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Anthony J Vitale
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NL-18-039

May 21, 2018

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

SUBJECT: Licensee Event Report # 2018-001-00, "Penetration Indications Discovered During Reactor Pressure Vessel Head Inspection"
Indian Point Unit No. 2
Docket No. 50-247
DPR-26

Dear Sir or Madam:

Pursuant to 10 CFR 50.73(a)(1), Entergy Nuclear Operations Inc. (ENO) hereby provides Licensee Event Report (LER) 2018-001-00. The enclosed LER identifies a degraded condition discovered during the reactor pressure vessel (RPV) head inspection conducted during the recently completed Indian Point Unit 2 Spring 2018 refueling outage (2R23). The degraded condition resulted from RPV head penetration indications that could not be dispositioned as acceptable per the requirements of 10 CFR 50.55a. This condition is reportable under 10 CFR 50.73(a)(2)(ii)(A) as a condition that resulted in the nuclear power plant, including its principal safety barriers, being degraded. In addition, one of the indications was found to have a through-weld component, with evidence of reactor coolant leakage. Because Technical Specification 3.4.13 (Reactor Coolant System (RCS) Operational Leakage) does not allow pressure boundary leakage, the condition is also reportable under 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by the plant's Technical Specifications. The described condition was recorded in the Entergy Corrective Action Program as Condition Reports CR-IP2-2018-01944, 01985, 02071, and 02122.

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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,

For A Vitale

AJV/cdm

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

NRC FORM 366 (02-2018) 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
Penetration Indications Discovered During Reactor Pressure Vessel Head Inspection

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
03	31	2018	2018	- 001	- 00	05	21	2018	Facility Name	Docket Number
										05000

9. Operating Mode	11. This Report is Submitted Pursuant to the Requirements of 10 CFR §: *(Check all that apply)*			
6	☐ 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)
0	☐ 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)(ii)
	☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(iii)
		☐ 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)
Nelson Azevedo, Supervisor, Engineering	(914) 254-6775

13. Complete One Line for each Component Failure Described in this Report

Cause	System	Component	Manufacturer	Reportable To ICES	Cause	System	Component	Manufacturer	Reportable To ICES
B	AB	RPV	W120	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 14 single-spaced typewritten lines)

On March 28, 2018, during the Indian Point Unit 2 Spring 2018 refueling outage (2R23), a white substance, indicative of reactor coolant leakage, was observed on the annulus region of reactor pressure vessel (RPV) head Penetration No. 3 while performing the visual testing (VT)-2 bare metal examination of the RPV head as required by American Society of Mechanical Engineering (ASME) Code Case N-729-4, as amended by the Nuclear Regulatory Commission (NRC) in 10 CFR 50.55a(g)(6)(ii)(D). No base material wastage was identified.

On March 31, 2018, liquid penetrant testing and eddy current testing of the partial penetration J-groove weld of RPV head Penetration No. 3 confirmed the presence of two relevant indications on the inside wetted surface of the J-groove weld. Both identified indications were determined to have lengths of approximately 0.085 inches. One of the two indications was characterized as being axially oriented with a measurable through-weld component, and it was concluded that this was the source of the reactor coolant leakage. The cause of the Penetration No. 3 indications was attributed to Primary Water Stress Corrosion Cracking (PWSCC) of the original Alloy 600 (Alloy 82/182) partial penetration J-groove weld. Penetration No. 3 was repaired in accordance with the NRC approved Westinghouse embedded flaw weld overlay method (WCAP-15987-P, Revision 2-P-A) and relief request IP2-ISI-RR-06.

This event had no effect on the public health and safety. The event was reported to the NRC on March 31, 2018 under 10 CFR 50.72(b)(3)(ii)(A) as a condition of the nuclear power plant, including its principal safety barriers, being degraded.

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NARRATIVE

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

DESCRIPTION OF EVENT

On March 28, 2018, during the Indian Point Unit 2 (IP2) Spring 2018 refueling outage (2R23), a white substance was observed on the annulus region of reactor pressure vessel (RPV) {AB, RPV} head Penetration {PEN} No. 3 while performing the visual testing (VT)-2 bare metal examination of the RPV head as required by American Society of Mechanical Engineering (ASME) Code Case N-729-4, as amended by the Nuclear Regulatory Commission (NRC) in 10 CFR 50.55a(g)(6)(ii)(D). The VT-2 visual examination did not identify any other relevant indications or base material wastage. Subsequent gamma spectroscopy testing of a sample of the white substance detected the presence of Cesium (Cs)-137, Cobalt (Co)-58, and Co-60, which is indicative of reactor coolant {AB} leakage, and that the leakage had commenced within the past 90 to 180 days.

