LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
Braidwood, Unit 2	0 5 0 0 0 4 5 7	1 of 0 3

Title (4) Feedwater Isolation Due to Hi-2 Steam Generator Level Caused by Steam Generator Sensitivity

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 6	2 6	8 8	[illegible] 8	0 1 5	0 0	0 7	1 9	8 8		0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 0 8

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

___	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
Daniel Stroh, Technical Staff Engineer Extension 2477	AREA CODE [illegible] 4 5 8 - 2 8 0 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)

At 0406 on June 26, 1988, a Feedwater (FW) Isolation and a Turbine Driven Feedwater Pump trip occurred. This was the result of the sensitivity of the Model D-5 Westinghouse Steam Generator. Immediate corrective action included resetting the FW Isolation and restarting the FW Pump. This event will be included in Operator required reading to increase Operator awareness of the differences between the Model D-4 and D-5 Westinghouse Steam Generators.

There have been no previous occurrences of a Feedwater Isolation and Turbine Driven Feedwater Pump due to sensitivity of the Model D-5 Westinghouse Steam Generator.

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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TEXT Energy Industry Identification System (EIIS) codes are identified in the text as [xx]

A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood Unit 2 ; Event Date: June 26, 1988 ; Event Time: 0625
MODE: 1 - Power Operation ; Rx Power: 8% ; RCS [AB] Temperature/Pressure: 559 Degrees F/2275 PSIG

B. DESCRIPTION OF EVENT:

There were no systems or components inoperable at the beginning of the event that contributed to the severity of the event.

At 0406 on June 26, 1988, a Unit Startup was under way. The Startup (S/U FWP) Feedwater Pump (FW) [SJ] was supplying water to the four Steam Generators (S/G). At 0608, the 2C Turbine Driven Feedwater Pump (TWFWP) was brought on line, in manual control, and the S/U FWP was secured. The TDFWP was carrying a large recirculation flow for this power level, and resulted in a decrease in FW discharge pressure, an increase in Condensate System (CD) [SD] Flow, and a high differential pressure on the Condensate/Condensate Booster (CD/CB) [SD] Pump Suction Strainer. At 0613 the 2C CD/CB Pump was shutdown for suction strainer cleaning, leaving the 2A and 2B CD/CB Pumps running to supply the TDFWP. During the realignment of the Feedwater System, FW discharge pressure had decreased between 120 and 130 PSIG. Concurrently, CD/CB Pump manipulations caused a S/G water level decrease of approximately 5%. Level was being maintained by directing FW flow through the FW regulating bypass valves which were in automatic control.

At 0620, reactor power had reached 8%, the Main Turbine had been placed on turning gear, and the steam dump valves were being used to maintain a constant steam line pressure. At 0625, a sudden increase in S/G water level occurred in all four S/G's. All FW regulating bypass valves were in automatic control at this time. The Nuclear Station Operator (NSO) throttled closed all FW valves in auto. However, the 2A S/G reached the HI-HI level of 78.1% and caused a FW Isolation trip of the 2C TDFWP. The NSO responded by resetting the FW Isolation and restarting the 2C TDFWP. All systems functioned as designed and at 0640, the plant was stable at 11.5% reactor power with S/G levels within the normal operating band.

Operator actions prevented a Reactor Trip by resetting the FW Isolation, and restarting the 2C TDFWP.

The appropriate NRC notification via the ENS Phone System was made at 0716 on June 26, 1988, 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 is extreme sensitivity of the Model D-5 Steam Generator Level Control System at low power. Computer point history was used to trace secondary side parameters before and during this event. The following facts were revealed:

1) Flow through the CD/CB pumps increased when the TDFWP was started and FW pump discharge pressure dropped between 120 and 130 PSIG;

2) The drop in FW pressure caused a rapid decrease in FW flow to the S/G's and was most pronounced in the 2A S/G.

C. CAUSE OF EVENT:

3) S/G levels came down slowly, approximately 1% per minute for about 5 minutes. This was determined to be caused by the Model D-5 S/G narrow range level control characteristics, and because the main turbine was not using steam.

4) During this time, the proportional plus integral plus derivative level control loop was beginning to drive the FW regulating bypass valves open.

5) FW temperature at the time of this event was only 112 degrees F, and within 2 minutes, the levels began to increase rapidly due to the increase in FW flow and a large swell of the cold feedwater.

D. SAFETY ANALYSIS:

There were no safety consequences as a result of this event. All equipment operated as designed. Under worst case conditions, operating at 100% power, FW flow would have been directed through the Main FW Regulating Valves and control of S/G level would have been much more stable. If the plant had been on the FW regulating bypass valves at the maximum power obtainable, 25% power, and the same event occurred, the same automatic actions would have occurred. Additionally, a reactor trip would possibly occur, with the possible initiation of an auxiliary feedwater pump on a LO-LO S/G Level.

E. CORRECTIVE ACTIONS:

Immediate corrective action was a resetting of the FW isolation circuitry and a restart of the 2C TDFWP.

Action to prevent recurrence will be to include this event in Operator required reading. This will be done to increase Operator awareness of the differences between the Model D-4 and D-5 Westinghouse Steam Geneators.

F. PREVIOUS OCCURRENCES:

There have been no previous occurrences of a feedwater isolation and trip of a turbine driven feedwater pump due to sensitivity of the Model D-5 Westinghouse Steam Generator.

G. COMPONENT FAILURE DATA:

This event was not caused by component failure, nor did any components fail as a result of this event.

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

BW/88-812

July 21, 1988

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 88-015-00; Docket No. 50-457.

Very truly yours,

for R. E. Querio
Station Manager
Braidwood Nuclear Station

REQ/PMB/jab
(7126z)

Enclosure: Licensee Event Report No. 88-015-00

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

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