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L-13-146

10 CFR 50.73(a)(2)(i)(B)

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001SUBJECT:
Perry Nuclear Power Plant
Docket No. 50-440, License No. NPF-58
Licensee Event Report Submittal

Enclosed is Licensee Event Report (LER) 2013-002, "Condition Prohibited by Technical Specifications Due to Scram Discharge Volume Level Switch Isolation." There are no regulatory commitments contained in this submittal.

If there are any questions or if additional information is required, please contact Mr. Thomas Veitch, Manager – Regulatory Compliance, at (440) 280-5188.

Sincerely,



Vito A. Kaminskas

Enclosure:
LER 2013-002cc: NRC Project Manager
NRC Resident Inspector
NRC Region IIIIE22
MUR

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: 80 hrs. 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.

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4. TITLE
Condition Prohibited by Technical Specifications Due to Scram Discharge Volume Level Switch Isolation

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
03	25	2013	2013	- 002	- 00	05	23	2013	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE 5	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)									
	<input type="checkbox"/> 20.2201(b)	<input type="checkbox"/> 20.2203(a)(3)(i)	<input type="checkbox"/> 50.73(a)(2)(i)(C)	<input type="checkbox"/> 50.73(a)(2)(vii)						
10. POWER LEVEL 000	<input type="checkbox"/> 20.2201(d)	<input type="checkbox"/> 20.2203(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(viii)(A)						
	<input type="checkbox"/> 20.2203(a)(1)	<input type="checkbox"/> 20.2203(a)(4)	<input type="checkbox"/> 50.73(a)(2)(ii)(B)	<input type="checkbox"/> 50.73(a)(2)(viii)(B)						
	<input type="checkbox"/> 20.2203(a)(2)(i)	<input type="checkbox"/> 50.36(c)(1)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(iii)	<input type="checkbox"/> 50.73(a)(2)(ix)(A)						
	<input type="checkbox"/> 20.2203(a)(2)(ii)	<input type="checkbox"/> 50.36(c)(1)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(iv)(A)	<input type="checkbox"/> 50.73(a)(2)(x)						
	<input type="checkbox"/> 20.2203(a)(2)(iii)	<input type="checkbox"/> 50.36(c)(2)	<input type="checkbox"/> 50.73(a)(2)(v)(A)	<input type="checkbox"/> 73.71(a)(4)						
	<input type="checkbox"/> 20.2203(a)(2)(iv)	<input type="checkbox"/> 50.46(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(v)(B)	<input type="checkbox"/> 73.71(a)(5)						
<input type="checkbox"/> 20.2203(a)(2)(v)	<input type="checkbox"/> 50.73(a)(2)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(v)(C)	<input type="checkbox"/> OTHER							
<input type="checkbox"/> 20.2203(a)(2)(vi)	<input checked="" type="checkbox"/> 50.73(a)(2)(i)(B)	<input type="checkbox"/> 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A							

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME Perry Nuclear Power Plant, John Pelcic, Compliance Engineer	TELEPHONE NUMBER (Include Area Code) (440) 280-5824
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13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED <input type="checkbox"/> YES (If yes, complete EXPECTED SUBMISSION DATE). <input checked="" type="checkbox"/> NO	15. EXPECTED SUBMISSION DATE MONTH: DAY: YEAR:
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ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On March 25, 2013, at approximately 1527 hours, maintenance personnel identified that a level instrument root valve for the control rod drive system, scram discharge volume (SDV) was closed and locked. With the valve in the closed position, the associated SDV float level switch, which provides input to the reactor protection system logic, could not perform its function and resulted in noncompliance with the Limiting Condition for Operation for Technical Specification 3.3.1.1.

The cause of the event was a failure to follow procedures. During the last performance of a calibration surveillance test, the instrumentation and control technicians did not open the valve as required and left the valve in the closed position. Independent verification of the valve position and locking device attachment failed to identify the error. Maintenance personnel promptly restored the valve to the locked open position. Corrective actions were administered through the internal performance management process. There were no system failures associated with this event.

The safety significance of this event is considered to be small. This event is being reported in accordance with 10 CFR 50.73(a)(2)(i)(B) as any operation or condition that was prohibited by the plant's Technical Specifications.

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NARRATIVE

Energy Industry Identification System (EIS) codes are identified in the text as [XX].

INTRODUCTION

On March 25, 2013, at approximately 1527 hours, maintenance personnel identified that control rod drive system [AA] valve 1C11-F0158A, SCRAM DISCH VOLUME LEVEL INST ROOT [RTV] was in the closed position and locked. With this valve closed, scram discharge volume (SDV) float switch 1C11-N013A [LS] could not perform its function. This configuration caused the limiting condition for operation (LCO) to not be met for Technical Specification (TS) 3.3.1.1, Reactor Protection System (RPS) Instrumentation. The plant was in Mode 5 (i.e., Refueling) at the time of discovery. This event is being reported in accordance with 10 CFR 50.73(a)(2)(i)(B) as any operation or condition prohibited by the plant's Technical Specifications.

EVENT DESCRIPTION

On February 18, 2013, with the plant in Mode 1 (i.e., Power Operation), surveillance instruction SVI-C11-T5376A, "SDV High Level Channel A Functional/Calibration for 1C11-N013A" was performed with acceptable results. This test is performed once every 92 days.

On March 18, 2013, following a manual RPS actuation (i.e., scram) to enter the current refueling outage, the operators observed that the SDV high level alarm remained locked-in after the RPS was reset. This was contrary to the expected response. A problem with the associated float switch 1C11-N0013A was initially suspected as the cause. The alarm remained locked-in and SDV level trip function was placed in bypass pending investigation and correction of the issue.

