UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 28, 2000

MEMORANDUM TO: Ashok Thadani, Director
Office of Nuclear Regulatory Research

FROM: for Samuel J. Collins, Director Roy P Zimmerman
Office of Nuclear Reactor Regulation

SUBJECT: REQUEST FOR INDEPENDENT REVIEWS OF MAY 26, 1999, SAFETY EVALUATION REGARDING STEAM GENERATOR TUBE INSPECTION INTERVAL AND FEBRUARY 13, 1995, SAFETY EVALUATION REGARDING F* REPAIR CRITERIA FOR INDIAN POINT STATION UNIT 2

In follow up to discussions with your staff on February 18, 2000, concerning the recent steam generator tube failure event at Indian Point Station Unit 2 (IP-2), this memorandum documents the Office of Nuclear Reactor Regulation's request that the Office of Nuclear Regulatory Research (RES) perform an independent review of the attached safety evaluation (SE) regarding the steam generator (SG) tube inspection interval for this Unit. In addition, this memorandum requests that RES perform an independent review of the attached safety evaluation allowing the F* repair criteria to be used at IP-2.

As you are aware, IP-2 shut down February 15, 2000, because of a sudden increase in primary to secondary leakage in SG 24. In 1999 the staff approved a license request to extend the SG tube inspection interval beyond the 24 calendar months required by the plant technical specifications. In particular, by letter dated December 7, 1998, as supplemented by letter dated May 12, 1999, Consolidated Edison Company of New York, Inc. (the licensee), proposed to amend the technical specifications for the Indian Point Station Unit 2. These letters are also attached. This was to allow a one-time extension of the SG inspection interval and remove the requirement of receiving NRC concurrence on the licensee's proposed SG examination program. By letter dated June 9, 1999, the staff issued the requested amendment and forwarded the SE of the licensee's proposed amendment request to the licensee (TAC No. MA4526).

In addition, by letter dated March 13, 1995, the staff issued an amendment allowing the repair of SG tubes via the implementation of an F* criteria, and forwarded the related February 13, 1995, SE (TAC No. M89373). The SE is attached. The F* criteria allowed tubes that are degraded in a location not affecting structural integrity of the tube to remain in service as an alternative to removal from service through the use of tube plugs. The amendment was issued in response to an application from the licensee transmitted by letter dated April 13, 1994, and supplemented by letters dated December 20, 1994, January 12, 1995, and January 31, 1995.

CONTACT: L. Lund, EMCB/DE
415-2786

J/81

[redacted]

We request that you perform an independent review of that part of the SE regarding the extension of the inspection interval, transmitted to the licensee on June 9, 1999. A written response is requested by March 3, 2000.

We also request that you perform an independent review of the SE regarding the implementation of the F* repair criteria, transmitted to the licensee on March 13, 1995. A written response is also requested by March 3, 2000.

The purpose of these independent reviews is to determine if the staff's conclusions are technically sound and that the data presented by the licensee provided reasonable assurance that the delayed inspection and the use of the F* repair criteria would not result in an appreciably increased probability of tube failure prior to the next scheduled inspection. Your support for this quick response is greatly appreciated.

Docket No.: 50-247

Attachments: As stated

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NAME	JRStrosnider*	BWSheron*	RPZimmerman	SJCollins
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We request that you perform an independent review of that part of the safety evaluation regarding the extension of the inspection interval, transmitted to the licensee on June 9, 1999. A written response is requested by March 3, 2000.

We also request that you perform an independent review of the safety evaluation regarding the implementation of the F* repair criteria, transmitted to the licensee on March 13, 1995. A written response is also requested by March 3, 2000.

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#1 2/23 #2 2/24

MEMORANDUM TO: Ashok Thadani, Director
Office of Nuclear Regulatory Research

FROM: Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

SUBJECT: REQUEST FOR INDEPENDENT REVIEW OF MAY 26, 1999 SAFETY EVALUATION REGARDING STEAM GENERATOR TUBE INSPECTION INTERVAL FOR INDIAN POINT STATION UNIT TWO

Based on discussions with your staff on February 18, 2000, concerning the recent steam generator tube failure event at Indian Point Station Unit 2 (IP-2), the Office of Nuclear Reactor Regulation requests that the Office of Nuclear Regulatory Research perform an independent review of the attached safety evaluation regarding the steam generator (SG) tube inspection interval for this Unit.

As you are aware, IP-2 shut down February 15, 2000, because of a sudden increase in primary to secondary leakage in steam generator 24. In 1999 the staff approved a license request to extend the SG tube inspection interval beyond the 24 calendar months required by the plant TS. In particular, by letter dated December 7, 1998, as supplemented by letter dated May 12, 1999, Consolidated Edison Company of New York, Inc. (the licensee), proposed to amend the technical specifications for the Indian Point Station Unit 2. These letters are also attached. This was to allow a one-time extension of the steam generator inspection interval and remove the requirement of receiving NRC concurrence on the licensee's proposed SG examination program. By letter dated June 9, 1999, the Commission issued the requested amendment and forwarded the related safety evaluation of the licensee's proposed amendment request to the licensee (TAC No. MA4526).

We request that you perform an independent review of that part of the safety evaluation regarding the extension of the inspection interval. The purpose of this independent review is to determine if you, given the same information, would have come to the same conclusion as NRR. A written response is requested by February 25, 2000. Your support for this quick response is greatly appreciated.

Attachments: As stated

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415-2786

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Based on discussions with your staff on February 18, 2000, concerning a possible tube rupture event at Indian Point Station Unit 2 (IP-2), the Office of Nuclear Reactor Regulation requests that the Office of Nuclear Regulatory Research perform an independent review of the attached safety evaluation regarding the steam generator (SG) tube inspection interval for this Unit.

As you are aware, IP-2 shut down February 15, 2000 because of a sudden increase in primary to secondary leakage in steam generator 24. A review of the technical specifications (TS) indicates that the staff approved a license request to extend the SG tube inspection interval beyond the 24 calendar months required by the plant TS. In particular, by letter dated December 7, 1998, as supplemented by letter dated May 12, 1999, Consolidated Edison Company of New York, Inc. (the licensee), proposed to amend the technical specifications for the Indian Point Station Unit 2. These letters are also attached. This was to allow a one-time extension of the steam generator inspection interval and remove the requirement of receiving NRC concurrence on the licensee's proposed SG examination program. By letter dated June 9, 1999, the Commission issued the requested amendment and forwarded the related safety evaluation of the licensee's proposed amendment request to the licensee (TAC No. MA4526).

We request that you perform an independent review of that part of the safety evaluation regarding the extension of the inspection interval. The purpose of this independent review is to determine if you, given the same information, would have come to the same conclusion as NRR, i.e., you would have issued a safety evaluation granting the licensee's request. A written response is requested by February 25, 2000. Your support for this quick response is greatly appreciated.

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We also request that you perform an independent review of the SE regarding the implementation of the F* repair criteria, transmitted to the licensee on March 13, 1995. A written response is also requested by March 8, 2000.

The purpose of these independent reviews is to determine if the staff's conclusions are technically sound and that the data presented by the licensee provided reasonable assurance that the delayed inspection and the use of the F* repair criteria would not result in an appreciably increased probability of tube failure prior to the next scheduled inspection. Your support for this quick response is greatly appreciated.

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Docket No.: 50-247

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 13, 1995

Mr. Stephen E. Quinn
Vice President, Nuclear Power
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, NY 10511

SUBJECT: ISSUANCE OF AMENDMENT FOR INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 (TAC NO. M89373)

Dear Mr. Quinn:

The Commission has issued the enclosed Amendment No. 180 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated April 13, 1994, as supplemented by letters dated December 20, 1994, January 12, 1995, and January 31, 1995.

The amendment revises TSs Sections 3.1.F and 4.13 to allow the repair of steam generator tubes via the implementation of an F* criteria. This would allow tubes that are degraded in a location not affecting structural integrity of the tube to remain in service as an alternative to removal from service through the use of tube plugs. Changes in your request related to tube sleeving will be treated in a subsequent amendment.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly <u>Federal Register</u> notice.

Sincerely,

Francis J. Williams, Jr., Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosures: 1. Amendment No. 180 to DPR-26
2. Safety Evaluation

cc w/encls: See next page

cc:

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Buchanan, NY 10511

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Senior Resident Inspector
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Regional Administrator, Region I
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 180
License No. DPR-26

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Consolidated Edison Company of of New York, Inc. (the licensee) dated April 13, 1994, as supplemented by letters dated December 20, 1994, January 12, 1995, and January 31, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-26 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 180, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

LB Marsh

Ledyard B. Marsh, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 13, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 180

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.1.F-3	3.1.F-3
3.1.F-9	3.1.F-9
4.13-1	4.13-1
4.13-2	4.13-2
4.13-3	4.13-3
4.13-4	4.13-4
4.13-5	4.13-5
4.13-6	4.13-6
4.13-7	4.13-7
4.13-8	4.13-8
Table 4.13-1 (p. 1 of 2)	Table 4.13-1 (p. 1 of 2)

2. Operational Leakage Limits

a. Primary to Secondary Leakage

(1) Primary to secondary leakage through the steam generator tubes shall not exceed 0.3 gpm in any steam generator. With any steam generator tube leakage greater than this limit, the reactor shall be brought to the cold shutdown condition within 24 hours.

(2) If leakage from two or more steam generators in any 20-day period is observed or determined, the reactor shall be brought to the cold shutdown condition within 24 hours and Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation. If tube leaks attributable to the tube denting phenomena are observed in two or more steam generators after the reactor is in cold shutdown, Nuclear Regulatory Commission approval shall be obtained before resuming reactor operation.

(3) Whenever the reactor is shut down in order to investigate steam generator tube leakage and/or to plug or otherwise repair a leaking tube, the NRC shall be informed before any tube is either plugged or repaired, or if no tube is either plugged or repaired, before the steam generator is returned to service.

b. RCS/RHR Pressure Isolation Valves Leakage

(1) Whenever the reactor is above cold shutdown, leakage through each of the RCS/RHR pressure isolation valves 897A, B, C and D, and 838A, B, C and D shall satisfy the following acceptance criteria:

(a) Leakage rates of less than or equal to 1.0 gpm are acceptable.

coolant system and the secondary coolant system. Leakage in excess of 0.3 gpm for any steam generator will require plant shutdown and the leaking tube(s) will be located and either plugged or repaired.

The 10 gpm limit for combined reactor coolant and non-reactor coolant leakage into the containment free volume provides allowance for a limited amount of leakage from sources other than the reactor coolant system within containment while conservatively limiting total leakage into the containment free volume to the same limit (i.e., 10 gpm) for identified reactor coolant leakage alone. This leakage is within the capabilities of the leakage detection and waste processing system and will not interfere with the detection of independent unidentified reactor coolant system leakage.

For those circumstances where high energy line failures occur inside containment resulting in flooding of the containment building sumps and/or floor, automatic actuation of reactor protection, safety injection and/or containment spray systems places the plant in a safe condition and, in some cases, provides intended flooding of the containment building. However, for those circumstances resulting from leakage or failure of low energy systems such as service water or component cooling inside containment, operator action is necessary to prevent accumulation of water on the containment floor to undesirable levels.

If the water level in the containment sump reaches EL. 45', or the water level in the recirculation sump reaches EL. 35', or the water level in the reactor cavity reaches EL. 20', the reactor is placed in cold shutdown within the next 36 hours. If the water level in the containment sump increases above EL. 45' and the water level in the recirculation sump increases above EL. 39' 9", or the water level in the reactor cavity increases above EL. 20' 5", the operator will immediately bring the reactor subcritical and initiate an expeditious cooldown of the plant.

4.13 STEAM GENERATOR TUBE INSERVICE SURVEILLANCE

Applicability

Applies to inservice surveillance of the steam generator tubes.

