LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
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Title (4) Reactor Trip Due to Control Rod Drop Caused by Intermittent Component Failure in the Rod Drive System

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)
									NONE	0 5 0 0 0
0 6	0 2	8 8	8 8	--- 0 0 6 ---	0 0	0 6	2 1	8 8		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
Don Brindle, Operating Engineer Ext. 2218	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 0640 on June 2, 1988, with the reactor at 94% power and the Rod Control Selector Switch in AUTOMATIC, the "Power Range Flux Rate High Reactor Trip" annunciator actuated and the reactor tripped. Control Room operators entered and complied with "Reactor Trip or Safety Injection Unit 2 Emergency Procedure." The Auxiliary Feedwater Pumps started due to low-low steam generator levels that resulted from the trip at high power. Stable plant conditions were achieved in Hot Standby at 0730 on June 2, 1988.

The intermediate cause of the reactor trip was the dropping of control rods into the reactor core, which resulted in the high flux negative rate reactor trip. Troubleshooting efforts failed to determine a root cause of the dropped rods. It is believed that an intermittent component failure in the rod drive system caused the event, but the component did not remain in the failed mode following the reactor trip.

Troubleshooting was conducted to locate component failures that may have caused the rod drop. A number of degraded fuses were identified in the rod drive power cabinets and all were replaced. On June 3, 1988, all approvals required to start up the plant following a reactor trip for which no root cause has been determined were obtained and the unit entered the Startup operational mode.

Previous occurrences of reactor trips caused by dropped rods on Unit 1 are documented in Licensee Event Reports 85-042, 85-063, 86-028 and 88-002.

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8806300205 880621
PDR ADOCK 05000455
S DCD

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

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 6/2/88 / 0640

Unit 2 MODE 1 - Power Operations Rx Power 94% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

The Rod Control Selector Switch was in the AUTOMATIC position and no rod motion was demanded by the rod control system. Shift relief was in progress and the oncoming Nuclear Station Operator (NSO) (licensed reactor operator) was clearing a mechanical jam of the Process Computer Alarm Typer. At 0640 on June 2, 1988, with Unit Two operating at 94 percent reactor power, the Power Range Nuclear Instruments [IG] sensed a high negative flux rate condition indicative of dropped control rods. The "Power Range Flux Rate High Reactor Trip" annunciator actuated in the Main Control Room, and the reactor trip breakers opened. The licensed control room operators entered and complied with "Reactor Trip or Safety Injection Unit 2 Emergency Procedure" (2BEP-0). A Feedwater Isolation occurred due to the Reactor Trip and low Reactor Coolant Average Temperature (T_{avg}), which is an expected Engineered Safety Features (ESF) actuation following a reactor trip. The Auxiliary Feedwater Pumps (AFP) [BA] automatically started in response to low-low steam generator levels caused by indicated level shrink after the trip. No annunciators related to the Rod Drive System (RD) [AA] actuated prior to the trip to indicate a possible cause. Expected post trip rod drive Urgent Failure alarms were actuated for voltage regulation failures due to the opening of the reactor trip breakers. The lack of urgent alarms prior to the trip suggested blown fuses as a possible cause for a rod drop event that would cause the negative rate reactor trip.

At 0655 the Startup Feedwater Pump [SJ] was started. At 0656 the NSO manually reenergized the Source Range Instruments because Intermediate Range Channel N35 appeared to be undercompensated and would delay the automatic reenergization. At 0707 the Feedwater Isolation signal was reset and a feedwater flow path from the Startup Feedwater Pump to the steam generators was established. At 0725 the diesel-driven AFP was stopped and at 0728 the motor-driven AFP was stopped. At approximately 0730 stable plant conditions were achieved with Unit Two in Hot Standby (Mode 3).

All Operator actions taken during this event were correct and contributed to its safe conclusion. This Licensee Event Report (LER) is submitted in accordance with 10CFR50.73 (a)(2)(iv) due to the automatic ESF actuations.

C. CAUSE OF EVENT:

The intermediate cause of the reactor trip was the dropping of control rods into the reactor core, which resulted in the power range flux negative rate high reactor trip. It was determined that rods dropped from the 1AC rod drive power cabinet, initiating the event. Three Stationary Gripper Phase fuses were blown during the event. Other Stationary Gripper Phase fuses remained intact and were capable of maintaining all control rods in their withdrawn positions. Therefore, the three blown fuses alone could not have caused multiple dropped rods. Fuses that conduct current to the Control Rod Drive Mechanism (CRDM) coils from the RD power cabinets were resistance checked using a Digital Low Resistance Ohmmeter, and several fuses had resistance readings slightly higher than permitted by acceptance criteria. The slight degradation of these fuses may have contributed to the control rod drop by limiting current flow to the CRDM coils, however, the minor nature of the fuse degradation does not provide conclusive evidence for designating these fuses as the root cause. Troubleshooting efforts failed to determine a root cause for the dropped rods. It is believed that an intermittent component failure in the rod drive system caused the event, but the component failure did not continue following the trip.

