May 20, 199

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. M93834)

Dear Mr. Quinn:

The Commission has issued the enclosed Amendment No. 192 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 August 29, 1995, as supplemented by letters dated August 7, 1996, and January 10, 1997.

The amendment revises TSs to incorporate the commitments made in connection with Amendment No. 183, which authorized the installation of laser welded steam generator tube sleeves inside defective tubes in steam generators.

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,

/S/

Jefferey F. Harold, 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. 192 to DPR-26
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\IP2\IP293834.AMD *See Previous Concurrence

DF01

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure **"E"** = Copy with attachment/enclosure **"N"** = No copy

OFFICE	PM:PDI-1	E	LA:PDI-1		OGC*		D:PDI-1			
NAME	JHarold:mw JH		SLittle		AHodgdon		SBajwa SSB			
DATE	05/20/97		05/20/97		05/13/97		05/20/97		05/ /97	

Official Record Copy

020009

NRC FILE CENTER COPY

9706020189 970520
PDR ADOCK 05000247
P PDR

DATED: ~~May 20, 1997~~

AMENDMENT NO. 192 TO FACILITY OPERATING LICENSE NO. DPR-26-INDIAN POINT UNIT 2

Docket File
PUBLIC
PDI-1 Reading
S. Varga, 14/E/4
S. Bajwa
S. Little
J. Harold
J. Tsao
OGC
G. Hill (2), T-5 C3
C. Grimes
ACRS
C. Cowgill, Region I

cc: Plant Service list

DATED: ~~May 20, 1997~~

AMENDMENT NO. 192 TO FACILITY OPERATING LICENSE NO. DPR-26-INDIAN POINT UNIT 2

Docket File
PUBLIC
PDI-1 Reading
S. Varga, 14/E/4
S. Bajwa
S. Little
J. Harold
J. Tsao
OGC
G. Hill (2), T-5 C3
C. Grimes
ACRS
C. Cowgill, Region I

cc: Plant Service list

May 20, 199

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. M93834)

Dear Mr. Quinn:

The Commission has issued the enclosed Amendment No. 192 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 August 29, 1995, as supplemented by letters dated August 7, 1996, and January 10, 1997.

The amendment revises TSs to incorporate the commitments made in connection with Amendment No. 183, which authorized the installation of laser welded steam generator tube sleeves inside defective tubes in steam generators.

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,

/S/

Jefferey F. Harold, 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. 192 to DPR-26
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\IP2\IP293834.AMD *See Previous Concurrence

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

OFFICE	PM:PDI-1	E	LA:PDI-1		OGC*		D:PDI-1			
NAME	JHarold:mw JH		SLittle		AHodgdon		SBajwa			
DATE	05/20/97		05/20/97		05/13/97		05/20/97		05/ /97	

Official Record Copy

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

May 20, 1997

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. M93834)

Dear Mr. Quinn:

The Commission has issued the enclosed Amendment No. 192 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 August 29, 1995, as supplemented by letters dated August 7, 1996, and January 10, 1997.

The amendment revises TSs to incorporate the commitments made in connection with Amendment No. 183, which authorized the installation of laser welded steam generator tube sleeves inside defective tubes in steam generators.

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,

Jefferey F. Harold

Jefferey F. Harold, 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. 192 to DPR-26
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:

Mayor, Village of Buchanan
236 Tate Avenue
Buchanan, NY 10511

Mr. F. William Valentino, President
New York State Energy, Research,
and Development Authority
Corporate Plaza West
286 Washington Ave. Extension
Albany, NY 12203-6399

Mr. Charles W. Jackson
Manager of Nuclear Safety and
Licensing
Consolidated Edison Company
of New York, Inc.
Broadway and Bleakley Avenue
Buchanan, NY 10511

Senior Resident Inspector
U. S. Nuclear Regulatory Commission
P.O. Box 38
Buchanan, NY 10511

Mr. Brent L. Brandenburg
Assistant General Counsel
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, NY 10003

Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Ms. Charlene D. Faison, Director
Nuclear Licensing
Power Authority of the State
of New York
123 Main Street
White Plains, NY 10601

Mr. Walter Stein
Secretary - NFSC
Consolidated Edison Company
of New York, Inc.
4 Irving Place - 1822
New York, NY 10003

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

Mr. Paul Eddy
New York State Department of
Public Service
3 Empire State Plaza, 10th Floor
Albany, NY 12223

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. 192
License No. DPR-26

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

   A. The application for amendment by Consolidated Edison Company of New York, Inc. (the licensee) dated August 29, 1995, as supplemented August 7, 1996, and January 10, 1997, 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:

9706020195 970520
PDR ADOCK 05000247
P PDR

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 192 , 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

S. Singh Bajwa

S. Singh Bajwa, Acting 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: May 20, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 192

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.1.F-1 thru 3.1.F-10	3.1.F-1 thru 3.1.F-9
4.13-1 thru 4.13-8	4.13-1 thru 4.13-7

F. REACTOR COOLANT SYSTEM LEAKAGE AND LEAKAGE INTO THE CONTAINMENT FREE VOLUME

Specifications

1. Leakage Detection And Removal Systems

a. The reactor shall not be brought above cold shutdown unless the following leakage detection and removal systems are operable:

(1) two containment sump pumps,

(2) two containment sump level monitors,

(3) a containment sump discharge line flow monitoring system,

(4) two recirculation sump level monitors,

(5) two reactor cavity level monitors,

(6) two of the following three systems:

(a) a containment atmosphere gaseous radioactivity monitoring system,

(b) a containment atmosphere particulate radioactivity monitoring system,

(c) the containment fan cooler condensate flow monitoring system.

b. When the reactor is above cold shutdown, the requirements of Specification 3.1.F.1.a may be modified as follows:

(1) One containment sump pump may be inoperable for a period not to exceed seven (7) consecutive days provided that, on a daily basis, the other containment sump pump is started and discharge flow is verified.

(2) One of the two required containment sump level monitors may be inoperable for a period not to exceed seven (7) consecutive days.

(3) The containment sump discharge line flow monitoring system may be inoperable for a period not to exceed seven (7) consecutive days provided a detailed Waste Holdup Tank water inventory balance is performed daily.

(4) One of the two required recirculation sump level monitors may be inoperable for a period not to exceed fourteen (14) consecutive days.

(5) One of the two required reactor cavity level monitors may be inoperable for a period not to exceed thirty (30) consecutive days.

(6) Two of the three monitoring systems specified in Specification 3.1.F.1.a.(6) may be inoperable for a period not to exceed thirty (30) consecutive days. If either of the radioactivity monitoring systems specified in Specification 3.1.F.1.a.(6) is inoperable, grab samples of the containment atmosphere shall be obtained and analyzed daily.

c. If the conditions of Specification 3.1.F.1.b cannot be met or an inoperable system(s) is not restored to operable status within the time period(s) specified therein, then either perform a visual inspection of containment once a shift, or place the reactor in the hot shutdown condition within the next 6 hours and, if the inoperability continues, place the reactor in the cold shutdown condition within the following 30 hours.

2. Operational ıkage 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 which does not contain tube sleeves. Primary to secondary leakage through the steam generator tubes and/or sleeves shall not exceed 150 gpd in any steam generator containing sleeves. 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.

(b) Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between the measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

(c) Leakage rates greater than 5.0 gpm are unacceptable.

(2) If any RCS/RHR pressure isolation valve listed in Specification 3.1.F.2.b.(1) is determined to be inoperable based on the acceptance criteria presented therein, an orderly plant shutdown shall be initiated and the reactor shall be placed in the cold shutdown condition within 24 hours.

c. <u>Total Reactor Coolant System Leakage</u>

(1) Whenever the reactor is above cold shutdown, reactor coolant system leakage shall be limited to:

(a) No pressure boundary leakage,

(b) 1 gpm unidentified leakage, and

(c) 10 gpm identified leakage.

