHANNEL TO FAIL TO ZERO. ANNUNCIATOR 10-B1 "SR HIGH VOLT FAILURE" IS ACTUATED .

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED SOURCE RANGE CHANNEL'S HIGH VOLTAGE POWER SUPPLY TO NORMAL.

EVENTS: NONE

NI04 FAILURE OF SR HIGH VOLTAGE TO DISCONNECT

TYPE: GENERIC, RB

A) N31
B) N32

CAUSE: HIGH VOLTAGE RELAY FAILURE

REF: 20E-1-4029 EF05
20E-1-4029 EF06
20E-1-4030 EF12
20E-1-4030 EF18
20E-1-4030 EF62

PLT STA: REACTOR START-UP

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE OPERATOR WILL BE UNABLE TO DEENERGIZE THE AFFECTED SR DETECTOR HIGH VOLTS USING THE SR BLOCK AND RESET SWITCH ON 1PM05J WHEN THE REACTOR IS ABOVE THE P-6 SETPOINT. THIS RESULTS IN THE SOURCE RANGE REMAINING ENERGIZED WHEN IT WOULD NORMALLY BE DEENERGIZED. THE P-10 SIGNAL WILL NOT DISCONNECT THE SR HIGH VOLTAGE EITHER.

THE AFFECTED SOURCE RANGE WILL CONTINUE TO INDICATE THE INCREASING REACTOR POWER LEVEL. THE REACTOR TRIP SIGNAL IS BLOCKED. ANNUNCIATOR 10-B1 "SR HIGH VOLT FAILURE" WILL ACTUATE ON THE UNAFFECTED DETECTOR BEING PLACED IN "BLOCK".

THE EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY PULLING THE INSTRUMENT FUSES FOR THE AFFECTED DETECTOR.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED DETECTOR CIRCUIT TO NORMAL.

EVENTS: NONE

NI05 SR DISCRIMINATOR FAILURE

TYPE: GENERIC, RV 0-100%

A) N31
B) N32

CAUSE: IMPROPER ADJUSTMENT OF DISCRIMINATOR

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR START-UP

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED SOURCE RANGE CHANNEL DISCRIMINATOR TO FAIL. THE VALUE OF THE DISCRIMINATOR OUTPUT WILL BE DETERMINED BY THE SELECTED SEVERITY. IF THE SELECTED SEVERITY IS LESS THAN THE INITIAL DISCRIMINATOR SIGNAL, THE AFFECTED SOURCE RANGE CHANNEL READING WILL BE GREATER THAN THE ACTUAL SOURCE RANGE POWER LEVEL. IF THE SELECTED SEVERITY IS GREATER THAN THE INITIAL DISCRIMINATOR SIGNAL, THE AFFECTED SOURCE RANGE CHANNEL READING WILL BE LESS THAN THE ACTUAL REACTOR POWER LEVEL.

MALFUNCTION REMOVAL RESTORES THE AFFECTED SOURCE RANGE DISCRIMINATOR TO NORMAL.

EVENTS: NONE

NI06 IR CHANNEL FAILURE

TYPE: GENERIC, RV 10E-11 TO 10E-3 AMPS

A) N35
B) N36

CAUSE: DETECTOR FAILURE

REF: SYSTEM DESCRIPTION

PLT STA: POWER IN INTERMEDIATE RANGE

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED INTERMEDIATE RANGE CHANNEL TO FAIL. THE VALUE OF THE DETECTOR OUTPUT WILL BE DETERMINED BY THE SELECTED SEVERITY AND WILL BE INDICATED ON THE FOLLOWING EQUIPMENT:

- -N35 NEUTRON LEVEL (AMPERES) ON 1PM07J
- -N36 NEUTRON LEVEL (AMPERES) ON 1PM07J
- -RATE METER, SELECTED TO AFFECTED CHANNEL, ON 1PM07J
- -IR CURRENT, 1NI-35B/36B, ON 1PM05J
- -IR START-UP RATE, 1NI-35D/36D, ON 1PM05J
- -NUCLEAR POWER RECORDER, 1NR-45, ON 1PM05J

AS THE MALFUNCTION SEVERITY IS INCREASED, ANNUNCIATOR 10-A2 "IR HIGH FLUX ROD STOP C-1" ACTUATES IF THE IR RX TRIP IS NOT BLOCKED. IF THE SELECTED SEVERITY IS INCREASED SO THAT THE DETECTOR OUTPUT IS THE CURRENT EQUIVALENT TO GREATER THAN 25% REACTOR POWER, WITHOUT BLOCKING THE IR REACTOR TRIP, THE REACTOR WILL TRIP. ANNUNCIATOR 11-B2 "IR HIGH FLUX RX TRIP" ACTUATES ON THE TRIP.

MALFUNCTION REMOVAL RESTORES THE AFFECTED INTERMEDIATE RANGE CHANNEL TO NORMAL.

EVENTS: NONE

NI07 IR CHANNEL GAMMA COMPENSATION FAILURE

TYPE: GENERIC, RV -2 TO +2 DECADES

A) N35
B) N36

CAUSE: LOSS OF COMPENSATION VOLTAGE

REF: SYSTEM DESCRIPTION

PLT STA: POWER IN INTERMEDIATE RANGE

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED INTERMEDIATE RANGE CHANNEL GAMMA COMPENSATION TO FAIL. THE VALUE OF THE RESULTANT DETECTOR OUTPUT WILL BE DETERMINED BY THE SELECTED SEVERITY AND WILL BE INDICATED ON THE FOLLOWING EQUIPMENT:

-N35 NEUTRON LEVEL (AMPERES) ON 1PM07J
-N36 NEUTRON LEVEL (AMPERES) ON 1PM07J
-RATE METER, SELECTED TO AFFECTED CHANNEL, ON 1PM07J
-IR CURRENT, 1NI-35B/36B, ON 1PM05J
-IR START-UP RATE, 1NI-35D/36D, ON 1PM05J
-NUCLEAR POWER RECORDER, 1NR-45, ON 1PM05J

IF MALF SET AT >1.1, ANNUNCIATOR 10-C2 "IR CMPSATING VOLT FAILURE" ACTUATES.

IF BOTH CHANNELS' OUTPUT ARE DECREASED BELOW THE P-6 LEVEL, THE SOURCE RANGE LEVEL TRIPS ARE AUTOMATICALLY REACTIVATED AND HIGH VOLTAGE IS RESTORED TO THE SOURCE RANGE DETECTORS. IF REACTOR POWER IS GREATER THAN 10^5 CPS, THE REACTOR WILL TRIP, AND ANNUNCIATOR 11-A2 "SR HIGH FLUX RX TRIP" ACTUATES.

MALFUNCTION REMOVAL RESTORES THE AFFECTED INTERMEDIATE RANGE CHANNEL GAMMA COMPENSATION TO NORMAL.

EVENTS: NONE

NI08 PR DETECTOR FAILURE

TYPE: GENERIC, RV 0-500 uAMPS

A) N41 UPPER DETECTOR
B) N42 UPPER DETECTOR
C) N43 UPPER DETECTOR
D) N44 UPPER DETECTOR
E) N41 LOWER DETECTOR
F) N42 LOWER DETECTOR
G) N43 LOWER DETECTOR
H) N44 LOWER DETECTOR

CAUSE: DETECTOR FAILURE

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED POWER RANGE DETECTOR TO FAIL. THE VALUE OF THE DETECTOR OUTPUT WILL BE DETERMINED BY THE SELECTED SEVERITY AND WILL BE INDICATED ON THE FOLLOWING EQUIPMENT:

N41 NEUTRON LEVEL (PERCENT FULL POWER) ON 1PM07J
N41 NEUTRON LEVEL (DET CURRENT MICROAMPS) ON 1PM07J
N42 NEUTRON LEVEL (PERCENT FULL POWER) ON 1PM07J
N42 NEUTRON LEVEL (DET CURRENT MICROAMPS) ON 1PM07J
N43 NEUTRON LEVEL (PERCENT FULL POWER) ON 1PM07J
N43 NEUTRON LEVEL (DET CURRENT MICROAMPS) ON 1PM07J
N44 NEUTRON LEVEL (PERCENT FULL POWER) ON 1PM07J
N44 NEUTRON LEVEL (DET CURRENT MICROAMPS) ON 1PM07J
PR 41 DELTA FLUX, 1NI-41C, ON 1PM05J
PR 42 DELTA FLUX, 1NI-42C, ON 1PM05J
PR 43 DELTA FLUX, 1NI-43C, ON 1PM05J
PR 44 DELTA FLUX, 1NI-44C, ON 1PM05J
PR 41 % FULL POWER, 1NI-41B, ON 1PM05J
PR 42 % FULL POWER, 1NI-42B, ON 1PM05J
PR 43 % FULL POWER, 1NI-43B, ON 1PM05J
PR 44 % FULL POWER, 1NI-44B, ON 1PM05J
LOWER FLUX RECORDER, 1NR-41, ON 1PM05J
LOWER FLUX RECORDER, 1NR-42, ON 1PM05J
UPPER FLUX RECORDER, 1NR-43, ON 1PM05J
UPPER FLUX RECORDER, 1NR-44, ON 1PM05J
NUCLEAR POWER RECORDER, 1NR-45, ON 1PM05J

WITH THE MALFUNCTION SEVERITY INCREASED ABOVE THE INITIAL POWER LEVEL, THE INDICATED REACTOR POWER WILL INCREASE FOR THE AFFECTED CHANNEL. IF THE AFFECTED CHANNEL DIFFERS BY 2% FROM THE LOWEST CHANNEL POWER, ANNUNCIATOR 10-C4 "PWR RNG CHANNEL DEV" ACTUATES. IF THE RATE OF CHANGE OF POWER FOR THE SELECTED CHANNEL IS +5% IN 2 SECONDS, ANNUNCIATOR 10-C3 "PWR RNG FLUX RATE RX TRIP ALERT" ACTUATES. ANNUNCIATOR 10-B5 "PWR RNG FLUX HIGH ROD STOP" IS ACTUATED AT 103% POWER. IF INDICATED REACTOR POWER EXCEEDS 109%, ANNUNCIATOR 10-A3 "PWR RNG HIGH STPT RX TRIP ALERT" IS ACTUATED. IF REACTOR POWER IS GREATER THAN 50% AND THE AFFECTED CHANNEL CAUSES AN UPPER OR LOWER DETECTOR OUTPUT RATIO OF 1.02, THE ASSOCIATED ANNUNCIATOR 10-A4/B4 "PWR RNG UPPER/LOWER DET FLUX DEV HIGH" ACTUATES. THE INPUT TO THE OVERPOWER DT AND OVERTEMPERATURE DT CIRCUITS MAY RESULT IN A RUNBACK CONDITION OCCURRING (2/4 COINCIDENCE). THE FOLLOWING ANNUNCIATORS MAY ALSO ACTUATE: 10-A7 "ROD DEV POWER RNG TILT" AND 10-C7 "DELTA I LIMITS EXCEEDED". IF INDICATED REACTOR POWER EXCEEDS 25% DURING A POWER ASCENSION AND THE PR LOW SETPOINT IS NOT BLOCKED, THEN ANNUNCIATOR 10-A3 "PWR RNG LOW STPT RX TRIP ALERT" IS ACTUATED.

WITH THE MALFUNCTION SEVERITY DECREASED BELOW THE INITIAL POWER LEVEL, THE INDICATED REACTOR POWER WILL DECREASE FOR THE AFFECTED CHANNEL. IF THE AFFECTED CHANNEL DIFFERS BY 2% FROM THE HIGHEST CHANNEL POWER, ANNUNCIATOR 10-C4 "PWR RNG CHANNEL DEV" ACTUATES. IF THE RATE OF CHANGE OF POWER FOR THE SELECTED CHANNEL IS -5% IN 2 SECONDS, ANNUNCIATOR 10-C3 "PWR RNG FLUX RATE RX TRIP ALERT" ACTUATES. IF REACTOR POWER IS GREATER THAN 50% AND THE AFFECTED CHANNEL CAUSES AN UPPER OR LOWER DETECTOR OUTPUT RATIO OF 1.02, THE ASSOCIATED ANNUNCIATOR 10-A4/B4 "PWR RNG UPPER/LOWER DET FLUX DEV HIGH" ACTUATES. THE FOLLOWING ANNUNCIATORS MAY ALSO ACTUATE: 10-A7 "ROD DEV POWER RNG TILT".

MALFUNCTION REMOVAL RESTORES THE AFFECTED POWER RANGE DETECTOR TO NORMAL.

EVENTS: NONE

NI09 PR CHANNEL FAILURE

TYPE: GENERIC, RV 0-120% FULL POWER

A) N41 C) N43
B) N42 D) N44

CAUSE: FAULTY GAIN ADJUST

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED POWER RANGE CHANNEL TO FAIL. THE VALUE OF THE CHANNEL OUTPUT WILL BE DETERMINED BY THE SELECTED SEVERITY AND WILL BE INDICATED ON THE FOLLOWING EQUIPMENT:

N41 NEUTRON LEVEL (PERCENT FULL POWER) ON 1PM07J
N42 NEUTRON LEVEL (PERCENT FULL POWER) ON 1PM07J
N43 NEUTRON LEVEL (PERCENT FULL POWER) ON 1PM07J
N44 NEUTRON LEVEL (PERCENT FULL POWER) ON 1PM07J
PR 41 % FULL POWER, 1NI-41B, ON 1PM05J
PR 42 % FULL POWER, 1NI-42B, ON 1PM05J
PR 43 % FULL POWER, 1NI-43B, ON 1PM05J
PR 44 % FULL POWER, 1NI-44B, ON 1PM05J
NUCLEAR POWER RECORDER, 1NR-45, ON 1PM05J

WITH THE MALFUNCTION SEVERITY INCREASED ABOVE THE INITIAL POWER LEVEL, THE INDICATED REACTOR POWER WILL INCREASE FOR THE AFFECTED CHANNEL. IF THE AFFECTED CHANNEL DIFFERS BY 2% FROM THE LOWEST CHANNEL POWER, ANNUNCIATOR 10-C4 "PWR RNG CHANNEL DEV" ACTUATES. IF THE RATE OF CHANGE OF POWER FOR THE SELECTED CHANNEL IS +5% IN 2 SECONDS, ANNUNCIATOR 10-C3 "PWR RNG FLUX RATE RX TRIP ALERT" ACTUATES. ANNUNCIATOR 10-B5 "PWR RNG FLUX HIGH ROD STOP" IS ACTUATED AT 103% POWER. IF INDICATED REACTOR POWER EXCEEDS 109%, ANNUNCIATOR 10-A3 "PWR RNG HIGH STPT RX TRIP ALERT" IS ACTUATED. IF INDICATED REACTOR POWER EXCEEDS 25% DURING A POWER ASCENSION AND THE PR LOW SETPOINT IS NOT BLOCKED, THEN ANNUNCIATOR 10-A3 "PWR RNG LOW STPT RX TRIP ALERT" IS ACTUATED.

WITH THE MALFUNCTION SEVERITY DECREASED BELOW THE INITIAL POWER LEVEL, THE INDICATED REACTOR POWER WILL DECREASE FOR THE AFFECTED CHANNEL. IF THE AFFECTED CHANNEL DIFFERS BY 2% FROM THE HIGHEST CHANNEL POWER, ANNUNCIATOR 10-C4 "PWR RNG CHANNEL DEV" ACTUATES. IF THE RATE OF CHANGE OF POWER FOR THE SELECTED CHANNEL IS -5% IN 2 SECONDS, ANNUNCIATOR 10-C3 "PWR RNG FLUX RATE RX TRIP ALERT" ACTUATES.

MALFUNCTION REMOVAL RESTORES THE AFFECTED POWER RANGE CHANNEL TO NORMAL.

EVENTS: NONE

NI10 INCORE MONITORING SYSTEM FAILURE

TYPE: GENERIC, RV 0-100% = DETECTOR RANGE

A) DETECTOR A
B) DETECTOR B
C) DETECTOR C
D) DETECTOR D
E) DETECTOR E
F) DETECTOR F

CAUSE: DETECTOR FAILURE

REF: SYSTEM DESCRIPTION

PLT STA: INCORE MONITORING SYSTEM IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED INCORE MONITORING SYSTEM DETECTOR TO FAIL. THE DETECTOR OUTPUT VALUE WILL BE DETERMINED BY THE SELECTED SEVERITY. MOVING THE DETECTOR TO ANOTHER POSITION WILL NOT CAUSE A CHANGE IN THE DETECTOR OUTPUT.

MALFUNCTION REMOVAL WILL RESTORE THE FAILED DETECTOR TO NORMAL.

EVENTS: NONE

NI11 STUCK INCORE DETECTOR

TYPE: GENERIC, RB

A) DETECTOR A
B) DETECTOR B
C) DETECTOR C
D) DETECTOR D
E) DETECTOR E
F) DETECTOR F

CAUSE: CABLE DAMAGE

REF: SYSTEM DESCRIPTION

PLT STA: INCORE MONITORING SYSTEM IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED DETECTOR TO STICK IN THAT POSITION. THE DETECTOR READOUT WILL REMAIN CONSTANT AS THE DETECTOR STOPS MOVING. THE OPERATOR WILL BE UNABLE TO EITHER INSERT OR WITHDRAW THE AFFECTED DETECTOR.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED DETECTOR CABLE TO NORMAL.

EVENTS: NONE

NI12 LEAK INTO GUIDE TUBE FOR INCORE DETECTOR

TYPE: DISCRETE, NRV 0-10 GPM @ 400 PSID

CAUSE: GUIDE TUBE FAILURE (L5)

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE REACTOR COOLANT SYSTEM TO LOSE MASS THROUGH THE INCORE INSTRUMENTATION GUIDE TUBE. THE RATE OF MASS LOSS WILL BE DETERMINED BY THE SELECTED SEVERITY. WHEN A HIGH LEVEL IS DEVELOPED IN THE DRAIN HEADER FOR THE 10 PATH TRANSFER DEVICE, AN ALARM ON 1PM08J WILL BE ACTUATED AND A LAMP WILL BE ILLUMINATED. ANNUNCIATOR 1-B2 "CNMT DRAIN LEAK DETECT FLOW HIGH" ACTUATES.

THE SIMULATOR MUST BE RESET TO REMOVE THIS MALFUNCTION.

EVENTS: NRC IN 87-44

NI/12

SSINS No.: 6835
IN 87-44

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

SEP 21 REC'D.

September 16, 1987

NRC INFORMATION NOTICE NO. 87-44: THIMBLE TUBE THINNING IN WESTINGHOUSE REACTORS

Addressees:

All pressurized water reactor facilities employing a Westinghouse nuclear steam supply system (NSSS) holding an operating license or a construction permit.

Purpose:

This information notice is being provided to alert addressees to potential problems resulting from thimble tube thinning in Westinghouse reactors. It is expected that recipients will review the information for applicability to their facilities and consider actions, if appropriate, to avoid similar problems. However, suggestions contained in this information notice do not constitute NRC requirements; therefore no specific action or written response is required.

Description of Circumstances:

During the recent refueling outage at North Anna Unit 1, eddy current (EC) testing identified wall thinning on approximately 23 out of 50 thimble tubes. The wall degradation occurred on the thimble tubes just above the lower core plate, between the lower core plate and the fuel assembly guide tubes. Several thimble tubes with greater than 35% wall thinning were identified, with one thimble tube thinned as much as 49%.

Discussion:

The movable incore neutron detectors travel within retractable thimble tubes. The thimble tubes normally extend (as indicated in Attachment 1) from a 10-path transfer device, through the seal table, through the bottom of the reactor vessel, and into selected fuel assemblies. The thimble tubes are supported by guide tubes within the lower vessel region and the fuel assemblies, and by high-pressure conduits between the reactor vessel and the seal table.

The thimble tubes are sealed at the leading (reactor) end, but are open at the 10-path transfer device to allow insertion of an incore neutron detector.

8709100056

Mechanical high-pressure seals, located at the seal table, are used to seal the area between the thimble tube and the high-pressure conduit. This seal serves as a reactor coolant system (RCS) pressure boundary since the area between the thimble tube and the high pressure conduit is at RCS pressure. Consequently, a leak in a thimble tube results in degradation of the RCS pressure boundary by creating a path for reactor coolant to bypass the mechanical seal. In order to halt the flow of leaking reactor coolant, the manual isolation valve must be closed.

As indicated, the thimble tubes are supported over most of their length. However, a small portion of the thimble tube is directly exposed to RCS flow. This exposed portion is between the top of the lower core plate and the bottom of the fuel assembly. This region is approximately 18.4 to 34.8 mm in length, depending on the reactor type. It is believed that flow-induced vibration on this exposed portion causes fretting at the adjacent guide tubes.

Undetected thinning of a thimble tube could lead to the development of a non-isolable leak and a corresponding loss of reactor coolant. As discussed previously, the manual isolation valve would have to be closed to halt the flow of leaking reactor coolant. The leaking coolant may create an environment in the vicinity of the isolation valves too hazardous for personnel to enter.

Leaking thimble tubes could result in degradation of the incore neutron monitoring system. If not isolated, reactor coolant from leaking thimble tubes can flow into the 10-path transfer device, allowing coolant to flood the other thimble tubes originating from that device. This could result in rendering inoperable more than just the leaking tube.

In addition to North Anna Unit 1, incore thimble tube thinning and leakage has been detected at facilities in France and Belgium. In this country, leaks in thimble tubes are known to have occurred at Salem Unit 1. In Licensee Event Report (LER) 81-028, Public Service Electric & Gas Co. (PSE&G) reported that three incore thimble tubes were known to have developed leaks because of fretting. One of these leaks resulted in the flooding of all six 10-path transfer devices, partially or completely flooding all the thimble tubes in the reactor. In addition, thinning has been detected on the Farley thimble tubes.

At North Anna Unit 1, the proposed corrective action was to retract selected thimble tubes approximately 2 inches. This would move the thinned area out of the region of high turbulence. In addition, the thimble tube that experienced the most degradation will be taken out of service by closing the corresponding isolation valve.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate regional office or this office.

Charles E. Rossi

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contact: Jack Ramsey, NRR
(301) 492-9081

Attachments:
1. Typical Westinghouse Incore Neutron Monitoring System
2. List of Recently Issued NRC Information Notices

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

RD01	ROD DRIVE M-G SET TRIP
RD02	DROPPED ROD
RD03	DROPPING ROD
RD04	ROD EJECTION
RD05	STUCK ROD
RD06	RODS FAIL TO MOVE
RD07	UNCONTROLLED ROD MOVEMENT
RD08	DRPI - DATA CABINET FAILURE
RD09	AUTO ROD SPEED CONTROLLER FAILURE
RD10	FAILURE IN LOGIC CABINET
RD11	POWER CABINET FAILURE
RD12	ROD STOPS FAIL
RD13	DRPI - OPEN OR SHORTED COIL

RD01 ROD DRIVE M-G SET TRIP

TYPE: GENERIC, RB

A) 1A M-G SET
B) 1B M-G SET

CAUSE: FAULTY ACTUATION OF SH-TR RELAY

REF: 20E-1-4030 RD01
20E-1-4030 RD02
20E-1-4030 RD10

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED M-G OUTPUT BRKR TO TRIP OPEN. ANNUNCIATOR 10-D8 "ROD DRIVE M/G SET TROUBLE" ACTUATES. IF THE OTHER M-G SET IS NOT ON-LINE, THEN THE RODS FALL INTO THE CORE INITIATING A REACTOR TRIP ON A NEGATIVE FLUX RATE. THE M-G SET BREAKER CANNOT BE RECLOSED WHILE THE MALFUNCTION IS STILL ACTIVE.

MALFUNCTION REMOVAL RESTORES THE SELECTED M-G SET TO NORMAL OPERATION.

EVENTS: 1) OE 1780
2) LER 20-02-88-031

OE 1780 STEPHENSON (APC) 14-JUL-86 12:30 PT

RDO1

Subject:WESTINGHOUSE MODEL #727 MOTOR-GENERATOR SET

UNIT: JOSEPH M. FARLEY UNIT 2

DOC NO/LER NO: LER 86-007-00

EVENT DATE: 6/8/86

NSSS/A-E: WESTINGHOUSE/BECHTEL/SCSI

RATING: 860 MWE

DATE OF COMMERCIAL OPERATION: 7/30/81

SUPPLEMENTAL DESCRIPTION:

WESTINGHOUSE MODEL #727 MOTOR-GENERATOR SET

EVENT DESCRIPTION:

ON 6-8-86, WHILE OPERATING AT 81% POWER, A REACTOR TRIP OCCURED ON HIGH NEGATIVE FLUX RATE. INVESTIGATION INTO THE CAUSE OF THE REACTOR TRIP REVEALED THAT BOTH MOTOR-GENERATOR (MG) SETS (WESTINGHOUSE MODEL #727) SUPPLYING POWER TO THE CONTROL ROD DRIVE SYSTEM MALFUNCTIONED, CAUSING A LOSS OF POWER TO THE CONTROL ROD DRIVE MECHANISM GRIPPERS. THIS ALLOWED ALL THE CONTROL RODS TO FALL INTO THE CORE RESULTING IN A HIGH NEGATIVE FLUX RATE.

IMMEDIATELY AFTER THE TRIP, IT WAS FOUND THAT THE 2A MG SET WAS STILL RUNNING WITH ITS OUTPUT BREAKER CLOSED BUT THE MG SET WAS GENERATING NO OUTPUT VOLTAGE. THE 2B MG SET WAS RUNNING BUT ITS OUTPUT BREAKER WAS OPEN.

FURTHER INVESTIGATION REVEALED THAT THE 2A MG SET HAD LOST ITS FIELD, WHICH SHOULD HAVE CAUSED ITS OUTPUT BREAKER TO OPEN. THE OUTPUT BREAKER, HOWEVER, DID NOT OPEN DUE TO A LOOSE PLUNGER SCREW ON AN AUXLIARY RELAY OF THE DIRECTIONAL OVERCURRENT RELAY (IRV) (WESTINGHOUSE TYPE IRV-2 STYLE NUMBER 290B089A09A) ON THE "C" PHASE OF THE 2A MG SET, WHICH CAUSED THE IRV TO BE INOPERABLE. THE EXACT CAUSE FOR THE LOSS OF FIELD OF THE 2A MG SET HAS NOT BEEN DETERMINED. A REGULATOR BOARD WHICH COULD HAVE CAUSED THIS PROBLEM HAS BEEN SENT TO WESTINGHOUSE FOR TESTING.

THE 2B MG SET SHOULD NOT HAVE TRIPPED. HOWEVER, IT WAS FOUND THAT A PLUNGER ON AN AUXILIARY RELAY OF THE IRV ON THE "C" PHASE OF THE 2B MG SET WAS OUT OF ADJUSTMENT, ALLOWING THE AUXILIARY RELAY TO REMAIN CLOSED IMPROPERLY DUE TO RESIDUAL MAGNETISM. THE CLOSED AUXILIARY RELAY CAUSED THE IRV TO LOSE ITS DIRECTIONAL CHARACTERISTICS AND CAUSED THE IRV TO ACTUATE MORE RAPIDLY THAN NORMAL. THEREFORE, WHEN THE 2A MG SET STOPPED GENERATING AND THE 2B MG SET ASSUMED THE FULL LOAD, THE IRV ON THE "C" PHASE OF THE 2B MG SET CAUSED THE 2B MG SET OUTPUT BREAKER TO OPEN.

DURING SUBSEQUENT INVESTIGATION, IT WAS FOUND THAT THE ADJUSTMENTS OF THE AUXULIARY RELAY OF THE IRV'S ON THE MG SETS IS SET AT THE FACTORY. THE TECHNICAL MANUAL FOR THE MG SETS DOES ADDRESS CHECKING OF THESE AUXILIARY RELAYS ON THE IRV'S IN THE TROUBLESHOOTING SECTION BUT IS NOT MENTIONED AS A ROUTINE PREVENTIVE MAINTENANCE TASK.

CORRECTIVE ACTION:

IN ORDER TO PREVENT RECURRENCE OF THIS TYPE OF EVENT, THE IRV'S ON ALL MG SETS AT FNP HAVE BEEN CHECKED AND RE-ADJUSTED AS REQUIRED. APPROPRIATE PROCEDURES WILL BE REVISED TO INCLUDE THE ADDITIONAL TASKS NECESSARY TO CHECK THE AUXILIARY RELAYS ON THE IRV'S.

Information Contact:LON BRADSHAW, 205-899-5156, EXT.3513

RD01

LICENSEE EVENT REPORT (LER)

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Facility Name (1)	Docket Number (2)	Page (3)
Braidwood Unit 2	0 5 0 0 0 4 5 7	1 of 0 3

Title (4) Reactor Trip Due to Negative Rate trip as a Result of Rod Control System Loss of Power.

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
1 1	0 5	8 8	8 8	0 3 1	0 0	1 1	1 8	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 8 8

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	X 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name		TELEPHONE NUMBER AREA CODE	
C. Wiegand, Technical Staff Engineer	Ext. 2492	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

At 1307 on November 5, 1988 during Rod Control (RD) System troubleshooting, the 2B Motor Generator (M/G) set's RV A relay (which was oscillating) was isolated to replace a blown fuse. This resulted in a loss of excitation to the 2B M/G set and the 2A M/G set was demanded to carry the entire RD System Load. The 2A M/G set's overvoltage relay (1H) was picked up, resulting in a total loss of power to the RD system. At 1308 a reactor trip due to a negative rate trip on all Nuclear Instrumentation System power range channels occurred. The cause of this event is an incorrect 1H relay setting due to conflicting information in the Technical Manual. The immediate corrective actions taken were to reset the 1H relay, replace the blown fuse, and to simulate the identical conditions that led to the trip (verifying that the incorrect relay setting was the cause). The 1H relays for each M/G set of both units have been verified at the correct setpoints. All future relay settings will be given by M/G output voltage. Testing of the M/G set setpoints will be performed during each refueling outage. There have been previous occurrences of Rod Control System perturbations that resulted in a reactor trip. Previous corrective actions are not applicable to this event.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. **PLANT CONDITIONS PRIOR TO EVENT:**

Unit: Braidwood 2; Event Date: November 5, 1988; Event Time: 1308;

Mode: 1 - Power Operation; Rx Power: 88%;

RCS [AB] Temperature/Pressure: 582 degrees F/2242 psig

B. DESCRIPTION OF EVENT:

There were no structures, systems, or components inoperable at the beginning of the event that contributed to the event.

At 1307 on November 5, 1988 Braidwood Station Unit 2 was in Mode 1 operating at approxiately 88% power. During normal operating rounds it was observed that the Rod Control Systems (RD) [AA] 2B M/G set IRV A directional current relay's moving contact was oscillating between the stationary contacts. A Nuclear Work Request (NWR) was written to troubleshoot the oscillating contact; troubleshooting was in progress at 1307. The Operational Analysis Department (OAD) was performing troubleshooting on the 2B M/G set and discovered a blown fuse (22FU). Subsequent discussions between OAD and licensed operators on duty resulted in the decision to isolate the IRV A relay when replacing the fuse. This would prevent a voltage spike which could potentially trip the 2B M/G set offline. At the time, isolating the IRV A relay was considered to be the most conservative method because it did not involve taking the 2B M/G set offline. At 1308 the IRV A relay was isolated which caused the 2R relay to dropout opening up the contacts, causing a loss of excitation current to the voltage regulator. With a loss of excitation current, the 2B M/G set could not carry any load, therefore, the 2A M/G set was demanded to carry the entire Rod Control System load. Upon assuming the entire system load, the 2A M/G set's exciter current rose to a point such that overvoltage relay (1H) was picked up, resulting in the 2A M/G set's output breaker opening. Although the 1H relay is called an overvoltage relay it really senses exciter current. The opening of the breaker resulted in a total loss of power to the Rod Control System which resulted in th elease of all 53 Rod Control Cluster Assemblies (RCCA's). The release of all RCCA's resulted in an automatic reactor trip due to a negative rate trip on all four of the Nuclear Instrumentation System (NIS) [IG] power range channels.

The licensed operators on duty performed a safe shutdown following station procedures, and stable conditions were achieved by 1330.

Although a potential problem with the 2B M/G set was being investigated, the 2B M/G set itself was operable and was sharing half of the Rod Control System Load with the 2A M/G set. However, the troubleshooting of the 2B M/G set did contribute to the initial cause of the event.

The appropriate NRC notification via the ENS phone system was made at 1456 pursuant to 10CFR50.72(b)(2)(ii).

This event is being reported pursuant to 10CFR50.73(a)(2)(iv) - any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature, including the reactor protection system.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

Form Rev 2.0

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Braidwood Unit 2	0 5 0 0 0 4 5 7	8 8	- 0 3 1 -	0 0	0 3 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

C. CAUSE OF EVENT:

Each of the M/G sets is designed to be capable of carrying the entire Rod Control System load by itself. The cause of this event is the failure of the 2A M/G set to carry the entire Rod Control System load. Subsequent investigations by OAD discovered that the 1H relay, which provides overvoltage protection, was set at a conservatively low value. The normal operating voltage of the M/G sets are 260 volts plus or minus 5 volts. The relay was set to pick up at approximately 260 volts. Consequently, a very small voltage rise was enough to trip the 2A M/G set offline. The intermediate cause of this event was an incorrectly set overvoltage relay. The root cause of this event is conflicting information, regarding the setting of this relay, reported in the Technical Manual governing the M/G sets (2702/386 book 4; Westinghouse Shop Order 82-S-988). One section of the manual indicates a setting of 3.0 amps for the pickup of the 1H relay. Another section of the manual indicates that the setting should be for 280 volts. Unfortunately, 3.0 amps does not convert to 280 volts, rather, 3.0 amps develops to a voltage setting of approximately 260 volts. The 2A M/G set overvoltage relay was checked by OAD and was found set at 3.05 amps, the setting for 280 volts was determined to be 3.65 amps.

D. SAFETY ANALYSIS:

This event had no effect on the safety of the plant or the public. All systems operated as designed. There would not have been any safety consequences if this event had occurred under more severe conditions. The worst case conditions would be the Unit operating at 100% power; the plant response would have been the same.

E. CORRECTIVE ACTIONS:

The immediate corrective actions taken were to reset the 2A M/G set overvoltage relay (1H) such that it would pickup at the correct value of 280 volts, replace the blown fuse that was found in the 2B M/G set circuitry, and to simulate the identical conditions which led to the trip. The simulations involved recreating the conditions to verify that the M/G set with a correct overvoltage setting would be able to carry the Rod Control System load. The resulting trials showed that the incorrect relay setting was the cause of the reactor trip. The overvoltage relays for each M/G set of both units have been rechecked and are at the correct setpoints.

Actions to prevent recurrence include having Production Services Department to validate that all future relay setting orders for these relays be given by M/G output voltage. This will be tracked to completion by Action Item 457-200-88-18501. Station Technical Staff will ensure that testing of the M/G set setpoints is performed during each refueling outage. This will be tracked to completion by Action Item 457-200-88-18502.

F. PREVIOUS OCCURRENCES:

There have been previous occurrences of Rod Control System perturbations that resulted in a reactor trip. The corrective actions were implemented addressing both root and contributing causes. Previous corrective actions are not applicable to this event.

G. COMPONENT FAILURE DATA:

This event was not the result of component failure, nor did any components fail as a result of this event.

RD02 DROPPED ROD

TYPE: GENERIC, RB

NOTE: MALFUNCTION ENTERED AS "RD02D02"

SB A ROD D2	CB A ROD H6
SB A ROD B12	CB A ROD H10
SB A ROD M14	CB A ROD F8
SB A ROD P4	CB A ROD K8
SB A ROD B4	CB B ROD F2
SB A ROD D14	CB B ROD B10
SB A ROD P12	CB B ROD K14
SB A ROD M2	CB B ROD P6
SB B ROD G3	CB B ROD B6
SB B ROD C9	CB B ROD F14
SB B ROD J13	CB B ROD P10
SB B ROD N7	CB B ROD K2
SB B ROD C7	CB C ROD H2
SB B ROD G13	CB C ROD B8
SB B ROD N9	CB C ROD H 14
SB B ROD J3	CB C ROD P8
SB C ROD E3	CB C ROD F6
SB C ROD C11	CB C ROD F10
SB C ROD L13	CB C ROD K10
SB C ROD N5	CB C ROD K6
SB D ROD C5	CB D ROD D4
SB D ROD E13	CB D ROD M12
SB D ROD N11	CB D ROD D12
SB D ROD L3	CB D ROD M4
SB E ROD H4	CB D ROD H8
SB E ROD D8	
SB E ROD H12	
SB E ROD M8	

CAUSE: STATIONARY GRIPPER COIL FAILURE

REF: SYSTEM DESCRIPTION

PLT STA: ALL CONTROL RODS WITHDRAWN FROM ROD BOTTOM

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED CONTROL ROD STATIONARY GRIPPER COIL TO FAIL. THIS CAUSES THE AFFECTED

CONTROL ROD TO DROP RAPIDLY INTO THE CORE AND ANNUNCIATOR 10-E6 "ROD AT BOTTOM" ACTUATES WHEN THE ROD IS ON THE BOTTOM. THE AFFECTED CONTROL ROD'S ROD BOTTOM LED WILL BE ILLUMINATED.

REACTOR POWER WILL DECREASE AS THE DROPPED CONTROL ROD ADDS NEGATIVE REACTIVITY TO THE CORE WHICH WILL DECREASE T_{ave}. THE CONTROL RODS WILL WITHDRAW IN AUTO TO RECOVER T_{ave} AND MATCH IT WITH T_{ref}. THE ROD STEP COUNTERS AND ROD SPEED METER, 1SI-412, IN ADDITION TO THE ROD DIRECTION LAMPS, WILL INDICATE THE CONTROL ROD MOTION. ANNUNCIATOR 10-A7 "ROD DEV POWER RNG TILT" WILL ACTUATE. ANY ATTEMPT BY THE OPERATOR TO RELATCH AND WITHDRAW THE AFFECTED CONTROL ROD, WHILE THE MALFUNCTION IS ACTIVE, IS UNSUCCESSFUL. AT POWER, CERTAIN RODS MAY CAUSE A NEGATIVE RATE TRIP.

MALFUNCTION REMOVAL RESTORES THE FAILED CONTROL ROD STATIONARY GRIPPER COIL TO NORMAL.

EVENTS. 1) DVR 06-02-87-004
2) LER 20-02-88-009

KDO2

LICENSEE EVENT REPORT (LER)

Facility Name (1): Braidwood, Unit 2

Docket Number (2): 0|5|0|0|0|4|5|7

Page (3): 1 of 0|3

Title (4): Manual Reactor Trip Due to Inoperable Rod Control System

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
0|5	3|0	8|8	8|8	— 0|0|9	— 0|0	0|6	1|4	8|8	NONE	0|5|0|0|0| |
										0|5|0|0|0| |

OPERATING MODE (9): 2

POWER LEVEL (10): 0|0|0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	X 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name: Christopher M. Wiegand, Nuclear Engineer, Ext. 2492

TELEPHONE NUMBER: AREA CODE 8|1|5 4|5|8|-|2|8|0|1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	V|L	5|1|•|•	W|1|2|0	N					

SUPPLEMENTAL REPORT EXPECTED (14)

__ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15): Month | Day | Year

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

At 0348 on May 30, 1988, while withdrawing Control Bank Rods, an urgent and non-urgent alarm occurred followed by the release of rods in Shutdown Banks C, D, E, and Group 2 Rods in Shutdown Bank A and Control Banks A and C. At 0406 the reactor was manually tripped. The root cause of this event was the failure of a miscellaneous Electric Room (MER) Ventilation Fan. This failure caused the temperature of the Rod Control power cabinets to increase to the Thermal Overload Protection setpoint. Actuation of the Thermal Overload Protection de-energized the power supplies which resulted in the rods being released.

Temporary cooling fans were installed in the MER until the ventilation fan repairs were completed. Procedural revisions are being processed to specify the personnel to be notified should the ventilation system become inoperable. This should allow appropriate actions to be taken in a timely manner to maintain MER Ambient Temperature within the limits of the power supplies. Additionally, an analysis and evaluation of the MER Ventilation System to verify its adequacy in cooling during the summer months will be performed. There have been no previous occurrences.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year		Sequential Number		Revision Number	Page (3)
Braidwood, Unit 2	0 5 0 0 0 4 5 7	8 8	-	0 0 9	-	0 0	0 2 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 2 ; Event Date: May 30, 1988 ; Event Time: 0406
MODE: 2 - Startup ; Rx Power: 0% ; RCS [AB] Temperature/Pressure: 558 Degrees F/2233 psig

B. DESCRIPTION OF EVENT:

The Miscellaneous Electric Room (MER) ventilation fan 2VE01C [VL] was inoperable at the beginning of this event and which contributed to the severity of the event.

At 0330 on May 30, 1988, mode change checklist 2BwGP 100-2T2, was completed. All shutdown banks were fully withdrawn, Boron Concentration was 884 ppm, and the estimated critical position was control bank D at 105 steps.

At 0343 rod control was placed in manual mode and control banks were withdrawn in overlap. At 0348 control bank A was fully withdrawn, control bank B was being withdrawn to 116 steps, and control bank C was being withdrawn to step 1. Although the unit was administratively in Mode 2, it was actually subcritical with a $K_{effective}$ of approximately 0.98.

At 0348, while withdrawing control bank rods, an urgent and non-urgent alarm occurred followed by the release of the rods in shutdown banks C, D, E, and the rods in Group 2 of Shutdown Bank A and the rods in group 2 of control banks A and C.

At 0406 the reactor was manually tripped due to the operational condition of the Rod Control System. All rods were fully inserted, and safe shutdown was accomplished.

Operator actions neither increased nor decreased the severity of the event.

The appropriate NRC notification via the ENS phone system was made at 0432 on May 30, 1988 pursuant to 10CFR50.72(b)(2)(ii).

This event is being reported pursuant to 10CFR50.73(a)(2)(iv) - any event or condition that resulted in manual or automatic actuation of any engineered safety feature, including the reactor protection system.

C. CAUSE OF THE EVENT

The intermediate cause of the event was a trip of the Main and Auxiliary Rod Control Power Supplies [AA] of the affected power cabinets. The root cause of the event was the inoperability of the 2VE01C Ventilation Fan which resulted in high ambient temperatures in the MER, where the rod control power cabinets are located. The affected power supplies are rated for a maximum ambient temperature of 104 degrees (deg) F. The room temperature at the time of the event was approximately 100 deg F. It is estimated that inside the cabinet, and at the level of the power supplies, the temperature was as high as 110 deg F. The power supplies are equipped with thermal and electrical overload protection. Both of these circuits will trip when the temperature exceeds a preset value. The thermal circuit will reset by itself when the temperature drops to an acceptable level, and the electrical trip is reset by removing power to the supply.

When the electrical trips were reset and the temperature had cooled to below 95 deg F, the system was restored to normal.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

D. SAFETY ANALYSIS:

There were no safety consequences as a result of this event. All equipment operated as designed. The manual trip was a conservative action. Under worst case conditions, operating at 100% power will all rods fully withdrawn, there would still have been no safety consequences as the reactor would have automatically tripped and safe shutdown would have been accomplished using plant procedures. Additionally, this event is described in section 15.4.3 of the Final Safety Analysis Report, "Rod Cluster Control Assembly Misoperation".

E. CORRECTIVE ACTIONS:

Immediate corrective actions were to repair ventilation fan 2VE01C and install temporary cooling fans in the MER.

Actions taken to prevent recurrence include revising Operating Rounds Procedures, BwOP 199-A53 and BwOP 199-A41, which monitor MER Temperature. The purpose of this change is to inform the Technical Staff Nuclear Group whenever the MER Temperature exceeds 95 deg F. This will allow appropriate actions to be taken in a timely manner to maintain MER ambient temperature within the design limits of the power supplies. These procedure revisions will be tracked to completion by action item numbers 457-200-88-08701, and 457-200-88-08702, respectively.

In addition, procedures BwAP 0-34-A3, and BwAR 0-31-A3, will be revised to include notifying the Technical Staff Heating Ventilation and Air Conditioning Group (HVAC) System Test Engineer whenever the 2VE01C fan becomes inoperable. These will be tracked to completion by item numbers 457-200-88-08703, and 457-200-88-08704, respectively.

Although the 2VE01C fan was responsible for the high ambient temperature in the MER that lead to this event, a similar event could happen even though the fan was operating properly. The MER is cooled by outside air, and therefore the room can be only as cool as the outside temperature. With the possibility of summertime temperatures reaching 95 deg or higher, the trip setpoint of the power supplies could again be reached. Therefore, an evaluation of the MER Ventilation System to verify its adequacy in cooling the MER during the summer months will be performed. This will be tracked to completion by item number 457-200-88-08705.

F. PREVIOUS OCCURRENCES:

There have been no previous occurrences of a reactor trip as the result of excessive ambient temperatures in the MER.

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Westinghouse	Relay, Overload	None	AA33A

DEVIATION INVESTIGATION REPORT

RD02

TITLE

CONTROL ROD D-2 DROPPED DUE TO BLOWN FUSE

PAGE 1 OF 0 2

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	2
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	0 0 3
0 1	3 1	8 7	0 6	0 2	8 7	0 0 4	0 0	0 3	1 6	8 7		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
T. Schuster, Assistant Technical Staff Supervisor Ext. 2244	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	A A	F U	B 1 5 6	Y					

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

MODE 2 - Startup Rx Power 3% RCS [AB] Temperature/Pressure 558°F/2235 psig

B. DESCRIPTION OF EVENT:

At 0942 hours on 01-31-87, while in Mode 2 at 3% power the Shutdown Bank A control rod (RD)[AA] corresponding to core location D-2 fell from 228 steps (fully withdrawn) to RB (rod at bottom, zero steps) as indicated by the Digital Rod Position Indication (PI) display (RD). It was determined that the control rod actually dropped (not a PI display error) based on power decrease to 1%. The unit NSO stabilized the unit at 1% power and troubleshooting of the RD system was started via Nuclear Work Request B41040. There were no components or systems that were inoperable at the beginning of the event which contributed to the event.

C. CAUSE OF EVENT:

Fuse FU7 in the 1AC Rod Control Power Cabinet 2RD06J was found blown. This fuse supplies Phase A Stationary Gripper power for power cabinet rod group C (corresponds to Shutdown Bank A, group 1). The transient change in current to the stationary gripper coils following the loss of phase A power allowed rod D-2 to drop but was not severe enough to drop the other three rods in Shutdown Bank A, group 1. The cause of the blown fuse is unknown, the electrical conditions at the phase A, B, and C fuses were essentially identical following fuse replacement.

TITLE	DIR NUMBER					PAGE
	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
CONTROL ROD D-2 DROPPED DUE TO BLOWN FUSE	0 6	0 2	8 7	— 0 0 4 —	0 0	2 OF 0 2

TEXT

D. SAFETY ANALYSIS:

There were no adverse safety consequences resulting from this event. The dropped rod added negative reactivity to the core and did not impair the ability of the other rods to drop if required. The dropped rod following a partial loss of coil power is a failure in the safe direction. Due to the very low power level at the time of the event (3%) there were no adverse affects on core power distributions. The short duration of the rod misalignment resulted in no significant changes in core Xenon distribution. If the rod had dropped at a higher power level, there would have been a significant reactor turbine power mismatch and core reactivity redistributions. This is an analyzed event and is predicted to not result in fuel damage of any kind.

E. CORRECTIVE ACTIONS:

Fuse FU7 was replaced and the control rod retrieved per BOA ROD-4 within 54 minutes of the time the rod dropped. Shutdown Bank A was moved 10 steps in and out and was declared operable. Fuses do occasionally blow and due to the low frequency of this type of event, no actions were taken to try to prevent recurrence.

F. PREVIOUS OCCURRENCES:

None, however DVR 6-1-85-133 was written for a blown phase B stationary gripper fuse that did not result in a dropped rod.

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Shawmut	"Amptrap" 30 amp 600V Fuse	A60 x 30 Type 1	N/A

b) RESULTS OF NPRDS SEARCH:

Not Applicable

RD03 DROPPING ROD

TYPE: GENERIC, RB

NOTE: MALFUNCTION ENTERED AS "RD03D02"

SB A ROD D2	CB A ROD H6
SB A ROD B12	CB A ROD H10
SB A ROD M14	CB A ROD F8
SB A ROD P4	CB A ROD K8
SB A ROD B4	CB B ROD F2
SB A ROD D14	CB B ROD B10
SB A ROD P12	CB B ROD K14
SB A ROD M2	CB B ROD P6
SB B ROD G3	CB B ROD B6
SB B ROD C9	CB B ROD F14
SB B ROD J13	CB B ROD P10
SB B ROD N7	CB B ROD K2
SB B ROD C7	CB C ROD H2
SB B ROD G13	CB C ROD B8
SB B ROD N9	CB C ROD H14
SB B ROD J3	CB C ROD P8
SB C ROD E3	CB C ROD F6
SB C ROD C11	CB C ROD F10
SB C ROD L13	CB C ROD K10
SB C ROD N5	CB C ROD K6
SB D ROD C5	CB D ROD D4
SB D ROD E13	CB D ROD M12
SB D ROD N11	CB D ROD D12
SB D ROD L3	CB D ROD M4
SB E ROD H4	CB D ROD H8
SB E ROD D8	
SB E ROD H12	
SB E ROD M8	

CAUSE: MOVABLE GRIPPER COIL FAILURE

REF: ROD CONTROL SYSTEM DESCRIPTION

PLT STA: ALL CONTROL RODS WITHDRAWN FROM THE BOTTOM

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED CONTROL ROD MOVABLE GRIPPER COIL WILL FAIL. WHEN THE ROD IS CALLED

UPON TO MOVE IN EITHER DIRECTION, THE AFFECTED CONTROL ROD WILL DROP WHEN THE MOVABLE GRIPPER COIL IS ENERGIZED DURING THE NORMAL SEQUENCE OF ROD OPERATION. REACTOR POWER WILL DECREASE AS THE DROPPING CONTROL ROD ADDS NEGATIVE REACTIVITY TO THE CORE WHICH WILL DECREASE T_{ave}. THE CONTROL RODS RECEIVE A WITHDRAW SIGNAL, IN AUTO, TO RECOVER T_{ave} AND MATCH IT WITH T_{ref}. AS THE RODS STEP OUT OF THE CORE, WITH THE AFFECTED ROD IN THE CONTROLLING GROUP, THE AFFECTED ROD WILL CONTINUE TO SLIP DOWN INTO THE CORE. THE ROD STEP COUNTERS AND ROD SPEED METER, 1SI-412, IN ADDITION TO THE ROD DIRECTION LAMPS, WILL INDICATE THE CONTROL ROD MOTION. ANNUNCIATOR 10-A7 "ROD DEV POWER RNG TILT" WILL ACTUATE WHEN ANY ROD IS GREATER THAN OR EQUAL TO 12 STEPS FROM ANY OTHER ROD IN ITS BANK. THE DIGITAL ROD POSITION INDICATION SYSTEM WILL ACCURATELY INDICATE THE POSITION OF THE DROPPING ROD. WHEN THE ROD IS ON THE BOTTOM, ANNUNCIATOR 10-E6 "ROD AT BOTTOM" WILL BE ACTUATED. THE AFFECTED CONTROL ROD'S ROD BOTTOM LED WILL BE ILLUMINATED WHEN THE ROD REACHES THE BOTTOM.

ANY ATTEMPT BY THE OPERATOR TO RELATCH AND WITHDRAW THE AFFECTED CONTROL ROD, WHILE THE MALFUNCTION IS ACTIVE, IS UNSUCCESSFUL.

MALFUNCTION REMOVAL WILL RESTORE THE FAILED CONTROL ROD MOVABLE GRIPPER COIL TO NORMAL.

EVENT: NONE

RD04 ROD EJECTION

TYPE: GENERIC, NRB

NOTE: MALFUNCTION ENTERED AS "RD04D02"

SB A ROD D2	CB A ROD H6
SB A ROD B12	CB A ROD H10
SB A ROD M14	CB A ROD F8
SB A ROD P4	CB A ROD K8
SB A ROD B4	CB B ROD F2
SB A ROD D14	CB B ROD B10
SB A ROD P12	CB B ROD K14
SB A ROD M2	CB B ROD P6
SB B ROD G3	CB B ROD B6
SB B ROD C9	CB B ROD F14
SB B ROD J13	CB B ROD P10
SB B ROD N7	CB B ROD K2
SB B ROD C7	CB C ROD H2
SB B ROD G13	CB C ROD B8
SB B ROD N9	CB C ROD H14
SB B ROD J3	CB C ROD P8
SB C ROD E3	CB C ROD F6
SB C ROD C11	CB C ROD F10
SB C ROD L13	CB C ROD K10
SB C ROD N5	CB C ROD K6
SB D ROD C5	CB D ROD D4
SB D ROD E13	CB D ROD M12
SB D ROD N11	CB D ROD D12
SB D ROD L3	CB D ROD M4
SB E ROD H4	CB D ROD H8
SB E ROD D8	
SB E ROD H12	
SB E ROD M8	

CAUSE: FAILURE OF ROD DRIVE ASSEMBLY HOUSING (NOTE: ONLY ONE EJECTED ROD MAY BE ACTIVATED AT ANY ONE TIME)

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED ROD DRIVE ASSEMBLY HOUSING TO FAIL. THE AFFECTED CONTROL ROD WILL BE EJECTED FROM THE CORE BY REACTOR COOLANT SYSTEM PRESSURE. IF THE REACTIVITY ADDED BY THE EJECTION OF THE SELECTED CONTROL ROD IS SUFFICIENT, ANNUNCIATOR 10-C3 "PWR RNG FLUX RATE RX TRIP ALERT" ACTUATES AT 5% POWER INCREASE IN 2 SECONDS. 2/4 CHANNELS HAVING A 5% POWER INCREASE IN 2 SECONDS ACTUATES A RX TRIP AND ANNUNCIATOR 11-E2 "PWR RNG FLUX RATE HIGH RX TRIP" ACTUATES. A TWO INCH REACTOR COOLANT SYSTEM LOSS OF COOLANT ACCIDENT INSIDE CONTAINMENT, CAUSED BY THE RUPTURE IN THE DRIVE ASSEMBLY HOUSING, WILL RESULT. DRPI WILL INDICATE DATA A, DATA B, URGENT FAILURE, GENERAL WARNING, AND ROD BOTTOM LIGHT FOR THE EJECTED ROD DUE TO DRPI COIL DAMAGE.

THE SIMULATOR MUST BE RESET TO RECOVER FROM THIS MALFUNCTION.

EVENTS: NONE

RD05 STUCK ROD

TYPE: GENERIC, RV 0-231 STEPS

NOTE: MALFUNCTION ENTERED AS "RD05D02"

SB A ROD D2	CB A ROD H6
SB A ROD B12	CB A ROD H10
SB A ROD M14	CB A ROD F8
SB A ROD P4	CB A ROD K8
SB A ROD B4	CB B ROD F2
SB A ROD D14	CB B ROD B10
SB A ROD P12	CB B ROD K14
SB A ROD M2	CB B ROD P6
SB B ROD G3	CB B ROD B6
SB B ROD C9	CB B ROD F14
SB B ROD J13	CB B ROD P10
SB B ROD N7	CB B ROD K2
SB B ROD C7	CB C ROD H2
SB B ROD G13	CB C ROD B8
SB B ROD N9	CB C ROD H14
SB B ROD J3	CB C ROD P8
SB C ROD E3	CB C ROD F6
SB C ROD C11	CB C ROD F10
SB C ROD L13	CB C ROD K10
SB C ROD N5	CB C ROD K6
SB D ROD C5	CB D ROD D4
SB D ROD E13	CB D ROD M12
SB D ROD N11	CB D ROD D12
SB D ROD L3	CB D ROD M4
SB E ROD H4	CB D ROD H8
SB E ROD D8	
SB E ROD H12	
SB E ROD M8	

CAUSE: MECHANICAL BINDING OF CONTROL ROD (NOTE: ONLY FOUR STUCK RODS MAY BE ACTIVATED AT ANY ONE TIME)

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE AFFECTED CONTROL ROD(S) TO BE STUCK AT THE SELECTED SEVERITY. THE CONTROL ROD(S) WILL OPERATE PROPERLY UNTIL IT REACHES THE SELECTED POSITION, AT WHICH TIME IT WILL NOT MOVE. WHEN THE SELECTED CONTROL ROD(S) DEVIATES FROM ITS BANK BY >12 STEPS, ANNUNCIATOR 10-A7 "ROD DEV POWER RNG TILT" ACTUATES. ANY ATTEMPT BY THE OPERATOR TO MOVE THE AFFECTED CONTROL ROD(S) IN MANUAL, WHILE THE MALFUNCTION IS ACTIVE, IS INEFFECTIVE.

MALFUNCTION REMOVAL RESTORES THE AFFECTED CONTROL ROD(S) TO NORMAL.

EVENTS: NONE

RD06 RODS FAIL TO MOVE

TYPE: DISCRETE, RB

CAUSE: MASTER CYCLER FAILURE

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR START-UP IN PROGRESS

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE MASTER CYCLER TO FAIL. THERE IS NO AFFECT ON PLANT OPERATION UNTIL AUTO OR MANUAL ROD MOTION IS DEMANDED. WHEN ROD MOTION IS DEMANDED (AUTO OR MANUAL) ANNUNCIATOR 10-C6 "ROD CONT URGENT FAILURE" ACTUATES RESULTING IN A LOSS OF ALL ROD MOTION EXCEPT FOR SD C, D & E RODS.

MALFUNCTION REMOVAL RESTORES THE MASTER CYCLER TO NORMAL OPERATION.

EVENTS: NONE

RD07 UNCONTROLLED ROD MOVEMENT

TYPE: DISCRETE, RV -76 STEPS/MIN TO +76 STEPS/MIN

CAUSE: MASTER CYCLER FAILURE

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: IF A NEGATIVE ROD SPEED IS SELECTED (RODS IN), THE CONTROL RODS MOVE INWARD AT THE SELECTED SPEED ON ANY DEMAND SIGNAL (IN OR OUT) AND STOP MOVING ONCE THE DEMAND SIGNAL HAS CLEARED. THE ROD DIRECTION LAMPS AND ROD SPEED METER RESPOND TO REQUESTED ROD SPEED AND DIRECTION. THE ROD STEP COUNTERS AND DRPI WILL INDICATE ACTUAL ROD POSITION.

IF A POSITIVE ROD SPEED IS SELECTED (RODS OUT), THE CONTROL RODS MOVE OUTWARD AT THE SELECTED SPEED ON ANY DEMAND SIGNAL (IN OR OUT) AND STOP MOVING ONCE THE DEMAND SIGNAL HAS CLEARED. THE ROD DIRECTION LAMPS AND ROD SPEED METER RESPOND TO REQUESTED ROD SPEED AND DIRECTION. THE ROD STEP COUNTERS AND DRPI WILL INDICATE ACTUAL ROD POSITION.

MALFUNCTION REMOVAL RESTORES THE FAILED MASTER CYCLER OUTPUT TO NORMAL.

EVENTS: NONE

RD08 DRPI - DATA CABINET FAILURE

TYPE: GENERIC, RB

A) DATA CABINET A
B) DATA CABINET B

CAUSE: FAILED I/O CARD

REF: ROD POSITION INDICATION SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION PREVENTS THE SELECTED DRPI DATA CABINET FROM PROVIDING ROD POSITION INDICATION DATA TO THE CONTROL BOARD DRPI DISPLAY UNIT. THIS CAUSES THE DATA A OR B FAILURE LEDs TO ILLUMINATE ON THE DRPI PANEL ALONG WITH THE GENERAL WARNING LEDs FOR ALL THE RODS. ANNUNCIATOR 10-D6 "ROD CONT NON-URGENT FAILURE" WILL ACTUATE. THE DRPI DISPLAY WILL NOW ONLY INDICATE EVERY OTHER LED AS THE RODS ARE MOVED UNDER THIS CONDITION. IF BOTH DATA CABINETS ARE SELECTED, THEN ANNUNCIATOR 10-C6 "ROD CONT URGENT FAILURE" WILL ACTUATE.

MALFUNCTION REMOVAL WILL RESTORE THE DATA CABINET TO NORMAL.

EVENTS: NONE

RD09 AUTO ROD SPEED CONTROLLER FAILURE

TYPE: DISCRETE, RV 0-72 STEPS PER MINUTE

CAUSE: AUTO ROD SPEED PROGRAMMER FAILURE

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE ROD SPEED METER TO READ THE SELECTED SEVERITY. WHEN AN AUTOMATIC ROD DEMAND SIGNAL IS RECEIVED, THE ROD MOTION WILL OCCUR AT THE SPEED DETERMINED BY THE SELECTED SEVERITY. THE ROD STEP COUNTERS AND ROD SPEED METER, 1SI-412, IN ADDITION TO THE ROD DIRECTION LAMPS WILL INDICATE THE CONTROL ROD MOTION. THE T_{ave} CONDITION INITIALLY BEING CORRECTED, WILL BE CORRECTED (SLOWER THAN NORMAL WITH A LOWER SEVERITY) AND SUBSEQUENTLY OVERSHOOT (WITH A SEVERITY HIGHER THAN ACTUAL DEMAND), SO THE ROD CONTROL SYSTEM MAY OSCILLATE FROM INSERT TO WITHDRAW AND VICE VERSA. REACTOR POWER, REACTOR COOLANT SYSTEM TEMPERATURE AND PRESSURE WILL VARY IN RESPONSE TO THE REACTIVITY CHANGE.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY PLACING THE ROD CONTROL BANK SELECT SWITCH IN MANUAL AND CONTROLLING THE RODS MANUALLY.

MALFUNCTION REMOVAL RESTORES THE FAILED ROD SPEED PROGRAMMER TO NORMAL.

EVENTS: NONE

RD10 FAILURE IN LOGIC CABINET

TYPE: DISCRETE, RB

CAUSE: SLAVE CYCLER FAILURE

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE LOGIC CABINET TO EXPERIENCE A SLAVE CYCLER FAILURE. ANNUNCIATOR 10-C6 "ROD CONT URGENT FAILURE" IS ACTUATED RESULTING IN LOSS OF ALL ROD MOTION EXCEPT FOR SD C, D & E RODS.

MALFUNCTION REMOVAL RESTORES THE SLAVE CYCLER TO NORMAL.

EVENTS: 1) DVR 06-01-89-030

KDIO

DEVIATION INVESTIGATION REPORT (DIR)

Facility Name: Byron Nuclear Power Station

Form Rev 2.0

PAGE 1 OF 0 3

Title: ROD DRIVE URGENT ALARM DUE TO LOGIC CABINET FAILURE

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	1
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	0 9 9
0 2	2 7	8 9	0 6	0 1	8 9	— 0 3 0 —	0 0	0 4	1 2	8 9		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER AREA CODE	
James Schrock, Unit 1 Operating Engineer Ext. 2216	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	A A	E C B D	W 1 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
[ ] YES (if yes, complete EXPECTED SUBMISSION DATE)	[X] NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 02/27/89 / 2314

Unit 1 MODE 1 - Power Operation Rx Power 99% RCS [AB] Temperature/Pressure Normal Operating

Unit 2 MODE 5 - Cold Shutdown Rx Power 0% RCS [AB] Temperature/Pressure 185°F/ 370 PSIG

B. DESCRIPTION OF EVENT:

At 2314 hours on 02/27/89, a "ROD CONT URGENT FAILURE" alarm [AA] was received at the Unit 1 Main Control Board, inhibiting Control Bank D (CBD) motion. Immediately prior to the alarm, the NSO (licensed) noticed abnormally fast automatic rod motion after he had initiated a load decrease with the Bank Selector Switch in Automatic. The NSO then selected manual on the Bank Selector Switch and attempted rod withdrawal. The rods had moved 1/2 step when the alarm was received. The alarm originated from the Logic Cabinet (1RD07J) where the failure light on the 2BD Slave Cycler Logic Card was energized.

