William J. Cahill, Jr.
Vice President

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4 Irving Place, New York, N Y 10003
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February 28, 1979

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

Director of Nuclear Reactor Regulation
ATTN: Mr. A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Schwencer:

Transmitted herewith are forty (40) copies of Supplement No. 3 to the "Inservice Inspection and Testing Program - Indian Point Nuclear Generating Unit No. 2". This supplement revises portions of our August 1977 program description including Supplements Nos. 1 and 2 dated September and October 1977, respectively. Attachment A to this letter summarizes these changes and their locations. In addition, Attachment B to this letter identifies certain requirements which we believe are impractical.

The fourth 20-month pump and valve testing period will begin on July 1, 1979. The applicable version of the ASME Code Section XI referenced in 10 CFR 50.55a(b)(2) on January 1, 1979 was the same version that is presently in effect for the third 20-month inspection period. Therefore, no code updating changes are necessary for the fourth 20-month inspection period. Accordingly, the testing program in effect for the third 20-month inspection period applies to the fourth 20-month inspection period as well. When the NRC Regulatory Staff issues their final approval of the Indian Point Unit No. 2 Inservice Inspection and Testing Program, this final approval will apply to both the inservice inspection program and the inservice pump and valve testing program at least through February 28, 1981.

Should you or your staff have any questions regarding our inservice inspection and testing program or the information contained herein, we will be pleased to discuss them with you.

Very truly yours,

William J. Cahill, Jr.

William J. Cahill, Jr.
Vice President

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<u>ATTACHMENT A</u>

Summary of
Supplement No. 3

Consolidated Edison Company of New York, Inc.

Indian Point Unit No. 2

Docket No. 50-247

February, 1979

Summary of Supplement No. 3

1. Alternate scheduling plans are provided for ultrasonic examination of reactor vessel welds if a malfunction of the automated reactor vessel inspection tool causes undue delay.

   Reference: Page 4

2. Ultrasonic examination of welds which have irregular "as welded surface conditions" is discussed and the documentation of such conditions is stipulated.

   Reference: Pgs. E1-3 Item B2.4
   E1-4 Item B3.3
   E1-8 Items B4.5, B4.6
   E1-17 Note 50
   E2-6 Items C2.1, C2.2, C2.3
   E2-13 Note 33

3. Alternate acceptance criteria are provided for integrally welded supports which may exhibit indications attributed to non-service related causes.

   Reference: Pgs. E1-8 Item B4.9
   E1-17 Note 49
   E2-2 Item C1.3 - Volume Control Tank
   E2-3 Item C1.3 - Reactor Coolant Filter
   E2-3 Item C1.3 - Seal Water Injection Filters
   E2-4 Item C1.3 - Seal Water Return Filter
   E2-6 Item C2.5
   E2-13 Note 32

4. Revisions to the inservice pump testing program are provided to reflect:

   a. alternate pump test schedules for both normal plant operation and shutdown conditions.

      Reference: Pgs. 8
      E5-1 through E5-5

   b. an alternate method for establishing bearing temperature.

      Reference: Pgs. E5-4
      E5-5
      E5-7 Note 5

   c. an alternate approach to establishing acceptance criteria for hydraulic reference quantities.

      Reference: Pgs. E5-4
      E5-5
      E5-7 Note 7

d. clarification of exceptions to the measurement/observation of lube oil level or pressure.

Reference: Pgs. E5-4
E5-5
E5-7 Note 8

5. Alternate means for establishing weld reference datum points are provided.

Reference: Pg. 5

6. Revisions to the inservice valve testing program are provided to reflect:

a. revised applicability of the Section XI code.

Reference: Pg. 1

b. alternate valve test schedules for both normal plant operation and shutdown conditions.

Reference: Pgs. 8
E6-1 through E6-2

c. clarification of valve stroke time measurement.

Reference: Pg. E6-1

d. addition of certain valves resulting from plant modifications.

Reference: Pgs. E6-4
E6-7

e. alternate testing requirements for certain relief valves.

Reference: Pgs. E6-9
E6-10

7. Various editorial and typographical oversights contained in previously transmitted documents are corrected and where applicable, more timely information is presented.

Reference: Pgs. 2
3
8
E1-1 Item B1.1
E1-2 Item B1.16
E1-11 Note 4
E1-12 Note 6
E1-13 Note 17
E1-15 Note 35
E1-16 Note 47

E2-1	Items C1.1, C1.2, C1.4 - steam generators
E2-1	Items C1.1, C1.2, C1.4 - residual ht xchgrs
E2-6	Items C2.2, C2.4, C2.5, C2.6
E2-8	Item C4.2
E2-9	Item C4.4
E5-6	Note 1
Encl. 5	Figure E5-1
Encl. 5	Figure E5-2
E6-3	Category A Valves
E6-5	Category A Valves
E6-7	Category B Valves
E6-8	Category B Valves
E6-9	Category C Valves
E6-10	Category C Valves
E6-11	Category C Valves

ATTACHMENT B

Inservice Examinations

Consolidated Edison Company of New York, Inc.

