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NL-17-008

January 26, 2017

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop O-P1-17
Washington, D.C. 20555-0001

SUBJECT: Licensee Event Report # 2016-001-01 "Safety System Functional Failure Due to an Inoperable Containment Caused by a Flaw on the 31 Fan Cooler Unit Service Water Return Coil Line Affecting Containment Integrity"
Indian Point Unit No. 3
Docket No. 50-286
DPR-64

Reference 1. Licensee Event Report # 2016-001-00, letter NL-16-139, dated December 21, 2017

Dear Sir or Madam:

Pursuant to 10 CFR 50.73(a)(1), Entergy Nuclear Operations Inc. (ENO) hereby provides Licensee Event Report (LER) 2016-001-01. The attached LER is a revision to an LER submitted by Reference 1. Subsequent to submittal it was identified that the transmittal letter date was incorrect as was Block 7 of NRC Form 366. The date that should have been identified was December 21, 2016. The error was recorded in the CAP as CR-IP2-2017-00172.

IE22
NRR

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

Sincerely,

AJV/cbr

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 (11-2015)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Indian Point 3	05000-286	1 OF 7

4. TITLE: Safety System Functional Failure Due to an Inoperable Containment Caused by a Flaw on the 31 Fan Cooler Unit Service Water Return Coil Line Affecting Containment Integrity

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
11	03	2016	2016	- 001	- 01	1	26	2017	FACILITY NAME	DOCKET NUMBER

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)(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)
☐ 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)
☐ 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 NUMER *(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	Yes					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☒ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☐ NO		8	4	2017

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

On November 3, 2016, as a result of a containment sump pump alarm, operations obtained from Chemistry a sample which indicated a Service Water (SW) leak due to abnormal chlorides levels. Technical Specification (TS) 3.6.1 (Containment) was entered and containment declared inoperable. Inspections identified a through wall leak on the 31 SW Fan Cooler Unit (FCU) from FCU coil 3 which feeds a SW return line header. TS 3.6.6 (Containment Spay and Fan Cooler System) was entered when the 31 FCU was secured and SW to the 31 FCU was isolated. TS 3.6.1, Condition A was exited after the 31 FCU was secured and SW was isolated to the 31 FCU. The leak is at a 3 inch butt-welded joint that is ISI Class 3, nuclear safety related. Leak rate estimate was 0.16 gpm. The direct cause was a leak in a SW pipe due to a through-wall flaw as a result of corrosion. The root cause is indeterminate. The specific cause for the pipe joint defect requires the component to be removed and a metallurgical failure analysis performed. Corrective actions included installation of a leak limiting clamp. The clamp is being monitored daily and UT monitoring will be performed every 90 days until the pipe is repaired. The pipe will be replaced in the next refueling outage in 2017. The affected pipe will be analyzed after removal. The event had no effect on public health and safety.

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NARRATIVE

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

DESCRIPTION OF EVENT

On November 3, 2016, while at 100 percent reactor power, operations received a "Vapor Containment (VC) Sump Pump Running," alarm at approximately 00:27 hours, following a VC sump pump out. In accordance with actions of 3-ARP-009, a check of the Unit Log identified that the last sump pump out was on October 30, 2016. Receipt of this alarm was earlier than expected and a possible indicator of a leak. VC radiation monitors R-11 and R-12 {IL}, VC Humidity {IJ}, and Fan Cooler Unit (FCU) Weir Levels {IJ} were normal. As a result of the early VC Sump Pump Running alarm {FQA}, operations requested Chemistry to obtain a sample of the VC pump out line. Results of the sample showed approximately 149 ppm chlorides, indicating a possible Service Water (SW) {BI} leak due to abnormal chloride levels. Operations entered Technical Specification (TS) 3.6.1 (Containment), Condition A (Containment Inoperable) at 03:00 hours, due to the possibility of a loss of containment {NH} integrity. At 3:19 hours, the 31 FCU {FCU} was secured due to suspected SW FCU coil leakage and entered TS 3.6.6 (Containment Spray System and Containment Fan Cooler System), Condition C (One Containment FCU Train Inoperable). At 3:19 hours, a Safety Function Determination was performed which concluded there had been a loss of safety function and VC became inoperable when indications of a possible SW leak was identified for the 31 FCU. At 3:44 hours, TS 3.6.1, Condition A was exited after the 31 FCU was secured and SW was isolated to the 31 FCU. The leak was recorded in Indian Point Energy Center (IPEC) corrective action program (CAP) as CR-IP3-2016-03607. An 8 hour non-emergency event notification (#52344) was made under 10 CFR 50.72(b)(3)(v) for a loss of safety function.