On March 29, 2018, ultrasonic testing (UT) and eddy current testing (ET) surface examinations of the control rod drive mechanism (CRDM) {AA} nozzle {NZL} of RPV head Penetration No. 3 were performed. UT is a volumetric technique used for examination of the CRDM tube, while ET is a technique designed to identify indications open to the wetted surface of the component. The UT and ET examinations were performed from the inside surface of the Alloy 600 penetration tube. Although no indications in the tube were identified, the UT leak path assessment did provide evidence of a change from previous outage examinations.

On March 31, 2018, follow up liquid penetrant testing (PT) and ET surface examinations of the partial penetration J-groove weld of RPV head Penetration No. 3 confirmed the presence of two relevant indications on the inside wetted surface of the J-groove weld. Both identified indications were determined to have linear components with estimated lengths of approximately 0.085 inches. One of the two indications was characterized as being axially oriented with a measurable through-weld component, and it was concluded that this was the source of the reactor coolant leakage previously identified in the annulus region of Penetration No. 3.

The two indications described above were determined to not meet the requirements of ASME Code N-729-4 due to evidence of reactor coolant leakage identified coincident with surface-connected flaws. ASME Code Case N-729-4 and ASME Boiler and Pressure Vessel Code Section XI do not allow through-wall flaws in pressure boundary material in Class 1 (i.e., reactor coolant pressure boundary) components. Indications that cannot be dispositioned as acceptable per ASME Code Section XI in a Reactor Coolant System (RCS) pressure boundary are reportable under 10 CFR 50.72(b)(3)(ii)(A) and 10 CFR 50.73(a)(2)(ii)(A) as a condition of the nuclear power plant, including its principal safety barriers, being degraded. The Penetration No. 3 examination that could not be dispositioned as acceptable per the requirements of 10 CFR 50.55(a) was reported to the NRC under 10 CFR 50.72(b)(3)(ii)(A) as required on March 31, 2018 (Event Number: 53305).

Primary Water Stress Corrosion Cracking (PWSCC) occurs in pressurized water reactors when a susceptible material is exposed to the borated primary water environment, and when in the presence of residual tensile stresses caused by welding. The IP2 RPV head Penetration No. 3 Alloy 600 (Alloy 82/182) J-groove weld in which the indications were discovered meets these criteria. The cause of the Penetration No. 3 indications was attributed to PWSCC of the original Alloy 600 partial penetration J-groove weld. An extent of condition analysis was performed, which identified the IP2 and Indian Point Unit 3 (IP3) locations with Alloy 600 materials susceptible to PWSCC. Since compensatory actions in the form of increased inspection frequencies as specified in 10 CFR 50.55(a) have already been implemented, no additional actions were deemed necessary.

The RPV heads for both IP2 and IP3 were fabricated by Combustion Engineering {C490} under a Westinghouse {W120} Nuclear Steam Supply System (NSSS) design specification. Because the indications in the IP2 RPV head

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Penetration No. 3 Alloy 600 J-groove weld could not be accepted by analytical evaluation, a weld repair of the J-groove weld was necessary. All of the RPV head penetrations were VT-2 and UT inspected during the IP2 Spring 2018 refueling outage (2R23) in accordance with ASME Code Case N-729-4, as amended by 10 CFR 50.55a(g)(6)(ii)(D), and only Penetration No. 3 was found to have indications requiring repair.

Pursuant to 10 CFR 50.55a(z)(1), Entergy Nuclear Operations, Inc. (Entergy) submitted relief request IP2-RR-ISI-06, which proposed to repair the J-groove weld of RPV head Penetration No. 3 using the Westinghouse embedded flaw repair process as an alternative to the defect removal and weld repair provisions of ASME Section XI, IWA-4000 and ASME Section III, NB-4450. The proposed embedded flaw repair process is described in WCAP-15987-P, Revision 2-P-A, and has been approved by the NRC. The embedded flaw repair methodology has been successfully employed multiple times for repairs of indications on the J-groove weld or the outside surface of penetration nozzles. Entergy received NRC verbal authorization on April 9, 2018 (ML18099A373) for relief request IP2-RR-ISI-06, allowing use of the embedded flaw repair method for the period of the relief request (i.e., IP2 Cycle 24 that will end in Spring 2020). The Penetration No. 3 weld repair utilizing the embedded flaw repair process involved sealing the entire Alloy 600 J-groove weld and a portion of the Alloy 600 penetration tube with an Alloy 690 (Alloy 52/52M) weld metal overlay. The weld repair was completed prior to IP2 entering Mode 5 (Cold Shutdown).

