May 19, 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. 183 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, January 31, March 17, and April 5, 1995.

The amendment revises the TS Sections 3.1.F and 4.13 to allow the repair of steam generator tubes by sleeving using laser welded sleeves. The April 13, 1994, application also proposed a new tube acceptance criteria, F*, which would allow tubes that are degraded in a location not affecting structural integrity of the tube to remain in service. On March 13, 1995, we issued License Amendment 180, which changed the TS to allow use of the F* criteria.

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 Daniel McDonald for:
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. 183 to DPR-26
2. Safety Evaluation

cc w/encls: See next page

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

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for Francis J. Williams, Jr., Project Manager
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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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Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
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DATED: May 19, 1995

AMENDMENT NO. 183 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
S. Little
F. Williams
OGC
D. Hagan, T-4 A43
G. Hill (2), T-5 C3
C. Grimes, 11/E/22
H. Conrad, 7/D/4
R. Hermann, 7/D/4
ACRS (4)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

260041

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. 183
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, January 31, March 17, and April 5, 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:

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(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 183, 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

LBMarsh

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: May 19, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 183

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Revise Appendix A as follows:

Remove Pages	Insert Pages
4.13-1	4.13-1
4.13-2	4.13-2
4.13-5	4.13-5

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, 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. 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.

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, 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 183 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, January 31, March 17, and April 5, 1995, the Consolidated Edison Company of New York, Inc. (Con Ed) 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 (SG) tubes via (1) the implementation of an F* criteria and (2) by sleeving. The supplemental letter dated January 31, 1995, requested that the NRC staff proceed with the F* criteria changes and to consider the sleeving changes separately, while Con Ed took the time necessary to respond to a staff Request for Additional Information (RAI) issued on February 15, 1995, which was related to sleeving. The January 31, 1995, submittal also provided revised TS pages to conform to the sleeving delay request. On March 13, 1995, License Amendment 180, which changed the TS to allow use of the F* criteria was issued. This Safety Evaluation (SE) covers the sleeving method of tube repair.

Con Ed's request of April 13, 1994, requested approval of sleeve repair by two Westinghouse processes. The first uses a laser welding technique to secure a sleeve inside a degraded steam generator tube. The second process uses mechanical expansion to secure the sleeve in position inside the SG tube. In either case, the sleeve is designed to substitute as the reactor coolant pressure boundary (RCPB) for the degraded portion of the SG tube. The supplemental letter dated March 17, 1995, acknowledged the RAI and the April 5, 1995, supplement responded to the RAI by committing to meet all NRC staff requirements for use of sleeved tubes. In addition, it advised that Con Ed did not intend using the mechanical sleeve process at this time. The subsequent NRC staff review, therefore, did not include the mechanical sleeving process and the attached SE only addresses the laser welded sleeving process.

The technical justification supporting the Westinghouse laser welded sleeve process is given in WCAP-13088, Rev. 3, "Westinghouse Series 44 and 51 Steam Generator Generic Sleeving Report, Laser Welded Sleeves" (Proprietary), and WCAP-13583, Rev. 1, "Specific Application of Laser Welded Sleeves for Indian Point 2 Steam Generators" (Proprietary). The NRC staff found that Con Ed's technical justification for using laser welded sleeves is within the Indian

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Point 2 licensing basis. The Indian Point 2 licensing basis as it relates to sleeving repairs of defective SG tubes is provided in the Attachment to this SE. The NRC staff's evaluation of Con Ed's submittal is summarized in Section 3.0 of this SE.

As stated above, the March 17, and April 5, 1995, supplements acknowledged and responded to the RAI and advised that only laser welded sleeves would be used. In addition, the April 5, 1995, supplement provided revised TS page changes. These supplements provided clarifying information which did not change the initial proposed no significant hazards consideration and was within the scope of the original Federal Register notice. The supplements dated December 20, 1994, January 12, 1995, and January 31, 1995, related to the F* criteria and were addressed in License Amendment 180 issued on March 13, 1995.

2.0 DISCUSSION

2.1 Background

Tubes in an operating pressurized water reactor (PWR) steam generator can be degraded by attack mechanisms, such as primary water stress corrosion cracking (PWSCC), outer diameter stress corrosion cracking (ODSCC), intergranular attack (IGA), or pitting, or by other phenomena as denting and vibration induced wear. Tubes that become excessively degraded reduce the integrity of the primary-to-secondary pressure boundary and must be removed from service or repaired. Degradation of SG tubes has typically been monitored using eddy current testing (ECT) techniques. TSs have historically required that SG tubes be plugged at both the inlet and outlet ends of the tubes, when tubes are determined to have degraded below a calculated minimum wall thickness value (termed the "plugging limit"); however, installing plugs in a SG tube reduces the heat transfer surface area available for reactor core cooling. For this reason, design restrictions limit the total number of SG tubes, which may be plugged in any one SG during the lifetime of a plant.