The SW System (SWS) {BI} 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 to include pump strainers, 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.

A VC entry was performed and inspections identified leak indications at the 31 FCU on the 3rd coil feeding the SW return line {PSP}. To confirm the specific leak location, scaffolding was erected and insulation was removed. A through wall leak was identified on the 31 SW Fan Cooler Unit (FCU) at weld B-297, in branch line C from FCU coil 3. This is one of the 3 inch SW return branch lines from the 31 FCU cooling coils which feeds a 10 inch SW return line header 12b upstream of containment penetration Mb. Line 12b is the SW system return piping from the 31 FCU back to the river.

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The piping for the 3 inch cooling coil return line is 904L stainless steel (SS) with a nominal pipe wall thickness for Schedule 40 pipe (0.216 inches). Leak rate estimate was 0.16 gpm. The leak is at a 3 inch butt-welded joint between a 904L stainless steel elbow {PSF} and pipe {PSP} located on approximately the 76 foot elevation in containment. The piping leak is in a moderate energy ASME ISI Code Class 3, nuclear safety related piping system. 904L SS material is susceptible to the development of corrosion pits. Pin hole leaks and weld defects in this piping have previously occurred and have been evaluated. The evaluation concluded the 904L piping does not have a general corrosion problem. Current analysis for SW pipe failures are postulated to be limited to small through-wall leakage flaws as opposed to guillotine breaks. There is no evidence of leakage at any other location on this weld or elsewhere on the piping adjacent to it.

Code Class 3 piping systems are addressed in ASME Code Case N-513-3. This Code Case provides the requirements for demonstrating structural integrity and therefore operability of a flawed pipe section. Characterization of the weld condition was performed by conducting an ultrasonic examination (UT) on November 4, 2016. The weld was examined circumferentially in a ½ inch by 1 inch grid pattern and by using a bulls-eye grid pattern in ¼ inch increments. These UT, (Non Destructive Examinations) NDE results were documented in an NDE report. However, due to physical obstructions presented by the FCU enclosure, two circumferential grid rows on the backside of the weld could not be reached. Due to the leak location on the bottom side of the pipe on the side opposite from the FCU enclosure, the bulls-eye grid was not obstructed, and all required readings were taken. As a result of limited access, the UT examination of weld B297 resulted in completion of only approximately 70 percent of the circumference of the weld. The remaining 30 percent (approximately 3 inches) was unable to be inspected due to space constraints between the weld and the adjacent FCU plenum wall. ASME Code Case N-513-3 requires the flaw geometry to be characterized by volumetric inspection methods or by physical measurement. It mandates that the full pipe circumference at the flaw location be inspected to characterize the length and depth of all flaws in the pipe section. Since a full volumetric examination could not be completed, the Code Case requirements could not be met. An immediate on-line weld repair of the defect was not considered feasible due to restrictions preventing 360 degree access, time required for work prep, and the potential for excessive sump filtration loading. As such, an NRC Relief Request to deviate from the 10CFR50.55a ASME Code requirements, specifically full compliance with ASME CC-N-513-3 was required. Therefore, pursuant to 10CFR50.55a(z)(2) Entergy requested relief by letter NL-16-133 (Request IP3-ISI-RR-10, Alternative to the Full Circumferential Inspection Requirement of Code Case N-513-3), dated November 7, 2016. The NRC approved the relief request which concluded the inability to obtain full circumference readings does not adversely impact the ability to fully characterize the weld condition vs the code case requirements.

