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10 CFR 50.73

March 2, 2011
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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Braidwood Station, Unit 1
Facility Operating License No. NPF-72
NRC Docket No. STN 50-456

Subject: Supplemental Licensee Event Report 2010-001-01 – Unit 1 Reactor Trip Due to Water Intrusion in Breakers Causing Circulating Water Pump Trips and Resulting in Loss of Condenser Vacuum

This is a supplement to Licensee Event Report 2010-001-00 which was submitted on October 15, 2010. This supplement contains updated information regarding the root cause and corrective actions of the issue.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this submittal, please contact Mr. Ronald Gaston, Regulatory Assurance Manager, at (815) 417-2800.

Respectfully,

Daniel J. Enright
Site Vice President
Braidwood Station

Enclosure: LER 2010-001-01
cc: NRR Project Manager – Braidwood Station
Illinois Emergency Management Agency – Division of Nuclear Safety
US NRC Regional Administrator, Region III
US NRC Senior Resident Inspector (Braidwood Station)

NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA/Privacy Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resource@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)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE

Reactor Trip Due to Water Intrusion in Breakers Causing Circulating Water Pump Trips and Resulting in Loss of Condenser Vacuum

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	DOCKET NUMBER
									N/A	N/A
08	16	2010	2010	- 001 -	01	03	02	2011	N/A	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

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
Ronald Gaston, Regulatory Assurance Manager	(815) 417-2800

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
N/A	N/A	N/A	N/A	N/A	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		N/A	N/A	N/A

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

On August 16, 2010, as a result of a Unit 2 reactor trip, condensate water overflowed from the auxiliary feedwater (AF) standpipes onto the turbine deck. The water spread through openings in the floor to the elevation below, and entered a Unit 1 substation cabinet. This water intrusion resulted in the trip of electrical buses that caused the 1A and 1C circulating water (CW) pumps to trip. Additionally, power to the respective CW pump discharge valves was lost. These conditions caused the Unit 1 condenser vacuum to degrade.

At 0219, the Unit 1 main turbine received an automatic trip on low condenser vacuum, resulting in an automatic reactor trip. Following the reactor trip, the auxiliary feedwater pumps auto started on low-low steam generator water levels.

The root causes were determined to be: 1) an inadequate design of the AF standpipes; and 2) a lack of sensitivity evolved that tolerated long-term uncontained water issues with inadequate evaluation for potential impacts.

The corrective action to prevent recurrence is to install a design feature on the AF suction standpipe which prevents water spill events.

There were no actual safety consequences impacting plant or public safety as a result of the event. This event is being reported pursuant to 10 CFR 50.73(a)(2)(iv)(A) due to actuation of the reactor protection system (reactor trip) and the auxiliary feedwater system.

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NARRATIVE

Background:

During a Unit trip from power, the condenser hotwell [SD] level is expected to increase. An increasing hotwell level often results in automatic condenser hotwell rejection flow to the condensate storage tank (CST) [SD] to prevent hotwell overfill.

The auxiliary feedwater system (AF) [BA] takes suction from the condensate storage tank (CST), via the CST header piping. In the late 1970s and early 1980s, during construction and start-up testing, Braidwood had problems with the start-up of the motor driven AF pumps. During start-up of the pumps, due to the motor driven pumps coming to speed quickly and the flow resistance of the suction piping to the CST, a temporary low net positive suction head would often occur, and result in an automatic pump trip. To address the issue, AF vent riser standpipes were added to the CST header piping in 1986, which corrected the condition. The standpipes fill with water to the level of the CST and acts as an accumulator by providing a momentary volume to the pump suction at pump start-up.

The installation of the standpipes improved the low net positive suction head problem with the motor driven AF pumps. However, it resulted in the water level in the standpipe momentarily rising and overflowing, and introduced the potential for water spills from the standpipe during condenser hotwell rejection operations.

A. Plant Operating Conditions Before The Event:

Event Date: August 16, 2010 Event Time: 0219 CDT

Unit: 1 MODE: 1 Reactor Power: 100 percent

Unit 1 Reactor Coolant System (RC) [AB]: Normal operating temperature and pressure

B. Description of Event:

No Unit 1 structures, systems, or components were inoperable at the start of this event that contributed to the event.

On August 16, 2010, at 0206, Unit 2 tripped off line due to a generator lockout. This event is addressed under Unit 2 LER 2010-003-00.

Due to the Unit 2 trip, condensate water was rejected to the CST to prevent hotwell overfill. The influx of condensate filled the AF standpipes and resulted in an overflow of the standpipes onto the turbine deck. The water spread to various gaps and openings in the floor and flowed down to the elevation below. The water entered a Unit 1 4160V/480V substation cabinet, through the cabinet roof panel joints which are not normally sealed.

At 0214, the water in this cabinet caused a ground over-current on breaker 1435VU and caused a trip of buses 133V and 133U. Loss of buses 133V and 133U caused the 1A and 1C circulating water (CW) [KE] pumps to trip due to the loss of excitation voltage. In addition, the power to the respective CW pump discharge valves was lost. As a result, when the CW pumps tripped, their discharge valves did not close automatically as expected on a pump trip. Therefore, flow from the remaining 1B CW pump was left with a full, and low restriction, recirculation path through the open discharge valves and through the idle 1A and 1C pumps. Due to this condition, either no, or very low, circulating water was flowing through the Unit 1 condenser, and resulted in a rapidly degrading condenser vacuum.

