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

March 20, 1992

Ltr: BYRON 92-0193

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 92-001; Docket No. 50-455.

Sincerely,

R. Pleniewicz
Station Manager
Byron Nuclear Power Station

RP/CW/mw

Enclosure: Licensee Event Report No. 92-001

cc: A. Bert Davis, NRC Region III Administrator
W. Kropp, NRC Senior Resident Inspector
INPO Record Center
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IE22

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)

Unit 2 Feedwater Isolation on P-14 in 2D Steam Generator

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 2	2 0	9 2	9 2	— 0 0 1	— 0 0	0 3	2 0	9 2		0 5 0 0 0

OPERATING MODE (9)	2	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)			
POWER LEVEL (10)	0 0 0	___ 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)	___ 50.3[illegible](2)	___ 50.73(a)(2)(vii)	___ Other (Specify in Abstract below and in Text)
		___ 20.405(a)(1)(iii)	___ [illegible])(i)	___ 50.73(a)(2)(viii)(A)	
		___ 20.405(a)(1)(iv)	___ 5[illegible])(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 Engineer	Ext. 2378		
W. Koube, U2 Operating Engineer	Ext. 2219	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
				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)

While performing a planned reactor shutdown, the 2D steam generator level increased to greater than the high-high level Feedwater Isolation setpoint of 78.1%. This resulted in a P-14 signal and feedwater system isolation. The turbine had been tripped previously. The feedwater isolation signal was reset and normal feedwater alignment was re-established. All equipment responded normally.

A modification (M6-2-89-033) is currently in progress to allow the D-5 Steam Generators to respond more like the D-4 Steam Generators. The modification will move the level instrument taps to reduce the effects of level instability.

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

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 02-28-92 / 0301

Unit 2 MODE 2 - Startup Rx Power 0% RCS [AB] Temperature/Pressure Normal Operating

B. DESCRIPTION OF EVENT:

At 0252, on 2-28-92, during a planned reactor shutdown for refueling, the Unit Two main turbine was manually tripped. This increased the main steam pressure, which caused collapse of the steam voids in the four steam generators (S/G), and a subsequent drop in S/G level. From 0252 to 0258, feedwater tempering line flow was increased to compensate for the decrease in S/G levels (level had decreased from 50% nominal to approximately 40%).

At 0301, the 2D Steam Generator (S/G) level increased to greater than the hi-2 setpoint of 78.1%, which initiated a P-14 signal and subsequent feedwater system isolation. At 0303, the feedwater isolation signal was manually reset and normal feedwater alignment was re-established.

The hi-2 level in the 2D S/G was the result of difficulty in controlling the Westinghouse model D-5 S/G levels at low power. The level instrument taps in the S/G are being moved during the present refueling outage to reduce this level instability. Modification testing will be performed during Startup to verify increased stability of the D-5 S/G level control system at Startup and low power operations.

C. CAUSE OF EVENT:

The root cause of the hi-2 level on the 2D steam generator was the inherent difficulty in controlling level in the Unit 2 (Westinghouse model D-5) steam generators at low power levels. The difficulty in controlling level is caused by the small span of the narrow range of the steam generators. This narrow range on Unit 2 is approximately 60% of the more stable Unit 1 (D-4) steam generator range. Level deviations in Unit 2 seem amplified as compared to Unit 1 S/G's.

The D-5 level tap locations are located above the transition cone in order to eliminate the potential for a 10% level indication error found by Westinghouse during its design testing. In the model D-4 the narrow range lower taps are located below the transition cone. In the model D-4 the velocity head (from the downflow of the recirculation water) below the transition cone is far less than the D-5. These different tap locations on the two model steam generators, have made the model D-5 steam generators more difficult to control than the D-4.

The D-5 design increases the effect of the shrink/swell phenomenon during low power operation. This creates an initial delay or even a response opposite to the anticipated long term effect. Thus, if an operator or FW control system senses the water level decreasing and reacts by increasing feedwater flow, the immediate effect will tend to decrease the indicated water level. In other words, the wide range level indication shows how actual level is changing, while the narrow range indication is temporarily affected by transitory errors due to changes in feedwater and steam flows.

D. SAFETY ANALYSIS:

There were no adverse safety consequences, all safeguard equipment functioned as designed. The 2D S/G hi-2 Level caused a Feedwater Isolation, as designed, to prevent overfill of the S/Gs. The safety significance would be the same if the same events occurred under any different initial conditions.

E. CORRECTIVE ACTIONS:

The Unit 2 (D-5) steam generator low power level instability will be corrected during the present refueling outage (B2R03) by relocating narrow range level lower taps to a position inside the transition cone. Modification M6-2-89-033 will allow the model D-5 steam generators to respond more like the model D-4 steam generators, which will decrease the possibility of future Feedwater Isolations due to level instabilities. The 10% error potential of the initial design concern, has been incorporated into the calculations for the new setpoints. The modification will lower the lower range sensing lines and result in the following changes:

1) Lower tap elevations changes from 438'2.375" to 429'5.25" (lowered by approximately 9 feet)

2) The current span will increase from 128 inches to 233 inches (Unit 1 span is 233 inches)

3) The steam generator level setpoints will be as follows:

	Unit 1(%/in)	Current Unit Two (%/in)	Proposed Unit Two (%/in)
High-High	81.4/523	78.1/538	80.8/522
Nominal	63.0/480	50.0/502	63.7/482
Low-Low	40.8/428	17.0/460	36.3/418

F. PREVIOUS OCCURENCES:

LER 87-002 Reactor Trips and Feedwater Isolations Due to Operator Difficulty in Controlling Steam Generator Level Transients at Low Power

LER 90-003 P-14 Feedwater Isolation Due to Inability to Control Level in D-5 Steam Generators at Low Power Levels

LER 91-005 Reactor Trip on Low-2 Steam Generator Level during Startup due to D-5 Steam Generator Level Control

G. COMPONENT FAILURE DATA.

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER

Not Applicable