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March 28, 2001

Re: Indian Point Unit No. 2
Docket No. 50-247
LER 2000-003-01
NL-01-035

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop PI-137
Washington, DC 20555-001

The attached Licensee Event Report Supplement 2000-003-01 is hereby submitted in accordance with the requirements of 10 CFR 50.73.

Sincerely,
A. Alan Blind

Attachment

cc: Mr. Hubert J. Miller
Regional Administrator - Region I
US Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. Patrick D. Milano, Senior Project Manager
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NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(6-1998)

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Steam Generators 21 and 24 Classified as Category C-3 per Tech Spec Table 4.13-1

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
										05000
03	23	2000	2000	-003-	01	03	28	2001	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): N

POWER LEVEL (10): 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)	X	50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)	X	OTHER -
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Richard T. Louie, Nuclear Safety & Licensing	(914) 734-5678

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
x	AB	SG	W351	Y					

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

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

On March 23, 2000, Indian Point Unit 2 steam generator inspection results were classified as Category C-3 in accordance with Technical Specification Table 4.13-1. Indian Point Unit 2 was manually shutdown on February 15, 2000 due to the detection of a primary to secondary leak in 24 steam generator. Eddy current inspections of all active steam generator tubes were subsequently initiated following the plant shutdown. On March 23, with approximately 90 percent of the inspections performed, 21 and 24 steam generators were determined to have more than 1 percent of their tubes inspected defective. Per Technical Specification Table 4.13-1, a steam generator would be classified as Category C-3 if more than 10 percent of the total tubes inspected were degraded, or if more than 1 percent of the tubes inspected were defective. The majority of the indications were located at the support plate intersections and at Row 2 U-bend areas. Several improved inspection techniques were used to assess steam generator tube degradation, including ultrasonic testing and use of high frequency eddy current inspection probes. The inspections have confirmed that the root cause for the tube failure in 24 steam generator was Primary Water Stress Corrosion Cracking (PWSCC) in the apex of the U-bend region of the tube identified as Row 2 Column 5.

This report is being made per 10 CFR 50.73(a)(2)(ii)(A) as a condition that resulted in the nuclear power plant being in an unanalyzed condition that significantly compromises plant safety. Pursuant to 10 CFR 50.72(b)(2)(I), this condition was reported to the NRC on March 23, 2000.

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

PLANT AND SYSTEM IDENTIFICATION

Westinghouse 4-Loop Pressurized Water Reactor

EVENT IDENTIFICATION

Steam Generators 21 and 24 Classified as Category C-3 per Technical Specification Table 4.13-1

EVENT DATE

March 23, 2000

REFERENCES

Condition Reporting System Number: 200002049

PAST SIMILAR EVENTS

None

EVENT DESCRIPTION

On March 23, 2000, Indian Point Unit 2 steam generator inspection results were classified as Category C-3 in accordance with Technical Specification Table 4.13-1 and a 10 CFR 50.72 notification to the NRC was made.

Indian Point Unit 2 was manually shutdown on February 15, 2000 due to the detection of a primary to secondary leak in 24 steam generator. Eddy current inspections of all active steam generator tubes were subsequently initiated following the plant shutdown. On March 23, with approximately 90 percent of the inspections completed, steam generators 21 and 24 were determined to have more than 1 percent of the tubes inspected defective. Per Technical Specification Table 4.13-1, a steam generator would be classified as Category C-3 if more than 10 percent of the total tubes inspected were degraded, or if more than 1 percent of the tubes inspected were defective. The majority of the indications were located at the support plate intersections and at row 2 U-bend areas.

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

On March 31, 2000, a 50.72 follow-up notification regarding the status of in-situ pressure testing on 24 steam generator was made. Although originally reported as a failure of the three-delta pressure requirement, subsequent review of test data concluded that the required performance criteria were met.

All required steam generator inspections were performed during the 2000 refueling outage. Inspection results are provided in the "2000 Refueling Outage Steam Generator Inspection Condition Monitoring and Operational Assessment" reports, which were transmitted to the NRC on June 2, 2000.

Subsequent to the completion of the inspection activities a project to replace the original steam generators was begun and completed by the end of 2000. The plant was returned to service shortly thereafter.

EVENT ANALYSIS

This report is provided in accordance with Technical Specification Table 4.13-1, which requires NRC notification if more than one steam generator is classified as Category C-3. Pursuant to 10 CFR 50.73(a)(2)(ii)(A) the basis of this notification has been determined to be a condition which results in the nuclear power plant being in an unanalyzed condition that significantly compromises plant safety. On March 23, 2000, eddy current inspection results on 21 and 24 steam generators were determined to be Category C-3. In 21 steam generator, a total of 41 indications (40 at the support plate intersections and 1 in the U-bend) were detected. In 24 steam generator, a total of 39 indications (36 at the support plate intersections and 3 in the U-bend) were detected. 100 percent eddy current tube examinations of all active tubes in the original steam generators were conducted.

