FENOC™
FirstEnergy Nuclear Operating Company

Perry Nuclear Power Plant
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Perry, Ohio 44081

David B. Hamilton
Vice President

440-280-5382

December 21, 2016
L-16-365

10 CFR 50.73(a)(2)(i)(A)
10 CFR 50.73(a)(2)(ii)(A)
10 CFR 50.73(a)(2)(iv)(A)
10 CFR 50.73(a)(2)(v)(A)
10 CFR 50.73(a)(2)(i)(B)

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

SUBJECT:
Perry Nuclear Power Plant
Docket No. 50-440, License No. NPF-58
Licensee Event Report Revision Submittal

Enclosed is Licensee Event Report (LER) 2016-001-01, "Pressure Boundary Leakage, Level 8 Automatic SCRAM, and APRM Loss of Safety Function". This revision is being submitted to clarify the 10 CFR 50.73 reporting requirements for the event. The original event was reported under 10 CFR 50.73(a)(2)(i)(A), 10 CFR 50.73(a)(2)(ii)(A), 10 CFR 50.73(a)(2)(iv)(A), and 10 CFR 50.73(a)(2)(v)(A). Revision 01 is being reported to also designate this event reportable pursuant to 10 CFR 50.73(a)(2)(i)(B). There are no regulatory commitments contained in this submittal.

If there are any questions or if additional information is required, please contact Mr. Nicola Conicella, Manager – Regulatory Compliance, at (440) 280-5415.

Sincerely,

David B. Hamilton
Vice President

Enclosure:
LER 2016-001-01

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III Regional Administrator

Enclosure
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LER 2016-001-01

NRC FORM 366 (02-2014) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 1/31/2017
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 feed back to industry. Send comments regarding burden estimate to the FOIA. Privacy Information Collection Branch (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
Pressure Boundary Leakage, Level 8 Automatic SCRAM, and APRM Loss of Safety Function

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
01	24	2016	2016	- 001 -	01	03	23	2016	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 Specify in Abstract below or in NRC Form 366A
☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Tony Kledzik – Regulatory Compliance	440-280-6188

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	AD	PSF	X999	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (If yes, complete EXPECTED SUBMISSION DATE).	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

At 2100 hours, on January 23, 2016, the Perry Nuclear Power Plant (PNPP) commenced a reactor shutdown to investigate unidentified leakage in the drywell. At 2122 hours, drywell unidentified leakage exceeded Technical Specification (TS) limits necessitating a plant shutdown as required by TSs. At 0357 hours, on January 24, 2016, while performing the shutdown required by plant TSs, the average power range monitors (APRM) became inoperable due to a calibration setpoint being out of tolerance in the nonconservative direction following a transfer of the reactor recirculation pumps to slow speed. This resulted in a loss of safety function for the APRMs. At 1007 hours, on January 24, 2016, with the plant at 8 percent power, during a feedwater shift to place the motor feed pump in service, reactor water level rose to the level 8 setpoint and the reactor protection system (RPS) automatically initiated, shutting down the reactor. Following the shutdown, a small leak was identified on the reactor recirculation loop "A" pump discharge valve vent line. The recirculation loop is part of the reactor coolant system; this resulted in a degraded condition and a condition prohibited by TS due to pressure boundary leakage.

The cause of the recirculation loop vent line leak was that the weld connecting the root appendage was not performed per the design drawing. The APRM calibration issue was caused by a change to the feedwater flow input to the heat balance. The cause of the reactor level rise and subsequent high water level scram was due to operator error in monitoring and manipulating feedwater system indications and controls.

The safety significance of this event is considered to be small. These events are being reported under; 50.73(a)(2)(i)(A), for completion of any plant shutdown required by the plant's TS; 50.73(a)(2)(ii)(A) for a condition resulting in the plant's principle safety barrier being seriously degraded; 50.73(a)(2)(i)(B) for a violation of Technical Specifications; 50.73(a)(2)(iv)(A) for actuation of the RPS while critical; and 50.73(a)(2)(v)(A) for a loss of safety function.

