Consolidated Edison Company of New York, Inc. Indian Point Station Broadway & Bleakley Avenue Buchanan, NY 10511 Telephone (914) 734-5340 Fax: (914) 734-5718 blinda@coned.com

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, 4. alan Bin

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 Project Directorate I Division of Licensing Project Management US Nuclear Regulatory Commission Mail Stop 0-8-C2 Washington, DC 20555

Senior Resident Inspector US Nuclear Regulatory Commission PO Box 38 Buchanan, NY 10511

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NRC FORM 3 (6-1998)	366 U.S. N	IUCLEAR REC	GULATORY CO	MMIS	SION			APF Estir	PROVE	DB) urden	OMB NO. 31	50-0104 comply w	EXPIRES	06/30/20	001 ormation
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In	dian Poi	nt. Unit	2						0500	1024	17		1	OF	6
TITLE (4)															
Steam Ge	nerators	21 and 2	24 Classi:	fied	l as Cat	egory	7 C-3	pe	r Teo	ch ;	Spec Tabl	e 4.13	8-1		
EVENT D	ATE (5)	LER	NUMBER (6)		REF	PORT D	ATE (7)				OTHER FAC	ILITIES I	NVOLVED	(8)	
				EVISIO N					FACILIT	YNAN	/E		DOCKET NU	MBER	
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Richar	d T. Lou	ie, Nucl	ear Safet	у&	Licensi	ing					(914) 734-	-5678		
			ONE LINE FO	REA	CH COMPC	NENT F	AILURE	E DE	SCRIBI	ED IN	THIS REPOR	IT (13)			
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x	AB	SG W351 Y													
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YES EXPECTED SUBMISSION															
I (If yes, complete EXPECTED SUBMISSION DATE). X NO DATE (15)															
ABSTRACT (Limit to 1400	spaces, i.e., a	pproximately 15	single	e-spaced typ	pewritten	lines) (16)							-
On March	23, 200	0, Indiar	n Point Ur	nit :	2 steam	gene	erator	: ir	ispec	tic	on result:	s were	class	ified	as
Category	C-3 in	accordanc	e with Te	echn	ical Sp	ecifi	catic	n 1	Cable	÷ 4.	.13-1. In	ndian	Point	Unit 2	2
leak in 2	24 steam	generato	repruary	LD, Cur	2000 a rent in	ue to spect	ions	aet	all	.on act	or a pru tive stear	nary t n gene	o seco rator	ndary tubes	
were sub	sequentl	y initiat	ed follow	ving	the pl	ant s	hutdo	own.	. On	n Ma	arch 23, v	with a	pproxi	mately	7 90
percent of	of the i	nspectior	ns perform	ned,	21 and	24 s	team	ger	lerat	ors	s were det	cermin	ed to	have r	more
a steam	generato	r would h	be classif	fied	as Cat	egorv	r C-3	if	more	tica th	an 10 per	rcent	of the	tota)-⊥, 1
tubes in	spected	were degi	aded, or	if	more th	an 1	perce	ent	of t	he	tubes ins	specte	d were		_
defective	e. The : w 2 II-b	majority end areas	of the in Severa	ndica al in	ations [·]	were	locat	ed	at t	he	support p	plate	inters	ection	ns
generato	r tube d	egradatio	on, includ	ling	ultras	onic	testi	.ng	and	use	e of high	frequ	ency e	ddy	am
current :	inspecti	on probes	. The in	nspe	ctions	have	confi	irm	ed tł	hat	the root	cause	e for t	he tu	be
of the U	-bend re	eam gener gion of t	cator was che tube i	den	mary wa tified	ter S as Ro	v 2 C	s Co Colu	orros umn 5	lor	1 Cracking	g (PWS	CC) in	the a	apex
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the nucle	ort is b ear powe	eing made r plant h	e per 10 C being in a	FR in ui	50.73(a nanalvz)(2)(ed co	ıi)(A nditi	.) a .on	is a that	cor si	dition th gnificant	nat re	sulted	in ses	
plant sat NRC on Ma	fety. P arch 23,	ursuant t 2000.	to 10 CFR	50.	72 (b) (2)(I),	this	cc	ondit	ior	i was repo	orted	to the		
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NBC FORM 36	6 (6-1998)							<u> </u>			,				

NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION						
LICENSEE EVENT REPORT (LER) TEXT CONTINUATION						
FACILITY NAME (1)	DOCKET NUMBER (2)	LEF	NUMBER (6)) BEVISI	PAG	E (3)
		YEAR	IAL NUMBER	ON NUMBE	P	
Indian Point, Unit 2	05000247	2000	-003-	01	2 OF	6
TEXT (If more space is required, use additional copies of NRC	C Form 366A) (17	7)				
IDANI AND SISTEM IDENTIFICATION						
Westinghouse 4-Loop Pressurized Wate	er Reactor					
EVENT IDENTIFICATION						
Steam Generators 21 and 24 Classifie Specification Table 4.13-1	ed as Catego:	ry C-3 p	er Techi	nical		
EVENT DATE						
March 23, 2000						
REFERENCES						
Condition Reporting System Number: 2	200002049					
PAST SIMILAR EVENTS						
None						
EVENT DESCRIPTION						
On March 23, 2000, Indian Point Unit were classified as Category C-3 in a Table 4.13-1 and a 10 CFR 50.72 noti	2 steam gen ccordance wi fication to	erator th Tech: the NRC	inspecti nical Sp was mad	on res ecific le.	sults catio	n
Indian Point Unit 2 was manually shu detection of a primary to secondary inspections of all active steam gene following the plant shutdown. On Ma the inspections completed, steam gen more than 1 percent of the tubes ins Specification Table 4.13-1, a steam C-3 if more than 10 percent of the t more than 1 percent of the tubes ins the indications were located at the U-bend areas.	tdown on Feb leak in 24 s rator tubes rch 23, with erators 21 a pected defec generator wo otal tubes i pected were support plat	oruary 1 steam gen were sul approx and 24 we tive. ould be defection e inters	5, 2000 nerator. bsequent imately ere dete Per Tech classifi d were d ve. The sections	due to Eddy ly ini 90 per rmined nical ed as egrade major and a	o the y cur itiat ccent l to l Cate ed, o city o at row	rent ed have gory r if of w 2
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Indian Point, Unit 2DS00247DS00247DS00XT (# 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 ANALYSISThis 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)(i)(A) the basi 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 resul on 21 and 24 steam generators 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.	FACILITY NAME (1)	DOCKET NUMBER (2)			PAGE (3
Indian Point, Unit 2050002472000 -003013 OFXI (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 			YEAR NUMBER	NUMB	
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NRC FORM 366A U.S. NUCLEAR REGULATORY COMMISSION (6-1998) LICENSEE EVENT REPORT (LER)	<u> </u>						
FACILITY NAME (1)	DOCKET			6)	P	AGE (3)	
		YEAR	SEQUENT IAL NUMBER	REVISI ON NUMB			
Indian Point, Unit 2	05000247	2000	-003	01	4	OF	6
TEXT (If more space is required, use additional copies of NRC	C Form 366A) (1	7)					
Degradation Assessment							
The following active degradation mec Indian Point 2:	hanisms were	e previ	ously id	lentifi	.ed	at	
 Denting at TSP (tube support pla 2) Pitting in the sludge pile regic 3) VOL (volumetric) indications in 4) PWSCC in the roll expanded regic 5) PWSCC and ODSCC at the TSP inter 6) ODSCC in the tube crevice region 7) PWSCC in Row 2 U-bends 8) Tube wear at AVB (anti-vibration Primary Water Stress Corrosion Crack was first observed during the 1997 e to secondary leakage in 24 steam gen apex of the Row 2 Column 5 (R2C5) tu root cause of the tube failure was P (PWSCC). This conclusion was based eddy current test data, industry exp tube stresses consistent with inside identified additional indications of Cracking (ODSCC) (intergraphics of the constraints) 	te) intersed on at the TT; the sludge p rs sections bar) inters ing (PWSCC) examinations erator was of be. Evaluat rimary Water upon a revie erience, and diameter cr Outside Dia	ctions S (top pile re section in the determine tions h c Stres ew of p d evalue cacking ameter	of tubes gion s Row 2 U location ned to b ave concos revious ation of . The i Stress C	(-bend of th e at t luded ion Cr and pr the m nspect orrosi	reg e p he ack ese axin ion on	ion rimar U-ben t the ing nt mum s	y Id
Consequently, the original examinati	ack (IGA) ir on program f	the ties the termination of termination	ube crev s region	ice re was e	gio: xpa:	n. nded.	
A contributing factor which led to t failure was the inability to detect during the 1997 inspections. This w of the low row U-bends, and to the d indication utilizing the technology at that time. Review of the 1997 da in the eddy current signal masked th of the flaw was due to signal distor (magnetite corpor) on the tube and	he occurrent a relatively as principal ifficulty of and industry ta indicates e flaw in the tion (noise)	ce of t y large lly due f detec y guide s that ne R2C5 cause	he R2C5 discont to the ting thi lines av backgrou tube. d by dep	tube inuity geomet ailabl nd noi Maskin osits	ry e se g		
the tube. A reduction in this noise refueling outage inspections with th Plus Point probe.	and potenti level was a e use of a h	achieve nigh fr	ovallzat d during equency	10n of the 2 800 kH	000 z		

NRC FORM 366A	U.S. NUCLEAR REGULATORY COMMISSION							
LICEI	NSEE EVENT REPORT (LER) TEXT CONTINUATION							
	FACILITY NAME (1)	DOCKET NUMBER (2)	LE	ER NUMBER (6)	F	AGE (3)
			YEAR	SEQUENT IAL NUMBER	REVISI ON NUMB			
Indian Po	int, Unit 2	05000247	2000	-003	01	5	OF	б
TEXT (If more spa	ace is required, use additional copies of NR	C Form 366A) (1	7)					

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

NŘC FORM 366A (6-1998)

FACILITY NAME (1)	DOCKET NUMBER (2)	LE		(6)	F	AGE (3)		
		YEAR	SEQUENT IAL NUMBER	REVISI ON NUMB					
Indian Point, Unit 2	05000247	2000	-003	01	6	OF	6		
TEXT (If more space is required, use additional copies of NRC	C Form 366A) (17	7)							
EVENT SAFETY SIGNIFICANCE:									
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 steam generator tubes not considered shall be plugged or repaired. Compr generator examinations including spe "2000 Refueling Outage Steam Generat and Operational Assessment" reports, staff on June 2, 2000. Subsequent t activities a project to replace the and completed by the end of 2000. T shortly thereafter.	2 Technica acceptable rehensive re acific repai for Inspecti which were to the compl original st whe plant wa	l Speci for co sults o rs are on Cond submit etion o eam gen s retur	ficatio ontinued of the s discuss dition M tted to of the i nerators cned to	n 4.13, servic ed in t onitori the NRC nspecti was be service	che ing ion egu	n			
On March 22, 2000 a new Station Admi "Administrative Steam Generator Prog implements Con Edison's commitment t of the nuclear industry initiative of Institute (NEI) "Steam Generator Pro	nistrative gram," was a to comply wi described in ogram Guidel	Order pproved th the the Nu ines 97	(SAO)-18 1. This latest uclear E 7-06"	0, SAO provisi nergy	ion	S			

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