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May 22, 2000

Re: Indian Point Unit No. 2
Docket No. 50-247
LER 2000-005-00

Document Control Desk
US Nuclear Regulatory Commission
Mail Station PI-137
Washington, DC 20555-0001

The attached Licensee Event Report 2000-005-00 is hereby submitted in accordance with the requirements of 10 CFR 50.73.

Sincerely,

Attachment

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

Mr. Jefferey Harold, Project Manager
Project Directorate I-1
Division of Reactor Projects I/II
US Nuclear Regulatory Commission
Mail Stop 14B-2
Washington, DC 20555

Senior Resident Inspector
US Nuclear Regulatory Commission
PO Box 38
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RGN-001

NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If 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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TITLE (4)

Steam Generator Primary To Secondary Side Design Pressure Differential Exceeded

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
04	20	2000	2000	-- 005 --	00	05	22	2000	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	N	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)
POWER LEVEL (10)	000		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)		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, Licensing Engineer	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

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 April 20, 2000 Consolidated Edison Company of New York, Inc., the licensee for Indian Point Unit 2 received a letter from Westinghouse Electric Company, the Nuclear Steam Supply System (NSSS) supplier, which stated that Westinghouse was in the process of completing a Nuclear Safety Advisory Letter regarding the primary to secondary design differential pressure in the steam generators [EIIS:SG]. At the time of discovery of this condition, the plant was in a refueling outage. The current design pressure listed in the steam generator equipment specification is 1550 psi. Based upon past operating cycle steam generator dome pressure values of approximately 700 psia, this design pressure would be exceeded. However, Westinghouse has performed an analysis which demonstrates that acceptable stress results may be obtained if a new design pressure differential of 1700 psi is assumed. This condition will be corrected prior to plant restart, by updating the design specification and stress reports in accordance with Section IWA-4312, Rerating, of Section XI of the ASME Code, 1995 Edition with 1996 Addenda, thereby demonstrating compliance with the Code. No equipment failures occurred as a result of this condition.

This report is being made per 10 CFR 50.73(a)(2)(ii)(B) as a condition that resulted in the nuclear power plant being in a condition that was outside the design basis of the plant. Pursuant to 10 CFR 50.72(b)(2)(i), this condition was reported to the NRC on April 20, 2000.

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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 Generator Primary To Secondary Side Design Pressure Differential Exceeded

EVENT DATE:

April 20, 2000

REFERENCES:

Condition Report (CR) No. 200002791

PAST SIMILAR EVENTS:

None

EVENT DESCRIPTION:

On April 20, 2000 Consolidated Edison Company of New York, Inc., the licensee for Indian Point Unit 2 received a letter from Westinghouse Electric Company, the Nuclear Steam Supply System (NSSS) supplier, which stated that Westinghouse was in the process of completing a Nuclear Safety Advisory Letter (NSAL)regarding the primary to secondary design differential pressure in the steam generators [EIIS:SG]. At the time of discovery of this condition, the plant was in a refueling outage. On May 5, 2000, Westinghouse subsequently issued the subject NSAL-00-007 to Consolidated Edison. It has been determined that previous design documentation, prepared to support changes in the full power operating conditions at Indian Point Unit 2, had not included a comparison of the maximum normal operating and / or abnormal condition primary-to-secondary side pressure differential to the steam generator design primary-to-secondary pressure differential. The current design pressure listed in the steam generator equipment specification is 1550 psi. Based upon past operating cycle steam generator dome pressure values of approximately 700 psia, this design pressure would be exceeded. However, Westinghouse has performed an analysis which demonstrates that acceptable stress results may be obtained if a new design pressure differential of 1700 psi is assumed. This condition will be corrected prior to plant restart, by updating the applicable design specification, stress calculations, and stress reports in accordance with Section IWA-4312, Rerating, of Section XI of the ASME Code, 1995 Edition with 1996 Addenda, thereby demonstrating compliance with the Code.

