Entergy

Indian Point Energy Center
450 Broadway, GSB
P.O. Box 249
Buchanan, N.Y. 10511-0249
Tel (914) 254-6700

John A. Ventosa
Site Vice President
Administration

NL-14-056

April 23, 2014

U.S. Nuclear Regulatory Commission
Document Control Desk
11545 Rockville Pike, TWFN-2 F1
Rockville, MD 20852-2738

SUBJECT: Licensee Event Report # 2014-002-00, "Technical Specification (TS) Prohibited Condition Due to an Inoperable 23 Steam Generator (SG) Caused by a Through Wall Defect in 23 SG Drain Line Valve MS-68 "
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) 2014-002-00. The attached LER identifies an event where there was a Technical Specification (TS) Prohibited Condition due to an inoperable steam generator required for an operable reactor coolant system loop, which is reportable under 10 CFR 50.73(a)(2)(i)(B). This condition was recorded in the Entergy Corrective Action Program as Condition Report CR-IP2-2014-00975.

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 JV

JAV/cbr

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

IE22
NRR

NRC FORM 366 (9-2007)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES: 10/31/2013

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME:	2. DOCKET NUMBER	3. PAGE
INDIAN POINT 2	05000-247	1 OF 4

4. TITLE: Technical Specification Prohibited Condition Due to an Inoperable 23 Steam Generator (SG) Caused by a Through Wall Defect in 23 SG Drain Line Valve MS-68

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
02	24	2014	2014-	002 -	00	04	23	2014	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 4

10. POWER LEVEL: 0%

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

NAME	TELEPHONE NUMBER (Include Area Code)
Nelson Azevedo	(914) 254-6775

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	AB	V	V139	Y					

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

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

On February 24, 2014, during initial Containment walkdowns after shutdown for a refueling outage, Operations identified a steam leak on 23 steam generator (SG) drain line valve MS-68. Valve MS-68 is a normally closed valve on a one inch drain pipe from the 23 SG shell side to the two inch SG Blowdown piping to the blowdown tank. The steam leak was due to a through wall defect in the valve body. The one inch SG drain pipe and valve MS-68 are safety related, ISI-ASME Code Class 2 High Energy (HE) and Seismic Class 1 components. Assessment of the condition determined the leak could not be isolated and Ultrasonic Testing (UT) could not be performed to determine the extent of the defect. The valve and associated piping are a pressure boundary for the SG and because there is no ASME Code method to evaluate the structural integrity of the through wall defect, Engineering concluded the valve was inoperable. Operations declared the 23 SG inoperable but the plant was in Mode 5 (Cold Shutdown) and one SG inoperable did not impact Technical Specification (TS) 3.4.7 [Reactor Coolant System (RCS) Loops-Mode 5, Loops Filled]. However, the condition had resulted in leaking prior to shutdown and therefore was applicable to TS 3.4.4 [Reactor Coolant System (RCS) Loops-Modes 1 and 2) which requires four operable RCS loops including four operable SGs. The cause of the steam leak was a pin hole in the body of valve MS-68. The pin hole was likely a defect in the original valve casting which over time propagated through the valve wall. Corrective action was removal and replacement of the valve. The event had no effect on public health and safety.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Indian Point Unit 2	05000-247	2014	- 002 -	00	2 OF 4

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

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

DESCRIPTION OF EVENT

On February 24, 2014, during initial Containment walkdowns after shutdown for a refueling outage, Operations identified a steam leak on 23 Steam Generator (SG) {AB} drain line valve MS-68. Valve MS-68 is a normally closed valve on a one inch drain pipe from the 23 SG shell side to a two inch SG Blowdown (SGBD) pipe that is routed to the blowdown tank. The one inch SG drain pipe and valve MS-68 are safety related, ASME Code Class 2 High Energy (HE) and Seismic Class 1 components. Engineering assessment of the condition determined the leak could not be isolated and Ultrasonic Testing (UT) could not be performed to determine the extent of the defect. The valve and associated piping are a pressure boundary for the SG and because there is no ASME Code method to evaluate the structural integrity of the through wall defect, engineering concluded the valve was inoperable. Operations declared the 23 SG inoperable but the plant was in Mode 5 (Cold Shutdown) and one SG inoperable did not impact Technical Specification (TS) 3.4.7 [Reactor Coolant System (RCS) Loops-Mode 5, Loops Filled]. TS 3.4.7 requires one Residual Heat Removal (RHR) loop to be operable and in operation, and either the non-operating RHR loop to be operable or the secondary side water level of at least two SGs to be equal to or greater than 0% narrow range. The condition was recorded in the Indian Point Energy Center corrective action program (CAP) as CR-IP2-2014-00975.

Leakage in Containment although unrecognized from MS-68 was previously identified in July 2013 as an increase in the fill rate of the Containment Sump and activity of the Fan Cooler Unit weirs. This condition was recorded in the IPEC CAP as CR-IP2-2013-03207. Several Containment entries were made to identify the source of the leakage but without success. An assessment of operability concluded there was no inoperable condition. There were no indications of actual RCS leakage and Pressurizer level, Volume Control Tank level, charging pump flow, RCS pressure and Radiation Monitors remained steady and within normal parameters. Containment sump and all sump levels remained within normal parameters. The calculation of RCS leakage maintained operability for TS 3.4.13 (RCS Operational leakage) and sump leakage maintained functionality for Technical Requirements Manual (TRM) 3.4.D (Containment Free Volume Leakage). An ODMI and troubleshooting plan was prepared and implemented.

