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NL-17-107

August 29, 2017

U.S. Nuclear Regulatory Commission
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SUBJECT: Licensee Event Report # 2015-001-02, "Technical Specification (TS) Prohibited Condition Due to an Inoperable Containment Caused by a Service Water Pipe Leak with a Flaw Size that Results in Exceeding the Allowed Leakage Rate for Containment"
Indian Point Unit No. 2
Docket No. 50-247
DPR-26

References:
1. Licensee Event Report # 2015-001-00, letter NL-15-124, dated October 9, 2015
2. Licensee Event Report # 2015-001-01, letter NL-16-108, dated September 29, 2016

Dear Sir or Madam:

Pursuant to 10 CFR 50.73(a)(1), Entergy Nuclear Operations Inc. (ENO) hereby provides Licensee Event Report (LER) 2015-001-02. The attached LER is a revision to an LER submitted by Reference 2, that identifies an event where there was a Technical Specification (TS) Prohibited Condition due to not meeting Containment integrity as a result of a Containment Fan Cooler Unit motor cooler service water return pipe defect whose leakage could result in post-LOCA air leakage out of containment in excess of that allowed by Technical Specification 3.6.1 (Containment), which requires leakage rates to comply with 10 CFR 50, Appendix J. This condition is reportable under 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(v)(C). This condition was recorded in the Entergy Corrective Action Program as Condition Report CR-IP2-2015-3550. This LER revision is being provided to report changes in the direct cause, the apparent cause, one corrective action, and to make minor editorial changes.

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NRR

There are no new commitments identified in this letter. Should you have any questions regarding this submittal, please contact Mr. Robert Walpole, Manager, Regulatory Assurance at (914) 254-6710.

Sincerely,

AJV/gd

Attachment: LER-2015-001-02

cc: Mr. Daniel H. Dorman, Regional Administrator, NRC Region I
NRC Resident Inspector's Office
Ms. Bridget Frymire, New York State Public Service Commission

NRC FORM 366 (04-2017) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 03/31/2020
LICENSEE EVENT REPORT (LER) (See Page 2 for required number of digits/characters for each block) (See NUREG-1022, R.3 for instruction and guidance for completing this form http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Information Services Branch (T-2 F43), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by 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.

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Technical Specification (TS) Prohibited Condition Due to an Inoperable Containment Caused by a Service Water Pipe leak with a Flaw Size that Results in Exceeding the Allowed Leakage Rate for Containment

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	11	2015	2015	- 001	- 02	08	29	2017	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
100%	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☒ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
	☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
		☐ 50.73(a)(2)(i)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A	

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Dennis Pennino, Engineer, Engineering Systems	(914) 254-7216

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
X	BI	PSP	U080	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 11, 2015, during operator investigations inside the reactor containment building, a through wall leak was discovered on the 24 Fan Cooler Unit (FCU) motor cooler service water (SW) return line. The leak was in a 2 inch copper-nickel pipe near a brazed joint upstream of containment penetration SS. The leak was located within the ASME Section XI Code ISI Class 3 boundary and estimated to be approximately 2 gpm. Since the pipe flaw was through wall and was located within the ASME Section XI boundary, it exceeds the flaw allowable limits provided per IWC-3000. The weld leak was evaluated and determined to meet the structural requirements of ASME Code Case N-513-3. The condition was determined to have no impact on SW cooling safety function or adverse impact on piping structural integrity. The pipe is considered a closed loop system inside containment and required to meet containment integrity. An engineering evaluation was performed to determine the potential air leakage out of containment based on the observed SW leakage into containment. This evaluation concluded that the leaking defect could result in post-LOCA air leakage out of containment in excess of that allowed by Technical Specification 3.6.1 (Containment) which requires leakage rates to comply with 10 CFR 50, Appendix J. The direct cause was localized flow accelerated corrosion attack. The apparent cause was higher than required flow conditions which caused localized higher velocities and flow separation at the sharp interior edge of the pipe elbow. The pipe and affected elbow were replaced in accordance with the requirements of ASME Section XI Code during the spring refueling outage in 2016. Flow through the FCU motor coolers was also reduced to prevent the conditions from occurring that lead to flow accelerated corrosion. The event had no significant effect on public health and safety.