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

EVENT ANALYSIS:

On May 5, 2000, Westinghouse Electric Company issued NSAL-00-007 entitled "Steam Generator Primary-to-Secondary Pressure Differential," informing all plants with Westinghouse steam generators that it has determined that previous design documentation prepared to support changes in the full power operating conditions for some plants, had not included a comparison of the maximum normal operating and/or upset condition primary-to-secondary side pressure differential to the steam generator design primary-to-secondary pressure differential. As a consequence, design primary-to-secondary side pressure differential requirements identified in applicable specifications were exceeded, as a result of changes in operating parameters associated with plant up-ratings or re-ratings, and tube plugging. Plant specific evaluations and corrective actions were subsequently provided to affected utilities via separate correspondence.

The Indian Point Unit 2 steam generators were originally constructed in accordance with the requirements of ASME Section III, 1965 Edition with Summer 1966 Addenda. Paragraph N-441 of the 1965 Code requires that "the specified design pressure shall not be less than the maximum difference in pressure between the inside and outside of the item, or between any two chambers of a combination unit, which exists under the specified operating conditions." No distinction between normal/abnormal conditions were provided in the 1965 Code. In later editions of the Code, specific requirements relative to the design pressure limitations for "abnormal loads" were identified. However those requirements were not applicable during the original design of the Indian Point Unit 2 steam generators.

The primary to secondary side design pressure differential for the Indian Point Unit 2 steam generators (Model 44 Series) is specified as 1550 psi. Based upon past operating cycle steam generator dome pressure values of approximately 700 psia, this design pressure would be exceeded. However, Westinghouse has performed an analysis which demonstrates that acceptable stress results may be obtained if a new design pressure differential of 1700 psi is assumed.

In accordance with Section IWA-4312, Rerating, of Section XI of the ASME Code, 1995 Edition with 1996 Addenda, this condition will be corrected by updating the applicable design specification, stress calculations, and stress reports. Per 10 CFR 50.55a(b)(2), this edition of the Code is acceptable for use. In accordance with IWA-4120(c) of Section XI of the ASME Code, 1989 Edition (the applicable Edition for the current IP-2 inspection interval), "later Editions and Addenda of Section XI... may be used... provided these Editions and Addenda at the time of the planned repair have been incorporated by reference in amended regulations of the regulatory authority having jurisdiction at the plant site."

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

EVENT SAFETY SIGNIFICANCE:

Evaluations have been conducted for the steam generator structural components (tubesheet, tubes, and plugs) that would be subjected to increases in the primary-to-secondary pressure differential resulting from the transients defined in the applicable design specification. Based on these evaluations, it concluded that steam generator structural integrity will be maintained if the design pressure is exceeded. Therefore, this condition does not represent a Substantial Safety Hazard pursuant to the requirements of 10 CFR 21.21(a), nor does it result in a violation of any Technical Specification safety limits.

CORRECTIVE ACTION:

To ensure compliance with ASME Code requirements, this condition will be corrected by a revision to the design basis of the Westinghouse Model 44 steam generators. The applicable steam generator design specification, stress calculations, and stress report will be revised to reflect a design primary to secondary side pressure differential of 1700 psi. The stress calculations demonstrate that the affected steam generator components satisfy the stress limits of the ASME Code. In conjunction with the revised design specification, Westinghouse determined that certain assumptions relative to limiting steam generator design transients were inconsistent with actual plant configuration and operation. Consequently, those transients which were most limiting with respect to the primary to secondary differential pressure were re-analyzed. This re-analysis resulted in changes to the following transients: 10 percent Step Load Increase, 10 percent Step Load Decrease, 50 percent Load Rejection, and Loss of Load. The revised transients from the re-analysis were incorporated into the revised design specification. Based upon a review of the revised design transients, Westinghouse has demonstrated that the steam generator design pressure differential of 1700 psi will not be exceeded during Cycle 15 operation as long as the steady state secondary side pressure in each steam generator is greater that 650 psia. It should be noted that by performing the aforementioned revisions to the Westinghouse Model 44 steam generator design basis, it was not necessary to invoke the higher (110 percent) stress limits for abnormal condition events that exist in later Editions of the Code, as discussed in NSAL-00-007. The subject design basis revisions will enable the Indian Point Unit 2 steam generators to meet the stress limits of their original construction Code edition. All other applicable requirements of IWA-4312, Rerating, of Section XI of the ASME Code will be completed prior to plant restart from the current refueling outage.

Pursuant to 10 CFR 50.59, these changes have been determined to have no adverse effects on the pressure boundary integrity or function of the steam generators, and does not represent an unreviewed safety question.