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 Federal Register notice.

Sincerely,

Original signed by:

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

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*See previous concurrence

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

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Consolidated Edison Company
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Docket No. 50-247

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2. Safety Evaluation

cc w/encls: See next page

Stephen E. Quinn
Consolidated Edison Company
of New York, Inc.

Indian Point Nuclear Generating
Station Units 1/2

cc:

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Buchanan, NY 10511

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Secretary - NFSC
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New York, NY 10003

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

DATED: March 13, 1995

AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NO. DPR-26-INDIAN POINT UNIT 2

Docket File
PUBLIC
PDI-1 Reading
S. Varga, 14/E/4
J. Zwolinski, 14/H/3
L. Marsh
C. Vogan
F. Williams
OGC
D. Hagan, T-4 A43
G. Hill (2), T-5 C3
C. Grimes, 11/E/22
PRush
ACRS (4)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

200009

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

L B 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

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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. % Degradation is an estimated % of the tube wall thickness affected or removed by degradation.

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

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.

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. <u>Additional Examination Criteria</u>

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

Steam Generator Tube Inspection

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

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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 TESTING TO DETERMINE F*

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 Fabrication of Test Specimens

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