NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1): Braidwood Station Unit 2

DOCKET NUMBER (2): STN 05000457

PAGE (3): 1 of 4

TITLE (4): Engineered Safety Feature Actuation (P-14) due to High Water Level in the 2C Steam Generator

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 NAME	DOCKET NUMBER
04	24	1999	99	002	00	05	21	99	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): 05

POWER LEVEL (10): 000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

[ ] 20.2201(b)	[ ] 20.2203(a)(3)(i)	[ ] 50.73(a)(2)(iii)	[ ] 73.71(b)
[ ] 20.2203(a)(1)	[ ] 20.2203(a)(3)(ii)	[X] 50.73(a)(2)(iv)	[ ] 73.71(c)
[ ] 20.2203(a)(2)(i)	[ ] 20.2203(a)(4)	[ ] 50.73(a)(2)(v)	[ ] OTHER
[ ] 20.2203(a)(2)(ii)	[ ] 50.36(c)(1)	[ ] 50.73(a)(2)(vii)	
[ ] 20.2203(a)(2)(iii)	[ ] 50.36(c)(2)	[ ] 50.73(a)(2)(viii)(A)	(Specify in Abstract
[ ] 20.2203(a)(2)(iv)	[ ] 50.73(a)(2)(i)	[ ] 50.73(a)(2)(viii)(B)	below and in Text,
[ ] 20.2203(a)(2)(v)	[ ] 50.73(a)(2)(ii)	[ ] 50.73(a)(2)(x)	NRC Form 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Robert Wegner, Braidwood Station Operations Manager	(815) 458-2801 Extension 2213

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)

On April 24, 1999, during cooldown of Unit 2 for refueling outage A2R07, an increase in Steam Generator (SG) water levels was recognized by operators. During efforts to identify and stop the source of in-leakage to the steam generators, a P-14 (SG Water Level High-High) Engineered Safety Feature (ESF) actuation occurred at 1936. As a result of the actuation signal, 2FW002A-C (Feedwater Pump Discharge Isolation Valves) closed. No other components changed status because other equipment receiving the actuation signal were in their actuated positions. The increasing steam generator water levels were determined to be caused by gravity feed from Unit 2 Condensate Storage Tank (CST) to the steam generators through the Auxiliary Feedwater (AF) System. The primary cause of this event is attributed to a procedural deficiency which allowed a flowpath to exist from the Condensate Storage Tank to the steam generators via the 2AF013A-H, Steam Generator Isolation Valves.

Following the event, the Steam Generator Blowdown System was placed in operation and the steam generator water level was returned to the normal operating range. In addition, an Emergency Notification System (ENS) notification was made in accordance with 10CFR50.72(b)(2)(ii), "Any event or condition that results in a manual or automatic actuation of any engineered safety feature (ESF), including the reactor protection system (RPS)..." Corrective Actions to be taken include revising the station's Shutdown procedure for both units to address identified deficiencies and discussing the event with operations personnel in Licensed Operator Requalification Training.

This event is being reported in accordance with 10CFR50.73(a)(2)(iv), "Any event or condition that results in a manual or automatic actuation of any Engineered Safety Feature (ESF)..."

A. **PLANT CONDITIONS PRIOR TO EVENT:**

Unit(s): 02 Event Date: 04/24/99 Event Time: 19:36
Reactor Mode(s): 05 Power Level(s): 000 RCS [AB] Temp./Press. 178 degrees/ 354 psig.

B. **DESCRIPTION OF EVENT:**

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

This event is being reported pursuant to 10CFR50.73(a)(2)(iv), "any event or condition that results in a manual or automatic actuation of any Engineered Safety Feature (ESF)..."

On April 24, 1999, during a cooldown of Unit 2 for refueling outage A2R07, an increase in SG water levels was recognized by operators at approximately 1800. The on shift crew suspected that the cause of the increasing levels was valve leakage in the main feedwater system (SJ)(FW) based on past operating experience. As a result, operators were dispatched to close the 2FW033A-D (Feedwater Tempering Line Manual Isolation Valves) and 2FW055A-D (Feedwater Bypass Manual Isolation Valves).

The computer point history recording SG water levels was reviewed for data obtained between 1100 and 2100 on April 24, 1999. The data showed that the water level for the 2C SG remained stable until approximately 1800, which was when Operations personnel recognized the level increase. At that time, the point history showed the SG water level to be approximately 60% narrow range.