1. FACILITY NAME	2. DOCKET NUMBER	3. LER NUMBER		
		YEAR	SEQUENTIAL NUMBER	REV NO.
Indian Point 2	05000-247	2015	- 001	- 02

NARRATIVE

Note: The Energy Industry Identification System Codes are identified within the brackets { }.

DESCRIPTION OF EVENT

On August 11, 2015, at approximately 16:45 hours, during operator investigations inside the reactor containment building {NH}, a through wall leak was discovered on the 24 Fan Cooler Unit (FCU) {FCU} motor cooler service water (SW) {BI} return line. The leak was in a 2 inch 90-10 Copper-Nickel pipe {PSP}, line #495-SWN-NF near a brazed joint upstream of containment penetration SS at elevation 76 feet. The condition was recorded in the Indian Point Energy Center (IPEC) Corrective Action Program (CAP) in Condition Report CR-IP2-2015-03550. NDE by UT was performed to characterize the flaw and the results evaluated using the structural margins provided in ASME Code Case N-513-3.

The leak was located within the ASME Section XI Code, Class 3 boundary and estimated to be approximately 2 gpm. Since the pipe flaw was through wall and was located within the ASME Section XI boundary, it exceeded the flaw allowable limits provided per IWC-3000. Since the leaking defect was determined to be within the structural limits of the ASME Code Case N-513-3, the condition was determined to have no adverse impact on SW cooling safety function or on the structural integrity of the system. There was no visual indication of weld or base metal degradation at the affected pipe section, and there was no evidence of leakage at any other location on this weld or elsewhere on the piping adjacent to it. The leakage, if not contained, would have eventually drained into the containment sump. The impact of the pipe flaw was evaluated for containment free volume in-leakage and the leak is not expected to exceed the limit. The pipe is considered a closed loop system inside containment and required to meet containment integrity. An engineering evaluation was performed to determine the potential air leakage out of containment based on observed SW leakage into containment. This evaluation concluded that the leaking defect could result in post-LOCA air leakage out of containment in excess of that allowed by Technical Specification (TS) Surveillance Requirement (SR) 3.6.1.1 (Containment) which requires leakage rates to comply with 10 CFR 50, Appendix J.

The leak in 2 inch line #495-SWN-NF at the affected location is downstream of 24 FCU motor cooler which can be supplied with SW via either 18 inch line #408 (4-5-6 SW Header) or 18 inch line #409 (1-2-3 SW Header). At the time of discovery, the SW System (SWS) was aligned with the 4-5-6 SW Header as the Essential Header for Modes 1-4 Operations per Technical Specification (TS) 3.7.8 (Service Water System). The SWS is designed to supply cooling water from the Hudson River to various heat loads in both the primary and secondary portions of the plant. The design ensures a continuous flow of cooling water to those systems and components necessary for plant safety during normal operation and under abnormal or accident conditions. The SWS consists of two separate, 100% capacity, safety related cooling water headers. Each header is supplied by 3 pumps each having its own strainer, with SWS heat loads designated as either essential or non-essential. The essential SWS heat loads are those which must be supplied with cooling water immediately in the event of a Loss of Cooling Accident (LOCA) and/or Loss of Offsite Power (LOOP). The essential SWS heat loads can be cooled by any two of the three SW pumps on the essential header. Either of the two SWS headers can be aligned to supply the essential heat loads or the non-essential SWS heat loads. The design pressure and temperature of the SWS is 150 psig and 160 degrees F. The function of line #495 is to return the SW that was used to cool the 24 FCU motor out of containment and discharge it to the discharge canal.

The FCUs consists of a motor, fan, cooling coils, dampers, duct distribution system and instrumentation and controls. The FCU motor is equipped with a cooler supplied by SW. SW is supplied to the cooling coils to perform the heat removal function. During normal operation, SW is supplied to all five FCUs and one or more FCUs may be operated for containment cooling to limit the ambient containment temperature to less than the limit specified in TS 3.6.5.