Troubleshooting of the condition was performed and on March 25, 2013, at approximately 1527 hours, instrumentation and control (I&C) personnel found that valve 1C11-F0158A SCRAM DISCH VOLUME LEVEL INST ROOT was in the closed position with a red locking device attached to the valve. With supervisor permission, the I&C technicians promptly reopened the valve. The level switch was reset and there were no further issues. All four SDV level instrument isolation valves were subsequently verified to be open and locked.

With 1C11-F0158A closed, float level switch 1C11-N013A could not perform its intended function, which is to provide a SDV high level input to the reactor protection system (RPS) [JC] trip logic. This switch is associated with Channel A of RPS trip system A. It is paired with float level switch 1C11-N013C which is associated with Channel C of RPS trip system A. A trip of either channel in RPS trip system A will cause a half-scram condition. Likewise, a trip of Channel B or Channel D in RPS trip system B will cause a half-scram condition. A trip of both RPS A and RPS B will result in a reactor scram.

Technical Specification (TS) 3.3.1.1, Reactor Protection System (RPS) Instrumentation, Table 3.3.1.1-1 requires two channels per trip system for Function 8.b, Scram Discharge Volume Water Level – High, Float Switch, in Modes 1 and 2. With only one required channel for RPS instrumentation Function 8.b, TS 3.3.1.1 Condition A requires that the channel or associated trip system be placed in trip within 12 hours. If the required action is not performed within the completion time, Condition D requires entry into the Condition referenced in Table 3.3.1.1-1 for the channel. The applicable Condition in this case is H, "Be in MODE 3." within 12 hours.

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Because it was not recognized that a condition existed requiring these TS actions to be performed, the actions were not taken, and the associated completion times were not satisfied. As a result, the plant was not in compliance with TS 3.3.1.1 since February 18, 2013. Not having both required channels for RPS Trip system A and not entering the applicable TS Action statements represented a condition prohibited by TS and is reportable in accordance with 10 CFR 50.73(a)(2)(i)(B).

CAUSE OF EVENT

The cause of the event was a failure to follow procedure in step-by-step sequence. The valve was last manipulated on February 18, 2013, during performance of SVI-C11-T5376A. I&C technicians who performed the test did not open the valve as required by the procedure steps and did not effectively perform independent verification of valve position and placement of locking devices as required by the procedure steps. The locking devices were installed prior to verification of valve position; the verifier relied upon the locking device tab color versus a hands-on manipulation. While the use of placekeeping was evident in the test package and the test steps were signed off as complete, these measures were not effective in preventing the error.

EVENT ANALYSIS

The SDV receives the water displaced by the motion of the control rod drive pistons during a reactor scram. The water level in each of the two SDVs is measured by two float type level switches and two transmitters and trip units for a total of eight level signals. Their outputs are arranged so that there is a signal from a level switch and a transmitter and trip unit to each RPS logic channel. Four channels of each type of SDV Level – High Function, with two channels of each trip system, are required to be operable to ensure that no single instrument failure will preclude a scram from these functions on a valid signal.

SDV Water Level-High functions 8.a and b are inputs to the RPS trip logic. However, no credit is taken for a scram initiated from these functions for any of the design basis accidents or transients analyzed in the USAR. They are retained in TS to ensure that the RPS remains operable.

The mispositioning of valve 1C11-F0158A prevented the TS required float level switch from properly sensing level in the SDV header. Since the reference leg of the SDV was isolated, the level indicator did not reset after the SDV drained when the scram was reset. The scram function of the control rod drive system was maintained. The other float switches in RPS trip system A and B were operable to insert a scram if an actual high level condition existed.

A qualitative probabilistic risk assessment (PRA) was performed for this event. While a single channel within the RPS A trip system was determined to be inoperable due to the valve misposition, the scram capability was not impacted due to the redundancy of the function. The event had no influence upon the plant's ability to successfully scram. Common cause/mode applications were not identified. On the basis that the scram function was not and would have not been affected, the PRA assessment concludes no change in core damage frequency (delta CDF) and no change in large early release frequency (delta LERF). The delta CDF and delta LERF values remain well below the acceptable thresholds of 1.0E-06/yr and 1.0E-07/yr respectively as discussed in Regulatory Guide 1.174. Plant configurations with changes in CDF of less than 1.0E-06 and LERF of less than 1.0E-07 are not considered to be significant risk events. Based on the PRA results, the safety significance of this event is considered to be small.

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CORRECTIVE ACTIONS

Corrective actions for this event were implemented under the FENOC performance management process. No procedure or hardware corrective actions were necessary. Dynamic learning activity (DLA) training for valve positioning, locking device placement, and independent verification techniques will be administered to PNPP I&C technicians. Field observations will then be performed by management personnel to verify that standards and expectations are being implemented.

PREVIOUS SIMILAR EVENTS

A review of LERs and the corrective action database for the past three years did not identify any previous similar events or condition reports associated with the SDV or the float level switches. A similar search for component misposition by I&C personnel did not identify previous instances where the corrective actions would have prevented the misposition of 1C11-F0158A. Condition Report 2013-00114, 'Water Overflow from SLC Test Tank during water sample due to valve misposition' was reviewed. This was a consequential mispositioning event for Operations. The cause was improper job assignment based on using the wrong work control document.

COMMITMENTS

There are no regulatory commitments contained in this report. Actions described in this document represent intended or planned actions, are described for the NRC's information, and are not regulatory commitments.