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February 21, 2005
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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Braidwood Station, Unit 2
Facility Operating License No. NPF-77
NRC Docket No. STN 50-457

Subject: Submittal of Licensee Event Report Number 2004-002-00, "Unit 2 Automatic Reactor Trip on 2C Steam Generator Low-Low Level Initiated by a Failure of the Controlling Channel Steam Flow Card"

The enclosed Licensee Event Report (LER) is being submitted in accordance with 10 CFR 50.73, "Licensee event report system", paragraph (a)(2)(iv)(A). 10 CFR 50.73(a) requires an LER to be submitted within 60 days after discovery of the event, therefore, this report is being submitted by February 21, 2005.

Should you have any questions concerning this submittal, please contact Mr. Dale Ambler, Regulatory Assurance Manager, at (815) 417-2800.

Respectfully,

for

Keith J. Polson
Site Vice President
Braidwood Station

Enclosure: LER Number 2004-002-00

cc: Regional Administrator - Region III
NRC Braidwood Senior Resident Inspector

IE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007

Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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4. TITLE
Unit 2 Automatic Reactor Trip on 2C Steam Generator Low-Low Level initiated by a Failure of the Controlling Channel Steam Flow Card

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME N/A	DOCKET NUMBER N/A
12	22	2004	2004	- 002 -	00	02	21	2005	FACILITY NAME N/A	DOCKET NUMBER N/A

9. OPERATING MODE: 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Michael Smith, Engineering Director	(815) 417-3800

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
X	JB	FB	W120	Y	N/A	N/A	N/A	N/A	N/A

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

Braidwood Unit 2 experienced a reactor trip on December 22, 2004 at 1308 due to a low-low level in the 2C Steam Generator (SG). The reactor trip was initiated by the failure of a circuit card in the 2C steam flow instrumentation loop. The steam flow signal from this card feeds into the 2C SG level control loop. The loss of the steam flow signal to the 2C SG level control loop resulted in two automatic actions; closure of the 2C SG feedwater regulating valve in response to a steam flow-feedwater flow mismatch, and a speed reduction of the two turbine-driven feedwater pumps to reduce feedwater header pressure in accordance with a programmed feedwater header pressure versus total steam flow control response. Operator actions were unable to recover SG level prior to the automatic reactor trip.

The root cause was determined to be inadequate understanding and incomplete evaluation of changes made to the Unit 2 feedwater system control settings for the feedwater regulating valves and the feedwater pump speed controller. The corrective actions to prevent recurrence include determining and setting the proper values for the feedwater regulating valve controller settings and main feed pump speed controller response for optimum plant response, and providing licensed operators the response characteristics of both Braidwood Units to feedwater type transients initiated by circuit card failures.

There were no safety consequences impacting plant or public safety as a result of this event. The reactor trip system responded as designed to the low SG level condition and shut down the reactor at the required setpoint for steam generator low-low level.

This event is being reported pursuant to 10 CFR 50.73(a)(2)(iv)(A).

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

A. **Plant Operating Conditions Before The Event:**

Event Date: December 22, 2004 Event Time: 1308

Unit: 2 MODE: 1 Reactor Power: 100 percent

Unit 2 Reactor Coolant System (RCS) [AB] Temperature: 581 degrees F, Pressure: 2233 psig

B. **Description of Event:**

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

On December 22, 2004 at 1306, a failure of a steam flow isolation card in the steam generator water level control system [JB] for the 2C Steam Generator (SG) caused the steam flow loop indication for the 2C SG steam flow indication to fail low. Because of this, the steam flow and feed flow to the 2C SG as sensed by the control circuitry were no longer matched, and resulted in two automatic actions; the control system mismatch resulted in an error demand to close the 2C feedwater [SJ] regulating valve (FRV), and the feedwater pump speed controller (FWPSC) reduced the speed of the feedwater pumps due to the apparent reduction in total steam flow. As a result, the 2C SG level began to decrease.

In response to the plant transient, the Nuclear Station Operator (NSO) took manual control of 2C SG FRV and increased the FRV controller manual-auto station demand signal to the full open position. Additionally, the NSO took manual control of the master FWPSC to begin raising the feedwater header pressure. These were expected operator responses, and in accordance with procedures.

