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

June 10, 1991

Ltr: BYRON 91-0420

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 in accordance with the requirements of 10CFR50.73(a)(2)(iv).

This report is number 90-010; Docket No. 50-455.

Sincerely,

R. Pleniewicz
Station Manager
Byron Nuclear Power Station

RP/DK/mw

Enclosure: Licensee Event Report No. 90-010

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

(0767R/0088R)

IE22
1/1

SUPPLEMENT TO DVR

DVR NO.

D- 06	- 02	- 90	- 057
STA	UNIT	YEAR	NO.

PART 1 TITLE OF EVENT

Manual Reactor Trip and Main Steam Isolation due to Sample Probe Failure.

OCCURRED

12/20/90 — DATE

0408 — TIME

REASON FOR SUPPLEMENTAL REPORT

Correct Error in Power Level.

PART 2

ACCEPTANCE BY STATION REVIEW: Paul O'Neill, J W Sch...

DATE: 6-12-91, 6-13-91

SUPPLEMENTAL REPORT APPROVED AND AUTHORIZED FOR DISTRIBUTION: G K Schwartz

STATION MANAGER

DATE: 6/17/91

(0767R/0088R-1)

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)

Manual Reactor Trip and Main Steam Isolation due to Sample Probe Weld Failure

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)
									Byron Unit 1	0| 5| 0| 0| 0| 4| 5| 4
1| 2	2| 0	9| 0	9| 0	— 0 | 1 | 0	— 0 | 1	0 | 6	1| 7	9| 1		0| 5| 0| 0| 0| |

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)			
POWER LEVEL (10)	7 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	
W. Scheffler, Technical Staff, ext. 2378 W. Kouba, Unit 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 (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
X	S | B			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 December 20, 1990 at 0400, a severe steam leak was reported in the Unit 2 Main Steam Tunnel. After verifying the size of the leak, the Reactor was manually tripped. By eliminating Steam Generator Blowdown and Feedwater as causes, it was determined that the leak was on the main steam side. The Main Steam Isolation Valves were then closed which isolated the leak. The Main Steam Dumps were opened to depressurize the Main Steam Header.

Upon entry into the Main Steam tunnel, the 2C Main Steam Sample probe was found lying on the floor. The weld for the probe had been improperly repaired during the previous refueling outage causing the probe and its isolation valve to be ejected leaving a one inch hole in the Main Steam line. Since this probe was needed only for initial start-up testing, the nozzle was capped.

This event is reportable pursuant to 10CFR50.73(a)(2)(iv) any event that results in a manual or automatic actuation of the Engineered Safety Features including Reactor Protection System.

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 12/20/90 / 0408

Unit 2 MODE 1 - Power Operation Rx Power 72% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

On December 20, 1990 at 0400 a severe steam leak was reported in the Unit Two Main Steam (MS) [SB] tunnel. At 0402, after shift confirmation of the leak and its magnitude, the Unit 2 NSO (RO, licensed) commenced a Unit 2 shutdown at 2 MW/minute. At 0403, in an attempt to determine the source of the leak, Steam Generator Blowdown (SD) [WI] was isolated; however, blowdown isolation had no affect on the leak. Security was notified to verify that no one was in the area of the steam leak. At 0406, the steam leak was determined severe enough to warrant a Reactor Trip. A shutdown electrical lineup was established and at 0408 the Unit 2 Reactor was manually tripped. Byron Emergency Procedure BEP-0 was entered at this time. A Feedwater (FW) [SJ] isolation occurred as a normal result of the trip. Feedwater system parameters were monitored to see if Feedwater was the source of the leak. At 0409, the Auxiliary Feedwater (AF) [BA] pumps were manually started to maintain S/G levels following the trip. At 0418, after determination that the leak was on the steam side, the Main Steam Isolation Valves were closed and at 0421 the Main Steam dumps were opened to de-pressurize the Main Steam header. The steam leak was fully isolated by this action. The plant was stabilized with temperature control provided by the Steam Generator Atmospheric Relief valves. The Startup Feedwater pump was started and normal feed was established. Entrance was made into the Main Steam tunnel and the 2C Main Steam sample probe was found lying on the floor. No systems were inoperable or were declared inoperable before or after the trip which contributed to the event. Operator action to mitigate the steam/water from the break was to manually trip the reactor and close the Main Steam isolation valves, which isolated the break in the Main Steam tunnel.