In order to facilitate troubleshooting during the initial investigation, the alarm sequence was repeated several times by resetting the alarm and attempting rod motion. It was noticed that rods usually stepped at a rate much faster than normal, but the rate seemed to vary. It was also noticed that the rods would move between 1/2 and 2 steps before the alarm would inhibit motion. In addition, the alarm would vary between the 1BD and 2BD Slave Cycler Logic Cards.

B. DESCRIPTION OF EVENT (Continued):

The Master Cycler Logic Card was then replaced in an attempt to correct the failure. The alarm was reset and rod motion requested. The rods moved 1 complete step and the alarm recurred, with the same Logic Cabinet error indicated.

Because of inconclusive troubleshooting, the control rods were declared inoperable and Limiting Condition for Operation Action Requirement (LCOAR) 1.3.1-1a was entered at 0555. Due to the imminent need to begin controlled shutdown of the plant per the LCOAR, the Pulser Oscillator Card, Master Cycler Logic Card, Master Cycler Selector Card, and the Supervisory Logic I Card were replaced at 0700. At 0705, the alarm was reset and rods were exercised without recurrence of the alarm. LCOAR 1.3.1-1a was exited at this time.

No systems were inoperable prior to this event that contributed to the event. Operator actions neither increased nor decreased the severity of this event. There were no manual or automatic safety system actuations and stable plant conditions were maintained throughout the event.

C. CAUSE OF EVENT:

The intermediate cause of this event was the urgent alarm produced by the Logic Cabinet. The root cause of this event was the failure of integrated circuit Z6 on the Supervisory Logic I Card. This failure caused an erratic pulse frequency to be sent to the CRDMs and an eventual "Go While Cycling" error. Electrical circuit failures of this type occur randomly at a very low frequency. This particular card had never failed before and the failure can be considered an isolated event.

D. SAFETY ANALYSIS:

There were no safety consequences as a result of this event. The health and safety of the public were at no time adversely affected or threatened. The control rods remained capable of being tripped at all times during this event. If this event had occurred under a more severe set of circumstances there would have been no safety consequences as the rods would still be capable of being tripped.

E. CORRECTIVE ACTIONS:

The corrective action was to replace the Supervisory Logic I Card in the Logic Cabinet, whereas the Pulser Oscillator Card, Master Cycler Logic Card, and Master Cycler Selector Card were replaced as a conservative measure.

F. RECURRING EVENTS SEARCH AND ANALYSIS:

a) EVENT SEARCH (DIR, LER)

None.

b) INDUSTRY SEARCH (OPEX's NPRDS)

No occurrences of a failed Supervisory Logic I Card have been found.

c) NWR

No occurrences of a failed Supervisory Logic I Card have been found.

d) ANALYSIS

No adverse trend indicated.

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Westinghouse	Supervisory Logic I Card		2260C97G01

H. OTHER RELATED DOCUMENTS:

None.

I. EFFECTIVENESS REVIEW:

None Scheduled.

J. ADDITIONAL DATA:

a) Affected Technical Specification: 3.1.3.1

b) Procedures: None.

c) Equipment Involved: See component failure data.

d) Other: Rod Urgent Failure Alarm.

RD11 POWER CABINET FAILURE

TYPE: GENERIC, RB

A) 1AC
B) 1BD
C) 2AC
D) 2BD
E) SD C, D, E

CAUSE: REGULATION FAILURE

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES ROD MOTION IN AUTO AND MANUAL TO STOP. ANNUNCIATOR 10-C6 "ROD CONT URGENT FAILURE" ACTUATES. THE OPERATOR MAY USE THE BANK SELECT SWITCH IN BANK SELECT TO MOVE ANY RODS THAT ARE NOT IN THE SELECTED CABINET.

MALFUNCTION REMOVAL RESTORES ROD CONTROL TO NORMAL OPERATION.

EVENTS: 1) DVR 06-01-86-163

DEVIATION INVESTIGATION REPORT

TITLE

ROD CONTROL URGENT ALARMS DUE TO CIRCUIT CARD FAILURES

PAGE 1 OF 0|2

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	OPERATING MODE	2
1|0	0|2	8|6	0|6	0|1	8|6	- 1|6|3 -	0|0	1|1	1|4	8|6	POWER LEVEL	0|0|0

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
Don Brindle, U2 Operating Engineer Ext. 2018	8|1|5	2|3|4|-|5|4|4|1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	R|D	|X|F|7	W|1|2|0	Yes					

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
_ YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

MODE 2 - Startup Rx Power 0 RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

No systems were inoperable at the beginning of this event which contributed to this event. On 10/02/86 at 0155 during a reactor start up with control bank "B" at 115 steps withdrawn, alarm window 1-10-C06 "ROD CONTROL URGENT FAILURE" annunciated. 1BOA ROD-2 was initiated and Limiting Condition for Operation Action Response (LCOAR) 1BOS 1.3.1-1a was entered into due to "All rods being inoperable but trippable". Circuit cards in the 2AC power cabinet were replaced by Instrument maintenance and the alarm was cleared. LCOAR 1BOS 1.3.1-1a was exited at 0550 on 10/02/86. Start up was resumed until a second urgent alarm occurred at 0650. A rod drop and manual reactor trip occurred during the second urgent alarm. Additional corrective action was taken and start up was resumed again. For further information see LER 86-028-00.

C. CAUSE OF EVENT:

Initially the root cause of the event was thought to be failures in the following cards in the 2AC power cabinet; regulation circuit cards for the stationary, moving and lift coils, and the firing circuit card for the moving coils. It was later discovered that a combination of an intermittently failing Alarm Circuit Card may have contributed to the urgent alarm. For further information see LER 86-028-00.

D. SAFETY ANALYSIS:

At all times all rods remained trippable and at no time was the health or safety of the public adversely affected.

(0900M)

RD12 ROD STOPS FAIL

TYPE: GENERIC, RB

A) IR C-1
B) PR C-2
C) OTDT C-3
D) OPDT C-4
E) LOW POWER C-5
F) BANK D WITHDRAWAL C-11

CAUSE: ROD STOP RELAY FAILURE

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR IN THE INTERMEDIATE RANGE

EFFECTS: INSERTING EITHER OF THESE MALFUNCTIONS CAUSES NO IMMEDIATE NOTICEABLE EFFECTS.

WHEN ONE OF THE TWO INTERMEDIATE RANGE NUCLEAR INSTRUMENTS EXCEEDS THE C-1 SETPOINT (CURRENT EQUIVALENT TO 20% REACTOR POWER), ANNUNCIATOR 10-A2 "IR HIGH FLUX ROD STOP C-1" ACTUATES. WITH THE CONTROL RODS MOVING OUTWARD, THE C-1 ROD STOP WILL NOT STOP ROD MOTION AS REQUIRED. THE CONTROL RODS WILL STILL BE ABLE TO BE WITHDRAWN IN EITHER THE MANUAL OR AUTOMATIC MODE. CONTINUED POWER INCREASE, WITH THIS MALFUNCTION ACTIVE, WILL RESULT IN AN INTERMEDIATE RANGE REACTOR TRIP AT CURRENT EQUIVALENT TO 25% REACTOR POWER.

WHEN ONE OF THE FOUR POWER RANGE NUCLEAR INSTRUMENTS EXCEEDS THE C-2 SETPOINT (103% REACTOR POWER), ANNUNCIATOR 10-B5 "PWR RNG FLUX HIGH ROD STOP" ACTUATES. WITH THE CONTROL RODS MOVING OUTWARD, THE C-2 ROD STOP WILL NOT STOP ROD MOTION AS REQUIRED. THE CONTROL RODS WILL STILL BE ABLE TO BE WITHDRAWN IN EITHER THE MANUAL OR AUTOMATIC MODE. CONTINUED POWER INCREASE, WITH THIS MALFUNCTION ACTIVE, COULD RESULT IN A POWER RANGE REACTOR TRIP AT 109% REACTOR POWER.

WHEN TWO OF THE FOUR LOOP DELTA T'S EXCEEDS THE C-3 SETPOINT (WITHIN 3% OF THE OTDT RX TRIP SETPOINT), ANNUNCIATOR 10-C5 "OTDT HIGH ROD STOP C-3" ACTUATES. WITH THE CONTROL RODS MOVING OUTWARD, THE C-3 ROD STOP WILL NOT STOP ROD MOTION AS REQUIRED. THE TURBINE RUNBACK IS NOT AFFECTED. THE CONTROL RODS WILL STILL BE ABLE TO BE WITHDRAWN IN EITHER THE MANUAL OR AUTOMATIC MODE. CONTINUED POWER INCREASE, WITH THIS MALFUNCTION ACTIVE, WILL RESULT IN A OVERTEMPERATURE DELTA T REACTOR TRIP WHEN THE SETPOINT IS REACHED.

WHEN TWO OF THE FOUR LOOP DELTA T'S EXCEEDS THE C-4 SETPOINT (<3% OF THE OPDT RX TRIP SETPOINT), ANNUNCIATOR 10-A5 "OPDT HIGH ROD STOP C-4" ACTUATES. WITH THE CONTROL RODS MOVING OUTWARD, THE C-4 ROD STOP WILL NOT STOP ROD MOTION AS REQUIRED. THE TURBINE RUNBACK IS NOT AFFECTED. THE CONTROL RODS WILL STILL BE ABLE TO BE WITHDRAWN IN EITHER THE MANUAL OR AUTOMATIC MODE. CONTINUED POWER INCREASE, WITH THIS MALFUNCTION ACTIVE, WILL RESULT IN A OVERPOWER DELTA T REACTOR TRIP WHEN THE SETPOINT IS REACHED.

WHEN THE TURBINE IMPULSE CHAMBER PRESSURE DECREASES BELOW THE C-5 SETPOINT (<15% TURBINE POWER) WITH THE CONTROL RODS MOVING OUTWARD IN THE AUTOMATIC MODE, THE C-5 ROD STOP WILL NOT STOP ROD MOTION AS REQUIRED. THE CONTROL RODS WILL STILL BE ABLE TO BE WITHDRAWN IN THE MANUAL OR AUTO MODE.

WHEN CONTROL ROD BANK D IS WITHDRAWN ABOVE THE C-11 SETPOINT (223 STEPS), ANNUNCIATOR 10-D5 "BANK D ROD STOP C-11" IS ACTUATED. WITH THE CONTROL RODS MOVING OUTWARD IN THE AUTOMATIC MODE, THE C-11 ROD STOP WILL NOT STOP ROD MOTION AS REQUIRED. THE CONTROL RODS WILL STILL BE ABLE TO BE WITHDRAWN IN THE MANUAL OR AUTO MODE.

MALFUNCTION REMOVAL RESTORES THE FAULTY ROD STOP RELAY TO NORMAL.

EVENTS: 1) DVR 06-02-87-072

DEVIATION INVESTIGATION REPORT

RD12

TITLE

FAILURE OF C-11 AUTO ROD STOP

EVENT DATE			DIR NUMBER						REPORT DATE			OPERATING MODE	POWER LEVEL
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER		REVISION NUMBER	MONTH	DAY	YEAR		
0 7	1 8	8 7	0 6	0 2	8 7	— 0 7 2	—	0 0	0 8	2 4	8 7	1	0 9 8

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
W. Kouba, Asst. Tech Staff Supervisor Ext. 2274	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	A A	A M P	W 1 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
_ YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 7/18/87 / 1834

Unit 1 MODE ___ - NA Rx Power NA RCS [AB] Temperature/Pressure NA

Unit 2 MODE 1 - Power Operation Rx Power 98% RCS [AB] Temperature/Pressure 584°F / 2200 psig

B. DESCRIPTION OF EVENT:

On 7/18/87 at 1834 hours, Startup Test 2.52.87, 10% Load Swing, was in progress. Turbine-generator power had been increased from 88% to 98% at a rate of 200%/minute per the test procedure. As a result of this power ramp Tave decreased and Tref increased. The rod control system was in automatic and attempted to restore Tave to match Tref by auto withdrawal of Control Bank D. Control Bank D was at 180 steps prior to the load increase and withdrew to an indicated 230 steps when the Unit 2 NSO placed rod control in manual, stopping Control Bank D motion. The C-11 interlock was suppose to block automatic rod withdrawal at 223 steps on Control Bank D but failed to do so. Subsequently, Control Bank D was returned to its normal operating range and the P/A converter, group demand step counters, bank overlap unit, and process computer rod supervision program indications were corrected to proper rod position within one hour. No safety system actuations resulted from this event, nor were any supposed to.

C. CAUSE OF EVENT:

During troubleshooting under Nuclear Work Request B47441 Zener Diode Z11-1 on summing amplifier 2ZY-442A was found to be failed. The diode was distorted in shape and was scorched. This diode serves to limit output voltage from this amplifier to +10 VDC. In its damaged state it was limiting output voltage to + 7.93 VDC. This had the effect of clipping the Control Bank D position signal which feeds C-11, at 182 steps. Any actual rod position exceeding 182 steps would be input as 182 steps to the C-11 voltage comparator 2AB-442C, which has an interlock setpoint of 223 steps (+9.697 VDC). Therefore, the C-11 setpoint would not be reached at any Control Bank D position. The cause of the diode failure is unknown and is attributed to a weak component.

TEXT

D. SAFETY ANALYSIS:

There were no adverse safety consequences as a result of this event. The reactor trip breakers were operable at all times, and all control rods were available to be tripped in if needed. Both automatic and manual control rod insertion capability was available at all times to control reactor power if needed. The reactivity worth of Control Bank D from the 223 step C-11 auto rod stop position to the fully withdrawn position is less than 1 pcm ($1 \times 10^{-5} \Delta K/K$) which had a negligible effect on reactor power response. The 2ZY -332A amplifier also feeds the Control bank D Low and Low-2 Rod Insertion Limit (RIC) alarms. In that the amplifier was found to work properly up to +7.93 VDC (182 steps) there was no adverse effect on RIL alarms. The highest RIL alarm occurs at 100% power at 171 steps on Control Bank D, which is below the 182 steps voltage clipping point. The resetting of the bank overlap unit counter ensured that proper overlap and other banks RILs would be met. Because rod travel was stopped at 230 steps, the overlap would be off by no more than two steps prior to the reset. The difference in physical rod position at 230 steps (as much as 231 steps is mechanically available) versus the 228 steps used for rod drop timing has a negligible impact because all Control Bank D rods have at least 0.99 sec. margin available with respect to the 2.40 sec. drop time test requirement.

E. CORRECTIVE ACTIONS:

Under Nuclear Work Request B47441 amplifier board 2ZY-442A was replaced. The new amplifier was calibrated per specification and the full range of proper output verified. The C-11 interlock at 223 steps on Control Bank D was also verified to properly function.

F. PREVIOUS OCCURRENCES:

LER NUMBER	TITLE
NONE	

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Westinghouse	NSA Summing Amplifier Card (7300 Series)	2837A14G01	Z11-1 (Style 743A403H11 10V Zener Diode)

b) RESULTS OF NPRDS SEARCH:

No Zener Diode failures were found.

RD13 DRPI - OPEN OR SHORTED COIL

TYPE: GENERIC, RB

NOTE: MALFUNCTIONS ENTERED AS "RD13AH06" & "RD13BH06" (CONTROL BANK RODS ONLY)

CB A ROD H6	(COIL A)	CB A ROD H6	(COIL B)
CB A ROD H10	(COIL A)	CB A ROD H10	(COIL B)
CB A ROD F8	(COIL A)	CB A ROD F8	(COIL B)
CB A ROD K8	(COIL A)	CB A ROD K8	(COIL B)
CB B ROD F2	(COIL A)	CB B ROD F2	(COIL B)
CB B ROD B10	(COIL A)	CB B ROD B10	(COIL B)
CB B ROD K14	(COIL A)	CB B ROD K14	(COIL B)
CB B ROD P6	(COIL A)	CB B ROD P6	(COIL B)
CB B ROD B6	(COIL A)	CB B ROD B6	(COIL B)
CB B ROD F14	(COIL A)	CB B ROD F14	(COIL B)
CB B ROD P10	(COIL A)	CB B ROD P10	(COIL B)
CB B ROD K2	(COIL A)	CB B ROD K2	(COIL B)
CB C ROD H2	(COIL A)	CB C ROD H2	(COIL B)
CB C ROD B8	(COIL A)	CB C ROD B8	(COIL B)
CB C ROD H14	(COIL A)	CB C ROD H14	(COIL B)
CB C ROD P8	(COIL A)	CB C ROD P8	(COIL B)
CB C ROD F6	(COIL A)	CB C ROD F6	(COIL B)
CB C ROD F10	(COIL A)	CB C ROD F10	(COIL B)
CB C ROD K10	(COIL A)	CB C ROD K10	(COIL B)
CB C ROD K6	(COIL A)	CB C ROD K6	(COIL B)
CB D ROD D4	(COIL A)	CB D ROD D4	(COIL B)
CB D ROD M12	(COIL A)	CB D ROD M12	(COIL B)
CB D ROD D12	(COIL A)	CB D ROD D12	(COIL B)
CB D ROD M4	(COIL A)	CB D ROD M4	(COIL B)
CB D ROD H8	(COIL A)	CB D ROD H8	(COIL B)

CAUSE: FAULTY COIL

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE DRPI A OR B DATA FOR THE SELECTED ROD(S) TO FAIL. THE ROD(S) WILL INDICATE THE POSITION BUT WITH ONLY DATA A OR DATA B (1/2) ACCURACY. REACTOR POWER WILL NOT CHANGE. THE SELECTED ROD "GENERAL WARNING" LED WILL BE LIT ON THE DRPI DISPLAY FOR DATA A AND/OR DATA B FAILURE. ANNUNCIATOR 10-D6 "ROD CONT NON-URGENT FAILURE" ACTUATES.

MALFUNCTION REMOVAL RESTORES THE SELECTED ROD(S) DRPI TO THE ACTUAL READING.

EVENTS: NONE

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

RH01	RHR PUMP FAILS TO START/TRIP
RH02	RHR HX FLOW CONTROL VALVE FAILURE
RH03	RHR HX BYPASS VALVE CONTROL FAILURE
RH04	RHR AUTO SWITCH-OVER MALFUNCTION
RH05	RWST LEVEL TRANSMITTER MALFUNCTION
RH06	RHR HX TUBE LEAK
RH07	RHR HX BYPASS LINE LEAK
RH08	RWST LEAK
RH09	RHR PUMP SUCTION HEADER BREAK
RH10	RHR PUMP DISCHARGE HEADER BREAK
RH11	SUCTION RELIEF VALVE FAILURE

RH01 RHR PUMP FAILS TO START/TRIP

TYPE: GENERIC, RB

A) 1A RH PUMP 1RH01PA
B) 1B RH PUMP 1RH01PB

CAUSE: FAULTY OVERCURRENT (450/451) RELAY ACTUATION

REF: 20E-1-4030 RH01,02

PLT STA: AFFECTED RH PUMP IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED RESIDUAL HEAT REMOVAL PUMP BREAKER TO TRIP. PUMP CURRENT INDICATION DECREASES TO ZERO. ANNUNCIATOR 6-A1 "RH PUMP TRIP" ACTUATES, AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. RESIDUAL HEAT REMOVAL PUMP DISCHARGE PRESSURE (1PI-614/615 @ 1PM06J), AND DISCHARGE FLOW (1FI-618/619 @ 1PM06J) DECREASES. THE LOSS OF RH COOLING TO THE REACTOR COOLANT SYSTEM WILL RESULT IN INCREASING RCS TEMPERATURE.

THE OPERATOR MAY RESET THE TRIP LIGHT BY PLACING THE CONTROL SWITCH IN THE TRIP POSITION. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL CLOSE THEN IMMEDIATELY TRIP OPEN.

MALFUNCTION REMOVAL RESTORES THE OVERCURRENT RELAY TO NORMAL OPERATION.

EVENTS: 1) SER 23-86

RHO1

IS 614 FORSYTH (INPO) 03-JUL-86 11:38 PT
Subject: SER 23-86, LOSS OF DECAY HEAT REMOVAL FLOW

SUBJECT: LOSS OF DECAY HEAT REMOVAL FLOW DUE TO INADEQUATE REACTOR COOLANT SYSTEM LEVEL CONTROL

UNIT (TYPE): CRYSTAL RIVER (PWR)
DOC NO/LER NO: 50-302/85003
EVENT DATE: 2/2/86
NSSS/AE: BABCOCK & WILCOX/GILBERT ASSOCIATES

REFERENCES: INPO SIGNIFICANT OPERATING EXPERIENCE REPORT (SOER) 85-4
INPO SIGNIFICANT EVENT REPORTS (SERs) 78-81, 87-81, 51-83, 60-83, 80-84, 17-86
OPERATIONS AND MAINTENANCE REMINDER (O&MR) NO. 295
NRC IE INFORMATION NOTICE 81-09

SUMMARY:

DECAY HEAT REMOVAL FLOW WAS LOST WHEN THE "1B" DECAY HEAT REMOVAL PUMP MOTOR BREAKER TRIPPED ON OVERLOAD. INVESTIGATION REVEALED THAT THE PUMP SHAFT HAD BROKEN DUE TO TORSIONAL FATIGUE. THE MOST LIKELY CAUSE OF THE FAILURE IS AIR ENTRAINMENT AT THE PUMP SUCTION THAT WAS THE RESULT OF EXTENDED OPERATION WITH INADEQUATE CONTROL OF THE REACTOR COOLANT SYSTEM WATER LEVEL. ATTEMPTS TO PLACE THE STANDBY DECAY HEAT REMOVAL TRAIN IN SERVICE WERE DELAYED 24 MINUTES WHEN THE SUCTION VALVE FROM THE REACTOR COOLANT SYSTEM DROP LINE TO THE STANDBY TRAIN FAILED TO OPEN DUE TO A TRIPPED BREAKER. AN ELECTRICAL SHORT EXISTED IN THE MOTOR OPERATOR FOR THE STANDBY DECAY HEAT REMOVAL TRAIN SUCTION VALVE.

THIS EVENT IS SIGNIFICANT BECAUSE IT DEMONSTRATES HOW SHUTDOWN CORE COOLING FLOW CAN BE LOST AND DECAY HEAT REMOVAL EQUIPMENT DAMAGED WHILE OPERATING WITH THE REACTOR COOLANT SYSTEM PARTIALLY DRAINED.

DESCRIPTION:

CRYSTAL RIVER UNIT 3 WAS SHUTDOWN TO REPLACE ALL FOUR REACTOR COOLANT PUMP SHAFTS. THE DECAY HEAT REMOVAL SYSTEM WAS IN OPERATION WITH THE "1B" PUMP RUNNING. THE REACTOR VESSEL HEAD WAS IN PLACE, AND THE REACTOR COOLANT SYSTEM TEMPERATURE MEASURED AT THE INCORE THERMOCOUPLES WAS BEING MAINTAINED AT 98 DEGREES FAHRENHEIT. REACTOR COOLANT SYSTEM WATER LEVEL HAD BEEN LOWERED TO THE HOT-LEG MIDPOINT FOR REACTOR COOLANT PUMP REMOVAL.

AT 2148 HOURS ON FEBRUARY 2, 1986, THE "1B" DECAY HEAT REMOVAL PUMP MOTOR BREAKER TRIPPED DUE TO ELECTRICAL OVERLOAD. ATTEMPTS TO PLACE THE STANDBvPjRAIN IN SERVICE WERE DELAYED 24 MINUTES WHEN THE SUCTION VALVE FROM THE REACTOR COOLANT SYSTEM DROP LINE TO THE "1A" DECAY HEAT REMOVAL PUMP (DHV-39) COULD NOT BE OPENED ELECTRICALLY DUE TO A TRIPPED MOTOR BREAKER. THE PLANT OPERATORS MANUALLY OPENED VALVE DHV-39, AND THE "1A" PUMP WAS PLACED IN SERVICE. THE REACTOR COOLANT TEMPERATURE INCREASED TO 131 DEGREES FAHRFNHEIT DURING THE PERIOD THAT FLOW WAS LOST.

INSPECTION OF THE "1B" DECAY HEAT REMOVAL PUMP REVEALED THAT THE PUMP SHAFT HAD BROKEN. TESTING PERFORMED ON THE MOTOR OPERATOR FOR DHV-39 INDICATED THAT AN ELECTRICAL SHORT EXISTED THAT RESULTED IN AN IMBALANCE BETWEEN THE PHASES OF THE MOTOR.

A REVIEW OF THE SHIFT OPERATING LOGS INDICATED THAT THE DECAY HEAT REMOVAL SYSTEM HAD BEEN OPERATED FOR APPROXIMATELY 30 DAYS WITH THE REACTOR COOLANT SYSTEM WATER LEVEL BELOW THE MINIMUM ALLOWED BY THE "RCS DRAINING AND NITROGEN BLANKETING" PROCEDURE. THE PROCEDURE IN USE, "DECAY HEAT REMOVAL SYSTEM", DID NOT CLEARLY IDENTIFY THE MINIMUM ALLOWABLE LEVEL OR THE FLOW-RATE ASSOCIATED WITH THE MINIMUM LEVEL.

ON FEBRUARY 14, 1986, AFTER THE "1B" DECAY HEAT REMOVAL PUMP SHAFT HAD BEEN REPLACED, REFILL OF THE "B" DECAY HEAT REMOVAL TRAIN WAS IN PROGRESS. AT THIS TIME, PLANT PERSONNEL OBSERVED MOVEMENT OF THE PIPING. SEVERAL RESTRAINTS IN THE VICINITY OF THE "1B" PUMP WERE FOUND LOOSE OR DAMAGED. IT IS SUSPECTED THAT THE PIPE HANGER DAMAGE TOOK PLACE WHEN THE PUMP FAILED.

ANALYSIS INDICATED THAT THE DECAY HEAT REMOVAL PUMP SHAFT FAILED DUE TO TORSIONAL FATIGUE. THIS FATIGjE FAILURE MOST LIKELY RESULTED FROM CAVITATION CAUSED BY AIR ENTRAINMENT AT THE PUMP SUCTION. WHEN THE LEVEL IN THE HOT-LEG PIPING DROPS TO A CERTAIN POINT, A VORTEX WILL FORM ABOVE THE DECAY HEAT REMOVAL SUCTION INLET AND ALLOW AIR TO BE DRAWN INTO THE SYSTEM. THE PLANT HAS CONCLUDED THAT BOTH THE SHAFT FAILURE AND THE DAMAGE TO THE DECAY HEAT REMOVAL SYSTEM PIPING RESTRAINTS MAY BE THE RESULT OF OPERATIONS WITH LOW REACTOR COOLANT SYSTEM WATER LEVEL.

THE CAUSE OF FAILURE OF THE DHV-39 VALVE OPERATOR MOTOR CANNOT BE POSITIVELY IDENTIFIED. THE SETTING FOR THE BREAKER MAGNETIC TRIP WAS BELOW THE REQUIRED VALUE FOR THIS APPLICATION. THE OPERATOR FOR DHV-39 IS LOCATED AT A DISTANCE FROM THE VALVE, AND THE VALVE IS DRIVEN BY REACH RODS WITH JOINTED COUPLINGS. PREVENTIVE MAINTENANCE PROCEDURES ARE BEING CHANGED TO REQUIRE MORE FREQUENT LUBRICATION OF THIS DRIVING ARRANGEMENT TO PREVENT MECHANICAL BINDING.

RH02 RHR HX FLOW CONTROL VALVE FAILURE

TYPE: GENERIC, RV 0-100% CONTROLLER OUTPUT

A) 1RH606
B) 1RH607

CAUSE: CONTROLLER OUTPUT FAILURE

REF: M-2062 SHEET 2

PLT STA: SELECTED RH HX ON-LINE

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED RH HEAT - EXCHANGER FLOW CONTROL VALVE TO FAIL. THE FAILED VALVE POSITION WILL BE DETERMINED BY THE SELECTED SEVERITY. IF THE SELECTED SEVERITY IS LESS THAN THE INITIAL VALVE POSITION, THE AFFECTED FLOW CONTROL VALVE WILL THROTTLE CLOSE, DECREASING RH SYSTEM FLOW THROUGH THE AFFECTED RH HEAT EXCHANGER, AND INCREASING THE BYPASS FLOW. THE DECREASED FLOW WILL BE INDICATED BY THE INCREASING RCS TEMPERATURE.

IF THE SELECTED SEVERITY IS GREATER THAN THE INITIAL VALVE POSITION, THE AFFECTED FLOW CONTROL VALVE WILL THROTTLE OPEN, INCREASING RH SYSTEM FLOW THROUGH THE RH HEAT EXCHANGER, AND DECREASING THE BYPASS FLOW. THE INCREASED FLOW WILL BE INDICATED BY THE DECREASING RCS TEMPERATURE. THE DECREASE IN REACTOR COOLANT SYSTEM TEMPERATURE MAY RESULT IN DECREASING PRESSURIZER LEVEL AND PRESSURE.

MALFUNCTION REMOVAL RESTORES THE AFFECTED RH HEAT EXCHANGER FLOW CONTROL VALVE TO NORMAL.

EVENTS: NONE.

RH03 RHR HX BYPASS VALVE CONTROL FAILURE

TYPE: GENERIC, RV 0-5000 GPM (50 GPM/1% VALVE POSITION)

A) 1RH0618
B) 1RH0619

CAUSE: CONTROLLER OUTPUT FAILURE (AUTO & MANUAL)

REF: M-2062 SHEET 2

PLT STA: SELECTED RH HX ON-LINE

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED RH HEAT EXCHANGER BYPASS FLOW CONTROL VALVE TO FAIL. THE FAILED VALVE POSITION WILL BE DETERMINED BY THE SELECTED SEVERITY. IF THE SELECTED SEVERITY IS GREATER THAN THE INITIAL VALVE POSITION, THE BYPASS VALVE OPENS, SLIGHTLY DECREASING RH SYSTEM FLOW THROUGH THE AFFECTED HEAT EXCHANGER, AND INCREASING THE TOTAL RH FLOW. THE DECREASED HX FLOW WILL BE INDICATED BY THE INCREASING RCS TEMPERATURE, AND RCS PRESSURE IF SOLID.

IF THE SELECTED SEVERITY IS LOWER THAN THE INITIAL VALVE POSITION, THE BYPASS VALVE WILL CLOSE, SLIGHTLY INCREASING RH SYSTEM FLOW THROUGH THE HEAT EXCHANGER, AND DECREASING THE BYPASS FLOW. THE INCREASED RH FLOW WILL BE INDICATED BY THE DECREASING RCS TEMPERATURE. THE DECREASE IN REACTOR COOLANT SYSTEM TEMPERATURE WILL RESULT IN DECREASING PRESSURIZER LEVEL AND PRESSURE.

MALFUNCTION REMOVAL RESTORES THE AFFECTED RH HEAT EXCHANGER FLOW CONTROL VALVE TO NORMAL.

EVENTS: NONE.

RH04 RHR AUTO SWITCH-OVER MALFUNCTION

TYPE: GENERIC, RB

A) 1SI8811A
B) 1SI8811B

CAUSE: K648 CONTACT FAILURE

REF: 20E-1-4030 SI14
20E-1-4030 EF11
20E-1-4030 EF60

PLT STA: LOCA IN PROGRESS

EFFECTS: INSERTING THIS MALFUNCTION CAUSES NO IMMEDIATE EFFECTS UNTIL THE RWST LO-2 LEVEL IS REACHED. AUTO SWITCH-OVER FROM THE RWST TO THE CONTAINMENT SUMP DOES NOT OCCUR, CAUSING THE RH PUMPS TO PUMP THE RWST LEVEL DOWN. RH PUMP SUCTION PRESSURE DECREASES CAUSING CAVITATION AT LOW SUCTION PRESSURE. RCS TEMPERATURE, AND PRESSURE INCREASE DUE TO THE LACK OF COOLING.

EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY MANUALLY CLOSING THE ASSOCIATED SI8812A/B, RH8701/RH8702 AND CS001A/B. THEN MANUALLY OPENING THE ASSOCIATED CONTAINMENT SUMP SUCTION VALVE.

MALFUNCTION REMOVAL RESTORES THE K648 CONTACT TO NORMAL.

EVENTS: NONE.

RH05 RWST LEVEL TRANSMITTER MALFUNCTION

TYPE: GENERIC, RV 0-100%

A) 1LT-SI930
B) 1LT-SI931
C) 1LT-SI932
D) 1LT-SI933

CAUSE: TRANSMITTER FAILURE

REF: M-2061 SHEET 2

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED LEVEL TRANSMITTER TO FAIL AT THE SELECTED SEVERITY LEVEL. THE FOLLOWING LISTS THE FUNCTIONS OF THE TRANSMITTERS:

1LT-930 & 1LT-931:

ANNUNCIATOR 6-A6 "RWST LEVEL HIGH"
ANNUNCIATOR 6-C7 "RWST LEVEL LOW"
ANNUNCIATOR 6-B7 "RWST LEVEL LO-2"
ANNUNCIATOR 6-A7 "RWST LEVEL EMPTY"
ESF LOGIC 2/4 TRANSFER TO CONTAINMENT SUMP
1PM06J LEVEL INDICATION
1PM06J LEVEL RECORDER
1PM06J RWST EMPTY LIGHTS

1LT-932 & 1LT-933:

ANNUNCIATOR 6-A6 "RWST LEVEL HIGH"
ANNUNCIATOR 6-C7 "RWST LEVEL LOW"
ANNUNCIATOR 6-B7 "RWST LEVEL LO-2"
ANNUNCIATOR 6-A7 "RWST LEVEL EMPTY"
ESF LOGIC 2/4 TRANSFER TO CONTAINMENT SUMP
1PM06J LEVEL INDICATION

MALFUNCTION REMOVAL RESTORES THE SELECTED TRANSMITTER TO NORMAL.

EVENTS: NONE.

RH06 RHR HX TUBE LEAK

TYPE: GENERIC, RV 0-500 GPM @ 300 PSID

A) 1A RH HX 1RH02AA
B) 1B RH HX 1RH02AB

CAUSE: TUBE FAILURE AT INLET TUBE SHEET

REF: M-62

PLT STA: SELECTED RH HX ON-LINE

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A LOSS OF MASS FROM THE RESIDUAL HEAT REMOVAL SYSTEM TO THE COMPONENT COOLING WATER SYSTEM. THE MASS BEING TRANSFERRED TO THE COMPONENT COOLING WATER SYSTEM WILL CAUSE THE CCW SURGE TANK LEVEL TO INCREASE AT A RATE DETERMINED BY THE SELECTED SEVERITY. ANNUNCIATOR 2-A5 "CC SURGE TANK LEVEL HIGH LOW" ACTUATES. THE CONTAMINATED RH WATER WILL CAUSE AN INCREASE IN CCW SYSTEM ACTIVITY LEVELS ON 1/0PR09J; CLOSING VENT VALVE 1CC017.

THE LOSS OF FLUID TO THE CCW SYSTEM REDUCES THE AMOUNT OF WATER PASSING THROUGH THE AFFECTED RH HEAT EXCHANGER BACK TO THE REACTOR COOLANT SYSTEM. THE ELEVATED TEMPERATURE OUT OF THE AFFECTED RH HEAT EXCHANGER RESULTS IN INCREASING RCS TEMPERATURE AND POSSIBLE DECREASING PRESSURIZER LEVEL AND PRESSURE, DEPENDING ON THE SELECTED SEVERITY.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY ISOLATING THE AFFECTED RH HEAT EXCHANGER AND PLACING THE OTHER RH HEAT EXCHANGER IN SERVICE.

MALFUNCTION REMOVAL ONLY RESTORES THE RESIDUAL HEAT REMOVAL SYSTEM PIPING INTEGRITY.

EVENTS: NONE.

RH07 RHR HX BYPASS LINE LEAK

TYPE: GENERIC, RV 0-500 GPM @ 300 PSID

A) 1A RH LOOP
B) 1B RH LOOP

CAUSE: PIPE BREAK IMMEDIATELY UPSTREAM 1RH618 (619)

REF: M-62

PLT STA: RESIDUAL HEAT REMOVAL SYSTEM IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A LOSS OF MASS FROM THE RESIDUAL HEAT REMOVAL SYSTEM TO THE AUXILIARY BUILDING. THE RATE OF MASS LOSS WILL BE DETERMINED BY THE SELECTED SEVERITY. ACTIVITY LEVELS IN THE AUXILIARY BUILDING IN THE LOCAL AREA AND IN THE VENTILATION SYSTEM FLOW PATH WILL INCREASE. THE LOSS OF MASS RETURNING TO THE REACTOR COOLANT SYSTEM WILL CAUSE A DECREASE IN PRESSURIZER LEVEL AND PRESSURE. THE DECREASE IN COOLED WATER FLOW RETURNING TO THE REACTOR COOLANT SYSTEM WILL RESULT IN AN INCREASE IN RCS TEMPERATURE. THE LOSS OF SYSTEM INTEGRITY WILL BE INDICATED BY THE DECREASED DISCHARGE PRESSURE ON 1PI-614/615 (1PM06J). AS MALFUNCTION SEVERITY IS INCREASED, THE RUNNING RH PUMP(S) AMPS WILL INCREASE AS ACTUAL FLOW THROUGH THE PUMP(S) INCREASES.

THE OPERATOR MAY LIMIT THE LEAK CONSEQUENCES OF THIS MALFUNCTION BY PLACING THE OTHER RH HX ON-LINE.

MALFUNCTION REMOVAL ONLY RESTORES THE RH HEAT EXCHANGER BYPASS LINE PIPING INTEGRITY.

EVENTS: NONE.

RH08 RWST LEAK

TYPE: DISCRETE, RV 0-100,000 GPM

CAUSE: TANK FAILURE

REF: M-61 SHEET 1B

PLT STA: RH ALIGNED FOR ECCS CL INJECTION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A LEAK FROM THE RWST TO THE AUXILIARY BLDG. LEAK RATE IS DETERMINED BY THE SEVERITY SELECTED. AS THE SEVERITY IS INCREASED, THE NPSH DECREASES CAUSING THE PUMPS TO CAVITATE AS INDICATED BY PRESSURE AND FLOW OSCILLATIONS. ANNUNCIATOR 6-C7 "RWST LEVEL LOW" ACTUATES AND LO-2 LEVEL ACTUATES AN AUTOMATIC TRANSFER TO THE CONTAINMENT SUMP PROVIDED AN SI SIGNAL IS PRESENT.

MALFUNCTION REMOVAL ONLY RESTORES THE TANK INTEGRITY TO NORMAL.

EVENTS: NONE.

RH09 RHR PUMP SUCTION HEADER BREAK

TYPE: GENERIC, RV 0-5000 GPM AT 350 PSID

A) TRAIN A RH SUCTION HEADER
B) TRAIN B RH SUCTION HEADER

CAUSE: PIPING FAILURE DOWNSTREAM OF MOTOR OPERATED VALVES 1RH8701A, AND 1RH-8702A RESPECTIVELY.

REF: M-62 SHEET 1

PLT STA: RH SYSTEM IN OPERATION

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, A LOSS OF MASS FROM THE RESIDUAL HEAT REMOVAL SYSTEM TO THE AUXILIARY BUILDING WILL RESULT. THE RATE OF MASS LOSS WILL BE DETERMINED BY THE SELECTED SEVERITY. ACTIVITY LEVELS IN THE AUXILIARY BUILDING, IN THE LOCAL AREA, AND IN THE VENTILATION SYSTEM FLOW PATH WILL INCREASE. THE LOSS OF MASS RETURNING TO THE REACTOR COOLANT SYSTEM WILL CAUSE A DECREASE IN PRESSURIZER LEVEL AND PRESSURE. THE LOSS OF SUCTION LINE MASS WILL CAUSE A DECREASE IN RH PUMP(S) DISCHARGE PRESSURE AND FLOW, AS INDICATED ON 1FI-618 (A PUMP) & 1FI-619 (B PUMP) AND 1PI-614/615 (A/B PUMP). THE DECREASED MASS RETURNING TO THE REACTOR COOLANT SYSTEM WILL CAUSE A DECREASE IN PRESSURIZER LEVEL AND PRESSURE. THE DECREASE IN COOLED WATER FLOW RETURNING TO THE REACTOR COOLANT SYSTEM, MAY RESULT IN AN INCREASE IN RCS TEMPERATURE.

THE OPERATOR MAY LIMIT THE LEAK CONSEQUENCES OF THIS MALFUNCTION BY SECURING THE RH PUMP AND MANUALLY ISOLATING THE RH SUCTION LINE.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE RH SUCTION LINE PIPING INTEGRITY.

EVENTS: NONE.

RH10 RHR PUMP DISCHARGE HEADER BREAK

TYPE: GENERIC, RV 0-5000 GPM AT 550 PSID

A) TRAIN A RH DISCHARGE HEADER
B) TRAIN B RH DISCHARGE HEADER

CAUSE: PIPING FAILURE IMMEDIATELY UPSTREAM OF 1SI8809A/B

REF: M-62 SHEET 1

PLT STA: RH SYSTEM IN OPERATION

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, A LOSS OF MASS FROM THE RESIDUAL HEAT REMOVAL SYSTEM TO THE AUXILIARY BUILDING WILL RESULT. THE RATE OF MASS LOSS WILL BE DETERMINED BY THE SELECTED SEVERITY. ACTIVITY LEVELS IN THE AUXILIARY BUILDING IN THE LOCAL AREA AND IN THE VENTILATION SYSTEM FLOW PATH WILL INCREASE. THE LOSS OF MASS RETURNING TO THE REACTOR COOLANT SYSTEM WILL CAUSE A DECREASE IN PRESSURIZER LEVEL AND PRESSURE. THE DECREASE IN COOLED WATER FLOW RETURNING TO THE REACTOR COOLANT SYSTEM WILL RESULT IN AN INCREASE IN RCS TEMPERATURE. THE LOSS OF SYSTEM INTEGRITY WILL BE INDICATED BY THE DECREASED DISCHARGE PRESSURE ON 1PI-614/615 (A/B PUMPS).

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY MANUALLY ISOLATING THE AFFECTED RH DISCHARGE HEADER AND PLACING THE UNAFFECTED RH TRAIN IN OPERATION.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE AFFECTED RH DISCHARGE HEADER LINE PIPING INTEGRITY.

EVENTS: NONE.

RH11 SUCTION RELIEF VALVE FAILURE

TYPE: GENERIC, RV 0-1500 GPM AT 400 PSID

A) TRAIN A RH SUCTION RELIEF 1RH-8708A
B) TRAIN B RH SUCTION RELIEF 1RH-8708B

CAUSE: MECHANICAL FAILURE OF RELIEF VALVE

REF: M-62 SHEET 1

PLT STA: RH SYSTEM IN OPERATION

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, A LOSS OF MASS FROM THE RH SYSTEM TO THE RECYCLE HOLDUP TANK WILL RESULT. DEPENDING ON THE RH PUMP SUCTION LINEUP, A MASS LOSS COULD OCCUR FROM EITHER THE REACTOR COOLANT SYSTEM OR THE RWST. THE MASS LOSS WILL BE INDICATED BY LEVEL DECREASING IN EITHER THE PRESSURIZER OR RWST DEPENDING ON THE LINEUP. THE RATE OF MASS LOSS WILL BE DETERMINED BY THE SELECTED SEVERITY. THE LOSS OF SUCTION LINE MASS WILL CAUSE A DECREASE IN PUMP(S) DISCHARGE PRESSURE AND FLOW AS INDICATED ON 1FI-618 (A PUMP) & 1FI-619 (B PUMP) AND 1PI-614/615 (A/B PUMP). THE DECREASED MASS RETURNING TO THE REACTOR COOLANT SYSTEM WILL CAUSE A DECREASE IN PRESSURIZER LEVEL AND PRESSURE. THE DECREASE IN COOLED WATER FLOW RETURNING TO THE REACTOR COOLANT SYSTEM, WILL RESULT IN AN INCREASE IN RCS TEMPERATURE.

THE OPERATOR MAY LIMIT THE LEAK CONSEQUENCES OF THIS MALFUNCTION BY SECURING THE RH PUMP AND MANUALLY ISOLATING THE RH SUCTION LINE.

MALFUNCTION REMOVAL WILL ONLY RESTORE THE FAILED RH SUCTION RELIEF VALVE TO NORMAL.

EVENTS: 1) LER 20-01-88-008

LICENSEE EVENT REPORT (LER)

KH11

Facility Name (1)	Docket Number (2)	Page (3)
Braidwood, Unit 1	0 5 0 0 0 4 5 6	1 of 0 5

Title (4) Reactor Coolant System Leakage Due to Broken Relief Valve Disc Pin

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
0 3	2 5	8 8	8 8 ---	0 0 8 ---	0 0	0 4	2 5	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 4

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	__ 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	X 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER AREA CODE	
Terry O'Brien, Technical Staff Engineer Ext. 2440	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	B P	R V * *	C 7 1 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

On March 25, 1988 and March 27,1988, Operators noted a decreasing volume control tank level which caused increased make-up. Reactor coolant water inventory balance surveillances confirmed that unidentified leakage was in excess of 1 gallon per minute (GPM). The source of the March 25, 1988 occurrence was thought to be an improperly locked closed valve which was inadvertently bumped off its closed seat. The Residual Heat Removal (RHR) pump suction relief valves may have contributed to the Generating Station Emergency Plan Unusual Event for both occurrences. Leakage past the seats by measuring the downstream temperature indicated the source of leakage. Subsequent investigation of one of the relief valves indicated that the disc insert pin was broken as a result of improper nozzle ring setting. The 1A RHR suction relief valve has been repaired and reinstalled. The 1B relief valve will be tested and repaired as necessary prior to restart of the unit. There have been no previous occurrences of Crosby Relief Valve failures.

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

RM01 AREA RADIATION MONITOR ACTUATION

TYPE: GENERIC, RV 1.0 E-05 TO 1.0 E+05 mr/HR

NOTE: LOGARITHMIC SCALE MODELED LINEARLY

A)	0RE-AR001	AE)	0RE-AR049
B)	0RE-AR002	AF)	0RE-AR050
C)	0RE-AR003	AG)	0RE-AR055
D)	0RE-AR004	AH)	0RE-AR056
E)	0RE-AR005	AI)	0RE-AR073
F)	0RE-AR006	AJ)	0RE-AR074
G)	0RE-AR007	AK)	1RE-AR001
H)	0RE-AR008	AL)	1RE-AR002
I)	0RE-AR009	AM)	1RE-AR003
J)	0RE-AR010	AN)	1RE-AR010
K)	0RE-AR011	AO)	1RE-AR011
L)	0RE-AR012	AP)	1RE-AR012
M)	0RE-AR013	AQ)	1RE-AR013
N)	0RE-AR014	AR)	1RE-AR022A
O)	0RE-AR015	AS)	1RE-AR022B
P)	0RE-AR016	AT)	1RE-AR022C
Q)	0RE-AR017	AU)	1RE-AR022D
R)	0RE-AR031	AV)	1RE-AR023A
S)	0RE-AR032	AW)	1RE-AR023B
T)	0RE-AR035	AX)	1RE-AR023C
U)	0RE-AR037	AY)	1RE-AR023D
V)	0RE-AR038	AZ)	1RE-AR024A
W)	0RE-AR041	BA)	1RE-AR024B
X)	0RE-AR042	BB)	1RE-AR025A
Y)	0RE-AR043	BC)	1RE-AR025B
Z)	0RE-AR044	BD)	1RE-AR026A
AA)	0RE-AR045	BE)	1RE-AR026B
AB)	0RE-AR046	BF)	1RE-AR027A
AC)	0RE-AR047	BG)	1RE-AR027B
AD)	0RE-AR048		

CAUSE: DETECTOR FAILURE

REF: AR/PR SYSTEM DESCRIPTION
M-78 SERIES

PLT STA: COLD SHUTDOWN

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED AREA RADIATION MONITOR TO FAIL. THE VALUE DISPLAYED IS DEPENDENT UPON THE SELECTED SEVERITY. IF THE SELECTED SEVERITY IS GREATER THAN THE ASSIGNED RM-11 ALERT SETPOINT, THEN THE RM-11 WILL ALARM AS ALERT. SELECTING A SEVERITY HIGHER THAN THE HIGH ALARM SETPOINT WILL CAUSE AN RM-11 HIGH ALARM. THE ONLY AREA MONITORS THAT HAVE INTERLOCKS ASSOCIATED WITH THEM ARE:

<u>0RE-AR055 & 0RE-AR056</u> - FUEL HANDLING BLDG FUEL HANDLING INCIDENT; FUEL HANDLING BLDG CHARCOAL BOOSTER FAN AUTO STARTS, AND FLOW IS DIRECTED THROUGH THE FILTER.

<u>1RE-AR011 & 1RE-AR012</u> - CNMT FUEL HANDLING INCIDENT; THE CONTAINMENT PURGE DAMPERS CLOSE.

MALFUNCTION REMOVAL RESTORES THE SELECTED AREA RADIATION MONITOR TO NORMAL.

EVENT: 1) LER 06-01-88-009
2) LER 20-01-88-003

KMO1

BRAIDWOOD-OPER
ACTION ITEM

DVR
20-1-88-01100

DATE 08 ·0·88
PAGE

ITEM NO 456-200-88-01100

OTHER UNIT NO

ITEM DATE: 01/13/88
MODE: 5

SCHEDULAR CAT: TEST CONDITION 1
COMMITMENT TO: NRC

CURRENT LOC (DEPT DNS   PERSON: CHOMACKE   DATE SENT 02/06/88)

SUBJECT: DVR 20-1-88-011, LOSS OF PULSES TO
FUEL HANDLING INCIDENT MONITOR
ORE-AR056 FOR UNKNOWN REASONS

TYPE: DEVIATION   SEVERITY LEVEL: LER NO 88-003   CRITERION

ORG CAUSING ITEM: BW
ORIG ORG/PERSON: OP /PROSPERO
RESP DEPT/SUPV : TSEL/STANCZAK
COG PERSON : /
COG PERSON : /
COG PERSON : /

RESPONSE DUE TO BY   DATE SET BY

INSPECTOR ARS
SYSTEM : IL IL
CORRECTIVE ACT
R/F OUTAGE :
PRIORITY :
TR: XIELEL4IM
BY/BW PROCEDURE:

TRANSMIT: BW: BY: D: L: Q: Z: NOD: DNS: ESS: PTC: NFS:

POWER: 0   ORIGINAL DUE DATE: 02/12/88
INTERIM REPORT: *
INTERIM REPORT: 10CFR50.73 (A)(2)(IV)
CLOSING REPORT:
SIGNATURE : E.E. FITZPATRICK

STATUS: COMPLETE
RDY FOR CLOSURE: 01/25/88
ORIG EXIT DATE : 01/29/88
ORIG CLOSED : 02/03/88
DATE COMPLETED : 02/17/88

REFER: DVR 20-1-88-011

DESCRIPTION:

AT 1910 ON 01/13/88 RAD MONITOR ORE-AR056 WENT INTO INTERLOCK
DUE TO A LOSS OF PULSES. THUS, THE TRAIN B FUEL HARDLING
CHARCOAL BOOSTER FAN STARTED AND THE CHARCOAL FILTER WAS PLACED
IN SERVICE. THE FAILURE OF ORE-AR056 WAS VERIFIED AS SPURIOUS
AND THE FUEL HANDLING CHARCOAL BOOSTER FAN WAS SHUTDOWN AND THE
FILTER RETURNED TO NORMAL
*
*
OPERATING ENGINEER'S COMMENTS
*
NONE W.B. MCCUE 01/14/88
*
30 DAY REPORTABLE/10CFR50.73 (A)(2)(IV)
*
LER NUMBER: 88-003

ITEM NO: 456-200-88-01100 (CONT)

DESCRIPTION (CONT):

ACTION SUMMARY:

A. PLANT CONDITIONS PRIOR TO EVENT:

*

UNIT: BRAIDWOOD 1, EVENT DATE: JANUARY 13, 1988; EVENT TIME: 1910
MODE: 5 - COLD SHUTDOWN; RX POWER: 0%; RCS (AB)
TEMPERATURE/PRESSURE: 100 DEGREES F/0 PSIG

*

*

B. DESCRIPTION OF EVENT:

*

THERE WERE NO SYSTEMS OR COMPONENTS INOPERABLE AT THE BEGINNING OF THE EVENT WHICH CONTRIBUTED TO THE SEVERITY OF THE EVENT. *
AT 1910 ON JANUARY 13, 1988, THE FUEL HANDLING BUILDING INCIDENT RADIATION MONITOR ORT-AR056 (IL) WENT INTO AN ALARM CONDITION ON A LOSS OF PULSES AS INDICATED AT THE CONTROL ROOM RADIATION MONITOR CONSOLE (RM-11). THIS STARTED THE AUXILIARY BUILDING VENTILATION (VF) FUEL HANDLING BUILDING CHARCOAL BOOSTER FAN 0VA04CA WITH THE FLOW THROUGH THE TRAIN B FUEL HANDLING BUILDING CHARCOAL FILTER. THE LOSS OF PULSES IMMEDIATELY CLEARED AND WAS CONSIDERED SPURIOUS. EQUIPMENT OPERAITON WAS IMMEDIATELY RETURNED TO NORMAL.

*

OPERATOR ACTION NEITHER INCREASED NOR DECREASED THE SEVERITY OF THE EVENT. PLANT CONDITIONS REMAINED STABLE THROUGHOUT THE EVENT.

*

THE APPROPRIATE NRC NOTIFICATION VIA THE ENS PHONE SYSTEM WAS MADE AT 1951 ON JANUARY 13, 1988, PURSUANT TO 10CFR50.72(B)(2)(II).

*

THIS EVENT IS BEING REPORTED PURSUANT TO 10CFR50.73(A)(2)(IV) - ANY EVENT OR CONDITION THAT RESULTED IN MANUAL OR AUTOMATIC ACTUATION OF ANY ENGINEERED SAFETY FEATURE, INCLUDING THE REACTOR PROTECTION SYSTEM. *

*

C. CAUSE OF EVENT:

*

THE ROOT CAUSE OF THE EVENT IS UNKNOWN. AN IMMEDIATE INVESTIGATION REVEALED NO WORK ACTIVITIES IN THE VICINITY OF MONITOR ORT-AR056. THE DETECTOR WAS INSPECTED, AND NO PHYSICAL DAMAGE WAS FOUND. THE DETECTOR CABLE WAS CHECKED FOR TIGHTNESS AND WAS ABLE TO BE TIGHTENED TWO TURNS. THIS CABLE SLACKNESS IS NOT CONSIDERED TO HAVE CAUSED THE LOSS OF PULSES WHICH RESULTED IN THE FUEL HANDLING BUILDING VENTILATION TO SHIFT TO ITS EMERGENCY MAKEUP MODE OF OPERATION. THE LOSS OF PULSES IMMEDIATELY CLEARED AND HAS NOT RECURRED.

*

*

D. SAFETY ANALYSIS:

ITEM NO: 456-200-88-01100 (CONT)

ACTION SUMMARY (CONT):

*

THERE WAS NO EFFECT ON THE SAFETY OF THE PLANT OR THE PUBLIC. [illegible] IS NO FUEL IN THE FUEL HANDLING BUILDING. BOTH UNIT 1 AND 2 [illegible] MODE 5.

*

HAD THIS EVENT OCCURRED UNDER WORST CASE CONDITIONS OF THE UNITS OPERATING WITH SPENT FUEL IN THE POOL, THERE WOULD BE NO EFFECT ON PLANT OR PUBLIC SAFETY. THE CHARCOAL BOOSTER FANS AND FILTER ARE DESIGNED TO ACTIVATE ON A FAILURE OF ORT-AR056 OR THE PRESENCE OF ACTUAL RADIATION. REDUNDANT MONITOR ORT-AR055 WAS AVAILABLE THROUGHOUT THE EVENT. *

*

E. CORRECTIVE ACTIONS:

*

THE IMMEDIATE CORRECTIVE ACTION WAS TO DETERMINE THAT THE SOURCE OF THE ACTUATION WAS SPURIOUS IN NATURE AND NOT DUE TO ACUTAL RADIOACTIVITY.

*

WORK REQUEST A19066 WAS WRITTEN TO FURTHER INSPECT THE MONITOR. IF THIS INVESTIGATION REVEALS ANY ADDITIONAL INFORMATION, IT WILL BE DOCUMENTED IN A SUPPLEMENT TO THIS REPORT. THIS WILL BE TRACKED TO COMPLETION BY ACTION ITEM 456-200-88-01101.

*

*

F. PREVIOUS OCCURRENCES:

*

DVR/LER NUMBER	TITLE
DVR 20-1-87-009 LER 87-003	CONTAINMENT VENTILATION ISOLATION SIGNAL DUE TO LOSS OF PULSES FROM 1RE-AR012

*

THIS HAS BEEN THE ONLY PREVIOUS OCCURRENCE OF A LOSS OF PULSES TO A RADIATION MONITOR. HOWEVER, THIS EVENT WAS DUE TO A FAILURE OF THE MONITOR'S SENSOR AS A RESULT OF CONSTRUCTION ACTIVITY WHICH PHYSICALLY DAMAGED IT.

*

*

G. COMPONENT FAILURE DATA:

*

COMPONENT FAILURE WAS NEITHER THE CAUSE NOR THE RESULT OF THIS EVENT.

*

*

---END---

RMD1

LICENSEE EVENT REPORT (LER)

Facility Name (1)

Byron, Unit 1

Form Rev 2.0

Docket Number (2)	Page (3)
0 5 0 0 0 4 5 4	1 of 0 3

Title (4) AUTOMATIC FUEL HANDLING BUILDING BOOSTER FAN ACTUATION DUE TO HIGH RADIATION CONDITION CAUSED BY A RADIOACTIVE PARTICLE

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
1 0	2 1	8 8	8 8	0 0 9	0 0	1 1	1 7	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

___ 20.402(b)	___ 20.405(c)	X 50.73(a)(2)(iv)	___ 73.71(b)
___ 20.405(a)(1)(i)	___ 50.36(c)(1)	___ 50.73(a)(2)(v)	___ 73.71(c)
___ 20.405(a)(1)(ii)	___ 50.36(c)(2)	___ 50.73(a)(2)(vii)	___ Other (Specify in Abstract below and in Text)
___ 20.405(a)(1)(iii)	___ 50.73(a)(2)(i)	___ 50.73(a)(2)(viii)(A)	
___ 20.405(a)(1)(iv)	___ 50.73(a)(2)(ii)	___ 50.73(a)(2)(viii)(B)	
___ 20.405(a)(1)(v)	___ 50.73(a)(2)(iii)	___ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER	
	AREA CODE	
Lee Sues, Assistant Superintendent Technical Services, Extension 2214	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

X Yes (If yes, complete EXPECTED SUBMISSION DATE) ___ NO

Expected Submission Date (15)	Month	Day	Year
	0 2	0 1	8 9

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

Unit 1 was in Cold Shutdown at a Reactor Coolant System temperature of 88°F and pressure of 175 psig. A submersible pump and portable filter assembly had been installed to pump borated water from the Fuel Transfer Canal to the Spent Fuel Pool following refueling operations. The pumping operation commenced at about 0700 on October 21, 1988. At 1437, the Fuel Building Isolation - Radioactivity High and Criticality Area Radiation Monitor (ORE-AR055) interlocked and alarmed. The ORE-AR055 interlock caused an automatic start of the OA Fuel Handling Building (FHB) Booster Fan which is an Engineered Safety Features actuation. Pumping operations were stopped immediately and access to the Spent Fuel Pool was controlled to limit radiation exposure. Contact dose rate measurements on the filter housing indicated as high as 16 Rem/hour. General area dose rates in the FHB ranged between 5 and 13 millirem/hour.

The cause of the event was a radioactive particle that was entrained with the water at the submersible pump suction and ultimately was trapped in the filter. Contact dose rate on the particle was 85 Rem/hour.

The filter element was removed and disposed of as high level radioactive waste. The remaining water in the canal was surveyed and the pumping operation was conducted to completion. At 2111 the OA FHB Booster Fan was stopped. Actions taken to prevent recurrence of this event will be reported in a supplemental report.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 10/21/88 / 1437

Unit 1 MODE 5 - Cold Shutdown Rx Power 0% RCS [AB] Temperature/Pressure 88°F / 175 psig

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of the event that contributed to the event. Unit 1 was in the Cold Shutdown Operational Mode (Mode 5) at a Reactor Coolant System (RCS) [AB] temperature of 88°F and pressure of 175 PSIG. The Unit's second refueling outage was coming to an end and activities were in progress to restore the Unit to an available condition. A submersible pump and portable filter assembly had been installed to pump borated water, that had remained in the Fuel Transfer Canal [DF] following refueling operations, to the Spent Fuel Pool. Temporary radiation shielding had been installed on the portable filter assembly, which was physically located on the Spent Fuel Pool crane platform above the pool. Fuel Handling Operators (non-licensed) notified licensed operators in the Main Control Room of the evolution, so letdown flow from the Spent Fuel Pool could be established. The Fuel Handling Operators also notified a Radiation Chemistry Technician (RCT) who required operation of a continuous air monitor with alarm capability. The RCT measured the contact dose rate on the portable filter assembly as 160 millirem/hour and initiated performance of shiftly dose rate surveillance on the filter assembly.

The pumping operation commenced at approximately 0700 on October 21, 1988 and continued until about 1300 when the submersible pump was replaced with a higher capacity pump. Post outage testing by the Technical Staff required access to the transfer canal, therefore it was desirable to pump the canal dry more rapidly. The RCT measured the portable filter assembly contact dose rate during the pump swap as 160 to 180 millirem/hour. The pumping operation was resumed using the higher capacity pump. At 1437, the Fuel Building Isolation - Radioactivity High and Criticality Area Radiation Monitor (ORE-AR055) [IL] interlocked and alarmed on the RM-11 Radiation Monitor Display Console in the Main Control Room. The ORE-AR055 measured 8 millirem/hour. Its interlock setpoint is 2.5 millirem/hour. The ORE-AR055 interlock caused an automatic start of the OA Fuel Handling Building Booster Fan [VG] which is an Engineered Safety Features (ESF) actuation. Pumping operations were stopped immediately and radiation dose rates in the area were measured. Contact dose rates on the filter housing were 16 Rem/hour at one location and 5 to 10 Rem/hour at other locations. General area dose rates in the Fuel Building ranged between 5 and 13 millirem/hour. Personnel access to the Spent Fuel Pool area was controlled to limit radiation exposure and preparations were made to change the portable filter assembly. This event had no effect on plant stability. This Licensee Event Report (LER) is submitted pursuant to 10CFR50.73(a)(2)(iv) due to the automatic actuation of an ESF System.

C. CAUSE OF EVENT:

The cause of the event was a radioactive particle that was entrained with the water at the submersible pump suction and ultimately was trapped in the filter. When the filter housing was disassembled, the particle fell onto the plastic sheeting beneath the work area. Contact dose rate on the particle was 85 Rem/hour. The particle was probably deposited in the Fuel Transfer Canal during irradiated fuel movements.

