ATTACHMENT A

PROPOSED TECHNICAL SPECIFICATIONS

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

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g. Plug[illegible]g 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 1999 only, shall be conducted during the 2000 Refueling Outage which will commence no later than June 3, 2000. The scheduled examinations will be completed prior to return to service from the 2000 Refueling Outage.

B. ACCEPTANCE CRITERIA AND CORRECTIVE ACTION

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

   a. depth of degradation is less than:

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

   AND

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

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

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

C. REPORTS AND REVIEW OF RESULTS

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

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

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

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

ATTACHMENT B

SAFETY ASSESSMENT

AND

BASIS FOR NO SIGNIFICANT HAZARDS
CONSIDERATION DETERMINATION

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SAFETY ASSESSMENT

BACKGROUND

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

REQUESTED CHANGE AND PURPOSE

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

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

JUSTIFICATION FOR REQUESTED CHANGES

BASES

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

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

identifying and removing from service defective steam generator tubes.

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

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

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

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

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

- WATER CHEMISTRY - The industry guide to water chemistry control is the EPRI Primary and Secondary Chemistry Guidelines. In October of 1997 the steam generators were placed in wet lay-up by adding the appropriate quantities of ammonium hydroxide and carbohydrazide. Carbohydrazide was added as an oxygen scavenger in place of hydrazine. The industry has recently recognized that carbohydrazide is a much more effective oxygen scavenger than is hydrazine at the cold temperatures of wet lay-up. In addition, carbohydrazide does not pose the industrial hygiene concerns that hydrazine does. During the outage when conditions permitted each of the steam generators were sparged with nitrogen for one hour each day to drive off any air that may have entered the steam generator gas space. With the exception of the time period during which maintenance activities associated with the blowdown isolation and main steam safety valves did not permit nitrogen sparging (16% of the total number of outage days), the steam generators were required to be sparging. Routine sampling and analysis of the lay-up solution indicated that the lay-up solution was maintained at an acceptable pH during the outage. A byproduct of the carbohydrazide and oxygen reaction is the formation of carbon dioxide. A slight depression of the pH of the lay-up solution was measured as a result of the dissolved carbon dioxide in the lay-up solution. The concentrations of impurities, including dissolved oxygen, were maintained well below industry

recommended maximums while the steam generators were in a lay-up condition. This careful control of impurities provides a measure of added confidence that there was no significant change in the condition of the steam generator tubes during the cold shutdown period.

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

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

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

CONCLUSION

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

BASIS FOR NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

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

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

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

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

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

proposed change to the Technical Specification.

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

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

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

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