At 1804, the Main Steam Isolation Valves (MSIVs) were closed with the steam header depressurized. At 1813, the main condenser vacuum was broken. At 1825, the operators previously dispatched to close the 2FW033A-D and 2FW055A-D valves reported to the control room that the valves were closed. At that time, the unit was also transitioned from Mode 4 to Mode 5. Operating crew turnover from the dayshift to nightshift was completed at approximately 1900. At that time, water levels in the 2C SG were approaching 75%.

Approximately twenty minutes after shift turnover, a Senior Reactor Operator (SRO) noticed that levels in all SGs were continuing to slowly increase with the 2B and 2C SGs approaching the P-14 ESF actuation setpoint. The Unit Supervisor was notified of this condition. Control Room personnel discussed the condition and decided to close Feedwater Shutoff valves 2FW006A-D, pursue further tightening down on the 2FW033A-D and 2FW055A-D valves, restart the Steam Generator Blowdown System (WI)(SD), and determine the feasibility of installing drain hoses on the SGs for manual draining.

At approximately 1930, the Shift Manager conferred with a Senior Reactor Operator in the field about the possibility of securing the remaining Condensate (SD)(CD)/ Condensate Booster (SD)(CB) pump. During this discussion, a P-14 ESF actuation occurred at 1936. As a result of the actuation signal, 2FW002A-C, Feedwater Pump Discharge Isolation Valves, closed. The other functions that actuate following a P-14 ESF actuation signal were already in their actuated positions.

At 1945, the Steam Generator Blowdown System was placed in operation and the SG water level was returned to the normal operating range. At approximately 2000, the 2AF013A-H valves were subsequently closed as specified in Procedure 2BwGP 100-5, thus stopping the inleakage to the SGs.

C. **CAUSE OF EVENT:**

Operations personnel were unable to prevent the 2C SG water level from exceeding the SG Water Level High-High setpoint which resulted in the P-14 ESF actuation. The primary cause of this event is attributed to a procedural deficiency which allowed a flowpath to exist from the Condensate Storage Tank (CST) to the SGs via the 2AF013A-H valves.

The procedural guidance provided to Operations personnel for the unit shutdown evolution, Procedure 2BwGP 100-5, "Plant Shutdown and Cooldown," Revision 15, did not provide sufficient direction to the operators. Although the Auxiliary Feedwater System (BA)(AF) is isolated in the procedure, the 2AF013A-H valves are not closed until after the Auxiliary Feedwater System is isolated, following the transition from Mode 4 to Mode 5. During the time period when operators recognized the urgency in responding to the increasing SG levels, they had not yet reached the step in 2BwGP 100-5 that requires closure of the 2AF013A-H valves. Based on the determination that the water flowpath into the SGs was via the 2AF013A-H valves, the condition that operators were working to avoid could have been mitigated if the 2AF013A-H valves had been closed earlier in the procedure after the Auxiliary Feedwater System was isolated.

Operators failed to consider the flowpath from the Condensate Storage Tank to the SGs through the open 2AF013A-H valves. Operators had the mindset that the leakage was coming from the main feedwater system because leakage problems with main feedwater valves had been encountered in the past.

D. **ASSESSMENT OF SAFETY CONSEQUENCES:**

There were no safety consequences as a result of this event. The SG Water Level High-High P-14 protective interlock is designed to protect the main turbine (TB) from water induction. This feature is designed for equipment protection, and functioned as designed. Under worst case conditions, operating at 100% power, a high-high water level in the SGs would cause an ESF actuation which would result in feedwater isolation, feedwater pump and turbine trips, and ultimately, a reactor trip. In this particular event, most Feedwater Isolation Valves were already in their closed positions with the exception of the Feedwater Pump Discharge Isolation Valves (2FW002A-C) which functioned as designed. Therefore, the health and safety of the public was not adversely affected by this event.

E. **CORRECTIVE ACTIONS:**

Operations will evaluate Procedures 1 / 2 BwGP 100-5 to determine the most appropriate sequence for closing the 1 / 2 AF013A-H valves and revise the procedures accordingly.

This event will be discussed with operations personnel in Licensed Operator Requalification Training.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Braidwood Station Unit 2	STN 05000457	99	002	00	4 of 4

(If more space is required, use additional copies of NRC Form 366A)(17)

F. PREVIOUS OCCURRENCES:

A review of previous reportable events was conducted to determine if any previous events occurred within the past two years that were similar to this event. No events were identified.

G. COMPONENT FAILURE DATA:

Since no component failure occurred, this section is not applicable.