D. SAFETY ANALYSIS:

The Fuel Handling Building (FHB) Ventilation System properly actuated to filter radioactive contaminants from the air in the FHB prior to exhausting the air to the environment. Actually there was no airborne contamination in the FHB, so the filtering function was not required. In any case the FHB Ventilation System actuation established a safer plant alignment with regard to limiting release of radioactive material to the environment. Therefore, there was no effect on public safety.

The abnormal radiation condition in the FHB was recognized immediately by Fuel Handling Operators, RCT's and Main Control Room Operators. Proper actions were taken immediately to minimize the radiological consequences of this incident to plant personnel. The Spent Fuel Pool area was established as a controlled access area, which ensured that no plant personnel exceeded any radiation exposure limits. Plant safety remained unaffected by this event. The initial conditions of this event could not hypothetically be more severe, such that the safety consequences of this event would be altered.

E. CORRECTIVE ACTIONS:

The filter element that entrapped the radioactive particle was removed and a new element was installed. All filters and floor coverings from the filter location were bagged and placed into a shielded drum. The drum was transferred to a High Level Radioactive Waste Storage Area. The filter housing and general area were surveyed to verify the removal of all particles.

The water remaining (about 6-inches deep) in the Fuel Transfer Canal was surveyed using an underwater probe. The highest dose rate measured 150 millirem/hour. The filter was replaced and the remaining water was pumped from the Fuel Transfer Canal to the Spent Fuel Pool. During this evolution the OA FHB Booster Fan continued to operate to preclude another automatic ESF actuation. A remote reading radiation probe with alarm capability was attached to the filter housing to permit continuous monitoring of dose rate. The maximum contact dose rate achieved during this portion of the pumping evolution measured 200 millirem/hour. Upon completion of the pumping operation at 2111 the OA FHB Booster Fan was stopped. Preventive actions to be taken in response to this event, if any, have not been finalized. A supplemental LER will be submitted to report preventive actions, when they are finalized.

F. PREVIOUS OCCURRENCES:

LER NUMBER	TITLE
NONE	

G. COMPONENT FAILURE DATA:

a) MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Not Applicable			

RM02 INOPERABLE RADIATION MONITOR

TYPE: GENERIC, RB

A) 0RE-PR003
B) 0RE-PR011
C) 0RE-PR012
D) 0RE-PR013
E) 0RE-PR014
F) 0RE-PR015
G) 0RE-PR021
H) 0RE-PR022
I) 0RE-PR024
J) 0RE-PR025
K) 0RE-PR026
L) 0RE-PR031
M) 0RE-PR032
N) 0RE-PR033
O) 0RE-PR034
P) 0RE-PR035
Q) 0RE-PR036
R) 0RE-PR037
S) 0RE-PR038
T) 1RE-PR001
U) 1RE-PR011
V) 1RE-PR013
W) 1RE-PR014
X) 1RE-PR015
Y) 1RE-PR016
Z) 1RE-PR017
AA) 1RE-PR018
AB) 1RE-PR021
AC) 1RE-PR027
AD) 1RE-PR028
AE) 0RE-PR001
AF) 0RE-PR006
AG) 0RE-PR007
AH) 0RE-PR008
AI) 0RE-PR009
AJ) 0RE-PR010
AK) 0RE-PR016
AL) 0RE-PR017
AM) 0RE-PR018
AN) 0RE-PR019
AO) 1RE-PR002
AP) 1RE-PR003
AQ) 1RE-PR006
AR) 1RE-PR007
AS) 1RE-PR008
AT) 0RE-PR041
AU) 0RE-PR040
AV) 0RE-PR005

CAUSE: CLOGGED SAMPLE LINE

REF: AR/PR SYSTEM DESCRIPTION
M-78 SERIES
BwOP AR/PR 11-T1

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED RADIATION MONITOR WILL BECOME INOPERABLE DUE TO THE LOSS SAMPLE FLOW THROUGH THE MONITOR. THE FLUID WHICH HAD BEEN FLOWING BY THE DETECTOR WILL BECOME STAGNANT. THE ACTIVITY OF THAT FLUID WILL BE INDICATED ON THE RM-11. THAT ACTIVITY LEVEL WILL STEADILY DECREASE AS THE RADIOACTIVE ISOTOPES DECAY AWAY. THE RM-11 MONITOR LOSS OF SAMPLE FLOW WILL ALARM FOR THE SELECTED MONITOR. THIS ALARM IS INDICATED AS AN OPERATIONS FAILURE ON THE RM-11 DISPLAY WHICH IS COLOR CODED DARK BLUE. THE OPERATIONS FAILURE WILL CAUSE THE ASSOCIATED MONITOR TO GO INTO INTERLOCK AND ANY ASSOCIATED AUTOMATIC ACTIONS WILL OCCUR EXCEPT FOR VC WHICH REQUIRES 2/2 FAILURES. CLOGGING OF THE SAMPLE LINE WILL RESULT IN A VACUUM TRIP OF THE ASSOCIATED SAMPLE PUMP AFTER APPROXIMATELY 50 SECONDS.

MALFUNCTION REMOVAL WILL ALLOW FLOW RESTORATION THROUGH THE AFFECTED RADIATION MONITOR.

EVENTS: NONE

RM03 INADVERTANT AUTO RADIATION MONITOR ACTUATION

TYPE: GENERIC, RB

A)	0RE-PR026	M)	0RE-PR019
B)	0RE-PR031	N)	1RE-PR008
C)	0RE-PR032	* O)	0RE-AR055
D)	0RE-PR033	P)	0RE-AR056
E)	0RE-PR034	Q)	1RE-AR011
F)	1RE-PR011	R)	1RE-AR012
G)	1RE-PR027	S)	0RE-PR041
H)	0RE-PR001	T)	0RE-PR005
I)	0RE-PR009	U)	0RE-PR040
J)	0RE-PR016		
K)	0RE-PR017		
L)	0RE-PR018		

* 15 sec. time delay on start prevents auto-start of fan.

CAUSE: CIRCUIT NOISE / VOLTAGE SPIKE

REF: AR/PR SYSTEM DESCRIPTION
M-78 SERIES
BwOP AR/PR 11-T1

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED RADIATION MONITOR OUTPUT WILL SPIKE MOMENTARILY INCREASING ITS OUTPUT TO GREATER THAN THE HIGH ALARM SETPOINT. THE DETECTOR OUTPUT WILL RETURN TO NORMAL IN 30 SECONDS TO 1 MINUTE. THIS WILL CAUSE ITS HIGH RADIATION AUTOMATIC ACTION TO OCCUR WITHOUT AN ACTUAL HIGH RADIATION LEVEL EXISTING. THE AUTOMATIC ACTION WILL FUNCTION PROPERLY AND ITS EFFECT ON PLANT OPERATION IS DEPENDENT ON WHICH MONITOR IS SELECTED. WHEN THE OPERATOR ATTEMPTS TO RECOVER FROM THE AUTOMATIC ACTION INITIATED BY THE SELECTED MALFUNCTION, REALIGNMENT OF THE AFFECTED SYSTEM(S) IS ALLOWED.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED RADIATION MONITOR TO NORMAL.

EVENTS:
1) LER 06-01-88-011
2) LER 06-01-88-006
3) LER 20-02-88-027
4) LER 20-01-88-019
5) LER 20-01-88-011
6) LER 20-01-88-020

LICENSEE EVENT REPORT (LER)

Name (1)	Docket Number (2)	Page (3)
Braidwood, Unit 1	0 5 0 0 0 4 5 6	1 of 0 3

e (4) Train A and Train B Control Room Ventilation Switchover Due to Radiation Monitors 0PR31J through 34J Experiencing a Momentary Loss of Power from 345 KV Line Loss

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0 8	0 1	8 8	8 8	0 1 9	0 0	0 8	1 8	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	1
POWER LEVEL (10)	1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

	20.402(b)		20.405(c)	X	50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		Other (Specify in Abstract below and in Text)
	20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER	
	AREA CODE	
Paul Stanczak, Tech Staff Engineer Ext. 2486	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X									

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

At 1201 on August 1, 1988, Control Room Train A and B ventilation automatically shifted to its emergency wake-up mode of operation due to a momentary loss of voltage on the Control Room Outside Air Intake Radiation Monitor. Radiation Monitors 0PR31J, 0PR33J, and 0PR34J alarmed simultaneously in the Control Room at the Radiation Monitor RM-11 console. This Power Fail alarm, indicated at the RM-11 console, occurred at the same time that the 345 Kilovolt line 2002 was de-energized from the station power distribution ring due the line coming in contact with a tree. This resulted in a momentary dip in the 345 Kilovolt line voltage which was sensed by the radiation monitors. The radiation monitors immediately returned to normal operation and the Control Room Ventilation was realigned to its normal line-up. There was no failure of plant equipment. System load dispatch has had the power line right-of-ways cleared. No further corrective action is considered necessary. There have been no previous occurrences of control room ventilation shifting to its emergency makeup mode as a result of power line perturbations.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Braidwood, Unit 1	0 5 0 0 0 4 5 6	8 8	- 0 1 9 -	0 0	0 2 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: 1 ; Event Date: August 1, 1988 ; Event Time: 1201
MODE: 1 - Power Operating ; Rx Power: 100% ; RCS [AB] Temperature/Pressure: NOT/NOP

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of the event that contributed to the event.

At 1201 on August 1, 1988, Control Room Train A and Train B ventilation [VI] automatically shifted to its emergency make-up mode of operation due to a momentary loss of voltage on the Control Room Outside Air Intake Radiation Monitor (IL). Radiation Monitors 0PR31J, 0PR33J, and 0PR34J alarmed simultaneously in the Control Room at the Radiation Monitor RM-11 console. This power fail alarm, indicated at the RM-11 console, occurred at the same time that the 345 Kilovolt (KV) line 2002 was de-energized from the station power distribution ring.

The radiation monitors immediately returned to normal operation and the Control Room ventilation was realigned to its normal line-up.

Operator actions neither increased nor decreased the severity of the event and plant operation were unaffected.

This event is being reported pursuant to 10CFR50.73(a)(2)(iv) - Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature, including the Reactor Protection System.

C. CAUSE OF EVENT:

The intermediate cause of the event was a momentary fluctuation in line voltage that dropped below the 90 volt Power Fail setpoint of the radiation monitors. This occurred when line 2002 was suddenly de-energized causing power line perturbations felt throughout the plant. The root cause was that line 2002 had sagged down onto a willow tree, causing a line overcurrent condition that de-energized the 345 KV line.

D. SAFETY ANALYSIS:

This event had no effect on the safety of the plant or the public. There was no abnormal level of radioactivity present. Equipment operation performed as designed in causing Control Room ventilation shift to its emergency makeup mode of operation as designed. Under worst case conditions of actual radioactivity being present, the Control Room ventilation system would have responded as it did in this event.

E. CORRECTIVE ACTIONS:

Equipment operation immediately returned to normal. There was no failure of plant equipment. System Load Dispatch has had the power line right-of-ways cleared. No further corrective action is considered necessary.

F. PREVIOUS OCCURRENCES:

There have been no previous occurrences of Control Room ventilation shifting to its emergency makeup mode as a result of power line perturbations.

G. COMPONENT FAILURE DATA:

This event was not caused by plant component failure, nor did any plant components fail as a result of this event.

RMD3

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Braidwood Unit 2	0 5 0 0 0 4 5 7	1 of 0 3

Title (4) Containment Ventilation Isolation Due to Transient Loss of 345KV Switchyard Line 0103

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
1 1	1 5	8 8	8 8	— 0 2 7 —	0 0	1 2	0 2	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 3

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	X 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER AREA CODE	
Richard Rountree Engineer Tech Staff Ext. 2487	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
__ Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

Prior to the event all containment ventilation isolation valves were closed. At 2340 on November 15, 1988 the Main Control Room received a Unit 2 Train B Containment Ventilation Isolation signal from the Containment Fuel Incident Monitor (2RT-AR012). The root cause of the event was a perturbation of the 345 Kilovolt Transmission System due to line 0103 opening as a result of a thunderstorm. The resulting power perturbation was felt from line 0103 through the System Auxiliary Transformer to Mntor Control Center (MCC) 232X2 AND 232X3. The radiation monitors fed from these MCC's indicated a Power Fail on their trend display screens also. However, the 2RT-AR012 radiation monitor initiated the interlock function, which was manually reset. This event was considered spurious and no further corrective action is considered necessary as the monitor performed its safety function. There have been previous occurrences of a Containment Ventilation Isolation as a result of a momentary loss of power.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 2; Event Date: November 15, 1988; Event Time: 2340;

Mode: 3 - Hot Standby; Rx Power: 0%;

RCS [AB] Temperature/Pressure: 553 degrees F/2243 psig:

B. DESCRIPTION OF EVENT:

There were no structures, systems, or components inoperable or degraded at the beginning of the event that contributed to the event.

At 2340 hrs on November 15, 1988 the Main Control Room received a Unit 2 Train B Containment Ventilation Isolation (EF) [JE] signal. All containment ventilation isolation valves were closed prior to the event. Plant conditions remained stable throughout the duration of the event. Operator action had no impact on the severity of the event. No equipment was declared inoperable as a result of this event.

The 2RT-AR012 Containment Fuel Incident Monitor (AR) [IL] initiated the Containment Ventilation Isolation interlock function at the same time 345 Kv Switchyard (MP) [EL] line 0103 opened momentarily during a thunderstorm and severe high winds. The event manifested itself on the trend display screen of the Unit 2 Control Room Radiation Monitoring Operator Console (RM-11). Because of the extremely short duration of the perturbation, the 2RT-AR012 Containment Fuel Incident Monitor did not alarm the loss of power, but it did initiate the Containment Ventilation Isolation, and the power fail was displayed on the trend display screen.

The Load Dispatcher verified that Bus Tie 3-4 circuit breaker opened, but did not reclose (automatically), as it should have. Seconds later, the Load Dispatcher manually reclosed the breaker, restoring power to the line.

The resulting power perturbation was felt from line 0103 through the System Auxiliary Transformer to Motor Control Center (MCC) 232X2 and 232X3. The radiation monitors fed from these MCC's indicated a Power Fail on their trend display screens and the 2RT-AR012 radiation monitor also initiated the interlock function.

The appropriate NRC notification via the ENS phone system was made at 0058 on November 16, 1988 pursuant to 10CFR50.72(b)(2)(ii).

This event is being reported pursuant to 10CFR50.73(a)(2)(iv), - any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature including the Reactor Protection System.

C. CAUSE OF EVENT:

The root cause of the event was a perturbation of the 345 Kilovolt Transmission System and is considered weather related, because of the thunderstorm and high winds present. The Containment Fuel Incident Monitor sensed an undervoltage condition that momentarily placed the monitor in a power fail mode of operation. This caused the generation of the Containment Ventilation Isolation signal.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

Form Rev 2.0

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					Page (3)
		Year	///	Sequential Number	///	Revision Number	
Braidwood Unit 2	0 5 0 0 0 4 5 7	8 8	-	0 2 7	-	0 0	0 3 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

D. SAFETY ANALYSIS:

This event had no effect on the safety of the plant or the public. All systems operated as designed.

The monitor was able to perform its safety function as described in Final Safety Analysis Report Section 12.3.4.1 throughout the duration of the event. In the worst case event of a permanent loss of power to the monitor, the Containment Ventilation Isolation signal would still have occurred, as the monitor would still have been able to sense the loss of power.

E. CORRECTIVE ACTIONS:

The Containment Ventilation Isolation was reset. This event was considered spurious, and no further action is considered necessary since the monitor performed its safety function.

F. PREVIOUS OCCURRENCES:

There have been previous occurrences of a Containment Ventilation Isolation as a result of a momentary loss of power.

DVR Number	Title
20-1-87-099/87-018	Train B Containment Isolation Signal Due to an Undervoltage Condition Sensed by Containment Incident Fuel Monitor 1RT-AR012
20-1-87-204/87-030	Containment Purge Isolation Due to Radiation Monitor Loss of Power

G. COMPONENT FAILURE DATA:

This event was not the result of component failure, nor did any components fail as a result of this event.

RM03

LICENSEE EVENT REPORT (LER)

Facility Name (1) Byron, Unit 1 | Form Rev 2.0

Docket Number (2) 0 5 0 0 0 4 5 4 | Page (3) 1 of 0 3

Title (4) Fuel Handling Building Charcoal Booster Fan OA Automatic Start Due to Inadequate Standard Out Of Service Procedure

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
1 1	2 9	8 8	8 8	0 1 1	0 0	1 2	2 7	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	X 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name: Tom Higgins, Operating Engineer, Ext. 2215

TELEPHONE NUMBER: AREA CODE 8 1 5 | 2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

__ Yes (If yes, complete EXPECTED SUBMISSION DATE) | X NO

Expected Submission Date (15): Month | Day | Year

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

At 0930 on November 29, 1988 the OB Fuel Handling Building (FHB) Charcoal Booster Fan was manually started to satisfy a Technical Specification for new fuel movements and for removing a FHB area radiation monitor from service for a planned modification. The radiation monitor was placed in an interlock condition, which would ordinarily cause an auto start of the OA FHB Charcoal Booster Fan, but the OA and OB fans are interlocked to permit operation of only one fan at any particular time. At about 1430 the Fuel Handlers reported the completion of fuel movements. At 1543 the OB fan was manually stopped. Due to the radiation monitor interlock, the OA fan auto started. The OA fan was stopped and the OB fan restarted. The modification to the radiation monitor was completed on December 2, 1988 and the OB fan was stopped.

The Out Of Service procedure for the radiation monitor failed to consider the interlock between the monitor and the OA FHB Charcoal Booster Fan. The procedure was in error, because it did not specify actions to prevent this event.

The standard Out Of Service procedures for radiation monitors with interlock functions will be modified to explain the interlock function as well as provide actions to prevent automatic equipment starts.

(0194R/0012R)

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 11-29-88 / 1543

Unit 1 MODE 1 - Power Operation Rx Power 100% RCS [AB] Temperature/Pressure Normal Operating

Unit 2 MODE 2 - Power Operation Rx Power 48% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

At 0930 on November 29, 1988 the OB Fuel Handling Building Charcoal Booster Fan [VG] was manually started by a Nuclear Station Operator (NSO) (licensed reactor operator) to satisfy Technical Specification 3.9.12 for new fuel movements in the Fuel Handling Building and for removing a Fuel Handling Building Fuel Handling Incident Area Radiation Monitor (ORT-AR055) [IL] from service for a planned internal wiring modification. At 0937 ORT-AR055 was removed from service, and placed in an interlock condition which would ordinarily cause an automatic start of the OA Fuel Handling Building Charcoal Booster Fan, but the OA and OB fans are interlocked so only one fan can operate at any particular time. Since the OB fan had been started and continued to operate, the OA fan remained off and in a standby condition.

At approximately 1430 Fuel Handling Operators (non-licensed operators) informed the control room operators that fuel movements were complete for the day. The NSO believed that operation of the OB fan was no longer required, since fuel movements had been ceased. At 1543 with Unit 1 in power operation (Mode 1) at 100 percent reactor power and Unit 2 in Mode 1 at 48 percent reactor power, the NSO stopped the OB Fuel Handling Building Charcoal Booster Fan. When the OB fan was stopped, the OA fan automatically started due to the interlock condition of ORT-AR055. The OA Booster Fan automatic start was an Engineered Safety Features (ESF) actuation, which is reportable by Licensee Event Report (LER) pursuant to 10CFR 50.73(a)(2)(iv). The OA fan was manually stopped by an NSO and the OB fan was manually started at 1543. At 1929 an event notification was telephoned to the Nuclear Regulatory Commission in accordance with 10CFR 50.72(b)(2)(ii). The modification to ORT-AR055 continued and was completed on December 2, 1988. ORT-AR055 was returned to service and the OB Fuel Handling Building Charcoal Booster Fan was stopped at 1655 on December 2, 1988. The stable operation of both Byron Units remained unaffected by this event.

C. CAUSE OF EVENT:

The Out Of Service procedure for the ORT-AR055 area radiation monitor did not consider that the monitor is interlocked with the OA Fuel Handling Building Charcoal Booster Fan. The Out Of Service also did not contain actions required to prevent this event. Therefore, the Out Of Service procedure was in error.

D. SAFETY ANALYSIS:

There was no effect on plant or public safety. The automatic start of the OA Fuel Handling Building Charcoal Booster Fan and realignment of dampers is an Engineered Safety Features (ESF) actuation which establishes a safer plant condition because the ESF lineup filters radioactive contaminants from the air in the Fuel Handling Building. The filtering capability was not required, since no radioactive contaminants were present during this event. The safety consequences would have been the same had this event occurred under a more severe set of initial conditions.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year		Sequential Number		Revision Number	Page (3)
Byron, Unit 1	0 5 0 0 0 4 5 4	8 8	-	0 1 1	-	0 0	0 3 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

E. CORRECTIVE ACTIONS:

The standard Out Of Service procedure for the ORT-AR055 area radiation monitor and other radiation monitors with interlock functions will be modified to add a note explaining the interlock function of the monitor as well as provide actions to prevent automatic equipment starts. Corrective action is tracked to completion by Action Item Record 454-225-88-0291.

F. PREVIOUS OCCURRENCES:

There have not been any automatic starts of the Fuel Handling Building Charcoal Booster Fans in the past from this root cause.

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Not Applicable			

RM03

LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
Byron, Unit 1	0 5 0 0 0 4 5 4	1 of 0 3

Title (4) FUEL HANDLING BUILDING BOOSTER FAN ACTUATION DUE TO VOLTAGE TRANSIENT CAUSED BY FALLEN STATIC LINE

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
0 8	2 6	8 8	8 8	— 0 0 6	— 0 0	0 9	0 7	8 8	Byron, Unit 2	0 5 0 0 0 4 5 5
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 8

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	X 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name: R. Flahive, Technical Staff Supervisor Extension 2243

TELEPHONE NUMBER: AREA CODE 8 1 5 2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
C	F C	I N S	X 9 9 9	N					

SUPPLEMENTAL REPORT EXPECTED (14)

__ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15) Month | Day | Year

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

On August 26, 1988, at 0512 with Unit 1 at 98% and Unit 2 at 33% reactor power, the Fuel Handling Building Fuel Handling Incident Area Radiation Monitor (ORT-AR056) sensed an undervoltage condition and transferred to the interlock mode. The OB Fuel Handling Building Charcoal Booster Fan automatically started and dampers aligned to filter the Fuel Handling Building atmosphere, although no actual airborne contamination existed. Following the voltage transient the ORT-AR056 monitor returned to its normal operating condition. The booster fan was stopped at 0540 by the licensed control room operators. The event had no effect on the stable power operation of either Unit.

The voltage transient, that caused the ORT-AR056 to interlock, occurred when an electrical distribution system transmission tower static line fell on one of the phases of the transmission line. The grounding of the phase automatically tripped distribution system breakers and resulted in the voltage transient. The electrical insulators for the static line had been severely damaged by lightning and failed mechanically.

The static line was repaired by Commonwealth Edison's Rock River Division Overhead Department. Previously installed plant modifications have effectively decreased radiation monitor sensitivity to distribution system transients. The voltage disturbance caused by the lightning induced static line failure is an acknowledged risk of transmission line operation and no further corrective actions are warranted.

Previous occurrences of radiation monitor power failure induced Engineered Safety Features actuations are documented in the following Unit 1 Licensee Event Reports: 85-036, 86-009, 86-026, 87-021.

(0100R/0012R)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					Page (3)
		Year	///	Sequential Number	///	Revision Number	
Byron, Unit 1	0 5 0 0 0 4 5 4	8 8	-	0 0 6	-	0 0	0 2 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 8/26/88 / 0512

Unit 1 MODE 1 - Power Operation Rx Power 98% RCS [AB] Temperature/Pressure Normal Operating

Unit 2 MODE 1 - Power Operation Rx Power 33% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On August 26, 1988, at 0512 with Unit 1 in power operation (Mode 1) at 98% reactor power, and Unit 2 in Mode 1 at 33% reactor power, the Fuel Handling Building Fuel Handling Incident Area Radiation Monitor (ORT-AR056) [IL] sensed an undervoltage condition and transferred to the interlock mode. The interlock signal automatically started the OB Fuel Handling Building Charcoal Booster Fan (VA)[VG] and transferred the associated dampers to their Engineered Safety Feature (ESF) positions. The monitor returned to its normal operating condition immediately after the voltage transient passed. At 0540 a licensed reactor operator stopped the booster fan and returned the system to a normal configuration. No plant systems or components were previously inoperable that contributed to this event. Both Units were maintained in a stable condition during this event. All operator actions taken were correct. This event is reportable per 10CFR50.73 (a)(2)(iv) due to the automatic ESF System actuation.

C. CAUSE OF EVENT:

The electrical insulators that anchored a static line to an electrical distribution transmission tower mechanically failed and allowed the static line (energized at 2300 Volts) tr fall onto one phase of the 345,000 Volt transmission line. Transmission line 0622 bus tie breakers 11 and 12-13 opened due to the line fault condition. The transmission line trip caused a voltage transient on the Station's electrical system. The bus voltage sensed by the ORT-AR056 momentarily dropped below the undervoltage setpoint of 90 ± 3 Volts which caused the monitor to transfer to the interlock mode of operation. The mechanical failure of the insulators was the result of severe, direct lightning damage.

D. SAFETY ANALYSIS:

There was no effect on plant or public safety. The automatic start of the OB Fuel Handling Building Charcoal Booster Fan and shifting of associated dampers to their ESF positions established a safer plant condition than the normal system lineup by filtering radioactive contaminants from the Fuel Handling Building atmosphere. This filtering capability was not required, since no airborne activity existed in the Fuel Handling Building during this event. The redundant area radiation monitor (ORT-AR055) was operable during this event and showed no increase in activity level. The safety consequences would have been the same had this event occurred under a more severe set of initial conditions.

E. CORRECTIVE ACTIONS:

New insulators were installed on the transmission tower and the static line was restored by Commonwealth Edison's Rock River Division Overhead Department. A plant modification was previously installed on ORT-AR056 to lower the undervoltage trip setpoint from 100 ± 3 to 90 ± 3 VAC in order to reduce the sensitivity of the monitor to distribution system voltage transients. Operating experience indicates that the setpoint modification has effectively reduced the monitor's sensitivity to voltage transients caused by large pump starts and most grid disturbances. The voltage disturbance caused by the lightning induced static line failure is an acknowledged risk of transmission line operation and no further corrective actions are warranted.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

F. PREVIOUS OCCURRENCES:

There have been several previous occurrences of radiation monitor power failures causing ESF actuations but each has been caused by a different initiating event.

LER NUMBER	TITLE
85-036-00 (Unit 1)	ESF Actuation Due To Radiation Monitor Power Fail
86-009-00 (Unit 1)	Containment Ventilation Actuation Due To 345KV Distribution System Voltage Transient
86-026-00 (Unit 1)	Control Room Ventilation Actuation Due To Lightning Induced Distribution System Voltage Transient
87-021-00 (Unit 1)	Control Room Ventilation Actuation Due To Distribution System Voltage Transient When Offsite Line Tripped

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Not Available	Electrical Insulator	Not Available	

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Braidwood 1	0 5 0 0 0 4 5 6	1 of 0 3

Title (4) Control Room Ventilation Switchover Due to Spurious Noise

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
0 9	1 6	8 8	8 8	--- 0 2 0	--- 0 0	0 9	2 7	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 3

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

___ 20.402(b)	___ 20.405(c)	X 50.73(a)(2)(iv)	___ 73.71(b)
___ 20.405(a)(1)(i)	___ 50.36(c)(1)	___ 50.73(a)(2)(v)	___ 73.71(c)
___ 20.405(a)(1)(ii)	___ 50.36(c)(2)	___ 50.73(a)(2)(vii)	___ Other (Specify in Abstract below and in Text)
___ 20.405(a)(1)(iii)	___ 50.73(a)(2)(i)	___ 50.73(a)(2)(viii)(A)	
___ 20.405(a)(1)(iv)	___ 50.73(a)(2)(ii)	___ 50.73(a)(2)(viii)(B)	
___ 20.405(a)(1)(v)	___ 50.73(a)(2)(iii)	___ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	Telephone Number Area Code	
Paul Stanczak, Tech Staff Engineer Ext. 2486	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	I L	P T * *	G 0 6 3	NO					

SUPPLEMENTAL REPORT EXPECTED (14)

___ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15) Month | Day | Year

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

On September 16, 1988 at 1752 the Control Room Train B Ventilation automatically switched to its make up mode of operation due to a high radiation signal on the Control Room Outside Air Intake gas Channel. A high radiation alarm also occurred in the Control Room at the Radiation Monitor (RM) - 11 console. The root cause of this event is not known. A noisy pressure transducer, located on the skid itself, was suspected of inducing noise into the monitor. Equipment operation is presently normal and no further corrective action is planned. Troubleshooting by the Instrument Maintenance Department did not reveal any problems with the equipment. The System Technical Staff Engineer had noticed the pressure transducer sticking, so he had it replaced. There have been two previous occurrences of Control Room Ventilation shift to Emergency Mode due to spurious noise.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION								Form Rev 2.0		
FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)						Page (3)		
		Year	///	Sequential Number	///	Revision Number				
Braidwood 1	0 5 0 0 0 4 5	8 8	-	0 2 0	-	0 0		0 2	OF	0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 1; Event Date: September 16, 1988; Event Time: 1752;

Mode: 3 - Hot Standby; Rx Power: 0%;

RCS [AB] Temperature/Pressure: NOT/NOP

Unit: Braidwood 2; Event Date: September 16, 1988; Event Time: 1752;

Mode: 2 - Startup; Rx Power: 2%;

RCS [AB] Temperature/Pressure: NOT/NOP

B. DESCRIPTION OF EVENT:

There were no structures, systems, or components inoperable or degraded at the beginning of the event that contributed to the event.

On September 16, 1988 at 1752 the Control Room Train B Ventilation (VC) [VI] automatically switched to its make up mode of operation due to a high radiation signal on the Control Room Outside Air Intake gas Channel ORE-PR033B (PR) [IL]. A high radiation alarm also occurred in the Control Room at the Radiation Monitor (RM) - 11 console.

There was no increase in activity levels on any other channels and the event was considered to be spurious. An investigation into the event revealed no work activity in the area and the VC system was returned to normal operation.

Operator actions neither increased nor decreased the severity of the event and plant conditions were always stable.

The appropriate NRC notification via the ENS phone system was made at 1930 on September 16, 1988, pursuant to 10CFR50.72(b)(2)(ii).

This event is being reported pursuant to 10CFR50.73(a)(2)(iv) - Any event or condition that resulted in manual or automatic actuation of any engineered safety feature, including the reactor protection system.

C. CAUSE OF EVENT:

The root cause of the event is not known. A noisy pressure transducer, located on the skid itself, was suspected of inducing noise into the monitor. Equipment operation is presently normal and no further corrective action is planned.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

D. SAFETY ANALYSIS:

There was no affect on the the plant or the public safety. There was no abnormal level of radioactivity present. ORE-PR033B operated as designed and generated an Engineered Safety Features actuation on a high radiation signal occurrence. ORT-PR034 was available for redundant indication of the activity level.

E. CORRECTIVE ACTIONS:

Troubleshooting of ORE-PR033B by the Instrument Maintenance Department did not reveal any problems with the equipment. The System Technical Staff Engineer had noticed the pressure transducer sticking, so he had it replaced.

F. PREVIOUS OCCURRENCES:

DVR/LER Number	Title
DVR 20-1-87-335/ LER 87-051	Control Room Ventilation Switchover Due to Spurious Noise on Channel ORE-PR003B
DVR 20-1-88-088/ LER 88-011	Control Room Ventilation Shift to Emergency Makeup Mode Due to Spurious Radiation Monitor Noise Spike

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	MFG Part Number
General Atomics (Sorento Electronics)	Pressure Transducer	P61K188	N/A

BP-ISUOGE-OPER
ACTION ITEM

DVR
20-1-88-088

ITEM NO: 456-200-88-08800

OTHER UNIT NO

ITEM DATE: 04/15/88
MODE: 5

SCHEDULAR CAT: TEST CONDITION
COMMITMENT TO:

CURRENT LOC (DEP): RA
PERSON: BERRY
DATE SENT

SUBJECT: DVR 20-1-88-088, CONTROL ROOM
VENTILATION SHIFT TO EMERGENCY
MAKE UP MODE DUE TO SPURIOUS
RADIATION MONITOR NOISE SPIKE

TYPE: DEVIATION
SEVERITY LEVEL:
LER NO: 88-011
CRITERION

ORG CAUSING ITEM: BW
ORIG ORG/PERSON: OP /WALRATH
RESP DEPT/SUPV: TSEL/STANCZAK
COG PERSON: A /DE
COG PERSON: /
COG PERSON: /

RESPONSE DUE TO BY
DATE SET BY

INSPECTOR HRS: SYSTEM VI VI
CORRECTIVE ACT:
R/F OUTAGE:
PRIORITY:
TR: XIEEL4IM
BY/BW PROCEDURE:

TRANSMIT: BW: BY: D: L: Q: Z: NOD: DNS: ESS: PTC: NFS:

POWER: 0
ORIGINAL DUE DATE: 05/15/88
INTERIM REPORT: *
INTERIM REPORT: 10CFR50.73(A)(2)(IV)
CLOSING REPORT:
SIGNATURE: D.E. O'BRIEN

STATUS: COMPLETE
RDY FOR CLOSURE: 04/26/88
ORIG EXIT DATE: 05/11/88
ORIG CLOSED: 05/13/88
DATE COMPLETED: 06/17/88

REFER: DVR 20-1-88-088

DESCRIPTION:

ON 04/15/88 AT 0132, THE OR CONTAROL ROOM VENTILATION SYSTEM RECEIVED A HIGH RAD SIGNAL FROM OPR033J GAS CHANNEL AND SWAPPED TO THE MAKEUP MODE OF OPERATION. RADCHEM WAS NOTIFIED TO PULL THE FILTERS AND CARTRIDGE FOR ANALYSIS. NO ABNORMAL AMOUNTS OF RADIATION WERE DETECTED. THE MONITOR APPEARED TO HAVE HAD A SPIKE. THE MONITOR OPERATED PROPERLY BOTH BEFORE AND AFTER THE EVENT. AN ENS PHONE CALL WAS COMPLETED PER BWAP 1350 REQUIREMENTS.

*

THE RCT THAT CHANGED OUT THE FILTER REPORTED THAT OPR5313 WAS FOUND OPEN WHICH IS AN INCORRECT POSITION. THIS SHOULD NOT HAVE AFFECTED THE ABILITY OF THE MONITOR TO PERFORM IT'S FUNCTION.

*

THE ONLY OTHER EVENT THAT MAY HAVE HAD AN EFFECT ON THE RAD MONITOR WAS A 0B VC CHILLER TRIP AT 2326 ON 04/14/88. THE CHILLER WAS OFF FOR ABOUT TEN MINUTES WHICH WOULD HAVE CAUSED A SLIGHT TEMPERATURE TRANSIENT ON THE SYSTEM.

*

ITEM NO 456-200-88-08800 (CONT)

DESCRIPTION (CONT):

10CFR50.72 NRC RED PHONE NOTIFICATION MADE, 0400, 04/15/88
*
*
OPERATING ENGINEER'S COMMENTS:
*
NONE. BARRY MCCUE 04/15/88
*
NOTIFICATIONS: RESIDENT INSPECTOR, NRC REGION III, 04/15/88, 1600
* T. J. MAIMAN/D. P. GALLE, VP/NSD, 04/15/88, 1600
*
30 DAY REPORTABLE/10CFR50.73(A)(2)(IV)
*
LER NUMBER: 88-011

ACTION SUMMARY:

RM04 PROCESS RADIATION MONITOR ACTUATION

TYPE: GENERIC, RV 1E-9 TO 1E+1 μCi/cc

NOTE: LOGARITHMIC SCALE MODELED LINEARLY

A)	0RE-PR001	K)	1RE-PR002
B)	0RE-PR006	L)	1RE-PR003
C)	0RE-PR007	M)	1RE-PR006
D)	0RE-PR008	N)	1RE-PR007
E)	0RE-PR009	O)	1RE-PR008
F)	0RE-PR010	P)	1RE-PR009
G)	0RE-PR016	Q)	0RE-PR005
H)	0RE-PR017	R)	0RE-PR040
I)	0RE-PR018	S)	0RE-PR041
J)	0RE-PR019		

CAUSE: DETECTOR FAILURE

REF: AR/PR SYSTEM DESCRIPTION
M-78 SERIES

PLT STA:REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES SELECTED PROCESS RADIATION MONITOR TO FAIL. THE VALUE DISPLAYED IS DEPENDENT UPON THE SEVERITY SELECTED. SELECTING A SEVERITY HIGHER THAN THE HIGH ALARM SETPOINT WILL CAUSE AN RM-11 ALARM. THE FOLLOWING MONITORS HAVE INTERLOCKS ASSOCIATED WITH THEM:

<u>0RE-PR001 - LIQUID RADWASTE EFFLUENT MONITOR:</u>
AUTO CLOSES THE RELEASE TANK DISCHARGE VALVE.

<u>0RE-PR005 - FIRE AND OIL SUMP:</u>
TRIPS TURB BLDG FIRE AND OIL SUMP PUMPS.

<u>0/1RE-PR009 - COMPONENT COOLING WATER HX 0/1 OUTLET MONITOR:</u>
AUTO CLOSES THE CC SURGE TANK VENT VALVE (1CC017).

<u>0RE-PR016 - BLOWDOWN AFTER FILTER A OUTLET MONITORS:</u>
AUTO DIRECTS DISCHARGE FLOW FROM THE BLOWDOWN MIXED-BED DEMIN TO THE BLOWDOWN MONITOR TANKS.

<u>1RE-PR008 - S/G BLOWDOWN MONITOR:</u>
AUTO CLOSES S/G SAMPLE VALVES (1PS179's).

MALFUNCTION REMOVAL RESTORES THE SELECTED PROCESS RADIATION MONITOR TO NORMAL.

EVENTS: NONE

RM05 RADIATION MONITOR INTERLOCK ACTUATION FAILURE

TYPE: GENERIC, RB

A)	0RE-PR001	P)	NOT USED
B)	0RE-PR005	Q)	NOT USED
C)	0RE-PR009	R)	NOT USED
D)	0RE-PR016	S)	0RE-PR041
E)	0RE-PR031	T)	0RE-PR040
F)	0RE-PR032	U)	NOT USED
G)	0RE-PR033		
H)	0RE-PR034		
I)	1RE-PR008		
J)	1RE-PR009		
K)	1RE-PR011		
L)	0RE-AR055		
M)	0RE-AR056		
N)	1RE-AR011		
O)	1RE-AR012		

CAUSE: FAULTY INTERLOCK RELAY (NOTE: MONITOR DETECTS AND ALARMS ON HIGH RADIATION BUT FAILS TO INITIATE AUTOMATIC ACTIONS)

REF: AR/PR SYSTEM DESCRIPTION
M-78 SERIES
BwOP AR/PR 11-T1

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THERE IS NO IMMEDIATELY NOTICEABLE EFFECT. WHEN THE SELECTED RADIATION DETECTOR INCREASES TO THE LEVEL WHICH WOULD NORMALLY CAUSE ITS HIGH RADIATION AUTOMATIC ACTION TO OCCUR, THE RM-11 CHANNEL ALERT AND HIGH ALARMS WILL ACTUATE, HOWEVER ANY AUTOMATIC ACTION(S) DO NOT OCCUR. THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY MANUALLY DUPLICATING THE AUTOMATIC ACTION.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED RADIATION MONITOR TO NORMAL.

EVENTS: NONE

RM06 GASEOUS AIR MONITOR FAILURE

TYPE: GENERIC, RV 1E-10 TO 1 μCi/cc

NOTE: LOGARITHMIC SCALE MODELED LINEARLY

A)	0RE-PR003	P)	0RE-PR035
B)	0RE-PR011	Q)	0RE-PR036
C)	0RE-PR012	R)	0RE-PR037
D)	0RE-PR013	S)	0RE-PR038
E)	0RE-PR014	T)	1RE-PR001
F)	0RE-PR015	U)	1RE-PR011
G)	0RE-PR021	V)	1RE-PR013
H)	0RE-PR022	W)	1RE-PR014
I)	0RE-PR024	X)	1RE-PR015
J)	0RE-PR025	Y)	1RE-PR016
K)	0RE-PR026	Z)	1RE-PR017
L)	0RE-PR031	AA)	1RE-PR018
M)	0RE-PR032	AB)	1RE-PR021
N)	0RE-PR033	AC)	1RE-PR027
O)	0RE-PR034	AD)	1RE-PR028

CAUSE: DETECTOR FAILURE

REF: AR/PR SYSTEM DESCRIPTION
M-78 SERIES

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED GASEOUS MONITOR TO FAIL. THE VALUE DISPLAYED IS DEPENDENT UPON THE SELECTED SEVERITY. IF THE SELECTED SEVERITY IS LOWER THAN THE ASSIGNED RM-11 SETPOINT, THEN THE RM-11 WILL READ LOW. SELECTING A SEVERITY HIGHER THAN THE HIGH ALARM SETPOINT WILL CAUSE AN RM-11 ALARM. THE FOLLOWING MONITORS HAVE INTERLOCKS ASSOCIATED WITH THEM:

<u>0RE-PR026 - RADWASTE AREA VENT EFFLUENT MONITOR:</u>
AUTO TRIPS THE RADWASTE BLDG VENT SYSTEM SUPPLY AND EXHAUST FANS.

0RE-PR031 & 0RE-PR032 - CR OUTSIDE AIR INTAKE "A" MONITORS:
AUTO CLOSES THE OUTSIDE AIR INTAKE "A" DAMPERS, TURBINE BLDG INTAKE AIR "A" DAMPERS OPEN, MAKEUP FAN "A" STARTS.

0RE-PR033 & 0RE-PR034 - CR OUTSIDE AIR INTAKE "B" MONITORS:
AUTO CLOSES THE OUTSIDE AIR INTAKE "B" DAMPERS, TURBINE BLDG INTAKE AIR "B" DAMPERS OPEN, MAKEUP FAN "B" STARTS.

1RE- PR011 - CONTAINMENT ATMOSPHERE MONITOR:
AUTO CLOSES 1PR035/037/038/041/043/044 TO ISOLATE THE DETECTOR.

1RE-PR027 - SJAE/GLAND STEAM EXHAUST MONITOR:
AUTO ACTIVATES THE OFF GAS VENT FILTER SYSTEM.

MALFUNCTION REMOVAL RESTORES THE SELECTED GASEOUS MONITOR TO NORMAL.

EVENT: NONE

BRAIDWOOD SIMULATOR

MALFUNCTION AND EFFECTS

RP01	AUTOMATIC REACTOR TRIP FAILURE
RP02	REACTOR TRIP BREAKER FAILS TO OPEN
RP03	REACTOR TRIP BYPASS BREAKER FAILS TO OPEN
RP04	FAILURE OF PHASE A CNMT ISOL TO ACTUATE
RP05	FAILURE OF PHASE B CNMT ISOL TO ACTUATE
RP06	TURBINE TRIP INTERLOCK C-8 FAILS
RP07	UNDER-FREQUENCY ON RCP BUS
RP08	UNDER-VOLTAGE ON RCP BUS
RP09	INADVERTENT FW ISOLATION
RP10	INADVERTENT PHASE A CONTAINMENT ISOLATION
RP11	INADVERTENT PHASE B CONTAINMENT ISOLATION
RP12	INADVERTENT CONTROL ROOM VENT ISOLATION
RP13	REACTOR TRIP PERMISSIVE P-4 FAILS TO ACTUATE
RP14	FAILURE OF SAFETY INJECTION TO ACTUATE
RP15	SAFEGUARD SEQUENCING FAILURE
RP16	PERMISSIVE P-6 FAILS TO ACTUATE
RP17	PERMISSIVE P-7 FAILS TO ACTUATE
RP18	PERMISSIVE P-8 FAILS TO ACTUATE
RP19	PERMISSIVE P-10 FAILS TO ACTUATE
RP20	PERMISSIVE P-11 FAILS TO ACTUATE
RP21	LO-LO TAVG PERMISSIVE P-12 FAILS TO ACTUATE
RP22	PERMISSIVE P-13 FAILS TO ACTUATE
RP23	PERMISSIVE P-14 FAILS TO ACTUATE
RP24	INADVERTENT SAFETY INJECTION
RP25	SSPS BLOWN GROUND RETURN FUSE

RP01 AUTOMATIC REACTOR TRIP FAILURE

TYPE: DISCRETE, RB

CAUSE: SOLID STATE PROTECTION SYSTEM FAILURE

REF: REACTOR PROTECTION SYSTEM DESCRIPTION

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES NO IMMEDIATE NOTICEABLE EFFECTS. WHEN THE REACTOR RECEIVES AN AUTOMATIC TRIP SIGNAL, THE REACTOR DOES NOT TRIP AS REQUIRED. THE INITIATING REACTOR TRIP SIGNAL WILL BE INDICATED BY THE FIRST-OUT ANNUNCIATOR. THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY MANUALLY TRIPPING THE REACTOR.

MALFUNCTION REMOVAL RESTORES THE SOLID STATE PROTECTION SYSTEM TO NORMAL.

EVENTS: 1) SER 10-85

OPEX 85-52

(SER 10-85)

RP01

NUCLEAR NETWORK INFORMATION

IS 491 GILLISPIE (INPO) 01-MAR-85 13:53 PT
Subject: INPO SIGNIFICANT EVENT REPORT (SER): 10-85

SUBJECT: REACTOR TRIP BREAKER FAILURE CAUSED BY IMPROPER TEST EQUIPMENT SETUP

UNIT (TYPE): SEQUOYAH 2 (PWR)
DOC NO/LER NO: 50-328/85002
EVENT DATE: 1-12-85
NSSS/AE: WESTINGHOUSE/TVA

SUMMARY:

DURING A PLANT TRIP, ONE REACTOR TRIP BREAKER FAILED TO OPEN AUTOMATICALLY BECAUSE OF A FAILED TRANSISTOR IN THE SOLID STATE PROTECTION SYSTEM. THE TRANSISTOR FAILURE WAS CAUSED BY AN ERROR IN SETUP OF TEST EQUIPMENT DURING SURVEILLANCE TESTING.

DESCRIPTION:

SEQUOYAH 2 WAS OPERATING AT 96-PERCENT POWER WHEN A FEEDWATER PUMP TRIP AND SUBSEQUENT TURBINE RUNBACK CAUSED A LOW-LOW STEAM GENERATOR LEVEL CONDITION. THE REACTOR TRIPPED AS DESIGNED, BUT THE CONTROL ROOM OPERATOR OBSERVED THAT THE "A" REACTOR TRIP BREAKER FAILED T OPEN AUTOMATICALLY. HE IMMEDIATELY OPENED THE BREAKER MANUALLY FR THE CONTROL ROOM IN ACCORDANCE WITH PROCEDURES.

AN INVESTIGATION OF THE SOLID STATE PROTECTION SYSTEM REVEALED THAT THE "A" REACTOR TRIP BREAKER UNDERVOLTAGE (UV) COIL REMAINED ENERGIZED DURING THE EVENT BECAUSE OF A FAILED OUTPUT TRANSISTOR WHICH CONTINUED TO MAINTAIN VOLTAGE TO THE COIL. THE CIRCUIT BOARD WAS REPLACED, AND BOTH THE CIRCUITRY AND THE REACTOR TRIP BREAKER WERE SUBSEQUENTLY TESTED SEVERAL TIMES WITH SATISFACTORY RESULTS.

A SIMILAR CIRCUIT BOARD FAILURE HAD OCCURRED PREVIOUSLY AT UNIT 2, BUT THE CAUSES OF THE FAILURE HAD NOT BEEN IDENTIFIED. AFTER THE RECENT EVENT, IT WAS DETERMINED THAT ERRORS DURING TESTING ACTIVITIES COULD SUBJECT THE TRANSISTORS TO HIGH CURRENTS AND POTENTIAL DAMAGE.

IN 1983, THE SURVEILLANCE TEST METHOD WAS REVISED TO SEPARATELY ACTUATE THE UV AND SHUNT TRIP CIRCUITS. THE TEST REQUIRES INSTALLATION OF A JUMPER IN THE MANUAL TRIP CIRCUIT AND USE OF A VOLT-AMMETER TO VERIFY THAT THE UV COIL REMAINS ENERGIZED WHEN THE SHUNT TRIP DEVICE IS ACTUATED.

FOLLOWING A REACTOR TRIP IN LATE DECEMBER 1984, A VOLT-AMMETER WAS CONNECTED ACROSS THE "A" UV COIL IN ACCORDANCE WITH THE TEST PROCEDURE, BUT THE METER WAS INADVERTENTLY SET TO MEASURE CURRENT RATHER THAN VOLTAGE. THE AMMETER FUNCTION CREATED A LOW RESISTANCE PATH (SHORT CIRCUIT) AROUND THE UNDERVOLTAGE COIL AND ALLOWED ABNORMALLY HIGH CURRENT TO PASS THROUGH THE TRANSISTOR CAUSING IT TO FAIL.

SINCE THE UV TRIP PORTION OF THE TEST HAD BEEN SUCCESSFULLY COMPLETED PREVIOUSLY, THE TRANSISTOR FAILURE WAS NOT DETECTED, AND THE PLANT WAS RETURNED TO SERVICE UNTIL THE JANUARY 12 EVENT. THE FAILURE WOULD HAVE BEEN DETECTED BY THE NEXT SURVEILLANCE TEST SCHEDULED FOR JANUARY 18.

COMMENTS:

1. THIS EVENT IS SIGNIFICANT BECAUSE UNDETECTED FAILURE OF THE ACTUATING SIGNAL TO ONE OF THE TWO REACTOR TRIP BREAKERS SUBSTANTIALLY REDUCED RELIABILITY OF THE REACTOR PROTECTION SYSTEM. IT IS CONCEIVABLE THAT ERRORS COULD DISABLE BOTH REACTOR TRIP BREAKERS. IT SHOULD BE NOTED THAT THE SHUNT TRIP FUNCTION IS UNAFFECTED BY UNDERVOLTAGE CARD FAILURES, AND THE CONTROL ROOM OPERATOR COULD OPEN THE BREAKERS IF REQUIRED (THE CIRCUITRY FOR AUTOMATIC SHUNT TRIP ACTUATION HAS NOT YET BEEN INSTALLED AT SEQUOYAH).

2. THE NORMAL TESTS OF THE SOLID STATE PROTECTION SYSTEM WOULD DETECT TRANSISTOR FAILURES OF THIS TYPE AND WOULD LIMIT THE PERIOD OF POTENTIAL BREAKER INOPERABILITY TO A MAXIMUM OF THIRTY DAYS.

3. ANOTHER PLANT HAS EXPERIENCED SIMILIAR TRANSISTOR FAILURES ON THREE OCCASIONS (ALL WERE DETECTED BY TESTING BEFORE THE UNIT W STARTED UP). EACH OF THESE FAILURES APPEARS TO HAVE RESULTED FROM MAINTENANCE OR MODIFICATION ACTIVITIES ON REACTOR TRIP BREAKER CIRCUITRY RATHER THAN TESTING ACTIVITIES.

4. IT WOULD BE ADVISABLE TO TEST THE SOLID STATE PROTECTION SYSTEM USING THE BUILT-IN, SEMI-AUTOMATIC TESTER FOLLOWING ANY MAINTENANCE OR TEST THAT COULD AFFECT THE UNDERVOLTAGE CARD.

5. FOR ELECTRONIC SYSTEMS SUBJECT TO SUCH DAMAGE, TEST PROCEDURES SHOULD CONTAIN CAUTIONS REGARDING CORRECT SETUPS AND USE OF TEST EQUIPMENT. IT WOULD ALSO BE ADVISABLE TO MODIFY CIRCUITRY SO THAT ROUTINE TESTS CAN BE PERFORMED WITHOUT LIFTING LEADS, JUMPERING, OR INSTALLING TEMPORARY METERS.

INPO'S EVALUATION OF THIS EVENT IS COMPLETE.

LIMITED DISTRIBUTION

Information Contact: DAVID HEMBREE, INPO, 404/953-7657

RP02 REACTOR TRIP BREAKER FAILS TO OPEN

TYPE: GENERIC, RB

A) RTA
B) RTB

CAUSE: RX TRIP BREAKERS ARC WELDED CLOSED

REF: 20E-1-4030 RD06
20E-1-4030 RD07

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED REACTOR TRIP BREAKER TO FAIL TO OPEN ON EITHER A MANUAL OR AUTOMATIC TRIP SIGNAL. THE OPERATOR MAY MITIGATE THE EFFECTS BY LOCALLY TRIPPING THE RX TRIP BREAKERS OR MANUALLY TRIPPING THE M-G SETS. NORMAL AFTER TRIP EFFECTS ARE OBSERVED.

FAILURE OF RTA TO OPEN WILL PREVENT ARMING OF THE STM DUMPS UNLESS THEY ARE ARMED FROM C-7 (TURB LOAD REJECTION OF >10%). FAILURE OF RTB TO OPEN WILL PREVENT THE STEAM DUMPS FROM SWITCHING TO THE PLANT TRIP CONTROLLER. THIS WILL STILL ALLOW THE DUMPS TO OPERATE ON THE LOAD REJECT CONTROLLER WITH A 3°F DEADBAND.

MALFUNCTION REMOVAL RESTORES THE SELECTED BREAKER TO NORMAL.

EVENTS: NONE

RP03 REACTOR TRIP BYPASS BREAKER FAILS TO OPEN

TYPE: GENERIC, RB

A) BYA
B) BYB

CAUSE: BYPASS BREAKERS ARC WELDED CLOSED

REF: 20E-1-4030 RD06
20E-1-4030 RD07

PLT STA: BYPASS BREAKER CLOSED

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED REACTOR TRIP BYPASS BREAKER TO FAIL TO TRIP OPEN ON EITHER AN AUTOMATIC OR MANUAL TRIP SIGNAL. THE OPERATOR MAY MITIGATE THE EFFECTS BY MANUALLY TRIPPING THE M-G SETS TO DROP THE RODS. NORMAL AFTER TRIP EFFECTS ARE OBSERVED.

MALFUNCTION REMOVAL RESTORES THE SELECTED BREAKER TO NORMAL.

EVENTS: NONE

RP04 FAILURE OF PHASE A CNMT ISOL TO ACTUATE

TYPE: GENERIC, RB

A) TRAIN A LOGIC
B) TRAIN B LOGIC

CAUSE: FAULTY LOGIC CARD(S)

REF: ESF SYSTEM DESCRIPTION
20E-1-4030 EF36
20E-1-4030 EF80

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THERE IS NO IMMEDIATE NOTICEABLE EFFECT. WHEN A CONDITION WHICH WOULD NORMALLY CAUSE A PHASE A CONTAINMENT ISOLATION SIGNAL TO BE GENERATED, THE AUTOMATIC ACTIONS WHICH NORMALLY OCCUR FOR THE SELECTED TRAIN DO NOT OCCUR. ANY ATTEMPT BY THE OPERATOR TO MANUALLY ACTUATE PHASE A CONTAINMENT ISOLATION, WHILE THE MALFUNCTION IS ACTIVE, IS UNSUCCESSFUL. IF BOTH TRAINS ARE SELECTED THEN THE FAILURE OF PHASE A CONTAINMENT ISOLATION TO FUNCTION PROPERLY MAY CAUSE THE INITIATING EVENT TO WORSEN. ALSO IF BOTH TRAINS ARE SELECTED, ANNUNCIATOR 5-B7 "CNMT PHASE A ISOLATION" WILL NOT ACTUATE. THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY MANUALLY CLOSING THE AFFECTED VALVES.

MALFUNCTION REMOVAL WILL RESTORE THE PHASE A CONTAINMENT ISOLATION CIRCUIT TO NORMAL.

EVENTS: NONE

RP05 FAILURE OF PHASE B CNMT ISOL TO ACTUATE

TYPE: GENERIC, RB

A) TRAIN A LOGIC
B) TRAIN B LOGIC

CAUSE: FAULTY LOGIC CARD(S)

REF: ESF SYSTEM DESCRIPTION
20E-1-4030 EF36
20E-1-4030 EF80

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THERE IS NO IMMEDIATE NOTICEABLE EFFECT. WHEN A CONDITION WHICH WOULD NORMALLY CAUSE A PHASE B CONTAINMENT ISOLATION SIGNAL TO BE GENERATED, THE AUTOMATIC ACTIONS WHICH NORMALLY OCCUR FOR THE SELECTED TRAIN DO NOT OCCUR. ANY ATTEMPT BY THE OPERATOR TO MANUALLY ACTUATE PHASE B CONTAINMENT ISOLATION, WHILE THE MALFUNCTION IS ACTIVE, IS UNSUCCESSFUL. IF BOTH TRAINS ARE SELECTED, THEN THE FAILURE OF PHASE B CONTAINMENT ISOLATION TO FUNCTION PROPERLY MAY CAUSE THE INITIATING EVENT TO WORSEN. ALSO IF BOTH TRAINS ARE SELECTED, ANNUNCIATOR 5-A7 "CNMT PHASE B ISOLATION" WILL NOT ACTUATE. (ANNUNCIATOR 5-A7 "CNMT PHASE B ISOLATION" WILL FLASH IF THE SSPS TRAINS HAVE CONFLICTING INFORMATION). THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY MANUALLY CLOSING THE AFFECTED CC VALVES.

MALFUNCTION REMOVAL WILL RESTORE THE PHASE B CONTAINMENT ISOLATION CIRCUIT TO NORMAL.

EVENTS: NONE

RP06 TURBINE TRIP INTERLOCK C-8 FAILS

TYPE: DISCRETE, RB

CAUSE: CIRCUIT FAILURE

REF: MAIN TURBINE SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE BYPASS-PERMISSIVE ANNUNCIATOR "TURBINE TRIP C8" WILL FAIL TO ACTUATE ON A TURBINE TRIP OR WILL FAIL TO CLEAR WHEN THE TURBINE TRIP SIGNAL TO SSPS IS RESET. THE TURBINE TRIP SIGNAL IS GENERATED FROM EITHER 4/4 TURBINE STOP VALVES BEING CLOSED, OR BY LOW PRESSURE ON 2/3 EMERGENCY TRIP HEADER PRESSURE TRANSMITTERS.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED CIRCUIT TO NORMAL.

EVEMTS: NONE

RP07 UNDER-FREQUENCY ON RCP BUS

TYPE: GENERIC, RB

A) BUS 156
B) BUS 157
C) BUS 158
D) BUS 159

CAUSE: FAULTY UNDERFREQUENCY RELAY ACTUATION

REF: 20E-1-4029 EF06
20E-1-4030 EF28
20E-1-4030 EF72
20E-1-4030 AP09
20E-1-4030 AP13
20E-1-4030 AP17
20E-1-4030 AP21

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES AN UNDERFREQUENCY SIGNAL ON THE SELECTED 6.9 KV RCP BUS. RCP BUS UNDERFREQUENCY STATUS LIGHTS ON 1PM05J INDICATE INDIVIDUAL RCP BUSSES THAT HAVE AN UNDERFREQUENCY CONDITION. ANNUNCIATOR 13-B2 "RCP BUS UNDERFREQ RX TRIP ALERT" ACTUATES. THE REACTOR PROTECTION LOGIC REQUIRES THAT 2/4 BUSSES HAVE AN UNDERFREQUENCY CONDITION BEFORE A REACTOR TRIP IS ACTUATED. ANNUNCIATOR 11-B5 "RCP BUS UNDER FREQ RX TRIP" ACTUATES ON 2/4 BUS UNDERFREQUENCY RELAY ACTUATIONS WHEN ABOVE P-7. ALL RCP BREAKERS OPEN WHEN THE CONDITIONS ABOVE ARE MET.

MALFUNCTION REMOVAL RESTORES THE SELECTED UNDERFREQUENCY RELAY(S) TO NORMAL.

EVENTS: NONE

RP08 UNDER-VOLTAGE ON RCP BUS

TYPE: GENERIC, RB

A) BUS 156
B) BUS 157
C) BUS 158
D) BUS 159

CAUSE: FAULTY UNDERVOLTAGE RELAY ACTUATION (SSV-T INPUT TO SSPS)

REF: 20E-1-4029 EF06
20E-1-4030 EF28
20E-1-4030 EF72
20E-1-4030 AP09
20E-1-4030 AP13
20E-1-4030 AP17
20E-1-4030 AP21

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES AN UNDERVOLTAGE ON THE SELECTED 6.9 KV RCP BUS. RCP BUS UNDERVOLTAGE STATUS LIGHTS ON 1PM05J INDICATE INDIVIDUAL RCP BUSSES THAT HAVE AN UNDERVOLTAGE CONDITION. ANNUNCIATOR 13-A2 "RCP BUS UNDERVOLT RX TRIP ALERT" ACTUATES. THE REACTOR PROTECTION LOGIC REQUIRES THAT 2/4 BUSSES HAVE AN UNDERVOLTAGE CONDITION BEFORE A REACTOR TRIP IS ACTUATED WHEN ABOVE P-7. ANNUNCIATOR 11-A5 "RCP BUS UNDERVOLT RX TRIP" ACTUATES ON 2/4 BUS UNDERVOLTAGE RELAY ACTUATIONS. AF PUMPS WILL AUTO START ON 2/4 BUS UNDERVOLTAGE ACTUATIONS.

MALFUNCTION REMOVAL RESTORES THE SELECTED UNDERVOLTAGE RELAY(S) TO NORMAL.

EVENTS: NONE

RP09 INADVERTENT FW ISOLATION

TYPE: GENERIC, RB

A) TRAIN A LOGIC
B) TRAIN B LOGIC

CAUSE: FAULTY LOGIC CARD(S)

REF: ESF SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, A FALSE FEEDWATER ISOLATION SIGNAL FOR THE SELECTED SSPS TRAIN (A OR B) WILL BE RECEIVED. THE FEEDWATER ISOLATION SIGNAL WILL CAUSE ALL ASSOCIATED VALVES THAT NORMALLY RECEIVE A FW ISOLATION SIGNAL FROM THE SELECTED TRAIN TO CLOSE. THE FW ISOLATION SIGNAL WILL CAUSE S/G WATER LEVELS TO DECREASE WHICH WILL RESULT IN A REACTOR TRIP AND AUX FEEDWATER ACTUATION AT LO-2 S/G LEVEL. PLANT ANNUNCIATORS WILL RESPOND ACCORDINGLY.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED FEEDWATER ISOLATION CIRCUIT TO NORMAL.

EVENTS: NONE

RP10 INADVERTENT PHASE A CONTAINMENT ISOLATION

TYPE: DISCRETE, RB

A) TRAIN A
B) TRAIN B

CAUSE: FAULTY K502/522 RELAY ACTUATION

REF: 20E-1-4030 EF11
20E-1-4030 EF36
20E-1-4030 EF53
20E-1-4030 EF97
20E-1-4030 EF60

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES TRAIN A AND/OR TRAIN B PHASE A CONTAINMENT ISOLATION TO BE ACTUATED. THIS WILL INITIATE THE REALIGNMENT OF ALL ASSOCIATED EQUIPMENT WITH PHASE A. ANNUNCIATOR 5-B7 "CNMT PHASE A ISOLATION" ACTUATES, AND THE ESF GROUP 3 MONITOR LIGHTS ON 1PM06J UPDATE IN RESPONSE TO THE ACTUATION. LETDOWN ISOLATION CAUSES PZR LEVEL TO INCREASE. RESETTING THE PHASE A ISOLATION HAS NO EFFECT WHILE THE MALFUNCTION IS STILL ACTIVE.

