Commonwealth Edison Company
Byron Generating Station
4450 North German Church Road
Byron, IL 61010-9794
Tel 815-234-5441

ComEd

October 22, 1996

LTR: BYRON 96-0239
FILE: 3.03.0800 (1.10.0101)

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

Dear Sir:

The enclosed Supplemental Licensee Event Report from Byron Generating Station is being transmitted to you.

This report is number 96-011, Supplement 1; Docket No. 50-454.

Sincerely,

K. L. Kofron
Station Manager
Byron Nuclear Power Station

KLK/WD/ld

Enclosure: Licensee Event Report No. 96-011, Supplement 1

cc: A. B. Beach, NRC Region III Administrator
NRC Senior Resident Inspector
INPO Record Center
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NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
BYRON NUCLEAR POWER STATION	05000454	1 OF 3

TITLE (4)

Manual Reactor Trip Due to Loss of Instrument Air to Feedwater Preheater Bypass Isolation 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 NAME	DOCKET NUMBER
										05000
07	02	96	96 --	011 --	01	9	10	96	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	15%		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Douglas J. Bowman, System Engineer	815-234-5441 X2385

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
AW	SJ								

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 15 single-spaced typewritten lines) (16)

On 07/02/96 at approximately 0312, the 1B Steam Generator Feedwater Preheater Bypass Isolation Valve failed close due to a broken Instrument Air line. This valve failure resulted in inadequate feedwater to the 1B steam generator. The Unit 1 Reactor was manually tripped prior to reaching the automatic trip on Lo-2 steam generator level. The valve closed due to a broken fitting that supplied air to the actuator. The piping that supplies air to this actuator is copper with soldered joints. One of these soldered joints broke causing the failure.

The failed tubing was replaced. The section of tubing that failed was sent to System Materials Analysis Dept. (SMAD) for determination of the specific failure mode.

This event is reportable pursuant to the requirements of 10CFR50.73(a)(1) and 50.73(a)(2)(iv).

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 07-02-96 / 0312

Unit 1 Mode 1 - 15% Rx Power RCS [AB] Temperature/Pressure NOT/NOP

Unit 2 Mode 1 - 100% Rx Power RCS [AB] Temperature/Pressure NOT/NOP

B. DESCRIPTION OF EVENT:

On 7/2/96 at approximately 0300, Unit 1 was in Mode 1 (Power Operations) at 15% reactor power (566° F Tave) and was making preparations to synchronize the main generator to the grid. There were no systems out of service at the time of the event that had any impact on the event; however, at 15% power, the plant is in a unique position with respect to the feedwater system alignment. At this power level both the S/G Tempering flow isolation valves (1FW035A-D) and the S/G preheater bypass valves (1FW039A-D) are required to be open to supply sufficient feedwater flow to the S/G. The Main Feedwater Isolation Valves cannot be opened until the plant is above approximately 20% power due to the purge and flow permissives interlocks associated with these valves. Had the plant been between 20% and 80% or less than 10%, there would have been little impact due to this event. Had the plant been above 80%, it would have to have been gradually ramped below 80%.

At 0312 the 1B Steam Generator Preheater Bypass Isolation Valve (1FW039B) failed closed. This caused a reduction in total feed flow from 616 gpm to 103 gpm. Steam generator levels had dropped from 66% narrow range to 39% narrow range when the plant was manually tripped at 0316. The levels in the S/G dropped to a low of 31% narrow range, two minutes after the trip. Auxiliary Feedwater automatically actuated as expected.

The valve closed due to a broken fitting that supplied air to the actuator. Preliminary analysis of the fitting indicates there was inadequate insertion of the nipple into the elbow when the joint was soldered. This fitting was installed during initial construction. The actuator is a piston-type air operated valve that fails closed on loss of air. Air to the valve is controlled by three solenoids. The solenoids are in series and each solenoid ports air to the actuator when energized, which opens the valve, or vents the actuator to atmosphere which closes the valve.

The section of piping was replaced and the original was sent to System Materials Analysis Department (SMAD) for analysis.

This event is reportable pursuant to 10CFR50.73(a)(i) and 50.73(a)(2)(iv).

C. CAUSE OF EVENT:

The reactor trip was caused by planned manual operator action as a result of decreasing steam generator level due to the failed closed feedwater valve. The feedwater valve failed closed when the instrument air supply line to the valve broke. This failed tubing joint was removed and sent to System Materials Analysis Department (SMAD) to determine the specific failure mode. Investigation of the failed air line revealed the copper tube had not been inserted far enough. The actual insertion was measured at 0.22". The typical insertion is 0.5" with a minimum recommended insertion of 0.25". This joint was made during construction. The elbow that was used for this joint was dented and may have contributed to the inadequate insertion. SMAD also indicated the joint had good fusion for the amount that was inserted.

D. SAFETY ANALYSIS:

The operators responded properly to the decreasing steam generator level by manually tripping the reactor. All systems responded as expected on the reactor trip and Lo-2 Steam Generator Level. Had the valve failed closed at a higher power level, (greater than 20% with the Main Feedwater Isolation valves open) the reactor trip would not have been required. Had the valve failed closed at a lower power level (10% Reactor Power or less) sufficient feedwater would have been available via the tempering line.

E. CORRECTIVE ACTIONS:

Immediate corrective actions were taken to repair the broken air line to the feedwater valve. The failed tubing was replaced under a work request (WR #960063506). The section of tubing that had failed was sent to SMAD for determination of the specific failure mode. The remainder of the valve population was inspected for similar defects and none were found.

F. RECURRING EVENTS SEARCH AND ANALYSIS:

No search performed due to the component that failed.

G. COMPONENT FAILURE DATA:

The failed component was a one-half inch copper tubing and elbow.