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process and transfer of this heat, via the SGs, to the secondary plant. The reactor coolant is circulated through four loops connected in parallel to the reactor vessel, each containing a SG, and a reactor coolant pump. The SGs provide the heat sink for the RCS. If the requirements of the TS Limiting Condition for Operation (LCO) are not met, the required action is to reduce power and bring the plant to Mode 3 (Hot Shutdown). This action lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating design limits.

The Main Steam (MS) {SB} and Feedwater (FW) {SJ} lines and the shell side of the SG are considered an extension of the containment boundary (closed system inside containment) (Class 7 Containment Isolation). Valve MS-68 is on a one inch pipe from the SG shell side that connects to the SGBD line which is then routed through a Containment penetration to the Blowdown Flash Tank and Sampling System. Two containment isolation valves in series are provided on the SGBD line as it exits Containment. The SGBD system is primarily used in maintaining the secondary side water chemistry of the SGs but also provides water samples and a means to drain the shell sides for inspection/maintenance. MS-68 is a one inch diameter gate valve {V} manufactured by Vogt {V139}, Model SW-2805.

Cause of Event

The cause of the steam leak was a pin hole in the body of valve MS-68. The pin hole was likely a defect in the original valve casting which over time propagated through the valve wall. Leakage was estimated to be approximately 0.4 gpm at SG secondary side pressure of approximately 750 psig.

Corrective Actions

The following corrective actions have been performed under Entergy's Corrective Action Program to address the cause and prevent recurrence:

- The defective MS-68 valve was removed and replaced with a new valve (WO#375380).

Event Analysis

The event is reportable under 10CFR50.73(a)(2)(i)(B). The licensee shall report any operation or condition which was prohibited by the plant's TS. This condition meets the reporting criteria because during past operation the defect in the valve body of ASME Code Class 2 HE component MS-68 leaked resulting in an increase in the fill rate of the Containment Sump and increased Fan Cooler Unit weir activity as recorded in CR-IP2-2013-03207. As a result of this defect, the valve and drain line can allow steam/feedwater on the SG secondary shell side to leak into containment. The valve and associated piping are a pressure boundary for the SG and because there is no ASME Code method to evaluate the structural integrity of the through wall defect, it was concluded the valve was inoperable. As the leak could not be isolated and Ultrasonic Testing (UT) could not be performed to characterize the defect, Operations declared the 23 SG inoperable. TS 3.4.4 [Reactor Coolant System (RCS) Loops-Modes 1 and 2] requires four RCS loops to be operable and in operation during Modes 1 and 2. An operable RCS loop consists of an operable reactor coolant pump in operation providing forced flow for heat transport and an operable SG. Because the condition was determined to be the cause of leakage in the Containment during past operation at full power and was considered to result in an inoperable 23 SG, the requirements of TS 3.4.4 were not complied with and is therefore a TS prohibited condition.

The condition is not reportable in accordance with 10CFR50.73(a)(2)(ii), Any event or condition that resulted in (A) The condition of the power plant, including its principal safety barriers, being seriously degraded; or (B) the nuclear power plant being in an unanalyzed condition that significantly degraded plant safety. Although there was no ASME Code method to evaluate the structural acceptability of the through wall defect, engineering judgment concluded the condition was structurally intact and did not represent a potential large loss to the SG and would not result in containment leakage in excess of that allowed by 10CFR, Appendix J under shutdown accident conditions. Therefore, there were no principal safety barriers seriously degraded nor was there any condition that significantly degraded plant safety.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Indian Point Unit 2	05000-247	2014	- 002	- 00	4 OF 4

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

Past Similar Events

A review was performed of the past three years for Licensee Event Reports (LERs) reporting a TS prohibited condition due to an inoperable ASME Code Class 2 boundary component. No LERs were identified. LER-2013-004 reported a TS prohibited condition due to an inoperable service water header as a result of a pin hole leak in Code Class 3 Moderate Energy SW piping. LER-2012-003 reported through wall defects in two reactor coolant pressure boundary (RCPB) components: #1) Defect on the horizontal leak-off pipe of Pressure Control Valve PCV-455A Spray Inlet Stop valve 4152, #2) Defect in the socket weld of a 3/8 inch diameter tubing fitting down stream of pressure transmitter isolation valve 4138. Neither of these were Code Class 2 valve body defects.

Safety Significance

There were no actual safety consequences for the event because there were no significant failures in components that are credited as closed systems inside containment for the Containment Isolation System (CIS). The CIS provides the means of isolating the various pipes passing through the containment walls as required to prevent the release of radioactivity to the outside environment in the event of a Design Basis Accident (DBA). A CIS design assumption is a piping rupture outside the containment at the same time as a DBA (i.e., LOCA) is not considered credible, as the penetrating lines are Seismic Class I up to and including the second isolation barrier. The MS, FW and SGBD system have redundant isolation capability to prevent any release of radioactivity to the environment in the event of a DBA.

Leakage through the valve defect was approximately 0.4 gpm at SG secondary side pressure of 750 psig. This amount of leakage would not have prevented the Auxiliary Feedwater System from providing its design cooling capability. TS surveillances provide a trend of leakage early before significant degradation. An early warning of unidentified leakage is provided by the systems that monitor containment atmosphere radioactivity, operation of the containment sump, sump level, Fan Cooler Unit weir condensate, and Containment temperature and dew point. Monitoring and periodic inspections identify leaks when they are small so that repairs can be performed to prevent further degradation.

Although there was no ASME code case or analytical means to evaluate the structural acceptability of the through wall defect, Engineering judgment concluded the condition was structurally intact and did not represent a potential large loss to the SG and would not result in containment leakage in excess of that allowed by 10CFR, Appendix J under shutdown accident conditions.