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

July 6, 1992

Ltr: BYRON 92-0462

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 92-003; Docket No. 50-455.

Sincerely,

R. Pleniewicz
Station Manager
Byron Nuclear Power Station

RP/CW/mw

Enclosure: Licensee Event Report No. 92-003

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

146694

(0932R/VS)

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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 4

Title (4)

Unit 2 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 5 0 0 0
0 6	1 0	9 2	9 2	---	0 2 6	--- 0	0 7	0 9	9 2		0 5 0 0 0

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)			
	1	___ 20.402(b)	___ 20.405(c)	X 50.73(a)(2)(iv)	___ 73.71(b)
POWER LEVEL (10)	1 0 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
		___ 20.405(a)(1)(iii)	___ 50.73(a)(2)(i)	___ 50.73(a)(2)(viii)(A)	in Abstract
		___ 20.405(a)(1)(iv)	___ 50.73(a)(2)(ii)	___ 50.73(a)(2)(viii)(B)	below and in
		___ 20.405(a)(1)(v)	___ 50.73(a)(2)(iii)	___ 50.73(a)(2)(x)	Text)

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER
	AREA CODE
W. Scheffler, Technical Staff Engineer, Ext. 2378	
W. Kouba, 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	
X	J B	F C V	F 1 3 0	Y							

SUPPLEMENTAL REPORT EXPECTED (14)		Expected Submission Date (15)	Month	Day	Year
X Yes (If yes, complete EXPECTED SUBMISSION DATE)	___ NO		0 8	2 5	9 2

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

On 06/10/92, at 1315, a Feedwater (FW) [JB] alarm (Steam Generator (S/G) 2C Level Deviation-Low) was received on the "C" loop steam generator. Initial attempts to stabilize steam generator level were successful, however, the 2FW530 ("C" loop Steam Generator Feedwater Regulating Valve) subsequently failed closed, and a manual reactor trip was initiated at 1325. All rods inserted and all systems functioned normally. In addition to the reactor trip, the Auxiliary Feedwater pumps Auto-started on Low-2 Steam generator level. Steam generator level was recovered to normal via the Auxiliary Feedwater system.

The cause of the this event was the failure of the valve operator diaphragm. This led to instantaneous loss of air pressure and immediate closure of the 2FW530 valve.

All Unit two feedwater regulating valve diaphragms were replaced based upon an external inspection of the diaphragm. Additionally the diaphragm casing torque was increased per vendor recommendations.

This event is reportable per 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.

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 06/10/92 / 1325

Unit 1 MODE 1 - Power Operation Rx Power 100% RCS [AB] Temperature/Pressure NOT / NOP

Unit 2 MODE 2 - Startup Rx Power 100% RCS [AB] Temperature/Pressure NOT / NOP

B. DESCRIPTION OF EVENT:

At 1315, on 06/10/92, a Feedwater (FW) [JB] alarm (Steam Generator (S/G) 2, C Level Deviation-Low) was received on the "C" loop steam generator. The Nuclear Station Operator (NSO) placed the 2FW530 Main Feedwater regulating valve in manual in order to increase FW flow and dispatched an equipment attendant (EA) to investigate locally. The EA reported an air leak from the valve operator diaphragm. At 1324, the NSO placed the FW pump Master Speed Controller (2SC-509A) in manual to increase the FW header pressure to force more flow through the partially open 2FW530 valve. At 1325, a load drop of 5 MW/minute was initiated to allow for closure of the 2FW530 valve at 20% Reactor power, but the 2FW530 valve failed full closed causing the loss of FW flow and subsequent rapid decrease in S/G 2C level. A manual reactor trip was initiated and Byron Emergency Procedure BEP-0 was entered. All rods inserted and all systems functioned normally. In addition to the manual reactor trip which caused a feedwater Isolation, the Auxiliary Feedwater pumps auto-started on Low-2 Steam generator level. Steam generator level was restored to normal via the Auxiliary feedwater system.

A review of the steam generator level trace, after the event by the root cause investigation team, showed a slight change in pattern approximately 2 hours prior to the event. It is postulated that this may have been the start of the diaphragm failure. This change was noted by the operator but was deemed to be within the normal variances seen in daily feedwater regulating valve control.

