

Commonwealth Edison Company
Byron Generating Station
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December 9, 1999

LTR: BYRON 99-0163
File: 3.03.0800

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Byron Station, Unit 2
Facility Operating License No. NPF-66
NRC Docket No. STN 50-455

Subject: Licensee Event Report (LER) 1999-002-00

Enclosed is an LER involving an inadvertent reactor trip and engineered safety feature system actuation signal generated during a refueling outage evolution of stroking open the Feedwater (FW) isolation valves.

The following corrective actions will be taken:

- The Operating Department will prohibit the stroking of the Feedwater isolation valves during refuel outages without the FW system properly filled until the detailed procedure is developed to properly control the evolution. This item will be tracked as Action Tracking #00019098-20.
- The Training Department will incorporate lessons learned from this event into the FW lesson plans for Initial and Continuing Operator License Training. This item will be tracked as Action Tracking #00019098-18.
- To prevent similar issues with other plant evolutions, Senior Operations Management will establish clear guidance and expectations as to when it is acceptable to perform actions without established procedural controls. This item will be tracked as Action Tracking #00019098-24 through 30.

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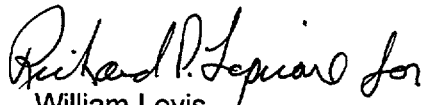
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- In addition, Operations Management will work with Outage Planners to develop a policy to require an independent team review of pending outage schedules. This item will be tracked as Action Tracking # 00019098-17.

If you need any additional information concerning this report, please contact Mr. Karl Moser, Acting Regulatory Assurance Manager, at (815) 234-5441, extension 2159.

Sincerely,


William Levis
Site Vice President
Byron Station

WL/JL/dpk

Enclosure: LER 1999-002-00

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Byron Station

NRC FORM 366 (4-95)				U.S. NUCLEAR REGULATORY COMMISSION				APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98				
LICENSEE EVENT REPORT (LER)								ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS 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 (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT				
FACILITY NAME (1) Byron, Unit 2						DOCKET NUMBER (2) STN 05000455		PAGE (3) 1 of 4				
TITLE (4) Inadvertent Reactor Protection and Engineered Safety Feature Systems Actuations in Mode 5 due to Unexpected Steam Generator Level Response When Stroking a Feedwater 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		
11	10	1999	1999	002	00	12	09	1999	FACILITY NAME	DOCKET NUMBER		
OPERATING MODE (9)		5		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)								
POWER LEVEL (10)		000										
		<input type="checkbox"/> 20.2201(b)		<input type="checkbox"/> 20.2203(a)(3)(i)		<input type="checkbox"/> 50.73(a)(2)(iii)		<input type="checkbox"/> 73.71(b)				
		<input type="checkbox"/> 20.2203(a)(1)		<input type="checkbox"/> 20.2203(a)(3)(ii)		<input checked="" type="checkbox"/> 50.73(a)(2)(iv)		<input type="checkbox"/> 73.71(c)				
		<input type="checkbox"/> 20.2203(a)(2)(i)		<input type="checkbox"/> 20.2203(a)(4)		<input type="checkbox"/> 50.73(a)(2)(v)		<input type="checkbox"/> OTHER				
		<input type="checkbox"/> 20.2203(a)(2)(ii)		<input type="checkbox"/> 50.36(c)(1)		<input type="checkbox"/> 50.73(a)(2)(vii)		(Specify in Abstract below and in Text, NRC Form 366A)				
		<input type="checkbox"/> 20.2203(a)(2)(iii)		<input type="checkbox"/> 50.36(c)(2)		<input type="checkbox"/> 50.73(a)(2)(viii)(A)						
		<input type="checkbox"/> 20.2203(a)(2)(iv)		<input type="checkbox"/> 50.73(a)(2)(i)		<input type="checkbox"/> 50.73(a)(2)(viii)(B)						
		<input type="checkbox"/> 20.2203(a)(2)(v)		<input type="checkbox"/> 50.73(a)(2)(ii)		<input type="checkbox"/> 50.73(a)(2)(x)						
LICENSEE CONTACT FOR THIS LER (12)												
NAME Karl Moser, Acting Regulatory Assurance Manager						TELEPHONE NUMBER (Include Area Code) (815) 234-5441 X2159						
COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)												
CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		
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)

Byron Station Unit 2 was in Mode 5 and making preparations for restart after successfully refueling the reactor. At this point in the refuel outage, activities are in progress to begin filling the secondary side and to perform necessary surveillances required per Technical Specifications. At 2220 hours on November 10, 1999, the priming of the hydraulic pump for the Main Feedwater Isolation valve (2FW009B) was in progress. The priming activity caused, as expected, the 2FW009B valve to open. As the valve opened water back flowed from the B Steam Generator into the feedwater lines. The B Steam Generator level dropped more than expected, approximately 15%, to below the Reactor Trip System (RTS) and Engineered Safety Feature (ESF) setpoints. The RTS and ESF signals were generated as designed and all plant systems and components responded normally for Mode 5 plant conditions. The extent of the possible consequences of allowing the 2FW009B valve to be stroked with the feedwater lines not filled was not fully understood by Licensed Operators. This Outage evolution will not be allowed in the future until a specific procedure is developed to control the evolution. There were no safety consequences from this event. The steam generator protection features were not required to be operable in this plant condition. This event is reportable in accordance with 10 CFR 50.73 (a) (2) (iv).

