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L-13-301

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

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 Submittal

Enclosed is Licensee Event Report (LER) 2013-003-01, "Shutdown Required by Technical Specifications Due to RCS Pressure Boundary Leakage." This supplement is being submitted to update the cause analysis and corrective actions associated with this event. 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,

Ernest J. Harkness

Enclosure:
LER 2013-003-01

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

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 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.

LICENSEE EVENT REPORT (LER)

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

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Perry Nuclear Power Plant, Unit 1	05000-440	1 OF 4

4. TITLE

Shutdown Required by Technical Specifications due to RCS Pressure Boundary Leakage

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
06	15	2013	2013	- 003	- 01	10	04	2013	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE: 1

10. POWER LEVEL: 008

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)*
Perry Nuclear Power Plant, John Pelcic, Compliance Engineer	(440) 280-5824

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

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)*

A planned power reduction commenced June 14, 2013, to inspect the Drywell for sources of unidentified leakage. Following a refueling outage completed in May 2013, the Drywell unidentified leakage was higher than levels prior to the outage. The Drywell inspection identified two leak sites, one of which was in the reactor coolant system (RCS) pressure boundary. A plant shutdown was conducted in accordance with Technical Specification 3.4.5, RCS Operational Leakage, to facilitate repairs. During the shutdown process and after the reactor was subcritical, the reactor protection system was actuated to insert the remaining withdrawn control rods.

The cause of the RCS pressure boundary leakage is a combination of stress corrosion cracking and fatigue or corrosion fatigue. A new vent valve assembly was fabricated and installed on the reactor recirculation system B flow control valve. Inspection of other vent and drain valves with similar configuration on the reactor recirculation system found no deficiencies. Design configuration options to address the cause will be evaluated.

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)(A), 10 CFR 50.73(a)(2)(ii)(A), and 10 CFR 50.73(a)(2)(iv)(A).

NARRATIVE

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

INTRODUCTION

On June 16, 2013, at 0200 hours, a controlled plant shutdown was conducted in accordance with Technical Specification (TS) 3.4.5, RCS Operational LEAKAGE, to repair reactor coolant pressure boundary leakage inside the Drywell. Both initiation of the shutdown required by TS and the principal safety barrier degradation were reported to the NRC at 0242 hours, June 16, 2013, in accordance with 10 CFR 50.72(b)(2)(i) and 10 CFR 50.72(b)(3)(ii)(A); reference ENF Notification No. 49121. The reactor shutdown was completed at 0340 hours on June 16, 2013, when the reactor went subcritical during control rod insertion. A manual reactor protection system (RPS) [JC] actuation was used to insert the remaining withdrawn control rods. This event is being reported in accordance with 10 CFR 50.73(a)(2)(i)(A), 10 CFR 50.73(a)(2)(ii)(A), and 10 CFR 50.73(a)(2)(iv)(A).

EVENT DESCRIPTION

Since Cycle 15 operations commenced on May 16, 2013, above normal levels of unidentified Drywell sump in-leakage were experienced. The unidentified Drywell sump in-leakage increased from 0.31 gallons per minute (gpm) to 0.58 gpm, which are above the typical Cycle 14 levels of approximately 0.2 gpm. This condition was entered in the corrective action program (CAP) and was being monitored in accordance with plant procedures and an approved Operational Decision Making Issue document. The Limiting Condition for Operation (LCO) TS 3.4.5.b limit for RCS unidentified operational leakage is less than or equal to 5 gpm.

On June 14, 2013, at 1900 hours with the plant operating in Mode 1, a power reduction from 100 percent rated thermal power (RTP) was commenced to establish plant conditions to perform a Drywell inspection in order to determine the source of the unidentified leakage. On June 15, 2013, at approximately eight percent RTP, an inspection of the Drywell was conducted.

The inspection identified two leak sites. One leak was at the reactor pressure vessel flange connection for control rod drive mechanism (CRDM) 30-15 [AA] in the undervessel region. The other leak was at the top of Reactor Recirculation system [AD] flow control valve (FCV) B [FCV] 1B33F0060B. Both leak sources were entered in CAP.

A Drywell entry determined that the leakage was at a three-quarter inch diameter vertical vent appendage, located off the top of the FCV. This vent appendage has double root isolation valves in series with a pipe cap at its end. Valves 1B33F0647B and 1B33F0648B [VTV] provide double isolation for venting and are welded in a socket weld joint configuration. The socket weld leg on the inlet to the first vent valve, 1B33F0647B, was cracked circumferentially.

The leakage, estimated at 0.2 gpm, represented RCS pressure boundary leakage. TS LCO 3.4.5.a states that RCS operational LEAKAGE shall be limited to no pressure boundary LEAKAGE. On June 16, 2013, at 0200 hours, the operators commenced a controlled plant shutdown from approximately eight percent RTP in compliance with TS to repair the leak. Both the shutdown required by TS and the degraded principal safety barrier were reported in ENF No. 49121. The operators had entered TS 3.4.5 Condition C, "Pressure boundary LEAKAGE exists." and performed Required Action C.1, "Be in MODE 3." (12 hours) and C.2, "Be in MODE 4." (36 hours).