This event is reportable in accordance with 10CFR50.73(a)(2)(iv) any event or condition that results in manual or automatic actuation of any Engineered Safety Feature.

C. CAUSE OF EVENT:

The root cause of the 2FW530 regulating valve closure was the failure of the operator diaphragm due to bolt hole elongation with insufficient clamping forces on the diaphragm in the diaphragm casing. The failure of the diaphragm along a "5 bolt length" section caused an 8 inch by one-half inch crescent-shaped piece of the diaphragm to separate at the five bolt holes. The diaphragm then blew inward and tore radially which led to complete instantaneous loss of air pressure and immediate closure of the 2FW530 valve, which is a "fail closed" valve. The insufficient clamping force may have been due to insufficient torquing of the bonnet bolts and/or the use of a Room Temperature Vulcanization (RTV) sealant on the sealing surface of the diaphragm. The RTV used was Permatex 6B. The Permatex was applied by a Mechanical Maintenance worker to prevent air leakage from the bonnet which was beyond the work instructions for the repair.

The diaphragm was sent to System Materials Analysis Department (SMAD) for analysis. The failure of the diaphragm was due to fatigue failure as opposed to transient failure. The differences in shear capability between the diaphragm material (Buna-N) and the Permatex, due to different coefficients of friction, may have allowed the diaphragm to move slightly during normal operation of the 2FW530 valve. The movement may have caused a loss of clamping forces on the diaphragm, which then tore away from the bolt holes in the diaphragm. Additionally, the torque value of 20 ft.-lbs., specified by the vendor manual, may have been inadequate to ensure proper clamping forces. The specified torque value is currently being reviewed by the vendor (Fisher).

D. SAFETY ANALYSIS:

All safeguard equipment functioned as designed which resulted in maintaining the Reactor in a safe condition throughout the event. The 2C S/G Lo-2 Level caused an Auxiliary Feedwater initiation, as designed, on a loss of main feedwater to maintain the secondary heat sink. The safety significance would be the same if the same events occurred under any different initial conditions.

E. CORRECTIVE ACTIONS:

A root cause investigation team (HPES 92-08) was immediately established to review the following items:

1). Impact on Unit One FW regulating valve operation.

2). Root cause and corrective actions required for Unit Two startup.

3). Performing a material analysis on the ruptured diaphragm.

4). Obtaining engineering assistance/recommendations.

5). Obtaining vendor (Fisher) recommendations with respect to proper torque values and the use of Permatex as a sealant.

A supplemental LER will be issued by August 25, 1992, when the final recommendations from the HPES team are issued, and will be tracked by NTS #4552009202600-51. Included in the investigation will be an evaluation of the maintenance worker going beyond the work instructions for the repair.

The 2FW530 regulating valve diaphragm was replaced along with the other three Unit Two regulating valve diaphragms based on an external inspection of the diaphragms which indicated the presence of Permatex.

In addition, the vendor (Fisher) recommended increasing the diaphragm casing torque from 20 to 40 foot pounds, which was done on all four Unit Two feedwater regulating valves. The Unit One feedwater regulating valves were also inspected. There was no Permatex sealant found on the edge of the exposed diaphragm. No increase in torque value was deemed necessary at this time. The vendor recommendations with respect to torque values will be reviewed by the Station and further actions will be taken for Unit 1 and Unit 2 as deemed appropriate. The vendor is reviewing the torque values specified. When the results of this review are received, Byron will review and take corrective actions as required. This will be tracked by NTS #4552009202600-01.

F. RECURRING EVENTS SEARCH AND ANALYSIS:

a) EVENT SEARCH (DIR, LER)

This is the first diaphragm failure recorded on these valves. Previous corrective action to prevent diaphragm failure occurred during Unit 1 refueling outage B1R04, when an inspection showed a leaky diaphragm that had become "hardened". This indication resulted in all four diaphragm changeouts on Unit One during B1R04 in September 1991; and the subsequent change out of all four Unit Two diaphragms during B2R03 in March, 1992. Of the eight diaphragms changed out, only the first diaphragm (1FW510), showed any signs of degradation.