An extent of condition (EOC) review determined other SW copper-nickel piping could be susceptible to similar degradation. EOC inspections are required by ASME Code Case N-513-3 in at least 5 similar and susceptible locations. The results of the inspections found all 5 selected locations to be structurally acceptable. There is no evidence of additional leakage at any other location in the 2 inch #495 line or at any other location in the other four FCU motor cooler return lines. The FCU motor cooler tubing is 6 percent molybdenum stainless steel which has a high resistance to corrosion vs the 90-10 copper-nickel piping material.

CAUSE OF EVENT

The direct cause was corrosion. Corrosion of the copper-nickel piping resulted in a through wall pipe defect and leakage of SW into the reactor containment. The leak was estimated to be approximately 2 gpm. Based on the UT wall thickness measurements the leak occurred at a degraded area approximately 0.5 inch long in both the axial and circumferential directions. Based on the Code Case N-513-3 evaluation, the allowable flaw length in the axial and circumferential direction was calculated to be 1.47 inches and 1.10 inches respectively. Using a calculated corrosion rate of 4.7 mil (0.0047 inches) per year based on measured conditions, the predicted flaw length by the next refueling outage in the spring of 2016 was estimated to be approximately 0.51 inches in both the axial and in the circumferential directions, which are within the allowable flaw lengths. Based on these results, the affected piping is structurally acceptable consistent with the requirements of ASME code Case N-513-3. The specific corrosion type was localized flow accelerated corrosion attack.

The apparent cause was higher than required flow conditions which caused localized higher velocities and flow separation at the sharp interior edge of the pipe elbow.

CORRECTIVE ACTIONS

The following corrective actions have been or will be performed under the Entergy Corrective Action Program to address the causes of this event.

- A leak-limiting engineered clam-shell type clamp was applied to the pipe flaw until a code repair could be completed.
- The clamp was monitored daily by a special operator log for any signs of increased leakage until a code repair could be completed.
- UT monitoring was performed every 90 days until the pipe was repaired.
- The pipe and affected elbow was replaced in accordance with the requirements of ASME Section XI Code during the spring refueling outage in 2016.
- The removed pipe/elbow was sent out to a vendor (LPI) for a metallurgical analysis to determine/confirm the specific cause.
- The flow balance test procedure was revised and the flow balance was performed to reduce the flow through the FCU motor coolers from 55-60 gpm to 25-30 gpm.

EVENT ANALYSIS

The event is reportable under 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(v). The licensee shall report any operation or condition which was prohibited by the plant's TS. This condition meets the reporting criteria because TS 3.6.1 Containment Operability was not met. Because the FCUs are utilized to maintain normal containment temperature within accident analysis input limits and for accident mitigation, SW flow is not isolated following a DBA. SW piping supply and discharge for the FCUs is considered to be a closed system in containment or an extension of the containment boundary. Consequently, defects discovered within this piping may adversely affect containment integrity, and the ability to control release of radioactive materials. The isolation valve for the faulted FCU SW piping was operable but the line was not isolated therefore the applicable TS not entered.

Initial operability determined the leak was within the Code Case Limits, that the inventory loss was not significant with no impact on other safety related structures, systems and components. Initially TS 5.5.14 requirements was not evaluated but subsequently it was determined the 10 CFR 50, Appendix J requirements were more stringent. The SW pipe flaw leakage was evaluated for containment in-leakage for potential flooding and out leakage for containment integrity per TS 5.5.14 and 10 CFR 50, Appendix J. TS 5.5.14.e requires that SW in-leakage into containment be limited to less than 0.36 gpm per FCU. Prior testing provided measured leakage that was less than the TS limit. Containment out leakage is required to be in accordance with TS Surveillance Requirement (SR) 3.6.1.1 whose leakage rate requirements comply with 10 CFR 50, Appendix J, Option B. The current total Appendix J leakage is 51,616.35 cc/min, therefore the total air leakage through the pipe defect must be limited to no more than 77,677.65 cc/min. A calculation based on a more limiting value of 70,000 cc/min of containment air leakage through the defect at 47 psig containment pressure and 7.41 psig of pipe internal pressure during LOCA conditions was determined to be equivalent to approximately 0.024 gpm of SW leakage out of the defect under current operating pressure (approximately 14.47 psig pipe internal pressure). Since TS 3.6.1 requires the containment to be operable in Modes 1-4 and TS SR 3.6.1.1 requires leakage rate requirements comply with 10 CFR 50, Appendix J, Option B and the estimated pipe flaw leak rate (2 gpm) exceeded the calculated equivalent containment air outleakage, TS 3.6.1 was not met. The application of an engineered clamp on the pipe flaw affected pipe section is structurally acceptable while degraded and the pipe considered operable.