MALFUNCTION REMOVAL RESTORES THE FAULTY K502/522 RELAYS TO NORMAL.

EVENTS: NONE

RP11 INADVERTENT PHASE B CONTAINMENT ISOLATION

TYPE: GENERIC, RB

A) TRAIN A
B) TRAIN B

CAUSE: FAULTY K506 RELAY ACTUATION

REF: 20E-1-4030 EF36
20E-1-4030 EF80

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES TRAIN A AND/OR TRAIN B PHASE B CONTAINMENT ISOLATION TO BE ACTUATED. THIS WILL INITIATE THE REALIGNMENT OF ALL ASSOCIATED PHASE B VALVES. ANNUNCIATOR 5-A7"CNMT PHASE B ISOLATION"ACTUATES (ANNUNCIATOR 5-A7 "CNMT PHASE B ISOLATION" WILL FLASH IF THE SSPS TRAINS HAVE CONFLICTING INFORMATION) AND THE GROUP 6 ESF MONITOR LIGHTS ON 1PM06J UPDATE IN RESPONSE TO THE ACTUATION. RESETTING THE PHASE B ISOLATION HAS NO EFFECT WHILE THE MALFUNCTION IS STILL ACTIVE.

MALFUNCTION REMOVAL RESTORES THE FAULTY RELAY TO NORMAL.

EVENTS: NONE

RP12 INADVERTENT CONTROL ROOM VENT ISOLATION

TYPE: GENERIC, RB

A) TRAIN A
B) TRAIN B

CAUSE: FAULTY K602 CONTACT 17-18 ACTUATION

REF: 20E-1-4030 EF36
20E-1-4030 EF80

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED VC TRAIN RECIRC CHARCOAL ABSORBER TO ALIGN TO THE ABSORB MODE. IF THE SELECTED VC TRAIN SUPPLY FAN IS OPERATING, THE M/U FAN AUTO STARTS, DAMPERS, AND FILTER UNIT FOR THAT TRAIN WILL ALIGN TO SUPPLY M/U AIR VIA THE M/U FAN FROM THE TURB BLDG.

MALFUNCTION REMOVAL RESTORES THE FAULTY K602 RELAY TO NORMAL. THE OPERATOR MUST THEN DEPRESS THE ASSOCIATED CRVIRA/B PUSHBUTTON TO RESET THE SIGNAL.

EVENTS: NONE

RP13 REACTOR TRIP PERMISSIVE P-4 FAILS TO ACTUATE

TYPE: GENERIC, RB

A) TRAIN A LOGIC
B) TRAIN B LOGIC

CAUSE: FAULTY LOGIC CARD(S)

REF: REACTOR PROTECTION SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE P-4 PERMISSIVE FAILS AS IS. IF INSERTED WITH THE RX TRIP BREAKERS CLOSED, WHEN THE CONDITIONS FOR SATISFYING THE P-4 PERMISSIVE ARE MET (A REACTOR TRIP BREAKER AND ITS BYPASS BREAKER BOTH OPEN), THE P-4 PERMISSIVE FAILS TO PERFORM ITS NORMAL FUNCTIONS. THE "RX TRIP P4" BYPASS PERMISSIVE LIGHT WILL FLASH IF THE SSPS TRAINS HAVE CONFLICTING INFORMATION.

IF THE MALFUNCTION IS INSERTED WITH THE P-4 LOGIC SATISFIED, (A REACTOR TRIP BREAKER AND ITS BYPASS BREAKER BOTH OPEN), THE CLOSURE OF EITHER BREAKER WILL NOT CLEAR THE P-4 PERMISSIVE FOR THE SELECTED TRAIN. THE P-4 SIGNAL WILL PREVENT THE STARTUP OF THE MAIN TURBINE WHEN THE SIGNAL WOULD NOT NORMALLY BE PRESENT. IF THE MALFUNCTION IS ACTIVE WITH AN ACTIVE SI SIGNAL PRESENT, THE SI SIGNAL CANNOT BE RESET FROM THE MCB.

THE P-4 PERMISSIVE NORMALLY DOES THE FOLLOWING: ACTUATES A TURBINE TRIP; ACTUATES FEEDWATER ISOLATION; SEALS IN A CIRCUIT TO PREVENT RE-OPENING THE MAIN FEED WATER VALVES WHICH WERE CLOSED BY EITHER A SAFETY INJECTION ACTUATION OR A HI-2 STEAM GENERATOR LEVEL; PROVIDES A SIGNAL TO THE SAFETY INJECTION BLOCK AND RESET LOGIC CIRCUIT.

MALFUNCTION REMOVAL WILL RESTORE THE FAULTY P-4 CIRCUIT TO NORMAL.

EVENTS: NONE

RP14 FAILURE OF SAFETY INJECTION TO ACTUATE

TYPE: DISCRETE, RB

A) TRAIN A
B) TRAIN B

CAUSE: FAILURE OF K501/K521 TO ACTUATE

REF: 20E-1-4030 EF11
20E-1-4030 EF36
20E-1-4030 EF60
20E-1-4030 EF80

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE FAILURE OF A MANUAL OR AUTOMATIC SAFETY INJECTION SIGNAL TO BE ACTUATED. IF A PLANT CONDITION CALLS FOR A SAFETY INJECTION ACTUATION, THE UNAFFECTED TRAIN WILL ACTUATE AN SI SIGNAL FOR THAT TRAIN ONLY. ONLY THE EQUIPMENT ASSOCIATED WITH UNAFFECTED TRAIN WILL START AND/OR REPOSITION. A REACTOR TRIP, TURBINE TRIP, PHASE A CONTAINMENT ISOLATION, FEEDWATER ISOLATION, AND DIESEL GENERATOR START SIGNAL WILL BE ACTUATED/STARTED ON THE AFFECTED TRAIN. THE EQUIPMENT ASSOCIATED WITH THE FAILED TRAIN MAY BE MANUALLY REPOSITIONED AND/OR STARTED TO MITIGATE THE EFFECTS OF THIS MALFUNCTION.

THE "SI ACTUATED" BYPASS PERMISSIVE LIGHT WILL FLASH IF THE SSPS TRAINS HAVE CONFLICTING INFORMATION.

MALFUNCTION REMOVAL RESTORES THE FAILED K501/521 RELAYS TO NORMAL.

EVENTS: NONE

RP15 SAFEGUARD SEQUENCING FAILURE

TYPE: GENERIC, RB

A)	T1A	(0 SEC)	CV
B)	T1B	(0 SEC)	CV
C)	T2A	(5 SEC)	SI
D)	T2B	(5 SEC)	SI
E)	T3A	(10 SEC)	RH
F)	T3B	(10 SEC)	RH
G)	T4A	(15 SEC)	WO
H)	T4B	(15 SEC)	WO
I)	T5A	(15 SEC)	CS
J)	T5B	(15 SEC)	CS
K)	T6A	(18 SEC)	CS
L)	T6B	(18 SEC)	CS
M)	T7A	(40 SEC)	CS
N)	T7B	(40 SEC)	CS
O)	T8A	(20 SEC)	CC
P)	T8B	(20 SEC)	CC
Q)	T9A	(25 SEC)	SX
R)	T9B	(25 SEC)	SX
S)	T10A	(35 SEC)	AF
T)	T10B	(35 SEC)	AF (NOT USED IN SSPS)

CAUSE: TIMER CONTACT FAILURE TO CLOSE

REF: 20E-1-4030 EF01
20E-1-4030 EF02

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE AFFECTED EQUIPMENT TO FAIL TO START. WHEN A PLANT CONDITION ARISES THAT CAUSES A LOSS OF THE 4160 KV ESF BUSSES (LOSS OF OFF SITE POWER), OR A LOSS OF OFF SITE POWER CONCURRENT WITH A SAFETY INJECTION ACTUATION, THE DIESEL GENERATORS START AND ENERGIZE THESE BUSSES. THE UNAFFECTED EQUIPMENT STARTS AT THE DESIGNATED SEQUENCE TIME AND OPERATES PROPERLY.

THE EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY MANUALLY STARTING THE EQUIPMENT AFFECTED BY THE FAILED CONTACT.

MALFUNCTION REMOVAL RESTORES THE FAILED TIMER CONTACT TO NORMAL.

EVENTS: NONE

RP16 PERMISSIVE P-6 FAILS TO ACTUATE

TYPE: GENERIC, RB

A) TRAIN A LOGIC B) TRAIN B LOGIC

CAUSE: FAULTY LOGIC CARD(S)

REF: REACTOR PROTECTION SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE P-6 PERMISSIVE FAILS AS IS. IF INSERTED ON A POWER INCREASE, WHEN THE CONDITIONS FOR SATISFYING THE P-6 PERMISSIVE ARE MET (EITHER OF TWO INTERMEDIATE RANGE CHANNELS INCREASES ABOVE 1E-10 AMPERES), THE P-6 PERMISSIVE FAILS TO PERFORM ITS NORMAL FUNCTIONS. THE "SR BLOCK PERMISSIVE P6" BYPASS PERMISSIVE LIGHT WILL FLASH IF THE SSPS TRAINS HAVE CONFLICTING INFORMATION.

IF THE MALFUNCTION IS INSERTED ON A POWER DECREASE WITH THE P-6 LOGIC SATISFIED, DECREASING POWER BELOW THE P-6 SETPOINT WILL NOT CLEAR THE P-6 PERMISSIVE. THIS RESULTS IN THE P-6 PERMISSIVE BEING PRESENT WHEN IT NORMALLY WOULD NOT BE THUS PREVENTING THE SELECTED SOURCE RANGE DETECTOR FROM RE-ENERGIZING AND FROM REACTIVATING THE SR HIGH FLUX REACTOR TRIP (DEPENDENT ON TRAIN SELECTED).

THE P-6 PERMISSIVE NORMALLY DOES THE FOLLOWING: ALLOWS THE OPERATOR TO MANUALLY DEENERGIZE THE HIGH VOLTAGE SUPPLY TO THE SOURCE RANGE NUCLEAR DETECTORS AND BLOCK THE SOURCE RANGE HIGH FLUX REACTOR TRIP.

FAILURE OF THE P-6 PERMISSIVE, WHILE INCREASING POWER, PREVENTS THE MCB BLOCK SWITCHES FROM BLOCKING THE SR HIGH FLUX TRIP WHICH WILL RESULT IN A REACTOR TRIP, IF POWER CONTINUES TO INCREASE. THE OPERATOR MAY PREVENT THE TRIP BY BYPASSING THE RESPECTIVE TRIP AT THE SOURCE RANGE DRAWER.

MALFUNCTION REMOVAL WILL RESTORE THE FAULTY P-6 CIRCUIT TO NORMAL.

EVENTS: NONE

RP17 PERMISSIVE P-7 FAILS TO ACTUATE

TYPE: GENERIC, RB

A) TRAIN A LOGIC
B) TRAIN B LOGIC

CAUSE: FAULTY LOGIC CARD(S)

REF: REACTOR PROTECTION SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE P-7 PERMISSIVE FAILS AS IS. IF DONE ON A POWER DECREASE, WHEN THE CONDITIONS FOR SATISFYING THE P-7 PERMISSIVE ARE MET (3 OF 4 POWER RANGE NUCLEAR INSTRUMENTS AND BOTH TURBINE IMPULSE PRESSURE SIGNALS INDICATE THAT POWER IS < 10%), THE P-7 PERMISSIVE FAILS TO PERFORM ITS NORMAL FUNCTIONS. THE "LOW POWER TRIPS BLOCKED P7" BYPASS PERMISSIVE LIGHT WILL FLASH IF THE SSPS TRAINS HAVE CONFLICTING INFORMATION.

IF THE MALFUNCTION IS INSERTED ON A POWER INCREASE WITH THE P-7 LOGIC SATISFIED, THE POWER INCREASE ABOVE 10% POWER WILL NOT CLEAR THE P-7 PERMISSIVE. THIS RESULTS IN THE P-7 PERMISSIVE BEING PRESENT WHEN IT NORMALLY WOULD NOT BE WHICH WILL PREVENT THE ASSOCIATED TRIPS FROM OCCURRING WHEN THAT TRIP SIGNAL IS GENERATED.

THE P-7 PERMISSIVE NORMALLY AUTOMATICALLY BLOCKS THE FOLLOWING REACTOR TRIPS: LOW REACTOR COOLANT FLOW IN MORE THAN ONE LOOP, RCP BREAKERS OPEN IN MORE THAN ONE LOOP, RCP BUS UNDERVOLTAGE/ UNDERFREQUENCY TRIP, PRESSURIZER LOW PRESSURE TRIP AND PRESSURIZER HIGH WATER LEVEL TRIP.

MALFUNCTION REMOVAL WILL RESTORE THE FAULTY P-7 CIRCUIT TO NORMAL.

EVENTS: NONE

RP18 PERMISSIVE P-8 FAILS TO ACTUATE

TYPE: GENERIC, RB

- A) TRAIN A LOGIC
- B) TRAIN B LOGIC

CAUSE: FAULTY LOGIC CARD(S)

REF: REACTOR PROTECTION SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE P-8 PERMISSIVE FAILS AS IS. IF INSERTED ON A POWER DECREASE, WHEN THE POWER IS REDUCED BELOW 30% THEN THE P-8 PERMISSIVE WILL BE MAINTAINED THUS MAINTAINING THE SINGLE LOOP LOSS OF FLOW REACTOR TRIP AND THE TURBINE TRIP-REACTOR TRIP. THE "LOW POWER TRIP BLOCKED P8" BYPASS PERMISSIVE LIGHT WILL FLASH IF THE SSPS TRAINS HAVE CONFLICTING INFORMATION.

IF THE MALFUNCTION IS INSERTED ON A POWER INCREASE WITH THE P-8 LOGIC SATISFIED, THE POWER INCREASE ABOVE 30% POWER WILL NOT CLEAR THE P-8 PERMISSIVE. THIS RESULTS IN THE P-8 PERMISSIVE BEING PRESENT WHEN IT NORMALLY WOULD NOT BE WHICH WILL PREVENT THE REACTOR FROM TRIPPING ON A LOW FLOW CONDITION IN ONE REACTOR COOLANT LOOP AND FROM TRIPPING ON A TURBINE TRIP SIGNAL.

AT POWERS BELOW 30%, THE P-8 PERMISSIVE NORMALLY ALLOWS FOR A LOSS OF COOLANT FLOW IN ONE REACTOR COOLANT LOOP AND OR A TURBINE TRIP WITHOUT CAUSING A REACTOR TRIP. IF P-8 FAILS TO CLEAR AS POWER IS REDUCED BELOW 30% THEN THE ABOVE TWO TRIPS WILL REMAIN ACTIVE AND TRIP THE REACTOR IF ACTUATED.

MALFUNCTION REMOVAL WILL RESTORE THE FAULTY P-8 CIRCUIT TO NORMAL.

EVENTS: NONE

RP19 PERMISSIVE P-10 FAILS TO ACTUATE

TYPE: GENERIC, RB

A) TRAIN A LOGIC
B) TRAIN B LOGIC

CAUSE: FAULTY LOGIC CARD(S)

REF: REACTOR PROTECTION SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE P-10 PERMISSIVE FAILS AS IS. IF INSERTED ON A POWER INCREASE, WHEN THE CONDITIONS FOR SATISFYING THE P-10 PERMISSIVE ARE MET (2 OF 4 POWER RANGE NUCLEAR INSTRUMENTS INDICATE THAT REACTOR POWER IS GREATER THAN 10%), THE P-10 PERMISSIVE FAILS TO PERFORM ITS NORMAL FUNCTIONS. THE "POWER RANGE PERMISSIVE P10" BYPASS PERMISSIVE LIGHT WILL FLASH IF THE SSPS TRAINS HAVE CONFLICTING INFORMATION.

IF THE MALFUNCTION IS INSERTED ON A POWER DECREASE WITH THE P-10 LOGIC SATISFIED, DECREASING POWER BELOW 10% WILL NOT CLEAR THE P-10 PERMISSIVE. THIS RESULTS IN THE P-10 PERMISSIVE BEING PRESENT WHEN IT NORMALLY WOULD NOT BE, THUS MAINTAINING A BLOCK ON THE INTERMEDIATE RANGE HIGH FLUX AND THE POWER RANGE HIGH FLUX LOW REACTOR TRIPS.

THE P-10 PERMISSIVE NORMALLY DOES THE FOLLOWING: GENERATES A SIGNAL TO ALLOW MANUAL BLOCKING OF THE INTERMEDIATE RANGE HIGH FLUX REACTOR TRIP; GENERATES A SIGNAL TO ALLOW MANUAL BLOCK OF THE POWER RANGE HIGH FLUX [LOW SETPOINT] REACTOR TRIP; GENERATES A SIGNAL TO REMOVE PERMISSIVE P-7; BLOCKS SOURCE RANGE HIGH FLUX REACTOR TRIP AND BLOCKS MANUAL REENERGIZING OF THE SOURCE RANGE DETECTOR HIGH VOLTAGE. IF THIS MALFUNCTION IS COMBINED WHEN MALFUNCTION RP22 IS SET THEN THE RESULTS WILL BE SIMILAR TO MALFUNCTION RP17

MALFUNCTION REMOVAL WILL RESTORE THE FAULTY P-10 CIRCUIT TO NORMAL.

EVENTS: NONE

RP20 PERMISSIVE P-11 FAILS TO ACTUATE

TYPE: GENERIC, RB

A) TRAIN A LOGIC
B) TRAIN B LOGIC

CAUSE: FAULTY LOGIC CARD(S)

REF: REACTOR PROTECTION SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE P-11 PERMISSIVE FAILS AS IS. IF INSERTED ON A PLANT SHUTDOWN, WHEN THE CONDITIONS FOR SATISFYING THE P-11 PERMISSIVE ARE MET (2 OF 3 PRESSURIZER PRESSURE SIGNALS ARE LESS THAN 1930 PSIG), THEN THE LOW PRESSURE SI SIGNALS CANNOT BE BLOCKED FROM THE MAIN CONTROL BOARD. IF THE SI ACCUMULATOR DISCHARGE VALVES 1SI8808 A-D ARE MANUALLY CLOSED, THE VALVES WILL GO FULL CLOSE THEN AUTO OPEN. THE "PZR LOW PRESS SI BLOCK PERMISSIVE P11" BYPASS PERMISSIVE LIGHT WILL FLASH IF THE SSPS TRAINS HAVE CONFLICTING INFORMATION.

IF THE MALFUNCTION IS INSERTED ON A PLANT STARTUP, INCREASING PRESSURIZER PRESSURE ABOVE 1930 PSIG WILL NOT CLEAR THE P-11 PERMISSIVE. THIS RESULTS IN THE P-11 PERMISSIVE BEING PRESENT WHEN IT NORMALLY WOULD NOT BE. THIS MAINTAINS THE BLOCK OF THE LOW PRESSURIZER PRESSURE SAFETY INJECTION AND LOW STEAMLINE PRESSURE SI AT ACTUAL PRESSURES ABOVE 1930 PSIG. IF AN SI ACCUMULATOR DISCHARGE VALVE 1SI8808 A-D IS CLOSED, THEN IT WILL NOT AUTO OPEN WHEN PRESSURE INCREASES ABOVE 1930 PSIG.

THE P-11 PERMISSIVE NORMALLY ALLOWS THE OPERATOR TO BLOCK THE LOW PRESSURIZER PRESSURE SAFETY INJECTION AND LOW STEAMLINE PRESSURE SI SIGNALS BELOW 1930 PSIG. ABOVE 1930 PSIG, IT SIGNALS THE SI ACCUMULATOR ISOLATION VALVES 1SI8808 A-D TO OPEN AND REINSTATES THE SI SIGNALS.

MALFUNCTION REMOVAL WILL RESTORE THE FAULTY P-11 CIRCUIT TO NORMAL.

EVENTS: NONE

RP21 LO-LO TAVG PERMISSIVE P-12 FAILS TO ACTUATE

TYPE: GENERIC, RB

A) TRAIN A LOGIC
B) TRAIN B LOGIC

CAUSE: FAULTY LOGIC CARD(S)

REF: REACTOR PROTECTION SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE P-12 PERMISSIVE FAILS AS IS. IF INSERTED ON A PLANT SHUTDOWN, WHEN THE CONDITIONS FOR SATISFYING THE P-12 PERMISSIVE ARE MET (2 OF 4 T_{ave} SIGNALS ARE LESS THAN 550 °F), THE P-12 PERMISSIVE FAILS TO PERFORM ITS NORMAL FUNCTIONS. THUS ALL STEAM DUMP VALVES WILL REMAIN OPERATIONAL. THE "LO-2 TAVE STM DUMP INTLK P12" BYPASS PERMISSIVE LIGHT WILL FLASH IF THE SSPS TRAINS HAVE CONFLICTING INFORMATION.

IF THE MALFUNCTION IS INSERTED ON A PLANT STARTUP, INCREASING T_{ave} ABOVE 550 °F WILL NOT CLEAR THE P-12 PERMISSIVE. THIS RESULTS IN THE P-12 PERMISSIVE BEING PRESENT WHEN IT NORMALLY WOULD NOT BE ALLOWING OPERATION OF ONLY 3 STEAM DUMP VALVES (1MS004A,E,J).

THE P-12 PERMISSIVE NORMALLY BLOCKS STEAM DUMP OPERATION BELOW 550 °F.

MALFUNCTION REMOVAL WILL RESTORE THE FAULTY P-12 CIRCUIT TO NORMAL.

EVENTS: NONE

RP22 PERMISSIVE P-13 FAILS TO ACTUATE

TYPE: GENERIC, RB

A) TRAIN A LOGIC
B) TRAIN B LOGIC

CAUSE: FAULTY LOGIC CARD(S)

REF: REACTOR PROTECTION SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE P-13 PERMISSIVE FAILS AS IS. WHEN THE CONDITIONS FOR SATISFYING THE P-13 PERMISSIVE ARE MET (EITHER OF THE TURBINE IMPULSE PRESSURE SIGNALS ARE GREATER THAN 10% POWER), THE P-13 PERMISSIVE FAILS TO PERFORM ITS NORMAL FUNCTIONS. THE "LOW TURB IMPULSE PRESSURE PERMISSIVE P13" BYPASS PERMISSIVE LIGHT WILL FLASH IF THE SSPS TRAINS HAVE CONFLICTING INFORMATION.

THE P-13 PERMISSIVE NORMALLY REMOVES THE P-7 PERMISSIVE. IF THIS MALFUNCTION IS COMBINED WHEN MALFUNCTION RP19 IS SET, THEN THE RESULTS WILL BE SIMILAR TO MALFUNCTION RP17

MALFUNCTION REMOVAL WILL RESTORE THE FAULTY P-13 CIRCUIT TO NORMAL.

EVENTS: NONE

RP23 PERMISSIVE P-14 FAILS TO ACTUATE

TYPE: GENERIC, RB

A) TRAIN A LOGIC
B) TRAIN B LOGIC

CAUSE: FAULTY LOGIC CARD(S)

REF: REACTOR PROTECTION SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE P-14 PERMISSIVE FAILS AS IS. INSERTED ON A STEAM GENERATOR LEVEL INCREASE, WHEN THE CONDITIONS FOR SATISFYING THE P-14 PERMISSIVE ARE MET (2 OF 4 STEAM GENERATOR NARROW RANGE LEVELS ARE ABOVE 81.4%), THE P-14 PERMISSIVE FAILS TO TRIP THE MAIN TURBINE AND THE FW PUMPS AND FAILS TO GENERATE A FW ISOLATION SIGNAL (DEPENDENT ON SELECTED TRAIN).

IF THE MALFUNCTION IS INSERTED ON A STEAM GENERATOR LEVEL DECREASE WITH THE P-14 PERMISSIVE SATISFIED, THE P-14 PERMISSIVE WILL STILL BE PRESENT WHEN THE CONDITIONS FOR SATISFYING THE P-14 PERMISSIVE ARE NO LONGER MET. THIS PREVENTS FEEDING THE STEAM GENERATORS AS A CLOSE SIGNAL IS MAINTAINED TO THE STEAM GENERATOR FEEDWATER VALVES. ALL MAIN FEED PUMPS AND THE MAIN TURBINE ALSO MAINTAIN A TRIP SIGNAL.

MALFUNCTION REMOVAL WILL RESTORE THE FAULTY P-14 LOGIC CARD TO NORMAL.

EVENTS: NONE

RP24 INADVERTENT SAFETY INJECTION

TYPE: GENERIC, RB

A) TRAIN A
B) TRAIN B

CAUSE: FAILURE OF SAFEGUARDS OUTPUT CARD A516

REF: 20E-1-4030 EF01
20E-1-4030 EF02
20E-1-4030 EF11
20E-1-4030 EF36
20E-1-4030 EF60
20E-1-4030 EF80

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED SAFETY INJECTION TRAIN TO ACTUATE. TRAIN A ACTUATION AUTOMATICALLY STARTS THE COMPONENTS LISTED ON SCHEMATIC DIAGRAM 20E-1-4030 EF01, AND ACTUATES A FW ISOLATION, CMNT PHASE A ISOLATION, RX TRIP/TURBINE TRIP, AND A CONTROL ROOM VENT ISOLATION. TRAIN B ACTUATION AUTOMATICALLY STARTS THE COMPONENTS LISTED ON SCHEMATIC DIAGRAM 20E-1-4030 EF02, AND ACTUATES A FW ISOLATION, CMNT PHASE A ISOLATION, RX TRIP/TURBINE TRIP, AND A CONTROL ROOM VENT ISOLATION.

THE "SI ACTUATED" BYPASS PERMISSIVE LIGHT WILL FLASH IF THE SSPS TRAINS HAVE CONFLICTING INFORMATION.

THE RX TRIP SIGNAL IS FROM A TURBINE TRIP ABOVE P8. IF RX POWER IS LESS THAN P8, A RX TRIP SIGNAL IS NOT GENERATED.

MALFUNCTION REMOVAL RESTORES THE FAILURE TO NORMAL AND WILL ALLOW THE SI TO BE RESET.

EVENTS: 1) LER 06-02-87-016
2) LER 06-02-89-001

LICENSEE EVENT REPORT (LER)

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Title (4) INADVERTANT SAFETY INJECTION DURING DIESEL GENERATOR OPERABILITY SURVEILLANCE DUE TO PROCEDURAL INADEQUACY

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
0 2	1 1	8 9	8 9	0 0 1	0 0	0 3	0 9	8 9	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 6

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	X 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER AREA CODE
R. Flahive, Technical Staff Supervisor Ext. 2243	8 1 5 2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	E K	R L Y	W 1 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

__ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15) Month | Day | Year

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

On February 11, 1989 at approximately 1240, with Byron Unit 2 in Mode 6, a safety injection (SI) actuation of Train A and B equipment occurred during the performance of a Diesel Generator Operability Surveillance. Actuation of B Train equipment was expected during the surveillance, but the A Train actuation was unanticipated. The inadvertent SI signal was generated when the required 2 of 3 coincidence was satisfied for Containment Pressure High. One high pressure signal was generated by Instrument Maintenance personnel, who were testing pressure channel 935. Channel 934 generated a high pressure signal when its instrument power bus was deenergized during the surveillance test. This occurred because the instrument bus was supplied by a constant voltage transformer from a bus that was intentionally deenergized during the surveillance procedure. The only unexpected automatic equipment actuations that occurred were the 2A Centrifugal Charging Pump and the 2A Diesel Generator. All other equipment was either running or in pull-to-lock. The surveillance was successfully completed later on February 11, 1989.

The root cause of the SI was inadequate precautions in the surveillance procedure. Because of this deficiency, operating personnel involved in this event did not recognize that the instrument bus would be deenergized during the surveillance test.

The procedures used to perform the Diesel Generator Operability Test will be revised to require consideration of instrument inverter status. This LER will be placed in the licensed operator required reading program.

There have been no previous occurrences of similar events.

(0255R/0031R/022889)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

Form Rev 2.0

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TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 2/11/89 / 1240

Unit 2 MODE 6 - Refueling Rx Power 0% RCS [AB] Temperature/Pressure 0 PSIG / 85° F

B. DESCRIPTION OF EVENT:

On February 11, 1989 at approximately 1240, during the performance of Technical Staff 18 Month Diesel Generator Operability Surveillance (2BVS 8.1.1.2.f-14) a safety injection actuation (SI) occurred on both A and B Engineered Safety Features (ESF) [JE] trains. A safety injection actuation of the Diesel Generator [EK] and the Safeguards Sequencer [JE] on B train was expected during the performance of this surveillance, but the A train SI was unanticipated.

This surveillance (2BVS 8.1.1.2.f-14) is performed to satisfy the requirements of Byron Technical Specification 4.8.1.1.2.f.6, for B train ESF equipment, by simulating a loss of ESF Bus Voltage concurrent with an SI actuation test signal. To accomplish this, the following activities are performed:

1. All B train ESF pumps are started and left running.

2. Test switches on degraded voltage relay 427 B242Y are opened which simulate a degraded voltage condition on ESF Bus 242 [EB].

3. At 310 seconds (± 30 seconds) Air Circuit Breakers (ACB) 2421, 2422 and 2424 trip, deenergizing ESF Bus 242.

4. All B train ESF pumps are shed from ESF Bus 242.

5. As soon as the ESF Bus is deenergized, an SI is initiated by manually actuating relays K608 and K611 in auxiliary electric panel, 2PA10J.

6. The 2B Diesel Generator (DG), auto starts and reenergizes ESF Bus 242.

7. The B train Safeguards Sequencer, starts and sequences B train ESF equipment back onto ESF Bus 242.

8. The loads on the 2B DG are transferred to offsite power and the surveillance is terminated.

The surveillance activities progressed as planned until ESF Bus 242 was deenergized in step 3 (above). At that point, all pumps were shed from the ESF Bus and the manual SI to the 2B Diesel Generator and the Safeguards Sequencer was initiated. However, concurrent with the manual (planned) SI, an inadvertent SI signal was generated on both A and B train when ESF Bus 242 was deenergized.

B. DESCRIPTION OF EVENT CONTINUED:

The inadvertent SI signal was generated when the required actuation logic was satisfied for CONTAINMENT PRESSURE HIGH (two of three coincidence). This logic was made up by two events which appeared to be independent of the surveillance activities. These events were:

Instrument Maintenance personnel (non-licensed) were performing instrument time response testing and had containment pressure channel 935 [EF] in TEST.

Instrument Bus 214 [EF], normally powered by instrument inverter 214, was being powered by ESF Bus 242 through a constant voltage transformer due to an out-of-service (OOS) condition on instrument inverter 214. This instrument bus provides power to containment pressure channel 934 (protection Channel IV).

When the undervoltage condition caused ESF Bus 242 to deenergize, instrument bus 214 was also deenergized causing a trip condition on Containment Pressure channel 934. This condition, in conjunction with channel 935 for Containment Pressure being in TEST, satisfied the 2 out of 3 logic required for an SI. Shift personnel responded using appropriate operating procedures.

Subsequent to the inadvertent SI, the 2B DG output breaker, (ACB 2423), did not close. This prevented the 2B Diesel Generator from reenergizing the ESF Bus 242. This condition also left instrument bus 214 and the B Train Safeguard Sequencer deenergized. Without a source of power, the B train Sequencer was unable to respond to the SI signal and no B Train ESF equipment, except the 2B DG, automatically started.

Investigation into the failure of the DG output breaker to close, revealed that a contact on 2B Diesel Generator auxiliary relay DG2BX [EK] failed to make up, preventing auxiliary relay DG2BX1 from energizing. With this relay deenergized, the logic required to close the Diesel Generator ouput breaker was not satisfied and the breaker did not close.

A review of the control board by the Nuclear Station Operator (NSO) (licensed reactor operator) following the SI, confirmed that no B train equipment had automatically started except the 2B DG. This review also indicated that the 2A DG and the 2A Centrifugul Charging Pump (CV) [CB] also auto started. All other A train equipment was either running prior to the event, or was in pull-to-lock due to other Mode 6 operation requirements. After assessing the situation, and after receiving concurrence from the Shift Engineer (licensed senior reactor operator), the NSO proceeded to shut down unneeded equipment and to restore normal plant conditions per operating procedures.

At approximately 1600 on February 11, 1989, with the plant conditions restored to normal and the causes of the event corrected, surveillance 2BVS 8.1.1.2.f-14 was reentered. The testing activities progressed without further incident and the surveillance was successfully completed.

Throughout this event the 2B Diesel Generator was not required to be operable. All operator actions taken during this event were correct. Except for the above mentioned components, no systems or components were inoperable that contributed to this event. This event is reportable in accordance with 10CFR50.73(a)(2)(iv) due to the automatic ESF actuations.

C. CAUSE OF EVENT:

The intermediate cause for the inadvertent Safety Injection was the deenergization of instrument bus 214 concurrent with testing on one channel of containment pressure. The root cause of the inadvertent Safety Injection was inadequate precautions in procedure 2BVS 8.1.1.2.f-14. The procedure did not caution that an ESF action could occur when Bus 242 is deenergized if any protection channels are being tested. The operating personnel involved in this event did not recognize that when ESF Bus 242 was deenergized instrument bus 214 and protection channel IV would also be deenergized. Had this fact been noted, it would also have been recognized that testing on containment pressure was not compatible with the surveillance activities in procedure 2BVS 8.1.1.2.f-14 and this event would have been prevented.

The immediate cause of the Diesel Generator output breaker failing to close was the failure of the auxiliary relay DG2BX. A dirty contact was found on this relay. After cleaning the relay functioned adequately.

D. SAFETY ANALYSIS:

There were no safety consequences as a result of the inadvertent SI. The reactor vessel head was not installed thus eliminating the possibility of any overpressurization. Had this event occurred with the reactor vessel head installed, the overpressurization protection system was available to provide its intended safety function.

There were no safety consequences as a result of the 2B Diesel Generator output breaker failing to close. The 2A Diesel Generator was fully operable during this event, and as evidenced by the results of the A train Safety Injection, was capable of performing its intended safety function. This surveillance is only performed during modes 5 and 6 when just one Diesel Generator is required to be operable.

The limiting case for the 2B Diesel Generator failure would have been if the 2A Diesel Generator was inoperable when the 2B Diesel Generator failed. In this case AC power could have been supplied to Bus 242 through the cross ties to the Unit 1 Diesel Generators.

E. CORRECTIVE ACTIONS:

The following corrective actions were taken and are planned to prevent reoccurrence of this event.

A caution will be placed in the 1/2BVS 8.1.1.2.f-14 and 1/2BVS 8.1.1.2.f-13 procedures to warn of possible ESF actuation with any protection channels in TEST. The caution will also instruct that the status of instrument inverters be evaluated prior to performance of the surveillance. Action Item Record (AIR) 454-225-89-0073 is tracking the completion of this activity.

When the instrument inverters are taken out-of-service, power is supplied to the instrument buses through the constant voltage transformers. A note has been placed in the Electrical Distribution Book which states "INTERRUPTING POWER TO THE CONSTANT VOLTAGE XFMR CAN CAUSE ESF ACTUATION". This book is available for review prior to taking electrical equipment out-of-service.

E. CORRECTIVE ACTIONS CONTINUED:

The DG2BX auxiliary relay contacts were cleaned. An investigation will take place to define any additional preventive maintenance or testing of the auxiliary relays in the DC undervoltage circuit. Action Item Record (AIR) 454-225-89-0072 is tracking the completion of this activity.

The account of this event will be placed in the Licensed Operator required reading program for dissemination to all license holders.

F. RECURRING EVENTS SEARCH AND ANALYSIS:

There have been other unintentional Safety Injection Actuations at Byron Station, but this is the first time that an inadvertent SI was caused by an instrument bus being deenergized.

There are no previous occurrences to the DG output breaker failing to close.

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Westinghouse	relay	AR2	ST No. 1456C88A01

A review of the NPRDS data base, and Byron's Work Request system revealed several failures as a result of dirty contacts. However, sufficient evidence to establish a trend was not found.

LICENSEE EVENT REPORT (LER)

RP24

Facility Name (1)	Docket Number (2)	Page (3)
Byron, Unit 2	0 5 0 0 0 4 5 5	1 of 0

Title (4): ADVERTENT TRAIN A SAFETY INJECTION DURING SURVEILLANCE TEST DUE TO PERSONNEL ERROR

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0 8	3 1	8 7	8 7	0 1 6	0 1	1 0	0 8	8 7	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 4

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	Other (Specify in Abstract below and in Text)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER	
	AREA CODE	
J. Schrock, Operating Engineer Ext. 2216	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

On August 31, 1987, two Nuclear Station Operators (NSO) were performing the Unit 2 Train A Solid State Protection System Bi-Monthly Surveillance. During the surveillance one NSO noticed a light bulb was burned out. He replaced the bulb and the surveillance was continued. The second NSO assumed that the previous step in the surveillance had been completed prior to the bulb replacement. However, Step 8 in the surveillance was never completed. Further in the surveillance, the Main Control Room Unit 2 Operator informed the two NSO's that a Safety Injection had just occurred. The NSOs performed the system restoration and exited the surveillance.

The root cause of the event was a communication breakdown between the two NSOs performing the surveillance.

All Train A Safety Injection equipment started as designed, except the 2A diesel generator, which was out-of-service. The Unit was properly recovered without incident.

Corrective Action included disciplinary action against the two NSOs performing the surveillance.

There have been two previous occurrences.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year		Sequential Number		Revision Number	Page (3)
Byron, Unit 2	0 5 0 0 0 4 5 5	8 7	-	0 1 1 6	-	0 1	0 2 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 8/31/87 / 0920

Unit 2 MODE 4 - Hot Shutdown Rx Power 0 RCS [AB] Temperature/Pressure 140°F / 150 psig

B. DESCRIPTION OF EVENT:

On August 31, 1987, at 0920, two Nuclear Station Operators (NSO) were performing the Unit 2 Train A Solid State Protection System (EF)[JE] Bi-Monthly Surveillance. One NSO was performing the steps of the procedure and the other NSO was reading and signing off the surveillance steps. At Step 8, in the surveillance procedure, the NSO performing the steps noticed a burned out light bulb on the output bay status panel. The NSO performing the steps obtained a new light bulb and replaced the bulb. The NSO reading and signing off the steps assumed that the NSO performing the steps had completed Step 8 prior to replacing the bulb. Therefore, when the surveillance was resumed, Step 8 was not performed. The surveillance was continued and at Step 16 the Unit 2 Control Room Operator (Licensed) notified the NSOs performing the surveillance that a Safety Injection (SI)[BQ][BR] had just occurred. The NSOs performed the system restoration steps and exited the surveillance.

In the Main Control Room the Train A Safety Injection functions were verified to perform properly with the exception of the 2A Diesel Generator (DG)[EK], which was out-of-service at the time. Since the surveillance was the Train A logic test, only the Train A equipment auto started. To prevent an RCS level and pressure transient, the 2A Safety Injection pump, the 2A Residual Heat Removal (RH)[BP] pump, and the 2A Auxiliary Feedwater (AF)[BA] pump were all placed in the pull-to-lock position to stop these pumps.

After the Safety Injection timer had timed out (60 seconds) the Safety Injection and Phase A Containment Isolation Signals were reset. In addition, the 2A Centrifugal Charging (CV)[CB] pump was realigned for normal injection and the 2SI8801A valve was closed. The Train A SI, RH, and AF pumps were then returned to the after trip position. 2BEP-0 (Reactor Trip or Safety Injection) and 2BEP ES-1.1 (SI Termination) were properly performed to restore the plant.

C. CAUSE OF EVENT:

The root cause of the Safety Injection was inadequate communication between the two NSOs performing the surveillance. A contributing factor to the event was the perceived time constraint associated with the completion of this lengthy surveillance.

During this surveillance the Tech Spec Action Statement time limit allows 2 hours for the completion of the surveillance. The 2 hour completion time requires two NSOs to be present in the Auxiliary Electric Equipment Room with one reading and signing off the steps while the other performs the steps. If any abnormal action occurs during this surveillance, more pressure is placed on the NSOs to complete it within the time limit.

(1647N/0196M)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year		Sequential Number		Revision Number	Page (3)
Byron, Unit 2	0 5 0 0 0 4 5 5	8 7	-	0 1 6	-	0 1	0 3 OF 0

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

D. SAFETY ANALYSIS:

The plant and public safety were not affected by this event. All safeguards equipment associated with Train A functioned as designed. The Control Room operator placed the Train A pumps in the pull-to-lock position to prevent an RCS level and pressure transient. The Safety Significance would be the same if the events had occurred under any different credible initial conditions.

E. CORRECTIVE ACTIONS:

The professionalism program will cover the use of complete and proper communications with the entire operating department on a weekly basis for the next six weeks starting 9/4/87.

An operating engineer has discussed this event with all operating shifts.

The conduct of operations procedure (BAP 300-1) will be revised to include guidance to the operators to exit any procedure and restore the system to a safe configuration if the Tech Spec action time limit is running out. The operating engineer has discussed this topic with each shift. Action Item Record 454-512-87-0244 will track the completion of the corrective action.

This surveillance was reperformed and was satisfactorily completed.

Disciplinary action was taken against the two NSOs involved.

. PREVIOUS OCCURRENCES:

LER NUMBER	TITLE
6-1-85-097	Inadvertent Safety Injection During Surveillance Test
6-1-87-004	Safety Injection on Train A Due to Personnel Error

G. COMPONENT FAILURE DATA:

a) MANUFACTURER NOMENCLATURE MODEL NUMBER MFG PART NUMBER

Not Applicable

b) RESULTS OF NPRDS SEARCH:

Not Applicable

RP25 SSPS BLOWN GROUND RETURN FUSE

TYPE: GENERIC, RB

A) TRAIN A
B) TRAIN B

CAUSE: BLOWN FUSE

REF: SYSTEM DESCRIPTION
20E-1-4030 EF SERIES

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A LOSS OF THE GROUND RETURN PATH IN THE LOGIC PORTION OF THE SSPS CAUSING A LOSS OF ALL TRIP INPUTS, PERMISSIVE INPUTS, AND MEMORY INPUTS (MCB INPUT BLOCKS). THIS RESULTS IN A GENERAL WARNING, AND RENDERS THE AFFECTED SSPS TRAIN INOPERABLE. A GENERAL WARNING ON BOTH TRAINS RESULTS IN A RX TRIP. ONE OF THE MOST APPARENT BLOCKS LOST IS THE SR NI BLOCK WHICH IS RESPONSIBLE FOR DEENERGIZING SR HIGH VOLTS TO THE DETECTOR. INSERTING THIS MALFUNCTION IN DIFFERENT POWER MODES WILL HAVE VARYING EFFECTS DUE TO THE BLOCKS PRESENT OR REQUIRED IN THAT MODE OF OPERATION.

THE EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY USING BwOP RP-6 TO REPLACE FUSES, PREVENTING A RX TRIP/SAFETY INJECTION. THIS PROCEDURE ENSURES THAT AN INADVERTENT TRIP IS PREVENTED BY PROPER ALIGNMENT OF THE INPUT ERROR INHIBIT SWITCHES IN THE AFFECTED LOGIC CABINET (VIA RF) TO THE INHIBIT POSITION. THE FOLLOWING IS A LIST OF BLOCK SWITCHES ON 1PM05J THAT MUST BE BLOCKED.

SR MAN BLOCK A/B TRAIN A/B
IR MAN BLOCK A/B TRAIN A/B
PR MAN BLOCK A/B TRAIN A/B
PZR PRESS SI RESET/BLOCK TRAIN A/B
STM LINE SI RESET/BLOCK TRAIN A/B

EVENTS: NONE

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

RX01	STEAM PRESSURE DETECTOR FAILURE
RX02	UNSTABLE S/G LEVEL CONTROLLER
RX03	STEAM FLOW DETECTOR FAILURE
RX04	FW FLOW TRANSMITTER FAILURE
RX05	STEAM LINE PRESS DETECTOR (PT-507) FAILURE
RX06	NARROW RANGE S/G LEVEL FAILURE
RX07	WIDE RANGE S/G LEVEL FAILURE
RX08	STEAM DUMP COOLDOWN VALVES CONTROL FAILURE
RX09	STEAM FLOW DETECTOR OSCILLATION - TIME
RX10	FIRST STAGE PRESS TRANSMITTER FAILURE
RX11	STEAM FLOW DETECTOR OSCILLATION - MAGNITUDE
RX12	TREF FAILURE
RX13	PZR LEVEL CHANNEL FAILURE
RX14	FW PUMP MASTER SPEED CONTROLLER FAILURE
RX15	PZR PRESS MASTER CONTROLLER FAILURE
RX16	PZR LEVEL MASTER CONTROLLER FAILURE
RX17	ROD CONTROL SYSTEM FAILURE
RX18	FAULTY PRIMARY RTD (NARROW RANGE) (Tc & Th)
RX19	LOSS OF LOAD INTERLOCK C-7 FAILS
RX20	CONDENSER AVAILABLE INTERLOCK C-9 FAILS
RX21	PZR PRESS CHANNEL FAILURE (455 & 456)
RX22	PZR PRESS CHANNEL FAILURE (457 & 458)
RX23	OVERPOWER DELTA T SETPOINT FAILURE
RX24	OVERTEMPERATURE DELTA T SETPOINT FAILURE
RX25	RCS PRESS TRANSMITTER FAILURE (403 & 405)
RX26	RCS PRESS TRANSMITTER FAILURE (406 & 407)
RX27	RCS PRESS TRANSMITTER FAILURE (408 & 409)
RX28	RCS LOOP FLOW TRANSMITTER FAILURE
RX29	FW REG VLV CONTROLLER FAILURE
RX30	FW BYP VLV CONTROLLER FAILURE

RX01 STEAM PRESSURE DETECTOR FAILURE

TYPE: GENERIC, RV 0-1300 PSIG

A)	1PT514	G)	1PT534
B)	1PT515	H)	1PT535
C)	1PT516	I)	1PT536
D)	1PT524	J)	1PT544
E)	1PT525	K)	1PT545
F)	1PT526	L)	1PT546

CAUSE: DETECTOR FAILURE

REF: M-2035 SHEET 1
20E-1-4029 EF03

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE AFFECTED DETECTOR TO FAIL AT A VALUE DEPENDENT UPON MALFUNCTION SEVERITY. THE ASSOCIATED STM FLOW AND STM PRESSURE INDICATORS ON 1PM04J INDICATE BASED ON THE SELECTED SEVERITY. IF THE SEVERITY IS GREATER THAN THE INITIAL VALUE, THE S/G WATER LEVEL CONTROL SYSTEM RESPONDS TO MAINTAIN FEED FLOW EQUAL TO INDICATED STM FLOW IF THE AFFECTED CHANNEL IS CONTROLLING.

IF THE SEVERITY IS LESS THAN THE INITIAL VALUE, THE S/G WATER LEVEL CONTROL SYSTEM RESPONDS TO MAINTAIN PROPER STEAM FLOW TO FEED FLOW VALUES, AND AT EVEN LOWER SEVERITIES THE FOLLOWING WILL OCCUR: ANNUNCIATORS 15-A1/B1/C1/D1 "S/G 1A/1B/1C/1D LOW PRESS STEAMLINE ISOL ALERT" ACTUATES, AND 15-E1 "MS PRESS RATE STM LINE ISOL ALERT" ACTUATES ON A 100 PSI DROP WITHIN 50 SECONDS. WHEN MORE THAN ONE OF THE ABOVE MALFUNCTIONS ARE SELECTED, A SAFETY INJECTION AND STEAM LINE ISOLATION OCCURS.

THE EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY TAKING MANUAL CONTROL OF THE FEED REGULATING VALVE TO MAINTAIN PROPER S/G LEVELS.

MALFUNCTION REMOVAL RESTORES THE AFFECTED DETECTORS TO NORMAL.

EVENTS: NONE.

RX02 UNSTABLE S/G LEVEL CONTROLLER

TYPE: GENERIC, RV 0-15 MIN

A)	1A FRV	1RC01BA
B)	1B FRV	1RC01BB
C)	1C FRV	1RC01BC
D)	1D FRV	1RC01BD

CAUSE: FAULTY CONTROLLER OUTPUT CARD - NCB1 (PROGRAM VARIES 15%)

REF: SYSTEM DESCRIPTION
4031-FW16-19

PLT STA: REACTOR OPERATING AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED S/G LEVEL CONTROLLER TO OSCILLATE. THE RANGE OF THE PROGRAM OSCILLATION IS 15% LEVEL DEVIATION OVER A SELECTABLE TIME CYCLE. AS THE PROGRAM LEVEL EXCEEDS THE ACTUAL LEVEL, THE ASSOCIATED FEED REG VALVE WILL OPEN TO RECOVER THE APPARENT LOW STEAM GENERATOR LEVEL. WHEN THE PROGRAM LEVEL IS LESS THAN ACTUAL STEAM GENERATOR LEVEL, THE STEAM GENERATOR WATER LEVEL CONTROL SYSTEM WILL DECREASE FEED FLOW TO RESTORE LEVEL. IF THE INDICATED LEVEL BECOMES GREATER THAN ±5% OF THE PROGRAM LEVEL ,THEN ANNUNCIATOR 15-A9/B9/C9/D9 "S/G 1A/1B/1C/1D LEVEL DEVIATION HIGH LOW" ACTUATES.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY OPERATING THE ASSOCIATED FEED REG VALVE IN MANUAL.

MALFUNCTION REMOVAL RESTORES THE FAULTY CONTROLLER OUTPUT CARD TO NORMAL.

EVENTS: NONE.

RX03 STEAM FLOW DETEC ꝛ FAILURE

TYPE: GENERIC, RV 0-4.8 MLB/HR

A)	1FT512	E)	1FT532
B)	1FT513	F)	1FT533
C)	1FT522	G)	1FT542
D)	1FT523	H)	1FT543

CAUSE: DETECTOR FAILURE

REF: M-2035 SHEET 3

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE AFFECTED STEAM FLOW DETECTOR TO FAIL TO A VALUE DEPENDENT UPON THE MALFUNCTION SEVERITY. STEAM FLOW METERS FOR EACH DETECTOR ARE LOCATED ON 1PM04J. SELECTING A SEVERITY HIGHER THAN THE INITIAL VALUE CAUSES THE FEED REG VALVE TO OPEN TO COMPENSATE. ANNUNCIATOR 15-A4/B4/C4/D4 "S/G 1A/1B/1C/1D FLOW MISMATCH FW FLOW LOW" ACTUATES. ASSOCIATED S/G ALARMS ACTUATE WHEN LEVEL DEVIATION IS GREATER THAN +5%, AND IF S/G LEVEL EXCEEDS THE HI-2 SETPOINT, A TURBINE TRIP/REACTOR TRIP WILL OCCUR.

SELECTING A SEVERITY LOWER THAN THE INITIAL VALUE CAUSES THE FEED REG VALVE TO CLOSE TO COMPENSATE, ANNUNCIATOR 15-A3/B3/C3/D3 "S/G 1A/1B/1C/1D FLOW MISMATCH STM FLOW LOW" ACTUATES. ASSOCIATED S/G ALARMS ACTUATE WHEN LEVEL DEVIATION IS GREATER THAN -5%, AND IF S/G LEVEL EXCEEDS THE LO-2 SETPOINT, THE REACTOR WILL TRIP.

EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY TAKING MANUAL CONTROL OF THE FEED REG VALVE.

MALFUNCTION REMOVAL RESTORES THE AFFECTED DETECTOR TO NORMAL.

EVENTS: 1) DVR 20-01-88-128

DEVIATION INVESTIGATION REPORT

RX03

TITLE Loss of Steam Flow Indication/Control on 1C Steam Generator Due to Personnel Error	PAGE 1 OF 0 2

EVENT DATE MONTH	DAY	YEAR	DIR NUMBER STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE MONTH	DAY	YEAR	OPERATING MODE	POWER LEVEL
0 5	2 3	8 8	2 0	0 1	8 8	1 2 8	0 0	0 7	0 3	8 8	1	0 7 5

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER AREA CODE	
Joe Doyle, Technical Staff Engineer Ext. 2660	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 1 ; Event Date: May 23, 1988 ; Event Time: 0836
MODE: 1 - Power Operation ; Rx Power: 75% ; RCS [AB] Temperature/Pressure: 578 Degrees F/2235 psig

B. DESCRIPTION OF EVENT:

On May 23, 1988 at 0836 with the Plant in Power Operation a loss of 1C Steam Generator Steam Flow [SB] indication occurred on 1FI-532 which was the controlling channel for steam flow/feed flow mismatch. Under the direction of the Station Control Room Engineer (SCRE) the Nuclear Station Operator (NSO) took the necessary actions using 1BwOA INST-2. While stabilizing the plant in manual for feedwater flow the Steam Flow indicator 1FI-533 was lost approximately 1 minute after the loss of indication on 1FI-532. With the loss of the second indicator all indication for steam flow on 1C Steam Generator was lost. During the loss of indication the Instrument Maintenance Department had been installing a Digital Oscilloscope in the Instrument racks for a retest of Startup Test BwSU FW-31, Calibration of Steam and Feedwater Flow for Steam Generator 1C.

The connection for the oscilloscope was removed about 1 minute after the loss of indication on 1FI-533 occurred. Upon removal of the connector steam flow indication for 1C Steam Generator returned. Stable plant conditions were obtained a few minutes afterward. No other operator actions were required.

C. CAUSE OF EVENT:

The root cause of the event was determined to be the improper installation of the Digital Oscilloscope. During the installation of the scope for FW-31 the scope was installed such that the leads from the steam flow transmitters output circuits were connected to the grounded inputs of the scope. This connection grounded the channel causing a false indication of zero on the affected channels. The NSO stopped all work at this time and the oscilloscope was disconnected from the instruments.

TEXT

D. SAFETY ANALYSIS:

There was no effect on plant or public safety. The event was of short duration and stable conditions were obtained very quickly. Under worst case conditions with the plant at 100% power, with no operator intervention, the Steam Generator Water Level Control System (SGWLC) would have drastically reduced feed flow to the 1C Steam Generator, ultimately resulting in Lo-Lo level condition. At this point the Auxiliary Feedwater Pumps would have started and a Reactor Trip would have occurred as per design with no effect on plant or public safety.

E. CORRECTIVE ACTIONS:

The oscilloscope was removed from the instrument racks and the leads changed from the grounded inputs to the differential inputs on the scope. A training class was held to acquaint all Instrument Maintenance personnel of this problem and its solution. No further action is required.

F. PREVIOUS OCCURRENCES:

NONE

G. COMPONENT FAILURE DATA:

NONE

RX04 FW FLOW TRANSMITTER FAILURE

TYPE: GENERIC, RV 0-4.8 MLB/HR

A)	1FT510	E)	1FT530
B)	1FT511	F)	1FT531
C)	1FT520	G)	1FT540
D)	1FT521	H)	1FT541

CAUSE: FAULTY TRANSMITTER

REF: M-2036 SHEET 2

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE AFFECTED FEEDWATER FLOW DETECTOR TO FAIL TO A VALUE DEPENDENT UPON THE MALFUNCTION SEVERITY. ALL FEED FLOW METERS FOR EACH DETECTOR ARE LOCATED ON 1PM04J. SELECTING A SEVERITY HIGHER THAN THE INITIAL VALUE CAUSES THE FEED REG VALVE TO CLOSE TO COMPENSATE, ANNUNCIATOR 15-A3/B3/C3/D3 "S/G 1A/1B/1C/1D FLOW MISMATCH STM FLOW LOW" ACTUATES. ASSOCIATED S/G ALARMS ACTUATE WHEN LEVEL DEVIATION IS GREATER THAN -5%, AND IF S/G LEVEL EXCEEDS THE LO-2 SETPOINT, THE REACTOR WILL TRIP.

SELECTING A SEVERITY LOWER THAN THE INITIAL VALUE CAUSES THE FEED REG VALVE TO OPEN TO COMPENSATE, ANNUNCIATOR 15-A4/B4/C4/D4 "S/G 1A/1B/1C/1D FLOW MISMATCH FW FLOW LOW" ACTUATES. ASSOCIATED S/G ALARMS ACTUATE WHEN LEVEL DEVIATION IS GREATER THAN +5%, AND IF S/G LEVEL EXCEEDS THE HI-2 SETPOINT, A TURBINE TRIP/REACTOR TRIP WILL OCCUR.

EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY TAKING MANUAL CONTROL OF THE FEED REG VALVE.

MALFUNCTION REMOVAL RESTORES THE AFFECTED DETECTOR TO NORMAL.

EVENTS: NONE.

RX05 STEAM LINE PRESS DETECTOR (PT-507) FAILURE

TYPE: DISCRETE, RV 0-1500 PSIG

CAUSE: DETECTOR FAILURE

REF: M-2035 SHEET 5
S/G WATER LEVEL CONTROL SYSTEM DESCRIPTION

PLT STA: REACTOR OPERATING AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE STEAM LINE PRESSURE DETECTOR FOR 1PT-507 TO FAIL. THE VALUE OF THE TRANSMITTER OUTPUT WILL BE DETERMINED BY THE SELECTED SEVERITY. THIS TRANSMITTER SUPPLIES SIGNALS TO THE STEAM DUMP CIRCUIT, MAIN FEED PUMP SPEED CONTROL SYSTEM AND PROVIDES INDICATION ON 1PI-507 (1PM04J) AND 1PI-MS021 (1PM02J). IF THE SELECTED SEVERITY IS GREATER THAN THE INITIAL VALUE, THE OPERATING MAIN FEED PUMP(S) SPEED WILL INCREASE TO RAISE THE FW TO STEAM PRESS ΔP DUE TO THE APPARENTLY INCREASED STEAM PRESSURE. FEEDWATER REG VALVES THROTTLE CLOSED TO CONTROL LEVEL AND MAY BECOME UNSTABLE.

IF THE SELECTED SEVERITY IS LOWER THAN THE INITIAL VALUE, THE OPERATING MAIN FEED PUMP(S) SPEED WILL DECREASE TO DECREASE THE DIFFERENCE BETWEEN FEED PRESSURE AND THE APPARENTLY DECREASED STEAM PRESSURE. THE FEEDWATER REG VALVES WILL OPEN TO MAINTAIN LEVEL. STEAM GENERATOR LEVEL MAY DECREASE, AND ANNUNCIATOR 15-A9/B9/C9/D9 "S/G 1A/B/C/D LEVEL DEVIATION HIGH LOW" ACTUATES. IF A LO-2 S/G LEVEL IS REACHED IN THE ASSOCIATED STEAM GENERATOR, A REACTOR TRIP OCCURS. IN THE STEAM PRESS MODE, THE STEAM DUMPS ATTEMPT TO MAINTAIN THE PROPER STEAM PRESS, CAUSING AN EXCESSIVE HEATUP OR COOLDOWN OF THE PRIMARY.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY TAKING MANUAL CONTROL OF THE MASTER FW PP SPEED CONTROLLER, 1SK-509A, TO CONTROL FEED PUMP SPEED.

MALFUNCTION REMOVAL RESTORES THE FAILED DETECTOR TO NORMAL.

EVENTS: 1) DVR 06-01-85-256

Commonwealth Edison

DEVIATION REPORT

D.R NO. 6 - 1 - 85 - 256
STA UNIT YEAR NO.

RX05

PART 1 TITLE OF DEVIATION

PT507 Failure

OCCURRED 8-12-85 DATE 07:10 TIME

SYSTEM AFFECTED: MS

PLANT STATUS AT TIME OF EVENT: MODE 1, POWER(%) 95

B-21345 WORK REQUEST NO.

TESTING ☐ YES ☒ NO

DESCRIPTION OF EVENT

Main Steam PT507 drifted low during normal operations affecting FW pump speed control and causing level deviations in 1A & 1D S/G's. Operators placed FW pp Mstr. Speed controller in manual and restored S/G levels. PT507 eventually drifted 100# lower than operating pressure over 1 hr.

POTENTIALLY SIGNIFICANT EVENT PER NSD DIRECTIVE A-07 ☐ YES ☒ NO

10CFR50.72 NRC RED PHONE NOTIFICATION MADE ☐ 1 HOUR ☐ 4 HOUR ______ TIME ☒ NO

S. L. Swan — RESPONSIBLE SUPERVISOR — 8-12-85 DATE

PART 2 OPERATING ENGINEER'S COMMENT

Recurring problem. Appears to be associated with noise spike from CD/CB Pp start.

Tech Staff, IM's to investigate

☒ NON REPORTABLE EVENT

☐ 30 DAY REPORTABLE/10CFR..........

☐ 5 DAY REPORT PER 10CFR21

☐ ANNUAL/SPECIAL REPORT REQUIRED

A.I.R. #.............

L.E.R. #.............

NOTIFICATION ______ REGION III ______ DATE ______ TIME

______ NSD ______ DATE ______ TIME

☐ CECO CORPORATE NOTIFICATION MADE IF ABOVE NOTIFICATION IS PER 10CFR21

TELECOPY D. Galle — CECO CORPORATE OFFICER — 8-15-85 DATE — 1115 TIME

PRELIMINARY REPORT COMPLETED AND REVIEWED: T. J. Tulon — OPERATING ENGINEER — 8-15-85 DATE

INVESTIGATION REPORT & RESOLUTION ACCEPTED BY STATION REVIEW: [signature] 9/24/85 [signature] 9-23-85 [signature] 9/25/85

RESOLUTION APPROVED AND AUTHORIZED FOR DISTRIBUTION: [signature] — STATION SUPERINTENDENT — 9/27/85 DATE

C.E.CO. 86-5176 (FORM 15-52-1) 1-84

RX06 NARROW RANGE S/G LEVEL FAILURE

TYPE: GENERIC, RV 0-100% LEVEL

A)	1A S/G LT-517	I)	1C S/G LT-537
B)	1A S/G LT-518	J)	1C S/G LT-538
C)	1A S/G LT-519	K)	1C S/G LT-539
D)	1A S/G LT-556	L)	1C S/G LT-558
E)	1B S/G LT-527	M)	1D S/G LT-547
F)	1B S/G LT-528	N)	1D S/G LT-548
G)	1B S/G LT-529	O)	1D S/G LT-549
H)	1B S/G LT-557	P)	1D S/G LT-559

CAUSE: FAULTY TRANSMITTER

REF: SGWLC SYSTEM DESCRIPTION
20E-1-4031 FW SERIES
C&ID M-2036

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED STEAM GENERATOR NARROW RANGE LEVEL TRANSMITTER WILL FAIL TO THE VALUE SELECTED.

THE VARIOUS LEVEL TRANSMITTERS RESPONSE WILL BE BASED ON WHETHER THE CHANNEL IS SELECTED AS A CONTROLLING CHANNEL. ALL CHANNELS WILL VARY THEIR SIGNALS TO THE SELECTED SEVERITY AND THIS WILL BE INDICATED ON THEIR ASSOCIATED LEVEL INDICATORS. PLANT ANNUNCIATORS AND STATUS LIGHTS WILL RESPOND ACCORDINGLY.

