LICENSEE EVENT REPORT (LER)

Form Rev 2.0

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

Title (4) Reactor Trip, Safety Injection and Main Steamline Isolation During Plant Heatup due to Management Deficiency

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 4	1 6	8 9	8 9	0 0 2	[illegible]	0 5	1 0	8 9		0 5 0 0 0

OPERATING MODE (9): 3

POWER LEVEL (10): 0 0 0

THIS REPORT IS SUBM[illegible] 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)	X	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	
Jerald Wagner, Regulatory Assurance	Ext. 2497	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	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	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 1450 on April 16, 1989 a plant heatup was being monitored using a graph display on a Control Room monitor (CRT). At 1601 the Nuclear Station Operator (NSO) attempted to repair a failed recorder. At 1640 a Main Steamline (MS) Low Pressure Reactor Trip, Safety Injection (SI), and MS Isolation occurred due to RCS pressure being above 1930 (P-11) psig and MS pressure less than 640 psig. At 1646 the SI signal was reset. At 1648 SI flow was terminated. Cause of this event was failure to assign adequate personnel to monitor the plant heatup. A formal policy on the use of the extra NSO during startup and heatup operations will be developed. The Plant Heatup procedure will be revised to add a hold point to verify that all Steam Generator pressures are greater than 640 psig before RCS pressure exceeds 1930 psig. This event will be included in Reactivity Management training sessions. The CRT graph display will be modified to include an alarm for P-11. There has been a previous occurrence of inadvertent safety injection due to testing the wrong channel during the performance of a surveillance. The corrective actions addressed both root and contributing causes for this event and previous corrective actions are not applicable.

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A. PLANT CONDITIONS PRIOR TO EVENT:

Unit: Braidwood 1; Event Date: April 16, 1989; Event Time: 1640;

Mode: 3 - Hot Standby; Rx Power: 0%;

RCS [AB] Temperature/Pressure: 500 degrees F/1935 psig

B. DESCRIPTION OF EVENT:

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

At 1450 on April 16, 1989 the afternoon shift Nuclear Station Operator (NSO) (Licensed Reactor Operator) relieved the day shift NSO on Unit 1. A plant heatup and pressurization were in progress in accordance with 1BwGP 100-1, Plant Heatup. Two pressure loops associated with 1A and 1B steam generators (SG) [AB] were simultaneously in test to facilitate Instrument Maintenance Department (IMD) calibrations. Plant conditions at this time were:

RCS Pressure:	1340 psig
RCS Temp:	460 degrees F
SG Pressure:	431 psig
Pressurizer Pressure Control:	Manual
Pressurizer Level Control:	Manual
SG Level Control:	Manual

The heatup and pressurization were being monitored using a computer graph displayed on a Control Room monitor (CRT). This graph displays the actual RCS Pressure and RCS Temperature over a green Target Value Line.

Surveillances 1BwOS 4.9.2-1, Pressurizer Temperature Limit Surveillance, and 1BwOS 4.9.1.1-1, RCS Pressure/Temperature Limit Surveillance were in progress in accordance with 1BwGP 100-1.

At 1530 the Unit 1 NSO observed that the 1B RCS Cold Leg RTD was providing erratic indication. He notified the Station Control Room Engineer (SCRE) and Technical Specification 3.3.3.5 Action Statement was entered.

From 1601 to 1639: The Unit 1 NSO observed that the chart recorder pen for the failed RTD was not inking. He attempted to repair the non-inking pen. During repair attempts he spilled ink on the chart, his hands, and the Main Control Board. During the process of cleaning up the spilled ink, the Unit 1 NSO periodically monitored the heatup and pressurization on the CRT. He was also periodically monitoring upper nozzle temperature on another CRT, making the required adjustments to the 1CV121, Pressurizer Level Control Valve, (CV) [CB] and the 1FW034A, B, C, and D, SG Level Control Valves, (FW) [SJ] for each SG.

At 1639 the Unit 1 NSO observed that the actual Pressure versus Temperature was deviating from target value on the heatup and pressurization display on the CRT. After noting that pressure was higher than desired for the temperature, he went to his desk to refer to procedure 1BwGP 100-1.

B. DESCRIPTION OF EVENT: (Cont'd)

At 1640 a Main Steamline Low Pressure Reactor Trip, Safety Injection (SI) [BQ], and Main Steamline Isolation occurred. This was due to RCS pressure being above 1930 psig, the P-11 setpoint, in conjunction with Main Steamline [SB] pressure being less than 640 psig. Braidwood Emergency Procedure, 1BwEP-0, Reactor Trip or Safety Injection Unit 1, was entered. Injection of the cool Refueling Water Storage Tank (RWST) water resulted in an increase in RCS pressure, a decrease in RCS temperature from 500 degrees F at the start of the event, and a decrease in the Main Steamline pressure.

At 1644 the resultant insurge of relatively cooler RCS water into the Pressurizer caused the Pressurizer Liquid Space water temperature to decrease from its initial value of 625 degrees F at the start of the event as expected.

At 1646 Braidwood Emergency Procedure, 1BwEP ES 1.1, SI Termination Unit 1 was entered. The SI signal was reset and termination of the SI flow was initiated.

At 1648 the High Head Safety Injection Isolation Valves were closed terminating the safety injection flow to the RCS. The Main Steamline pressure reached a minimum value of 612 psig.

At 1649 RCS pressure from the SI achieved a maximum value of 2242 psig with a Main Steamline pressure of 612 psig. This resulted in the Administrative Limit of 1600 psid to be exceeded by 30 psid.