NRC FORM 366A (11-2015) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2018
LICENSEE EVENT REPORT (LER) CONTINUATION SHEET	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 FOIA, Privacy and Information Collections 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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The approval of the relief request allowed Entergy to re-establish SW to the FCU to verify that leakage limits are met using a qualified clamp over the pinhole leak. The leakage will be inspected daily in accordance with the Code Case requirements. SW piping to the 31 FCU must be isolated when an allowable leakage is exceeded. The qualified clamp is an engineered clam-shell type clamp comprised on an outer metal jacket and rubber gasket. The clamp is classified as a temporary modification. The clamp will withstand post-accident containment pressure, temperature and environment, is located away from potential missiles and pipe whip effects, is dedicated for safety-related use, and is rigidly attached to Seismic Category 1 piping. The clamp over the defect will return the system to its original containment integrity configuration and allow the 31 FCU to remain operable.

The defect in the SW return pipe from the 31 FCU was evaluated with respect to TS 5.5.15 (Containment Leak Rate Testing Program), and the Appendix J Leakage program. TS 5.5.15 requires that the SW in-leakage into containment must be limited to less than 0.36 gpm per FCU when pressurized equal to or greater than 1.1Pa. This limit protects the internal recirculation pumps from flooding during the 12 month period of post-accident recirculation. TS 5.5.15 also implements the leakage rate testing of the containment as required by 10CFR50, Appendix J. The maximum leakage to assure that the post-accident containment leakage remains within allowable limits is 0.023 gpm. This limit is based upon an evaluation to calculate the amount of SW which can leak through this pinhole under normal system operating conditions to ensure that 10CFR50 appendix J containment leakage limits are not exceeded under any mode of operation including accident conditions. The leak does not impinge upon any safety related equipment. As a result, no damage from the leakage is expected to occur.

Based on UT measured readings from the NDE Report, a new calculation was generated (IP-CALC-16-00079 FCU 31 Leak) per the ASME CC-N-513-3 requirements. The minimum required thickness for the elbow containing the weld B297 is 0.073 inches. The minimum measured thickness was 0.117 inches. The maximum allowable axial flaw size is 4.11 inches and the maximum allowable circumferential flaw size is 3.65 inches. The existing flaw is characterized as approximately 0.50 inches by 0.50 inches, and the uninspected arc length (approximately 3 inches) of the pipe circumference is less than the allowable circumferential flaw length. Therefore, if the entirety of the uninspected portion of the pipe were to be considered a flaw, the pipe would still retain its structural integrity as evaluated in the new calculation. The pinhole flaw is opposite the uninspected portion and the flaw sizes of the two areas are independent and not additive. Based on this information, the pipe is structurally adequate for service consistent with the requirements of ASME Code Case N-513-3. The remaining service life was calculated to be 3.3 years, which is beyond the next scheduled refueling outage in the spring 2017 when a permanent repair will be made.

An extent of condition review determined the Code Case requires five similar and susceptible locations in the SW system to be volumetrically examined. NDEs were performed on November 5, 2016, on five 31 FCU SW pipe welds and recorded in NDE reports. All five weld locations were found to be structurally acceptable and documented in CR-IP3-2016-03607, corrective action (CA-6). The additional inspections confirm the integrity of the SW piping inspected since all UT data measurements were above the 87.5 percent of pipe nominal wall thickness. Also, there was no evidence of additional leakage at any other place in the 31 FCU 3 inch return line or at any other location in the other four FCU return lines. Unit 2 does not apply as it does not have similar 904L SS FCU supply or return lines.

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CAUSE OF EVENT

The direct cause was a SW leak associated with the 31 FCU at weld B297 in branch line C from coil 3 feeding the 10 inch SW return line 12b. The leak was from a through-wall pinhole flaw at a butt-welded joint between a 904L SS elbow and pipe in containment. The likely degradation mechanism leading to the leak was corrosion. The pipe with the flaw resulted in containment out leakage in excess of 10CFR50, Appendix J limits. The root cause is indeterminate. The specific cause for the pipe joint defect requires the component to be removed and a metallurgical failure analysis performed.