Alternatively, SG tubes experiencing localized degradation can be fitted with sleeves over the degraded area to reestablish the integrity of the RCPB. The sleeves are expanded and sealed inside the tubes to provide an acceptable leak resistant load-carrying path. The reductions in heat transfer area and primary flow caused by sleeving are slight in comparison to that caused tube plugging. Since the sleeve now serves as the RCPB in the vicinity of the degraded portion of the tube, ECT methods have now been developed for monitoring the extent of SG sleeve degradation. However, licensees are required to submit amendments to their TS for review and approval before the NRC will authorize a given sleeving technique as an acceptable SG tube repair method. Licensees typically implement the TS amendment by referencing the generic or plant-specific sleeving topical report(s) in the appropriate SG Limiting Condition for Operation (LCO) or Surveillance Requirement (SR).

In this case, where a licensee has amended its TS to allow for sleeving of SG tubes, Con Ed may use either plugging or sleeving as a SG tube repair method. Should subsequent ECT measurements of installed sleeves indicate that a sleeve

has degraded beyond the plugging limit for its design, the TS would then require that the defective SG tube be plugged and removed from service. The tube and sleeve plugging limits conservatively account for the uncertainties in ECT measurements and contain additional margins for expected or postulated degradation during service.

Westinghouse has installed laser welded sleeves in recirculating SG tubes in the United States and European nuclear plants since 1988. The sleeves are of two basic designs: tubesheet sleeves and tube support plate sleeves. Tube degradation historically has occurred in the tube adjacent to the tube support plates, at the top of the tubesheet, or within the section of tube adjacent to the tubesheet. Laser welded tubesheet sleeves span from the end of the tube, at the bottom surface of the tubesheet, to a point above the secondary side surface of the tubesheet. Tubesheet sleeves can vary from 27 to 36 inches in length. In comparison, the Indian Point 2 tubesheets are approximately 22-inches thick. Sleeve installation criteria require that the sleeve to tubesheet joint be located a specified minimum distance from the degradation. In contrast, laser welded sleeves installed at the tube support plate are 12 inches in length, and are approximately centered about the tube support plate intersection. Tube degradation at the tube support plate intersection is generally confined to the thickness of the tube support plate.

Tubesheet sleeves are secured by first performing a hydraulic expansion of the upper and lower portions of the sleeve. The hydraulic expansion serves to bring the sleeve in contact with the parent tube to optimize weld performance. A laser produced weld is implemented in the area of the hydraulic expansion region of the upper joint. This weld structurally supports the sleeve in addition to forming a seal.

At the tubesheet lower joint, a mechanical roll expansion is performed, which provides sleeve structural integrity during all plant conditions. Each joint is designed to meet the structural integrity requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III.

A structural analysis of the sleeve and sleeve joints using bounding temperature and pressure differences for Series 44 and 51 SGs was performed. Corrosion testing and both mechanical and leak testing of prototypic sleeve test specimens was also performed. The results of these evaluations and test programs are provided in the documents referenced in Section 1.0 to this SE.

2.2 Licensee Commitments

Recent experience at operating plants has emphasized the sensitivity of the Alloy 600 parent tube material to stress corrosion cracking when unfavorable residual stresses are introduced by processes such as sleeving. For this reason, the NRC staff position on sleeving considers sleeving unable to assure an unlimited service life for a repaired tube. The conservative view is that sleeving potentially creates new locations in the parent tube, which may be susceptible to cracking after new incubation times are expended. Incubation

times have been observed to vary between individual SGs and the various conditions of tubes within them. According to Con Ed, there have been no instances of parent tube cracking attributable to the Westinghouse laser welded sleeving process reported on installations dating back to 1988, when 54 laser welded sleeves were installed in an European plant. More recently, 860 sleeves were installed in the United States between 1992 and 1994. All laser welded sleeves were given a postweld heat treatment (PWHT). Con Ed has committed to the use of a PWHT and to performing additional confirmatory testing to determine the design life of the sleeves and to confirm that leakage detection requirements are met. Con Ed further committed to amending its license requirements to provide for appropriate inservice inspection for tubes containing sleeves and for license requirements pertaining to primary-to-secondary leakage limits to account for the installation of any sleeves into the SG. It will administratively limit primary-to-secondary leakage to no more than 150 gallon per day for any SG containing sleeves.