C. CAUSE OF EVENT: (Continued)

Intermediate Range Channel N35 was determined to be properly compensated. The P-6 permissive that automatically reenergizes the Source Range Instruments actuated shortly after the manual reenergization, within the expected 15 to 18 minute time frame following the reactor trip.

D. SAFETY ANALYSIS:

All ESF systems actuated and performed as designed. The Reactor Protection System [JG] responded properly to a power range flux high negative rate condition sensed by two of the four power range nuclear instruments. The Auxiliary Feedwater Pumps automatically started due to low-low steam generator levels that resulted following the reactor trip from high power. At no time during this event was the health or safety of the public adversely affected or threatened.

E. CORRECTIVE ACTIONS:

Immediately following the reactor trip on June 2, 1988, the rod drive power cabinet fuses were inspected for blown fuse indicators and the Process Computer Alarm Typer jam was cleared. Blown fuse indicators were actuated on the following three fuses:

Fuse	Cabinet	
Phase A Stationary Grippers Group 1	2RD03J	2BD
Phase B Stationary Grippers Group 3	2RD02J	SCDE
Phase B Stationary Grippers Group 1	2RD05J	1AC

The rod drive motor-generator (MG) sets' protective relays were checked for targets indicating trouble, and none were found. By 0830 the rod drive conditions at the time of the trip had been documented. At 1130 the RD bus duct was checked for shorts, both phase to phase and phase to ground, but no shorts were found. At 1200 the three fuses associated with the blown fuse indicators were verified to be bad. The balance of the stationary phase fuses were checked using a Digital Low Resistance Ohmmeter (DLRO). By visual and DLRO checks, 23 fuses were found to be either cracked or electrically unacceptable, so all stationary phase fuses in all power cabinets were replaced. From 1530 to 2150 the fuses leading out to the Control Rod Drive Mechanism's (CRDMs) coils were resistance checked utilizing "CONTAINMENT PENETRATION CONDUCTOR PROTECTIVE DEVICES 260 VAC RCD POWER (FUSES) Electrical Surveillance Procedure" (2BHS 8.4.1.a.3-1) and the fuse clips tightened. Although 14 fuses were found to be outside acceptance criteria for the surveillance, none had degraded enough to have caused rods to drop. All fuses failing the acceptance criteria were replaced, and the replacement fuses were checked before installation. In parallel with the fuse checks, the cables to the stationary coils on the reactor head package from RD power cabinet 1AC were checked for circuit shorts, and none were found. On June 3, 1988 at 0307, the RD system was energized and reset, but an urgent alarm remained actuated on RD power cabinet SCDE. The card edge lights on the error detector circuit cards indicated a stationary gripper voltage regulation failure. The voltages into the thyristors and out of the power supplies were checked and found to be acceptable. After troubleshooting and swapping out the associated firing circuit card and the signal processing circuit card, the urgent alarm was cleared and determined to be due to a loose card edge connector on the stationary gripper firing circuit card. A cracked solder connection on a capacitor was fixed and the card edge connectors tightened. The rods were exercised from 0915 to 1115 to verify proper operations. Approval for startup following a reactor trip for which no root cause has been determined was obtained and the plant entered the Startup operational mode at 1227 on June 3, 1988.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

E. CORRECTIVE ACTIONS: (Continued)

As a result of the fuse discrepancies discovered on Unit 2, fuses on Unit 1 were resistance checked on June 9, 1988. One fuse was determined to be broken and was replaced with a new fuse.

Personnel from Commonwealth Edison's Pressurized Water Reactor facilities and Corporate Offices have joined to form a task force, whose goal is to reduce reactor trips caused by rod drive system problems.

F. PREVIOUS OCCURRENCES:

LER NUMBER	TITLE
85-042 (Unit 1)	Reactor Trips Due to Dropped Rods
85-063 (Unit 1)	Reactor Trip Due to Turbine Trip Above P-7
86-028 (Unit 1)	Manual Reactor Trip Due to Rod Drop Caused by Faulty Circuit Cards
88-002 (Unit 1)	Reactor Trip Due to Rod Drop During Manual Control Rod Motion

G. COMPONENT FAILURE DATA:

a) MANUFACTURER NOMENCLATURE MODEL NUMBER MFG PART NUMBER

Not Applicable

b) RESULTS OF NPRDS SEARCH:

Not Applicable

Commonwealth Edison
Byron Nuclear Station
4450 North German Church Road
Byron, Illinois 61010

DATE: June 21, 1988

LTR: BYRON 88-0608

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Dear Sir:

The enclosed Licensee Event Report from Byron Generating Station is being transmitted to you in accordance with the requirements of 10CFR50.73(a)(2)(iv).

This report is number 88-006; Docket No. 50-455.

Very truly yours,

T. P. Joyce
for R. Pleniewicz
Station Manager
Byron Nuclear Power Station

RP/RJP/bb (1921M/0206M)

Enclosure: Licensee Event Report No. 88-006-00

cc: A. Bert Davis, NRC Region III Administrator
P. Brochman, NRC Senior Resident Inspector
INPO Record Center
CECo Distribution List

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