CAUSE OF EVENT

The cause of the identified RPV head Penetration No. 3 indications was attributed to PWSCC of the original Alloy 600 (Alloy 82/182) partial penetration J-groove weld. This is a known failure mechanism that is addressed by the requirements of 10 CFR 50.55a(g)(6)(ii)(D). The repair of Penetration No. 3 utilized the Westinghouse WCAP-15987-P, Revision 2-P-A, embedded flaw weld overlay process to encapsulate the indication with an Alloy 690 (Alloy 52/52M) PWSCC resistant material. A weld overlay of PWSCC resistant material to isolate the indications from the borated primary water environment arrests any further PWSCC on the penetration. This was the first weld overlay repair performed at either IP2 or IP3 using the alternative embedded flaw weld overlay methodology.

CORRECTIVE ACTIONS

RPV head Penetration No. 3 has been repaired in accordance with the NRC approved Westinghouse embedded flaw weld overlay method (WCAP-15987-P, Revision 2-P-A) and relief request IP2-ISI-RR-06.

EVENT ANALYSIS

The event is reportable under 10 CFR 50.73(a)(2)(ii)(A). The licensee shall report any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being degraded. This condition meets the reporting criteria because the indications identified in the J-groove weld of RPV head Penetration No. 3, which is considered part of the RPS pressure boundary, could not be dispositioned as acceptable under ASME Code Section XI. In addition, one of the indications was found to have a through-weld component, with evidence of reactor coolant leakage. Because Technical Specification 3.4.13 (Reactor Coolant System (RCS) {AB} Operational Leakage) does not allow pressure boundary leakage, the condition is also reportable under 10 CFR 50.73(a)(2)(i)(B) as a condition prohibited by the plant's Technical Specifications.

PAST SIMILAR EVENTS

No previous similar Licensee Event Reports (LERs) were identified for either IP2 or IP3.

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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 the through-weld reactor coolant leakage in the annulus region of RPV head Penetration No. 3 was minor and the timely RPV head inspection and repair of the identified indications prevented significant propagation of the indications. All of the RPV head penetrations were VT-2 and UT inspected during the IP2 Spring 2018 refueling outage (2R23), and only Penetration No. 3 was found to have indications requiring repair. During the period of plant operation with the through-weld reactor coolant leakage, the primary containment function to act as an additional fission product barrier was not affected, the Control Rod Drive System {AA} remained structurally capable of performing its intended safety functions, and IP2 Technical Specification 3.4.13 would have required a plant shutdown in the unlikely event that the reactor coolant leakage had exceeded the operational limit. With the implementation of the NRC approved Westinghouse embedded flaw repair process on Penetration No. 3, the PWSCC was mitigated and no longer a credible crack growth mechanism. The embedded flaw repair is considered bounded by previous analyses and the structural integrity of the RPV closure head and Penetration No. 3 are maintained for one fuel cycle of operation in accordance with the NRC approved relief request (IP2-ISI-RR-06).

In addition, an extensive industry safety assessment of PWSCC in RPV head penetrations concluded that an inspection program of periodic nonvisual, non-destructive examinations at appropriate intervals supplemented by periodic bare metal visual examinations provides adequate protection against safety-significant failures. The safety assessment also demonstrates that the bare metal visual examination of the head top surface performed at appropriate intervals provides assurance against significant wastage of the low-alloy steel head material, even given a leaking penetration nozzle. Based on the industry safety assessment and the results of the IP2 examinations, it is reasonable to conclude that the Indian Point Energy Center inspection program, which is in accordance with ASME Code Case N-729-4, as modified by the additional limitations set forth in 10 CFR 50.55a(g)(6)(ii)(D), provides assurance against any credible PWSCC degradation event that would challenge nuclear safety.