



Commonwealth Edison
Braidwood Nuclear Power Station
Route #1, Box 84
Braceville, Illinois 60407
Telephone 815/458-2801

July 6, 1990
BW/90-0694

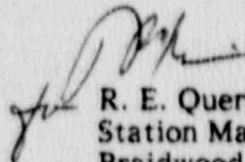
U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Dear Sir:

The enclosed Licensee Event Report from Braidwood Generating Station is being transmitted to you in accordance with the requirements of 10CFR50.73(a)(2)(iv) which requires a 30-day written report.

This report is number 90-010-00; Docket No. 50-457.

Very truly yours,



R. E. Querio
Station Manager
Braidwood Nuclear Station

REQ/JDW/sjs
(7126z)

Enclosure: Licensee Event Report No. 90-010-00

cc: NRC Region III Administrator
NRC Resident Inspector
INPO Record Center
CECo Distribution List

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LICENSEE EVENT REPORT (LER)

Form Rev 2.0

Facility Name (1)

Docket Number (2)

Page (3)

Braidwood 2

0150004571 of 03

Title (4) Reactor Trip During Plant Start Up From Low Steam Generator Water Level due to a Malfunctioning Bypass Feedwater Regulating 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 Names	Docket Number(s)
01	06	019	910	0110	010	01	07	013	910	0150004571
OPERATING MODE (9)			THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR (Check one or more of the following) (11)							
POWER LEVEL (10)			20.402(b)		20.405(c)		X		50.73(a)(2)(iv)	
			20.405(a)(1)(i)		50.36(c)(1)				50.73(a)(2)(v)	
			20.405(a)(1)(ii)		50.36(c)(2)				50.73(a)(2)(vii)	
			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)(ix)	
									73.71(b)	
									73.71(c)	
									Other (Specify in Abstract below and in Text)	

LICENSEE CONTACT FOR THIS LER (12)

Name	TELEPHONE NUMBER
Dave Ibrahim Tech Staff Eng.	AREA CODE 8115458-2801
Ext. 2402	

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPDOS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPDOS
X	2	B	FICV	F1130	Yes				

SUPPLEMENTAL REPORT EXPECTED (14)

Expected Submission Date (15)	Expected Month	Expected Day	Expected Year
Yes (If yes, complete EXPECTED SUBMISSION DATE)	X	NO	

ABSTRACT (Limit to 1400 spaces, i.e. approximately fifteen single-space typewritten lines) (16)

A unit start up was in progress. Feedwater (FW) flow was being controlled in automatic by the Bypass FW Regulating valves (BFRV). Steamline Header Pressure was being controlled in automatic by the Steam Dumps. At 0030 on June 9, 1990 a Reactor Operator (RO), who was monitoring the FW panel, observed that indicated level on the 2B Steam Generator (SG) had decreased to 35%. This was below the set point of 50%. The RO placed the controller in manual and increased the output to raise SG level. 2B SG level continued to decrease from the 'shrink' effect of the cold FW. The Supervisor (SRO) directed the RO who was monitoring Reactor Control Panel, to withdraw control rods to increase temperature and "Swell" the level. SG level increased from an initial value of 20% to 24%. At 0039 the effects of the increased heat input caused steamline pressure to increase which caused the Steam Dump Valves to cycle. This created a level perturbation which caused the level in the 2B SG to decrease below the Reactor Trip Set point of 17% and a Reactor Trip occurred. The cause of the event was a malfunctioning BFRV which would stick during operation in the lower third of valve travel. The valve packing was loosened and valve travel was smooth and acceptable. Operator training will be provided. Previous corrective actions are not applicable.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			Page (3)		
		Year	Sequential Number	Revision Number			
Braidwood 2	01510101014517	910	-	0110	-	010	
TEXT Energy Industry Identification System (EIIIS) codes are identified in the text as [XX]							

A. Plant Conditions Prior to Event:

Unit: Braidwood 2; Event Date: June 9, 1990; Event Time: 0039;
 Mode: 1 - Power Operation; Rx Power: 15%;
 RCS [ABe] Temperature / Pressure: NOT/NOP;

B. Description of Event:

A unit start up was in progress in accordance with approved procedures. Reactor power was at 5% and being increased to approximately 15% in preparations for Main Turbine start up. Feedwater (FW) [SJ] flow was being controlled in automatic by the Bypass FW Regulating Valves (BFRV). Steamline Header Pressure [SB] was being controlled in automatic at 1092 psig by the Steam Dumps.