Objective

To assure the continued integrity of the steam generator tubes that are a part of the primary coolant pressure boundary.

Specifications

Steam generator tubes shall be determined operable by the following inspection program and corrective measures.

A. INSPECTION REQUIREMENTS

1. Definitions

   a. Imperfection is a deviation from the dimension, finish, or contour required by drawing or specification.

   b. Deformation is a deviation from the initial circular cross-section of the tubing. Deformation includes the deviation from the initial circular cross-section known as denting.

   c. Degradation means service-induced cracking, wastage, pitting, wear or corrosion (i.e., service-induced imperfections).

   d. Degraded Tube is a tube, that contains imperfections caused by degradation large enough to be reliably detected by eddy current inspection. This is considered to be 20% degradation.

e. <u>% Degradation</u> is an estimated % of the tube wall thickness affected or removed by <u>degradation</u>.

f. <u>Defect</u> is a degradation of such severity that it exceeds the <u>plugging limit</u>. A tube containing a <u>defect</u> is <u>defective</u>.

g. <u>Plugging Limit</u> is the degradation depth at or beyond which the tube must be plugged or repaired.

h. <u>Hot-Leg Tube Examination</u> is an examination of the hot-leg side tube length. This shall include the length from the point of entry at the hot-leg tube sheet around the U-bend to the top support of the cold leg.

i. <u>Cold-Leg Tube Examination</u> is an examination of the cold-leg side tube length. This shall include the tube length between the top support of the cold leg and the face of the cold-leg tube sheet.

j. <u>F* Distance</u> is the distance of the expanded portion of a tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube from the tubesheet. The F* distance is equal to 1.25 inches and is measured down from the bottom of the roll transition.

k. <u>F* Tube</u> is a tube:

a) With degradation equal to or greater than 40% below the F* distance, and b) which has no indication of degradation within the F* distance, and c) that remains in service.

2. Extent and Frequency of Examination

a. Steam generator examinations shall be conducted not less than 12 months nor later than twenty four calendar months after the previous examination.

b. Scheduled examinations shall include each of the four steam generators in service.

c. Unscheduled steam generator examinations shall be required in the event there is a primary to secondary leak exceeding technical specifications, a seismic occurrence greater than an operating basis earthquake, a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steamline or feedwater line break.

d. Unscheduled examinations may include only the steam generator(s) affected by the leak or other occurrence.

e. In case of an unscheduled steam generator examination, the profilometry tensile strain criterion shall be the same as contained in the approved program for the last scheduled steam generator inspection.

3. Basic Sample Selection and Examination

a. At least 12% of the tubes in each steam generator to be examined shall be subjected to a hot-leg examination.

b. At least 25% of the tubes inspected in Specification 4.13.A.3.a above shall be subjected to a cold-leg examination.

c. Tubes selected for examination shall include, but not be limited to, tubes in areas of the tube bundle in which degradation has been reported, either at Indian Point 2 in prior examinations, or at other utilities with similar steam generators.

d. Examination for deformation ("dents") shall be either by eddy current or by profilometry.

e. Examination for degradation other than deformation shall be by eddy current techniques, using a 700-mil diameter probe. If the 700-mil diameter probe cannot pass through the tube, a 610-mil diameter probe shall be used. For examination of the U-bends and cold-legs of tubes in rows 2 through 5, a 540-mil diameter probe may be used, provided it is justified by profilometry measurement within the tensile strain criterion.

f. In addition to the minimum sample size as determined by Table 4.13-1, all F* tubes shall be inspected within the pertinent tubesheet region. The results of F* tube inspections are not to be utilized as a basis for additional inspections per Table 4.13-1.

4. Additional Examination Criteria

1. Degradation Not Caused by Denting

a. If 5% of more of the tubes examined in a steam generator exhibit degradation or if any of the tubes examined in a steam generator are defective, additional examinations shall be required as specified in Table 4.13-1.

b. Tubes for additional examination shall be selected from the affected area of the tube array and the examination may be limited to that region of the tube where degradation or defective tube(s) were detected.

c. The second and third sample inspections in Table 4.13-1 may be limited to the partial tube inspection only, concentrating on tubes in the areas of the tube sheet array and on the portion of the tube where tubes with imperfections were found.

2. Degradation Caused by Denting

a. Additional examinations, for degradation caused by denting, shall be performed as described in the most recent steam generator examination program approved by the NRC.

B. ACCEPTANCE CRITERIA AND CORRECTIVE ACTION

1. Tubes shall be considered acceptable for continued service if:

a. depth of degradation is less than 40% of the tube wall thickness,

AND

b. the tube will permit passage of a 0.540" diameter probe and the strain in the tube wall (if measured) is less than the tensile strain criterion as specified in the approved examination program, or the tube will permit passage of a 0.610" diameter probe in the absence of strain measurement.

c. the tube is an F* tube and meets a. and b. above the F* region.

2. Tubes that are not considered acceptable for continued service shall be plugged or repaired.

C. REPORTS AND REVIEW AND APPROVAL OF RESULTS

1. The proposed steam generator examination program shall be submitted for NRC staff review and concurrence at least 60 days prior to each scheduled examination.

2. The results of each steam generator examination shall be submitted to NRC within 45 days after the completion of the examination. A significant increase in the rate of denting or significant change in steam generator condition shall be reportable immediately.

3. An evaluation which addresses the long term integrity of small radius U-bends beyond row 1 shall be submitted within 60 days of any finding of significant hour-glassing (closure) of the upper support plate flow slots.

4. Restart after the scheduled steam generator examination need not be subject to NRC approval.

Basis

Inservice examination of steam generator tubing is essential if there is evidence of mechanical damage or progressive deterioration in order to assure continued integrity of the tubing. Inservice examination of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

An essentially 100% tube examination was performed on each tube in each steam generator by eddy current techniques prior to service in order to establish a baseline condition for the tubing. No significant baseline imperfections were identified. In addition, prior to the discontinuance of phosphate treatment and the institution of all-volatile treatment (AVT), a baseline inspection was conducted in March, 1975 before the resumption of power operation.

Wastage-type defects are unlikely with the all-volatile treatment (AVT) of secondary coolant; however, even if this type of defect occurs, the steam generator tube examination will identify tubes with significant degradation from this effect.

The results of steam generator tube burst and collapse tests have demonstrated that tubes having wall thickness of not less than 0.025 inch have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents. An allowance of 10% for tube degradation that may occur between inservice tube examinations added to the 40% degradation depth provided in the acceptance criteria provides an adequate margin to assure that tubes considered acceptable for continued operation will not have a minimum tube wall thickness of less than the acceptable 50% of normal tube wall thickness (i.e. 0.025 inch) during the service life-time of the tubes. Steam generator tube examinations of other operating plants have demonstrated the capability to reliably detect wastage type defects that have penetrated 20% of the original 0.050 inch wall thickness.

Examination of samples of tubes and support plates removed from steam generators have revealed that "denting" is caused by the accretion of steel corrosion products in the tube/support plate annuli. As these corrosion products are more voluminous than the support plate material from which they are derived, a compressive force is exerted on the tubes in the plane of the support plates, resulting in deformation of the tubes. If the deformation results in an ovalization of the tubes, the resulting strain is low and there is no risk of development of stress corrosion cracking in the tubes. However, if the deformation results in an irregular tube shape, the resulting strain may be high enough for the tube to become susceptible to stress corrosion cracking inservice, and it should be preventively repaired. Beginning with the steam generator examination to be conducted during the Cycle 5/6 Refueling Outage, the tensile strain criterion for profilometry shall be 25%. The 25% strain criterion is based on a review of data currently available from operating steam generators, and will be revised as necessary as more experience is gained with the evaluation of this measurement. In the future, this criterion may be revised, either higher or lower, based on steam generator examination results. The profilometry criterion to be used for any steam generator examination shall be established in the most recent program approved by NRC.

A first report on the R&D work leading to the development of profilometry, entitled "Profilometry of Steam Generator Tubes" dated August, 1980, was forwarded to the NRC by Con Edison. Additional R&D work has improved the accuracy of the profilometer and the calculation of strain in a deformed tube.

Before the development of profilometry, a minor diameter of 0.610" was established as the criterion for continuing a tube inservice. This criterion was used successfully for several years at Indian Point Unit 2 and at other plants, and appears to be sufficiently conservative so that it can be continued in the absence of more accurate strain determination by means of profilometry.

A sound roll expansion throughout the F* distance provides a tube to tubesheet interface that ensures the requirements of Regulatory Guide 1.121 are met regardless of the severity of any tube degradation below the F* distance. The F* distance of 1.25 inches is comprised of 1.01 inches of sound roll that ensures tube integrity requirements are met plus 0.24 inches which allows for eddy current measurement uncertainty. The testing and analysis supporting the F* distance is documented in B&W Nuclear Technologies Qualification Report No. BAW-10195P.

Testing performed as documented in BAW-10195 P demonstrates the maximum postulated leakage under accident conditions for repair of 100% of the tube ends using the F* criteria is well below the allowable leakage limits for Indian Point 2 steam generators. If, in the future, steam generator tubes are allowed to remain in service by the use of F* and, in addition, other tube acceptance criteria, then the aggregate maximum postulated accident leakage must be below the allowable leakage limits for Indian Point 2 steam generators.

This program for inservice inspection of steam generator tubes exceeds the requirements of Regulatory Guide 1.83, Revision 1, dated July 1975.

Table 4.13-1

<u>Steam Generator Tube Inspection</u>

First Sample Inspection			Second Sample Inspection		Third Sample Inspection	
Minimum Size	Result	Action	Result	Action	Result	Action
12% tubes per steam generator hot leg plus 3% tubes per steam generator cold leg	C-1	-------	---	-------	- - >	Go to power.
	C-2	Plug or repair defective tubes. Inspect additional 6% tubes in this S.G.	C-1 -	-------	- - >	Go to power.
			C-2	Plug or repair defective tubes. Inspect additional 12% tubes in this S.G.	C-1->	Go to power.
					C-2->	Plug or repair defective tubes. Go to power.
					C-3->	Go to first sample. C-3 action.
			C-3	Go to first sample. C-3 action.		
	C-3	Inspect all tubes this S.G. Plug or repair defective tubes. Inspect 6% tubes in each other S.G. if not included in the examination program.	All other S.G.s C-1 -	------------	- - - >	Go to power.
			Some S.G.s C-2 but no add'l C-3	Go to second sample. C-2 action.		
			Add'l S.G. C-3	Inspect all tubes in all S.G.s. Plug or repairs defective tubes.	- - >	Report to NRC. NRC approval req'd prior to startup.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated April 13, 1994, as supplemented by letters dated December 20, 1994, January 12, 1995, and January 31, 1995, the Consolidated Edison Company of New York (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 2 (IP2) Technical Specifications (TSs). The requested changes would revise TSs Sections 3.1.F and 4.13 to allow the repair of steam generator tubes via the implementation of an F* criteria. This would allow tubes that are degraded in a location not affecting structural integrity of the tube to remain in service as an alternative to removal from service through the use of tube plugs. The requested changes related to tube sleeving will be addressed in a separate amendment. The December 20, 1994, submittal provided responses to an NRC staff Request for Additional Information (RAI). The January 12, 1995, submittal clarified one of the RAI responses and the January 31, 1995, submittal requested that the part of the proposed request dealing with steam generator tube sleeving be separated from the original request to allow additional review of current sleeving issues and to permit processing of the F* part of the request. The December 20, 1994, January 12, 1995, and January 31, 1995, letters provided clarifying information which did not change the initial proposed no significant hazards consideration and that was within the scope of the original *Federal Register* notice.

