Constellation Energy
Nine Mile Point Nuclear Station

P.O. Box 63
Lycoming, New York 13093

October 29, 2004
NMP1L 1878

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Nine Mile Point Unit 1
Docket No. 50-220
Facility Operating License No. DPR-63

Licensee Event Report 04-004, "Manual Reactor Scram Due to Failure of #13 Feedwater Flow Control Valve Positioner"

Gentlemen:

In accordance with 50.73(a)(2)(iv)(A), we are submitting Licensee Event Report 04-004, "Manual Reactor Scram Due to Failure of #13 Feedwater Flow Control Valve Positioner."

Very truly yours,

Tim O'Connor
Plant General Manager

TO/KSE/jm
Attachment

cc: Mr. S. J. Collins, NRC Regional Administrator, Region I
Mr. G. K. Hunegs, NRC Senior Resident Inspector

IE22

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

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

LICENSEE EVENT REPORT (LER)

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

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Nine Mile Point, Unit 1	05000220	1 OF 4

4. TITLE
Manual Reactor Scram Due to Failure of #13 Feedwater Flow Control Valve Positioner

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 05000
08	30	2004	2004	- 004 -	00	10	29	2004	FACILITY NAME	DOCKET NUMBER 05000

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)
M. Steven Leonard	315-349-4039

#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
B	SJ	FCV	Hartman-Braun	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 August 30, 2004, at approximately 0825 hours, with the mode switch in "RUN" and reactor thermal power at approximately 100%, Nine Mile Point Unit 1 experienced feedwater flow oscillations. Efforts to manually stabilize feedwater flow were unsuccessful. At approximately 0833 hours, a manual scram was initiated to shutdown the plant.

The feedwater flow oscillations were caused by degradation of the #13 feedwater flow control valve positioner due to a ruptured diaphragm in the output pilot valve. The root cause was determined to be the original design did not adequately establish a service life that took into account operating the positioner at the maximum recommended air supply pressure, thereby decreasing the service life below the established preventive maintenance frequency.

The positioner, including the failed pneumatic module, for the #13 feedwater flow control valve was replaced. Subsequent testing demonstrated satisfactory flow control capabilities. To prevent recurrence, the positioner will be replaced every two years.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Nine Mile Point, Unit 1	05000220	2004	-- 004 --	00	2 OF 4

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

I. Description of Event

On August 30, 2004, at approximately 0825 hours, Unit 1 experienced feedwater flow oscillations due to a loss of control of the #13 feedwater flow control valve (FCV). Operators transferred feedwater control from automatic to manual in accordance with operating procedures but were not able to obtain stable control of the valve. Operators then inserted a manual reactor scram at approximately 0833 hours and entered Emergency Operating Procedures (EOPs) due to low reactor water level. All control rods fully inserted as designed. High Pressure Coolant Injection (HPCI) initiated as designed due to reactor water low level (53 inches) and restored reactor water level from 22 inches (the lowest level reached during the transient). The reactor water high level signal (95 inches) tripped the feedwater pumps. When the reactor high level trip bypass switches were cycled, the #11 Feedwater Pump restarted as expected. Operators stabilized reactor pressure and level and transitioned out of the EOPs into normal operating procedures. The plant was placed in Cold Shutdown at 2036 hours.

No maintenance, testing, or plant evolutions were in progress related to the #13 feedwater flow control valve, its instrumentation, or its support systems.

II. Cause of Event

The direct cause for the feedwater flow oscillations was the failure of a diaphragm in the pneumatic module of the positioner for the feedwater flow control valve.

The failed valve positioner was removed from the station and sent to Constellation Energy's Materials Laboratory for failure analysis. A vendor representative assisted in the troubleshooting and analysis to determine the direct cause of the failure. The vendor representative determined the failure was most likely caused by a rupture in one or both of the diaphragms in the pilot assembly of the pneumatic module. Construction of the pneumatic module did not allow disassembly to directly inspect the diaphragms. Due to construction of the component, the diaphragms were destroyed during disassembly of the component.

Several other U.S. and Canadian nuclear plants that use a similar positioner were contacted. Based on the analysis of possible causes, the most probable cause for the positioner failure is the duty cycle in combination with operating the positioner at the maximum vendor recommended air supply pressure of 90 psi. This reduced the service life of the pilot valve diaphragms. The service life published by the vendor appears to be non-conservative. This was supported by discussions with personnel from other stations.

The root cause was the original design did not adequately establish a service life taking into account operating at the maximum air supply pressure and duty cycles, thereby decreasing the service life below the established preventive maintenance frequency.

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Nine Mile Point, Unit 1	05000220	2004	-- 004 --	00	3 OF 4

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

III. Analysis of Event

On August 30, 2004, at approximately 0825 hours, a failure of the #13 feedwater flow control valve positioner resulted in feedwater flow oscillations. Efforts to manually stabilize feedwater flow were unsuccessful. At approximately 0833 hours, a manual reactor scram was inserted to shutdown the reactor.