At 0030 on June 9, 1990 the Nuclear Station Operator (NSO) (Licensed Reactor Operator) who was monitoring the FW panel, observed that indicated level on the 2B Steam Generator (SG) [JB] had decreased to 35% narrow range scale. This was significantly below the normal control set point of 50%. The NSO observed that the output from the controller for the 2B BFRV, indicated 0. The NSO placed the controller in manual and increased the controller output in an attempt to raise SG level. 2B SG level indicated 32% at this time and continued to decrease from the "shrink" effect of the cold FW addition.

The Shift Engineer (SE) (Licensed Senior Reactor Operator) directed the NSO who was monitoring Reactor Control Panel, to withdraw control rods in an effort to increase temperature and "Swell" the level in the SGs. The NSO withdrew the control rods and observed the resultant increases in RCS temperature and SG level. SG level increased from an initial value of 20% to 24% and reactor power was indicating 11%.

At 0039 the effects of the increased heat input to the SGs caused steamline pressure to increase. This caused the Steam Dump valves, which were controlling in auto, to cycle. This created a level perturbation on the SG level control system which caused the level in the 2B SG to decrease below the Reactor Trip Set point of 17% and a Reactor Trip occurred. The NSOs verified all automatic actions. All systems functioned as designed. Stable conditions were immediately established.

The appropriate NRC notification via the ENS phone system was made at 0108 pursuant to 10CFR50.72(b)(2)(ii).

This event is being reported pursuant to 10CFR50.73(a)(2)(iv) - any event or condition that resulted in manual or automatic actuation of any Engineered Safety feature, including the Reactor Protection System.

C. Cause of Event:

The root cause of this event was a malfunctioning valve. The malfunction of the 2B BFRV created a low level condition in the 2B SG. During the action to recover level, the addition of the cold FW combined with the cycling of the Steam Dumps created a condition that caused the indicated level in the 2B SG to decrease below the Reactor trip set point. The sequence of events initiated by the malfunction of the 2B BFRV caused the event.

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D. Safety Analysis:

This event had no effect on the safety of the plant or the public. All systems operated as designed. Auxiliary Feedwater actuated and restored SG Levels as designed.

Under the worst case condition of a low SG level occurring with the Unit operating at 100% power there would still be no effect. The Engineered Safety Feature Actuation System (ESF) would initiate a Reactor Trip, Turbine Trip and Auxiliary Feedwater Actuation from either of the two redundant Trains of ESF, both of which were operable and available for this event.

E. Corrective Actions:

Trouble shooting was performed on both the instrument control loop for the 2B BFRV and the valve itself, 2FW0520A. All portions of the control loop tested out satisfactorily. The valve experienced sticking and jerky motion when operated in the lower third of valve travel. The valve packing was loosened and valve travel was smooth and acceptable during additional testing. The valve has performed acceptably at maintaining SG level in automatic since making the packing adjustment.

This event will be included in the Licensed Operator Regualification Training Program as part of the continuing effort to share experience on level control of the Model D-5 SGs during low power operation. This action will be tracked to completion by action item 457-200-90-02801.

F. Previous Occurrences:

There have been previous occurrences of ESF actuations as a result of level perturbations occurring on the Unit 2 Model D-5 Steam Generators during low power operations. The corrective actions were implemented addressing both root and contributing causes. Previous corrective actions are not applicable to this event.

G. Component Failure Data:

<u>Manufacturer</u>	<u>Nomenclature</u>	<u>Mfg. Part No.</u>
Fisher Controls Co.	Level Control Valve	667-55-79