At 0219, Unit 1 main turbine received an automatic trip on low condenser vacuum, resulting in an automatic reactor trip.

NARRATIVE

Following the reactor trip, the AF pumps auto started on low-low steam generator [SJ] water levels. Due to the loss of bus 133V, steam dumps [SB] were unavailable and core cooling was maintained from the main steam [SB] power operated relief valves (PORVs).

Operator response to the trip was proper and safety systems and controls performed as expected with the exception of the following:

- The main steam relief valve 1MS016D lifted early due to age related spring relaxation, and did not reseat until main steamline [SB] pressure was reduced to 918 psig. Unit 1 subsequently transitioned to Mode 5 (cold shutdown) and the valve replaced.
- The motor control center 131X1 did not energize, preventing two valves from being energized - the safety injection pumps [BQ] cold leg isolation valve 1SI8835, and the residual heat removal [BP] to cold legs 1A and 1D isolation valve 1SI8809A.
- As a result of CW forebay material stirred up by the recycle flow from the 1B CW pump through the idle 1A and 1C CW pumps, essential service water (SX) [BI] pump discharge pressure was low and the differential pressure across the SX strainer was high. A second SX pump was started to restore SX system pressure.

This event is reportable under 10 CFR 50.73(a)(2)(iv)(A), any event or condition that resulted in manual or automatic actuation of any of the systems listed in 10 CFR 50.73(a)(2)(iv)(B) including any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical, and actuation of the PWR auxiliary feedwater system.

C. Cause of Event

The root causes for this event were determined to be:

1. An inadequate design of the AF standpipes. Braidwood did not implement an effective design configuration which would have prevented the water spills from the AF suction standpipes.
2. A lack of sensitivity evolved that tolerated long-term uncontained water issues with inadequate evaluation for potential impacts. This organizational cause was considered a historical issue rather than a current issue and is being addressed via the corrective action program.

The installation of the standpipes in 1986 improved the low net positive suction head problem with the motor driven AF pumps. However, it resulted in the water level in the standpipe momentarily rising and overflowing, and introduced the potential for water spills from the standpipe during condenser hotwell rejection operations.

The CST header, in addition to containing the suction line to the AF pumps and the standpipe, is also used as the flow path of condenser hotwell reject water from the hotwell to the CST. When the hotwell level becomes high, the excess water is rejected back to the CST through this line. With unit perturbations, ranging from unit trips to condensate and condensate booster pump swap operations, the level in the hotwell often changes. If the level becomes high enough, the controls initiate automatic hotwell rejection flow to the CST. When an automatic hotwell rejection occurs, the level in the standpipe will rise, and if sufficient, would result in an overflow. The overflow was onto the floor of the 451 elevation of the turbine building.

NARRATIVE

D. Safety Consequences:

There were no safety consequences impacting plant or public safety as a result of this event.

For the loss of condenser vacuum, the systems and controls for managing this type of condition worked as expected. The low vacuum trip setpoint removed the main turbine from service as expected. With the trip of the main turbine, an automatic reactor trip also occurred as expected. The main steam relief valve 1MS016D lifted early due to age related spring relaxation, and reseated when main steamline pressure was reduced to 918 psig during Unit 1 cooldown. All other safety systems and controls performed as expected.

The steam released from the opened 1MS016D valve and the PORVs contained tritium. After the 1MS016D valve reseated, steam release continued through the PORVs during the Unit 1 cooldown until shutdown cooling was established from the residual heat removal system [BP]. The offsite dose resulting from this release was not significant since tritium was the only isotope released and the dose impact of tritium is low. Calculated off-site dose was 4.59E-6 millirem.

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

E. Corrective Actions:

The corrective action to prevent recurrence is to install a design feature on the AF suction standpipe which prevents water spill events. The organizational cause was considered a historical issue rather than a current issue and is being addressed via the corrective action program.

Other corrective actions include:

- CST level was limited to prevent another AF standpipe siphon event.
- An operating configuration change was implemented to valves controlling a condenser hotwell rejection, to prevent AF suction standpipe overflows from hotwell rejections.
- Develop a plan to identify long-term water leaks, spills, other uncontained fluid conditions or any degraded or abnormal conditions that are not captured by the station processes.
- Develop a process for Byron and Braidwood to identify differences in their critical systems in design, alignment, operations or maintenance practices, to evaluate any differences for the best practices, define risks, and have actions assigned to gain commonality.

Both the inability to energize 1SI8835 and 1SI8809A valves and the issue with the high differential pressure across Unit 1 SX strainer have been addressed in the corrective action program.

F. Previous Occurrences:

There have been no previous, similar events identified at the Braidwood Station.

G. Component Failure Data:

Manufacturer	Nomenclature	Model	Mfg. Part Number
N/A	N/A	N/A	N/A