3.0 EVALUATION

3.1 Sleeve Design and Analysis

The laser welded sleeves (both tubesheet and tube support plate sleeves) have been analyzed to maintain SG tube and leak integrity during all plant conditions. The sleeve joint designs have been qualified through laboratory testing and analysis. Analytical verification has been performed using design and operating transient parameters selected to envelop loads imposed during normal operation, upset and accident conditions for all plants with Westinghouse Series 44 and 51 steam generators, (which includes Indian Point 2).

The sleeves are manufactured from thermally treated ASME SB-163, Alloy 690. The function of the sleeve is to restore the integrity of the RCPB in the region between the sleeve joints to a level, which is consistent with the original tube. The sleeve has been designed according to Section III, of the ASME Code. Fatigue and stress analyses of sleeved tube assemblies have been completed in accordance with the requirements of the ASME Code, Section III. The analyses include a primary stress intensity evaluation, primary plus secondary stress intensity range evaluation, and a fatigue evaluation for mechanical and thermal conditions which envelop the loading conditions for the Indian Point 2 SGs. For all analyzed conditions, the calculated stress levels and fatigue usage factors for both the sleeve and weld were found to be bounded by the ASME Code allowable values. The analysis conditions consider bounding values, which envelop the operating regimes of all Westinghouse Series 44 SGs.

The material used for sleeving, Alloy 690 TT (thermally treated) has been demonstrated to be highly resistant to intergranular stress corrosion cracking (IGSCC) under SG conditions. The resistance of the laser welded sleeve joint to inservice cracking also depends on the resistance of the parent Alloy 600 tubing to IGSCC. As mentioned previously, stresses in the tubing, either service operating stresses or residual stresses, can potentially cause

cracking. Two sources of residual stresses are related to hydraulic expansion during sleeve placement and to stresses introduced as a result of welding.

A testing program was conducted under conditions, which accelerate corrosion in SG materials to simulate long term steam generator service. Each test contained a rolled to unrolled SG tube transition, which served as a control sample. The stress levels in the control sample were representative of the residual stress conditions in the hard rolled transitions found at the top of the tubesheets in operating SGs.

The accelerated corrosion testing was performed in high temperature-high pressure autoclaves. A primary-to-secondary pressure differential, which bounds the conditions in the Indian Point 2 SGs was applied, and a doped steam environment was utilized to accelerate crack propagation. Results of the accelerated corrosion tests indicate that laser welded sleeve joints with post-weld heat treatment have a resistance to corrosion of greater than 10 times compared to the as-welded joint. Con Ed has committed to the use of the post weld stress relief heat treatment for the laser welded sleeving method.

3.3 Sleeve Plugging Limits

3.3.1 Sleeve Minimum Wall Thickness Determination

The sleeve minimum acceptable wall thickness is determined using the criteria of Regulatory Guide (RG) 1.121, ASME Code Section III allowable material strength values, and the pressure stress equation of Section III of the ASME Code. With respect to the design of the sleeve, the limiting criterion of RG 1.121, which applies to partial through-wall degradation, is such that, under normal operating conditions, the sleeve wall must maintain a safety factor of three against tube failure. In addition, the ASME Section III stress limit rules for upset conditions require that the membrane stress should not exceed the yield stress at temperature.

A bounding set of input conditions, which enveloped the operating parameters for all Series 44 and 51 SGs, was used to determine the minimum wall thickness. The minimum acceptable sleeve wall thickness was determined to be 55% of the original sleeve wall thickness. For postulated accident condition loadings, the calculated minimum wall thickness was determined to be bounded by that calculated using normal operating condition parameters. Similarly, the minimum wall thickness calculated for the loading conditions during a loss of load transient, which has the highest pressure differential of all transients during upset conditions, was also determined to be bounded by that calculated using normal operating condition parameters. Therefore, the minimum acceptable tube wall thickness that satisfies RG 1.121 criteria, based on generic conditions, was correctly determined to be 55% of the nominal sleeve wall thickness.

3.3.2 Sleeve Plugging Limit Determination

According to RG 1.121 criteria, an allowance for non-destructive evaluation (NDE) uncertainty and operational growth of tube wall degradation within the sleeve must be accounted for when using NDE to determine sleeve plugging limits. Therefore, a conservative tube wall combined allowance for postulated degradation growth and eddy current uncertainty of 20% through-wall per cycle have been assumed for the purpose of determining the sleeve plugging limit.