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS 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

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
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Byron, Unit 2	STN 05000455	1999 - 002 - 00			2 of 4

(If more space is required, use additional copies of NRC Form 366A)(17)

A. Plant Conditions Prior to Event:

Event Date/Time: November 10, 1999/2220 hours

Unit 2 - Mode 5 ❄ Cold Shutdown, Reactor Power - 0%

Reactor Coolant System [AB] Temperature/Pressure: 180°F/40 psig

No structures, systems, or components were inoperable at the start of the event that contributed to the event.

B. Description of Event:

Byron Station Unit 2 was in Mode 5 and making preparations for restart after successfully refueling the reactor. At this point in the refuel outage, activities are in progress to begin filling the secondary side and to perform necessary surveillances required by Technical Specifications. These surveillances include Solid State Protection System [JG] tests, which require the Reactor Trip Breakers to be closed.

The four Steam Generators [SB] had been filled to approximately 50% level and were isolated from the Main Feedwater [SJ] (FW) by the closed hydraulically operated steam generator Feedwater Isolation Valves (i.e., 2FW009A, B, C, and D.) The FW lines had not yet been completely filled up to the 2FW009 valves. A normal preventive maintenance activity performed during each refuel outage is to change the hydraulic fluid in the actuator's pump. The pump then must be primed after refill.

Each steam generator has four level channels, which provide input into the Reactor Trip System (RTS) and Engineered Safety Features [JB] (ESF) System. When two out of four channels on any Steam Generator indicate low level of less than 36.3%, then an automatic RTS signal is generated to trip the reactor and a ESF signal is generated to start the Auxiliary Feedwater [BA] (AF) Pumps.

On November 10, 1999, at 2140 hours, the maintenance activity of priming of the hydraulic pump for the 2FW009A began. This caused, as expected, the valve to stroke open. The Licensed Operators in the Main Control Room expected, from performing 2FW009 valve strokes in previous refuel outages, a small steam generator level drop knowing the feedwater lines on the other side of the 2FW009s were not completely filled. The level dropped in the A steam generator approximately seven percent to 43%, which was as expected and was above the RTS and ESF actuation setpoint.

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B. Description of Event, cont.:

At 2220 hours on the same day, the priming of the hydraulic pump for the 2FW009B was in progress. Again, this priming activity caused, as expected, the 2FW009B valve to open. As the valve opened water back flowed from the B Steam Generator into the feedwater lines. However, the B Steam Generator level dropped more than expected, approximately 15%, to 35% which is below the RTS and ESF setpoint.

RTS and ESF signals were generated as designed and all plant systems and components responded normally for Mode 5 plant conditions. The reactor trip breakers opened, however the control rod drive system was out of service due to plant conditions and not capable of control rod withdrawal. The Main AF pumps were properly defeated from automatically starting due to plant conditions, however the auxiliary lube oil pumps for the main AF pumps did automatically start.

This event is reportable as an Emergency Notification in accordance with 10 CFR 50.72 (b) (2) (ii) and as a Licensee Event Report (LER) in accordance with 10 CFR 50.73 (a) (2) (iv), ☒ any event or condition that results in a manual or automatic actuation of any Engineered Safety Feature, including the Reactor Protection System☒.

C. Cause of Event:

The extent of the possible consequences of allowing the 2FW009B valve to be stroked with the feedwater lines not completely filled was not fully understood by Licensed Operators. They knew a steam generator level drop would ensue but based on prior experience they believed the low level RTS and ESF would not be challenged. Initial system variables, such as steam generator levels and the amount of water existing in the FW lines play a significant role in the resulting consequences of stroking the FW009 valves.

D. Safety Analysis:

There were no adverse safety consequences from this event. The steam generator low water level protection features ensures that protection is provided against a loss of reactor heat sink by shutting down the reactor and starting the Auxiliary Feedwater system to refill the steam generators. This protection is only applicable when the steam generators are used as the heat sink in Modes 1 through 3. In Mode 5 the reactor is already shutdown and the Residual Heat Removal [BP] system is used as the heat sink and steam generator protective features are not necessary.

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E. Corrective Actions:

The Operating Department will prohibit the stroking of the FW009 valves during refuel outages without the FW system properly filled or until a detailed procedure is developed to properly control the evolution.

The Training Department will incorporate lessons learned from this event into the FW lesson plans for Initial and Continuing Operator License Training.

To prevent similar issues with other plant evolutions, Senior Management will establish clear guidance and expectations as to when it is acceptable to perform actions without established procedural controls.

In addition, Operations Management will work with Outage Planners to develop a policy to require an independent team review of the pending outage schedules. This team will encompass a broad base of operating positions to maximize perspectives on the schedule's impact on the plant.

F. Previous Occurrences:

None

G. Component Failure Data:

None