VARYING A SELECTED CONTROL CHANNEL 1LT-519/556, 529/557, 539/558, 549/559 WILL CAUSE ITS ASSOCIATED FEED REG VALVE TO MODULATE TO RECOVER FROM THE APPARENT LEVEL CHANGE. IF THE DEVIATION BETWEEN THE MODULATED FEED FLOW AND THE STEAM FLOW TO THE ASSOCIATED STEAM GENERATOR IS 750K LB/HR, ANNUNCIATOR 15-A3,B3,C3,D3 (A4,B4,C4,D4) "SG FLOW MISMATCH STM (FW) FLOW LOW" WILL BE ACTUATED. HIGH/LOW LEVEL AND LEVEL DEVIATION ANNUNCIATORS RESPOND ACCORDINGLY. THE FAILURE OF THE TRANSMITTER WILL LEAD TO A PLANT TRIP IF NO OPERATOR ACTION IS TAKEN. DEPENDENT ON SEVERITY LEVEL, S/G LEVELS WILL EITHER CONTINUE TO INCREASE OR DECREASE RESPECTIVELY. THE RESULT IS A TURBINE TRIP AT HI-2 S/G LEVEL OR A REACTOR TRIP AT LO-2 S/G LEVEL.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY CONTROLLING THE ASSOCIATED STEAM GENERATOR LEVEL MANUALLY.

MALFUNCTION REMOVAL WILL RESTORE THE FAILED LEVEL TRANSMITTER TO NORMAL.

EVENTS: 1) LER 20-02-88-026

LICENSEE EVENT REPORT (LER)

RXO6 Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Braidwood Unit 2	0 5 0 0 0 4 5 7	1 of 0 4

Title (4)
Turbine Trip caused by Hi-Hi S/G Level

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0 9	2 3	8 8	8 8	--- 0 2 6 ---	0 0	1 0	2 4	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 3 8

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR
(Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	X 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract Below and in Text)
__ 20.405(a)(1)(iii)	__ 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name		TELEPHONE NUMBER	
		AREA CODE	
Paul Stanczak, Technical Staff Engineer	Ext. 2486	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
__ Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

In response to an anomaly identified on one of the 2A steam generator level transmitters, it was determined that simultaneous data collection was required. The method was reviewed by the appropriate personnel involved with the task and approval was given to start. At 1347 when the second channel of the second phase of data collection was attached, a turbine trip and reactor trip occurred. Cause of the event was personnel error in that the review process failed to identify the normally grounded connection on the test equipment. Contributing factors were: 1) equipment had been used in the past however, not the configuration, 2) proper communications between the personnel performing the task were not maintained, and 3) connection on the test equipment was not properly labelled. The test equipment was removed and the plant stabilized. Event has been formally reviewed with the personnel involved and will be formally reviewed with station personnel by upper station management stressing the lessons learned from this event in the areas of communications, work practices and concurrent activities. Other action items relative to the startup testing program will be reviewed to ensure appropriate dispositioning. The test equipment will be properly labelled. No previous occurrences.

(2586i)/23

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

Form Rev 2.0

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year		Sequential Number		Revision Number	Page (3)
Braidwood Unit 2	0 5 0 0 0 4 5 7	8 8	-	0 2 6	-	0 0	0 2 OF 0 4

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. Plant Conditions Prior to Event:

Unit: Braidwood 2; Event Date: September 23, 1988; Event Time: 1347;

Mode: 1 - Power Operation; Rx Power: 38%

RCS/[AB] Temperature/Pressure: NOT / NOP

B. Description of Event:

Unit 2 FW70D Post Test Review Board (TRB) established that an anomaly exists on the four (4) level transmitters for the 2A steam generator. One of the 4 transmitters appeared to be reading differently that the other 3. TRB comment #3 stated that 2LT556 is suspect. Action Item Record (AIR) number 88-262 was written to track closure of the anomaly.

Nuclear Work Request (NWR) A25065 was written on August 23, 1988 to troubleshoot and repair 2LT556. It was determined that the loop and transmitter were in calibration. A Unit 2 containment entry was made to determine if a significant difference in level tap elevation existed. The review of the data indicated that the tap elevation difference was insignificant. Based on these results, it was decided that data should be obtained on all 4 level transmitters simultaneously using a Nicolet 4 channel oscilloscope.

This proposed method of data gathering was reviewed by the work group Instrument Maintenance (IM) technicians, the Shift Test Director (STD), Station Control Room Engineer (SCRE), Nuclear Station Operator (NSO) and Shift Engineer (SE). The review focused on the available method to obtain accurate data which was to monitor all 4 loops simultaneously. Data had been successfully gathered from these test points in the past with data logging equipment. Approval was given to obtain the data from TP2 and then TP1 on all four loops on the 2A steam generator.

Head phone communications were established between the Control Room and the Auxiliary Electrical Equipment Room. The Nicolet was successfully attached to TP2 and the data was obtained. Based on the completion of this phase of data collection and the belief that the same actions would be repeated for TP1, head phone communications were terminated.

Concurrent with the data gathering effort, the Unit 2 NSO's attention was shifted to secondary plant evolutions.

The IMs and STD started to attach the test equipment to TP1. When the first channel of test equipment was attached to TP1, the loop failed high due to normal internal ground on the test equipment. The Unit 2 NSO recognized that the bistables had tripped and prior to communicating with the Auxiliary Electrical Equipment Room, the second channel was attached. This resulted in the completion of the 2/4 coincidence logic for P-14, turbine trip and reactor trip at 1347 on September 29, 1988.

The appropriate NRC notification via the ENS phone system was made at 1550 on September 29, 1988 pursuant to 10CFR50.72(b)(2)(ii).

This event is being reported pursuant to 10CFR50.73(a)(2)(iv) - any event or condition that resulted in manual or automatic actuation of any engineered safety feature, including the reactor protection system.

C. Cause of Event:

The cause of the event was personnel error by the maintenance personnel. The review process failed to identify the normally grounded connection on the test equipment.

Contributing to this event were:

1. The equipment had been used in the past; however, the configuration required for this check had not been previously used.

2. Proper communications between the personnel performing the check in the Auxiliary Electrical Equipment Room and the Control Room were not maintained throughout the duration of the testing.

3. The connection on the equipment was not properly labelled to indicate that the connection was grounded.

D. Safety Analysis:

There was no affect on plant or public safety as all systems operated as designed. Under worst case conditions of operating at full power and a loss of steam flow for steam generator, the results would have been the same as in this event. The other 3 steam generator steam flow instruments were operable throughout this event.

E. Corrective Actions:

The immediate corrective actions were to remove the test equipment and stabilize the plant.

This event has been formally reviewed with the personnel involved by upper station management.

This event will be formally reviewed with station personnel by upper station management stressing the lessons learned from this event in the areas of communications, work practices and concurrent activities. This will be tracked to completion by action item 457-200-88-16401.

A review of other action items relative to the startup testing program will be performed to ensure appropriate dispositioning. This will be tracked to completion by action item 457-200-88-16402.

The equipment connection will be properly labelled indicating that it is grounded. This will be tracked to completion by action item #57-200-88-16403.

F. Previous Occurrences:

There have been previous occurrences of a personnel error resulting in a reactor trip. The corrective actions were implemented addressing both root and contributing causes. Previous corrective actions are not applicable to this event.

G. Component Failure Data:

This event was not caused by component failure nor did any components fail as a result of this event.

RX07 WIDE RANGE S/G LEVEL FAILURE

TYPE: GENERIC, RV 0-100%

A)	1A S/G	LT-501
B)	1B S/G	LT-502
C)	1C S/G	LT-503
D)	1D S/G	LT-504

CAUSE: FAULTY TRANSMITTER

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED STEAM GENERATOR WIDE RANGE LEVEL TRANSMITTER WILL FAIL. THE VALUE OF THE FAILED WIDE RANGE LEVEL WILL BE DETERMINED BY THE SELECTED SEVERITY. THE WIDE RANGE LEVEL TRANSMITTER HAS NO CONTROL FUNCTION, BUT SENDS SIGNALS TO A STEAM GENERATOR LEVEL METER AND RECORDER INDICATION AND ALARM. ANNUNCIATOR 15-A7,B7,C7,D7 "S/G 1A/1B/1C/1D LEVEL HIGH" IS ACTUATED WHEN THE AFFECTED WIDE RANGE SIGNAL INCREASES.

MALFUNCTION REMOVAL WILL RESTORE THE FAILED LEVEL TRANSMITTER TO NORMAL.

EVENTS: NONE.

RX08 STEAM DUMP COOLDOWN VALVES CONTROL FAILURE

TYPE: DISCRETE, RV 0-1500 PSIG

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**************************************************
*                    NOTE                        *
*                                                *
*     0 PSIG=0% ON THE CONTROLLER AND            *
*     1500 PSIG=100% ON THE CONTROLLER.          *
*     (15 PSIG PER % GAIN)                       *
*                                                *
**************************************************
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CAUSE: CONTROLLER 1PK507 AUTO OUTPUT FAILURE

REF: M-2035 SHEET 5

PLT STA: HOT STANDBY

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE STEAM DUMPS TO ATTEMPT TO CONTROL THE COOLDOWN AT THE SELECTED SEVERITY ONLY IN THE "STEAM PRESSURE" MODE. THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY TAKING MANUAL CONTROL OF THE STEAM DUMP SYSTEM TO REDUCE REACTOR COOLANT SYSTEM TEMPERATURE .

MALFUNCTION REMOVAL RESTORES THE CONTROLLER TO NORMAL.

EVENTS: NONE.

RX09 STEAM FLOW DETECTOR OSCILLATION - TIME

TYPE: GENERIC, RV 0-900 SECONDS (DEFAULT MAGNITUDE 5%)

A)	1A S/G FT-512	E)	1C S/G FT-532
B)	1A S/G FT-513	F)	1C S/G FT-533
C)	1B S/G FT-522	G)	1D S/G FT-542
D)	1B S/G FT-523	H)	1D S/G FT-543

CAUSE: FAULTY TRANSMITTER

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED STEAM FLOW TRANSMITTER WILL BEGIN TO OSCILLATE. THE OSCILLATION WILL BE INDICATED ON THE ASSOCIATED INDICATORS AND RECORDERS. THE DEFAULT MAGNITUDE OF THE OSCILLATION IS 5%. IF MALFUNCTION RX11 (MAGNITUDE) IS ACTIVE, THEN THE OSCILLATION WILL BE AT THE MAGNITUDE SELECTED. IF THE SELECTED TRANSMITTER IS A CONTROLLING CHANNEL INPUT FOR THE ASSOCIATED STEAM GENERATOR LEVEL CONTROL SYSTEM, THEN THE FEED REG VALVE WILL BEGIN TO OSCILLATE IN RESPONSE TO THE VARIED STEAM FLOW. IF THE DEVIATION BETWEEN THE MODULATED FEED FLOW AND THE STEAM FLOW TO THE ASSOCIATED STEAM GENERATOR IS 750K LBS/HR, ANNUNCIATOR 15-A3,B3,C3,D3 (A4,B4,C4,D4) "SG FLOW MISMATCH STM (FW) FLOW LOW" WILL BE ACTUATED. PLANT ANNUNCIATORS AND STATUS LIGHTS WILL RESPOND ACCORDINGLY.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY SELECTING THE UNAFFECTED STEAM FLOW TRANSMITTER AS AN INPUT TO THE STEAM GENERATOR WATER LEVEL CONTROL SYSTEM OR BY TAKING MANUAL CONTROL OF THE ASSOCIATED FW REG VALVE.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED STEAM FLOW TRANSMITTER TO NORMAL.

EVENTS: NONE.

RX10 FIRST STAGE PRESS TRANSMITTER FAILURE

TYPE: GENERIC, RV 0-850 PSIG

A) 1PT-505
B) 1PT-506

CAUSE: TRANSMITTER FAILURE

REF: M-2035 SHEET 8

PLT STA: REACTOR OPERATING AT POWER

EFFECTS: INSERTING MALFUNCTION RX10A CAUSES THE FIRST STAGE PRESSURE TRANSMITTER (1PT-505) TO FAIL. THE VALUE OF THE TRANSMITTER OUTPUT WILL BE DETERMINED BY THE SELECTED SEVERITY. THE OUTPUT FROM THIS TRANSMITTER IS USED IN THE ROD CONTROL SYSTEM Tref PROGRAMMER, INPUT TO THE P-13 INTERLOCK CIRCUIT OF THE REACTOR PROTECTION SYSTEM, C-20 FOR ATWS MITIGATION SYSTEM, AND FOR PRESSURE INDICATION ON 1PM05J. IF THE SELECTED SEVERITY IS GREATER THAN THE INITIAL VALUE WITH RODS IN AUTO, THE CONTROL RODS WILL AUTOMATICALLY WITHDRAW TO RAISE T_{ave} TO THE ELEVATED Tref. THE INCREASED TRANSMITTER SIGNAL WILL ALSO BE SENT TO THE INDICATIONS AND TO THE P-13 CIRCUITRY WHICH MAY SATISFY THE LOGIC FOR ENABLING P-7. IF THE SELECTED SEVERITY IS LESS THAN THE INITIAL VALUE, THE CONTROL RODS WILL AUTOMATICALLY BE INSERTED TO REDUCE T_{ave} TO THE LOWERED Tref. THE DECREASED TRANSMITTER SIGNAL WILL ALSO BE SENT TO THE INDICATION AND TO THE P-13 CIRCUITRY WHICH MAY SATISFY THE LOGIC FOR P-7.

IF FIRST STAGE PRESSURE TRANSMITTER (1PT-506) IS FAILED, THE VALUE OF THE TRANSMITTER OUTPUT WILL BE DETERMINED BY THE SELECTED SEVERITY. THE OUTPUT FROM THIS TRANSMITTER IS USED IN THE LOAD REJECTION CONTROL INTERLOCK C-7 CIRCUIT, INPUT TO THE P-13 INTERLOCK CIRCUIT OF THE REACTOR PROTECTION SYSTEM, C-20 FOR ATWS MITIGATION SYSTEM, AND FOR PRESSURE INDICATION ON 1PM02J (1PI-MS004) & 1PM05J. IF THE SELECTED SEVERITY IS GREATER THAN THE INITIAL VALUE, THE INCREASED TRANSMITTER SIGNAL WILL ALSO BE SENT TO THE INDICATION AND TO THE P-13 CIRCUITRY WHICH MAY SATISFY THE LOGIC FOR ENABLING P-7. IF THE SELECTED SEVERITY IS LESS THAN THE INITIAL VALUE, THE DECREASED TRANSMITTER SIGNAL WILL BE SENT TO THE INDICATIONS AND TO THE P-13 CIRCUITRY WHICH MAY SATISFY THE LOGIC FOR P-7. IF THE FAILURE OF 1PT-506 CAUSES A 10% DECREASE IN <120 SEC, THEN C-7 WILL BE ACTUATED AND THE STM DUMPS WILL BE ARMED, IF IN THE T_{ave} MODE.

MALFUNCTION REMOVAL RESTORES THE FAILED TRANSMITTER TO NORMAL.

EVENTS: 1) DVR 06-01-88-062

DEVIATION INVESTIGATION REPORT

RXIO

TITLE	PAGE
TURBINE IMPULSE PRESSURE TRANSMITTER PT-505 FAILED HIGH	1 OF 0 2

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	POWER LEVEL
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR		
0 4	1 8	8 8	0 6	0 1	8 8	— 0 6 2 —	0 0				1	8 7

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
Roger Flahive, Technical Staff Supervisor Ext. 2243	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	S B	P I	I 2 0 4	Y					

SUPPLEMENTAL REPORT EXPECTED		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 4/18/88 / 0816

Unit 1 MODE 1 - Power Operation Rx Power 87 RCS [AB] Temperature/Pressure Normal Operation

Unit 2 MODE 1 - Power Operation Rx Power 94 RCS [AB] Temperature/Pressure Normal Operation

B. DESCRIPTION OF EVENT:

On April 18, 1988 at 0809 hours, while operating at 87% Reactor Power, Unit 1 experienced difficulties with the Turbine Impulse Pressure Transmitter 1PT-505. The problem was discovered during the routine performance of Train "B" Solid State Protection System Bi Monthly Surveillance 1BOS 3.1.1-21 when the Pressure Transmitter was noted to be failed high.

The failure of 1PT-505 caused the T_{avg} - T_{ref} Deviation Alarm to annunciate, and Limiting Condition for Operations Action Requirement (LCOAR) 1BOS 3.1-1a, Action Number 8 was entered at 0816 hours. The licensed personnel in the Control Room immediately implemented Byron Abnormal Procedure 1BOA INST-2 as required. All affected bistables were verified to be in their required positions by 0825 hours per BOA INST-2, "Operation With A Failed Instrument Channel", and Station Nuclear Work Request (NWR) number B55120 was generated to the Instrument Maintenance Department to investigate and repair the failed pressure transmitter.

Maintenance activities were completed on April 19, 1988, and post maintenance testing was performed by Operating Department personnel to document continued component operability. LCOAR 1BOS 3.1-1a, Action Number 8, was exited on April 19, 1988, at 2110 hours, with the affected bistables being reset at that time.

There were no systems, subsystems or components considered inoperable at the beginning of this event which would have contributed to or exacerbated this event. No manual or automatic safety system actuations occurred during the event. All operator actions taken throughout the event were correct. Plant conditions remained stable throughout the event.

DEVIATION INVESTIGATION REPORT TEXT CONTINUATION

TITLE	DIR NUMBER							PAGE
	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
TURBINE IMPULSE PRESSURE TRANSMITTER PT-505 FAILED HIGH	0 6	0 1	8 8	—	0 6 2	—	0 0	2 OF 0 2

TEXT

C. CAUSE OF EVENT:

The root cause of this pressure transmitter failure is indeterminate. However, during maintenance troubleshooting activities, evidence of moisture was noted in the terminal junction box for PT-505. The moisture had no readily apparent source, as all associated conduits leading into it were noted as being dry.

D. SAFETY ANALYSIS:

There were no safety consequences resulting from this event which would have adversely impacted plant or public safety. At the time of this occurrence, Operating Department personnel entered the appropriate Operating Abnormal Procedure, BOA INST-2, and aligned the system for correct and safe interim operation.

E. CORRECTIVE ACTIONS:

The moisture noted in the terminal junction box for 1PT-505 was dried out and the pressure transmitter internals were also inspected for signs of moisture with none being evidenced.

Associated O-Rings for the pressure transmitter were replaced in kind and the calibration of the component was verified to be within tolerance and indicating properly. No further corrective action is planned at this time.

F. PREVIOUS OCCURRENCES:

There have been previous Deviation Reports written against failed pressure transmitter channels. However to date, all instances have had a definite mode of failure and none were determined to be of an indeterminate root cause.

DVR NUMBER	TITLE
06-01-85-353	1B S/G Level Loop 529 Failure Due to a Circuit Card Failure
06-01-86-002	1C Steam Generator Loop Failure
06-01-86-101	Failure of Steam Pressure Instrument 1PI-524A
06-01-86-207	1D Steam Generator Pressure Channel 544 Failure Low
06-01-88-031	S/G 1B Pressure Channel 525 Failure Due to Lead/Lag and Multiplier/Divider Card Failures

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
ITT Barton	Pressure Transmitter	753-2706	N/A

b) RESULTS OF NPRDS SEARCH:
An NPRDS search was not conducted, since the data obtained would not lend itself useful in this case where the root cause was determined to be indeterminate.

c) RESULTS OF NWR SEARCH:
A review of "TJM History" file for the Unit 1 and 2 Turbine Impulse Pressure Transmitter 1/2 PT-505 indicated only calibration histories of the transmitters

(0006R/0001R)

RX11 STEAM FLOW OSCILLATION - MAGNITUDE

TYPE: GENERIC, RV 0-1.5 MLB/HR (DEFAULT TIME 30 SEC)

A)	1A S/G FT-512	E)	1C S/G FT-532
B)	1A S/G FT-513	F)	1C S/G FT-533
C)	1B S/G FT-522	G)	1D S/G FT-542
D)	1B S/G FT-523	H)	1D S/G FT-543

CAUSE: FAULTY TRANSMITTER

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED STEAM FLOW TRANSMITTER WILL BEGIN TO OSCILLATE. THE DEFAULT TIME OF THE OSCILLATION IS 30 SECONDS. THE MAGNITUDE OF THE OSCILLATION IS DETERMINED BY THE SELECTED SEVERITY. IF MALFUNCTION RX09 (TIME) IS ACTIVE, THEN THE OSCILLATION WILL BE AT THE TIME SELECTED. IF THE SELECTED TRANSMITTER IS A CONTROLLING CHANNEL INPUT FOR THE ASSOCIATED STEAM GENERATOR LEVEL CONTROL SYSTEM, THEN THE FEED REG VALVE WILL BEGIN TO OSCILLATE IN RESPONSE TO THE VARIED STEAM FLOW. IF THE DEVIATION BETWEEN THE MODULATED FEED FLOW AND THE STEAM FLOW TO THE ASSOCIATED STEAM GENERATOR IS 750K LBS/HR, ANNUNCIATOR 15-A3,B3,C3,D3 (A4,B4,C4,D4) "SG FLOW MISMATCH STM (FW) FLOW LOW" WILL BE ACTUATED.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY SELECTING THE UNAFFECTED STEAM FLOW TRANSMITTER AS AN INPUT TO THE STEAM GENERATOR WATER LEVEL CONTROL SYSTEM, OR BY TAKING MANUAL CONTROL OF THE ASSOCIATED FW REG VALVE. PLANT ANNUNCIATORS AND STATUS LIGHTS WILL RESPOND ACCORDINGLY.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED STEAM FLOW TRANSMITTER TO NORMAL.

EVENTS: NONE.

RX12 Tref FAILURE

TYPE: DISCRETE, RV 557°F - 584°F

CAUSE: Tref PROGRAMMER OUTPUT FAILS

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION AT 557°F CAUSES THE Tref PROGRAMMER TO FAIL TO A 557°F PROGRAM TEMPERATURE. WITH THE ROD CONTROL SYSTEM IN AUTOMATIC, THE CONTROL RODS WILL BEGIN TO INSERT TO LOWER T_{ave} TO THE LOWER Tref SETPOINT. IF THE DEVIATION BETWEEN AUCTIONEERED HIGH T_{ave} AND Tref REACHES 3°F, ANNUNCIATOR 14-D1 "T_{ave} CONT DEV HIGH" ACTUATES. IF THE SEVERITY IS HIGHER THAN T_{ave}, WITH THE ROD CONTROL SYSTEM IN AUTOMATIC, THE CONTROL RODS WILL BEGIN TO WITHDRAWAL TO RAISE T_{ave} TO THE HIGHER T_{ref} SETPOINT. THE FAILED VALUE WILL BE INDICATED ON 1TR-412, AND WILL AFFECT INDICATION OF 1TI-412A.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY CONTROLLING THE ROD CONTROL SYSTEM IN MANUAL TO MAINTAIN REACTOR COOLANT SYSTEM TEMPERATURE.

MALFUNCTION REMOVAL RESTORES THE FAILED Tref PROGRAMMER TO NORMAL.

EVENTS: NONE.

RX13 PZR LEVEL CHANNEL FAILURE

TYPE: GENERIC, RV 0-100%

A) 1LT459
B) 1LT460
C) 1LT461

CAUSE: TRANSMITTER FAILURE

REF: M-2060 SHEET 6,7,8

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED PRESSURIZER LEVEL TRANSMITTER TO FAIL. THE VALUE OF THE TRANSMITTER OUTPUT WILL BE DETERMINED BY THE SELECTED SEVERITY AND WILL BE INDICATED ON THE ASSOCIATED METER AND DISPLAYED ON RECORDER 1LR-459, IF SELECTED. 1LT-460 & 1LT-461 ARE NOT NORMALLY CONTROLLING LEVEL CHANNELS. IF THE SELECTED SEVERITY IS LESS THAN THE INITIAL VALUE, CHARGING WILL MODULATE TO RAISE THE APPARENT LOW LEVEL, WHICH WILL ALSO RESULT IN AN INCREASE IN PRESSURIZER PRESSURE. ANNUNCIATOR 12-B4 "PZR LEVEL CONT DEV LOW" ACTUATES WHEN THE SELECTED SEVERITY IS 5% LESS THAN PROGRAM LEVEL.

IF THE SELECTED SEVERITY IS < 17% THEN THE FOLLOWING OCCURS: PZR HTRS TRIP OFF, LETDOWN ORIFICES ISOLATE, CV459 AND/OR CV460 CLOSE, ANNUNCIATORS 12-B3 "PZR LEVEL HIGH" & 12-A3 "PZR LEVEL HIGH RX TRIP STPT ALERT" ACTUATE.

IF THE SELECTED SEVERITY IS GREATER THAN THE INITIAL VALUE, CHARGING WILL MODULATE TO LOWER THE APPARENT HIGH LEVEL, WHICH WILL ALSO RESULT IN A DECREASE IN PRESSURIZER LEVEL. ANNUNCIATOR 12-C3 "PZR LEVEL CONT DEV HIGH HTRS ON" WILL ACTUATE.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY TRANSFERRING PRESSURIZER LEVEL CONTROL, OR CHARGING PUMP SPEED OR FLOW CONTROL TO MANUAL.

MALFUNCTION REMOVAL RESTORES THE FAILED LEVEL TRANSMITTER TO NORMAL.

EVENTS: NONE.

RX14 FW PUMP MASTER SPEED CONTROLLER FAILURE

TYPE: DISCRETE, RV 0-100% CONTROLLER OUTPUT

CAUSE: AUTO CONTROLLER FAILURE (1SK-509A)

REF: SGWLC SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE OUTPUT OF THE FW PUMP MASTER SPEED CONTROLLER TO VARY ACCORDING TO THE SELECTED SEVERITY LEVEL. DECREASING THE CONTROLLER OUTPUT WILL CAUSE THE CONTROL SYSTEM TO ATTEMPT TO DECREASE THE ΔP BETWEEN FW HEADER PRESSURE AND STEAM HEADER PRESSURE AND INCREASING THE CONTROLLER OUTPUT WILL DO THE OPPOSITE.

IF THE SELECTED SEVERITY IS LESS THAN THE ACTUAL CONTROLLER OUTPUT, THE GOVERNOR VALVES OF THE MAIN FEEDWATER PUMP TURBINES WILL RECEIVE A SIGNAL TO CLOSE TO LOWER THE APPARENT HIGH FEED TO STEAM ΔP. THIS WILL CAUSE ALL STEAM GENERATORS' LEVELS TO DECREASE AND ALL FW REG VALVES (1FW510,520,530,540) TO MODULATE OPEN. IF THE MOTOR DRIVEN MAIN FEEDWATER PUMP IS OPERATING, THE DISCHARGE VALVE, 1FW016, WILL THROTTLE DOWN TO DECREASE THE ΔP. PLANT ANNUNCIATORS WILL RESPOND ACCORDINGLY.

IF THE SELECTED SEVERITY IS GREATER THAN THE ACTUAL CONTROLLER OUTPUT, THE GOVERNOR VALVES OF THE MAIN FEEDWATER PUMP TURBINES WILL RECEIVE A SIGNAL TO OPEN TO RAISE THE APPARENT LOW FEED TO STEAM ΔP. THIS WILL CAUSE ALL STEAM GENERATORS' LEVELS TO INCREASE AND ALL FW REG VALVES (1FW510,520,530,540) TO MODULATE CLOSED. IF THE MOTOR DRIVEN MAIN FEEDWATER PUMP IS OPERATING, THE DISCHARGE VALVE, 1FW016, WILL MODULATE OPEN TO INCREASE THE ΔP.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY CONTROLLING THE MAIN FEEDWATER PUMP SPEED OR FLOW, FOR 1A FW PUMP, IN MANUAL.

MALFUNCTION REMOVAL WILL RESTORE THE FAILED CONTROLLER TO NORMAL.

EVENTS: NONE.

RX15 PZR PRESS MASTER CONTROLLER FAILURE

TYPE: DISCRETE, RV 2195-2355 PSIG

CAUSE: CONTROLLER 1PK455A AUTO OUTPUT FAILURE

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

```
**************************************************
*                    NOTE                        *
*                                                *
*     2195 PSIG=0% ON THE CONTROLLER AND         *
*     2355 PSIG=100% ON THE CONTROLLER.          *
*     (1.6 PSIG PER % GAIN)                      *
*                                                *
**************************************************
```

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE PRESSURIZER PRESSURE MASTER CONTROLLER (1PK-455A) TO FAIL. THE OUTPUT OF THE CONTROLLER WILL BE DETERMINED BY THE SELECTED SEVERITY. IF THE SELECTED SEVERITY IS LESS THAN THE INITIAL VALUE, PRESSURIZER PRESSURE WILL RISE. DEPENDING ON THE SELECTED SEVERITY, ANNUNCIATOR 12-C1 "PZR PRESS CONT DEV LOW HTRS ON" MAY ACTUATE. ANNUNCIATOR 12-C2 "PZR PRESS HIGH" ACTUATES.

IF THE SELECTED SEVERITY IS GREATER THAN THE INITIAL VALUE, PRESSURIZER PRESSURE WILL DROP. DEPENDING ON THE SELECTED SEVERITY, ANNUNCIATOR 12-C1 "PZR PRESS CONT DEV LOW HTRS ON" MAY ACTUATE. ANNUNCIATOR 12-A1 "PZR PRESS LOW RX TRIP STPT ALERT" IS ACTUATED AS IS ANNUNCIATOR 11-C3 "PZR PRESS LOW RX TRIP". ALSO ANNUNCIATOR 12-D2 "PZR PRESS CONT DEV HIGH" ACTUATES.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY TAKING MANUAL CONTROL OF THE FAILED MASTER PRESSURE CONTROLLER TO MAINTAIN PRESSURIZER PRESSURE.

MALFUNCTION REMOVAL RESTORES THE FAULTY AUTO OUTPUT TO NORMAL.

EVENTS: NONE.

RX16 PZR LEVEL MASTER CONTROLLER FAILURE

TYPE: DISCRETE, RV 0-100% OUTPUT

CAUSE: CONTROLLER 1LK459 AUTO OUTPUT FAILURE

REF: M-2060 SHEET 6

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE PRESSURIZER LEVEL MASTER CONTROLLER (1LK-459) TO FAIL. THE OUTPUT OF THE CONTROLLER WILL BE DETERMINED BY THE SELECTED SEVERITY. IF THE SELECTED SEVERITY IS GREATER THAN THE INITIAL VALUE, CHARGING FLOW WILL INCREASE AND PRESSURIZER LEVEL WILL RISE.

IF THE SELECTED SEVERITY IS LESS THAN THE INITIAL VALUE, PRESSURIZER LEVEL WILL DROP DUE TO A REDUCTION IN CHARGING FLOW. ON FAILURES < 1.5%, ANNUN 9-D3 "CHG LINE FLOW HIGH LOW" WILL ACTUATE.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY TAKING MANUAL CONTROL OF THE FAILED MASTER LEVEL CONTROLLER TO MAINTAIN PRESSURIZER LEVEL.

MALFUNCTION REMOVAL RESTORES THE FAULTY PZR LEVEL MASTER CONTROLLER TO NORMAL.

EVENTS: NONE.

RX17 ROD CONTROL SYSTEM FAILURE

TYPE: DISCRETE, RV -10 TO +10°F error

CAUSE: FALSE Terror SIGNAL FROM SUMMING AMP (1UY-0412)

REF: ROD CONTROL SYSTEM DESCRIPTION
20E-1-4031 MS16
20E-1-4031 RD03

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE ROD CONTROL SYSTEM SUMMING AMP WILL GENERATE AN OUTPUT DEPENDENT ON THE SELECTED SEVERITY REGARDLESS OF ACTUAL INPUT SIGNALS. IF THE SELECTED SEVERITY IS GREATER THAN 1.5°F, WITH THE ROD CONTROL SYSTEM IN AUTOMATIC, THE CONTROL RODS WILL BEGIN TO WITHDRAW TO RAISE ACTUAL T_{ave}. IF THE DEVIATION BETWEEN AUCTIONEERED HIGH T_{ave} AND Tref REACHES 3°F, ANNUNCIATOR 14-D1 "T_{ave} CONT DEV HIGH" WILL BE ACTUATED.

IF THE SELECTED SEVERITY IS LESS THAN -1.5°F, WITH THE ROD CONTROL SYSTEM IN AUTOMATIC, THE CONTROL RODS WILL BEGIN TO INSERT TO LOWER ACTUAL T_{ave}. IF THE DEVIATION BETWEEN AUCTIONEERED HIGH T_{ave} AND Tref REACHES 3°F, ANNUNCIATOR 14-E1 "T_{ave} CONT DEV LOW" WILL BE ACTUATED. IF THE CONTROL RODS ARE INSERTED SUFFICIENTLY TO LOWER T_{ave} TO 550°F, BYPASS PERMISSIVE ANNUNCIATOR FOR P12 WILL ACTUATE. ROD INSERTION LIMIT ALARMS AND DELTA FLUX ALARMS WILL RESPOND ACCORDINGLY.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY CONTROLLING THE ROD CONTROL SYSTEM IN MANUAL TO MAINTAIN REACTOR COOLANT SYSTEM TEMPERATURE .

MALFUNCTION REMOVAL WILL RESTORE THE FAILED SUMMING AMP TO NORMAL.

EVENTS: NONE.

RX18 FAULTY PRIMARY RTD (NARROW RANGE) (Tc & Th)

TYPE: GENERIC, RV (see below)

A)	1TE411B	510-630°F (Loop 1 Tc)		I)	Loop 1-Th2	530-650°F
B)	1TE421B	510-630°F (Loop 2 Tc)		J)	Loop 2-Th2	530-650°F
C)	1TE431B	510-630°F (Loop 3 Tc)		K)	Loop 3-Th2	530-650°F
D)	1TE441B	510-630°F (Loop 4 Tc)		L)	Loop 4-Th2	530-650°F
E)	Loop 1-Th1	530-650°F		M)	Loop 1-Th3	530-650°F
F)	Loop 2-Th1	530-650°F		N)	Loop 2-Th3	530-650°F
G)	Loop 3-Th1	530-650°F		O)	Loop 3-Th3	530-650°F
H)	Loop 4-Th1	530-650°F		P)	Loop 4-Th3	530-650°F

CAUSE: RTD FAILURE

REF: M-2060 SHEET 2

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED NARROW RANGE RTD TO FAIL. THE VALUE OF THE RTD OUTPUT WILL BE DETERMINED BY THE SELECTED SEVERITY. THE 3 HOT LEG RTD'S ARE AVERAGED PRIOR TO BEING UTILIZED. THE RTD'S PROVIDE AN INPUT TO THE ΔT (WHICH ARE FED TO THE OTΔT AND OPΔT CIRCUITRY) AND T_{ave} CALCULATIONS AND ARE DISPLAYED ON THEIR RESPECTIVE METERS AND RECORDERS ON 1PM05J.

THE EFFECTS ON T_{ave} AND ΔT FROM A FAILED RTD ARE AS FOLLOWS:

RTD	FAILS	ΔT	Tave
HOT LEG	HIGH	INCREASES	INCREASES
HOT LEG	LOW	DECREASES	DECREASES
COLD LEG	HIGH	DECREASES	INCREASES
COLD LEG	LOW	INCREASES	DECREASES

IF THE AFFECTED LOOP ΔT DECREASES TO 4°F LESS THAN THE AUCTIONEERED HIGH ΔT, ANNUNCIATOR 14-A4/B4/C4/D4 "LOOP 1A/B/C/D ΔT DEV LOW" ACTUATES. ANNUNCIATOR 14-A3/B3/C3/D3 "LOOP 1A /B/C/D TAVE DEV LOW" ACTUATES AT 3°F DEVIATION LESS THAN THE AUCTIONEERED HIGH T_{ave}. THE AUCTIONEERED HIGH T_{ave} IS USED FOR AUTOMATIC ROD CONTROL, STEAM DUMP CONTROL AND PRESSURIZER LEVEL CONTROL CIRCUITS. AUCTIONEERED HIGH ΔT IS USED AS AN INPUT IN DETERMINING CONTROL ROD INSERTION LIMITS.

IF THE AFFECTED LOOP ΔT INCREASES TO 4°F GREATER THAN THE AUCTIONEERED HIGH ΔT, ANNUNCIATOR 14-A5/B5/C5/D5 "LOOP 1A/B/C/D ΔT DEV HIGH" ACTUATES. ANNUNCIATOR 14-A3/B3/C3/D3 "LOOP 1A /B/C/D TAVE DEV HIGH" ACTUATES AT 3°F DEVIATION GREATER THAN THE AUCTIONEERED HIGH T_{ave}. IF THE MALFUNCTION SEVERITY IS VARIED, THE OTΔT AND OPΔT OUTPUT FOR THAT LOOP MAY EXCEED THEIR SETPOINTS.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY DEFEATING THE AFFECTED LOOP T_{ave} AND ΔT UTILIZING THEIR DEFEAT SWITCHES.

MALFUNCTION REMOVAL RESTORES THE FAILED RTD CIRCUIT TO NORMAL.

EVENTS: 1) LER 06-01-90-002
2) DVR 20-01-87-398

LICENSEE EVENT REPORT (LER)

RX18

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Byron, Unit 1	0 5 0 0 0 4 5 4	1 of 0 3

Title (4) Reactor Trip on OTΔT Due to the Failure of a Resistance Temperature Detector Card Coincident with One Channel in Test.

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0 3	0 1	9 0	9 0	0 0 2	0 0	0 3	2 7	9 0	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 2

POWER LEVEL (10): 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

___ 20.402(b)	___ 20.405(c)	X 50.73(a)(2)(iv)	___ 73.71(b)
___ 20.405(a)(1)(i)	___ 50.36(c)(1)	___ 50.73(a)(2)(v)	___ 73.71(c)
___ 20.405(a)(1)(ii)	___ 50.36(c)(2)	___ 50.73(a)(2)(vii)	___ Other (Specify in Abstract below and in Text)
___ 20.405(a)(1)(iii)	___ 50.73(a)(2)(i)	___ 50.73(a)(2)(viii)(A)	
___ 20.405(a)(1)(iv)	___ 50.73(a)(2)(ii)	___ 50.73(a)(2)(viii)(B)	
___ 20.405(a)(1)(v)	___ 50.73(a)(2)(iii)	___ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER
A. Javorik, Assistant Technical Staff Supervisor Ext. 2106	AREA CODE 8 1 5 2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	A B	A M P	W 1 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

___ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15): Month | Day | Year

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

On March 1, 1990, at 0939, while in Mode 2, a Unit 1 reactor trip occurred when the 2 out of 4 coincidence was satisfied on Over Temperature Delta Temperature (OTΔT). The failure of the loop 1B Reactor Coolant Resistance Temperature Detector amplifier card concurrent with the previously tripped loop 1A bistables, which was necessary to accommodate low power physics testing, satisfied the trip logic.

The failed card was replaced, but the root cause of the failure is indeterminate.

All systems responded as required, and the Unit was stabilized in Mode 3. This event is reportable per 10CRF 50.73 (a)(2)(iv) for an event that resulted in automatic actuation of an Engineered Safety Feature including the Reactor Protection System.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 03/01/90 / 0939

Unit 1 MODE 2 - Startup Rx Power 0% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On March 1, 1990, at 0939, with Unit 1 in Mode 2 at 10^{-7} amperes on Intermediate Range, a reactor trip occurred due to a Reactor Coolant [AB] Resistance Temperature Detector (RTD) amplifier card failure concurrent with previously tripped bistables required to accommodate low power physics testing. Per BVS XPT-4 "Initial Criticality After Refueling and Nuclear Heating Levels", the reactivity computer was connected to a Nuclear Instrumentation Power Range channel. Nuclear Instrumentation (NR) [IG] channel N41 was taken Out-of-Service, which resulted in loop 1A bistables TB411C (Over Temperature Delta Temperature Reactor Trip), and TB411D (Over Temperature Delta Temperature Turbine Runback Rod Withdrawal Stop) being tripped. Limiting Condition for Operation Action Requirement (LCOAR) 1BOS 3.1-1a was in effect for the tripped channel. The loop 1B amplifier card (1TY-0421A) failure, coincident with loop 1A channel in test satisfied the logic on an Over Temperature Delta Temperature (OTΔT) reactor trip.

Following the reactor trip signal, all rods fully inserted, and the plant was stabilized in Mode 3. All plant systems responded as required. Operator actions were correct and aided in the immediate recovery and stabilization of the Unit.

This event is reportable pursuant to 10CFR 50.73 (a)(2)(iv) for an event that resulted in an automatic actuation of an Engineered Safety Feature including the Reactor Protection System.

C. CAUSE OF EVENT:

The intermediate cause of the reactor trip was the loop 1B hot leg RTD amplifier card temporarily failing high. The spurious failure resulted in loop 1B OTΔT Reactor Trip bistable actuation. The failure mechanism is indeterminate because the card returned to a normal operating condition. The root cause of the failure is believed to be heat related based on previous failure history, although troubleshooting did not identify a failure mechanism. A contributing factor was the previously tripped loop 1A OTΔT bistables while Power Range Channel N41 was Out-of-Service to accommodate low power physics testing.

D. SAFETY ANALYSIS:

Plant and public safety were not jeopardized by this event. All safety systems operated as designed in response to the trip signal. At no time did actual conditions exist in the reactor core which would cause the OTΔT setpoint to be exceeded. The safety systems would have shutdown the plant under more severe circumstances such as an actual OTΔT transient at full power operation.

E. CORRECTIVE ACTIONS:

The failed RTD amplifier card was replaced and the replacement calibrated under NWR B74473. A replacement for this model of card is not being pursued at this time.

As a conservative measure, the Reactivity Computer was reconnected to Power Range channel N42 (loop 1B) to preclude another spurious trip.

No further corrective actions are planned.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

F. PREVIOUS OCCURRENCES:

A similar event is documented in LER 87-001 (Docket 455), "Reactor Trip due to 2 out of 4 Logic on Over Temperature channel - 1 channel Out for Required Reactivity Computer, 1 Channel out due to failed Reactor Coolant Resistance Temperature Detector with unknown Cause".

A Total Job Management (TJM) search of the station's maintenance history identified the same card had been replaced on 1/19/90 oue to an out of tolerance found during calibration.

In addition Problem Analysis Data Sheet (PADS) 89-109 identified the high incidence of failure of these cards.

G. COMPONENT FAILURE DATA:

a.

MANUFACTURER	NOMENCLATURE	MFG PART NUMBER
Westinghouse	RTD Amplifier Card	2837A15G01

b. Results of NPRDS search:

A Nuclear Plant Reliability Data System (NPRDS) search of this model card identified 124 events of failures or out of tolerances. Of these failures, 4 similar events occurred on Unit 1 and 8 similar events occurred on Unit 2. The Component Failure Analysis Report (CFAR) showed a comparable failure rate on Unit 1 as compared to the industry failure rate, and a higher failure rate on Unit 2 when compared to the industry. A review of the NPRDS search did not reveal a common mode failure mechanism.

DEVIATION INVESTIGATION REPORT

RX 18

TITLE	Inoperable Reactor Coolant System Narrow Range Resistance Temperature Detector Channel Due to Failed RTD Amplifier	PAGE 1 OF 0 3

EVENT DATE MONTH	DAY	YEAR	DIR NUMBER STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE MONTH	DAY	YEAR	OPERATING MODE	POWER LEVEL
1 1	1 9	8 7	2 0	0 1	8 7	3 9 8	0 0	1 2	2 1	8 7	1	0 2 8

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER AREA CODE	
Jenny D. Tolar, Technical Staff Engineer Ext. 2454	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	A B	C N V *	W 1 2 0	YES					

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 1 ; Event Date: 11-19-87 ; Event Time: 2100
MODE: 1 - Power Operation ; Rx Power: 28% ; RCS [AB] Temperature/Pressure: 566°F/2235 psig

B. DESCRIPTION OF EVENT:

On November 19, 1987, Unit 1 was in Operating Mode 1 at 28 percent reactor power. During Shift 3, the Unit 1 Nuclear Station Operator (NSO) noted unexpected control rod motion. A check of Control Room instrumentation revealed that 1TI-0411A (Loop 1A Delta T) and 1TI-0412 (Loop 1A Tave) at panel 1PM05J were failing in the high direction. The instruments were erratic, drifting up slowly off-scale high and then returning to normal indication. Both of these indicators receive input from 1TE-0411A, Loop 1A Hot Leg RTD (Resistance Temperature Detector) and 1TE-0411B, Loop 1A Cold Leg RTD. The Delta T portion of the loop provides input to Overpower Delta T and Overtemperature Delta T (Solid State Reactor Protection System), while the Tave portion of the loop provides input to Rod Control.

At 2100 hours, Abnormal Operating Procedure 1BwOA INST-2 was entered for a failed Reactor Coolant System (RCS) Narrow Range RTD Channel, and Limiting Condition for Operation Action Requirement (LCOAR) 3.1-1a was entered for inoperable Overpower and Overtemperature Delta T Channels. At 2125 hours, the bistables for 1TE-0411A were tripped per procedure, and 1BwOA INST-2 was exited. Nuclear Work Request (NWR) A17974 was written to verify the calibration of 1TE-0411A.

The outputs of 1TY-411A and B were monitored for 24 hours using a strip chart recorder. No spiking was observed. Using Procedure BwIS 3.1.1-001, the Instrument Maintenance Department (IMD) began to troubleshoot the loop. The hot leg portion of the Delta T loop was considered the more probable cause of instrument drift and was investigated first. Card 1TY-411A was replaced and no subsequent spiking was observed. The LCOAR was exited at 1605 hours on November 21, 1987. Shortly thereafter spiking was again observed, LCOAR 3.1-1a was again entered at 1839 hours on November 21, 1987 and Procedure BwIS 3.1.1-001 performed. Card 1TY-411A was recalibrated and card 1TY-411B replaced. No further spiking was observed. The LCOAR was exited at 0405 hours on November 22, 1987.

Stable plant conditions were maintained throughout this event.

DEVIATION INVESTIGATION REPORT TEXT CONTINUATION

TITLE	DIR NUMBER						PAGE		
	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER		
Inoperable Reactor Coolant System Narrow Range Resistance Temperature Detector Channel Due to Failed RTD Amplifier	2 0	0 1	8 7	—	3 9 8	—	0 0	2	OF 0 3

TEXT

C. CAUSE OF EVENT:

This event was caused by a drifting resistance to voltage converter card 1TY-0411A in the Westinghouse 7300 Process Computer racks and a faulty voltage converter card 1TY-0411B, also in the 7300 racks. Instrument Maintenance Department troubleshooting found the output voltage from the 1TY-0411A card to be higher than the other loops. The output was monitored and a step change in voltage was identified. This fluctuating output from the card contributed to the erratic indication on 1TI-0411A and 1TI-0412. Since Tave is affected by 1TY-0411A also, the fluctuating output from 1TY-0411A resulted in unexpected control rod motion as well.

Subsequent troubleshooting under blanket Nuclear Work Request A00219 revealed that, following replacement of card 1TY-411A, recalibration was required due to excessively low output voltage. Additionally, the cold leg card 1TY-411B was replaced because of faulty and abnormally low output voltages.

D. SAFETY ANALYSIS:

No safety consequences existed from the occurrence of this event. Redundant RCS loop temperature instrumentation allowed for isolation of the failed channel and continued stable plant operation. The reactor trip functions were not affected by this event due to the required 2 out of 4 coincidence trip logic.

The most limiting condition for this event would occur if an RCS Narrow Range RTD Channel were to fail while another Channel was in test. In this case, only 2 out of 4 RCS Narrow Range RTD Channels would be operable. With the bistables tripped for the loop in test, the required reactor trip coincidence logic would be 1 out of 3. Thus, a failed channel in the high or low direction would make up the 2 out of 4 logic and trip the reactor. In this event, the plant would operate to shut down as designed, and no adverse safety consequences would result.

E. CORRECTIVE ACTIONS:

Operating personnel took the appropriate Technical Specification (Tech Spec) corrective actions. IMD troubleshooting isolated the problem, and the failed card 1TY-0411A was replaced. At 1605 hours on November 21, 1987, LCOAR 3.1-1a was exited. More extensive troubleshooting followed (due to reoccurring problems) and at 1839 hours on November 21, LCOAR 3.1-1a was again entered. Card 1TY-0411A was recalibrated and card 1TY-0411B (cold leg) was replaced. At 0405 hours on November 22, 1987, LCOAR 3.1-1a was exited. No further corrective action is required.

F. PREVIOUS OCCURRENCES:

DVR/LER NUMBER	TITLE
NONE	

DEVIATION INVESTIGATION REPORT TEXT CONTINUATION

TITLE	DIR NUMBER							PAGE
Inoperable Reactor Coolant System Narrow Range Resistance Temperature Detector Channel Due to Failed RTD Amplifier	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
	2 0	0 1	8 7	—	3 9 8	—	0 0	3 OF 0 3

TEXT

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Westinghouse	NRA G01 Card 7300 Computer	2837A15G01	NRA G01 1TY-0411A
Westinghouse	NRA G01 Card 7300 Computer	2837A15G01	NRA G01 1TY-0411B

RX19 LOSS OF LOAD INTERLOCK C-7 FAILS

TYPE: DISCRETE, RB

CAUSE: CARD FAILURE

REF: STEAM DUMP SYSTEM DESCRIPTION
20E-1-4030 MS09
20E-1-4031 MS13

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE C-7 INTERLOCK FAILS AS IS. IF INSERTED DURING NORMAL OPERATION, WHEN THE TURBINE EXPERIENCES A LOSS OF LOAD CONDITION (>10% DROP IN TURBINE IMPULSE PRESSURE). THIS MALFUNCTION WILL PREVENT ARMING THE STEAM DUMPS IF OPERATING IN THE T_{ave} MODE. THE BYPASS PERMISSIVE ANNUNCIATOR "LOSS OF TURB LOAD INTLK C7" WILL NOT ACTUATE AND THE STEAM DUMP ARMED STATUS LIGHT ON 1PM02J WILL NOT LIGHT. THUS ON AN ACTUAL LOSS OF LOAD, THE S/C PORV's WILL LIFT TO CONTROL REACTOR COOLANT SYSTEM TEMPERATURE (THE STEAM GENERATOR SAFETY VALVES MAY LIFT TO REDUCE RCS TEMPERATURE IF REQUIRED).

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS BY MANUALLY OPERATING THE STEAM DUMPS IN THE STEAM PRESSURE MODE.

IF THE MALFUNCTION IS INSERTED WHEN THE LOSS OF LOAD LOGIC IS SATISFIED, RETURNING TO NORMAL OPERATING CONDITIONS WILL NOT CLEAR THE C-7 INTERLOCK. THIS RESULTS IN THE C-7 INTERLOCK BEING PRESENT WHEN IT NORMALLY WOULD NOT BE. THE STEAM DUMPS WILL CONTINUE TO OPERATE

MALFUNCTION REMOVAL WILL RESTORE THE FAULTY C-7 INTERLOCK TO NORMAL.

EVENTS: NONE.

RX20 CONDENSER AVAILABLE INTERLOCK C-9 FAILS

TYPE: DISCRETE, RB

CAUSE: CONTROL CIRCUIT FAILURE

REF: STEAM DUMP SYSTEM DESCRIPTION
20E-1-4030 MS09
20E-1-4030 ES21

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE C-9 INTERLOCK FAILS AS IS. INSERTION OF THIS MALFUNCTION WHEN THE CONDITIONS FOR SATISFYING THE C-9 INTERLOCK ARE NOT MET, PREVENTS THE C-9 INTERLOCK FROM BEING SATISFIED AND ARMING THE STM DUMPS. THIS PREVENTS ARMING THE STEAM DUMPS IN ANY MODE OF OPERATION. THEREFORE, THE STEAM DUMP VALVES WILL NOT OPEN AS REQUIRED.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY UTILIZING THE S/G PORV'S TO CONTROL REACTOR COOLANT SYSTEM TEMPERATURE (THE STEAM GENERATOR SAFETY VALVES LIFT TO REDUCE RCS TEMPERATURE IF REQUIRED).

IF THE MALFUNCTION IS INSERTED DURING PLANT POWER OPERATION, THEN WHEN THE CONDITIONS FOR SATISFYING THE C- 9 INTERLOCK ARE NOT MET, THE C-9 PERMISSIVE WILL STILL SHOW THE CONDENSER AS AVAILABLE WHEN THE MAIN CONDENSER VACUUM IS <24 INCHES VACUUM AND THE CIRCULATING WATER PUMPS ARE NOT OPERATING. THIS RESULTS IN THE STEAM DUMP SYSTEM BEING OPERABLE WHEN IT NORMALLY WOULD NOT BE, RESULTING IN AN EXCESSIVE REACTOR COOLANT SYSTEM COOLDOWN AND POSSIBLE OVERPRESSURIZATION OF THE CONDENSER.

MALFUNCTION REMOVAL WILL RESTORE THE FAULTY C-9 PERMISSIVE CIRCUITRY TO NORMAL.

EVENTS: NONE.

RX21 PRESSURIZER PRESSURE CHANNEL FAILURE (455 & 456)

TYPE: GENERIC, RV 1700-2500 PSIG

A) 1PT455 B) 1PT456

CAUSE: DETECTOR FAILURE

REF: M-2060 SHEET 7,8
20E-1-4029 EF04

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING MALFUNCTION RX21A CAUSES 1PT-455 DETECTOR TO FAIL TO A VALUE DEPENDENT UPON THE SELECTED SEVERITY. 1PI-455A AND 1PR-455 ON 1PM05J RESPOND ACCURATELY TO THE SELECTED SEVERITY. IF THE SELECTED SEVERITY IS GREATER THAN THE INITIAL VALUE, THE MASTER PZR PRESSURE CONTROLLER WILL OPEN THE SPRAY VALVES, OPEN PORV-455A, TURN OFF ALL PZR HEATERS, AND GENERATE HIGH PRESSURE ALARMS. ACTUAL PZR PRESS DECREASES UNTIL THE REACTOR TRIP AND SAFETY INJECTION ARE ACTUATED (PORV WILL CLOSE WHEN PORV BLOCK PRESSURE IS MET AT 2185 PSIG).

IF THE SEVERITY IS LOWER THAN THE INITIAL VALUE, THE MASTER PZR PRESSURE CONTROLLER WILL TURN ON THE HEATERS, AND THE LOW PRESSURE ALARMS WILL BE ACTUATED. PZR SPRAY WILL NOT ACTUATE, AND THE ACTUAL PZR PRESSURE WILL INCREASE UNTIL THE PORV-456 SETPOINT IS REACHED (PORV-455A IS CONTROLLED BY THE AFFECTED PRESSURE TRANSMITTER). RCS PRESSURE WILL OSCILLATE DUE TO PORV LIFTING.

ON A PT-456 FAILURE LOW, PORV-456 WILL NOT AUTO OPEN ON HIGH RCS PRESSURE. ON A FAILURE HIGH, PORV-456 WILL OPEN UNTIL CHANNEL 457 IS < 2185 PSIG.

THE EFFECTS CAN BE MITIGATED BY PLACING THE MASTER PZR PRESSURE CONTROLLER OR THE SPRAY VALVE CONTROLLERS IN MANUAL TO MAINTAIN PZR PRESSURE.

MALFUNCTION REMOVAL RESTORES THE FAILED DETECTOR TO NORMAL.

EVENTS: 1) LER 20-02-88-018
2) DVR 06-02-88-122

DEVIATION INVESTIGATION REPORT (DIR) RX21

Form Rev 2.0

Facility Name	PAGE
Byron Nuclear Power Station	1 OF 0 3

Title

Pressurizer Pressure Channel Failure

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	1
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	034
12	31	88	06	02	88	— 122 —	01	01	09	89		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER AREA CODE	
R. A. Flahive, Tech Staff Superviosr Ext. 2243	815	234-5441

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	AB	FT	204	Y					

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 12-31-88 / 1428

Unit 2 MODE 1 - Power Operation Rx Power 34 RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On December 31, 1988, at 1428, the 2P-456 Pressurizer Pressure Channel failed low. This event was discovered by the unit operator (NSO- Licensed). The associated Pressurizer Pressure and OTΔT bistables were tripped per procedure 2BOA INST-2. Limiting Conditions for Operation Action Requirements (LCOAR's) 3.1-1a and 3.2-1a were entered. No safety systems were activated and NWR B63671 was initiated to troubleshoot and correct the problem.

At the time of failure the plant was stable, with no other systems inoperable that may have contributed to the failure. This failure caused no automatic safety system actuations and the plant subsequently remained in a stable condition. Operation actions neither increased nor decreased the severity of this event.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

C. CAUSE OF EVENT:

Investigation into the failure, under NWR B63671, traced the failure to card 2PQY-0456. This card is the loop power supply and its failure caused the loop to go to the low (trip) condition by design. The internal mode of failure within the power supply is indeterminable, and the entire supply card has been replaced.

There were no personnel errors contributing to this event, and no unusual plant or system characterstics at the time. This failure should be considered a random failure due to normal wear.

D. SAFETY ANALYSIS:

The failure of this power supply conservatively failed the entire loop to the low pressurizer pressure condition. This condition gave one (1) logic make-up to a 2/4 trip permissive. The subsequent operator action per 2BOA INST-2 of tripping the bistables, for both pressurizer pressure low and OTΔT, created a 1/3 trip permissive for the loop, which is conservative. Therefore, there were no adverse safety consequences due to this failure.

E. CORRECTIVE ACTIONS:

Since the failure is considered to be due to normal wear out, the corrective action was to replace the card. This was done per NWR B63761, and the loop was re-calibrated per normal instrument procedures.

No other corrective actions are required at this time.

F. RECURRING EVENTS SEARCH AND ANALYSIS:

a) EVENT SEARCH (DIR, LER)

81-006-01 Failed Power Supply

b) INDUSTRY SEARCH (OPEX's NPRDS)

There have been previously reported failures of this type of card at Byron. All of these failures are considered to be normal wear within the accepted industry average.

c) NWR

B63663 - card replacement.

d) ANALYSIS

No adverse trend.

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Westinghouse Electric Corporation	Loop Power Supply Isolated	NLPG02	2837A12G02

H. OTHER RELATED DOCUMENTS:

None

I. EFFECTIVENESS REVIEW:

None Scheduled

J. ADDITIONAL DATA:

a) Affected Technical Specification: 3.3.1, 3.3.2

b) Procedures: 2BOA INST-2

c) Cause Code: XIELEL1IM

d) Equipment Involved: Westinghouse Loop Power Supply Card

e) Other: None

LICENSEE EVENT REPORT (LER)

RX21

Facility Name (1)	Docket Number (2)	Page (3)
Braidwood, Unit 2	0 5 0 0 0 4 5 7	1 of 0 3

Title (4) Reactor Trip Due to Defective Circuit Card

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0 7	0 2	8 8	8 8	0 1 8	0 0	0 7	2 1	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 4 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

___	20.402(b)	___	20.405(c)	X	50.73(a)(2)(iv)	___	73.71(b)
___	20.405(a)(1)(i)	___	50.36(c)(1)	___	50.73(a)(2)(v)	___	73.71(c)
___	20.405(a)(1)(ii)	___	50.36(c)(2)	___	50.73(a)(2)(vii)	___	Other (Specify in Abstract below and in Text)
___	20.405(a)(1)(iii)	___	50.73(a)(2)(i)	___	50.73(a)(2)(viii)(A)		
___	20.405(a)(1)(iv)	___	50.73(a)(2)(ii)	___	50.73(a)(2)(viii)(B)		
___	20.405(a)(1)(v)	___	50.73(a)(2)(iii)	___	50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER (AREA CODE)
Joe Doyle, Technical Staff Engineer Ext. 2660	8 1 5 4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
B	J G	P I C *	W 1 2 3	N					

SUPPLEMENTAL REPORT EXPECTED (14)

___ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15): Month | Day | Year

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

At 2304 on July 2, 1988, a reactor trip occurred while conductirg startup test BwSU NR-75A, Axial Flux Difference Calibration. Per the test procedure, Pressurizer Pressure Channel 455C Bistable was placed in the tripped condition. During the test, Pressurizer Pressure (AB) Bistable 457C failed, completing the 2 out of 4 logic needed for a reactor trip. Post trip response was normal except that bus 243, a non safety related bus, did not auto transfer from the Unit Auxiliary transformer to the system auxiliary transformer. This was done manually by an Equipment Attendant. A feedwater isolation occurred because of the low average temperature and reactor trip. Both auxiliary feedwater pumps auto started because of a LO-LO Level in the Steam Generators (SJ). Conditions were stabilized by 2330. The root cause of the event was the failure of pressurizer pressure bistable 457C in conjunction with pressurizer pressure bistable 455C being in a tripped condition. The failure of the 457C bistable was caused by a failure of the comparator card. Corrective actions included replacing the failed card and to repairing the breaker. There have been no previous occurrences of a reactor trip as the result of a failed bistable card while conducting a startup test.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Braidwood, Unit 2	0 5 0 0 0 4 5 7	8 8	- 0 1 8 -	0 0	0 2 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 1 ; Event Date: July 2, 1988 ; Event Time: 2304
MODE: 1 - Power Operation ; Rx Power: 40% ; RCS [AB] Temperature/Pressure: NOT/NOP

B. DESCRIPTION OF EVENT:

Pressurizer Pressure Channel 455C Bistable had been placed in the tripped condition per the requirements of the startup that was in progress.

Prior to the reactor trip, the Instrument Maintenance Department (IMD) was conducting startup test BwSU NR-75A, Axial Flux Difference Calibration. Per the test prerequisites, pressurizer pressure channel 455C bistable was placed in the test condition. At 2304 pressurizer pressure (AB) bistable 457C failed, and with 2 out of 4 of the pressurizer pressure channels tripped, a reactor trip occurred. Post trip response was normal except that bus 243, a non safety related bus, did not auto transfer from the unit auxiliary transformer to the system auxiliary transformer. An Equipment Operator was sent to do this manually. A feedwater isolation occurred because of the low average temperature and reactor trip. Both Auxiliary Feedwater Pumps auto started because of a LO-LO level in the Steam Generators (SJ). Conditions were stabilized by 2330. No other actions were required.

Operator actions neither increased nor decreased the severity of the event.

The appropriate NRC notification via the ENS Phone System was made at 0043 on July 3, 1988 pursuant to 10CFR50.72(b)(2)(ii).

This event is being reported pursuant to 10CFR50.73(a)(2)(iv) - any event or condition that resulted in manual or automatic actuation of any engineered safety feature, including the reactor protection system.

C. CAUSE OF EVENT:

The critical factor leading to the Reactor Trip was the failure of pressurizer pressure bistable 457C in conjunction with pressurizer pressure bistable 453C being in a tripped condition in accordance with Start-up Test BwSU NR-75A. It takes 2 out of 4 pressurizer pressure channels being in a tripped condition to cause a Reactor Trip. The failure of the 457C bistable was caused by a failure of the comparator card.

D. SAFETY ANALYSIS:

There was no effect on plant or public safety. The plant reached stable conditions in 30 minutes. Under worst case conditions, of operating at 100% power, the results would have been the same.

E. CORRECTIVE ACTIONS:

Work Request A23995 was written and the failed card was replaced by the Instrument Maintenance Department. Work Request A23974 was written to repair the apparent breaker auto transfer problem. No problems were found with the breaker. It is believed that the breaker was not fully racked in and the contacts were not engaged, thus preventing an auto transfer. The breaker was returned to service.

E. CORRECTIVE ACTIONS: (cont'd)

The placement of Pressurizer Pressure Bistable 455C into test was required by Start-up Test BwSU MR-75A. The periodic AFD Surveillance does not require placing the 455C loop in test.

This event is considred to be an isolated event. No further corrective actions are required.

F. PREVIOUS OCCURRENCES:

There have been no previous occurrences where a reactor trip has occurred due to a failure of a pressurizer bistable while the axial flux distribution startup test was being performed.