The sleeve plugging limit, which was calculated based on bounding generic conditions, and which would require the plugging of sleeved tubes if exceeded, was calculated to be 25% of the sleeve nominal wall thickness. In contrast to the generic plugging limit, the sleeve plugging limit for Indian Point 2 was calculated to be 23% of the sleeve nominal wall thickness. The difference in allowable through-wall extent is due to a larger primary-to-secondary pressure differential during operating conditions for Indian Point 2, as compared to the generic analysis. For Indian Point 2, a primary-to-secondary pressure differential of 1600 psi (primary pressure = 2250 psia, secondary pressure = 650 psia) was used to determine the minimum allowable wall thickness. This pressure differential will bound all normal and faulted/upset plant conditions. Removal of tubes and/or sleeves from service when degradation reaches the plugging limit provides assurance that the minimum acceptable wall thickness will not be exceeded during the next subsequent cycle of operation.

3.4 Leakage

Historically, the NRC has not required that sleeve repairs be leak-tight (only leak-limiting). The basis for this is that the first SG sleeving systems utilized sleeve to tube joints, which were basically mechanical seals, and which were not considered intrinsically leaktight. Therefore, the NRC staff does not require that sleeves be qualified for leak-tightness; rather, the staff determines whether the sleeves will meet the requirements for primary to secondary leakage limits under normal and accident conditions contained in the plant TSs.

While the laser weld should be inherently leak-tight, in practice, the lower joint of a tubesheet sleeve may be installed with or without a seal weld; therefore, the leakage characteristics must be considered. Con Ed has analyzed the effects of an abnormal tubesheet sleeve lower joint seal weld because of the inability of the ultrasonic testing inspection process to confirm the width of the fusion zone. The analysis shows that even under extreme postulated conditions, it will have satisfactory leakage integrity.

3.5 Non-Destructive Examination

The welding parameters are computer controlled. The essential variables, in accordance with the ASME Code are monitored and documented to produce repeatability of the weld process. In addition, the NDE of the laser welded sleeves utilizes two techniques. UT is performed after welding to confirm that the laser welds placed in service are consistent with critical process

dimensions and are of acceptable weld quality. Con Ed has presented data on an UT system that demonstrates post weld examinations of the sleeve-tube assembly will be adequate. Standards, which simulated a leakage path along the weld metal were used in qualification of the UT technique. The results of the qualification tests demonstrate that the system can confirm that there is a metallurgical bond between the sleeve and the tube and determine if a leak path exists across the weld.

Eddy testing (ET) is used to establish baseline inspection data for every installed sleeve/tube to be utilized during subsequent inservice inspections. Furthermore, in accordance with the criteria in RG 1.83, Revision 1, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," Con Ed has presented data, which demonstrates that structural integrity of sleeved tubes can be monitored by performing periodic present eddy current examinations. The eddy current technique qualified involved a double crosswound eddy current coil, which minimizes the effects of geometry and weld zone changes. However, for future inservice inspections, Con Ed, in its letter of April 5, 1995, has committed to utilize enhanced and improved inspection techniques as they are developed and verified for use. Con Ed described a number of proprietary advanced inspection techniques that are currently under development, and stated that alternate inspection techniques may be applied as they become available, as long as they can be demonstrated to provide the same or a greater degree of inspection accuracy as the method described in the reports submitted to and accepted by the NRC staff.

It should be noted that inspections of installed sleeves necessitate the use of an eddy current bobbin probe, which is of sufficiently small outer diameter (OD) to allow the probe to pass through the sleeve ID. This probe diameter, however, is not optimized for the span between the sleeves (reduced "fill-factor"). Con Ed has committed that for any tube indication in this area, a further inspection will be performed by an alternate technique, such as a surface riding probe, in order to determine the acceptability of the sleeved tube for further service. In addition, Con Ed has committed that, for any change in the eddy current signature of the sleeve or sleeve/tube joint region, a further inspection will also be performed by an alternate eddy current technique in order to determine the acceptability of the sleeved tube for continued service.

3.6 Sleeving of Previously Plugged Tubes

In the event that previously plugged tubes are unplugged and returned to service by using the sleeving process, the TSs would require that the sleeving requirements be applied to the tubes designated for sleeving. This includes provisions to ensure that the new sleeve joints are located a minimum acceptable distance apart from the degraded tube area. Following installation, a new "baseline" inspection of the tube and sleeve would then be required for any sleeved tube placed back in service.