At 1308, the Unit Supervisor ordered a manual reactor trip, based on the recognition that the 2C SG level was approaching the low-low level trip setpoint. This is consistent with licensed operator training and management expectations. Prior to the manual reactor trip, the 2C SG reached the low-low level reactor trip setpoint, and an automatic reactor trip on low-low 2C SG level occurred. This also initiated an Engineered Safety Feature [JE] actuation of the auxiliary feedwater (AF) [BA] system.

C. **Cause of Event**

The root cause was determined to be inadequate understanding and incomplete evaluation of historical changes made to the Unit 2 feedwater system control settings for the feed regulating valves and the feedwater pump speed controller.

The current Unit 2 feedwater control system settings are based on activities that took place in the 1987 to 1993 time period. In 1987, the Braidwood Unit 2 settings were based initially on recommended startup controller settings by Westinghouse, with the understanding that the settings may change as necessary to achieve stable plant operations. The final settings were largely based on the operating experience from Byron Unit 2 startup and the difficulties in controlling the D5 model SGs. The only other change occurred in 1993, when the Unit 2 main feedwater pump power program lag time constant was changed in support of a Braidwood and Byron standardization of the Instrument Test Report Packages.

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

The Unit 2 trip was initiated by the failure of a 2C SG steam flow isolation card. The steam flow signal from this card feeds into the 2C SG level control loop. The failed circuit card was sent to the Exelon Power Labs (East) for failure analysis, which found a failed Op amp on the circuit card.

The loss of the steam flow signal to the 2C SG level control loop resulted in two automatic actions; closure of the 2C SG FRV in response to a steam flow-feedwater flow mismatch, and a speed reduction of the two turbine-driven feedwater pumps to reduce feedwater header pressure in accordance with a programmed feedwater header pressure versus total steam flow control response.

The root cause investigation identified that such automatic actions are manageable on Unit 1, based on simulator performance and previous events, providing control room operators the chance to stabilize SG level and restore the plant to normal operating conditions. The feedwater system controller settings on Unit 2 result in a slower FRV response to re-open on low SG level and a much faster speed reduction of the feedwater pumps, making a similar recovery more difficult.

D. **Safety Consequences:**

There were no safety consequences impacting plant or public safety as a result of this event. The reactor trip system responded as intended to the low SG level condition and shut down the reactor at the required setpoint for steam generator low-low level. All control rods inserted. All feedwater pumps were available at the time of the reactor trip. This includes the two turbine driven feedwater pumps, the motor driven feedwater pump and the startup feedwater pump. In addition, the two AF pumps were automatically started on initiation of the low-low 2C SG level. The plant was maintained in Mode 3 with cooling provided by the condenser steam dumps.

There was no concern for removal of reactor heat based on the series of events and the operator actions following the reactor trip initiation. The heat sink (steam generators) had sufficient water inventory to maintain reactor heat removal capability. The ability to use the reactor coolant feed and bleed method was still available and all equipment was operable.

The fourth quarter 2004 Unit 2 NRC Performance Indicator for Unplanned Scrams per 7,000 Critical Hours is well within the green band (i.e., low safety significance) at a value of 0.8.

This event did not result in a safety system functional failure.

E. **Corrective Actions:**

There are three corrective actions to prevent recurrence:
1. Determine and set the proper values for the FRV controller settings for optimum plant response.
2. Determine and set the proper values for the FWPSC response for optimum plant response.
3. Provide Licensed Operators the response characteristics of both Braidwood units to feedwater type transients initiated by circuit card failures. This will include a discussion of the differences in the Unit 1 and Unit 2 SGs and the importance of operator response timing, and will include a demonstration on the simulator to reinforce the significant difference for the Unit 2 SGs and their characteristics.

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

F. **Previous Occurrences:**

There have been no similar Licensee Event Report events at Braidwood Station in the last three years. This is the first known steam flow circuit card failure on Unit 2.

An event similar to this occurred on Braidwood Unit 1 in 2002, with the exception that Braidwood Unit 1 remained at full power and the SG level was restored.

G. **Component Failure Data:**

Manufacturer	Nomenclature	Model	Mfg. Part Number
Westinghouse	7300-Isolator	NLP G03	2837A12G03