The licensee proposed an alternative repair criteria for defects found in the tube expansion region within the tubesheet. Steam generator tubes with degradation in excess of the current plugging limits could remain inservice without repair provided the indications existed below a specified distance, F* (F-star), from the bottom of the roll transition region. To support the licensee's request, Babcock & Wilcox Nuclear Technologies (BWNT) completed a test program to demonstrate that the proposed F* distance satisfies the necessary structural and leakage integrity requirements of Appendix A to 10 CFR Part 50 and the IP2 TSs.

Surveillance requirements within the plant TSs require a periodic inspection of steam generator tubes for the detection of potential degradation (i.e., cracks, dents, corrosion, etc.), which could diminish the structural margins and leakage integrity of the tubes. For IP2, detection of tube degradation in excess of the TSs limits requires removal of the tube from service. The licensee has proposed a revised repair criteria that would allow steam generator tube defects to remain in place without repair provided the defects reside a specified distance below the roll transition region. This distance is called F*. Degradation identified in a steam generator tube below the F* length would be allowed to remain in service without repair. This is based on the results of the testing which determined the minimum interference fit engagement length necessary to retain steam generator tubes within the tubesheet.

The NRC staff requested additional information by letter sent to the licensee dated November 21, 1994. In addition, a phone call was held on January 11, 1995, between the NRC staff and the licensee to clarify several answers in the licensee's response to this request. The NRC staff has reviewed the information supplied by the licensee and completed an evaluation of the licensee's request to amend the IP2 TSs with the F* criteria.

The licensee also included in their submittal dated April 13, 1994, a proposed amendment to permit steam generator tube sleeving as an alternative to removing defective tubes from service through plugging. This safety evaluation only addresses the proposed changes associated with F*. Steam generator tube sleeving will be evaluated in a separate safety evaluation.

2.0 BACKGROUND

Steam generator tubes comprise a significant portion of the reactor coolant pressure boundary. Maintenance of this barrier is provided by the integrity of the steam generator tube wall and the tube to tubesheet connection. The connection between the tube and tubesheet is an interference fit made by roll expanding the tube into a bore through the tubesheet. The inelastically deformed steam generator tube is held in place by the elastic springback of the tubesheet. The IP2 steam generator tubes are roll expanded from the bottom of the tubesheet and welded at the tubesheet primary face. Undegraded, this joint provides sufficient strength to maintain adequate structural and pressure boundary (leakage) integrity.

General Design Criteria 14, "Reactor Coolant Pressure Boundary," and 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," of Appendix A to 10 CFR Part 50 state the requirements applicable to maintaining adequate structural and leakage integrity for steam generator tubes. Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," describes an acceptable method to the NRC staff for establishing the limiting safe conditions of tube degradation of steam generator tubing. Although RG 1.121 conservatively focuses on tube degradation in the freespan regions, the methods described apply to other tube regions, such as the roll expansion length.

In order to demonstrate adequate structural margin for steam generator tube degradation, the bases must address the limiting conditions during normal operation, anticipated operational occurrences, and postulated accident conditions. The margin to failure under normal operating conditions as recommended in RG 1.121 should not be less than 3 at any tube location. Subsection NB-3225 of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) specifies the margins of safety under postulated accident conditions.

Structural loads imposed on the steam generator tube-to-tubesheet connections primarily result from the differential pressure between the primary and secondary sides of the tubes. The peak postulated loading occurs during a steam line break due to a lowering of the secondary side pressure. However, normal operating loads, cyclic joint loading from major plant transients (i.e., startup/shutdown), and potential thermal expansion loads can also be significant. The analysis (BAW-10195P) supporting the licensee's proposed modification to the IP2 TSs addressed the limiting conditions necessary to maintain adequate integrity of the tube-to-tubesheet interference fit. Specifically, the tube must not experience excessive displacement relative to the tubesheet.

Leakage through steam generator tubes is limited by plant TSs. For IP2, the limits for the allowable primary to secondary leakage are stated in TS 3.1.F. The total primary to secondary leakage must be less than 0.3 gpm in any steam generator.

The elastic preload between the tube and tubesheet not only prevents pullout of the tube from the tubesheet, but also provides a leaktight barrier minimizing the potential for primary to secondary coolant leakage. With sufficient length of hardroll, the tube-to-tubesheet connection will not allow any leakage under normal and faulted conditions. Steam generator tube through-wall degradation within the roll expanded joint would decrease the path length necessary for primary to secondary leakage. The licensee's proposed amendment would permit such degradation to remain in service provided there exists a sufficient length of undegraded hardroll below the bottom of the roll transition region. Therefore, an acceptable F* distance must be such that leakage integrity is not jeopardized during all analyzed conditions.

3.0 <u>TESTING TO DETERMINE F*</u>

The licensee completed a test program to determine the F* distance. Two failure criteria were considered for testing -- pullout of the tube from the tubesheet and primary to secondary leakage requirements. The following describes the methodology used for the tests.

3.1 <u>Fabrication of Test Specimens</u>

Mockup blocks were fabricated to simulate the actual installed rolled expansion fabrication variations and loading conditions with the IP2 Steam Generators. Lengths of steam generator tubing were roll expanded into holes drilled through the mockup blocks. Several peripheral tubes were roll expanded in the test block to simulate additional constraint by surrounding tubes. The interior tubes were used for testing. In order to simulate tube wall degradation, tubes were severed at a certain distance below the bottom of the roll transition region. This configuration is representative of a 360° through-wall crack present in the tube.

Several mockup blocks were fabricated for testing. After the tubes were expanded into the blocks and the hardroll length was verified by nondestructive evaluation methods, each block was thermally soaked to simulate the effects of actual steam generator service temperature. Heating the test block would theoretically lead to thermal stress relaxation in the roll expansion joint.

To account for potential factors, which might affect the calculated F* length, several variables were changed within the test matrix. For example, tubing with high and low yield strengths were tested. In addition, tubesheet bore surface roughness, as well as tubesheet bore diameter were varied in the test matrix. The results from the tests revealed the effects from these variables.

In the qualification test program the effects of boric acid corrosion and post-weld heat treatment on the integrity of the tube-to-tubesheet joint were considered. BWNT postulated that if primary coolant penetrated through the steam generator tube wall and came in contact with the carbon steel tubesheet the potential exists for initiating stress corrosion cracking in the tubesheet. Based on previous studies, the likelihood of developing significant corrosion of the tubesheet bore due to boric acid corrosion is low.

After the roll expansion process to secure steam generator tubes into the tubesheet, the channel head to tubesheet weld attachment was stress relieved via heat treatment. BWNT assessed the effects of this fabrication step on the tube-to-tubesheet joint by exposing several test specimens to temperatures similar to that experienced by the tubes during the original fabrication process. Any stress relaxation due to creep would theoretically lower the pullout strength of the tube from the tubesheet. Based on ultimate load testing, it was concluded that the post-weld heat treatment did not adversely affect the strength of the joint retaining the tube within the tubesheet.

3.2 Testing for F* Determination

To determine the necessary roll expansion joint engagement length the licensee completed a series of mechanical tests on the simulated steam generator tubes. The testing involved subjecting tubes to combined internal pressure and axial loading to determine the F* distance. Under actual service conditions, the differential pressure acting over the cross section of the tube provides an axial force tending to force the tube out of the tubesheet. This axial load is counterbalanced by the frictional force between the tube and tubesheet due to the roll expanded interference fit. The primary to secondary differential pressure also increases radial force between the tube and tubesheet by slightly expanding the tube and bowing the tubesheet. The increased radial forces due to these effects will increase the frictional force between the tube and tubesheet resisting pullout. The use of both internal pressure and axial loading during testing simulates actual loading on the roll expanded joint.

Three different mechanical tests were conducted to determine F*: a locked tube test, pressure cycling, and an ultimate load test. All tests were conducted at ambient temperatures. The locked tube test simulated the loading applied to a steam generator tube during cooldown of the plant assuming the tube was locked at a tube support plate location. The unequal coefficients of thermal expansion of the tube wrapper and the tube would lead to an applied tensile load on the tube. For the pressure cycling test, several tubes were subjected to pressure cycling between low and normal operating pressures. Motion of the tube was monitored during the cyclic loading. Finally, tubes were subjected to an ultimate load test. Tubes were internally pressurized and subjected to an increasing axial tensile load until failure. Failure was defined as a relative movement of a specified distance between the tube and tubesheet.

As part of the test program to provide the basis for the proposed F* length, steam generator tubes were subject to leak rate testing. Tubes were internally pressurized to simulate differential pressures during normal operating and faulted conditions. The acceptance criteria for these tests specified an allowable leakage limit. Tube displacements were also monitored during the tests.

3.3 F* Test Results

Based on the results of the leakage rate and mechanical testing the licensee determined a nominal engagement length necessary to ensure adequate margins of safety. Accounting for limited sample size, statistical scatter in the data, and NDE inspection error this value was increased appropriately. The licensee has proposed that steam generator tubes with degradation in the roll expanded portion of the tube can remain in service if all degradation lies below the F* distance. The F* distance is equal to the 1.25 inches and is measured down from the bottom of the roll transition.

4.0 EVALUATION OF PROPOSED TECHNICAL SPECIFICATION AMENDMENT

The licensee proposed a revision to the IP2 TS to implement the F* criterion by letter dated April 13, 1994. The following summarizes the proposed changes:

1. The TSs define the F* distance as the distance of expanded portion of a tube which is sufficient to resist pullout of the tube from the tubesheet. This distance is equal to 1.25 inches.

2. The revised TSs include a definition of an F* tube, which is a tube with degradation of 40% through-wall or greater below the F* distance, has no degradation within the F* distance, and remains in service.

3. In addition to the minimum sample size for steam generator tube inspection, all F* tubes will be inspected within the pertinent tubesheet region at every outage.

As a bases for the proposed amendment to the IP2 TSs, tests were completed to determine an acceptable F* distance. The testing utilized specimens which reflect the actual tube-to-tubesheet joint configuration within the IP2 Steam Generators. Unknown variables, which could potentially affect the calculated F* distance were taken into consideration in developing the test matrix. Applied loads for structural assessment and leakage rate testing were specified in accordance with staff recommendations in RG 1.121 and the ASME Code. The licensee's proposed changes to the IP2 TSs are consistent with the conclusions from the test program to determine F*.

To ensure continued integrity of F* tubes, the licensee has included a requirement in the plant TSs to reinspect F* tubes during each steam generator examination. Although F* tubes are specified as part of the Basic Sample Selection for examination, F* steam generator tubes are not considered as part of the 12 percent Minimum Size specified for examination in Table 4.13-1 of the IP2 TSs. The inclusion of a requirement in the proposed TSs to reexamine F* tubes during each examination should ensure that no degradation exists within the F* distance of all F* tubes.

One issue not discussed in the licensee's original request to amend the IP2 TSs, was the option of rerolling a degraded steam generator tube so that it would then be acceptable by the F* criterion, pending approval of this amendment. The licensee indicated during a phone call on January 11, 1995, that rerolling has been considered and was evaluated using the same methodology as that described previously in Section 3 of this Safety Evaluation. However, the licensee stated that the test program addressed potential impurities, which could become trapped between the outer tube surface and the tubesheet bore during the rerolling process. Results from these tests concluded that the proposed F* distance is acceptable for rerolled tubes as well. Based on one assumption made in the qualification test program the licensee could only reroll a steam generator tube up to one-half the tubesheet thickness. Additional rerolling beyond this height would be considered an unanalyzed condition.