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NARRATIVE

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

INTRODUCTION

On January 23, 2016 at 2100 hours, with the plant in mode 1 at 100 percent power the Perry Nuclear Power Plant (PNPP) commenced a reactor shutdown to investigate unidentified leakage in the drywell [VB]. At 2122 hours, drywell unidentified leakage exceeded the Technical Specification (TS) 3.4.5.d limit of 'less than or equal to 2 gpm increase in unidentified leakage within the previous 24 hour period in Mode 1.' The unidentified leakage increased to approximately 2.85 gpm at 2122 hours. TS 3.4.5 actions allow 4 hours to reduce the leakage within limits or be in Mode 3 within 12 hours and Mode 4 within 36 hours. On January 24, 2016 at 0056 hours, an event notification was made to the NRC Operations Center in accordance with 10CFR50.72(b)(2)(i) for the initiation of a plant shutdown required by TS due to the expected inability to restore the leakage within limits prior to exceeding the limiting condition of operation (LCO) action time. (Reference ENF No. 51679)

On January 24, 2016 at 0357 hours, with the plant in mode 1 at 26 percent power, 7 of 8 Average power range monitor's (APRMs) [IG] were declared inoperable in accordance with TS 3.3.1.1 due to being greater than 2 percent out of calibration when compared to the calculated heat balance. APRM "A" was previously declared inoperable and bypassed due to a failed input. The APRMs were out of calibration in the nonconservative direction, (i.e. reading greater than 2 percent below the calculated reactor power) which resulted in a loss of safety function. On January 24, 2016 at 1123 hours, an event notification was made to the NRC Operations Center in accordance with 10CFR50.72(b)(3)(v)(A) for the loss of safety function . (Reference ENF No. 51681)

On January 24, 2016 at 1007 hours, with the plant at 8 percent power and during a feedwater shift to place the motor feed pump [SJ] in service, reactor level rose to the level 8 setpoint and the Reactor Protection System (RPS) automatically initiated, shutting down the reactor. On January 24, 2016 at 1123 hours, an event notification was made to the NRC Operations Center in accordance with 50.72(b)(2)(iv)(B) for an RPS initiation while critical. (Reference ENF No. 51679)

On January 24, 2016 at 1617 hours, after entry into the drywell, a small leak was identified on the reactor recirculation [AD] loop "A" pump discharge valve vent line. The recirculation loop is part of the reactor coolant system, this resulted in a degraded condition. Due to the discovered pressure boundary leakage it was determined that a plant cool down was required in accordance with TS 3.4.5, Action C, which requires the plant to be in MODE 4 within 36 hours. TS 3.4.5 was previously entered for increased unidentified leakage in the drywell. On January 24, 2016 at 1915 hours, an event notification was made to the NRC Operations Center in accordance with 50.72(b)(3)(ii)(A)) for a condition resulting in a principal safety barrier being seriously degraded. (Reference ENF No. 51679)

EVENT DESCRIPTION

Increased Drywell Unidentified Leakage:
On January 19, 2016, unidentified drywell sump [WK] in-leakage began to show a rising trend, with a starting rate of 0.07 gpm. Over the next three days, drywell unidentified leakage continued to rise and the particulate channel of the drywell radiation monitor also began to rise. At 2100 hours, on January 23, 2016, the PNPP commenced a reactor shutdown to investigate the unidentified leakage. Unidentified leakage prior to commencing the shutdown was 1.75 gpm.

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At 2122 hours, on January 23, 2016, the unidentified leakage increased to approximately 2.85 gpm exceeding the TS 3.4.5.d limit of 'less than or equal to 2 gpm increase in unidentified leakage within the previous 24 hour period in Mode 1.' On January 24, 2016 at 0056 hours, an event notification was made to the NRC Operations Center in accordance with 10CFR50.72(b)(2)(i) for the initiation of a plant shutdown required by TS due to the expected inability to restore the leakage within limits prior to exceeding the LCO action time.

A drywell entry was made and at 1617 hours, on January 24, 2016. A small leak was identified on the reactor recirculation loop "A" pump discharge valve vent line. The recirculation loop is part of the reactor coolant system and a leak in this vent line results in a degraded condition and a condition prohibited by TS due to pressure boundary leakage. On January 24, 2016 at 1915, an event notification was made to the NRC Operations Center in accordance with 50.72(b)(3)(ii)(A) for a condition resulting in a principal safety barrier being seriously degraded.