CORRECTIVE ACTIONS

The following corrective actions have been or will be performed under the Corrective Action Program (CAP) to address the causes of this event:

- A leak-limiting engineered clam-shell type clamp was applied to the pipe flaw.
- The clamp is being monitored daily by a special operator log for any signs of increased leakage. The maximum allowed leakage rate past the clamp is 0.023 gpm.
- UT monitoring will be performed every 90 days until the pipe is repaired.
- The pipe/elbow will be replaced in the next refueling outage (RO) in the spring 2017.
- The removed pipe/elbow will be inspected and a metallurgical analysis performed by an independent vendor to determine the specific cause.
- A volumetric inspection of a sample of 3 inch 904L SS butt-welds at the 32, 33, 34, and 35 FCUs will be performed in the spring 2017 RO.
- The Generic Letter 89-13 Program will be revised to include a requirement to conduct a definitive number of 904L weld volumetric inspections each pre-outage interval.
- This LER will be updated after engineering review of the metallurgical analysis and revision as necessary of the cause analysis.

EVENT ANALYSIS

The event is reportable under 10 CFR 50.73(a)(2)(v)(C). 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.

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The pipe flaw leakage was approximately 0.16 gpm which was greater than the calculated 10 CFR 50, Appendix J allowable leak rate of 0.023 gpm. TS 3.6.1 (Containment) requires the containment to be operable in Modes 1-4. TS Surveillance Requirement (SR) 3.6.1.1 requires visual examinations and leakage rate testing in accordance with the containment Leakage Rate Testing Program specified in TS 5.5.15. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B. As SW is required in an accident, the SW to the FCU would not be isolated in DBA and the piping credited as a closed system inside containment for containment integrity.

Consequently, defects discovered within the FCU SW piping may adversely affect containment integrity and the ability to control releases of radioactive material. The condition also meets the reporting criteria of 10 CFR 50.73(a)(2)(i)(B). The licensee shall report any operation or condition which was prohibited by the plant's TS. During the previous period of operation for an unknown period of time the SW pipe contained a through wall leak that did not meet code requirements. This previously unrecognized condition required entry into TS 3.7.9 and corrective actions implemented to return the pipe to operable. Failure to comply with the TS LCO and perform required actions is a TS prohibited condition.

PAST SIMILAR EVENTS

A review of the past three years of Licensee Event Reports (LERs) for events that involved containment integrity due to flawed piping credited as a closed system inside containment. No applicable LERs were identified. There was one LER, LER-2014-002 reporting a Technical Specification prohibited condition for a flaw discovered on a SW pipe connected to the Component Cooling Water Heat Exchanger. This LER is not similar as the impacted piping is outside containment and not credited as a closed system for containment integrity.

SAFETY SIGNIFICANCE

This condition 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 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 and instrumentation to alert operators to a degraded condition.

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The containment consists of the concrete reactor building, its steel liner, and the penetrations through the structure. The containment building is designed to contain radioactive material that might be released from the reactor following a design basis accident (DBA). The containment building steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment operable limits the leakage of fission product radioactivity from the containment to the environment. The DBA analysis assumes that the containment is operable such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage.

The containment was designed with an allowable leakage rate of 0.1 percent of containment air weight per day. Containment isolation valves form a part of the containment pressure boundary. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analysis. One of these barriers may be a closed system such as the SW piping for the FCUs. The only time containment integrity can be affected is post accident when the FCUs safety function is being performed and SW pressure for the FCU cooling piping and coils fall below peak accident pressure. Mitigation of radiation release by the degraded SW pipe pathway can be by use of radiation monitors R-16A and R-16B which monitor containment fan cooling water for radiation indicative of a leak from the containment atmosphere into the cooling water. If radiation is detected, each FCU heat exchanger can be individually sampled to determine the leaking unit. The SW for the 31 FCU can be isolated to prevent radioactive effluent releases. During the time the FCU SW piping was degraded there was no leakage out of containment.

A risk assessment was performed to determine the overall probability of a core damage event which could cause a loss of containment integrity due to a SW to FCU leak assuming it would take 5 days to detect a SW leak to a FCU. The risk result was 7.8E-8 which is considered negligible in terms of both core damage and large early release.