4.0 CONCLUSION

The effects of sleeving, as evaluated by Con Ed have been shown to not adversely affect the dose consequence estimates required by 10 CFR Part 100 and established in the Indian Point updated final safety analysis report (UFSAR). Con Ed has shown that installation of tube sleeves, based on test results, will not increase the offsite dose consequences calculated for a postulated main steam line break event, the design basis accident for the plant. Con Ed has shown that the consequences of a SG tube rupture event, as evaluated in the Indian Point UFSAR, are also unaffected by tube sleeving. Therefore, the NRC staff concludes that Con Ed's amendment request, as supplemented, will not result in: (1) a significant increase in the probability or consequence of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously analyzed; or (3) result in a significant reduction in a margin of safety, and therefore meet the acceptance criteria for implementing TS amendments, as delineated in 10 CFR 50.59(c), 50.90, and 50.92.

Based on the preceding analysis, the NRC staff concludes that at Indian Point 2, the repair of SG tubes using laser welded sleeves in accordance with the proposed amendment, as supplemented, is acceptable based on the commitments made by Con Ed in their April 5, 1995, submittal to: (1) using enhanced and improved eddycurrent testing inspection techniques as they are developed and verified for use; (2) performing post-weld heat treatment of installed laser welded sleeves; (3) performing additional confirmatory testing to establish the design life of the sleeves and to confirm that leakage detection requirement will be met; (4) amending the license requirements to provide for appropriate inservice inspection for any SG tubes containing sleeves; and (5) amending the license requirement for primary-to-secondary leakage limits to account for any installation of sleeves into the SGs. Con Ed committed to submitting a license amendment request for items 3, 4, and 5 by September 1, 1995, and until a license amendment is issued they have committed to administratively limit primary-to-secondary leakage to no more than 150 gpd for any SG containing sleeves.

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. 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: H. Conrad

Date: May 19, 1995

Attachment: As stated

DISCUSSION OF LICENSING BASIS

1. 10 CFR 50.55a

10 CFR 50.55a, "Codes and Standards," requires that components, which are a part of the reactor coolant pressure boundary to be built to the requirements of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). 10 CFR 50.55a also requires that throughout the service life of plant, that licensees meet the inservice inspection requirements of the ASME Code Section XI for ASME Code Class 1, 2, or 3 components.

Section 5.2.1.1 of the Standard Review Plan, entitled "Compliance with the Codes and Standards Rule, 10 CFR 50.55a," provides an outline of the standards used for evaluation by the NRC staff. Any modification, repair or replacement of these components must also meet the requirements of the ASME Code to assure that the basis on which the unit was originally evaluated is unchanged.

2. ASME Code Requirements

The design of the sleeves is predicated on the requirements of the ASME Code Section III, Subarticles NB-3200, "Analysis" and NB-3300, "Wall Thickness." The ASME Code provides criteria for evaluation of the stress levels in the tubes for design, normal operating, and postulated accident conditions. The margin of safety is provided, in part, by the inherent safety factors in the criteria and requirements of the ASME Code.

Section IX of the ASME Code, Subsection QW, and Section III, including Code Case N-395, define the applicable essential variables for the welding procedure specification and welding procedure qualification test.

Section XI, IWB-4334 of the ASME Code defines the extent of examination requirements for installation of laser welded sleeves.

3. Regulatory Guide 1.121

Regulatory Guide 1.121, issued for comment, entitled "Bases for Plugging Degraded PWR Steam Generator Tubes," addresses tubes with defects. The criteria of Regulatory Guide 1.121 are extended to the laser welded sleeve in order to determine the level of degradation, which will require removal of the sleeve from service by plugging. ASME Code allowable strength values were used for this evaluation. By utilizing the requirements for sleeve design according to the ASME Code and Regulatory Guide 1.121 to define acceptance criteria, the sleeve meets

the requirements of General Design Criterion (GDC) 14, "Reactor Coolant Pressure Boundary," GDC 15, "Reactor Coolant System Design," and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

4. Regulatory Guide 1.83

Regulatory Guide 1.83, "Inservice Inspection of Pressurizer Water Reactor Steam Generator Tubes" (and the Indian Point Unit 2 Technical Specifications) is used as the basis to determine the inservice inspection requirements for the sleeve.

5. 10 CFR Part 100

Total plant allowable primary-to-secondary leakage rates, derived from the requirements of 10 CFR Part 100, are determined on a plant specific basis. Offsite doses during either a main steam line break or tube rupture event are not to exceed a small fraction of the 10 CFR 100 limits per the Bases to the Indian Point Unit 2 Technical Specifications.