LICENSEE EVENT REPORT (LER)

Facility Name (1)	Docket Number (2)	Page (3)
Braidwood, Unit 1	0 5 0 0 0 4 5 6	1 of 0 3

Title (4) Reactor Trip and Safety Injection Due to Cognitive Personnel Error

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 1	2 5	8 8	8 8	--- 0 0 2	--- 0 0	0 2	1 7	8 8		0 5 0 0 0

OPERATING MODE (9): 5

POWER LEVEL (10): 0 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)	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	
	AREA CODE	
Fred Ramos, Technical Staff Engineer Ext. 2487	8 1 5	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
X Yes (If yes, complete EXPECTED SUBMISSION DATE)	NO		0 4	0 1	8 8

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

At 1449 on January 25, 1988, a reactor trip signal and a safety injection were generated as a result of a cognitive personnel error by an Instrument Mechanic during the performance of Instrument Surveillance Analog Operational Test and Channel Verification/Calibration for Loop P-0935, Containment Pressure III. Instead of making a connection to Channel 935, he made the connection to Channel 936, which was an active engineered safeguard feature channel. This resulted in a reactor trip signal and a safety injection, which injected approximately 1250 gallons of water into the reactor coolant system. All systems performed as designed.

Immediate corrective actions included terminating the safety injection, securing the running equipment, and restoring Loop 936 to its original configuration.

Long term corrective action is currently being finalized. A supplemental report will be issued.

There have been no previous occurrences.

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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 1 ; Event Date: January 25, 1988 ; Event Time: 1449
MODE: 5 - Cold Shutdown ; Rx Power: 0% ; RCS [AB] Temperature/Pressure: 1[illegible]°F/0 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. The 1A Centrifugal Charging Pump (CV) [CB] was running at the time of the event.

At 0730 on January 25, 1988, a team of four Instrument Mechanics (IM's) were performing BwIS 3.2.1-01[illegible], Analog Operational Test and Channel Verification/Calibration for Loop P-0935, Containment Pressure III. The loop was placed in a tripped condition and the solid state portion of the surveillance was completed. The IM's proceeded to Containment to perform the sensor calibration portion. Two IM's entered Containment to apply a test pressure to the loop under test and the remaining IM's were to monitor the transmitter signal outside the Containment. A 0 psig signal was input to the bellows of the transmitter and a corresponding 4 milliamp signal was observed at the transmitter. At 1449, a 16 psi pressure was input to the bellows of the transmitter, but the IM's outside Containment observed no change in the transmitter output. At this time, the IM in Containment realized that he had applied the test pressure to the wrong loop, 936, which was an active Engineered Safeguards Feature (ESF) (JE) channel. As a result of the application of the 16 psi test pressure to the wrong loop, 936, the following occurred as designed. A Reactor Trip signal was generated. A Safety Injection signal was generated which resulted in an auto-start signal of the 1A Centrifugal Charging pump and alignment to the Refueling Water Storage Tank. Approximately 1250 gallons of water was injected into the Reactor Coolant System (RCS) (AB) causing an approximate 6 psi pressure increase and a 10% [illegible]izer level increase. Also occurring was a Containment Ventilation Isolation Signal [VA], a Containm[illegible] [illegible]e A Isolation, Auto-Start of the 1B Essential Service Water Pump [BI], an auto-start of the Make-up Air [illegible]ter Unit Fan in the Control Room Ventilation System [VI], an auto-start of the A Charcoal Booster Fan in the Control Room Ventilation System (VC) (VI), auto-start of the [illegible] Charcoal Booster Fan in the Fuel Handling Building Plenum (VG), auto-start of the B and D Charcoal Booster Fans in the non-accessible area Exhaust Filter Plenums OA and OB (VL), respectively, a Main Steam Isolation Signal (SB), and auto-start of the 1A Diesel Generator (EK). Major equipment out of service at the time of the event included: the Train-B Emergency Core Cooling System, the 1B Diesel Generator and both of the Safety Injection Pumps (BQ).

Immediate corrective actions were to terminate the safety injection operation, secure the running equipment, and restore Loop 936 to its original configuration. No other operations were required.

Operators responded appropriately to the event by immediately securing the safety injection once it was determined that the signal was not valid, thereby mitigating the pressure transient on the Reactor Coolant System.

The appropriate NRC notification via the ENS Phone System was made at 1607 on January 25, 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 was a cognitive personnel failure on the part of the IM inside the Containment. When he checked the identification of the bellows associated with the 936 Loop, he read the tag correctly and thought he was at the proper instrument. Loop 935 was the loop under calibration.

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D. SAFETY ANALYSIS:

There was no effect on plant or public safety from this event as it occurred, as the plant was in Cold Shutdown at the time of the event. All plant equipment operated as designed given the plant configuration at the time of the event. Had this event occurred under worst case conditions, in Mode 5 with the RCS in a water solid condition, there would still be no adverse effect on the plant or public. Cold overpressurization of the RCS is an analyzed transient and is mitigated by the use of the Power Operated Relief Valves armed in their cold-overpressure protection mode and/or Residual Heat Removal suction relief valves. The operability of these features is governed by Technical Specifications.

E. CORRECTIVE ACTIONS:

Immediate corrective actions were to terminate the safety injection, secure the running equipment, and restore Loop 936 to its original configuration.

This event has been reviewed with the individuals involved.

Details of corrective actions designed to preclude a recurrence of this event are being finalized. This action will be reported in a supplement to this report.

F. PREVIOUS OCCURRENCES:

There have been no previous occurrences of a safety injection and reactor trip due to performance of a surveillance on the wrong channel.

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

EEF/88-316

February 23, 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 88-002-00; Docket No. 50-456.

Very truly yours,

E E Fitzpatrick 2/24/88

E. E. Fitzpatrick
Station Manager
Braidwood Nuclear Station

EEF/PMB/jab
(6669z)

Enclosure: Licensee Event Report No. 88-002-00

cc: NRC Region III Administrator
T. Tongue, NRC Resident Inspector
INPO Record Center
CECo Distribution List

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