Exelon.
Nuclear

Clinton Power Station
8401 Power Road
Clinton, IL 61727-9351

U-604048
January 12, 2012

10 CFR 50.73
SRRS 5A.108

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

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Licensee Event Report 2011-004-00

Enclosed is Licensee Event Report (LER) No. 2011-004-00: Automatic Reactor Scram During Removal of Main Generator. This report is being submitted in accordance with the requirements of 10 CFR 50.73.

There are no regulatory commitments contained in this report.

Should you have any questions concerning this report, please contact A. Khanifar, at (217)-937-3800.

Respectfully,

William G. Noll
Site Vice President
Clinton Power Station

RSF/blf

Enclosures: Licensee Event Report 2011-004-00

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Clinton Power Station
Office of Nuclear Facility Safety – IEMA Division of Nuclear Safety

IE22
NRR

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
Automatic Reactor Scram During Removal of Main Generator

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
11	29	2011	2011	- 004 -	00	01	12	2012	FACILITY NAME: N/A	DOCKET NUMBER: N/A

9. OPERATING MODE: 1

10. POWER LEVEL: 016

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)*
A. Khanifar, Site Engineering Director	(217) 937-3800

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
E	JI	CBD	G082	Y					

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)*

On November 29, 2011, with the reactor at 16.1 percent thermal power, Operators were reducing generator load to 50 Megawatts in order to trip the Main Turbine to start refueling outage C1R13. Turbine Bypass Valve (BPV) #1 had begun opening and was approximately 8 percent open when it ramped closed contrary to system demand. A Steam Bypass Control Pressure Regulator error alarm was received in the Main Control Room. Due to the loss of the steam flow path through the bypass valve to the Main Condenser, Reactor Pressure Vessel (RPV) pressure began to increase and subsequently reached the high reactor pressure scram setpoint resulting in an automatic reactor scram. Reactor pressure control was established manually utilizing the Steam Bypass Valve opening jack. Troubleshooting determined that BPV #1 closed due to failure of a Bypass Valve Demand (BVD) card in the Steam Bypass and Pressure Control (SBPC) System causing RPV pressure to increase. The cause of this event is attributed to the failure to replace or refurbish the BVD card prior to it failing. Corrective actions include replacement of both the A and B BVD cards, establishment of periodic preventive maintenance for the cards, and performing an extent of condition review for other cards in the SBPC system.

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NARRATIVE

PLANT AND SYSTEM IDENTIFICATION

General Electric – Boiling Water Reactor, 3473 Megawatts Thermal Rated Core Power
Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

EVENT IDENTIFICATION

Automatic Reactor Scram During Removal of Main Generator

A. Plant Operating Conditions Before the Event

Unit: 1	Event Date: 11/29/11	Event Time: 1728 hours CST
Mode: 1	Mode Name: Power Operation	Reactor Power: 16.1 percent

B. DESCRIPTION OF EVENT

On November 29, 2011, with the reactor in Mode 1 (Power Operation), Operators were in the process of a planned shutdown and were reducing load in preparation for refueling outage C1R13.

At approximately 1728 hours, with the reactor at about 16.1 percent thermal power, Operators were reducing Main Generator [TB] [TG] load to 50 Megawatts in order to trip the Main Turbine [TA][TRB]. At this point in time, Turbine Bypass Valve (BPV) #1 [V] had begun opening as expected, as observed by the Operators and confirmed by a review of plant computer data. BPV #1 was approximately 8 percent open when it ramped closed contrary to system demand. Coincident with the BPV #1 sudden closure, a Steam Bypass Control Pressure Regulator error alarm [ALM] was received in the Main Control Room. Due to the loss of the steam flow path through the bypass valve to the Main Condenser, Reactor Pressure Vessel (RPV) pressure began to increase and subsequently reached the high reactor pressure scram setpoint of 1065 pounds per square inch gage (psig) resulting in an automatic reactor scram before operators were able to take manual action. The highest RPV pressure recorded was 1074 psig. Following the reactor scram RPV water level lowered to the Level 3 trip setpoint of 8.9 inches. The lowest RPV water level recorded was 7.8 inches.

The Operations crew responded in accordance with Emergency Operating Procedure – 1, “Reactor Pressure Vessel Level Control,” and the Scram Off-Normal procedure. Reactor pressure control was established manually utilizing the Steam Bypass Valve opening jack, which was found capable of opening and positioning the turbine bypass valves. At 1830 hours, RPV water level was at 30 inches and RPV pressure was at 775 psig, normal band for the current conditions.

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

Troubleshooting determined that BPV #1 unexpectedly ramped closed as a result of a failure of a Bypass Valve Demand (BVD) [CBD] card in the Steam Bypass and Pressure Control (SBPC) System [JI] (B Channel), causing RPV pressure to increase. The BVD card had a faulty op-amp output; the op-amp was found to be failed low and caused a close demand signal to be applied to the BPV.

This is a 4-hour reportable event under 10 CFR 50.72 (b)(2)(iv)(B) as a valid actuation of the Reactor Protection System [JC] while the reactor was critical (Completed Event Notification 47489).

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NARRATIVE

Issue Report 1295617 was initiated to evaluate this event.

C. CAUSE OF EVENT

The cause of this event has been determined to be the failure to replace or refurbish the BVD card prior to its Mean Time Between Failure (MTBF) lifetime, resulting in the card failure. A MTBF analysis completed following this event concluded that the MTBF for the BVD card is 20 years. The card had been in service for 24 years prior to failure.

The BVD card was not identified as a critical component during the 2001 Plant Material Condition Excellence Initiative (PMCEI) assessment and 2007 Performance Centered Maintenance (PCM) assessment, and therefore, was not reviewed for maintenance strategy.

C. SAFETY CONSEQUENCES

This event is reportable under the provisions of 10 CFR 50.73 (a)(2)(iv)(A) due to a valid actuation of the Reactor Protection System [JC] while the reactor was critical.

The actuation of the Reactor Protection System placed the plant in a safe and stable condition. There were no plant safety limits exceeded, and no other Engineered Safety Feature (ESF) actuations, and risk significance was low. Safety related systems functioned correctly in response to this event with critical plant parameters remaining within the bounds of plant design, Technical Specifications, Updated Safety Analysis Report, Offsite Dose Calculation Manual, and Core Operating Limits Report. The affected system (SBPC System) is non-safety related.

No loss of safety function occurred during this event.

E. CORRECTIVE ACTIONS

The failed BVD card (B channel) and the A channel BVD card were replaced during the refueling outage with new cards that were tested with satisfactory results.

Periodic 16-year replacement refurbishment PMs are being established for the A and B BVD cards in the SBPC system.

The SBPC System is being reviewed to ensure that critical cards have been identified and PMs are established as needed.

F. PREVIOUS OCCURRENCES

A review of reactor scrams over the past 10 years did not identify any events occurring as a result of the same cause.

G. COMPONENT FAILURE DATA

Component Description: Circuit Board Assembly, Bypass Valve Demand

Manufacturer	Nomenclature	Model	Mfg. Part Number
General Electric	N/A	30-992107-002	N/A