The condition was also a safety system functional failure reportable under 10 CFR 50.73(a)(2)(v). The licensee shall report any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to (C) Control the release of radioactive material. This condition meets the reporting criteria because TS 3.6.1 Containment Operability was not met. Because the FCUs are utilized to maintain normal containment temperature within accident analysis input limits and for accident mitigation, SW flow is not isolated following a Design Basis Accident (DBA). SW piping supply and discharge for the FCUs is considered to be a closed system in containment or an extension of the containment boundary. Defects discovered within this piping may adversely affect containment integrity, and the ability to control release of radioactive materials. As a result of revised reportability guidance provided by Entergy fleet experience, the inoperability of single train systems (e.g., containment barrier) are to be considered as a safety system functional failure (SSFF). This event was not at the time considered a SSFF therefore an event notification under 10CFR50.72(b)(3)(v) was not provided.

PAST SIMILAR EVENTS

A review was performed of the past three years of Licensee Event Reports (LERs) for events reporting a TS violation due to inoperable SW piping caused by leaks and one LER was identified. This LER is a result of an extent of condition review. LER-2013-004 reported pin hole leaks in Code Class 3 SW piping elbows for series 300 stainless steel. The pin hole leaks were due to pitting corrosion. The cause of the event reported in LER-2013-004 was not the same as this event as the piping material was different (copper-nickel vs stainless steel), was in an elbow and the cause was failure to follow procedure EN-DC-336 (Plant Health Committee).

SAFETY SIGNIFICANCE

This event had no effect on the health and safety of the public. There were no actual safety consequences for the event because there were no accidents or events during the degraded condition.

There were no significant potential safety consequences of this event. The leakage from the affected SW pipe was within the capability of the SW system to provide adequate SW flow to SW loads. The degraded piping was on the discharge of the FCU motor therefore any failure would not prevent the SW cooling function. Current analysis for SW pipe failures are postulated to be limited to small through-wall leakage flaws as SW is defined as a moderate energy fluid system. The SW leak would eventually drain to the containment sump. The containment sumps have pumps with sufficient capacity to remove excessive leakage.

The impact of the pipe flaw was evaluated for containment free volume in-leakage per the limits of TRM 3.4.D. The leak did not and is not expected to exceed the TRM 3.4.D limit. The pipe leak was just upstream of outboard containment isolation valve SWN-71-4A.

SW effluent is monitored by radiation monitors R-46 and R-56 prior to discharge. If radiation is detected, each FCU heat exchanger can be individually sampled to determine the leaking unit. The SW for the 24 FCU can be isolated to prevent radioactive effluent releases. The Containment Spray System and Containment Fan Cooler System are Engineered Safety Feature systems designed to ensure that the heat removal capability required during the post-accident period can be attained. The CSS and the Containment FCU System provide redundant methods to limit and maintain post accident conditions to less than the containment design value. Five FCUs alone, or 3 FCUs and 1 CSP, or no FCUs and 2 CSPs possess this capability. The configuration with one CS train and two FCU trains is the configuration available following the loss of any safeguards power train (e.g., diesel failure). Accident analysis assumptions regarding containment air cooling and iodine removal are met by one CS train and any two FCU trains (i.e., at least three FCUs). The Containment FCU System consisting of five 20 percent capacity FCUs and the CSS consisting of two 50% trains are divided into trains based on the safeguards power train which supports them. During the period of time the FCU SW pipe leak was being addressed there was minimum safeguards capability available. FCU 24 is associated with Fan Cooler Train 2A/3A which consists of the 24 FCU and the 23 FCU. An evaluation of the leak concluded that it did not result in any structural, flooding, or spraying condition that would adversely impact the capability of SSCs to perform their safety function.