Constellation Energy
Nine Mile Point Nuclear Station

P.O. Box 63
Lycoming, NY 13093

October 12, 2005
NMP1L 1990

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 05-003, "Automatic Reactor Scram during Surveillance Testing due to Inadvertently De-energizing 4160 VAC Power Board 11"

Gentlemen:

In accordance with 10 CFR 50.73(a)(2)(iv)(A), we are submitting Licensee Event Report 05-003, "Automatic Reactor Scram during Surveillance Testing due to Inadvertently De-energizing 4160 VAC Power Board 11."

Very truly yours,

Timothy J. O'Connor
Plant General Manager

TJO/DEV/sac
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)	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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Automatic Reactor Scram during Surveillance Testing due to Inadvertently De-energizing 4160 VAC Power Board 11

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	18	2005	2005	- 003 -	00	10	12	2005	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)
James A Hutton, Director Licensing	(315) 349-1041

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
A	EA	SWGR	GE	Yes					

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 18, 2005, at approximately 0950 hours, with Nine Mile Point Unit 1 operating at approximately 100 percent power, an automatic reactor scram was experienced. The scram occurred when Power Board 11 (PB-11) was de-energized due to inadvertent protective over-current relay actuations. The resulting loss of power to the Channel 11 side of the Reactor Trip Bus (RTB), concurrent with planned in-progress surveillance testing that introduced a half-scram signal on the opposite (Channel 12) side of the RTB, caused a full automatic reactor scram signal to be generated.

PB-11 was de-energized due to inadvertent protective over-current relay actuations caused by mechanical jarring of the relays mounted on the cubicle door. This occurred when a 4.16 kV circuit breaker was being rolled into position within its PB-11 cubicle. The underlying cause of the event is failure to adequately identify, assess, and mitigate potential risks of conducting work. The potential plant impacts that could be caused by inadvertent actuation or damage to surrounding equipment due to planned work were not identified or effectively managed.

Corrective actions have been developed such that, when completed, the work control process will yield consistent and effective risk assessments of work activities that will prevent the same type of inadvertent relay actuation and other similar events (i.e., events caused by failure to identify and effectively manage potential risks associated with surrounding equipment).

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

I. Description of Event

On August 18, 2005, at approximately 0950 hours, with Nine Mile Point Unit 1 (NMP1) operating at approximately 100 percent power, an automatic reactor scram was experienced. The scram occurred when 4160 VAC Power Board 11 (PB-11) was de-energized due to inadvertent protective over-current relay actuations. PB-11 provides power primarily to non-safety related station power production equipment. The loss of PB-11 resulted in loss of power to one of the two sources of power to the Reactor Trip Bus (RTB). Concurrently with the loss of PB-11, planned surveillance testing was in progress that introduced a half-scram signal on the opposite (Channel 12) side of the RTB. With a half-scram signal on the Channel 12 side and loss of power on the Channel 11 side of the RTB, a full automatic reactor scram signal was generated. Had the surveillance testing on the Channel 12 side not been in progress, the loss of PB-11 and the downstream loads would have resulted in a significant power reduction or plant shutdown due to the loss of key components, including two strings of feedwater heating, a condensate pump, two reactor recirculation pumps, a service water pump, three drywell cooling fans, a condenser circulating water pump, and numerous other loads.

Following the automatic scram, all control rods fully inserted and the reactor was shutdown.

Reactor pressure did not rise above its initial value during the event. The turbine bypass valves were used to control reactor pressure during scram recovery and plant cool down. The cooldown rate was maintained less than 75 degrees Fahrenheit per hour.

Reactor water level lowered as expected following the scram to approximately 27 inches. The Low Reactor Level scram/High Pressure Coolant Injection (HPCI) initiation signals (53 inches) were received as expected. At the time of the scram, feedwater pump 11 was operating in manual and feedwater pump 12 was in standby. Upon HPCI initiation on the turbine trip signal, feedwater pump 12 started and injected. Feedwater pump 13 was secured and the HPCI logic was reset per procedure. Level continued to increase, reaching the High Reactor Level trip (~93 inches), at which time feedwater pump 12 tripped as designed. Maximum reactor water level reached was 104 inches, partially covering the Emergency Cooling System steam line nozzles but remaining below the Main Steam Line nozzles. The Emergency Cooling System was not required to operate for this event.

House loads on PB-12 transferred to offsite power as expected. PB-11 did not fast transfer per design as there was a lockout signal present to the offsite feeder breaker. There were no challenges to operators with respect to any electrical system beyond the initiating event.

II. Cause of Event

PB-11 was de-energized due to inadvertent protective over-current relay actuations that occurred when a 4.16 kV circuit breaker was being rolled into position within its PB-11 cubicle. Though workers did not recall creating any abnormal mechanical shock or jarring of the cubicle during this activity, analysis of the relays' physical and electrical characteristics, their control circuits, and surrounding electrical equipment, did not reveal any other credible mode of actuation. Actuation of multiple relays mounted on the cubicle door due to mechanical jarring was demonstrated using a mockup of a 4.16 kV power board and circuit breaker.

The underlying cause of the event is failure to adequately identify, assess, and mitigate potential risks of conducting work. The potential plant impacts that could be caused by inadvertent actuation or damage to surrounding equipment due to planned work were not identified or effectively managed.

III. Analysis of Event

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 the HPCI system is also reportable under 10 CFR 50.73(a)(2)(iv)(A).

There were no actual safety consequences associated with this event. All available plant safety systems operated as designed and the operators effectively stabilized reactor parameters. Water level was maintained well above the top of active fuel throughout the event and was restored to its desired value without the need for emergency core cooling system actuation. The conditional core damage probability for the event was calculated as 2.8E-6.

Based on the above, the event did not pose a threat to the health and safety of the public or plant personnel.

IV. Corrective Actions

A. Action Taken to Return Affected Systems to Pre-Event Normal Status

Plant shutdown was accomplished without assistance from PB-11. The over-current (51B) relays were tested and found to be within the design specification requirements. The power board was successfully re-energized from offsite power and loads were sequenced back on. No further issues were found with PB-11.

B. Action Taken or Planned to Prevent Recurrence

NOTE: There are no NRC regulatory commitments in this Licensee Event Report.

Corrective actions have been developed such that, when completed, the work control process will yield consistent and effective risk assessments of work activities that will prevent the same type of inadvertent relay actuation and other similar events (i.e., events caused by failure to identify and effectively manage potential risks associated with surrounding equipment). These actions include:

- Identifying other relays on switchgear/power boards and other trip sensitive equipment that, if inadvertently impacted, could cause undesirable equipment and plant impacts. Based on potential risk impacts, compensatory measures will be established to prevent significant adverse plant impacts.

- Improving risk assessment effectiveness by work planners, supervisors, and operations staff through enhanced administrative controls, improved tools, training, and qualification, and by re-affirming the lead role of the Operations department in the risk assessment process.

LICENSEE EVENT REPORT (LER)

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		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Nine Mile Point Unit 1	05000220	2005	-- 003 --	00	4 OF 4

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

V. Additional Information

A. Failed Components:

None

B. Previous similar events:

None

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

Components	IEEE 805 System ID	IEEE 803.1 Function
Reactor Protection System	JC	N/A
High Pressure Coolant Injection System	BJ	N/A
Feedwater System	SJ	N/A
Emergency Cooling System	BL	N/A
Main/Reheat Steam System	SB	N/A
Control Rods	AA	ROD
Turbine Bypass Valves	SB	PCV
Power Board	EA	SWGR
Pump	SJ, BJ	P
Breaker	EA	52
Relay	EA	51