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Westinghouse	NAL Card	2837A13G01	A42348

RX22 PRESSURIZER PRESSURE CHANNEL FAILURE

TYPE: GENERIC, RV 1700-2500 PSIG

A) 1PT-457 B) 1PT-458

CAUSE: DETECTOR FAILURE

REF: M-2060 SHEET 6
20E-1-4029 EF04

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING EITHER MALFUNCTION CAUSES THE AFFECTED DETECTOR TO FAIL AT A VALUE DEPENDENT UPON THE SELECTED SEVERITY. THE ASSOCIATED PRESSURE INDICATOR ON 1PM05J RESPONDS ACCURATELY TO THE MALFUNCTION SEVERITY. IF THE SELECTED SEVERITY IS BELOW THE PORV INTERLOCK SETPOINT FOR THE AFFECTED CHANNEL, THE FOLLOWING WILL OCCUR:

<u>1PT-457</u>
PORV-456 WILL NOT LIFT AT THE SETPOINT IN THE AUTOMATIC MODE DUE TO THE LACK OF THE INTERLOCK SIGNAL. ANNUN 12-C1 "PZR PRESS CONT DEV LOW HTRS ON" WILL BE ACTUATED.

<u>1PT-458</u>
PORV-455A WILL NOT LIFT AT THE SETPOINT IN THE AUTOMATIC MODE DUE TO THE LACK OF THE INTERLOCK SIGNAL. ANNUN 12-C1 "PZR PRESS CONT DEV LOW HTRS ON" WILL BE ACTUATED.

IF THE SELECTED SEVERITY IS ABOVE THE PORV INTERLOCK SETPOINT FOR THE AFFECTED CHANNEL, THE FOLLOWING WILL OCCUR:

<u>1PT-457</u>
PORV-456 AUTOMATIC CLOSE INTERLOCK WILL ALWAYS BE SATISFIED. ANNUN 12-A2 "PZR PRESS HIGH Rx TRIP SETPOINT ALERT" WILL BE ACTUATED.

<u>1PT-458</u>
PORV-455A AUTOMATIC CLOSE INTERLOCK WILL ALWAYS BE SATISFIED. ANNUN 12-A2 "PZR PRESS HIGH Rx TRIP SETPOINT ALERT" WILL BE ACTUATED.

MALFUNCTION REMOVAL RESTORES DETECTOR TO NORMAL.

EVENTS: NONE.

RX23 OVERPOWER DELTA T SETPOINT FAILURE

TYPE: GENERIC, RV 0-150%

A) CHANNEL A
B) CHANNEL B
C) CHANNEL C
D) CHANNEL D

CAUSE: CARD FAILURE

REF: 20E-1-4031 RC SERIES

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED OVERPOWER DELTA T SETPOINT TO FAIL TO SELECTED SEVERITY. THE VALUE OF THE SETPOINT IS INDICATED ON ITS ASSOCIATED METER AND 1TR-411, IF SELECTED (1PM05J). ANNUNCIATORS 10-A5 "OPΔT HIGH ROD STOP C-4" AND 14-A1 "OPΔT HIGH RX TRIP ALERT" ACTUATE WHEN THE VARIABLE SETPOINT IS EXCEEDED IN THE AFFECTED LOOP. USING TWO MALFUNCTIONS, IF THE OVERPOWER DELTA T VALUES OF TWO LOOPS EXCEED THE DECREASED SETPOINTS, ANNUNCIATOR 11-A4 "OPΔT RX TRIP" ACTUATES AND THE REACTOR WILL TRIP.

MALFUNCTION REMOVAL RESTORES THE FAILED CARD TO NORMAL.

EVENTS: NONE.

RX24 OVERTEMPERATURE DELTA T SETPOINT FAILURE

TYPE: GENERIC, RV 0-150%

A) CHANNEL A
B) CHANNEL B
C) CHANNEL C
D) CHANNEL D

CAUSE: CARD FAILURE

REF: 20E-1-4031 RC SERIES

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED OVERTEMPERATURE DELTA T SETPOINT TO FAIL TO THE SELECTED SEVERITY. THE VALUE OF THE SETPOINT IS INDICATED ON ITS ASSOCIATED METER AND 1TR-411, IF SELECTED (1PM05J). ANNUNCIATORS 10-C5 "OTΔT HIGH ROD STOP C-3" AND 14-B1 "OTΔT HIGH RX TRIP ALERT" ACTUATE WHEN THE VARIABLE SETPOINT IS EXCEEDED IN THE AFFECTED LOOP. USING TWO MALFUNCTIONS, IF THE OVERTEMPERATURE DELTA T VALUES OF TWO LOOPS EXCEED THE DECREASED SETPOINTS, ANNUNCIATOR 11-B4 "OTΔT RX TRIP" ACTUATES AND THE REACTOR WILL TRIP.

MALFUNCTION REMOVAL RESTORES THE FAILED CARD TO NORMAL.

EVENTS: 1) DVR 06-02-88-036

DEVIATION INVESTIGATION REPORT

RX24

TITLE LOOP 2C OTΔT SETPOINT FAILURE HIGH

PAGE 1 OF 0 2

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	POWER LEVEL
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR		
0 3	2 4	8 8	0 6	0 2	8 8	— 0 3 6 —	0 0	0 5	0 6	8 8	1	9 3

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
Lee Sues, Asst. Superintendent Technical Services Ext. 2214	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	A B	E C B D	R 1 3 5	Y					

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 3/24/88 / 1516

Unit 1 MODE N/A - N/A Rx Power N/A RCS [AB] Temperature/Pressure N/A

Unit 2 MODE 1 - Power Operations Rx Power 93% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

At 1516 on March 24, 1988, during the shiftly/daily Surveillance 2BOS 0.1-1.2.3, the Loop 2C OTΔT Setpoint Indicator 2TI-431C was found in a failed high condition. This was confirmed at the OTΔT Setpoint Pen Recorder 2TR 411. Abnormal procedure 2BOA INST-2 was entered and the bistables were placed in a tripped condition. Limiting Condition for Operation Action Requirement (LCOAR) 3.1-1a was entered and Nuclear Work Request (NWR) B54937 was initiated to troubleshoot and correct the failure.

C. CAUSE OF EVENT:

The root cause of the high setpoint indications, ΔT_{sp}, on 2TI-431C and 2TR-411 was the failure of NLL Card 2TY-0432A. NLL Card 2TY-0432A provides the temperature input to the ΔT_{sp} summing and 2TI-411L. ΔT_{sp} is calculated using the following equation and is electronically produced in this loop.

$$[\Delta T_{sp} = K_1 - K_2 \frac{1 + \tau_{1s}}{1 + \tau_{2s}} (T-588.4) + K_3 (P-2235) - f_1 (\Delta q)]$$

Since it failed low, the -588.4 term with the $-K_2$ term summed to a higher setpoint value. The cause of the failure is indeterminate. No further action is to be taken to determine the cause of the failure.

DEVIATION INVESTIGATION REPORT TEXT CONTINUATION

TITLE	DIR NUMBER							PAGE		
	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER			
LOOP 2C OTΔT SETPOINT FAILURE HIGH	0 6	0 2	8 8	—	0 3 6	—	0 0	2	OF	0 2

TEXT

D. SAFETY ANALYSIS:

The abnormal procedure 2BOA INST-2 was entered and the bistables were placed in a tripped condition. Due to the redundancy of the Reactor Protection System (RPS), the tripped bistables resulted in a 1/3 coincident logic for RPS actuation which is conservative. The normal logic is 2/4 channels coincidence for the RPS actuation. Thus, there was no effect in the health and safety of the public.

E. CORRECTIVE ACTIONS:

NLL Card 2TY-0432A has been replaced and the affected loop recalibrated. LCOAR 3.1-1a was exited on 3/25/88 at 0420 and the channel was declared operable.

F. PREVIOUS OCCURRENCES:

There have been no previous occurrences of this particular failed component as documented by a DVR though an RTD failed for the 2C loop as documented by the following DVR. No trend has been identified.

DVR NUMBER	TITLE
6-2-87-003 (87-001)	Reactor Trip Due to 2 of 4 Logic on Over Temperature Delta Temperature

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Westinghouse	7300 Circuit Card	NLL	819579

b) RESULTS OF NPRDS SEARCH:

Search shows that although this type of card has failed before, there is nothing consistent (or trending) in the types of failures. 1TY-0421C at Byron, which is a similar loop, had previously failed due to a defective relay (see DVR above), however, most card failures are not normally tracked to this degree of discreteness.

c) RESULTS OF NWR SEARCH:

None Found

RX25 RCS PRESS TRANSMITTER FAILURE (403 & 405)

TYPE: GENERIC, RV 0-3000 PSIG

A) 1PT-403
B) 1PT-405

CAUSE: PRESSURE TRANSMITTER FAILURE

REF: M-2060 SHT 17,18
20E-1-4030 RC17

PLT STA: S/D COOLING IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED TRANSMITTER OUTPUT TO FAIL TO A VALUE DEPENDENT UPON THE SELECTED SEVERITY. OUTPUT FOR PT-403/405 IS INDICATED ON 1PR-403, PI-403 AND PI-405 (1PM05J), 1PI-403A, AND 1PI-405A (1PM06J). THESE TRANSMITTERS ALSO PROVIDE PERMISSIVE SIGNALS FOR THE RHR LOOP ISOLATION VALVES OPEN INTERLOCK CIRCUIT (<360 PSIG).

WITH THE RCS IN A LOW PRESSURE CONDITION, AND THE RHR SYSTEM IN OPERATION, THE FAILURE OF A TRANSMITTER HIGH CAUSES ANNUN 6-A3 "RH SUCT PRESS HIGH" TO ACTUATE. IF THIS SIGNAL IS COINCIDENT WITH AN SI AND ANOTHER TRANSMITTER (403, 405, 408, 409) BEING > 1643 PSIG, THEN AN OPEN SIGNAL WILL BE SENT TO THE CV MINIFLOW VALVES. THE VALVES WILL OPEN AND ANNUNCIATOR 9-E1 "CHG PUMP ISOLATION VALVE CLOSED" WILL CLEAR.

FAILURE OF A TRANSMITTER <360 PSIG WILL SATISFY THE OPEN PERMISSIVE, AND ALLOW THE OPERATOR TO OPEN THE ASSOCIATED RH ISOLATION VALVE WITH RCS PRESSURE ACTUALLY HIGHER THAN PERMISSIBLE. IF THIS SIGNAL IS COINCIDENT WITH AN SI AND ANOTHER TRANSMITTER (403, 405, 408, 409) BEING < 1448 PSIG, THEN A CLOSE SIGNAL WILL BE SENT TO THE CV MINIFLOW VALVES. THE VALVES WILL CLOSE AND ANNUNCIATOR 9-E1 "CHG PUMP ISOLATION VALVE CLOSED" WILL ACTUATE.

MALFUNCTION REMOVAL RESTORES THE FAILED TRANSMITTER TO NORMAL.

EVENTS: 1) DVR 20-02-88-115

DEVIATION INVESTIGATION REPORT

RX25

TITLE Failure of Wide Range RCS Pressure Channel 405 Due to Loss of Fill Through Leaking Fittings

PAGE 1 OF 0 2

EVENT DATE MONTH	DAY	YEAR	DIR NUMBER STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE MONTH	DAY	YEAR	OPERATING MODE	POWER LEVEL
0 6	2 8	8 8	2 0	0 2	8 8	1 1 5	0 0	0 7	2 1	8 8	1	0 4 8

CONTACT FOR THIS DIR

NAME			TELEPHONE NUMBER AREA CODE	
Jenny D. Tolar,	Technical Staff Engineer	Ext. 2484	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 2; Event Date: June 28, 1988; Event Time: 1700
MODE: 1 - Power Operation; Rx Power: 48%; RCS [AB] Temperature/Pressure: 569 Degrees F/2230 psig

B. DESCRIPTION OF EVENT:

During Shift 3 on June 28, 1988, Unit 2 was operating at 48 percent power. While performing a Main Control Board walkdown, the Unit 2 Nuclear Station Operator (NSO) identified that the Wide Range (WR) Reactor Coolant System (RCS) [AB] Loop 2C pressure channel 405 was showing only 2100 psig on indicator 2PI-405A at 2PM06J. The Loop 2A WR RCS pressure indicator 2PI-403A was showing normal indication of approximately 2250 psig. At 1700 hours, channel 405 was declared inoperable and Limiting Condition for Operating Action Requirement (LCOAR) 3.3.6-1A was entered. Nuclear Work Request (NWR) A23893 was written to investigate/calibrate the loop. Stable plant conditions were maintained throughout this event.

C. CAUSE OF EVENT:

This event was caused by leaking fittings at transmitter 2PT-405. The 405 loop operates using a capillary fill system. Instrument 2PIS-405 indicates the fill level in the instrument lines from 2PIS-405 to the volume sensor in containment and from 2PIS-405 to transmitters 2PT-405 and 2PT-409. If 2PIS-405 indicates zero, both lines are properly filled. A reading to the right of zero indicates a fill abnormality on the transmitter side, and a reading to the left of zero indicates a fill abnormality on the volume sensor side. Instrument Maintenance Department (IMD) troubleshooting identified that 2PIS-405 was pegged to the right indicating there was a leak in the transmitter side at 2PT-405. This leak allowed the fill to escape from the instrument lines between the transmitters and 2PIS-405 causing a lower than actual signal to be transmitted. This resulted in the lower than actual indication on Instrument 2PI-405A.

DEVIATION INVESTIGATION REPORT TEXT CONTINUATION

TITLE	DIR NUMBER					PAGE
Failure of Wide Range RCS Pressure Channel 405 Due to Loss of Fill Through Leaking Fittings	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
	2 0	0 2	8 8 —	1 1 5 —	0 0	2 OF 0 2

TEXT

D. SAFETY ANALYSIS:

This event did not create any adverse safety consequences. The redundant channel 403 was available to provide WR RCS pressure indication. The interlocks associated with channel 405 are not required to be in effect at normal operating temperature and pressure. The most limiting condition for this event would occur when the interlock functions were effected. For the Chemical and Volume Control System (CVCS) [CB] miniflow valves, this would occur at 1448 psig on decreasing pressure and 1643 psig on increasing pressure coincident with a Safety Injection (SI) [BQ] signal. Since the interlock requires 2 out of 4 logic, redundant channels 403 and 408 would be sufficient to complete the required logic. For the Residual Heat Removal System (RHR) valves, the 405 loop provides an Auto Closure signal on 662 psig increasing pressure and a permissive to open signal at 360 psig decreasing pressure to prevent RHR piping overpressurization. Redundant channel 403 providing signals to valves in series with valves controlled by the 405 channel ensures that an overpressure event would not occur. Due to Technical Specification 4.4.9.3.2 requirements, the redundant channel 403 would be sufficient to assure an operable RHR suction relief valves.

This event has no adverse safety consequences either now or at the most limiting condition.

E. CORRECTIVE ACTIONS:

Operating personnel took the appropriate Technical Specification corrective actions. IMD contacted the vendor to assist in troubleshooting, and the leak was identified. The fittings were tightened, and the loop was monitored for 12 hours and found to be stable. Transmitter 2PT-405 was calibrated and the loop was returned to service. LCOAR 3.3.6-1A was exited at 2020 hours on July 1, 1988. No further corrective action is required.

F. PREVIOUS OCCURRENCES:

DVR/LER NUMBER	TITLE
DVR 20-2-88-028	Failure of Wide Range Reactor Coolant System Pressure Channel 403 Due to Loss of Fill Through Leaking Fitting

G. COMPONENT FAILURE DATA:

NONE

RX26 RCS PRESSURE TRANSMITTER FAILURE (406 & 407)

TYPE: GENERIC, RV 0-3000 PSIG

A) WIDE RANGE RCS PT-406
B) WIDE RANGE RCS PT-407

CAUSE: FAULTY TRANSMITTER

REF: RCS/PRESSURIZER SYSTEM DESCRIPTION
C&ID M-2060 SHEET 17,18
20E-1-4031 RC26, RC33, RC34
20E-1-4030 RC31

PLT STA: PZR PORV IN ARM LOW TEMP

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED TRANSMITTER OUTPUT TO FAIL TO A VALUE DEPENDENT UPON THE SELECTED SEVERITY. THIS SIGNAL WILL BE SENT TO THE PRESSURIZER PORV CONTROL CIRCUITS ACCORDINGLY (PT-406 SIGNAL TO THE PORV 1RY456 AND PT-407 TO PORV 1RY455A). IF THE PORVS ARE ARMED IN THE COLD OVERPRESSURIZATION MODE, THEY WILL RESPOND ACCORDING TO THEIR PROGRAM WITH THE FALSE TRANSMITTER INPUT SIGNAL. ANNUNCIATOR 12-D4 "RC SYSTEM COLD PRESS HIGH" ACTUATES AT 20 PSIG BELOW THE VARIABLE SETPOINT. ANNUNCIATOR 12-C4 "RC PRESS HIGH AT LOW TEMP PORV OPEN" ACTUATES AT THE PORV VARIABLE SETPOINT.

MALFUNCTION REMOVAL WILL RESTORE THE TRANSMITTER TO NORMAL.

EVENTS: NONE.

RX27 RCS PRESSURE TRANSMITTER FAILURE (408 & 409)

TYPE: GENERIC, RV 0-3000 PSIG

A) WIDE RANGE RCS PT-408
B) WIDE RANGE RCS PT-409

CAUSE: FAULTY TRANSMITTER

REF: RCS/CVCS SYSTEM DESCRIPTIONS
C&ID M-2060 SHEET 17,18
20E-1-4031 RC35, RC36
20E-1-4030 CV37, CV38, EF17, EF61

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED TRANSMITTER OUTPUT TO FAIL TO A VALUE DEPENDENT UPON THE SELECTED SEVERITY. AT A SEVERITY LEVEL CORRESPONDING TO A PRESSURE OF <1448 PSIG, THE TRANSMITTER(S) WILL SIGNAL THE CV PUMP MINI FLOW VALVES (1CV8114/8116) SSPS LOGIC CIRCUITS AS EVIDENCED BY THE ILLUMINATION OF THE RCS PRESS LOW TRIP STATUS LIGHTS ON 1PM05J. IF THIS SIGNAL IS COINCIDENT WITH AN SI AND ANOTHER TRANSMITTER (403, 405, 408, 409) BEING <1448 PSIG, THEN A CLOSE SIGNAL WILL BE SENT TO THE CV MINIFLOW VALVES. THE VALVES WILL CLOSE AND ANNUNCIATOR 9-E1 "CHG PUMP ISOLATION VALVE CLOSED" WILL ACTUATE.

AT A SEVERITY LEVEL CORRESPONDING TO A PRESSURE OF >1643 PSIG, THE TRANSMITTER(S) WILL SIGNAL THE CV PUMP MINI FLOW VALVES (1CV8114/8116) SSPS LOGIC CIRCUITS AS EVIDENCED BY THE ILLUMINATION OF THE RCS PRESS HIGH TRIP STATUS LIGHTS ON 1PM05J. IF THIS SIGNAL IS COINCIDENT WITH AN SI AND ANOTHER TRANSMITTER (403, 405, 408, 409) BEING >1643 PSIG, THEN AN OPEN SIGNAL WILL BE SENT TO THE CV MINIFLOW VALVES. THE VALVES WILL OPEN AND ANNUNCIATOR 9-E1 "CHG PUMP ISOLATION VALVE CLOSED" WILL CLEAR.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED TRANSMITTER TO NORMAL.

EVENTS: NONE.