The NRC staff has reviewed the TSs change related to the implementation of the F* criterion proposed by the licensee in their submittal dated April 13, 1994. Based on information provided in the submittal, additional information provided by letter dated December 20, 1994, and during a phone conversation on January 11, 1995, the NRC staff finds the licensee's proposed changes acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (59 FR 27051). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: P. Rush

Date: March 13, 1994

May 26, 1999

MEMORANDUM TO: S. Singh Bajwa, Chief
Project Section 1-1
Division of Licensing Project Management

FROM: Edmund J. Sullivan, Chief
NDE & Metallurgy Section
Materials and Chemical Engineering Branch

SUBJECT: SAFETY EVALUATION REGARDING STEAM GENERATOR TUBE INSPECTION INTERVAL FOR INDIAN POINT STATION UNIT 2 (TAC NO. MA4526)

By letter dated December 7, 1998, as supplemented by letter dated May 12, 1999, Consolidated Edison Company of New York, Inc. (the licensee), proposed to amend the technical specifications (TSs) for the Indian Point Station Unit 2 (IP-2). The proposed amendment would allow a one-time extension of the steam generator (SG) inspection interval in TS 4.13A.2.a. The amendment would also remove the requirement of receiving NRC concurrence on the licensee's proposed SG examination program in TS 4.13C.1.

The Materials and Chemical Engineering Branch has reviewed the licensee's proposed amendment request and supporting documentation. The staff finds the proposed amendment to be acceptable, because the modification will not impact the IP-2 SGs' ability to safely operate for the entire fuel cycle and receiving formal NRC concurrence on the licensee's proposed SG examination program is not necessary.

Our safety evaluation is attached. This completes our review for TAC number MA4526.

Docket No.: 50-247

Attachment: As stated

CONTACT: Andrea T. Keim, EMCB/DE
415-1671

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 26, 1999

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415-1671

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
STEAM GENERATOR TUBE INSPECTION INTERVAL
INDIAN POINT STATION UNIT 2
CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
DOCKET NUMBER 50-247

1.0 INTRODUCTION

By letter dated December 7, 1998, as supplemented by letter dated May 12,1999, Consolidated Edison Company of New York, Inc. (the licensee), submitted a request to modify the technical specifications (TSs) for Indian Point Station Unit 2 (IP-2). The proposed amendment would allow a one-time extension of the steam generator (SG) inspection interval in TS 4.13A.2.a. The amendment involves adding a statement allowing the SG inspection interval to coincide with the year 2000 refueling outage (the units 14th refueling outage) and no later than June 3, 2000. The amendment would also remove the requirement of receiving NRC concurrence on the licensee's proposed SG examination program in TS 4.13C.1.

IP-2 is a Westinghouse four-loop pressurized water reactor with Model 44 SGs. Each SG contains 3260 mill-annealed (MA), Inconel 600 tubes.

2.0 BACKGROUND

The applicable surveillance requirement for IP-2 at this time is TS 4.13A.2.a. This requirement specifies that the SG inspections are to occur at intervals not exceeding 24 calendar months. The licensee's last surveillance was performed during the 13th refueling outage and was completed on June 13, 1997. The licensee did not perform an inspection during the unscheduled maintenance outage commencing on October 25, 1997, because a minimum interval of 12 months (as stated in Surveillance 4.13A.2.a) is required before taking credit for a subsequent inservice inspection. IP-2 was shut down for an unscheduled maintenance outage from October 1997 until August 1998.

3.0 EVALUATION

The objective of the staff's evaluation is to determine the impact of the proposed extended inspection interval on the structural and leakage integrity of the tubes, considering the extended period that the plant was shut down. The staff has focused it's evaluation on the licensee's evaluations of 1) SG tube integrity for the previous and current operating cycles, 2) SG lay-up in accordance with industry guidelines and the present cycle (cycle 14) chemistry control, and 3) leakage monitoring and leakage guidelines.

3.1 June 1997 SG Inspection

The licensee performed an extensive eddy current inspection in June 1997 (end of cycle 13). The inspection included 100 percent examination using a bobbin probe on all inservice tubes. If the tight-radius U-bends in rows two and three precluded passage of the Cecco-5/bobbin probe,

ATTACHMENT

a rotating pancake coil (RPC) was utilized. Row one tubes were previously preventively plugged. Any locations with distorted bobbin coil signals were resolved by the Cecco-5 coils. An RPC probe was utilized for further characterization of indications as necessary.

Tubes with indications evaluated at 40 percent of the wall thickness or larger, linear indications (axial or circumferential), Cecco-5 indications at tube support plate intersections, and tube roll transition cracks that were not rerolled, or indications that did not meet the F* distance were plugged. Twenty tubes were plugged due to passage restrictions of the 610 mil diameter bobbin probe. Seventeen tubes were administratively plugged due to passage restrictions of the Zetec +point dent inspection probe (gimbaled +point probe). These tubes were examined by the Cecco-5/bobbin probe but did not allow access of the +point probe. Eighteen tubes were preventively plugged based upon an IP-2 study of tube support plate deformation.

Prior to tube plugging, the licensee performed in-situ pressure testing on selected tubes exceeding EPRI/Westinghouse screening criteria. Four tubes in the tubesheet :revice area were found to have exceeded the screening criteria and were subsequently in-situ tested. Two additional tubes were in-situ tested even though they were below the screening criteria. Those two tubes were selected because one was typical of tube roll transition cracking, and the other was an axial indication above the top of the tubesheet. No leakage was detected from the six tubes that were in-situ tested. Test pressures of 1710 psi, 2500 psi, 2840 psi, and 5075 psi were used to simulate indications under normal operating differential pressure, intermediate pressure, steam line break pressure, and three times normal operating pressure, respectively. Each pressure was met and held for two minutes. The in-situ pressure tests showed that the SG tubes have maintained adequate structural integrity in accordance with Regulatory Guide (RG) 1.121. The in-situ pressure tests demonstrated that RG 1.121 margins were met over the past operating cycle (cycle 13). On the basis of the licensee's assessment, the staff finds that the structural and leakage integrity of tubes during cycle 13 was acceptable.

The licensee assessed the SG tube integrity for the remainder of the present operating cycle (cycle 14) on the basis of the end of cycle 13 inspection and testing results. The severity of degradation at the end of cycle 14 was projected considering BOC degradation status, degradation growth rates, and EOC allowable degradation. The severity of degradation at the EOC 14 was projected to determine if required structural and leakage integrity margins would be maintained. The scope of the licensee's evaluation included the following forms of degradation: 1) top of tubesheet (TTS) pitting, 2) outer diameter stress corrosion cracking (ODSCC) in the TTS sludge pile region, 3) ODSCC in the tubesheet crevice, 4) primary water stress corrosion cracking (PWSCC) at the roll transition region, 5) PWSCC at dented TSP intersections, 6) ODSCC at dented TSP intersections, 7) PWSCC at row two U-bends, and 8) wear. The licensee's evaluation determined that the forms of degradation listed above did not present a challenge to the 3ΔP structural margin criteria for the expected operating cycle length of 21.4 effective full power months (EFPM). Based on a review of this portion of the licensee's assessment the staff expects the steam generator tubes will continue to satisfy structural and leakage integrity requirements under normal and accident conditions through the end of the current operating cycle (cycle 14). This conclusion is based on: 1) the licensee's comprehensive eddy current examination and plugging practice at EOC 13, 2) the growth rates of the degradation mechanisms are expected to be similar to what was seen for cycle 13 operation, and 3) the licensee's acceptable in-situ testing results on the limiting EOC 13 indications.

3.2 Chemistry Assessment for the SG During Shutdown and the Present Operating Cycle

After the June 13, 1997, inspection, IP-2 commenced operation. The Unit was subsequently shut down for an extended maintenance outage. During the outage the Unit remained in cold shutdown condition for 304 days prior to restart on August 5, 1998. The licensee maintained the SG in wet lay-up conditions in accordance with EPRI guidelines by adding the appropriate quantities of ammonium hydroxide and carbohydrazide. The ammonium hydroxide was added to control pH and the carbohydrazide was added as an oxygen scavenger. For one hour each day, when conditions permitted, each of the steam generators were sparged with nitrogen for one hour. This was done to drive off any air that may have entered the SG gas space.

The licensee performed routine sampling and analysis of the lay-up solution. The licensee determined that the lay-up solution was maintained at acceptable alkaline and reducing conditions during the outage. However, the licensee did detect a slight depression of the pH which was attributed to dissolved carbon dioxide in the lay-up solution. The carbon dioxide was due to the reaction of the carbohydrazide and oxygen. The samples taken during the outage indicated that no detectable dissolved oxygen (less than 10 ppb) was identified in any of the SGs.

The concentrations of other potentially corrosive impurities in the lay-up solution were routinely monitored during the outage period. The concentrations of chloride, sulfate and sodium were each maintained well below the 1000 ppm maximum that the EPRI guidelines recommend.

The staff believes that the SG lay-up was maintained in accordance with industry guidelines which were designed to minimize the potential for corrosion during wet lay-up conditions. Based on the above, the staff concludes that, during shutdown, the SG were maintained at reduced temperatures and with water chemistry conditions that should have prevented further degradation of the SG tubes.

Chemistry Control During Operation of Cycle 14
(August 1998 - April 1999)

Each of the SG were drained and refilled with condensate quality water prior to exceeding 200°F during startup for resuming cycle 14. SG chemistry has been maintained in accordance with EPRI guidelines for the present operating period (August 1998 - April 1999). SG impurities have been maintained well below EPRI recommended action levels. No intrusions of impurities into the secondary plant have been observed that would indicate a condenser tube leak (chloride, sulfate, or sodium). Iron and copper corrosion products during this operating period have been below the EPRI guideline recommended action levels for these corrosion products.

The staff finds the licensee's water chemistry monitoring and procedures provide assurance that corrosion during the operation period of August 1998 - April 1999 has been minimized.

3.3 Leakage Monitoring and Leakage Guidelines

The licensee stated that should unforeseen circumstances cause SG tube leakage, there are multiple methods available to monitor primary-to-secondary leakage through the SGs. They employ radiation monitors in the condenser air ejector, the SG blowdown line, and the main steamline (MSL). In addition, MSL N-16 monitors are installed, which significantly enhance monitoring of MSL activity. In addition, TS 3.1.F.2.a.(1) limits the primary-to-secondary leakage to 0.3 gallons per minute (gpm) for any one SG. However, the licensee maintains an administrative limit of 0.1gpm. This administrative limit provides added assurance that, should a leak develop during the operating cycle, it would be quickly detected to allow immediate mitigating actions to be taken.

The staff finds the licensee's leakage monitoring program provides assurance that should a leak develop during the operating cycle it would be quickly detected allowing immediate mitigating actions to be taken before tube rupture occurs.

4.0 MODIFICATION OF THE PLANT TSs

The licensee proposes to add the following footnote to page 4.13-2:

> *Examinations scheduled for 1999 only, shall be conducted during the 2000 refueling outage which will commence no later than June 3, 2000. The scheduled examinations will be completed prior to return to service from the 2000 Refueling Outage.

The licensee proposes to modify TS 4.13C.1. to state:

> The proposed steam generator examination program shall be submitted for NRC staff review at least 60 days prior to each scheduled examination.

The modification will require the licensee to submit for staff review their proposed SG examination program 60 days prior to the scheduled examination. The licensee will no longer be required to get formal NRC approval for their proposed SG examination program. The 60 days notice of the licensee's proposed SG examination program provides time for the NRC to review the examination program and determine if there are any concerns to be addressed.

Modifications to TSs Sections 4.13A.2.e, 4.13A.4.2.a, and Bases Section 4.13 will be modified to be consistent with other licensees regarding NRC approval of the proposed SG examination program.

The staff has reviewed the proposed modifications and finds them acceptable.

5.0 SUMMARY

Based on the above evaluation, the staff finds that conducting TS 4.13A.2.a during a mid-cycle surveillance in June 1999 to be unnecessary. NRC staff concludes that the licensee's proposal to allow a one time extension to the SG tube inspection interval is acceptable and that there is reasonable assurance that SG tubes will maintain structural and leakage integrity for the entire cycle 14 operation. The staff also finds that since the licensee is required to submit their proposed SG examination program 60 days prior to the scheduled outage, receiving formal NRC concurrence is not necessary.

