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

October 5, 1990

Ltr: BYRON 90-0982

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

Dear Sir:

The enclosed Licensee Event Report from Byron Gener--ing Station is being transmitted to you in accordance with the requirements of 10CFR50.73(a)(2)(v).

This report is number 90-012; Docket No. 50-454.

Sincerely,

R. Pleniewicz
Station Manager
Byron Nuclear Power Station

RP/mlm

Enclosure: Licensee Event Report No. 90-012

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

(0651R/0077R)

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LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)	Docket Number (2)	Page (3)
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Title (4)

Auxiliary Feedwater Discharge Isolation Valves Design Inadequacy

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 2	0 5 0 0 0 4 5 5
0 9	0 6	9 0	9 0	0 1 2	0 0	1 0	0 5	9 0		0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	X 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	
P. J. O'Neill, Assistant Technical Staff Supervisor Ext. 2244	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

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 9-6-90, with Unit 1 in Mode 1 at 100% reactor power and Unit 2 in Mode 5, it was determined by the Nuclear Engineering Department that the Auxiliary Feedwater (AF) [BA] Discharge Header Isolation valves 1/2 AF013 could not be relied upon to fully close during a Main Steam Line Break Inside Containment accident. Based on the results of calculations that had been performed in response to NRC Generic Letter 89-10, it had been concluded that the valve operators potentially could not develop adequate thrust. This could potentially affect the assumptions of the Updated Final Safety Analysis Report (UFSAR) for a Main Steamline Break Inside Containment. The UFSAR assumed that the AF flow to a faulted Steam Generator (SG) would be isolated within ten minutes. The methodology specified in the emegency procedures for this function depended exclusively on the use of the associated AF013 valves for the affected SG. The emergency procedures for Unit 1 were immediately temporarily revised to provide alternative measures to isolate AF flow if the AF013 valves failed to fully close. An On-Site Review (90-221) was completed on 9-7-90 which concluded that the AF system continued to be operable and that applicable Technical Specifications and Updated Final Safety Analysis Report (UFSAR) bases continue to be satisfied.

Recently, motor operated valve performance models have been improved to consider additional factors which could degrade valve performance. The motor operators on the AF013 valves were originally designed to close at a smaller differential pressure than is required. Permanent procedure revisions are in progress for Unit 1 to provide additional actions to take if an AF013 valve fails to close. Minor Change M6-2-90-698 is in progress during the current refuel outage to increase the overall gear ratio in order to provide additional thrust capability of the Unit 2 AF013 valves.

This event is being reported pursuant to 10CFR50.73(a)(2)(v)(D)as a result of a condition that alone could have prevented fulfillment of the safety function.

(0651R/0077R-1)

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)					Page (3)
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TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [XX]

A. PLANT CONDITIONS PRIOR TO EVENT:

Event Date/Time 9-6-90 / 1510

Unit 1 MODE 1 - Power Operation Rx Power 100% RCS [AB] Temperature/Pressure Normal Operating

Unit 2 MODE 5 - Cold Shutdown Rx Power 0% RCS [AB] Temperature/Pressure 100°F/Atmospheric

B. DESCRIPTION OF EVENT:

On 9-6-90 at 1510, the Nuclear Engineering Department (NED) notified the station that the Unit 1 and Unit 2 Auxiliary Feedwater (AF) [BA] Discharge Header Isolation motor operated valves (1/2 AF013 A through H) could not be relied upon to fully close to isolate Auxiliary Feedwater to a faulted Steam Generator (MS) [SB] during a main steam line break inside containment.

Prior to this in response to Generic letter 89-10, Bechtel Corporation performed an analysis on the AF013 valves to determine the required differential pressure that the valves must be capable of closing against during a postulated accident. NED reviewed the Bechtel analysis and previous testing data on the valves. Based on the NED review, it was determined that the AF013 valves could not be relied upon to fully close under certain accident conditions.

AF flow is required to be isolated to the faulted steam generator within 10 minutes following a main steam line break inside containment, as discussed in the Byron Updated Final Safety Analysis Report (UFSAR), Section 6.2.1.4.1.g. The existing Emergency Procedures (BEP 2, "Faulted Steam Generator Isolation") rely on the AF013 valves alone to isolate AF flow to a faulted steam generator. Additionally, the response not obtained (RNO) column to address the failure of one of these valves to close directed the operator to locally close the affected valve. Due to the location and accessibility of the valves, it would take more than 10 minutes to close. Nuclear Station Operators (NSO, Reactor Operators) are trained to take alternative actions to isolate AF flow if the AF013 valves do not fully isolate flow. However, the procedure did not address these alternatives. Since the Emergency Procedures relied on the AF013 valves alone to isolate AF flow to a faulted steam generator, the procedures were considered inadequate. The appropriate procedures were immediately temporarily revised to include the specific actions to take if an AF013 valve failed to fully close. On-Site Review 90-221 was completed on 9-7-90 to document that the AF system is operable and all Technical Specifications and USFAR bases continue to be satisfied.