The failure occurred at the weld that connects vent valve appendage 1B33F0068A and 1B33F0069A to the reactor recirculation loop "A" pump discharge valve, 1B33F0067A. This appendage had been replaced in the last refuel outage in May of 2015 (1R15) due to an extent of condition from a previous appendage failure. Visual inspection showed a crack propagating between the toes of the fillet welds on the pipe. Based on the as found condition of the weld described above, it was concluded that the field weld was not made per the weld design drawing.

Average Power Range Monitor Calibration:
On January 24, 2016 at 0357, with the plant in mode 1 at 26 percent power, 7 of 8 APRMs were declared inoperable due to being greater than 2 percent out of calibration when compared to the calculated heat balance (APRM "A" was previously declared inoperable and bypassed due to a failed LPRM). The APRMs were out of calibration in the nonconservative direction, (reading greater than 2 percent below the calculated reactor power) which resulted in a loss of safety function. TS 3.3.1.1 was entered with the most limiting action to restore RPS trip capability within 1 hour. On January 24, 2016 at 1123 hours, an event notification was made to the NRC Operations Center in accordance with 10CFR50.72(b)(3)(v)(A) for the loss of safety function. This event occurred following a transfer of the reactor recirculation pumps to slow speed and a subsequent 3A feedwater heater [SM] isolation. The calibrated APRM input to the RPS was restored at 0445 hours, on January 24, 2016.

Automatic RPS Actuation on Level 8 during a Feedwater Shift:
On January 24, 2016, the control room operators were in the process of shutting down to support a drywell entry to identify the source of unidentified leakage in the drywell. Reactor power had been reduced to 8 percent and the generator was off-line. As part of the shutdown sequence, the Control Room operators were in the process of shifting feed water from the reactor feed pump turbine "A" (RFPT "A") [SJ] to the motor feed pump (MFP).

At approximately 1003 hours, the control room operator performing the feed pump shift began feeding the vessel with the MFP. When the operator began to see the anticipated slow rise in vessel level, he reduced the RFPT "A" flow. The change in RFPT "A" flow resulted in a lowering trend in the reactor vessel level. The operator began increasing the MFP flow to the vessel in response to the lowering vessel level. At approximately 1005 hours, the operator was successful in stopping the lowering vessel trend; however the MFP was now feeding the vessel at a rate that resulted in an increasing level trend.

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As the vessel level neared its original level, the operator began reducing the RFPT "A" controller output in an attempt to reduce feed flow and stabilize vessel level. However, the initial reduction in flow from the RFPT "A" had lowered flow to a point that RFPT "A" was no longer feeding the vessel. Vessel level continued to rise. At 1007, reactor vessel level reached the level 8 setpoint (219") and an automatic RPS actuation occurred. On January 24, 2016 at 1123 hours, an event notification was made to the NRC Operations Center in accordance with 50.72(b)(2)(iv)(B) for an RPS initiation while critical.

CAUSE OF EVENT

Increased Drywell Leakage:
The organization failed to effectively manage the work performed on reactor pressure retaining equipment, resulting in a deficient weld. No additional process requirements or barriers were in place to heighten awareness of work performed on critical systems, like the reactor coolant pressure boundary.

The instruction in the work order to replace the vent appendage on discharge valve 1B33F0067A in 1R15 did not contain sufficient detail to ensure the final weld satisfied the design drawing requirements, resulting in a failed weld and a plant shutdown.

The reactor recirculation appendages at PNPP have a unique weld profile to prevent cracking. This weld profile is referred to as the "hourglass" configuration. The recirculation loop valve appendages all have the hourglass configuration. If applied correctly, the weld buildup has mitigated cracking.

The failed weld was performed as a standard weld with no additional precautions or controls in the work planning process for working on the reactor coolant pressure boundary.

The primary document guiding the welder referenced the design engineering drawing in the remarks. The design drawing requirements were not clarified on the primary document. Based upon the observed results, the design drawing was not interpreted correctly.