A. Alan Blind
Vice President

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December 7, 1998

Re: Indian Point Unit No. 2
Docket No. 50-247

Document Control Desk
US Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555-0001

SUBJECT: Proposed Amendment to Technical Specifications Regarding Steam Generator Tube Inservice Inspection Frequency

Pursuant to 10 CFR 50.90, Consolidated Edison Company of New York, Inc. (Con Edison) transmits herewith one (1) signed original and two (2) copies of "Application for Amendment to Operating License," sworn to on December 7, 1998. This application proposes amendments to the Indian Point Unit No. 2 Technical Specifications that would permit a one-time only extension of the steam generator tube inservice inspection interval for fuel cycle 14, and revises specific reporting requirements to reflect an NRC request. Technical Specification Section 4.13A.2.a requires steam generator tube inspections to be conducted at not less than 12 months and no later than twenty- four calendar months after the previous inspection. Technical Specification Section 4.13C.1 requires the submittal and NRC concurrence of the proposed steam generator examination program. The proposed amendment to remove the requirement to receive concurrence is the result of an NRC request and is administrative in nature.

During the 1997 refueling outage, steam generator tube inspections were completed (i.e., steam generator manway closed) on June 13, 1997. Following the completion of those inspections, the unit was heated-up above 200 F on June 30, 1997. Upon return to service from the refueling outage, the unit was subsequently shutdown on October 25, 1997 for an unscheduled maintenance outage. A cumulative duration of 304 days with the plant at cold shutdown (i.e., below 200 F) had occurred before the plant was re-started on August 5, 1998. During the period of cold shutdown the steam generators were maintained in a wet lay-up condition by adding appropriate quantities of ammonium hydroxide and carbohydrazide. The concentrations of impurities, including dissolved oxygen, were maintained well below industry recommended maximums while the steam generators were in a lay-up condition. A mid-cycle outage to perform periodic refueling cycle testing is currently scheduled for November 1999. The length of this outage is estimated to be 15 days.

As a result of this maintenance outage, and the scheduled mid-cycle outage in November 1999, Con Edison has rescheduled the start of the 1999 refueling outage to June 3, 2000.

In accordance with Technical Specification 4 . 2.a. operation of Indian Point 2 beyond June 13, 1999 would not be permitted. Con Edison hereby requests that the steam generator inspection period, based upon 24 calendar months, be extended to no later than June 3, 2000. The proposed Technical Specification pages which support this amendment to Technical Specification Sections 4.13A.2.a, and 4.13 C.1 are provided as Attachment A for your review. Attachment B to this application for amendment reflects the safety assessment supporting this change as well as the conclusion that the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92.

No new regulatory commitments are being made by Con Edison in this correspondence.

Should you or your staff have any questions regarding this matter, please contact Mr. Charles W. Jackson, Manager, Nuclear Safety and Licensing.

Very truly yours,

A. Alan Blind

Attachment

C: Mr. Hubert J. Miller
Regional Administrator-Region I
US Nuclear Regulatory Commission
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Mr. Paul Eddy
State of New York Department of Public Service
3 Empire Plaza
Albany, NY 12223

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the matter of	)	
	)	
CONSOLIDATED EDISON COMPANY	)	Docket No. 50-247
OF NEW YORK, INC.	)	
(Indian Point Station,	)	
Unit No. 2)	)	

APPLICATION FOR AMENDMENT TO OPERATING LICENSE

Pursuant to Section 50.90 of the Regulations of the Nuclear Regulatory Commission ("NRC"), Consolidated Edison Company of New York, Inc. ("Con Edison"), as holder of Facility Operating License No. DPR-26, hereby applies for amendment of the Technical Specifications contained in Appendix A of that license. Specifically, Con Edison requests that the change specified in Attachment A to this submittal be approved. The proposed change would permit a one-time only extension of the steam generator tube inservice inspection frequency for fuel cycle 14. Additionally the proposed administrative change would delete the requirement to receive NRC concurrence of the proposed steam generator examinations.

The specific proposed change to the Indian Point Unit No. 2 Technical Specifications is to Sections 4.13A.2.a, and 4.13C.1 as set forth in Attachment A to this Application. A Safety Assessment of the proposed change is set forth in Attachment B to this Application. This assessment demonstrates that the proposed change does not represent a significant hazards consideration as defined in 10 CFR 50.92(c).

As required by 10 CFR 50.91(b)(1), a copy of this application and our analysis concluding that the proposed change does not constitute a significant hazards consideration has been provided to the appropriate New York State official designated to receive such amendments.

BY: A. Alan Blind

A. Alan Blind
Vice President

9812160012 981207
PDR ADOCK 05000247
P PDR

Subscribed and sworn to
before me this 7th day
of December, 1998

Karen L. Lancaster
Notary Public

KAREN L. LANCASTER
Notary Public, State of New York
No 60-4643689
Qualified in Westchester County
Term Expires 9/30/99

ATTACHMENT A

PROPOSED TECHNICAL SPECIFICATIONS

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
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DECEMBER 1998

g. Plugging Limit is the degradation depth at or beyond which the tube must be plugged or repaired.

h. Hot-Leg Tube Examination is an examination of the hot-leg side tube length. This shall include the length from the point of entry at the hot-leg tube sheet around the U-bend to the top support of the cold leg.

i. Cold-Leg Tube Examination is an examination of the cold-leg side tube length. This shall include the tube length between the top support of the cold leg and the face of the cold-leg tube sheet.

j. F* Distance is the distance of the expanded portion of a tube which provides a sufficient length of undegraded tube expansion to resist pullout of the tube from the tubesheet. The F* distance is equal to 1.25 inches and is measured down from the bottom of the roll transition.

k. F* Tube is a tube:

a) With degradation equal to or greater than 40% below the F* distance, and b) which has no indication of degradation within the F* distance, and c) that remains in service.

l. Sleeving refers to tube repair achieved by laser welded sleeving, as described by Westinghouse Report WCAP-13583 and 13088. Sleeving is used to maintain a tube in service or return a previously plugged tube to service.

2. Extent and Frequency of Examination

a. Steam generator examinations shall be conducted not less than 12 months nor later than twenty four calendar months after the previous examination.*

b. Scheduled examinations shall include each of the four steam generators in service.

* Examinations scheduled for 1999 only, shall be conducted during the 2000 Refueling Outage which will commence no later than June 3, 2000. The scheduled examinations will be completed prior to return to service from the 2000 Refueling Outage.

B. ACCEPTANCE CRITERIA AND CORRECTIVE ACTION

1. Tubes shall be considered acceptable for continued service if:

   a. depth of degradation is less than:

      - 40% of the tube wall thickness, or
      - 23% of the sleeve wall thickness

   AND

   b. the tube will permit passage of a 0.540" diameter probe and the strain in the tube wall (if measured) is less than the tensile strain criterion as specified in the approved examination program, or the tube will permit passage of a 0.610" diameter probe in the absence of strain measurement.

   c. the tube is an F* tube and meets a. and b. above the F* region.

2. Tubes or sleeves that are not considered acceptable for continued service shall be plugged or repaired.

C. REPORTS AND REVIEW OF RESULTS

1. The proposed steam generator examination program shall be submitted for NRC staff review at least 60 days prior to each scheduled examination.

2. The results of each steam generator examination shall be submitted to NRC within 45 days after the completion of the examination. A significant increase in the rate of denting or significant change in steam generator condition shall be reportable immediately.

3. An evaluation which addresses the long term integrity of small radius U-bends beyond row 1 shall be submitted within 60 days of any finding of significant hour-glassing (closure) of the upper support plate flow slots.

4. Restart after the scheduled steam generator examination need not be subject to NRC approval.

ATTACHMENT B

SAFETY ASSESSMENT

AND

BASIS FOR NO SIGNIFICANT HAZARDS
CONSIDERATION DETERMINATION

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
DECEMBER 1998

SAFETY ASSESSMENT

BACKGROUND

During the 1997 refueling outage, inspections of the Indian Point Unit No. 2 steam generators were completed (i.e., steam generator manway closed) on June 13, 1997. The original examination program submitted to the NRC was expanded to include full length examination of all steam generator tubes. Following the completion of those inspections, the unit was heated-up above 200 F on June 30, 1997. Upon return to service after the 1997 refueling outage, the unit was subsequently shutdown on October 25, 1997 for an unscheduled maintenance outage. During this extended maintenance outage, the steam generators were maintained in a cold shutdown condition, minimizing the effects of corrosion and deterioration. On August 5, 1998, Indian Point Unit No. 2 was heated above 200 F and returned to service. A duration of 304 days in cold shutdown (i.e., below 200 F) had accumulated before the plant was re-started on August 5, 1998. A mid-cycle outage to perform instrument calibrations is currently scheduled for November 1999. The duration of this outage is estimated to be 15 days.

REQUESTED CHANGE AND PURPOSE

The proposed change would permit a one-time only extension of the steam generator tube inservice inspection interval for fuel cycle 14. The extension would permit steam generator tube inspections to be conducted during the next refueling outage which would commence no later than June 3, 2000.

Technical Specification Section 4.13C.1 requires the submittal and NRC concurrence of the proposed steam generator examination program. The proposed amendment to remove the requirement to receive concurrence is the result of an NRC request and is administrative in nature.

JUSTIFICATION FOR REQUESTED CHANGES

BASES

In accordance with Indian Point Unit No. 2 Technical Specification Section 4.13A.2.a, and the completion date of the last steam generator inspections on June 13, 1997, the next inspection would normally be required by June 13, 1999. If this requirement is maintained, an additional mid-cycle outage will be required, which will incur unnecessary personnel radiation exposures and increased challenges to engineered safety features.

The Indian Point Unit No. 2 steam generator inservice inspection program is based upon the guidance contained within Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," Revision 1, dated July 1975. The purpose of this surveillance is to provide reasonable assurance of equipment integrity necessary to operate without experiencing tube rupture or tube leakage in excess of specified limits. This is accomplished by identifying and removing from service defective steam generator tubes.

Details of the Indian Point Unit No. 2 steam generator tube inservice inspection program proposed for the thirteenth refueling outage were submitted to the NRC in Con Edison letter dated February 7, 1997. The original scope was subsequently expanded during the outage, to include full length examination of all steam generator tubes. This change was primarily due to the indications discovered at the hot leg and cold leg upper support plate locations. A comprehensive inspection was performed. Based on the results of the inspections, assessments, and associated tube repairs, the steam generators were determined to be acceptable for continued service at full power. The results of the 1997 refueling outage steam generator inspections were submitted to the NRC via Con Edison letter dated July 29, 1997.

