Exelon Generation Company, LLC
Braidwood Station
35100 South Route 53, Suite 84
Braceville, IL 60407-9619

www.exeloncorp.com

10 CFR 50.73

August 2, 2018
BW180082

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

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

Subject: Licensee Event Report 2018-006-00 – Manual Unit Trip on Low Steam Generator Level Following Trip of a Turbine Feedwater Pump Due to a Design Issue

The enclosed Licensee Event Report (LER) is being submitted in accordance with 10 CFR 50.73, "Licensee Event Report System."

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

Respectfully,

for Marri Marchionda-Palmer
Site Vice President
Braidwood Station

Enclosure: LER 2018-006-00

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)
Illinois Emergency Management Agency – Braidwood Representative

NRC FORM 366 (04-2018) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 03/31/2020

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 Information Services Branch (T-2 F43), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by 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 Page 2 for required number of digits/characters for each block)

(See NUREG-1022, R.3 for instruction and guidance for completing this form http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)

1. Facility Name	2. Docket Number	3. Page
Braidwood Station, Unit 1	05000456	1 OF 3

4. Title

Manual Unit Trip on Low Steam Generator Level Following Trip of a Turbine Feedwater Pump Due to a Design Issue

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
06	04	2018	2018	- 006 -	00	08	02	2018	Facility Name N/A	Docket Number N/A

9. Operating Mode	11. This Report is Submitted Pursuant to the Requirements of 10 CFR §: *(Check all that apply)*			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
10. Power Level	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
100	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
		☐ 50.73(a)(2)(i)(C)	☐ Other (Specify in Abstract below or in NRC Form 366A)	

12. Licensee Contact for this LER

Licensee Contact	Telephone Number (Include Area Code)
Francis Jordan	815-417-2800

13. Complete One Line for each Component Failure Described in this Report

Cause	System	Component	Manufacturer	Reportable to ICES	Cause	System	Component	Manufacturer	Reportable to ICES
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 14 single-spaced typewritten lines)

On June 4, 2018, at 0920 hours, Braidwood Unit 1 was manually tripped due to lowering steam generator water levels following a trip of the 1C turbine driven feedwater pump. The 1A and 1B auxiliary feedwater (AF) pumps auto started as expected due to low steam generator levels.

The cause of the event was due to the inclusion of a digital input module in the turbine driven feedwater pump control circuitry, which affected only the safeguards testing function of the circuit, and which was not identified as a design input per the design input and configuration change impact screening procedure. Corrective actions planned include implementing the changes to eliminate the design deficiency, and revising the design input and configuration change impact screening procedure.

This event is reportable in accordance with 10 CFR 50.73(a)(2)(iv)(A) for "Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section, ..." Specifically, for 1) 10 CFR 50.73(a)(2)(iv)(B)(1) for the "Reactor protection system (RPS) including: reactor scram or reactor trip," and 2) 10 CFR 50.73(a)(2)(iv)(B)(6) for the "PWR auxiliary or emergency feedwater system."

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET

(See NUREG-1022, R.3 for instruction and guidance for completing this form http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Braidwood	05000456	2018	- 006	- 00

NARRATIVE

A. Plant Operating Conditions Before the Event:

Event Date: June 4, 2018

Unit: 1 Mode: 1 Reactor Power: 100 percent

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

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

B. Description of Event:

On June 4, 2018 at 0917 hours, while performing an Engineered Safety Feature actuation relay surveillance test associated with the main feedwater (FW) [SJ] pumps, the 1C main turbine driven FW pump tripped. Operations initiated a main turbine runback. At 0920 hours, Braidwood Unit 1 was manually tripped before reaching the steam generator low water level trip setpoint. The 1A and 1B auxiliary feedwater (AF) [BA] pumps auto started as expected due to low steam generator levels. At 1044 hours, after the startup feedwater pump was placed in service, the 1A and 1B AF pumps were secured.

Operator response to the trip was proper and all systems and controls performed as expected.

This event is reportable in accordance with 10 CFR 50.73(a)(2)(iv)(A) for "Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section, ..." Specifically, for 1) 10 CFR 50.73(a)(2)(iv)(B)(1) for the "Reactor protection system (RPS) including: reactor scram or reactor trip," and 2) 10 CFR 50.73(a)(2)(iv)(B)(6) for the "PWR auxiliary or emergency feedwater system." This LER is being submitted in follow-up to ENS 53443 made on June 4, 2018.

C. Cause of Event

The cause of the event was, during a modification of the turbine driven FW pump control circuitry, the inclusion of a digital input module affected only the safeguards testing function of the circuit, and this impact was not identified as a design input per the design input and configuration change impact screening procedure. As a result, the impact of the module on the circuit's electrical impedance was not evaluated and the design error was not discovered during the subsequent evaluation and design change testing. The change to the circuit caused an unexpected loss of a turbine driven feedwater pump during surveillance testing that was performed consistent with past performances.

D. Safety Consequences:

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

The operating crew responded appropriately to lowering steam generator levels and manually tripped the reactor prior to automatic actuation. All equipment operated appropriately as designed. There was no loss of any function that would have prevented fulfillment of actions necessary to 1) Shutdown the reactor and maintain it in a safe shutdown condition, 2) Remove residual heat, 3) Control the release of radioactive material, or 4) Mitigate the consequences of an accident.

NARRATIVE

There was no loss of safety function for this event.

E. Corrective Actions:

Planned Corrective Actions:

- Implement the changes to eliminate the design deficiency associated with the testing of the FW pump control circuitry.
- Revise the design input and configuration change impact screening procedure to include consideration of changes to circuits for electrical impacts to the circuit such as impedance.

F. Previous Occurrences:

None

G. Component Failure Data:

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