On 9-6-90 at 1729, the Nuclear Regulatory Commission was notified of the AF013 design discrepancy pursuant to 10CFR50.72(b)(2)(iii)(D). This report is being submitted pursuant to 10CFR50.73(a)(2)(v) as a result of a condition that alone could have prevented fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident.

C. CAUSE OF EVENT:

Nuclear Engineering Department analysis of the Byron Station valves in accordance with Generic Letter 89-10 included new design considerations. For example, the effects of degraded voltage and valve mispositioning. NED's review indicated that the AF013 valves may not close under postulated worst case circumstances.

C. CAUSE OF EVENT: (Cont)

The original differential pressure specified for the valves was 1450 psid. As a result of the Generic Letter 89-10 analysis, the valves are required to close at a significantly higher differential pressure. The valves cannot be relied upon to fully close at the revised differential pressure of 2048 psid which assumes the AF005 valves are left in their normally throttled position.

The review of the station specific actions to implement the generic requirements of the Westinghouse Owners Group Emergency Procedures, failed to identify that local operation of the AF013 valves could not be performed within the ten minute time interval assumed in the UFSAR because of the location and accessibility of the valves. Had an AF013 valve not fully closed, the procedures directed the operators to locally close the valve. The emergency procedures gave no further direction on how to isolate AF flow other than using the AF013 valves. Therefore, the emergency procedures were considered inadequate.

D. SAFETY ANALYSIS:

There were no adverse safety consequences. The AF013 valves are normally full open. All critical AF system valves are verified to be in their correct position monthly per Byron Operating Surveillances 1/2BOS 7.1.2.1-1/2, "Train A/B Auxiliary Feedwater Monthly Surveillance". The AF013 valves do not receive any Engineered Safety Features (ESF) signals to open or close. Operator action is taken if a position change is required. The valves are capable of opening and closing under static, no flow conditions. The valves are also capable of opening under all conditions, had a valve been mispositioned closed. Only valve closure against full flow differential pressure to a faulted Steam Generator is of concern.

At no time during either unit's operation were the AF013 valves required to be closed during an accident. Had an accident occurred which required AF flow to be isolated from a faulted Steam Generator other alternatives were available for isolation other than the AF013 valves. The AF005 upstream flow control valves can be throttled from the main control board. Additionally, the B train AF005 valves have handwheels and can be locally closed. If both AF pumps are running, one pump can be shutdown at a time to relieve the differential pressure, and the AF013 valve could then be closed. The control of the AF005 valves can also be transferred to the Remote Shutdown Panel and then closed which again would allow the associated AF013 valve to be closed. The AF013 valves could also be closed locally. However, a bolted flood seal would first have to be removed to gain access to the tunnel where the AF013 valves are located. As a result, it cannot be assumed the valves can be closed locally in 10 minutes. An initial review of the basis for the 10 minute time period was completed by Westinghouse Electrical Corporation. The review indicated that the existing steamline break mass and energy releases and containment integrity analyses are valid with AF Flow continuing for 30 minutes. This additional time provided an opportunity for the operator to evaluate the situation and take effective action not specified in the emergency procedures ensuring AF flow isolation.

E. CORRECTIVE ACTIONS:

The Station's preference is to provide AF isolation capability using the AF013 valves. Minor Change M6-2-90-698 was initiated to replace various components in the Unit 2 AF013 valves to improve the capability of the valves to close against the required differential pressure. The Minor Change will be completed during the current refuel outage on Unit 2. As part of the Minor Change testing, a differential pressure test (SPP 90-26) will be performed at the required differential pressure. Based on pressure test data, further corrective actions (i.e. procedure revisions) will be evaluated. Action Item Record (AIR) 90-231 will track Unit 2 corrective actions.

As interim corrective action for Unit 1, emergency procedures 1BFR-S.1, "Response to Nuclear Power Generation/ATWS", 1BFR-Z.1, "Response to High Containment Pressure", and 1BEP-2 were temporarily revised to specifically include the alternative steps to take if the AF013 valves fail to close. Permanent procedure changes are in progress for Unit 1 and Unit 2 to incorporate the added steps to take to isolate AF flow. A review of the Unit 2 Corrective Actions will be performed to determine additional actions for Unit 1. Air 90-230 will track the corrective actions for Unit 1.

F. RECURRING EVENTS SEARCH AND ANALYSIS:

Zion Station LER 89-025 (Docket 50-295), "Auxiliary Feedwater Discharge Motor Operated Valve Failure", documents a similar event. The engineering evaluation of Byron's valves was in progress at the time, and dissimilarities between the Byron and Zion design and operation did not allow for drawing direct comparisons.

This event also applies to Braidwood Station and is documented in LER 90-18 (Docket 50-456).

G. COMPONENT FAILURE DATA:

Although no components failed, the components involved are:

MANUFACTURER	NOMENCLATURE	MODEL NUMBER	MFG PART NUMBER
Limitorque	Actuator	SMB-00	-----
Velan	Valve	4" globe	-----