(2) With any pressure boundary leakage, the reactor must be placed in hot shutdown within 6 hours and in cold shutdown within the following 30 hours.

(3) If the Reactor Coolant System leakage exceeds the limits in either c.(1)(b) or c.(1)(c) above, the leakage rate must be reduced to within limits within 4 hours or the reactor must be placed in hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

d. Leakage Into The Containment Free Volume

(1) Whenever the reactor is above cold shutdown, the total leakage into the containment free volume from both reactor coolant and non-reactor coolant sources combined shall not exceed 10 gpm.

(2) Notwithstanding the action which may be required by Specification 3.1.F.2.d.(3) below, with the combined leakage into the containment free volume greater than the above limit, the leakage rate must be reduced to within the specified limit within 12 hours or the reactor must be placed in cold shutdown within the following 36 hours.

(3) If 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 shall be placed in a cold shutdown condition within the next 36 hours unless the water level(s) is reduced below the specified limit(s).

(4) 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", immediately place the reactor in a subcritical condition and initiate an expeditious cooldown of the reactor to the cold shutdown condition.

Basis

Water inventory balances, monitoring equipment, radioactive tracing, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not, can be a serious problem with respect to in-plant radioactivity contamination and cleanup or it could develop into a still more serious problem; therefore, first indications of such leakage will be followed up as soon as practicable.

Although some leak rates on the order of gpm may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks on the order of drops per minute through any pressure boundary of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross pipe rupture.

If leakage is to the containment, it may be identified by one or more of the following methods:

a. The containment air particulate monitor is sensitive to low rates. The rates of reactor coolant leakage to which the instrument is sensitive are are within the recommended sensitivity guidelines of Regulatory Guide 1.45.

b. The containment radiogas monitor.

c. A leakage detection system collects and measures moisture condensed from the containment atmosphere by cooling coils of the main air recirculation units including leaks from the cooling coils themselves. This system provides a dependable and accurate means of measuring the total leakage from these sources. Condensate flows from approximately 1 gpm to 15 gpm per detector can be measured by this system. Condensate flows greater than 15 gpm can be determined using weir calibration curves. Condensate flows less than 1 gpm may be determined by periodic observation of the water accumulation in the standpipes of the condensate collection system.

d. Leakage detection via the containment sump level and discharge flow monitoring systems will determine leakage losses from all fluid systems to the containment free volume. Water collecting on the containment floor will normally be delivered to the containment sump via the containment floor trench system. Level monitoring of the containment sump is in part provided by two level instruments which actuate control room lights at discrete sump/containment water levels and provide an audible alarm for certain discrete levels within the containment sump. In addition, another level monitoring device provides a continuous level readout in the control room. When the water level in the containment sump reaches predetermined levels, one or both containment sump pumps will automatically start and pump the fluid out of containment to the liquid waste disposal system. Flow in the containment sump pump discharge line from containment to the Waste Holdup Tank is monitored on a continuous basis. Thus, monitoring of both flow indication systems will provide a positive means for determining leakage into the containment free volume.

e. Water may also collect in the recirculation sump and/or the reactor cavity depending on the size and location of the leak. However, under most circumstances, the containment sump will be filled prior to the recirculation sump filling and both sumps will be filled prior to water level increasing on the containment floor (EL. 46') sufficient to initiate filling of the reactor cavity. Level monitoring of the recirculation sump is provided by two level instruments which actuate control room lights at discrete sump/containment water levels and provide an audible alarm for certain discrete levels within the recirculation sump. In addition, another level monitoring device provides a continuous level readout in the control room. Level monitoring of the reactor cavity is provided by a single analog continuous level indication in the control room and by two separate and independent level switches, each of which actuates an audible alarm in the control room.

Total reactor coolant leakage can be determined by means of periodic water inventory balances. If leakage is into another closed system, it will be detected by the plant radiation monitors and/or inventory balances. Determined leakage rates are an average over the applicable surveillance interval. Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure detection of additional leakage.