F. RECURRING EVENTS SEARCH AND ANALYSIS: (continued)

b) INDUSTRY SEARCH (OPEX's NPRDS)

O&MR 386 Reactor Transients Resulting from Feedwater Regulating Bypass Valve Problems.

c) NWR

2FW510	B93608	and	B93717
2FW520	B93610	and	B93718
2FW530	B93594	and	B93719
2FW540	B93609	and	B93720

d) ANALYSIS

None

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Fisher Controls	16" AOV Flow Control Valve	SS-137 ENA	Type 80

DEVIATION REPORT

DVR NO. 06 - 02 - 92 - 026
STA UNIT YEAR NO.

Form Rev 2.0

PART 1 | TITLE OF DEVIATION

Unit 2 Reactor Trip due to C Loop Feedwater Regulating Valve Failing Closed.

OCCURRED 06/10/92 1315
DATE TIME

SYSTEM AFFECTED	PLANT STATUS AT TIME OF EVENT	WORK REQUEST NO.	TESTING
FW	MODE 1 POWER(%) 100%	93594/93719	[ ] YES [X] NO

DESCRIPTION OF EVENT

Unit 2 was manually tripped after the "C" loop Feed Reg Valve (2FW530) failed closed.

POTENTIALLY PART 21 [ ] YES* *FORWARD TO THE NUCLEAR ENGINEERING MANAGER
[X] NO FOR AN EXPANDED REVIEW

POTENTIALLY SIGNIFICANT EVENT PER NOD DIRECTIVE OP.10 [ ] YES [X] NO

10CFR50.72 NRC RED PHONE [ ] 1 HOUR
NOTIFICATION MADE [X] 4 HOUR 1357 TIME [ ] NO

RESPONSIBLE SUPERVISOR D. Flowers DATE 06/10/92

PART 2 | OPERATING ENGINEER'S COMMENTS

[ ] NON REPORTABLE EVENT

[X] 30 DAY REPORTABLE/10CFR 50.72(1)(2)(ii)
50.73(a)(2)(iv)

[ ] ANNUAL/SPECIAL REPORT REQUIRED

A.I.R. #

L.E.R. #

NOTIFICATION

REGION III	DATE	TIME
Office of M. Wallace	06/11/92	1515
NSD	DATE	TIME

[ ] CECO CORPORATE NOTIFICATION MADE IF ABOVE NOTIFICATION IS PER 10CFR21

TELECOPY
CECO CORPORATE OFFICER DATE TIME

PRELIMINARY REPORT
COMPLETED AND REVIEWED T. Didier 06/11/92
OPERATING ENGINEER DATE

INVESTIGATION REPORT & RESOLUTION
ACCEPTED BY STATION REVIEW W Koelsa 7-2-92 R. Wegner 7-6-92

RESOLUTION APPROVED AND
AUTHORIZED FOR DISTRIBUTION [signature] 7/2/92
STATION MANAGER DATE

86-5176 (Form 15-52-1) 4/12/90

DOCUMENT ID

(0932R/VS-1)

SHEET 7

EVENT SUMMARY AND CAUSE CODES

DVR Number
86-2-98-126

- [ ] Lost generation
- [ ] Cost > $25,000
- [ ] Hazard or Spill
- [ ] Personnel injury
- [X] Reactor trip
- [ ] ESF actuation
- [X] NRC reportable
- [X] LER
- [ ] PSE
- [ ] NRC violation, level___
- [ ] GSEP event, class_____
- [ ] Tech Spec LCO
- [ ] Potential or future loss
- [ ] SALP functional area___

Component type | Failure mode | Department

	Component type	Failure mode	Department
X	C	M	M	Valve diaphragm Bleach
X				
X				

Licensed? L or blank | Level | Department | Type | Detail code

	Licensed? L or blank	Level	Department	Type	Detail code
A					
A					
A					

Type | Detail Code | Department

	Type	Detail Code	Department
B			
B			
B			

	Type	Detail code
C		

Type of deficiency | Detail code | Procedure type

	Type of deficiency	Detail code	Procedure type
D			
D			
D			

Type | Detail code | Department

	Type	Detail code	Department
E			
E			
E			