RX28 RCS LOOP FLOW TRANSMITTER FAILURE

TYPE: GENERIC, RV 0-110% SEVERITY

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*****************************************************
*                      NOTE:                        *
*                                                   *
*       0-110% SEVERITY EQUATES TO 0-100%           *
*       INDICATED FLOW.                             *
*****************************************************
```

A)	1FT414	G)	1FT434
B)	1FT415	H)	1FT435
C)	1FT416	I)	1FT436
D)	1FT424	J)	1FT444
E)	1FT425	K)	1FT445
F)	1FT426	L)	1FT446

CAUSE: DETECTOR FAILURE

REF: M-2060 SHEET 1
20E-1-4029 EF05

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE AFFECTED DETECTOR TO FAIL TO A VALUE DEPENDENT UPON MALFUNCTION SEVERITY (SEE NOTE ABOVE). FLOW INDICATIONS RESPOND ACCURATELY TO THE SEVERITY. WHEN THE SELECTED SEVERITY IS DECREASED TO LESS THAN 90% FLOW, ANNUNCIATOR 13-A3/B3/C3/D3 "RCP 1A/B/C/D BRKR OPEN OR FLOW LOW ALERT" ACTUATES. IF TWO MALFUNCTIONS FROM THE SAME LOOP ARE DECREASED BELOW 90% FLOW WITH REACTOR POWER GREATER THAN P8, ANNUNCIATOR 11-C5 "RCP LOW FLOW ABOVE P8 RX TRIP" ACTUATES AND THE REACTOR WILL TRIP. IF TWO MALFUNCTIONS FROM THE SAME LOOP FOR TWO LOOPS ARE DECREASED BELOW 90% FLOW WITH P7 SATISFIED, ANNUNCIATOR 11-D5 "RCP LOW FLOW ABOVE P7 RX TRIP" ACTUATES AND THE REACTOR WILL TRIP.

MALFUNCTION REMOVAL RESTORES THE FAILED LOOP FLOW DETECTOR TO NORMAL.

EVENTS: 1) DVR 20-01-88-121
2) LER 06-02-88-012

RX28

LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Byron, Unit 2	0 5 0 0 0 4 5 5	1 of 0 3

Title (4)

Reactor Trip On Indicated Low Reactor Coolant Flow Rate Caused By Cleaning Flow Transmitter Vent Valve

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
1 2	1 5	8 8	8 8	0 1 2	0 0	0 1	0 9	8 9	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 4 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	Other (Specify in Abstract below and in Text)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER AREA CODE	
Dale St. Clair, Assistant Superintendent Work Planning, Ext. 2888	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO	Expected Submission Date (15)	Month	Day	Year

ABSTRACT (Limit to 1400 spaces, i.e, approximately fifteen single-space typewritten lines) (16)

At 0849 on December 15, 1988 personnel entered the Unit 2 Containment Building to inspect for active boric acid leakage as a pre-refueling outage function. The unit was at 40 percent power. The inspection team observed boric acid crystals and a minor packing leak on an instrument valve labeled "2FT-434 Vent". While removing the encrusted boric acid and tightening the packing nut about one flat at 1002, an automatic reactor trip occurred due to indicated low reactor coolant loop C flow. The inspection team was unaware of the reactor trip and the Control Room Operators could not correlate any specific inspection team activities with the trip. An automatic Feedwater Isolation and automatic Auxiliary Feedwater Pump starts occurred as expected following the trip. The plant was stabilized in Hot Standby at approximately 1030.

The low reactor coolant flow signals were caused by a pressure transient in the high pressure sensing line, which is common to all three flow transmitters in the reactor coolant elbow flow meter design. By removing encrusted boric acid from the vent valve and tightening its packing, a mechanic induced sufficient vibration in the high pressure sensing line to cause a pressure decrease at the connections to all three flow transmitters. The low flow indication persisted for approximately 200 milliseconds.

To prevent recurrence of the event, valves associated with reactor coolant loop flow transmitters will be prominently marked to prohibit physical contact unless reactor power is below the P-8 permissive setpoint (30 percent nuclear power).

A previous similar occurrence is described in Unit 1 LER 85-090.

(0209R/0023R)

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 12-15-88/ 1002

Unit 2 MODE 1 - Power Operations Rx Power 40% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

There were no systems inoperable at the beginning of this event that contributed to its occurrence or severity. Unit 2 was in the Power Operations Mode at 40 percent reactor power. At 0849 on December 15, 1988 eight personnel entered the Unit 2 Containment Building to conduct a leak inspection of systems containing boric acid and to evaluate identified leakage for corrective action to be taken during an imminent refueling outage. The inspection team consisted of a Shift Control Room Engineer (licensed senior reactor operator), an Inservice Inspection Engineer, a Mechanical Maintenance Foreman, Mechanical Maintenance Technicians, a Health Physics Foreman and Health Physics Technicians. The inspection team observed an accumulation of boric acid crystals and a minor packing leak on a manual instrument valve (1/2 inch Anderson Greenwood, single packing nut with integral packing follower). The valve was labeled with a construction tag that read " 2FT-434 Vent". While removing encrusted boric acid from the instrument valve with a stainless steel bristle brush and tightening the packing nut about one flat at 1002, an automatic reactor trip occurred due to all three reactor coolant [AB] loop C flow instruments detecting less than 90 percent of rated flow. The Nuclear Station Operator (NSO) (licensed reactor operator) was alerted to the reactor trip in the Main Control Room by the actuation of the "RCP Flow Low Above P-8 Reactor Trip" annunciator. The inspection team was unaware that a reactor trip had occurred, and although the NSO was aware of the inspection team in the Containment, he lacked the specific information to permit a correlation between inspection team activities and the reactor trip. Plant response to the trip was normal and included an automatic Feedwater Isolation [SJ] due to low average reactor coolant temperature coincident with the open reactor trip breakers and automatic starts of both Auxiliary Feedwater Pumps [BA] due to lo-2 steam generator levels. Licensed operators in the Main Control Room entered and complied with the "Reactor Trip or Safety Injection Unit 2 Emergency Operating Procedure" and the "Reactor Trip Response Unit 2 Emergency Operating Procedure". The inspection team was notified of the trip and ordered to leave the Containment Building. At 1019 the NSO started the Startup Feedwater Pump and aligned flow to feed the steam generators. By 1020 all personnel had exited the Containment Building. At 1033 the NSO stopped both Auxiliary Feedwater Pumps, since the Startup Feedwater Pump was maintaining steam generator levels. The plant was stable in the Hot Standby Operational Mode at approximately 1030. The NRC Operations Center was notified of the automatic Engineered Safety Feature (ESF) actuations via the Emergency Notification System at 1342. This Licensee Event Report (LER) is submitted pursuant to 10CFR 50.73(a)(2)(iv) due to the automatic actuations of ESF systems.

C. CAUSE OF EVENT:

All three reactor coolant loop C elbow flow meters sensed a flow signal less than 90 percent of rated flow. Two of the three low flow signals satisfied the coincidence to trip the reactor coolant loop C loss of flow bistable. Since reactor power was above the P-8 permissive circuit setpoint of 30 percent nuclear power, the loss of flow signal in a single reactor coolant loop induced an automatic reactor trip.

C. CAUSE OF EVENT: (cont.)

The low flow signals were caused by a pressure transient in the high pressure sensing line, which is common to all three flow transmitters in the reactor coolant elbow flow meter design. By removing encrusted boric acid from the 2FT-434 vent valve and tightening its packing, a Mechanical Maintenance technician induced sufficient vibration in the high pressure sensing line to cause a pressure decrease at the connections to all three flow transmitters. The pressure decrease was sensed by the transmitters as a low reactor coolant flow condition, which endured for approximately 200 milliseconds. Subsequent investigation revealed that a differential pressure decrease of approximately 2.3 psid is equivalent to a reactor coolant flow decrease from rated flow to 90 percent of rated flow. Considering that the high pressure tap senses reactor coolant crossover pipe pressure of about 2235 psig, the relatively insignificant pressure decrease needed to trip the low flow bistable clearly demonstrates the sensitivity of the reactor coolant flow instrumentation. The vent valve serves extremely sensitive instrumentation and was not clearly marked to indicate its specific function nor its potential impact on power operation.

D. SAFETY ANALYSIS:

Actual reactor coolant flow was maintained at its design rating throughout this event. The automatic reactor trip was caused by an incorrect indication of low flow, that did not reflect actual flow conditions. The plant responded normally following the trip and all ESF systems actuated properly. Had this event occurred at full power, the ESF system response would have been identical. The event did not impact plant or public safety.

E. CORRECTIVE ACTIONS:

To prevent recurrence of this event, valves associated with reactor coolant loop flow transmitters will be prominently marked to prohibit physical contact unless reactor power is below the P-8 setpoint. Corrective actions are tracked by Action Item Records 454-225-89-0001 and 455-225-89-0002. Tightening the packing nut on the 2FT-434 vent valve one flat was successful in stopping the minor packing leak.

F. PREVIOUS OCCURRENCES:

LER NUMBER	TITLE
454-85-090	"Unit Reactor Trip On Low Loop Flow While Venting Flow Transmitter"

The applicable instrument maintenance procedure was revised to prohibit reactor coolant flow transmitter maintenance unless reactor power is below the P-8 setpoint. This preventive action was not intended to prohibit the agitation of valves connected to the flow transmitters, as occurred during the December 15, 1988 event. The currently proposed preventive action expands the previous actions and should minimize reoccurrence of low reactor coolant flow trips caused by instrument sensitivity.

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Not Applicable			

RXZ8

DEVIATION INVESTIGATION REPORT

TITLE Loop 1B Low Flow Rx Trip Alert Annunciation Due to Blown Fuse On Circuit Card 1FB-0424A	PAGE 1 OF 0 2

EVENT DATE MONTH	DAY	YEAR	DIR NUMBER STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE MONTH	DAY	YEAR	OPERATING MODE 1 / POWER LEVEL
0 5	1 2	8 8	2 0	0 1	8 8	1 2 1	0 0	0 5	2 7	8 8	0 7 6

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER AREA CODE	
Freddie Ramos, Technical Staff Engineer Ext. 2487	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	A B	E C B D	W 1 2 0	NO					

SUPPLEMENTAL REPORT EXPECTED		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 1 ; Event Date: May 12, 1988 ; Event Time: 2257
MODE: 1 - Power Operation ; Rx Power: 76% ; RCS [AB] Temperature/Pressure: 578 Degrees F/2240 psig

B. DESCRIPTION OF EVENT:

No systems or components were inoperable at the beginning of the event which contributed to the event. At 2257 hrs on May 12, 1988 annunciator window 13-B-3 "RCP 1B Brkr Open or Flow Low Alert" and test status light Reactor Coolant System (TSLB-3) cube 5.1 "RC 1B Low Flow FB424A" illuminated. The channel was declared inoperable and the appropriate bistable tripped per operating procedure BwOA INST-2. Instrument Maintenance Department (IMD) was contacted and asked to investigate/troubleshoot. An IM Technician found a failed signal comparator card located in protection cabinet 1PA01J, slot 325. The technician traced the failure to a blown fuse on the circuit card. The card failed in the conservative direction which reduced the coincidence logic on Loop 1B from 2/3 to 1/2. The technician installed a new fuse and replaced the card. An extra Unit One Nuclear Station Operator (NSO) manually switched the bistable for loop 0424 back to normal (i.e. operable status) and the annunciations in the Control Room cleared.

C. CAUSE OF EVENT:

The root cause of the event is unknown. The intermediate cause was due to a blown fuse on signal comparator card 1FB-0424A. This occurrence is considered an isolated event because there have not been any previous failures, of this kind, recorded for card 1FB-0424A.

TITLE	DIR NUMBER					PAGE
Loop 1B Low Flow Rx Trip Alert Annunciation Due to Blown Fuse On Circuit Card 1FB-0424A	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
	2 0	0 1	8 8	1 2 1	0 0	2 OF 0 2

TEXT

D. SAFETY ANALYSIS:

There was no effect on plant or public safety. This event resulted in only annunciation with no components actuating. Under worst case conditions with a failed 1FB-0424A circuit card, any Final Safety Analysis Report (FSAR) postulated accident would have been covered. The failure of the card immediately reduced the coincidence logic from 2/3 to 1/2 which is in the conservative direction.

E. CORRECTIVE ACTIONS:

The immediate corrective action by the Unit One NSO was to put loop 0424 to trip which maintained the coincidence logic at 1/2 and allowed IMD to troubleshoot the affected channel. Loops 0424, 0425, and 0426 monitor flow on Loop 1B. After troubleshooting, IMD replaced the blown fuse on circuit card 1FB-0424A and reinstalled the card. The extra Unit One NSO returned loop 0424 to normal and the annunciations in the Control Room were able to be cleared. No further corrective action is necessary as this is considered an isolated event.

F. PREVIOUS OCCURRENCES:

NONE

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Westinghouse	Signal Comparator	2837A13G01	428B04

RX29 FW REG VLV CONTROLLER FAILURE

TYPE: GENERIC, RV 0-100%

A) 510
B) 520
C) 530
D) 540

CAUSE: CONTROLLER AUTO OUTPUT FAILURE

REF: 20E-1-4031 FW16
20E-1-4031 FW17
20E-1-4031 FW18
20E-1-4031 FW19

PLT STA: REACTOR OPERATING AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED FEED REG VALVE TO FAIL TO A VALUE DEPENDENT UPON MALFUNCTION SEVERITY. FEED FLOW INDICATORS WILL RESPOND TO THE FAILURE. IF THE SEVERITY IS LESS THAN THE INITIAL VALUE, ANNUNCIATORS 15-A4/B4/C4/D4 "S/G FLOW MISMATCH FW FLOW LOW," 15-A9/B9/C9/D9 "S/G LEVEL DEVIATION HIGH LOW," 15-A6/B6/C6/D6,"S/G LEVEL LOW," 15-A11/B11/C11/D11 "S/G MAIN FW NOZZLE FLOW HIGH LOW," AND 11-A8/B8/C8/D8 "S/G LEVEL L0-2 RX TRIP" WILL ACTUATE. IF THE SEVERITY IS GREATER THAN THE INITIAL VALUE, ANNUNCIATORS 15-A3/B3/C3/D3 "S/G FLOW MISMATCH STM FLOW LOW," 15-A9/B9/C9/D9 "S/G LEVEL DEVIATION HIGH LOW," 15-A8/B8/C8/D8 "S/G LVL HI-2 TURB TRIP P-14 ALERT," AND 18-A1/B1/C1/D1 "S/G LEVEL HI-2 TURB TRIP" WILL ACTUATE.

THE EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY TAKING MANUAL CONTROL OF THE FEED REGULATING VALVES TO MAINTAIN PROPER S/G LEVELS.

THE CONTROL VALVES WILL STILL CLOSE ON RECEIPT OF ANY FEEDWATER ISOLATION SIGNAL.

MALFUNCTION REMOVAL WILL RESTORE THE MAIN FEEDWATER AUTO CONTROL TO NORMAL.

EVENTS: NONE.

RX30 FW BYP VLV CONTROLLER FAILURE

TYPE: GENERIC, RV 0-100%

A) 510A
B) 520A
C) 530A
D) 540A

CAUSE: CONTROLLER AUTO OUTPUT FAILURE

REF: 20E-1-4031 FW26
20E-1-4031 FW27
20E-1-4031 FW71
20E-1-4031 FW72

PLT STA: REACTOR POWER IN THE POWER RANGE

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED S/G BYPASS CONTROL VALVE TO FAIL TO A VALUE DEPENDENT UPON THE MALFUNCTION SEVERITY. FEED FLOW INDICATORS WILL RESPOND TO THE FAILURE. IF THE SEVERITY IS LESS THAN THE INITIAL VALUE, ANNUNCIATORS 15-A4/B4/C4/D4 "S/G FLOW MISMATCH FW FLOW LOW," 15-A9/B9/C9/D9 "S/G LEVEL DEVIATION HIGH LOW," 15-A6/B6/C6/D6 "S/G LEVEL LOW" WILL ACTUATE. IF THE SEVERITY IS GREATER THAN THE INITIAL VALUE, ANNUNCIATORS 15-A3/B3/C3/D3 "S/G FLOW MISMATCH STM FLOW LOW," 15-A9/B9/C9/D9 "S/G LEVEL DEVIATION HIGH LOW," 15-A8/B8/C8/D8 "S/G LVL HI-2 TURB TRIP P-14 ALERT," AND 18-A1/B1/C1/D1 "S/G LEVEL HI-2 TURB TRIP" WILL ACTUATE.

THE EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY TAKING MANUAL CONTROL OF THE FEEDWATER BYPASS VALVE TO MAINTAIN PROPER S/G LEVELS.

THE BYPASS VALVES WILL STILL CLOSE ON A RECEIPT OF ANY FEEDWATER ISOLATION SIGNAL.

MALFUNCTION REMOVAL WILL RESTORE THE FEEDWATER BYPASS AUTO CONTROL TO NORMAL.

EVENTS: NONE.

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

SI01	SAFETY INJECTION PUMP FAILS TO START/TRIP
SI02	SI ACCUMULATOR LEVEL XMITTER FAILURE
SI03	COLD LEG INJ CHECK VALVE LEAKAGE (SI8818)
SI04	COLD LEG INJ CHECK VALVE LEAKAGE (SI8819)
SI05	COLD LEG INJ CHECK VALVE LEAKAGE (SI8948)
SI06	COLD LEG INJ CHECK VALVE LEAKAGE (SI8956)
SI07	HOT LEG INJ CHECK VALVE LEAKAGE (SI8905)
SI08	HOT LEG INJ CHECK VALVE LEAKAGE (SI8841)
SI09	HOT LEG INJ CHECK VALVE LEAKAGE (SI8949)
SI10	HIGH HEAD SI LEAK INSIDE CONTAINMENT
SI11	SI ACCUMULATOR TANK RUPTURE

SI01 SAFETY INJECTION PUMP FAILS TO START/TRIP

TYPE: GENERIC, RB

A) 1SI01PA
B) 1SI01PB

CAUSE: FAULTY OVERCURRENT (450/451) RELAY ACTUATION

REF: 20E-1-4030 SI01
20E-1-4030 SI02

PLT STA: SELECTED PUMP IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED SAFETY INJECTION PUMP BREAKER TO TRIP. PUMP CURRENT INDICATION DECREASES TO ZERO. ANNUNCIATOR 5-A4 "SI PUMP TRIP" ACTUATES, AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. PUMP FLOW, AS INDICATED ON 1FI-918/922 (1PM06J), WILL DECREASE TO ZERO.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND THE TRIP LIGHT BY PLACING THE CONTROL SWITCH IN THE TRIP POSITION. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL CLOSE THEN IMMEDIATELY TRIP OPEN.

MALFUNCTION REMOVAL RESTORES THE OVERCURRENT RELAY TO NORMAL OPERATION.

EVENTS: NONE.

SI02 SI ACCUMULATOR LEVEL XMITTER FAILURE

TYPE: GENERIC, RV 0-100%

A) 1A SI ACCUMULATOR LT-950
B) 1A SI ACCUMULATOR LT-951
C) 1B SI ACCUMULATOR LT-952
D) 1B SI ACCUMULATOR LT-953
E) 1C SI ACCUMULATOR LT-954
F) 1C SI ACCUMULATOR LT-955
G) 1D SI ACCUMULATOR LT-956
H) 1D SI ACCUMULATOR LT-957

CAUSE: FAULTY TRANSMITTER

REF: M-2061 SHEET 1

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED LEVEL TRANSMITTER TO FAIL TO A VALUE DEPENDENT UPON THE SELECTED SEVERITY. THIS WILL BE INDICATED ON ITS ASSOCIATED LEVEL INDICATOR 1LI-950-957. ANNUNCIATORS 5-A1/B1/C1/D1 "ACCUM 1A (1B,1C,1D) LEVEL HIGH LOW" MAY ACTUATE DEPENDING ON SEVERITY.

MALFUNCTION REMOVAL WILL RESTORE THE FAULTY TRANSMITTER TO NORMAL.

EVENTS: NONE.

SI03 COLD LEG INJ CHECK VALVE LEAKAGE (SI8818)

TYPE: GENERIC, RV 0-500 GPM AT 2200 PSID WITH MF SI05() SET AT 500 GPM

A) 1A CL 1SI8818A
B) 1B CL 1SI8818B
C) 1C CL 1SI8818C
D) 1D CL 1SI8818D

CAUSE: WORN VALVE SEAT

REF: M-61 SHEET 4

PLT STA: REACTOR AT POWER

EFFECTS: THIS MALFUNCTION REQUIRES THAT MALFUNCTION SI05 "COLD LEG INJECTION CHECK VALVE LEAKAGE 1SI8948 A-D" BE RUN CONCURRENTLY. WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED COLD LEG INJECTION CHECK VALVE WILL BEGIN TO LEAK FROM THE REACTOR COOLANT SYSTEM INTO THE RESIDUAL HEAT REMOVAL (RH) SYSTEM. THE RATE OF IN-LEAKAGE WILL BE DETERMINED BY THE SELECTED SEVERITY. THE MASS TRANSFER TO THE RH SYSTEM WILL RESULT IN A DECREASE IN PRESSURIZER LEVEL. THE RH SYSTEM IN-LEAKAGE WILL RESULT IN AN INCREASE IN RH SYSTEM PRESSURE AS INDICATED ON THE DISCHARGE HEADER PRESSURE INDICATORS 1PI-614 FOR A TRAIN AND 1PI-615 FOR B TRAIN. WHEN THE PRESSURE INCREASES TO THE PRESSURE SETPOINT OF RH PUMP SUCTION RELIEF VALVE 1RH-8708A/B (450 PSIG), THE RELIEF VALVE WILL LIFT AND DISCHARGE TO THE HOLDUP TANK. IF THE PRESSURE INCREASES TO THE PRESSURE SETPOINT OF DISCHARGE RELIEF VALVE 1SI-8856A/B (600 PSIG), THE RELIEF VALVE WILL LIFT AND DISCHARGE TO THE HOLDUP TANK.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED CHECK VALVE TO NORMAL.

EVENTS: NONE.

SI04 COLD LEG INJ CHECK VALVE LEAKAGE (SI8819)

TYPE: GENERIC, RV 0-500 GPM AT 2200 PSID WITH MF SI05() SET AT 500 GPM

A) 1A CL 1SI8819A
B) 1B CL 1SI8819B
C) 1C CL 1SI8819C
D) 1D CL 1SI8819D

CAUSE: WORN VALVE SEAT

REF: M-61 SHEET 3

PLT STA: REACTOR AT POWER

EFFECTS: THIS MALFUNCTION REQUIRES THAT MALFUNCTION SI05 "COLD LEG INJECTION CHECK VALVE LEAKAGE 1SI8948 A-D" BE RUN CONCURRENTLY. WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED COLD LEG INJECTION CHECK VALVE WILL BEGIN TO LEAK FROM THE REACTOR COOLANT SYSTEM INTO THE SAFETY INJECTION (SI) SYSTEM. THE RATE OF IN-LEAKAGE WILL BE DETERMINED BY THE SELECTED SEVERITY. THE MASS TRANSFER TO THE SAFETY INJECTION SYSTEM WILL RESULT IN A DECREASE IN PRESSURIZER LEVEL. THE SI SYSTEM IN-LEAKAGE WILL RESULT IN AN INCREASE IN SI SYSTEM PRESSURE AS INDICATED ON THE DISCHARGE HEADER PRESSURE INDICATORS 1PI-919 FOR A TRAIN AND 1PI-923 FOR B TRAIN. THE IN-LEAKAGE WILL RESULT IN AN INCREASE IN PRESSURE IN THE SI INJECTION HEADER. WHEN THE PRESSURE INCREASES TO THE PRESSURE SETPOINT OF RELIEF VALVE 1SI-8851 (1750 PSIG), THE RELIEF VALVE WILL LIFT AND DISCHARGE TO THE HOLDUP TANK.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED CHECK VALVE TO NORMAL.

EVENTS: NONE.

SI05 COLD LEG INJ CHECK VALVE LEAKAGE (SI8948)

TYPE: GENERIC, RV 0-500 GPM AT 2200 PSID

A) 1A CL 1SI8948A
B) 1B CL 1SI8948B
C) 1C CL 1SI8948C
D) 1D CL 1SI8948D

CAUSE: WORN VALVE SEAT

REF: M-61 SHEET 5 & 6

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED COLD LEG INJECTION CHECK VALVE WILL BEGIN TO LEAK FROM THE REACTOR COOLANT SYSTEM INTO THE SAFETY INJECTION SYSTEM (SI) OR RH SYSTEM. THE RATE OF IN-LEAKAGE WILL BE DETERMINED BY THE SELECTED SEVERITY. THE MASS TRANSFER TO THE SI/RH SYSTEM WILL RESULT IN A DECREASE IN PRESSURIZER LEVEL AND WILL CONTINUE UNTIL PRESSURE IS EQUALIZED ACROSS THE CHECK VALVE.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED CHECK VALVE TO NORMAL.

EVENTS: NONE.

SI06 COLD LEG INJ CHECK VALVE LEAKAGE (SI8956)

TYPE: GENERIC, RV 0-500 GPM AT 1600 PSID WITH MF SI05() SET AT 500 GPM

A) 1A CL 1SI8956A
B) 1B CL 1SI8956B
C) 1C CL 1SI8956C
D) 1D CL 1SI8956D

CAUSE: WORN VALVE SEAT

REF: M-61 SHEET 5 & 6

PLT STA: REACTOR AT POWER

EFFECTS: THIS MALFUNCTION REQUIRES THAT THE ASSOCIATED MALFUNCTION SI05 "COLD LEG INJECTION CHECK VALVE LEAKAGE 1SI8948 A-D" BE RUN CONCURRENTLY. WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED COLD LEG INJECTION CHECK VALVE WILL BEGIN TO LEAK FROM THE REACTOR COOLANT SYSTEM INTO THE SAFETY INJECTION (SI) ACCUMULATOR. THE RATE OF IN-LEAKAGE WILL BE DETERMINED BY THE SELECTED SEVERITY. THE MASS TRANSFER TO THE SI ACCUMULATORS WILL RESULT IN A DECREASE IN PRESSURIZER LEVEL AND AN INCREASE IN SI ACCUMULATOR LEVEL AND PRESSURE AS INDICATED ON ITS ASSOCIATED LEVEL AND PRESSURE INDICATORS (1LI-950/957 AND 1PI-960/967). WHEN ACCUMULATOR LEVEL IS HIGH, ANNUNCIATORS 5-A1,B1,C1 AND/OR D1 "ACCUM 1A (1B,1C,1D) LEVEL HIGH LOW" WILL ACTUATE. WHEN ACCUMULATOR PRESSURE IS HIGH, ANNUNCIATORS 5-A2,B2,C2 AND/OR D2 "ACCUM 1A (1B,1C,1D) PRESS HIGH LOW" WILL ACTUATE. IF ACCUMULATOR PRESSURE IS INCREASED TO >700 PSIG, THEN THE ACCUMULATOR RELIEF VALVE, 1SI8855 A-D, WILL LIFT TO RELIEVE THE OVERPRESSURE TO THE CONTAINMENT ATMOSPHERE.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY ISOLATING, VENTING AND/OR DRAINING THE AFFECTED ACCUMULATOR.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED CHECK VALVE TO NORMAL.

EVENTS: NONE.

SI07 HOT LEG INJ CHECK VALVE LEAKAGE (SI8905)

TYPE: GENERIC, RV 0-500 GPM AT 2200 PSID

A) 1A HL 1SI8905A
B) 1B HL 1SI8905B
C) 1C HL 1SI8905C
D) 1D HL 1SI8905D

CAUSE: WORN VALVE SEAT

REF: M-61 SHEET 3

PLT STA: HL INJ PATH IN-SERVICE

EFFECTS: THIS MALFUNCTION REQUIRES THAT MALFUNCTION SI09 "HOT LEG INJECTION CHECK VALVE LEAKAGE 1SI8949 A-D" BE RUN CONCURRENTLY. WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED HOT LEG INJECTION CHECK VALVE WILL BEGIN TO LEAK FROM THE REACTOR COOLANT SYSTEM INTO THE SAFETY INJECTION (SI) SYSTEM. THE RATE OF IN-LEAKAGE WILL BE DETERMINED BY THE SELECTED SEVERITY. THE MASS TRANSFER TO THE SAFETY INJECTION SYSTEM WILL RESULT IN A DECREASE IN PRESSURIZER LEVEL AND WILL CONTINUE UNTIL PRESSURE IS EQUALIZED ACROSS THE CHECK VALVE. IF THE SI TO RCS HOT LEG INJECTION ISOLATION VALVES 1SI8802A AND/OR B ARE OPEN, THE IN-LEAKAGE WILL RESULT IN AN INCREASE IN SI SYSTEM PRESSURE AS INDICATED ON THE DISCHARGE HEADER PRESSURE INDICATORS 1PI-919 FOR A TRAIN AND 1PI-923 FOR B TRAIN. THE IN-LEAKAGE WILL RESULT IN AN INCREASE IN PRESSURE IN THE SI INJECTION HEADER. WHEN THE PRESSURE INCREASES TO THE PRESSURE SETPOINT OF RELIEF VALVE, 1SI-8853A/B, (1750 PSIG), THE RELIEF VALVE WILL LIFT AND DISCHARGE TO THE HOLDUP TANK.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED CHECK VALVE TO NORMAL.

EVENTS: NONE.

SI08 HOT LEG INJ CHECK VALVE LEAKAGE (SI8841)

TYPE: GENERIC, RV 0-500 GPM AT 2200 PSID

A) 1A HL 1SI8841A
B) 1C HL 1SI8841B

CAUSE: WORN VALVE SEAT

REF: M-61 SHEET 3

PLT STA: HL INJ PATH IN-SERVICE

EFFECTS: THIS MALFUNCTION REQUIRES THAT MALFUNCTION SI09 "HOT LEG IN8ECTION CHECK VALVE LEAKAGE 1SI8949 A-D" BE RUN CONCURRENTLY. WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED HOT LEG INJECTION CHECK VALVE WILL BEGIN TO LEAK FROM THE REACTOR COOLANT SYSTEM INTO THE RESIDUAL HEAT REMOVAL (RH) SYSTEM. THE RATE OF IN-LEAKAGE WILL BE DETERMINED BY THE SELECTED SEVERITY. THE MASS TRANSFER TO THE RH SYSTEM WILL RESULT IN A DECREASE IN PRESSURIZER LEVEL AND WILL CONTINUE UNTIL PRESSURE IS EQUALIZED ACROSS THE CHECK VALVE.

IF THE RH TO THE RCS HOT LEG INJECTION ISOLATION VALVE (1SI8840) IS OPEN, THEN THE IN-LEAKAGE WILL RESULT IN AN INCREASE IN RH SYSTEM PRESSURE. THIS PRESSURE IS INDICATED ON THE DISCHARGE HEADER PRESSURE INDICATORS 1PI-614 FOR A TRAIN AND 1PI-615 FOR B TRAIN. WHEN THE PRESSURE INCREASES TO THE PRESSURE SETPOINT OF RH PUMP SUCTION RELIEF VALVE, 1RH-8708A/B, (450 PSIG), THE RELIEF VALVE WILL LIFT AND DISCHARGE TO THE HOLDUP TANK. IF THE PRESSURE INCREASES TO THE PRESSURE SETPOINT OF DISCHARGE RELIEF VALVE, 1SI-8842, (600 PSIG), THE RELIEF VALVE WILL LIFT AND DISCHARGE TO THE HOLDUP TANK.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED CHECK VALVE TO NORMAL.

EVENTS: NONE.

SI09 HOT LEG INJ CHECK VALVE LEAKAGE (SI8949)

TYPE: GENERIC, RV 0-500 GPM AT 2200 PSID

A) 1A HL 1SI8949A
B) 1B HL 1SI8949B
C) 1C HL 1SI8949C
D) 1D HL 1SI8949D

CAUSE: WORN VALVE SEAT

REF: M-61 SHEET 3

PLT STA: HL INJ PATH IN-SERVICE

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED HOT LEG INJECTION CHECK VALVE WILL BEGIN TO LEAK FROM THE REACTOR COOLANT SYSTEM INTO THE SAFETY INJECTION (SI)/RH SYSTEM. THE RATE OF IN-LEAKAGE WILL BE DETERMINED BY THE SELECTED SEVERITY. THE MASS TRANSFER TO THE SI/RH SYSTEM WILL RESULT IN A DECREASE IN PRESSURIZER LEVEL AND WILL CONTINUE UNTIL PRESSURE IS EQUALIZED ACROSS THE CHECK VALVE.

MALFUNCTION REMOVAL WILL RESTORE THE AFFECTED CHECK VALVE TO NORMAL.

EVENTS: NONE.

SI10 HIGH HEAD SI LEAK INSIDE CONTAINMENT

TYPE: DISCRETE, RV 0-1000 GPM AT 600 PSID

CAUSE: PIPE BREAK AT OUTLET OF CHECK VALVE 1SI8815

REF: M-61 SHEET 2

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THERE ARE NO IMMEDIATELY NOTICEABLE EFFECTS. WHEN A SAFETY INJECTION IS ACTUATED AND THE CHARGING PUMPS ARE DISCHARGING TO THE RCS COLD LEGS, A LOSS OF CHARGING WATER MASS TO THE CONTAINMENT WILL OCCUR. THE RATE OF MASS LOSS WILL BE DETERMINED BY THE SELECTED SEVERITY.

CONTAINMENT ACTIVITY LEVELS, AREA RADIATION LEVELS AND SUMP LEVELS MAY INCREASE DEPENDENT UPON THE SELECTED MALFUNCTION SEVERITY. THE EFFECTS ON CONTAINMENT TEMPERATURE AND PRESSURE FROM THIS LEAK WILL BE MINIMAL DUE TO THE LOW TEMPERATURE OF THE CHARGING/RWST WATER. IF THE CHARGING WATER IS REQUIRED TO COOL THE REACTOR CORE, THE DURATION OF ELEVATED CORE TEMPERATURE WILL BE LONGER DUE TO REDUCED COOLING WATER FLOW.

MALFUNCTION REMOVAL WILL ONLY RESTORE HIGH HEAD SI LINE PIPING INTEGRITY.

EVENTS: NONE.

SI11 SI ACCUMULATOR TANK RUPTURE

TYPE: GENERIC, RV 0-500 GPM @ 600 PSID

A) 1SI04TA
B) 1SI04TB
C) 1SI04TC
D) 1SI04TD

CAUSE: TANK RUPTURE

REF: M-61 SHEET 5
M-61 SHEET 6

PLT STA: RX AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE AFFECTED ACCUMULATOR TO LEAK TO THE CONTAINMENT. THE AFFECTED TANK LEVEL AND PRESSURE WILL DECREASE (ON 1PM06J) AT A RATE DEPENDENT UPON THE MALFUNCTION SELECTED SEVERITY. ANNUNCIATORS 5-A1/B1/C1/D1 "ACCUM 1A/B/C/D LEVEL HIGH LOW", AND 5-A2/B2/C2/D2 "ACCUM 1A/B/C/D PRESS HIGH LOW" WILL ACTUATE.

MALFUNCTION REMOVAL RESTORES THE AFFECTED ACCUMULATOR INTEGRITY TO NORMAL.

EVENTS: 1) LER 06-01-88-010

SI 11

LICENSEE EVENT REPORT (LER)

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Title (4)

SAFETY INJECTION ACCUMULATOR FILL LINE BREAK DUE TO FATIGUE REQUIRING PLANT SHUTDOWN PER TECHNICAL SPECIFICATIONS

Event Date (5) Month	Day	Year	LER Number (6) Year		Sequential Number		Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
1 1	0 5	8 8	8 8	—	0 1 0	—	0 0	1 2	0 2	8 8	NONE	0 5 0 0 0
												0 5 0 0 0

OPERATING MODE (9): 2

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	__ 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	X 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name		TELEPHONE NUMBER AREA CODE	
Tim Tulon, Assistant Superintendent Operating	Extension 2213	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	B Q	P S P	X 9 9 9	N					

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
__ Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

On November 5, 1988 Unit One was in the Startup Operational Mode at 1E-8 Amps in the intermediate range. Shortly after equalizing 1A and 1C Safety Injection Accumulator levels the licensed reactor operator observed lowering level and pressure in the 1A Accumulator. At 1100, the 1A Accumulator was declared inoperable. The operator acted to refill and repressurize the accumulator and the accumulator was declared operable at 1124. Personnel entered the Containment Building and identified a leak on the 1A Accumulator fill line, which could not be isolated from the accumulator. At 1137, the 1A Accumulator was declared inoperable and action was initiated to place Unit One in Hot Shutdown.

Visual examination of the cracked section of the 1A Accumulator fill line revealed outside diameter cracking, which had propagated circumferentially about 350 degrees around the pipe at the toe of the socket weld. The existence of beach marks (ripples on fracture surface) and the transgranular cracking mode of failure indicated fatigue induced cracking from the outside diameter to the inside diameter.

The 1B, 1C, and 1D Accumulator fill lines were dye penetrant inspected. The 1C and 1D indicated cracking. The 1A, 1C, and 1D flawed sections were removed and replaced with new sections of pipe. All four Accumulator fill lines have been provided with additional support to minimize vibrction of the lines, which should preclude the fatigue failure. Periodic dye penetrant examinations will continue until the success of preventive actions is verified.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION							Form Rev 2.0
FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					Page (3)
		Year	///	Sequential Number	///	Revision Number	
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TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 11/05/88 / 1137

Unit 1 MODE 2 - Startup Rx Power 1E-8 Amps RCS [AB] Temperature/Pressure 557°F / 2235 PSIG

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable prior to this event that contributed to its severity. On November 5, 1988 Unit One was in the Startup Operational Mode (Mode 2) at 1E-8 Amps in the intermediate range conducting low power physics testing following a recent refueling. At 1055 the Nuclear Station Operator (NSO) (Licensed Reactor Operator) equalized 1A and 1C Safety Injection Accumulator [BQ] levels in accordance with the "Lowering SI Accumulator level by Equalizing Level with a Lower Pressure Accumulator Operating Procedure" (BOP SI-7). At 1058 the NSO observed that 1A Accumulator level and pressure were decreasing, even though the level equalizing evolution had been completed. At 1100, both level and pressure in the 1A Accumulator dropped below the operability requirements of Technical Specification Limiting Condition for Operation 3.5.1 and the applicable Action Requirement was initiated. At 1102, the "Containment Drain Leak Detection Flow High" annunciator alarmed in the Main Control Room due to high floor drain flow. The NSO implemented the applicable annunciator response procedure and determined that the leakage source was a support system and not the Reactor Coolant System [AB]. At 1105 the NSO aligned the Safety Injection System to fill the 1A Accumulator from the Refueling Water Storage Tank and started the 1A Safety Injection Pump. By 1124, 1A Accumulator level and pressure were restored to specification and at 1130, the Action Requirement was exited. At 1135 the NSO stopped the 1A Safety Injection Pump. A Shift Foreman (Licensed Senior Reactor Operator) and an Equipment Attendant (Non-Licensed Operator) entered the Containment Building to identify the leak location. The Shift Foreman reported that the leakage was from a break in the 1A Accumulator fill line and that the break location could not be isolated from the accumulator. Based on this information, the 1A Accumulator was declared inoperable at 1137. Technical Specification 3.5.1 Action Requirements are to restore the inoperable Accumulator to operable status within one hour or be in at least Hot Standby within six hours and in Hot Shutdown within the following six hours. Since the 1A Accumulator could not be restored to an operable status, a plant shutdown was initiated at 1158 and an Unusual Event was declared. At 1214, the Unit entered Hot Standby (Mode 3). An Event Notification System telephone call was made to the Nuclear Regulatory Commission (NRC) to report the shutdown required by Technical Specifications and the declaration of an Unusual Event at 1215. At 1300, a plant cooldown and depressurization was initiated and at 1746, the Unit entered Hot Shutdown (Mode 4). The Unusual Event was terminated and appropriate telephone notifications were made to that effect. The plant shutdown was a controlled evolution and therefore stable conditions were maintained throughout the event. This Licensee Event Report (LER) is submitted pursuant to 10CFR50.73 (a)(2)(i)(A) due to the completion of a plant shutdown required by the plant's Technical Specifications.

C. CAUSE OF EVENT:

Visual examination of the cracked section of the 1A Accumulator fill line revealed outside diameter cracking, which had propagated circumferentially approximately 350 degrees around the pipe at the toe of the socket weld. Close visual examination of one of the fracture surfaces revealed faint beach marks (ripples on fracture surface), which indicate fatigue crack propagation. Study of the beach marks determined that the cracking had initiated on the pipe's outside diameter surface at the toe of the socket weld where the fill line connects to the accumulator. Metallographic examination of the fracture surface revealed a transgranular crack which propagated from the weld toe on the pipe outside diameter through the wall to the inside diameter. The microstructure of the pipe material and weld were typical of 304 stainless steel. Based upon the transgranular mode of crack propagation and the existence of beach marks on the fracture surface, the cause of the line failure is attributed to fatigue induced cracking. It is postulated that the line was subject to vibration, resulting in reverse bending loads which induced cyclic stresses at the socket weld. These stresses initiated fatigue cracks at the weld toe where the stress concentration is high.

D. SAFETY ANALYSIS:

The operability of four Safety Injection Accumulators ensures that a sufficient volume of borated water will be immediately forced into the core through each of the Reactor Coolant System (RCS) cold legs in the event RCS pressure falls below accumulator pressure. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures. Throughout this event the active functioning of the accumulators was unnecessary, since no large RCS pipe ruptures occurred. Therefore, neither plant nor public safety were literally affected.

In the event of an actual large RCS pipe rupture concurrent with the failure of an additional accumulator, unacceptable peak fuel cladding temperatures may have resulted. Cladding damage, however, is unlikely due to the very low power history of the core, which contained approximately one-third new fuel and two-thirds previously irradiated fuel that had been subcritical for over two months. Additionally three of the four accumulators remained operable and all other Emergency Core Cooling Systems were operable.

E. CORRECTIVE ACTIONS:

Following the plant shutdown, the fill lines for the 1B, 1C and 1D Accumulators were dye penetrant inspected for cracking in the vicinity of the socket weld. The magnitude of the 1A Accumulator fill line crack was sufficient to permit detection by visual examination. The dye penetrant testing revealed surface cracks in the 1C and 1D Accumulator fill lines. No indications of cracking were detected on the 1B Accumulator fill line. The flawed sections of piping were removed and replaced with new sections of pipe. The flawed sections were sent to Commonwealth Edison's System Materials Analysis Department (SMAD) for visual examinations and metallographic analyses. Results for the 1A Accumulator are reported in Section C of this LER. The 1C fill line crack was not found during visual examination. The 1D fill line crack was located at the toe of the socket weld and had propagated approximately 25 percent through the wall thickness at the 12 O'clock position. The crack was transgranular in nature.

All four accumulator fill lines have been provided with additional support to minimize vibration of the lines, which should preclude fatigue failure. Periodic dye penetrant examinations of the fill lines had been conducted since April 1988, as recommended by Commonwealth Edison's Pressurized Water Reactor Engineering Department, following the occurrence of a similar event at Braidwood Station. The examinations were not conducted during the Unit 1 refueling outage. Therefore the cracking was not detected prior to fracture. Examinations will continue until corrective actions (provision for additional support) have been deemed adequate in preventing fatigue failure. The examinations have been scheduled on the Byron Station General Surveillance program, which ensures their completion.

F. PREVIOUS OCCURRENCES:

No LER's have been written in the past to document Safety Injection Accumulator fill line cracking, because none of the 10CFR 50.73 reportability criteria have been met. However, accumulator fill lines have cracked at Byron and are documented in Deviation Reports (DVR), which are internal Commonwealth Edison documents:

DVR NUMBER	TITLE
6-1-88-057 (Unit 1)	"1C and 1D Accumulator Cracked Fill Lines Detected By Dye Penetrant Test on April 8, 1988".
6-2-86-018 (Unit 2)	"2D Accumulator Fill Line Crack Detected By Lowering Water Level Due to Leakage on December 18, 1986".

The failure mechanism in the previous events is the same as the mechanism described in this event. Preventive actions implemented had been effective in preventing recurrence of the failure when the plant operated in modes requiring accumulator operability. The suspension of inspections during the refueling outage resulted in a failure to identify fill line cracking prior to failure. Preventive actions implemented in response to this event are expected to be successful.

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE
Not Available	1-inch ANSI Schedule 40 Type 304 Stainless Steel Pipe

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

SW01	SX PUMP FAILS TO START/TRIP
SW02	SX BREAK INSIDE CONTAINMENT
SW03	LOSS OF SX COOLING TO D/G
SW04	SX DISCHARGE HEADER BREAK
SW05	WS HEADER BREAK
SW06	WS PUMP FAILS TO START/TRIP

SW01 SX PUMP FAILS TO START/TRIP

TYPE: GENERIC, RB

A)	1A SX PUMP	1SX01PA
B)	1B SX PUMP	1SX01PB

CAUSE: FAULTY OVERCURRENT (450/451) RELAY ACTUATION

REF: 20E-1-4030 SX01
20E-1-4030 SX02

PLT STA: SELECTED SX PUMP IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED ESSENTIAL SERVICE WATER PUMP BREAKER TO TRIP. PUMP CURRENT INDICATION DECREASES TO ZERO. ANNUNCIATOR 2-A1 "SX PUMP TRIP" ACTUATES, AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. ESSENTIAL SERVICE WATER HEADER PRESSURES, AS INDICATED ON 1PI-SX007/008 (1PM06J), DECREASES TO APPROX. 30 PSIG. ANNUNCIATOR 2-A2 "SX PUMP DSCH PRESS LOW" ACTUATES. SYSTEMS COOLED BY THE SX SYSTEM WILL RESPOND ACCURATELY TO THE LOSS OF SX COOLING.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND THE TRIP LIGHT BY PLACING THE CONTROL SWITCH IN THE TRIP POSITION. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL CLOSE THEN IMMEDIATELY TRIP OPEN.

MALFUNCTION REMOVAL RESTORES THE OVERCURRENT RELAY TO NORMAL OPERATION.

EVENTS: NONE

SW02 SX BREAK INSIDE CONTAINMENT

TYPE: GENERIC, RV 0-30,000 GPM @ 90 PSID

A) A TRAIN
B) B TRAIN

CAUSE: PIPE BREAK DOWNSTREAM OF 1SX016

REF: M-42 SHEETS 5 & 5A

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED SX TRAIN PIPING BREAK INSIDE CONTAINMENT AT THE SELECTED SEVERITY. THIS IN TURN CAUSES A LOSS OF MASS FROM THE SX SYSTEM AT A RATE DEPENDENT ON THE SEVERITY LEVEL. THE MASS LOSS RESULTS IN A DECREASE IN SX SYSTEM PRESSURE AS INDICATED ON 1PI-SX007/008. ALL SX SYSTEM COOLING LOADS WILL BEGIN TO HEAT UP DUE TO A REDUCTION IN SX COOLING. CONTAINMENT TEMPERATURES AND SUMP LEVELS WILL INCREASE AS THE SX WATER IS LOST FROM THE RCFC'S TO THE CONTAINMENT SUMP. AS SYSTEM PRESSURE DECREASES, ANNUNCIATOR 2-A2 "SX PUMP DSCH HDR PRESS LOW" WILL ACTUATE.

THE OPERATOR CAN MITIGATE THE EFFECTS BY CLOSING THE ASSOCIATED SX016A/B & SX027A/B VALVES TO ISOLATE THE LEAK.

MALFUNCTION REMOVAL RESTORES THE SX PIPE INTEGRITY.

EVENTS: NONE

SW03 LOSS OF SX COOLING TO D/G

TYPE: GENERIC, RB

A) 1SX169A
B) 1SX169B

CAUSE: SOLENOID REMAINS ENERGIZED IN AUTO & OPEN

REF: 20E-1-4030 SX17
M-42 SHEET 3

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED DIESEL SX INLET VALVE TO REMAIN CLOSED. THERE ARE NO INITIAL EFFECTS UNTIL THE ASSOCIATED DIESEL GENERATOR IS STARTED. AT >280 RPM, THE SX INLET VALVE DOES NOT OPEN, CAUSING THE DIESEL TO HEAT UP.

IF THE D/G IS EMERGENCY STARTED AND LOADED ON THE ESF BUS, THE D/G WILL SEIZE IN APPROXIMATELY 40-50 MINUTES DEPENDING ON THE D/G LOADING. AT 4000 KW LOADING, D/G SEIZURE OCCURS IN APPROXIMATELY 33 MINUTES.

IF THE D/G IS TEST STARTED, THE D/G WILL TRIP AT A JACKET WATER TEMP OF 205_F (APPROXIMATELY 21 MINUTES AT 4000 KW).

MALFUNCTION REMOVAL RESTORES THE DIESEL GENERATOR SX INLET VALVE TO NORMAL.

EVENTS: NONE

SW04 SX DISCHARGE HEADER BREAK

TYPE: DISCRETE, RV 0-30,000 GPM AT 90 PSID

CAUSE: PIPING FAILURE ON 36" LINE, 1SX13A, BETWEEN VALVES 1SX033 AND 1SX034

REF: M-42 SHEET 1A & 1B

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A PIPING FAILURE ON THE ESSENTIAL SERVICE (SX) PUMP DISCHARGE HEADER. THIS IN TURN CAUSES A LOSS OF MASS FROM THE SX SYSTEM AT A RATE DEPENDENT ON THE SEVERITY LEVEL. THE MASS LOSS RESULTS IN A DECREASE IN SX SYSTEM PRESSURE AS INDICATED ON 1PI-SX007/008. ALL SX SYSTEM COOLING LOADS WILL BEGIN TO HEAT UP DUE TO A REDUCTION IN SX COOLING. AS SYSTEM PRESSURE DECREASES, ANNUNCIATOR 2-A2 "SX PUMP DSCH HDR PRESS LOW" WILL ACTUATE.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY ISOLATING THE LEAK. CLOSING THE SX TRAIN CROSSTIE VALVES, 1SX033/034, WILL ISOLATE THE LEAK.

MALFUNCTION REMOVAL WILL RESTORE THE SX DISCHARGE PIPING INTEGRITY.

EVENTS: NONE

SW05 WS HEADER BREAK

TYPE: DISCRETE, RV 0-100,000 GPM @ 100 PSID

CAUSE: PIPE BREAK IMMEDIATELY UPSTREAM 1WS137

REF: M-43 SHEET 8

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES MASS TO BE LOST FROM THE WS COMMON HEADER. THE RATE AT WHICH MASS IS LOST IS DEPENDENT UPON MALFUNCTION SEVERITY. WS DISCHARGE PRESSURE DECREASES AND THE RUNNING WS PUMP AMPS INCREASE AS THE FLOW INCREASES. AS THE WS HEADER PRESSURE DECREASES, THE STANDBY WS PUMP AUTO STARTS AND ANNUNCIATOR 38-A10 "WS PUMP TRIP OR AUTO START" ACTUATES. ANNUNCIATOR 38-B10 "WS HDR PRESS LOW" ACTUATES. ALL COMPONENTS COOLED BY WS WILL HEATUP.

MALFUNCTION REMOVAL RESTORES THE PIPE INTEGRITY TO NORMAL

EVENTS: NONE

SW06 WS PUMP FAILS TO START/TRIP

TYPE: GENERIC, RB

A) 0WS01PA
B) 0WS01PB
C) 0WS01PC

CAUSE: FAULTY OVERCURRENT (450/451) RELAY ACTUATION

REF: 20E-0-4030-WS01
20E-0-4030-WS02
20E-0-4030-WS03

PLT STA: SELECTED SERVICE WATER PUMP IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED NON-ESSENTIAL SERVICE WATER PUMP BREAKER TO TRIP. PUMP CURRENT INDICATION DECREASES TO ZERO. ANNUNCIATORS 38-A10 "WS PUMP TRIP OR AUTO START" AND 38-B10 "WS HDR PRESS LOW" ACTUATE, AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. NON-ESSENTIAL SERVICE WATER HEADER PRESSURES, AS INDICATED ON 0PI-WS008 (0PM01J), DECREASES UNTIL THE STBY WS PUMP AUTO STARTS. WS HEADER PRESSURE WILL THEN RETURN TO NORMAL.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND THE TRIP LIGHT BY PLACING THE CONTROL SWITCH IN THE TRIP POSITION. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL CLOSE THEN IMMEDIATELY TRIP OPEN.

MALFUNCTION REMOVAL RESTORES THE OVERCURRENT RELAY TO NORMAL OPERATION.

EVENTS: NONE.

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

TC01	INADVERTENT TURBINE RUNBACK
TC02	TURBINE TRIP ON LOW LOAD INDICATION (PDS-TO071)
TC03	TURBINE AUTO TRIP FAILURE
TC04	TURBINE AUTO RUNBACK FAILURE
TC05	OPC - LP TURB INLET PRESS SENSOR (PT-MS003) FAILURE
TC06	DEHC - IMP PRESS TRANSMITTER (PT-MS002) FAILURE
TC07	DEHC - MW TRANSDUCER FAILURE
TC08	DEHC - GV/TV OSCILLATION - TIME
TC09	DEHC - GV/TV OSCILLATION - MAGNITUDE
TC10	LOSS OF DEHC SPEED CONTROL CHANNEL(S)
TC11	LOSS OF DEHC SUPERVISORY SPEED CHANNEL
TC12	EHC PILOT OPERATED IA VALVE FAILS (1EH-5042)
TC13	TV SERVO FAILURE - VALVE FAILS
TC14	GV SERVO FAILURE - VALVE FAILS
TC15	EH SYSTEM LEAK
TC16	GOVERNOR VALVES NOT TRACKING AUTO
TC17	EH PUMP FAILS TO START/TRIP
TC18	INADVERTENT OTΔT TURBINE RUNBACK

TC01 INADVERTENT TURBINE RUNBACK

TYPE: DISCRETE, RV 0-1175 MW

CAUSE: TURB LOAD REF REDUCTION CONTACT FAILS CLOSED

REF: 20E-1 4030 MS17
DEHC SYSTEM DESCRIPTION

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A TURBINE RUNBACK AT 200%/MIN TO THE SELECTED LOAD. THE LOSS OF LOAD INCREASES S/G PRESSURES, RCS TEMPERATURE AND CAUSES THE ROD CONTROL SYSTEM TO STEP RODS IN. THE "RUNBACK OPER" LIGHT ON DEHC IS LIT WHILE RUNBACK IS ACTIVE, THEN EXTINGUISHES. THE TURBINE LOAD DECREASE IS INDICATED ON THE DIGITAL AND DEH REFERENCE, REFERENCE DEMAND INDICATORS, AND THE MW RECORDER. IF THE TURBINE IS RUNBACK TO <60 MW (5%) THE GENERATOR WILL TRIP ON REVERSE POWER/ANTI-MOTORING.

MALFUNCTION REMOVAL RESTORES THE FAULTY RUNBACK CIRCUIT TO NORMAL.

EVENTS: NONE.

TC02 TURBINE TRIP ON LOW LOAD INDICATION (PDS-TO071)

TYPE: DISCRETE, RB

CAUSE: FAULTY 1PDS-TO071 CONTACT IN 63TDR RELAY CIRCUIT

REF: M-2035 SHEET 8
20E-1-4030 TO09
20E-1-4030 MP02

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES NO IMMEDIATE NOTICEABLE EFFECT. 60 SECONDS LATER THE TURBINE TRIPS DUE TO A FAULTY-DIFFERENTIAL PRESSURE CONTACT IN THE ANTI-MOTORING CIRCUIT. ANNUNCIATOR 19-B2 "TURBINE MOTORING GEN TRIP" ACTUATES.

MALFUNCTION REMOVAL RESTORES THE FAULTY CONTACT TO NORMAL.

EVENTS: 1) LER 06-02-87-005

LICENSEE EVENT REPORT (LER)

TC02

Facility Name (1): Byron, Unit 2

Docket Number (2): 0|5|0|0|0|4|5|5

Page (3): 0|1 of 0|1

Title (4): REACTOR TRIP FROM TURBINE TRIP ABOVE 10% POWER DUE TO A SPURIOUS MAIN GENERATOR MOTORING SIGNAL WITH AN UNKNOWN CAUSE

Event Date (5)			LER Number (6)			Report Date (7)			Other Facilities Involved (8)	
Month	Day	Year	Year	Sequential Number	Revision Number	Month	Day	Year	Facility Names	Docket Number(s)
0|3	3|1	8|7	8|7	0| 0| 5	0|0	0|4	2|9	8|7	NONE	0|5|0|0|0| |
										0|5|0|0|0| |

OPERATING MODE (9): 1

POWER LEVEL (10): 0 | 1 | 7

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

___ 20.402(b)	___ 20.405(c)	X 50.73(a)(2)(iv)	___ 73.71(b)
___ 20.405(a)(1)(i)	___ 50.36(c)(1)	___ 50.73(a)(2)(v)	___ 73.71(c)
___ 20.405(a)(1)(ii)	___ 50.36(c)(2)	___ 50.73(a)(2)(vii)	___ Other (Specify in Abstract below and in Text)
___ 20.405(a)(1)(iii)	___ 50.73(a)(2)(i)	___ 50.73(a)(2)(viii)(A)	
___ 20.405(a)(1)(iv)	___ 50.73(a)(2)(ii)	___ 50.73(a)(2)(viii)(B)	
___ 20.405(a)(1)(v)	___ 50.73(a)(2)(iii)	___ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name: Tom Joyce, Assistant Superintendent Technical Services Ext. 2214

TELEPHONE NUMBER: AREA CODE 8|1|5 | 2|3|4|-|5|4|4|1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)

___ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15): Month | Day | Year

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

The Unit 2 reactor tripped due to a spurious actuation of the Turbine Motoring Generator Trip circuitry. The root cause of this event could not be determined. The calibration of the differential pressure switch which feeds the turbine motoring generator trip circuitry was checked and found to be out of tolerance. however, the direction of the out of tolerance would have resulted in a trip at a lower than normal power level rather than the higher than normal level at which this trip occurred. The pressure switch was replaced, nevertheless. The pressure switch was checked during the Unit Startup and no abnormal functioning was detected. There have been no previous occurrences of this type.

(1358M/0157M)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Byron, Unit 2	0 5 0 0 0 4 5 5	8 7	- 0 0 5 -	0 0	0 2 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

UNIT 2 Event Date/Time 03/31/87/0603
MODE 1 - Power Operation Rx Power 17% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On March 31, 1987, at 0603, the Unit 2 reactor tripped. Prior to the trip, the unit was being shut down for a planned outage. All parameters were nominal. Reactor power was at 17% and the Turbine [TB] load was at 135 megawatts (MW) decreasing to 125 MW at 3 MW per minute. The first out annunciator indication was Turbine Motoring Generator Trip. The Reactor, the Turbine, and the Generator tripped as required. Auxiliary Feedwater [BA] started due to low steam generator [SB] level.

All safeguard systems operated as designed. Licensed Operators recovered from the event using the appropriate emergency procedures. A Reactor Protection Actuation and an Engineered-Safeguard Actuation are reportable pursuant to 10CFR 50.73(a)(2)(iv).

C. CAUSE OF EVENT:

The intermediate cause of the trip was a spurious actuation of the Turbine Motoring Generator Trip circuitry. The root cause of this event could not be determined. The Turbine Motoring Generator Trip is fed from a differential pressure switch which senses the pressure difference across the high pressure turbine. When the differential pressure drops to 16.4 pounds per square inch delta (psid) for one minute the result is a generator trip. Instrument Maintenance personnel verified the calibration of the differential pressure switch and found it to be out of tolerance. However, this out of tolerance condition would have resulted in the anti-motoring trip at a lower than normal power, not the abnormally high power level at which this trip occurred. No problems were found with the time delay relay associated with the Turbine Moturing Generator Trip circuitry.

D. SAFETY ANALYSIS:

All Safeguard equipment functioned as designed which resulted in a safe shutdown of the reactor. The Auxiliary Feedwater pumps started and provided water to maintain steam generator levels. The safety consequences would be the same had this event occurred under any different credible set of initial conditions.

E. CORRECTIVE ACTIONS:

The Turbine Motoring Generator Trip differential pressure switch was replaced. Temporary instrumentation was installed to monitor the output of the Turbine Motoring Generator Trip circuitry during the startup synchronization, following the trip. The pressure differential switch worked as designed. No further corrective action is planned at this time.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

F. PREVIOUS OCCURRENCES:

LER NUMBER	TITLE
NONE	

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
No Components actually failed			

b) RESULTS OF NPRDS SEARCH:

Not Applicable

TC03 TURBINE AUTO TRIP FAILURE

TYPE: DISCRETE, RB

CAUSE: MULTIPLE ELECTRICAL/MECHANICAL FAILURES (20ET AND 20-2/AST FAIL)

REF: DEHC SYSTEM DESCRIPTION

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES NO IMMEDIATE NOTICEABLE EFFECT. WHEN THE TURBINE RECEIVES A SIGNAL TO 20ET OR 20-2/AST WHICH WOULD NORMALLY RESULT IN AN AUTOMATIC TURBINE TRIP, THE MAIN TURBINE DOES NOT TRIP. WHEN THE OPERATOR ATTEMPTS TO MANUALLY TRIP THE TURBINE, THE TURBINE WILL TRIP. ALL TRIPS OFF THE PROTECTION BLOCK (CND. VACUUM, THRUST BEARING, BEARING OIL PRESSURE) WILL STILL FUNCTION.

IF USED AT HIGH POWER, A MSLI AND SI ON LOW STEAM LINE PRESSURE MAY RESULT AFTER A RX TRIP IS INITIATED.

MALFUNCTION REMOVAL RESTORES THE AUTOMATIC TURBINE TRIP TO NORMAL.

EVENTS: 1) OE 3729

OE 3729 I MOFFATT (FPC) 27-DEC-89 09:30 EST

TC 03

Subject: REACTOR TRIP WITHOUT MAIN TURBINE TRIP

SUBJECT: REACTOR TRIP WITHOUT MAIN TURBINE TRIP

Operating Experience Report

UNIT..........................CRYSTAL RIVER UNIT 3
DOC NO/LER NO.................5000302/88-006
EVENT DATE....................Feb. 28, 1988
NSSS/A-E......................B&W/GILBERT
RATING........................2544 MW THERMAL
DATE OF COMMERCIAL OPERATION..MARCH 13, 1977

EVENT DESCRIPTION:

On Sunday Feb 28, 1988 during a power reduction from 99% to 40 %, Crystal River 3 experienced a reactor trip caused by a feedwater transient. At approximately 45% power the main feedwater block valves began to close as designed. After receiving the closed indication on both main feedwater block valves (MFBV), a feedwater transient developed. Feedwater flow to the "B" steam generator (SG) began to increase and remained excessive for approximately one minute. The Reactor Coolant System (RCS) experienced a cooldown and depressurization, and reactor power increased to approximately 56% due to the excessive cooling.

Operator manipulations of the Feedwater valves resulted in thFBVs going closed resulting in an underfeed condition to the SGs. Approximately one minute later, a reactor trip occurred due to high RCS pressure.

The Main Turbine (MT) failed to trip when the reactor tripped and operators tried subsequently to manually trip it from the Main Control Board but were unsuccessful on several attempts. They manually opened the generator output breakers and closed the main steam isolation valves. Minutes later the MT was manually tripped from the local trip mechanism. Emergency Feedwater (EFW) actuated on low level in the "B" SG. Secondary steam pressure control was accomplished manually with the atmospheric dump valves and steam safety valves.

CAUSE:

The feedwater transient was caused by a broken stem nut on the "B" MFBV. The Identified cause of the nut failure is cyclic fatigue. The valve was manufactured by Crane and was an "18X16X18 1-900 U pressure seal gate valve". The nut was originally fabricated of material identified by B&W as B505 C 93200. It has been replaced with a nut made of B584 C86300 and which has a different design. The new design reduces the stress concentration factor by 76% of the original nut.

The MT failure to trip was caused by a faulty turbine trip solenoid. It was manufactured by Westinghouse and was Part No. 439a936601.

CORRECTIVE ACTION:

A nut of new design and material was installed on the MFBVs. The MT solenoid was replaced with a new solenoid. The solenoid was evaluated and a 10 CFR PART 21 report was made. Improved requirements were imposed for solenoid installation.

INFORMATION CONTACT: LARRY MOFFATT (904) 795-6486 EXT 4390

TC04 TURBINE AUTO RUNBACK FAILURE

TYPE: DISCRETE, RB

CAUSE: FAULTY RUNBACK SIGNAL (DEHC DOES NOT PROCESS RUNBACK SIGNAL)

REF: DEHC SYSTEM DESCRIPTION

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES NO IMMEDIATE NOTICEABLE EFFECT. WHEN A RUNBACK CONDITION DEVELOPS, (3% BELOW EITHER THE OPΔT OR OTΔT TRIP SETPOINT) THE TURBINE DOES NOT RUNBACK. AUTOMATIC AND MANUAL ROD WITHDRAWAL WILL NOT OCCUR IF A OPΔT OR OTΔT ROD STOP IS ACTIVE. THE REACTOR WILL EVENTUALLY TRIP DUE TO AN OPΔT OR OTΔT TRIP.

MALFUNCTION REMOVAL RESTORES THE RUNBACK SIGNAL TO NORMAL.

EVENTS: NONE.

TC05 OPC - LP TURB INLET PRESS SENSOR (PT-MS003) FAILURE

TYPE: DISCRETE, RV 0-105 PSIG

CAUSE: TRANSMITTER FAILURE

REF: M-2035 SHEET 9
DEHC SYSTEM DESCRIPTION
TECH MANUAL F-317, FIGURE 3-7

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION AT THE LOWEST SEVERITY CAUSES PT-MS003 TO FAIL TO 0 PSIG. "OPC PRESS TRANSD MONITOR" LIGHT ON.THE "DEH TURBINE CONTROL" PANEL LIGHTS WHEN THIS SIGNAL IS NOT WITHIN THE PRESCRIBED LIMITS.

MALFUNCTION REMOVAL RESTORES THE FAILED TRANSMITTER TO NORMAL.

EVENTS: NONE.

TC06 DEHC - IMPULSE PRESS TRANSMITTER (PT-MS002) FAILURE

TYPE: DISCRETE, RV 0-700 PSIG

CAUSE: TRANSMITTER FAILURE

REF: M-2035 SHEET 8
DEHC SYSTEM DESCRIPTION

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE INPUT TO THE IMPULSE PRESSURE FEEDBACK LOOP OF DEHC TO FAIL TO THE SELECTED SEVERITY. THE "IMP PRESS TRANSD MONITOR" LIGHT ON THE "DEH TURBINE CONTROL" PANEL LIGHTS WHEN THIS SIGNAL IS NOT WITHIN THE PRESCRIBED LIMITS. COMPENSATION FOR STEAM INLET PRESSURE CHANGES WILL BE INACCURATE RESULTING IN LARGE VARIATIONS IN GENERATOR MEGAWATT OUTPUT. THE IMPULSE FEEDBACK LOOP IS AUTOMATICALLY KICKED OUT ON LARGE ERRORS. THE "LED" MW INDICATION IS AFFECTED, BUT THE MW METER AND RECORDER STILL FUNCTION PROPERLY.

MALFUNCTION REMOVAL RESTORES THE FAILED TRANSMITTER TO NORMAL.

EVENTS: NONE.

TC07 DEHC - MW TRANSDUCER FAILURE

TYPE: DISCRETE, RV 0-1250 MW

CAUSE: TRANSDUCER (DEV WX- EHC) FAILURE

REF: 20E-1-4015A
DEHC SYSTEM DESCRIPTION

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION AT THE LOWEST SEVERITY CAUSES A FAILURE OF THE MW TRANSDUCER TO 0%. "MW TRANSD MONITOR" LIGHT ON THE "DEH TURBINE CONTROL" PANEL LIGHTS WHEN THIS SIGNAL IS NOT WITHIN THE PRESCRIBED LIMITS. THE MW FEEDBACK LOOP IS AUTOMATICALLY KICKED OUT ON LARGE ERRORS. THE "LED" MW INDICATION WILL FOLLOW THE MALFUNCTION SEVERITY.

MALFUNCTION REMOVAL RESTORES THE FAILED TRANSDUCER TO NORMAL.

EVENTS: NONE.

TC08 DEHC - GV/TV OSCILLATION - TIME

TYPE: GENERIC, RV 0-200 SECONDS

A)	GV1	E)	TV1
B)	GV2	F)	TV2
C)	GV3	G)	TV3
D)	GV4	H)	TV4

CAUSE: FAULTY SIGNAL TO SERVO (DEFAULT MAGNITUDE 10%)

REF: DEHC SYSTEM DESCRIPTION

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED GV OR TV TO OSCILLATE. THE TIME REQUIRED TO COMPLETE AN OSCILLATION WILL BE DETERMINED BY THE SELECTED SEVERITY. THE HIGHER THE SELECTED SEVERITY, THE LONGER IT WILL TAKE TO COMPLETE ONE OSCILLATION. THE VALVE FLUCTUATION WILL BE DISPLAYED ON THE ASSOCIATED GOVERNOR VALVE POSITION RECORDER ON 1PM02J, AND THE LIGHTS ON THE DEH OPERATOR'S VALVE TEST PANEL. THE VALVE OSCILLATIONS WILL AFFECT GENERATOR MEGAWATT OUTPUT AND REACTOR COOLANT SYSTEM TEMPERATURE/PRESSURE DUE TO THE VARIED HEAT REMOVAL DURING THE COURSE OF THE OSCILLATION. IF THE TURBINE IS IN THE "IMP OUT" MODE (AT LOWER POWER LEVELS), ONLY THE SELECTED VALVE WILL OSCILLATE AND THE EFFECTS ON THE PLANT WILL BE MORE VISIBLE. IN THE "IMP IN" MODE, SOME OF THE OTHER GOVERNOR VALVES WILL MODULATE IN THE OPPOSITE DIRECTION TO ATTEMPT TO MAINTAIN A CONSTANT IMPULSE CHAMBER PRESSURE, THUS LIMITING THE OVERALL PLANT EFFECTS OF THE VALVE FAILURE.

THIS MALFUNCTION IN COMBINATION WITH TC09 (MAGNITUDE) WILL CAUSE THE DESIRED EFFECTS.

MALFUNCTION REMOVAL RESTORES THE FAILED SERVO SIGNAL TO NORMAL.

EVENTS: NONE.

TC09 DEHC - GV/TV OSCILLATION - MAGNITUDE

TYPE: GENERIC, RV 0-100% VALVE SWING

A)	GV1	E)	TV1
B)	GV2	F)	TV2
C)	GV3	G)	TV3
D)	GV4	H)	TV4

CAUSE: FAULTY SERVO VALVE (DEFAULT TIME 30 SECONDS)

REF: DEHC SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED VALVE WILL BEGIN TO OSCILLATE. THE MAGNITUDE OF THE OSCILLATION WILL BE DETERMINED BY THE SELECTED SEVERITY. THE HIGHER THE SELECTED SEVERITY, THE GREATER THE OSCILLATION IN VALVE POSITION. THE VALVE FLUCTUATION WILL BE DISPLAYED ON THE ASSOCIATED VALVE POSITION METER ON 1PM02J. THE VALVE OSCILLATIONS WILL AFFECT GENERATOR MEGAWATT OUTPUT AND REACTOR COOLANT SYSTEM TEMPERATURE/PRESSURE DUE TO THE VARIED HEAT REMOVAL DURING THE COURSE OF THE OSCILLATION. IF THE TURBINE IS IN THE "IMP OUT" MODE, ONLY THE SELECTED VALVE WILL OSCILLATE AND THE EFFECTS ON THE PLANT WILL BE MORE VISIBLE. IN THE "IMP IN" MODE, THE OTHER VALVES WILL MODULATE IN THE OPPOSITE DIRECTION TO ATTEMPT TO MAINTAIN A CONSTANT IMPULSE CHAMBER PRESSURE, THUS LIMITING THE OVERALL PLANT EFFECTS OF THE VALVE FAILURE. AT HIGHER SEVERITIES, THE EFFECTS ON THE PRIMARY PLANT MAY BE SUFFICIENT TO CAUSE A REACTOR/PLANT TRIP.

IF TC08 MALFUNCTION IS ALSO SET THEN THE OSCILLATION WILL BE AT THE TIME SELECTED.

MALFUNCTION REMOVAL WILL RESTORE THE FAILED SERVO VALVE TO N´ MAL.

EVENTS: NONE.

TC10 LOSS OF DEHC SPEED CONTROL CHANNEL(S)

TYPE: GENERIC, RB

A) SPEED PICKUP A
B) SPEED PICKUP B

CAUSE: SPEED PICKUP FAILURE (FAILS TO 0 RPM)

REF: DEHC SYSTEM DESCRIPTION

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED SPEED PICKUP CHANNEL TO FAIL LOW. THE "CONTROL SPEED CHANNEL MONITOR" LAMP ILLUMINATES ON THE DEHC TURBINE CONTROL PANEL. THIS MEANS THAT THE COMPUTER HAS DETERMINED THAT CHANNEL UNRELIABLE. IF BOTH CHANNELS ARE AFFECTED THEN THE "CONTROL SPEED CHANNEL OUT" LAMP ILLUMINATES. DURING "SPEED CONTROL" THE DEHC WILL TRANSFER TO "TURBINE MANUAL" CONTROL, AND DURING "LOAD CONTROL" THE DEHC WILL TRANSFER TO "SPEED OUT" CONTROL. THE "OPC SPEED CHANNEL MONITOR" LAMP ILLUMINATES INDICATING AN UNRELIABLE CHANNEL TO THE OPC NETWORK.

THIS MALFUNCTION IN CONJUNCTION WITH A LOSS OF THE DEHC SUPERVISORY SPEED CHANNEL (MALF TC11) WILL CAUSE THE SAME EFFECTS AS ABOVE.

MALFUNCTION REMOVAL RESTORES THE FAILED SPEED PICKUP CHANNEL TO NORMAL.

EVENTS: NONE.

TC11 LOSS OF DEHC SUPERVISORY SPEED CHANNEL

TYPE: DISCRETE, RB

CAUSE: SUPERVISORY SPEED SIGNAL FAILURE (FAILS TO 0 RPM)

REF: DEHC SYSTEM DESCRIPTION

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SUPERVISORY SPEED CHANNEL TO FAIL LOW. THE "CONTROL SPEED CHANNEL MONITOR" LAMP ILLUMINATES ON THE DEHC TURBINE CONTROL PANEL. THIS MEANS THAT THE COMPUTER HAS DETERMINED THAT CHANNEL - UNRELIABLE. THE "OPC SPEED CHANNEL MONITOR" LAMP ILLUMINATES INDICATING AN UNRELIABLE CHANNEL TO THE OPC NETWORK.

THIS MALFUNCTION IN CONJUNCTION WITH A LOSS OF THE DEHC SPEED PICKUP CHANNEL (MALF TC10) WILL CAUSE THE SAME EFFECTS AS MENTIONED IN MALF TC10.

MALFUNCTION REMOVAL RESTORES THE FAILED SUPERVISORY SPEED CHANNEL TO NORMAL.

EVENTS: NONE.

TC12 EHC PILOT OPERATED IA VALVE FAILS (1EH-5042)

TYPE: DISCRETE, RB

CAUSE: MECHANICAL VALVE FAILURE

REF: EXTRACTION STEAM (ES) SYSTEM DESCRIPTION
DEHC SYSTEM DESCRIPTION

PLT STA: PRIOR TO MAIN GEN SYNCHRONIZATION

EFFECTS: IF THIS MALFUNCTION IS ACTIVATED, THE NON-RETURN CHECK VALVES WILL CLOSE AND THE VALVES WILL NOT BE ABLE TO BE OPENED. IF THIS MALFUNCTION IS USED ONCE THE MAIN GENERATOR IS ON-LINE, THE FLOW THROUGH THE NON-RETURN CHECK VALVES WILL KEEP THE VALVES OPEN.

MALFUNCTION REMOVAL WILL RESTORE THE FAILED VALVE TO NORMAL.

EVENTS: NONE.

TC13 TV SERVO FAILURE - VALVE FAILS

TYPE: GENERIC, RV 0-100%

A) TV1
B) TV2
C) TV3
D) TV4

CAUSE: SERVO FAILURE

REF: DEHC SYSTEM DESCRIPTION
ANNUNCIATOR 18-A4

PLT STA: TURBINE ON LINE

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED HP TURBINE THROTTLE VALVE TO FAIL. THE VALVE POSITION WILL BE DETERMINED BY THE SELECTED SEVERITY. VALVE POSITION IS INDICATED ON 1PM02J ON THE DEH TURBINE CONTROL VALVE TEST PANEL LIGHTS. IF THE SELECTED SEVERITY IS LESS THAN THE INITIAL VALVE POSITION, THE THROTTLING EFFECT OF THE THROTTLE VALVE CLOSURE WILL BE COUNTERED BY THE GOVERNOR VALVE(S) OPENING WHILE IN THE "IMP IN" MODE. IN THE "IMP OUT" MODE, THE THROTTLE VALVE CLOSURE WILL RESULT IN A SMALL DECREASE IN GENERATOR MEGAWATT OUTPUT. ANNUNCIATOR 18-A4 "TURB STOP VLV CLOSED ALERT" ACTUATES WHEN A TURBINE THROTTLE VALVE IS FULLY CLOSED. IF THE AFFECTED THROTTLE VALVE IS OPENED, WHEN IT WOULD NORMALLY BE CLOSED, THERE IS NO NOTICEABLE EFFECT OTHER THAN THE OPEN LAMP ILLUMINATING EXCEPT WHEN IN TURBINE SPEED CONTROL AN OVERSPEED CONDITION CAN OCCUR.

MALFUNCTION REMOVAL RESTORES THE FAILED HP TURBINE THROTTLE VALVE SERVO TO NORMAL.

EVENTS: NONE.

TC14 GV SERVO FAILURE - VALVE FAILS

TYPE: GENERIC, RV 0-100%

A) GV1 C) GV3
B) GV2 D) GV4

CAUSE: SERVO FAILURE

REF: DEHC SYSTEM DESCRIPTION

PLT STA: TURBINE ON LINE

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED GOVERNOR CONTROL VALVE TO FAIL. THE VALVE POSITION WILL BE DETERMINED BY THE SELECTED SEVERITY AND WILL BE INDICATED ON THE DEH TURBINE CONTROL VALVE TEST PANEL LIGHTS AT 1PM02J. IF THE SELECTED SEVERITY IS GREATER THAN THE INITIAL VALVE POSITION, THE AFFECTED GOVERNOR VALVE WILL OPEN. IF THE TURBINE CONTROL SYSTEM IS IN THE "IMP OUT" MODE, THE VALVE OPENING WILL RESULT IN AN INCREASED MEGAWATT OUTPUT AND DECREASED REACTOR COOLANT SYSTEM TEMPERATURE AND PRESSURE. IF THE TURBINE CONTROL SYSTEM IS IN THE "IMP IN" MODE, THE OPENING OF THE AFFECTED GOVERNOR WILL BE COUNTERED BY THE CLOSURE OF THE OTHER GOVERNOR CONTROL VALVES TO MAINTAIN A CONSTANT IMPULSE CHAMBER PRESSURE.

IF THE SELECTED SEVERITY IS LESS THAN THE INITIAL VALVE POSITION, THE AFFECTED GOVERNOR VALVE WILL CLOSE. IF THE TURBINE CONTROL SYSTEM IS IN THE "IMP-OUT" MODE, THE VALVE CLOSURE WILL RESULT IN AN DECREASED MEGAWATT OUTPUT AND INCREASED REACTOR COOLANT SYSTEM TEMPERATURE AND PRESSURE. IF THE TURBINE CONTROL SYSTEM IS IN THE "IMP IN" MODE, THE CLOSURE OF THE AFFECTED GOVERNOR WILL BE COUNTERED BY THE OPENING OF THE OTHER GOVERNOR CONTROL VALVES TO MAINTAIN A CONSTANT IMPULSE CHAMBER PRESSURE.

MALFUNCTION REMOVAL RESTORES THE FAILED GOVERNOR CONTROL VALVE SERVO TO NORMAL.

EVENTS: 1) DVR 06-02-88-011

DEVIATION INVESTIGATION REPORT

TC14

TITLE

FAILURE OF #3 GOVERNOR VALVE TO CLOSE

PAGE 1 OF 0 2

EVENT DATE			DIR NUMBER						REPORT DATE			OPERATING MODE	5
MONTH	DAY	YEAR	STA	UNIT	YEAR		SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	0 0 0
1 2	1 5	8 7	0 6	0 2	8 8	—	0 1 1 —	0 0	0 3	0 3	8 8		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER AREA CODE	
Fred Hornbeak, Senior Staff Engineer Ext. 2822	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
M2	T B	F C V	W 1 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED

		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
X YES (if yes, complete EXPECTED SUBMISSION DATE)	NO		0 6	0 1	8 9

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 12/15/87 / 0917

Unit 1 MODE 1 - Power Operations Rx Power 80% RCS [AB] Temperature/Pressure Normal Operating

Unit 2 MODE 5 - Cold Shutdown Rx Power 0 RCS [AB] Temperature/Pressure 178 / 340

B. DESCRIPTION OF EVENT:

On December 15, 1987, with Unit 2 in Mode 5 during the performance of a calibration of the governor valve Linear Variable Differential Transformer, governor valve number 3 stuck in the open position after being stroked open. After repeated efforts to stroke the valve full closed, the valve eventually was successfully stroked closed. A Nuclear Work Request (NWR) was written and the governor valve number 3 was replaced with a new valve. The new valve was successfully stroked by the Operational Analysis Department (OAD).

C. CAUSE OF EVENT:

An investigation into the cause of the event was conducted by station and Westinghouse personnel. The investigation showed that the valve plug was rotating while the valve was in the open position. The governor valve seal rings are designed to minimize steam leakage along the valve shaft. The rings move up and down as the valve closes and opens. Apparently the rings were in their highest position while the plug was rotating. The seal ring wore a groove into the plug at the top of the seal ring gap. When an attempt to close the valve was made the ring pressed against this groove and would not compress in its normal manner. As a result the seal ring bound against the valve cylinder, preventing valve closure.

TEXT

D. SAFETY ANALYSIS:

The Westinghouse probabilistic risk assessment group issued a 10CFR50.59 notification on the valve binding problem. The probability of an overspeed was still at an acceptable level (less than 1×10^{-5}). The event did not pose any danger to the health and safety of the plant or public as the unit was shut down with the turbine off line during the duration of the event. If the event had occurred at power with resultant turbine overspeed, the turbine would have tripped on overspeed (throttle stop valves would go closed) with a Reactor Trip and the plant response would have followed as designed and analyzed in the accident analysis. Thus, there would have been no effect on the health and safety of the public.

E. CORRECTIVE ACTIONS:

In conjunction with the 10CFR50.59 notification, Westinghouse provided corrective actions for the governor and throttle valves (Availability Improvement Bulletins 8714 and 8715). The bulletins provide various suggested modifications to the valves as well as the turbine steam chest. Station and Westinghouse personnel are currently discussing the proposed modifications to determine which, if any, will be implemented on both units. A supplemental report will be issued after action is taken.

In addition, per the 10CFR50.59 notification, the governor valves are now tested before the throttle valves. Also on Unit 2 the throttle valves were stroked closed and left closed for 5 minutes, stroked open and stroked closed and open again to verify no evidence of binding existed. The governor valves are currently being tested on a weekly basis on Unit 2.

The governor valve was replaced with a new valve from Westinghouse under a Nuclear Work Request (NWR). The new valve was stroked successfully by Operational Analysis Department. The damaged valve was sent to Westinghouse for analysis. Governor valves 1, 2 and 4 were stroked with no visible or audible problems.

F. PREVIOUS OCCURRENCES:

A turbine governor valve has not previously failed to close due to binding on either Byron unit.

DVR NUMBER	TITLE
NONE	

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Westinghouse	Control Valve	BB296	

b) RESULTS OF NPRDS SEARCH:

Not Applicable

c) NUCLEAR WORK REQUEST (NWR) SEARCH:

No previous NWR's were found that were written on binding of turbine governor valves.

TC15 EH SYSTEM LEAK

TYPE: DISCRETE, RV 0-50 GPM

CAUSE: PIPE LEAK ON THE EH SUPPLY HEADER

REF: 20E-1-4030 EH01 & EH02
DEHC SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE EH SUPPLY HEADER TO DEVELOP A LEAK AT THE SELECTED SEVERITY. THIS WILL CAUSE A DECREASING EH RESERVOIR LEVEL. AS EH RESERVOIR LEVEL DECREASES, ANNUNCIATORS 18-C15 "EH FLUID RSRVR LVL HIGH LOW", 18-D15 "EH FLUID RSRVR LVL LO-2", AND 18-A3 "EH RSRVR LEVEL LO-2 LOCKOUT TURB TRIP" ACTUATE. DECREASING EH HEADER PRESSURE MAY CAUSE AN AUTO START OF THE STANDBY PUMP, ANNUNCIATOR 18-B15 " EH SYS TROUBLE," WILL ACTUATE.

ON A LO-2 EH RESERVOIR LOCKOUT, BOTH EH PUMPS WILL TRIP.

LOW EH PRESSURE WILL GENERATE A TURB TRIP AND RX TRIP IF >P-8. IN ADDITION, BOTH TDFWP's WILL TRIP.

LEAKS ABOVE 35 GPM MAY CAUSE TURBINE VALVES TO DRIFT SHUT ON LOW EH HEADER PRESSURE AND/OR A TURBINE TRIP.

MALFUNCTION REMOVAL WILL RESTORE ONLY THE EH PIPING INTEGRITY.

EVENTS: NONE.

TC16 GOVERNOR VALVES NOT TRACKING AUTO

TYPE: DISCRETE, RB

CAUSE: COUNTER FAILURE

REF: DEHC SYSTEM DESCRIPTION

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE GOVERNOR VALVES MANUAL CONTROLLER TO NOT TRACK THE AUTOMATIC CONTROLLER. THE "MANUAL NOT TRACKING AUTO" LAMP ILLUMINATES ON THE "DEH TURBINE CONTROL" PANEL. THE MANUAL CONTROLLER WILL STOP TRACKING THE AUTO CONTROLLER AT THE TIME OF MALFUNCTION INSERTION AND STAY IN THAT POSITION. WITH A LOAD CHANGE THE AUTO CONTROLLER WILL MODULATE TO ACCOMODATE THE CHANGE. PLACING DEHC IN THE MANUAL MODE WILL CAUSE THE MANUAL CONTROLLER TO SEND THE SIGNAL BEING SENT AT TIME OF MALF INSERTION TO THE GOVERNOR VALVES. THE TURBINE WILL RESPOND ACCURATELY TO THE CHANGE.

MALFUNCTION REMOVAL RESTORES THE MANUAL CONTROLLER TO NORMAL.

EVENTS: NONE.

TC17 EH PUMP FAILS TO START/TRIP

TYPE: GENERIC, RB

A) 1A EH PUMP
B) 1B EH PUMP

CAUSE: FAULTY LOCKOUT (86LFT) RELAY

REF: 20E-1-4030 EH01
20E-1-4030 EH02
DEHC SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: WHEN THIS MALFUNCTION BECOMES ACTIVE, THE SELECTED EH PUMP BREAKER WILL TRIP. ANNUNCIATOR 18-A15 "EH PUMP TRIP" ACTUATES, AND THE CONTROL SWITCH TRIP LIGHT ILLUMINATES. THE STANDBY EH PUMP WILL AUTO START AT 1600 PSIG DISCHARGE PRESSURE, MAINTAINING NORMAL TURBINE OPERATION.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND THE TRIP LIGHT BY PLACING THE CONTROL SWITCH IN THE TRIP POSITION. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL CLOSE THEN IMMEDIATELY TRIP OPEN.

MALFUNCTION REMOVAL WILL RESTORE THE FAULTY LOCKOUT RELAY TO NORMAL OPERATION.

EVENTS: NONE.

TC18 INADVERTENT OTΔT TURBINE RUNBACK

TYPE: DISCRETE, RB

CAUSE: OTΔT TX RELAY CONTACT 7/8 FAILS CLOSED

REF: 20E-1 4030 MS17
DEHC SYSTEM DESCRIPTION

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A TURBINE RUNBACK AT A 200%/MIN RATE. THE LOSS OF LOAD INCREASES S/G PRESSURES, RCS TEMPERATURE AND CAUSES THE ROD CONTROL SYSTEM TO STEP RODS IN. "RUNBACK OPER" LIGHT ON THE DEH IS LIT WHILE RUNBACK IS IN PROGRESS (FOR 1.5 SECONDS) THEN EXTINGUISHES. THE TURBINE LOAD DECREASE IS INDICATED ON THE DIGITAL AND DEH REFERENCE, REFERENCE DEMAND INDICATORS, AND THE MW RECORDER. RUNBACK WILL STOP AT 235 MW IF IN AUTO OR ≈ 650 MW IF IN MANUAL.

MALFUNCTION REMOVAL RESTORES THE FAULTY RUNBACK CIRCUIT TO NORMAL.

EVENTS: NONE.

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

TH01	PZR STEAM SPACE LEAK
TH02	PZR RELIEF TANK LEAK
TH03	S/G TUBE RUPTURE
TH04	RCS LEAK, HOT LEG (HIGH)
TH05	RCS LEAK, HOT LEG (MEDIUM)
TH06	RCS LEAK, COLD LEG
TH07	REACTOR VESSEL FLANGE LEAK
TH08	RCS FUEL ELEMENT FAILURE
TH09	RTD MANIFOLD FAULTY FLOW CONDITIONS
TH10	PZR SPRAY VALVE FAILURE
TH11	PZR POWER OPERATED RELIEF VALVE FAILURE
TH12	PZR SAFETY VALVE FAILURE
TH13	PZR LEVEL DETECTOR REFERENCE LEG LEAK
TH14	PZR RELIEF LINE RTD FAILURE
TH15	RCS WIDE RANGE RTD FAILURE
TH16	RCP FAILS TO START/TRIP
TH17	RCP DEGRADED PERFORMANCE/LOCKED ROTOR
TH18	RCP SHAFT BREAK
TH19	REACTOR VESSEL BOTTOM CRACK

TH01 PZR STEAM SPACE LEAK

TYPE: DISCRETE, NRVI 0-.5 MLB/HR @ 2235 PSID

CAUSE: VESSEL FAILURE AT SAFETY NOZZLE PENETRATION

REF: M-60 SHEET 5

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES LEAKAGE OF REACTOR COOLANT TO THE CONTAINMENT ATMOSPHERE FROM THE PRESSURIZER STEAM SPACE. THE RATE OF MASS LEAKAGE WILL BE DETERMINED BY THE SELECTED SEVERITY. CONTAINMENT TEMPERATURE AND PRESSURE, AIRBORNE ACTIVITY LEVELS, AREA RADIATION LEVELS AND SUMP LEVELS WILL INCREASE DEPENDENT UPON THE SELECTED SEVERITY. PRESSURIZER PRESSURE WILL DECREASE AT A RATE PROPORTIONAL TO THE SELECTED SEVERITY. AS MALFUNCTION SEVERITY IS INCREASED, THE RATE OF DECREASE OF PRESSURIZER PRESSURE WILL INCREASE. THE PRESSURIZER HEATERS WILL ENERGIZE IN AN ATTEMPT TO RECOVER PRESSURIZER PRESSURE. ANNUNCIATOR 11-B4 "OTDT RX TRIP" WILL ACTUATE. P-11 IS AT 1930 PSIG, ANNUNCIATOR 12-A1 "PZR PRESS LOW RX TRIP STPT ALERT" ACTUATES AT 1885 PSIG. WHEN PRESSURIZER PRESSURE DECREASES BELOW 1829 PSIG, ANNUNCIATORS 12-B1 "PZR PRESS LOW", AND 11-C1 "PZR PRESS LOW SI/RX TRIP" (2/4 LOGIC) ACTUATES, AND A SAFETY INJECTION SIGNAL WILL BE GENERATED.

MALFUNCTION SEVERITY MAY ONLY BE INCREASED FOR THIS MALFUNCTION AND THE SIMULATOR MUST BE RESET TO RECOVER FROM THIS MALFUNCTION.

EVENTS: NONE.

TH02 PZR RELIEF TANK LEAK

TYPE: DISCRETE, NRVI 0-100 GPM @ 3 PSID

CAUSE: TANK FAILURE AT DRAIN LINE PENETRATION

REF: M-60 SHEET 6

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE PRT TO LOSE MASS TO THE CONTAINMENT FLOOR DRAIN SUMP. THE RATE OF MASS LOSS WILL BE DETERMINED BY THE SELECTED SEVERITY. AS MALFUNCTION SEVERITY INCREASES, THE RATE OF MASS LOSS THROUGH THE OPENING INCREASES. PRT LEVEL AND PRESSURE, AS INDICATED ON 1LI-470/1PI-469 (1PM05J), WILL DECREASE. ANNUNCIATOR 12-A7 "PRT LEVEL HIGH LOW" ACTUATES.

MALFUNCTION REMOVAL RESTORES THE PRT TANK INTEGRITY TO NORMAL.

EVENTS: NONE.

TH03 S/G TUBE RUPTURE

TYPE: GENERIC, NRVI 0-4500 GPM @ 1500 PSID

A) 1A S/G
B) 1B S/G
C) 1C S/G
D) 1D S/G

CAUSE: TUBE FAILURE

REF: M-60 SHEET 1A,2,3,4

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED STEAM GENERATOR TUBE(S) TO LEAK. THE RATE OF MASS LEAKAGE WILL BE DETERMINED BY THE SELECTED SEVERITY AND THE RELATIVE DIFFERENTIAL PRESSURE ACROSS THE STEAM GENERATOR U-TUBES. THE AFFECTED STEAM GENERATOR ACTIVITY AND LEVEL (IF THE S/G IS ISOLATED), WILL INCREASE AT A RATE PROPORTIONAL TO THE INPUT SEVERITY (N-16 RESPONSE WILL CAUSE THE MS LINE RAD MONITORS TO RISE TO APPROX. 575-625 mr/hr IF AT 100% FOR A 650 GPM SGTR). FW FLOW TO THE AFFECTED S/G WILL DECREASE. ANNUNCIATOR 15-A9/B9/C9/D9 "S/G 1A/1B/1C/1D LEVEL DEVIATION HIGH LOW" IS ACTUATED AT +5% DEVIATION FROM PROGRAM LEVEL. FEEDWATER FLOW TO THE AFFECTED STEAM GENERATOR WILL DECREASE AS THE STEAM GENERATOR WATER LEVEL CONTROL SYSTEM COMPENSATES FOR THE PRIMARY IN-LEAKAGE. PRESSURIZER PRESSURE AND LEVEL WILL DECREASE AT A RATE PROPORTIONAL TO THE SELECTED SEVERITY, AND ANNUNCIATE WHEN THEIR RESPECTIVE SETPOINTS ARE REACHED. THE SJAE/GS EXH. RAD MONITOR, S/G BLOWDOWN RAD MONITOR, AND ASSOCIATED MAIN STEAM LINE RAD MONITORS WILL INDICATE THE RADIOACTIVITY TRANSPORT (N-16 RESPONSE WILL CAUSE A DROP IN THE MS LINE RAD MONITOR AS POWER IS REDUCED). AS MALFUNCTION SEVERITY INCREASES, THE FEED FLOW TO STEAM FLOW MISMATCH AND THE SECONDARY ACTIVITY LEVELS WILL INCREASE.

MALFUNCTION SEVERITY MAY ONLY BE INCREASED FOR THIS MALFUNCTION AND THE SIMULATOR MUST BE RESET TO RESTORE THE S/G TO NORMAL OPERATION.

EVENTS: 1) SER 33-87
2) OE 3249

IS 722 I FORSYTH (INFO) 11-NOV-87 12:05 EST
SUBJECT: SER 33-87, STEAM GENERATOR TUBE RUPTURE

TH03

UNIT (TYPE): NORTH ANNA 1 (PWR)
DOC NO/LER: 50-338/87017
EVENT DATE: 7/15/87
NSSS/AE: WESTINGHOUSE/STONE & WEBSTER

SUMMARY:

WHILE OPERATING AT 100 PERCENT POWER, THE UNIT EXPERIENCED A PRIMARY TO SECONDARY LEAK IN THE "C" STEAM GENERATOR CALCULATED TO BE BETWEEN 550 AND 637 GALLONS PER MINUTE. AN UNUSUAL EVENT WAS DECLARED AND WAS SUBSEQUENTLY UPGRADED TO AN ALERT. APPROXIMATELY 0.159 CURIES OF RADIOACTIVE GAS WAS RELEASED TO THE ATMOSPHERE FROM THE CONDENSER STEAM JET AIR EJECTOR DISCHARGE VIA THE PLANT VENT AND THE STEAM-DRIVEN AUXILIARY FEEDWATER PUMP EXHAUST. THE STEAM GENERATOR WITH THE RUPTURED TUBE WAS ISOLATED APPROXIMATELY 18 MINUTES AFTER THE EVENT INITIATED. THIS ACTION ISOLATED THE RELEASE PATHS TO THE ENVIRONMENT. RESIDUAL RADIOACTIVE GAS RELEASES WERE TERMINATED IN APPROXIMATELY AN HOUR AND A HALF. THE UNIT WAS BROUGHT TO COLD SHUTDOWN WITHOUT COMPLICATIONS.

THIS EVENT IS SIGNIFICANT BECAUSE A LARGE PRIMARY TO SECONDARY SIDE LEAK OCCURRED IN ONE OF THE PLANT'S STEAM GENERATORS THAT RESULTED IN A SIGNIFICANT PRIMARY TRANSIENT AND A RELEASE OF RADIOACTIVE GASES TO THE ENVIRONMENT. THE PLANT WAS SHUTDOWN FOR APPROXIMATELY THREE MONTHS TO EVALUATE THE CAUSE OF THE EVENT AND IMPLEMENT NECESSARY MODIFICATIONS.

THE CONDITIONS THAT CONTRIBUTED TO THE TUBE FAILURE ARE SUMMARIZED BY THE FOLLOWING TABLE:

FATIGUE REQUIREMENT	PREREQUISITE CONDITIONS
CHANGE OF MEAN STRESS IN STEAM GENERATOR TUBES	DENTING - CAUSED BY RESIN INTRUSION INTO STEAM GENERATOR DURING THE INITIAL OPERATING CYCLE
ALTERNATING STRESS IN STEAM GENERATOR TUBES	TUBE VIBRATION CAUSED BY o DENTING AT THE TOP SUPPORT PLATE THAT CAPTURES THE TUBE AND REDUCES THE TUBE DAMPING o HIGH LOCAL FLUID VELOCITIES IN THE SECONDARY SIDE OF THE STEAM GENERATOR o ABSENCE OF ANTIVIBRATION BARS WHERE HIGH LOCAL FLUID VELOCITIES OCCUR

NOTE: only 1 tube broke
Entire tube seperated
1/2" gap between break

COMMENTS:

1. SIMULATOR TRAINING SHOULD INCORPORATE REALISTIC PLANT CASUALTY SCENARIOS THAT REQUIRE THE CONTROL ROOM OPERATING TEAMS TO PRACTICE HANDLING THE SITUATIONS WITH APPROVED PROCEDURES.

   IN THIS EVENT, OPERATORS RESPONDED WELL AND DEALT WITH THE CASUALTY IN A PROFESSIONAL AND EFFECTIVE MANNER. IN THE VIEW OF VIRGINIA POWER MANAGEMENT AND INPO, PRACTICE OF THIS SPECIFIC EVENT ON THE SIMULATOR CONTRIBUTED DIRECTLY TO THEIR SUCCESS.

... DUKE (DUKE) 28-MAR-89 11:04 EST
Subject: MCGUIRE UNIT 1 STEAM GENERATOR TUBE RUPTURE EVENT ON MARCH 7, 1989

TH03

UNIT - McGUIRE UNIT 1
DOCKET NO - 50-369
EVENT DATE - MARCH 7-8, 1989
NSSS/A-E - WESTINGHOUSE/UTILITY
RATING - 1180 MWe

SUBJECT: McGUIRE UNIT 1 STEAM GENERATOR TUBE RUPTURE EVENT ON MARCH 7, 1989

SUMMARY

ON MARCH 7, 1989, A STEAM GENERATOR TUBE RUPTURE (SGTR) OCCURRED IN McGUIRE UNIT 1 STEAM GENERATOR 'B'. THE MAXIMUM PRIMARY-TO-SECONDARY LEAK RATE RESULTING FROM THE RUPTURE WAS ESTIMATED TO BE 540 GALLONS PER MINUTE. A TOTAL OF 43.4 CURIES OF XENON-133 EQUIVALENT WAS RELEASED AS A RESULT OF THE SGTR AND THE SUBSEQUENT DEGASSING OF THE SECONDARY SYSTEM. THE CALCULATED WHOLE BODY DOSE AT THE SITE BOUNDARY FROM THIS RELEASE WAS 0.015 MILLIREM AND THE CALCULATED CHILD THYROID DOSE WAS 0.03 MILLIREM. THE RELEASE WAS WELL WITHIN TECHNICAL SPECIFICATION LIMITS FOR THE SITE BOUNDARY.

EVENT DESCRIPTION

UNIT 1 HAD BEEN OPERATING AT 100% POWER AND HAD BEEN ON-LINE FOR 66 DAYS. ON MARCH 7, 1989 AT 2338 HOURS, A HIGH RADIATION ALARM OCCURRED IN THE STEAM LINE 'B' RADIATION MONITOR, INDICATING THE PRESENCE OF A SUBSTANTIAL TUBE LEAK. AT 2340, THE CONDENSATE AIR EJECTOR HIGH RADIATION ALARM ACTUATED, INDICATING RADIOACTIVE MATERIAL IN THE CONDENSER OFFGAS SYSTEM. AT THIS TIME, OPERATORS BEGAN DECREASING ELECTRICAL LOAD AT 30MW PER MINUTE. AT 2342, THE UNIT VENT HIGH RADIATION ALARM ACTUATED. THIS INDICATED THAT A RADIOACTIVE RELEASE WAS OCCURRING THROUGH THE UNIT VENT.

AN ALERT WAS DECLARED AT 2345. PRIMARY-TO-SECONDARY LEAKAGE WAS ESTIMATED AT THIS TIME TO BE 100-150 GPM. (LATER ESTIMATES OF LEAK RATE WOULD SHOW THAT THIS INITIAL ESTIMATE WAS LOW.)

AT 2346. THE REACTOR WAS MANUALLY TRIPPED. STEAM GENERATOR

'A', 'C', AND 'D' POWER OPERATED RELIEF VALVES (PORVs) OPENED BRIEFLY. (THE 'B' PORV HAD BEEN ISOLATED PRIOR TO THIS EVENT.) BY 2349, THE MAIN STEAM ISOLATION VALVE FOR STEAM GENERATOR 'B' WAS CLOSED, THEREBY ISOLATING THE AFFECTED GENERATOR.

NOTIFICATION OF THE EVENT WAS FIRST MADE TO COUNTY AND STATE GOVERNMENTS AT 2358. BY 0030 HOURS ON MARCH 8, THE LEAK RATE HAD BEEN REDUCED TO NEAR ZERO BY EQUALIZING PRIMARY AND SECONDARY SYSTEM PRESSURES. THE STATION OPERATIONS SUPPORT CENTER AND TECHNICAL SUPPORT CENTER WERE ACTIVATED AT 0128 AND 0132, RESPECTIVELY.

AT 0322, PLANT STAFF BEGAN A UNIT COOLDOWN IN ORDER TO REDUCE REACTOR COOLANT SYSTEM PRESSURE. OFF-SITE DOSE MONITORING TEAMS WERE DISPATCHED AS A PRECAUTION BY 0640.

AT 0705, A CHANGE WAS MADE TO THE INITIAL LEAK RATE ASSESSMENT. A NEW ASSESSMENT INDICATED A LEAK RATE OF APPROXIMATELY 540 GPM. BY 0740, THE CORPORATE CRISIS MANAGEMENT CENTER WAS FULLY ACTIVATED.

AT 1015, COOLDOWN WAS COMMENCED ON THE AFFECTED STEAM GENERATOR USING THE BACKFILL METHOD. (IN THIS METHOD, PRIMARY SYSTEM PRESSURE IS REDUCED SO THAT SECONDARY WATER FLOWS BACK THROUGH THE TUBE BREAK INTO THE PRIMARY SYSTEM.) THE UNIT ACHIEVED HOT SHUTDOWN BY 1025.

BY 1520, THE RESIDUAL HEAT REMOVAL SYSTEM WAS PLACED INTO SERVICE AND THE UNIT ACHIEVED COLD SHUTDOWN BY 1745. THE ALERT WAS TERMINATED AT 1815.

DISCUSSION

AN INSPECTION OF STEAM GENERATOR 'B' (WESTINGHOUSE MODEL D-2 WITH 4670 TUBES) HAS REVEALED THE RUPTURED TUBE (ROW 18, COLUMN 25 IN THE COLD LEG PORTION OF TUBE) TO CONTAIN AN APPROXIMATELY 4 INCH LONG, 1/4 INCH WIDE AXIAL CRACK AT THE LOCATION OF THE FIRST TUBE SUPPORT PLATE. 197 TUBES HAD BEEN PREVIOUSLY PLUGGED IN THIS STEAM GENERATOR. THE RUPTURED TUBE HAD NOT BEEN INSPECTED SINCE THE INITIAL PRE-SERVICE BASELINE INSPECTION.

EDDY CURRENT TESTING IS CURRENTLY UNDERWAY IN ALL FOUR STEAM GENERATORS. VISUAL INSPECTION OF THE CRACKED TUBE IS IN PROGRESS ON STEAM GENERATOR 'B'. THREE TUBES HAVE BEEN IDENTIFIED TO BE REMOVED FROM THIS STEAM GENERATOR. TUBE PULLING EQUIPMENT IS CURRENTLY BEING SET UP.

AT THIS POINT, THE CAUSE OF THE TUBE RUPTURE IS STILL UNKNOWN; HOWEVER, THE EVIDENCE SO FAR DOES NOT SUGGEST A GENERIC FAILURE MECHANISM.

THE RECOVERY EFFORT IS BEING COORDINATED BY A TECHNICAL REVIEW GROUP COMPOSED OF REPRESENTATIVES FROM VENDORS, STEAM GENERATOR CONSULTANTS, UTILITY PERSONNEL, AND THE ELECTRIC POWER RESEARCH INSTITUTE. THIS GROUP WILL REVIEW THE TECHNICAL FINDINGS RESULTING FROM THE EVENT AND MAKE RECOMMENDATIONS FOR THE INDUSTRY AS APPROPRIATE.

CURRENT PLANS CALL FOR UNIT 1 TO BE RETURNED TO SERVICE BY MID-APRIL.

INFORMATION CONTACT: J.S. REESIDE (704) 875-4182

TH04 RCS LEAK, HOT LEG (HIGH)

TYPE: GENERIC, NRVI 0-540,000 GPM @ 2235 PSID

A) LOOP A
B) LOOP B
C) LOOP C
D) LOOP D

CAUSE: PIPE RUPTURE @ HOT LEG RX VESSEL NOZZLE (NOTE: ONLY ONE RCS LEAK MAY BE ACTIVE AT ANY ONE TIME)

REF: M-60 SHEET 1B

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A RAPID LOSS OF MASS INVENTORY FROM THE REACTOR COOLANT SYSTEM. PZR PRESSURE AND LEVEL DECREASE RAPIDLY, WHILE THE CONTAINMENT PRESSURE, TEMPERATURE, RADIATION LEVELS, AND SUMP LEVELS INCREASE. A REACTOR TRIP, SAFETY INJECTION, AND SUBSEQUENT SAFEGUARDS ACTUATIONS OCCUR DUE TO THE LOW PZR PRESSURE AND/OR HIGH CONTAINMENT PRESSURE. CONTAINMENT SPRAY AND A MAIN STEAM LINE ISOLATION ACTUATE ON INCREASING CONTAINMENT PRESSURE.

AS REACTOR COOLANT PRESSURE DECREASES, THE SAFETY INJECTION PUMPS DISCHARGE BORATED WATER INTO THE REACTOR FROM THE RWST, FOLLOWED BY THE SI ACCUMULATORS, THEN FROM THE RHR SYSYEM. THESE SOURCES WILL FLOOD THE REACTOR VESSEL UNTIL THE OPERATOR IS REQUIRED TO LINE-UP FOR LONG TERM CORE COOLING.

MALFUNCTION SEVERITY CAN ONLY BE INCREASED, AND THE SIMULATOR MUST BE RESET TO RETURN THE REACTOR COOLANT SYSTEM TO NORMAL OPERATION.

EVENTS: NONE.

TH05 RCS LEAK, HOT LEG (MEDIUM)

TYPE: GENERIC, NRVI 0-500 GPM @ 2235 PSID

A) LOOP A
B) LOOP B
C) LOOP C
D) LOOP D

CAUSE: PIPE RUPTURE @ HOT LEG RX VESSEL NOZZLE (NOTE: ONLY ONE RCS LEAK MAY BE ACTIVE AT ANY ONE TIME)

REF: M-60 SHEET 1B

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A LOSS OF REACTOR COOLANT INVENTORY TO THE CONTAINMENT ATMOSPHERE AT THE SELECTED SEVERITY. CONTAINMENT PRESSURE, TEMPERATURE, SUMP LEVELS, AND RADIATION LEVELS INCREASE AT A RATE DEPENDENT ON SELECTED SEVERITY. PZR LEVEL WILL DECREASE AS THE MALFUNCTION SIZE IS INCREASED AND AS THE SYSTEM LEAKAGE EXCEEDS THE TOTAL MAKEUP CAPACITY OF THE CHARGING SYSTEM. PZR PRESSURE AND LEVEL DECREASE WILL DECREASE FASTER CAUSING LETDOWN ISOLATION, AND EVENTUALLY A LOW PZR PRESSURE OR OTDT REACTOR TRIP, FOLLOWED BY A SAFETY INJECTION ON LOW PZR PRESSURE.

THE CHARGING SYSTEM WILL EXHAUST THE VCT WATER SUPPLY, THEN SWITCH TO THE RWST RESULTING IN THE BORATION OF THE REACTOR COOLANT SYSTEM UNLESS AN SI OCCURS FIRST.

MALFUNCTION SEVERITY CAN ONLY BE INCREASED, AND THE SIMULATOR MUST BE RESET TO RETURN THE REACTOR COOLANT SYSTEM TO NORMAL OPERATION.

EVENTS: NONE.

TH06 RCS LEAK, COLD LEG

TYPE: GENERIC, NRVI 0-540,000 GPM @ 2235 PSID

A) LOOP A
B) LOOP B
C) LOOP C
D) LOOP D

CAUSE: PIPE RUPTURE @ COLD LEG RX VESSEL NOZZLE (NOTE: ONLY ONE RCS LEAK MAY BE ACTIVE AT ANY ONE TIME)

REF: M-60 SHEET 1B

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A LOSS OF REACTOR COOLANT INVENTORY TO THE CONTAINMENT ATMOSPHERE AT THE SELECTED SEVERITY. CONTAINMENT PRESSURE, TEMPERATURE, SUMP LEVELS, AND RADIATION LEVELS INCREASE AT A RATE DEPENDENT ON THE SELECTED SEVERITY. PZR LEVEL WILL DECREASE AS THE MALFUNCTION SEVERITY IS INCREASED. PZR PRESSURE AND LEVEL DECREASE WILL DECREASE FASTER CAUSING LETDOWN ISOLATION, AND EVENTUALLY A LOW PZR PRESSURE REACTOR TRIP OR OTDT REACTOR TRIP, FOLLOWED BY A SAFETY INJECTION ON LOW PZR PRESSURE.

THE CHARGING SYSTEM WILL EXHAUST THE VCT WATER SUPPLY THEN SWITCH TO THE RWST RESULTING IN THE BORATION OF THE REACTOR COOLANT SYSTEM UNLESS AN SI OCCURS FIRST.

MALFUNCTION SEVERITY CAN ONLY BE INCREASED, AND THE SIMULATOR MUST BE RESET TO RETURN THE REACTOR COOLANT SYSTEM TO NORMAL OPERATION.

EVENTS: 1) LER 20-02-88-032

LICENSEE EVENT REPORT (LER)

THD6

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
Braidwood Generating Station Unit 2	0 5 0 0 0 4 5 7	1 of 0 3

Title (4) Unit 2 Shutdown due to RCS leakage in excess of Technical Specification Limits.

Event Date (5) Month	Day	Year	LER Number (6) Year	Sequential Number	Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
1 2	1 4	8 8	8 8	0 3 2	0 0	1 2	2 3	8 8	NONE	0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 9 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

__ 20.402(b)	__ 20.405(c)	__ 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	X 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER AREA CODE	
Randall A. Smith Technical Staff Engineer Ext. 2481	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	A B	I S V *	A 4 1 5	Yes					

SUPPLEMENTAL REPORT EXPECTED (14)

__ Yes (If yes, complete EXPECTED SUBMISSION DATE) X NO

Expected Submission Date (15): Month | Day | Year

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

At 1320 on December 14, 1988 a Reactor Coolant System (RCS) Leak Rate Surveillance was initiated as a result of a suspected RCS Leak. At 1600 Unit 2 Containment (outside the missile barrier) and the Auxiliary Building were visually inspected for leaks. At 1604 the RCS Leak Rate calculation was completed indicating an unidentified RCS Leakage of 2.95 gallons per minute (gpm) which is greater than the Technical Specification Limit of 1 gpm. At 1635 water was observed inside the missile barrier. Preparations were made for an orderly shutdown of Unit 2. At 2000 Reactor Power was stabilized at 30.6% (350 Mwe) to allow entry into the missile barrier. At 0429 on December 15, 1988 the leak was isolated by closing the RCS Loop "B" RTD Bypass Manifold Isolation valves. At 0857 a new RCS Leak Rate Surveillance was completed indicating an RCS unidentified leak rate of 0.51 gpm. The cause of this event was subsequently determined to be a result of packing leaks on instrument isolation valves 2RC028B and 2RC029B. The corrective actions were to repack instrument isolation valves 2RC028B and 2RC029B. There have been no previous reportable occurrences of completing a reactor shutdown due to exceeding the RCS Leakage Limits as a result of valve packing leakage.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

Form Rev 2.0

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) Year	Sequential Number	Revision Number	Page (3)
Braidwood	0 5 0 0 0 4 5 7	8 8	- 0 3 2	- 0 0	0 2 OF 0 3

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 2; Event Date: December 14, 1988; Event Time: 1320;

Mode: 1 - Power Operation; Rx Power: 90%;

RCS [AB] Temperature/Pressure: 580 degrees F/ 2240 psig

B. DESCRIPTION OF EVENT:

At 1320 on December 14, 1988 a Reactor Coolant System (RCS) [AB] Leak Rate Surveillance was initiated as a result of a suspected RCS leak. Indications of the RCS leak included a decreasing Volume Control Tank (VCT) [CB] level, increased Containment Sump (RF) [WK] flow, and increased activity on 2PR11J (PR) [IL] Unit 2 containment Atmosphere Radiation Monitor.

At 1600 an inspection team entered Unit 2 Containment to visually inspect for leaks outside the missile barrier. Additionally, an inspection of the Auxiliary Building was performed to check for any sources of leakage.

At 1604 the RCS Leak Rate calculation was completed indicating an unidentified RCS leakage of 2.95 gallons per minute (gpm). The Technical Specification Limit for RCS unidentified leakage is 1 gpm. At 1635 Unit 2 Control Room was notified that water was observed inside the missile barrier by the Unit 2 Containment inspection team.

As a result of exceeding the limit for unidentified RCS leakage, Limiting Condition for Operation Action Requirement (LCOAR) 2BwOS 4.6.2-1a, Reactor Coolant System - Operational Leakage, was entered effective 1320 on December 14, 1988. Preparations were made for an orderly shutdown of Unit 2 in accordance with 2BwGP 100-4, Power Descension. At 1720, a power reduction of Unit 2 was initiated.

At 1726 a Generating Station Emergency Plan (GSEP) Unusual Event was declared.

At 2000 Reactor Power was stabilized at 30.6% (350 MWe) to allow entry into the missile barrier.

At 2100, while at 30.6% reactor power a containment inspection team entered Unit 2 Containment Missile Barrier in an attempt to identify the source of leakage. It was determined that a leak existed in the RCS loop "B" Resistance Temperature Detector (RTD) Bypass Manifold.

At 2119 the shutdown of Unit 2 re-commenced until Mode 3 was subsequently entered at 2311.

At 0241 on December 15, 1988 Operations personnel entered Unit 2 Containment for RCS leak identification and isolation. At 0429, the Shift Foreman reported that the leak appeared to be from the 2FE-427 RTD Bypass Manifold Flow Element and was isolated by closing the "B" RTD Bypass Manifold Isolation valves.

At 0857 a new RCS Leak Rate Surveillance was completed indicating an RCS unidentified leak rate of 0.51 gpm. At 0945 LCOAR 2BwOS 4.6.2-1a was exited and the GSEP Unusual Event was terminated.

The appropriate NRC notification via the ENS phone system was made at 1729 on Decembere 14, 1988 pursuant to 10CFR50.72(b)(1)(i).

This event is being reported pursuant to 10CFR50.73(a)(2)(i) - The completion of any nuclear plant shutdown required by the plant's Technical Specifiction's.

TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

C. CAUSE OF EVENT:

The cause of this event was subsequently determined to be a result of packing leaks on instrument isolation valves 2RC028B and 2RC029B "RCS Loop 2B RTD Loop Flow Instrument Isolation Valves".

D. SAFETY ANALYSIS:

This event had no effect on the safety of the plant or the public. All systems operated as designed. Upon determination that an unidentified RCS Leak Rate of greater than 1 gpm existed, Unit 2 was promptly shutdown in accordance with Technical Specification Requirements. The leakage was within the Chemical and Volume Control System makeup capability.

E. CORRECTIVE ACTIONS:

The immediate corrective actions included performing an RCS Leak Rate Surveillance to quantify the leakage, walkdowns to determine the source of and isolate the leak, and an orderly shutdown of Unit 2 in accordance with the Technical Specifications.

The long term corrective actions were to repack instrument isolation valves 2RC028B and 2RC029B.

F. PREVIOUS OCCURRENCES:

There have been no previous reportable occurrences of completing a reactor shutdown due to exceeding the RCS leakage limits as a result of valve packing leakage.

G. COMPONENT FAILURE DATA:

Manufacturer	Nomenclature	Model Number	MFG Part Number
Anderson Greenwood & Co.	Isol Valve	M5YS-46BC-N	N02-8255-593

TH07 REACTOR VESSEL FLANGE LEAK

TYPE: DISCRETE, RV 0-75 GPM @ 2235 PSID

CAUSE: INNER O-RING FAILURE

REF: M-60 SHEET 1B
M-2060 SHEET 5

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE REACTOR VESSEL FLANGE O-RING TO FAIL. THE RATE OF LEAKAGE TO THE REACTOR COOLANT DRAIN TANK WILL BE DETERMINED BY THE SELECTED SEVERITY AND WILL BE INDICATED BY AN INCREASED TEMPERATURE ON 1TI-401 (1PM05J). ANNUNCIATOR 14-E5 "RX VESSEL FLNG LEAKOFF TEMP HIGH" ACTUATES. THE MASS LOSS TO THE REACTOR COOLANT DRAIN TANK WILL RESULT IN VCT LEVEL DECREASING AT A RATE PROPORTIONAL TO THE INPUT SEVERITY. THE CHARGING SYSTEM WILL MAKEUP MORE FREQUENTLY DUE TO THE LOSS.

EFFECTS OF THIS MALFUNCTION CAN BE MITIGATED BY DELETING THE MALFUNCTION.

EVENTS: NONE.

TH08 RCS FUEL ELEMENT FAILURE

TYPE: DISCRETE, NRVI 0-10% CLADDING FAILURE

CAUSE: CLADDING FAILURE

REF: SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A RELEASE OF FUEL INTO THE REACTOR COOLANT. THE MAJOR INDICATIONS WILL BE INCREASED RADIATION READINGS ON THE LETDOWN LINE GROSS FAILED FUEL MONITOR. OTHER SYSTEM RADIATION MONITORS THAT COME IN CONTACT WITH REACTOR COOLANT WILL ALSO SHOW AN INCREASE IN ACTIVITY DEPENDENT UPON THE SEVERITY LEVEL.

MALFUNCTION SEVERITY CAN ONLY BE INCREASED, AND THE SIMULATOR MUST BE RESET TO RETURN THE REACTOR COOLANT SYSTEM TO NORMAL OPERATION.

EVENTS: NONE.

TH09 RTD MANIFOLD FAULTY FLOW CONDITIONS

TYPE: GENERIC, RV 0-100% (100% = FULLY CLOSED VALVE)

A) LOOP 1
B) LOOP 2
C) LOOP 3
D) LOOP 4

CAUSE: 1RC8074( ) CLOSED

REF: M-60 SHEET 1A
M-60 SHEET 2
M-60 SHEET 3
M-60 SHEET 4

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE RTD MANIFOLD RETURN VALVE TO BE THROTTLED CLOSED ALLOWING A SELECTED AMOUNT OF MANIFOLD FLOW. AT HIGHER SEVERITIES (VALVE ALMOST CLOSED) A LOWER FLOW RATE WILL CAUSE THE RTDs TO READ RELATIVELY COLDER TEMPERATURES. THIS IN TURN WILL CAUSE ERRONEOUS SIGNALS TO THE T_{ave}, AND ΔT CIRCUITS. LOOP T_{ave} AND LOOP ΔT DEVIATION ANNUNCIATORS ACTUATE FOR THE SELECTED LOOP.

MALFUNCTION REMOVAL RESTORES THE RETURN VALVE TO ITS NORMAL POSITION.

EVENTS: NONE.

TH10 PZR SPRAY VALVE FAILURE

TYPE: GENERIC, RV 0-100%

A) 1RY455B
B) 1RY455C

CAUSE: SPRAY VALVE CONTROLLER (1PK-455B/C) AUTO FAILURE

REF: M-60 SHEET 5

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED PRESSURIZER SPRAY VALVE TO FAIL. THE VALVE POSITION AND FLOW RATE INTO THE PRESSURIZER WILL BE DETERMINED BY THE SELECTED SEVERITY. PRESSURIZER PRESSURE WILL DECREASE AT A RATE PROPORTIONAL TO THE SELECTED SEVERITY. AS MALFUNCTION SEVERITY IS INCREASED, THE RATE OF DECREASE OF PRESSURIZER PRESSURE WILL INCREASE. THE PRESSURIZER HEATERS WILL ENERGIZE IN AN ATTEMPT TO RECOVER PRESSURIZER PRESSURE. ANNUNCIATOR 12-A1 "PZR PRESS LOW RX TRIP STPT ALERT" ACTUATES AT 1885 PSIG. 2/4 LOGIC CAUSES ANNUNCIATOR 11-C3 "PZR PRESS LOW RX TRIP" ACTUATION. WHEN PRESSURIZER PRESSURE DECREASES BELOW 1829 PSIG, ANNUNCIATORS 12-B1 "PZR PRESS LOW", AND 11-C1 "PZR PRESS LOW SI/RX TRIP" (2/4 LOGIC) ACTUATE, AND A SAFETY INJECTION SIGNAL WILL BE GENERATED.

MALFUNCTION REMOVAL WILL RESTORE THE FAILED PRESSURIZER SPRAY VALVE TO NORMAL.

EVENTS: NONE.

TH11 PZR POWER OPERATED RELIEF VALVE FAILURE

TYPE: GENERIC, RV 0-100%

A) 1RY455A
B) 1RY456

CAUSE: SPRING FAILURE (STICKS ON LOSS OF AIR OR PWR)

REF: M-60 SHEET 5

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED PRESSURIZER PORV TO LEAK TO THE PRT. THE RATE OF MASS TRANSFER TO THE PRESSURIZER RELIEF TANK AND PORV OPENING WILL BE DETERMINED BY THE SELECTED SEVERITY. ANNUNCIATORS 12-B2 "PZR PORV OR SAF VLV OPEN", AND 12-C6 "PZR PORV DSCH TEMP HIGH" ARE ACTUATED WHEN THE VALVE OPENS. THE INCREASED TEMPERATURE IS INDICATED ON 1TI-463 (1PM05J). PRT PRESSURE (1PI-469), TEMPERATURE (1TI-468), AND LEVEL (1LI-470) INCREASE AT A RATE PROPORTIONAL TO THE SELECTED SEVERITY. PRESSURIZER PRESSURE AND LEVEL WILL DECREASE AS THE STEAM LEAKS BY THE PORV TO THE PRT. THE PRESSURIZER HEATERS WILL ENERGIZE IN AN ATTEMPT TO RECOVER THE DECREASING PRESSURIZER PRESSURE. AS MALFUNCTION SEVERITY INCREASES, THE RATE OF MASS LOSS FROM THE PRESSURIZER WILL INCREASE. ANNUNCIATOR 12-A1 "PZR PRESS LOW RX TRIP STPT ALERT" ACTUATES. 2/4 LOGIC CAUSES ANNUNCIATOR 11-C3 "PZR PRESS LOW RX TRIP" ACTUATION AT 1885 PSIG. WHEN PRESSURIZER PRESSURE DECREASES BELOW 1829 PSIG, ANNUNCIATORS 12-B1 "PZR PRESS LOW", AND 11-C1 "PZR PRESS LOW SI/RX TRIP" (2/4 LOGIC) ACTUATE, AND A SAFETY INJECTION SIGNAL WILL BE GENERATED.

ANY ATTEMPT BY THE OPERATOR TO CLOSE THE AFFECTED PORV, WHILE THE MALFUNCTION IS ACTIVE, IS UNSUCCESSFUL. THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY CLOSING THE ASSOCIATED PORV BLOCK VALVE.

MALFUNCTION REMOVAL RESTORES THE FAILED PORV CONTROLLER TO NORMAL.

EVENTS: NONE.

TH12 PZR SAFETY VALVE FAILURE

TYPE: GENERIC, NRVI 0-100%

A) 1RY8010A
B) 1RY8010B
C) 1RY8010C

CAUSE: MECHANICAL FAILURE

REF: M-60 SHEET 5

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED PRESSURIZER SAFETY VALVE TO LEAK TO THE PRT. THE RATE OF MASS TRANSFER TO THE PRESSURIZER RELIEF TANK AND SAFETY VALVE OPENING WILL BE DETERMINED BY THE SELECTED SEVERITY. ANNUNCIATOR 12-D6 "PZR SAF RLF DSCH TEMP HIGH" IS ACTUATED WHEN THE VALVE OPENS. THE INCREASED TEMPERATURE IS INDICATED ON 1TI-464/465/466 (1PM05J). PRT PRESSURE (1PI-469), TEMPERATURE (1TI-468), AND LEVEL (1LI-470) INCREASE AT A RATE PROPORTIONAL TO THE SELECTED SEVERITY. PRESSURIZER PRESSURE AND LEVEL WILL DECREASE AS THE STEAM LEAKS BY THE SAFETY VALVE TO THE PRT. THE PRESSURIZER HEATERS WILL ENERGIZE IN AN ATTEMPT TO RECOVER THE DECREASING PRESSURIZER PRESSURE. AS MALFUNCTION SEVERITY INCREASES, THE RATE OF MASS LOSS FROM THE PRESSURIZER WILL INCREASE. P-11 IS AT 1930 PSIG, AND ANNUNCIATOR 12-A1 "PZR PRESS LOW RX TRIP STPT ALERT" ACTUATES AT 1885 PSIG. WHEN PRESSURIZER PRESSURE DECREASES BELOW 1829 PSIG, ANNUNCIATORS 12-B1 "PZR PRESS LOW", AND 11-C1 "PZR PRESS LOW SI/RX TRIP" (2/4 LOGIC) ACTUATE, AND A SAFETY INJECTION SIGNAL WILL BE GENERATED.

THE SEVERITY OF THIS MALFUNCTION CAN ONLY BE INCREASED, AND THE SIMULATOR MUST BE RESET TO RESTORE TO NORMAL.

EVENTS: 1) LER 06-01-86-023

LICENSEE EVENT REPORT (LER)

THIZ

Facility Name (1)	Docket Number (2)	Page (3)
Byron, Unit 1	0 5 0 0 0 4 5 4	1 of 6

Title (4) INSTALLATION OF A DEFECTIVE PRESSURIZER SAFETY VALVE DUE TO PERSONNEL ERRORS

Event Date (5) Month	Day	Year	LER Number (6) Year		Sequential Number		Revision Number	Report Date (7) Month	Day	Year	Other Facilities Involved (8) Facility Names	Docket Number(s)
0 7	1 8	8 6	8 6	---	0 2 3	---	0 0	0 8	1 5	8 6	NONE	0 5 0 0 0
												0 5 0 0 0

OPERATING MODE (9): 3

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

POWER LEVEL (10): 0 0 0

__ 20.402(b)	__ 20.405(c)	__ 50.73(a)(2)(iv)	__ 73.71(b)
__ 20.405(a)(1)(i)	__ 50.36(c)(1)	__ 50.73(a)(2)(v)	__ 73.71(c)
__ 20.405(a)(1)(ii)	__ 50.36(c)(2)	__ 50.73(a)(2)(vii)	__ Other (Specify in Abstract below and in Text)
__ 20.405(a)(1)(iii)	X 50.73(a)(2)(i)	__ 50.73(a)(2)(viii)(A)	
__ 20.405(a)(1)(iv)	__ 50.73(a)(2)(ii)	__ 50.73(a)(2)(viii)(B)	
__ 20.405(a)(1)(v)	__ 50.73(a)(2)(iii)	__ 50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER AREA CODE	
D. St.Clair, Assistant Superintendent Work Planning, Ext. 2288	8 1 5	2 3 4 - 5 4 4 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
A	A B	R V	C 7 1 0	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

		Expected Submission Date (15)	Month	Day	Year
Yes (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

On July 17, 1986 with the Reactor Coolant System (RCS) [AB] at 1750 psig, excessive seat leakage of a recently installed pressurizer safety valve precluded a further RCS pressure increase. The safety valve was declared inoperable and the plant placed in Cold Shutdown on July 18, 1986. Removal and disassembly of the defective valve revealed that the disc insert (a seating surface) was missing.

The root cause of the defective valve installation was a series of personnel errors that occurred during valve testing in October of 1985. The maintenance and quality control personnel involved in the testing failed to verify that the serial numbers listed in the test package matched those on the tested component. The working environment in the Hot Shop contributed to the personnel errors. Also contributing to this event was the lack of storage facilities for contaminated equipment that is acceptable for use.

Maintenance Department meetings will be conducted to discuss this event. Maintenance Departments will be required to administratively close out work packages more expeditiously. Several secure storage cages are being constructed to facilitate the storage of contaminated equipment that has been deemed acceptable for use.

TH13 PZR LEVEL DETECTOR REFERENCE LEG LEAK

TYPE: GENERIC, NRVI 0-8 GPM

A) 1LT459 (AFFECTS PT-455 ALSO)
B) 1LT460 (AFFECTS PT-456 ALSO)
C) 1LT461 (AFFECTS PT-457 & 458)

CAUSE: LEAK IN REFERENCE LEG

REF: M-60 SHEET 5
M-2060 SHEET 6

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES LEAKAGE OF REACTOR COOLANT TO THE CONTAINMENT ATMOSPHERE FROM THE PRESSURIZER LEVEL DETECTOR REFERENCE LEG TAP. THE RATE OF MASS LEAKAGE WILL BE DETERMINED BY THE SELECTED SEVERITY. CONTAINMENT TEMPERATURE AND PRESSURE, AIRBORNE ACTIVITY LEVELS, AREA RADIATION LEVELS AND SUMP LEVELS WILL INCREASE DEPENDENT UPON THE SELECTED SEVERITY. INDICATED PRESSURIZER PRESSURE WILL DECREASE AT A RATE PROPORTIONAL TO THE INPUT SEVERITY. INDICATED PRESSURIZER LEVEL WILL READ AT A HIGHER LEVEL THAN ACTUAL LEVEL. CONTROL FUNCTIONS WILL RESPOND ACCURATELY TO THE FAILURE. ANNUNCIATOR 12-A3 "PZR LEVEL HIGH RX TRIP SETPT ALERT" ACTUATES.

THE SEVERITY OF THIS MALFUNCTION CAN ONLY BE INCREASED, AND THE SIMULATOR MUST BE RESET TO RESTORE TO NORMAL.

EVENTS: NONE.

TH14 PZR RELIEF LINE RTD FAILURE

TYPE: GENERIC, RV 50-400°F

A)	1TE0463	(PORV'S)
B)	1TE0464	(1A SAFETY VALVE)
C)	1TE0465	(1B SAFETY VALVE)
D)	1TE0466	(1C SAFETY VALVE)

CAUSE: RTD FAILURE

REF: M-60 SHEET 5
M-2060 SHEET 9

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED PRESSURIZER RELIEF LINE RTD TO FAIL. THE VALUE OF THE RTD OUTPUT WILL BE DETERMINED BY THE SELECTED SEVERITY. IF THE SELECTED SEVERITY IS GREATER THAN THE INITIAL VALUE, THE APPARENTLY INCREASED RTD OUTPUT WILL BE INDICATED ON THE ASSOCIATED METER, 1TI-463/464/465/466 (1PM05J). ANNUNCIATOR 12-C6 "PZR PORV DSCH TEMP HIGH" OR ANNUNCIATOR 12-D6 "PZR SAF RLF DSCH TEMP HIGH" ACTUATES. IF THE SELECTED SEVERITY IS LESS THAN THE INITIAL VALUE, THE FALSE RTD OUTPUT COULD HIDE AN ACTUAL LEAKING VALVE CONDITION.

MALFUNCTION REMOVAL WILL RESTORE THE FAILED RTD TO NORMAL.

EVENTS: NONE.

TH15 RCS WIDE RANGE RTD FAILURE

TYPE: GENERIC, RV 0-700°F

A) 1TW413A
B) 1TW413B
C) 1TW423A
D) 1TW423B
E) 1TW433A
F) 1TW433B
G) 1TW443A
H) 1TW443B

CAUSE: RTD FAILURE

REF: M-2060 SHEET 3
20E-1-4030 RC22

PLT STA: RH SYSTEM IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED RTD TO FAIL TO THE SELECTED SEVERITY AS INDICATED ON THE ASSOCIATED METER AND RECORDER. ATTEMPTING TO OPEN AN ISOLATED COLD LEG LOOP ISOLATION STOP VALVE WITH EITHER OF THE ASSOCIATED LOOP HOT OR COLD LEG RTDS' FAILED AT GREATER THAN A 20 °F DIFFERENCE, WILL NOT COMPLETE THE PERMISSIVE CIRCUIT TO OPEN THE VALVE. THIS ASSUMES THAT ALL OTHER REQUIRED CONDITIONS HAVE BEEN MET. OPERATION OF THE PZR PORV IN ARM LOW TEMP MAY ALSO BE AFFECTED.

MALFUNCTION REMOVAL RESTORES THE SELECTED RTD TO NORMAL OPERATION.

EVENTS: 1) DVR 20-01-89-133
2) DVR 06-01-88-064

DEVIATION REPORT

DVR NO.
20 - 1 - 89 - 133
STA UNIT YEAR NO.

TH15

Form Rev 2.0

PART 1 | TITLE OF DEVIATION
1B W/R RTD slowly failing high.

OCCURRED 08/30/89 0640
DATE TIME

SYSTEM AFFECTED
RCS

PLANT STATUS AT TIME OF EVENT
MODE 1 POWER(%) 47

WORK REQUEST NO.

TESTING
|___| YES |_X_| NO

DESCRIPTION OF EVENT

1B W/R RTD has been slowly failing high on 08/30/89 at 0640 the 1B W/R RTD had exceeded the acceptance criteria for Remote Shutdown Monitoring Tech Spec surveillance 1BwOS 3.3.5-1.

POTENTIALLY SIGNIFICANT EVENT PER NSD DIRECTIVE A-07 |___| YES |_X_| NO

10CFR50.72 NRC RED PHONE |___| 1 HOUR
NOTIFICATION MADE |___| 4 HOUR ______ TIME |_X_| NO

K. Boyle 08/30/89
RESPONSIBLE SUPERVISOR DATE

PART 2 | OPERATING ENGINEER'S COMMENTS

LCOAR entered.

|_X_| NON REPORTABLE EVENT

|___| 30 DAY REPORTABLE/10CFR________

|___| 5 DAY REPORT PER 10CFR21

|___| ANNUAL/SPECIAL REPORT REQUIRED

A.I.R. #________

L.E.R. #________

NOTIFICATION N/A ______ ______
REGION III DATE TIME

OFC of VP PWR OPS 08/31/89 1600
NSD DATE TIME

|___| CECO CORPORATE NOTIFICATION MADE
IF ABOVE NOTIFICATION IS PER 10CFR21

TELECOPY N/A ______ ______
CECO CORPORATE OFFICER DATE TIME

PRELIMINARY REPORT
COMPLETED AND REVIEWED Paul Smith 08/31/89
OPERATING ENGINEER DATE

INVESTIGATION REPORT & RESOLUTION
ACCEPTED BY STATION REVIEW David J. Miller 11OCT89 Carl [illegible] 10-11-89

RESOLUTION APPROVED AND
AUTHORIZED FOR DISTRIBUTION K C Kofron 10/13/89
STATION MANAGER DATE

-5176 (Form 15-52-1) 11-20-85

2878m(101089)

DEVIATION INVESTIGATION REPORT

TITLE
1B WIDE RANGE T_{COLD} LOOP FAILED HIGH

THIS

EVENT DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	PAGE
MONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	1	1 OF 0 2
0 4	1 5	8 8	0 5	0 1	8 8	0 6 4	0 0	0 5	2 7	8 8	POWER LEVEL 3 3	

NAME — CONTACT FOR THIS DIR

Warren Walter, Asst. Tech Staff Supervisor Ext. 2106

TELEPHONE NUMBER: AREA CODE 8 1 5 2 3 4 - 5 4

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	R C	R T D	W 1 2 0	Y					

SUPPLEMENTAL REPORT EXPECTED

__ YES (if yes, complete EXPECTED SUBMISSION DATE) X NO

EXPECTED SUBMISSION DATE: MONTH | DAY | YEAR

TEXT

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 4/15/88 / 2300

Unit 1 MODE 1 - Power Operations   Rx Power 33%   RCS [AB] Temperature/Pressure Normal Operating

Unit 2 MODE N/A - N/A   Rx Power N/A   RCS [AB] Temperature/Pressure N/A

B. DESCRIPTION OF EVENT:

At 2300 on 4/15/88, the Unit 1 licensed Nuclear Station Operator (NSO) noticed that the 1B Wide Range T_{Cold} Indicator (1TI-423B) was reading higher than the other three by approximately 25°F. The indication was trending upwards with time. The Unit 1 NSO entered LCOAR 1BOS 3.3.5-1a and generated Nuclear Work Request (NWR) B55090 to investigate and repair the problem. No safety system actuations occurred as a result of this event. There were no systems or components inoperable at the time of this event which contributed to this event.

C. CAUSE OF EVENT:

The cause of the increasing temperature drift was the failure of one of four leads 1TE RC023B. The root cause of the lead opening is indeterminate.

TITLE	DIR NUMBER							PAGE
	STA	UNIT	YEAR		SEQUENTIAL NUMBER		REVISION NUMBER	
1B WIDE RANGE T_{COLD} LOOP FAILED HIGH	0 6	0 1	8 8	—	0 6 4	—	0 0	2 OF 0 2

TEXT

D. SAFETY ANALYSIS:

Plant and public safety were not affected by this failure. The 423 Wide Range Cold Leg Temperature Loop is one of four channels. The other three instrument loops were functioning normally. Each Wide Range Cold Leg Temperature Loop supplies an input to the Loop Stop Valve Opening Logic for its respective RCS loop, and to the Cold Over Pressurization System Circuitry which actuates Pressurizer Power Operated Relief Valves (PORV). At the time of occurrence, the loop stop valves were out of service (OOS) open. With the plant operating in Mode 1, the input from the Cold Over Pressurization System Control to the PORV circuit is inhibited. Thus, this incident affected only the temperature monitoring and did not affect normal plant operation. Had plant conditions required the Cold Over Pressurization System to be inservice at the time of this event, Reactor Coolant Pressure would not have exceeded the PORV setpoint as at least one PORV would have remained operable. (One PORV uses T_{Hot} circuitry).

E. CORRECTIVE ACTIONS:

The Resistance Temperature Device (RTD) 1TE RC023B was found to have one of four leads open. Three normally used, therefore, the defective and remaining good leads were interchanged and the RTD placed bac in service. NWR B55161 was generated to replace the RTD during the next outage.

F. PREVIOUS OCCURRENCES:

There have been 2 previous occurrences due to failed RTDs on Wide Range Channels. DVR 6-2-87-049 was written for a failed RTD due to an indetermined cause on the 2A T_{Cold} channel. DVR 6-2-87-010 was written for a failed RTD due to unknown cause on the 2D T_{Hot} channel. In addition, a narrow-range RTD failure was attributed to a poor splice connection (LER 87-001) which is considered an unrelated cause.

DVR NUMBER	TITLE
6-2-87-010	Failure of Wide Range T_{Hot} RTD
6-2-87-049	Wide Range Temperature Indication Erratic Due to a Faulty RTD

G. COMPONENT FAILURE DATA:

a)

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Westinghouse Elect. Corporation	RCS Cold Leg RTD	P/N 21205	Spin No: QAELRT-01

b) RESULTS OF NPRDS SEARCH:

The NPRDS search shows that numerous failures have occurred within the industry. The RTD failure rate at Byron is believed to be within acceptable limits as RTDs are expected to fail within a reasonable number of plant cycles due to temperature changes. No adverse trend has been identified at Byron Station.

c) RESULTS OF NWR SEARCH:

No relevant NWR's exist to show a trend in the failure of these RTDs.

TH16 RCP FAILS TO START/TRIP

TYPE: GENERIC, RB

A)	1A RCP	1RC01PA
B)	1B RCP	1RC01PB
C)	1C RCP	1RC01PC
D)	1D RCP	1RC01PD

CAUSE: FAULTY OVERCURRENT (550/551) RELAY ACTUATION

REF: 20E-1-4030 RC01
20E-1-4030 RC02
20E-1-4030 RC03
20E-1-4030 RC04

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED RCP BREAKER TO OPEN. THE SELECTED RCP TRIP LIGHT AND STOP LIGHT ENERGIZE, MOTOR AMPS DECREASE TO ZERO, AND THE ASSOCIATED LOOP FLOW DECREASES TO 20%. ANNUNCIATORS 13-E3 "RCP TRIP" AND 13-A3/B3/C3/D3 "RCP 1A/1B/1C/1D BRKR OPEN OR FLOW LOW ALERT" ACTUATE. THE UNAFFECTED LOOP FLOWS WILL INCREASE AND THE REACTOR WILL TRIP DUE TO LOW COOLANT FLOW IF > 30% POWER. < 30% POWER, THE REACTOR WILL NOT TRIP UNLESS 2 OR MORE RCP'S ARE TRIPPED.

AFTER THE PLANT TRIPS, THE TEMPERATURE AND PRESSURE WILL DECREASE DUE TO CORE HEAT PRODUCTION DECREASING.

ATTEMPTS TO RECLOSE THE BRKR WILL RESULT IN THE IMMEDIATE RETRIPPING OF THE BRKR.

MALFUNCTION REMOVAL RESTORES THE SELECTED RCP BRKR TO NORMAL OPERATION.

EVENTS: 1) DVR 20-01-86-108

DEVIATION INVESTIGATION REPORT

TH16

: 1D REACTOR COOLANT PUMP TRIP ON OVERCURRENT

PAGE 1 OF 0 2

DATE			DIR NUMBER					REPORT DATE			OPERATING MODE	5
ONTH	DAY	YEAR	STA	UNIT	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	POWER LEVEL	0 0 0
1 2	3 0	8 6	2 0	0 1	8 6	— 1 0 8	— 0 0	0 2	0 2	8 7		

CONTACT FOR THIS DIR

NAME	TELEPHONE NUMBER	
	AREA CODE	
SUSAN M ZAPINSKI TECH STAFF ENGINEER Ext. 2481	8 1 5	4 5 8 - 2 8 0 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED		EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

:T

PLANT CONDITIONS PRIOR TO EVENT

Mode 5- Cold Shutdown. Rx Power 0% RCS [AB] Temperature/Pressure 102 °F/375 psig.

RIPTION OF EVENT

At 1945 on December 30, 1986, the 1D Reactor C~lant Pump (RCP) [AB] was started for a one minute run per BwCP RC-3, Reactor Coolant System Fill and Vent. At er 12 seconds of run time the pump tripped on 0A phase timer overcurrent, relay PR10A551. The electrical maintenance department investigated. No apparent breaker problem was found. The target was reset and at 2052 the 1D RCP was started a second time. Again after 12 seconds of run time the pump tripped on 0A Phase timer overcurrent, relay PR10A551. Southern Division Operational Analysis Division (OAD) was called to assist in troubleshooting. Stable plant conditions were maintained during this event.

CAUSE OF EVENT

OAD found that the overcurrent relay was not resetting due to a layer of dust on the induction disk of the relay. This layer of dust interfered with the field between the inductive disk and the stationary magnet in the relay. This prevented the disk from fully winding, which shortened the time for the disk to unwind to approximately 12 seconds after pump start, causing the pump to trip on overcurrent.

The overcurrent relay is an inductive type relay with an inductive disk that rotates on high current and resets. The speed the disk turns is governed by a stationary magnet just above the disk. On a RCP start, a large starting current is seen. The overcurrent relay winds and resets. The magnet allows the disk to slowly unwind. The time for the disk to unwind and t~ip is long enough to allow starting current to drop below the overcurrent trip point. Shortening the travel time of the disk, as in this event, shortens the time allowed for starting current to drop, resulting in pump trip on overcurrent.

~ dust was left from construction which had taken place in the switchgear room since the last time the relays : cleaned and calibrated.

- CAUSE OF EVENT: (Continued)

The 1A, 1B, and 1C RCP's had previously been run, and no problems of this type were encountered. The relays on all for RCP's are on an 18 month maintenance schedule for cleaning and calibrating.

- SAFETY ANALYSIS

There were no safety consequences for this event or existing conditions. This event would cause no safety consequences under the most limiting conditions of pump start. Starting a pump is not required by any accident analysis.

- CORRECTIVE ACTION

The relays were cleaned at Bus 159 cubicle 5 for 1D RCP. The relay calibrations were checked and verified to be correct. Construction is no longer taking place in any of the switchgear rooms.

- PREVIOUS OCCURRENCES

None

- COMPONENT FAILURE DATA

ne

TH17 RCP DEGRADED PERFORMANCE/LOCKED ROTOR

TYPE: GENERIC, NRB

A)	1A RCP	1RC01PA
B)	1B RCP	1RC01PB
C)	1C RCP	1RC01PC
D)	1D RCP	1RC01PD

CAUSE: SHAFT SEIZURE

REF: M-60 SHEET 1A
M-60 SHEET 2
M-60 SHEET 3
M-60 SHEET 4

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED RCP TO START SEIZING UP. THE SELECTED RCP MOTOR AMPS START TO INCREASE, AND THE ASSOCIATED LOOP FLOW INDICATORS START TO DECREASE SLOWLY. THE RCP MOTOR STATOR WINDING TEMPERATURES INCREASE.

THE REACTOR WILL TRIP DUE TO LOW COOLANT FLOW IN APPROXIMATELY 10 MINUTES AFTER ACTIVATING THE MALFUNCTION.

THE RCP WILL TRIP ON OVERCURRENT IN APPROXIMATELY 20 MINUTES WITH NO OPERATOR ACTION. THE SELECTED RCP TRIP LIGHT, AND STOP LIGHT ENERGIZE, MOTOR AMPS DECREASE TO ZERO, AND THE ASSOCIATED LOOP FLOW INDICATORS DECREASE TO 20%. ANNUNCIATORS 13-E3 "RCP TRIP" AND 13-A3/B3/C3/D3 "RCP 1A/1B/1C/1D BRKR OPEN OR LOW FLOW ALERT" ACTUATE. THE UNAFFECTED LOOP FLOWS WILL INCREASE.

THE SIMULATOR HAS TO BE RESET TO RETURN THE RCP TO NORMAL OPERATION.

EVENTS: NONE.

TH18 RCP SHAFT BREAK

TYPE: GENERIC, NRB

A)	1A RCP	1RC01PA
B)	1B RCP	1RC01PB
C)	1C RCP	1RC01PC
D)	1D RCP	1RC01PD

CAUSE: SHAFT FAILURE

REF: M-60 SHEET 1A
M-60 SHEET 2
M-60 SHEET 3
M-60 SHEET 4

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED RCP SHAFT TO BREAK. THE SELECTED MOTOR AMPS DECREASE TO THE MOTORS NO LOAD VALUE, AND THE ASSOCIATED LOOP FLOWS DECREASE TO 20%. ANNUNCIATOR 13-A3/B3/C3/D3 "RCP 1A/1B/1C/1D BRKR OPEN OR LOW FLOW ALERT" ACTUATES. THE UNAFFECTED LOOP FLOWS WILL INCREASE AND THE REACTOR WILL TRIP DUE TO LOW Rx COOLANT FLOW.

PZR LEVEL AND PRESSURE WILL DECREASE DUE TO THE Rx TRIP.

THE SIMULATOR HAS TO BE RESET TO RESTORE THE SELECTED RCP TO NORMAL.

EVENTS: 1) NRC IN 89-15

TH18

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

February 16, 1989

INFORMATION NOTICE NO. 89-15: SECOND REACTOR COOLANT PUMP SHAFT FAILURE AT CRYSTAL RIVER

Addressees:

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose:

This information notice is being provided to alert addressees to indications of potential sudden failure of a reactor coolant pump (RCP) shaft. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On January 18, 1989, the Crystal River Unit 3 plant experienced a loop "A" low coolant flow alarm and an automatic power runback from 75 percent of full power to 64 percent of full power. Operators noted a drop in the "A" reactor coolant pump motor current from 90 percent to 25 percent.

A preliminary review of the vibration and other coastdown data suggests that the pump shaft and the impeller have decoupled. This may be due either to fracture of the shaft itself or to failure of the cap screws and drive pins which hold the impeller to the shaft. The root cause of the failure will be more fully known when the pump is disassembled. The pump was manufactured by Byron Jackson.

Both the low flow alarm and motor current decreases were also symptomatic of the previous pump shaft failure in 1986.* During the 1986 event, pump vibration remained high after the shaft break, indicating interference to motor spin at the fracture interface, and after the pump was tripped, the pump motor rotation stopped within a few seconds. The licensee believes that the lack of pump vibration and the longer post trip motor coastdown after the recent pump failure indicate a lack of interference at the fracture interface.

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*The 1986 failure is described in Information Notice 86-19, "Reactor Coolant Pump Shaft Failure at Crystal River."

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Following the 1986 pump shaft failure, the licensee replaced the shafts in all four coolant pumps. Two of the reactor coolant pumps received new shafts of a different material (Alloy A-479 XM-19) and a different design. The new design did not contain the groove that was determined to be the crack initiation location for the 1986 fracture. One pump was fitted with a new shaft of the same material as that of the shaft that failed (Alloy A-286), but the licensee believes the new shaft did not contain a groove. The "A" pump was fitted with a new shaft of the same design and material as that of the shaft that had failed previously.

In addition, following the 1986 failure, the licensee refurbished and improved the vibration monitoring equipment on each coolant pump and located vibration monitor alarms on the main control panel. The reactor coolant pump vibration is continuously monitored by the Bently-Nevada Dynamic Data Manager System. This system monitors the motor casing accelerometer inputs along with the pump shaft proximity probes (X & Y, Keyphasor) on all four reactor coolant pumps.

Increased vibration on the "A" RCP was noted in November 1988. A review of the vibration monitoring data revealed a loss of rotor stiffness. The vibration monitor vendor (Bently-Nevada) believed that the pump shaft had cracked. The licensee examined the "A" RCP shaft with ultrasonic testing equipment and concluded that the shaft had not cracked. Cracks in the lower motor housing support were identified and corrected. After repair of the lower motor housing support, the licensee reported normal pump vibration. However, pump vibrations of varying magnitudes were again noted shortly thereafter.

The ultimate objective of the vibration monitoring system is to correlate the vibration data with crack growth and to provide an early warning such that a shaft break can be avoided. The program depends on an early detection of shifts in steady state values of maximum shaft displacement, first and second harmonics and corresponding phase angles. Since shifts in the second harmonic and its phase angle are sensitive indicators of changes in shaft stiffness and crack growth, particular attention to these parameters is important.

Additional RCP shaft failures are discussed in Information Notice 85-03, "Separation of Primary Reactor Coolant Pump Shaft and Impeller," and its supplement.

TH19 REACTOR VESSEL BOTTOM CRACK

TYPE: DISCRETE, NRVI 0-5,000 GPM

CAUSE: BRITTLE FRACTURE

REF: SYSTEM DESCRIPTION

PLT STA: REFUELING MODE

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A RUPTURE ON THE BOTTOM OF THE REACTOR VESSEL AT THE DESIRED SEVERITY. ANNUNCIATOR 6-C3 "REFUELING CAVITY LEVEL LOW ACTUATES. ANNUNCIATOR 12-A4 "PZR LEVEL LOW HTRS OFF LTDWN SECURED" ACTUATES AT 17% PZR LEVEL. ANNUNCIATOR 1-A2 "CNMT DRAIN LEAK DETECT FLOW HIGH" WILL ACTUATE AS WATER ACCUMULATES IN THE SUMPS. AT A LOW SEVERITY, THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY MAKING UP TO THE REFUELING CAVITY FROM THE RWST.

THE REFUELING CAVITY LEVEL INDICATORS, 1LI-RY046, 047, 048, AND 049, WILL INDICATE DECREASING LEVEL UNTIL THE INDICATORS ARE OFFSCALE LOW OR WHEN LEVEL HAS DECREASED TO APPROXIMATELY THE 400 FT LEVEL AT THE REACTOR VESSEL FLANGE.

CONTAINMENT AREA RADIATION MONITORS WILL SHOW INCREASING RADIATION LEVELS AS THE REFUELING CAVITY LEVEL DECREASES. AS THE CORE BECOMES UNCOVERED, RADIATION LEVELS WILL INCREASE DRAMATICALLY FROM THE RELEASE OF FISSION PRODUCTS TO THE ATMOSPHERE.

THE SIMULATOR MUST BE RESET TO RECOVER FROM THIS MALFUNCTION.

EVENTS: NONE.

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

TP01 STATOR COOLING WATER PUMP FAILS TO START/TRIP

TYPE: GENERIC, RB

A) 1GC01PA
B) 1GC01PB

CAUSE: FAULTY SH-TR RELAY ACTUATION

REF: 20E-1-4030 GC01
20E-1-4030 GC02

PLT STA: SELECTED PUMP IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED PUMP TO TRIP DUE TO A FAULTY TRIP RELAY. PUMP MOTOR AMPS DECREASE TO ZERO, THE TRIP LIGHT ILLUMINATES, AND ANNUNCIATOR 18-A14 "STATOR CLG WTR PUMP TRIP" ACTUATES. AT 20 PSID ACROSS THE AFFECTED PUMP, THE STANDBY PUMP AUTO STARTS, AND ANNUNCIATOR 18-D13 "H2/STATOR CLG PANEL TROUBLE" ACTUATES .

THE OPERATOR MAY RESET THE ANNUNCIATOR AND THE TRIP LIGHT BY PLACING THE CONTROL SWITCH IN THE TRIP POSITION. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL CLOSE THEN IMMEDIATELY TRIP OPEN.

MALFUNCTION REMOVAL RESTORES THE AFFECTED TRIP RELAY TO NORMAL.

EVENTS: NONE.

TP02 STATOR COOLING WATER HIGH CONDUCTIVITY

TYPE: DISCRETE, RB

CAUSE: DAMAGED DEMIN

REF: SYSTEM DESCRIPTION
20E-1-4030 HY03
20E-1-4030 HY04

PLT STA: 100% REACTOR POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE DEMINERALIZER TO BE DAMAGED, RESULTING IN A 10 μMHO/CM CONDUCTIVITY INCREASE OVER A 15 MINUTE PERIOD. ANNUNCIATOR 18-D13 "H2/STATOR CLG PANEL TROUBLE" ACTUATES AT 1.5 μMHO/CM. AFTER 15 MINUTES, THE GENERATOR INTERNALS WILL DEGRADE AND CAUSE A GENERATOR TRIP AND A REACTOR TRIP.

MALFUNCTION REMOVAL RESTORES THE DEMIN OUTPUT TO NORMAL.

EVENTS: NONE.

TP03 SEAL OIL SYSTEM PUMP FAILS TO START/TRIP

TYPE: GENERIC, RB

A)	1TO04SA	AIR SIDE SEAL OIL PUMP
B)	1TO04SB	H2 SIDE SEAL OIL PUMP
C)	1TO04SC	AIR SIDE SEAL OIL BACKUP PUMP

CAUSE: FAULTY M COIL

REF: 20E-1-4030 TO07,08
SYSTEM DESCRIPTION

PLT STA: AFFECTED PUMP IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE AFFECTED PUMP TO TRIP DUE TO A FAULTY M COIL. LOSS OF H2 PURITY AND PRESSURE IS INDICATED ON THE MCB. DESCRIPTIONS FOR EACH PUMP FAILURE ARE AS FOLLOWS:

1TO04SA - AIR SIDE SEAL OIL PUMP
TRIPPING THIS PUMP CAUSES THE AIR SIDE PRESS TO DECREASE AND ANNUNCIATOR 18-B12 "GEN AIR SIDE SEAL OIL PUMP TRIP" TO ACTUATE. AT 8 PSID, TURBINE OIL BACK-UP SUPPLIES PRESSURE TO THE AIR SIDE. AT 5 PSID, THE AIR SIDE SEAL OIL BACK-UP PUMP AUTO STARTS, AND ANNUNCIATORS 18-B13 "GEN AIR SIDE SEAL OIL B/U PUMP RUNNING" AND 18-D13 "H2/STATOR CLG PANEL TROUBLE" ACTUATE.

1TO04SB - H2 SIDE SEAL OIL PUMP
TRIPPING THIS PUMP CAUSES MORE AIR TO LEAK INTO THE H2 SIDE FROM THE AIR SIDE THAN NORMAL. ANNUNCIATORS 18-A12 "GEN H2 SIDE SEAL OIL PUMP TRIP" AND 18-D13 "H2/STATOR CLG PANEL TROUBLE" ACTUATE.

1TO04SC - AIR SIDE SEAL OIL BACK-UP PUMP
TRIPPING THIS PUMP WHILE THE AIR SEAL OIL PUMP IS NOT OPERATING CAUSES AIR SIDE PRESS TO DECREASE AND ANNUNCIATORS 18-A13 "GEN AIR SIDE SEAL OIL B/U PUMP TRIP" AND 18-D13 "H2/STATOR CLG PANEL TROUBLE" ACTUATE.

MALFUNCTION REMOVAL RESTORES THE AFFECTED M RELAY TO NORMAL.

EVENTS: NONE.

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

TU01	TURBINE VIBRATION
TU02	TURBINE BEARING OIL PUMP FAILS TO START/TRIP
TU03	TURBINE HP SEAL OIL B/U PUMP FAILS TO START/TRIP
TU04	TURBINE DC EMER OIL PUMP FAILS TO START/TRIP
TU05	TURBINE OIL SYSTEM LEAK
TU06	BEARING LIFT PUMP SUCTION STRAINER CLOGS

TU01 TURBINE VIBRATION

TYPE: GENERIC, RV 0-15 MILS (ADDITIVE)

A) BEARING 1
B) BEARING 2
C) BEARING 3
D) BEARING 4
E) BEARING 5
F) BEARING 6
G) BEARING 7
H) BEARING 8
I) BEARING 9
J) BEARING 10
K) BEARING 11

CAUSE: BEARING FAILURE

REF: SYSTEM DESCRIPTION

PLT STA: MAIN TURBINE IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED BEARING TO FAIL AND BEGIN TO VIBRATE. THE VIBRATION VALUE WILL BE DETERMINED BY THE SELECTED SEVERITY (MALFUNCTION VALUE IS ADDITIVE TO PRE-EXISTING VALUE) AND WILL BE INDICATED ON 1VR-TS002 (1PM02J). BEARING VIBRATION WILL INCREASE AS SELECTED SEVERITY IS INCREASED AND WILL ALSO INCREASE IN PROPORTION TO THE INPUT SEVERITY ON THE BEARINGS NEXT TO THE AFFECTED BEARING. ANNUNCIATOR 18-B16 "TURBINE SUPERVSRY ALARM STPT EXCEEDED" WILL BE ACTUATED AT 7 MILS ALONG WITH THE ASSOCIATED ALARM LIGHT ON THE TURBINE SUPERVISORY MONITOR PANEL. ANNUNCIATOR 18-B3 "TURBINE SUPERVSRY TRIP STPT EXCEEDED" ACTUATES AT 14 MILS ALONG WITH THE ASSOCIATED TRIP LIGHT ON THE TURBINE SUPERVISORY MONITOR PANEL. BRNG METAL AND OIL RETURN TEMPERATURES WILL INCREASE.

MALFUNCTION REMOVAL WILL RESTORE THE FAILED BEARING TO NORMAL.

EVENTS: NONE.

TU02 TURBINE BEARING OIL PUMP FAILS TO START/TRIP

TYPE: DISCRETE, RB

CAUSE: FAULTY SH-TR RELAY ACTUATION (1TO06P)

REF: 20E-1-4030 TO01

PLT STA: TURBINE START-UP

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE TURBINE BEARING OIL PUMP TO TRIP. THE TRIP LIGHT ILLUMINATES AND BEARING OIL PRESSURE (1PI-TO066 ON 1PM02J) DECREASES. BEARING OIL PUMP PRESSURE (1PI-TO067 ON 1PM02J) ALSO DECREASES. ANNUNCIATOR 18-B9 "BRNG OIL PUMP TRIP" ACTUATES AND THE EMERGENCY OIL PUMP AND TURBINE SEAL OIL B/U PUMP AUTO-START.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND THE TRIP LIGHT BY PLACING THE CONTROL SWITCH IN THE TRIP POSITION. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL CLOSE THEN IMMEDIATELY TRIP OPEN.

MALFUNCTION REMOVAL RESTORES THE FAULTY TRIP RELAY TO NORMAL.

EVENTS: NONE.

TU03 TURBINE HP SEAL OIL B/U PUMP FAILS TO START/TRIP

TYPE: DISCRETE, RB

CAUSE: FAULTY M RELAY (1TO07P)

REF: 20E-1-4030 TO02

PLT STA: SEAL OIL BACK-UP PUMP IN OPERATION

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SEAL OIL B/U PUMP TO TRIP. THE TRIP LIGHT ILLUMINATES, AND ANNUNCIATOR 18-A10 "SEAL OIL BACKUP PUMP TRIP" ACTUATES. WHEN THE TURBINE IS RUNNING AT <1800 RPM WITH MAIN OIL PUMP PRESSURE NOT ADEQUATE, AND THIS MALFUNCTION IS ACTIVATED, THE TURBINE WILL TRIP SINCE THE SEAL OIL B/U PUMP SUPPLIES THE AUTO STOP OIL PRESSURE.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND THE TRIP LIGHT BY PLACING THE CONTROL SWITCH IN THE TRIP POSITION. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL NOT CLOSE.

MALFUNCTION REMOVAL RESTORES THE FAULTY M COIL TO NORMAL.

EVENTS: NONE.

TU04 TURBINE DC EMER OIL PUMP FAILS TO START/TRIP

TYPE: DISCRETE, RB

CAUSE: FAULTY CR RELAY (1TO05P)

REF: 20E-1-4030 TO04

PLT STA: DC EMERGENCY OIL PUMP ON-LINE

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE DC EMERGENCY OIL PUMP TO TRIP. THE TRIP LIGHT ILLUMINATES AND BEARING OIL PRESSURE (1PI-TO066 ON 1PM02J) DECREASES. ANNUNCIATOR 18-C10 "EMERGENCY OIL PUMP TRIP" ACTUATES AND THE BEARING OIL PUMP AUTO STARTS (IF IN STANDBY). THE SEAL OIL BACK-UP PUMP AUTO STARTS. IF THE BEARING OIL PUMP AND SEAL OIL B/U PUMP ARE NOT IN STANDBY, THEN ANNUNCIATORS 18-D5 "BRNG OIL PRESS LOW", AND 18-D2 "BRNG OIL PRESS LOW TURBINE TRIP" (ALL OIL PUMPS OFF-LINE) ACTUATES.

THE OPERATOR MAY RESET THE ANNUNCIATOR AND THE TRIP LIGHT BY PLACING THE CONTROL SWITCH IN THE TRIP POSITION. IF THE OPERATOR ATTEMPTS TO RESTART THE PUMP, THE BREAKER WILL NOT CLOSE.

MALFUNCTION REMOVAL RESTORES THE FAULTY OVERLOAD TO NORMAL.

EVENTS: NONE.

TU05 TURBINE OIL SYSTEM LEAK

TYPE: DISCRETE, NRV 0-1000 GPM AT 350 PSID

CAUSE: PIPE BREAK ON DISCHARGE OF SHAFT DRIVEN MAIN TURBINE OIL PUMP.

REF: 20E-1-4030 TO02
20E-1-4030 TO09
M-152 SHEET 2
TURBINE OIL SYSTEM DESCRIPTION

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A LEAK IN THE DISCHARGE LINE OF THE SHAFT DRIVEN MAIN OIL PUMP. THE LEAK SIZE IS DEPENDENT UPON THE SELECTED SEVERITY. THE LEAK RESULTS IN A LOSS OF OIL FROM THE TURBINE OIL (TO) SYSTEM AS INDICATED ON THE OIL RESERVOIR LEVEL INDICATOR 1LI-TO004. THE OIL SYSTEM PRESSURE WILL ALSO DECREASE AS INDICATED ON THE SYSTEM PRESSURE INDICATORS 1PI-TO065 AND 1PI-TO066. AS THE TO RESERVOIR LEVEL DECREASES, ANNUNCIATORS 18-B11 "TURB OIL RSRVR LVL HIGH LOW", AND 18-C11 "TURB OIL RSRVR LVL L0-2" ACTUATE. AS BEARING OIL PRESSURE DECREASES, THE BEARING OIL PUMP (1TO06P) AND THE SEAL OIL B/U PUMP (1TO07P) WILL AUTO START. THE DC EMERGENCY OIL PUMP (1TO05P) MAY ALSO AUTO START. THE OIL PUMPS WILL EVENTUALLY CAVITATE AS INVENTORY IS LOST. THE MAIN TURBINE WILL ULTIMATELY TRIP WHICH WILL RESULT IN A REACTOR TRIP IF POWER IS >30%.

THE OPERATOR MAY LIMIT THE CONSEQUENCES OF THIS MALFUNCTION BY TRIPPING THE TURBINE EARLY IN THE MALFUNCTION.

MALFUNCTION REMOVAL WILL RESTORE THE DISCHARGE LINE PIPING INTEGRITY TO NORMAL.

EVENTS: NONE.

TU06 BEARING LIFT PUMP SUCTION STRAINER CLOG

TYPE: GENERIC, RV 0-100%

A) 1A BEARING LIFT PUMP 1TO08PA
B) 1B BEARING LIFT PUMP 1TO08PB
C) 1C BEARING LIFT PUMP 1TO08PC
D) 1D BEARING LIFT PUMP 1TO08PD
E) 1E BEARING LIFT PUMP 1TO08PE
F) 1F BEARING LIFT PUMP 1TO08PF

CAUSE: CLOGGED FILTER ON PUMP SUCTION

REF: 20E-1-4030 TO11 - TO14
M-152 SHEET 2
TURBINE OIL SYSTEM DESCRIPTION

PLT STA: MAIN TURBINE BELOW 600 RPM

EFFECTS: INSERTING THIS MALFUNCTION CAUSES THE SELECTED TURBINE OIL BEARING OIL LIFT PUMP STRAINER TO CLOG. THIS IN TURN WILL CAUSE THE ASSOCIATED BEARING OIL LIFT PUMP TO TRIP ON LOW SUCTION PRESSURE. WHEN THE PUMP TRIPS, THE SUCTION PRESSURE MAY BE RESTORED AND THE BEARING OIL LIFT PUMP MAY AUTO-START. THIS CYCLING PROCESS WILL CONTINUE AT A RATE DEPENDENT UPON THE STRAINER BLOCKAGE. AT 100% SEVERITY, THE BEARING OIL LIFT PUMP WILL NOT RESTART DUE TO THE MOTOR THERMAL OVERLOADS TRIPPING. THE BEARING OIL LIFT PUMP RUNNING/STOPPED INDICATION ON 1PM03J WILL CYCLE AS INDICATED ON 1EL-TO094. ANNUNCIATOR 18-C9 "TURB BRNG LIFT PUMP TROUBLE" IS ACTUATED ON LOW SUCTION PRESSURE. THE TURNING GEAR WILL STOP IF IT IS IN LOCAL CONTROL WHENEVER ANY LIFT PUMP STOPS.

MALFUNCTION REMOVAL WILL RESTORE THE SUCTION STRAINER TO NORMAL.

EVENTS: NONE.

BRAIDWOOD SIMULATOR

MALFUNCTION CAUSE AND EFFECTS

WD01 GAS DECAY TANK RUPTURE

WD01 GAS DECAY TANK RUPTURE

TYPE: GENERIC, RV 0-2000 SCFM AT 95 PSID

A) TANK A
B) TANK B
C) TANK C
D) TANK D
E) TANK E
F) TANK F

CAUSE: TANK RUPTURE

REF: M-78 SHEETS 1,3,9
M-95 SHEETS 2,9,11,12, & 14

PLT STA: REACTOR AT POWER

EFFECTS: INSERTING THIS MALFUNCTION CAUSES A RUPTURE OF THE SELECTED GAS DECAY TANK. THE TANK RELEASES ITS CONTENTS TO THE SURROUNDINGS. THE RADIOACTIVE GAS WILL BE TRANSPORTED THROUGH AUX BUILDING VENTILATION SYSTEM CAUSING RADIATION MONITORS 0PR13J AND 1PR28J TO ALARM ON THE RM-11.

MALFUNCTION REMOVAL WILL RESTORE THE GAS DECAY TANK TO NORMAL.

EVENTS: NONE.