Following completion of the steam generator inspections on June 13, 1997, and commencement of power operation, Indian Point Unit No. 2 was subsequently shutdown for an extended maintenance outage. During this outage the unit remained in cold shutdown condition for 304 days prior to restart on August 5, 1998. An additional 5 days at cold shutdown is anticipated as a result of the planned mid-cycle outage to perform periodic refueling cycle tests. Based upon similar criteria previously accepted and documented in NRC letter to Con Edison dated April 9, 1997, the resulting 309 days at cold shutdown would theoretically permit extending the current inspection interval to April 16, 2000. The basis for acceptance of this increase in the technical specification limit is the "non-operating" steam generator time between the last inspection and the upcoming inspection. An additional 48 days is requested in order to support commencement of the refueling outage scheduled for June 3, 2000. Extending the steam generator "operating" duration by 48 days would not significantly increase wear which might lead to tube failure. A review of past steam generator eddy current data indicates no appreciable growth trend. The wear indications identified in the 1993 and 1995 examinations were reviewed against the 1997 results. For the 21 indications identified in 1993 and 1995, seven (7) indications showed no change, four (4) disappeared, four (4) decreased in depth, and six (6) increased in depth. The nominal increase or decrease of the indications, excluding disappeared indications, was 3-4%, which is within the accuracy of the eddy current measurements. Thus, wear growth was not appreciable. Degradation of the steam generator tubes due to stress corrosion and pitting is a chemical process which is dependent upon temperature. Corrosion rate generally decreases by a factor of two for each 18 F temperature reduction. Since the steam generators were maintained in cold shutdown temperature conditions, the environment for corrosion was reduced to an inconsequential level. No appreciable steam generator tube wear or degradation is expected as a result of this extension. As a result, Con Edison has a high level of confidence that corrosion growth and new corrosion initiation during the cold shutdown were essentially non-existent, and the steam generators are prepared to operate for a full fuel cycle without incident. In support of this conclusion, we provide the following:

- TUBE INSPECTION - The results of the steam generator tube eddy current examinations conducted during the 1997 refueling outage were submitted to the NRC via Con Edison letter dated July 29, 1997. A combination Cecco-5/bobbin probe was utilized for the majority of the eddy current testing. The Cecco-5/bobbin probe was qualified to the EPRI

PWR Steam Generator Examination Guidelines. As part of the Cecco-5 (Cecco) qualification program, a C-Scan or topographical presentation graphics package was developed and incorporated into the Cecco-5 data analysis guidelines. This data presentation graphics is an enhancement to the eddy current data analysis. The original examination program was expanded to include full length examination of all steam generator tubes. This scope expansion was made because of the Cecco indications found at the hot leg and cold leg upper support plate locations. Additionally, all sludge pile pit indications were characterized by the +Point prol e to determine if crack-like indications could be associated with the pits. Sludge pile pitting and AVB wear were dispositioned by the bobbin analysis in the absence of a +Point linear indication. All other indications were dispositioned based on the examinations with the Cecco probe. Tubes with indications evaluated at 40 percent or greater of the wall thickness, linear indications (axial or circumferential), Cecco-5 indications at tube support plate intersections (both characterized by +Point and not confirmed by the +Point probe), and tube roll transition cracks that were not rerolled, or did not meet F*, were plugged.

- REDUCED TEMPERATURE - Intergranular attack/stress corrosion cracking (IGA/SCC) growth is well understood to be accelerated by increasing temperature. Reducing temperature is a proven method to slow both initiation and growth. The effect of reduced temperature can be estimated using the Arrhenius equation and assuming an average value of activation energy of 57 kcal/mole. This results in decrease by a factor of two in corrosion rate for each 18 F temperature reduction. Since the steam generators were maintained at cold shutdown conditions (below 200 F), instead of the normal operational hot leg temperature of 590 F, the corrosion rate during the cold shutdown period can be considered to have been essentially halted.

- WATER CHEMISTRY - The industry guide to water chemistry control is the EPRI Primary and Secondary Chemistry Guidelines. In October of 1997 the steam generators were placed in wet lay-up by adding the appropriate quantities of ammonium hydroxide and carbohydrazide. Carbohydrazide was added as an oxygen scavenger in place of hydrazine. The industry has recently recognized that carbohydrazide is a much more effective oxygen scavenger than is hydrazine at the cold temperatures of wet lay-up. In addition, carbohydrazide does not pose the industrial hygiene concerns that hydrazine does. During the outage when conditions permitted each of the steam generators were sparged with nitrogen for one hour each day to drive off any air that may have entered the steam generator gas space. With the exception of the time period during which maintenance activities associated with the blowdown isolation and main steam safety valves did not permit nitrogen sparging (16% of the total number of outage days), the steam generators were required to be sparging. Routine sampling and analysis of the lay-up solution indicated that the lay-up solution was maintained at an acceptable pH during the outage. A byproduct of the carbohydrazide and oxygen reaction is the formation of carbon dioxide. A slight depression of the pH of the lay-up solution was measured as a result of the dissolved carbon dioxide in the lay-up solution. The concentrations of impurities, including dissolved oxygen, were maintained well below industry recommended maximums while the steam generators were in a lay-up condition. This careful control of impurities provides a measure of added confidence that there was no significant change in the condition of the steam generator tubes during the cold shutdown period.

Indian Point Unit No. 2 employs radiation monitors in the condenser air ejector, in the steam generator blow down line, and in the main steam line. Main Steam line N-16 monitors have also been installed to enhance monitoring of main steam line activity. Furthermore, Technical Specification 3.1.F.2.a.(1) limits primary to secondary leakage to 0.3 gpm in any steam generator which does not contain tube sleeves; however, Con Edison maintains an administrative limit of 0.1 gpm. This provides assurance that, should a leak develop during the operating cycle, it would be quickly detected to allow immediate mitigating actions to be taken. There are currently no tube sleeves installed in the Indian Point Unit No. 2 steam generators.

Activities associated with a steam generator inspection for Indian Point Unit No. 2 typically incur a radiation exposure of approximately 40 person-rem per inspection. Performing a mid-cycle steam generator inspection solely to conform to the letter of the technical specification, without any corresponding safety benefit, is inconsistent with the principles of As Low As Reasonably Achievable (ALARA).

Finally, the Indian Point Unit No. 2 steam generator inservice inspection program is based upon the guidance contained within Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," Revision 1, dated July 1975. The regulatory guide requires that subsequent inservice inspections be no more than of 24 calendar months after the previous inspection. However, EPRI Report TR-107569-V1R5, "PWR Steam Generator Examination Guidelines: Revision 5" dated September 1997, specifies that subsequent inservice inspections be performed at the end of each fuel cycle or 24 EFPM, whichever is less. The cycle 14 core design life is 643 EFPD or approximately 21.4 EFPM.

CONCLUSION

In conclusion, the extensive inspection completed in June 1997 and the benefits of the extended cold shutdown condition of the steam generators work together to ensure that the Indian Point Unit No. 2 steam generators are in a condition that can be reasonably expected to safely and reliably support full power operation for the entire fuel cycle. However, should an unforseen circumstance cause leakage which exceeds guidelines, a number of systems are available for timely detection and mitigation. Additionally the proposed administrative change to delete the requirement to receive NRC concurrence of the proposed steam generator examinations will have no bearing on the actual results of the steam generator examinations. Thus, there is no adverse consequence to public health and safety that might result from granting this request for technical specification amendment.

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The proposed changes do not involve a significant hazards consideration since:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change does not involve any physical modifications to the plant or modification in the methods of plant operation which could increase the probability or consequences of previously evaluated accidents. The proposed change permits an extension of the current steam generator tube inservice inspection cycle. This extension would allow the steam generator tube examinations to be conducted during the 2000 refueling outage which will commence no later than June 3, 2000. The basis for acceptance of this increase in the technical specification limit is the "non-operating" steam generator time between the last examination and the upcoming examination. Extending the steam generator "operating" duration by 48 days would not significantly increase wear which might lead to tube failure. No appreciable steam generator tube wear or degradation is expected as a result of this extension. This change will not affect the scope, methodology, acceptance limits and corrective measures of the existing steam generator tube examination program. The probability and consequences of failure of the steam generators due to leaking or degraded tubes is not increased by the proposed change. Additionally the proposed administrative change to delete the requirement to receive NRC concurrence of the proposed steam generator examinations will have no bearing on the actual results of the steam generator examinations. Therefore, the probability and the consequence of a design basis accident are not being increased by the proposed change.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   Plant systems and components will not be operated in a different manner as a result of the proposed Technical Specification change. The proposed change permits the upcoming steam generator tube examination to be conducted during the 2000 refueling outage that will commence no later than June 3, 2000. There are no plant modifications or changes in methods of operation. This extension is based upon the "non-operating" steam generator time between the last examination and the upcoming examination. Extending the steam generator "operating" duration by an additional 48 days would not significantly increase wear which might lead to tube failure. The proposed extension will not increase the probability of occurrence of a tube rupture, increase the probability or consequences of an accident, or create any new accident precursor. Additionally the proposed administrative change to delete the requirement to receive NRC concurrence of the proposed steam generator examinations will have no bearing on the actual results of the steam generator examinations. Therefore, the possibility for an accident of a different type than was previously evaluated in the safety analysis report is not created by the proposed change to the Technical Specification.

3. The proposed change does not involve a significant reduction in a margin of safety.

The proposed change to Technical Specification section 4.13A.2.a will not reduce the margin of safety. This amendment involves an extension of the current steam generator tube inservice inspection cycle. The basis for acceptance of this increase in the technical specification limit is the "non-operating" steam generator time between the last examination and the upcoming examination. Extending the steam generator "operating" duration by an additional 48 days would not significantly increase wear which might lead to tube failure. No appreciable steam generator tube wear or degradation is expected as a result of this extension. Additionally the proposed administrative change to delete the requirement to receive NRC concurrence of the proposed steam generator examinations will have no bearing on the actual results of the steam generator examinations. Therefore, the accident analysis assumptions for design basis accidents are unaffected and the margin of safety is not decreased by the proposed Technical Specification change.

Based on the preceding analysis it is concluded that operation of Indian Point Unit No. 2 in accordance with the proposed amendment does not increase the probability of an accident previously evaluated, does not create the possibility of a new or different kind of accident from any accident previously evaluated, nor reduce any margin of plant safety. Therefore, the license amendment does not involve a Significant Hazards Consideration as defined in 10 CFR 50.92.

The proposed change has been reviewed by both the Station Nuclear Safety Committee (SNSC) and the Nuclear Facilities Safety Committee (NFSC). Both Committees concur with the proposed change.

James S. Baumstark
Vice President
Nuclear Engineering

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May 12, 1999

Re: Indian Point Unit No. 2
Docket No. 50-247

Document Control Desk
US Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555-0001

Subject: Response to Request for Additional Information - Proposed Amendment to Technical Specifications Regarding Steam Generator Tube Inservice Inspection Frequency

Reference: 1) Con Edison Letter to USNRC de[illegible] December 7, 1998 entitled "Proposed Amendment to Tech[illegible]al Specifications Regarding Steam Generator Tube Inservi[illegible] Inspection Frequency"

2) USNRC Letter to Con Edison dated May 5, 1999 entitled "Request For Additional Information – Regarding Indian Point Nuclear Generating Unit 2 Steam Generator Inspection Interval One-Time Extension (TAC No. MA4526)"

By letter dated December 7, 1998, Consolidated Edison Company of New York, Inc. (Con Edison) submitted a proposed amendment to the Indian Point Unit No. 2 (IP-2) Technical Specifications that would permit a one-time extension of the steam generator tube inservice inspection interval for fuel cycle 14. This amendment would be a one-time change to the 24 calendar month steam generator inspection period for IP-2

On May 4, 1999, Con Edison and NRC personnel discussed areas where additional information was needed in order to assist the staff in its review of the proposed amendment request. As a result of the discussion, a list of requested additional information was transmitted to Con Edison via Reference 2. Pursuant to 10 CFR 50.54(f), this letter and attachments provide the response of Con Edison to NRC's request for additional information dated May 5, 1999. Attachment 1 to this letter contains the requested information. Attachment 2 provides additional proposed pages for Technical Specification Sections 4.13A.2.e, 4.13A.4.2.a, and 4.13 Basis that support this amendment. Con Edison has evaluated this additional information and determined that it does not invalidate the original no significant hazards determination provided in our letter dated December 7, 1998.

The steam generator tube inspections that this amendment would extend become due on June 13, 1999. Therefore it is requested that this proposed amendment receive an expedited review.

No new regulatory commitments are being made by Con Edison in this correspondence.

Should you have any questions regarding this matter, please contact Mr. John McCann, Manager, Nuclear Safety and Licensing.

Very truly yours,

Subscribed and sworn to
before me this 12th day
of May 1999

Karen L. Lancaster
Notary Public
KAREN L. LANCASTER
Notary Public, St... New York
No. 60-4645654
Qualified in W... County
Term Expires 9/30/99

Attachments

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ATTACHMENT 1

**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REVIEW OF TECHNICAL SPECIFICATIONS AMENDMENT REGARDING
A ONE-TIME SG INSPECTION INTERVAL EXTENSION**

**CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
MAY 1999**

Question 1

For each degradation mechanism, please provide a general description of the operational assessment methodology used to ensure that SG tube integrity will be maintained for the entire fuel cycle (cycle 14). The description should include an explanation of the predictive methodology, flaw growth rates, and NDE uncertainty used to determine structural and accident leakage integrity.

