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

October 21, 1994

LTR: BYRON 94-0417
FILE: 3.03.0800 (1.10.0101)

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 94-003; Docket No. 50-455.

Sincerely,

G.K. Schwartz

G.K. Schwartz
Station Manager
Byron Nuclear Power Station

GKS/DSK/bl

Enclosure: Licensee Event Report No. 94-003

cc: J. Martin, NRC Region III Administrator
NRC Senior Resident Inspector
INPO Record Center
CECo Distribution List

9410270260 941021
PDR ADOCK 05000455
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(9915R/WPF/101394-5)

JE22

SIGNATURE PAGE FOR LICENSE EVENT REPORT

LER Number
455:94-003

Title of Event: Auto Start of the 2B Auxiliary Feedwater Pump on Lo Lo Steam Generator Level Due to Equipment Failure

Occurred: 09/24/94 / 1045
Date Time

OSR DISCIPLINES REQUIRED: ABFG

JFS / 10/18/94
SES DATE

Acceptance by Station Review:

M Rasmussen / 10-20-94 ABCFG
OE Date

Joseph Lange / 10/21/94
SES Date

D Brindle / 10/21/94
RAS ABG Date

_____ / _____
OTHER Date

Approved by: GK Schwartz / 10/21/94
Station Manager Date

(9915R/WPF/101.94-1)

LICENSEE EVENT REPORT (LER)

FACILITY NAME	DOCKET NUMBER	PAGE
BYRON NUCLEAR POWER STATION	0 5 0 0 0 4 5 5	1 OF 0 3

TITLE

AUTO START OF THE 2B AUX FEED PUMP ON LO LO STEAM GENERATOR LEVEL DUE TO EQUIPMENT FAILURE.

EVENT DATE			LER NUMBER			REPORT DATE			OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQ. NUMBER	REVISION	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
0 9	2 4	9 4	9 4	- 0 0 3 -	0 0	1 0	2 1	9 4	NONE	0 5 0 0 0
										0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (CHECK ONE OR MORE OF THE FOLLOWING)

OPERATING MODE: 3		20.402(b)		20.405(e)	X	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL: 0		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, NRC Form 366A)
		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

NAME	TELEPHONE NUMBER
BRAD JACOBSEN, OPERATING DEPARTMENT EXT. 2622	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				

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

While performing a surveillance for post maintenance testing of the Unit 2 Rod Control System, a rod dropped into the core resulting in a reactor trip and corresponding Aux Feedwater actuation. When Steam Generator levels had been restored above the Lo Lo level setpoint, the 2B Aux Feed Pump was secured. When the Steam Dumps responded to an erratic signal from the Main Steam Header Pressure Controller, level in the Steam Generator dipped below the Lo L level setpoint which caused the 2B Aux Feed Pump to automatically start.

The root cause of this event was a failure of the Main Steam Header Pressure Controller.

To prevent recurrence, the Controller has been replaced.

This event is reportable per 10CFR 50.73(A)(2)(iv) any condition that results in the actuation of the ESF System.

(9915R/WPF/101394-2)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME	DOCKET NUMBER	LER NUMBER					PAGE
		YEAR		SEQ. NUMBER		REVISION	
BYRON NUCLEAR POWER STATION	0 5 0 0 0 4 5 5	9 4	-	0 0 3	-	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 09/24/94 / 1045

Unit 1 MODE 6 - 0% Rx Power RCS [AB]Temperature/Pressure ~86°F Refueling

Unit 2 MODE 3 - 0% Rx Power RCS [AB] Temperature/Pressure ~557°F Trip Recovery

B. DESCRIPTION OF EVENT:

On 09/24/94, testing was being performed on the Unit 2 Rod Drive System (RD)[AA]. When Rod C-7 dropped in the core at 1012 hrs, a Unit 2 Reactor Trip occurred on Power Range Flux Negative Rate. All plant safety systen responded as required, including automatic start of the 2A and 2B Aux Feedwater (AF)[BA] Pump. (LER 455:94-002 discusses this event.)

At 1036 hrs, the Unit 2 Operator started the Startup Feedwater (FW)[SJ] Pump in preparation for transitioning from Aux Feedwater to normal feedwater flowpath. When Steam Generator levels reset above the Lo Lo level reset setpoint, flow was throttled from the 2B Aux Feedwater Pump to minimize the plant cooldown. At 1040 hr with feedwater flow from the 2B Aux Feedwater Pump dialed to a minimum and Steam Generator levels relatively stable above the Lo Lo setpoint, the 2B Aux Feedwater Pump was secured and reset to provide designed Steam Generator level protection.

With 2PK-507, Main Steam Header Pressure Controller, in Steam Pressure Mode, a potentiometer is dialed to provide a setpoint for Steam Generator pressure. The controller compares this setpoint to actual Steam Generatc pressure and sends an appropriate signal to control the Steam Dump valve positions. Unknown to the Operator, the potentiometer was not working properly and erratic signals were being sent to the Steam Dumps. At 1045 hrs, a spike in the signal from the faulty potentiometer caused Steam Dumps to open, then rapidly close. Rapid closure of the Steam Dumps caused the level in the 2C Steam Generator to drop below the Lo Lo Steam Generatc level setpoint which resulted in an Auto Start of the 2B Aux Feed Pump. 2PK-507 was placed in manual to bypa the failed potentiometer.

After the Auto Start of the 2B Aux Feedwater Pump, it remained in operation until 1154 hrs, when it was again secured and reset to provide designed Steam Generator level protection.

This event is reportable per 10CFR 50.73(a)(2)(iv) any condition that results in the actuation of the Engineered Safeguards System.

C. CAUSE OF EVENT:

The cause of this event was a faulted potentiometer in 2PK-507, Main Steam Header Pressure Controller.

D. SAFETY ANALYSIS:

There were no safety consequences impacting plant or public safety as a result of this event. Throughout the incident, all Safety Systems operated as designed.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME	DOCKET NUMBER	LER NUMBER					PAGE
		YEAR		SEQ. NUMBER		REVISION	
BYRON NUCLEAR POWER STATION	0 5 0 0 0 4 5 5	9 4	–	0 0 3	–	0 0	0 3 OF 0 3

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

E. CORRECTIVE ACTIONS:

2PK-507, Main Steam Header Pressure Controller was replaced under Work Request #940065554 on 09/25/94

G. COMPONENT FAILURE DATA:

Manufacturer	Description	Model Number	Serial Number
Westinghouse	M/A Station	6629D75-G01	2224