Indian Point Unit No. 2

Docket No. 50-247

February, 1979

Inservice Examinations

Your October 28, 1977 letter granted interim approval of our proposed inservice inspection and testing program and required that we comply with the existing Technical Specifications and our proposed program dated August 3, 1977 as supplemented September 22, 1977. The effect of this interim approval is to require particular inspections, during the current forty month period, of the reactor vessel closure head cladding and reactor vessel nozzle safe ends which we believe are impractical and/or unwarranted. The background details and justification for not performing these examinations are presented in the following pages. Accordingly, performance of these examinations during the current forty-month inspection period (11/1/77-2/28/81) is not planned at this time.

Reactor Vessel Closure Head Cladding

1. Inspection method(s) - PT&V

2. Source Requirement -

   1) Present Plant Technical Specification, Section 4.2, Item 1.13

   2) Section XI-1974 edition through Summer 1975 addenda, Item B1.13

3. Basis for exception

   The cladding's intended function is to minimize corrosion products not to provide structural support. Potential surface indications in cladding are generally of little or no significance. These considerations have been recognized in later editions of Section XI as evidenced by the deletion of these cladding examinations from the code effective with the Summer 1976 addenda. Furthermore, a laydown area suitable for visual and liquid penetrant examination of the closure head cladding is not currently available. Modification to the existing head laydown stand to support these examinations would require the removal of six concrete sections to provide access to each clad patch requiring examination. In addition, these cladding examinations typically involve relatively high personnel exposure levels due to the radiation field emanating from the cladding in combination with the focusing effect of the concave closure head internal surface. Nominally, a 4 R/hr. field has been measured.

4. Alternate examinations

   Since potential surface indications of the cladding are generally of little or no significance, an alternate examination intended specifically to detect such indications is of little technical value. However potential propagation of such indications into

base material is an important consideration. Volumetric examination of the circumferential and meridional welds in the closure head (from the head O.D.) is currently performed as required by the code. These examinations will detect such potential indications in the base material.

5. Conclusion

Based on the questionable value of these examinations, latest code requirements, lack of a suitable laydown area, required modifications and expected personnel exposure levels associated with these modifications and inspections, we conclude that performance of the required visual and penetrant examinations are neither desirable nor justified. Furthermore, a practical examination is already required to be implemented. Exception is therefore taken to performing the required visual and liquid penetrant examinations of the reactor vessel closure head cladding.

Reactor Vessel Nozzle Safe Ends

1. Inspection method(s) - Visual and Surface

2. Source Requirement(s) -

   Visual - Present Plant Technical Specifications - Section 4.2, Item 1.7

   Surface - Section XI - 1974 edition through Summer 1975 addenda, Item B1.6

3. Basis for exception

Visual and surface examination of the reactor vessel nozzle safe ends are precluded by fixed non-removable insulation about the nozzle circumference, extremely limited physical access and high radiation fields.

Plant design provides only limited access to these welds through removable sand plugs located in the refueling cavity floor. The sand plug cavity is a very confining box-like structure measuring 36" wide x 16" length x 60" depth set into the biological shield. The nozzle to safe end welds are located at the bottom of their respective sand plug cavities, approximately five feet below the refueling cavity floor. Access to the top 90° of nozzle/pipe surface is afforded with the plugs removed but exposure levels in the sand plug cavity are approximately 150 mrem/hr. Protective clothing and breathing apparatus are required particularly since the radiation fields where personnel would be located (i.e., the refueling cavity floor near the reactor vessel flange) will be higher than in the sand plug cavity. Due to the sand plug cavity configuration there is effectively no access to the bottom 270° of nozzle/pipe surface area.

Rigid insulation covering the nozzle to safe end welds precludes any examination requiring visual or physical contact with the nozzle/pipe surface and plant design imposes severe access restrictions to its removal. Installation of a removable insulation panel providing access to the top 90° of nozzle/pipe surface would require an estimated total exposure exceeding 10 man-rem. Removal of the existing insulation to provide access to 360° of the nozzle/pipe surface as required by code and replacement with removable insulation panels would require significantly higher exposure levels. Finally even if removable insulation panels were installed, both visual and surface examination would require the use of remote application and observation equipment for the bottom 270° of nozzle/pipe surface due to the 4" annular clearance between the nozzle/pipe surface and biological shield.

4. Alternate Examinations

Volumetric examination of the nozzle to safe end welds as required by the code, using the automated reactor vessel inspection tool from the nozzle bore (I.D.) is a practical alternative to the visual and surface examinations. This technique supports the detection of indications throughout the nozzle/pipe thickness including the area near the outside surface.

5. Conclusion

Based on the severe access limitations, the installed nonremovable insulation and the high personnel exposures necessary to accomplish insulation replacement and to inspect in this area, we conclude that performance of the required visual and surface examinations are neither desirable nor justified. A practical alternative examination is available and currently implemented, thus exception is taken to performing visual and surface examination of the reactor vessel nozzle safe ends.