Response

The degradation mechanisms that were evaluated for the conditional monitoring assessment following Cycle 13, leading to the operational assessment for the upcoming Cycle 14 are summarized below.

Pitting Above Top of Tubesheet

The majority of freespan indications detected at Indian Point 2 have been due to pitting attack on the tube OD in the sludge pile region. The pit indications observed at EOC 13 were judged to be normal for IP-2 and to not represent either a burst or leakage potential. Tubes which exhibited numerical % depth bobbin calls or "PIT" based on +Point characterization were reviewed to determine the relative severity of the indications. Where possible, +Point voltages were compared with the bobbin % and voltage values, since the bobbin technique is qualified for depth estimation under EPRI's Appendix H criteria.

The maximum indicated pitting depth of 45% found in 1997 will not challenge the tube structural or leakage integrity. Similarly, sludge pile pitting would not be expected to represent a burst or steam line break leakage potential at EOC 14, since the reported maximum lengths and depths were well below the in-situ burst screening threshold parameters. While specific growth rate analyses of pit indications at IP-2 were not performed for EOC 13, historical information suggests that the average growth characteristics of pits are less than 10% through-wall per cycle. For pit indications to represent a challenge to structural or leakage integrity, the pit diameter must approach 0.3 to 0.4 inch, and depths must approach the 80% range. Bobbin voltages of such indications would exceed 10 volts. The 60% deep, 0.187 inch diameter flat bottom hole of the ASME standard yields a voltage of about 4 volts. The non-degraded tube material surrounding the pit acts to significantly reinforce the degradation, and prevents tearing of the base metal. Based on historical data, typical pit diameters are expected to be about 0.10 to 0.15 inch. The expected failure mode of a postulated pit indication would be rupture of the degraded tube material in the pit region in an axial mode. This opening is expected to be blunted by the non-degraded tube material surrounding the pit.

The 45% depth for R35 C19 SG23 was the production call recorded using the bobbin coil from the Cecco-5/bobbin combination probe, with a reported voltage of 0.52 volts. The bobbin coil signal from the combination Cecco-5/bobbin probe was used for pit sizing in 1997. Historical review of the eddy current data reports shows that the 1995 depth of the pit indication on R35 C19 SG23 was 34% through-wall, 0.43 volts, using the combination Cecco-5/bobbin probe and 27% through-wall, 0.40 volts, using the standard bobbin. The bobbin coils on the combination

Cecco-5/bobbin probe and the standard bobbin are the same, and the 7% difference in through-wall depth is within limits of probe repeatability and part of the NDE uncertainty. This implies a depth growth of 11% through-wall, and 0.11 volts voltage growth using the combination Cecco-5/bobbin probe, and 18% through-wall using the combination Cecco-5/bobbin probe from 1997 and standard bobbin from 1995. The 41% depth call for R31 C15 SG24 in 1997 was recorded using the bobbin coil from the combination Cecco-5/bobbin probe, with a reported voltage of 1.12 volts. In 1997, the standard bobbin depth was 22%, with a 1.00 volt response. Historical review of the eddy current data reports shows that the 1995 bobbin depth of the pit indication on R31 C15 SG24 was 33% through-wall, with a voltage of 0.66 volts using the standard bobbin probe. The combination Cecco-5/bobbin probe was not used on this tube in 1995. This implies a growth of 8% or (-) 11%, and voltage growth of 0.46 volts or 0.34 volts, dependent upon the probe. The 1993 standard bobbin depth of the pit indication on R31 C15 was 32%, with a voltage of 0.26 volts.

ODSCC (Outside Diameter Stress Corrosion Cracking) Above Top of Tubesheet (Sludge Pile)

ODSCC in the sludge pile was detected for the first time in the 1997 inspection. Comparisons of 1997 and 1995 bobbin data suggest that significant change had occurred, since in 1995 only one tube exhibited a signal similar to those observed in 1997. The 1997 indication production eddy current data for the +Point probe was 0.71 volt with a 48% maximum depth and a 0.54 inch length. In-situ pressure testing at a maximum pressure of 5075 psi (equivalent to approx. 4613 psi at operating temperature) did not result in burst or leakage. Other OD sludge pile indications were detected but did not meet more than one of the screening parameters. Recent +Point depth sizing evaluations performed by Westinghouse for axial ODSCC indicate that flaw average depth standard deviation measurement error is about 10% through-wall. A 20% measurement uncertainty allowance is provided in the in-situ screening parameters.

Based on +Point depth profile analysis, the maximum voltage was reported as 0.58 volts, the maximum depth was reported as 69%, average depth as 48% and length as 0.55 inch. The calculated burst pressure based on the depth profile input and lower tolerance limit (LTL) material property values was 6126 psi at operating conditions. Therefore, a 1536 psi margin against the 3ΔP structural capability requirement of 4590 psi for Indian Point 2 was provided for this flaw at EOC 12. The indication on this represented the limiting sludge pile ODSCC indication. It should be noted that the average depth from profiling did not exceed the screening criteria for in-situ testing.

This indication was not +Point inspected in 1995, but was inspected using Cecco-5. In 1997, it represented the limiting indication. Average depth detection thresholds for axial ODSCC are in the range of 20% to 30% through-wall [Probability of Detection (POD) about 0.2 to 0.5] for both the Cecco-5 and +Point, therefore, assuming the +Point depth profile to be accurate, the growth in average depth for Cycle 13 is bounded by about 18% to 28% for sludge pile ODSCC indications. Since all detected indications were repaired, the modest growth would lead to acceptable EOC structural integrity even if 40% to 50% average depth indications were not detected.

While ODSCC in the sludge pile region is a new mechanism at Indian Point 2, the 22 indications detected represent 0.17% of the total tube population. Therefore, based upon the observed sludge pile flaw eddy current characteristics at IP-2 and in-situ testing results, from more limiting flaws at similar plants, it can be concluded that this corrosion mechanism would not represent either a burst or steam line break leakage potential at EOC 14.

ODSCC in Tubesheet Crevice

During the 1997 inspection, there were several ODSCC tubesheet crevice indications. The condition monitoring assessment methodology used concluded that the tubesheet crevice ODSCC detected at EOC 13 would not have contributed to primary to secondary leakage during a postulated steam line break event. The indications were characterized with the +Point probe. The indication with the largest +Point indication was in-situ pressure tested, as well as the three others that exceeded the screening parameters. The burst screening parameters were based on 1) crack voltage, critical or threshold, 2) maximum depth, and 3) depth profiling. The characterized indication lengths, based on depth profiling ranged from 5.02 to 6.70 inches. The four tested tubes showed no evidence of leakage. The tubes were tested at a nominal cold pressure of 2900 psi (which is equivalent to approx. 2636 psi at operating temperature).

The four ODSCC crevice indications in-situ tested at EOC 13 ranged in maximum voltage from 1.18 to 3.64 volts (by +Point). No leakage was detected at 2900 psi. Axial ODSCC indications within the crevice are restrained from burst by the presence of the tubesheet, and therefore represent only a leakage potential. As the most limiting indications at EOC 13 were successfully tested without leakage, it is highly unlikely that leakage would occur at EOC 14.

Since no changes are anticipated in the operating temperatures for Cycle 14, and the anticipated cycle length is approximately 7% longer compared to Cycle 13, a more extensive crevice degradation condition is not anticipated. During the cold shutdown period, the cold shutdown metal temperatures would not contribute to degradation of the tubing.

Roll Transition PWSCC (Primary Water Stress Corrosion Cracking)

An approximately ½ inch long tube indication located in the original hard roll region was in-situ tested. The indication was leak tested to show that indications typically reroll repaired did not represent a leakage potential at EOC 13 conditions. The indication was characterized with a combination bobbin/RPC probe. No leakage was detected. The testing was done with a maximum pressure of 2900 psi (which is equivalent to approx. 2636 psi at operating temperature).

In addition, the largest number of indications were detected in the original roll expanded regions. These indications were located below the additional F* reroll expansion repair, as such, these indications do not represent a leakage potential under any condition since the sound roll length above the indication prevents primary to secondary leakage.

There were three circumferential indications in this region of the steam generators. One of the tubes in-situ tested had both an axial ODSCC in the tubesheet crevice as well a circumferential indication. The location of the testing mandrel tested both indications. The tube showed no evidence of leakage. The other circumferential indications were characterized by the +Point probe, and subsequently analysis determined that the remaining sound cross section was capable of resisting a tensile overload failure. The tube was later rerolled and met the F* requirement for continued service.

PWSCC at Dented Tube Support Plate Intersections

Plugging determinations at dented tube support plate intersections used the most conservative inspection technique available. Prior to the start of the 1997 examination, EOC 13, Con Edison made a decision to conservatively use the Cecco-5 probe as the probe-of-record. PWSCC at dented tube support plate intersections were plugged based primarily on the Cecco-5 response. Of the 34 Cecco-5 dented tube support plate possible indications plugged, 23 were located at hot leg dented tube support plates. Of the 23 total hot leg Cecco-5 indications inspected with +Point, 22 produced NDD (No Degradation Detected) responses, and the single confirmed ID indication was reported with a length of approximately 0.34 inch, entirely contained by the tube support plate. The global denting condition at IP-2 tube support plate intersections will act to preclude tube support plate motion during a postulated steam line break event. Therefore, as the indications are expected to remain within the tube support plates, these indications would not represent a burst potential. Postulated axial PWSCC flaws extending out of the TSP would require a through-wall length outside of the plate of about 0.5 inch. For Indian Point 2, a 0.42 inch long, 100% through-wall over the entire length flaw would be expected to provide integrity consistent with the 3ΔP structural requirement. EPRI data for roll transition burst tests shows that the restraining effect of the roll interaction with the tubesheet prevents the lower crack tip from opening, thereby "reinforcing" the flaw, and providing burst pressures in excess of those calculated assuming the flaw were a freespan flaw. This reinforcing effect produced burst pressures equivalent to flaw lengths reduced by about 0.1 inch. A similar effect is expected at the edge of a dented TSP. The constraining effect of the dent will prevent the crack tip adjacent to the edge of the TSP from opening, and therefore will provide added structural integrity. Lack of +Point probe responses in these tubes may suggest that the degradation extent was below the detection threshold of the +Point probe for this condition. It is expected that the +Point coil is able to reliably detect structurally significant PWSCC flaws in a dented tube support plate intersection. Lack of a +Point response may also suggest that the Cecco-5 probe may have been influenced by some other causal mechanism, such as OD tube deposits. As through wall lengths on the order of 0.15 inch may represent a lower bound for steam line break leakage at a non-dented tube support plate intersections and would be expected to be reliably detected by the +Point, the lack of a +Point response strongly suggests that these intersections would not represent a leakage potential during a postulated steam line break event as the denting condition would require even longer through wall lengths to support steam line break leakage.

The denting phenomenon at IP-2 was a concern entering into the EOC 13 examination due to the observance of PWSCC at other plants with large number of dented tube support plate intersections. However, only 34 tubes were plugged at dented tube support plate intersections. All of the 34 tubes were plugged based on the Cecco-5 suggestion of the presence of possible degradation. The 1997 inspection program strongly suggests that the denting occurrence at IP-2 has not accentuated a PWSCC concern, and may be attributed in part to the relatively low T_{hot} value of the Unit.