C. CAUSE OF EVENT:

The root cause for this event was improper installation of the sample probe during initial construction. Inadequate clearance was left for the stainless steel probe to expand in the thirty-two and three quarters inch diameter Main Steam carbon steel pipe. This caused the probe to expand far enough to bend and crack the exterior weld which held both the probe and its isolation valve (2MS032C). Thermal expansion was verified by observation (through the penetration in the steam line after failure) of a shiny wear mark on the opposite end of the line where the probe had been seated. The presence of the the tough undiluted original stainless steel filler metal prevented a brittle fracture in the original leak. This crack was blowing steam and was Furmanited prior to the second refueling outage (B2R02). A weld repair (carbon to carbon) was performed during B2R02, which consisted of removing the original stainless fillet weld by grinding and welding a new fillet using a carbon steel filler rod. The repaired fillet weld subsequently failed in service. Visual examination of the failed fillet weld revealed it had fractured through the throat of the weld. No documentation existed for the original non-safety related weld, which prevented verification of the material used originally. All the stainless steel filler from the original weld was not removed (as verified by laboratory analysis) prior to the reweld which contaminated the carbon steel weld causing embrittlement. This weld then completely failed, causing the probe and its isolation valve to be ejected from the Main Steam line, leaving a one-inch hole in the line.

The only function of the probe was to provide a sample of Main Steam from the 2C Steam Generator for moisture carryover during initial startup testing.

D. SAFETY ANALYSIS:

Safety of the plant and public was not affected by this event. The reactor was tripped because of Operating Department concern over the magnitude of the steam leak. No abnormal system parameters were indicated as a result of the steam leak because of its relative size. Under a worst case scenerio i.e. a large steam leak, the Main Steam isolation valves would automatically close at a Main Steam pressure of 640 psig.

E. CORRECTIVE ACTIONS:

Upon notification of steam and water in the Main Steam tunnel, with no way to identify the source of the leak, it was decided to perform a manual reactor trip. After Main Steam isolation, the sample probe was found laying in the Main Steam tunnel. Since the probe was not required for Chemistry concerns, the decision was made to cap the one-inch nozzle which remained on the outside of the 2C Main Steam line. After receiving engineering concurrence, the defective end of the nozzle was cut back 1/2" from the original defective weld to ensure complete removal of weld material. The nozzle weld to the MS pipe was ground to 50% throat depth in three locations and verified to have no underlaying defects by MT examination. The nozzle was capped and welded in place by Temporary Alteration 90-2-66. BOP MS-M2 was also changed to reflect that 2MS032C was removed. The other three sample probes were inspected at their external welds and no defects were found. In addition, the welds were verified to have a stainless steel filler. Unit One probes were visually inspected to verify no leakage was present. Since the other probes' welds have never exhibited any signs of leakage, nor required any weld repair, the probability of another catastrophic weld failure is highly unlikely. In addition based on experience with the 2C sample probe initial leak, leakage before failure is the likely scenerio for similar failure of any probe with an unrepaired weld. This inappropriate weld repair was an isolated event since any weld that would have been repaired in this way would quickly fail.

Unit 2 was brought back on-line, and inspection of the remaining probes will occur during B2R03 and B1R04 for structural integrity or removal. Action Item Records 454-225-91-0020 and 454-225-91-0030 will track the inspections and/or removal of the remaining seven probes for Unit One and Unit Two respectively. Braidwood Station was notified of the event and evaluated their probe installations. Zion Station was verified not to have any probes. The event was reviewed with the Station Welding Supervisor and welders.

F. PREVIOUS OCCURRENCES

None

G. COMPONENT FAILURE DATA:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER

No component failed, it was improper installa[illegible]of the sample probe that caused the initial weld stresses and subsequent improper repair that allowed the catastrophic failure.