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

Degradation Assessment

The following active degradation mechanisms were previously identified at Indian Point 2:

1) Denting at TSP (tube support plate) intersections
2) Pitting in the sludge pile region at the TTS (top of tubesheet)
3) VOL (volumetric) indications in the sludge pile region
4) PWSCC in the roll expanded regions
5) PWSCC and ODSCC at the TSP intersections
6) ODSCC in the tube crevice region
7) PWSCC in Row 2 U-bends
8) Tube wear at AVB (anti-vibration bar) intersections

Primary Water Stress Corrosion Cracking (PWSCC) in the Row 2 U-bend region was first observed during the 1997 examinations. The location of the primary to secondary leakage in 24 steam generator was determined to be at the U-bend apex of the Row 2 Column 5 (R2C5) tube. Evaluations have concluded that the root cause of the tube failure was Primary Water Stress Corrosion Cracking (PWSCC). This conclusion was based upon a review of previous and present eddy current test data, industry experience, and evaluation of the maximum tube stresses consistent with inside diameter cracking. The inspections identified additional indications of Outside Diameter Stress Corrosion Cracking (ODSCC) / intergranular attack (IGA) in the tube crevice region. Consequently, the original examination program for this region was expanded.

A contributing factor which led to the occurrence of the R2C5 tube failure was the inability to detect a relatively large discontinuity during the 1997 inspections. This was principally due to the geometry of the low row U-bends, and to the difficulty of detecting this indication utilizing the technology and industry guidelines available at that time. Review of the 1997 data indicates that background noise in the eddy current signal masked the flaw in the R2C5 tube. Masking of the flaw was due to signal distortion (noise) caused by deposits (magnetite, copper) on the tube end, and potentially, ovalization of the tube. A reduction in this noise level was achieved during the 2000 refueling outage inspections with the use of a high frequency 800 kHz Plus Point probe.

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Stress Evaluations

An analysis of the low row U-bends, R2, R3 and R4, was initiated with the objective to evaluate the sensitivity of the tube stresses to pinching of the straight legs resulting from hourglassing of the top tube support plate flowslots. The results of this analysis support the observation that the leak was due to PWSCC at the apex of the tube, since the point of maximum predicted stress is at the apex of the extrados of the tube. The stress results also indicate that Row 3 tubes are expected to be less susceptible to PWSCC due to the larger TSP displacement required to reach the maximum stress condition, their larger radius U-bends, and because of the lower displacement of the tube legs due to TSP hourglassing as compared to the Row 2 tubes.

Inspection Techniques

Several improved inspection techniques were used to assess steam generator tube degradation and to establish corrective actions to reduce the probability of future tube degradation. These techniques included the qualification and use of a high frequency, 800 kHz Plus Point probe to supplement the conventional Plus Point low row U-bend examinations, ultrasonic testing of select sludge pile indications, stress analysis modeling of the upper tube support plate and U-bend area, and the installation of new hillside ports in 21 and 24 steam generators to further evaluate degradation at the sixth tube support plate.

Pluggable Tube Summary - Original Steam Generators

SG	Tubes Plugged
21	190
22	237
23	192
24	172
Total	791

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

EVENT SAFETY SIGNIFICANCE:

This report is being submitted in accordance with Technical Specification Table 4.13-1 which requires NRC notification if more than one steam generator is classified as Category C-3. The actual safety consequences and implications of this required notification are not significant. Based upon the analysis of data collected during the inspections, an assessment of the plant's steam generators, including degradation mechanisms, was performed.

CORRECTIVE ACTION:

In accordance with Indian Point Unit 2 Technical Specification 4.13, steam generator tubes not considered acceptable for continued service shall be plugged or repaired. Comprehensive results of the steam generator examinations including specific repairs are discussed in the "2000 Refueling Outage Steam Generator Inspection Condition Monitoring and Operational Assessment" reports, which were submitted to the NRC staff on June 2, 2000. Subsequent to the completion of the inspection activities a project to replace the original steam generators was begun and completed by the end of 2000. The plant was returned to service shortly thereafter.

On March 22, 2000 a new Station Administrative Order (SAO)-180, "Administrative Steam Generator Program," was approved. This SAO implements Con Edison's commitment to comply with the latest provisions of the nuclear industry initiative described in the Nuclear Energy Institute (NEI) "Steam Generator Program Guidelines 97-06"