ODSCC at Dented Tube Support Plate Intersections

A total of eleven cold leg dented tube support plate intersections were identified with the Cecco-5 probe as possible indications; all were +Point inspected. Two were found to contain ODSCC indications and one was reported to contain a volumetric indications based on the +Point responses. As with the hot leg intersections, the tube support plates would be expected to remain adjacent to the indication, thereby precluding burst during a postulated steam line break event. Only one of these dented intersections restricted passage of the 0.640 inch diameter Cecco-5 probe, due to a restriction at a lower elevation support plate. None of the indications at dented tube support plate (either ID or OD initiated) confirmed by the +Point extended out of the plate while the longest indication was reported at 0.37 inch.

PWSCC at Row 2 U-bend

For the first time, a Row 2 U-bend PWSCC indication was found. The dimension of the indication by +Point characterization was below the in-situ screening threshold for Row 2 U-bend flaws. All Row 1 tubes were preventively plugged prior to operation for a non-tube related issue. As this represents the first detected U-bend indication after approximately 23 years of operation, any growth rates associated with this indication would be considered minimal.

Question 2

Please discuss the results of your condition monitoring assessment conducted during your most recent inspection. Include, what degradation mechanisms were evaluated using the Westinghouse and/or EPRI screening criteria? What mechanisms were not evaluated using the screening criteria? What assurance is provided that the structural integrity would be maintained?

Response

All of the above listed mechanisms were evaluated to the Westinghouse screening parameters, with the exception of sludge pile pitting (pitting above top of tubesheet). The pit indications were not assessed against the Westinghouse screening criteria since screening criteria specifically for pits is not included. As discussed above, pit indications generally do not represent structural or leakage integrity issues, and as such, in-situ testing of such indications will not provide additional support for tube integrity assessments. Pit indications during the outage were assessed for in-situ testing based on a maximum bobbin coil depth of 50% and voltage of 3 volts. No indications met this criteria. Discussions of the approaches used for each mechanism, including sludge pile pitting, are provided as part of the Question 1 response.

Question 3

Please provide an assessment of the water chemistry performance during the extended period of SG wet lay-up and during the current cycle of operation.

Response

Assessment of Steam Generator Wet Lay-Up October 1997 through July 1998

Each steam generator was adequately treated with a lay-up solution to minimize corrosion during the outage period. An appropriate reducing and alkaline condition was maintained and aggressive impurities were maintained at very low concentrations.

In October of 1997 the steam generators were placed in wet lay-up by adding the appropriate quantities of ammonium hydroxide and carbohydrazide. Carbohydrazide was added as an oxygen scavenger in place of hydrazine. During the outage when conditions permitted, each of the steam generators were sparged with nitrogen for one hour each day to drive off any air that may have entered the steam generator gas space.

Routine sampling and analysis of the lay-up solution indicated that the lay-up solution was maintained at an acceptable pH during the outage. A byproduct of the carbohydrazide and oxygen reaction is the formation of carbon dioxide. A slight depression of the pH of the lay-up solution was measured as a result of the dissolved carbon dioxide in the lay-up solution.

Oxygen was well excluded from the lay-up solution for the entire outage period. All samples taken during the outage indicated no detectable dissolved oxygen (less than 10 ppb) in each steam generator. This demonstrates the effective oxygen scavenging capabilities of carbohydrazide, which ensured reducing conditions were present throughout the outage period.

The concentrations of potentially corrosive impurities in the lay-up solution were routinely monitored during the outage period. The concentrations of chloride, sulfate and sodium were each maintained well below the 1000 ppm maximum that the EPRI guidelines recommend. Maximum concentrations of chloride, sulfate and sodium were 126 ppb, 287 ppb and 490 ppb, respectively. In addition, each of the steam generators were drained and refilled with condensate quality water prior to exceeding 200°F during the unit start-up.

Assessment of Operating Steam Generator Chemistry August 1998 through April 1999

Steam generator chemistry was maintained in accordance with EPRI guidelines for the entire operating period. The secondary system ETA/Ammonia/Hydrazine/Boric Acid chemical treatment was well controlled in accordance with the station's secondary system chemistry pH control program.

Steam generator impurities were well maintained below EPRI recommended action levels. Steam generator blowdown chloride, sulfate and sodium concentrations averaged 0.9 ppb, 1.5 ppb and 0.5 ppb, respectively, while operating during this period. There were no intrusions of impurities into the secondary plant during this operating period such as may occur as a result of a condenser tube leak. In addition, EPRI guidelines limits and hold points for chloride, sulfate and sodium concentrations were satisfied prior to the power escalations associated with the August and September startups.

The iron and copper corrosion products transported to the steam generators in the final feedwater during this operating period averaged 4.3 ppb and 0.5 ppb, respectively. These concentrations are below the EPRI guideline recommended action levels for these corrosion products. It should be noted however that the copper corrosion product concentration is higher than what is generally seen in this industry due to the presence of copper in the low pressure feedwater heaters.

<u>Question 4</u>

In TS sections 4.13A.2.e. and 4.13A.4.2.a. there are references to the most recent steam generator examination program approved by the NRC. Please modify these sections to maintain consistency of the TS when the NRC concurrence is no longer required.

<u>Response</u>

See Attachment 2 for Technical Specification Sections 4.13A.2.e, 4.13A.4.2.a, and 4.13.Basis revised pages. These pages have been modified to be consistent with the removal the requirement for NRC concurrence of the proposed steam generator examination program.

ATTACHMENT 2

ADDITIONAL PROPOSED TECHNICAL SPECIFICATIONS

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
MAY 1999

c. Unscheduled steam generator examinations shall be required in the event there is a primary to secondary leak exceeding technical specifications, a seismic occurrence greater than an operating basis earthquake, a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steamline or feedwater line break.

d. Unscheduled examinations may include only the steam generator(s) affected by the leak or other occurrence.

e. In case of an unscheduled steam generator examination, the profilometry tensile strain criterion shall be the same as contained in the program for the last scheduled steam generator inspection.

3. <u>Basic Sample Selection and Examination</u>

a. At least 12% of the tubes in each steam generator to be examined shall be subjected to a hot-leg examination.

b. At least 25% of the tubes inspected in Specification 4.13.A.3.a above shall be subjected to a cold-leg examination.

c. At least 20% of a random sample of tubes containing sleeves shall be Subjected to an examination throughout the sleeved portion of the tube.

d. Tubes selected for examination shall include, but not be limited to, tubes in areas of the tube bundle in which degradation has been reported, either at Indian Point 2 in prior examinations, or at other utilities with similar steam generators.

e. Examination for deformation ("dents") shall be either by eddy current or by profilometry.

f. Examination for degradation other than deformation shall be by eddy current techniques, using a 700-mil diameter probe. If the 700-mil diameter probe cannot pass through the tube, a 610-mil diameter probe shall be used. For examination of the U-bends and cold-legs of tubes in rows 2 through 5, a 540-mil diameter probe may be used, provided it is justified by profilometry measurement within the tensile strain criterion.

g. In addition to the minimum sample size as determined by Table 4.13-1, all F* tubes shall be inspected within the pertinent tubesheet region. The results of F* tube inspections are not to be utilized as a basis for additional inspections per Table 4.13-1.

4. Additional Examination Criteria

1. Degradation Not Caused by Denting

a. If 5% of more of the tubes examined in a steam generator exhibit degradation or if any of the tubes examined in a steam generator are defective, additional examinations shall be required as specified in Table 4.13-1 with the exception of degraded or defective tube sleeves.

b. Tubes for additional examination shall be selected from the affected area of the tube array and the examination may be limited to that region of the tube where degradation or defective tube(s) were detected.

c. The second and third sample inspections in Table 4.13-1 may be limited to the partial tube inspection only, concentrating on tubes in the areas of the tube sheet array and on the portion of the tube where tubes with imperfections were found.

d. If as tube sleeve exhibits degradation of greater than 23% or is otherwise defective, an additional 20% (minimum) of the unsampled sleeves shall be examined. If a sleeve exhibits degradation of greater than 23% or is otherwise defective in the second sample, all remaining sleeves shall be examined.

2. Degradation Caused by Denting

a. Additional examinations, for degradation caused by denting, shall be performed as described in the most recent steam generator examination program.

Basis

Inservice examination of steam generat tubing is essential if there is evidence of mechanical damage or progressive deterioration in order to assure continued integrity of the tubing. Inservice examination of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

An essentially 100% tube examination was performed on each tube in each steam generator by eddy current techniques prior to service in order to establish a baseline condition for the tubing. No significant baseline imperfections were identified. In addition, prior to the discontinuance of phosphate treatment and the institution of all-volatile treatment (AVT), a baseline inspection was conducted in March, 1975 before the resumption of power operation.

Wastage-type defects are unlikely with the all-volatile treatment (AVT) of secondary coolant; however, even if this type of defect occurs, the steam generator tube examination will identify tubes with significant degradation from this effect.

The results of steam generator tube burst and collapse tests have demonstrated that tubes having wall thickness of not less than 0.025 inch have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents. An allowance of 10% for tube degradation that may occur between inservice tube examinations added to the 40% degradation depth provided in the acceptance criteria provides an adequate margin to assure that tubes considered acceptable for continued operation will not have a minimum tube wall thickness of less than the acceptable 50% of normal tube wall thickness (i.e. 0.025 inch) during the service life-time of the tubes. Steam generator tube examinations of other operating plants have demonstrated the capability to reliably detect wastage type defects that have penetrated 20% of the original 0.050 inch wall thickness.

Examination of samples of tubes and support plates removed from steam generators have revealed that "denting" is caused by the accretion of steel corrosion products in the tube/support plate annuli. As these corrosion products are more voluminous than the support plate material from which they are derived, a compressive force is exerted on the tubes in the plane of the support plates, resulting in deformation of the tubes. If the deformation results in an ovalization of the tubes, the resulting strain is low and there is no risk of development of stress corrosion cracking in the tubes. However, if the deformation results in an irregular tube shape, the resulting strain may be high enough for the tube to become susceptible to stress corrosion cracking inservice, and it should be preventively repaired. Beginning with the steam generator examination to be conducted during the Cycle 5/6 Refueling Outage, the tensile strain criterion for profilometry shall be 25%. The 25% strain criterion is based on a review of data currently available from operating steam generators, and will be revised as necessary as more experience is gained with the evaluation of this measurement. In the future, this criterion may be revised, either higher or lower, based on steam generator examination results. The profilometry criterion to be used for any steam generator examination shall be established in the most recent program.

A first report on the R&D work leading to the development of profilometry, entitled "Profilometry of Steam Generator Tubes" dated August, 1980, was forwarded to the NRC by Con Edison. Additional R&D work has improved the accuracy of the profilometer and the calculation of strain in a deformed tube.

Before the development of profilometry, a minor diameter of 0.610" was established as the criterion for continuing a tube inservice. This criterion was used successfully for several years at Indian Point Unit 2 and at other plants, and appears to be sufficiently conservative so that it can be continued in the absence of more accurate strain determination by means of profilometry.

A sound roll expansion throughout the F* distance provides a tube to tubesheet interface that ensures the requirements of Regulatory Guide 1.121 are met regardless of the severity of any tube degradation below the F* distance. The F* distance of 1.25 inches is comprised of 1.01 inches of sound roll that ensures tube integrity requirements are met plus 0.24 inches which allows for eddy current measurement uncertainty. The testing and analysis supporting the F* distance is documented in B&W Nuclear Technologies Qualification Report No. BAW-10195P.

Testing performed as documented in BAW-10195 P demonstrates the maximum postulated leakage under accident conditions for repair of 100% of the tube ends using the F* criteria is well below the allowable leakage limits for Indian Point 2 steam generators. If, in the future, steam generator tubes are allowed to remain in service by the use of F* and, in addition, other tube acceptance criteria, then the aggregate maximum postulated accident leakage must be below the allowable leakage limits for Indian Point 2 steam generators.

This program for inservice inspection of steam generator tubes exceeds the requirements of Regulatory Guide 1.83, Revision 1, dated July 1975.