The 10 gpm identified leakage limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of unidentified leakage by the leakage detection systems.

Pressure boundary leakage of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any pressure boundary leakage requires the unit to be promptly placed in cold shutdown. Primary system leakage through packing, gaskets, seal welds or mechanical joints is not considered to be pressure boundary leakage.

The leakage limit and surveillance testing for RCS/RHR Pressure Isolation Valves provide added assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA.

Leakage fr~ the RCS/RHR Pressure Isolation Val~ is identified leakage and will be considered as a portion of the allowed limit.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those limits found to result in negligible corrosion of the steam generator tubes. If stress corrosion cracking occurs, the extent of cracking during plant operation would be limited by limitation of steam generator leakage between the reactor 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 lower allowable leak rate of 150 gpd for a steam generator containing sleeved tubes is based on industry operating experience.

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 expeditious cooldown of the plant.

The above actions are necessary to (1) preclude accumulation of water inside containment so that if a LOCA were to occur safety-related equipment would not become submerged, (2) prevent the reactor cavity from becoming filled with water, (3) prevent the reactor vessel from being wetted while it is at an elevated temperature, and (4) prevent the immersion of the in-core instrument conduits. The amount of water estimated to be inside containment after actuation of the emergency core cooling system following a loss of coolant accident is approximately 423,000 gallons. This amount of water would, by itself, reach approximately EL. 50' 1". An additional 28,000 gallons (a total of approximately 451,000 gallons) would have to accumulate inside containment before any safety-related electrical component would be submerged (approximately EL. 50' 5"). The combined volume of the containment sump, the recirculation sump and the containment floor trenches is approximately 18,000 gallons. Since operator action is required by these specifications to shut the reactor down before these volumes are filled, sufficient margin between the water level inside containment following a loss of coolant accident and the level at which a safety-related electrical component may become submerged is maintained. Furthermore, since both sumps, the floor trenches and the containment floor up to EL. 46' 5 3/8" must be flooded (i.e., approximately 50,000 gallons) before the water level is sufficiently high to flood over the curb leading to the reactor cavity, the forementioned operator actions taken to preclude excessive flooding plus LOCA water levels will conservatively preclude flooding of the reactor cavity and subsequent wetting of the reactor vessel at an elevated temperature.

<u>References</u>

UFSAR Sections 6.7, 11.2.3 and 14.2.4

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, or sleeved 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 or sleeve wall thickness affected or removed by degradation.

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

g. Plug[illegible] Limit is the degradation depth at or [illegible]ond 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 1997 only, shall be conducted during the 1997 Refueling Outage which will commence no later than May 2, 1997. The scheduled examinations will be completed prior to return to service from the 1997 Refueling Outage.

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. 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. <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 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 a 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 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, 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 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.192 TO FACILITY OPERATING LICENSE NO. DPR-26

CONSOLIDATED EDISON COMPANY OF NEW YORK

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated August 29, 1995, as supplemented by letters dated August 7, 1996, and January 10, 1997, Consolidated Edison Company of New York (the licensee) requested an amendment to change the Technical Specifications (TSs) for Indian Point Unit No. 2 (IP2). The licensee proposed the amendment to incorporate items contained in a letter to the staff dated April 5, 1995, concerning the installation of laser welded sleeves inside of defective tubes in steam generators. The proposed changes would modify TS Section 4.13.A., Steam Generator Tube Inservice Surveillance; and TS Sections 3.1.F.2.a.(1), Primary to Secondary Leakage. The August 7, 1996, and January 10, 1997, letters provided clarifying information that did not change the initial proposed no significant hazards consideration.

IP2 uses four Westinghouse model 44 steam generators. The tube is fabricated with mill annealed alloy 600 material. Each tube has an outside diameter of 7/8 inches with a wall thickness of 0.050 inches.