Average Power Range Monitor Calibration:
The station failed to effectively manage the impacts of the recirculation pump downshift evolution on the calculated thermal power. The plant computer calculates the reactor heat balance that is used to calibrate the APRMs. As part of that calculation, the feedwater flow to the vessel is calculated using either the feedwater flow from venturi in the feedwater lines or the suction minus recirculation flow through the reactor feed pumps. If the reactor recirculation pumps are in fast speed, then the computer will use the venturi feedwater flow exclusively. If the reactor recirculation pumps are running in slow speed, then the computer will use the larger of the venturi or suction minus recirculation feedwater flows.

In this event, the reactor recirculation pumps had just been shifted to slow speed and the plant computer used suction minus recirculation flow for the heat balance input. This feed flow input to the heat balance was larger than the venturi flow. With a larger feedwater flow input to the heat balance, calculated reactor power was higher than calculated power using the venturi flow. This caused APRMs to read greater than 2 percent below the calculated reactor power.

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Automatic RPS Actuation on Level 8 during a Feedwater Shift:
The operators were focused on the efforts to stabilize the reactor vessel level fluctuations resulting in an RPS actuation on reactor pressure vessel (RPV) level 8 during the feedwater pump shift. This type of feedwater shift is typically performed at a higher power level. When performed at a higher power level, the shift utilizes the automatic functions of the digital feedwater system. The shift that caused the RPV level 8 RPS actuation was performed at a lower power level, and in this configuration the shift is performed in manual.

EVENT ANALYSIS

The unidentified leakage condition within the drywell, and the corresponding controlled shutdown in accordance with the TS are not evaluated as an initiating event based on the fact that these conditions did not result in an immediate scram of the reactor vessel and did not require advanced mitigative strategies to be employed. The Plant Probabilistic Risk Assessment (PRA) does not consider controlled plant shutdowns as initiating events.

The out-of-calibration condition of the APRMs are outside of the PRA modeling boundaries. PRA analyzes the event sequences from the onset of a plant transient to a given end state from the mitigative perspectives. Reactor core operational aspects prior to the given transient are managed and assessed via TS. The APRMs were calibrated within the required TS time frame, therefore the risk is small.

A PRA evaluation was performed for the January 24, 2016 RPV level 8 scram. A conservative analysis of this uncomplicated plant scram indicates a conditional core damage probability (CCDP) of 2.81x10-7, which corresponds to a delta CDF of 2.81x10-7 /yr. This delta CDF is well below the acceptable threshold of 1x10-6/yr, as discussed in Regulatory Guide 1.174. The risk of this event is therefore considered small in accordance with the Regulatory Guidance.

CORRECTIVE ACTIONS

Increased Drywell Leakage:
To correct the direct cause, the vent appendage was removed and replaced with a more robust pipe and cap. In addition, the weld process will be modified to contact the weld engineer to ensure that the design drawing requirements are properly reflected within the primary document guiding the welder. A new process will be implemented to heighten awareness and to include criteria and guidelines for work performed on critical equipment, such as the reactor pressure boundary.

Average Power Range Monitor Calibration:
No physical repairs were needed as this is a known plant response to a recirculation pump down shift. Operations procedures have been updated to reflect the known plant response. Procedural steps were also added to ensure that one APRM from each reactor protection system channel is calibrated to the estimated power level of the new heat balance prior to a pump down shift.

Automatic RPS Actuation on Level 8 during a Feedwater Shift:
Operations procedures were revised to add a caution and steps to direct shifting from the reactor feed pump turbine to the motor feed pump at 15 to 18 percent power to ensure the automatic level control

LICENSEE EVENT REPORT (LER) CONTINUATION SHEET

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functions of the digital feedwater system are utilized. Individual performance issues for this event were addressed under the FENOC performance management process.

PREVIOUS SIMILAR EVENTS

A review of LERs and the corrective action database for the past three years showed one similar event. On June 15, 2013, a similar leak was discovered on the vent appendage for the reactor recirculation system flow control valve on the "B" recirculation loop. An event notification was made at 0242 hours, on June 16, 2013 . The leak in the appendage was at the weld to the first vent valve, not at the pipe. The failure was from the inner wall to the outer wall and determined to be corrosion fatigue. This is different from the current failure, which occurred due to an improper weld. None of the corrective actions from this previous similar event would have reasonably been expected to prevent the events from this current LER.

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.