At 1650 the Pressurizer Liquid Space temperature reached a minimum value of 518 degrees F. This indicated a Pressurizer Liquid space cooldown of 107 degrees F in a 6 minute period which is in excess of the Administrative Limit for cooldown of 100 degrees F in a one hour period. However, it was still well within the 200 degrees F in one hour limit allowed by the Technical Specifications. As a result of the termination of the SI flow, the insurge flow to the pressurizer stopped and as a result the Pressurizer Liquid Space water temperature started to increase.

At 1651 the differential pressure between the RCS and the Main Steamlines decreased below 1600 psid with the RCS at 2155 psig and the Main Steamlines at 617 psig.

At 1659 an Unusual Event was declared and terminated pursuant to the Generating Stations Emergency Plan (GSEP) Emergency Action Level (EAL) 2.g - ECCS initiation signal and resultant injection to the vessel (Not spurious).

At 1704 the Nuclear Accident Reporting System (NARS) notification was made to declare and terminate the Unusual Event.

At 1726 the Pressurizer Liquid Space temperature reached a peak value of 623 degrees F. This indicated a Pressurizer Liquid Space temperature increase of 105 degrees F in 36 minutes which is in excess of the Technical Specification Limiting Condition for Operation (LCO) of 100 degrees F in any one hour period. Stable plant conditions were achieved.

B. DESCRIPTION OF EVENT: (Cont'd)

The appropriate NRC notification via the ENS phone system was made at 1732 pursuant to:

10CFR50.72(b)(2)(iv) - Any event that results or should have resulted in Emergency Core Cooling System discharge into the reactor coolant system as a result of a valid signal.

10CFR50.72(a)(i) - The declaration of any of the Emergency Classes specified in the Licensee's approved Emergency Plan.

10CFR50.72(c)(1)(iii) - A termination of the Emergency Class.

The NRC notification via the ENS phone system was also made incorrectly pursuant to 10CFR50.36(c)(1)(ii)(A) based on a deficient Administrative Procedure for identifying and classifying events. The SI automatically actuated at the correct setpoint. Therefore, this reporting requirement is inappropriate.

At approximately 2200 while reviewing the Pressurizer Temperature Limit Surveillance, it was discovered that the heatup of the Pressurizer Liquid Space was in excess of the Technical Specification Limit of 100 degrees F in a one hour period. The LCO action statement was entered. An engineering evaluation was initiated to determine the effects of the heatup on the Pressurizer in accordance with the Technical Specifications.

At 0556 on April 17, 1989, the Onsite Review of the engineering evaluation was completed. The evaluation concluded that the structural integrity of the Pressurizer was acceptable for continued operation.

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.

10CFR50.73(a)(2)(i) - Any operation or condition prohibited by the plant's Technical Specifications.

Based on the initial information associated with this event a "Braidwood Station Error Evaluation Presentation" was held to review this event with the personnel directly involved and their supervisor. The corrective actions addressing both root and contributing causes are detailed below.

C. CAUSE OF EVENT:

The root cause of this event was a management deficiency. The Unit 1 NSO was allowed to become distracted without consideration by either the NSO or the SCRE to assign additional personnel.

D. SAFETY ANALYSIS:

There was no effect on plant or public safety from this event as it occurred, as the plant was in hot standby and all plant equipment operated as designed.

All engineered safety features and the reactor protection system, including manual reactor trip, were operable to mitigate the consequences of this event.

E. CORRECTIVE ACTIONS:

Immediate corrective actions:

1. The SI signal was reset and stable plant conditions were established.

2. An engineering evaluation of the structural integrity of the Pressurizer was performed for the pressure transient. The evaluation concluded that the structural integrity of the Pressurizer was acceptable for continued operation.

3. Westinghouse has performed an analysis of the effects of this event on the structural integrity of the RCS. The analysis has concluded that the impact of this event on the structural integrity of RCS components is insignificant.

Based on the initial information associated with this event the personnel directly involved with this event participated in a "Braidwood Station Error Evaluation Presentation" to identify the root and contributing causes of this event. Based on the conclusions of this presentation the following corrective actions will be taken:

1. Operating Department will develop and establish a formal policy on the use of the extra NSO during startup and heatup operations. This will be tracked to completion of Action Item 456-200-89-06102.

2. 1(2) BwGP 100-1, Plant Heatup will be revised to establish a hold point to verify that all steam generator pressures are greater than 640 psig before RCS pressure exceeds P-11 setpoint. This will be tracked to completion by Action Item 456-200-89-06103.

3. This event will be reviewed with Operating Department personnel as part of the training associated with Reactivity Management. This will be tracked to completion by Action Item 456-200-89-06104.

4. The heatup and pressurization computer graph display will be modified to include the setpoint for P-11. This will be tracked to completion by Action Item 456-200-89-06105.

F. PREVIOUS OCCURRENCES:

There was a previous occurrence of inadvertent safety injection, DVR 20-1-88-019/LER 50-456-88-002. However, that event was due to an Instrument Mechanic testing the wrong channel during the performance of a surveillance. The corrective actions were implemented addressing both root and contributing causes for this event. Previous corrective actions are not applicable to this event.

G. COMPONENT FAILURE DATA:

This event was not the result of 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

May 15, 1989
BW/89-582

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 require a 30 day written report.

This report is number 89-002-00; Docket No. 50-456.

Very truly yours,

for R. E. Querio
Station Manager
Braidwood Nuclear Station

REQ/AJS/jab
(7126z)

Enclosure: Licensee Event Report No. 89-002-00

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

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