NARRATIVE

At 0313 hours, the plant entered Mode 2 when the operators placed the mode switch in the Startup/Hot Standby position.

The type of plant shutdown conducted was a rapid soft shutdown where the operators insert control rods manually. At 0340 hours, the Unit Supervisor declared the reactor subcritical based on in-core nuclear instrumentation readings. The operators continued to insert control rods in accordance with the established shutdown sequence. At 0353 hours, the rod control and information system (RC&IS) [AA] malfunctioned preventing normal control rod movement. The operators entered the off-normal instruction for inability to move control rods. At 0403 hours, a manual reactor scram was inserted to complete the shutdown in accordance with normal operating procedures and the evolution specific reactivity plan termination criteria. All withdrawn control rods at the time fully inserted and there were no complications experienced in the scram. Mode 3 was entered at 0403 hours. The scram was reset at 0410 hours.

Mode 4 was entered at 1358 hours.

CAUSE OF EVENT

The cause of the through wall leak at the inlet socket weld to 1B33F0647B is a combination of stress corrosion cracking and fatigue, which is referred to as corrosion fatigue. Age-related coincident factors worked together to result in a breakdown of the material and to create a susceptibility to corrosion fatigue.

Stress corrosion cracking requires a susceptible material, a corrosive environment, and a tensile stress. Stainless steel is a material that is susceptible to stress corrosion cracking and to corrosion fatigue. The crevice formed from the socket weld provides for a natural accumulation of contaminants that are not readily flushed as they are isolated from the flow stream. Fatigue stresses can be either thermal fatigue or vibration fatigue. In this case, the fatigue stress was caused by vibration. Therefore, the cause of this event has been attributed to corrosion fatigue. The vent valve appendage had been in-service approximately 27 years prior to failure of the weld.

EVENT ANALYSIS

The initial reactor downpower to eight percent RTP and the rapid soft shutdown were performed in accordance with plant operating procedures and the reactivity plan. The RC&IS malfunction and the manual RPS actuation occurred after the reactor was subcritical. No plant parameters experienced in the shutdown process challenged the transients described in the Updated Safety Analysis Report Chapter 15, Accident Analysis.

The reactor recirculation system (RRS) provides a forced coolant flow through the core to remove heat from the fuel to allow operation at significantly higher power levels than would otherwise be possible. The system consists of two recirculation flow loops each consisting of a motor driven pump and a flow control valve. The RCS pressure boundary leak, as described, would not have prevented RRS flow control valve B from performing its design function.

A qualitative probabilistic risk assessment (PRA) was performed for this event. While RCS pressure boundary leakage was present, this leakage did not require an immediate scram. The PRA model does not consider a controlled plant shutdown as an initiating event. Furthermore, the given condition did not make any PRA modeled equipment/functions unavailable. Therefore, the PRA assessment concludes there are no changes in core damage frequency (CDF) or large early release frequency (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.

NARRATIVE

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.

CORRECTIVE ACTIONS

Visual (VT-2) and liquid penetrant (PT) inspections were performed on the other vent and drain appendages for the RRS flow control valves. Examinations of all welds on these valves were satisfactory with no indications of surface cracks.

A new vent valve assembly was fabricated and installed on 1B33F0060B. The schedule 80 pipe originally installed between 1B33F0060B and 1B33F0647B was replaced with schedule 160 pipe. The socket weld was built up to the design dimensions. This new configuration is stiffer, resulting in a higher natural frequency of the appendage. This new configuration is less likely to be impacted by resonant vibrations that adversely affected the previous designed appendage.

The removed vent valve assembly containing the cracked weld was sent off-site to a qualified testing facility for metallurgical failure analysis. The results are discussed earlier in this LER.

Design configuration options will be evaluated to minimize the sensitivity of unsupported appendages on the RRS piping to corrosion fatigue. The options include use of additional supports, modifying the line stiffness, elimination of appendages, and use of corrosion resistant material.

The flange connection for CRDM 30-15 is a mechanical joint and was reworked to eliminate any reactor water leakage. Corrective actions to repair the RCS pressure boundary leakage and the CRDM flange connection restored Drywell unidentified leakage back to less than the previous operating cycle leakage.

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 RCS pressure boundary leakage. Two LERs were written for completion of shutdown required by TS. These include LER 2011-002-01, Condition Prohibited by Technical Specifications and Plant Shutdown due to Unit 1 Startup Transformer Issues, and LER 2010-003, Loss of Control Rod Drive Header Pressure Results in Manual RPS Actuation. None of the corrective actions for these LERs would have been reasonably expected to prevent the event documented in LER 2013-003.

COMMITMENTS

There are no regulatory commitments for these LERs 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.