2.0 EVALUATION

By letter dated April 13, 1994, as supplemented by letters dated December 20, 1994; January 12, January 31, March 17, and April 5, 1995; the licensee requested to use two Westinghouse-designed sleeving processes to disposition defective steam generator tubes. One process uses a laser welding technique to secure a sleeve inside a degraded tube. The other process uses mechanical expansion to secure the sleeve in position inside the degraded tube. In either process, the sleeve is designed to substitute as the reactor coolant pressure boundary for the degraded portion of the tube. At the time, the licensee informed the staff that the mechanical sleeve process would not be used. The NRC staff, therefore, only reviewed the laser welded sleeving process.

By letter dated February 15, 1995, the staff requested the licensee to address certain aspects of the sleeve installation. The staff requested that the licensee 1) use enhanced and improved eddy current inspection techniques as they are developed and verified for use; 2) perform post-weld heat treatment of installed laser welded sleeves; 3) perform additional confirmatory testing to establish the design life of the sleeves and to confirm that leakage

9706020199 970520
PDR ADOCK 05000247
P PDR

detection requirement will be satisfied; 4) amend the TSs to provide for appropriate inservice inspection for any tubes containing sleeves; and 5) amend the TS for primary-to-secondary leakage limits to account for any installation of sleeves into the steam generator tubes. In the April 5, 1995, letter, the licensee agreed to the above five items and stated that it would submit a license amendment request to incorporate items 4 and 5 into the plant TSs.

By letter dated May 19, 1995, the staff approved the tube sleeving process for IP2 technical specifications as documented in License Amendment No. 183. The staff concluded that the licensee's technical justification for using laser welded sleeves was acceptable.

In the August 29, 1995, letter, the licensee proposed to formally incorporate the aforementioned items 4 and 5 into the IP2 TSs. During its review, the staff raised two issues regarding the wording of the licensee's proposed TSs. One was related to the specific area of inspection and the other was related to the expansion criteria of the sleeve inspection.

By letter dated July 29, 1996, the staff provided comments on the proposed wording and requested clarification from the licensee. The staff recommended that sleeve joint inspections include the original tube in addition to the sleeve and that the sample expansion of sleeve inspection be based on pluggable indications on the sleeve joint not just in the sleeve. The staff's recommendation was based on industry experience that shows that the most likely area for degradation is in the sleeve-to-tube joint. In its August 7, 1996 letter, the licensee revised the wording in TS Section 4.13.A.3.c to include the staff's recommendations.

With respect to the criteria for expanding sleeve inspection, the proposed TS Section 4.13.A.4.1.d in the August 29, 1995 letter stated that "..(i)f a 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." The staff requested the licensee to clarify the proposed expansion criteria because it was unclear whether this wording applied to the sleeves in one steam generator or all the steam generators. Subsequently, in the January 10, 1997, letter, the licensee revised the wording in TS Section 4.13.A.4.1.d to state that "...(i)f a tube sleeve or associated sleeve to tube joint region exhibits degradation of greater than 23% of wall thickness or is otherwise defective, all unsampled sleeves shall be examined." The staff finds that the revised wording clarifies the staff's concern and is acceptable because a single defective sleeve found in the first sample will result in expansion of the inspection to the sleeves in all the steam generators.

In TS section 3.1.F.2.a.(1), the licensee proposed to limit primary-to-secondary leakage through the steam generator tubes and/or sleeves to 150 gallons per day in any steam generator containing sleeves. The staff finds that the proposed limit on primary-to-secondary leakage is acceptable because it provides improved defense-in-depth for assuring leakage integrity during normal operation and is consistent with previous staff recommendations for primary-to-secondary leakage for plants that repair tubes by sleeving.

3.0 CONCLUSION

Based on the staff review of the information submitted, the staff concludes that the licensee proposed revisions to IP2 TSs regarding the inspection of laser welded sleeves and primary-to-secondary leakage monitoring are acceptable. The licensee may incorporate the proposed changes into the TSs for IP2.

4.0 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.

5.0 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. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent 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 (60 FR 56365). 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.

6.0 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: J. Tsao

Date: May 20, 1997