All control rods fully inserted as expected.

As expected, High Pressure Coolant Injection (HPCI) initiated on low Reactor Pressure Vessel (RPV) water level due to the level transient associated with the scram.

The Emergency Operating Procedures (EOPs) were entered for RPV Control on reactor water level < 53 inches, as expected. When conditions stabilized, the EOPs were exited and normal operating procedures were used to continue the plant shutdown.

During recovery, reactor water level lowered as expected following the scram to approximately 22 inches. Before the scram, operators had placed #13 feedwater flow control valve in manual in an attempt to mitigate feedwater flow oscillations. At the time of the scram, #11 Feedwater Pump was operating in manual per plant procedures and the #12 Feedwater Pump was in standby. Upon HPCI initiation, #12 Feedwater Pump started and remained in operation, supplying flow to the reactor until level had recovered. The #11 Feedwater Pump flow, as designed, was held at minimum flow until total feedwater flow lowered to 4.5 Mlbm/hr. At that time, #11 Feedwater Pump began supplying flow to the reactor to assist with reactor water level recovery. The #13 Feedwater Pump was secured and HPCI logic was reset. Reactor water level continued to increase, reaching the High Reactor Level trip (</=95 inches), at which time both #11 and #12 Feedwater Pumps tripped as designed. Maximum reactor water level reached was 96 inches. The high-level pump trips were reset and the #11 Feedwater Pump auto started as expected since reactor water level was below the High Reactor Level trip.

Reactor pressure was 1024 psig before the scram. No appreciable rise in reactor pressure was observed during this event. The turbine bypass valves were used to control reactor pressure during plant cooldown. No abnormalities were encountered in pressure control.

Reactor pressure and cooldown rates were properly controlled with no abnormalities or challenges to the operators experienced. The Electromatic Relief Valves (ERVs) remained closed during the event as expected. The reactor cooldown rate was maintained less than 75 degrees F/hr until the reactor reached Cold Shutdown in accordance with Technical Specifications.

As designed, house electrical loads auto transferred to the reserve supply following the scram. No adverse or unexpected electrical transients occurred. No automatic start signals were initiated or expected for the emergency diesel generators during this event.

No systems or components were inoperable at the start of the event that contributed to the severity of the event.

Operators inserted a manual scram, which was a conservative decision based on the inability to obtain stable reactor water level. The reactor scram did not pose a threat to the health and safety of the public or plant personnel.

An NRC 10 CFR 50.72 report (Event Number 40998) was made on August 30, 2004 at 1142 hours to report the scram and was amended on August 30, 2004 at 2358 hours to include the required report of the HPCI initiation. The Emergency Plan was not activated because no entry conditions were met.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Nine Mile Point, Unit 1	05000220	2004	-- 004 --	00	4 OF 4

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

IV. Analysis of Event (Continued)

This event is reportable in accordance with 10 CFR 50.73(a)(2)(iv)(A) as an event or condition that resulted in manual or automatic actuation of the Reactor Protection System (RPS). The automatic initiation of HPCI is also reportable under 10 CFR 50.73(a)(2)(iv)(A) as an event or condition that resulted in manual or automatic actuation of an Emergency Core Cooling System.

IV. Corrective Actions

The valve positioner, which includes the failed pneumatic module, for the #13 feedwater flow control valve was replaced. Post maintenance testing and subsequent performance were satisfactory.

To prevent recurrence, the valve positioner for the #13 feedwater flow control valve will be replaced approximately every two years. This make and model of positioner is unique to the #13 feedwater flow control valve.

V. Additional Information

A. Failed Components:

- Pneumatic Module of Positioner for Feedwater Flow Control Valve FCV-29-134
  Manufacturer: Hartmann & Braun (Now ABB)
  Model: TZIDA535220

B. Previous similar events:

LER 96-04, "Reactor Scram Caused by Turbine Trip Due to Feedwater Oscillations" is similar in that the same component failed, though for different reasons. In 1996, feedwater flow oscillations resulted in an automatic reactor scram. The flow oscillations were caused by a degraded actuator for the #13 feedwater flow control valve. In this case, the actuator degradation involved pneumatic and mechanical alignment problems. The pneumatic problems are different. Since the 1996 event, the pneumatic controls were replaced with a new design to improve valve performance (change was not a direct result of the 1996 event). Therefore, though the symptoms and consequences are similar, the direct and root causes are different.

C. Identification of systems and components referred to in this Licensee Event Report:

Components	IEEE 805 System ID	IEEE 803A Function
#13 Feedwater Flow Control Valve	SJ	FCV
#11 Feedwater Pump	SJ	P
#12 Feedwater Pump	SJ	P
#13 Feedwater Pump	SJ	P
High Pressure Coolant Injection System	BJ	N/A
Electromatic Relief Valves	SB	PSV
Control Rods	AA	ROD
Reactor Protection System	JC	N/A
Turbine Bypass Valves	SB	PCV
Emergency Diesel Generators	EK	DG